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March 13, 2015
NND-15-0096
10 CFR 50.90
10 CFR 52.63

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

Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3
Combined License Nos. NPF-93 and NPF-94
Docket Nos. 52-027 & 52-028

Subject: LAR 14-07 S1 License Amendment and Exemption Request: CA04 Structural Module ITAAC Dimensions Change

- Reference:
1. South Carolina Electric & Gas Company (SCE&G) Request for License Amendment and Exemption: CA04 Structural Module ITAAC Dimensions Change September 27, 2014 (NND-14-0430) (ML14268A388)
 2. Letter from Denise L. McGovern (NRC) to Ronald Jones (SCE&G), Virgil C. Summer Nuclear Station Units 2 and 3: Request for Additional Information Letter No. 7749 (ML15007A328)

In accordance with the provisions of 10 CFR 50.90, South Carolina Electric & Gas Company (SCE&G) requests an amendment to the Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3 combined licenses (COLs) numbers NPF-93 and NPF-94, respectively. The proposed amendment would allow changes to adjust the concrete wall thickness tolerances of four Nuclear Island walls found in Tier 1. Because the proposed changes impact Tier 1 of the plant-specific DCD and Appendix C of the COL, this activity has been determined to require prior NRC approval. Also, because the change requires a departure from Tier 1 information, an exemption is requested from the requirements of the Generic DCD Tier 1 pursuant the provisions of 10 CFR 52.63(b) and 10 CFR 52.7.

On January 13, 2015, SCE&G received RAI 7749 (Reference 2). As a result, SCE&G is providing Enclosure 4 which contains the RAI response, to address NRC's questions. The RAI response commits to update the LAR as well as the UFSAR. The LAR, Exemption Request and UFSAR markup update can be found in Enclosure 5, 6 and 7 respectively. The LAR request date has been revised further as shown below.

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) are contained in Enclosure 1. Enclosure 2 provides the

background and supporting basis for the requested exemption. Enclosure 3 provides the markups depicting the requested changes to the licensing basis documents. Enclosure 1 through 3 can be found in Reference 1. As stated above Enclosure 1, 2 and 3 are effectively updated by the information contained in Enclosure 5, 6 and 7 respectively.

In order to support the VCSNS Unit 2 construction schedule, SCE&G requests NRC staff review and approval of the license amendment by May 19, 2015. Approval by this date will allow sufficient time to implement the licensing basis changes prior to affected construction activities. SCE&G expects to implement the proposed amendment within 30 days of approval.

The supplemental information provided in this letter does not impact the scope of the requested amendment, nor the conclusions of the regulatory evaluation (including the significant hazards consideration determination and the environmental considerations).

This letter contains no regulatory commitments.

In accordance with 10 CFR 50.91, SCE&G is notifying the State of South Carolina 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. Justin R. Bouknight by telephone at (803) 941-9828, or by email at justin.bouknight@scana.com.

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

Executed on this 13th day of March, 2015.

Sincerely,



April R. Rice
Manager
Nuclear Licensing

MMD/ARR/mmd

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- Enclosure 4: Virgil C. Summer Nuclear Station Units 2 and 3 – SCE&G Response to NRC Request for Additional Information Letter No. 7749 (**LAR 14-07 S1**)
- Enclosure 5: Virgil C. Summer Nuclear Station Units 2 and 3 – Updates to the License Amendment Request: CA04 Structural Module ITAAC Dimensions Change (**LAR 14-07 S1**)
- Enclosure 6: Virgil C. Summer Nuclear Station Units 2 and 3 – Updates to the Request for Exemption Regarding CA04 Structural Module ITAAC Dimensions Change (**LAR 14-07 S1**)
- Enclosure 7: Virgil C. Summer Nuclear Station Units 2 and 3 – Supplement to Proposed Changes (Public Information) (**LAR 14-07 S1**)

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**South Carolina Electric and Gas Company
Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3**

NND-15-0096

Enclosure 4

**SCE&G Response to NRC Request for Additional Information Letter No. 7749
(LAR 14-07 S1)**

Question (26401):

On page 5 of 13 of the License Amendment Request (LAR 14-07), Enclosure 1, "License Amendment Request: CA04 Structural Module ITAAC Dimensions Change," the second paragraph states, "As stated in UFSAR Subsection 3.8.3.6.1, structural module tolerances conform to the requirements of section 4 of ACI117, Sections 3.3 and 3.4 of AWS D1.1, and Section 01.23 of AISC N690, and FSAR Subsection 3.8.4.4.1 requires that the design and analysis procedures are in accordance with ACI 349." The staff interprets this statement to indicate that the proposed tolerance changes for the structural modules meet the requirements in ACI 117. The staff noticed a difference between the tolerances in ACI 117 and the tolerances proposed in the LAR. ACI 117, Section 4.5, "Deviation from Cross-Sectional Dimensions," stipulates that the thickness of elements, except slabs, where specified cross-sectional dimension is more than 36 inches (3 feet), the tolerances is plus 1 (+1) and minus 3/4 (-3/4) of an inch. These values are different than those proposed in the LAR. The licensee is requested to modify the LAR, conveying the difference between the standard and the LAR proposed wall thickness tolerance or providing a technical justification that validates this statement.

Response:

The Tier 1 Table 3.3-1 tolerances apply to wall thickness ITAAC values and do not correspond to tolerances provided in ACI 117. Construction is performed in accordance with ACI 117, which specifies the tolerances on cross-sectional dimensions as +1 and -3/4 in. for dimensions over 3 ft. ITAAC measurements are performed relative to the acceptance criteria specified by Table 3.3-1. The tolerances in Table 3.3-1 are in some cases larger than those in ACI 117 and are appropriate for the corresponding ITAAC. The ITAAC in Tier 1 Table 3.3.6 Item 3.3.00.02a.II.a that references Table 3.3-1 does not verify that the Containment Internal Structure as-built structures are in conformance with ACI 117.

The proposed changes of the tolerances in table 3.3-1 of plant-specific DCD Tier 1 of LAR 14-07 (total CA04/CB65 wall tolerance of $\pm 1-1/4$ " and total CA04/CA01 wall tolerance of $\pm 1-5/8$ ") are greater than the values described 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.

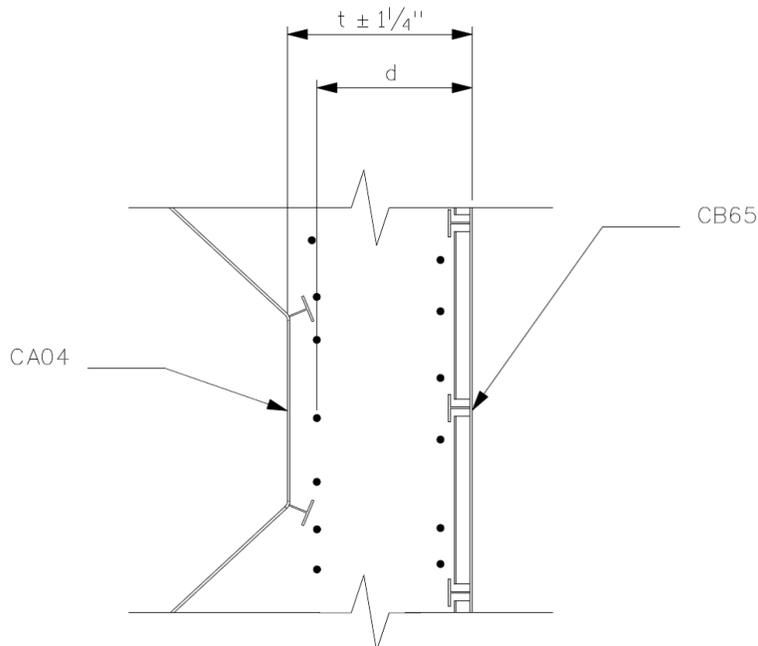


Figure 1: Concrete Cross-Section between CA04 and CB65

In Figure 1 the thickness of the wall and the proposed change requested in LAR 14-07 are shown. For evaluation of the impact and the margin of the wall design variation, the wall thickness and corresponding "d" value was set to $\pm 1\text{-}1/4\text{'}$, which is greater than ACI 117 tolerances on cross-sectional dimensions. For evaluation of the impact of the thickness differences between CA01 and CA04, the variation of the concrete thickness was set to $\pm 1\text{-}5/8\text{'}$. It should also be noted that the CA01/CA04 walls are 7'-6" to 9'-0" thick and designed as mass concrete.

As noted in the RAI UFSAR Subsection 3.8.3.6.1 is not consistent with the larger wall thickness tolerance. A proposed change to the LAR and UFSAR Subsection 3.8.3.6.1 is provided below to identify that some thickness tolerances may exceed ACI 117 tolerances. The proposed change clarifies that for walls around the reactor vessel cavity, the evaluation of the reinforcement design in the concrete conforms to the reinforcement design requirements of ACI 349.

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.

Question 26402:

On page 7 of 13 of the License Amendment Request (LAR 14-07), Enclosure 1, "License Amendment Request: CA04 Structural Module ITAAC Dimensions Change," the first paragraph states, "The proposed installation tolerance of $\pm 1\text{-}1/4$ " between the CA04/CB65 wall is greater than the ± 1 " specified in ACI 349-01, Section 7.5.2.1. To address this deviation, the total wall width was evaluated for both the larger and smaller tolerances from the actual design wall width of 36 inches. The review found that the required reinforcement in each direction increased slightly, but that the amount of reinforcement used in the wall design was still adequate." The licensee's justification for the increase in tolerances and not meeting both standards ACI 117 and ACI 349 is not clear to the staff. The licensee is requested to provide quantitative values for establishing the safety margin in the modules that do not meet ACI 117 and ACI 349 wall thickness and reinforcement location tolerance requirements.

SCE&G Response to Question 26402

The evaluation of the margin is described in reply to question 26401. For the wall between CA04 and CB65 the margin was conservatively evaluated. The minimum margin for vertical reinforcement is 48.4%, for horizontal reinforcement 55.3% and for shear 61.3 %. Since the thickness of the wall influences also the d value, an assessment of the shear reinforcement spacing was performed which found the provided spacing sufficient.

The change in tolerance between CA01 and CA04 of $\pm 1\text{-}5/8$ " leads to a decrease of the surrounding concrete by 1.3 %. This decrease is insignificant and negligible.

**South Carolina Electric and Gas Company
Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3**

NND-15-0096

Enclosure 5

**Update to License Amendment Request
CA04 Structural Module ITAAC Dimensions Change
(LAR 14-07 S1)**

With this enclosure, South Carolina Electric & Gas Company (SCE&G) has updated the LAR content submitted in Enclosure 1 of Reference 1. Text additions are denoted in purple and deletions are in red.

1. Summary Description

In accordance with 10 CFR 50.90, South Carolina Electric and Gas Company (SCE&G), the licensee for Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3, requests an amendment to Combined License (COL) Numbers NPF-93 and NPF-94, for VCSNS Units 2 and 3, respectively.

The proposed changes would revise the Combined Licenses (COLs) by increasing the concrete wall thickness tolerances of four Nuclear Island walls identified in COL Appendix C and corresponding plant-specific Tier 1 Table 3.3-1, "Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building." The changes also include an update to Tier 2 Subsection 3.8.3.6.1 as discussed further below.

The requested amendment requires changes to plant-specific Tier 1, corresponding change to COL Appendix C, and Tier 2 information. For the COL Appendix C changes proposed below, the exemption necessary to implement the corresponding change to plant-specific Tier 1 is also requested in Enclosure 2. ~~There is no UFSAR (Tier 2) change associated with the proposed COL Appendix C changes.~~ This enclosure requests approval of the license amendment necessary to implement the changes described below.

2. Detailed Description

This License Amendment Request (LAR) proposes to revise COL Appendix C Table 3.3-1, "Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building" for the concrete thickness of one wall from $\pm 1"$ to $\pm 1-1/4"$ and for the concrete thicknesses of three other walls from $\pm 1"$ to $\pm 1-5/8"$. In addition, this LAR requests the associated Tier 2 changes to identify the departure from the ACI 117 code requirements for the four walls discussed below. The need for this change was identified during a survey performed of an installed CA04 module and identification that the tolerance specified in COL Appendix C information presented 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, and 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 Figure 3.8.3-1. Plant north on 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 (termed the RCDT Compartment).

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 plant-specific DCD and 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 Reactor Coolant Drain Tank (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 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 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 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 Figure 3.8.3-1 (Sheet 3 of 7).

As stated in UFSAR Subsection 3.8.3.6.1, structural module tolerances conform to the requirements of Section 4 of ACI 117, Sections 3.3 and 3.4 of AWS D1.1, and Section Q1.23 of AISC-N690, and UFSAR Subsection 3.8.4.4.1 that requires that the design and analysis procedures are in accordance with ACI 349. The design documentation conforming to these requirements provides installation tolerances of $\pm 1/2"$ and where wall thickness measurements are not taken at internal wall module stiffeners, fabrication wall thickness tolerances of $\pm 1/2"$. Where wall thickness measurements are taken at internal wall module stiffeners, wall thickness tolerances of $\pm 1/8"$ are utilized.

A review was performed of the total ~~fabrication and assembly, and placement tolerances for these modules~~ wall tolerances, and it was identified that the total concrete thickness tolerance after fabrication and placement exceeded the $\pm 1"$ 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 Tier 1. 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	$\pm 1/8"$	$\pm 1/2"$	$\pm 5/8"$
CB65 Wall	$\pm 1/8"$	$\pm 1/2"$	$\pm 5/8"$
Total CA04/CB65 Wall Tolerance	$\pm 1/4"$	$\pm 1"$	$\pm 1-1/4"$

For the CA04/CA01 walls, measurements for the CA01 module are not taken between internal wall module stiffeners and thus have the following concrete thickness total tolerances.

	Fabrication + Assembly Tolerance	Installation Tolerance	Total Individual Wall Tolerance
CA04 Wall	$\pm 1/8"$	$\pm 1/2"$	$\pm 5/8"$
CA01 Wall	$\pm 1/2"$	$\pm 1/2"$	$\pm 1"$
Total CA04/CA01 Wall Tolerance	$\pm 5/8"$	$\pm 1"$	$\pm 1-5/8"$

Thus, to acknowledge this issue, it is proposed that the concrete thickness tolerance specified in COL Appendix C for these different concrete thicknesses be increased 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 two notes 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 first added note to COL Appendix C Table 3.3-1, labeled Note 10, is footnoted to the concrete thickness in the first item in the list (Shield Wall between Reactor Vessel Cavity and (RCDT) Room), and specifies that this concrete thickness has a construction tolerance of $\pm 1\text{-}1/4"$.

The second added note to COL Appendix C Table 3.3-1, labeled Note 11, is footnoted to the second through fourth items in the list (West Reactor Vessel Cavity Wall, North Reactor Vessel Cavity Wall, and East Reactor Vessel Cavity Wall), and specifies that these concrete thicknesses have a construction tolerance of $\pm 1\text{-}5/8"$.

In order to identify the departure from the requirements of ACI 117 for the four walls identified in this LAR, this request further proposes an update to UFSAR Tier 2 Subsection 3.8.3.6.1 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. For the affected walls, the concrete thickness tolerances listed in COL Appendix C Table 3.3-1 ~~are conservative relative to the codes and~~ do not utilize the complete allowance specified in these codes with the exception of ACI 349 and ACI 117.

The proposed installation tolerance of $\pm 1\text{-}1/4"$ between the CA04/CB65 wall is greater than the $\pm 1"$ specified in ACI 349-01 Section 7.5.2.1 and the $+1\text{'}/\text{-}3/4"$ 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.

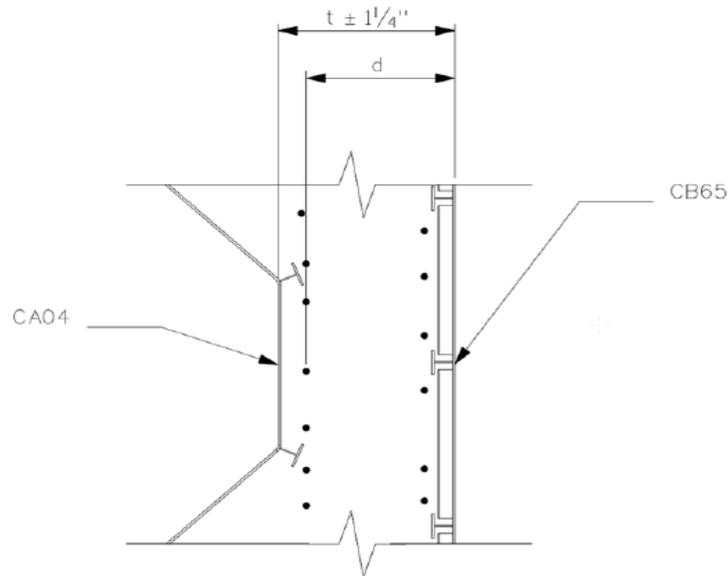


Figure 1: Concrete Cross-Section between CA04 and CB65

In Figure 1 the thickness of the wall and the proposed change requested in LAR 14-07 are shown. For evaluation of the impact and the margin of the wall design variation, the wall thickness and corresponding "d" value was set to +/- 1-1/4", 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 48.4%, for horizontal reinforcement 55.3% and for shear 61.3 %. Since the thickness of the wall influences also the d value, an assessment of the shear reinforcement spacing was performed which found the provided spacing sufficient. ~~To address this deviation, the total wall width was evaluated for both the larger and smaller tolerances from the actual design wall width of 36 inches. The review found that the required reinforcement in each direction increased slightly, but that the amount of reinforcement used in the wall design was still adequate.~~

Similarly, the CA04/CA01 walls were examined to determine the impact of the extra ±5/8" beyond the ACI 349-01 requirements and the 7/8" beyond the ACI 117 requirements. 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". It should be noted that these walls are 7'-6" to 9'-0" thick and designed as mass concrete. ~~Code allowable tolerance,~~ This assessment conservatively assumed ~~ing~~ 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 ~~minor,~~ ~~0.51.3%~~ decrease to the area of the surrounding concrete and that this ~~change in tolerance decrease has no significant impact on the in-place structural analyses or qualifications is insignificant and negligible.~~

Therefore, despite increasing the concrete thickness tolerance of these four walls, the functions of the walls are maintained and are able to perform adequately. By conforming to the other applicable codes, conformance with SRP Section 3.8.4 is also maintained.

Because the increased concrete thickness tolerances ~~either maintains~~ conformance with the applicable construction codes ~~or sufficient margin exists to justify the deviation, and do not~~

~~result in a significant dimension change~~, 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 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. Accordingly, there is no impact on the physical layout of other internal structures or system components since they utilized the tolerances found for each individual module. 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 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.

Summary

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 information, and a corresponding change to plant-specific Tier 1, Inspections, Tests, Analyses and Acceptance Criteria information; 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 50, Appendix A, General Design Criterion (GDC) 1, *Quality standards and records*, requires 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 structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunamis, 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 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

No precedent is identified.

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 and plant-specific Tier 1 Table 3.3-1, "Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building," from $\pm 1"$ to $\pm 1-1/4"$ for one wall and from $\pm 1"$ to $\pm 1-5/8"$ for the remaining three walls. The COL Appendix C changes also involve proposed changes to the corresponding information in the plant-specific Tier 1. For the COL Appendix C changes proposed, the exemption necessary to implement the corresponding change to plant-specific Tier 1 is also requested in Enclosure 2. ~~There is no UFSAR Tier 2 change associated with the proposed COL Appendix C changes.~~ 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 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 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 do not change the performance of the affected containment internal structures. As demonstrated by the continued conformance to the 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.

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 does not alter any ~~design code compliance~~, design function, design analysis, or safety analysis input or result, and sufficient margin exists to justify a departure from the standards identified in the underlying Tier 2 information with respect to 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 Consideration

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 via an exemption requested in Enclosure 2. 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 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.

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Enclosure 5 – Update to License Amendment Request: CA04 Structural Module ITAAC
LAR 14-07 S1

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.

**South Carolina Electric and Gas Company
Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3**

NND-15-0096

Enclosure 6

**Update to Exemption Request
CA04 Structural Module ITAAC Dimensions Change
(LAR 14-07 S1)**

With this enclosure, South Carolina Electric & Gas Company (SCE&G) has updated the Exemption Request content submitted in Enclosure 2 of Reference 1. Text additions are denoted in purple and deletions are in red.

1.0 Purpose

South Carolina Electric and Gas Company (SCE&G) 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 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 certification 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 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 Notes 10 and 11 of 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 UFSAR have identified the need for proposed design changes to certain concrete wall thickness tolerance 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

South Carolina Electric and Gas Company (SCE&G) is the holder of Combined License Nos. NPF-93 and NPF-94, which authorize construction and operation of two Westinghouse Electric Company AP1000 nuclear plants, named Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3, 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 add a Notes 10 and Note 11 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)
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 $\pm 1"$ to $\pm 1-1/4"$ (i.e., Note 10) for the Shield Wall between Reactor Vessel Cavity and RCDT Room wall and from $\pm 1"$ to $\pm 1-5/8"$ (i.e., Note 11) for the other three walls. The reason as to why the two altered tolerances are different is due to the location of the stiffeners relative to where the overall concrete wall thickness measurements are taken. A wall thickness measurement taken at the location of a stiffener will have a lower tolerance than a wall thickness measurement taken between stiffeners.

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. 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 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 are found to be acceptable. because for the affected walls, the concrete thickness tolerances listed in Tier 1 Table 3.3-1 are conservative relative to ACI 117, AWS D1.1, and AISC N690. The wall thickness tolerances do not utilize the complete allowance specified in these codes except ACI 349-01 and ACI 117.~~

The proposed installation tolerance of $\pm 1-1/4"$ between the CA04/CB65 wall is greater than the $\pm 1"$ 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 $\pm 1-1/4"$, 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 48.4%, for horizontal reinforcement 55.3% and for shear 61.3 %. Since the thickness of the wall influences also the d value, an assessment of the shear reinforcement spacing was performed which found the provided spacing sufficient. ~~The effects of the proposed installation tolerance of $\pm 1-1/4"$ and $\pm 1-5/8"$ were also assessed with respect to ACI 349 which specifies a $\pm 1"$. For the $\pm 1-1/4"$ the review found that the required reinforcement in each direction increased slightly, but that the amount of reinforcement used in the wall design was still adequate.~~

Similarly, the CA04/CA01 walls were examined to determine the impact of the extra $\pm 5/8$ " beyond the ACI 349-01 requirements and the $7/8$ " beyond the ACI 117 requirements. ~~For the $\pm 1-5/8$ " the impacted~~ Those walls were examined and the review found that there was a potential for a minor, ~~0-5~~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. By conforming to the other applicable codes, conformance with SRP Section 3.8.4 is also maintained. 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.

Additional detail for justification ~~for~~ of this exemption is provided in Enclosure 5. ~~Section 3 of the accompanying License Amendment Request in Enclosure 1.~~

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 SCE&G 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 §§ 50.12, 52.7, and 52.63 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 Tier 1 DCD will continue to reflect the approved licensing basis for the Licensee, and will maintain a consistent level of detail with that which is currently provided elsewhere in Tier 1 of the plant-specific DCD. The revised tolerances for the concrete wall thicknesses discussed in Section 2, do not have an adverse impact on the response of the nuclear island structures to safe shutdown earthquake ground motions or loads due to anticipated transients or postulated accident conditions, nor do they change the seismic Category I classification. The revised tolerances also do not impact the support, design, or operation of mechanical and fluid systems. There is no change to plant systems or the response of systems to postulated accident conditions. There is no change to the predicted radioactive releases due to postulated accident conditions. The plant response to previously evaluated accidents or external events is not adversely affected, and the change described does not create any new accident precursors. Therefore, no adverse safety impact that would present any additional risk to the health and safety is present. The affected Design Description in the plant-specific Tier 1 DCD will also continue to provide the detail necessary to support the performance of the associated ITAAC.

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 exemption from the requirements of 10 CFR 52, Appendix D, Section III.B would change elements of the plant-specific Tier 1 DCD by departing from the AP1000 certified (Tier 1) design information. The 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) list 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 VCSNS Units 2 and 3 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 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 this exemption request and the associated marked-up tables demonstrate that the change from the standard information provided in the generic AP1000 DCD 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. In fact, as described in item 6 below, the exemption will result in no reduction in the level of safety.

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 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 determined to be not applicable to address the acceptability of this request.

6.0 Precedent

No precedent for this request is identified.

7.0 Environmental Consideration

A review of the requested amendment, has 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.

8.0 Conclusion

The proposed changes to Tier 1 are necessary to update Note 2, and add new Notes 10 and 11 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,”* 10 CFR 51.22, *“Criteria for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review,”* 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, presents special circumstances, does not present a significant decrease in safety as a result of a reduction in standardization, and meets the eligibility requirements for categorical exclusion.

**South Carolina Electric and Gas Company
Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3**

NND-15-0096

Enclosure 7

**Supplement to Proposed Changes (Public Information)
(LAR 14-07 S1)**

With this enclosure, South Carolina Electric & Gas Company (SCE&G) has supplemented the Proposed Changes content submitted in Enclosure 3 of Reference 1. While the Tier 1 changes to Table 3.3-1 changes presented in the Enclosure 3 of Reference 1 are still valid, the changes below portray additional changes consistent with the Tier 2 changes discussed in the RAI response (Enclosure 4).

The last paragraph of UFSAR Subsection 3.8.3.6.1 is updated as follows:

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.