SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS

RELATED TO EXEMPTION AND AMENDMENT NO. 42

TO THE COMBINED LICENSE NOS. NPF-91 AND NPF-92

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MEAG POWER SPVM, LLC

MEAG POWER SPVJ, LLC

MEAG POWER SPVP, LLC

CITY OF DALTON

VOGTLE ELECTRIC GENERATING PLANT UNITS 3 AND 4

DOCKET NOS. 52-025 AND 52-026

1.0 INTRODUCTION

By letter dated September 18, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15261A757), Southern Nuclear Operating Company, Inc. (SNC/licensee) submitted license amendment request (LAR) 15-015 and requested that the U.S. Nuclear Regulatory Commission (NRC/Commission) amend the combined licenses (COL) for Vogtle Electric Generating Plant (VEGP) Units 3 and 4, COL Numbers NPF-91 and NPF-92, respectively.

The proposed LAR revises the concrete wall thickness tolerances of four containment internal structural wall modules (Shield Wall between the Reactor Vessel Cavity and the Reactor Coolant Drain Tank (RCDT) Room, West Reactor Vessel Cavity Wall, North Reactor Vessel Cavity Wall, and East Reactor Cavity Wall). These structural walls are formed by modules CA04 and CB65 as well as modules CA04 and CA01. The proposed changes to Tier 2 information in the Updated Final Safety Analysis Report (UFSAR), plant-specific Tier 1 information, and corresponding COL Appendix C information would allow an increase of the concrete wall thickness tolerances. The proposed changes would allow:

(1) a change to Tier 2 information in UFSAR Subsection 3.8.3.6.1, "Fabrication, Erection, and Construction of Structural Modules," to allow an increase in wall thickness tolerance beyond the American Concrete Institute (ACI) 349-01, "Code Requirements for Nuclear Safety Related Concrete Structures," and ACI 117, "Standard Specifications for Tolerance for Concrete Construction and Material," specified tolerance for some Containment Internal Structure (CIS) walls.

(2) the addition of Note 10 to Tier 1 Table 3.3-1, which provides the wall thickness tolerance deviations.

The licensee has also requested an exemption from the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, Appendix D, Section III.B, "Design Certification Rule for the AP1000 Design, Scope and Contents," to allow a departure from the elements of the certification information in Tier 1 of the generic Design Control Document (DCD).[1]

In order to modify the UFSAR (the plant-specific DCD) Tier 1 information, the NRC must find the licensee's exemption request included in its submittal for the LAR acceptable. The staff's review of the exemption request as well as the LAR is included in this safety evaluation.

The NRC staff issued an initial *Federal Register* notice of opportunity to request a hearing and a proposed No Significant Hazards Determination on October 8, 2015 (80 FR 60937).

By letter dated September 25, 2014 (ADAMS Accession No. ML14268A388), and supplemented by letter dated March 13, 2015 (ADAMS Accession No. ML15072A306), South Carolina Electric and Gas Company (SCE&G), the licensee for Virgil C. Summer Nuclear Station (VCSNS) Units 2 and 3, submitted LAR 14-07. VEGP's LAR 15-015 is consistent in technical content to that of the LAR submitted to the NRC by SCE&G for VCSNS Units 2 and 3 with one exception. VEGP's LAR 15-015 proposes to revise the concrete thickness tolerance for all four walls from ±1" to ±1-5/8" and VCSNS's LAR 14-07 proposed to revise the concrete thickness of one wall from ±1" to ±1-1/4" and for the concrete thicknesses of three other walls from ±1" to ±1-5/8". On August 24, 2015, the NRC issued License Amendment 29 for VCSNS Unit 2 and 3 for LAR 14-07 (ADAMS Accession No. ML15216A071).

2.0 REGULATORY EVALUATION

Tier 1 information is defined in 10 CFR Part 52, Appendix D, Section II.D, "Definitions." Information in 10 CFR Part 52, Appendix D, Section II.D.3 lists inspections, tests, analyses, and acceptance criteria (ITAAC) as part of the definition for Tier 1 information. The information that the licensee is requesting to change is referenced in ITAAC Tables. Therefore, the information is considered Tier 1 information.

In accordance with 10 CFR Part 52, Appendix D, Section VIII.A.4, exemptions from Tier 1 information are governed by the requirements of 10 CFR 52.63(b)(1) and 10 CFR 52.98(f). It also states that the Commission will deny such a request if the design change causes a significant reduction in the level of plant safety otherwise provided by the design.

Regulations in 10 CFR 52.63(b)(1) allow the licensee to request NRC approval for an exemption from one or more elements of the certification information. The Commission may only grant such a request if it complies with the requirements of 10 CFR 52.7, "Specific Exemptions," which in turn points to the requirements listed in 10 CFR 50.12, "Scope of Subpart," for specific exemptions, and if the special circumstances present outweigh the potential decrease in safety

[1] While the licensee describes the requested exemption as being from Section III.B of 10 CFR Part 52, Appendix D, the entirety of the exemption pertains to proposed departures from Tier 1 information in the generic DCD. In the remainder of this evaluation, the NRC will refer to the exemption as an exemption from Tier 1 information to match the language of Section VIII.A.4 of 10 CFR Part 52, Appendix D, which specifically governs the granting of exemptions from Tier 1 information.

due to reduced standardization. Therefore, any exemption from the Tier 1 information certified by Appendix D to 10 CFR Part 52, “Design Certification Rule for the AP1000 Design,” must meet the requirements of 10 CFR 50.12, 52.7, and 52.63(b)(1).

Regulations in 10 CFR 52.98(f) state that any modification to, addition to, or deletion from the terms and conditions of a COL including any modification to, addition to, or deletion from the ITAAC contained in the license is a proposed amendment to the license. Appendix C of COLs NPF-91 and NPF-92 contain tables and a figure, which the licensee is proposing to modify. Therefore, the proposed change requires a license amendment.

Regulations in 10 CFR Part 52, Appendix D, Section VIII.B.5.a require prior NRC approval for Tier 2 departures that involve changes to Tier 1, Tier 2* information, or the Technical Specifications. The proposed changes affect Tier 1 information and thus require NRC approval.

Regulations in 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix A, “General Design Criteria for Nuclear Power Plants,” General Design Criterion (GDC) 1, “Quality Standards and Records,” require 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.

Regulations in 10 CFR Part 50, Appendix A, GDC 2, “Design Bases for Protection against Natural Phenomena,” require 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.

Regulations in 10 CFR Part 50, Appendix A, GDC 4, “Environmental and Dynamic Effects Design Bases,” require 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.

3.0 TECHNICAL EVALUATION

3.1 EVALUATION OF EXEMPTION

INTRODUCTION

The regulations in Section III.B of Appendix D to 10 CFR Part 52 require a holder of a COL referencing Appendix D to 10 CFR Part 52 to incorporate by reference and comply with the requirements of Appendix D, including certified information in Tier 1 of the generic AP1000 DCD.

As defined in Section II of Appendix D to 10 CFR Part 52, Tier 1 information includes ITAAC. Therefore, a licensee referencing Appendix D incorporates by reference all the ITAAC contained in the generic DCD. These ITAAC, along with the plant-specific ITAAC, were enumerated in Appendix C of the COL at its issuance. The proposed changes would depart from the plant-specific DCD by revising Note 2 of Table 3.3-1, “Definition of Wall Thickness for Nuclear Island Buildings, Turbine Building, and Annex Building.” Specifically, the note is revised to add Note 10, which depicted those walls as having a tolerance of plus or minus one and five eighths of an inch (± 1-5/8"). The proposed change will also correct inconsistencies between Tier 1 and

UFSAR Tier 2. An exemption is needed because Section III.B of Appendix D to 10 CFR Part 52 requires a licensee to comply with the Tier 1 information of the generic AP1000 DCD.

In summary, the end result of this exemption would be that the licensee can implement modifications to Tier 1 information described and justified in LAR 15-015 if the NRC approves LAR 15-015. This is a permanent exemption limited in scope to the particular Tier 1 information specified.

As stated in Section VIII.A.4 of Appendix D to 10 CFR Part 52, an exemption from Tier 1 information is governed by the requirements of 10 CFR 52.63(b)(1) and 52.98(f). Additionally, the Commission will deny a request for an exemption from Tier 1 if it finds that the design change will result in a significant decrease in the level of safety. Pursuant to 10 CFR 52.63(b)(1), the Commission may, upon application by an applicant or licensee referencing a certified design, grant exemptions from one or more elements of the certification information, so long as the criteria given in 10 CFR 52.7 are met, and that the special circumstances as defined by 10 CFR 50.12 outweigh any potential decrease in safety due to reduced standardization.

Pursuant to 10 CFR 52.7, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 52. Regulations in 10 CFR 52.7 further states that the Commission's consideration will be governed by 10 CFR 50.12, "Specific exemptions," which states that an exemption may be granted when: (1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and (2) special circumstances are present. Regulations in 10 CFR 50.12(a)(2) list six special circumstances for which an exemption may be granted. It is necessary for one of these special circumstances to be present in order for NRC to consider granting an exemption request. The licensee stated that the requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii). That subsection defines special circumstances as when "[a]pplication 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 staff's analysis of each of these findings is presented below.

3.1.1 Authorized by Law

This exemption would allow the licensee to implement approved changes to Tier 1 Table 3.3-1. This is a permanent exemption limited in scope to particular Tier 1 information, and subsequent changes to Tier 1 Table 3.3-1 or any other Tier 1 information, would be subject to the exemption process specified in Section VIII.A.4 of Appendix D to 10 CFR Part 52. As stated above, 10 CFR 52.63(b)(1) allows the NRC to grant exemptions from one or more elements of the Tier 1 information. The NRC staff has determined that granting of the licensee's proposed exemption will not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations. Therefore, as required by 10 CFR 50.12(a)(1), the exemption is authorized by law.

3.1.2 No Undue Risk to Public Health and Safety

The underlying purpose of Appendix D to 10 CFR Part 52 is to ensure that the licensee will construct and operate the plant based on the approved information found in the DCD incorporated by reference into the licensee's licensing basis. The changes to the design details for the structural wall modules do not have an adverse impact on the response of the nuclear island structures to safe shutdown earthquake ground motions, loads due to anticipated transients, or postulated accident conditions, nor do they change the seismic Category I classification. These changes will not impact the ability of the structures to perform their design function, including with respect to structural integrity and radiation protection. Because the changes will not alter the operation of any plant equipment or systems, these changes do not present an undue risk from existing equipment or systems. These changes do not add any new equipment or system interfaces to the current plant design. The changes 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, operational controls, or safeguards intended to mitigate any existing onsite 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 significant fuel cladding failures. Accordingly, these changes do not present an undue risk from any new equipment or systems. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that there is no undue risk to public health and safety.

3.1.3 Consistent with the Common Defense and Security

This exemption would allow the licensee to implement approved changes to Tier 1 Table 3.3-1. This is a permanent exemption limited in scope to particular Tier 1 information. Subsequent changes to Table 3.3-1 or any other Tier 1 information would be subject to Appendix D to 10 CFR Part 52. The change does not alter or impede the design, function, or operation of any plant structures, systems, or components (SSCs) associated with the facility's physical or cyber security, and therefore does not affect any plant equipment that is necessary to maintain a safe and secure plant status. In addition, the change has no impact on plant security or safeguards. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the common defense and security is not impacted by this exemption.

3.1.4 Special Circumstances

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present whenever 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 underlying purpose of the Tier 1 information is to ensure that the licensee will safely construct and operate the plant based on the certified information found in the AP1000 DCD that was incorporated by reference into the licensee's licensing basis. The corresponding changes to the design details for the structural wall modules result in no appreciable decrease in the design margins of the internal containment structures, as explained further below in the staff's evaluation in Section 3.2. These changes are necessary to enhance the ability of the licensee to construct the plant based on the information in the certified design, by clarifying the information found in Table 3.3-1. If this exemption is not granted and the proposed changes in the LAR are not allowed to be implemented, then the Tier 1 ITAAC would not conform to the UFSAR Tier 2 design descriptions, and the performance of the Tier 1 ITAAC would not accurately verify construction of the proposed design. Therefore the staff finds the special circumstances exist for granting of an exemption from the Tier 1 information as required by 10 CFR 50.12(a)(2)(ii).

3.1.5 Special Circumstances Outweigh Reduced Standardization

This exemption would allow the implementation of changes to Table 3.3-1 proposed in the LAR. Based on the nature of the proposed changes to the generic Tier 1 information and the understanding that these changes were identified during the design finalization process for the AP1000, this exemption may be requested by other AP1000 licensees and applicants. However, the staff's review of the reduction in standardization resulting from 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 containment internal structural wall modules associated with this request will continue to be maintained with no appreciable decrease in the design margins for those modules. While the text in the Table 3.3-1 may be changed, the changes have no effect on any SSCs meeting their design function. Therefore, as required by 10 CFR Part 52.63(b)(1), the staff finds that the special circumstances outweigh the effects the departure has on the standardization of the AP1000 design.

3.1.6 No Significant Reduction in Safety

This exemption would allow the implementation of changes to Table 3.3-1 proposed in the LAR. The corresponding changes to the design details for the structural wall modules result in no appreciable decrease in the design margins of the internal containment structures, as discussed further below in Section 3.2. The proposed changes to Table 3.3-1 will not adversely affect the ability of the SSCs to perform their design functions and the level of safety provided by the SSCs is unchanged. Therefore, as required by 10 CFR Part 52, Appendix D, Section VIII.A.4, the staff finds that granting the exemption would not result in a significant decrease in the level of safety otherwise provided by the design.

3.2 Evaluation of Proposed Changes

To perform the technical evaluation, the NRC staff considered UFSAR Tier 1 Section 3.3, "Buildings," and Tier 2 Section 3.8, "Design of Category I Structures." The staff also examined the portions of NUREG–1793, Supplement 2, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design" (ADAMS Accession No. ML112061231) and portions of NUREG-2124, Volume 1, "Final Safety Evaluation Report Related to the Combined Licenses for Vogtle Electric Generating Plant, Units 3 and 4," (ADAMS Accession No. ML12271A045) documenting the staff's technical evaluation of those aspects of the AP1000 DCD and Vogtle COL applications, respectively. The NRC staff reviewed the licensee's proposed UFSAR changes to wall thickness tolerances to confirm that the safety of the affected CIS walls is not compromised by the proposed increase in tolerance and that the changes to wall thickness tolerances will not not result in increased radiation exposures to plant personnel.

UFSAR Subsection 3.8.3.1 states that the AP1000 CIS walls are designed using reinforced concrete and structural steel. The AP1000 CIS walls are a mix of steel-concrete (SC) composite modules and reinforced concrete walls.

The SC wall modules consist of steel faceplates connected by trusses. Shear studs are welded to the interiors of the module faceplates. Steel faceplate connections are complete joint penetration welds such that the full capacity of the steel plates is developed across the joint. Concrete is poured between the steel faceplates, which serve as forms. Once the concrete in the wall modules cures, the concrete, trusses, faceplates, and the shear studs act as a lateral force resisting system, behaving as a shear wall, to resist design basis loads. Examples of SC walls are those that form the CA01 module.

Some reinforced concrete walls are attached to permanent steel plate formwork, which lines the perimeter of compartments or cavities. Examples of cavities are the reactor vessel cavity (RVC) and the RCDT. The RVC is formed by the CA04 module, and the RCDT is formed by the CB65 module. The formwork steel plates in CA04 and CB65 have welded steel angles and "T" sections embedded in the concrete to anchor the steel plates once concrete is poured.

In some locations, CIS walls hold reinforced concrete between SC wall modules and formwork steel plates. These sections (steel plate formwork – concrete – SC wall module) are to be treated as a single wall. An example is the CA04/CA01 wall. Steel plates on both sides of the wall shall be anchored to the concrete, and the section should behave as a reinforced concrete wall. In other locations, CIS walls hold reinforced concrete between formwork steel plates. These sections (steel plate formwork – concrete – steel plate formwork) are also to be treated as a single wall. An example is the CA04/CB65 wall.

The shear studs and trusses are designed in accordance with the provisions of the American Institute of Steel Construction (AISC) Standard Specification, AISC N690-1994, "Specification for the Design, Fabrication, and Erection of Steel Safety Related Structures for Nuclear Facilities." The SC wall modules and composite sections are designed in accordance with the provisions of the American Concrete Institute (ACI) Code, ACI 349-01, "Building Code Requirements for Nuclear Safety Related Structures."

Under this LAR the licensee proposed to depart from Tier 1 material in UFSAR Table 3.3-1, "Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building." The proposed changes add a footnote to Tier 1 Table 3.3-1 indicating increased thickness tolerances for walls around the RVC. The changes apply to specific walls between elevations 71'-6" and 98'-0". These structural walls are formed by modules CA04 and CB65 as well as modules CA04 and CA01. The original wall thickness tolerance for CIS walls (as approved for the AP1000 DCD and specified in ACI 349-01) is ±1". The LAR proposes a revised thickness tolerance of ±1-5/8" for the wall formed by modules CA04 and CB65 and for the walls formed by modules CA04 and CA01. The 3'-0" thick CA04/CB65 wall is located between the RVC and the RCDT. The 7'-6" and 9'-0" thick CA04/CA01 walls are located around the West, North, and East directions of the RVC. The new tolerance exceeds the approved wall thickness tolerance of ±1".

The licensee also proposed a change to Tier 2 material in UFSAR Subsection 3.8.3.6.1. The proposed change modifies the text to allow an increase in wall thickness tolerance beyond ACI 349-01 specified tolerances for certain CIS walls.

The staff's evaluation of these design changes is summarized below.

3.2.1 Effect of increase of wall thickness tolerance on structural integrity

In the LAR, the licensee stated that UFSAR Subsection 3.8.3.6.1 requires structural module tolerances to conform to ACI 117, American Welding Society (AWS) D1.1, "Structural Welding Code – Steel," and AISC N690 and that UFSAR Subsection 3.8.4.4.1, "Seismic Category I Structures," requires design and analysis procedures to conform with ACI 349. The licensee stated that for the affected walls, the proposed new thickness tolerance (±1-5/8") exceeds the tolerance requirements of industry codes ACI 117 (+1" and -3/4") and ACI 349-01(± 1"). The staff concluded that the increased tolerance changes for the aforementioned walls do not affect the tolerance requirements prescribed in AWS D1.1 and AISC N690 codes for the fabrication, assembly, and installation of the structural modules.

The affected walls, which are a mix of SC composite modules and reinforced concrete, are designed using reinforced concrete requirements per ACI 349-01 and structural steel requirements per AISC N690. The aforementioned codes are robust in that they include safety factors that account for the uncertainties that exist in structural design. To address the tolerance deviation from that required by ACI 349-01 and ACI 117, the licensee performed an assessment of the smallest of the affected walls (CA04/CB65), which is 36" thick, in order to determine the potential impact on the margin of safety (ratio of the reinforcement required by the Code to the reinforcement provided by the design) as a result of the increased tolerance.

That assessment showed that the minimum margin for the vertical reinforcement is 47.9%, the horizontal reinforcement is 54.8%, and the shear reinforcement is 61.3%. The results of the tolerance increase indicate that sufficient margin exists because, even assuming the minimum tolerance permitted under the proposed change, the provided reinforcement of the affected walls continues to exceed the minimum reinforcement required by ACI 349-01 and ACI 117. In addition, the licensee stated that the shear reinforcement spacing is sufficient for the increased tolerances.

The licensee also performed a similar assessment on the thicker CA01/CA04 walls, which are 7'-6" to 9'-0" thick. Because of their size, the walls 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. The results showed an insignificant change in the volume of concrete. The licensee then conservatively assumed the worst case tolerance in all directions of the thicker wall (CA04), instead of only the affected walls (north, east and west side of the CA04 Module). Even with these conservative assumptions, the results of the assessment on the thicker wall indicate that there is a minimal decrease in volume, less than 1.5%, to the area of the surrounding concrete which forms the reactor vessel cavity. The licensee concluded that the reinforcement used in the affected wall design was adequate and that the potential decrease in the volume of the surrounding concrete is insignificant and has negligible effect on the structural analyses.

The staff reviewed the LAR and compared the results of the licensee’s assessment to the requirements in ACI 349-01 and ACI 117. The staff’s evaluation concluded that the margin provided by the reinforcement (for the thinner walls), and the volume of concrete (for the thicker walls) provides reasonable assurance that the tolerance changes will not compromise the intended safety functions of the affected walls (CA04/CB65 and CA01/CA04). In particular, the staff confirmed that for the thinner walls, even assuming the minimum tolerance, the provided reinforcement would still remain well above the minimum reinforcement required by the Code. Likewise, for the thicker walls, the staff confirmed that even conservatively assuming the worst case tolerance, any decrease in concrete volume would be minimal, less than 1.5%, which the staff agreed would constitute an insignificant change that would have negligible effect on structural analyses. For these reasons, neither change in tolerance would compromise the intended safety functions of any of the affected walls. Therefore, the tolerance deviation from the ACI 349-01 and ACI 117 codes is acceptable.

In Enclosure 3 of the LAR, the licensee proposed the licensing basis change descriptions to the affected subsections of the UFSAR to be consistent with the proposed wall thickness tolerances for the CA04/CB65 and CA01/CA04 walls. The licensee proposed to revise Subsection 3.8.3.6.1 of the UFSAR for consistency as a result of the proposed changes to the tolerances of the CA04/CB65 and CA01/CA04 walls.

The staff reviewed the proposed changes along with the referenced enclosures and the additional information in Subsection 3.8.3.6.1 of the UFSAR and agrees that the changes to the UFSAR descriptions accurately reflect the proposed changes to wall thickness tolerances; for the reasons stated above in Section 3.2.1, the staff agrees that the proposed changes are acceptable.

3.2.2 Effect of increase of wall thickness tolerance on radiation exposures to plant personnel

The staff evaluated the effects of the increase of wall thickness tolerances for the four walls listed in the LAR on radiation exposures to plant personnel.

The West and East Reactor Vessel Cavity Walls both have a nominal thickness of 7'-6" and separate the Reactor Vessel Cavity from the lower sections of the Steam Generator 1 and 2 Compartments. These steam generator compartments are designated as Zone VI areas (less than or equal to 10 rem/hr) during full power operation and, because of the high expected dose rates in these areas, workers would be expected to have very limited access to these areas during full power operation. Since the steam generator compartments are designated as high radiation areas, access to these rooms is controlled according to the requirements of 10 CFR 20.1601. Based on the 7'-6" nominal thickness of the reactor vessel cavity shield walls in these two areas, the specified increase of wall thickness tolerance will have a negligible effect on the dose rates in the Steam Generator Compartments and will not result in an increase in the designated plant radiation zones for these areas.

The North Reactor Vessel Cavity Wall has a nominal thickness of 9'-0" and separates the Reactor Vessel Cavity from a stairwell area. This stairwell area is designated as a Zone VI area (less than or equal to 10 rem/hr) during full power operation and, based on the expected dose rates in this stairwell area, workers would be expected to have very limited access to this area during full power operation. Since this stairwell area is designated as a high radiation area, access to the stairwell area is controlled according to the requirements of 10 CFR 20.1601. On the basis of the 9'-0" nominal thickness of the shield wall separating the reactor cavity from this stairwell area, the specified increase of wall thickness tolerance will have a negligible effect on the dose rates in this stairwell area and will not result in an increase in the designated plant radiation zone for this area.

The fourth wall specified in the LAR has a nominal thickness of 3'-0" and is the shield wall between the Reactor Vessel Cavity and the RCDT room. Both of these areas are located at the lowest elevation (elevation 66'-6") in the Containment Building. The Reactor Vessel Cavity is designated as a Zone IX area (less than or equal to 500 rads/hr) during full power operation and the RCDT room is designated as a Zone VI area (less than or equal to 10 rem/hr) during full power operation. The Reactor Vessel Cavity is designated as a very high radiation area; that area is locked and access to it is controlled according to the requirements of 10 CFR 20.1602. The Reactor Vessel Cavity would not be accessible during power operation. With respect to the RCDT room, because the dose rate in the RCDT room is higher than 100 mrem/hr, it is designated as a high radiation area and access to this room is controlled according to the requirements of 10 CFR 20.1601. The licensee stated in the LAR that the specified increase of wall thickness tolerance will not result in an increase in the designated plant radiation zone for the RCDT room.

The areas discussed above are all within the Radiation Controlled Area (RCA) of the Vogtle plant. As discussed in Section 12.5.4, "Controlling Access and Stay Time," of the AP1000 DCD, entrance to the RCA area is normally through the access control area at the health physics area entry/exit location in the annex building. High and very high radiation areas are segregated and identified in accordance with the applicable requirements of 10 CFR Part 20. The entrances to high and very high radiation areas are locked or barricaded and equipped with audible and/or visible alarms, as required.

On the basis that the four shield walls are located in remote, high radiation area locations in containment, where personnel access is restricted or not permitted during power operations and the fact that the proposed increased wall tolerance for each of these four shield walls will not result in an increase in the designated plant radiation zones for the adjacent areas, the staff finds the proposed tolerance changes for these shield walls to be acceptable.

It should be noted, however, that the staff's evaluation and acceptance of these tolerance changes pertains only to the specified tolerances for the four shield walls described in this LAR.

CONCLUSION

The staff reviewed the licensee's proposed changes provided in the LAR. Based on the staff's technical evaluation, the staff finds that:

(1) The proposed change to include a tolerance increase of ± 1-5/8 of an inch (Note 10 in Tier 1 Table 3.3-1) provides wall thickness tolerance deviations which do not affect the structural integrity of the affected CIS walls, and the safety margin remains substantial for the range of potential wall thicknesses. The proposed changes to provide additional thickness tolerances to the CA01/CB65 and CA01/CA04 walls are acceptable because the licensee performed an analysis of the walls and determined that sufficient safety margin exists for the proposed thickness tolerances. In addition, the change in the tolerance will not result in an increase in the designated plant radiation zones for these areas.

(2) The proposed change to identify a departure from the ACI 349-01 and the ACI 117 code requirements for some CIS wall thickness tolerances is acceptable because the increased tolerance on the RVC walls does not impair the ability of the walls to perform their safety function including with respect to structural integrity and radiation protection.

For the reasons specified above, the staff finds that the proposed UFSAR amendments to Tier 1 Table 3.3-1 and Tier 2 Subsection 3.8.3.6.1 of the UFSAR are acceptable. Furthermore, the supporting analysis provided in the LAR does not alter the relevant conclusions made for the AP1000 standard design.

Based on these findings, the NRC staff concludes that there is reasonable assurance that the requirements of General Design Criterion (GDC) 1, GDC 2, and GDC 4 of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A ("General Design Criteria for Nuclear Power Plants"), and Appendix D ("Design Certification Rule for the AP1000 Design") to 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," will continue to be met. Therefore, the staff finds the proposed changes to be acceptable.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in the margin of safety.

The staff provides its final no significant hazards consideration below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The increase in tolerance associated with the concrete thickness of four 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. (Note that the four walls range in nominal thicknesses from 3', 7'-6", and 9'). The new wall thicknesses will be in accordance with the Final Safety Analysis Report and adhere to other applicable codes and standards and are adequate for safe construction and eventual operation of the AP1000 at the Vogtle site. The proposed change to wall thickness tolerances (an additional plus or minus five eighths of an inch beyond what was previously approved in the AP1000 design certification and VEGP COLs) does not change the functionality of the walls nor affect the functions of any containment internal structures, systems or components. The increase of the wall thickness tolerances does not negatively impact any safety-related or important to safety design feature nor will it affect the ability of the walls to perform their structural or radiation protection functions. There is no effect on the likelihood that a design basis accident or an accident for which the facility was designed to experience will occur. Likewise, the change in wall tolerances does not impacts any safety margins and therefore will not significantly increase the consequences of an accident previously evaluated. 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.

2. Does the proposed change 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 with respect to structural or radiation protection functions. 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. The proposed changes have no impact on any of the factors that might lead to a new or different kind of accident from any accident previously evaluated. There is no new design feature introduced based on the proposed changes nor is there any effect on the performance capabilities of systems, structures, and components that may contribute to the possibility of a new or different kind of accident from any accident previously evaluated. The suite of design basis accidents or other accidents for which the AP1000 is designed remains unaffected by the proposal to increase by five eighths of an inch the wall thickness tolerance for these four walls. 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.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change to increase the concrete thickness tolerance for four containment internal walls does not alter any design function, design analysis, or safety analysis input or result. The increase in the acceptable wall thickness tolerance may, in one potential case, result in a wall that is five eighths of an inch thinner than what was previously approved; however, the safety margin is not significantly reduced because of the significant conservatism that was (and still is) incorporated into the design and construction of the walls. The walls will still be constructed to high standards as detailed in the Final Safety Analysis Report and in accordance with other nuclear codes and standards. The result of potentially decreasing the wall thickness by five eighths of an inch does not decrease the margin of safety significantly because of the highly conservative nature of the design and construction of the walls. Likewise, a potential increase in any of the wall thicknesses by five eighths of an inch will not negatively impact the safety functions of and structures, systems, or components. No safety margins will be significantly affected and, in fact, some margins may increase if the wall is thicker. 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.

For the reasons set forth above, the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff finds that the license amendment request involves no significant hazard considerations.

On November 9, 2015, the staff received a public comment from the Blue Ridge Environmental Defense League and its Chapter Concerned Citizens of Shell Bluff (BREDL) regarding the technical aspects of this proposed amendment request (Reference 11). In its public comment, BREDL expressed concerns regarding the requested changes in concrete thickness tolerance and their potential significance for safety margins. For the reasons set forth in Section 3.2 of this Safety Evaluation, and as summarized in the staff Final No Significant Hazards

Determination, the staff has determined that the requested changes in wall thickness tolerance do not affect the structural integrity of the affected CIS walls, that the safety margin remains substantial for the range of potential wall thicknesses, and that the changes do not impair the ability of the walls to perform their safety function including with respect to structural integrity and radiation protection.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations in 10 CFR 50.91(b) (2), the Georgia State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, "Standards for Protection Against Radiation." The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration (80 FR 60937; published on October 8, 2015) and the discussion in Section 4.0 above continues to support that proposed finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

Because the exemption is necessary to allow the changes proposed in the license amendment and because the exemption does not authorize any activities other than those proposed in the license amendment, the environmental consideration for the exemption is identical to that of the license amendment. Accordingly, the exemption meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the exemption.

7.0 CONCLUSION

The Commission has determined that pursuant to Section VIII.A.4 of Appendix D to 10 CFR Part 52, the exemption (1) is authorized by law, (2) presents no undue risk to the public health and safety, (3) is consistent with the common defense and security, (4) is a special circumstance, (5) that outweighs the reduction in standardization, and (6) does not significantly reduce the level of safety at the licensee's facility. Therefore, the staff grants the licensee an exemption from the Tier 1 information specified by the licensee.

The Commission has concluded, based on the considerations discussed above that: (1) the amendment does not (a) involve a significant increase in the probability or consequences of an accident previously evaluated, (b) create the possibility of a new or different kind of accident from any accident previously evaluated, or (c) involve a significant reduction in a margin of safety and therefore, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by construction activities in the proposed manner; (3) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations; and (4) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. Therefore, the staff finds the changes proposed in this license amendment acceptable.

8.0 REFERENCES

1. Request for License Amendment and Exemption 15-015: CA04 Structural Module ITAAC Dimensions Change, letter from Southern Nuclear Operating Company, dated September 18, 2015 (ADAMS Accession No. ML15261A757).

2. Request for License Amendment and Exemption 14-07: CA04 Structural Module ITAAC Dimension Change, letter from South Carolina Electric & Gas, dated September 25, 2014 (ADAMS Accession No. ML 14268A388).

3. Request for License Amendment and Exemption 14-07 S1: CA04 Structural Module ITAAC Dimension Change, letter from South Carolina Electric & Gas, dated March 13, 2015 (ADAMS Accession No. ML15072A306).

4. Vogtle Electric Generating Plant Updated Final Safety Analysis Report, Revision 3, dated June 27, 2014 (ADAMS Accession No. ML14183A994).

5. AP1000 Design Control Document, Revision 19, dated June 13, 2012 (ADAMS Accession No. ML11171A500).

6. Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design, NUREG-1793, Supplement 2, dated August 5, 2011 (ADAMS Accession No. ML112061231).

7. Vogtle Electric Generating Plant, Final Safety Evaluation Report, dated August 5, 2011 (ADAMS Accession No. ML111950510).

8. American Concrete Institute (ACI), ACI-349-01, "Building Code Requirements for Nuclear Safety Related Structures."

9. American Institute of Steel Construction (AISC), AISC-N690-1994, "Specification for the Design, Fabrication, and Erection of Steel Safety Related Structures for Nuclear Facilities."

10. American Concrete Institute (ACI), ACI-117, "Specification for Tolerances for Concrete Construction and Materials and Commentary."

11. Comment from Blue Ridge Environmental Defense League on Vogtle Electric Generating Station, Units 3 and 4; License Amendment Application, dated September 18, 2015 (ADAMS Accession No. ML15320A016).