



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 173 AND 171

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

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, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT UNITS 3 AND 4

DOCKET NOS. 52-025 AND 52-026

1.0 INTRODUCTION

By letter dated July 16, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML19197A278), as supplemented by letters dated October 3, 2019 (ADAMS Accession No. ML19276G437) and December 12, 2019 (ADAMS Accession No. ML19346E598) Southern Nuclear Operating Company, Inc. (SNC) requested that the U.S. Nuclear Regulatory Commission (NRC or Commission) amend the Vogtle Electric Generating Plant (VEGP) Units 3 and 4, Combined License (COL) Nos. NPF-91 and NPF-92, respectively. The License Amendment Request (LAR) 19-003 requested to depart from Updated Final Safety Analysis Report (UFSAR) Tier 2 information (which includes the plant-specific Design Control Document (PS-DCD) Tier 2 information) and involved related changes to PS-DCD Tier 1 information, with corresponding changes to the associated COL Appendix C information.

Pursuant to Section 52.63(b)(1) of Title 10 of the *Code of Federal Regulations* (10 CFR), SNC also requested an exemption from the provisions of 10 CFR Part 52, Appendix D, "Design Certification Rule for the AP1000 Design," Section III.B, "Scope and Contents." The requested exemption would allow a departure from the corresponding portions of the certified information

in Tier 1 of the generic DCD.<sup>1</sup> In order to modify the PS-DCD Tier 1 information, the NRC must find SNC's exemption request included in its submittal for the LAR to be acceptable. The staff's review of the exemption request, as well as the LAR, is included in this safety evaluation.

The supplements dated October 3, 2019, and December 12, 2019, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 8, 2019 (84 FR 53777).

## 2.0 REGULATORY EVALUATION

LAR 19-003 requested changes to COL Appendix C and corresponding PS-DCD Tier 1 and Tier 2 information to incorporate design basis passive residual heat removal (PRHR) heat exchanger (HX) leakage to the in-containment refueling water storage tank (IRWST) as a radiation source.

In terms of environmental qualification (EQ) for structures systems and components (SSCs), the purpose of the LAR is to update the EQ analyses from those of the originally licensed design to accommodate new source term assumptions. This update includes radiation zoning changes and the evaluation of the increased doses to equipment, due to the IRWST source term. In addition, although the north-east wall and the west wall of the IRWST have existed in the design, they were not credited in the original radiation shielding analysis. The LAR now proposes to credit these walls for radiation shielding purpose.

The staff considered the following regulatory requirements in reviewing the LAR that included the proposed changes:

Appendix D, Section VIII.A.4 to 10 CFR Part 52 states that exemptions from Tier 1 information are governed by the requirements in 10 CFR 52.63(b)(1) and 10 CFR 52.98(f). It also states that 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 otherwise provided by the design.

10 CFR 52.63(b)(1) allows the licensee who references a design certification rule 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 determines that the exemption will comply with the requirements of 10 CFR 52.7, which, in turn, points to the requirements listed in 10 CFR 50.12 for specific exemptions. In addition to the factors listed in 10 CFR 52.7, the Commission shall consider whether the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. Therefore, any exemption from the Tier 1 information certified by Appendix D to 10 CFR Part 52 must meet the requirements of 10 CFR 50.12, 52.7, and 52.63(b)(1).

10 CFR 52.98(f) requires NRC approval for any modification to, addition to, or deletion from the terms and conditions of a COL. These activities involve a change to COL Appendix C ITAAC

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<sup>1</sup> While SNC 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 PS-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.

information, with corresponding changes to the associated PS-DCD Tier 1 information. Therefore, NRC approval is required prior to making the plant specific proposed changes in the LAR.

10 CFR 52.47(a)(5) requires applicants to identify the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radiation exposures within the limits set forth in part 20 of this chapter.

10 CFR Part 50, General Design Criterion (GDC) 61 - "Fuel storage and handling and radioactivity control," requires, in part, that systems which may contain radioactivity be designed with a capability to permit appropriate periodic inspection and testing of components important to safety, with suitable shielding for radiation protection, and with appropriate containment, confinement, and filtering systems.

10 CFR Part 50, Appendix A, GDC 4, "Environmental and dynamic effects design bases," requires that structures systems and components (SSCs) important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions (including radiation) associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions inside and outside the nuclear power unit.

10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants" specifically addresses EQ for electrical equipment including temperature, pressure, humidity, radiation, chemicals and submergence at the location where the equipment must perform its intended function(s).

Since the LAR does not include any physical changes to the approved design and is based solely upon the specific changes associated with analyses concerning the level of potential radiation exposure to localized equipment in and around the IRWST, radiation dose is the only environmental factor of concern for this proposed change.

### 3.0 TECHNICAL EVALUATION

#### 3.1 EVALUATION OF THE REQUESTED CHANGES

As described in LAR-19-003, SNC proposed to revise the COLs to incorporate the source term contribution of design basis PRHR HX leakage to the IRWST. The source term for the IRWST is based on the technical specification operational leakage from the PRHR HX into the IRWST of 500 gallons per day (gpd). The 500 gpd limit is assumed to be leaking from a single crack. The leakage, assumed to occur with the maximum fuel failure percentage allowed by technical specifications, for the design basis source term, results in the IRWST liquid and vapor source term being a significant radiological source inside containment. The IRWST source term affects normal operating plant zoning inside containment and results in a significant normal operation dose increase to equipment located within the IRWST.

Consideration of the IRWST normal operating doses results in the need to credit the north-east wall and west wall of the IRWST as radiation shielding walls in PS-DCD Tier 1, and associated

COL Appendix C, Table 3.3-1. These walls were previously not credited for plant radiation shielding or radiation zoning analysis.

In addition to the Tier 1 changes to Table 3.3-1, the LAR also includes Tier 2 changes, including: 1) a description of the IRWST source in UFSAR Sections 11.1.1.5 and 12.2.1.1.11; 2) the addition of the normal and design basis IRWST gamma source strengths and radionuclide specific activities in UFSAR Tables 12.2-30 and 12.2-31; 3) revisions to the normal operation radiation zone maps provided in UFSAR Figure 12.3-1, and; 4) the addition of the IRWST to UFSAR Table 3D.5-2, which provides the environmental qualification gamma dose rate and the 60-year total integrated gamma dose inside the IRWST.

As stated in the LAR, in combination with the proposed addition of Sections 11.1.1.5 and 12.2.1.1.11 of the UFSAR, the design basis IRWST source term provided in Table 12.2-30 is based on 500 gpd PRHR HX leaking into the IRWST at the design basis fuel defects (i.e., the leakage is assumed to be at the concentrations provided in UFSAR Table 11.1-2). The staff verified the source term for some of the radionuclides in Table 12.2-30 to be based on leakage into the IRWST at these concentrations for the entire operating cycle. The spent fuel pool cooling system (SFS) purifies the IRWST and is assumed to be operating at 10 gpm. The noble gas source term continually builds up in the IRWST with the only removal mechanism being radioactive decay. SNC assumes that all halogens are in the liquid source term and that there are no significant halogen concentrations in the vapor source term. SNC indicated that conversion to elemental iodine will be minimal and that partitioning across the surface of the water is minimal. SNC also indicated that the methodology used in concluding that no insignificant iodine would be found in the IRWST vapor space is based on NUREG/CR-5950. In performing a review of the source terms and potential doses, the staff determined that a large portion of the iodine would need to partition to the vapor space to result in a significant contribution to the dose from the vapor source term.

The staff concurs with the LAR's analysis of the maximum source term that could be leaked into the IRWST, without exceeding plant technical specification limits, since the concentration of the leakage is based on the maximum failed fuel percentage allowed by the technical specifications and the leakage is at the maximum rate allowed by technical specifications. Since the IRWST fluid is not expected to be at high temperature during normal operation and the concentrations of iodine in the IRWST from the leaked fluid are spread over the large water volume of the IRWST, the staff would not expect a large portion of the iodine to partition to the vapor space. Therefore, the staff finds it reasonable to assume that there is a negligible contribution to dose rates from iodine in the vapor space of the IRWST compared to the contribution of noble gases in the vapor space source term.

The design basis source terms are based at the maximum concentration at the end of the operating cycle. For calculating the total integrated dose (TID) to equipment, the source term is assumed to be at this maximum concentration (at the end of the operating cycle) continuously for 60 years for determining the TID to equipment. Since the IRWST is drained during each refueling outage to fill the reactor cavity, it is not expected that the radionuclide concentrations would be at the concentrations found at the end of the operating cycle during much of plant operation. The staff reviewed the LAR and found the above assumptions to be acceptable for determining the maximum source term for the IRWST for calculating radiation doses inside containment and for calculating the doses to equipment. Therefore, the staff determined that SNC appropriately characterized the kinds and quantities of radioactive material inside the IRWST in accordance with 10 CFR 52.47(a)(5).

The staff reviewed the source terms provided in the proposed additions to Tables 12.2-30 and 12.2-31 and verified that the radionuclide concentrations provided for numerous radionuclides in both the liquid and vapor source terms were consistent with the assumptions described above. In addition, the staff performed independent radiation dose calculations using the liquid and vapor source terms and found the gamma doses calculated outside the tank to be consistent with doses provided in the LAR and the UFSAR, with credit given for the north-east wall and west wall of the IRWST as radiation shielding walls.

SNC also included details on the updated UFSAR Chapter 12 radiation zone maps, provided in the LAR, specifying that shine above the IRWST vents and hatches may exceed the designated dose values. The staff determined that this is acceptable because workers would not be expected to normally spend a significant amount of time in those areas when the tank is full of water and radioactive materials. In addition, SNC specified that there is no safety-related equipment above the IRWST hatches and that there is no qualified or survivable equipment affected by the proposed changes besides the IRWST vents and the hydrogen ignitors inside the IRWST. According to SNC, as provided on Page 9 of 20 of the LAR, there is no equipment impacted by the additional IRWST source term, except the IRWST vents and the non-safety related hydrogen ignitors. Workers would not be expected to enter the IRWST when it is full of highly radioactive fluids and the water and radioactive material will be drained during outages to fill the refueling cavity. A worker entering the IRWST during outage to perform inspections or maintenance would be exposed to residual radioactivity from the fluids inside the tank during operation. This exposure would be controlled by the radiation protection program.

Based on the above discussion, the staff concludes that doses will be appropriately controlled in accordance with 10 CFR Part 20 and therefore meets 10 CFR 52.47(a)(5) requirements. In addition, Staff finds that the design permits appropriate inspection, is appropriately shielded and contained, and includes appropriate filtering systems, since the tank can be cleaned by the spent fuel pool purification system filters and demineralizers, and therefore meets GDC 61.

The staff also performed radiation dose analysis of the gamma doses inside the IRWST during normal operation and found that the dose rate and 60-year TID provided in the proposed updates to UFSAR Table 3D.5-2 for the qualification of equipment were appropriate and consistent with values calculated by the staff. The staff finds that the addition of the IRWST as a radiation source did not result in any EQ dose changes in Table 3D.5-2 to areas outside the IRWST, when credit is appropriately given for existing radiation shielding barriers. SNC indicated that the normal operation IRWST gamma dose values were also used for the equipment survivability for the hydrogen ignitors inside the IRWST. The staff determined that the normal operation dose should be the same for equipment in the equipment qualification and equipment survivability program. As such, the staff finds the doses to the equipment to be acceptable.

In the December 12, 2019, supplement letter SNC clarified that normal operation beta radiation need not be considered because the important components of the IRWST vents are metal and would not be sensitive to beta radiation, and the hydrogen igniters are contained within metal enclosures which shields the equipment from beta radiation under normal operation. Therefore, SNC concluded that beta radiation need not be considered during normal operation. Staff finds that, since beta radiation would not be expected to penetrate through the metal, and since there are conservatism in the gamma EQ dose assumptions, it is acceptable to not consider beta radiation in the normal operation beta radiation dose. Staff also finds that at the concentrations and energies of the radioactive materials in the vapor space, secondary radiation resulting from

the beta radiation would not be expected to be a significant contributor to the total dose to the equipment. Staff concludes that beta radiation would not be expected to be significant in the radioactive material in the water because of self-shielding by the water. As a result, the staff concludes that the equipment survivability doses during normal operation are appropriate.

The LAR presents analyses to demonstrate that when accounting for leakage into the IRWST from the PRHR HX that a situation will not be created in which the projected radiation doses will exceed the dose levels to which the in-situ equipment has been qualified when the shielding provided by the north-east wall and the west wall of the IRWST is considered. SNC stated that “the proposed change credits the north-east wall and the west wall of the IRWST as applicable radiation shielding walls. Without shielding credit applied to these walls based upon their design, radiation levels in Rooms 11300 and 11400 would exhibit significant increases. The downstream impacts of this would likely cause physical changes to equipment or qualification impacts...” The acceptability of these radiation dose analyses is evaluated above in this section.

For equipment inside the IRWST, SNC states that, “the only equipment affected by the addition of the IRWST to the Table 3D.5-2 60-year normal operating doses are the safety-related IRWST vents and the non-safety-related hydrogen igniters located in the IRWST and at the IRWST vents.” SNC also states that, “The IRWST vents and the hydrogen igniters located at the vents are qualified to the design basis 60-year normal total integrated dose (TID) of  $2.2 \times 10^7$  rads air. Dynamic effects are not changed by accounting for design basis leakage of the PRHR HX into the IRWST. The SSCs important to safety are qualified for the conditions anticipated in the IRWST due to the increased source term from design basis leakage from the PRHR HX into the IRWST concurrent with design basis fuel defects, thereby maintaining compliance with GDC 4 and 10 CFR 50.49.” Furthermore, SNC states that “there is no impact to EQ because such equipment is either not located where it would incur the estimated dose or is qualified for more severe doses (e.g., severe accident doses). Therefore, there is no impact to the capability of safety-related and important-to-safety equipment to perform their functions credited in reducing the probability, or mitigating the consequences, of an accident.” The staff finds that EQ is not impacted for the safety-related IRWST vents and non-safety-related hydrogen igniters since the equipment are qualified to the 60-year normal operating dose of  $2.2 \times 10^7$  rads air and that SNC continues to comply with GDC 4 and 10 CFR 50.49.

The staff reviewed the proposed changes to COL Appendix C, and corresponding plant-specific Tier 1, Table 3.3-1 to indicate the north-east wall and the west wall of the IRWST are applicable radiation shielding walls during normal operations as well as UFSAR Table 3D.5-2 to include the 60-year normal operating doses for the IRWST. The staff finds these changes acceptable since they document the equipment credited as well as the updated radiation dose.

The staff determined that there are no physical changes to the approved design, that the submitted analyses demonstrate that the already included SSCs are still shielded with similar margin from the maximum integrated doses calculated originally, and that the newly added SSCs within or above the IRWST (e.g., vents and hydrogen igniters) are also qualified to the total projected integrated lifetime radiation doses, the staff concludes that the design maintains compliance with GDC 4 and 10 CFR 50.49.

The staff concludes that the changes proposed in the LAR are in accordance with all applicable regulatory requirements, including 10 CFR 52.47(a)(5), 10 CFR 50.49, GDC 4, and GDC 61. As a result, the staff finds the changes proposed in LAR 19-003, with supplements, to be acceptable.

### 3.2 EVALUATION OF EXEMPTION

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. Exemptions from Tier 1 information are governed by the change process in Section VIII.A.4 of Appendix D of 10 CFR Part 52. Because the licensee has identified changes to plant-specific DCD Tier 1 information, with corresponding changes to the associated COL Appendix C information resulting in the need for a departure, an exemption from the certified design information within plant-specific Tier 1 material is required to implement the LAR.

The Tier 1 information for which a plant-specific departure and exemption was requested is described above. The result of this exemption would be that SNC could implement modifications to Tier 1 information in the PS-DCD and associated COL Appendix C. Pursuant to the provisions of 10 CFR 52.63(b)(1), an exemption from elements of the design as certified in the 10 CFR Part 52, Appendix D, design certification rule is requested for the involved Tier 1 information described and justified in LAR 19-003. This exemption is a permanent exemption limited in scope to the specific 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, Section VIII.A.4 of Appendix D to 10 CFR Part 52 provides that the Commission will deny a request for an exemption from Tier 1 if it finds that the requested change will result in a significant decrease in the level of safety otherwise provided by the design. Pursuant to 10 CFR 52.63(b)(1), the Commission may grant exemptions from one or more elements of the certification information, so long as the criteria given in 10 CFR 52.7, which, in turn, references 10 CFR 50.12, are met and the Commission must consider whether the special circumstances, which are defined by 10 CFR 50.12(a)(2), 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. As 10 CFR 52.7 further states, 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 the public health and safety, and are consistent with the common defense and security; and (2) special circumstances are present. Specifically, 10 CFR 50.12(a)(2) lists six special circumstances for which an exemption may be considered. It is necessary for one of these special circumstances to be present for the NRC to consider granting an exemption request. SNC stated that the requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii). That subparagraph 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.

#### 3.2.1 AUTHORIZED BY LAW

The requested exemption would allow SNC to implement the amendment described above. This exemption is a permanent exemption limited in scope to specific Tier 1 information. Subsequent changes to this plant-specific Tier 1 information, and corresponding changes to Appendix C, 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 and the requirements of 10 CFR 52.63(b)(1). As stated above, 10 CFR Part 52, Appendix D, Section VIII.A.4 allows the NRC to grant exemptions from one or more elements of the Tier 1 information. The staff has determined that granting of SNC's proposed exemption will not result in a violation of the AEA, as amended, or the Commission's regulations. Therefore, as required by 10 CFR 50.12(a)(1), the exemption is authorized by law.

### 3.2.2 NO UNDUE RISK TO PUBLIC HEALTH AND SAFETY

As discussed above in the technical evaluation, the proposed changes comply with the NRC's substantive safety regulations. Therefore, there is no undue risk to the public health and safety.

### 3.2.3 CONSISTENT WITH COMMON DEFENSE AND SECURITY

The proposed exemption would allow changes as described above in the technical evaluation, thereby departing from the AP1000 certified (Tier 1) design information. The change does not alter or impede the design, function, or operation of any plant structures, systems, or components 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 changes have 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.

### 3.2.4 SPECIAL CIRCUMSTANCES

Special circumstances, in accordance with 10 CFR 50.12(a)(2), are present, in part, 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 ITAAC information is to meet 10 CFR 52.97(b) by providing reasonable assurance that the facility has been constructed and will be operated in accordance with the pertinent requirements in the license, AEA, and NRC rules and regulations. The proposed changes described in the above technical evaluation do not impact the ability of any SSCs to perform their functions or negatively impact safety.

Special circumstances are present in the circumstances discussed in LAR 19-003, because the application of the specified Tier 1 ITAAC information is not necessary to achieve the underlying purpose of the rule. As discussed above, the change revises the COLs to incorporate the contribution of design basis PRHR HX leakage to the IRWST. This exemption request and associated revisions to the Tier 1 information and corresponding changes to Appendix C demonstrate that the applicable regulatory requirements will continue to be met. Therefore, for the above reasons, the staff finds that the special circumstances required by 10 CFR 50.12(a)(2)(ii) for the granting of an exemption from the Tier 1 information exist.

### 3.2.5 SPECIAL CIRCUMSTANCES OUTWEIGH REDUCED STANDARDIZATION

This exemption would allow the implementation of changes to Tier 1 information in the PS-DCD and corresponding changes to COL Appendix C that are being proposed in the LAR. The justification provided in LAR 19-003, the exemption request, and the associated licensing basis mark-ups demonstrate that there is a limited change from the standard information provided in



the generic AP1000 DCD. The design functions of the system associated with this request will continue to be maintained because the associated revisions to the Tier 1 information do not change or otherwise affect the design function of IRWST. Consequently, the safety impact that may result from any reduction in standardization is minimized, because the proposed change does not result in a reduction in the level of safety. Based on the foregoing reasons, in accordance with 10 CFR Part 52.63(b)(1), the staff finds that the special circumstances outweigh any decrease in safety that may result from the reduction of standardization of the AP1000 design.

### 3.2.6 NO SIGNIFICANT REDUCTION IN SAFETY

This exemption would allow the implementation of changes discussed above. The exemption request proposes to depart from the certified design by allowing changes discussed above in the technical evaluation. The changes will not impact the functional capabilities of this system. The proposed changes will not adversely affect the ability of the NI, annex building, turbine building, and radwaste building to perform its design functions, and the level of safety provided by the current systems and equipment therein is unchanged. Therefore, based on the foregoing reasons and 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.

### 3.3 SUMMARY

As discussed above, the staff concludes that the changes proposed in the LAR are in accordance with all applicable regulatory requirements, including 10 CFR 52.47(a)(5), 10 CFR 50.49, GDC 4, and GDC 61. As a result, the staff finds the changes proposed in the LAR to be acceptable.

### 4.0 STATE CONSULTATION

In accordance with the Commission's regulations in 10 CFR 50.91(b)(2), on December 30, 2019, the Georgia State official was notified of the proposed issuance of the amendment. The State official had no comments.

### 5.0 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 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, and there has been no public comment on such finding as published in the *Federal Register* on October 8, 2019 (84 FR 53777). 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 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the exemption.

## 6.0 CONCLUSION

The staff 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) presents special circumstances, and (5) does not reduce the level of safety at the licensee's facility. Therefore, the staff grants the licensee an exemption from the Tier 1 information requested by the licensee.

The staff has concluded, based on the considerations discussed in Section 3.1 that there is reasonable assurance that: (1) the health and safety of the public will not be endangered by the proposed changes, (2) the changes are 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. Therefore, the staff finds the changes proposed in this license amendment acceptable.

## 7.0 REFERENCES

1. Southern Nuclear Operating Company, Vogtle Electric Generating Plant Units 3 and 4, "Request for License Amendment and Exemption: Addition of in-Containment Refueling Water Storage Tank to Radiation analyses (LAR 19-003)," July 16, 2019 (ADAMS Accession No. ML19197A278).
2. Southern Nuclear Operating Company, Vogtle Electric Generating Plant Units 3 and 4, "Supplement to Request for License Amendment and Exemption: Addition of in-Containment Refueling Water Storage Tank to Radiation analyses (LAR 19-003S1)," October 3, 2019 (ADAMS Accession No. ML19276G437).
3. Southern Nuclear Operating Company, Vogtle Electric Generating Plant Units 3 and 4, "Supplement to Request for License Amendment and Exemption: Addition of in-Containment Refueling Water Storage Tank to Radiation analyses (LAR 19-003S2)," December 12, 2019 (ADAMS Accession No. ML19346E598).
4. Vogtle Electric Generating Plant Units 3 and 4, Updated Final Safety Analysis Report, Revision 7 and Tier 1, Revision 6, June 15, 2018 (ADAMS Accession No. ML18179A227).
5. AP1000 Design Control Document, Revision 19, June 13, 2011 (ADAMS Accession No. ML11171A500).
6. Combined License NPF-91 for Vogtle Electric Generating Plant Unit 3, Southern Nuclear Operating Company (ADAMS Accession No. ML14100A106).

7. Combined License NPF-92 for Vogtle Electric Generating Plant Unit 4, Southern Nuclear Operating Company (ADAMS Accession No. ML14100A135).