SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS

RELATED TO EXEMPTIONS AND AMENDMENT NOS. 107 AND 106

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, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT, UNITS 3 AND 4

DOCKET NOS. 52-025 AND 52-026

1.0 INTRODUCTION

By letter dated March 31, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17090A570), Southern Nuclear Operating Company (SNC) submitted a license amendment request (LAR) 17-010 and requested that the U.S. Nuclear Regulatory Commission (NRC) amend the combined licenses (COL) for Vogtle Electric Generating Plant (VEGP) Units 3 and 4, COL Numbers NPF-91 and NPF-92, respectively. The application proposed, for each of VEGP Units 3 and 4, revisions to COL License Conditions, COL plant-specific Tier 1 information and corresponding changes to COL Appendix C, and changes to the associated Tier 2 material incorporated into the VEGP Updated Final Safety Analysis Report (UFSAR), to address the need for mitigation of fire protection system (FPS) flooding of the Auxiliary Building identified during completion of the pipe rupture hazards analysis (PRHA). SNC also requests permanent exemptions, one for each unit, to allow departures from elements of the certified information in Tier 1 of the AP1000 certified Design Control Document (DCD) as specified in LAR-17-010. These exemptions are related to, and necessary for the granting of the amendments, which are being issued concurrently with these exemptions.

License Condition 2.D.(12)(b) requires that before commencing installation of individual piping segments and connected components in their final locations, SNC shall complete the as-designed PRHA for compartments (rooms) containing those segments in accordance with the criteria outlined in the AP1000 DCD, Revision 19, Sections 3.6.1.3.2 and 3.6.2.5. After completion of the PRHA for several of the Auxiliary Building compartments (rooms) SNC identified the need for revision to the description of the evaluation and results provided in the licensing basis documents. Some of the proposed changes to the UFSAR involve impacts to the COL Appendix C information (and the corresponding plant-specific Tier 1 information).

The proposed changes would revise the COLs to modify the design of the power plant by adding two floodup level sensors to the Auxiliary Building radiologically controlled area (RCA). The floodup level sensors provide the main control room (MCR) notification of a rise in water level that may indicate flooding in the Auxiliary Building. Changes are also proposed to the piping and procedures to limit the volume of FPS water that will be available for flooding the Auxiliary Building, and to provide other mitigating changes to limit the flooding on Levels 1 and 2 of the Auxiliary Building RCA. Changes are also proposed to address flooding of Levels 3, 4, and 5 of the Auxiliary Building RCA.

In a letter dated August 21, 2017 (ADAMS Accession No. ML17233A325), SNC submitted Supplement 1 to LAR-17-010, which, in Enclosure 5, provided SNC's responses to 5 of 7 staff requests for additional information (RAIs) and in Enclosure 6 corrected the original LAR by identifying one additional licensing basis document that was not mentioned in the LAR.

In a letter dated October 9, 2017, (ADAMS Accession No. ML17282A014), SNC submitted Supplement 2 to LAR-17-010, which, in Enclosure 7, provided SNC's responses to the balance of the RAIs not responded to in the August 21, 2017, letter and additional supplemental information in response to public discussions held on September 7, 2017, and a revision to Enclosure 1 from the original LAR.

In a letter dated November 1, 2017, (ADAMS Accession No. ML17305B507), SNC submitted Supplement 3 to LAR-17-010, which, in Enclosure 12, provided SNC's responses to a second staff RAI, and proposed changes to the licensing basis in Enclosure 13.

In a letter dated December 1, 2017, (ADAMS Accession No. ML17335A762), SNC submitted Supplement 4 to LAR-17-010, which, in Enclosure 15, provided SNC's responses to a third staff RAI and proposed associated revisions to licensing basis documents. Supplement 4 was updated and revised by a letter dated December 15, 2017, ((called Supplement 4, Revision 1) ADAMS Accession No. ML17349A928). This revision updated Enclosure 15 to the December 1, 2017, letter and included submitting an additional statement in response to Question 1 regarding moisture incursion in the waste gas system or associated control system. The update to the UFSAR markup in Enclosure 16 added a paragraph that, in general, identified the electrical classification thereof and that the two water level sensors provide input to the Protection and Safety Monitoring (PMS) subsystem.

In a letter dated January 3, 2018, (ADAMS Accession No. ML18003B082), SNC submitted Supplement 5 to LAR-17-010, which, in Enclosure 17, provided SNC's responses to a fourth staff RAI regarding loads on the safety-related Class 1E batteries.

In order to modify the UFSAR containing the associated Tier 2 information, the NRC must find SNC's proposed changes in the LAR acceptable. The staff's safety evaluation (SE) of the LAR is presented in this report. The licensee has also requested permanent exemptions from the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, Appendix D, "Design Certification Rule for the AP1000 Design," Section III.B, "Scope and Contents," to allow departures from Tier 1 of the certified generic AP1000 DCD[1]. The staff's review of the exemption requests is also included in this SE.

By the letters previously discussed, SNC submitted supplements and revisions to the original LAR. The submitted supplements and revisions did not expand the scope of the original LAR and did not change the staff's original proposed no significant hazards consideration determination published in the *Federal Register* on June 6, 2017 (82 FR 26123).

2.0 REGULATORY BASIS

The LAR and exemptions concern changes to plant-specific Tier 1 information, corresponding changes to COL Appendix C, and changes to the plant-specific DCD Tier 2 material incorporated into the VEGP UFSAR, by revising the COLs to modify the design of the power plant by adding two floodup level sensors to the Auxiliary Building RCA. The staff considered the following regulatory requirements in reviewing SNC's proposed LAR and exemptions:

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

10 CFR Part 52, Appendix D, Section VIII.B.5.a allows an applicant or licensee who references this appendix to depart from Tier 2 information without prior NRC approval unless the proposed departure involves a change to or departure from Tier 1 information, Tier 2* information, or the Technical Specifications, or requires a license amendment under paragraphs B.5.b or B.5.c of this section.

10 CFR 52.63(b)(1) allows a licensee who references a design certification rule to request NRC's approval for an exemption from one or more elements of the certification information. The Commission may grant such a request only if it determines that the exemption will comply with the requirements of 10 CFR 52.7, "Specific Exemptions," which in turn points to the requirements listed in 10 CFR 50.12 for specific exemptions. In addition to the factors listed in § 52.7, the Commission shall consider whether the special circumstances that § 52.7 requires to be present outweigh any decrease in safety due to 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), 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 inspections, tests, analyses and acceptance criteria (ITAAC) contained in the license is a proposed amendment to the license. Appendix C of COLs NPF-91 and NPF-92 contain information that SNC is proposing to modify. Therefore, the proposed changes require a license amendment.

10 CFR 20.1101(b) requires that SNC shall use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational

[1] While SNC described the requested exemptions as being from Section III.B of 10 CFR Part 52, Appendix D, the entirety of the exemptions pertain to proposed departures from Tier 1 information in the generic AP1000 DCD. In the remainder of this evaluation, the NRC will refer to the exemptions as exemptions 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.

doses and doses to members of the public that are as low as is reasonably achievable "ALARA."

10 CFR 20.1406 requires that applicants shall describe in the application how facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

10 CFR 50.34(f)(2)(iii) requires that the preliminary design of a facility must contain Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.

10 CFR 50.34(f)(2)(vii) requires that applicants perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment.

10 CFR 50.34(f)(2)(viii) requires that applicants provide a capability to promptly obtain and analyze samples from the reactor coolant system (RCS) and containment that may contain accident source term radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities.

10 CFR 50.48, requires a fire protection plan that satisfies the specific criteria outlined in this subsection.

10 CFR 50.49, requires, in part, that each combined license holder, issued under part 52 of this chapter, other than a nuclear power plant for which the certifications required under § 50.82(a)(1) or § 52.110(a)(1) of this chapter have been submitted, shall establish a program for qualifying the electric equipment defined in paragraph (b) of this section. 10 CFR 50.49(e)(6) requires that submergence must be included in the equipment qualification program

10 CFR 50.55a(h), requires, in part, that protection systems must be consistent with their licensing basis or may meet the requirements of Institute of Electrical and Electronic Engineers (IEEE) Standard (Std.) 603–1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995, and for safety systems, issued under a COL under part 52 of this chapter, must meet the requirements for safety systems in IEEE Std. 603–1991 and the correction sheet dated January 30, 1995.

10 CFR Part 50.55a, requires, in subsection (h)(2) Safety Systems, in part, that COLs under Part 52 of this chapter must meet the requirements for safety systems in the IEEE Std. 603–1991 and the correction sheet dated January 30, 1995.

10 CFR 52.79(a)(10), requires the list of electric equipment important to safety that is required by 10 CFR 50.49(d).

10 CFR 52.80(a), which require that a COL application contain the proposed inspections, tests, and analyses, that SNC shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in

conformity with the COL, the provisions of the Atomic Energy Act, and the Commission's rules and regulations.

The 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 (SSC) important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

GDC 2, "Design Bases for Protection against Natural Phenomena," require that SSCs 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.

GDC 3, "Fire Protection," requires and specifies, in part, that (1) SSCs important to safety must be designed and located to minimize the probability and effects of fires and explosions, (2) noncombustible and heat-resistant materials must be used wherever practical, and (3) fire detection and suppression systems must be provided to minimize the adverse effects of fires on SSCs important to safety.

GDC 4, "Environmental and Dynamic Effects Design Bases," requires that SSCs 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.

GDC 17, "Electric Power Systems," requires that an onsite and offsite electric power system shall be provided to permit functioning of SSCs ts important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

GDC 21, "Protection system reliability and testability," requires that the protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

GDC 22, "Protection system independence," requires that the protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

GDC 23, “Protection system failure modes,” requires that the protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

GDC 24, “Separation of protection and control systems,” requires that the protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

GDC 60, “Control of Releases of Radioactive Materials to the Environment,” requires that the nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

GDC 61, “Fuel Storage and Handling and Radioactivity Control,” requires that the fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal (RHR) capability having reliability and testability that reflects the importance to safety of decay heat and other RHR, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

3.0 TECHNICAL EVALUATION

3.1 EVALUATION OF EXEMPTIONS

Section VIII.A.4 of Appendix D to 10 CFR Part 52 requires a licensee to obtain an exemption to depart from the Tier 1 information of the generic AP1000 DCD. Because SNC has identified changes to plant-specific Tier 1 information, with corresponding changes to the associated COL Appendix C information, resulting in the need for a departure, exemptions from the certified design information within plant-specific Tier 1 material is required under 10 CFR 52.63(b)(1) to implement the LAR. The end result of these exemptions would be that SNC can implement modifications to Tier 1 information described and justified in LAR-17-010 if and only if the NRC approves LAR-17-010. These are permanent exemptions limited in scope to the particular Tier 1 information specified. As defined in Section II of Appendix D to 10 CFR Part 52, Tier 1 information includes ITAAC and design descriptions, among other things. Therefore, a licensee referencing Appendix D incorporates by reference Tier 1 information contained in the generic AP1000 DCD. The Tier 1 ITAAC and design descriptions, along with the plant-specific ITAAC, were included in Appendix C of the VEGP COL at its issuance.

As stated in Section VIII.A.4 of Appendix D to 10 CFR Part 52, exemptions from Tier 1 information are 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 requests for exemptions to permit deviations from Certified Tier 1 information 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, is met and that the special circumstances, 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 states, the Commission's consideration will be governed by 10 CFR 50.12, "Specific exemptions." 10 CFR 50.12 states that exemptions 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 exemptions may be considered. It is necessary for one of these special circumstances to be present in order for the NRC to consider granting requests for exemptions.

The licensee stated that the requested exemptions met 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 the exemption requests is presented below.

3.1.1 Authorized by Law

These exemptions would allow SNC to implement a revision to Tier 1 of the plant-specific DCD, specifically related to the changes described in LAR-17-010 and the exemption requests in Enclosure 2 to LAR-17-010. These exemptions are permanent exemptions limited in scope to particular Tier 1 information. Subsequent changes to any 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). Based on the review of LAR-17-010, the staff has determined that granting of SNC's proposed exemptions will not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations. Therefore, in accordance with 10 CFR 50.12(a)(1), the exemptions are authorized by law.

3.1.2 No Undue Risk to Public Health and Safety

The underlying purpose of Appendix D to 10 CFR 52 is to ensure that a licensee will construct and operate the plant based on the approved information found in the DCD incorporated by reference into a licensee's licensing basis. The changes proposed in the LAR will not negatively impact any design function. The changes to plants systems is limited to the FPS and components, the new floodup water level sensors and associated equipment, and the revised identification of flood barrier walls, are necessary to continue to maintain the level of necessary flood protection afforded in the certified design. There are no other changes to plant systems or the response of systems to postulated accident conditions. Furthermore, 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. The changes described do not introduce any new industrial, chemical, or radiological hazards that would represent a public health or safety risk, nor do they modify or remove any design or operational controls or safeguards intended to mitigate any existing on-site hazards. There is no detriment to the predicted radioactive releases due to postulated accident conditions. 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. Therefore, in accordance with 10 CFR 50.12(a)(1), the granting of the exemptions will not present undue risk to the public health and safety.

3.1.3 Consistent with Common Defense and Security

The proposed exemptions would allow changes to elements of the plant-specific Tier 1 DCD. These proposed exemptions would be permanent exemptions limited in scope to particular Tier 1 Table information. Any changes to Tier 1 information would be subject to the exemption process in Section VIII.A.4 of Appendix D to 10 CFR Part 52. The change does not alter or impede the design, function, or operation of any plant 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 changes have no impact on plant security or safeguards. Therefore, in accordance with 10 CFR 50.12(a)(1), the staff finds that the exemptions are consistent with the common defense and security.

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 rule under consideration in these exemption requests is 10 CFR Part 52, Appendix D, Section III.B., which requires that a licensee referencing the AP1000 Design Certification Rule in 10 CFR Appendix D shall incorporate by reference and comply with Appendix D, including Tier 1 information. The underlying purpose of the Tier 1 information is to ensure that a licensee will safely construct and operate a plant based on the certified information found in the AP1000 DCD, which was incorporated by reference into the VEGP's licensing bases. The proposed changes in Tier 1 of the plant-specific DCD would change the information related to the design of the power plant by changing the FPS and components, adding new floodup water level sensors and associated equipment, and the revising the identification of flood barrier walls. These changes will enable SNC to safely construct and operate the AP1000 facility consistent with the design certified by the NRC, by updating information related to the two floodup level sensors in the Auxiliary Building RCA in Tier 1 of the plant-specific DCD.

Special circumstances are present in the particular circumstances discussed in LAR-17-010 because the application of the specified Tier 1 information does not serve the underlying purpose of the rule. More specifically, the underlying purpose of this Tier 1 information is to provide system configurations that are acceptable to safely construct and operate the plant. The changes in Tier 1 of the plant-specific DCD would change the information related to the design of the power plant by changing the FPS and components, adding new floodup water level sensors and associated equipment, and the revising the identification of flood barrier walls, and related changes. Therefore, staff concludes these proposed changes serve the underlying purpose of the rule. These exemption requests and the associated proposed revisions to Tier 1 demonstrate that the applicable regulatory requirements will continue to be met. Based on the foregoing reasons, the staff finds that the special circumstances required by 10 CFR 50.12(a)(2)(ii) for the granting of exemptions from the Tier 1 information exist.

3.1.5 Special Circumstances Outweigh Reduced Standardization

Under 10 CFR 52.63(b)(1) "[i]n addition to the factors listed in § 52.7, the Commission shall consider whether the special circumstances that § 52.7 requires to be present outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption." These exemptions would allow the implementation of changes to Tier 1 in the plant-specific DCD as proposed in the LAR so that the design functions of the system associated with this request are consistent with the current design of the plant in supporting the actual system functions, even in off-normal and postulated accident conditions. Specifically, the proposed changes in Tier 1 of the plant-specific DCD would change the information related to the design of the power plant by changing the FPS and components, adding new floodup water level sensors and associated equipment, and the revising the identification of flood barrier walls. These exemptions from the certification information will enable SNC to safely construct and operate the AP1000 facility consistent with the levels of safety afforded in the design certified by the NRC in 10 CFR Part 52, Appendix D. Consequently, any decrease in safety impact that may result from any reduction in standardization caused by the exemptions is minimized, because the changes ensure that the normal system functions are maintained during various modes of operation and off-normal and postulated accident conditions. In addition, the design functions of the systems associated with this request will be maintained. Based on the foregoing reasons, 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

These exemptions would allow the implementation of changes to Tier 1 in the plant-specific DCD as proposed in the LAR-17-010. The exemption requests propose to depart from the certified design by making departures in the plant-specific DCD from the generic AP1000 DCD, and the changes ensure that the system functions are maintained during various modes of operation. In this case, the flooding of the Auxiliary Building from a moderate energy line break would not affect the functioning of safety-related equipment or safe-shutdown capability. The proposed changes do not involve or interface with any SSC, accident initiator, or initiating sequence of events related to the accidents evaluated, and therefore do not have an adverse effect on any SSC's design function. The proposed changes would not adversely affect the ability of the Auxiliary Building or safety-related and safe-shutdown equipment located in the Auxiliary Building, to perform its design functions, and the level of safety provided by the current systems and equipment would be 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 exemptions would not result in a significant decrease in the level of safety otherwise provided by the design.

3.2 TECHNICAL EVALUATION OF PROPOSED CHANGES

By letter dated March 31, 2017 (ADAMS Accession No. ML17090A570), SNC requested NRC approval of LAR-17-010, for the VEGP Units 3 and 4, COL Numbers NPF-91 and NPF-92, respectively. By letters dated August 21, October 9, November 1, December 1, and December 15, 2017, and January 3, 2018, SNC submitted Supplement 1 (ADAMS Accession No. ML17233A325), Supplement 2 (ADAMS Accession No. ML17282A014), Supplement 3 (ADAMS Accession No. ML17305B507), Supplement 4 (ADAMS Accession No. ML17335A762), Supplement 4, Revision 1 (ADAMS Accession No. ML17349A928), and Supplement 5 (ADAMS

Accession No. ML18003B082), respectively, SNC supplemented and refined their requested LAR-17-010.

License Condition 2.D.(12)(b) requires that before commencing installation of individual piping segments and connected components in their final locations, SNC shall complete the as-designed PRHA for rooms containing those segments in accordance with the criteria outlined in the AP1000 DCD, Revision 19, Sections 3.6.1.3.2 and 3.6.2.5. After completion of the PRHA for several of the Auxiliary Building rooms SNC identified the need for revision to the description of the evaluation and results provided in the licensing basis documents.

The changes proposed in the LAR revise the COLs to modify the design of the power plant by adding two floodup level sensors to the Auxiliary Building RCA. Changes are also made to the piping and procedures to limit the volume of FPS water that can be available for flooding the Auxiliary Building, and to provide other mitigating changes to limit the flooding on the Auxiliary Building RCA.

The PRHA of final piping routing has determined that the limiting moderate-energy line breaks (MELBs) within the Auxiliary Building RCA have been redefined. The FPS moderate-energy lines have been determined to provide flooding of the Auxiliary Building RCA that bounds flooding from high energy (chemical and volume control system) line breaks and other MELBs. For the new bounding break, the fire water storage tanks are the source of the flood water to the Auxiliary Building RCA. In order to limit the flood level within the Auxiliary Building RCA, changes are made to the size and lineup of the fire water storage tanks and the location of the motor-driven fire water pump. These changes affect the COL Appendix C (and plant-specific Tier 1) fire protection ITAAC information. New level instrumentation (floodup sensors) is added with indication in the MCR to alert the MCR operators of an Auxiliary Building RCA flood condition. These floodup sensors are added to the Liquid Radwaste System (WLS) ITAAC. In addition, and a seismic Category 1 safety-related flood relief louver is installed in the wall next to stairwell S04 in the RCA. Also, a 1.5 square foot (ft^2) flooding relief flapper is being installed at the bottom of the door that opens into Room 12362 in the Auxiliary Building RCA. The revised flooding locations affect the ITAAC that identify the flood boundaries within the Nuclear Island (NI). This section presents the staff's SE of various aspects of the proposed changes included in the LAR in the following subsections.

3.2.1 Break Selection

The staff reviewed the LAR to verify that SNC correctly identified the most limiting pipe break locations. In Enclosure 1, "License Amendment Request: Pipe Rupture Hazard and Flooding Analyses," of LAR-17-010, SNC described its PRHA and flooding analyses performed for various levels of the Auxiliary Building RCA and the non-RCA. The licensee stated that each Auxiliary Building plant area containing safety-related systems or equipment is reviewed to determine the postulated fluid system failures which would result in the most adverse internal flooding conditions. The licensee stated that completion of the PRHA for several of the Auxiliary Building rooms has resulted in the need for revision to the description and results in the flood protection analyses. The staff reviewed the LAR to verify that SNC appropriately identified the most limiting pipe break locations for flood protection of the Auxiliary Building. The staff's review of SNC's criteria for the PRHA postulated pipe break/crack locations and for the flood protection analyses to determine the most limiting break locations which would result in the most adverse internal flooding conditions in the Auxiliary Building are described below.

3.2.1.1 PRHA Criteria for Postulating Pipe Break/Crack Locations

In LAR-17-010, there is no change proposed to the PRHA criteria for defining the high- and moderate-energy piping and for postulating the respective pipe break/crack locations and types (i.e., break or crack). To support the review of the LAR, the staff planned a regulatory audit of supporting documentation as outlined in an audit plan (ADAMS Accession No. ML17156A426). In the regulatory audit staff reviewed four non-docketed reports (APP-GW-GLR-075, Revision 3, “PRHA Summary Report for Auxiliary Building,” APP-GW-N1-007, Revision 5, “AP1000 Design Criteria for Protection from Flooding,” APP-GW-POC-005, Revision 3, “AP1000 Break Locations for Pipe Rupture Hazard Analyses,” and APP-GW-POC-170, Revision 2, “Identification of Crack Exclusion Zones and Piping Without Stress-Based Cracks within the AP1000 Auxiliary Building”) provided by SNC. The staff reviewed the LAR and audited the four non-docketed reports to verify that the criteria used in the LAR for defining the high- and moderate-energy piping and the associated postulated pipe breaks/cracks locations and types were in accordance with the reviewed and approved UFSAR criteria.

3.2.1.2 Staff Evaluation

Based on the review of the information provided in the above documents, the staff determined that the SNC’s criteria used in LAR-17-010 for defining high- and moderate-energy piping are consistent with the reviewed and approved UFSAR criteria. The staff also determined that in LAR-17-010, SNC appropriately considered breaks in high-energy piping for environmental and flooding analysis, except for piping which has been shown to meet the break exclusion criteria. For moderate-energy seismic Category I or seismic Category II piping, through-wall cracks are postulated, except for piping which has been shown to meet the crack exclusion criteria. In addition, for evaluating consequences of flooding, SNC appropriately considered full circumferential pipe breaks in non-seismically supported moderate-energy piping (e.g., fire-protection system moderate-energy lines). The staff found that the PRHA criteria used in the LAR for determining the postulated pipe breaks/cracks locations and types for high- and moderate-energy piping as described above are in accordance with the pertinent reviewed and approved criteria and are therefore acceptable.

3.2.2 Internal Flood Protection

In LAR-17-010, Enclosure 1, SNC described its internal flooding analysis performed for the various levels of the Auxiliary Building. The licensee states that the limiting MELBs within the Auxiliary Building RCA have been redefined for the flooding protection. The FPS moderate-energy lines have been determined to be the limiting case flooding in the Auxiliary Building. The staff reviewed the LAR to verify that SNC appropriately identified the most limiting pipe break for internal flood protection in the Auxiliary Building and to ensure that all safety-related SSCs in the Auxiliary Building are adequately protected against internal flooding.

The relevant regulatory guidance for this area of review and the associated acceptance criteria are given in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” (SRP) Section 3.4.1, “Internal Flood Protection for Onsite Equipment Failures,” Revision 3. The applicable regulatory requirements are 10 CFR Part 50, Appendix A, GDC 2 and GDC 4. The requirements of GDC 4 are met if SSCs important to safety are designed to accommodate the flooding of discharged fluid resulting from high and moderate energy line cracks that are postulated in SRP Sections 3.6.1 and 3.6.2. The requirements of GDC 2 are met if SSCs important to safety are designed to accommodate the flooding of discharged fluid resulting from non-seismic pipe breaks under seismic events.

3.2.2.1 Limiting Case for Flooding within the Auxiliary Building from Pipe Failures

Section 3.4.1.2.2.2 of the UFSAR describes the flooding of the Auxiliary Building RCA, and identified a normal RHR suction line break as the limiting break. A subsequent PRHA flooding analysis performed by SNC has determined that the limiting MELB within the Auxiliary Building RCA to be FPS pipe break, which bounds flooding from HELBs and other MELBs. Therefore, PRHA has redefined the limiting pipe failure in the Auxiliary Building RCA.

The proposed revision to Section 3.4.1.2.2.2 of the UFSAR states that the RCA of the Auxiliary Building is subject to flooding from a variety of potential sources of pipe breaks including the component cooling water, central chilled water, hot water, spent fuel pool cooling, normal RHR, FPS, and chemical and volume control systems, as well as various tanks. The staff noted that changes to the FPS are newly identified as a result of the PRHA, and that all the other systems were previously considered in the certified design.

3.2.2.1.1 Staff Evaluation

In Section 3.2.1.1 of this SE, entitled "PRHA Criteria for Postulating Pipe Break/Crack Locations," the staff found that the PRHA criteria used in the LAR for determining the postulated pipe breaks/cracks locations and types for high- and moderate-energy piping are in accordance with the pertinent reviewed and approved UFSAR criteria, and are acceptable.

In addition, SNC's PRHA flooding analysis identified that a flood level of 19 feet from the rupture of the FPS piping is worse than the flood level of 12 inches from the previously identified limiting break of the normal RHR suction line. The staff finds the change to include the FPS acceptable because it is consistent with the guidance in SRP Section 3.4.1 which indicates that FPS pipe ruptures should be considered in the spectrum of pipe breaks for the flooding analysis. The guidance is based on the requirements of GDC 2 that the failure of a non-seismically designed FPS in a seismic event should be considered for the flooding analysis. The limiting case of 19 feet flood level for the spectrum of pipe breaks including FPS is determined by the PRHA flooding analysis in terms of water volumes, source of water, and plant layout that will be evaluated in the following sections.

Based on above, the staff determined that SNC's flooding analysis adequately considered all the possible sources of internal flooding in the Auxiliary Building because, in determining the limiting flooding source, SNC has considered the spectrum of breaks and cracks including FPS as discussed in SRP, Section 3.4.1, Paragraph III.3, for all the compartments in the Auxiliary Building.

3.2.2.2 Source of Water for Internal Flooding Analysis

The staff found the following relevant information in LAR-17-010 regarding the source of water for the flood protection analysis:

- PRHA and flooding analysis were performed for Levels 1 through 5, in both the RCA and non-RCA of the Auxiliary Building. Each area of the plant containing safety-related systems or equipment is reviewed to determine the postulated fluid system failures which would result in the most adverse internal flooding conditions.

- For the limiting case, the flooding source from MELB in a 10-inch fire water line of FPS in the Auxiliary Building RCA is assumed to be one whole fire water storage tank (525,000 gallons), which results in a maximum flooding level of 19 feet in the RCA of Auxiliary Building.

- This volume of 525,000 gallons is to limit the maximum water volume for the flooding analysis. Each of the two fire water storage tanks has 504,000 gallons, and a 21,000 gallons automatic makeup flow (resulting from a new 2-inch 50 gpm reduced flow makeup line) over a 7 hour period. The operator action to terminate the FPS flow within 7 hours is credited for this event.

3.2.2.2.1 Staff Evaluation

Based on above, the staff determined that SNC's flooding analysis adequately considered all the possible sources of internal flooding in the Auxiliary Building because, in determining the limiting flooding source, SNC has considered the spectrum of breaks and cracks including FPS for all the compartments in all five levels of the Auxiliary Building and all the important parameters such as break flows, crack flows, effective flood area for each compartment, line isolation time, and tank volumes. The source of water from the FPS is limited by the design changes on tank size, piping realignment, and isolation by operator action. Therefore, the staff finds SNC's flooding analysis in identifying sources for internal flooding acceptable because the analysis is consistent with the guidance in SRP Section 3.4.1 Paragraph III.3 for the assessment of potential flooding.

3.2.2.3 Flood Level Determination for Auxiliary Building

Based on the review of the applicant's flooding analysis in the enclosures of LAR-17-010 and a regulatory audit of calculations (ADAMS Accession No. ML17156A426), the staff found the following information regarding the flood level determination.

- Changes are made to the piping and procedures to limit the volume of FPS water that can be available for flooding the Auxiliary Building, and to provide other mitigating changes to limit the flooding on Levels 1 and 2 of the Auxiliary Building RCA.

- Doors, other than watertight doors, are assumed to be closed for individual room equilibrium flood heights and then opened for adjacent room bounding flood height determinations. Compartment floor penetrations such as gratings, pipe chases, stairways, and open hatches are assumed to be unclogged for flooding analysis.

- UFSAR Subsection 3.4.1.2.2.2 describes the flooding assessment for the Auxiliary Building RCA at Level 3 and Level 4 and specifies that the potential sources of flooding drain directly to Level 1. The UFSAR also describes the flooding assessment for the Auxiliary Building RCA at Level 5 and specifies that accumulation of water in this area is prevented by floor drains and by flow to the stairwell and elevator shafts, which drain to Level 1. The flooding levels on Level 1 caused by these breaks are either bounded by or, in the case of the Level 3 break, result in a maximum flooding of 19 feet on Level 1.

- LAR-17-010 provides as-designed PRHA including flood protection analyses. The FPS moderate-energy lines has been determined to provide flooding of the Auxiliary Building that bounds all pipe breaks and pipe cracks. Enclosure 1 of LAR-17-010 provides the information on the flooding analysis. The amount of flooding water is limited to be 520,000 gallons, and the flood level in the Auxiliary Building RCA is determined to be 19 feet.

3.2.2.3.1 Staff Evaluation

The staff reviewed the above information regarding SNC's flooding analysis in accordance with SRP, Section 3.4.1, Paragraph III.2, which provides the guidance regarding the evaluation of the flooding analysis using plant arrangement, layout drawings, and other methods. The staff reviewed the information in the enclosures of LAR-17-010 and audited the calculations as described in the audit plan (ADAMS Accession No. ML17156A426) on the methodology, assumptions, plant layout, flood barriers, flow path, and calculated results of the flood levels. The staff found sufficient details in the LAR and calculations. For each pipe break, the flow paths and floodable volumes of the affected rooms, and flood levels are provided in the calculations. All the important parameters such as break flows, crack flows, effective flood area for each compartment line isolation time, and tank volumes are described in the calculations. The analysis method and assumptions are consistent with the guidance in SRP, Section 3.4.1.

Tier 1, Table 3.3-2, "Nuclear Island Building Room Boundaries Required to Have Flood Barrier Floors and Walls," identifies rooms and corresponding flood levels that need flood barrier floors and walls. Tier 2, Table 3D.5-4, identifies rooms and corresponding flood levels that need consideration for equipment qualification of submergence. The flood levels for other rooms that do not have impact on the walls or on the safety-related equipment may not be listed in the LAR. The revised UFSAR Tier 2, Table 3.11-1, "Environmentally Qualified Electrical and Mechanical Equipment," specifies that submergence testing or operation with spray, as applicable, is required for these safety-related valves and associated subcomponents. In a subsequent submittal of LAR-17-010S2, Enclosure 11 (ADAMS Accession No. ML1728282A013) provides Table 2, "Flood Heights for Auxiliary Building RCA Rooms Affected by Flooding," that addresses additional information of the flood levels for all the rooms being affected by the flooding in the Auxiliary Building RCA. The staff finds the information adequate and therefore acceptable.

Based on above, the staff found SNC's flooding analysis on the flood level determination and limiting flood level of 19 feet to be acceptable because the analysis, as detailed in the audited document, is consistent with the guidance in SRP Section 3.4.1, Paragraph III.2, regarding flooding analysis using the plant arrangement, layout drawings and other acceptable methods.

3.2.2.4 Changes in the Design and Procedures

Due to the reevaluation of the limiting case flood level, there are changes in the system design, pipe routing, pump relocation, procedures, and consequences of the flooding. These changes are listed in bullets and evaluated below:

- For the new bounding break, the fire water storage tanks provide the flooding water to the Auxiliary Building RCA. In order to limit the flooding level within the Auxiliary Building RCA, changes are made to the size and lineup of the fire water storage tanks and the location of the motor-driven fire water pump.

Changes in the FPS re-routed piping and relocated electric motor-driven fire pump and jockey pumps are evaluated in Section 3.2.8 of this SE.

- The volume of two fire water tanks are modified to 525,000 gallons to limit the maximum flooding level to 19 feet. The cross-connect valves are proposed to be changed so that the position of the cross-connect valves between the two fire water storage tanks is procedurally controlled locked-open and the supply line valve to the respective pump suction from the secondary fire water storage tank (the out-of-service tank) is changed to procedurally controlled locked-closed. This allows only one water tank to be connected to the fire water header at any time and limits the volume of water that is available for flooding of the Auxiliary Building RCA.

Because the cross-connect line is no longer an open line, the check valves on the outlets of the fire water storage tanks are no longer needed. A new 2-inch makeup path, limited to 50 gpm, is added to automatically makeup to the fire water storage tank, providing 21,000 gallons of water over 7 hours for flooding water. Operator action is required to isolate the FPS fire water storage tank makeup within a period of 7 hours.

- Two floodup level sensors to the Auxiliary Building RCA are added. A safety-related display in the MCR provides indication of the flooding situation. These sensors are safety-related and Class 1E because they are connected to the PMS system. These level sensors are added to the WLS ITAAC.

- A seismic Category 1 safety-related flood relief louver is installed in the wall next to stairwell S04 at Elevation 66'-6." Also, a 1.5 square foot (ft^2) flooding relief flapper is being installed at the bottom of the door that opens into Room 12362 in the Auxiliary Building RCA.

3.2.2.4.1 <u>Staff Evaluation</u>

The staff finds that the design modifications of the volume of fire water tanks, procedure controlled cross-connect valves, the 2-inch makeup path, and operator actions to isolate the flow path within 7 hours are acceptable, because these changes limit the source of water for the limiting case flooding, and are consistent with SRP Section 3.4.1. Additional evaluation regarding the operator actions is provided in Section 3.2.5 below.

The added level sensors are acceptable because these sensors (which are safety-related, Class 1E and included in ITAAC) can provide the MCR notification of a rise in water level that indicates flooding in the Auxiliary Building so that proper operator actions can be taken, and the Class 1E design of these sensors are consistent with SRP Section 3.4.1 paragraph III. (7). The staff finds that the design modifications of the volume of fire water tanks, procedure controlled cross-connect valves, the 2-inch makeup path, and operator actions to isolate the flow path within 7 hours are acceptable because these changes limit the water volume of the limiting case flooding.

The installed seismic Category 1 safety-related flood relief louver and relief flapper are acceptable because these changes are necessary to limit the flood level to 19 feet in the RCA stairwell S04 area, and the seismic category designation is consistent with SRP Section 3.4.1.

3.2.2.5 <u>Protection of the Safety-Related Equipment and the Adequacy of Structures</u>

In Section 3.2.7.2 of this SE, entitled "Environmental Qualification of Safety-Related Valves due to Submergence or Spray," the staff evaluates the provisions for the environmental qualification of safety-related valves and associated valve subcomponents that are submerged or exposed to spray as a result of the PRHA as specified in the proposed revision to UFSAR Tier 2, Table 3.11-1. In Section 3.2.3 of this SE, the staff evaluates the provisions for environmental qualification of safety-related digital instrumentation and control system. In Section 3.2.4 of this SE, the staff reviewed the structure adequacy of the changes proposed in the LAR.

3.2.2.5.1 Staff Evaluation

Based on the review in Sections 3.2.3 and 3.2.4 of this SE, the staff finds that SNC has adequately addressed the protection of the safety-related equipment resulting from the PRHA and the adequacy of the structure design, because the equipment required for safe shutdown of the plant are not affected by the postulated flooding of the Auxiliary Building.

3.2.2.6 Proposed Changes to Licensing Basis Documents

Enclosure 3 of LAR-17-010 summarizes the proposed changes to licensing basis documents, which includes the following:

- Combined License Changes,
- COL Appendix C and Plant-Specific Tier 1 Changes, and
- Plant-Specific Tier 2 Changes.

The staff reviewed the above documents relating to the flood protection in the Auxiliary Building and finds the changes in the licensing documentation consistent with the design and analysis changes that have been reviewed earlier and, therefore, acceptable.

3.2.2.6.1 Staff Evaluation

The staff reviewed SNC's analysis provided in LAR-17-010 and LAR-17-010S1, LAR-17-010S2, and LAR-17-010S3 and finds that the PRHA criteria used in the LAR for determining the postulated pipe breaks/cracks locations and types for high-energy and moderate-energy piping are in accordance with reviewed and approved UFSAR criteria and are therefore acceptable and the provisions for the environmental qualification safety-related valves and associated valve subcomponents that are submerged or exposed to spray as a result of the PRHA, are acceptable, because applicable environmental qualification testing is specified in the proposed revision to UFSAR Tier 2, Table 3.11-1. Based on these findings, the staff concludes that there is reasonable assurance that the requirements of 10 CFR Part 50, Appendix A, GDC 4, will continue to be met. Therefore, the staff finds the proposed change acceptable.

The licensee's internal flooding analysis shows that safety-related SSCs are not prevented from performing their required safe shutdown functions due to effects of the postulated failures. In addition, the analysis identifies the protection features that mitigate the consequences of flooding in areas that contain safety-related equipment. The staff's review, as discussed above, concludes that SNC has adequately discussed the proposed changes, the consequences, and documentation changes, resulting from the newly identified FPS pipe break resulting from PRHA. Therefore, the staff concludes that LAR-17-010 is acceptable because the changes to the design, and associated flooding analysis and equipment protection are consistent with the guidance in SRP Section 3.4.1.

3.2.3 Instrumentation and Controls

LAR-17-010 proposes adding two new safety-related level sensors and accompanying equipment to support an updated PRHA and flooding analysis that resulted in higher water levels being postulated in the event of a pipe rupture or flooding event in the 66'-6" level of the Auxiliary Building.

The newly proposed level sensors (WLS-400A and WLS-400B) and associated equipment (level transmitters WLS-JE-LT-400A and WLS-JE-LT-400B and related signal cabling) interface with the AP1000 safety-related digital instrumentation and controls system, the PMS. The PMS provides detection of off-nominal conditions and, when system conditions require, actuates the appropriate safety-related components and/or functions necessary to place and maintain the plant in a safe shutdown condition. Per WCAP-16675-P, "AP1000 Protection and Safety Monitoring System Architecture Technical Report," Revision 5, a secondary reference to Chapter 7 of UFSAR, in Chapter 1, AP1000 Functional Requirements, it states, in part, "The Protection and Safety Monitoring System performs the reactor trip (RT) functions, the engineered safety features (ESF) actuation and the Qualified Data Processing System[2] (QDPS) functions." Additionally, in Section 1.4, "Component Control Functions," it states, in part, "Control of individual safety-related components that perform Class 1E functions is provided."

Based upon information contained in the original LAR and Enclosure 5 of LAR-17-010, the proposed sensors are classified as Class 1E and are environmentally qualified to operate during the limiting postulated flood(s) in the event of flooding at the Elevation 66'-6" level of the Auxiliary Building (250F and $\leq$10 psig [19 feet of static head]), including submergence. Staff evaluated the environmental qualification of the proposed sensors in Section 3.2.7 of this report. Additionally, the cabling for the new level instrumentation is routed to the PMS in safety-related raceways following Class 1E separation principles. Per the LAR, there is only a minor increase to the power requirements of the newly proposed level equipment, which, per SNC, is within total Class 1E battery capacity. The acceptability of the additional loading of the 1E batteries is evaluated in Section 3.2.7.1.3 of this SE. No new safety functions or modifications to existing safety functions of any safety-related equipment that interfaces with the protection system are being added as a result of the proposed LAR.

Based upon the information within the LAR, and SNC's subsequent responses to the staff's RAIs as described in Enclosure 12 of SNC's letter dated November 1, 2017 and Enclosure 16 of SNC letter dated December 15, 2017, SNC stated a new functionality subset within the PMS is being added. Specifically the newly proposed level monitoring equipment provides an input into the PMS that performs an alarm and indication only function.

Within Enclosure 12 of SNC letter dated November 1, 2017, SNC describes the signal path of the new system as presenting the level signal to the analog input modules of the PMS; then directing the signals to bistable processor logic microprocessors (BPLs), that perform the bistable function of the protection system; then to the integrated communications processor for signal resolution; prior to sending the signals to the safety displays on the primary dedicated safety panel (PDSP) for indication and alarm.

Since the BPLs also simultaneously perform the safety functions for the RT and ESF functions, the requirements of IEEE Std. 603-1991, as incorporated by reference in 10 CFR 50.55a(a)(2),

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[2] Per Section 4.2 of WCAP 16675-P, the QDPS provides process data for main control room display and to meet Regulatory Guide 1.97 post-accident monitoring system requirements.

apply. Although the newly proposed level sensors and associated equipment do not perform a safety function, IEEE Std. 603-1991 requires, in part, in Section 5.12, "Auxiliary Features" that, "Auxiliary supporting features shall meet all requirements of this standard," and that, "Other auxiliary features that (1) perform a function that is not required for the safety systems to accomplish their safety functions, and (2) are part of the safety system by association (that is, not isolated from the safety system) shall be designed to meet those criteria necessary to ensure that these components, equipment and systems do not degrade the safety systems below an acceptable level."

As such, within Enclosure 16, SNC provided new text to be added to UFSAR, Section 7.1.2.12, "Safety Related Display Information," to ensure this new PMS functionality was adequately described and its system impact understood. This is due to the additional application level tasks being added to the BPLs' microprocessor as a result of this LAR, which in turn causes additional loading or burden on the microprocessors that must be assessed and managed in relation to SNC's, design control (or design change) process and in its verification and validation process for PMS development. In particular, it should address the maximum load testing of the microprocessor's central processing unit and associated time response testing per the IEEE Standard and other licensee commitments. These design change and associated testing activities will be managed via SNC's design change and verification and validation programs and processes.

Although this new functionality of the proposed level sensors and associated equipment does not perform a new safety function, its use within the portions of the PMS that perform a safety function, must be designed and tested in a sufficient manner such that its failure will not negatively impact the inservice testability or reliability of the system. The related activities to confirm system reliability were addressed with the processes identified in WCAP-16096-P, some of which are more closely delineated in the text above, and are therefore acceptable.

The information provided by SNC specifies the new equipment will be Class 1E and will follow Class 1E separation principles to ensure independence is maintained. Based upon the fulfillment of these commitments, and the requirements of GDC 22, the staff finds the proposed changes to the PMS acceptable.

The licensee committed to following the guidance within BTP 7-17, "Guidance on Self-test and Surveillance Test Provisions" and BTP 7-21, "Guidance on Digital Computer Real-Time Performance," Revision 5, within Chapter 7 of the SRP. Based upon the information contained within the LAR and SNC's design change and verification and validation programs, processes and procedures that will address the changes to the design and its impact upon the PMS, including any dealing with failure states, and the requirements of GDC 23, the staff finds the proposed design change acceptable.

Per the information provided in the LAR and its supplements, since the proposed design change does not perform any control or safety-related function, and only provides indication and alarm functionality on the PDSP and the equipment is qualified as Class 1E and follows Class 1E separation principles to provide that functionality, and based on the requirements of GDC 24, the staff finds the proposed design change acceptable.

Per the information contained in LAR-17-010 that delineate the proposed change does not impact any RG 1.97 data or parameters as part of the QDPS subsystem within the PMS or the related display of that information on the PDSP, the staff finds the proposed design change as it applies to RG 1.97 acceptable.

3.2.3.1 Staff Evaluation

Based upon the information contained in the LAR, its supplements that describe the new level equipment's functionality and impact to the PMS subsystems, along with the information previously provided by SNC that describes how its design change, software development and verification and validation processes described in WCAP-16096-P will account for and incorporate the design change within the LAR, the staff finds the proposed design to be acceptable.

3.2.4 Structural Engineering

The staff reviewed the proposed changes in the LAR to the COL, COL Appendix C (and to plant-specific Tier 1 information) and associated Tier 2 information related to the Auxiliary Building flooding identified by the PRHA. Currently, UFSAR Subsection 3.4.1.2.2.2 describes the flooding level assessment for the Auxiliary Building RCA, and specifies that the Auxiliary Level 1 (El. 66'-6") is less than 12 inches. The licensee stated that the PRHA of final piping routing has determined that MELBs (moderate-energy line breaks) in a 10" fire water line in the Auxiliary Building RCA floods Level 1 and 2 of the Auxiliary Building RCA to rise 19'-0." The source of the flooding is from the fire water storage tanks which can cause no more than 19 feet of flooding in seven hour time in the Auxiliary Building RCA. In this section of SE, the staff evaluates the effect of the change in water level from 12'' to 19'-00", in the Auxiliary Building RCA and non-RCA area, flood affected rooms shown in the revised UFSAR Table 3.3-2, "Nuclear Building Room Boundaries Required to have Flood Barrier Floor and Walls," in the LAR Enclosure 3. The UFSAR Table 3.3-2 is revised to address the new flood levels for the flooded RCA Rooms 12156, 12158, and 12258 and respective adjacent non-RCA Rooms 12111, 12112, 12211 and 12212 to incorporate the PRHA results. UFSAR Figure 3.3-11B, "Turbine Building Arrangement Plan at Elevation 100'," Sensitive Unclassified Non-Safeguards Information (SUNSI) is revised to delete the walls for the room at the north-west corner of the Turbine Building that formerly housed the motor driven fire pump and jockey pump, and (SUNSI) UFSAR Figure 1.2-23 is revised to delete the walls for Room 20303. Also, a seismic Category 1 safety-related flood relief louver is proposed for installation to alleviate concern of a FPS MELB in stairwell S04 of Auxiliary Building RCA that can result in an isolated flooding which could accumulated water above 19'-0." The flood louver opens at 1 psi and relieves the flooding from the stairwell to the adjacent area, designated Room 12161.

The staff also evaluated the impact of the flooding on the Auxiliary Building walls and floors, as stated above, in the RCA and non-RCA areas with respect to the structural analysis and design. In performing their evaluation, the staff considered SNC's design criteria described in UFSAR Subsection 3.8.4, "Other Category 1 structures." The Auxiliary Building is classified as Seismic Category I and is designed for the safe shutdown earthquake. According to UFSAR Section 3.8.4.2, "Applicable Codes, Standards and Specifications," the concrete walls and floors are designed to the criteria of the American Concrete Institute (ACI) 349-01, "Code Requirements for Nuclear Safety Related Concrete Structures," and American Institute of Steel Construction (AISC) N690, "Specification for Safety-Related Steel Structures for Nuclear Facilities." Also, the staff considered SRP, Section 3.8.4, for guidance on the acceptance of the wall design.

As a result of the staff review of the LAR, the following additional information was requested from SNC on July 20, 2017 (ADAMS Accession No. ML17201Q412) to complete SE:

a) Provide a visual characterization of the area affected by the flooding including the current wall thickness and the height to which the flood water is expected to rise.

b) Explain using the load combinations that govern the wall design, how the existing wall thickness was re-evaluated to ensure that the new demand was accommodated by the existing wall capacity.

c) Provide configuration and mounting details of the flood relief louver installed in the wall.

d) Provide the minimum distance between the NI structures and the new tanks in the yard to prevent external flooding and distances between objects with seismic II/I interaction consideration.

e) Where will the flood water collected from the RCA be stored? If the collected water is stored in the Auxiliary Building or any other adjacent building, provide a design for the storage of this contaminated water and explain how this additional weight was considered in the building design.

In Supplement 1 (ADAMS Accession No. ML17233A325) and Supplement 2 (ADAMS Accession No. ML17282A013) to LAR-17-010, SNC addressed the above questions as follows:

a) The licensee stated that the current wall thicknesses surrounding the affected area are found in Tier 1 ITAAC Table 3.3-1. Table 2 of Enclosure 11, Supplement 2, provides PRHA and tank/vessel rupture flood heights for rooms in the Auxiliary Building RCA. Figures 1 through 5 correlate the room listed in Table 2 to rooms and wall thicknesses shown in UFSAR Figures 3.7.2-12, Sheet 1 through 5 for NI key structural dimensions. The staff finds that the provided information is adequate.

b) In the response SNC provided details of the flood height criteria how it applied to the Auxiliary Building wall calculations. The licensee included flood loads in the ANSYS model as a surface hydrostatic triangular force distribution on the walls according to flood height. The licensee followed the ACI 349-01 Section 9.2.7, Load Combination 4 (Dead load+ Liquid+Live+Earth+Normal Reaction+Normal Thermal +Tornado). ACI 349-01 code is a commitment in SNC UFSAR Section 3.8.4. It was determined by SNC that Load Combination 4, does not govern the walls design. The tornado load is replaced by the flooding load since the elevations are below ground level. Instead of Load Condition 4, Load Condition 7 (SSE+Accident Thermal) governs for the majority of the walls of the CA20 module.

   Additionally, SNC provided their assessment of the Auxiliary Building walls within RCA from El 66-'6" to El 100"-0" for the highest peak demand ratio. The licensee provided table below shows the demand ratio, wall and elevation.

Direction	Wall	Elevation	Maximum Demand Ratio	Load Combination
Horizontal	1	82'-6"	0.999	LC 7 (SSE + Accident Thermal)
Vertical	I	66'-6"	0.996	LC 7 (SSE + Accident Thermal)

Using the information in the above table, the maximum demand ratio are less than 1. The demand ratio is the demand Vs capacity. Based on the above, the staff finds that SNC followed the committed UFSAR code ACI 349-01 methodology and it is consistence with SRP Section 3.8.4 guidance. Therefore, SNC response is acceptable to the staff.

c) In response to the question on the flood relief louver, SNC noted that in Supplement 1 and 2 of the LAR the louver configuration and mounting is proposed to be installed on the floor of El 66'6" and not on the wall as stated in the LAR-17-010, in Enclosure 1 to LAR 17-010, the license stated that the louver is categorized as AP1000 equipment class C, seismic Category 1, in accordance with AISC N690. The louver is fastened to the floor using anchor bolts in accordance with ACI 349-01, Appendix B, "Anchoring to Concrete." Sealant is applied to the frame to create sealed boundary. Also, SNC stated that the louver is constructed with bolts that meet American Society for Testing and Materials A193, "Standard Specification for Alloy-Steel and Stainless Steel Bolting for High temperature or High Pressure Service." The Auxiliary Building stairwell SO4 Louver dampers are to open and remain open when the differential pressure exceeds 1.3 psi. Based on the SNC response, the staff finds that the design and construction of the louver is in accordance with the UFSAR, ACI 349-01, and AISC N690 code commitments and is consistence with SRP Section 3.8.4 guidance. Therefore, the SNC response is acceptable to the staff.

d) The licensee stated in the response that the tanks are located 388 feet from the north-west corner of the Auxiliary building to prevent interaction between seismic Category 1 NI and the fire protection tank. In Enclosure 4 of the LAR, UFSAR Figure 1.2-2 shows the proposed tank location. The staff reviewed commitment in UFSAR Section 3.7.2.8 and Section 3.7.2.8. The staff finds that the tanks are located an appropriate distance from the NI structures so that there is no possibility for interaction. The staff finds that UFSAR Section 3.7.2.8 meets the requirements of GDC 1, 2, and 4. Therefore, SNC response is acceptable to the staff.

e) The licensee responded that the collected water from the flooding will not be stored in the Auxiliary Building or any other adjacent building. Instead, it will be processed and removed by temporary equipment. The response is acceptable to the staff since there will be no additional loading in other adjacent buildings as a result of the flooding. Therefore, SNC's response is acceptable.

In the LAR, SNC revised following figures:

a) COL Appendix C (and Plant-specific Tier 1) (SUNSI) UFSAR Figure 3.3-11B, "Turbine Building Arrangement Plan at Elevation 100'-0"," is revised to delete the walls for the room at the north-west corner of the Turbine Building that formerly housed the motor-driven fire pump and jockey pump.

b) Plant-specific Tier 2, (SUNSI) UFSAR Figure 1.2-23, "Turbine Building General Arrangement Plan at Elevation 100'-0"," is revised to delete the walls for Room 20303 at the north-west corner of the Turbine Building that formerly housed the motor-driven fire pump.

The staff reviewed above two figure changes of the walls removal and concluded that the walls are located in the non-safety-related portion of the Turbine Building and considering removal of the interior walls, will reduce load which have no significant effect on the structural qualification. Even with the removal of the wall, the design of the Turbine Building will be consistent with the codes and standards included in the design bases in UFSAR Section 3.7.2.8.3, "Turbine Building."

3.2.4.1 Staff Evaluation

The staff reviewed SNC's proposed changes provided in LAR-17-010. Based on the staff's technical evaluation described in this SE, the staff found:

1. The proposed changes to the plant specific license bases to the UFSAR and COL Appendix C (and Plant Specific Tier 1) to mitigate the FPS flooding in the Auxiliary Building Level 1 and Level 2 as evaluated above are acceptable. The staff review of the affected walls and floors of the Auxiliary Building considered the new flood level that increased from 1'-0" to 19'-0" and the wall and floor design review was performed to include new loads to ensure the design was in accordance with codes and standards committed to in the UFSAR including the provisions of ACI 349-01 and AISC N690 codes. Based on this review, effect of flood on the structure design meets regulatory requirements, therefore the changes proposed are acceptable.

2. The proposed changes to COL Appendix C (and Plant-specific Tier 1) Subsection 3.3, Buildings, (SUNSI) UFSAR Figure 3.3-11B, Turbine Building Arrangement Plan at Elevation 100'-0" and Plant-specific Tier 2, (SUNSI) UFSAR Figure 1.2-23, "Turbine Building General Arrangement Plan at Elevation 100'-0"," are revised to remove walls for the room at the north-west corner of the Turbine Building that formerly housed the motor-driven fire pump and jockey pump. The affected the Turbine Building wall will see reduction in the design loads and the design will continue to meet codes and standards. Based on this review, the proposed changes in Turbine Building wall design meets regulatory requirements, therefore the changes proposed are acceptable.

Based on these findings, the staff concludes that there is reasonable assurance that the requirements of GDC 1, GDC 2, and GDC 4 of 10 CFR Part 50, Appendix A, will continue to be met. Therefore, the staff finds the changes proposed in this LAR acceptable.

3.2.5 Operator Actions

In NRC RAI #2 (ADAMS Accession No. ML17265A357) dated September 22, 2017, the staff asked for additional information about the credited manual operator actions including environmental conditions that may interfere with credited actions to mitigate the cascading CVS HELB scenario. In LAR Supplement 3 (ADAMS Accession No. ML17305B507), dated November 1, 2017, SNC provided additional information about the credited operator actions and environmental conditions.

In the case of flooding resulting from a MELB of the FPS, operators receive a signal in the MCR from the two proposed seismically qualified level indicators (WLS-LE-400A/B) on safety-related displays indicating flooding is occurring in the Auxiliary Building. Plant operators dispatched by the MCR proceed to the yard area next to the Fire Water Storage Tank to locally close one of the two locked 2” manual valves (FPS-V804 or FPS-V805) to secure the 50 gpm automatic refilling of the FPS tanks. This credited operator action, if taken at or before 7 hours after pipe rupture initiation, limits flooding to the design basis level.

In the case of a cascading Chemical and Volume Control System (CVS) High Energy Line Break (HELB) in the Auxiliary Building, operator action to isolate the demineralized water storage tank (DWST) lines into the Auxiliary Building and boric acid storage tank (BAST) supply lines to CVS makeup pumps must be taken in addition to the isolation performed for the MELB FPS break. The credited operator action to isolate the DWST involves closing the supply header valve in the containment access corridor located in the Annex Building. The credited operator action to isolate the BAST involves closing the BAST outlet header isolation valve in the demineralized water deoxygenating room in the Annex Building. Operator action to terminate the FPS and CVS within seven hours is credited for this event in the HELB analysis. The credited operator actions to mitigate flooding are procedurally controlled via abnormal operating procedures (AOPs) which direct the operator's action to respond to this condition. Operators have approximately seven hours to complete the above actions to prevent the flood levels from exceeding the design basis limit of 19 feet.

For the flooding events detailed in this LAR, environmental conditions are not expected to impact the ability of operators to access these valves because the valves are not in the Auxiliary Building. Loss of coolant accident and steam line rupture are not postulated with this flooding event, therefore high radiation levels do not preclude the operator actions.

In NRC RAI #2 (ADAMS Accession No. ML17265A357) dated September 22, 2017, the staff asked for additional information about the credited manual operator actions including environmental conditions that may interfere with credited actions to mitigate the cascading CVS HELB scenario. In LAR Supplement 3 (ADAMS Accession No. ML17305B507), dated November 1, 2017, Southern Nuclear Company provided additional information about the credited operator actions and environmental conditions.

3.2.5.0.1 General Deterministic Review

A formal risk analysis was not completed during this review. The staff used engineering judgment to scope the appropriate level of human factors review. Because safety-related equipment was not impacted by the postulated flooding of the Auxiliary Building, even without operator action, for up to seven hours, the staff conducted a Level II review. This strategy is consistent with NUREG-1764 “Guidance for the Review of Changes to Human Actions,” guidance which allows the reviewer flexibility in adjusting the review contents based on qualitative factors and with SRP “Introduction – Part 2” which describes conditions that allow for adjusting the scope of reviews. The LAR and supplemental information describe the human factors considerations for the new operator actions resulting from the design change. This includes consideration of task characteristics necessary for successful task completion, use of procedures, and addition of reliable control room indications.

The staff review determined that the actions introduced by this LAR do not reduce defense in depth. The proposed plant modification primarily introduces new Fire Water Storage Tank size and lineup changes that minimize the flood levels within the Auxiliary Building RCA to below the design basis level within a seven hour flooding period. The operator actions to isolate automatic make-up add to the defense in depth, and do not overly rely on the operator to mitigate the flooding. Operator action can also be taken at any time to isolate the line break and minimize flooding. Additionally, the inclusion of flood relief louvers increases defense in depth by using automatic passive actions to reduce the likelihood of flooding in some areas of the plant.

3.2.5.0.1.1 Staff Evaluation

The proposed plant changes do not reduce defense in depth with respect to human factors because passive features such as tank size and flood relief louvers are capable of mitigating and control flood levels within the first seven hours of an event. Operator actions are not overly relied upon during the first seven hours. Relatively simple actions are credited only to halt the flooding beyond seven hours.

The modifications described above are consistent with the human factors process described in NUREG-1764 and are therefore adequate to meet 10 CFR 50.34(f)(2)(iii). The staff concluded that the acceptance criteria are met, therefore staff concludes that modifications described in the LAR continue to meet current human factors requirements.

3.2.5.0.2 Analysis

In LAR Supplement 3, SNC described the tasks that must be completed by an operator to limit flooding. The description identifies relevant task related concerns (such as task timing and reasonable consideration of environmental concerns). The tasks are relatively simple, and have a large amount of time margin to allow for recovery from errors. Multiple success paths are available; the safety-related indications being added to the control room (as well as existing indications of flooding in the Auxiliary Building) will help operators to understand when actions are necessary.

3.2.5.0.2.1 Staff Evaluation

The changes created by this LAR do not change the plant function allocation and do not affect the staffing levels. Therefore, descriptions of these analyses are not applicable to this review. The licensee has analyzed the new credited operator actions and identified appropriate modifications to the human interfaces and AOPs. AOPs are covered in the scope of SNC's training program. Therefore, the staff finds this analysis to be acceptable to meet this NUREG-1764 criterion.

3.2.5.0.3 Design of Human System Interfaces, Procedures and Training

LAR-17-010 implements new seismic Category 1, Class 1E flood-up level sensors and associated safety-related MCR indications to cue operators of a flooding event in the Auxiliary Building RCA. In addition to the flooding instrumentation added by this LAR, operators have several existing redundant indications of flooding such as sump level indication and high-level alarm, running sump pumps, and an unexpected actuation of the FPS pumps running alarm. The credited operator actions described in this LAR are procedurally controlled via AOPs. AOPs are included in the plant training program.

3.2.5.0.3.1 Staff Evaluation

The licensee is adding new instrumentation and indications to notify the MCR of the need to take actions to reduce flooding levels in addition to existing indications for helping operators understand plant status. Procedures and training are also employed to ensure predictable and reliable plant operation. Therefore, the staff finds this treatment to be consistent with the acceptance criteria in NUREG-1764 and thus acceptable.

3.2.5.0.4 Human Action Verification

The licensee will use ITAAC item 2.3.10.12 to ensure that the safety-related flood-up level sensors and displays are available and can be retrieved in the MCR.

3.2.5.0.4.1 Staff Evaluation

Because the plant is still under-construction at the time of this LAR submittal, walkthrough verification of the new credited operator actions is not possible, but not required in order for the staff to complete this SE. In choosing to perform a Level II review, the staff took a conservative approach; NUREG-1764 allows the reviewer to further adapt the guidance to meet the unique demands of particular reviews. Thus, the staff did not review validation or testing of the operator actions because the change involves simple, proceduralized actions, new instrumentation and very large time margins. Additional testing would add little value to the review. The use of ITAAC ensures that the as-built plant will conform to the design as described in this LAR. Given this, the staff finds this treatment to be consistent with this NUREG-1764 criterion and thus acceptable.

3.2.5.1 Conclusion

The staff completed a review of the credited operator actions proposed by this LAR. Based on the information provided in the LAR and Supplement 3, the actions were found to be in accordance with NUREG-1764 are therefore acceptable.

3.2.6 Radiation Protection

The staff evaluated potential radiological impacts from the flood scenarios discussed in the LAR. Areas that could be flooded include many of the rooms containing radwaste processing equipment and tanks in the lower elevations of the Auxiliary Building, as well as other systems expected to contain radioactive material. The potentially flooded areas include rooms with valves used to align spent fuel pool (SFP) makeup. Aligning the valves to provide SFP makeup is a vital mission which is included in Chapter 12 of the UFSAR as an area which is required to be accessible following an accident in accordance with the requirement 10 CFR 50.34(f)(2)(vii). The valves in Room 12365 were of particular concern to staff. The LAR specifies that Room 12365 could be flooded to a maximum flood level of approximately 108 inches. The staff issued Question 1 (ADAMS Accession No. ML17201Q412), requesting that SNC explain how the flooding events described in the LAR would or would not impact the ability for operators to access the SFP make-up valve alignment areas and any other vital access paths or areas. In the response to Question 1 (ADAMS Accession No. ML17233A325), SNC stated that operations to align SFP makeup and passive containment water inventory makeup are performed at an elevation greater than 96’-6”, including Room 12365. The licensee stated that any manual actions in these room are not required until after flooding has been terminated and floodwater has drained to lower elevations leaving these room unflooded (access to the valves are not needed until at least 18 hours after an event, as specified in UFSAR Table 9.1-4). The staff reviewed LAR-17-010 and found that the maximum flooding level in the Auxiliary Building is 19 feet (elevation 85'-6"). Since the water will be drained from these areas when access is required and since any residual radioactivity at those elevations from drained floodwater which could come from equipment damaged from the flooding (the FPS, which is the initial source of the floodwater, is not radiologically contaminated), would be expected to be minimal. As a result, the staff found the response to be acceptable.

10 CFR 50.34(f)(2)(viii) requires that licensees provide a capability to promptly obtain and analyze samples from the RCS and containment that may contain accident source term radioactive materials. The licensee explained that the need to take post-accident samples is not coincident with the pipe-rupture events described in LAR-17-010. In addition, SNC specified that NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design," indicated that the AP1000 plant is not required to have a dedicated post-accident sampling system. The acceptance criteria of NUREG-1793 does not require a dedicated location to take post-accident samples. The staff found that there was no single design basis event that would both flood the post-accident sampling system room and damage the reactor core requiring the need to take post-accident samples. Therefore, the staff found the staff found SNC's response to Question 1 to be acceptable and that the requirements of 10 CFR 50.34(f)(2)(vii) and 10 CFR 50.34(f)(2)(viii) continue to be met.

As discussed previously, the flooding events assumed in the LAR would flood much of the radwaste processing system equipment. Therefore, the staff issued Question 2 (ADAMS Accession No. ML17201Q412) requesting SNC to describe how the radwaste systems and components continue to meet the guidance of Regulatory Guide (RG) 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components in Light Water Cooled Nuclear Power Plants." Compliance with RG 1.143 ensures compliance with some aspects of the requirements of 10 CFR Part 50 Appendix A, GDC 2, and GDC 61.

In the response to Question 2 (ADAMS Accession No. ML17233A325), SNC specified that the design of the radwaste management systems is not affected by LAR-17-010 because the external flooding design was not changing and that the pipe rupture events described in LAR 17-10 do not affect the RG 1.143 commitments. The staff notes that for the AP1000 the radioactive waste management systems and components in the Auxiliary Building containing high enough activity (such as the gaseous waste management system) are classified RW-IIa, in accordance with RG 1.143, which is the highest classification of such systems in the guidance document. The licensee also specified that the hydrostatic pressure exerted on equipment by the postulated flood water only serves to counter-act the internal design pressure of the systems and that the internal design pressures remain limiting for pressurized equipment, which includes the liquid and gaseous radwaste systems. The staff agrees that internal flooding events are not explicitly addressed in RG 1.143 and that the seismic and external flooding would not be impacted. However, SNC did not provide information demonstrating that there would not be a significant release of radioactive material due to the assumed flooding and submergence of the systems, for example, due to water intrusion into the systems or buoyancy effects. This is discussed in more detail below, regarding the evaluation of the response to Question 3.

In Question 3, (ADAMS Accession No. ML17193A265), the staff requested that SNC provide additional information regarding whether the facility continues to meet the guidance (to which the AP1000 design was certified) of RG 1.29, "Seismic Design Classification for Nuclear Power Plants," and RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants." RG 1.29 indicates that systems that contain or may contain radioactive material and the postulated failure of which would result in conservatively calculating potential offsite doses that are more than 500 mrem total effective dose equivalent be designed as seismic Category 1, and that those portions of SSCs of which failure could reduce the functioning of any plant feature and result in exceeding this criteria, also be designed to seismic Category I criteria. RG 1.26 contains similar guidance for classifying systems as Quality Group C (and therefore, designed to those standards). Therefore, the staff requested that SNC update the LAR to describe the worst case radiological release from flooding scenarios due to the piping failures and evaluate if the release exceeds the offsite dose criteria described in the RGs. In addition, if the potential for offsite doses exceeding the dose criteria exists, the staff requested that SNC provide additional details about how the facility will meet the applicable regulatory requirements (such as by designing a portion of the FPS to a higher standard, in accordance with RGs 1.26 and 1.29). Finally, if no design changes were considered to the system from which the flooding originates and the potential for the flooding events described in the LAR still exist, the staff requested that SNC provide additional information regarding how the requirements of 10 CFR 20.1101(b) and 10 CFR 20.1406 were being met.

In the response to Question 3 (ADAMS Accession No. ML17282A014), SNC specified that the worst-case radiological release resulting from the postulated FPS driven flooding events described in LAR-17-010 would involve flooding of the Railcar Bay (Room 12371), which would then be available to transport any available surface contamination below the flood height to the environment through the exterior door. In undertaking the analysis, SNC assumed that only surface contamination would be released from the building because resin and high integrity containers in the area are located above the maximum flood height and would not be affected by the flood water. The licensee indicated that the results were within applicable safety and regulatory limits. The licensee also indicated that the analysis of the failure of small lines carrying primary coolant outside containment in UFSAR Subsection 15.6.2 would be bounding for airborne doses of the liquid-only releases postulated for breaks in the rail car bay. The staff agrees that the radiological consequences of a surface contamination release would not be significant given the scenario described in the LAR and RAI responses. The licensee also provided additional information regarding design features to limit the spread of potential contamination and the release of radioactive material if a release of radioactive liquids were to occur. This includes that any penetration in the RCA of the Auxiliary Building are designed to serve as flooding and ventilation barriers and that any radioactive material collected due to a failure would be processed and monitored, as necessary before release. The staff finds these design features to minimize the spread of contamination and minimize exposure to be consistent with the requirements of 10 CFR 20.1406 and 10 CFR 20.1101(b). Staff also notes that the dose to workers would be controlled by the plant radiation protection program, if such a flooding event described in the LAR were to occur.

While the staff agrees that the radiological consequences of a surface contamination release would not be significant, as stated previously, SNC did not originally provide sufficient information to conclude that there would not be damage and a subsequent significant release of radioactive material from non-safety-related systems due to flooding. In the initial Vogtle combined license review, a dose of 5.3 mrem was calculated from the maximum liquid tank failure. This was based on an assumed failure of the effluent holdup tank containing the design basis RCS activity. There are only about seven radwaste storage tanks that could contain significant amounts of radioactive materials and that could be impacted by the maximum flooding scenario. In addition, while chemical and volume control system and liquid radwaste system demineralizers and filters could potentially be flooded, these components are below ground level and their contents would not be expected to be released to the environment in significant quantities by any flooding scenario. Based on this, the staff determined that there is reasonable assurance that any potential public dose received from a liquid or resin containing system failure would be significantly less than the public dose limit and the requirements of 10 CFR Part 20 would continue to be met

UFSAR Section 11.3.3.4 specifies that using the site boundary (0 to 2 hour) atmospheric dispersion factor, and an assumed 1 percent fuel defects, the radiological consequence of a 1-hour bypass of the delay beds and 30 minute decay before release to the environment would result in a site boundary whole body dose of 0.1 rem. However, damage to the guard and delay beds could potentially release a greater quantity of radioactive material because the beds are designed to hold large quantities of radioactive material (holding up Xenon for 38 days prior to release and Krypton for 2 days prior to release). Therefore, staff requested a clarification to the RAI response requesting that SNC provide sufficient information demonstrating that there would not be a significant release of radioactive material if the gaseous radioactive waste management system were flooded or if a significant release could not be prevented to provide a dose analysis and to justify why the release is acceptable. The licensee clarified, in a revision to the previous RAI response, in Supplement 4, Revision 1, (ADAMS Accession No. ML17349A928) that the weight of the delay beds, with charcoal, is at least 6,047 lbs, and that the weight of displaced water is 5,910 lbs. Likewise, SNC indicated that the guard bed has a total component mass of 2409 lbs, while due to the small size, the displaced water is only 755.7 lbs. Therefore the delay beds will not be subjected to upward buoyant forces even if completely submerged. In addition, the beds are bolted to a mechanical module that is then welded to the floor. Based on this, the staff finds that the beds are appropriately protected from buoyancy effects when flooded.

In the response, SNC also specified that the operating pressure of the system exceeds the hydrostatic pressure of the postulated flood, therefore, there is no driving force to send moisture from the flood into the system given the system operating parameters, thus, there is minimal risk of moisture intrusion. In the LAR 17-010 Supplement 4, Revision 1, response, SNC specified that the valve which isolates the normal system release (post-delay bed) is both fail closed and controlled to close on a high-high radiation signal. In addition, SNC specified that the postulated flooding does not result in submergence of the electrical cabinets or control cabinets of the gaseous waste management system, so there is no need to postulate a spurious control signal, i.e., the release valve that releases radioactive gases periodically under normal operation would not release gases under the flooding scenario. The staff notes that other valves associated with the beds are hermetically sealed and, therefore, would be expected to prevent leakage into the system. Therefore, the staff finds that there is reasonable assurance that water would not enter the system through the valves and that the release valve would prevent an accidental release. In the supplemental information, SNC also specified that there would be no breach of the gaseous waste management system to allow moisture incursion into the system. If there is no breach of the system, there is no risk of a release.

The staff audited the pipe rupture hazard analysis safety-related equipment dose evaluation in the Auxiliary Building for equipment important to safety that is required to be environmentally qualified in accordance with 10 CFR 50.49 and 10 CFR Part 50, Appendix A, GDC 4 (ADAMS Accession No. ML17156A426 for the audit plan). For the EQ analysisc the licensee assumed that the radiologically contaminated systems failed and that the radionuclide contents dispersed in the area, including resin transfer lines and that the safety-significant component is submerged in the radioactive fluid for two weeks. This assumption is conservative in that it assumes essentially the worst possible radiological scenario for the equipment in those areas for a two week period. It is reasonable that appropriate action could be taken within two weeks of a radioactive waste release to maintain safety (such as waste cleanup, equipment replacement, and/or reactor shutdown, as appropriate). In addition, in the very unlikely scenario that during actual plant operation doses to equipment are higher than those calculated or could result in higher total integrated dose than a piece of equipment is qualified, the licensee still has to appropriately maintain the equipment, as required by 10 CFR 50.49 and GDC 4. As a result, the staff found the radiation doses calculated for equipment impacted to be acceptable.

3.2.6.1 Conclusion for Radiation Protection Evaluation

As discussed above, SNC demonstrated that all post-accident vital missions will be able to be adequately performed in accordance with 10 CFR 50.34(f)(2)(vii) and 10 CFR 50.34(f)(2)(viii), that equipment will remain appropriately qualified in accordance with 10 CFR 50.49 and GDC 4, and that there is reasonable assurance that a flooding event will not result in any significant radiological releases. The design includes features to minimize radiation exposure and limit the potential for the spread of contamination to non-radiological areas in accordance with the applicable regulatory requirements including GDC 60 and 61, 10 CFR 20.1101b, and 10 CFR 20.1406. Consequently, the staff finds the staff finds that the changes as detailed in the LAR are acceptable.

3.2.7 Electrical Engineering

3.2.7.1 Environmental Qualification of Electrical Equipment

The staff reviewed the proposed revisions to UFSAR Tier 2, Table 3.11-1 specified in Enclosure 1 of the LAR. Based on the results of the PRHA, SNC proposed to revise the COLs Appendix C, Table 3.3-2, to modify the design of the power plant by changing the flooding of Levels 3 through 5 of the Auxiliary Building RCA. The flooding level changes require that all the equipment subject to environmental qualification be qualified for submergence. 10 CFR 50.49(e)(6) requires that submergence must be included in the equipment qualification program. The environmental qualification of the valves and their subcomponents in regards to submergence is in Section 3.2.8.1 of this report. In addition SNC proposed to add two floodup level sensors to the Auxiliary Building RCA. Based on the result of the PRHA the following changes are proposed to UFSAR Tier 2, Table 3.11-1:

- Adding WLS Auxiliary Building RCA Floodup Level Sensors WLS-JELT400A and WLS-JE-LT400B to table with the Environmental Zone from UFSAR Table 3D.5-1.

3.2.7.1.1 Staff Evaluation

The staff finds the changes to the UFSAR Tier 2, Table 3.11-1, are acceptable because it provides sufficient information to verify the equipment added and revised will be environmentally qualified in accordance with the environmental qualification requirements specified in 10 CFR 52.79(a)(10), which requires the list of electric equipment important to safety that is required by 10 CFR 50.49(d). Therefore, SNC continues to meet 10 CFR 50.49 and specifically, SNC is updating the list of electric equipment important to safety that will be qualified.

The staff also evaluated the proposed changes to UFSAR Tier 2, Subsection 3D.5.2.2. SNC proposed to add a sentence to this section to specify that qualification for the flooded/wetted abnormal environment can be accomplished to be consistent with UFSAR Tier 2 Subsection 3.11.1.2. UFSAR Tier 2, Subsection 3.11.1.2, states, “In the event of potential flooding/wetting, one of the following criteria is applied for protection of equipment for service in such an environment; Equipment will be qualified for submergence due to flooding/wetting; Equipment will be protected from wetting due to spray; and will be evaluated to show that failure of the equipment due to flooding/wetting is acceptable since its safety-related function is not required or has otherwise been accomplished." Based on the proposed changes in UFSAR Tier 2, Table 3.11-1, the applicant will be qualifying the equipment added or revised for submergence. 10 CFR 50.49(e)(6) requires that submergence must be included in the equipment qualification program. Therefore, the staff finds the proposed changes to UFSAR Tier 2, Subsection 3D.5.2.2, are acceptable, since SNC continues to meet 10 CFR 50.49 by including flooding/wetting.

3.2.7.2 Electrical Separation of Equipment

The staff reviewed the proposed revisions to UFSAR Tier 1, Subsection 2.3.10, "Liquid Radwaste System," specified in Enclosure 1 of the LAR. SNC proposes to revise Section 2.3.10 to "add the Class 1E, divisionalized and separated cabling as Items 11.a) and 11.b) of the Design Description for the Class 1E WLS Auxiliary Building RCA Floodup Level Sensors." The following were revised with respect electrical separation of equipment;

- UFSAR Tier 1 Table 2.3.10-1 was revised to add the WLS Auxiliary Building RCA floodup level sensors, WLS-400A and WLS-400B, with notation the level sensors are seismic Category I, are Class 1E, and have safety-related display for flood indication.

- Table 2.3.10-4, "Inspections, Tests, Analyses, and Acceptance Criteria," was revised to add Design Commitments; Inspections, Tests, Analyses, and Acceptance Criteria to the ITAAC for items 2.3.10.11.a for verifying Class 1E sensors are powered from respective Class 1E divisions (note: Separation of the WLS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable as required by Section 2.3.10 Item 11.b) is provided by Table 3.3-6, item 7.d which confirms that minimum separation within the areas of the plant which have Class 1E cables.

3.2.7.2.1 Staff Evaluation

10 CFR 50.55a(h) requires that the safety systems for plants with construction permits must meet the requirements of IEEE Std. 603-1991. Section 5.6.1 of IEEE Std. 603-1991 states, "Redundant portions of a safety system provided for a safety function shall be independent of, and physically separated from, each other to the degree necessary to retain the capability of accomplishing the safety function during and following any design basis event requiring that safety function."

The staff evaluated the proposed changes to UFSAR Tier 1, Subsection 2.3.10, to verify that it meet the requirements of 10 CFR 50.55a(h) and GDC 17. The proposed changes UFSAR Table 2.3.10-4 are acceptable because it requires that the Class 1E components identified in UFSAR Table 2.3.10-1 are powered from their respective Class 1E division, and separation is provided between WLS Class 1E divisions and between Class 1E divisions and non-Class 1E cable. Therefore, the staff finds the proposed changes to UFSAR Tier 1, Subsection 2.3.10, are acceptable and meet the requirements of 10 CFR 50.55a(h).

3.2.7.3 Additional Battery Loading

In the LAR, SNC discussed the addition of the WLS Auxiliary Building RCA Floodup Level sensors WLS-JELT400A and WLS-JE-LT400B to the PMS System. These new sensors resulted in additional battery loading.

3.2.7.3.1 Staff Evaluation

The staff evaluated the impact of adding the floodup level sensors to the class 1E systems batteries' electrical loading. In RAI, Question 4 (ADAMS Accession No. ML17201Q412), the staff requested the applicant to describe the impact to the battery loading by adding the floodup level sensors. In response to RAI, Question 4, (ADAMS Accession No. ML17233A325), SNC stated that new level instrumentation (floodup sensors WLS-400A and WLS-400B) are added and that there is a minor increase to the power requirements of the Class 1E batteries which is within the available battery capacity. Based on SNC's response, the staff requested the applicant to describe the added load on the Class 1E batteries, to discuss which battery banks or inverters are powering the loads, to discuss any changes or confirm no changes to UFSAR Tables 8.3.2-1 through 8.3.2-7, to discuss the changes to the battery sizing calculation, and to describe the impact of the changes to the margin for battery sizing.

In Supplement 5, dated January 3, 2018, (ADAMS Accession No. ML18003B082) SNC stated that the floodup level sensor consume a maximum of 1.8 Watts (W) per transmitter, or 3.6 W total for the 2 added transmitters, and that the sensors are continuous loads on the 24-hour batteries. The floodup level sensors are powered from Divisions A and C of the PMS, on the IDSA-DU-1 inverter and IDSC-DU-2 inverter. The licensee stated that there are no impacts on UFSAR Tables 8.3.2-1 through 8.3.2-7 or to the battery sizing calculation. The applicant stated that because the sensors are connected to the PMS, and the input to the battery sizing calculation as it pertains to the PMS loads presume the cabinet supports the full number of sensors, the cabinet is capable of handling in determination of power consumption and there is no impact on battery loading. Therefore, SNC determined there is no impact to UFSAR Tables 8.3.2-1 through 8.3.2-7 and that the battery sizing calculation, including design margins, is not affected since SNC presumes the PMS is fully loaded prior to the addition of the sensor.

The staff evaluated SNC response to the RAI questions and finds that SNC has considered the impact of the loading created based on the addition of the new floodup level sensors, because the loads in the battery sizing calculation considered the PMS at the full capacity at the time of the calculation. Therefore, the staff concludes that the loads created by addition of the floodup level sensors has been adequately evaluated and does not impact the capacity or the margin of the battery sizing calculation.

3.2.8 Mechanical Engineering

3.2.8.1 Environmental Qualification of Safety-Related Valves due to Submergence or Spray

In LAR-17-010 and Supplements 1 through 3, SNC identified safety-related valves that will be submerged or exposed to spray due to the PRHA and require submergence or spray testing as part of the environmental qualification program. The licensee proposed to revise UFSAR Tier 2, Table 3.11-1 to specify that submergence testing or operation with spray, as applicable, is required for these safety-related valves and associated subcomponents.

By RAI Question 5 dated July 20, 2017 (ADAMS Accession No. ML17201Q412), the staff requested SNC to provide additional information to describe the safety-related valves and associated valve subcomponents that are submerged as a result of the PRHA and the basis for concluding that submergence testing is not required for certain valves. In its response to RAI Question 5 dated October 9, 2017 (ADAMS Accession No. ML17282A014), SNC listed all safety-related valves and associated valve subcomponents that are submerged as a result of the PRHA and stated that submergence testing is not required for the valves and subcomponents where flooding has no impact to the valve safety function. For example, the safety-related function is not required during the event or has otherwise been accomplished, the valves are fail-closed or remain closed and flooding does not affect the valve operation, or the additional external pressure of flooding does not affect the valve operation. In the response to RAI Question 5, SNC proposed to revise UFSAR Tier 2, Table 3.11-1, to specify submergence or spray testing, as applicable, for the safety-related valves, solenoid valves, and limit switches that required submergence or spray testing. The staff reviewed SNC's response to RAI Question 5 and determined that UFSAR Tier 2, Table 3.11-1, and the proposed revision to UFSAR Tier 2, Table 3.11-1, as specified in LAR-17-010, identify the safety-related valves and associated valve subcomponents that require submergence or spray testing as a result of the PRHA. The staff finds the proposed revision to UFSAR Tier 2, Table 3.11, acceptable because it identifies submergence or spray testing for the safety-related valves and associated valve subcomponents that are submerged as a result of the PRHA, and it satisfies the GDC 4 requirements that SSCs 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. Therefore, RAI Question 5 was closed.

In LAR-17-010, SNC stated that containment isolation valve VFS-PL-V003 body is submerged but the staff could not determine whether the associated solenoid valve and limit were also submerged. By RAI Question 6 dated July 20, 2017 (ADAMS Accession No. ML17201Q412), the staff requested SNC to describe if the solenoid valve and limit switch for valve VFS-PL-V003 are submerged. In its response to RAI Question 6 dated August 21, 2017 (ADAMS Accession No. ML17233A325), SNC stated that VFS-PL-V003 valve body is submerged but the limit switch, air operator, and solenoid valve are located above the flood level and are not submerged. The staff confirmed that UFSAR Tier 2, Table 3.11-1, specifies submergence testing for VFS-PL-V003 valve body and that submergence testing is not required for the associated limit switch, air operator, and solenoid valve. The staff finds SNC's response acceptable because submergence testing is specified for Valve VFS-PL-V003 body and SNC clarified that the limit switch, air operator, and solenoid valve are located above the flood level and are not submerged. Therefore, RAI Question 6 was closed.

3.2.8.1.1 Staff Evaluation

The staff has reviewed SNC's analysis provided in LAR-17-010 and applicable Supplements 1-3 and finds that provisions for the environmental qualification safety-related valves and associated valve subcomponents that are submerged or exposed to spray as a result of the PRHA, are acceptable, because applicable environmental qualification testing is specified in the proposed revision to UFSAR Tier 2, Table 3.11-1. Based on these findings, the staff concludes that there is reasonable assurance that the requirements of 10 CFR Part 50, Appendix A, GDC 4, will continue to be met. Therefore, the staff finds the proposed change acceptable.

3.2.9 Fire Protection

Modifications to the FPS in order to limit flooding levels in the Auxiliary Building RCA impacted the automatic refilling capability of the fire water storage tanks and the existing separation provision of the redundant fire pumps. With the modification to the motor-operated valves used in the 6-inch refilling lines for the fire water storage tanks from automatic to manually-operated lock-closed valves, the staff is concerned that the requirement for a fire water storage tank to be refilled in 8 hours or less in accordance with RG 1.189, Revision 2, “Fire Protection for Nuclear Power Plants,” cannot be met without additional compensatory design and/or operational features. As such, the staff issued an RAI dated September 22, 2017 (ADAMS Accession No. ML17265A357), seeking further justification for the acceptability of the modification, specifically the reliability and feasibility of the manual action required to refill the fire water storage tanks. In an RAI response dated November 1, 2017 (ADAMS Accession No. ML17305B507), SNC indicated that both primary and secondary fire water storage tanks are equipped with multiple level indicators and alarms in the Control Room as well as the main fire protection panel to alert operators of various low/high water level conditions. The multiple alarms allow operators to coordinate the activity to either open the manual 6" fill supply lines and/or align the redundant fire water tank with the FPS. Procedurally, once the appropriate low level alarm is received, an operator will proceed to the fire water storage tank that has a low water level and refill the tank by opening the 6-inch manual valve. Once initiated, the minimum required amount of fire protection water (396,000 gallons) can be replenished in 8 hours at the minimum fill rate of 825 gpm. The staff reviewed plant equipment layout drawings and found the location of the manual valves reasonably accessible for manual operation by the operator. Based on the above, the staff finds the proposed manual refilling of the fire water storage tanks adequately reliable and feasible, and therefore, continue to meet the guidance of RG 1.189, Revision 2, and, therefore, the requirement of 10 CFR 50.48.

With the relocation of the electric motor-driven fire pump and jockey pump from the Turbine Building to the yard area adjacent to the diesel-driven fire water pump, the staff was concerned that required separation criteria of these redundant fire pumps is not maintained in accordance with 10 CFR 50.48, GDC 3, and RG 1.189, Revision 2. The staff issued an RAI dated September 22, 2017 (ADAMS Accession No. ML17265A357), seeking further clarification and confirmation that the electric motor-driven fire pump and its associated power and control cables are adequately separated from the diesel-driven fire pump and/or adequately protected such that a postulated fire at any location in the plant area cannot render both fire pumps inoperable. In an RAI response dated November 1, 2017 (ADAMS Accession No. ML17305B507), SNC stated that the diesel-driven fire pump package and motor-driven fire pump package are separated by a minimum 3-hour fire barrier. The equipment and controls for the diesel-driven and motor-driven fire pumps are located within their dedicated separated enclosures, and the enclosures are separated by physical distance as well as a 3-hour fire barrier. Both fire pumps have circuits/cables that are utilized for remote start from the MCR. The plant control system (PLS) cabinets that control the relay for the remote start are located in the same electrical room within the Turbine Building. However, in an event of a fire within the MCR or the room containing the PLS cabinets, both fire pumps maintain their automatic start capability as required by National Fire Protection Association (NFPA) 20, “Standard for the Installation of Stationary Pumps for Fire Protection,” and both pumps maintain manual starting capability from the applicable fire pump controller.

3.2.9.1 Staff Evaluation

Based on the above, the staff finds the proposed relocation of the motor-driven fire pump and associated jockey pump from the Turbine Building to the yard area adjacent to the diesel-driven fire pump maintained adequate separation between the redundant fire pumps, and has no adverse impact on the required function of the fire pumps, and therefore, continue to meet the guidance of RG 1.189, Revision 2, and, therefore, the requirement of 10 CFR 50.48. Consequently, the staff concludes that modifications to the FPS to limit flooding level in the Auxiliary Building RCA as proposed by LAR-17-010 has no adverse impact on the FPS required function and capability. Therefore, SNC continues to meet the requirement of 10 CFR 50.48.

3.3 SUMMARY OF TECHNICAL EVALUATION

Based on these findings, the staff concludes that there is reasonable assurance that the requirements of GDC 1, GDC 2, GDC 3, GDC 4, GDC 17, GDC 21, GDC 22, GDC 23, GDC 24, GDC 60 and GDC 61 in Appendix A to Title 10 CFR 50, 10 CFR 50.150, Appendix S to 10 CFR Part 50, and Appendix B to 10 CFR Part 50, will continue to be met. Therefore, the staff finds the proposed changes to be acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission regulations in 10 CFR 50.91(b)(2), the designated Georgia State official was consulted on the proposed issuance of the amendment. The State official had no comment (dated December 27, 2017).

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.” Based on the staff evaluation and conclusion stated in Section 3.2, the staff 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 previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (82 FR 26123 published on June 6, 2017). 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 exemptions are necessary to allow the changes proposed in the license amendment, and because the exemptions do not authorize any activities other than those proposed in the license amendment, the environmental consideration for the exemptions is identical to that of the license amendment. Accordingly, the exemptions meet 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 exemptions.

6.0 CONCLUSION

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

The Commission has concluded, based on the considerations discussed in Section 3.2 of this SE and the staff's confirmation that the changes proposed in this LAR do not change an analysis methodology, or assumptions that there is reasonable assurance that (1) the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission 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 LAR to be acceptable.

7.0 REFERENCES

1. LAR-17-010, Vogtle Electric Generating Plant, Units 3 and 4, Request for License Amendment and Exemption: Pipe Rupture Hazard and Flooding Analyses, dated March 31, 2017 (ADAMS Accession No. ML17090A570).

2. LAR-17-010, Supplement 1, Vogtle Electric Generating Plant, Units 3 and 4, Request for License Amendment and Exemption: Pipe Rupture Hazard and Flooding Analyses, dated August 21, 2017 (ADAMS Accession No. ML17233A325).

3. LAR-17-010, Supplement 2, Vogtle Electric Generating Plant, Units 3 and 4, Request for License Amendment and Exemption: Pipe Rupture Hazard and Flooding Analyses, dated October 9, 2017 (ADAMS Accession No. ML17282A014).

4. LAR-17-010, Supplement 3, Vogtle Electric Generating Plant, Units 3 and 4, Request for License Amendment and Exemption: Pipe Rupture Hazard and Flooding Analyses, dated November 1, 2017 (ADAMS Accession No. ML17305B507).

5. LAR-17-010, Supplement 4, Vogtle Electric Generating Plant, Units 3 and 4, Request for License Amendment and Exemption: Pipe Rupture Hazard and Flooding Analyses, dated December 1, 2017 (ADAMS Accession No. ML17335A762).

6. LAR-17-010, Supplement 4, Revision 1, Vogtle Electric Generating Plant, Units 3 and 4, Request for License Amendment and Exemption: Pipe Rupture Hazard and Flooding Analyses, dated December 15, 2017 (ADAMS Accession No. ML17349A928).

7. LAR-17-010, Supplement 5, Vogtle Electric Generating Plant, Units 3 and 4, Request for License Amendment and Exemption: Pipe Rupture Hazard and Flooding Analyses, dated January 3, 2018 (ADAMS Accession No. ML18003B082).

8. Vogtle Electric Generating Plant, Units 3 and 4, Updated Final Safety Analysis Report, Revision 6, dated June 15, 2017 (ADAMS Accession No. ML17172A218).

9. AP1000 Design Control Document, Revision 19, dated June 13, 2011 (ADAMS Accession No. ML11171A500).

10. Final Safety Evaluation Report for Vogtle Electric Generating Plant, Units 3 and 4, Combined License Application, dated August 5, 2011 (ADAMS Accession No. ML110450302).

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

12. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants" (ADAMS Accession No. ML070290283).

13. Regulatory Guide 1.29, "Seismic Design Classification for Nuclear Power Plants" (ADAMS Accession No. ML070310052).

14. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3 (ADAMS Accession No. ML003740282).

15. Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components in Light Water Cooled Nuclear Power Plants" (ADAMS Accession No. ML013100305).

16. Regulatory Guide 1.189, Revision 2, "Fire Protection for Nuclear Power Plants" (ADAMS Accession No. ML092580550).

17. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (ADAMS Accession No. ML070660036).

18. NUREG-1764, "Guidance for the Review of Changes to Human Actions" (ADAMS Accession No. ML072640413).

19. IEEE Standard 603–1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," 1991.

20. American Society for Testing and Materials A193, "Standard Specification for Alloy-Steel and Stainless Steel Bolting for High temperature or High Pressure Service."

21. NFPA 20, "Standard for the Installation of Stationary Pumps for Fire Protection," 2007.

22. IEEE Std. 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations."

23. ACI 349-01, "Code Requirements for Nuclear Safety Related Concrete Structures."

24. AISC N690, “Specification for Safety-Related Steel Structures for Nuclear Facilities.”

25. WCAP-16675-P, “AP1000 Protection and Safety Monitoring System Architecture Technical Report,” Revision 5 (Proprietary).