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Vogtle Electric Generating Plant Units 1 and 2
License Amendment Request to Revise Technical Specification
Section 5.5.17 "Containment Leakage Rate Testing Program"
Response to NRC Request for Additional Information

Ladies and Gentlemen:

On September 12, 2017, Southern Nuclear Operating Company (SNC) submitted a license amendment request (LAR) to revise Vogtle Electric Generating Plant, Unit 1 and Unit 2, Technical Specifications (TS) 5.5.17, "Containment Leakage Rate Testing Program." On February 8, 2018, the Nuclear Regulatory Commission (NRC) staff, upon a determination that additional information was needed to complete its review, issued a request for additional information (RAI) letter. Enclosed is SNC's response.

This letter contains no NRC commitments. If you have any questions, please contact Ken McElroy at 205.992.7369.

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

Executed on April 5, 2018.

Cheryl Gayheart
Regulatory Affairs Director

efb/cbg

Enclosure:
1. Response to NRC Request for Additional Information

cc: NRC Regional Administrator, Region II
NRC NRR Project Manager – Vogtle Unit 1 and 2
NRC Senior Resident Inspector – Vogtle Unit 1 and 2
SNC Records RTYPE: CVC-7000

Vogtle Electric Generating Plant Units 1 and 2
License Amendment Request to Revise Technical Specification Section 5.5.17
"Containment Leakage Rate Testing Program"

Enclosure 1

Response to NRC Request for Additional Information

NRC RAI 1:

Section 2.5.3 of Regulatory Guide (RG) 1.174, Revision 2 (ADAMS Accession No. ML100910006), states:

> The development of the PRA [probabilistic risk assessment] model is supported by the use of models for specific events or phenomena. In many cases, the industry's state of knowledge is incomplete, and there may be different opinions on how the models should be formulated. Examples include approaches to modeling human performance, common-cause failures, and reactor coolant pump seal behavior upon loss of seal cooling. This gives rise to model uncertainty.

Regarding model uncertainty, Section 2.5.3 of RG 1.174, Revision 2, states:

...The impact of using alternative assumptions or models may be addressed by performing appropriate sensitivity studies or by using qualitative arguments, based on an understanding of the contributors to the results and how they are impacted by the change in assumptions or models.

In addition, Section 2.5.5 states:

> [I]n general, the results of the sensitivity studies should confirm that the guidelines are still met even under the alternative assumptions (i.e., change generally remains in the appropriate region)."

In Section 3.3.2 of the license amendment request (LAR), the licensee stated:

> The Westinghouse reactor coolant pump (RCP) shutdown seals have been installed at [Vogtle Electric Generating Plant] VEGP, and are credited in the PRA.

Please provide the following information to validate and confirm the PRA technical acceptability for use in the risk evaluation performed to support the requested permanent 15-year integrated leak rate test (ILRT) extension.

a. Provide a summary of the PRA modeling of the RCP shutdown seals, addressing the following aspects:

    1. Specify the PRA models (e.g., internal events, fire, seismic) that credit the RCP shutdown seals.

    2. Describe the PRA modeling of the RCP shutdown seals. Demonstrate how any limitations and conditions delineated in the NRC-approved guidance are being met [e.g., those in Section 3 of Topical Report PWROG-14001-P, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal," and Section 5 of NRC safety evaluation (SE) for PWROG-14001-P (ADAMS Accession Number ML17200A116)].

3. Indicate, and provide justification, whether the incorporation of the RCP shutdown seals into the PRA model is PRA maintenance or PRA upgrade, as defined in Section 1-5.4 of the American Society of Mechanical Engineers/ American Nuclear Society, ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2 (ADAMS Accession No. ML090410014). This discussion should be of sufficient detail to allow NRC staff to independently assess whether this change is a PRA maintenance or PRA upgrade (e.g., summarize the original method in the PRA and the new method, summarize the impact that this change has on significant accident sequences or the significant accident progression sequences).

b. If PRA modeling of the RCP shutdown seals is considered a PRA upgrade and a peer review(s) was performed for this upgrade, then discuss this peer review(s). In this discussion, describe the peer review process applied to the shutdown seal model; identify the guidance used to perform this peer review(s) (e.g., ASME/ANS RA-Sa-2009, Nuclear Energy Institute NEI 05-04, RG 1.200, Revision 2); include any necessary gap- or self-assessments if current guidance/standards were not used in the peer review(s); provide all facts and observations (F&Os) characterized as findings from the peer review(s) and the associated dispositions as it pertains to this application.

c. If PRA modeling of the RCP shutdown seals is considered a PRA upgrade and a peer review was not performed for this upgrade, then perform an appropriate sensitivity and/or bounding analysis for the RCP shutdown seals (e.g., remove credit for RCP shutdown seals) that assesses the contribution of risk for permanently extending the ILRT to 15 years. This analysis should also address the below consideration on the regulatory guides related to fire PRA (FPRA). Discuss this sensitivity/bounding analysis and provide updated risk values that include the increase in total large early release frequency (LERF), change in LERF (ΔLERF), population dose rate (PDR), and conditional containment failure probability (CCFP) for each unit to assess the risk impact. Confirm that the results of this analysis still meet the acceptance guidelines in RG 1.174, Revision 2, and Electric Power Research Institute EPRI Technical Report 1009325, Revision 2-A. If the acceptance guidelines are exceeded, then provide qualitative or quantitative arguments, based on an understanding of the contributors to the results and how they are impacted by the change in assumptions or models, to support the conclusion of the LAR. This discussion should include which metrics are exceeded and the conservatisms in the analysis and the risk significance of these conservatisms.

   If the FPRA applied in the sensitivity/bounding analysis does not incorporate the recently approved FPRA methodologies below, provide a detailed justification for why the integration of these NRC-approved FPRA methods and studies would not change the conclusions of the LAR.

   - The NRC issued a letter dated June 21, 2012, "Recent Fire PRA Methods Review Panel Decisions and EPRI 1022993, 'Evaluation of Peak Heat Release Rates in Electrical Cabinet Fires'" (ADAMS Accession No. ML12171A583), providing staff positions on (1) frequencies for cable fires initiated by welding and cutting, (2) clarifications for transient fires, (3)

alignment factor for pump oil fires, (4) electrical cabinet fire treatment refinement details, and (5) the EPRI 1022993 report.

- The NRC published NUREG/CR-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)," Volume 2 (ADAMS Accession No. ML14141A129), which is supported by a letter from the NRC to NEI, "Supplemental Interim Technical Guidance on Fire-Induced Circuit Failure Mode Likelihood Analysis," and supplemented in April 2014 (ADAMS Accession Nos. ML14017A135 and ML14086A165).

- The NRC published NUREG-2169, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database: United States Fire Event Experience Through 2009" (ADAMS Accession No. ML15016A069).

- Guidance on the credit taken for very early warning fire detection system (VEWFDS) is available in NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities, (DELORES-VEWFIRE)" (ADAMS Accession No. ML16343A058). The guidance provided in FAQ 08-0046, "Closure of National Fire Protection Association 805 Frequently Asked Question 08-0046 Incipient Fire Detection Systems" (ADAMS Accession No. ML093220426), has been rescinded.

SNC Response to RAI 1.a.1:

The PRA models that credit the RCP shutdown seals are: internal events including flooding (IEIF) and seismic (SPRA).

SNC Response to RAI 1.a.2:

The RCP shutdown seals (SDS) were modeled in the VEGP PRA models (IEIF and SPRA) by adding events and operator actions with corresponding human error probabilities consistent with Topical Report PWROG-14001-P, Rev. 1 and the associated NRC safety evaluation including the Limitations and Conditions in Section 5. Limitations and Conditions 2, 4, and 5 were addressed either with explicit modeling in a sensitivity study model or in a revised Model of Record (MOR) as explained below.

Limitation and Condition 2

Limitation and Condition 2 is addressed in the VEGP PRA models (IEIF and SPRA) as follows:

- For the VEGP IEIF model, a sensitivity study was performed to determine the CDF and LERF impacts of RCP seal LOCAs if the rated temperature of the RCP shutdown seal is exceeded. The sensitivity study results showed an increase of 13.6% in Core Damage Frequency (CDF) and an increase of 6.2% in Large Early Release Frequency (LERF) in comparison to the IEIF values submitted in SNC's September 12, 2017 LAR.

- The VEGP SPRA model was revised after the September 12, 2017 ILRT LAR submittal was transmitted to the NRC. The revised SPRA model incorporates the impacts of RCP seal LOCA scenarios if the rated temperature of the RCP shutdown seal is exceeded.

An initial analysis, performed by Westinghouse for asymmetric cooling based on the most bounding case, concluded that the temperature in the idle loop cold leg, where the steam generator (SG) is not getting adequate feed flow, will eventually exceed the temperature limitation. The analysis did not evaluate the time when the idle loop cold leg would exceed the limitation, but instead concluded that if cooldown of the reactor coolant system (RCS) via the secondary side was initiated before the SG in the idle loop dried out, the cold leg temperature in the idle loop will remain below the temperature limitation. The assumption for dry out time of the idle loop SG was 45 minutes.

Information from the Westinghouse analysis, which was not based on MAAP, was used in modeling the impact of asymmetric cooling in the VEGP IEIF sensitivity analysis model and the revised VEGP SPRA model. For these VEGP PRA models, additional time was assumed, beyond the 45 minutes, to account for a) time for the water in the idle cold leg to heat up after the SG dried out; and b) time for the polymer ring in the RCP shutdown seal to heat up and lose its material properties. Consequently, the VEGP IEIF sensitivity analysis model and the revised SPRA model were based on the more realistic assumption that if the operator failed to initiate cooldown within 1 hour, the RCP shutdown seal would fail.

Lastly, if the IEIF CDF and LERF shown in Table 6.2 of Attachment 1 of the September 12, 2017 LAR increased 13.6% and 6.2% respectively by the sensitivity analysis values, and the Seismic CDF and LERF values in Table 6.2 were replaced with the values of 3.98E-06/yr and 1.38E-07/yr from the revised Seismic PRA which also includes revisions to address Condition #2, then the total CDF and LERF values would remain below the NRC RG 1.174 acceptance criteria of 1E-04/yr for CDF and 1E-05/yr for LERF.

Limitation and Condition 4

Limitation and Condition 4 is not applicable because the limitation is specifically for RCP model 93A while VEGP has RCP model 93A-1. Model 93A-1 seals directly on the shaft with no shaft sleeve O-ring to consider.

Limitation and Condition 5

Limitation and Condition 5 has been addressed by modeling plant-specific operator actions with corresponding human error probabilities in the IEIF and SPRA models as described in PWROG-14001-P, Rev. 1.

SNC Response to RAI 1.a.3:

Incorporation of the RCP shutdown seals into the VEGP PRA models is PRA maintenance as defined in ASME/ANS RA-Sa-2009 and qualified by RG 1.200, Revision 2.

The peer-reviewed VEGP IEIF PRA and SPRA did not include the Westinghouse Generation III low-leakage (shutdown) seals. However, the peer-reviewed PRA model did include an RCP seal leakage model (WOG 2000 model) to assess the plant response to events that result from

a total loss of cooling to the RCP seals. Implementation of the new low-leakage RCP seal model into the IEIF PRA was performed consistent with the PRA method, modeling, and framework that had already been peer-reviewed.

ASME/ANS RA-Sa-2009 defines a PRA upgrade as a new methodology, or a change in scope or change in capability, that impacts the significant accident sequences or the significant accident progression sequences. PRA maintenance is defined as changes within the framework of an existing model structure. The change in the seal leakage model is not a new methodology because the new seal leakage model is simply an expansion of the current peer-reviewed model with different failure probabilities and associated human actions. There is no change in the model scope because the equipment, dependencies, and types of accident sequences remain the same. Finally, there is no change in PRA modeling capability, i.e., the peer reviewed PRA model can still evaluate the risk associated with station blackout and total loss of cooling events related to RCP seal failures. Therefore, implementation of the new seal leakage model is a change implemented within the framework of the existing peer-reviewed PRA model structure.

The seal leakage model change is only a change in the expected seal leakages associated with the new seals. The framework of the model remains essentially the same, and the High Level and Supporting Requirements (HLRs) in the PRA Standard for the Technical Elements associated with RCP seal modeling (e.g., those within the Accident Sequence Analysis, Data Analysis, Human Reliability Analysis, and Quantification technical elements) will continue to be Met or Not Met regardless of implementation of the change from the WOG2000 RCP seal model to the shutdown seal model. The five HLRs of interest associated with quantification (HLR-QU-A, -B, -C, -D and -E) continue to be met. Although the lower seal failure rates affect the ordering of the associated accident sequences and reduce CDF and LERF overall, the associated sequences were not significantly changed and new sequences that were not already modeled in the PRA and peer-reviewed were not generated. Consequently, this change in the internal events PRA does not constitute a PRA upgrade and does not require a focused scope peer review.

Because the incorporation of the RCP shutdown seals into the VEGP PRA models is considered PRA maintenance, not PRA upgrade, parts b and c of NRC RAI #1 are not applicable.

NRC RAI 2:

Table 6-2, "Vogtle Units 1 and 2 Internal and External Events Summary," provided LERF values for Units 1 and 2. The LERF values for the Units 1 and 2 internal events PRA (IEPRA) are both 6.45E-09/year. For fire events, the LERF values for Units 1 and 2 are 1.39E-06/year and 1.56E-06/year, respectively. The LERF values across the IEPRA and FPRA hazards are approximately two orders of magnitude (i.e., 10^{-2}) in difference, while that for core damage frequency (CDF) is one order of magnitude. Confirm the LERF values across the IEPRA and FPRA hazards.

SNC Response to RAI 2:

VEGP has confirmed that the LERF values reported in the LAR are correct. VEGP FPRA LERF is much higher than VEGP IEPRA LERF because the FPRA LERF has a major contribution unique to FPRA and not IEPRA.

IEPRA LERF does not have contributions from Main Control Room (MCR) abandonment scenarios (MCRAB) (where operators should abandon the MCR due to fire). Indeed, 43% of the VEGP FPRA LERF is due to MCRAB due to fire. LERF from MCRAB scenarios is 9.59E-8 (for Unit 1). Containment isolation (to prevent LERF after core damage) is not credited in MCRAB scenarios because there are no directions for containment isolation in the remote shutdown panel (RSP) operation procedures which are used after MCR abandonment.

The MCRAB model only credits systems/components available at the RSP. The RSP procedure does not provide direction for containment isolation; therefore, containment isolation is assumed to fail in the MCRAB model. Also, LERF scenarios related to fire induced Multiple Spurious Operation (MSO) scenarios or common cause scenarios (e.g. failure of containment isolation valves due to fire) are major contributors for FPRA LERF. Such contributors are not in the IEPRA LERF.

RAI No. 3

In the section of the LAR entitled, "Assessment of PRA Model Technical Adequacy," the licensee stated that in 2013 "[a] significant upgrade in [Modular Accident Analysis Program] MAAP capabilities was initiated" for the IEPRA. Further review of the technical adequacy of the VEGP IEPRA does not readily identify whether a peer review in accordance with the ASME/ANS RA-Sa-2009 PRA standard, as qualified by RG 1.200, Revision 2, was performed to assess the upgrade in MAAP capabilities.

Provide a summary of the upgrade in MAAP capabilities and justify whether the "significant upgrade" constitutes a PRA upgrade in accordance with ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2. If the significant upgrade in MAAP capabilities constitutes a PRA upgrade, provide the results from the focused-scope peer review, including the associated F&Os and their dispositions for any impact to the ILRT application; otherwise, provide a quantitative evaluation (e.g., sensitivity or bounding analysis) of its effect until a focused-scope peer review can be completed. For this quantitative evaluation, confirm that the results of this analysis still meet the acceptance guidelines in RG 1.174, Revision 2, and EPRI Technical Report 1009325, Revision 2-A. If the acceptance guidelines are exceeded, then provide qualitative or quantitative arguments, based on an understanding of the contributors to the results and how they are impacted by the change in assumptions or models, to support the conclusion of the LAR. This discussion should include which metrics are exceeded and the conservatisms in the analysis and the risk significance of these conservatisms.

SNC Response to RAI 3:

The use of the term "upgrade" was improperly used to designate updates/changes to the MAAP version 4.0.5 parameter file. MAAP4.0.8 included many enhancements to the reactor core, reactor coolant system, containment models, engineered safeguards, and other miscellaneous models in MAAP4. These enhancements included new parameters which allow the user to more accurately model the physical phenomena and allow the user to view the accident progression to aid in interrupting the results. Updating MAAP 4.0.5 to MAAP 4.0.8 has a small, but insignificant, impact on the results. Therefore, the success criteria used in the PRA models have not been revised due to this update. Accordingly, the changes made to the MAAP parameter file are characterized as an update; and therefore, a focused-scope peer review was not performed.

RAI No. 4

In Table 6-2 of the LAR, a CDF value of 2.52E-06/year and a LERF value of 6.45E-09/year for Units 1 and 2, respectively, are provided for the internal events hazard. These CDF and LERF values are identical for each unit. Typically, differences in CDF and LERF results exist for multiple-unit plants, even if the differences are not significant. Accordingly, it is not clear whether the risk values reported in the LAR are the results of separate PRAs performed for each unit or whether PRAs were performed only for a given unit and assumed to represent both units.

a. If the PRAs were performed only for a given unit and assumed to represent both units, then for the internal events hazard justify that the PRA model is an adequate representation of Units 1 and 2. Include a discussion of systems, structures, and components that are shared between units and how these were implicitly or explicitly modeled.

b. If the PRAs were performed for each unit separately, briefly explain why the risk results are identical.

SNC Response to RAI 4:

VEGP has a single PRA logic model that represents both units.

VEGP, Unit 1 and Unit 2 are constructed almost identical. Unit 1 and Unit 2 PRA models were found to be identical during the Initial Plant Examination (IPE), therefore a single PRA model is used. Plant changes are reviewed during model updates to identify unit difference that may require creating separate unit PRAs. There are differences between the units, such as spent fuel pools size, room numbers for internal flooding or room cooling but these differing items are not explicitly modeled in the PRA. These differences are identified in appropriate documentation as needed.

Because Vogtle is a two-unit site, shared systems are evaluated for the potential to cause dual-unit initiating events. The shared systems have been reviewed and determined not to have the capability of causing a dual unit trip. The following are the VEGP Unit 1&2 shared systems:

Boron Recycle System (BRS): Processes reactor coolant effluent that can be readily reused as makeup. The BRS is designed to tolerate equipment faults with critical functions being met using two pieces of equipment so that the failure of one will, at most, reduce the capacity of the BRS but not completely shut it down. Because of the large surge capacity of the BRS, the occasional unavailability of the system can be tolerated for brief periods of time. Also, backup is provided by a portable filtration system and portable demineralizer system located in the radwaste processing facility. Thus, this system has no capability to cause a dual-unit trip.

Liquid Waste Processing System (LWPS): Controls, collects, processes, handles, stores, and disposes of liquid radioactive waste generated as the result of normal operation, including anticipated operational occurrences. The LWPS design can accept equipment malfunctions without affecting the capability of the system to handle both anticipated liquid waste flows and possible surge load due to excessive leakage. The LWPS services both units, but most equipment is not shared but is associated with either Unit 1 or Unit 2. This system has no capability to cause a dual-unit trip.

Gaseous Waste Processing System (GWPS): Collects, processes, and stores gaseous wastes generated by plant operations including anticipated operational occurrences. The GWPS services both units, but most equipment is not shared but is associated with either Unit 1 or Unit 2. This system has no capability to cause a dual-unit trip.

Solid Waste (Radwaste) System: Collects, processes, and packages all solid radioactive wastes generated because of normal plant operation, including anticipated operational occurrences. This system has no capability to cause a dual-unit trip.

Fire Protection System (FPS): Provides the equipment throughout the plant that can suppress a fire. Because the FPS runs throughout the plant, sprays/leaks from the FPS could affect sensitive electrical areas within a unit. Because most electrical equipment is associated with either Unit 1 or Unit 2, problems caused by spray/leaks are mostly limited to a single unit. However, this capability is assessed in the flooding PRA.

Compressed Air System (Instrument and Service Air): Provides normally filtered and dried compressed air for service outlets located throughout the plant and a continuous supply of filtered, dried, and essentially oil-free air for pneumatic instruments. Loss of Instrument Air (IA) is a special initiating event. There is a cross-connect between units for instrument air; however, loss of IA in one unit does not affect the other unit.

Auxiliary Gas System: Provides hydrogen, oxygen, and nitrogen gases to the plant systems as needed. This system has no capability to cause a dual-unit trip.

Plant Makeup Water Treatment System: Provides reactor makeup water to maintain the volume control tank level within a predetermined band and acts in conjunction with the chemical and volume control system (CVCS) to maintain the required reactor coolant system (RCS) boron concentration. This system has no ability to cause a dual unit trip because its function is supplementary and because operators would have ample time to respond to a system failure before a reactor trip would occur.

Plant Demineralized Water System: Receives water from the well water storage tank, processes this water to remove soluble and insoluble impurities and dissolved gases, and

provides for storage and transfer of the demineralized water. This system has no capability to cause a dual-unit trip.

Auxiliary Steam System: Provides all auxiliary steam to balance-of-plant systems prior to the generation and supply of steam from the nuclear steam system. Although the system performs no safety function, it is protected from over pressurization by the following: Isolation from the higher-pressure systems using double-block valves and vent valve of the higher-pressure rating safety-relief valves. This system has no capability to cause a dual-unit trip because most equipment is associated with either Unit 1 or Unit 2. Also, operators would have ample time to respond and mitigate a system failure before a reactor trip would occur.

Turbine-Generator Hydrogen Gas Cooling System: Controls the temperature rise of the generator rotor components by circulating hydrogen gas through water-cooled hydrogen coolers using single-stage fans attached to both ends of the field. A seal oil system controls leakage of hydrogen (and carbon dioxide during purging) from the generator shaft seals. This system has no capability to cause a dual-unit trip because most equipment is associated with either Unit 1 or Unit 2. Also, operators would have ample time to respond and mitigate a system failure before a reactor trip would occur.

Laboratory Gas and Liquid Supply System: Provides gases and liquids needed for operation of the shared plant chemistry laboratory. This system is completely off-line and has no capability to cause a dual-unit trip.

Fuel Handling Building HVAC System: Provides the fuel handling building (which is shared between Units 1 and 2) with ventilation and filtration to maintain a suitable atmosphere for personnel and equipment during normal operation. This system has no capability to cause a dual-unit trip.

Control Building Normal HVAC System: Provides a suitable environment for operating personnel during normal conditions. The system is not required to operate after an accident. This system has no capability to cause a dual-unit trip. Failure of the system will not compromise safety-related systems nor prevent safe shutdown.

Control Room HVAC System: Provides a suitable and safe environment for operating personnel and control equipment in the Unit 1 and Unit 2 control rooms, conference room, emergency storage room, record storage room, kitchen, janitor room, shift superintendent/shift clerk office, and control room toilet during normal, accident, and post-accident plant operations. Operators have ample time to respond and mitigate a system failure before a reactor trip would occur. Therefore, this system is very unlikely to cause a dual-unit trip.

Control Room Lighting System: Provides adequate illumination levels in the plant under all operating conditions. Lighting includes the following categories: Normal lighting, which provides required levels of illumination throughout the plant. Essential lighting, which provides required illumination levels throughout the plant upon loss-of-offsite power (LOSP). Emergency lighting, made up of fixed and portable units, which provides sufficient illumination in support of station blackout (SBO) in areas manned for a safe shutdown and for access and egress routes to and from all fire and safe shutdown areas. Emergency Lighting (Main Control Room, Remote Shutdown Panel, Diesel Generator Panel, and Auxiliary Feedwater Pump House Panel Areas) is normally energized. Power for the system is available from offsite sources or the onsite

source (standby diesel generators). Upon interruption of the ac power supply, the main control room units will continue to be supplied ac power from integral battery/inverter units. Emergency lighting units operate in the event of loss of ac power to the integral battery charging units. Under normal plant operating conditions, fixtures are not lit. Security lighting provides illumination for certain controlled areas. This system has no capability to cause a dual-unit trip.

Spent Fuel Cask Bridge Crane: Transports spent fuel casks between the railcar loading and unloading area and the spent fuel storage area. The crane may be in operation during normal plant operation and when the plant is shut down for refueling or maintenance. This system has no capability to cause a dual-unit trip.

Plant Communications System: Provides intra-plant communications and effective plant-to-offsite communications during normal, transient, fire, and accident conditions, including loss of offsite power. The communication system consists of the following subsystems:
1) Telephone/page system; 2) Private automatic branch exchange (PABX) system; and 3) Sound-powered system. These communication systems are independent of each other; therefore, a failure in one system does not degrade performance of the other systems. Communication systems are non-safety-related and serve no safety function. This system has no capability to cause a dual-unit trip.

Fuel Handling Machine: The fuel handling machine is a wheel-mounted walkway spanning the spent fuel pools; it carries a trolley-mounted electric hoist on an overhead structure. This machine is used for handling fuel assemblies within the fuel storage area by means of a long-handled tool suspended from the hoist. This system has no opportunity to cause a dual-unit trip because it is used primarily when a unit is shutdown.

Condensate Filter Demineralizer Spent Resin Dewatering System: Consists of the condensate filter demineralizer (CFD) system, the backwash recovery system, the spent resin disposal system, and the spent resin dewatering system. This system collects and dehydrates spent resins to prepare them for storage and disposal. This system has no opportunity to cause a dual-unit reactor trip.

CVCS Chiller: Cools down process stream during storage of boron on the resin and maintains the outlet temperature from the BTRS at or below 115° F during the release of boron. There is one CVCS chiller shared between Units 1 and 2. This system's function is supplemental and has no opportunity to cause a dual-unit trip.

4160 VAC Non-Safety-Related Electrical Distribution System: The 4160 VAC system for each generating unit consists of a Class 1E 4160 VAC system, which performs a safety-related function, and a non-Class 1E 4160 VAC system, which performs no safety-related function. The principal function of the 4160 VAC system is to distribute electrical power to the plant 4160 VAC Class 1E and non-Class 1E loads. Non-Class 1E buses common to both units have supply sources from both units. The two supply breakers, one from each unit, are electrically interlocked so that only one can be closed at a time. While failures in the non-Class 1E 4160 VAC system can cause single unit trips, the interlocks preventing tying both units to common buses at the same time will prevent loss of a bus from causing a dual-unit trip.

Switchyard: Connects the plant generator and electrical systems to the grid. Losses of offsite power caused by switchyard-related and grid-related failures are included in the LOSP initiating

event. Even though switchyard/grid failure can affect both units simultaneously, the systems for mitigating the loss of power are associated with either Unit 1 or Unit 2. Cross-connections exist, but are used only if all power sources for one-unit fail.

Intake Structure: The intake structure at Vogtle is shared between the two units and serves as a backup to the NSCW well supply. Because river water taken in through the intake structure backs up the NSCW well supply and there is over a 30-day supply of water in the wells, loss of the intake structure would not result in a dual-unit trip.

Non-Nuclear Service Water: Non-nuclear service water is shared between the two Vogtle units and supplies cooling water to balance-of-plant components. Loss of the non-nuclear cooling water system will not result in a dual-unit trip