Southern Nuclear

Cheryl A. Gayheart
Regulatory Affairs Director

3535 Colonnade Parkway
Birmingham, AL 35243
205 992 5316 tel
205 992 7795 fax

cagayhea@southernco.com

OCT 2 1 2019

Docket Nos.: 50-424
50-425

NL-19-1193

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Vogtle Electric Generating Plant - Units 1 & 2
Revision 22 to the Updated Final Safety Analysis Report, Technical Specification Bases Changes, Technical Requirements Manual Changes,
10 CFR 50.59 Summary Report, and Revised NRC Commitments Report

Ladies and Gentlemen:

In accordance with 10 CFR 50.4(b) and 50.71(e), Southern Nuclear Operating Company (SNC) hereby submits Revision 22 to the Vogtle Electric Generating Plant (VEGP) Units 1 and 2 Updated Final Safety Analysis Report (UFSAR). The revised VEGP Units 1 and 2 UFSAR pages, indicated as Revision 22, reflect changes through September 30, 2019.

The VEGP Units 1 and 2 Technical Specifications, Section 5.5.14, "Technical Specifications (TS) Bases Control Program," provides for changes to the Bases without prior NRC approval. In addition, TS Section 5.5.14 requires that Bases changes made without prior NRC approval be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). Pursuant to TS 5.5.14, SNC hereby submits a complete copy of the VEGP TS Bases. The revised VEGP TS Bases pages, indicated as Revision 57, reflect changes to the TS Bases through September 30, 2019.

In accordance with Regulatory Issue Summary (RIS) 2001-05, "Guidance on Submitting Documents to the NRC by Electronic Information Exchange or on CD-ROM," all of the current pages of the VEGP Units 1 and 2 UFSAR, the VEGP Units 1 and 2 UFSAR reference drawings, the TS Bases, and the Technical Requirements Manual (TRM) are being submitted on CD-ROM in portable document format (PDF). The revised VEGP Units 1 and 2 TRM pages, indicated as Revision 49, reflect changes to the TRM through September 30, 2019.

In accordance with 10 CFR 50.59(d)(2), SNC hereby submits the 10 CFR 50.59 Summary Report containing a brief description of any changes, tests, or experiments, including a summary of the safety evaluation of each. This report is based on the same time period as Revision 22 of the UFSAR.

In accordance with NEI 99-04, "Guidelines for Managing NRC Commitment Changes," Revision 0, SNC reviewed its Commitment Database and identified no commitment changes for the applicable reporting period (March 1, 2018 to September 30, 2019).

A053
IE47
NRR

Enclosure 1 provides a table of contents with associated file names for the set of three CD-ROMs (Enclosure 2). Enclosure 3 provides the 10 CFR 50.59 Summary Report.

This letter contains no NRC commitments. If you have any questions, please contact Jamie Coleman at (205) 992-6611.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 18TH day of October 2019.

Respectfully submitted,

Cheryl Gayheart
Regulatory Affairs Director

CAG/TLE/scm

Enclosures:
1. CD-ROM Table of Contents
2. CD-ROMs (3 discs) containing Files 001 – 029
3. 10 CFR 50.59 Summary Report

cc: Regional Administrator, Region II (w/o enclosures)
Senior NRR Project Manager – VEGP Units 1 and 2 (w/o enclosures)
Senior Resident Inspector – VEGP Units 1 and 2 (w/o enclosures)
INPO Emergency Management Manager (Enclosure 2, CD ROMs, only)
RType: CVC7000

NL-19-1193

Vogtle Electric Generating Plant, Units 1 & 2
Revision 22 to the Updated Final Safety Analysis Report,
Technical Specification Bases Changes, Technical Requirements Manual
Changes, 10 CFR 50.59 Summary Report, and Revised NRC Commitments Report

Enclosure 1
CD-ROM Table of Contents

SEQ	FILENAME CONTENT	EXTENSION
	DISC 1	
	VEGP-File Nomenclature	.doc
001	VEGP FSAR_CH1, CH2 (PRT 1) Chapter 1 Chapter 2 (PRT 1) Section 2.1 to 2.3 to Figure 2.3.5-4	.pdf
002	VEGP FSAR_CH2 (PRT 2) & CH3 (PRT 1) Chapter 2 (PRT 2) Figure 2.3.5-5 to 2.3.5-6 Section 2.4 Table 2.4.1-1 to Table 2.4.13-1 Figures 2.4.1-1 Section 2.5 Appendix 2A Appendix 2B Chapter 3 (PRT 1) Section 3.1 to 3.6 Section 3.7 to Figure 3.7.B.2-24	.pdf
003	VEGP FSAR_CH3 (PRT 2) Chapter 3 (PRT 2) Figures 3.7.B.2-25 to 3.7.4-1 Section 3.8 to 3.11 Appendix 3A to 3C Appendix 3D to Figure 3D-17	.pdf
004	VEGP FSAR_CH3 (PRT 3) Chapter 3 (PRT 3) Appendix 3D Figures 3D-18 to 3D-40	.pdf
005	VEGP FSAR_CH3 (PRT 4) Chapter 3 (PRT 4) Appendix 3D Figures 3D-41 to 3D-65	.pdf
006	VEGP FSAR_CH3 (PRT 5) Chapter 3 (Part 5) Appendix 3D Figure 3D-66 to 3D-92	.pdf
007	VEGP FSAR_CH 3 (PRT 6) Chapter 3 (Part 6) Appendix 3D-93 to 3D-136	.pdf

SEQ	FILENAME CONTENT	EXTENSION
008	VEGP FSAR_CH 3 (PRT 7) & CH 4 (PRT 1)	.pdf
	Appendix 3D-137 to 3D-148	
	Appendix 3E	
	Appendix 3F	
	Chapter 4 (PRT 1)	
	Section 4.1 to 4.2	
	Section 4.3	
009	VEGP FSAR_CH 4 PRT 2, CH5, & CH6 (PRT 1)	.pdf
	Chapter 4 (PRT 2)	
	Tables 4.3-1 to 4.3-12	
	Figures 4.3-1 to 4.3-57	
	Section 4.4	
	Section 4.5	
	Section 4.6	
	Appendix 4A	
	Chapter 5	
	Chapter 6 (PRT 1)	
	Section 6.1	
	Section 6.2 to Figure 6.2.1-15 (SH 14)	
010	VEGP FSAR_CH6 (PRT 2)	.pdf
	Figure 6.2.1-15 (SH 15) to 6.2-21 (SH 74)	

DISC 2

SEQ	FILENAME CONTENT	EXTENSION
011	VEGP FSAR_CH6 (PRT 3) & CH7 (PRT 1)	.pdf
	Figure 6.2.1-22 (SH 1) to 6.2.5-7	
	Section 6.3 to Section 6.6	
	Chapter 7 (PRT 1)	
	Section 7.1 to 7.5	
	Section 7.6 to Figure 7.6.4-1	
012	VEGP FSAR_CH7 (PRT 2), CH8 (PRT 1)	.pdf
	Chapter 7 (PRT 2)	
	Figure 7.6.5-1	
	Section 7.7 to Figure 7.7.2-2	
	Chapter 8 (PRT 1)	
	Section 8.1 to 8.2	
	Section 8.3 to Figure 8.3.1-1 (SH 4)	

SEQ	FILENAME CONTENT	EXTENSION
013	VEGP FSAR_CH8 (PRT 2) Chapter 8 (PRT 2) Figure 8.3.1-1 (SHs 5 thru 24)	.pdf
014	VEGP FSAR_CH 8 (PRT 3), CH 9, CH 10 (PRT 1) Figure 8.3.1-1 (SHs 25 thru 35) Section 8.4 Chapter 9 Section 10. 1 Section 10.2 to Figure 10.2.2-2b	.pdf
015	VEGP FSAR CH10 (PRT2), CH11, CH12, CH13, CH14, CH 15 (PRT 1) Section 10.3 Section 10.4 Appendix 10A Chapter 11 Chapter 12 Chapter 13 Chapter 14 Chapter 15 to Figure 15.1.5-11	.pdf
016	VEGP FSAR CH15 (PRT2), CH16, CH17, CH18, CH19, TABLE OF CONTENTS, REV 22 EFFECTIVE PAGE LIST	.pdf
017	VEGP TECHNICAL REQUIREMENTS MANUAL	.pdf
018	VEGP BASES	.pdf

DISC 3

SEQ	FILENAME CONTENT	EXTENSION
019	VEGP FSAR REF DWGS PART 1 (1K5-1305-058-01 thru 1X3D-BD-J02D)	.pdf
020	VEGP FSAR REF DWGS PART 2 (1X3DG001 thru 1X4DB150-1)	.pdf
021	VEGP FSAR REF DWGS PART 3 (1X4DB151-1 thru 1X4DB203)	.pdf
022	VEGP FSAR REF DWGS PART 4 (1X4DB205-1 thru 1X4DE507)	.pdf

SEQ	FILENAME CONTENT	EXTENSION
023	VEGP FSAR REF DWGS PART 5 (1X4DE508 thru 1X6AA02-00238)	.pdf
024	VEGP FSAR REF DWGS PART 6 (1X6AA02-00239 thru 2X4DB170-2)	.pdf
025	VEGP FSAR REF DWGS PART 7 (2X4DB174-2 thru AX1D11A04-1)	.pdf
026	VEGP FSAR REF DWGS PART 8 (AX1D11A04-3 thru AX4DB241)	.pdf
027	VEGP FSAR REF DWGS PART 9 (AX4DB242-1 thru AX4DJ8047)	.pdf
028	VEGP FSAR REF DWGS PART 10 (AX4DJ8048 thru AX6DD303)	.pdf
029	VEGP FSAR REF DWGS PART 11 (AX6DD304 thru CX5DT101-58)	.pdf

NL-19-1193

Vogtle Electric Generating Plant, Units 1 & 2
Revision 22 to the Updated Final Safety Analysis Report,
Technical Specification Bases Changes, Technical Requirements Manual
Changes, 10 CFR 50.59 Summary Report, and Revised NRC Commitments Report

Enclosure 2
CD-ROMs (3 discs) containing Files 001 – 029

NL-19-1193

Vogtle Electric Generating Plant, Units 1 & 2
Revision 22 to the Updated Final Safety Analysis Report,
Technical Specification Bases Changes, Technical Requirements Manual
Changes, 10 CFR 50.59 Summary Report, and Revised NRC Commitments Report

Enclosure 3
10 CFR 50.59 Summary Report

10 CFR 50.59 Summary Report

Activity: Various (related to RER SNC918782)

Title: RWST Back-leakage Operator Actions

10 CFR 50.59 Evaluation Summary:

RER SNC918782 was generated as part of the re-evaluation of IN 91-56, "Potential Radioactive Leakage to Tank Vented to Atmosphere." The return of highly contaminated sump water to the RWST (which is vented to atmosphere) after a Loss of Coolant Accident (LOCA) challenged the allowable dose rates for onsite and offslte areas. This RER provides the technical justification for the determination of Emergency Core Cooling System (ECCS) flow paths to the RWST susceptible to contaminated back-leakage following a LOCA once the ECCS swaps over to cold leg recirculation. In the RER, flow paths deemed susceptible to contaminated back-leakage to the RWST were identified and evaluated. Recommended actions for the susceptible flow paths were to add specific valves to the Vogtle IST program and periodically quantify the leakage rates past these valves.

Specifically, contaminated back-leakage to the RWST from the Safety Injection (SI) and Centrifugal Charging (CC) recirculation lines presented a greater challenge to Control Room and offsite dose rates. Since the water level in the RWST will be lower when swap-over to recirculation is required, back-leakage from these lines will empty into the RWST vapor space based on the elevation of the combined return header. This changes the dose calculations such that even a very small amount of leakage from this header will challenge onsite and offsite dose limits. Additionally, the analyses in X6CAJ.14 and in RER SNC949800, "Dose Impact of Increased Post-LOCA RWST Back-leakage," implicitly assume this release path is isolated before entering recirculation mode 30 minutes after event initiation.

Manual action to shut the three SI pump mini-flow valves (HV-8813, HV-8814, and HV-8920) and the two isolation valves in each CC pump's alternate mini-flow line (HV-8508A/B and HV-8509A/B) is already credited in the Vogtle UFSAR (see Table 6.3.2-7). However, within Operations procedures, there are currently no steps to ensure that actions are taken to verify all these valves have actually shut before the ECCS must be completely swapped over to cold leg recirculation mode. This evaluation is for a change to Operations procedures to add steps ensuring that operators shut the SI and CC pump mini-flow valves prior to completion of swap-over to cold leg recirculation. This includes verbiage requiring that operators be dispatched to locally close any valves that are unable to be closed from the Control Room. Operators are trained to operate valves of this type and therefore there is minimal potential likelihood of inadequate manual closure of these valves.

This proposed activity does not change the frequency or likelihood of occurrence of any accident previously evaluated in the UFSAR. The added time-critical steps to direct operators to shut the mini-flow isolation MOVs are only applicable if plant conditions meet the entry conditions for the EOP and if Safety Injection is required. This administrative control prevents unnecessary or inadvertent manipulation of these MOVs within the procedure. Since shutting these MOVs is a credited action in the UFSAR as part of re-alignment for cold leg recirculation, there is no expected change in system response beyond what has already been evaluated. These verification steps do not logically introduce any new type of accident outside of what has already been evaluated within Vogtle's safety analysis.

Regarding the consequences of accidents evaluated In the UFSAR (specifically, large break LOCAs), the added verification steps help ensure assumptions made in the existing analyses are valid. The analysis of record for Control Room and offsite dose, X6CAJ.12, does not credit dose resulting from contaminated back-leakage to the RWST. Calculation X6CAJ.14 and RER SNC949B00, which take this release path into account, implicitly assume that the SIP and CCP mini-flow lines are not a credible leakage path back to the RWST (given the direction to isolate these lines in the emergency procedure and credited in the USFAR). However, since it is conceivable that a single failure could result in only one MOV in these miniflow lines providing isolation from leakage and none of these valves are periodically tested for leak tightness, this procedure change will help guarantee double valve isolation in these mini-flow lines prior to switch-over to recirculation. The single failure criteria for the plant's safety systems does not require that we account for secondary equipment failures unrelated to the first failure (per DC-1009). This means that even if one MOV is assumed to passively fail by have leakage past its seat (towards the RWST), then seat leakage past the next valve in series does not have be assumed. Thus, the SIP and CCP miniflow lines do not have be treated as credible leakage paths to the RWST and the assumptions for Control Room and offsite dose remain valid.

These procedure changes do not impact the design basis of the ECCS system, so there Is no impact to fission product barriers or their associated limits as described in the UFSAR.

Activity: Caution Tagout 1-CA-17-1204-00118

Title: 10 CFR 50.59 Evaluation for Caution Tagout 1-CA-17-1204-00118

10 CFR 50.59 Evaluation Summary:

Caution tagout 1-CA-17-1204-00118 identifies that Accumulator "B" vent path isolation valve, 12402U4576, has been closed due to leak-by through Accumulator Gas Vent Valve, 1HV0943B, when closed. The 10 CFR 50.59 screening (Ver. 2.0) of the tagout addressed the most-limiting event of a control room fire, during which, manual operator action would be required to enter Containment and manually open valve 12402U4576 before progression to cold shutdown could begin. It was determined this represented a change to procedure that adversely affected performance of a design function. Therefore, this evaluation addresses the vent path valve isolation, and the previously-unevaluated manual operator action that has been introduced to restore that vent path.

Since Accumulator venting is employed in response to an accident or for entry into a safe shutdown condition, the manual operator action would not cause an increase in the likelihood of a previously evaluated accident condition, or an accident of a different type. Applicable questions answered addressed the likelihood of component malfunction, and the potential consequences of that malfunction, introduced by the manual operator action.

References in the UFSAR and the Technical Specification Bases both allow time for actions to be taken external to the control room or shutdown panels to repair or otherwise restore unavailable equipment prior to beginning RCS depressurization. Operation procedures provide evidence of preparation for, and performance of, RCS depressurization being driven solely by manual operator actions and interfaces. It is also determined that personnel availability and Containment environmental conditions would not adversely affect the timely performance of manually opening valve 12402U4576. Therefore, it was determined that introduction of the manual operator action does not more than minimally increase the likelihood of equipment

malfunction, nor cause more severe or different consequences than those previously evaluated in the UFSAR. It is, therefore, determined that NRC approval of the manual operator action to enter Containment and manually open valve 12402U4576 prior to Accumulator and RCS depressurization is not required.

Activity: Caution Tagout 1-CA-18-1204-00140

Title: 10 CFR 50.59 Evaluation for Caution Tagout 1-CA-18-1204-00140

10 CFR 50.59 Evaluation Summary:

Caution Tagout 1-CA-18-1204-00140 has removed the Unit 1 Sludge Mixing Pump from service due to a degradation in the pump motor condition. Per UFSAR sections 6.2.2.2.2.3.1 and 6.3.2.2.9, the Sludge Mixing Pump and associated Recirculation Heater are credited for Refueling Water Storage Tank (RWST) freeze protection, and maintaining RWST temperature above 50°F. Technical Specification (TS) 3.5.4 sets a required action at a RWST temperature below 44°F. A 10 CFR 50.59 screening determined the tagout results in a change to a procedure that adversely affects performance of a design function.

Since the RWST is the primary source of water for Safety Injection (SI), and with SI only required in response to an accident, it is determined inadequate RWST temperature would not increase the frequency of an accident, nor cause an accident of a different type than evaluated in the UFSAR. Credit was taken for operator monitoring and the TS 3.5.4 eight-hour shutdown LCO requirement in minimizing the possibility of an SI malfunction, or changes in consequences or results of accidents, including fuel cladding impact. In summary, "No" was determined to be the appropriate response to all evaluation questions, indicating NRC approval is not required for the tagout to remain in place for more than ninety days.

Activity: Caution Tagout 1-CA-18-1204-00243(002)

Title: 10 CFR 50.59 Evaluation for Caution Tagout 1-CA-18-1204-00243(002)

10 CFR 50.59 Evaluation Summary:

Caution tagout 1-CA-18-1204-00243(002) identifies that Accumulator "B" vent path isolation valve, 12402U4576, has been closed due to leak-by through Accumulator Gas Vent Valve, 1HV0943B, when closed. The 10 CFR 50.59 screening of the tagout addressed the most-limiting event of a control room fire, during which, manual operator action would be required to enter Containment and manually open valve 12402U4576 before progression to cold shutdown could begin. It was determined this represented a change to procedure that adversely affected performance of a design function. Therefore, this evaluation addresses the vent path valve isolation, and the previously-unevaluated manual operator action that has been introduced to restore that vent path.

Since Accumulator venting is employed in response to an accident or for entry into a safe shutdown condition, the manual operator action would not cause an increase in the likelihood of a previously evaluated accident condition, or an accident of a different type. Applicable questions answered addressed the likelihood of component malfunction, and the potential consequences of that malfunction, introduced by the manual operator action.

References in the UFSAR and the Technical Specification Bases both allow time for actions to be taken external to the control room or shutdown panels to repair or otherwise restore unavailable equipment prior to beginning RCS depressurization. Operation procedures provide evidence of preparation for, and performance of, RCS depressurization being driven solely by manual operator actions and interfaces. It is also determined that personnel availability and Containment environmental conditions would not adversely affect the timely performance of manually opening valve 12402U4576. Therefore, it was determined that introduction of the manual operator action does not more than minimally increase the likelihood of equipment malfunction, nor cause more severe or different consequences than those previously evaluated in the UFSAR. It is, therefore, determined that NRC approval of the manual operator action to enter Containment and manually open valve 12402U4576 prior to Accumulator and RCS depressurization is not required.

Activity: Danger Tagout 2-DT-18-1204-00162(002)

Title: 10 CFR 50.59 Evaluation for Danger Tagout 2-DT-18-1204-00162(002)

10 CFR 50.59 Evaluation Summary:

Danger Tagout 2-DT-18-1204-00162(002) has removed the Unit 2 Sludge Mixing Pump from service due to a degradation in pump and motor conditions. Per UFSAR sections 6.2.2.2.2.3.1 and 6.3.2.2.9, the Sludge Mixing Pump and associated Recirculation Heater are credited for Refueling Water Storage Tank (RWST) freeze protection, and maintaining RWST temperature above 50°F. Technical Specification (TS) 3.5.4 sets a required action at a RWST temperature below 44°F. A 10 CFR 50.59 screening determined the tagout results in a change to a procedure that adversely affects performance of a design function.

Since the RWST is the primary source of water for Safety Injection (SI), and with SI only required in response to an accident, it is determined inadequate RWST temperature would not increase the frequency of an accident, nor cause an accident of a different type than evaluated in the UFSAR. Credit was taken for operator monitoring and actions, and the TS 3.5.4 eight-hour shutdown LCO requirement in minimizing the possibility of an SI malfunction, or changes in consequences or results of accidents, including fuel cladding impact. In summary, "No" was determined to be the appropriate response to all evaluation questions, indicating NRC approval is not required for the tagout to remain in place for more than ninety days.

Activity: Procedure 17019-1/2

Title: Major Procedure Revision to 17019-1/2 Regarding Operator Response to ALB19-B04 Turbine Condenser Low Vacuum / High Rate of Change Lowering

10 CFR 50.59 Evaluation Summary:

Summary:
The proposed change will modify the immediate operator action contained within 17019-1/2 relative to ALB19-B04 "Turbine Condenser Low Vacuum / High Rate of Change Lowering". Current guidance states that it is desired that operators immediately lower turbine load in response to the alarm in order to maintain the C-9 Interlock. The C-9 interlock arms the steam

dumps when condenser vacuum is above 24.7" Hg. The proposed guidance will give operations the discretion to allow the plant to operate below the C-9 setpoint which would render the steam dumps unavailable for the duration of operation below C-9. Section 15.2.2.1 describes the function of the "automatic turbine bypass system" or steam dumps relative to a generator load rejection response. The purpose of the steam dumps is to reject excess heat from the secondary loop in the event of a primary to secondary power mismatch in order to prevent primary loop transients. This procedural change would allow for operations to run without this automatic action in order to eliminate the risk associated with moving reactor power around due to environmental conditions when the low vacuum can be attributed to the "Circulating Water System" conditions. UFSAR chapter 15 clearly documents the plant response associated with a turbine trip or loss of load event with or without the steam dumps; therefore, this is not an adverse change requiring prior approval.

For reference:

Per UFSAR 7.7.1.8
The plant is designed to accept a 50-percent loss of net load without tripping the reactor. The automatic steam dump system is able to accommodate this abnormal load rejection and to reduce the effects of the transient imposed upon the reactor coolant system. By bypassing main steam directly to the condenser, an artificial load is thereby maintained on the primary system. The rod control system can then reduce the reactor temperature to a new equilibrium value without causing overtemperature and/or overpressure conditions. The steam dump steamflow capacity is 40 percent of full load steamflow at full load steam pressure.

Per UFSAR 10.4.4.1
10.4.4.1.1 Safety Design Bases
The turbine bypass system serves no safety function and has no safety design basis
1 0.4.4.1.2 Power Generation Design Bases

A. The turbine bypass system has the capacity to bypass 40 percent of the valves wide open (VWO) main steamflow to the main condenser.
B. The turbine bypass system is designed to bypass steam to the main condenser during plant startup and to permit a normal manual cooldown of the reactor coolant system from a hot shutdown condition to a point consistent with the initiation of residual heat removal system operation.
C. The turbine bypass system will permit a 50-percent electrical step-load reduction without reactor trip. The system under some upset conditions will also allow a turbine and reactor trip from full power without lifting the main steam relief and safety valves.

Activity: DECP SNC942152

Title: U1 & U2 NSCW Level Control Temporary Piping

10 CFR 50.59 Evaluation Summary:

The proposed activity involves design and guidance to provide an alternate method for making up to the NSCW Towers on Units 1 & 2 due to the normal level control piping and control valve described in the FSAR Section 9.2.5.5 not being available. This includes guidance to route temporary hose and fittings to supply makeup water from the normal NSCW Transfer Pumps. The activity does not increase the likelihood of an accident or the consequences of an accident.

There will be operational guidance provided with the Temporary Configuration Change to ensure that all NSCW Tower levels are maintained adequately during normal operating conditions to maintain Technical Specification level requirements.

Activity: DCP 1071817701

Title: Unit 1 Solid State Protection System (SSPS) Modifications

10 CFR 50.59 Evaluation Summary:

This Design Change Package (DCP) replaces Unit 1 Westinghouse Solid State Protection System (SSPS) obsolete components with Westinghouse designed and supplied replacement components. The components covered by this full evaluation are: the Undervoltage Driver, Universal Logic, Safeguards Driver, and Semi-Automatic Tester printed circuit boards (PCBs). The circuit boards are safety-related and seismic category 1. The nuclear safety and seismic classifications of the replacements are the same as the components being replaced. Implementation of these new-design replacement boards constitute a digital upgrade of the existing SSPS board because the boards utilize a CPLD to replace the MHTL logic devices used on the original-design boards. This design is a component level upgrade and not a system upgrade. All aspects of this digital replacement were evaluated and determined to result in no new or additional failure modes. The SSPS design was developed by Westinghouse under a safety-related Appendix B program. Westinghouse performed an analysis of all the circuits on the CPLD using the appropriate vendor supplied tool, with the intention of demonstrating that the testing that was already performed met the "testability" criteria in BTP 7-19, Section 1.9(2), in order to eliminate consideration of CCF. The analyses and testing are sufficiently rigorous and complete to eliminate consideration of CCF. No diverse system is required to address CCF of the CPLD based SSPS boards.

In section 4.0 of NRC Final Safety Evaluation (SE) for PWROG Topical Report WCAP-17867-P-A, Revision 1, "Westinghouse SSPS Board Replacement Licensing Summary Report", dated September 19, 2014, the SE states, "Based on the evaluations, audits, and technical reviews summarized in this Safety Evaluation, the NRC staff concludes that the new design SSPS boards can be used to replace the original design boards." Section 4.2 prescribes four (4) plant-specific action items related to atmospheric operating environment, lifetime total integrated dose, EMI/RFI levels, and actuation logic testing. Each of these action items is explicitly addressed in this 10 CFR 50.59 Evaluation as well as the DCP implementation requirements. Section 4.0 also states, "The NRC staff also finds that the unique configuration of each plant makes it important that each licensee analyze whether the new design boards can be installed under 10 CFR 50.59 without prior NRC approval." This Vogtle Nuclear Plant Unit 1 specific 10 CFR 50.59 Evaluation provides a documented basis for implementation of the Westinghouse new design SSPS ULB, SGD, UVD and/or SAT boards in any combination of new and original design SSPS boards without prior NRC approval.

All aspects of this change were evaluated for a change in all the parameters listed in Section C of this 50.59. Based on the evaluation above, following has been determined:

- Implementation of new-design SSPS replacement boards will not result in a more than a minimal increase in the frequency of occurrence of an accident previously evaluated, since no new accident initiators are being introduced, and the reliability of the replacement boards was determined to exceed those of the existing boards based on a MTBF calculation.

- Implementation of new-design SSPS replacement boards does not increase consequences of an accident previously evaluated in the FSAR because the RTS and ESFAS will continue to respond as assumed in the accident analyses.
- It was also determined that implementation of new-design SSPS replacement boards will not cause more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety since the replacement boards was determined to be more reliable than the existing boards, and introduce no new system malfunctions as the result of any failures.
- Implementation of new-design SSPS replacement boards does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR.
- Implementation of new-design SSPS replacement boards will not have any adverse impact on other equipment and does not create the possibility of an accident of a different type than was previously evaluated in the FSAR.
- Implementation of new-design SSPS replacement boards does not increase the possibility for a malfunction of an SSC important to safety with a different result than was previously evaluated in the FSAR as the failure modes and effects determined that the replacement boards are essentially transparent to the plant, as compared with the original-design boards with respect to plant response at the system level to failures or malfunctions.
- The new-design SSPS replacement boards do not have any impact on the integrity of the fuel cladding, fuel pellet, reactor pressure coolant boundary or containment structure. Thus, this design change does not result in a design basis safety limit change or new transient challenge for a fission product barrier (i.e., numerical limiting value for controlling the integrity of the fuel cladding and pellet, reactor coolant pressure boundary and/or containment building) being revised or altered. In addition, the replacement boards will not alter nor affect the validity of the existing ANS Condition II, III and IV transient and accident analyses.
- Implementation of new-design SSPS replacement boards will not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.

Therefore, the activity to which this evaluation applies does not represent a change to the parameters that have already been evaluated in the FSAR.

Activity: DCP 2071817801

Title: Unit 2 Solid State Protection System (SSPS)

10 CFR 50.59 Evaluation Summary:

This Design Change Package (DCP) replaces Unit 2 Westinghouse Solid State Protection System (SSPS) obsolete components with Westinghouse designed and supplied replacement components. The components covered by this full evaluation are: the Undervoltage Driver, Universal Logic, Safeguards Driver, and Semi-Automatic Tester printed circuit boards (PCBs). The circuit boards are safety-related and seismic category 1. The nuclear safety and seismic classifications of the replacements are the same as the components being replaced. Implementation of these new-design replacement boards constitute a digital upgrade of the

existing SSPS board because the boards utilize a CPLD to replace the MHTL logic devices used on the original-design boards. This design is a component level upgrade and not a system upgrade. All aspects of this digital replacement were evaluated and determined to result in no new or additional failure modes. The SSPS design was developed by Westinghouse under a safety-related Appendix B program. Westinghouse performed an analysis of all the circuits on the CPLD using the appropriate vendor supplied tool, with the intention of demonstrating that the testing that was already performed met the "testability" criteria in BTP 7-19, Section 1.9(2), in order to eliminate consideration of CCF. The analyses and testing are sufficiently rigorous and complete to eliminate consideration of CCF. No diverse system is required to address CCF of the CPLD based SSPS boards.

In section 4.0 of NRC Final Safety Evaluation (SE) for PWROG Topical Report WCAP-17867-P-A, Revision 1, "Westinghouse SSPS Board Replacement Licensing Summary Report", dated September 19, 2014, the SE states, "Based on the evaluations, audits, and technical reviews summarized in this Safety Evaluation, the NRC staff concludes that the new design SSPS boards can be used to replace the original design boards." Section 4.2 prescribes four (4) plant-specific action items related to atmospheric operating environment, lifetime total integrated dose, EMI/RFI levels, and actuation logic testing. Each of these action items is explicitly addressed in this 10 CFR 50.59 Evaluation as well as the DCP implementation requirements. Section 4.0 also states, "The NRC staff also finds that the unique configuration of each plant makes it important that each licensee analyze whether the new design boards can be installed under 10 CFR 50.59 without prior NRC approval." This Vogtle Nuclear Plant Unit 1 specific 10 CFR 50.59 Evaluation provides a documented basis for implementation of the Westinghouse new design SSPS ULB, SGD, UVD and/or SAT boards in any combination of new and original design SSPS boards without prior NRC approval.

All aspects of this change were evaluated for a change in all the parameters listed in Section C of this 50.59. Based on the evaluation above, following has been determined:

- Implementation of new-design SSPS replacement boards will not result in a more than a minimal increase in the frequency of occurrence of an accident previously evaluated, since no new accident initiators are being introduced, and the reliability of the replacement boards was determined to exceed those of the existing boards based on a MTBF calculation.
- Implementation of new-design SSPS replacement boards does not increase consequences of an accident previously evaluated in the FSAR because the RTS and ESFAS will continue to respond as assumed in the accident analyses.
- It was also determined that implementation of new-design SSPS replacement boards will not cause more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety since the replacement boards was determined to be more reliable than the existing boards, and introduce no new system malfunctions as the result of any failures.
- Implementation of new-design SSPS replacement boards does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR.
- Implementation of new-design SSPS replacement boards will not have any adverse impact on other equipment and does not create the possibility of an accident of a different type than was previously evaluated in the FSAR.
- Implementation of new-design SSPS replacement boards does not increase the possibility for a malfunction of an SSC important to safety with a different result than was previously

evaluated in the FSAR as the failure modes and effects determined that the replacement boards are essentially transparent to the plant, as compared with the original-design boards with respect to plant response at the system level to failures or malfunctions.

- The new-design SSPS replacement boards do not have any impact on the integrity of the fuel cladding, fuel pellet, reactor pressure coolant boundary or containment structure. Thus, this design change does not result in a design basis safety limit change or new transient challenge for a fission product barrier (i.e., numerical limiting value for controlling the integrity of the fuel cladding and pellet, reactor coolant pressure boundary and/or containment building) being revised or altered. In addition, the replacement boards will not alter nor affect the validity of the existing ANS Condition II, III and IV transient and accident analyses.
- Implementation of new-design SSPS replacement boards will not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.

Therefore, the activity to which this evaluation applies does not represent a change to the parameters that have already been evaluated in the FSAR.

Activity: DCP SNC847051

Title: RAT 1A Open Phase Trip

10 CFR 50.59 Evaluation Summary:

The proposed activity will add a new function to the OPP panels and existing protective relaying for the RAT. The individual OPP panel isolates the monitored transformer on detection of a loss of phase on the high side (upstream) of that transformer. The OPP panel is an addition to existing protective relays which lockout (isolate) the transformer. The transformer is the offsite source of power to the onsite AC power distribution system during normal start-up, normal shutdown, and emergency shutdown. When the nuclear plant is producing power, the separate unit auxiliary transformers, sourced from the main generator, provide the nonemergency power to the plant. Thus, isolation of the RAT while the nuclear plant is producing power will not result in complete loss of the non emergency onsite AC power distribution system. Automatic bus transfer from the UAT to the RAT will occur based on residual voltage. If the SAT is in service (replacing one of the RATs), then automatic bus transfer does not occur.

Energizing the transformer generates a large inrush current. To avoid false indication of an OPC and inadvertent transformer lockout, procedures for energizing the transformer must be modified to turn off the OPP panel manually when energizing the monitored transformer and re-energized manually after energizing the transformer.

Modification of the OPP system at the Vogtle Nuclear Plant has been evaluated based on the requirements of 10 CFR 50.59 and following the guidance provided in NEI 96-07 and NEI 01-01. The conclusion of the Evaluation is that the proposed activities may be implemented under 10 CFR 50.59 without requiring prior US NRC review or approval.

Activity: DCP SNC847106

Title: RAT 1B Open Phase Trip

10 CFR 50.59 Evaluation Summary:

The proposed activity will add a new function to the OPP panels and existing protective relaying for the RAT. The individual OPP panel isolates the monitored transformer on detection of a loss of phase on the high side (upstream) of that transformer. The OPP panel is an addition to existing protective relays which lockout (isolate) the transformer. The transformer is the offsite source of power to the onsite AC power distribution system during normal start-up, normal shutdown, and emergency shutdown. When the nuclear plant is producing power, the separate unit auxiliary transformers, sourced from the main generator, provide the nonemergency power to the plant. Thus, isolation of the RAT while the nuclear plant is producing power will not result in complete loss of the non emergency onsite AC power distribution system. Automatic bus transfer from the UAT to the RAT will occur based on residual voltage. If the SAT is in service (replacing one of the RATs), then automatic bus transfer does not occur.

Energizing the transformer generates a large inrush current. To avoid false indication of an OPC and inadvertent transformer lockout, procedures for energizing the transformer must be modified to turn off the OPP panel manually when energizing the monitored transformer and re-energized manually after energizing the transformer.

Modification of the OPP system at the Vogtle Nuclear Plant has been evaluated based on the requirements of 10 CFR 50.59 and following the guidance provided in NEI 96-07 and NEI 01-01. The conclusion of the Evaluation is that the proposed activities may be implemented under 10 CFR 50.59 without requiring prior US NRC review or approval.

Activity: DCP SNC872971

Title: RAT 2A Open Phase Trip

10 CFR 50.59 Evaluation Summary:

The proposed activity will add a new function to the OPP panels and existing protective relaying for the RAT. The individual OPP panel isolates the monitored transformer on detection of a loss of phase on the high side (upstream) of that transformer. The OPP panel is an addition to existing protective relays which lockout (isolate) the transformer. The transformer is the offsite source of power to the onsite AC power distribution system during normal start-up, normal shutdown, and emergency shutdown. When the nuclear plant is producing power, the separate unit auxiliary transformers, sourced from the main generator, provide the nonemergency power to the plant. Thus, isolation of the RAT while the nuclear plant is producing power will not result in complete loss of the non emergency onsite AC power distribution system. Automatic bus transfer from the UAT to the RAT will occur based on residual voltage. If the SAT is in service (replacing one of the RATs), then automatic bus transfer does not occur.

Energizing the transformer generates a large inrush current. To avoid false indication of an OPC and inadvertent transformer lockout, procedures for energizing the transformer must be modified to turn off the OPP panel manually when energizing the monitored transformer and re-energized manually after energizing the transformer.

Modification of the OPP system at the Vogtle Nuclear Plant has been evaluated based on the requirements of 10 CFR 50.59 and following the guidance provided in NEI 96-07 and NEI 01-01. The conclusion of the Evaluation is that the proposed activities may be implemented under 10 CFR 50.59 without requiring prior US NRC review or approval.

Activity: DCP SNC872972

Title: RAT 2B Open Phase Trip

10 CFR 50.59 Evaluation Summary:

The proposed activity will add a new function to the OPP panels and existing protective relaying for the RAT. The individual OPP panel isolates the monitored transformer on detection of a loss of phase on the high side (upstream) of that transformer. The OPP panel is an addition to existing protective relays which lockout (isolate) the transformer. The transformer is the offsite source of power to the onsite AC power distribution system during normal start-up, normal shutdown, and emergency shutdown. When the nuclear plant is producing power, the separate unit auxiliary transformers, sourced from the main generator, provide the nonemergency power to the plant. Thus, isolation of the RAT while the nuclear plant is producing power will not result in complete loss of the non emergency onsite AC power distribution system. Automatic bus transfer from the UAT to the RAT will occur based on residual voltage. If the SAT is in service (replacing one of the RATs), then automatic bus transfer does not occur.

Energizing the transformer generates a large inrush current. To avoid false indication of an OPC and inadvertent transformer lockout, procedures for energizing the transformer must be modified to turn off the OPP panel manually when energizing the monitored transformer and re-energized manually after energizing the transformer.

Modification of the OPP system at the Vogtle Nuclear Plant has been evaluated based on the requirements of 10 CFR 50.59 and following the guidance provided in NEI 96-07 and NEI 01-01. The conclusion of the Evaluation is that the proposed activities may be implemented under 10 CFR 50.59 without requiring prior US NRC review or approval.

Activity: DCP SNC872975

Title: SAT Open Phase Trip

10 CFR 50.59 Evaluation Summary:

The proposed activity will add a new function to the OPP panels and existing protective relaying for the RAT. The individual OPP panel isolates the monitored transformer on detection of a loss of phase on the high side (upstream) of that transformer. The OPP panel is an addition to existing protective relays which lockout (isolate) the transformer. The transformer is the offsite source of power to the onsite AC power distribution system during normal start-up, normal shutdown, and emergency shutdown. When the nuclear plant is producing power, the separate unit auxiliary transformers, sourced from the main generator, provide the nonemergency power to the plant. Thus, isolation of the RAT while the nuclear plant is producing power will not result in complete loss of the non emergency onsite AC power distribution system. Automatic bus

transfer from the UAT to the RAT will occur based on residual voltage. If the SAT is in service (replacing one of the RATs), then automatic bus transfer does not occur.

Energizing the transformer generates a large inrush current. To avoid false indication of an OPC and inadvertent transformer lockout, procedures for energizing the transformer must be modified to turn off the OPP panel manually when energizing the monitored transformer and re-energized manually after energizing the transformer.

Modification of the OPP system at the Vogtle Nuclear Plant has been evaluated based on the requirements of 10 CFR 50.59 and following the guidance provided in NEI 96-07 and NEI 01-01. The conclusion of the Evaluation is that the proposed activities may be implemented under 10 CFR 50.59 without requiring prior US NRC review or approval.

Activity: DCP SNC889319

Title: Vogtle 2 Cycle 21 Reload Core and Associated COLR with FRAMATOME GAIA Lead Test Assemblies with EATF Features

10 CFR 50.59 Evaluation Summary:

The Vogtle 2 Cycle 21 Reload Core will be refueled with 88 fresh Westinghouse VANTAGE+ fuel assemblies and 101 previously burned VANTAGE+ fuel assemblies. The fresh VANTAGE+ fuel assemblies (OFA) have the same design features as the Cycle 20 fresh fuel assemblies. The enrichment and burnable poison loading is consistent with past cycles. Four FRAMATOME GAIA lead test assemblies (LTAs), including Enhanced Accident Tolerant Fuel (EATF) features, will be inserted in Vogtle Unit 2 Cycle 21. The EATF features include Chromia-doped pellets and full-length Chromium-coated fuel rod cladding. The cladding of all fuel rods is M5, an optimized Zirconium alloy. Each assembly will be composed of the following four fuel rod types:

- 228 rods incorporating M5 cladding, Chromia-doped pellets at 3.80 w/o U-235, zircaloy-4 end caps, and a nickel alloy plenum spring,
- 4 rods (located in the four corners of the assembly) incorporating Chromium-coated M5 cladding, Chromia-doped pellets at 3.80 w/o U-235, zircaloy-4 end caps, and a nickel alloy plenum spring,
- 20 rods incorporating M5 cladding, 8% Gadolinia pellets at 3.11 w/o U-235, blanket pellets at 2.60 w/o U-235, zircaloy-4 end caps, and a nickel alloy plenum spring, and
- 12 rods incorporating M5 cladding, 4% Gadolinia pellets at 3.19 w/o U-235, blanket pellets at 2.60 w/o U-235, zircaloy-4 end caps, and a nickel alloy plenum spring.

The Core Operating Limits Report (COLR) for Vogtle Unit 2 will be updated to reflect the new core design and cycle-dependent data.

The purpose of the LTA program is to collect data around not only the EATF features, but also the GAIA design as a whole in a high duty core. The data obtained through core monitoring and post-irradiation inspections will be used to enhance FRAMATOME codes and methods and facilitate the research and development of accident tolerant fuel.

In accordance with the guidance in NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation," Section 4.2.2, Example 4; the GAIA LTAs are conservatively being deemed a

test or experiment. Answering in this manner on the 50.59 screening required the performance of a 50.59 evaluation.

The GAIA LTAs have been evaluated for mechanical and functional compatibility including interfaces associated with reactor internals, handling and storage equipment, co-resident fuel assemblies, in-core instrumentation, non-fuel components, and auxiliary equipment. The GAIA LTAs have also been evaluated for hydraulic compatibility, normal operation component stresses and loads, and faulted component stresses and loads. In all cases the requirements for the GAIA LTAs, resident Westinghouse OFA fuel assemblies, and the overall reload core were met. The design of the reload core and GAIA LTAs are such that the GAIA LTAs have the neutronic equivalence of a resident OFA fuel assembly that would be sufficiently non-limiting with respect to power peaking. There are four (4) GAIA LTAs loaded into the core, which represents approximately two percent of the total core loading. The locations of the GAIA LTAs were chosen such that they would be non-rodded, further ensuring safety analyses are not adversely affected and power peaking remains non-limiting. These criteria are in compliance with Technical Specification 4.2.1. Along with the neutronic equivalence, the GAIA LTAs were shown to have mechanical equivalence to the Westinghouse RFA fuel design. The GAIA LTAs were modeled by Westinghouse in their evaluations as the RFA fuel design. Penalties were conservatively applied as appropriate due to the mixed core environment between the resident OFA fuel design and GAIA LTAs. The Westinghouse evaluations, which form the licensing basis for the reload core and associated COLR, demonstrated that all applicable design criteria and pertinent licensing basis acceptance criteria were met. Neither the reload core nor the associated COLR result in existing acceptable safety limits for any accident being exceeded, and do not result in any adverse changes. Confirmatory evaluations were also performed by FRAMATOME and SNC which demonstrated that no changes to the established design bases or safety analysis limits were necessitated by the utilization of the GAIA LTAs.

The GAIA LTAs will be utilized and controlled with the same handling, storage, and operational requirements as the resident Westinghouse OFA fuel assemblies. Operations in compliance with the limits specified in the COLR and Technical Specifications is sufficient to ensure that margin to the Specified Acceptable Fuel Design Limits (SAFDLS) is maintained, as required by the reactor core design basis.

Therefore, the responses to Questions 1-7 all result in “NO” answers. Question 8 was not addressed because NRC-approved methods were used to ensure compliance with the SAFDLS.