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

SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS

RELATED TO AMENDMENT NOS. 138 AND 137

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MEAG POWER SPVM, LLC

MEAG POWER SPVJ, LLC

MEAG POWER SPVP, LLC

CITY OF DALTON, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT UNITS 3 AND 4

DOCKET NOS. 52-025 AND 52-026

1.0 INTRODUCTION

By letter dated July 28, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17209A755) as supplemented by letters dated January 23 and March 23, June 21, and August 9, 2018 (ADAMS Accession Nos. ML18023A440, ML18082B370, ML18172A154, and ML18221A364, respectively), the Southern Nuclear Operating Company (SNC) requested that the Nuclear Regulatory Commission (NRC) amend Vogtle Electric Generating Plant (VEGP) Units 3 and 4, Combined License (COL) Numbers NPF-91 and NPF-92, respectively. License Amendment Request (LAR) 17-024 proposes changes to the COLs to depart from approved COL Appendix A, Technical Specifications (TS) related to rod control and other miscellaneous updates. The requested amendment proposes changes to revise COL Appendix A, plant-specific TS by modifying specified TS to make them consistent with the design, licensing basis, and other TS. The NRC staff's review of the LAR is included in this safety evaluation.

In the supplements dated January 23, March 23, June 21, and August 9, 2018, SNC provided additional information that supplemented the application. This information did not expand the scope of the application, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 5, 2017 (82 FR 57469).

2.0 REGULATORY EVALUATION

The LAR summarizes the changes as follows: The requested amendment proposes changes to revise COL Appendix A, plant-specific TS by modifying the TS to make them consistent with the design, licensing basis, and the TS.

The NRC staff considered the following regulatory requirements in reviewing the proposed LAR:

Title 10 of the *Code of Federal Regulations* (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 TS, or requires a license amendment under paragraphs B.5.b or B.5.c of the section. A departure was proposed to the TS, thus a license amendment is required.

10 CFR Part 52, Appendix D, Section VIII.C.6 states that after issuance of a license, "Changes to the plant-specific TS will be treated as license amendments under 10 CFR 50.90." 10 CFR 50.90 addresses applications for amendments of licenses, construction permits, and early site permits. As discussed above, a change to COL Appendix A was requested, and thus a license amendment is required.

10 CFR 50.36, "Technical specifications," impose limits, operating conditions, and other requirements upon reactor facility operation for the public health and safety. The TS are derived from the analyses and evaluations in the safety analysis report. TS must contain: (1) safety limits and limiting safety system settings; (2) limiting conditions for operation; (3) surveillance requirements (SR); (4) design features; and (5) administrative controls.

10 CFR Part 50, Appendix A, General Design Criterion (GDC) 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

10 CFR Part 50, Appendix A, GDC 13, "Instrumentation and control," requires that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

10 CFR Part 50, Appendix A, GDC 25, "Protection system requirements for reactivity control malfunctions," requires that the protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of the control rods.

10 CFR Part 50, Appendix A, GDC 26, "Reactivity control system redundancy and capability," requires that two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits

are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure that the acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

10 CFR Part 50, Appendix A, GDC 27, "Combined reactivity control systems capability," requires that the reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

10 CFR Part 50, Appendix A, GDC 28, "Reactivity limits," requires that the reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

10 CFR Part 50, Appendix A, GDC 29, "Protection against anticipated operational occurrences," requires that the protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

10 CFR Part 50, Appendix A, GDC 35, "Emergency core cooling," requires that a system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

The staff also considered the following guidance in reviewing the proposed amendment:

Regulatory Guide (RG) 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants."

3.0 TECHNICAL EVALUATION OF PROPOSED CHANGES

In LAR 17-024, SNC proposed changes to the COL Appendix A, plant-specific TS for VEGP Units 3 and 4 for the following "items" (designated in parentheses starting with the letter "L"):

- TS 1.1 Definitions - Shutdown Margin (L01)

  Change Shutdown Margin (SDM) definition c. from "In MODE 2 with k_{eff}<1.0, and MODES 3, 4, and 5, the worth of fully inserted Gray Rod Cluster Assemblies (GRCAs) will be included in the SDM calculation." to read "In MODE 2 with k_{eff}<1.0, and in MODES 3, 4, and 5, the worth of the verified fully inserted Gray Rod Cluster Assemblies (GRCAs) which have passed the acceptance criteria for GRCA bank worth measurements performed during startup physics testing may be included in the SDM calculation."

- TS 3.1.4 Rod Group Alignment Limits (L02A through L02M)

  (L02A) Change Limiting Condition of Operation (LCO) from "All shutdown and control rods shall be OPERABLE." to "Each rod cluster control assembly (RCCA) shall be OPERABLE."

  (L02B) Change LCO AND statement from "Individual indicated rod positions shall be within 12 steps of their group step counter demand position." to "Individual indicated rod positions of each RCCA and Gray Rod Cluster Assembly shall be within 12 steps of their group step counter demand position."

  (L02C) Delete LCO 3.1.4 note.

  (L02D) Change Action Condition A from "One or more rod(s) inoperable." to "One or more RCCA(s) inoperable."

  (L02E) Acronym defined in change to Required Action (RA) B.1 Completion Time from "1 hour with the OPDMS not monitoring parameters" to "1 hour with the On-Line Power Distribution Monitoring System not monitoring parameters."

  (L02F) Add RA B.2.3.1 where the RA will be to "Perform SR 3.2.5.1" with a Completion Time of "Once per 12 hours," OR perform B.2.3, which is renumbered as B.2.3.2.1.

  (L02G) Delete RA B.2.4 Note, and renumber the RA to B.2.3.2.2.

  (L02H) Delete RA B.2.5 Note, and renumber the RA to B.2.3.2.3.

  (L02I) Renumber RA B.2.6 to B.2.4.

  (L02K) Change SR 3.1.4.2 from "Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥10 steps in either direction." to "Verify rod freedom of movement (trippability) by moving each RCCA not fully inserted in the core ≥10 steps in either direction."

  (L02L) Delete the Note to SR 3.1.4.3

  (L02M) Change SR 3.1.4.3 from "Verify rod drop time of each rod…" to "Verify rod drop time of each RCCA…"

- TS 3.1.6 Control Bank Insertion Limits (L03)

  Change LCO 3.1.6, Note 2 from "This LCO is not applicable to Gray Rod Cluster Assembly (GRCA) banks during GRCA bank sequence exchange with On-Line Power Distribution Monitoring System monitoring parameters." to "This LCO is not applicable to Gray Rod Cluster Assembly (GRCA) banks for up to one hour during GRCA bank sequence exchange."

- TS 3.1.7 Rod Position Indication (L04)

  Delete RA B.2 and renumber the remaining Condition B RAs.

- Table 3.3.1-1 Reactor Trip System Instrumentation (L05)

  Change surveillance requirements for Function 4 Overpower ΔT by adding two new Surveillance Requirements SR 3.3.1.4 and SR 3.3.1.5.

- Table 3.3.1-1 Reactor Trip System Instrumentation (L06)

  Change Function 12 Passive Residual Heat Removal Actuation by deleting Surveillance Requirement SR 3.3.1.9.

- Table 3.3.5-1, Function 1; Function 2; Function 4 (L07)

  Change the Table 3.3.5-1, Function 4, Core Makeup Tank Actuation Input from Engineered Safety Feature Actuation System – Manual Required Channels from to "2 switch sets" to "2."

- Table 3.3.17-1 Post-Accident Monitoring Instrumentation (L08)

  1. Delete "Monitor" from Function 5 "RCS Subcooling Monitor."
  2. Add new Function 20 "Pressurizer Pressure," with Required Channels, "2," and Condition Referenced From Required Action D.1, "E."

- TS 3.3.19 Diverse Actuation System Manual Controls (L09)

  Change Note (c) for Table 3.3.19-1, "Diverse Actuation System Manual Controls" from "With reactor internals in place." to "With upper internals in place."

- TS 3.5.4 Passive Residual Heat Removal (PRHR) Heat Exchanger (HX)-Operating (L10)

  Change Surveillance Requirements SR 3.5.4.6 from "Verify both PRHR HX air operated outlet isolation valves and both In-Containment Refueling Water Storage Tank (IRWST) gutter isolation valves stroke open." to "Verify both PRHR HX air operated outlet isolation valves stroke open and both IRWST gutter isolation valves stroke closed."

- TS 3.8.3 Inverters – Operating (L11)

  1. Change Action Condition A. from "One inverter inoperable." to "One or two inverter(s) within one division inoperable."
  2. Change RA A.1 from "Restore inverter to OPERABLE status." to "Restore inverter(s) to OPERABLE status."

The staff's review of each proposed TS changes described above is provided in the following sections.

3.1 TS 1.1 Definitions - Shutdown Margin (L01)

TS 1.1, "Definitions," SHUTDOWN MARGIN (SDM) c. currently states:

> In MODE 2 with k_{eff} < 1.0, and MODES 3, 4, and 5, the worth of fully inserted Gray Rod Cluster Assemblies (GRCAs) will be included in the SDM calculation.

SNC proposes to change TS 1.1, SDM definition c. to:

> In MODE 2 with k_{eff} < 1.0, and MODES 3, 4, and 5, the worth of verified fully inserted Gray Rod Cluster Assemblies (GRCAs) which have passed the acceptance criteria for GRCA bank worth measurements performed during startup physics testing may be included in the SDM calculation.

SDM requirements provide sufficient reactivity margin to assure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences. SNC states in LAR 17-024 that, when SDM is calculated during power operations, it is conservative to ignore the effects of GRCAs on SDM upon reactor trip. When SDM is calculated in reactor shutdown and refueling MODES, the calculation of SDM can include the GRCAs, but is unnecessary for maintaining adequate SDM.

The staff reviewed the change to the SDM definition in TS 1.1 (c.) and, for the reasons set forth below, finds it acceptable. The change maintains the requirement for the SDM calculation while leaving the use of the GRCAs optional for ensuring adequate SDM exists. The change is proposed to maintain consistency with WCAP 16943-P-A, "Enhanced Gray Rod Cluster Assembly Rodlet Design," discussed below, and other SDM TS Bases discussions.

The technical basis for the proposed changes to the shutdown margin definition of TS 1.1 is provided in WCAP 16943-P-A. This proprietary document presents generic design and evaluation information for the introduction of an enhanced AP1000 GRCA design intended to replace the Ag-In-Cd GRCA absorber rodlet that was originally approved for use in the AP1000 Standard Plant (public version - ADAMS Accession No. ML12284A086). In the final Topical Report Safety Evaluation (TRSE) dated June 21, 2018, WCAP 16943-P-A was found by the staff to be acceptable for referencing in licensing applications for AP1000 designed pressurized water reactors to the extent specified and under the limitations delineated in WCAP 16943-P-A and in the final TRSE (ADAMS Accession No. ML18172A042).

As presented in LAR 17-024 for the change to the SDM definition, the staff did confirm per Section 3.5 of WCAP 16943-P-A that GRCAs will not be credited in SDM calculations made prior to the startup of the core but will be accounted for in analyses in which their use leads to more limiting results. The staff also notes that Section 3.5 of WCAP 16943-P-A states the following:

> It is expected, however, that when the reactor is in a shut down or hot standby condition with GRCA banks confirmed to be inserted by position indication, that utility and Westinghouse personnel may credit the presence of the inserted GRCAs for any shutdown margin or shutdown boron calculations made subsequent to startup physics testing for the cycle, providing that the total measured GRCA rod worth has been confirmed to meet the physics testing acceptance criteria.

To this end, Section 4.4.6, "Operational Monitoring Program," of the TRSE also specifies in regard to startup physics testing: "[t]he applicant will also measure GRCA bank worth during each cycle startup to confirm the adequacy of the nuclear design calculations." Therefore, measuring GRCA bank worth during each cycle startup would support the option of crediting the worth of the GCRAs in the SDM calculations. Finally, the staff conclusions provided in Section 5.0 of the WCAP 16943-P-A TRSE also apply to this LAR. Namely:

The staff notes that, while the GRCAs do not have a safety function and therefore do not have the same regulatory requirement burden that RCCAs do, the applicant has demonstrated compliance with GDCs 27 and 35 by demonstrating that the GRCAs will not fail in such a way as to impact the surrounding fuel assemblies and RCCAs.

From the above staff review, the proposed change of the shutdown margin definition in TS 1.1 does not change the prior staff conclusion and plant operation in accordance with the proposed amendment will remain in compliance with GDCs 27 and 35. To summarize, this part of the proposed TS will allow, but not require, the licensee to credit the GRCAs for SDM in the specified operational modes only if (1) they are verified inserted and (2) the GRCA bank worth has been measured during startup cycle physics testing and pass the specified acceptance criteria, and this is conservative. Based on the above, the staff finds that the proposed changes to VEGP Units 3 and 4 TS 1.1 to be acceptable.

3.2 TS 3.1.4 Rod Group Alignment Limits (L02)

3.2.1 L02A, L02B, L02D, L02K, L02L, and L02M

The series of TS changes proposed under LAR 17-024 Items L02A, L02B, L02D, L02K, L02L, and L02M address the operability of the RCCAs and provide additional clarification about the operability of the GRCAs and their inclusion in SDM calculations. As was previously noted for LAR 17-024 Item L01, the GRCAs do not have a safety function and their removal from TS 3.1.4 would not affect any function or feature used for the prevention and mitigation of accidents or their safety analyses. Essentially, this series of TS changes re-defines the term “rods” to “RCCAs.” Thus, the proposed TS changes would not adversely affect any fuel design limits or design analysis, nor would they adversely affect any safety analysis input or result, or design/safety margins. Since the GRCAs are not credited to perform any safety function governed by TS 3.1.4, they need not be operable under that TS, and the proposal to remove the GRCAs from TS 3.1.4 is acceptable in that regard.

In response to a staff request for additional (RAI) information regarding LAR 17-024 Item L02J (Question 1 of ADAMS Accession No. ML18054B559), SNC provided a second (March 23, 2018) supplement to LAR 17-024 that removed information from the LAR pertaining to the exclusion of the AO control bank RCCAs from SR 3.1.4.2 (ADAMS Accession No. ML18082B370). In the March 23 supplement, the SR 3.1.4.2 note associated with L02J was deleted altogether from LAR 17-024. Since the TS changes would not affect the prior approved safety analyses and design/safety margins, plant operation in accordance with the proposed amendment will remain in compliance with GDC 27. Further, the AO control bank RCCAs remain subject to SR 3.1.4.2. Removal of the note to SR 3.1.4.2 only affects the GRCAs, which need not be controlled by TS 3.1.4 as explained above. Because the existing safety analysis remains applicable, the staff finds the proposed TS changes regarding L02A, L02B, L02D, L02K, L02L, and L02M to be acceptable.

3.2.2 L02C AND L02E

TS LCO 3.1.4 currently states the following:

> All shutdown and control rods shall be OPERABLE.
>
> <u>AND</u>

Individual indicated rod positions shall be within 12 steps of their group step counter demand position.

TS LCO 3.1.4 Note currently states:

Not applicable to Gray Rod Cluster Assemblies (GRCAs) during GRCA bank sequence exchange with the On-Line Power Distribution Monitoring System (OPDMS) monitoring parameters.

This note was added when it was thought that the GRCA bank sequence exchange would be accomplished by moving pairs of GRCA rods from the two banks exchanging positions, thus purposely misaligning rods from their group during normal operation. The final mechanical shim (MSHIM) design has the sequence exchange occurring by moving the gray rods in groups, thus maintaining alignment. SNC proposed to remove this Note from LCO 3.1.4 for GRCA applicability.

The staff reviewed this change and agrees that the note is not needed. The current note addresses a bank sequence exchange maneuver which would purposely misalign GRCAs from their bank for a short period of time.

TS LCO 3.1.4, “Rod Group Alignment Limits,” Condition B. “One rod not within alignment limits.” Completion Time B.1 currently states:

1 hour with the OPDMS not monitoring parameters

SNC proposes to change TS LCO 3.1.4 Condition B.1 to:

1 hour with the On-Line Power Distribution Monitoring System not monitoring parameters

The proposed changes (reviewed above) to LCO 3.1.4 includes the GRCA to be within 12 steps of their group step counter demand position. The MSHIM design will move the GRCA in groups and maintain alignment. Therefore, the staff finds the removal of the Note in LCO 3.1.4 acceptable.

SNC also proposed to define OPMDS as the “On-Line Power Distribution Monitoring System” in LCO 3.1.4 RA B.1 Completion Time. The staff reviewed this change and determined that it is editorial and does not change the Completion Time or purpose of the RA. Therefore, the staff finds the change acceptable.

3.2.3 L02F, L02G, L02H, AND L02I

TS LCO 3.1.4, “Rod Group Alignment Limits,” Condition B. “One rod not within alignment limits.”

Add RA B.2.3.1 where the RA will be to “Perform SR 3.2.5.1” with a Completion Time of “Once per 12 hours, OR” perform “B.2.3.2.1,” which was renumbered from B.2.3.

Delete RA B.2.4 Note, and renumber the RA to B.2.3.2.2.

Delete RA B.2.5 Note, and renumber the RA to B.2.3.2.3.

Renumber RA B.2.6 to B.2.4.

The safety analysis of record for one rod not within alignment limits (or statically misaligned) is addressed by the analysis in UFSAR Section 15.4.3, “Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error).” This analysis for a statically misaligned RCCA demonstrates that conditions worse than one rod not within alignment limits do not cause the departure from nucleate boiling ratio to exceed a safety analysis limit value nor do they result in linear heat generation that causes fuel melting. Therefore, staff finds that the proposed changes would not affect any function or feature used for the prevention and mitigation of accidents or their safety analyses. The proposed changes would not adversely affect any fuel design limits or design analysis, nor would they adversely affect any safety analysis input or result, or design/safety margin. Since the TS changes would not affect the prior approved safety analyses and design/safety margins, plant operation in accordance with the proposed amendment will remain in compliance with GDCs 10, 25, 26, 27, 28, 29, and 35. For these reasons, the safety analysis of record in UFSAR Section 15.4.3 remains bounding. Because the existing safety analysis remains applicable, the staff finds the proposed TS changes regarding L02F, L02G, L02H, and L02I to be acceptable.

3.3 TS 3.1.6 Control Bank Insertion Limits (L03)

TS LCO 3.1.6, “Control Bank Insertion Limits,” Note 2 currently states:

> This LCO is not applicable to Gray Rod Cluster Assembly (GRCA) banks during GRCA bank sequence exchange with On-Line Power Distribution Monitoring System monitoring parameters.

SNC proposes to change TS LCO 3.1.6, Note 2 to read:

> This LCO is not applicable to Gray Rod Cluster Assembly (GRCA) banks for up to one hour during GRCA bank sequence exchange.

SNC notes in LAR 17-024 that the final MSHIM design established that the GRCA bank sequence exchange procedure needed to be accomplished by moving both banks at the same time instead of individually. Such an operation is expected to take a few minutes from the time the banks to be exchanged begin to move. While the prior GRCA exchange sequence would require OPDMS monitoring and several hours to complete, under short duration of the revised GRCA bank sequence exchange, OPDMS may not detect a significant change in the core radial power distribution. Thus, functionally, the OPDMS would not be available to monitor the power distribution during the short duration and would be essentially inoperable. However, this situation is addressed in the current LCOs within TS 3.1.4, 3.2.1, 3.2.2, 3.2.3, and 3.2.4 to allow operations for one hour with OPDMS inoperable with the operators monitoring the key power distribution limits. UFSAR Section 4.3, “Nuclear Design,” discusses actions related to the unlikely event that the online monitoring system is out of service (or cannot be relied upon). It is specifically noted under UFSAR Section 4.3.2.2.6, “Limiting Power Distributions,” that, “power distribution controls based on bounding, precalculated analysis are also provided to the operator such that the online monitoring system is not a required element for short term reactor operation” (see page 4.3-11 of UFSAR Section 4.3.2.2.6). Thus, the online monitoring system is not a required element for the GRCA bank sequence exchange operation, which would need to be completed within 1 hour.

For the initial design certification of the AP1000, as presented in Section 4.3 of NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design,” (ADAMS Accession No. ML112061231) the staff reviewed the online monitoring system and related technical specifications given in AP1000 Design Control Document (DCD) Chapter 16, “Technical Specifications,” and found that core monitoring is acceptable because it meets the acceptance criteria of GDC 13. As noted in the VEGP COL Final Safety Evaluation Report (FSER), the VEGP UFSAR incorporated by reference Chapter 4 of the AP1000 DCD Revision 19, thus the staff’s finding from the AP1000 DCD are valid for the VEGP UFSAR in regard to the nuclear design information and safety analysis. Since the TS changes would not affect the prior approved online monitoring system and associated TSs, operation of the online monitoring system remains in compliance with GDC 13. Based on the above, the information provided in the approved AP1000 DCD and the associated VEGP UFSAR is still valid along with the allowance in TS for the OPDMS not to be a required element for short term reactor operation. Accordingly, the staff finds the proposed TS revisions of LAR 17-024 Item L03 acceptable.

3.4 TS 3.1.7 Rod Position Indication (L04)

Delete RA B.2, and renumber the remaining RAs to B.2 and B.3.

To justify the deletion of monitoring and recording of Reactor Coolant System (RCS) average temperature in TS 3.1.7, “Rod Position Indication,” RA B.2, SNC cites in LAR 17-024 the NRC conclusions reached in Section 3.5 of the final Safety Evaluation (SE) to TSTF-547, Revision 1, “Clarification of Rod Position Requirements” (see page 13 and 14 of TSTF-547, Revision 1, with ADAMS Accession No. ML15328A350). The staff considered TSTF-547, Revision 1, (ADAMS Accession No. ML15365A610) and the aforementioned final SE and verified that the requested change to VEGP Unit 3 and 4 TS 3.1.7, RA B.2, is supported by and consistent with the cited documents. The technical rationale for the deletion of TS 3.1.7, RA B.1, is provided on page 10 of TSTF-547, Revision 1, with the results of the staff’s review provided in Section 3.5 of the final SE on pages 13 and 14 with the following conclusion:

> The NRC staff concludes that the proposed changes to LCO 3.1.7 are acceptable because the LCO continues to specify the minimum performance level of equipment needed for safe operation of the facility. As described in the preceding paragraph the appropriate remedial measures are prescribed when the LCO is not met. SRs are not being changed by the deletion of RA B.2. The NRC staff finds that the requirements of 10 CFR 50.36(c)(2) continue to be met.

Therefore, based on the staff’s findings for TSTF-547, Revision 1, and its use in justifying the VEGP TS change, the staff finds that plant operation as proposed in L04 remains in compliance with GDCs 10, 25, 26, 27, 28, 29, and 35 along with 10 CFR 50.36. Based on the its review of the above supporting documents and the applicability of the staff’s finding for TSTF-547, Revision 1, to the proposed changes, the staff finds that the proposed changes to VEGP Units 3 and 4 TS 3.1.7, RA B.2, presented in LAR 17-024 Item L04 to be acceptable.

3.5 TS Table 3.3.1-1 Reactor Trip System Instrumentation (L05 & L06)

3.5.1 Function 4. “Overpower △T” (L05)

Change TS Surveillance Requirements for Function 4 “Overpower ΔT” by adding two new Surveillance Requirements SR 3.3.1.4 and SR 3.3.1.5.

In LAR 17-024 Item L05, SNC describes a more restrictive change being made to SRs within TS Table 3.3.1-1, “Reactor Trip System Instrumentation.” Within Function 4, “Overpower ΔT,” SNC proposes to add two new SRs, SR 3.3.1.4 and SR 3.3.1.5. SR 3.3.1.4 compares the results of the incore detector instruments to the excore power range (PR) axial flux difference signal ΔI, which provides an input signal related to the difference in neutron flux being produced in the top portion of the core minus the neutron flux being produced in the bottom portion of the core. SR 3.3.1.5 proposes to calibrate the power range channels to agree with the incore detector measurements.

In relation to SR 3.3.1.4, the ΔI signal serves as one of the inputs to the setpoints being calculated for two reactor protection system trips, the Overtemperature ΔT (OTΔT) and the Overpower ΔT (OPΔT) reactor trips within the AP1000 safety-related digital instrumentation and controls (I&C) system, the Protection and Safety Monitoring System (PMS). The OTΔT trip provides core protection to prevent departure from nucleate boiling for existing combinations of system pressure, power, coolant temperature and axial power distribution, where the axial power distribution is represented as ΔI. The OPΔT reactor trip protects the core against excessive power being generated in the core and provides confidence of fuel integrity during overpower conditions by limiting the required range for OTΔT protection. The OPΔT reactor trip also serves as a backup to the power range high neutron flux reactor trip.

The ΔI signal, which acts as a variable bias input signal to both setpoints, (OTΔT and OPΔT), depends upon the axial differential core power, and is designed to lower the respective trip setpoint when axial flux offset of the core is at a greater value. Since the signal serves as an input to one or, in this case, two reactor trips, it is appropriate that the instrumentation that generates the signal be periodically calibrated to ensure the instrumentation is operating properly. In accordance with LAR 17-024 Item L05, the calibration of the ΔI instrumentation will occur by comparing the ΔI signal to the signals from the incore instrumentation system (IIS), which provides a three-dimensional (3-D) representation of the flux being produced at various points throughout the core. The signal processing software used is integral to the IIS and allows the fixed incore detector signals to be used to calculate an accurate 3-D core power distribution suitable for deriving calibration information for the excore nuclear instrumentation input to OTΔT and OPΔT reactor trip setpoints. The staff verified the addition of the two SRs to the OPΔT setpoint will accomplish its stated goal and, as such, finds the addition of SR 3.3.1.4 to Function 4, “Overpower ΔT,” within Table 3.3.1-1 acceptable.

Regarding SR 3.3.1.5, the IIS is designed to generate a 3-D neutron flux map of the core, which is then used to calibrate the neutron detectors used by the PMS. Based upon the information provided in the LAR, the staff verified that SR 3.3.1.5 requires that exact activity, therefore the staff finds the addition of SR 3.3.1.5, within Table 3.3.1-1 acceptable.

3.5.2 Function 12, “Passive Residual Heat Removal Actuation” (L06)

> Change Function 12, “Passive Residual Heat Removal Actuation” by deleting Surveillance Requirement SR 3.3.1.9.

In LAR 17-024 Item L06, SNC describes a less restrictive change in which it proposes to remove SR 3.3.1.9 within TS Table 3.3.1-1, “Reactor Trip System instrumentation” for Function 12, “Passive Residual Heat Removal Actuation.” SR 3.3.1.9 requires that a channel calibration be conducted in accordance with the setpoint program at a frequency of every 24 months. The setpoint associated with this particular reactor trip involves actuating a reactor trip whenever either of the parallel PRHR discharge valves (PXS-PL-V-108A and PXS-PL-V-108B) is not fully closed – as determined by the proper positioning and setup of the associated valve position indication contact sets.

SR 3.3.1.10 directs the licensee to perform a trip actuating device operational test (TADOT) every 24 months. In Section 1.1 Definitions, a TADOT consists of operating the trip actuating device and verifying the OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY. The TADOT includes adjustment, as necessary, of the trip actuating device so that it actuates at the setpoint determined in accordance with the setpoint program within the necessary accuracy. The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps. Based upon information contained in the LAR as supplemented by RAI Response No. 2 in LAR 17-024 Supplement 2 (ADAMS Accession No. ML18082B370), SNC described that SR 3.3.1.10 requires the verification of the proper setup and adjustment of the same valve position indication contact sets (PXS-PL-V-108A and PXS-PL-V-108B) at the same periodicity as SR 3.3.1.9. Accordingly, the staff determined that both SR 3.3.1.9 for Function 12 and the TADOT apply for the identical position indication contact sets and are redundant. As such, since SNC did not propose to alter its commitment to perform the TADOT under SR 3.3.1.10 for the PRHR valve position contact sets, the staff finds the removal of the requirement to perform Function 12 SR 3.3.1.9 acceptable

3.6 TS Table 3.3.5-1, "Reactor Trip System Manual Actuation," Function 4 (L07)

> Change the Table 3.3.5-1, Function 4, "Core Makeup Tank Actuation Input from Engineered Safety Feature Actuation System – Manual," Required Channels from to "2 switch sets" to "2."

LCO 3.3.5 states that the Reactor Trip System (RTS) manual actuation channels for each function in Table 3.3.5-1 shall be OPERABLE. Two independent actuation channels are required to be OPERABLE so that no single random failure will disable the Manual Reactor Trip Function. Table 3.3.5-1 Function 4 requires two manual actuation channels for the Core Makeup Tank (CMT) actuation input from the Engineered Safety Feature (ESF) Actuation System to be OPERABLE.

SR 3.3.5.1 requires the performance of a TADOT of the RTS inputs for Manual Reactor Trip, and from the ESF logic for Safeguards Actuation, automatic depressurization system (ADS) Stage 1, 2, and 3 Actuation, and CMT Actuation. This TADOT is performed every 24 months. The test independently verifies the OPERABILITY of the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers.

The staff reviewed UFSAR Section 7.3.1.2.3, "Core Makeup Tank Injection," in Chapter 7, "Instrumentation and Controls," within Revision 6 of the licensee's UFSAR, which states, in part, "Condition 5 [Manual initiation] consists of two manual controls. Manual actuation of either of the two controls will align the core makeup tanks for injection." Additionally, Sheet 12 of 21 of Figure 7.2-1, "Functional Diagram Core Makeup Tank Actuation," depicts the manual activation of the core makeup tank being carried out via an "or" gate, such that either of the two switches will initiate the requested functionality, in this case that being a manual reactor trip.

Since the staff did not understand the information in the technical evaluation related to Item L07, it requested that SNC provide additional information to clarify the licensee's request regarding Function 4 in RAI No. 3 dated May 31, 2018 (ADAMS Accession No. ML18151B056). When the staff received the licensee's response related to the technical evaluation of Item L07 in Supplement 3 (ADAMS Accession No. ML18172A154), the staff continued to be unable to understand the licensee's reasoning related to the acceptability of the technical evaluation for the requested change. Specifically, the technical evaluation for Item L07 of LAR 17-024 Supplement 3 first indicated that there are two sets of redundant switches, one on the Primary Dedicated Safety Panel (PDSP) and one on the Secondary Dedicated Safety Panel (SDSP). Then, in the second paragraph, the applicant indicated that the Protection and Safety Monitoring System design provides two redundant controls on the PDSP, with no corresponding controls on the SPDS. In the case of Item L07, the staff understands there to be two redundant switches located on the PDSP for Function 4, such that, when either is operated this will cause Function 4 to actuate.

After the staff explained this issue to SNC during a public phone call on July 12, 2018, (ADAMS Accession No. ML18197A241) to discuss RAI No. 4 dated July 17, 2018 (ADAMS Accession No. ML18198A060), SNC responded by submitting Supplement 4 of LAR 17-024 in a letter dated August 9, 2018 (ADAMS Accession No. ML18221A364). Supplement 4 to LAR 17-024 revised the technical evaluation for the proposed change regarding the "Core Makeup Tank Actuation Input from Engineered Safety Feature Actuation System – Manual" function (Function 4 in TS Table 3.3.5-1). As revised, the LAR explains that there are redundant switches on the PDSP, either of which will initiate manual core makeup tank injection and which also sends a reactor trip signal to the reactor trip switchgear breakers. Since either switch will initiate the reactor trip signal, the term 'switch set' does not apply. Therefore, SNC proposed to change the "REQUIRED CHANNELS" column from "2 switch sets" to "2."

The staff reviewed this response and finds it to be acceptable, as follows: The change is needed to properly describe the function of the two manual actuation channels for the CMT actuation input from the ESF actuation system – that is, there are two redundant switches on the PDSP, either of which will initiate a manual CMT injection and send a reactor trip signal. The proposed change from "2 switch sets" to "2" continues to meet the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii), "Limiting conditions of operation," and 10 CFR 50.36(c)(3), "Surveillance requirements." LCO 3.3.5 and SR 3.3.5 continue to be met by requiring 2 channels to be OPERABLE and the performance of a TADOT on 2 channels every 24 months. Therefore, the staff concludes that the proposed changes to Table 3.3.5-1 Function 4 are acceptable.

3.7 TS Table 3.3.17-1, "Post-Accident Monitoring Instrumentation" (L08)

Delete the word "Monitor" from Function 5 "RCS Subcooling Monitor."

Add new Function 20, "Pressurizer Pressure," with Required Channels, "2," and Condition Referenced From Required Action D.1 to be "E."

TS LCO 3.3.17, "Post Accident Monitoring (PAM) Instrumentation," ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident. The instrument channels required to be operable by this LCO include two classes of parameters identified by RG 1.97 as Type A and Category 1 variables. This LCO addresses those RG 1.97 instruments, which are listed in Table 3.3.17-1.

SNC proposed the deletion of the word "Monitor" from the Function 5, "RCS Subcooling Monitor." SNC states in the LAR that the deletion is made for consistency with the other monitors and sensors listed in Table 3.3.17-1. The staff reviewed this change and concluded that it does not change the intended purpose of Function 5 and is an editorial change.

Therefore, the staff finds the deletion of “Monitor” from Function 5 in Table 3.3.17-1 to be acceptable.

SNC proposed to add a new Function 20, “Pressurizer Pressure,” to Table 3.3.17-1. SNC stated in the LAR that Function 20 is included to provide the operators with information to assess the process for accomplishing or maintaining the safety-related function of RCS control.

The staff concludes that the new function will continue to provide the control room operating staff with information to assess reactor coolant system inventory control. The new Function 20 is consistent with other monitoring requirements listed in TS Table 3.3.17-1, in that it requires two channels to be operable and lists Condition E referenced from RA D.1. It also continues to meet the guidance of RG 1.97 to provide instrumentation to monitor variables and systems over their anticipated ranges for accident conditions as appropriate to ensure adequate safety. Therefore, the staff finds the addition of Function 20 to LCO 3.3.17 in Table 3.3.17-1 acceptable.

3.8 TS 3.3.19, “Diverse Actuation System Manual Controls” (L09)

> Change Note (c) for TS Table 3.3.19-1, “Diverse Actuation System Manual Controls” from “With reactor internals in place.” to read “With upper internals in place.”

As discussed in LAR 17-024 Item L09, SNC has identified an inconsistency in the operability requirements for the ADS Stage 4 valves between Function 7 for Table 3.3.19-1, the applicability requirement under MODE 6 for LCO 3.4.13, and Function 7 for MODE 6 in TS 3.3.9. LCO 3.4.13 and Function 7 in TS 3.3.9 state ADS Stage 4 valves are operable with upper internals in place while Table 3.3.19-1, Function 7, requires the ADS Stage 4 valves to be operable with reactor internals in place. With the upper internals removed, the reactor internals are covered with water and the RCS is open for refueling shutdown conditions. Under these conditions, there is not a need for the ADS Stage 4 valves to be operable to depressurize the RCS to assure IRWST makeup water to the core.

The staff finds that the proposed TS change would not affect any function or feature used for the prevention and mitigation of accidents or their safety analyses. This TS change would not adversely affect any fuel design limits or design analysis, nor would it adversely affect any safety analysis input or result, or design/safety margin. In view of the above information, the staff finds that compliance with GDCs 10 and 35 is maintained. Therefore, the staff finds the proposed change to Note (c) for the TS Table 3.3.19-1 adequately addresses the identified inconsistency with other TS sections and is acceptable.

3.9 TS 3.5.4, “PRHR HX-Operating” (L10)

TS 3.5.4, “PRHR HX - Operating,” Surveillance Requirement SR 3.5.4.6 currently states:

> Verify both PRHR HX air operated outlet isolation valves and both IRWST gutter isolation valves stroke open.

SNC proposes to change SR 3.5.4.6 to read:

> Verify both PRHR HX air operated outlet isolation valves stroke open and both IRWST gutter isolation valves stroke closed.

SNC has identified an inconsistency for the stated safety-related design function for both IRWST gutter isolation values. As presented in LAR 17-024 Item L10, the IRWST gutter isolation valves safety-related design function is to stroke closed. The operation of the IRWST gutter is described in VEGP UFSAR Section 6.3.2.1.1, "Emergency Core Decay Heat Removal at High Pressure and Temperature Conditions," as revised by VEGP Units 3 and 4, License Amendments 72 and 71, as discussed in NRC safety evaluation dated February 27, 2017 (ADAMS Accession No. ML17024A317) as follows:

> The passive residual heat removal heat exchanger, in conjunction with the in-containment refueling water storage tank, the condensate return features, and the passive containment cooling system, can provide core cooling for at least 72 hours. After the in-containment refueling water storage tank water reaches its saturation temperature (in several hours), the process of steaming to the containment initiates. Containment pressure increases as steam is released from the in-containment refueling water storage tank. As containment temperature increases, condensation begins to form on the subcooled metal and concrete surfaces inside containment. Condensation on these heat sink surfaces transfers energy to the bulk metal and concrete until they come into equilibrium with the containment atmosphere. Condensation that is not returned to the in-containment refueling water storage tank drains to the containment sump.
>
> Condensation occurs on the steel containment vessel, which is cooled by the passive containment cooling system. Most of the condensate formed on the containment vessel wall is collected in a safety-related gutter arrangement. A gutter is located near the operating deck elevation, and a downspout piping system is connected at the polar crane girder and internal stiffener, to collect steam condensate inside the containment during passive containment cooling system operation and return it to the in-containment refueling water storage tank. The gutter normally drains to the containment sump, but when the passive residual heat removal heat exchanger actuates, safety-related isolation valves in the gutter drain line shut and the gutter overflow returns directly to the in-containment refueling water storage tank. Recovery of the condensate maintains the passive residual heat removal heat exchanger heat sink for greater than 14 days.

This TS change also assists in assuring that the IRWST gutter system would perform as intended if challenged, which ensures compliance with GDCs 10, 29, and 35. The staff determined that the safety analysis credits the valves as closing when the passive residual heat removal heat exchanger actuates, and these systems are designed to perform their safety functions with the valves closed, as described in LAR 17-024 Item L10 and the IRWST gutter operation discussion in Section 6.3.2.1.1 of the VEGP UFSAR. Accordingly, the staff finds the change to SR 3.5.4.6 adequately addresses the identified inconsistency and is acceptable.

3.10 TS 3.8.3 Inverters – Operating (L11)

> Change Action Condition A. from "One inverter inoperable." to "One or two inverter(s) within one division inoperable."
>
> Change RA A.1 from "Restore inverter to OPERABLE status." to "Restore inverter(s) to OPERABLE status."

LAR 17-024 Item L11 focuses on the plant Class 1E inverters. As part of this LAR, SNC recognized that the current TS do not cover the possibility that for Divisions B and C, which have two inverters (one for the 24-hour battery bus and one for the 72-hour battery bus), both inverters could be inoperable concurrently. The RA change maintains the same Completion

Time of 24 hours. The accompanying note directs the licensee to LCO 3.8.5 "Distribution Systems – Operating" with any I&C bus de-energized. Under this LCO, restoration time for an inoperable I&C bus is 6 hours.

The staff has verified that the loss of one or both inverters within a division (B or C) has no different effect on any given function powered by the respective (24-hour or 72-hour battery-backed) bus as the two buses are not back-ups for each other. In addition, the basis for the 24-hour restoration time for one or two inverters inoperable in the same division remains unaffected by the loss of the second inverter. Also, loss of an inverter does not result in loss of the bus as each bus is backed up by a transformer from the ac portion of the power system. This condition of the bus being unavailable is covered by LCO 3.8.5 as discussed above.

The staff finds that the proposed changes to TS 3.8.3 covers the possibility of two inverters in one division becoming concurrently inoperable and finds that accounting for a second inverter being inoperable (due to their non-redundancy) does not increase the risk of losing a given safety function or create a new risk and is therefore acceptable.

3.11 SUMMARY OF TECHNICAL EVALUATION

Based on the foregoing, the staff concludes the LAR proposed changes meet the requirements of 10 CFR 50.36 and comply with GDCs 10, 13, 25, 26, 27, 28, 29, and 35, and do not change the safety analysis presented in the VEGP UFSAR; therefore the existing safety analysis remains applicable. Accordingly, the staff finds the changes proposed to the TSs to be acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations in 10 CFR 50.91(b)(2), on August 13, 2018, the Georgia State official was consulted regarding the proposed issuance of the amendment. The State official had no comment.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, "*Standards for Protection Against Radiation*." The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite. Also, there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (83 FR 6227, published on February 13, 2018). 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.

6.0 CONCLUSION

The staff has reviewed the changes associated with the license amendment and concluded, based on the considerations discussed in Section 3.2 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's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. Therefore, the staff finds the changes proposed in this license amendment acceptable.

7.0 REFERENCES

1. Southern Nuclear Operating Company, Vogtle Electric Generating Plant Units 3 and 4, "Request for License Amendment: Technical Specification Updates for Reactivity Controls and other Miscellaneous Changes (LAR 17-024)," July 28, 2017 (ADAMS Accession No. ML17209A755).

2. Southern Nuclear Operating Company, Vogtle Electric Generating Plant Units 3 and 4, "Supplement to Request for License Amendment: Technical Specification Updates for Reactivity Controls and other Miscellaneous Changes (LAR 17-024S1)," January 23, 2018 (ADAMS Accession No. ML18023A440).

3. Southern Nuclear Operating Company, Vogtle Electric Generating Plant Units 3 and 4, "Supplement to Request for License Amendment: Technical Specification Updates for Reactivity Controls and other Miscellaneous Changes (LAR 17-024S2)," March 23, 2018 (ADAMS Accession No. ML18082B370).

4. Southern Nuclear Operating Company, Vogtle Electric Generating Plant Units 3 and 4, "Supplement to Request for License Amendment: Technical Specification Updates for Reactivity Controls and other Miscellaneous Changes (LAR 17-024S3)," June 21, 2018 (ADAMS Accession No. ML18172A154).

5. Southern Nuclear Operating Company, Vogtle Electric Generating Plant Units 3 and 4, "Supplement to Request for License Amendment: Technical Specification Updates for Reactivity Controls and other Miscellaneous Changes (LAR 17-024S4)," August 9, 2018 (ADAMS Accession No. ML18221A364).

6. Southern Nuclear Operating Company, Vogtle Electric Generating Plant Units 3 and 4 Updated Final Safety Analysis Report, Revision 6 and Tier 1, Revision 5, March 12, 2017 (ADAMS Accession No. ML17172A218).

7. U.S. Nuclear Regulatory Commission, NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design" August 5, 2011 (ADAMS Accession No. ML112061231).

8. U.S. Nuclear Regulatory Commission, Final Safety Evaluation of TSTF-547, Revision 1, "Clarification of Rod Position Requirements," March 4, 2016 (ADAMS Accession No. ML1532A350).

9. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," June 2006 (ADAMS Accession No. ML061580448).

10. Westinghouse Electric Company, LLC, AP1000 Design Control Document, Revision 19, June 13, 2011 (ADAMS Accession No. ML11171A500).

11. Westinghouse Electric Company, LLC, “Enhanced GRCA Rodlet Design,” WCAP 16943-P-A, Non-Proprietary Version, 2012 (ADAMS Accession No. ML12284A086).

12. U.S. Nuclear Regulatory Commission, Final Topical Report Safety Evaluation for WCAP 16943, “Enhanced Gray Rod Cluster Assembly Rodlet Design,” June 21, 2018 (ADAMS Accession No. ML18172A042).

13. U.S. Nuclear Regulatory Commission, Request for Additional Information Regarding Vogtle 3 & 4 LAR-17-024, February 23, 2018 (ADAMS Accession No. ML18054B559).

14. U.S. Nuclear Regulatory Commission, Draft Request for Additional Information # 3 Regarding Vogtle 3 & 4, dated May 31, 2018 (ADAMS Accession No. ML18151B056).

15. U.S. Nuclear Regulatory Commission, Summary of a Public Meeting with Southern Nuclear Operating Company on July 12, 2018, (ADAMS Accession No. ML18197A241).

16. U.S. Nuclear Regulatory Commission, Final RAI No. 4 on LAR-17-024, TS Updates for Reactivity Control and Miscellaneous,” dated July 17, 2018 (ADAMS Accession No. ML18198A060).

17. U.S. Nuclear Regulatory Commission, Safety Evaluation Related to Amendment Nos. 72 and 71 to the Combined License Nos. 91 and 92, Respectively; Vogtle Electric Generating Plant Units 3 and 4; February 27, 2017; (ADAMS Accession No. ML17024A317).

12. Westinghouse Electric Company, LLC, “AP1000 Protection and Safety Monitoring System Architecture Technical Report.” WCAP-16675-P, Rev. 5, Proprietary Version, November 2010 (ADAMS Accession No. ML103370224).

13. Institute of Electrical and Electronic Engineers, IEEE 100, “The Authoritative Dictionary of IEEE Standards Terms,” Seventh Edition, 2000.