COMBINED LICENSE

VOGTLE ELECTRIC GENERATING PLANT UNIT 4

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

Docket No. 52-026

License No. NPF-92

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for a combined license (COL) for Vogtle Electric Generating Plant (VEGP) Unit 4 filed by Southern Nuclear Operating Company, Inc. (SNC) acting on behalf of Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia,[1] and the City of Dalton, Georgia, an incorporated municipality in the state of Georgia acting by and through its Board of Water, Light and Sinking Fund Commissioners (City of Dalton), herein referred to as “the VEGP owners,” which incorporates by reference Appendix D to 10 CFR Part 52 and Early Site Permit No. ESP-004, complies with the applicable standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. There is reasonable assurance that the facility will be constructed and will operate in conformity with the application, as amended, the provisions of the Act, and the Commission regulations set forth in 10 CFR Chapter I, except as exempted from compliance in Sections 2.F and 2.G below;

   C. There is reasonable assurance (i) that the activities authorized by this COL can be conducted without endangering the health and safety of the public and (ii) that such activities will be conducted in compliance with the Commission regulations set forth in 10 CFR Chapter I, except as exempted from compliance in Sections 2.F and 2.G below;

[1] On June 24, 2014, Municipal Electric Authority of Georgia transferred its ownership interest to its wholly owned subsidiaries: MEAG Power SPVM, LLC; MEAG Power SPVJ, LLC; and MEAG Power SPVP, LLC as described in the SNC letter dated December 2, 2013 and in the Commission’s Safety Evaluation available in the Agencywide Document Access and Management System (ADAMS) under Accession No. ML14072A340.

D. SNC[2] is technically qualified to engage in the activities authorized by this license in accordance with the Commission regulations set forth in 10 CFR Chapter I. SNC and the VEGP owners together are financially qualified to engage in the activities authorized by this COL in accordance with the Commission regulations set forth in 10 CFR Chapter I;

E. SNC and the VEGP owners have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements;"

F. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

G. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering reasonable available alternatives, the issuance of this license subject to the conditions for protection of the environment set forth herein is in accordance with Subpart A of 10 CFR Part 51 and all applicable requirements have been satisfied; and

H. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this license will be in accordance with the applicable regulations in 10 CFR Parts 30, 40, and 70.

2. On the basis of the foregoing findings regarding this facility, COL No. NPF-92 is hereby issued to SNC, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and the City of Dalton, Georgia (the licensees) to read as follows:

A. This license applies to the VEGP Unit 4, a light-water nuclear reactor and associated equipment (the facility), owned by the VEGP Owners. The facility would be located adjacent to existing VEGP Units 1 and 2 on a 3,169-acre coastal plain bluff on the southwest side of the Savannah River in eastern Burke County, GA, approximately 15 miles east-northeast of Waynesboro, GA, and 26 miles southeast of Augusta, GA, and is described in the licensees' updated final safety analysis report (UFSAR), as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) SNC pursuant to Sections 103 and 185b. of the Act and 10 CFR Part 52, to construct, possess, use, and operate the facility at the designated location in accordance with the procedures and limitations set forth in this license;

(2) The VEGP owners pursuant to the Act and 10 CFR Part 52, to possess but not operate the facility at the designated location in Burke County, GA, in accordance with the procedures and limitations set forth in this license;

[2] SNC is authorized by the VEGP owners to exercise responsibility and control over the physical construction, operation, and maintenance of the facility.

(3) (a) SNC pursuant to the Act and 10 CFR Part 70, to receive and possess at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and in amounts necessary for reactor operation, described in the UFSAR, as supplemented and amended;

(b) SNC pursuant to the Act and 10 CFR Part 70, to use special nuclear material as reactor fuel, after a Commission finding under 10 CFR 52.103(g) has been made, in accordance with the limitations for storage and in amounts necessary for reactor operation, described in the UFSAR, as supplemented and amended;

(4) (a) SNC pursuant to the Act and 10 CFR Parts 30 and 70, to receive, possess, and use, at any time before a Commission finding under 10 CFR 52.103(g), such byproduct and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts, as necessary;

(b) SNC pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, after a Commission finding under 10 CFR 52.103(g), any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as necessary;

(5) (a) SNC pursuant to the Act and 10 CFR Parts 30 and 70, to receive, possess, and use, before a Commission finding under 10 CFR 52.103(g), in amounts not exceeding those specified in 10 CFR 30.72, any byproduct or special nuclear material that is (1) in unsealed form; (2) on foils or plated surfaces, or (3) sealed in glass, for sample analysis or instrument calibration or other activity associated with radioactive apparatus or components;

(b) SNC pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, after a Commission finding under 10 CFR 52.103(g), in amounts as necessary, any byproduct, source, or special nuclear material without restriction as to chemical or physical form, for sample analysis or instrument calibration or other activity associated with radioactive apparatus or components but not uranium hexafluoride; and

(6) SNC pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. The license is subject to, and the licensees shall comply with, all applicable provisions of the Act and the rules, regulations, and orders of the Commission, including the conditions set forth in 10 CFR Chapter I, now or hereafter in effect.

D. The license is subject to, and SNC shall comply with, the conditions specified and incorporated below:

(1) Changes during Construction

(a) SNC may request use of a preliminary amendment request (PAR) process, for license amendments, at any time before a Commission finding under 10 CFR 52.103(g). To use the PAR process, SNC shall submit a written request to the Office of New Reactors (NRO) in accordance with COL-ISG-025, "Changes during Construction under Part 52."

(b) Before NRO's issuance of a written PAR notification, SNC shall submit the license amendment request (LAR). Thereafter, NRO will issue a written PAR notification, setting forth whether SNC may proceed in accordance with the PAR, LAR, and COL-ISG-025. If SNC elects to proceed and the LAR is subsequently denied, SNC shall return the facility to its current licensing basis.

(2) Pre-operational Testing

(a) SNC shall perform the design-specific pre-operational tests identified below:

(b) SNC shall review and evaluate the results of the tests identified in Section 2.D.(2)(a) of this license and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with UFSAR Section 14.2.9,

(c) SNC shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of the design-specific pre-operational tests identified in Section 2.D.(2)(a) of this license; and

(d) (Removed by Amendment No. 194)

(3) Nuclear Fuel Loading and Pre-critical Testing

(a) Until the submission of the notification required by Section 2.D.(2)(c) of this license, SNC shall not load fuel into the reactor vessel;

(b) (Removed by Amendment No. 194)

(c) SNC shall perform the pre-critical tests identified in UFSAR Section 14.2.10.1;

(d) SNC shall review and evaluate the results of the tests identified in Section 2.D.(3)(c) of this license and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with UFSAR Section 14.2.10; and

(e) SNC shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of the pre-critical tests identified in Section 2.D.(3)(c) of this license.

(4) Initial Criticality and Low-Power Testing

(a) Upon submission of the notification required by Section 2.D.(3)(e) of this license, SNC is authorized to operate the facility at reactor steady-state core power levels not to exceed 5-percent thermal power in accordance with the conditions specified herein;

(b) SNC shall perform the initial criticality and low-power tests identified in UFSAR Sections 14.2.10.2 and 14.2.10.3, respectively;

(c) SNC shall review and evaluate the results of the tests identified in Section 2.D.(4)(b) of this license and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with UFSAR Sections 14.2.10.2 and 14.2.10.3; and

(d) SNC shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of initial criticality and low-power tests identified in Section 2.D.(4)(b) of this license, including the design-specific tests identified therein.

(5) Power Ascension Testing

(a) Upon submission of the notification required by Section 2.D.(4)(d) of this license, SNC is authorized to operate the facility at reactor steady-state core power levels not to exceed 100-percent thermal power in accordance with the conditions specified herein, but only for the purpose of performing power ascension testing;

(b) SNC shall perform the power ascension tests identified in UFSAR Section 14.2.10.4;

(c) SNC shall review and evaluate the results of the tests identified in Section 2.D.(5)(b) of this license and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with UFSAR Section 14.2.10.4; and

(d) SNC shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of power ascension tests identified in Section 2.D.(5)(b) of this license, including the design-specific tests identified therein.

(6) Maximum Power Level

Upon submission of the notification required by Section 2.D.(5)(d) of this license, SNC is authorized to operate the facility at steady state reactor core power levels not to exceed 3400 MW thermal (100-percent thermal power), as described in the UFSAR, in accordance with the conditions specified herein.

(7) Reporting Requirements

(a) Within 30 days of a change to the initial test program described in UFSAR Section 14, Initial Test Program, made in accordance with 10 CFR 50.59 or in accordance with 10 CFR Part 52, Appendix D, Section VIII, "Processes for Changes and Departures," SNC shall report the change to the Director of NRO, or the Director's designee, in accordance with 10 CFR 50.59(d).

(b) SNC shall report any violation of a requirement in Section 2.D.(3), Section 2.D.(4), Section 2.D.(5), and Section 2.D.(6) of this license within 24 hours. Initial notification shall be made to the NRC Operations Center in accordance with 10 CFR 50.72, with written follow up in accordance with 10 CFR 50.73.

(8) Incorporation

The Technical Specifications and Environmental Protection Plan in Appendices A and B, respectively, of this license, as revised through Amendment No. 196, are hereby incorporated into this license.

(9) Technical Specifications

The technical specifications in Appendix A to this license become effective upon a Commission finding that the acceptance criteria in this license (ITAAC) are met in accordance with 10 CFR 52.103(g) with the following exceptions:

(a) Prior to initial criticality of the reactor core while operating in plant operational Mode 5 (Cold Shutdown) or Mode 6 (Refueling) the following TS are temporarily excluded from becoming effective:

- TS 3.3.8, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," Table 3.3.8-1
  - Function 14, RCS Wide Range Pressure – Low
  - Function 15, Core Makeup Tank (CMT) Level – Low 3
  - Function 16, CMT Level – Low 6
  - Function 18, IRWST Lower Narrow Range Level – Low 3
- TS 3.3.9, "Engineered Safety Feature Actuation System (ESFAS) Manual Initiation," Table 3.3.9-1
  - Function 1, Safeguards Actuation - Manual Initiation
  - Function 6, ADS Stages 1, 2 & 3 Actuation – Manual Initiation
  - Function 7, ADS Stage 4 Actuation – Manual Initiation
  - Function 8, Passive Containment Cooling Actuation – Manual Initiation
  - Function 9, Passive Residual Heat Removal Heat Exchanger Actuation – Manual Initiation

- Function 12, In-Containment Refueling Water Storage Tank (IRWST) Injection Line Valve Actuation – Manual Initiation
- Function 13, IRWST Containment Recirculation Valve Actuation – Manual Initiation

- TS 3.3.10, "Engineered Safety Feature Actuation System (ESFAS) Reactor Coolant System (RCS) Hot Leg Level Instrumentation"
- TS 3.3.14, "Engineered Safety Feature Actuation System (ESFAS) In-containment Refueling Water Storage Tank (IRWST) and Spent Fuel Pool Level Instrumentation," Table 3.3.14-1
  - Function 1, Spent Fuel Pool Level – Low 2
- TS 3.3.19, "Diverse Actuation System (DAS) Manual Controls," Table 3.3.19-1
  - Function 2, Passive Residual Heat Removal Heat Exchanger (PRHR HX) control and In-Containment Refueling Water Storage Tank (IRWST) gutter control valves
  - Function 4, Automatic Depressurization System (ADS) stage 1 valves
  - Function 5, ADS stage 2 valves
  - Function 6, ADS stage 3 valves
  - Function 7, ADS stage 4 valves
  - Function 8, IRWST injection squib valves
  - Function 9, Containment recirculation valves
  - Function 10, Passive containment cooling drain valves
  - Function 11, Selected containment isolation valves
- TS 3.3.20, "Automatic Depressurization System (ADS) and In-containment Refueling Water Storage Tank (IRWST) Injection Blocking Device," Table 3.3.20-1
  - Function 2, ADS and IRWST Injection Block Switches for Manual Unblocking
- TS 3.4.12, "Automatic Depressurization System (ADS) – Shutdown, RCS Intact"
- TS 3.4.13, "Automatic Depressurization System (ADS) – Shutdown, RCS Open"
- TS 3.5.5, "Passive Residual Heat Removal Heat Exchanger (PRHR HX) – Shutdown, Reactor Coolant System (RCS) Intact"
- TS 3.5.7, "In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 5"
- TS 3.5.8, "In-containment Refueling Water Storage Tank (IRWST) – Shutdown, MODE 6"
- TS 3.6.7, "Containment Penetrations"
- TS 3.7.13, "Spent Fuel Pool Cooling System (SFS) Containment Isolation Valves"

(b) Prior to initial criticality of the reactor core while operating in plant operational Mode 4 (Safe Shutdown) when any cold leg temperature is ≤ 275°F the following TS are temporarily excluded from becoming effective:

- TS 3.3.8, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," Table 3.3.8-1
  - Function 14, RCS Wide Range Pressure – Low
  - Function 15, Core Makeup Tank (CMT) Level – Low 3
  - Function 16, CMT Level – Low 6
  - Function 18, IRWST Lower Narrow Range Level – Low 3
  - Function 19, Reactor Coolant Pump Bearing Water Temperature – High 2
- TS 3.3.9, "Engineered Safety Feature Actuation System (ESFAS) Manual Initiation," Table 3.3.9-1
  - Function 3, Containment Isolation – Manual Initiation
  - Function 6, ADS Stages 1, 2 & 3 Actuation – Manual Initiation
  - Function 7, ADS Stage 4 Actuation – Manual Initiation
  - Function 8, Passive Containment Cooling Actuation – Manual Initiation
  - Function 12, In-Containment Refueling Water Storage Tank (IRWST) Injection Line Valve Actuation – Manual Initiation
  - Function 13, IRWST Containment Recirculation Valve Actuation – Manual Initiation
- TS 3.3.13, "Engineered Safety Feature Actuation System (ESFAS) Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization," Table 3.3.13-1
  - Function 1, Main Control Room Air Supply Iodine or Particulate Radiation – High 2
- TS 3.3.19, "Diverse Actuation System (DAS) Manual Controls," Table 3.3.19-1
  - Function 4, Automatic Depressurization System (ADS) stage 1 valves
  - Function 5, ADS stage 2 valves
  - Function 6, ADS stage 3 valves
  - Function 7, ADS stage 4 valves
  - Function 8, IRWST injection squib valves
  - Function 9, Containment recirculation valves
  - Function 10, Passive containment cooling drain valves
  - Function 11, Selected containment isolation valves

- TS 3.3.20, “Automatic Depressurization System (ADS) and In-containment Refueling Water Storage Tank (IRWST) Injection Blocking Device,” Table 3.3.20-1
    - Function 2, ADS and IRWST Injection Block Switches for Manual Unblocking
- TS 3.4.11, “Automatic Depressurization System (ADS) – Operating”
- TS 3.5.1, “Accumulators”
- TS 3.5.6, “In-containment Refueling Water Storage Tank (IRWST) – Operating”
- TS 3.6.1, “Containment”
- TS 3.6.2, “Containment Air Locks”
- TS 3.6.3, “Containment Isolation Valves”
- TS 3.6.6, “Passive Containment Cooling System (PCS)”
- TS 3.6.8, “pH Adjustment”
- TS 3.7.4, “Secondary Specific Activity”
- TS 3.7.10, “Steam Generator (SG) Isolation Valves” only for PORV and PORV block valves (SG blowdown isolation valve not excluded)

(c) Prior to initial criticality of the reactor core while operating in plant operational Mode 4 (Safe Shutdown) with all four cold leg temperatures > 275°F the following TS are temporarily excluded from becoming effective:

- TS 3.3.8, “Engineered Safety Feature Actuation System (ESFAS) Instrumentation,” Table 3.3.8-1
    - Function 3, Containment Radioactivity – High
    - Function 18, IRWST Lower Narrow Range Level – Low 3
    - Function 19, Reactor Coolant Pump Bearing Water Temperature – High 2
- TS 3.3.9, “Engineered Safety Feature Actuation System (ESFAS) Manual Initiation,” Table 3.3.9-1
    - Function 3, Containment Isolation – Manual Initiation
    - Function 6, ADS Stages 1, 2 & 3 Actuation – Manual Initiation
    - Function 7, ADS Stage 4 Actuation – Manual Initiation
    - Function 8, Passive Containment Cooling Actuation – Manual Initiation
    - Function 13, IRWST Containment Recirculation Valve Actuation – Manual Initiation
    - Function 14, SG Power Operated Relief Valve and Block Valve Isolation – Manual Initiation

- TS 3.3.13, “Engineered Safety Feature Actuation System (ESFAS) Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization,” Table 3.3.13-1
    - Function 1, Main Control Room Air Supply Iodine or Particulate Radiation – High 2
- TS 3.3.19, “Diverse Actuation System (DAS) Manual Controls,” Table 3.3.19-1
    - Function 4, Automatic Depressurization System (ADS) stage 1 valves
    - Function 5, ADS stage 2 valves
    - Function 6, ADS stage 3 valves
    - Function 7, ADS stage 4 valves
    - Function 9, Containment recirculation valves
    - Function 10, Passive containment cooling drain valves
    - Function 11, Selected containment isolation valves
- TS 3.4.11, “Automatic Depressurization System (ADS) – Operating”
- TS 3.5.1, “Accumulators”
- TS 3.5.6, “In-containment Refueling Water Storage Tank (IRWST) – Operating” only for containment recirculation flow paths (injection flow paths not excluded)
- TS 3.6.1, “Containment”
- TS 3.6.2, “Containment Air Locks”
- TS 3.6.3, “Containment Isolation Valves”
- TS 3.6.6, “Passive Containment Cooling System (PCS)”
- TS 3.6.8, “pH Adjustment”
- TS 3.7.4, “Secondary Specific Activity”
- TS 3.7.10, “Steam Generator (SG) Isolation Valves” only for PORV and PORV block valves (SG blowdown isolation valve not excluded)

(10) Operational Program Implementation

SNC shall implement the programs or portions of programs identified below, on or before the date SNC achieves the following milestones:

(a) Environmental Qualification Program implemented before initial fuel load;

(b) Reactor Vessel Material Surveillance Program implemented before initial criticality;

(c) Preservice Testing Program implemented before initial fuel load;

(d) Containment Leakage Rate Testing Program implemented before initial fuel load;

(e) Fire Protection Program

1. The fire protection measures in accordance with Regulatory Guide (RG) 1.189 for designated storage building areas (including adjacent fire areas that could affect the storage area) implemented before initial receipt

of byproduct or special nuclear materials that are not fuel (excluding exempt quantities as described in 10 CFR 30.18);

2. The fire protection measures in accordance with RG 1.189 for areas containing new fuel (including adjacent areas where a fire could affect the new fuel) implemented before receipt of fuel onsite;

3. All fire protection program features implemented before initial fuel load;

(f) Standard Radiological Effluent Controls implemented before initial fuel load;

(g) Offsite Dose Calculation Manual implemented before initial fuel load;

(h) Radiological Environmental Monitoring Program implemented before initial fuel load;

(i) Process Control Program implemented before initial fuel load;

(j) Radiation Protection Program (RPP) (including the ALARA principle) or applicable portions as identified in UFSAR Section 12.5 thereof:

1. RPP features applicable to receipt of by-product, source, or special nuclear materials (excluding exempt quantities as described in 10 CFR 30.18) implemented before initial receipt of such materials;

2. RPP features (including the ALARA principle) applicable to new fuel implemented before receipt of initial fuel on site;

3. All other RPP features (including the ALARA principle) except for those applicable to control radioactive waste shipment implemented before initial fuel load;

4. RPP features (including the ALARA principle) applicable to radioactive waste shipment implemented before first shipment of radioactive waste;

(k) Reactor Operator Training Program implemented 18 months before the scheduled date of initial fuel load;

(l) Motor-Operated Valve Testing Program implemented before initial fuel load;

(m) Initial Test Program (ITP)

1. Preoperational Test Program implemented before the first preoperational test; and

2. Startup Test Program implemented before initial fuel load;

(n) Special Nuclear Material Control and Accounting Program implemented before initial receipt of special nuclear material; and

(o) Special Nuclear Material Physical Protection Program implemented before initial receipt of special nuclear material on site.

(11) <u>Operational Program Implementation Schedule</u>

No later than 12 months after issuance of the COL, SNC shall submit to the Director of NRO, or the Director's designee, a schedule for implementation of the operational programs listed in UFSAR Table 13.4-201, including the associated estimated date for initial loading of fuel. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until all the operational programs listed in UFSAR Table 13.4-201 have been fully implemented.

(12) <u>Site- and Unit-specific Conditions</u>

(a) SNC shall either remove and replace, or shall improve, the soils directly above the blue bluff marl for soils under or adjacent to Seismic Category I structures, to eliminate any liquefaction potential.

(b) Before commencing installation of individual piping segments and connected components in their final locations, SNC shall complete the as-designed pipe rupture hazards analysis for compartments (rooms) containing those segments in accordance with the criteria outlined in the UFSAR Sections 3.6.1.3.2 and 3.6.2.5, and shall inform the Director of NRO, or the Director's designee, in writing, upon the completion of this analysis and the availability of the as-designed pipe rupture hazards analysis reports.

(c) Before commencing installation of individual piping segments, identified in UFSAR Section 3.9.8.7, and connected components in their final locations in the facility, SNC shall complete the analysis of the as-designed individual piping segments and shall inform the Director of NRO, or the Director's designee, in writing, upon the completion of these analyses and the availability of the design reports for the selected piping packages.

(d) No later than 180 days before initial fuel load, SNC shall submit to the Director of NRO, or the Director's designee, in writing, a fully developed set of plant-specific emergency action levels (EALs) for VEGP Unit 4 in accordance with the criteria defined in Amendment No. 76. The EALs shall have been discussed and agreed upon with State and local officials.

No later than 180 days before initial fuel load, SNC shall submit to the Director of NRO, or the Director's designee, in writing, an assessment of emergency response staffing performed in accordance with NEI 10-05, "Assessment of On-Shift Emergency Response Organization Staffing and Capabilities," Revision 0.

(e) SNC shall not revise or modify the provisions of Sections 5.3, 5.4, 5.6, 5.9, and 5.10 of the Special Nuclear Material (SNM) Physical Protection Program until the requirements of 10 CFR 73.55 are implemented.

(f) No later than 12 months after issuance of the COL, SNC shall submit to the Director of NRO, or the Director's designee, a schedule for implementation of the following license conditions. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until each license condition has been fully implemented. The schedule shall identify the completion of or implementation of the following:

1. The construction and inspection procedures for steel concrete composite (SC) construction activities for seismic Category I nuclear island modules (including shield building SC modules) described in UFSAR Section 3.8.4.8;

2. The spent fuel rack Metamic Coupon monitoring program (before initial fuel load);

3. Implementation of the flow accelerated corrosion (FAC) program including construction phase activities (before initial fuel load);

4. A turbine maintenance and inspection program, which must be consistent with the maintenance and inspection program plan activities and inspection intervals identified in UFSAR Section 10.2.3.6 (before initial fuel load);

5. The availability of documented instrumentation uncertainties to calculate a power calorimetric uncertainty (before initial fuel load);

6. The availability of administrative controls to implement maintenance and contingency activities related to the

power calorimetric uncertainty instrumentation (before initial fuel load);

7. The site-specific severe accident management guidelines (before startup testing);

8. The operational and programmatic elements of the mitigative strategies for responding to circumstances associated with loss of large areas of the plant due to explosions or fire developed in accordance with 10 CFR 50.54(hh)(2) (before initial fuel load); and

9. The ITP procedures identified in UFSAR Section 14.2.3:

    a. administrative manual (before the first preoperational test)
    b. preoperational testing (before scheduled performance)
    c. startup testing (before initial fuel load)

(g) Before initial fuel load, SNC shall:

1. Update the seismic interaction analysis in UFSAR Section 3.7.5.3 to reflect as-built information, which must be based on as-procured data, as well as the as-constructed condition;

2. Reconcile the seismic analyses described in Section 3.7.2 of the UFSAR, to account for detailed design changes, including, but not limited to, those due to as-procured or as-built changes in component mass, center of gravity, and support configuration based on as-procured equipment information;

3. Calculate the instrumentation uncertainties of the actual plant operating instrumentation to confirm that either the design limit departure from nucleate boiling ratio (DNBR) values remain valid or that the safety analysis minimum DNBR bounds the new design limit DNBR values plus DNBR penalties;

4. Update the pressure temperature (P-T) limits using the pressure temperature limits report (PTLR) methodologies approved in the UFSAR, using the plant-specific material properties or confirm that the reactor vessel material properties meet the specifications of and use the Westinghouse generic PTLR curves;

5. Verify that plant-specific belt line material properties are consistent with the properties given in UFSAR Section 5.3.3.1 and Tables 5.3-1 and 5.3-3. The verification must include a pressurized thermal shock (PTS) evaluation based on as-procured reactor vessel material data and the projected neutron fluence for the

plant design objective. Submit this PTS evaluation report to the Director of NRO, or the Director's designee, in writing, at least 18 months before initial fuel load;

6. Review differences between the as-built plant and the design used as the basis for the AP1000 seismic margin analysis. SNC shall perform a verification walkdown to identify differences between the as-built plant and the design. SNC shall evaluate any differences and must modify the seismic margin analysis as necessary to account for the plant-specific design and any design changes or departures from the certified design. SNC shall compare the as-built structures, systems, and components (SSC) high confidence, low probability of failures (HCLPFs) with those assumed in the AP1000 seismic margin evaluation, before initial fuel load. SNC shall evaluate deviations from the HCLPF values or assumptions in the seismic margin evaluation due to the as-built configuration and final analysis to determine if vulnerabilities have been introduced;

7. Review differences between the as-built plant and the design used as the basis for the AP1000 probabilistic risk assessment (PRA) and UFSAR Table 19.59-18. SNC shall evaluate the plant-specific PRA-based insight differences and shall modify the plant-specific PRA model as necessary to account for the plant-specific design and any design changes or departure from the PRA certified in Rev. 19 of the AP1000 DCD;

8. Review differences between the as-built plant and the design used as the basis for the AP1000 internal fire and internal flood analysis. SNC shall evaluate the plant-specific internal fire and internal flood analyses and shall modify the analyses as necessary to account for the plant-specific design and any design changes or departures from the design certified in Rev. 19 of the AP1000 DCD; and

9. Perform a thermal lag assessment of the equipment listed in UFSAR Tables 19D-8 and 19D-9 to provide additional assurance that this equipment can perform its severe accident functions during environmental conditions resulting from hydrogen burns associated with severe accidents. SNC shall perform this assessment for equipment used for severe accident mitigation that has not been tested at severe accident conditions. SNC shall assess the ability of the equipment to perform during accident hydrogen burns using the environment enveloping method or the test based thermal analysis

method described in Electric Power Research Institute (EPRI) NP-4354, “Large Scale Hydrogen Burn Equipment Experiments.”

10. Implement a surveillance program for explosively actuated valves (squib valves) that includes the following provisions in addition to the requirements specified in the edition of the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) as incorporated by reference in 10 CFR 50.55a.

    a. Preservice Testing

       All explosively actuated valves shall be preservice tested by verifying the operational readiness of the actuation logic and associated electrical circuits for each explosively actuated valve with its pyrotechnic charge removed from the valve. This must include confirmation that sufficient electrical parameters (voltage, current, resistance) are available at the explosively actuated valve from each circuit that is relied upon to actuate the valve. In addition, a sample of at least 20% of the pyrotechnic charges in all explosively actuated valves shall be tested in the valve or a qualified test fixture to confirm the capability of each sampled pyrotechnic charge to provide the necessary motive force to operate the valve to perform its intended function without damage to the valve body or connected piping. The sampling must select at least one explosively actuated valve from each redundant safety train. Corrective action shall be taken to resolve any deficiencies identified in the operational readiness of the actuation logic or associated electrical circuits, or the capability of a pyrotechnic charge. If a charge fails to fire or its capability is not confirmed, all charges with the same batch number shall be removed, discarded, and replaced with charges from a different batch number that has demonstrated successful 20% sampling of the charges.

    b. Operational Surveillance

       Explosively actuated valves shall be subject to the following surveillance activities after commencing plant operation:

       i. At least once every 2 years, each explosively actuated valve shall undergo visual external examination and remote internal examination (including evaluation and removal of fluids or contaminants that may interfere with operation of the valve) to verify the operational readiness of the valve and its actuator. This examination shall also verify the appropriate position of the internal actuating mechanism and proper operation of remote position indicators. Corrective action shall be taken to resolve any deficiencies identified during the examination with post-maintenance testing conducted that satisfies the preservice testing requirements.

ii. At least once every 10 years, each explosively actuated valve shall be disassembled for internal examination of the valve and actuator to verify the operational readiness of the valve assembly and the integrity of individual components and to remove any foreign material, fluid, or corrosion. The examination schedule shall provide for both of the two valve designs used for explosively actuated valves at the facility to be included among the explosively actuated valves to be disassembled and examined every 2 years. Corrective action shall be taken to resolve any deficiencies identified during the examination with post-maintenance testing conducted that satisfies the preservice testing requirements.

iii. For explosively actuated valves selected for test sampling every 2 years in accordance with the ASME OM Code, the operational readiness of the actuation logic and associated electrical circuits shall be verified for each sampled explosively actuated valve following removal of its charge. This must include confirmation that sufficient electrical parameters (voltage, current, resistance) are available for each valve actuation circuit. Corrective action shall be taken to resolve any deficiencies identified in the actuation logic or associated electrical circuits.

iv. For explosively actuated valves selected for test sampling every 2 years in accordance with the ASME OM Code, the sampling must select at least one explosively actuated valve from each redundant safety train. Each sampled pyrotechnic charge shall be tested in the valve or a qualified test fixture to confirm the capability of the charge to provide the necessary motive force to operate the valve to perform its intended function without damage to the valve body or connected piping. Corrective action shall be taken to resolve any deficiencies identified in the capability of a pyrotechnic charge in accordance with the preservice testing requirements.

This license condition shall expire upon (1) incorporation of the above surveillance provisions for explosively actuated valves into the facility's inservice testing program, or (2) incorporation of inservice testing requirements for explosively actuated valves in new reactors (i.e., plants receiving a construction permit, or combined license for construction and operation, after January 1, 2000) to be specified in a future edition of the ASME OM Code as incorporated by reference in 10 CFR 50.55a, including any conditions imposed by the NRC, into the facility's inservice testing program.

(13) <u>Departures from Plant-specific DCD Tier 2* Information</u>

(a) SNC is exempt from the requirements of 10 CFR Part 52, Appendix D, Paragraphs VIII.B.6 and VIII.B.5.a for prior NRC approval of departures from Tier 2* information and departures from Tier 2 information involving a change to or departure from Tier 2* information; except for departures that:

1. Involve a deviation from a code or standard credited in the plant-specific DCD for establishing the criteria for the design or construction of a structure, system, or component (SSC) important to safety,

2. Result in a change to a design process described in the plant-specific DCD that is material to implementation of an industry standard or endorsed regulatory guidance,

3. (i) Result in a change to the fuel criteria evaluation process, the fuel principal design requirements, or the nuclear design of the fuel or the reactivity control system that is material to a fuel or reactivity control

system design function, or the evaluation process in WCAP-12488, "Westinghouse Fuel Criteria Evaluation Process," or

(ii) Result in any change to the maximum fuel rod average burn-up limits; or the small break LOCA analysis information in UFSAR Subsections 15.6.5.4B.2.2 or 15.6.5.4B.2.3,

4. Adversely affect the containment debris limits or debris screen design criteria,

5. Change the Reactor Coolant Pump (RCP) type from a canned motor to a different type of RCP,

6. Result in a change to the Passive Residual Heat Removal Heat Exchanger natural circulation test (first plant test), the Core Makeup Tank Heated Recirculation Tests (first three plants test), or the Automatic Depressurization System Blowdown Test (first three plants test) that is material to the test objectives or test performance criteria,

7. Involve structural materials or analytical or design methods, including design codes and analytical assumptions, that deviate from those credited in the plant-specific DCD for critical sections,

8. Result in a change to the design of the steel faceplates, internal trusses, tie bars, or headed studs of the steel-concrete (SC) module walls in the Nuclear Island or the Shield Building, including SC-to-reinforced concrete (RC) connections,

9. Result in an increase in the demand to capacity (D/C) ratio of a critical section of the structure. SNC shall determine the D/C ratio under this condition for each critical section structural member including, but not limited to, wall segments, wall sections, concrete panels, slabs, or basemat sections, affected by a departure by:

   (i) Using the Tier 2* information in the UFSAR Section 3.8 or Appendix 3H table that directly states the D/C ratio or states the area of steel provided and the area of steel required for the affected structural member, or

   (ii) Providing the same total area of steel across the entire critical section using any combination of rebar sizes and spacing allowed by the design basis codes used in the UFSAR as the total area of steel specified in UFSAR Section 3.8 and Appendix 3H tables marked Tier 2*;

(b) For a departure from Tier 2* information that does not require prior NRC approval under the exemption in License Condition 2.D.(13)(a), SNC may take the departure provided that SNC complies with the requirements for Tier 2 departures in 10 CFR Part 52, Appendix D, Paragraph VIII.B.5, as modified by the exemption in License Condition 2.D.(13)(a). For each departure authorized by this License Condition:

1. The departure or change to Tier 2* information shall remain Tier 2* information in the plant-specific DCD.

2. SNC shall prepare and maintain a written evaluation that provides the bases for its determinations regarding the criteria in License Condition 2.D.(13)(a). In the report that 10 CFR Part 52, Appendix D, Section X.B.1 requires SNC to submit, SNC shall include a brief description of each departure and a summary of the evaluation of the departure.

E. The licensees shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

F. Exemptions

(1) The following exemption from any part of the referenced design certification rule meets the requirements of 10 CFR 52.7 and Section VIII.A.4, VIII.B.4, or VIII.C.4 of Appendix D to 10 CFR Part 52, is authorized by law, will not present an undue risk to the public health or safety, and is consistent with the common defense and security. Special circumstances are present in that the application of the regulation in this particular circumstance is not necessary to achieve the underlying purpose of the rule (10 CFR 50.12(a)(2)(ii)) as described in the application and the staff SER dated August 5, 2011.

(a) The licensees are exempt from the requirement of 10 CFR Part 52, Appendix D, Section IV.A.2.a to include a plant-specific DCD containing the same type of information and using the same organization and numbering as the generic DCD for the AP1000 certified design. This exemption is specific to the organization and numbering scheme in the FSAR and is related to departure number VEGP DEP 1-1.

(2) The following exemptions from regulations were granted in the rulemaking for the design certification rule that is referenced in the application. In accordance with 10 CFR Part 52, Appendix D, Section V, Applicable Regulations, Subsection B, and pursuant to 10 CFR 52.63(a)(5), the licensees are exempt from portions of the following regulations:

(a) Paragraph (f)(2)(iv) of 10 CFR 50.34—Plant Safety Parameter Display Console;

(b) Paragraph (c)(1) of 10 CFR 50.62—Auxiliary (or emergency) feedwater system; and

(c) Appendix A to 10 CFR Part 50, GDC 17—Second offsite power supply circuit.

(3) For the reasons set forth below, the following specific exemptions, which are outside the scope of the design certification rule referenced in the application, are granted:

(a) The licensees are exempt from the requirements of 10 CFR 70.22(b), 10 CFR 70.32(c), 10 CFR 74.31, 10 CFR 74.41, and 10 CFR 74.51 because the licensees meet the requirements of 10 CFR 70.17 and 74.7 as follows: The exemption is authorized by law, will not present an undue risk to the public health or safety, and is consistent with the common defense and security. Additionally, special circumstances are present in that the application of the regulations in this particular circumstance is not necessary to achieve the underlying purpose of the rule (10 CFR 50.12(a)(2)(ii)) as described in the COL Application and the staff SER dated August 5, 2011.

(b) The licensees are exempt from the requirements of 10 CFR 52.93(a)(1) as it relates to the exemption granted in Section 2.F.(1)(a) of this license because the exemption meets the requirements of 10 CFR 52.7, because the exemption is authorized by law, will not present an undue risk to the public health or safety, and is consistent with the common defense and security. Additionally, special circumstances are present in that the application of the regulation in this particular circumstance is not necessary to achieve the underlying purpose of the rule (10 CFR 50.12(a)(2)(ii)) as described in the staff SER dated August 5, 2011.

G. Variances

Having applied the technically relevant criteria applicable to the application for the Early Site Permit No. ESP-004, to the variances requested in the application, as described in NUREG-1923, the staff SER dated July 2009, the following variances from the early site permit (ESP) are granted:

(1) A variance (VEGP VAR 1.6-1) from Section 1.6 of the VEGP ESP site safety analysis report (SSAR) as it references Revision 15 of the AP1000 DCD instead of Revision 19 of the AP1000 DCD, which is incorporated by reference in the FSAR;

(2) The variance (VEGP VAR 1.6-2) from Section 3.8.5, Foundations, of the VEGP ESP SSAR, which references Revision 15 of the AP1000 DCD, to reference Revision 19 of the AP1000 DCD, which is incorporated by reference in the FSAR;

(3) The variance (VEGP VAR 1.6-3) from Chapter 15, Accident Analysis, of the VEGP ESP SSAR which references Revision 15 of the AP1000 DCD, to reference Revision 19 of the AP1000 DCD, which is incorporated by reference in the FSAR;

(4) The variance (VEGP VAR 1.2-1) from the site layout information in Figures 1-4, 1-5, 13.3-2, and Part 5 Figure ii, of the VEGP ESP SSAR, which is superseded by the corresponding information in FSAR Section 1.1, Figure 1.1-202;

(5) The variance (VEGP VAR 2.2-1) from the information related to onsite chemical hazards in Section 2.2.3.2.3 and Table 2.2-6 of the VEGP ESP SSAR, which is superseded by the corresponding information contained in FSAR Sections 2.2 and 6.4; and

(6) The variance (VEGP VAR 2.3-1) from the information related to design-basis temperature characteristics in Section 2.3.1.5 and Table 1-1 of the VEGP ESP SSAR, which is superseded by the corresponding information contained in FSAR Section 2.3.1.5 and Table 2.0-201, which conforms to AP1000 DCD, Revision 19.

H. Following SNC's ITAAC closure notifications under paragraph (c)(1) of 10 CFR 52.99 until the Commission makes the finding under 10 CFR 52.103(g), SNC shall notify the NRC, in a timely manner, of new information that materially alters the bases for determining that either inspections, tests, or analyses were performed as required, or that acceptance criteria are met. The notification must contain sufficient information to demonstrate that, notwithstanding the new information, the prescribed inspections, tests, or analyses have been performed as required, and the prescribed acceptance criteria are met.

I. SNC shall maintain the guidance and strategies developed in accordance with 10 CFR 50.54(hh)(2).

J. This license is effective as of February 10, 2012, and shall expire at midnight on the date 40 years from the date that the Commission finds that the acceptance criteria in the combined license are met in accordance with 10 CFR 52.103(g).

FOR THE NUCLEAR REGULATORY
COMMISSION

/RA/

Michael R. Johnson, Director
Office of New Reactors

Appendices:
Appendix A – Technical Specifications
Appendix B – Environmental Protection Plan
Appendix C – Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

APPENDIX A

VOGTLE ELECTRIC GENERATING PLANT

UNITS 3 AND 4

TECHNICAL SPECIFICATIONS

See Vogtle 3 License for Common Technical Specifications (ML14100A106)

APPENDIX B

VOGTLE ELECTRIC GENERATING PLANT UNIT 4

ENVIRONMENTAL PROTECTION PLAN

(NONRADIOLOGICAL)

1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) objectives are to ensure compliance with Biological Opinions issued pursuant to the Endangered Species Act of 1973, as amended (ESA), and to ensure that the Commission is kept informed of other environmental matters. The EPP is intended to be consistent with Federal, State, and local requirements for environmental protection.

2.0 Environmental Protection Issues

In the Final Supplemental Environmental Impact Statement (FSEIS) dated March 2011, the staff considered the environmental impacts associated with the construction and operation of Vogtle Electric Generating Plant Unit Nos. 3 and 4. This EPP applies to the licensees' actions affecting the protected environmental resources evaluated in the FSEIS and the licensees' actions that may affect any newly discovered protected environmental resources.

2.1 Aquatic Resources Issues

Federal agencies other than the U.S. Nuclear Regulatory Commission (NRC), such as the U.S. Environmental Protection Agency (EPA) and the U.S. Army Corps of Engineers (Corps), have jurisdiction to regulate aquatic resources under the Federal Water Pollution Control Act (Clean Water Act or CWA) and the Rivers and Harbors Appropriation Act of 1899 (RHA). Certain water quality environmental considerations identified in the FSEIS, including effluent limitations, monitoring requirements, and mitigation measures, are regulated under the licensees' CWA permits, such as National Pollutant Discharge Elimination System (NPDES) and Section 404 permits, and RHA Section 10 permit. Nothing within this EPP shall be construed to place additional requirements on the regulation of aquatic resources except the imposition of the requirements in a Biological Opinion under the ESA (see section 2.3). The licensees are required to inform the NRC of events or situations concerning aquatic resources consistent with the provisions of 10 CFR 50.72(b)(2)(xi), and this EPP does not expand any reporting requirement required by that regulation.

2.2 Terrestrial Resources Issues

Several statutes govern the regulation of terrestrial resources. For example, the U.S. Fish and Wildlife Service (FWS) regulates matters involving migratory birds and their nests in accordance with the Migratory Bird Treaty Act. Activities affecting migratory birds or their nests may require permits under the Migratory Bird Treaty Act. The FWS also regulates matters involving the protection and taking of bald and golden eagles in accordance with the Bald and Golden Eagle Protection Act. The licensees shall inform NRC of any events or situations concerning terrestrial resources consistent with the provisions of 10 CFR 50.72(b)(2)(xi), and this EPP does not expand any reporting requirement required by that regulation.

2.3 Endangered Species Act of 1973

The NRC may be required to protect some aquatic resources and terrestrial resources in accordance with the ESA. If a Biological Opinion is issued to the NRC in accordance with ESA Section 7 prior to the issuance of the combined license, the licensees shall comply with the terms and conditions set forth in the Incidental Take Statement of the Biological Opinion. If any Federally listed species or critical habitat occurs in an area affected by construction or operation of the plant that was not previously identified as occurring in such areas, including species and critical habitat that were not previously Federally listed, the licensees shall inform the NRC within four hours of discovery. The time of discovery is identified as the specific time when a decision is made to notify another agency or to issue a press release. Similarly, the licensees shall inform the NRC within four hours of discovery of any take, as defined in the ESA, of a Federally listed species or destruction or adverse modification of critical habitat. The four-hour discovery notifications shall be made to the NRC Operations Center via the Emergency Notification System. The licensees shall provide any necessary information to the NRC if the NRC initiates or reinitiates consultation under the ESA.

Unusual Event - The licensees shall inform the NRC of any onsite mortality, injury, or unusual occurrence of any species protected by the ESA within four hours of discovery, followed by a written report in accordance with Section 4.1. The time of discovery is identified as the specific time when a decision is made to notify another agency or to issue a press release. Such incidents shall be reported regardless of the licensees' assessment of causal relation to plant construction or operation.

3.0 Consistency Requirements

The licensees shall notify the NRC of proposed changes to permits or certifications concerning aquatic or terrestrial resources by providing the NRC with a copy of the proposed change(s) at the same time it is submitted to the permitting agency. The licensees shall provide the NRC with a copy of the application for renewal of permits or certifications at the same time the application is submitted to the permitting agency.

Changes to or renewals of such permits or certifications shall be reported to the NRC within 30 days following the later of the date the change or renewal is approved or the date the change becomes effective. If a permit or certification, in part or in its entirety, is appealed and stayed, the NRC shall be notified within 30 days following the date the stay is granted.

4.0 Administrative Procedures

4.1 Plant Reporting Requirements: Non-routine Reports

A written report shall be submitted to the NRC within 30 days of occurrence of any unusual event described in Section 2.3 of this EPP. The report shall: (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact and plant operating characteristics at the time of the event, (b) describe the probable cause of the event, (c) indicate the action taken to correct the reported event, (d) indicate the corrective action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection, which also require reports to other Federal, State, or local agencies, shall be reported in accordance with those reporting requirements in lieu of the requirements of this subsection. The NRC shall be provided a copy of such report at the same time it is submitted to the other agency.

4.2 Review and Audit

The licensees shall provide for review and audit of compliance with Section 2.3 of this EPP. The audits shall be conducted independently of the individual or groups responsible for performing the specific activity. A description of the organizational structure utilized to achieve the independent review and audit function and results of the audit activities shall be maintained and made available for inspection.

4.3 Records Retention

Records required by this EPP shall be made and retained in a manner convenient for review and inspection. These records shall be made available to the NRC on request. The records, data, and logs relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

4.4 Changes in Environmental Protection Plan

A request for a change in the EPP shall include an assessment of the environmental impact of the proposed change and a supporting justification. Implementation of such changes in the EPP shall not commence prior to NRC approval of the proposed changes in the form of a license amendment incorporating the appropriate revision to the EPP.

The licensees shall request a license amendment to incorporate the requirements of any Terms and Conditions set forth in the Incidental Take Statement of applicable Biological Opinions issued subsequent to the effective date of this EPP.

APPENDIX C

VOGTLE ELECTRIC GENERATING PLANT UNIT 4

INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA

(Removed by Amendment No. 194)