COMBINED LICENSE

VOGTLE ELECTRIC GENERATING PLANT UNIT 3

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

Docket No. 52-025

License No. NPF-91

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 3 filed by Southern Nuclear Operating Company, Inc. (SNC) acting on behalf of Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, 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;

D. SNC[1] 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-91 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 3, 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' final safety analysis report (FSAR), 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;

[1] 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 FSAR, 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 FSAR, 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:

1. In-Containment Refueling Water Storage Tank (IRWST) Heatup Test (first plant test as identified in AP1000 Design Control Document (DCD), Rev. 19, Section 14.2.9.1.3 Item (h));

2. Pressurizer Surge Line Stratification Evaluation (first plant test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.7 Item (d));

3. Reactor Vessel Internals Vibration Testing (first plant test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.9);

4. Core Makeup Tank Heated Recirculation Tests (first three plants test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.3 Items (k) and (w)); and

5. Automatic Depressurization System Blowdown Test (first three plants test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.3 Item (s)).

(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 AP1000 DCD Rev. 19, 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) SNC shall notify the Director of NRO, or the Director's designee, in writing, upon the successful completion of all the ITAAC included in Appendix C to this license.

(3) <u>Nuclear Fuel Loading and Pre-critical Testing</u>

(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) Upon submission of the notification required by Section 2.D.(2)(c) of this license and upon a Commission finding in accordance with 10 CFR 52.103(g) that all the acceptance criteria in the ITAAC in Appendix C to this license are met, SNC is authorized to perform pre-critical tests in accordance with the conditions specified herein;

(c) SNC shall perform the pre-critical tests identified in AP1000 DCD Rev. 19, 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 AP1000 DCD Rev. 19, 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) <u>Initial Criticality and Low-Power Testing</u>

(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 AP1000 DCD Rev. 19, Sections 14.2.10.2 and 14.2.10.3, respectively, the Natural Circulation (first plant test) identified in AP1000 DCD Rev. 19, Section 14.2.10.3.6, and the

Passive Residual Heat Removal Heat Exchanger (first plant test) identified in AP1000 DCD Rev. 19, Section 14.2.10.3.7;

(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 AP1000 DCD Rev. 19, 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) <u>Power Ascension Testing</u>

(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 AP1000 DCD Rev. 19, Section 14.2.10.4, the Rod Cluster Control Assembly Out of Bank Measurements (first plant test) identified in the AP1000 DCD, Rev. 19, Section 14.2.10.4.6, and the Load Follow Demonstration (first plant test) identified in AP1000 DCD, Rev. 19, Section 14.2.10.4.22;

(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 AP1000 DCD Rev.19, 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) <u>Maximum Power Level</u>

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 FSAR, in accordance with the conditions specified herein.

(7) Reporting Requirements

(a) Within 30 days of a change to the initial test program described in FSAR 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, Environmental Protection Plan, and ITAAC in Appendices A, B, and C, respectively, of this license 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).

(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 FSAR 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

1. Construction Test Program implemented before the first construction test;

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

3. 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) Operational Program Implementation Schedule

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 FSAR 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 FSAR Table 13.4-201 have been fully implemented.

(12) Site- and Unit-specific Conditions

(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 AP1000 DCD, Rev. 19, 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 AP1000 DCD, Rev. 19, 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 3 in accordance with Nuclear Energy Institute (NEI) 07-01, "Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors," Revision 0, with no deviations. The EALs shall have been discussed and agreed upon with State and local officials.

(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 AP1000 DCD Rev. 19, 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 FSAR 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 pre-operational and startup procedures (including the site-specific startup administration manual) identified in FSAR Section 14.2.3 (before initiating the initial test program).

(g) Before initial fuel load, SNC shall:

1. Update the seismic interaction analysis in AP1000 DCD, Rev. 19, Section 3.7.3.5 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 AP1000 DCD, Rev. 19, 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 AP1000 DCD, Rev. 19, 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 AP1000 DCD Rev. 19, 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 the AP1000 DCD, Rev. 19, 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 as-built equipment listed in Tables 6b and 6c in Attachment A of APP-GW-GLR-069, "Equipment Survivability Assessment," 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 as-built 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.

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 FSAR 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)