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

SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS

RELATED TO AMENDMENT NOS. 151 AND 150

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MEAG POWER SPVM, LLC

MEAG POWER SPVJ, LLC

MEAG POWER SPVP, LLC

CITY OF DALTON, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT UNITS 3 AND 4

DOCKET NOS. 52-025 AND 52-026

1.0 INTRODUCTION

By letter dated August 3, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18215A382), the Southern Nuclear Operating Company (SNC) requested that the Nuclear Regulatory Commission (NRC or the Commission) amend Vogtle Electric Generating Plant (VEGP) Units 3 and 4, Combined License (COL) Numbers NPF-91 and NPF-92, respectively. The requested amendment requires changes to the initial test program (ITP) in the Updated Final Safety Analysis Report (UFSAR) in the form of departures from the incorporated plant-specific Design Control Document (DCD) Tier 2* and Tier 2 information and related changes to the VEGP Units 3 and 4 COL and plant-specific Tier 1 information, with corresponding changes to the associated COL Appendix C information.

In license amendment request (LAR) 18-019, SNC seeks approval to utilize and evaluate the results of three tests performed in China on new AP1000 power reactor facilities at Sanmen Units 1 and 2 and Haiyang Unit 1 as part of the ITP for SNC's VEGP Units 3 and 4. These tests are used to further establish unique phenomenological performance parameters of certain AP1000 design features beyond testing performed for the Design Certification of the AP600 that will not change from plant to plant. Some of these tests are required only for the first plant and others are required only for the first three plants and thereafter, because of the standardization of the AP1000 design, would not be required as part of the ITP for subsequent plants. "First plant only" and "first three plant only" tests are defined and listed in AP1000 Design Control Document (DCD) Revision 19 Tier 2 Section 14.2.5. The requested amendment includes changes to COL Condition 2.D.(2)(a) and plant-specific Tier 1 Section 2.1.3 to credit previously

completed first plant only and first three plant only testing performed in China at Sanmen Units 1 and 2 and Haiyang Unit 1, and revise the COL to delete conditions requiring that the following first plant only and first three plant only tests be conducted on VEGP Units 3 and 4: In-Containment Refueling Water Storage Tank (IRWST) Heatup Test, Reactor Vessel Internals Vibration Testing, and Core Makeup Tank (CMT) Heated Recirculation Tests. The staff's review of LAR 18-019 considers topics associated with crediting first plant only and first three plant only tests communicated to SNC by letter dated January 13, 2012 (ADAMS Accession No. ML120040121).

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 52.63(b)(1), SNC also requested an exemption from the provisions of 10 CFR Part 52, Appendix D, "Design Certification Rule for the AP1000 Design," Section III.B, "Scope and Contents." This exemption request will allow a departure from the corresponding portions of the certified information in Tier 1 of the generic DCD.[1] In order to modify the UFSAR (the plant-specific design control document (PS-DCD)) Tier 1 information, the NRC must find the licensee's exemption request included in its submittal for the license amendment request (LAR) to be acceptable. The staff's review of the exemption request, as well as the LAR, is included in this safety evaluation.

2.0 REGULATORY EVALUATION

SNC requests approval to utilize and evaluate the results of the first plant only tests and the first three plant only tests performed in China at Sanmen Units 1 and 2 and Haiyang Unit 1 as part of the ITP for SNC's VEGP Units 3 and 4. The requested amendment seeks to credit these tests by proposing changes to COL License Condition 2.D.(2)(a), plant-specific Tier 1 information, and UFSAR Tier 2 information. Specifically, the proposed changes would revise the COL to delete conditions requiring that the following first plant only and first three plant only tests be conducted on VEGP Units 3 and 4: In-Containment Refueling Water Storage Tank (IRWST) Heatup Test, Reactor Vessel Internals Vibration Testing, and Core Makeup Tank (CMT) Heated Recirculation Tests. Tier 1 information is defined in 10 CFR Part 52, Appendix D Section II.D. The license condition (2).D.(2), "Pre-operational Testing", requires, in part, that: (a) SNC perform the design-specific preoperational tests set forth in UFSAR Section 14.2.9.1, (b) SNC review and evaluate the tests and confirm that the systems perform their specified functions in UFSAR Section 14.2.9, and (c) SNC notify the Director of NRO upon successful completion of the tests in 2.D.(2)(a). The deletion of the conditions in (2).D.(2)(a) to perform the tests outlined above per the amendment request would obviate the need for the licensee to perform the tests and associated evaluation and notification required by the license in (2).D.(2)(b) and (2).D.(2)(c). Approval of the amendment and exemption would allow SNC to use the results of these tests that were performed on Sanmen Units 1 and 2 and Haiyang Unit 1 as evaluated by the licensee as part of its ITP.

The staff considered the following regulatory requirements in reviewing the LAR that included the proposed UFSAR changes.

[1] While SNC describes the requested exemption as being from Section III.B of 10 CFR Part 52, Appendix D, the entirety of the exemption pertains to proposed departures from Tier 1 information in the PS-DCD. In the remainder of this evaluation, the NRC will refer to the exemption as an exemption from Tier 1 information to match the language of Section VIII.A.4 of 10 CFR Part 52, Appendix D, which specifically governs the granting of exemptions from Tier 1 information.

10 CFR Part 52, Appendix D, Section VIII.A.4, states that exemptions from Tier 1 information are governed by 10 CFR 52.63(b)(1) and 52.98(f). It also states that the Commission will deny such a request if the design change causes a significant reduction in plant safety otherwise provided by the design.

10 CFR Part 52, Appendix D, Section VIII.B.5.a allows an applicant or licensee who references this appendix to depart from Tier 2 information, without prior NRC approval, unless the proposed departure involves a change to or departure from Tier 1 information, Tier 2* information, or the Technical Specifications, or requires a license amendment under paragraphs B.5.b or B.5.c of the section.

10 CFR 52.63(b)(1) allows the licensee who references a design certification rule to request NRC approval for an exemption from one or more elements of the certification information. The Commission may only grant such a request if it determines that the exemption will comply with the requirements of 10 CFR 52.7, which, in turn, points to the requirements listed in 10 CFR 50.12 for specific exemptions. In addition to the factors listed in 10 CFR 52.7, the Commission shall consider whether the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. Therefore, any exemption from the Tier 1 information certified by Appendix D to 10 CFR Part 52 must meet the requirements of 10 CFR 50.12, 52.7, and 52.63(b)(1).

10 CFR 52.98(f) requires NRC approval for any modification to, addition to, or deletion from the terms and conditions of a COL. These changes involve a change to COL License Condition 2.D.(2)(a) and COL Appendix C Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) information, with corresponding changes to the associated PS-DCD Tier 1 information. Therefore, NRC approval is required prior to making the plant specific proposed changed in this license amendment request.

10 CFR Part 50, Appendix B requires that licensees apply a quality assurance (QA) program to the design, fabrication, construction, and testing of structures, systems, and components.

3.0 TECHNICAL EVALUATION

3.1 TECHNICAL EVALUATION OF THE REQUESTED CHANGES

By letter dated January 13, 2012 (ADAMS Accession No. ML120040121), the staff communicated six general topics that should be considered in a submittal requesting to credit previously conducted first plant only or first three plant only tests performed in China. These six topics were considered as part of the staff's review of LAR 18-019. Specifically:

1. Details regarding the staff's audit and review of documents regarding the quality assurance program review, test control program procedures, test specifications, and post-test analysis are discussed in Sections 3.1.1 through 3.1.4 of this safety evaluation (SE). The staff confirmed that these documents were written in English.

2. Staff's review of quality assurance documents is provided in Section 3.1.1 of this SER. Considerations specific to instrument calibration are discussed in Section 3.1.4 of this SE and in the staff's audit report (ADAMS Accession No. ML18338A090).

3. Fidelity of the design, construction, and as-built conditions of Sanmen Units 1 and 2, and Haiyang Unit 1 as compared to VEGP Units 3 and 4 is discussed in Sections 3.1.2 through 3.1.4 of this SE.

4. The adequacy of this ITP, including a description of the administrative controls governing the ITP at Sanmen Units 1 and 2, and Haiyang Unit 1 is discussed in Section 3.1.1 of this SE and in the staff's inspection report (ADAMS Accession No. ML18176A395).

5. Critical test parameters, calculations, and verification methods used during the ITP are discussed in Sections 3.1.2 and 3.1.4 of this SE, and in the staff's audit report (ADAMS Accession No. ML18338A090).

6. Staff's review of quality assurance documents and procedures is provided in Section 3.1.1 of this SE. During the audit, the staff reviewed documents in English that were subsequently translated into Chinese. Although the staff did not verify the process used to translate the documents from English, NRC inspectors reviewed the tests being used and were able to observe testing while following the procedures in English. There were no issues identified with the conduct of the tests that would suggest a problem with translation.

3.1.1 TECHNICAL EVALUATION OF THE QUALITY ASSURANCE PROGRAM

SNC LAR-18-019 seeks NRC approval for SNC to credit previously completed first plant only tests and the first three plants only test performed in China at Sanmen Units 1 and 2 and Haiyang Unit 1 to satisfy the applicable testing requirements for SNC's VEGP Units 3 and 4. In LAR-18-019, SNC requests changing the design-specific preoperational tests listed in COL Condition 2.D.(2)(a), and the first plant and first three plant only tests described in UFSAR Sections 14.2.5, 14.2.9.1.3 and 14.2.9.1.9. The requested changes would revise the COL by deleting the following conditions: 2.D.(2)(a)1 "In-Containment Refueling Water Storage Tank (IRWST) Heatup Test," 2.D.(2)(a)3 "Reactor Vessel Internals Vibration Testing," and 2.D.(2)(a)4 "Core Makeup Tank Heated Recirculation Tests."

SNC is required to apply a QA program consistent with 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," for activities affecting the safety-related functions of structures, systems, or components (SSCs), which also includes testing. SNC reviewed the applicability and acceptability of the data derived from preoperational testing at the Sanmen Units 1 and 2 and Haiyang Unit 1 AP1000 nuclear plants that was conducted with assistance from Westinghouse Electric Company (Westinghouse). Westinghouse assisted the Sanmen Units 1 and 2 and Haiyang Unit 1 owners with testing activities and used its own (Westinghouse's) QA program to analyze the collected data. SNC evaluated the test results to ensure that adequate QA processes were in place to verify the validity and applicability of the data collected. This included ensuring (1) that the testing methods and conditions were properly controlled, (2) that any deviations or anomalies identified during testing were properly evaluated, (3) that the test data was evaluated for acceptability against appropriate acceptance criteria, (4) that there were no changes to the standard AP1000 design implemented at Sanmen Units 1 and 2 and Haiyang Unit 1 that could impact the applicability of the data collected.

In performing its review, the staff assessed Westinghouse's and SNC's evaluations with regards to the four objectives listed above. To address the adequacy of the quality controls

implemented at Sanmen and Haiyang, the staff focused on the evaluation of anomalies and the evaluation test data. During the audit (ADAMS Accession No. ML18338A090), the staff reviewed SNC's evaluation study that was conducted to compare Appendix B to 10 CFR Part 50 against the applicable portions of the Sanmen Unit 1 and 2 and Haiyang Unit 1 Q4 programs. In addition, the staff considered documented observations by NRC inspectors of selected portions of the Sanmen Unit 1 and 2 preoperational test program under the auspices of the bilateral agreement between the NRC and the Chinese National Nuclear Safety Administration (CNNSA). The staff also considered the results of NRC vendor inspections performed at Westinghouse associated with the development and implementation of the preoperational test program and with design control activities of the AP1000 plants under construction (ADAMS Accession No. ML18176A395). Staff actions performed as part of this review are discussed in more detail below.

Westinghouse developed the AP1000 first plant only tests and the first three plants test used in China at Sanmen Units 1 and 2 and Haiyang Unit 1. The testing methodology, acceptance criteria, and evaluation of test results were conducted under Westinghouse's QA program. SNC's surveillance report, CMP-ITP-2018-7-13761, dated June 9, 2018, evaluated Westinghouse's preoperational test program and QA program controls. Specifically, SNC's surveillance assessed Westinghouse's preoperational tests specifications, integrated test program, qualification records of personnel, selection of qualified personnel, and applicability of QA program requirements. The staff reviewed surveillance report CMP-ITP-2018-7-13761 during the audit (ADAMS Accession No. ML18338A090) and noted that the findings identified from this surveillance report were entered into the applicable corrective action programs and that the proposed corrective actions appeared appropriate and were evaluated by appropriate personnel.

The staff also reviewed a report of an NRC inspection performed at Westinghouse on May 21-25, 2018 (ADAMS Accession No. ML18176A395) to evaluate, in part, the design control and configuration management processes of Westinghouse's AP1000 ITP. This inspection also assessed whether adequate processes were in place for determining the impact of activities (e.g., design change, non-conformance, issuance of new drawings, specifications or procedures, etc.) on licensing basis documents for the domestic AP1000 design. This assessment included identifying any changes made to the Sanmen or Haiyang units that would constitute a departure from the standardized AP1000 design. The NRC inspection team reviewed the processes to ensure that any changes affecting the design were adequately implemented and formally captured, tracked, and incorporated into future revisions of the affected design.

During the May 21-25, 2018 inspection, NRC inspectors also selected and reviewed samples of preoperational test specifications and administrative procedures based on the ITP for the Sanmen Units 1 and 2 and Haiyang Unit 1 plants, to verify appropriate implementation of the technical bases for the changes identified. Additionally, the NRC inspection team reviewed current versions of test specifications and test procedures to confirm that proposed changes were adequately incorporated into design documentation, or were appropriately identified and tracked for formal configuration management of the affected design or testing documentation.

During the May 21-25, 2018 inspection, NRC inspectors selected a sample from Westinghouse's preoperational test specifications and procedures for VEGP Units 3 and 4. The NRC inspection team determined the test procedures contained adequate acceptance criteria that were consistent with the guidance contained in Regulatory Guide (RG) 1.68, "Initial Test

Programs for Water-Cooled Nuclear Power Plants." The NRC inspection team also determined that the test procedures documented sufficient information to permit adequate evaluation of the test results for those cases where a calculation or post-test evaluation is required to demonstrate that the test objective was met. The NRC inspection team confirmed, for the sample reviewed during this inspection, that Westinghouse's data and analysis supported the conclusion that the test acceptance criteria were met.

In addition, during the May 21-25, 2018 inspection, NRC inspectors performed a traceability review of test specifications and test procedures to confirm the test acceptance criteria were adequately translated from the design information into test specifications and test procedures. Lastly, the NRC inspection team reviewed the methods of testing, data acquisition, and reference to performance of either calculations or analysis credited for meeting the acceptance criteria to confirm consistency with the ITAAC descriptions in the domestic AP1000 design. The staff concluded through its review of SNC's oversight of Westinghouse's ITP controls and the NRC's inspection of Westinghouse's process and procedures that SNC and Westinghouse demonstrated adequate QA processes were in place in order to ensure the proper development of test procedures, appropriate development of acceptance criteria, and sufficient control over the design of the standardized AP1000. Additionally, the NRC inspection team reviewed the adequacy of the test procedures to be used to implement the ITP. No findings of significance were identified.

SNC stated that Sanmen Units 1 and 2 and Haiyang Unit 1 AP1000 first plant and first three plant only tests were performed in accordance with China's regulatory QA requirements specified in HAF-003-1991, "Safety Regulations for Quality Assurance of Nuclear Power Plants." In addition, SNC also stated that testing was performed in the presence of Westinghouse personnel as the design authority for the SSCs involved in the first plant and first three plant only testing. SNC performed a comparison of the 13 sections of HAF-003-1991 and the 18 criteria of Appendix B to 10 CFR Part 50. SNC concluded that the first plant and first three plant only tests performed at Sanmen Units 1 and 2 and Haiyang Unit 1 were conducted following QA standards similar to requirements for VEGP Units 3 and 4. The staff reviewed the applicable section of SNC's comparison associated with the first plant and first three plant only testing and determined there were adequate quality controls for testing activities. In addition, SNC performed a design oversight review ND-EN-006-F01, dated July 27, 2018, to evaluate Westinghouse's test procedures, test results, and calculations used to support crediting the first plant and first three plant only testing. SNC determined that the first plant and first three plant only tests performed at Sanmen Units 1 and 2 and Haiyang Unit 1 were conducted satisfactorily in accordance with appropriate quality procedures and oversight for crediting VEGP Units 3 and 4 first plant and first three plant only test requirements. The staff determined, after the review of documents provided during the audit (ADAMS Accession No. ML18338A090) that the regulatory requirements are similar and are sufficient for this specific application. However, the staff's agreement with SNC's comparison of HAF-003-1991 to Appendix B to 10 CFR Part 50 does not constitute the NRC's endorsement of the Chinese regulatory requirements or its equivalency to Appendix B to 10 CFR Part 50. The staff's audit evaluation of ND-EN-006-F01 determined that SNC's review of administrative manual procedures, test procedures, test reports and post-test analysis oversight was appropriate and no issues were identified that would invalidate the first plant and first three plant only testing results.

NRC inspectors also observed the Sanmen Unit 1 preoperational tests from October 6 to December 17, 2017, at the invitation of the Chinese regulator (CNNSA). The inspectors interfaced with the site owner, Westinghouse personnel, and associated contractors and

subcontractors as necessary to evaluate the oversight and controls of the tests. The NRC inspectors conducted in-plant walk downs and observed most major plant components. While on-site, the NRC inspectors observed testing, reviewed administrative and test procedures, interviewed Westinghouse and owner personnel to identify relevant start-up testing operating experience, and met with local regulators on numerous occasions to discuss various aspects of the NRC expectations for observing testing, training and qualification requirements, and conducting inspection activities. Documents supplied to NRC inspectors, for their review, were available in English.

The NRC inspectors performed an additional observation at the Sanmen Unit 1 associated with the Power Ascension Start-up Test Program from June 25 to October 5, 2018. The purpose of the observation was to assess power ascension start-up testing and any first plant only tests conducted at Sanmen Units 1 and 2 to inform the NRC's construction and operational inspection program for VEGP Units 3 and 4.

SNC observed preoperational testing at Sanmen Unit 2 from January 9 to February 8, 2018, and documented the observations in report ND-CX-001-F04. Two SNC individuals, with backgrounds in engineering and operations, were on site at Sanmen Unit 2 to perform observations of the pre-operational testing including the CMT recirculation first three plants only test. The objective of the visit was to observe the following activities for those specific tests: performance of pre-test requirements; confirmation of measuring and test equipment usage; adherence to the approved procedure; execution of test changes; handling of anomalies, problems, and/or interruptions; handling of deficiencies; recording of data; maintenance of the test narrative log; and maintenance of operator logs. Based on their observations, SNC concluded that the first three plants only tests at Sanmen Unit 2 were conducted in accordance with the test procedures.

During the audit (ADAMS Accession No. ML18338A090), the staff reviewed SNC's: oversight of Westinghouse, comparison of regulatory requirements, and evaluation of Sanmen's test procedures. The staff reviewed SNC's documented observations at the Sanmen Units 1 and 2. The staff also reviewed a report documenting NRC observations of testing activities at Sanmen during SNC's oversight and a report of an NRC staff inspection of Westinghouse's AP1000 ITP. In addition, the staff performed technical evaluations of the objective evidence supporting the proposed changes associated with IRWST heatup testing, reactor vessel internals vibration testing, and CMT recirculation testing, as detailed in Sections 3.1.2, 3.1.3, and 3.1.4 of this safety evaluation, respectively. The staff determined, from the objective evidence review, that QA program controls for the first plant and the first three plants only tests used were consistent with 10 CFR Part 50, Appendix B and support the changes proposed in Enclosure 4 of LAR 18-019 to delete conditions requiring SNC to meet the ITP requirements in COL condition 2.D(2)(a), which include the performance of the following tests: IRWST Heatup Test, Reactor Vessel Internals Vibration Testing, and CMT Heated Recirculation Tests.

3.1.2 TECHNICAL EVALUATION OF PROPOSED CHANGES ASSOCIATED WITH IRWST HEATUP TEST

As stated in the LAR, the primary function of the passive core cooling system (PXS) is to provide safety-related core cooling following a design basis event. This function is verified via testing of the passive residual heat removal (PRHR) heat exchanger. The PRHR is a heat exchanger that is submerged in the IRWST, through which reactor coolant system (RCS) fluid flows during transient conditions. One test, designated in the AP1000 DCD as being required

only for the first plant, involves observing the heatup of the IRWST resulting from actuation of the PXS.

SNC proposes to remove COL condition 2.D.(2)(a)1, which requires SNC to perform the IRWST heatup test described in UFSAR 14.2.9.1.3, as well as 14.2.9.1.3(h), which contains the test itself. SNC further proposes to add a statement in UFSAR 14.2.5 stating this test will not be performed based on successful completion of the testing at the first AP1000. In this case, the test was performed at the Sanmen Unit 1.

The stated motivation for the test in the UFSAR is to confirm the results of the PRHR tests conducted as part of the design certification process with regards to IRWST mixing, and to help quantify the conservatism in the Chapter 15 transient analyses. The test acceptance criteria, as stated in UFSAR subsection 14.2.9.1.3, item (h), is to demonstrate that the average IRWST heatup is consistent with the PRHR heat transfer modeling in the Chapter 15 analysis. In order to determine whether SNC's proposed changes are acceptable, staff reviewed SNC's evaluation of the testing in order to determine whether these test purposes were adequately satisfied by the tests performed at the Sanmen Unit 1 and establish the phenomenological performance of the IRWST heatup in accordance with the analysis.

SNC stated the test has been performed at the Sanmen Unit 1 with satisfactory results. In reaching this conclusion, SNC performed an evaluation of the test and determined the as-built average IRWST heatup characteristics are consistent with the analytical model used in UFSAR Chapter 15 analyses, and the PRHR performance as modeled is conservative. SNC stated the observed temperature profiles were developed in accordance with assumed profiles, with strong vertical temperature gradients developed in the IRWST. To confirm the conservative nature of the test, the vendor performed an analysis using LOFTRAN, comparing the test conditions to the data obtained. Staff audited the analysis (ADAMS Accession No. ML18338A090) and found that the analysis and test show very good agreement and that the LOFTRAN analysis heat transfer is conservative with respect to the test data.

Although the tests showed very little heatup in the region below the PRHR in the IRWST, this behavior is explained by the duration of the test. Consistent with the test duration, the IRWST is not heated to the point of saturation where further mixing might be expected. Further, the Chapter 15 LOFTRAN analysis accounts for this behavior by conservatively neglecting heatup of the water volume below the heat exchanger. Based on the expected behavior during the duration of the test and the considerations included in the Chapter 15 LOFTRAN analysis, the staff concludes that Chapter 15 is conservative with respect to the observed test behavior in accordance with the UFSAR subsection 14.2.9.1.3 test acceptance criteria.

The staff audited (ADAMS Accession No. ML18338A090) the detailed evaluation performed by the vendor and SNC and found it to be appropriate. This determination is based on the staff review of SNC's evaluation, which shows that data was collected under an appropriate QA program (Reference 6), test conditions for the PRHR heat transfer tests that were used to measure IRWST heatup characteristics were appropriate at the time of testing, and that Sanmen Unit 1 and VEGP Units 3 and 4 are sufficiently identical with respect to the design of the IRWST and PRHR that test data from Sanmen Unit 1 apply to VEGP Units 3 and 4. As such, staff agrees with SNC's assessment that data obtained from Sanmen Unit 1 are applicable to VEGP Units 3 and 4.

Further, reasonable assurance that the PRHR and IRWST will perform their function will be verified by tests that will be performed at VEGP Units 3 and 4. For the PXS, including the IRWST and PRHR, relevant parameters will be verified by ITAAC Nos. 2.2.03.08b and 08c, for both parameters associated with the physical dimensions of the components and specific required thermal performance characteristics. Additionally, SNC is required to perform testing of the PXS, including the PRHR, as part of tests documented in Section 14.2.9.1.3 of the UFSAR. These tests will also involve heatup of the IRWST and demonstrate efficacy of the PRHR heat exchanger.

As such, the staff finds that SNC has satisfied the intent of COL Condition 2.D.(2)(a)1 by meeting COL Condition 2.D.(2)(b) and 2.D.(2)(c) applicable to the IRWST heatup test through this LAR. The testing referenced in the LAR and the associated evaluations performed by the licensee and audited by the staff fulfill the stated purpose of the first plant test, to establish a unique phenomenological performance parameter (here, of the IRWST) beyond testing performed for the Design Certification and that will not change from plant to plant. Based on the evaluation above, the staff finds the proposed change to delete COL Condition 2.D.(2)(a)1 acceptable because SNC has demonstrated that the test requirements for the IRWST heatup test have been satisfied by the testing performed at Sanmen Unit 1 and the associated data obtained are applicable to VEGP Units 3 and 4. By extension, the associated language changes in UFSAR Section 14.2 are acceptable, as the intended purposes for the first plant IRWST heatup test have been satisfied.

3.1.3 TECHNICAL EVALUATION OF PROPOSED CHANGES ASSOCIATED WITH REACTOR VESSEL INTERNALS VIBRATION TESTING

The purpose of the Comprehensive Vibration Assessment Program (CVAP) is to verify the structural integrity of the reactor vessel internals (RVI) for flow-induced vibration prior to commercial operation. The dynamic flow-related loads considered are those associated with steady-state and anticipated transients during preoperational, initial startup, and normal operating conditions. Components instrumented for vibration measurement during the hot functional test for the AP1000 design include the core barrel (CB), core shroud (CS), secondary core support structure, lower guide tube, upper guide tube, upper support column, upper support assembly, instrumentation grid assembly (IGA), incore instrument thimble assembly (IITA) tube, IGA IITA tube support, IGA arch support, IGA column support, and IGA Quickloc column. These locations were reviewed and approved by staff as part of the standard AP1000 design certification. SNC proposed in LAR-18-019 to classify Sanmen Unit 1 as a prototype for reactor internals, as defined in RG 1.20 “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing,” and VEGP Units 3 and 4 as a non-prototype, for Category I reactor internals as defined in RG 1.20.

UFSAR subsection 14.2.5 states that “…justification shall be provided that the results of the first plant only test… are applicable to the subsequent plant.” UFSAR subsection 14.2.5 also states that “[b]ecause of the standardization of the AP1000 design, these special tests (designated as first plant only tests) are not required on follow-on plants.” RG 1.20 states that “non-prototype, Category I reactor internals are those configurations that have substantially the same arrangement, design, size and operating conditions as specified for the “valid prototype,” for which nominal differences in arrangement, design, size, and operating conditions have been shown (by test or analysis) to have no significant effect on the vibratory response and excitation of those reactor internals important to safety.” Thus, establishing Sanmen Unit 1 as the prototype for VEGP Units 3 and 4, in accordance with RG 1.20, would allow SNC to take credit

for the Sanmen Unit 1 CVAP test results to satisfy ITAAC and COL requirements of the VEGP UFSAR.

SNC has proposed several changes to the COL documentation to take credit for the Sanmen Unit 1 testing instead of conducting the tests at VEGP Units 3 and 4. Specifically, the proposed changes are as follows:

- COL Condition 2.D.(2)(a)3 requires SNC to perform reactor vessel internals vibration testing as described in UFSAR subsection 14.2.9.1.9. This COL condition is proposed to be deleted based on the successful completion of the test at Sanmen Unit 1.

- COL Appendix C (and plant-specific Tier 1) Table 2.1.3-2, ITAAC No. 2.1.03.07.i requires a vibration type test be conducted on the first unit reactor internals representative of AP1000 and that a report exists and concludes the reactor internals have no observable damage or loose parts as a result of the vibrations type test. This portion of the ITAAC is proposed to be deleted because under the proposed LAR, VEGP Units 3 and 4 will not perform an instrumented vibration test. The Sanmen Unit 1 report is being submitted as part of this amendment request to serve as the report for the first AP1000 unit referenced in the acceptance criteria of the ITAAC.

- COL Appendix C, subsection 2.1.3, is editorially revised to renumber the items under the Design Description consistent with the plant-specific Tier 1 information numbering.

- UFSAR subsection 3.9.2.4 describes the pre-operational flow-induced vibration testing of reactor internals. This subsection is proposed to be revised to describe Sanmen Unit 1 as a prototype for reactor internals and VEGP Units 3 and 4 as a non-prototype for Category I reactor internals consistent with the guidance in RG 1.20. Discussion of reference plants used prior to Sanmen Unit 1 testing is proposed to be deleted.

- UFSAR Table 3.9-4 provides the locations for the first plant reactor internals vibration measurement program transducer locations. This table is proposed to be deleted because under the proposed LAR, VEGP Units 3 and 4 will not perform an instrumented CVAP and the instruments described in the table will not be used.

- UFSAR subsection 14.2.5 describes the first plant only tests, including reactor internals vibration testing. The LAR would add a statement to UFSAR subsection 14.2.5 that the instrumented vibration test will not be run at VEGP Units 3 and 4 based on the successful completion of the test at the first AP1000 (Sanmen Unit 1) and that VEGP Units 3 and 4 would perform a non-instrumented CVAP which is consistent with the guidance of RG 1.20 for non-prototype, Category I reactor internals.

- UFSAR subsection 14.2.9.1.9, describes the RVI vibration testing. The test requirements for an instrumented CVAP are proposed to be deleted from this subsection because under the proposed LAR, VEGP Units 3 and 4 will not perform an instrumented CVAP. The LAR would add a statement to UFSAR subsection 14.2.9.1.9 that VEGP Units 3 and 4 would perform a non-instrument CVAP which is consistent with the guidance of RG 1.20 for non-prototype, Category I reactor internals.

The staff assessed SNC's LAR and determined that if the following four criteria are met, the Sanmen Unit 1 CVAP tests can be credited for use at Vogtle Units 3 and 4:

- There are no loose parts or abnormal wear due to adverse vibration following the Sanmen Unit 1 reactor internals preoperational testing.
- Measurements from the Sanmen Unit 1 reactor internals preoperational testing meet strain limits.
- Reactor internals design and operating conditions of Sanmen Unit 1 are substantially the same as those of VEGP Units 3 and 4.
- As-built dimensions associated with secondary flow paths through narrow gaps between components and subcomponents, which parallel the primary flow path, in Sanmen Unit 1 are substantially the same as those of VEGP Units 3 and 4.

3.1.3.1 LOOSE PARTS AND ABNORMAL WEAR

The staff reviewed APP-GW-GLR-179 (SM1-CVAP-T2R-200), Rev. 0, "Comprehensive Vibration Assessment Program (CVAP) Preliminary Report for the Sanmen 1 AP1000 Plant," Table 4-1, "Pre- and Post-Hot Functional Test Inspections," during the audit (ADAMS Accession No. ML18338A090) to determine if Sanmen Unit 1 experienced loose parts or abnormal wear due to adverse flow-induced vibration after testing. This report documents the locations in the RVI where the inspections revealed indications (issues identified during non-destructive examination that require further evaluation). Items of interest follow.

APP-GW-GLR-179, Location 5 (Guide Tube Flange Bolts and Locking Devices) states that locking cups were not crimped and one locking cup was damaged after Sanmen Unit 1 reactor internals preoperational testing. During the audit from September 17 to October 31, 2018 (ADAMS Accession No. ML18338A090), the staff evaluated this report and inquired as to the reason for not crimping the locking cup and the cause for damage to the locking cup. Westinghouse responded during the audit that the locking cups on the guide tube flange bolts are not crimped until after the hot functional test (HFT) is completed and the profile inspection of the guide tubes are completed satisfactorily in case there is a need to remove or relocate a guide tube. The locking cups were crimped after the profile inspection was completed. During the audit, the staff reviewed document SM1-MI01-GNR-4177 (SM1-MI01-GQR-5060), Rev. 0, "Field Deviation Notice for Dimple/Nick in Locking Cup Rim (CVAP Inspection Location 5);" the damage to the locking cup was minor. Westinghouse stated that a possible cause is a dropped tool sometime during fabrication or construction. The deviation notice states that the dimple/nick in the guide tube holddown bolt locking cup does not affect the function of the locking cup to retain the head of the bolt in the event of a bolt failure.

APP-GW-GLR-179, Table 4-1, Locations 15 (Core Barrel Outlet Nozzle Interface Surfaces), 28 (Vessel Nozzle Interface Surface Condition), and 85 (Energy Absorber to Housing Welds) state that light contact marks and some indentations due to loose parts were detected after Sanmen Unit 1 reactor internals preoperational testing. During the audit, the staff evaluated this report and inquired about the sources of the loose parts and whether design changes were made to the reactor internals to prevent future loose parts. Westinghouse responded that: (1) the source of the loose parts was the failed lead wire cover plate and associated hardware that was installed temporarily as part of the instrumented CVAP hardware, (2) as part of the lessons learned, the lead wire cover plate and attachment hardware would be modified for any future

instrumented CVAP, and (3) there were no broken or loose parts of the RVI assembly that would require a design change.

APP-GW-GLR-179, Section 4.2.2, "Post-HFT Inspection Results," states that no broken or loose parts in the reactor internals assembly were found after Sanmen Unit 1 reactor internals preoperational testing. However, SM1-CVAP-T2R-200 Table 4-1, also evaluated as part of the staff's audit, indicates that there were loose parts. Westinghouse explained during the audit that the loose parts referred to in Table 4-1 refer to the failed lead wire cover plate and associated hardware that was installed temporarily as part of the instrumented CVAP hardware (mentioned in last bullet on page 4-20 of APP-GW-GLR-179), whereas Westinghouse's statement made in Section 4.2.2 was made to document that there were no broken or loose parts that were part of the RVI themselves.

Based the above discussion, the staff concludes that there is no abnormal wear nor loose parts that would preclude use of or invalidate the CVAP from complying with RG 1.20.

3.1.3.2 MEASURED STRAIN LIMITS

SNC stated in the CVAP Preliminary Report for the Sanmen 1 AP1000 Plant (APP-GW-GLR-179) that, "[t]he measured responses are due primarily to turbulence-induced excitation. The measured data were also evaluated for evidence of vortex shedding and for excitation by reactor coolant pump (RCP)-related acoustic pressure pulsations. No evidence of vortex shedding lock-in was observed in the data for any components. Several discrete tones were observed in the RVI forced response associated with RCP pressure pulsations, but these tones did not significantly contribute to the overall forced response of the RVI structures." If the vortex shedding frequency is near a natural frequency of the structure being excited, then lock-in can occur. In this situation, the vortex shedding frequency locks-in, or assumes the frequency of the structure's natural frequency. If the structure is not damped sufficiently, resonance can produce large deflections and associated strains. SNC stated that the ASME Boiler and Pressure Vessel Code (BPV Code), Section III, Appendix N, "Dynamic Analysis Methods," Section N-1324.1(a) was used to evaluate the potential for lock-in.

According to ASME Code Section N-1324.1(a), "[i]f the reduced velocity for the fundamental vibration mode (n = 1) satisfies, $V/f_1D<1$, then both lift and drag direction, lock-in are avoided." Where V is the flow velocity, f_1 is the Strouhal frequency which corresponds to the natural frequency of the cylinder, and D is the diameter of the cylinder. A tabulation of reduced velocities and Strouhal frequencies was not found in the LAR, and as such, the staff was unable to confirm that cylindrical structures met the reduced velocity criterion nor was the staff able to examine the power spectral density (PSD) for corresponding peaks at the respective Strouhal frequencies.

However, during the staff's audit (ADAMS Accession No. ML18338A090), the staff reviewed the overall strain levels compared to criteria (acceptable strain limits), which also ensures that lock-in had not occurred. In Table 2-1, "Acceptance Criteria and Measured Responses for AP1000 Reactor Vessel Internals," of APP-GW-GLR-179, the total root mean square measured strain and corresponding acceptance criterion for all components of the Sanmen Unit 1, RV upper internals is tabulated for cylindrical components such as the upper guide tube, upper support column, IGA IITA Tube, IGA column support and the IGA Quickloc column. The staff found that the strain measurement locations are consistent with Westinghouse Document APP-CVAP-GER-004 (WCAP-17983), Rev. 0, "Comprehensive Vibration Assessment Program (CVAP)

Measurement and Inspection Programs for the AP1000 Plant," March 27, 2015. Therefore, the test data show that the acceptance criteria are met with margin to spare. Since the measured strains for the cylindrical components met the strain criterion with margin, the staff concurs that lock-in has not occurred. The staff also reviewed APP-GW-GLR-181, Rev. 0, "Comprehensive Vibration Assessment Program (CVAP) Final Report for the Sanmen 1 AP1000 Plant," during the audit and confirmed that measured reactor internals displacements meet the acceptance criteria.

The staff concludes that, based on discussion above and the strains for all components meeting the criterion, all allowable stresses are met.

3.1.3.3 REACTOR INTERNALS DESIGN AND OPERATING CONDITIONS

RG 1.20 states that "non-prototype, category I reactor internals are those configurations that have substantially the same arrangement, design, size and operating conditions as specified for the 'valid prototype,' for which nominal differences in arrangement, design, size, and operating conditions have been shown (by test or analysis) to have no significant effect on the vibratory response and excitation of those reactor internals important to safety." Verifying standardization of the reactor internal configuration between Sanmen Unit 1 and VEGP Units 3 and 4 provides the basis for concluding that the VEGP Units 3 and 4 reactor internals are substantially the same as the valid prototype, Sanmen Unit 1.

If Sanmen Unit 1 is an implementation of the AP1000 plant standard design, as approved by the NRC, then the RVI and operating conditions should be in accordance with 10 CFR Part 52, Appendix D. However, SNC must confirm that there are no departures with the PS-DCD for Sanmen Unit 1. SNC stated, "Sanmen Unit 1 is an implementation of the AP1000 plant standard design, with the RVI designed in accordance with the generic AP1000 RVI design specification and qualified in the generic AP1000 RVI design report. The differences between the generic RVI design and the Sanmen Unit 1 RVI as-built configuration were reconciled in the Sanmen Unit 1 plant-specific RVI design report in accordance with ASME BPV Code, Section III, NCA-3554. Considering the as-built configuration, the Sanmen Unit 1 RVI are substantially similar to, and therefore representative of, the generic RVI design." SNC further stated, "[f]or Sanmen Unit 1, the electrical grid operates at 50 Hz, while the RCPs are driven by variable frequency drives (VFD) operating at 60 Hz. For VEGP Units 3 and 4, both the electrical grid and RCP VFDs operate at 60 Hz. Since the VFDs operate at the same frequency in both cases, there is no impact on the CVAP. Furthermore, potential line noise or VFD noise is accounted for in the analysis/post-processing of the test data." The staff evaluated these statements and agrees that, if the VFDs operate at the same frequency there is no impact to the CVAP.

VEGP UFSAR, Table 4.1-1, "Reactor Design Comparison Table," lists the AP1000 coolant flow rates and temperatures. The RCS flow rate is 113.5×10^6 lbm/hr and the inlet coolant temperature is 535 °F. AP1000 DCD Tier 2, Table 4.1-1, "Reactor Design Comparison Table," also lists the same RCS flow rate and inlet coolant temperature. As stated in the LAR, Sanmen Unit 1 is an implementation of the AP1000 plant standard design, the Sanmen Unit 1 operating conditions are consistent with the standard AP1000 design. The operating conditions in VEGP UFSAR Table 4.1-1 are the same as in the standard AP1000 DCD Table 4.1-1. During the audit (ADAMS Accession No. ML18338A090), the staff also confirmed through the evaluation of Generic Standard AP1000 drawings that the reactor internals designs for Sanmen Unit 1 and VEGP Units 3 and 4 are identical.

Based on the discussion above, the staff concludes that reactor internals design and operating conditions of Sanmen Unit 1, and VEGP Units 3 and 4 are implementations of the AP1000 Generic Standard Design and therefore, the reactor internal configuration between Sanmen Unit 1 and VEGP Units 3 and 4 are substantially the same. Consistent with RG 1.20, SNC has demonstrated that Sanmen Unit 1 is a valid prototype for reactor internals and VEGP Units 3 and 4 are appropriately considered non-prototype for reactor internals.

3.1.3.4 AS-BUILT DIMENSIONS ASSOCIATED WITH SECONDARY FLOW

During the audit (ADAMS Accession No. ML18338A090), the staff inquired about the as-built dimensions associated with secondary flow paths through narrow gaps between components and subcomponents at Sanmen Unit 1. SNC advised that only the core plate to core barrel gap could cause instability. During the audit, SNC furnished a series of drawings associated with the gap between the core barrel and upper core plate. The staff then reviewed the discussion of the design to address leakage-flow induced instability found in the CVAP pre-test report, APP-CVAP-GER-003, Section 5.4.1.5.4. In this document, SNC identified one location in the internals where leakage instability is a potential issue, but ultimately concluded that the leakage-flow induced instability did not impact the measured strain response. The staff also reviewed the CVAP report for this location. After evaluation of the provided information, the staff agrees that instability is not a concern at this location, since the strain gauge response for the upper support assembly met the CVAP acceptance criterion.

Staff concludes that the as-built dimension comparison supports the finding that the VEGP unit configurations have substantially the same arrangement, design, size and operating conditions as the Sanmen Unit 1 reactor. Therefore, leakage flow instability is not an issue with the measured strains.

<u>Summary</u>

Based on the information provided in Enclosure 1 and Enclosure 7 to LAR 18-019 and the information documented in the staff's audit report (ADAMS Accession No. ML18338A090) as summarized above, the staff finds that SNC satisfied the CVAP requirements for the RVI test because (1) these tests have been performed for Sanmen Unit 1; (2) the CVAP test performed at Sanmen Unit 1 is applicable to VEGP Units 3 and 4; (3) the criteria of Section 3.1.3 of the SE are met, i.e. no loose parts, strain limits met, same design and operating conditions, and substantially same as-built dimensions; and (4) SNC reviewed the Westinghouse evaluations and found them acceptable. The staff confirmed this evaluation. Finally, the provisions in VEGP Units 3 and 4 UFSAR subsection 14.2.9.1.9 are satisfied by the CVAP test in Sanmen Unit 1. Specifically, the preoperational vibration test program for the reactor internals of the AP1000 conducted on the first AP1000 is consistent with the guidelines of RG 1.20 for a CVAP.

Based on the evaluations in Sections 3.1.3.1, 3.1.3.2, 3.1.3.3 and 3.1.3.4 of this SE, the staff finds that the changes requested as part of LAR 18-019 with regard to reactor vessel internals vibration testing discussed in 3.1.3 are acceptable.

3.1.4 TECHNICAL EVALUATION OF PROPOSED CHANGES ASSOCIATED WITH CMT HEATED RECIRCULATION TESTING

Section 2.3.3 of Enclosure 1 for LAR 18-019 describes a proposed change to delete COL item 2.D.(2)(a)4 and make corresponding updates to UFSAR subsection 14.2.5 and 14.2.9.1.3, items (k) and (w) to clarify that the CMT recirculation and draindown tests will not be performed for VEGP Units 3 and 4. SNC proposes these changes because the corresponding tests are first three plant only tests, and these tests have been conducted for Sanmen Units 1 and 2, and Haiyang Unit 1.

3.1.4.1 TEST APPLICABILITY TO VEGP UNITS 3 AND 4

Section 2.3.3.2 of Enclosure 1 for LAR-18-019 identifies the critical design and construction attributes for the CMT recirculation and draindown tests and states that standard design and procurement documentation is used for Sanmen Units 1 and 2, Haiyang Unit 1, and VEGP Units 3 and 4. SNC further stated that there are no site-specific design changes for Sanmen Units 1 and 2, or Haiyang Unit 1 that alter the standard design features for any of the components involved in the CMT recirculation and draindown tests. Although the LAR identifies tests that SNC proposes for removal, several ITAAC are retained that verify CMT performance (ITAAC Nos. 2.2.03.08c.i.01, 2.2.03.08c.ii, 2.2.03.08c.iii, 2.2.03.08c.iv.03, 2.2.03.08c.v.01, 2.2.03.08c.vi.01, 2.2.03.08c.xi, and 2.2.03.08c.xii). Section 2.3.3.2 of Enclosure 1 for LAR 18-019 states that Sanmen Units 1 and 2, and Haiyang Unit 1 use the same acceptance criteria for inspections and tests as VEGP Units 3 and 4. Based on the information provided in Section 2.3.3.2 of Enclosure 1 for LAR 18-019, the staff finds SNC has demonstrated the fidelity of the design, construction, and as-built condition of Sanmen Units 1 and 2 and Haiyang Unit 1 to VEGP Units 3 and 4 in relation to CMT performance because (1) all of these units use the same standard design and procurement documentation for the components involved in the tests, (2) ITAAC are retained for VEGP Units 3 and 4 to verify as-built CMT performance, and (3) the same inspections and tests are performed for all of the units and have the same acceptance criteria. Accordingly, the staff finds that the CMT recirculation and draindown tests performed at Sanmen Units 1 and 2, and Haiyang Unit 1 are applicable to VEGP Units 3 and 4.

3.1.4.2 CMT RECIRCULATION TEST

Section 2.3.3.1 of Enclosure 1 to LAR 18-019 discusses the CMT recirculation test. UFSAR subsection 14.2.9.1.3, item (k) states that this test verifies the net mass injection rate from the CMTs following reactor coolant pump coastdown. The staff performed an audit of the analyses and reports supporting LAR 18-019, which includes test predictive analysis, instrument calibration and data analysis, and test evaluation for the CMT recirculation tests for Sanmen Units 1 and 2, and Haiyang Unit 1 (ADAMS Accession No. ML18338A090). During this audit, the staff observed that for the CMT recirculation tests performed at Sanmen Units 1 and 2, and Haiyang Unit 1: (1) pretest analyses were performed by Westinghouse using NOTRUMP, which the staff previously found to be acceptable for modeling CMT behavior for the AP1000 in Section 21.6.2.2 of NUREG-1793 (ADAMS Accession No. ML043450274); (2) CMT flow was measured using redundant and diverse instrumentation; (3) pre-test calibration was performed on the flow measurement instrumentation; (4) test measurement uncertainty was determined in accordance with ASME PTC 19.1-2005, “Test Uncertainty;” and (5) the test evaluation performed by Westinghouse concluded that CMT injection flow rate is consistent with the predictive analysis and that the test demonstrated that the CMT will recirculate and inject its

cold water into the reactor coolant system. In addition, during the audit the staff confirmed that that (1) SNC performed a review of the Westinghouse evaluations for the CMT recirculation tests, and (2) SNC found the Westinghouse evaluations acceptable.

Based on the information provided in Section 2.3.3 of Enclosure 1 to LAR 18-019 and the information documented in the staff's audit report (ADAMS Accession No. ML18338A090) and summarized above, the staff finds that SNC satisfied the first three plant only test requirements for the CMT recirculation test because the staff confirmed (1) these tests have been performed for Sanmen Units 1 and 2, and Haiyang Unit 1, (2) CMT recirculation and draindown tests performed at Sanmen Units 1 and 2, and Haiyang Unit 1 are applicable to VEGP Units 3 and 4 (see Section 3.1.4.1 of this SE), (3) Westinghouse evaluated the tests and concluded that the CMT mass injection rate was verified, and (4) SNC reviewed the Westinghouse evaluations and found them acceptable.

3.1.4.3 CMT DRAINDOWN TEST

Section 2.3.3.2 of Enclosure 1 to LAR 18-019 discusses the CMT draindown test. UFSAR subsection 14.2.9.1.3, item (w) states that this test verifies the proper operation of the CMTs to transition from their recirculation mode to their draindown mode, and also verifies the proper operation of the CMT level instrumentation. The staff notes that a test to verify CMT level instrumentation is retained in ITAAC No. 2.2.03.08c.i.01 as described in UFSAR subsection 14.2.9.1.3, item (v). The staff performed an audit of the analyses and reports supporting LAR 18-019, which includes test predictive analysis, and test evaluation for the CMT draindown tests for Sanmen Units 1 and 2, and Haiyang Unit 1 (ADAMS Accession No. ML18338A090). During this audit, the staff observed that for the CMT draindown tests performed at Sanmen Units 1 and 2, and Haiyang Unit 1: (1) pretest analyses were performed by Westinghouse using NOTRUMP, which the staff previously found to be acceptable for modeling CMT behavior for the AP1000 in Section 21.6.2.2 of NUREG-1793 (ML043450274); and (2) the test evaluation performed by Westinghouse concluded that the CMT instrumentation performed properly and that, consistent with the predictive analysis, the CMT properly transitioned from recirculation to draindown mode of operation as reactor coolant system inventory was lost. In addition, during the audit the staff confirmed that (1) SNC performed a review of the Westinghouse evaluations of the CMT draindown tests, and (2) SNC found the Westinghouse evaluations acceptable.

Based on the information provided in Section 2.3.3 of Enclosure 1 to LAR 18-019 and the information documented in the staff's audit report (ADAMS Accession No. ML18338A090) and summarized above, the staff finds that SNC satisfied the first three plant only test requirements for the CMT draindown test because the staff confirmed (1) these tests have been performed for Sanmen Units 1 and 2, and Haiyang Unit 1, (2) CMT recirculation and draindown tests performed at Sanmen Units 1 and 2, and Haiyang Unit 1 are applicable to VEGP Units 3 and 4 (see Section 3.1.4.1 of this SE), (3) a test to verify CMT level instrumentation is retained in ITAAC No. 2.2.03.08c.i.01, (4) Westinghouse evaluated the tests and concluded that the transition from recirculation to draindown mode was verified, and (5) SNC reviewed the Westinghouse evaluations and found them acceptable.

Based on the evaluations in Sections 3.1.4.1, 3.1.4.2, and 3.1.4.3 of this SE, the staff finds that the proposed change to delete COL item 2.D.(2)(a)4 and make corresponding updates to UFSAR subsection 14.2.5 and 14.2.9.1.3, items (k) and (w) acceptable because SNC demonstrated that the first three plant only test requirements for the CMT recirculation and CMT

draindown tests have been satisfied by the testing performed at Sanmen Units 1 and 2, and Haiyang Unit 1.

3.2 EVALUATION OF EXEMPTION

The regulations in Section III.B of Appendix D to 10 CFR Part 52 require a holder of a COL referencing Appendix D to 10 CFR Part 52 to incorporate by reference and comply with the requirements of Appendix D, including certified information in Tier 1 of the generic AP1000 DCD. Exemptions from Tier 1 information are governed by the change process in Section VIII.A.4 of Appendix D of 10 CFR Part 52. Because SNC has identified changes to plant-specific Tier 1 information, with corresponding changes to the associated COL Appendix C information resulting in the need for a departure, an exemption from the certified design information within plant-specific Tier 1 material is required to implement the LAR.

The Tier 1 information for which a plant-specific departure and exemption was requested relates to revisions to ITAAC on the first plant reactor internal flow induced vibration testing system as described in the licensing basis documents, including COL Condition 2.D(12)(g)9 and plant-specific Tier 1 Sections 2.1.3. The result of this exemption would be that SNC could implement the requested modifications to Tier 1 information, with corresponding changes to COL Appendix C. Pursuant to the provisions of 10 CFR 52.63(b)(1), an exemption from elements of the design as certified in the 10 CFR Part 52, Appendix D, design certification rule is requested for the involved Tier 1 information described and justified in LAR 18-019. This exemption is a permanent exemption limited in scope to the particular Tier 1 information specified.

As stated in Section VIII.A.4 of Appendix D to 10 CFR Part 52, an exemption from Tier 1 information is governed by the requirements of 10 CFR 52.63(b)(1) and 52.98(f). Additionally, Section VIII.A.4 of Appendix D to 10 CFR Part 52 provides that the Commission will deny a request for an exemption from Tier 1 if it finds that the requested change will result in a significant decrease in the level of safety otherwise provided by the design. Pursuant to 10 CFR 52.63(b)(1), the Commission may grant exemptions from one or more elements of the certification information, so long as the criteria given in 10 CFR 52.7, which, in turn, references 10 CFR 50.12, are met and that the special circumstances, which are defined by 10 CFR 50.12(a)(2), outweigh any potential decrease in safety due to reduced standardization.

Pursuant to 10 CFR 52.7, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 52. As 10 CFR 52.7 further states, the Commission's consideration will be governed by 10 CFR 50.12, "Specific exemptions," which states that an exemption may be granted when: (1) the exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security; and (2) special circumstances are present. Specifically, 10 CFR 50.12(a)(2) lists six circumstances for which an exemption may be granted. It is necessary for one of these bases to be present in order for the NRC to consider granting an exemption request. SNC stated that the requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii). That subparagraph defines special circumstances as when "[a]pplication of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule." The staff's analysis of these findings is presented below.

3.2.1 AUTHORIZED BY LAW

The requested exemption would allow SNC to implement the amendment described above. This exemption is a permanent exemption limited in scope to particular Tier 1 information. Subsequent changes to Tier 1, subsection 2.1.3 and Table 2.1.3-2 or any other Tier 1 information would be subject to the exemption process specified in Section VIII.A.4 of Appendix D to 10 CFR Part 52 and the requirements of 10 CFR 52.63(b)(1). As stated above, 10 CFR Part 52, Appendix D, Section VIII.A.4 allows the NRC to grant exemptions from one or more elements of the Tier 1 information. The staff has determined that granting of SNC's proposed exemption will not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations. Therefore, as required by 10 CFR 50.12(a)(1), the exemption is authorized by law.

3.2.2 NO UNDUE RISK TO PUBLIC HEALTH AND SAFETY

As discussed above in the technical evaluation, the proposed changes comply with the NRC's substantive safety regulations. Therefore, there is no undue risk to the public health and safety.

3.2.3 CONSISTENT WITH COMMON DEFENSE AND SECURITY

The proposed exemption would allow a change to the ITAAC on first plant reactor internal flow induced vibration testing system as described in the licensing basis documents, and as presented in plant-specific Tier 1 information, thereby departing from the AP1000 certified (Tier 1) design information. The change does not alter or impede the design, function, or operation of any plant SSCs associated with the facility's physical or cyber security and, therefore, does not affect any plant equipment that is necessary to maintain a safe and secure plant status. In addition, the changes have no impact on plant security or safeguards. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the common defense and security is not impacted by this exemption.

3.2.4 SPECIAL CIRCUMSTANCES

Special circumstances, in accordance with 10 CFR 50.12(a)(2), are present, in part, whenever application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. The underlying purpose of the Tier 1 information is to ensure that a licensee will safely construct and operate the plant based on the certified information found in the AP1000 DCD, which was incorporated by reference into the VEGP Units 3 and 4 licensing basis. The proposed changes described in the above technical evaluation do not impact the ability of any SSCs to perform their functions or negatively impact safety.

Special circumstances are present in the particular circumstances discussed in LAR 18-019 because the application of the specified Tier 1 information is not necessary to achieve the underlying purpose of the rule. The proposed exemption would provide revisions to ITAAC on the first plant reactor internal flow induced vibration testing system. The proposed changes do not affect any function or feature used for the prevention and mitigation of accidents or their safety analyses, and no safety-related SSC or function is involved. This exemption requests revisions to Tier 1, subsection 2.1.3 and Table 2.1.3-2 that continue to demonstrate that the applicable regulatory requirements will be met. Therefore, for the above reasons, the staff finds

that the special circumstances required by 10 CFR 50.12(a)(2)(ii) for the granting of an exemption from the Tier 1 information exist.

3.2.5 SPECIAL CIRCUMSTANCES OUTWEIGH REDUCED STANDARDIZATION

This exemption would allow the implementation of changes to Tier 1 information in the plant-specific DCD and corresponding changes to Appendix C. The justification provided in LAR 18-019, the exemption request, and the associated licensing basis mark-ups demonstrate that there is a limited change from the standard information provided in the generic AP1000 DCD. The design functions of the equipment associated with this request will continue to be maintained because the associated revisions to the Tier 1 ITAAC information support the design function of the reactor internal flow induced vibration system. Consequently, the safety impact that may result from any reduction in standardization is minimized, because the proposed design change does not result in a reduction in the level of safety. Based on the foregoing reasons, as required by 10 CFR Part 52.63(b)(1), the staff finds that the special circumstances outweigh any decrease in safety that may result from the reduction of standardization of the AP1000 design.

3.2.6 NO SIGNIFICANT REDUCTION IN SAFETY

This exemption would allow the implementation of changes discussed above. The exemption request proposes to depart from the certified design by allowing changes discussed above in the technical evaluation. The proposed ITAAC changes will not adversely affect the ability of the reactor internal flow induced vibration system to perform its design functions, and the level of safety provided by the current systems and equipment therein is unchanged. Therefore, based on the foregoing reasons and as required by 10 CFR 52.7, 10 CFR 52.98(f), and 10 CFR Part 52, Appendix D, Section VIII.A.4, the staff finds that granting the exemption would not result in a significant decrease in the level of safety otherwise provided by the design.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations in 10 CFR 50.91(b)(2), on December 10, 2018, the Georgia State official was consulted regarding the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (*Federal Register,* 83 FR 48463, dated September 25, 2018). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

Because the exemption is necessary to allow the changes proposed in the license amendment, and because the exemption does not authorize any activities other than those proposed in the license amendment, the environmental consideration for the exemption is identical to that of the license amendment. Accordingly, the exemption meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the exemption.

6.0 CONCLUSION

The staff has determined that pursuant to Section VIII.A.4 of Appendix D to 10 CFR Part 52, the exemption (1) is authorized by law, (2) presents no undue risk to the public health and safety, (3) is consistent with the common defense and security, (4) is a special circumstance that outweighs the reduction in standardization, and (5) does not significantly reduce the level of safety at SNC's facility. Therefore, the staff grants SNC an exemption from Tier 1 information specified by SNC in the LAR.

The staff has concluded, based on the considerations discussed in Section 3.1 and staff's confirmation that the changes proposed in this LAR do not change an analysis methodology, or assumptions that there is reasonable assurance that: (1) the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. Therefore, the staff finds the changes proposed in this license amendment acceptable.

7.0 REFERENCES

1. Southern Nuclear Operating Company, Vogtle Electric Generating Plant Units 3 and 4, Request for License Amendment and Exemption: Crediting Previously Completed First Plant and First Three Plant Tests (LAR-18-019), August 3, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18215A382).

2. Vogtle Electric Generating Plant Units 3 and 4, Audit Report: Crediting Previously Completed First Plant and First Three Plant Tests (LAR 18-019), December 3, 2018, (ADAMS Accession No. ML18338A090).

3. Nuclear Regulatory Commission Vendor Inspection of Westinghouse Electric Company LLC, Cranberry Township, Report No. 99900404/2018-201, July 6, 2018 (ADAMS Accession No. ML18176A395).

4. Vogtle Electric Generating Plant, Unit 3, Current Facility Combined License NPF-91, Revised September 11, 2018 (ADAMS Accession No. ML14100A106).

5. Vogtle Electric Generating Plant, Unit 4, Current Facility Combined License NPF-92, Revised September 11, 2018 (ADAMS Accession No. ML14100A135).

6. Vogtle Electric Generating Plant, Units 3 and 4 Updated Final Safety Analysis Report, Revision 6 and Tier 1, Revision 5, August 11, 2017 (ADAMS Accession No. ML17172A218).

7. 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” January 1, 2008.

8. NRC RG 1.68, Revision 3, “Initial Test Programs for Water-Cooled Nuclear Power Plants,” June 2013, (ADAMS Accession No. ML13051A027).

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