Southern Nuclear Operating Company

ND-18-0826

Enclosure 2

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Request for License Amendment:

Crediting Previously Completed First Plant and First Three Plant Tests (Non-Proprietary)

(LAR-18-019)

(This Enclosure consists of 38 pages, including this cover page)

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Pursuant to 10 CFR 52.98(c) and in accordance with 10 CFR 50.90, Southern Nuclear Operating Company (SNC, or the "Licensee") hereby requests an amendment to Combined License (COL) Nos. NPF-91 and NPF-92 for Vogtle Electric Generating Plant (VEGP) Units 3 and 4, respectively.

1. SUMMARY DESCRIPTION

The requested amendment involves changes to the design-specific pre-operational tests listed in COL Condition 2.D.(2)(a), and the first plant only tests and first three plant only tests described in UFSAR Sections 3.9.2.4, 14.2.5, 14.2.9.1.3 and 14.2.9.1.9. It also involves changes to COL Appendix C (and plant-specific Tier 1) Inspections, Tests, Analyses and Acceptance Criteria (ITAAC) 2.1.03.07.i for reactor internal flow induced vibration testing. The proposed changes would revise the COLs concerning the performance of first plant testing and first three plant testing during preoperational testing, as described by License Condition 2.D.(2)(a). The proposed changes would revise the COL to delete conditions requiring the following tests:

2.D.(2)(a)1. In-Containment Refueling Water Storage Tank (IRWST) Heatup Test;

2.D.(2)(a)3. Reactor Vessel Internals Vibration Testing;

2.D.(2)(a)4. Core Makeup Tank Heated Recirculation Tests.

The requested amendment proposes changes to the COL Conditions, COL Appendix C (and corresponding plant-specific Tier 1) and Updated Final Safety Analysis Report (UFSAR) in the form of departures from the plant-specific DCD Tier 2* and Tier 2 information (as detailed in Section 2). This enclosure requests approval of the license amendment necessary to implement these changes.

UFSAR Subsection 14.2.5 provides the basis that "[b]ecause of the standardization of the AP1000 design, these special tests (designated as first plant only tests) are not required on follow plants." UFSAR Subsection 14.2.5 also states for subsequent plants "…justification shall be provided that the results of the first plant only tests or first three plant tests are applicable to the subsequent plant." The purpose of the first plant and first three plant tests is to further establish the unique phenomenological performance parameters of the AP1000 design features. These special tests are in addition to preoperational testing that will be completed at every AP1000 unit. The three preoperational first plant and first three plant tests listed above have been completed at the first AP1000 units at Sanmen Unit 1 and Sanmen Units 1 & 2 and Haiyang Unit 1, respectively.

A review of the Quality Assurance Regulations governing Sanmen and Haiyang has been performed to demonstrate that the requirements governing these tests are equivalent to 10 CFR 50 Appendix B. To confirm appropriate adherence to these requirements, applicability of the first plant and first three plant only tests and acceptability of the results, SNC has worked with the Sanmen and Haiyang owners and Westinghouse to review the applicable Administrative Procedures, test procedures, test reports and results. Additionally, SNC had individuals on site at Sanmen Unit 2 observing the performance of the first three plant tests. The reviews of the Quality Assurance Regulations, Administrative Procedures governing testing and the test procedures and reports have been documented by SNC. The observations made while on site at Sanmen Unit 2 have also been documented. These reviews have concluded that the test results from Sanmen Units 1 & 2 and Haiyang Unit 1 are acceptable.

The applicability of the tests to Vogtle Units 3 & 4 was validated by determining the systems, structures and components (SSCs) within the scope of the tests are designed and procured using the same standard AP1000 design requirements across Sanmen Units 1 & 2, Haiyang Unit 1 and Vogtle Units 3 & 4. Reviews of design changes were completed to confirm Sanmen Units 1 & 2 and Haiyang Unit 1 did not have any changes that would take the SSCs involved in the testing outside of the standard plant design such that the test performance would be impacted. Vogtle Units 3 & 4 have ITAAC and pre-operational test requirements associated with the SSCs involved in these tests. Completion of these ITAAC and pre-operational tests will verify that Vogtle Units 3 & 4 are within the standard plant AP1000 design as described in the Vogtle Units 3 & 4 UFSAR. The reviews completed by SNC have concluded that the first plant and first three plant testing and results completed at Sanmen Units 1 & 2 and Haiyang Unit 1 are also applicable to Vogtle Units 3 & 4.

Additionally, the Reactor Vessel Internals Vibrations Testing completed at Sanmen Unit 1 was performed following Regulatory Guide 1.20, Rev. 2. The preliminary and final reports, as specified in Regulatory Guide 1.20, Rev. 2, for Sanmen Unit 1 have been reviewed by SNC and are submitted with this requested amendment with the intent of classifying the Sanmen 1 reactor vessel internals as the valid prototype. The reactor internals design at Sanmen Unit 1 is substantially the same as the Vogtle Units 3 & 4 reactor internals.

2. DETAILED DESCRIPTION and TECHNICAL EVALUATION

As described in the Combined License (COL) Condition 2.D.(2)(a), the licensee shall perform design-specific pre-operational tests including In-Containment Refueling Water Storage Tank (IRWST) Heatup Test, Reactor Vessel Internals Vibration Testing, and Core Makeup Tank (CMT) Heated Recirculation Tests.

The three tests listed above are designated as first plant only tests or first three plant only tests. The first plant only tests and first three plant only tests are described in UFSAR Subsection 14.2.5. The tests are described as “[s]pecial tests to further establish a unique phenomenological performance parameter of the AP1000 design features beyond testing performed for Design Certification of the AP600 and that will not change from plant to plant….” UFSAR Subsection 14.2.5 also provides the basis that “[b]ecause of the standardization of the AP1000 design, these special tests (designated as first plant only tests) are not required on follow plants.”

UFSAR Subsection 14.2.5 also states for subsequent plants “…justification shall be provided that the results of the first plant only tests or first three plant tests are applicable to the subsequent plant.”

There are four AP1000 units being completed ahead of the Vogtle Units 3 & 4 schedule. These are Sanmen Units 1 & 2 and Haiyang Units 1 & 2. Sanmen Unit 1 has already performed the pre-operational first plant only tests described in UFSAR Subsection 14.2.5 and Sanmen Units 1 & 2 and Haiyang Unit 1 have performed the pre-operational first three plant tests described in UFSAR Subsection 14.2.5. The results of these tests have been provided to SNC for review for applicability to Vogtle Units 3 & 4. To determine if the test results are acceptable and applicable to Vogtle Units 3 & 4, various efforts have been completed to evaluate the performance of the tests and the results. The reviews performed focused on key areas including the Quality Assurance (QA) regulations and administrative procedures governing the performance of the testing, evaluation of test procedures and results, and the use of standard AP1000 designed

Systems, Structures and Components (SSCs). SNC determined that the completed tests and test results accomplished their purpose and are applicable to Vogtle Units 3 & 4.

The following sections describe assessments of the QA regulations applicable to the first plant only and first three plant only tests, the Westinghouse oversight of the design and testing, the SNC observation and review of the testing, and the test results and applicability to Vogtle Units 3 & 4.

2.1 Assessment of Quality Assurance Regulation

10 CFR Part 50 Appendix B requirements apply to all activities affecting the safety-related functions of those structures, systems, or components including testing. For the first plant and first three plant tests, the design of the SSCs, the testing methods and acceptance criteria and evaluation of test results were developed by Westinghouse under a 10 CFR Part 50 Appendix B compliant program.

Performance of the first plant and first three plant only tests at Sanmen Units 1 & 2 and Haiyang Unit 1 were performed following the China regulatory quality assurance requirements specified in HAF-003-1991, “Safety Regulations for Quality Assurance of Nuclear Power Plants.” A review comparing the requirements of 10 CFR Part 50 Appendix B and HAF-003-1991 was completed. This review used a matrixed table approach to compare the two regulations. The review compares the introduction and 13 sections of HAF-003-1991 to the introduction and 18 criteria of 10 CFR Part 50 Appendix B. The review concluded that the requirements of HAF-003-1991, as implemented, are comparable to and encompass the requirements of 10 CFR Part 50 Appendix B. For any specific requirements in 10 CFR Part 50 Appendix B that are not directly included in HAF-003-1991, other standards were identified that implement the same requirements of 10 CFR Part 50 Appendix B.

Based on the review of the QA regulations, the first plant and first three plant tests performed at Sanmen Units 1 & 2 and Haiyang Unit 1 were conducted following QA standards that encompass the 10 CFR Part 50 Appendix B requirements applied at Vogtle Units 3 & 4.

2.2 Oversight and Design Control

Westinghouse is the design authority for the scope of systems and components involved in the first plant and first three plant testing. Because Westinghouse is the design authority for Sanmen Units 1 & 2, Haiyang Unit 1 and Vogtle Units 3 & 4 for this scope, they play a key role in maintaining standardization of the design across the units and in oversight of testing.

Westinghouse has an NRC approved 10 CFR Part 50 Appendix B program including governance of design and document control.

10 CFR Part 50 Appendix B, Criterion III states “Design control measures shall be applied to items such as the following: …delineation of acceptance criteria for inspections and tests.” Westinghouse engineering developed and approved the test specifications for the first plant and first three plant tests including the acceptance criteria for the tests. After initial issuance of the test specifications, changes were authorized and approved using the Westinghouse design control process. Westinghouse also worked directly with the Sanmen Units 1 & 2 and Haiyang Unit 1 owners to develop the test procedures for the first plant and first three plant tests. The test procedures and test reports were authored, verified and approved by Westinghouse test engineers and cosigned by the owners’ engineers. The procedures were approved by a Test

Review Board, which included Westinghouse personnel, prior to use. Major changes to the test procedures were also approved by the Test Review Board. Minor changes were reviewed and approved by Westinghouse personnel. Using this process, the acceptance criteria for the first plant and first three plant tests were developed and maintained under the Westinghouse Appendix B process.

10 CFR Part 50, Appendix B, Criterion III states "Design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design and be approved by the organization that performed the original design unless the applicant designates another responsible organization."

The AP1000 plant is a standard design across the Sanmen, Haiyang and Vogtle units for the scope of the first plant and first three plant tests. Westinghouse created, approved and maintained the design documents governing the scope of SSCs in the first plant and first three plant tests. Any design change made to an SSC involved in the first plant or first three plant tests has been reviewed and approved by Westinghouse under the Westinghouse design control process. In the Westinghouse design control process, each of these design changes is designated with an applicability for any plant where the change would be applied. Changes which were only applicable to Sanmen Units 1 & 2 or Haiyang Unit 1 AP1000 units were reviewed for potential impacts to the first plant and first three plant test design parameters. The review was completed after completion of the testing to confirm all design changes were captured. The purpose of the review was to confirm the design of the SSCs involved in the first plant and first three plant tests at Sanmen Units 1 & 2 and Haiyang Unit 1 were not altered to be outside the standard AP1000 design such that test results could be impacted. The review of design changes concluded that there were no site-specific design changes for the Sanmen Units 1 & 2 or Haiyang Unit 1 AP1000 units which would alter any of the critical design attributes for first plant or first three plant only tests. The components involved in the testing were procured using the same design specification requirements. There is no difference between the units for these design requirements. Based on the standard AP1000 design, review of design changes, and procurement to the same quality requirements imposed by the design specification, the Sanmen Units 1 & 2 and Haiyang Unit 1 SSCs for the first plant and first three plant tests are within the standard AP1000 design. The Vogtle 3&4 units follow the same design control process described above to maintain the standard plant design. Additionally, Vogtle 3 & 4 have ITAAC on the SSCs involved in the first plant and first three plant tests. The specific ITAAC relating to each test are described in the subsections below on each specific test. The completion of these ITAAC will confirm the SSCs for the first plant and first three plant tests meet the AP1000 standard design described in the Vogtle 3&4 UFSAR.

HAF-003-1991, criterion XII requires calibration and testing of test equipment, which is comparable to 10 CFR Part 50 Appendix B criterion XII requirements. Calibration requirements for measurement and test equipment (M&TE) are specified in the Sanmen Units 1 & 2 and Haiyang Unit 1 localized procedures. Westinghouse had a team of engineers on site at both Sanmen Units 1 & 2 and Haiyang Unit 1 during performance of the first plant and first three plant tests. The Westinghouse engineers were embedded in the startup organization at the sites and worked alongside the owners as the testing was performed. Prior to the testing, Westinghouse and the owners walked-down instrumentation to confirm proper installation. Calibration records for temporary instrumentation used for the engineering analysis were provided by the owners.

Post-test analysis of the test data was performed by Westinghouse to confirm the test results met the acceptance criteria. Westinghouse compared the test data with the predictive analysis models. All computer codes used for the models used existing computer codes described in the UFSAR. No new computer codes were used for the first plant or first three plant testing. Test reports for each first plant and first three plant tests included Westinghouse engineering and safety analysis reports. The test reports were created and approved under the Westinghouse QA Program.

SNC reviewed the administrative manual procedures, test procedures, test reports and post-test analysis. The reviews were performed by knowledgeable individuals in engineering, testing and operations. NRC Inspection Procedure 70367, “Inspection of Preoperational Test Program,” was used for guidance in creating criteria to review the Sanmen and Haiyang administrative manual procedures. The purpose of the review was to assess the procedural processes and controls for the conduct of testing at Sanmen Units 1 & 2 and Haiyang Unit 1. NRC Inspection Procedure 70702, “Part 52, Inspection of Preoperational Test Performance,” was used for guidance in creating criteria to review the test procedures and reports against. The conclusion of the administrative manual procedures review is that the procedures satisfy the requirements of the NRC Inspection Procedure. For instances where the NRC Inspection Procedure requirement was not explicitly addressed in a single administrative procedure, the requirement was addressed through a combination of procedures. There were no issues identified from the administrative manual procedures reviewed that would challenge any test results. The test procedures were reviewed line-by-line using the established criteria. For any inconsistencies identified in the procedures, the issue was reviewed and dispositioned for impact to the test results. Additionally, these items were reviewed by Westinghouse. The review concluded that none of these inconsistencies would impact the test results. The post-test analysis was reviewed to confirm proper documentation, the analysis methods were appropriate and the test results meet the acceptance criteria in the Vogtle 3&4 UFSAR for each test. All reviews completed by SNC have been documented following SNC procedures.

In addition to reviewing test documentation and results, SNC performed observations of pre-operational testing at Sanmen Unit 2. 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 plant test. The objective of the visit was to observe the following activities for those specific tests:

- performance of pre-test requirements,
- confirmation of M&TE 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.

The observations were documented in a report. The report chronicles the daily observations and access the individuals had throughout their time on site. The observations concluded the first three plant test at Sanmen Unit 2 was conducted in accordance with the test procedures.

SNC concludes that the test results are sufficient to support crediting first plant and first three plant testing completed at Sanmen Units 1 & 2 and Haiyang Unit 1 for Vogtle Units 3 & 4.

2.3 Sanmen and Haiyang Test Results and Applicability to Vogtle Units 3 & 4

2.3.1 IRWST Heatup Test – UFSAR Subsection 14.2.9.1.3, item (h)

The primary function of the passive core cooling system (PXS) is to provide emergency core cooling following postulated design basis events. The passive core cooling system emergency core decay heat removal function is verified by testing of the passive residual heat removal (PRHR) heat exchanger. One of these tests is a first plant only test to observe heatup of the IRWST. The IRWST is a large, stainless-steel lined tank located underneath the operating deck inside the containment. The IRWST is AP1000 Equipment Class C and is designed to meet seismic Category I requirements. The tank is constructed as an integral part of the containment internal structures and is isolated from the steel containment vessel except for the bottom portion of the tank wall, which is separated from the containment vessel by concrete.

The PRHR heat exchanger consists of inlet and outlet channel heads connected by vertical C-shaped tubes. The tubes are supported inside the IRWST. The top of the tubes is several feet below the IRWST water level. The passive residual heat removal heat exchanger is AP1000 Equipment Class A and is designed to meet seismic Category I requirements.

During preoperational testing of the passive core cooling system, a natural circulation test of the PRHR heat exchanger is conducted. In accordance with UFSAR Subsection 14.2.5, for the first plant only, thermocouples are placed in the IRWST to observe the thermal profile developed during the heatup of the IRWST water during PRHR heat exchanger operation. The purpose of this test is to confirm the results of the AP600 Design Certification Program PRHR tests with regards to IRWST mixing, and to quantify the conservatism in the UFSAR Chapter 15 transient analyses.

In addition to this first plant IRWST heatup test, Vogtle Units 3 and 4 will perform a PRHR heat transfer natural circulation test and a PRHR forced flow test. These tests are described in UFSAR Subsection 14.2.9.1.3, items f and g.

During the first plant test, the heatup characteristics of the IRWST water are verified by measuring the vertical water temperature gradient that occurs in the IRWST water at the PRHR heat exchanger tube bundle and at several distances from the tube bundle during the PRHR natural circulation preoperational test and the PRHR forced flow test. The acceptance criterion demonstrates that the average IRWST heatup is consistent with the PRHR heat transfer modeling in the Chapter 15 analysis. These results (in conjunction with Items f and g) are evaluated to demonstrate that the overall PRHR heat transfer performance, i.e., heat removal from the Reactor Coolant System (RCS), is conservative with respect to the analysis documented in Chapter 15.

The acceptance criterion listed in UFSAR Subsection 14.2.9.1.3, Item (h) is not intended to represent a specific quantitative value, rather it is a part of the acceptance of the tests described in UFSAR Subsection 14.2.9.1.3, Items f and g, to demonstrate that the PRHR heat exchanger heat transfer is conservatively predicted in the Chapter 15 analyses. The purpose of the IRWST Heatup Test is to collect data of the IRWST heatup profile to be used in conjunction with the other data obtained during the PRHR heat exchanger tests described in Items f and g to determine the as built PRHR heat exchanger heat transfer performance. As such, no explicit predictive analysis is performed for the IRWST heatup test alone.

The heat transfer rates calculated from the PRHR heat exchanger test data are compared to the Chapter 15 LOFTRAN code. The input parameters in the LOFTRAN code are adjusted to account for differences in the IRWST temperatures, among other parameters, as stated in Items (f) and (g) of UFSAR Subsection 14.2.9.1.3. This adjustment is necessary to normalize the Chapter 15 LOFTRAN code (which is calculated with specified conditions as described in Item (f) and Item (g) with an inlet temperature of 520°F and 250°F respectively, initial IRWST temperature of 80°F, and the design basis number of tubes plugged) with the test results to compare the allowable heat transfer rates predicted versus measured. (The LOFTRAN computer code is described in UFSAR Subsection 15.0.11.2.) The normalized test results are reviewed to confirm that the PRHR heat exchanger performs as predicted (or better than predicted) by the analysis model. Since the analysis model used for the tests is the same analysis model used for the UFSAR Chapter 15 analyses, with differences in input parameters due to the differences in the test conditions from the postulated accident conditions, the test results demonstrate that the UFSAR Chapter 15 LOFTRAN model and methodology are sufficient to conservatively predict PRHR heat exchanger performance documented in the safety analyses. Therefore, the acceptance criteria for the IRWST Heatup Test is considered satisfied if the measured IRWST temperatures demonstrate that the average IRWST heatup during the test supports a normalized PRHR heat exchanger heat transfer rate that is greater than or equal to the LOFTRAN predicted heat transfer rate for the specific conditions described for the acceptance criteria of the PRHR heat exchanger heat transfer tests.

<u>Test Overview</u>

During Hot Functional Testing (HFT), the IRWST thermal profile developed during the heatup of the IRWST water during PRHR heat exchanger operation is measured during two tests; the PRHR heat exchanger natural circulation test and the PRHR heat exchanger forced flow test. The results of the IRWST heatup measurement along with the measurement of the PRHR heat transfer performance of both tests are evaluated together to demonstrate that the overall PRHR heat transfer performance is conservative with respect to the UFSAR Chapter 15 LOFTRAN PRHR heat exchanger performance. A description of the PRHR heat exchanger tests and the relationship to the IRWST heatup test is provided below.

The PRHR heat exchanger forced flow test is conducted with the four reactor coolant pumps (RCPs) in operation and the RCS at a reduced temperature. Flow through the heat exchanger is initiated with the four RCPs running at 50% pump speed. The initial RCS hot leg temperature for the test is ≥ 350˚F. The test continues until the RCS hot leg temperature decreases to ≤ 250˚F. The acceptance criteria for this test is from UFSAR Chapter 3 Table 3.9-17. The calculated heat transfer rate is ≥ 8.46E7 Btu/hr with an inlet temperature of 250°F and an initial IRWST temperature of 80°F and the design basis number of tubes plugged. The allowable calculated heat transfer rate (calculated with the Chapter 15 LOFTRAN model) is adjusted after the test to account for differences in the initial conditions of the actual test and is compared to the test calculated heat transfer rate to validate the acceptance criterion.

The PRHR heat exchanger natural circulation test is conducted by tripping the four RCPs and initiating the PRHR heat exchanger in a natural circulation mode of operation. The elevation difference between the heat exchanger and the rest of the RCS and the density change over the heat exchanger provides the driving head to circulate flow and cool the RCS. The test is initiated with an initial RCS hot leg temperature of ≥ 540°F. The test continues until the RCS hot leg temperature decreases to ≤ 420°F. The acceptance criteria for this test is from UFSAR Subsection 14.2.9.1.3, item (f), the heat transfer rate is ≥ 1.78E+08 Btu/hr based on a 520°F hot leg temperature and ≥ 1.11 E+08 Btu/hr based on 420°F hot leg temperature with an 80°F IRWST temperature and the design number of tubes plugged. The allowable calculated heat transfer rate (calculated with the Chapter 15 LOFTRAN model) is adjusted after the test is run to account for differences in the initial conditions of the actual test and is compared to the test calculated heat transfer rate to verify the acceptance criterion is met.

The primary objective of the test is to determine the IRWST heatup characteristics and demonstrate that the overall PRHR heat transfer performance is conservative with respect to the UFSAR Chapter 15 safety analyses.

The IRWST heatup characteristics are verified by measuring the vertical water temperature gradient occurring in the IRWST water at the PRHR tube bundle and at different distances from the tube bundle during the PRHR heat exchanger performance tests. Resistance Temperature Detectors (RTDs) are mounted on standpipes and placed at several locations inside the IRWST for the IRWST heatup test. RTDs are mounted along each standpipe at the same vertical elevations across all the standpipes, providing a direct comparison of the vertical temperature readings and gradient. The data gathered from the RTDs for both PRHR tests provides the information necessary to characterize the IRWST heatup profile. The RTDs are separate from the permanent plant instrumentation.

The heat transfer performance of the PRHR heat exchanger is calculated using data from instruments along the flow loop from the RCS Loop 1 hot leg, through the PRHR heat exchanger, and into the Loop 1 Steam Generator. RTDs are located on the inlet and outlet lines of the PRHR heat exchanger flowpath to measure the process fluid temperature directly. A local pressure transmitter is located on the inlet line prior to the PRHR heat exchanger. These instruments are permanently installed plant instruments. In addition to the temperature measurement at the outlet of the heat exchanger, there is a flow element (pitot tube) in the PRHR heat exchanger outlet line which is used in conjunction with two temporary differential pressure transmitters to provide the volumetric flowrate in the outlet line.

The results are evaluated using a steady flow energy balance using the data taken from the instrumentation and the PRHR heat exchanger flow rate and heat transfer rate calculated. Engineering evaluations of the PRHR performance account for measurement errors and uncertainties as identified in Regulatory Guide 1.68. ASME PTC 19.1-2005 “Test Uncertainty” is used as code of reference for the test evaluations. A two-sided 95% confidence level is chosen for the calculation results.

Sanmen Unit 1 has successfully performed the first plant IRWST heatup test with satisfactory results. [

] There were no deviations observed for this test. The Westinghouse review and evaluation of the test results to verify acceptability is documented in a test report. SNC reviewed the test report and concurred with the conclusions.

Applicability of Test to Vogtle Units 3 & 4

Vogtle 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 provides the basis that "[b]ecause of the standardization of the AP1000 design, these special tests (designated as first plant only tests) are not required on follow plants." Therefore, verifying standardization of the component design between Sanmen Unit 1 and Vogtle Units 3 & 4 provides the basis that the successful test results from Sanmen Unit 1 are applicable to Vogtle Units 3 & 4.

The critical design and construction attributes for this test and for the overall LOFTRAN PRHR heat transfer model are:

- PRHR Heat Exchanger design including inlet and outlet channel heads;
- PRHR inlet and outlet piping and fittings, and valves design;
- Location of the PRHR heat exchanger relative to the hot and cold legs, steam generator, and IRWST; and
- Free IRWST volume.

For these components, standard design and procurement documentation is used for Sanmen Unit 1 and Vogtle Units 3 & 4. The reactor vessel, steam generators, reactor coolant pumps, PRHR heat exchanger and RCS pressure boundary piping including the PRHR inlet and outlet lines are all manufactured using the same design specification, are procured to the same quality requirements imposed by the design specification and are built to the same standard tolerances. The IRWST is formed by six structural modules (CA01, CA02, CA03, CA55, CA56 and CA57). All of these structural modules are designed using the same standard design drawings. The IRWST has a standardized minimum volume requirement that is the same for all AP1000 Units. The use of standard design documentation confirms that the PXS and RCS system components used for this test are within the standard AP1000 design parameters. Any design changes made to any of these standard components are captured in the Westinghouse design change process. A review has confirmed that there are no site-specific design changes for either Sanmen Unit 1 or Vogtle Units 3 & 4 that alter the standard design features for any of the components involved in this test.

In addition to using standard design and procurement requirements, Vogtle Units 3 & 4 have multiple ITAAC which are applicable to the components involved in this test. Sanmen Unit 1 uses these same acceptance criteria. The following ITAAC will confirm the critical design and construction attributes related to this test for Vogtle Units 3 & 4:

- ITAAC No. 2.2.03.08b.02 (PRHR center line elevation);
- ITAAC No. 2.2.03.08c.vi.03 (PRHR inlet line sloping);
- ITAAC No. 2.2.03.08c.iv.04 (PRHR outlet line to SG elevation); and
- ITAAC No. 2.2.03.08c.iii (IRWST volume).

Vogtle Units 3 & 4 will still perform all other PXS pre-operational tests described in UFSAR Subsection 14.2.9.1.3 which will verify the installed components and associated piping and valves properly perform their design function. These preoperational tests include the PRHR Heat Transfer capability and a natural circulation test.

Based on the use of standard designed components, ITAAC for critical design features and pre-operational tests, the boundary conditions for the IRWST heatup test are the same for Sanmen Unit 1 and Vogtle Units 3 & 4.

Therefore, the successful completion of the first plant IRWST heatup test at Sanmen Unit 1 is applicable to Vogtle Units 3 & 4 and the first plant test is not required to be performed at Vogtle Units 3 & 4.

<u>Change Description</u>

As stated above, the first plant IRWST heatup test was successfully completed at the first AP1000, Sanmen Unit 1, and the results are applicable to Vogtle Units 3 & 4. Therefore, the IRWST heatup test is proposed to be deleted from the UFSAR Subsection 14.2.5, first plant tests, and UFSAR Subsection 14.2.9.1.3, PXS pre-operational tests. Specifically, the proposed changes are:

- COL Condition 2.D.(2)(a)1 requires the licensee to perform an IRWST heatup test as described in UFSAR 14.2.9.1.3, item (h). This COL condition is proposed to be deleted based on the successful completion of the test at the first AP1000.
- UFSAR Subsection 14.2.5 describes the first plant only tests, including IRWST heatup.

A statement is proposed to be added that the test will not be run at Vogtle Units 3 & 4 based on the successful completion of the test at the first AP1000.

- UFSAR Subsection 14.2.9.1.3, item (h), describes the general test methods and acceptance criteria for the IRWST heatup test. The IRWST heatup test is proposed to be deleted from this section because the test will not be run at Vogtle Units 3 & 4 based on the successful completion of the test at the first AP1000.

2.3.2 Reactor Vessel Internals Vibration Testing – UFSAR Subsection 14.2.9.1.9

The Comprehensive Vibration Assessment Program (CVAP) confirms the long-term, steady-state vibration response of the reactor internals for operating steady-state and transient conditions. The three aspects of this evaluation are the following: a prediction of the vibrations of the reactor internals (based in part on previous scale model tests), a preoperational vibration test program of the internals of the first plant, and a correlation of the analysis and test results. AP1000 vibration assessment program requirements are currently described in topical reports WCAP-17983 and WCAP-17984. These reports are incorporated by reference documents in the UFSAR. These program requirements are applicable to the testing completed at Sanmen Unit 1 and are also applicable to Vogtle Units 3 & 4.

UFSAR Subsections 14.2.5 and 14.2.9.1.9 describes the reactor internals instrumented vibration testing as a first plant only test. UFSAR Subsection 14.2.5 states “the preoperational vibration test program for the reactor internals of the AP1000 conducted on the first AP1000 is consistent with the guidelines of Regulatory Guide 1.20 for a comprehensive vibration assessment program. This program is discussed in UFSAR Subsection 3.9.2.” UFSAR Subsection 3.9.2.4 describes the first AP1000 reactor internals as a prototype. The UFSAR states “with respect to the reactor internals preoperational test program, the first AP1000 plant reactor vessel internals are classified as a prototype as defined in Regulatory Guide 1.20.” Sanmen Unit 1, as the first AP1000 reactor internals, is proposed to be classified as a valid prototype in accordance with Regulatory Guide (RG) 1.20. Conformance to RG 1.20, Revision 2, is described in UFSAR Appendix 1A. In accordance with RG 1.20, to classify Sanmen as the valid prototype, the detailed results of the program should be included in an application related to a non-prototype and should address all the provisions within the Regulatory Guide. Vogtle Units 3 & 4, as a subsequent AP1000 plant, is proposed to be classified as the non-prototype, category 1 as defined in RG 1.20.

Test Overview

As described in UFSAR Subsection 14.2.9.1.9, during hot functional testing (HFT) of the AP1000 plant, the reactor vessel internals (RVI) are monitored as a part of the CVAP. The program demonstrates that the RVI are adequately designed to withstand flow-induced vibration (FIV) forces during normal and anticipated transient plant operating conditions for the design life of the plant. The program is performed in accordance with Regulatory Guide 1.20, Revision 2, “Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing”.

In accordance with RG 1.20, Revision 2, the first constructed AP1000 plant RVI assembly at Sanmen Unit 1 is classified as a Prototype. The CVAP for a Prototype RVI configuration includes the following elements:

- **Vibration Analysis Program**

  The analysis program consists of a vibration analysis for steady-state and anticipated transient conditions corresponding to preoperational and initial startup test and normal operating conditions. This includes creating structural and hydraulic models, determining natural frequencies and associated mode shapes, and estimating random and deterministic forcing functions. The analysis program also calculates expected and acceptable responses for selected vibration measurement program sensor locations and develops acceptance criteria for the vibration measurement program, including permissible deviations.

- **Vibration Measurement Program**

  The in-plant vibration measurement program verifies the structural integrity of the RVI for FIV, determines the margin of safety associated with steady-state and anticipated transient conditions for normal operation, and verifies the results of the analysis program. Margin of safety in a particular component is established by comparing the limiting measured response in the component to the maximum allowable response at the measurement location. Appropriate transducers are placed throughout the RVI to monitor significant lateral, vertical, and torsional motions of major RVI components in the significant modes of vibration, and their hydraulic responses. The transducer data is recorded for the steady-state and bounding anticipated transient modes of operation (flow transients), including expected reactor coolant pump (RCP) speeds and combinations permissible during the hot functional test (HFT). The selected test duration with plant operation at normal operating modes confirms that each critical component experiences at least 10^6 cycles of vibration (computed at the lowest frequency for which the component has a significant structural response) prior to the final inspection.

- **Inspection Program**

  The inspection program consists of pre-HFT and post-HFT inspections of the RVI. The inspection program includes a tabulation of the RVI components and local inspected areas, and a description of the inspection procedure including inspection method, documentation, access provisions, and any specialized equipment used during inspection.

- **Documentation of Results**

  A review and correlation of the results of the analysis, vibration measurement, and inspection program are conducted following completion of the inspection program to determine if the acceptance criteria are satisfied. Evaluation of the results and a description of any modifications or actions necessary to demonstrate the structural adequacy of the RVI are documented in preliminary and final reports.

The analysis, measurement, and inspection programs for the AP1000 plant CVAP are provided in two WCAPs that satisfy the requirements of Regulatory Guide 1.20, Revision 2, Sections C.2.1, C.2.2, and C.2.3. WCAP-17984, "Comprehensive Vibration Assessment Program (CVAP) Vibration Analysis Program for the AP1000 plant" contains a description of the AP1000 plant RVI, vibration analysis methodology, response predictions for the RVI components, and acceptance criteria for applicable sensor locations. WCAP-17983, "Comprehensive Vibration

Assessment Program (CVAP) Measurement and Inspection Programs for the AP1000 Plant" contains the description of measurement and inspection programs for the AP1000 plant CVAP. It is important to note that WCAP-17983 provides additional sensor locations that were not included in DCD, Rev. 19 (via the Incorporated by Reference document WCAP-15949). The additional sensor locations provide redundancy that was not included at the time of the AP1000 design certification.

For the CVAP performed at Sanmen Unit 1, Westinghouse wrote both a Preliminary Report (APP-GW-GLR-179 (Proprietary) and APP-GW-GLR-180 (Non-Proprietary)) and a Final Report (APP-GW-GLR-181 (Proprietary) and APP-GW-GLR-182 (Non-Proprietary)). The Preliminary Report contains the evaluation of the Sanmen Unit 1 CVAP Vibration Measurement and Inspection Program results with respect to the test acceptance criteria. [

] The Preliminary Report satisfies the requirements in Section C.2.4.1 of Regulatory Guide, 1.20 Revision 2.

The Final Report provides a comparison of analytical predictions, test measurements, and inspections. Descriptions of any significant deviations, comparisons between measured and analytical responses, determination of high-cycle fatigue margins for component responses, and evaluation of unanticipated observations are included. The Final Report concluded that there were no modifications or actions necessary to demonstrate the structural adequacy of the RVI. The Final Report satisfies the requirements in Section C.2.4.2 of Regulatory Guide 1.20, Revision 2.

SNC reviewed the test reports and concurred with the conclusions.

<u>Framework for Implementing Regulatory Guide 1.20</u>

Guidance for the CVAP for prototype reactor internals is outlined in Section C.2 of Regulatory Guide 1.20, Revision 2. The framework established for the AP1000 CVAP to implement these regulatory guidelines is presented in Table 1. The regulatory guidelines are presented along with the corresponding elements of the AP1000 CVAP.

Table 1. Framework for AP1000 RVI CVAP Implementation of U.S. Regulatory Guide 1.20

U.S. Regulatory Guide 1.20, Part C.2		**CVAP Report(s)**	
Section(s)	**Guidelines**	**Section(s)**	**Program Elements**
CVAP Vibration Analysis Program (WCAP-17984)			
2.1	Vibration Analysis Program	5	Description of vibration analysis program
		6, 7	Justification of the CVAP configuration and acceptance criteria
2.1.1	The theoretical structural and hydraulic models and analytical formulations or scaling laws and scale models used in the analysis.	5.2, 5.3	Structural models
		5.4	Hydraulic models
		7.1.2	Scaling relationships

Table 1. Framework for AP1000 RVI CVAP Implementation of U.S. Regulatory Guide 1.20			
U.S. Regulatory Guide 1.20, Part C.2		CVAP Report(s)	
Section(s)	**Guidelines**	**Section(s)**	**Program Elements**
2.1.2	The structural and hydraulic system natural frequencies and associated mode shapes which may be excited during steady state and anticipated transient operation.	5.3	Structural modes and frequencies
2.1.3	The estimated random and deterministic forcing functions, including any very-low-frequency components, for steady state and anticipated transient operation.	5.4	Forcing function development
2.1.4	The calculated structural and hydraulic responses for steady state and anticipated transient operation.	5.5	Predictions are provided of RVI component structural responses and subsequent limiting locations relative to normal operating and test related plant operating conditions.
2.1.5	A comparison of the calculated structural and hydraulic responses for preoperational and initial startup testing with those for normal operation.	6	General analysis methodologies are described, including the approach for extrapolating preoperational test results to normal operating conditions.
2.1.6	The anticipated structural or hydraulic vibratory response (defined in terms of frequency, amplitude, and modal contributions) that is appropriate to each sensor location for steady-state and anticipated transient pre-operational and startup conditions.	5.4	Component structural evaluations provide predictions of anticipated structural responses at CVAP sensor locations during CVAP testing.
2.1.7	The test acceptance criteria with permissible deviations and the basis for the criteria.	7, 8	Acceptance criteria include consideration of predictive analysis and measurement uncertainties.
CVAP Measurement Program (WCAP-17983)			
2.2	Vibration Measurement Program	4	Description of the Measurement Program
2.2.1	Description of data acquisition and reduction system.	4.2, 4.3	Transducer types, specifications, frequency/amplitude ranges
		4.1, 4.2	Transducer locations, descriptions
		4.3	Precautions during design, installation
		4.3	Transducer redundancy
		4.3	Transducer testing

Table 1. Framework for AP1000 RVI CVAP Implementation of U.S. Regulatory Guide 1.20			
U.S. Regulatory Guide 1.20, Part C.2		**CVAP Report(s)**	
Section(s)	**Guidelines**	**Section(s)**	**Program Elements**
		4.4	Data acquisition system and analysis, including frequency and modal content, precautions during data collection, signal conditioning, real-time frequency and time-domain analysis.
		4.4	Discussion of data analysis
2.2.2	Test operating conditions	5.1, 5.2, 5.3	CVAP steady-state and transient test conditions, test data collection points, required duration of testing, description and justification of test vs. normal operating conditions and configurations, disposition of fuel assemblies.
		3.1, 3.2	Design configuration (normal operating), test configuration
2.3	Inspection Program	6	Description of the Inspection Program
2.3.1	Tabulation of RVI components and areas to be inspected.	6.1	Detailed tabulation of inspection locations, type of inspections performed, and inspection methods.
2.3.2	A tabulation of specific inspection areas that can be used to verify segments of the vibration analysis and measurement program.	6.2	Areas to be inspected, basis for inspections.
2.3.3	A description of the inspection procedure	6.1, 6.3	Detailed inspection methods noted for each of the tabulated inspection locations.
CVAP Preliminary Report (APP-GW-GLR-179 (Proprietary) and APP-GW-GLR-180 (Non-Proprietary))			
2.4, 2.4.1	The preliminary report should summarize an evaluation of the raw and, as necessary limited processed data and the results of the inspection program with respect to the test acceptance criteria. Anomalous data that could bear on the structural integrity of the reactor internals should be identified, as should the method to be used for evaluating such data.	3.2.1	[ ]
		3.5.5	[ ]
		3.6	Component measured responses are compared to the acceptance criteria
		4.2	Pre-HFT and post-HFT inspection results
		2, 5	Conclusions demonstrating acceptable measurement and inspection results.
CVAP Final Report (APP-GW-GLR-181 (Proprietary) and APP-GW-GLR-182 (Non-Proprietary)**)**			
2.4, 2.4.2	The final report should include:		
2.4.2.a	A description of any deviations from the specified measurement and	4.2.1	[ ]
		4.5.5	[ ]

Table 1. Framework for AP1000 RVI CVAP Implementation of U.S. Regulatory Guide 1.20			
U.S. Regulatory Guide 1.20, Part C.2		**CVAP Report(s)**	
Section(s)	**Guidelines**	**Section(s)**	**Program Elements**
	inspection programs, including instrumentation reading and inspection anomalies, instrumentation malfunctions, and deviations from the specified operating conditions.	4.7.2.3	[ ]
2.4.2.b	A comparison between the measured and analytically determined modes of structural and hydraulic response (including those parameters from which the input forcing function is determined) for the purpose of establishing the validity of the analytical technique.	4.6	Comparison of predicted vs. measured responses, predicted vs. measured damping.
		4.7	Discussion of forcing functions
2.4.2.c	A determination of the margin of safety associated with normal steady-state and anticipated transient operation.	4.6	Detailed evaluation of measured margins
		2, 6	Overall conclusions demonstrating successful analysis program and acceptable measurement and inspection results.
2.4.2.d	An evaluation of measurements that exceeded acceptable limits not specified as test acceptance criteria or of observations that were unanticipated and the disposition of such deviations.	N/A	No acceptance limits were exceeded; no observations represented deviations in the measured responses.
2.4.3	If necessary, include an evaluation and description of the modifications or actions planned in order to justify the structural adequacy of the reactor internals.	N/A	No modifications or actions to the RVI were necessary to demonstrate the structural adequacy of the RVI.
2.4.4	The collection, storage and maintenance of all records relevant to the analysis, measurement and inspection phases of the CVAP should be consistent with U.S. Regulatory Guide 1.88.	CVAP Functional Specification	Records related to the CVAP are stored in accordance with the CVAP Functional Specification which comply with NQA-1 1994. Note that Regulatory Guide 1.88 is withdrawn and not applicable to Vogtle Units 3 & 4.
2.5	This section covers schedule during the construction permit review which is not applicable to Vogtle Units 3 & 4 at this point.		

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Applicability to Vogtle Units 3 & 4

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 provides the basis that "[b]ecause of the standardization of the AP1000 design, these special tests (designated as first plant only tests) are not required on follow 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 "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." Based on these UFSAR and RG 1.20 requirements, verifying standardization of the reactor internal configuration between Sanmen Unit 1 and Vogtle Units 3 & 4 provides the basis that the Vogtle Units 3 & 4 reactor internals are substantially the same as the valid prototype, Sanmen Unit 1.

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 Boiler and Pressure Vessel 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.

As with Sanmen Unit 1, the Vogtle Units 3 & 4 RVI are designed and qualified in accordance with the generic RVI design specification and design report, with as-built conditions to be reconciled in a plant-specific design report. The as-built configuration at Vogtle Units 3 & 4 is to be evaluated and any necessary ASME Code reconciliations will be performed. It is expected that the as-built configuration will be substantially similar to the generic design; therefore, the RVI would be appropriately represented by the Sanmen Unit 1 RVI.

For Sanmen Unit 1, the electrical grid operates at 50 Hz, while the reactor coolant pumps (RCP) are driven by variable frequency drives (VFD) operating at 60 Hz. For Vogtle Units 3 & 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.

In addition to using standard design and procurement requirements, Vogtle Units 3 & 4 has multiple ITAAC which are applicable to the components involved in CVAP. Sanmen 1 uses these same acceptance criteria. The following ITAAC will confirm the critical design and construction attributes related to CVAP:

- ITAAC No. 2.1.02.01 (Inspection of As-built System – RCS functional arrangement);
- ITAAC No. 2.1.03.01 (Inspection of As-built System – RXS functional arrangement);
- ITAAC No. 2.1.03.02a (Inspection of As-built System – Fuel assembly positions and CRDMs);
- ITAAC No. 2.1.03.02b (Inspection of As-built System – Control assemblies and drive rods); and
- ITAAC No. 2.1.03.02c (Inspection of As-built System – Reactor Vessel Arrangement).

Based on the standard AP1000 reactor internals design, use of standard design and procurement requirements and ITAAC confirming the as-built reactor internals design, the Vogtle Units 3 & 4 reactor internals are substantially the same as Sanmen Unit 1 reactor internals, the specified valid prototype. Therefore, Vogtle Units 3 & 4 satisfies the RG 1.20 guidelines to be classified as non-prototype, category I and is proposed to be classified as such. As a proposed non-prototype, category I reactor internal design, Vogtle Units 3 & 4 would not perform an instrumented CVAP as specified in RG 1.20 and described in UFSAR Subsections 3.9.2.4, 14.2.5 and 14.2.9.1.9, and ITAAC 2.1.03.07.i. Vogtle Units 3 & 4 would perform a non-instrumented CVAP which is consistent with RG 1.20 for non-prototype, category I. This is specified in ITAAC 2.1.03.07.ii.

In accordance with RG 1.20, the results of the prototype reactor internals CVAP should be submitted to the Commission in the form of preliminary and final reports. These reports are also required to satisfy the acceptance criteria of ITAAC 2.1.03.07.i. These reports (SM1-CVAP-T2R-200 and SM1-CVAP-T2R-300) are being submitted as part of this amendment request. Therefore, the intent of ITAAC 2.1.03.07.i is satisfied by submittal of the preliminary and final reports and is no longer necessary for Vogtle Units 3 & 4 and is proposed to be deleted.

<u>Change Description</u>

As stated above, the first plant vibration test program for the reactor internals was successfully completed at the first AP1000, Sanmen Unit 1. Sanmen Unit 1 is proposed to be classified as the prototype reactor internals as defined in RG 1.20. Vogtle Units 3 & 4 are proposed to be classified as non-prototype, category I reactor internals as defined in RG 1.20. Specifically, the proposed changes are:

- COL Condition 2.D.(2)(a)3 requires the licensee 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 the first AP1000.

- COL Appendix C (and plant-specific Tier 1) Table 2.1.3-2, ITAAC 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 Vogtle Units 3 & 4 will not perform an instrumented vibration test. The report for the first AP1000 unit referenced in the acceptance criteria of the ITAAC is being submitted as part of this amendment request.

- 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 the prototype reactor internals and Vogtle Units 3 & 4 as non-prototype, 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 Vogtle Units 3 & 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. A statement is proposed to be added that the instrumented vibration test will not be run at Vogtle Units 3 & 4 based on the successful completion of the test at the first AP1000. Vogtle Units 3 & 4 will 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 reactor vessel internals vibration testing. The test requirements for an instrumented CVAP are proposed to be deleted from this subsection because Vogtle Units 3 & 4 will not perform an instrumented CVAP. Vogtle Units 3 & 4 will perform a non-instrument CVAP which is consistent with the guidance of RG 1.20 for non-prototype, category I reactor internals.

2.3.3. Core Makeup Tank Heated Recirculation Tests – UFSAR Subsection 14.2.9.1.3, items (k) and (w)

The PXS has two core makeup tanks (CMTs). The CMTs are vertical, cylindrical tanks with hemispherical upper and lower heads. They are made of carbon steel, clad on the internal surfaces with stainless steel. The core makeup tanks are AP1000 Equipment Class A and are designed to meet seismic Category I requirements. They are located inside containment on the 107-foot floor elevation. The core makeup tanks are located above the direct vessel injection line connections to the reactor vessel, which are located at an elevation near the bottom of the hot leg. The core makeup tanks provide injection for an extended time after core makeup tank actuation when in recirculation mode.

As described in UFSAR Subsection 14.2.5, during preoperational testing of the PXS, a test is performed for each plant to verify the CMT inlet piping resistances. In addition, cold draining tests of the CMTs are conducted that verify the discharge piping resistance and proper drain rate of the CMTs for each plant. For the first three plants, two additional CMT tests are conducted during hot functional testing of the RCS. These tests are a natural circulation heatup of the CMTs followed by a test to verify the ability of the CMTs to transition from a recirculation mode to a draindown mode while at elevated temperature and pressure.

Operation of the CMTs in their natural circulation mode is conducted on the first three plants only for the following reasons:

- Natural circulation of the CMTs will not vary from plant to plant, provided that the other verifications discussed above are performed as specified;
- Natural circulation testing of the CMTs was extensively tested as part of the Design Certification Tests; and
- Performance of this test results in a significant thermal transient on Class 1 components including the CMTs and the direct vessel injection nozzles. The current component and system design considers 5 transient cycles. Elimination of this test will result in additional conservatism in the fatigue analysis.

2.3.3.1 Core Makeup Tank Recirculation Test

UFSAR Subsection 14.2.9.1.3 describes the general test method and acceptance criteria for the CMT recirculation test. Proper operation of the core makeup tanks to perform their reactor water makeup and boration function is verified by initiating recirculation flow through the tanks during hot functional testing with the reactor coolant system at ≥ 530°F. This testing is initiated by simulating a safety signal which opens the tank discharge isolation valves and stops reactor coolant pumps after the appropriate time delay. The proper tank recirculation flow after the pumps have coasted down is verified. Based on the cold leg temperature, CMT discharge temperature, and temporary CMT flow instrumentation, the net mass injection rate into the reactor is verified.

<u>Predictive Analysis</u>

The CMT recirculation test was simulated using the NOTRUMP computer code used for small break loss-of-coolant accident (SBLOCA) analyses and described in UFSAR Subsection 15.6.5.4B.2.1.

The UFSAR Chapter 15 SBLOCA NOTRUMP model was used as the starting model for the CMT recirculation predictive analysis. The model was updated to reflect the initial RCS conditions, including a reduced RCS pressure and nominal operating temperature. Additionally, the NOTRUMP code does not contain a thermal stratification model, which along with coarse CMT noding, will lead to differences between the CMT fluid temperatures predicted by NOTRUMP and the test. This was observed and discussed within the NOTRUMP final validation report for the AP600. Due to the importance of the movement of the hot layer within the CMT for this test, increased CMT nodalization was implemented to help account for the lack of a CMT thermal stratification model in NOTRUMP and improve the prediction of the CMT temperature and flow rate.

The CMT recirculation test was initialized by manual CMT actuation which causes the CMT outlet valves to open and the RCPs to trip. To provide a range of expected flow rates, the predictive analysis modeled the minimum and maximum CMT line resistances.

The acceptance criteria are listed in UFSAR Subsection 14.2.9.1.3, Item (k). A number of the specific test acceptance criteria are demonstrated by proper operation of the equipment and are not dependent on the predictive analysis.

The test acceptance criteria to verify proper tank recirculation flow and the net mass injection rate into the reactor are based on the predictive analysis performed with the NOTRUMP code. The predictive analysis varied the CMT line resistance (maximum and minimum line resistances were modeled) to provide the expected range of flow from the CMT outlet. The results of the predictive analysis are compared directly to the CMT outlet flow rates from the tests to demonstrate that the appropriate flow rate is obtained. The CMT temperatures in the NOTRUMP predictive analysis will diverge from the test results as the hot layer descends through the tank due to the perfect mixing that occurs in each control volume. Considering this expected behavior, the time to recirculate approximately half of the CMT tank was calculated using the NOTRUMP results and provides a direct comparison to when the hot layer will reach the thermocouple at the midpoint elevation of CMT in the test. This validates that the appropriate mass is injected into the reactor.

Test Overview

During HFT, a CMT recirculation test is performed to verify that both CMTs provide sufficient reactor water makeup and boration. The initial conditions of the test include a uniform CMT temperature full of cold water and the RCS ≥ 530°F. This test is initiated by opening the CMT discharge isolation valves of both CMTs simultaneously which trips the RCPs after a certain delay. The CMT recirculation flow after the pumps have coasted down is measured with temporary instrumentation and is compared to the acceptance criteria established with the USFAR Chapter 15 LOCA NOTRUMP computer code. (The NOTRUMP computer code is described in UFSAR Subsection 15.6.5.4B.2.1.) In addition to verifying the CMT recirculation flow rates, this test confirms that the CMT recirculation behavior consistent with the AP600 separate effect and integral effect tests.

The CMTs are located at an elevation above the core and are filled with borated water and provide the RCS makeup and boration for the LOCA and non-LOCA events. Each CMT is made of carbon steel, clad on the inside surfaces with stainless steel and have a volume of greater than 2487 ft^3 (COL Appendix C ITAAC 2.2.03.08c.vi.01) with an inlet line that connects one of the cold legs to the top of the CMT and an outlet line that connects the bottom of the CMT to the direct vessel injection (DVI) line. The DVI line is connected to the reactor vessel downcomer. The CMT inlet valve is normally open, therefore the CMT is at primary system pressure. The

CMT outlet valves are normally closed, preventing natural circulation during normal operation. When one of the outlet valves is open, a natural circulation path is established. Cold borated water flows into the reactor vessel and hot primary fluid is drawn upward into the top of the CMT through the inlet line.

The CMTs can operate in two different modes, depending on the RCS conditions. If the cold legs are filled with water, CMTs operate in a water recirculation mode with the driving force based on gravity and on the density difference between the hot reactor coolant in the CMT balance line and the colder water in the CMT. If the cold legs become voided, as they do during LOCAs, the CMTs will operate in a steam displacement injection or steam drain-down mode. In this mode, the driving force is based on gravity and the density difference between steam from the cold legs and water in the CMTs.

The primary objective of the CMT recirculation test is to measure the CMT recirculation flow rates and the CMT temperature distribution in a natural circulation mode.

CMT temperature distribution is measured with permanently installed nonsafety related RTDs located inside the CMTs. These RTDs are used to confirm that the CMT temperature is maintained within the bounds of the plant Technical Specifications. Temperature indications of these RTDs are also used to monitor CMT temperature during PXS operation. Each channel provides a signal to the PLS, which provides temperature indication and alarms in the main control room.

Temporary test instrumentation is also installed on the CMT discharge line to measure flow and calibrated under cold conditions prior to the recirculation test. Differential pressure transmitters and ultrasonic flowmeters (UFMs) are installed to provide diverse flow measurements which allow for the full range of flows from the CMT to be measured with sufficient accuracy. UFMs provide direct flow measurement while the differential pressure transmitters provide the differential pressure of the CMT injection line. Differential pressure is then converted to flow.

Engineering evaluations of the CMT flow rates account for measurement errors and uncertainties as required by the Regulatory Guide 1.68. ASME PTC 19.1-2005, “Test Uncertainty,” is used as code of reference for the test evaluations. A two-sided 95% confidence level is chosen for the calculation results.

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The reproducibility of the results between the units demonstrates that the CMT performance during recirculation does not vary significantly from plant to plant. This reproducibility of plant performance is the result of the AP1000 standardization concept. No deviations were observed during these tests. The Westinghouse review and evaluation of the test results to verify acceptability is documented in a test report. Haiyang Unit 1 successfully performed the CMT recirculation test with similar results to Sanmen Units 1 & 2. SNC reviewed the test reports for Sanmen Units 1 & 2 and Haiyang Unit 1 and concurred with the conclusions.

2.3.3.2 Core Makeup Tank Draindown Test

UFSAR Subsection 14.2.9.1.3 describes the general test method and acceptance criteria for the CMT draindown test. In conjunction with the verification of the core makeup tanks to perform their reactor water makeup function and boration function described above, the proper operation of the core makeup tanks to transition from their recirculation mode of operation to their draindown mode of operation after heatup is verified. This testing also verifies the proper operation of the core makeup tank level instrumentation to operate during draining of the heated tank fluid. The IRWST initial level is reduced to at least 3 feet below the spillway level as a prerequisite condition for this testing in order to provide sufficient ullage to accept the mass discharged from the reactor coolant system via the automatic depressurization stage 1.

The recirculation operation in the CMT, described above, is continued until the core makeup tank fluid has been heated to ≥ 350°F. The core makeup tank isolation valves are then closed, the reactor coolant pumps are started, and the reactor coolant system is reheated up to hot functional testing conditions. This testing is initiated by shutting off the reactor coolant pumps, opening the core makeup tank isolation valves, and by opening one of the automatic depressurization stage 1 flow paths to the IRWST. This initiates a large loss of mass from the reactor coolant system, depressurization of the reactor coolant system to the bulk fluid saturation pressure, and additional recirculation through the core makeup tank. Core makeup tank draindown initiates in response to the continued depressurization and mass loss from the reactor coolant system. The automatic depressurization stage 1 flow path is closed after the core makeup tank level has decreased below the level at which stage 4 actuation occurs.

Predictive Analysis

The CMT draindown test was simulated using the NOTRUMP computer code used for small break loss-of-coolant accident (SBLOCA) analyses and described in UFSAR Subsection 15.6.5.4B.2.1.

The UFSAR Chapter 15 SBLOCA NOTRUMP model was used as the starting model for the CMT draindown predictive analysis. The model updates performed for the CMT recirculation predictive analysis were also used for the CMT draindown predictive analysis in addition to updates modeling a single ADS stage 1 flow path in operation during the test.

The CMT draindown simulation was initialized with CMT temperatures from the CMT recirculation predictive analysis representing the expected conditions at the end of the CMT recirculation test. The RCS initial conditions were nominal operating pressure and temperature consistent with the expected conditions during the test following the heat-up after the CMT recirculation test.

The CMT draindown simulations were initiated by manually actuating the CMTs and the RCPs trip as a result. After CMT actuation, a delay period was modeled to simulate a time for the operators to confirm that the CMT discharge valves have been opened before opening ADS Stage 1 valves. After that delay, a single ADS Stage 1 flow path was opened causing the RCS to depressurize and RCS inventory to be discharged to the IRWST. The hot CMTs initially operated in recirculation mode until sufficient inventory had been lost to allow the CMTs to transition to a draindown mode of operation. The predictive analysis was run until the CMTs drained through the level at which ADS Stage 4 actuation occurs. The predictive analysis confirmed that the test design was expected to allow for the transition from CMT recirculation to draindown and sufficient RCS inventory discharge to allow the CMTs to drain through the necessary setpoints.

Consistent with the CMT recirculation test, two cases were modeled for the CMT draindown test, one with minimum and one with maximum CMT piping resistances, to provide a potential range of behavior.

Additional NOTRUMP simulations were performed following the tests at Sanmen Unit 1. After the test at Sanmen Unit 1, it was observed that the NOTRUMP analysis over predicted the discharge through the single ADS Stage 1 flow path. To better match the RCS conditions during the test, and the resulting CMT recirculation and draindown behavior, the ADS Stage 1 model was adjusted in an updated simulation. The updated simulation showed agreement of CMT recirculation and draindown behavior between the prediction and the test for Sanmen Unit 1 and Sanmen Unit 2.

The test acceptance criteria are demonstrated by proper operation of the equipment and are not dependent on the predictive analysis. There is no specific acceptance criteria obtained from the predictive analysis. However, the predictive analysis provides the expected RCS and CMT behavior that can be compared to the actual test behavior.

Test Overview

During HFT, a test is performed to verify that the CMTs properly transition from recirculation mode to draindown mode. This test is performed by opening automatic depressurization system (ADS) Stage 1 and the CMT discharge valves and continues until the CMT level is below the level at which ADS Stage 4 occurs. This test is conducted with an initial Reactor Coolant System (RCS) hot leg temperature of ≥ 550˚F and the temperature at the midpoint elevation of the CMT ≥ 350˚F. The test is initiated by stopping the RCPs and manually actuating the CMTs (i.e., opening the CMT discharge isolation valve and opening ADS Stage 1). In addition to verifying proper transition between recirculation and draindown mode, the test monitors the dynamic effects of the steam injection and mixing with CMT liquid on the CMT level instrumentation. This test provides confirmation that the CMTs properly transition to draindown following recirculation and proper operation of the CMT upper narrow and lower narrow range instrumentation used for the actuation of the ADS Stages 1, 2, 3, and 4 during LOCA events. The test also verifies that the CMT level instrumentation is operating properly by comparing CMT wide range level instrumentation to CMT upper narrow and lower narrow range level instrumentation.

The primary objective of the CMT draindown test is to examine the CMT performance over the aforementioned temperature conditions to verify that the level instrumentation is properly operating and that the CMT transitions from recirculation to draindown mode. As described previously, proper operation of the CMT level instrumentation is verified by comparing the CMT upper narrow range level instrumentation and lower narrow range level instrumentation to the CMT wide range level instrumentation. During the CMT draindown test, permanent CMT wide range and upper/narrow range levels are collected. Test data of the CMT wide, narrow and upper range levels obtained are used to demonstrate that the acceptance criteria are met for the CMT draindown test. No temporary instrumentation is used for this test.

Sanmen Units 1 & 2 successfully performed the CMT draindown test. [

]

The Westinghouse review and evaluation of the Sanmen Units 1 & 2 test results to verify acceptability is documented in a test report. Haiyang Unit 1 successfully performed the CMT draindown test with similar results to Sanmen Units 1 & 2. SNC reviewed the test reports for Sanmen Units 1 & 2 and Haiyang Unit 1 and concurred with the conclusions.

<u>Applicability of CMT Recirculation and Draindown Tests to Vogtle Units 3 & 4</u>

Vogtle UFSAR Subsection 14.2.5 states that "…justification shall be provided that the results of the first plant only test or first three plant test are applicable to the subsequent plant." UFSAR Subsection 14.2.5 also provides the basis that "[b]ecause of the standardization of the AP1000 design, once these special tests have affirmed consistent passive system function they are not required on follow plants." Therefore, verifying standardization of the component design between Sanmen Units 1 & 2, Haiyang Unit 1 and Vogtle Units 3 & 4 provides the basis that the successful test results are applicable to Vogtle Units 3 & 4.

The critical design and construction attributes for the CMT recirculation and draindown tests are:

- CMT Balance Line design (line resistance and layout);
- CMT design (geometry, volume, location);
- CMT Injection Line (line resistance and layout); and
- Direct Vessel Injection (DVI) Line design (line resistance, layout).

For these components, standard design and procurement documentation is used for both Sanmen Units 1 & 2, Haiyang Unit 1, and Vogtle Units 3 & 4. The reactor vessel, steam generators, reactor coolant pumps and CMTs are manufactured using the same design specifications, are procured to the same quality requirements imposed by the design specifications and are built to the same standard tolerances. The use of standard design documentation confirms that the PXS and RCS system components used for this test are the within the standard AP1000 design parameters. Design changes made to any of these standard components are captured in the Westinghouse design change process. A review has confirmed that there are no site-specific design changes for either Sanmen Units 1 & 2 or Haiyang Unit 1 that alter the standard design features for any of the components involved in this test.

In addition to using standard design and procurement requirements, Vogtle Units 3 & 4 has multiple ITAAC which are applicable to the components involved in this test. Sanmen Units 1 & 2 and Haiyang Unit 1 use these same acceptance criteria. The following ITAAC will confirm the critical design and construction attributes related to this test:

- ITAAC No. 2.2.03.08vi, CMT volume
- ITAAC No. 2.2.03.08cii, CMT Balance Line resistance
- ITAAC No. 2.2.03.08ci, CMT Injection Line resistance

Vogtle Units 3 & 4 will still perform all other PXS pre-operational tests described in UFSAR Subsection 14.2.9.1.3 which will verify the installed components and associated piping and valves properly perform their design function. Based on the use of standard designed components, ITAAC for critical design features and pre-operational tests, the boundary conditions for the CMT recirculation tests are the same for Sanmen Unit 1 & 2, Haiyang Unit 1, and Vogtle Units 3 & 4.

Therefore, the successful completion and results of the first three plant CMT recirculation tests at Sanmen Unit 1 & 2 and Haiyang 1 are applicable to Vogtle Units 3 & 4, and the tests are not required to be performed at Vogtle Units 3 & 4.

<u>Change Description</u>

As stated above, the first three plant CMT recirculation tests were successfully completed at the first three AP1000 units, Sanmen Units 1 & 2 and Haiyang Unit 1, and the results are applicable to Vogtle Units 3 & 4. Therefore, the CMT recirculation tests are proposed to be deleted from the UFSAR Subsection 14.2.5, first three plant tests, and UFSAR Subsection 14.2.9.1.3, PXS pre-operational tests. Specifically, the proposed changes are:

- COL Item 2.D.(2)(a)4 requires the licensee to perform an ADS blowdown test as described in UFSAR 14.2.9.1.3, items (k) and (w). This COL condition is proposed to be deleted based on the successful completion of the tests at the first three AP1000 units.

- UFSAR Subsection 14.2.5 describes the first three plant only tests, including CMT recirculation tests. A statement is proposed to be added that the tests will not be run at Vogtle Units 3 & 4 based on the successful completion of the tests at the first three AP1000 units.

- UFSAR Subsection 14.2.9.1.3, items (k) and (w), describe the general test methods and acceptance criteria for the CMT recirculation tests. The CMT recirculation tests are proposed to be deleted from this section because the tests will not be run at Vogtle Units 3 & 4 based on the successful completion of the tests at the first three AP1000 units.

2.4 Response to 2012 NRC Letter on Crediting First Plant and First Three Plant Tests

By letter dated January 1, 2012 (ML 120040121) the NRC provided SNC with six topics that should be addressed in order to support crediting the tests at Sanmen Units 1 & 2 and Haiyang Unit 1. As much has changed in the six years since this letter was sent, SNC is providing responses to each of the six topics identified in the letter below.

<u>NRC Topic No. 1</u>
Provide to the NRC staff copies of the quality assurance program, test control program, test specifications, and test procedures (which we understand are in English) for staff review.

<u>SNC Response</u>
SNC has performed assessments of the quality assurance regulations and the test control program procedures. These assessments are available for NRC review. SNC has made the test procedures, reports and post-test analysis available for NRC review for each of the preoperational tests included in the scope of the LAR. All of these documents are in English.

NRC Topic No. 2

Validate the adequacy of the quality assurance program for construction and performance of the initial test program by providing a comparison of the quality assurance program implemented for the construction and initial test program at Sanmen Units 1 & 2 and Haiyang Units 1 & 2 to the quality assurance program as described in Section 17 of the FSAR for VEGP Units 3 and 4. This includes validating the adequacy of the quality assurance program for instrument calibration.

SNC Response

The purpose of the first plant and first three plant tests is to further establish the unique phenomenological performance parameters of the AP1000 design features. These special tests are in addition to preoperational testing that will be completed at every AP1000 unit. Additionally, the use of standard design documentation confirms that the system components used for these tests are within the standard AP1000 design parameters. Any design changes made to any of these standard components are captured in the Westinghouse design change process. A review has confirmed that there are no site-specific design changes for either Sanmen Units 1 & 2, Haiyang Unit 1, or Vogtle Units 3 & 4 that alter the standard design features for any of the components involved in this test. Because the purpose of these tests is to verify design phenomena and Westinghouse design control maintains a standard plant, a validation of the adequacy of the quality assurance program for construction is not necessary.

SNC has validated the adequacy of the quality assurance regulations and assessed adherence to these regulations through reviews of the tests. The reviews completed by SNC focused on the governing QA regulations, implementing procedures for the scope of tests in the LAR and the output documentation. SNC has reviewed the Quality Assurance regulations governing the tests to 10 CFR 50 Appendix B. SNC has also reviewed the implementing administrative procedures governing the testing. SNC reviewed the test procedures, reports and post-test analysis. These reviews in total demonstrated the QA requirements were implemented appropriately for the scope of the first plant and first three plant tests. All of these documents and the documentation of the reviews completed by SNC are available for NRC review.

Performance of the first plant and first three plant only tests at Sanmen Units 1 & 2 and Haiyang Unit 1 were performed following the China regulatory quality assurance requirements specified in HAF-003-1991, “Safety Regulations for Quality Assurance of Nuclear Power Plants.” A review comparing the requirements of 10 CFR Part 50 Appendix B and HAF-003-1991 was completed. This review used a matrixed table approach to compare the two regulations. The review compares the introduction and 13 sections of HAF-003-1991 to the introduction and 18 criteria of 10 CFR Part 50 Appendix B. The review concluded that the requirements of HAF-003-1991, as implemented, are comparable and encompass the requirements of 10 CFR Part 50 Appendix B. For any specific requirements in 10 CFR Part 50 Appendix B that are not directly included in HAF-003-1991, other standards were identified that implement the same requirements of 10 CFR Part 50 Appendix B.

HAF-003-1991, criterion XII requires calibration and testing of test equipment, which is comparable to 10 CFR Part 50 Appendix B criterion XII requirements. Calibration requirements for measurement and test equipment (M&TE) are specified in the Sanmen Units 1 & 2 and Haiyang Unit 1 localized procedures. Westinghouse had a team of engineers on site at both Sanmen Units 1 & 2 and Haiyang Unit 1 during performance of the first plant and first three plant tests. The Westinghouse engineers were embedded in the startup organization at the sites and worked alongside the owners as the testing was performed. Prior to the testing, Westinghouse and the owners walked-down instrumentation to confirm proper installation. Calibration records for temporary instrumentation used for the engineering analysis were provided by the Owners.

NRC Topic No. 3
Demonstrate the fidelity of the design, construction, and as-built condition of Sanmen Units 1 & 2 and Haiyang Units 1 & 2 to VEGP Units 3 & 4 in relation to the applicability of the FPOT/F3POT tests. This includes identifying any physical verification(s) that have been or will be performed.

SNC Response
The AP1000 plant is a standard design across the Sanmen, Haiyang and Vogtle units for the scope of the first plant and first three plant tests. Westinghouse created, approved and maintained the design documents governing the scope of SSCs in the first plant and first three plant tests. Any design change made to an SSC involved in the first plant or first three plant tests has been reviewed and approved by Westinghouse under the Westinghouse design control process. In the Westinghouse design control process, each of these design changes is designated with an applicability for any plant where the change would be applied. Changes which were only applicable to Sanmen Units 1 & 2 or Haiyang Unit 1 AP1000 units were reviewed for potential impacts to the first plant and first three plant test design parameters. The purpose of the review was to confirm the design of the SSCs involved in the first plant and first three plant tests at Sanmen Units 1 & 2 and Haiyang Unit 1 is within the standard AP1000 design and therefore substantially the same design as Vogtle 3&4. The review of design changes concluded that there were no site-specific design changes for the Sanmen Units 1 & 2 or Haiyang Unit 1 AP1000 units which would alter any of the critical design attributes for first plant or first three plant only tests. The components involved in the testing were procured using the same design specification requirements. There is no difference between the units for these design requirements. Based on the standard AP1000 design, review of design changes, and the use of the same procurement requirements, the Sanmen Units 1 & 2 and Haiyang Unit 1 SSCs for the first plant and first three plant tests are within the standard AP1000 design. The Vogtle 3 & 4 units follow the same design control process described above to maintain the standard plant design. Additionally, Vogtle Units 3 & 4 have ITAAC on the SSCs involved in the first plant and first three plant tests. The specific ITAAC relating to each test are described in the subsections below on each specific test. The completion of these ITAAC will confirm the SSCs for the first plant and first three plant tests meet the AP1000 standard design described in the Vogtle Units 3 & 4 UFSAR.

NRC Topic No. 4
Validate the adequacy of the initial test program. This includes a description of the administrative controls governing the initial test program at Sanmen Units 1 & 2 and Haiyang Units 1 & 2, including a comparison to the administrative controls governing the initial test program as described in Section 14.2 of the FSAR for VEGP Units 3 & 4.

SNC Response
SNC reviewed the Sanmen and Haiyang administrative manual procedures governing the first plant and first three plant tests. NRC Inspection Procedure (IP) 70367, “Inspection of Preoperational Test Program”, was used for guidance in creating criteria to review the Sanmen and Haiyang administrative manual procedures. The purpose of the review was to assess the

procedural processes and controls for the conduct of testing at Sanmen Units 1 & 2 and Haiyang Unit 1. This review followed the same process as the review of the Vogtle Units 3 & 4 administrative manual procedures against the NRC Inspection Procedure. Each line item in NRC IP Section 70367-02 was assessed individually against the Sanmen and Haiyang procedures. The overall conclusion of the review is that the Sanmen and Haiyang administrative manual procedures satisfy the requirements within NRC IP 70367. For instances where the NRC Inspection Procedure requirement was not explicitly addressed in a single administrative procedure, the requirement was addressed through a combination of procedures. There were no issues identified from the administrative manual procedures reviewed that would challenge any test results.

NRC Topic No. 5
Identify all the test critical parameters, calculations, and verification methods used during the initial test program. For calculated values this includes the calculated validation methodology, software verification and validation (when applicable), and the actual data inputs and outputs from the initial test results to support calculated values.

SNC Response
Within the LAR, each test critical parameter is described. All post-test calculations have been identified and made available to the NRC for review. These calculations described the methodology, any codes used and the evaluation of the test data.

NRC Topic No. 6
Document the controls that WEC will have in place during the initial test program to ensure that work performed to non-English procedures or instructions are correctly translated from English.

SNC Response
For the first plant and first three plant tests, all necessary test documentation is available in English. SNC has reviewed the administrative manual procedures, test procedures, test reports and post analysis, which are all in English. Based on SNC reviews of these test documents, SNC is confident that there are no communication concerns due to language.

2.5 Changes to Current Licensing Basis Documents

COL Condition Changes

Combined License Condition 2.D.(2)(a), Pre-operational Testing, is revised to remove the requirements to perform design-specific pre-operational first plant and first three plant tests including In Containment Refueling Water Storage Tank (IRWST) Heatup Test, Reactor Vessel Internals Vibration Testing, and Core Makeup Tank Heated Recirculation Tests. The changes credit previously completed tests which confirmed the design functions of the involved SSCs.

COL Appendix C (and Plant-Specific Tier 1) Changes

3. 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.

4. Table 2.1.3-2 is revised to delete ITAAC 2.1.03.07.i; item i) in the ITA and the AC

columns are identified as not used.

UFSAR Tier 2 Changes

6. UFSAR Subsection 3.9.2.4 is revised to describe Sanmen Unit 1 as the prototype for AP1000 reactor internals and Vogtle Units 3 & 4 reactor internals as non-prototype, Category 1.

7. UFSAR Table 3.9-4 is deleted and identified as not used.

8. UFSAR Subsection 14.2.5 is revised to add a statement that the IRWST heatup, CVAP, and CMT recirculation tests will not be run at Vogtle Units 3 & 4 based on the successful completion of the tests at the first AP1000 units.

9. UFSAR Subsection 14.2.9.1.3 is revised to remove the test descriptions for IRWST heatup first plant test and CMT recirculation first three plant tests, A statement is added that the test will not be run at Vogtle Units 3 & 4 based on the successful completion of the test at the first AP1000 units.

10. UFSAR Subsection 14.2.9.1.9 is revised to reflect the RG 1.20 guidance for a non-prototype, category I reactor internal design.

2.6 Summary

The proposed changes credit previously completed first plant and first three plant tests which confirmed the design functions of the involved SSCs. There are no changes to any pre-operational testing requirements from Regulatory Guide 1.68. The proposed changes involving CVAP comply with Regulatory Guide 1.20. The proposed changes do not affect any function or feature used for the prevention and mitigation of accidents or their safety analyses. No changes were made to the assumptions used in the Chapter 15 analyses. No safety-related structure, system, or component (SSC) function is changed. The proposed changes do not involve nor interface with any SSC accident initiator or initiating sequence of events related to the accidents evaluated in the plant-specific Design Control Document (DCD) or UFSAR. The proposed changes do not affect the radiological source terms (i.e., amounts and types of radioactive materials released, their release rates and release durations) used in the accident analyses. No system or design function or equipment qualification is adversely affected by the proposed changes. The changes do not result in a new failure mode, malfunction or sequence of events that could adversely affect a radioactive material barrier or safety-related equipment. The proposed changes do not allow for a new fission product release path, result in a new fission product barrier failure mode, or create a new sequence of events that would result in significant fuel cladding failures. The proposed changes do not adversely affect any design code limit allowable value, design analysis, nor do they adversely affect any safety analysis input or result, or design/safety margin. The proposed changes do not revise any aspects of the plant that could have any adverse effect on safety or security, including the site emergency plan.

3. TECHNICAL EVALUATION (Included in Section 2)

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR Part 52.98(c) requires an amendment to the license for any modification to, addition to, or deletion from the terms and conditions of a combined license, including modification to, addition to, or deletion from the inspections, tests, analyses, or related acceptance criteria contained in the license. This change involves changes to UFSAR Subsections 14.2.5 and 14.2.9, which requires a revision to the COL 2.D.(2)(a) and COL Appendix C ITAAC. Therefore, a license amendment request (LAR) (as supplied herein) is required.

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. The proposed changes to first plant and first three plant tests include changes to Tier 1 and Tier 2* information in UFSAR Subsections 14.2.5 and 14.2.9.1.3. Therefore, NRC approval is required for the departures.

10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1, requires that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. The proposed changes involve crediting first plant only and first three plant tests which were previously completed at the lead AP1000 units and do not need to be repeated at Vogtle Units 3 & 4. The test results confirmed the design functions of the involved SSCs. The proposed changes do not alter any design, analysis or test acceptance criteria. Therefore, the proposed changes comply with the requirements of GDC 1.

10 CFR Part 50, Appendix A, GDC 35 requires that a system to provide abundant emergency core cooling be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. The proposed changes to credit previously completed first plant and first three plant tests involving PXS do not include changes to any design feature or function described in the UFSAR. The changes credit previously completed tests which confirmed the design functions of the involved SSCs. Therefore, the proposed changes comply with the requirements of GDC 35.

10 CFR Part 50, Appendix A, GDC 36 requires that the emergency core cooling system be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system. The proposed changes to credit previously completed first plant and first three plant tests involving PXS do not include physical changes to any component. The changes credit previously completed tests which confirmed the design functions of the involved SSCs. Therefore, the proposed changes do not adversely affect the capability to perform appropriate inspections and comply with the requirements of GDC 36.

10 CFR Part 50, Appendix A, GDC 37 requires that the emergency core cooling system be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak tight integrity of its components, (2) the operability and performance of the

active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system. The proposed changes to credit previously completed first plant and first three plant tests involving PXS do not included changes to any design feature or function described in the UFSAR. The changes credit previously completed tests which confirmed the design functions of the involved SSCs. Therefore, the proposed changes comply with the requirements of GDC 37.

Regulatory Guide (RG) 1.20 describes the approved methodology to be used for vibratory stress analysis and measurement, inspections, documentation of results and schedule for CVAP. The proposed changes follow the guidance of RG 1.20 and do not adversely impact the UFSAR in terms of conformance to RG 1.20.

Regulatory Guide 1.68 describes the Initial Test Program (ITP) requirements. The proposed changes to first plant and first three plant tests do not alter compliance with RG 1.68 and the SSCs within the scope of RG 1.68 are still included in the ITP. The proposed changes to credit previously completed first plant and first three plant testing do not adversely impact the UFSAR in terms of conformance to RG 1.68.

The proposed changes have been evaluated to determine whether applicable regulations continue to be met. It was determined that the proposed changes do not affect conformance with the General Design Criteria differently than described in the plant-specific DCD or UFSAR.

4.2 Precedent

No precedent is identified.

4.3 Significant Hazards Consideration

The requested amendment involves changes to credit previously completed first plant and first three plant preoperational testing requirements in the VEGP Units 3&4 COLs and UFSAR Subsections 14.2.5 and 14.2.9 for IRWST heatup test, reactor vessel internals vibration testing, and CMT recirculation tests based on the successful completion of the tests at the lead AP1000 units.

The requested amendment proposes a change to COL Condition 2.D.(2)(a) and associated Tier 1 and UFSAR information supporting this change.

An evaluation to determine whether or not a significant hazards consideration is involved with the proposed amendment was completed by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of Amendment,” as discussed below.

4.3.1 Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change does not affect the operation of any systems or equipment that initiates an analyzed accident or alter any structures, systems, or components (SSC) accident initiator or initiating sequence of events. The proposed changes remove first plant and first three plant only tests including the IRWST heatup test, reactor vessel internals vibration testing, and CMT recirculation tests based on the successful completion of the tests at the lead AP1000 units. The change does not adversely affect any methodology which would increase the probability or consequences of a previously evaluated accident.

The change does not impact the support, design, or operation of mechanical or fluid systems. There is no change to plant systems or the response of systems to postulated accident conditions. There is no change to predicted radioactive releases due to normal operation or postulated accident conditions. The plant response to previously evaluated accidents or external events is not adversely affected, nor does the proposed change create any new accident precursors.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of a previously evaluated accident.

4.3.2 Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed change does not affect the operation of any systems or equipment that may initiate a new or different kind of accident, or alter any SSC such that a new accident initiator or initiating sequence of events is created.

The proposed change credits previously completed first plant and first three plant only tests including the IRWST heatup test, reactor vessel internals vibration testing, and CMT recirculation tests based on the successful completion of the tests at the lead AP1000 units. The proposed changes do not adversely affect any design function of any SSC design functions or methods of operation in a manner that results in a new failure mode, malfunction, or sequence of events that affect safety-related or non-safety-related equipment. This activity does not allow for a new fission product release path, result in a new fission product barrier failure mode, or create a new sequence of events that result in significant fuel cladding failures.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

4.3.3 Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

The proposed change maintains existing safety margin and provides adequate protection through continued application of the existing requirement in the UFSAR. The proposed change satisfies the same design functions in accordance with the same codes and standards as stated in the UFSAR. This change does not adversely affect any design code, function, design analysis, safety analysis input or result, or design/safety margin. No safety analysis or design basis acceptance limit/criterion is challenged or exceeded by the proposed change.

Since no safety analysis or design basis acceptance limit/criterion is challenged or exceeded by this change, no significant margin of safety is reduced.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, it is concluded that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of “no significant hazards consideration” is justified.

4.4 Conclusions

This assessment addresses the considerations discussed above. The plant licensing basis, safety analyses, and design bases evaluations demonstrate that the requested change is accommodated without an increase in the probability or consequences of an accident previously evaluated, without creating the possibility of a new or different kind of accident from any accident previously evaluated, and without a significant reduction in the margin of safety. In conclusion, based on the considerations discussed above, (1) there is a reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) 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. Having arrived at negative declarations with regard to the criteria of 10 CFR 50.92, this assessment determined that the requested change does not involve a Significant Hazards Consideration.

5. ENVIRONMENTAL CONSIDERATIONS

This review supports a request to amend the Combined License (COL) to revise various elements of the certification information related to pre-operational first plant and first three plant test requirements including IRWST heatup test, reactor vessel internals vibration testing, and CMT recirculation tests in Updated Final Safety Analysis Report (UFSAR), the plant-specific Tier 1 information, and COL Condition 2.D.(2)(a).

Sections 2 and 3 of this license amendment request provide the details of the proposed change.

The Licensee has determined that the anticipated construction and operational effects of the proposed amendment meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), in that:

(i) *There is no significant hazards consideration.*

As documented in Section 4.3, Significant Hazards Consideration, of this license amendment request, an evaluation was completed to determine whether or not a significant hazards consideration is involved by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of Amendment.” The significant hazards consideration determined that (1) the requested amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) the requested amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) the requested amendment does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the requested amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of “no significant hazards consideration” is justified.

(ii) *There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.*

The proposed change is unrelated to any aspect of plant construction or operation that would introduce any change to effluent type (e.g., effluents containing chemicals or biocides, sanitary systems effluents, and other effluents), or affect any plant radiological or non-radiological effluent release quantities. Furthermore, the proposed changes do not affect any effluent release path or diminish the design function or operational features that are credited with controlling the release of effluents during plant operation. Therefore, it is concluded that the requested amendment does not involve a significant change in the types or a significant increase on the amounts of any effluents that may be released offsite.

(iii) *There is no significant increase in individual or cumulative occupational radiation exposure.*

The proposed changes do not adversely affect walls, floors, or other structures that provide shielding. Plant radiation zones are not affected, and there are no changes to the controls required under 10 CFR Part 20 that preclude a significant increase in occupational radiation exposure. Therefore, the requested amendment does not involve a significant increase in individual or cumulative occupational radiation exposure.

Based on the above review of the requested amendment, it has been determined that the anticipated construction and operational impacts of the requested amendment do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the requested amendment 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 need be prepared in connection the requested amendment.

6. REFERENCES

None.

Southern Nuclear Operating Company

ND-18-0826

Enclosure 3

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Exemption Request:

Crediting Previously Completed First Plant and First Three Plant Tests

(LAR-18-019)

(This Enclosure consists of 7 pages, including this cover page**)**

1.0 Purpose

Southern Nuclear Operating Company (SNC, the Licensee) requests a permanent exemption from the provisions of 10 CFR Part 52, Appendix D, Section III.B, *Design Certification Rule for the AP1000 Design, Scope and Contents*, to allow a departure from elements of the certification information in Tier 1 of the generic AP1000 Design Control Document (DCD). The regulation, 10 CFR Part 52, Appendix D, Section III.B, requires an applicant or licensee referencing Appendix D to 10 CFR Part 52 to incorporate by reference and comply with the requirements of Appendix D, including certified information in DCD Tier 1. The Tier 1 information for which a plant-specific departure and exemption is being requested includes revisions to Inspections, Tests, Analysis and Acceptance Criteria (ITAAC) for reactor internal flow induced vibration first plant test.

This request for exemption provides the technical and regulatory basis to demonstrate that 10 CFR 52.63, §52.7, and §50.12 requirements are met and will apply the requirements of 10 CFR Part 52, Appendix D, Section VIII.A.4 to allow departures from generic Tier 1 information to delete ITAAC 2.1.03.07.i for the first plant reactor internal flow induced vibration report in Table 2.1.3-2.

2.0 Background

As described in the Combined License (COL) Condition 2.D.(2)(a), the licensee shall perform design-specific pre-operational tests including Reactor Vessel Internals Vibration Testing. The Reactor Vessel Internals Vibration Testing is designated as a first plant only test. First plant only tests are described in UFSAR Subsection 14.2.5. The tests are described as "special tests to further establish a unique phenomenological performance parameter of the AP1000 design features beyond testing performed for Design Certification of the AP600 and that will not change from plant to plant…."

The reactor vessel internals vibration testing is part of the Comprehensive Vibration Assessment Program (CVAP). CVAP is performed in accordance with Regulatory Guide 1.20 Revision 2. The AP1000 vibration assessment program requirements are described in UFSAR Subsection 3.9.2.4 and topical reports WCAP-17983 and WCAP-17984. CVAP is directed toward confirming the long-term, steady-state vibration response of the reactor internals for operating steady-state and transient conditions. The three aspects of this evaluation are the following: a prediction of the vibrations of the reactor internals (based in part on previous scale model tests), a preoperational vibration test program of the internals of the first plant, and a correlation of the analysis and test results.

3.0 Technical Justification of Acceptability

Vogtle UFSAR Subsection 14.2.5 states "…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 provides the basis that "because of the standardization of the AP1000 design, these special tests (designated as first plant only tests) are not required on follow plants."

In addition to Vogtle Units 3 & 4, there are four other AP1000 units. These are Sanmen Units 1 & 2 and Haiyan Units 1 & 2. Sanmen Unit 1 has already performed the first plant only Reactor Vessel Internals Vibration Testing described in UFSAR Subsections 3.9.2.4, and 14.2.5. AP1000 vibration assessment program requirements are currently described in topical reports WCAP-17983 and WCAP-17984. These program requirements are applicable to the testing completed at Sanmen Unit 1 and are also applicable to Vogtle Units 3 & 4.

As described in UFSAR Subsection 3.9.2.4, with respect to the reactor internals preoperational test program, the first AP1000 plant reactor vessel internals are classified as a prototype as defined in Regulatory Guide 1.20. Although the AP1000 reactor internals do not represent a first-of-a-kind or unique design based on the arrangement, design, size, or operating conditions, for the purposes of the reactor internals preoperational test program, the first operational AP1000 reactor vessel internals are classified as a prototype. Sanmen Unit 1, as the first AP1000, is proposed to be classified as prototype as defined in RG 1.20. In accordance with RG 1.20, the results of the prototype reactor internals CVAP should submitted to the Commission in the form of preliminary and final reports. These reports are also required to satisfy the acceptance criteria of ITAAC 2.1.03.07.i. These reports are being submitted to the Commission in support of this exemption. Based on the proposed classification of Sanmen Unit 1 as prototype, Vogtle Units 3 & 4 reactor internals are proposed to be classified as non-prototype, Category I. Based on this classification, the instrumented first plant test is not required for Vogtle Units 3 & 4. Vogtle Units 3 & 4 will perform the required inspections per RG 1.20 for non-prototype, Category I. ITAAC 2.1.03.07.ii captures this requirement. Therefore, ITAAC 2.1.03.07.i is satisfied by submittal of the preliminary and final reports and is no longer necessary for Vogtle Units 3 & 4 and is proposed to be deleted.

Detailed technical justification supporting this request for exemption is provided in Section 2 of the associated License Amendment Request in Enclosure 1 of this letter.

4.0 Justification of Exemption

10 CFR Part 52, Appendix D, Section VIII.A.4 and 10 CFR 52.63(b)(1) govern the issuance of exemptions from elements of the certified design information for AP1000 nuclear power plants. Since SNC has identified changes to the Tier 1 information as discussed in Enclosure 1 of the accompanying License Amendment Request, an exemption from the certified design information in Tier 1 is needed.

10 CFR Part 52, Appendix D, and 10 CFR 50.12, §52.7, and §52.63 state that the NRC may grant exemptions from the requirements of the regulations provided six conditions are met: 1) the exemption is authorized by law [§50.12(a)(1)]; 2) the exemption will not present an undue risk to the health and safety of the public [§50.12(a)(1)]; 3) the exemption is consistent with the common defense and security [§50.12(a)(1)]; 4) special circumstances are present [§50.12(a)(2)]; 5) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption [§52.63(b)(1)]; and 6) the design change will not result in a significant decrease in the level of safety [Part 52, App. D, VIII.A.4].

The requested exemption satisfies the criteria for granting specific exemptions, as described below.

1) This exemption is authorized by law

The NRC has authority under 10 CFR 52.63, §52.7, and §50.12 to grant exemptions from the requirements of NRC regulations. Specifically, 10 CFR 50.12 and §52.7 state that the NRC may grant exemptions from the requirements of 10 CFR Part 52 upon a proper showing. No law exists that would preclude the changes covered by this exemption request. Additionally, granting of the proposed exemption does not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations.

Accordingly, this requested exemption is "authorized by law," as required by 10 CFR 50.12(a)(1).

2) This exemption will not present an undue risk to the health and safety of the public

The proposed exemption from the requirements of 10 CFR Part 52, Appendix D, Section III.B would allow changes to elements of the plant-specific Tier 1 DCD to depart from the AP1000 certified (Tier 1) design information. The plant-specific DCD Tier 1 will continue to reflect the approved licensing basis, and will maintain a consistent level of detail with that which is currently provided elsewhere in Tier 1 of the DCD. Therefore, the affected plant-specific DCD Tier 1 ITAAC will continue to serve its required purpose.

The revisions to ITAAC on first plant reactor internal flow induced vibration testing do not represent an adverse impact to the design functions supported by the equipment, or the associated systems, structures and components and will continue to protect the health and safety of the public in the same manner. The clarifications and additional exceptions do not introduce any new industrial, chemical, or radiological hazards that would represent a public health or safety risk, nor do they modify or remove any design or operational controls or safeguards intended to mitigate any existing on-site hazards. Furthermore, the proposed change would not allow for a new fission product release path, result in a new fission product barrier failure mode, or create a new sequence of events that would result in fuel cladding failures. Accordingly, this change does not present an undue risk from any existing or proposed equipment or systems.

Therefore, the requested exemption from 10 CFR Part 52, Appendix D, Section III.B would not present an undue risk to the health and safety of the public.

3) The exemption is consistent with the common defense and security

The requested exemption from the requirements of 10 CFR Part 52, Appendix D, Section III.B would allow the licensee to depart from elements of the plant-specific Tier 1 design information. The proposed exemption does not alter the design, function, or operation of any structures or plant equipment that is necessary to maintain a safe and secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures.

Therefore, the requested exemption is consistent with the common defense and security.

4) Special circumstances are present

10 CFR 50.12(a)(2) lists six "special circumstances" for which an exemption may be granted. Pursuant to the regulation, it is necessary for one of these special

circumstances to be present in order for the NRC to consider granting an exemption request. The requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii). That subsection defines special circumstances as when "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 rule under consideration in this request for exemption is 10 CFR Part 52, Appendix D, Section III.B, which requires that a licensee referencing the AP1000 Design Certification Rule (10 CFR Part 52, Appendix D) shall incorporate by reference and comply with the requirements of Appendix D, including Tier 1 information. The VEGP Units 3 and 4 COLs reference the AP1000 Design Certification Rule and incorporate by reference the requirements of 10 CFR Part 52, Appendix D, including Tier 1 information. The underlying purpose of Appendix D, Section III.B is to describe and define the scope and contents of the AP1000 design certification, and to require compliance with the design certification information in Appendix D.

The proposed exemption would provide revisions to ITAAC on reactor internal flow induced vibration. The proposed revisions reflect the design functions of the associated systems and components as described in the licensing basis documents. Accordingly, this exemption from the certification information enables the Licensee to safely construct and operate the facility consistent with the design certified by the NRC in 10 CFR Part 52, Appendix D.

Therefore, special circumstances are present, because application of the current generic certified design information in Tier 1 as required by 10 CFR Part 52, Appendix D, Section III.B, in the particular circumstances discussed in this request is not necessary to achieve the underlying purpose of the rule.

5) The special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption.

Based on the nature of the changes to the plant-specific Tier 1 information and the understanding that these changes support the design function of the supported equipment, it is expected that this exemption may be requested by other AP1000 licensees and applicants. However, a review of the reduction in standardization resulting from the departure from the standard DCD determined that even if other AP1000 licensees and applicants do not request this same departure, the special circumstances will continue to outweigh any decrease in safety from the reduction in standardization because the key design functions of the equipment associated with this request will continue to be maintained. Furthermore, the justification provided in the license amendment request and this exemption request and the associated mark-ups demonstrate that there is a limited change from the standard information provided in the generic AP1000 DCD, which is offset by the special circumstances identified above.

Therefore, the special circumstances associated with the requested exemption outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption.

6) The design change will not result in a significant decrease in the level of safety.

The exemption revises the plant-specific DCD Tier 1 information by revising ITAAC on reactor internal flow induced vibration. The revisions do not change the design requirements of the associated equipment. Because these functions continue to be met, there is no reduction in the level of safety.

5.0 Risk Assessment

A risk assessment was not determined to be applicable to address the acceptability of this proposal.

6.0 Precedent Exemptions

None

7.0 Environmental Consideration

The Licensee requests a departure from elements of the certified information in Tier 1 of the generic AP1000 DCD. The Licensee has determined that the proposed departure would require a permanent exemption from the requirements of 10 CFR Part 52, Appendix D, Section III.B, *Design Certification Rule for the AP1000 Design, Scope and Contents,* with respect to installation or use of facility components located within the restricted area, as defined in 10 CFR Part 20, or which changes an inspection or a surveillance requirement; however, the Licensee evaluation of the proposed exemption has determined that the proposed exemption meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Based on the above review of the proposed exemption, the Licensee has determined that the proposed activity does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed 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), an environmental impact statement or environmental assessment of the proposed exemption is not required.

Specific details of the environmental considerations supporting this request for exemption are provided in Section 5 of the associated License Amendment Request provided in Enclosure 1 of this letter.

8.0 Conclusion

The proposed changes to Tier 1 are necessary to revise ITAAC on reactor internal flow induced vibration. The exemption request meets the requirements of 10 CFR 52.63, *Finality of design certifications*, 10 CFR 52.7, *Specific exemptions*, 10 CFR 50.12, *Specific exemptions*, and 10 CFR Part 52 Appendix D, *Design Certification Rule for the AP1000*. Specifically, the exemption request meets the criteria of 10 CFR 50.12(a)(1) in that the request is authorized by law, presents no undue risk to public health and safety, and is consistent with the common defense and security. Furthermore, approval of this request does not result in a significant decrease in the level of safety, satisfies the underlying purpose of the AP1000 Design Certification Rule, and does not present a significant decrease in safety as a result of a reduction in standardization.

9.0 References

None

Southern Nuclear Operating Company

ND-18-0826

Enclosure 4

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Proposed Changes to the Licensing Basis Documents

(LAR-18-019)

Note:

Added text is shown as Blue Underline

Deleted text is shown as ~~Red Strikethrough~~

Relocated text is show in Green Underline ~~and Strikethrough~~

Omitted text is shown as three asterisks (*...*...*)

(This Enclosure consists of 11 pages, including this cover page)

COL Changes

Combined License Condition 2.D.(2)(a), Pre-operational Testing – Revise information related to design-specific pre-operational first plant and first three plant tests including In Containment Refueling Water Storage Tank (IRWST) Heatup Test, Reactor Vessel Internals Vibration Testing, and Core Makeup Tank Heated Recirculation Tests as shown below.

(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 UFSAR Section 14.2.9.1.3 Item (h));~~

1~~.2.~~ Pressurizer Surge Line Stratification Evaluation (first plant test as identified in UFSAR, Section 14.2.9.1.7 Item (d));

~~3. Reactor Vessel Internals Vibration Testing (first plant test as identified in UFSAR Section 14.2.9.1.9);~~

~~4. Core Makeup Tank Heated Recirculation Tests (first three plants test as identified in UFSAR Section 14.2.9.1.3 Items (k) and (w)); and~~

2~~.5.~~ Automatic Depressurization System Blowdown Test (first three plants test as identified in UFSAR Section 14.2.9.1.3 Item (s)).

COL Appendix C, Subsection 2.1.3 is revised to renumber items under Design Description.

The component locations of the RXS are as shown in Table 2.1.3-3.

1. The functional arrangement of the RXS is as described in the Design Description of this Section 2.1.3.

2. a) The reactor upper internals rod guide arrangement is as shown in Figure 2.1.3-1.

   b) The rod cluster control and drive rod arrangement is as shown in Figure 2.1.3-2.

   c) The reactor vessel arrangement is as shown in Figure 2.1.3-3.

3~~2~~. The components identified in Table 2.1.3-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

4~~3~~. Pressure boundary welds in components identified in Table 2.1.3-1 as ASME Code Section III meet ASME Code Section III requirements.

5~~4~~. The pressure boundary components (reactor vessel [RV], control rod drive mechanisms [CRDMs], and incore instrument QuickLoc assemblies) identified in Table 2.1.3-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

6~~5~~. The seismic Category I equipment identified in Table 2.1.3-1 can withstand seismic design basis loads without loss of safety function.

7~~6~~. The reactor internals will withstand the effects of flow induced vibration.

8~~7~~. The reactor vessel direct injection nozzle limits the blowdown of the reactor coolant system (RCS) following the break of a direct vessel injection line.

COL Appendix C (and plant-specific Tier 1), Subsection 2.1.3, Table 2.1.3-2, Revise to delete ITAAC 2.1.03.07.i

Table 2.1.3-2
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
***				
78	2.1.03.07.i	7. The reactor internals will withstand the effects of flow induced vibration.	i) Not Used per Amendment XXX. ~~A vibration type test will be conducted on the (first unit) reactor internals representative of AP1000.~~	i) Not Used per Amendment XXX. ~~A report exists and concludes that the (first unit) reactor internals have no observable damage or loose parts as a result of the vibration type test.~~
***				

UFSAR Tier 2 Changes

UFSAR Subsection 3.9.2.4, Pre-operational Flow-Induced Vibration Testing of Reactor Internals, revise to describe Sanmen Unit 1 as the valid protype and Vogtle Units 3 & 4 as non-prototype.

~~The pre-operational vibration test program for the reactor internals of the AP1000 conducted on the first AP1000 is consistent with the guidelines of Regulatory Guide 1.20 for a comprehensive vibration assessment program. Design features that have not previously been tested in the reference plants or subsequent testing are tested to verify the vibration analysis.~~ The VEGP Units 3 & 4 reactor internals are classified as non-prototype, Category I as defined in Regulatory Guide 1.20. Sanmen Unit 1, as the first AP1000 plant, is the valid prototype for the AP1000 reactor internal design. Based on this classification, an instrumented test is not required. The recommendations of Regulatory Guide 1.20 are satisfied by conducting the confirmatory pre- and post-hot functional examination for integrity. Conformance with Regulatory Guide 1.20 is summarized in Subsection 1.9.1.

~~The program is directed toward confirming the long-term, steady-state vibration response of the reactor internals for operating conditions. The three aspects of this evaluation are the following: a prediction of the vibrations of the reactor internals, a preoperational vibration test program of the internals of the first plant, and a correlation of the analysis and test results.~~

~~With respect to the reactor internals preoperational test program, the first AP1000 plant reactor vessel internals are classified as a prototype as defined in Regulatory Guide 1.20. The AP1000 reactor vessel internals do not represent a first-of-a-kind or unique design based on the arrangement, design, size, or operating conditions. The units referenced in Subsection 3.9.2.3 as supporting the AP1000 reactor vessel internals design features and configuration have successfully completed vibration assessment programs including vibration measurement programs. These units have subsequently demonstrated extended satisfactory inservice operation.~~

The reactor internals flow-induced vibration assessment program is documented in WCAP-17983 (Reference 40) and WCAP-17984 (Reference 41), including comparison of the AP1000 plant and operating plants that have undergone vibration measurement programs.

~~The pre-operational test program of the first AP1000 plant includes a vibration measurement program and a pre- and post-hot functional inspection program. This program satisfies the guidelines for a Regulatory Guide 1.20 Prototype Category plant. The program for plants subsequent to the first plant satisfies the guidelines for the appropriate Non-prototype Category plant.~~

The acceptance criteria for the vibration predictions are established and related to the ASME Code allowables for long term steady-state conditions.

During the hot functional test, the internals are subjected to flow conditions representative of normal operation for a sufficient length of time go generate a cyclic loading of greater than 10^6 cycles on the main structural elements of the internals. In addition, there is some operating time with one, two, or three pumps operating.

~~Instrumentation is designed and installed to measure the vibration of the internals during hot~~

~~functional testing. The instrumentation includes devices attached to reactor vessel internals to measure component strains and accelerations.~~

Inspection before and after the hot functional test serves to confirm the structural integrity of the internals with regard to flow-induced vibrations. When no indications of harmful vibrations or signs of abnormal wear are detected and no apparent structural changes take place, the core support structures are considered to be structurally adequate and sound for operation. If such indications are detected, further evaluation is required.

The testing and inspection plan ~~for the first plant~~ includes features with emphasis on the areas susceptible to FIV, including flow-induced wear. Consistent with Regulatory Guide 1.20, these areas include as a minimum:

- The major load-bearing elements of the reactor internals relied upon to retain the core support structures in position,
- The lateral, vertical, and torsional restraints provided within the vessel,
- The locking and bolting components whose failure could adversely affect the structural integrity of the reactor internals,
- Those surfaces that are known to be or may become contact surfaces during operation,
- Those critical locations on the reactor vessel internal components as identified by the vibration analysis, and
- The interior of the reactor vessel for evidence of loose parts or foreign material.

~~The reactor internals flow-induced vibration measurement program will be conducted during preoperational tests of the first AP1000. The response of the reactor and the internals due to flow-induced vibration will be measured during the hot functional test. Data will be acquired at several temperatures from cold startup to hot standby conditions. The location of the is outlined in Table 3.9-4.~~

***

Table 3.9-4, First Plant AP1000 Reactor Internals Vibration Measurement Program Transducer Locations, delete table as shown below

Table 3.9-4 Not Used

~~**Table 3.9-4**~~

~~**Selected First Plant AP1000 Reactor Internals**~~

~~**Vibration Measurement Program Transducer Locations**~~

~~Instrumented Component~~	~~Number and Type of Transducers~~[1]	~~Approximate Transducer Locations~~	~~Direction of Sensitivity~~
~~Core Shroud (Inner Wall)~~	~~4 accelerometers~~	~~0°, 45°, 180°, 315°~~	~~Radial~~
~~Core Shroud to Core Barrel~~	~~4 relative displacement transducers~~	~~225°, 315°~~	~~2 Radial~~ ~~2 Tangential~~
~~Core Barrel Flange (Outer Wall)~~	~~4 strain gages~~	~~0°, 90°, 180°, 270°~~	~~Axial~~
~~Core Barrel (Inner Wall)~~	~~1 strain gage~~	~~180°~~	~~Axial~~
~~Core Barrel Mid-elevation (Outer Wall)~~	~~3 accelerometers~~	~~0°, 45°, 180°~~	~~Radial~~
~~Core Barrel Mid-elevation~~	~~1 pressure transducer~~	~~90°~~	~~Radial~~
~~Upper Support Skirt (Inner Wall)~~	~~3 strain gages~~	~~0°, 90°, 180°~~	~~Axial~~
~~Upper Support Plate (Outer Wall)~~	~~1 strain gage~~	~~90°~~	~~Axial~~
~~Lower Core Support Plate~~	~~1 accelerometer~~	~~Near the center of the plate~~	~~Veritical~~
~~Vortex Suppression Plate Support Columns (2)~~	~~4 strain gages~~	~~On outside of columns at an elevation near LCSP with 3 gages on one column and 1 gage on another column; these two columns are 180° apart~~	~~Axial~~
~~Reactor Vessel (Head Studs)~~	~~4 accelerometers~~	~~Studs at 0°, 90°, 180°, 270°~~	~~Vertical~~
	~~3 accelerometers~~	~~Studs at 0°, 90°, stud at 180° (x-direction), stud at 180°~~ ~~(y-direction)~~	~~Horizontal~~
~~IGA Guide Tubes (2)~~	~~4 strain gages~~	~~0°, 90°~~	~~Axial~~
~~IGA Guide Tube Support~~	~~3 strain gages~~	~~Bottom of support~~	~~Axial~~
~~Lower Guide Tube on B-6~~	~~4 strain gages~~	~~0°, 90°, 180°, 270°~~	~~Axial~~
~~Upper Guide Tube on B-6~~	~~2 strain gages~~	~~0°, 90°~~	~~Axial~~
~~Upper Support Column on B-7~~	~~4 strain gages~~	~~0°, 90°, 180°, 270°~~	~~Axial~~

~~Notes:~~

~~1. Additional sensors are installed throughout the reactor internals assembly for redundancy and enhanced data collection. WCAP-17983 (Reference 40) Table 4-1 provides additional detailed information on the sensors.~~

UFSAR Subsection 14.2.5, Utilization of Reactor Operating and Testing Experience in the Development of Initial Test Program, revise text as shown below.

IRWST Heatup Test (14.2.9.1.3, item (h))

<u>This first plant only test was completed at the first AP1000 unit. This test is not required to be conducted at Vogtle Units 3 & 4.</u>

During preoperational testing of the passive core cooling system, a natural circulation test and a forced flow test of the passive residual heat removal (PRHR) heat exchanger is conducted (items f and g). For the first plant only, temperature sensors are placed in the IRWST to observe the thermal profile developed during the heatup of the IRWST water during PRHR heat exchanger operation. This test will be useful in confirming the results of the AP600 Design Certification Program PRHR tests with regards to IRWST mixing, and is useful in quantifying the conservatism in the Chapter 15 transient analyses.

Due to the standardization of the AP1000, the heatup and thermal stratification characteristics of the IRWST will not vary from plant to plant. The PRHR heat exchanger design, and the size and configuration of the IRWST are standardized, such that the heatup characteristics will not significantly change from plant to plant.

Therefore, since the phenomenon to be tested (i.e., heatup and mixing characteristics of the IRWST) will not vary significantly from plant to plant due to standardization, a first plant only test of the IRWST heatup characteristics is justified.

Core Makeup Tank Heated Recirculation Tests (14.2.9.1.3, Items (k) and (w))

<u>This first three plant only tests were completed at the first three AP1000 units. These tests are not required to be conducted at Vogtle Units 3 & 4.</u>

During preoperational testing of the passive core cooling system, a test is performed for each plant to verify the CMT inlet piping resistances. In addition, cold draining tests of the CMTs are conducted that verify the discharge piping resistance and proper drain rate of the CMTs for each plant. For the first three plants, two additional CMT tests are conducted during hot functional testing of the RCS. These tests are a natural circulation heatup of the CMTs followed by a test to verify the ability of the CMTs to transition from a recirculation mode to a draindown mode while at elevated temperature and pressure.

Operation of the CMTs in their natural circulation mode is conducted on the first three plants only for the following reasons:

- Natural circulation of the CMTs will not vary from plant to plant, provided that the other verifications discussed above are performed as specified.
- Natural circulation testing of the CMTs was extensively tested as part of the Design Certification Tests.
- Performance of this test results in significant thermal transients on Class 1 components including the CMTs and the direct vessel injection nozzles.

Reactor Vessel Internals Vibration Testing (14.2.9.1.9)

This first plant only instrumented vibration test was completed at the first AP1000 unit. This test is not required to be conducted at Vogtle Units 3 & 4.

The preoperational vibration test program for the reactor internals of the AP1000 conducted on the first AP1000 is consistent with the guidelines of Regulatory Guide 1.20 for a comprehensive vibration assessment program. This program is discussed in Subsection 3.9.2.

UFSAR Subsection 14.2.9.1.3, delete general test method and acceptance criteria for items h, k and w as shown below.

14.2.9.1.3 Passive Core Cooling System Testing

* * *

h) This first plant only test was completed at the first AP1000 unit. This test is not required to be conducted at Vogtle Units 3 & 4. ~~The heatup characteristics of the in-containment refueling water storage tank water are verified by measuring the vertical water temperature gradient that occurs in the in-containment refueling water storage tank water at the passive residual heat removal heat exchanger tube bundle and at several distances from the tube bundle, during testing in Items f) and g), above.~~ ~~**Note that this verification is required only for the first plant.**~~ ~~The acceptance criterion demonstrates that the average IRWST heatup is consistent with the PRHR heat transfer modeling in the Chapter 15 analysis. These results (in conjunction with Items f) and g)) are evaluated to demonstrate that the overall PRHR heat transfer performance, i.e., heat removal from the RCS, is conservative with respect to the analysis documented in Chapter 15.~~

* * *

k) This first three plant only test was completed at the first three AP1000 units. This test is not required to be conducted at Vogtle Units 3 & 4. ~~[*Proper operation of the core makeup tanks to perform their reactor water makeup and boration function is verified by initiating recirculation flow through the tanks during hot functional testing with the reactor coolant system at≥ 530°F. This testing is initiated by simulating a safety signal which opens the tank discharge isolation valves, and stops reactor coolant pumps after the appropriate time delay. The proper tank recirculation flow after the pumps have coasted down is verified. Based on the cold leg temperature, CMT discharge temperature, and temporary CMT flow instrumentation, the net mass injection rate into the reactor is verified.* ***Note that this verification is required only for the first three plants.***]*~~

* * *

w) This first three plant only test was completed at the first three AP1000 units. This test is not required to be conducted at Vogtle Units 3 & 4.~~[*In conjunction with the verification of the core makeup tanks to perform their reactor water makeup function and boration function described in item k) above, the proper operation of the core makeup tanks to transition from their recirculation mode of operation to their draindown mode of operation after heatup will be*~~

~~*verified. This testing will also verify the proper operation of the core makeup tank level instrumentation to operate during draining of the heated tank fluid. The in-containment refueling water storage tank initial level is reduced to at least 3 feet below the spillway level as a prerequisite condition for this testing in order to provide sufficient ullage to accept the mass discharged from the reactor coolant system via the automatic depressurization stage 1.*~~

~~*The recirculation operation in Item k) above, should be continued until the core makeup tank fluid has been heated to≥ 350°F. The core makeup tank isolation valves are then closed, the reactor coolant pumps are started, and the reactor coolant system is reheated up to hot functional testing conditions. This testing is initiated by shutting off the reactor coolant pumps, opening the core makeup tank isolation valves, and by opening one of the automatic depressurization stage 1 flow paths to the in-containment refueling water storage tank. This will initiate a large loss of mass from the reactor coolant system, depressurization of the reactor coolant system to the bulk fluid saturation pressure, and additional recirculation through the core makeup tank. Core makeup tank draindown initiates in response to the continued depressurization and mass loss from the reactor coolant system. The automatic depressurization stage 1 flow path is closed after the core makeup tank level has decreased below the level at which stage4 actuation occurs.*~~ ~~***Note that this verification is required only for the first three plants.***~~~~*]**~~

* * *

UFSAR Subsection 14.2.9.1.9, Reactor Vessel Internals Vibration Testing, revise to remove instrumented first plant CVAP test requirements as shown below.

Purpose
The AP1000 reactor internals testing is part of a comprehensive vibration assessment program performed in accordance with Regulatory Guide 1.20 as discussed in Subsection 3.9.2.4. This ~~testing obtains data to~~ verifies ~~verify~~ the structural integrity of the AP1000 reactor internals with regard to flow-induced vibrations, as part of an internals vibration assessment program. This program ~~also~~ includes visual examination of the reactor internals before and after hot functional testing.~~, and analysis of the test data.~~

~~AP1000 plants~~ The reactor internals are visually inspected before and after the hot functional test to confirm the structural integrity of the reactor internals with regard to flow-induced vibrations. The major features of the reactor internals outlined in Subsection 3.9.2.4 are visually inspected for signs of abnormal wear and structural changes.

Prerequisites
The component testing of the reactor coolant system has been completed. The first plant only portions of this test have been completed at the first AP1000 unit. The ~~testing and calibration of the required test instrumentation has been completed. The test instrumentation is located on the internals as specified in Table 3.9-4 and the.~~ internals pre-test visual inspection has been completed. The internals~~, test instrumentation, and instrumentation lead wires~~ are installed in the reactor vessel. The reactor vessel head is installed in preparation for the cold hydrostatic test of the reactor coolant system. ~~and instrument leads have been properly sealed. The proper~~

~~operation and calibration of the test instrumentation and recording equipment is verified during the hydrostatic testing of the reactor coolant system.~~

General Test Method and Acceptance Criteria

~~Reactor vessel internals testing is performed by measuring and recording component strains and accelerations in order to determine actual displacements that occur with the reactor coolant pumps operating. This testing is performed at several reactor coolant system temperatures during the system hot functional test. The analysis of data obtained from this testing, combined with a p~~Pre-test and post-test visual inspections of the internals are performed to confirm that the stresses and wear on the AP1000 internals, due to flow induced vibration during plant operation, are acceptably low. The criteria for evaluating testing results are established in the AP1000 reactor internals flow-induced vibration assessment program (see Section 7 of WCAP-17984), and appropriate design specifications.

~~When an instrumented test is performed, the internals are instrumented to obtain data during the following reactor coolant system operating conditions:~~

~~a) Background noise in the instrumentation and recording equipment is recorded with no reactor coolant pumps running~~

~~b) Data is recorded during the initial startup of the reactor coolant pumps and with all four pumps operating and with the reactor coolant at cold temperature~~

~~c) Data is recorded at several increasing coolant temperatures with the pumps operating~~

~~d) Data is recorded at the hot functional testing temperature with all four pumps operating~~

~~e) Data is recorded at the hot functional testing temperature with the appropriate combinations of reactor coolant pumps operating, including pump start and stop transients~~

~~f) Shutdown conditions including temperature cooldown and pump speed reduction~~

~~When viusals inpsections are performed, v~~Visual inspections are performed before and after the hot functional test. When no indications of harmful vibrations or signs of abnormal wear are detected and no structural damage or changes are apparent, the core support structures are considered to be structurally adequate and sound for operation. If such indications are detected, further evaluation is required.

~~For the first plant, the vibration assessment program includes an instrumented test, analysis of the data, and visual inspections.~~

Southern Nuclear Operating Company

ND-18-0826

Enclosure 5

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Westinghouse Proprietary Information Notice, Copyright Notice and CAW-18-4753, Application for Withholding Proprietary Information from Public Disclosure and Affidavit

(LAR-18-019)

(This Enclosure consists of 8 pages, including this cover page)

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CAW-18-4753

June 12, 2018

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Comprehensive Vibration Assessment Program (CVAP) Post-Test Reports

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-18-4753 signed by the owner of the proprietary information, Westinghouse. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Southern Nuclear Operating Company.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-18-4753, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2 Suite 259, Cranberry Township, Pennsylvania 16066.

Jill S Monahan

Jill S. Monahan, Manager
Licensing Inspections and Special Programs

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, Jill S. Monahan, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse") and declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

Executed on: 6-12-2018

Jill S. Monahan

Jill S. Monahan, Manager
Licensing Inspections and Special Programs

(1) I am Manager, Licensing Inspection and Special Programs, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in APP-GW-GLR-179 Revision 0, "Comprehensive Vibration Assessment Program (CVAP) Preliminary Report for the Sanmen 1 AP1000 Plant (Proprietary)" and APP-GW-GLR-181 Revision 0, "Comprehensive Vibration Assessment Program (CVAP) Final Report for the Sanmen 1 AP1000 Plant (Proprietary)," for submittal to the Commission, being transmitted by Southern Nuclear Operating Company letter ND-18-0826. The proprietary information as submitted by Westinghouse is that associated with the First Plant Only license amendment request, Southern LAR number LAR-18-019, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to manufacture and deliver products to utilities based on proprietary designs.

(b) Further, this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of licensing new nuclear power stations.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of a document, furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.390 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

Southern Nuclear Operating Company

ND-18-0826

Enclosure 6

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Affidavit from Southern Nuclear Operating Company for Withholding Under 10 CFR 2.390

(LAR-18-019)

(This Enclosure consists of 3 pages, including this cover page)

Affidavit of Michael J. Yox

1. My name is Michael J. Yox. I am the Vogtle Units 3 & 4 Regulatory Affairs Director for Southern Nuclear Operating Company (SNC). I have been delegated the function of reviewing proprietary information sought to be withheld from public disclosure and am authorized to apply for its withholding on behalf of SNC.
2. I am making this affidavit on personal knowledge, in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations, and in conjunction with SNC's filing on dockets 52-025 and 52-026 requesting license amendment LAR-18019. I have personal knowledge of the criteria and procedures used by SNC to designate information as a trade secret, privileged or as confidential commercial or financial information.
3. Based on the reason(s) at 10 CFR 2.390(a)(4), this affidavit seeks to withhold from public disclosure Enclosures 1, 5 and 7 of SNC letter ND-18-0826 for 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).
4. The following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.
   a. The information sought to be withheld from public disclosure has been held in confidence by SNC, Sanmen Nuclear Power Company Ltd., Shandong Nuclear Power Company Ltd., and/or Westinghouse Electric Company.
   b. The information is of a type customarily held in confidence by SNC, Sanmen Nuclear Power Company Ltd., Shandong Nuclear Power Company Ltd., and/or Westinghouse Electric Company and not customarily disclosed to the public.

c. The release of the information might result in the loss of an existing or potential competitive advantage to SNC, Sanmen Nuclear Power Company Ltd., Shandong Nuclear Power Company Ltd., and/or Westinghouse Electric Company.

d. Other reasons identified in Enclosures 5, 11, and 12 of SNC letter ND-18-0826 for 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). Those reasons are incorporated here by reference.

5. Additionally, release of the information may harm SNC because SNC has a contractual relationship with Westinghouse Electric Company regarding proprietary information. SNC is contractually obligated to seek confidential and proprietary treatment of the information. Further, SNC is obligated to seek confidential treatment of Sanmen and Haiyang information.

6. The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

7. To the best of my knowledge and belief, the information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method.

I declare under penalty of perjury that the foregoing is true and correct.

Michael J. Yox — Executed on 8/3/2018
Date

Southern Nuclear Operating Company

ND-18-0826

Enclosure 8

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

APP-GW-GLR-180, Revision 0, Comprehensive Vibration Assessment Program (CVAP) Preliminary Report for the Sanmen 1 AP1000 Plant (Non-Proprietary)

(LAR-18-019)

(This Enclosure consists of 94 pages, including this cover page)

DOCUMENT REVIEW SHEET

DOCUMENT NO. SM1-CVAP-T2R-200 REVISION 0

ALTERNATE DOC. NO. N/A March 2017

ORGANIZATION Westinghouse Electric Company LLC – WNS

TITLE Comprehensive Vibration Assessment Program (CVAP) Preliminary Report for the Sanmen 1 AP1000 Plant

WORK BREAKDOWN STRUCTURE NUMBER: CVAP

WESTINGHOUSE PROPRIETARY CLASS: ☐ CLASS 1 ☒ CLASS 2 ☐ CLASS 3

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3. David C. DiBasilio	Section 3.6.1.3	**Electronically Approved*****
4. Adam P. Walker	Sections 3.6.2.1, 3.6.2.2	**Electronically Approved*****
5. Gregory A. Banyay	Sections 3.6.2.3, 3.6.3, App. C	**Electronically Approved*****
6. Richard A. Basel	Section 3.5.6, Section 4	**Electronically Approved*****
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*****Electronically approved records are authenticated in the electronic document management system.***

*** This record was final approved on 6/7/2018 9:46:30 AM. (This statement was added by the PRIME system upon its validation)

Comprehensive Vibration Assessment Program (CVAP) Preliminary Report for the Sanmen 1 AP1000 Plant

RECORD OF REVISIONS

Rev.	Date	Revision Description
0	See EDMS	Original Issue

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LIST OF TABLES

LIST OF FIGURES

ACRONYMS AND TRADEMARKS

CB	Core Barrel
[	] a,c
CS	Core Shroud
CSS	Core Support Structures
CVAP	Comprehensive Vibration Assessment Program
DAQ	Data Acquisition System
DCD	Design Control Document
ECNC	Eddy Current Non-Contact
FFT	Fast Fourier Transform
FIV	Flow-Induced Vibration
HFT	Hot Functional Test
IGA	Instrumentation Grid Assembly
IITA	Incore Instrument Thimble Assembly
LCSP	Lower Core Support Plate
LGT	Lower Guide Tube
LRS	Lower Radial Support
NRC	Nuclear Regulatory Commission
PSD	Power Spectral Density
QIN	Quickloc Instrument Nozzle
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RMS	Root Mean Square
RVCH	Reactor Vessel Closure Head
RVI	Reactor Vessel Internals
SCSS	Secondary Core Support Structure
UGT	Upper Guide Tube
U.S.	United States
USA	Upper Support Assembly
USC	Upper Support Column
USP	Upper Support Plate
VSP	Vortex Suppression Plate
VADE	Vibration and Diagnosis Expansion
VSE	VADE Satellite Enclosures

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1 BACKGROUND

The United States (U.S.) Nuclear Regulatory Commission (NRC) Regulatory Guide 1.20, Revision 2, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing" (Ref. 1), provides guidance for the comprehensive vibration assessment program (CVAP) for nuclear power plants during preoperational and initial startup testing. The program demonstrates that the reactor vessel internals (RVI) are adequately designed to withstand flow-induced vibration (FIV) forces during normal and anticipated transient plant operating conditions for the design life of the plant.

In accordance with the commitment to Reg. Guide 1.20, Revision 2, in the **AP1000®** Design Control Document (DCD) (Ref. 2), the first constructed AP1000 plant RVI assembly at Sanmen 1 is classified as a Prototype, as defined in Reg. Guide 1.20. The CVAP for a Prototype RVI configuration includes the following elements:

- **Vibration Analysis Program**

  The analysis program (Ref. 3) consists of a vibration analysis for steady-state and anticipated transient conditions corresponding to preoperational and initial startup test and normal operating conditions. This includes creating structural and hydraulic models, determining natural frequencies and associated mode shapes, and estimating random and deterministic forcing functions. The analysis program also calculates expected and acceptable responses for selected vibration measurement program sensor locations and develops acceptance criteria for the vibration measurement program, including permissible deviations.

- **Vibration Measurement Program**

  The in-plant vibration measurement program (Ref. 4) verifies the structural integrity of the RVI for FIV, determines the margin of safety associated with steady-state and anticipated transient conditions for normal operation, and verifies the results of the analysis program. Margin of safety in a particular component is established by comparing the limiting measured response in the component to the maximum allowable response at the measurement location. Appropriate transducers are placed throughout the RVI to monitor significant lateral, vertical, and torsional motions of major RVI components in all significant modes of vibration, and their hydraulic responses. These transducer data are recorded for all steady-state and bounding anticipated transient modes of operation (flow transients), including expected reactor coolant pump (RCP) speeds and combinations permissible during the hot functional test (HFT). The selected test duration with plant operation at normal operating modes ensures that each critical component experiences at least 10^6 cycles of vibration (computed at the lowest frequency for which the component has a significant structural response) prior to the final inspection.

- **Inspection Program**

  The inspection program (Ref. 4) consists of pre- and post-HFT inspections of the RVI. The inspection program includes a tabulation of all RVI components and local inspected areas, and a description of the inspection procedure including inspection method, documentation, access provisions, and any specialized equipment used during inspection.

- **Documentation of Results**

  A review and correlation of the results of the analysis, vibration measurement, and inspection program are conducted following completion of the inspection program to determine if the acceptance criteria are satisfied. This evaluation of results and a description of any modifications or actions necessary to demonstrate the structural adequacy of the RVI are documented in preliminary (the present document) and final reports as specified in Reg. Guide 1.20 (Ref. 1).

The CVAP for the AP1000 plant is provided in two documents. The Vibration Analysis Program (Ref. 3) includes a description of the AP1000 plant RVI, vibration analysis methodology, response predictions for the RVI components, and acceptance criteria for applicable sensor locations. The Measurement and Inspection Programs for the AP1000 plant CVAP are documented in the CVAP Measurement and Inspection Program Report (Ref. 4).

The Vibration Analysis Program and Measurement and Inspection Program satisfy the requirements of Reg. Guide 1.20, as outlined in Table 2-2 of Ref. 3 and Table 2-1 of Ref. 4.

This Preliminary Report contains the evaluation of the Sanmen 1 CVAP Vibration Measurement and Inspection Program results with respect to the test acceptance criteria. Anomalous data that could affect the structural integrity of the RVI are identified and evaluated on a preliminary basis, including instrumentation failures that have occurred during the HFT. This Preliminary Report satisfies the requirements of Reg. Guide 1.20, Section C.2.4.1.

The Final Report will provide a detailed comparison of analytical predictions, test measurements, and inspections. Descriptions of any significant deviations, comparisons between measured and analytical responses, determination of high-cycle fatigue margins for component responses, and evaluation of unanticipated observations will be included. The Final Report satisfies the requirements of Reg. Guide 1.20, Section C.2.4.2.

2 SUMMARY OF RESULTS

The purpose of the CVAP is to verify the structural integrity of the 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.

The RVI assembly is depicted in Figure 2-1. Components instrumented for vibration measurement during the HFT include [

] a,c The instrumentation is described in detail in Section 3.2.

The Sanmen 1 HFT began in August 2016 and was completed in December 2016. Testing was performed at numerous steady-state and transient operating conditions during HFT, including the limiting conditions for the RVI shown in Table 3-2. The cumulative duration of testing at normal operating conditions was greater than 15 days, exposing the RVI to normal operating modes for greater than [ ] a,c cycles of vibration (see Section 3.4). This exceeds the Reg. Guide 1.20 (Ref. 1) requirement for a minimum of 10^6 cycles of vibration.

During HFT, vibration data were acquired for the RVI components listed in Table 2-1 to compare with acceptance criteria [ ] a,c Transducer signals were conditioned and recorded for post-test reduction and analysis. Responses during testing were monitored online and the signals were evaluated for spectral content. Root mean square (RMS) values of the signals related to structural response were computed and compared to the acceptance criteria based on analysis as described in the Vibration Analysis Program (Ref. 3). The limiting measured response values and the associated acceptance criteria are compared in Table 2-1; these results indicate positive measured margin for high-cycle fatigue in these critical RVI components.

To assess wear or fatigue of the components, the RVI were visually inspected before and after the HFT as specified in the Inspection Program (Ref. 4). The inspections included major load bearing surfaces, contact surfaces, welds, and maximum stress locations identified by analysis. Photographic records of the pre-HFT and post-HFT inspections were made. The inspections are documented in the Owner's Pre-HFT and Post-HFT Inspection Reports (Ref. 9 and Ref. 10). Comparisons of the visual inspection results before and after HFT were performed (see Section 4.2); based on these results, no signs of abnormal wear or contact for the RVI components were found.

The evaluations documented in this report demonstrate the structural integrity of the Sanmen 1 AP1000 reactor vessel internals with respect to flow-induced vibration for the 60-year design life of the plant, in accordance with Reg. Guide 1.20, Revision 2.

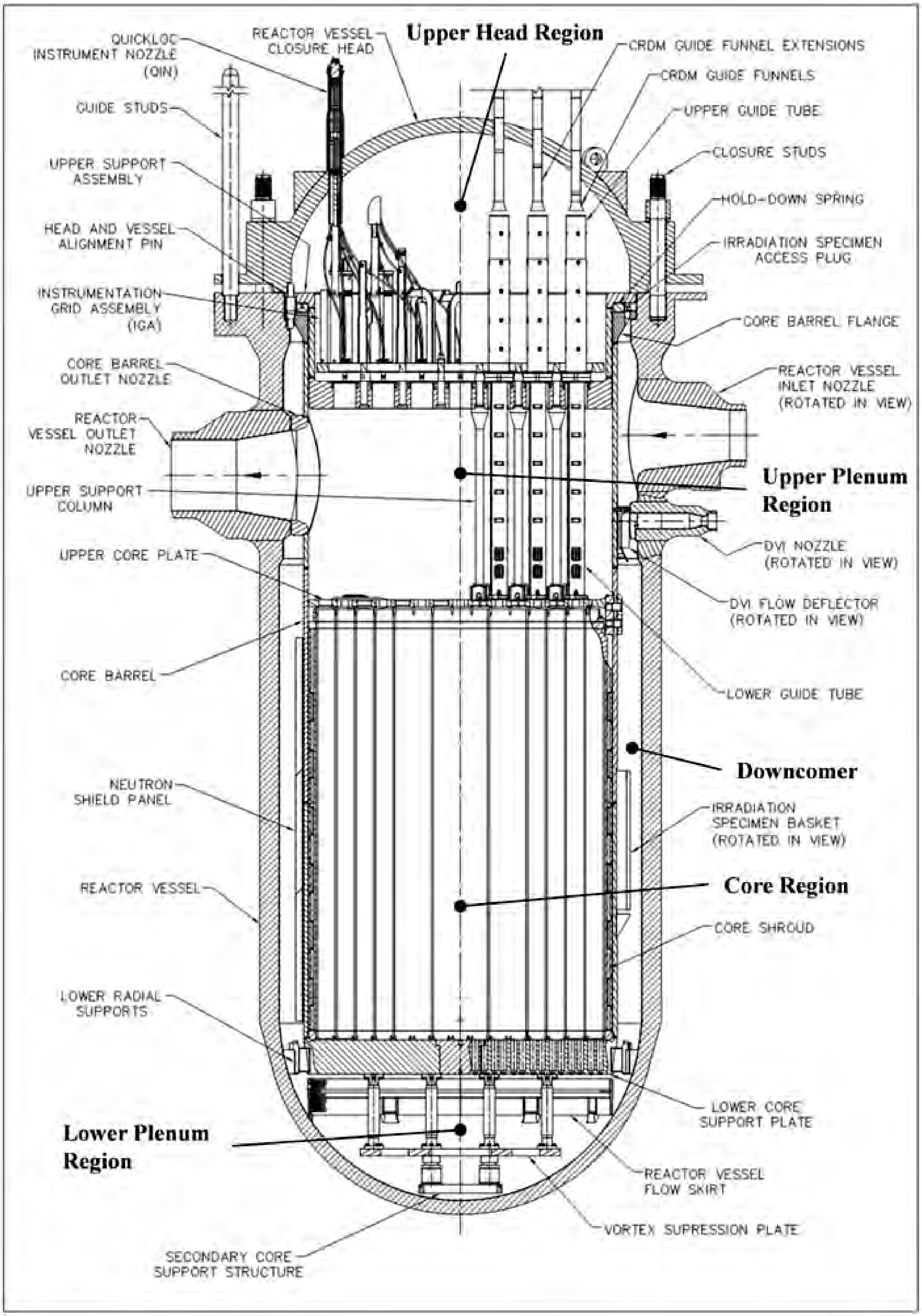

Figure 2-1. AP1000 Reactor Vessel Internals

Table 2-1 Acceptance Criteria and Measured Responses for AP1000 Reactor Vessel Internals

RVI Component(1)	Gage Identifier	Location	Total RMS Measured(2)	Total RMS Measured + Uncertainty(2)	Total RMS Acceptance Criterion(3)	CVAP Measured Margin(4)

a,c

3 MEASUREMENT PROGRAM

3.1 INTRODUCTION

The objective of the measurement program is to obtain sufficient data to confirm predictions of the margin of safety at operating conditions for steady-state and transient normal operation. Hence, the RVI instrumentation includes transducers to measure the structural response.

3.2 INSTRUMENTATION

The CVAP instrumentation mounted on the RVI is described in detail in the Vibration Measurement Program (Ref. 4) and summarized in Table 3-1. This table summarizes the types and locations of the CVAP sensors, and includes comments recorded for these sensors during HFT. A total of [ ] [a,c] were monitored during testing. The sensor locations are illustrated in Appendix A.

3.2.1 Disposition of Failed Sensors

a,c

[ ]

Table 3-1 Transducer Locations for AP1000 CVAP Vibration Measurement Test					
Instrumented Component	**ID**	**Transducer Type**	**Approximate Transducer Location[(1)]**	**Direction of Sensitivity**	**Comments / Status**

a,c

Table 3-1 Transducer Locations for AP1000 CVAP Vibration Measurement Test

Instrumented Component	ID	Transducer Type	Approximate Transducer Location[1]	Direction of Sensitivity	Comments / Status

a,c

Table 3-1 Transducer Locations for AP1000 CVAP Vibration Measurement Test

Instrumented Component	ID	Transducer Type	Approximate Transducer Location(1)	Direction of Sensitivity	Comments / Status

a,c

Table 3-1 Transducer Locations for AP1000 CVAP Vibration Measurement Test

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a,c

Table 3-1 Transducer Locations for AP1000 CVAP Vibration Measurement Test					
Instrumented Component	**ID**	**Transducer Type**	**Approximate Transducer Location(1)**	**Direction of Sensitivity**	**Comments / Status**

a,c

3.3 DATA ACQUISITION

3.3.1 Data Acquisition System

The CVAP data acquisition is performed using the Westinghouse Vibration and Diagnosis Expansion (VADE) system (Figure 3-1), which is described in detail in the Vibration Measurement Program (Ref. 4).

a,c

During the HFT, the DAQ performed as expected, i.e., there were no issues that affected the collection or quality of the measurement data.

a,c

3.3.2 Data Acquisition Method

The DAQ records electrical signals from the transducers mounted on the reactor vessel internals. These recorded time histories are post-processed and analyzed using Fast Fourier Transform (FFT) techniques. [ ] a,c Broadband RMS levels of selected channels are determined to establish that the response of the reactor vessel internals are within the acceptable design limits.

3.3.2.1 Signal Conditioning, Calibration, and Baseline Measurements

A detailed description of the signal conditioning equipment is given in the Vibration Measurement Program (Ref. 4). All sensors and signal conditioning equipment were calibrated before installation. Each transducer and its associated cables are uniquely identified. Confirmation of correct channel location and identification was performed during the installation of the instrumentation.

After installation and field wiring of the transducers, field system verification checks were performed for each channel prior to initial data acquisition. Thereafter, baseline data are collected to determine the channel noise level.

3.3.2.2 Data Recording

Signals from [ ] a,c are recorded and monitored during testing. For transient testing [ ] a,c the transient event. Plant parameters are recorded in the control room during each test condition, [ ] a,c These parameters are used to confirm the applicable test conditions for each CVAP Test Point.

3.3.2.3 Data Monitoring

The DAQ is configured to monitor the incoming signals for "over range" and "no signal" conditions before and during data recording. [ ] a,c

The measurements are compared to the established acceptance criteria (Table 2-1) for each location to ensure that the structural response is within the acceptable limits. The acceptance criteria are based on the allowable vibratory stress limits for the instrumented components and the installed location of the transducers.

a,c

[ ] a,c

3.4 TEST CONDITIONS

Vibration data required for the CVAP were obtained during HFT at Sanmen 1, which began in August 2016 and completed in December 2016. Testing was performed at numerous steady-state and transient operating conditions during HFT (Ref. 12), including the baseline condition and limiting test conditions for the RVI given in Table 3-2 (Ref. 4). The cumulative duration of testing at normal operating conditions was greater than 15 days (Ref. 12). Per Section 5.3 of Ref. 4, a duration of [ ] a,c generates over 10^6 cycles of vibration based on the lowest significant structural frequency of the RVI components [ ] a,c, as required by Reg. Guide 1.20 (Ref. 1). Therefore, the actual duration of 15+ days exposed the RVI to normal operating modes for a minimum of [ ] a,c cycles of vibration, which exceeds the Reg. Guide 1.20 requirement.

Table 3-2 Baseline and Limiting Test Conditions for AP1000 CVAP						
CVAP Test Point(1)(2)	Description of Test Point	Inlet Coolant Temp. (°F)	Number of Operating Pumps	Pump Speed (rpm)	Best Estimate RCS Flow Rate (gpm)	Comments/Basis

a,c

3.5 DATA REDUCTION AND ANALYSIS

During the HFT, data signal time histories are monitored on-line for signal amplitude, electrical noise level, and frequency content. Broadband RMS levels of selected channels are compared to previously defined acceptance criteria to demonstrate the responses of the RVI are acceptable.

3.5.1 General Discussion

[ ] a,c Analyses are performed for each component to identify the amplitude of the FIV response, the frequency and mode shape of the resonance response peaks in the spectra, [ ] a,c

Broadband measured responses, acceptance criteria, and CVAP margins are provided for components in the Lower Internals Assembly, Upper Internals Assembly, and IGA in Sections 3.6.1, 3.6.2, and 3.6.3, respectively. [ ] a,c The specific frequency range for the RMS calculations is identified for each component. [ ] a,c

3.5.2 Calculation Methods

The following concepts apply to the data reduction as used in the analysis of the CVAP test results reported in Section 3.6.

3.5.2.1 Power Spectral Density, Cross-Spectral Density, and Coherence Functions

Chapter 19 of Ref. 5 describes the power spectral density (PSD), cross-spectral density, and coherence functions as follows. The finite Fourier transform of a time sample of duration T (where $j = \sqrt{-1}$) is defined as:

$$X(f,T) = \int_0^T x(t)e^{-j2\pi ft}\,dt = \int_0^T x(t)\cos(2\pi ft)\,dt - j\int_0^T x(t)\sin(2\pi ft)\,dt \quad \text{(Ref. 5, Eq. 19.3)}$$

The PSD, also referred to as the autospectral density function, is the Fourier transform of the autocorrelation function of a dynamic signal.

$$G_{xx}(f) = \lim_{T\to\infty}\frac{2}{T}E[|X(f,T)|^2]\,, f > 0 \quad \text{(Ref. 5, Eq. 19.13)}$$

where *E[]* denotes the expected value of *[]*, which implies an ensemble average (i.e., arithmetic average). Given two stationary random vibrations, *x(t)* and *y(t)*, the cross-spectral density function is defined as:

$$G_{xy}(f) = \lim_{T\to\infty}\frac{2}{T}E[X(f,T)Y(f,T)]\,, f > 0 \quad \text{(Ref. 5, Eq. 19.17)}$$

Then, the coherence function between two random vibrations *x(t)* and *y(t)* is:

$$\gamma_{xy}{}^2(f) = \left|G_{xy}(f)\right|^2 \Big/ G_{xx}(f)G_{yy}(f)\,, f > 0 \quad \text{(Ref. 5, Eq. 19.20)}$$

Physically, the coherence function is an indication of the degree of linear relationship between two signals. For frequencies where $\gamma_{xy}{}^2(f) = 0$ there is no linear relationship between *x(t)* and *y(t)* (vibrations are uncorrelated), whereas at frequencies where $\gamma_{xy}{}^2(f) = 1$ there is a perfect relationship between *x(t)* and *y(t)* (one vibration can be exactly predicted from the other).

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3.5.2.2 Root Mean Square (RMS) Calculation Method

To determine the narrow band RMS result (equivalent RMS value of the time history result that is filtered to include only a specified frequency band), it is recognized that the mean square value over a specified frequency range is the sum of the autospectral density coefficients for that range (Ref. 5). The equivalent RMS value of the signal over that range is the square root of the mean square value for the range.

3.5.3 Measurement Uncertainty

Appendix B documents the uncertainty applicable to the measured responses, including uncertainty associated with data analysis and post-processing. Uncertainty in the acceptance criteria is addressed with conservative bias as part of the predictive analysis and acceptance criteria development (Ref. 3).

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3.5.4 Measured CVAP Margin

The measured CVAP margins are calculated by comparing the measured RMS response, including uncertainty, to the corresponding RMS acceptance criterion from Appendix C. Thus, margin is defined as:

$$Margin = \frac{Acceptance\ Criterion}{Measurement + Uncertainty} - 1$$

Where:

$Acceptance\ Criterion$	Maximum allowable sensor response (RMS) at the limiting condition (Appendix C)
$Measurement$	Measured sensor response (RMS) at the limiting condition
$Uncertainty$	Measurement uncertainty (Appendix B)

In general, margins are reported for the limiting condition for each component, that is, the condition that produces the smallest measured margin. For components with large measured margins, the component margin may be conservatively calculated using the highest overall measured response and the smallest overall acceptance criterion.

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3.5.6 Contact at Structural Interfaces

There are three guidance features in the RVI where small structural gaps that exist during normal operation may come into contact under certain conditions:

- Between the lower radial support (LRS) keys and clevis inserts
- Between the core barrel and reactor vessel outlet nozzles
- Between the alignment plates on the core barrel and inserts in the core shroud and in the upper core plate

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3.6 COMPONENT MEASURED RESPONSES

The measured responses are due [

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The component measured responses, acceptance criteria, and calculated CVAP margin are presented in the following sections.

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For the results presented in these sections, the units for strain are micro-strain ($\mu\varepsilon$, 10^{-6} in/in), and the units for displacement are mils (10^{-3} in).

3.6.1 Lower Internals Response

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3.6.2 Upper Internals Response

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3.6.3 Instrumentation Grid Assembly (IGA)

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4 INSPECTION PROGRAM

4.1 INTRODUCTION

The planned pre-HFT and post-HFT inspection methods and locations are provided on inspection drawings (Ref. 6 and Ref. 7, respectively), with inspection activities specified in Ref. 6.g, Ref. 7.g, and Ref. 13. The inspections are visual, checking for cracks, wear, and debris or loose parts. The RVI are removed from the reactor vessel for these inspections. Inspections were performed in accordance with the RVI installation manual (Ref. 8) and CVAP inspection procedure (Ref. 11).

The inspection locations of interest were selected in accordance with guidance given in Ref. 1. The components and areas of inspection include all major load-bearing elements that retain the position of the CSS; lateral, vertical, and torsional restraints; locking and bolting components; contact surfaces; and the reactor vessel interior for debris and loose parts. The inspection locations and methods are described in Ref. 4 and presented in detail in Ref. 6, Ref. 7, and Ref.10. A representative subset of the inspection locations is also shown in Figure 4-1.

The results of the pre-HFT and post-HFT inspections at each location are recorded in the form of inspection notes and photographs, and are documented in the pre-HFT and post-HFT inspection reports (Ref. 9 and Ref. 10). Pertinent observations from these inspection reports are summarized in Table 4-1. The inspections consider the items listed in Ref. 6.g and Ref. 7.g, including:

- Identification of any broken or loose parts detected before or after the HFT.
- Examination of weld surfaces on the CB for abnormal markings or cracks.
- Examination of interface surfaces such as the contact surfaces of the lower radial support keys and clevis inserts, the CB and reactor vessel outlet nozzle contact surfaces, the alignment plate contact surfaces, the CB flange top and bottom surfaces, the upper support assembly flange top and bottom surfaces, and the hold-down spring interfaces with the CB and upper support assembly flanges.

It should be noted that the inspection drawings (Ref. 6) were updated after the pre-HFT inspections to incorporate lessons learned from those inspections, specifically to clarify instructions and level of detail for the inspections (see Ref. 7 and Ref. 13). The "features to be inspected" in Table 4-1 and inspection locations shown in Figure 4-1 are based on the updated inspection drawings (Ref. 7).

Table 4-1 Pre- and Post-Hot Functional Test Inspections

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Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
1	Core Shroud Studs, Nuts, Locking Caps, and Dowel Pin Welds	1, 2	7, 11		
2	Hold-Down Spring Interface Surface Condition	6	9		
3	Upper Support Column Screw Locking Devices	1, 2, 13	7, 11		
4	Upper Core Plate Inserts	3, 6	9, 12		

Table 4-1 Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
5	Guide Tube Flange Bolts and Locking Devices	1, 2, 13	7, 11		
6	USP Skirt Longitudinal Weld (Inner and Outer Surface) (*For the pre-HFT inspection, Location 6 duplicated the Location 10 inspections and was omitted as redundant)*	N/A	6, 13		
7	Upper Support Skirt to Upper Support Plate Girth Weld	1	6		
8	Upper Support Skirt to Upper Support Flange Girth Weld	1	6		
9	Guide Tube Welds (*The upper guide tubes were not installed at core locations D-12, E-11 and F-10 during the Hot Functional Test. Special cover plates were installed at these three locations to accommodate the CVAP instrumentation.)*	1, 19	6, 16		

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Table 4-1 Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
10	Upper Core Plate Insert Locking Devices and Welds	1, 2	7, 11		
11	Upper Core Barrel to Core Barrel Flange Girth Weld (Inner and Outer Surface)	1	6		
12	Upper Core Barrel to Mid Barrel Girth Weld (Inner and Outer Surface)	1	6		
13	Lower Core Barrel to Lower Core Support Plate Girth Weld	1	6		
14	Alignment Plate Interface Surfaces	9	9		
15	Core Barrel Outlet Nozzle Interface Surfaces	6	9		

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Table 4-1 Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
16	Neutron Shield Panel Dowel Pin Welds	1	6		
17	Neutron Shield Panel Screw Locking Devices and Welds	1, 2	7, 11		
18	Interface Surfaces at the Spacer Pads Along the Top and Bottom Ends of the Neutron Panels	8	12		
19	Core Shroud C-Panel to W-Panel Welds	1	6		
20	Lower Core Support Plate Fuel Alignment Pin Lockwasher Welds	1	7		
21	Secondary Core Support Assembly to Base Plate Weld	1	6		

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Table 4-1 Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
22	Locking Devices and Contact of the Butt Columns Where Attached to the Lower Core Support Plate and Vortex Suppression Plate and Welds	1, 2, 8	7, 11, 12		
23	Locking Devices and Contact of the Secondary Core Support Columns at the Lower Core Support Plate and at the Vortex Suppression Plate and Welds	1, 2, 8	7, 11, 12		
24	Radial Support Key Welds	1	6		
25	Radial Support Key Interface Surfaces	4	9		

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Table 4-1 Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
26	Head And Vessel Alignment Pins, Contact Surfaces and Lock Bar Welds	1,4	7, 8		
27	Irradiation Specimen Basket Screw Locking Devices, Welds, and Dowel Pins	1, 2	7, 11		
28	Vessel Nozzle Interface Surface Condition	6	9		
29	Vessel Clevis Interface Surfaces, Locking Devices, and Dowel Pin Welds	1, 3, 9	7, 9, 12		

a,c

Table 4-1 Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
30	Reinforcement Pad Welds	1	6		
31	Core Barrel Longitudinal Welds (Inner and Outer) *(For the pre-HFT inspection, this location included the USP skirt longitudinal weld)* *(For the post-HFT inspection, the USP skirt longitudinal weld was deleted from Location 31 and renumbered as Location 6)*	1, 12	6, 13, 15		
32	Direct Vessel Injection (DVI) Deflector Stud Lock Bar Welds	1, 6	7		
33	Core Shroud Top Plate Inserts (*The radial inserts were not installed at any of the four locations during the Hot Functional Test to accommodate CVAP instrumentation fixtures. The CVAP instrumentation fixtures were replaced with the standard radial inserts after the Hot Functional Test)*	6	9, 12		
34	Core Shroud Top Plate Insert Locking Devices (*The radial inserts were not installed at any of the four locations during the Hot Functional Test)*	1, 2	7, 11		

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Table 4-1 Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
35	Alignment Plate Screws and Locking Devices *(The lock bars for the middle elevation screws at 225 and 315 degrees extend beyond the reinforcement pad surface as part of the special CVAP hardware and were replaced with the standard items after the Hot Functional Test)*	1	7		
36	Roto-Lock Inserts and Locking Tab Welds	1, 6	6, 9		
37	Head Cooling Spray Nozzle Weld	1	6		
38	Inside Diameter of the Quickloc Instrument Nozzle (QIN)	5, 9	9		
39	Direct Vessel Injection (DVI) Deflector Stud Tack Welds	1	7		
40	Reactor Vessel Direct Vessel Injection (DVI) Nozzle Interior Face	9	9		

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Table 4-1 Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
41	DVI Deflector Outer Surface	9	9		
42	Upper Support Assembly Flange Top and Bottom Surface	6	9		
43	Core Barrel Flange Top and Bottom Surface	6	9		
44	Reactor Vessel Closure Head Bottom Surface	6	9		
45	Reactor Vessel Support Ledge Top Surface	6	9		

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Table 4-1 Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
46	Head and Vessel Alignment Pin Keyways in Reactor Vessel Closure Head	4	9		
47	Head and Vessel Alignment Pin Keyways in Reactor Vessel	4	8, 9		
48	Core Barrel Outlet Nozzle Weld	1	6		
49	Mid to Lower Core Barrel Girth Weld	1	6		

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Table 4-1 Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
50	Interface Surface of Upper Support Assy Skirt Outer Wall and Core Barrel Inner Wall	6	9		
51	Bottom Of Base Plate and Reactor Vessel	6	9		
52	Upper Core Plate Fuel Alignment Pin Nuts and Lock Welds	1, 13	7		
53	Lower Guide Tube Support Pin Locking Devices	1, 2, 13	11, 15		
54	Lower Core Support Plate Access Plug Cap Screw Lock Bar Welds and Lock Pin Welds	1	7		
55	Interior of Reactor Vessel and Outside of Flow Skirt	11	10		

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Table 4-1 Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
56	Flow Skirt to Reactor Vessel Support Lug Welds	1	6		
57	Inner Surfaces of Flow Skirt	4	8		
58	Top Surface of Flow Skirt (Absence of Contact)	6	9		
59	Flow Skirt Tabs, Slots and Area Around Adjacent Holes	4	8		
60	Vertical Welds Between Flow Skirt and Flange Sections	1	6, 13		
61	Flow Skirt Shell to Flange Welds	1	6		
62	IITA Tubes at Compression Fitting Assemblies	9, 12	15, 17		
63	Welds at Compression Fitting Locations	1, 12	15, 17		
64	IITA Tubes as They Exit the Quickloc Upper Support Flange Assembly	9	17		

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Table 4-1 Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
65	Threaded Structural Fasteners and Locking Cups at All Locations on the IGA Plate	1, 2, 9, 12	11, 17		
66	Locking Nut to Support Pin and Locking Nut to IGA Plate Welds	1	17		
67	Locking Nut to Locating Pin and Locking Nut to IGA Plate Welds	1	17		
68	Interface Surfaces of Locating Pins, Support Pins, and Corresponding Seating Surface and the Locating Holes and Seating Surface on the Upper Support Plate	9	9		
69	Joints Between The IGA Instrument Tube, IGA Split Collar Threaded Structural Fasteners and Locking Cups	1, 2, 9, 13	11, 17		
70	All Welds in the Quickloc Upper & Lower Support Assemblies	1	6		
71	IGA Instrument Tube Outside Surface at the IGA Spring Can Bottom Interface	7, 9, 13	14, 15		

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Table 4-1 Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
72	IGA Spring Can Bottom Surface Interface with the IGA Instrument Tube Sleeve Top Surface	9, 13	9, 15		
73	Top Surface and the Top 2 Inches of the OD of the IGA Instrument Tube Sleeve	9, 13	9, 15		
74	OD Surfaces of the Quickloc Stalk Can and the IGA Quickloc Upper Support Flange Assembly	9	9		
75	IGA Guide Studs at the Interface Location of the Guide Bushings	9	17		
76	IGA IITA Tube Support Welds to IGA Plate	1, 12	6, 15		
77	IGA Guide Stud Attachment Bolts & Locking Devices	1, 2	7, 11		
78	Upper Support Column Nut to Upper Support Column Lock Welds	1, 2, 13	7		

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Table 4-1 Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
79	IGA Guide Bushing Attachment (Bolts, Locking Cups and Welds)	1, 2	11, 17		
80	Welds on Upper Support Column Instrumentation Adaptor	1	17		
81	Quickloc Stalk Alignment Screws and Hold-Down Screws	1, 2	17		
82	Contact Locations Between Full Flow Restrictors and Upper Core Plate and Fuel Alignment Pins	6	9		
83	IGA IITA Tube Support Welds to Quickloc Lower Support Assemblies	1, 12	6, 15		
84	SCSS guide post to energy absorber welds	1	6		
85	SCSS energy absorber to housing welds	1	6		

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Table 4-1 Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)

Notes:
1. See Ref. 6.g for pre-HFT inspection notes.
2. See Ref. 7.g for post-HFT inspection notes.
3. Refer to the indicated page number(s) in Ref. 9.
4. Refer to the indicated page number(s) in Ref. 10.

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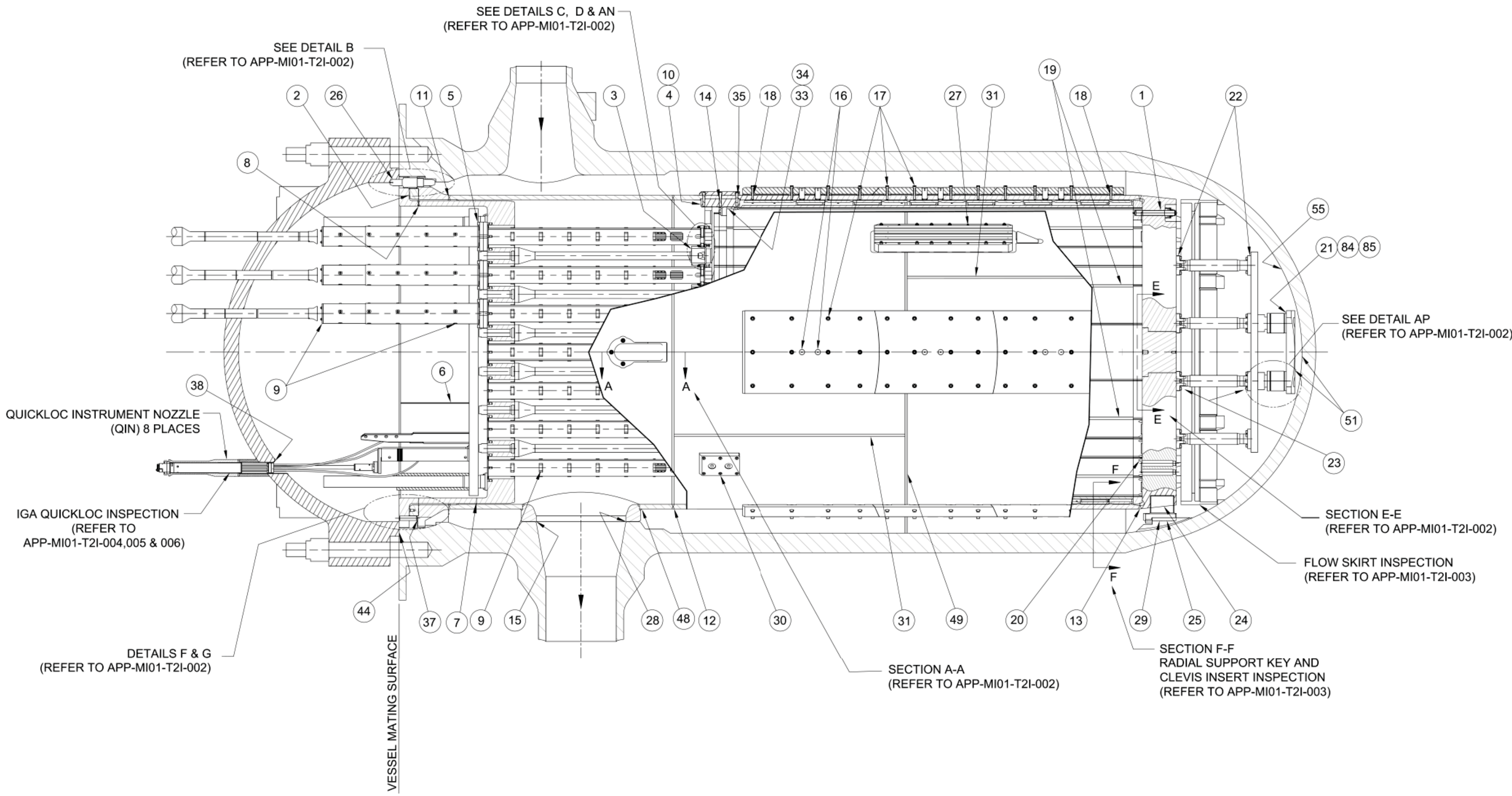

Figure 4-1. Selected Pre-HFT and Post-HFT Inspection Locations

4.2 SUMMARY OF INSPECTIONS

4.2.1 Pre-HFT Inspection Results

The pre-HFT inspections of the reactor vessel internals are complete and were performed in accordance with the Inspection Program (Ref. 4) and inspection drawings (Ref. 6). The pre-HFT inspection results are documented in Ref. 9 and summarized in Table 4-1. As noted in Table 4-1, no major defects were observed and the pre-HFT inspection results are acceptable. The pre-HFT inspections are summarized below:

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4.2.2 Post-HFT Inspection Results

The post-HFT inspections of the reactor vessel internals are complete and were performed in accordance with the Inspection Program (Ref. 4), inspection instructions (Ref. 13), and inspection drawings (Ref. 7). The post-HFT inspection drawings were updated (from those in Ref. 6) and Reference 13 was added based on the experience gained during the pre-HFT inspections, specifically to clarify instructions and level of detail for the inspections. The inspection results are documented in Ref. 10 and summarized in Table 4-1. The post-HFT inspections are summarized below:

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5 SUMMARY AND CONCLUSIONS

The HFT at Sanmen 1 was conducted between August 2016 and December 2016. This period of operation resulted in greater than [ ] a,c cycles of vibration at normal operating conditions accumulated by the core support structures, which exceeds the Reg. Guide 1.20 requirement for a minimum of 10^6 cycles of vibration (see Section 3.4).

Based on the evaluation of the measurement results and comparison of the pre-HFT and post-HFT inspection results described below, it is concluded that the dynamic response of the RVI due to FIV loading is acceptable for long-term operation of the reactor.

5.1 VIBRATION MEASUREMENT RESULTS

An evaluation was conducted of the vibration measurements recorded during HFT [ ] a,c As documented in Section 3.6 and summarized in Table 2-1, the RVI component measured responses are acceptable, i.e., lower than their respective acceptance criteria.

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5.2 INSPECTION RESULTS

Inspections of the RVI before and after HFT were compared to evaluate evidence of unusual contact, wear, or broken or damaged parts. The inspection results were also compared for consistency with the measured vibration results.

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6 REFERENCES

1. U.S. NRC Regulatory Guide 1.20, Rev. 2, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," May 1976.

2. Westinghouse Document, APP-GW-GL-700, Rev. 19, "AP1000 Design Control Document," June 17, 2011.

3. Westinghouse Document, APP-CVAP-GER-003 (WCAP-17984), Rev. 0, "Comprehensive Vibration Assessment Program (CVAP) Vibration Analysis Program for the AP1000 Plant," July 28, 2015 (as modified by APP-CVAP-GEF-024, Rev. 0, "Changes to IGA Strain Gage Locations and Routing").

4. Westinghouse Document, APP-CVAP-GER-004 (WCAP-17983), Rev. 0, "Comprehensive Vibration Assessment Program (CVAP) Measurement and Inspection Programs for the AP1000 Plant," March 26, 2015.

5. A. G. Piersol and T. L. Paez, "Harris Shock and Vibration Handbook," 6th Edition, McGraw Hill, 2009.

6. Westinghouse Inspection Drawings (Westinghouse Proprietary) (as modified by APP-MI01-GEF-371, Rev. 0, "Updates to CVAP Inspections"):

   a. APP-MI01-T2I-001, Rev. 2, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Inspection Data."

   b. APP-MI01-T2I-002, Rev. 1, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Inspection Data."

   c. APP-MI01-T2I-003, Rev. 1, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Flow Skirt Inspection Data."

   d. APP-MI01-T2I-004, Rev. 1, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Instrumentation Grid Assembly Inspection (IGA Lowered Operating Elevation)."

   e. APP-MI01-T2I-005, Rev. 1, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Instrumentation Grid Assembly Inspection (IGA Raised Refueling Elevation)."

   f. APP-MI01-T2I-006, Rev. 1, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Instrumentation Grid Assembly Inspection."

   g. APP-MI01-T2I-007, Rev. 0, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Inspection Data."

7. Westinghouse Inspection Drawings (Westinghouse Proprietary):

   a. APP-MI01-T2I-001, Rev. 3, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Inspection Data."

b. APP-MI01-T2I-002, Rev. 2, “AP1000 Reactor Internals Vibrational Check-Out Functional Test Inspection Data.”

c. APP-MI01-T2I-003, Rev. 2, “AP1000 Reactor Internals Vibrational Check-Out Functional Test Flow Skirt Inspection Data.”

d. APP-MI01-T2I-004, Rev. 2, “AP1000 Reactor Internals Vibrational Check-Out Functional Test Instrumentation Grid Assembly Inspection (IGA Lowered Operating Elevation).”

e. APP-MI01-T2I-005, Rev. 2, “AP1000 Reactor Internals Vibrational Check-Out Functional Test Instrumentation Grid Assembly Inspection (IGA Raised Refueling Elevation).”

f. APP-MI01-T2I-006, Rev. 2, “AP1000 Reactor Internals Vibrational Check-Out Functional Test Instrumentation Grid Assembly Inspection.”

g. APP-MI01-T2I-007, Rev. 1, “AP1000 Reactor Internals Vibrational Check-Out Functional Test Inspection Data.”

8. Westinghouse Document, APP-MI01-Z0M-001, Rev. 4, “AP1000 RVI Installation Requirements Manual,” (Westinghouse Proprietary), December 8, 2016.

9. Westinghouse Document, SM1-CVAP-T2R-100, Rev. 0 (SMRC-VT-RS-03), “Sanmen Unit 1 Pre-Hot Functional Test Visual Inspection Report of the AP1000 Reactor Vessel Internals,” July 13, 2016.

10. Westinghouse Document, SM1-CVAP-T2R-400, Rev. 0 (SMRC-M-00557-SCPP), “Sanmen Unit 1 Post-Hot Functional Test Visual Inspection Report of the AP1000 Reactor Vessel Internals,” March 10, 2017.

11. Westinghouse Document, SM1-RXS-T2R-50006, Rev. A (SM1-RXS-T1P-501), “Pre and Post-Hot Functional Inspection Test of Reactor Vessel Internals Preoperational Test Procedure,” March 1, 2017 (attached to this report in EDMS).

12. Westinghouse Document, SM1-RXS-T2R-50007 Rev. A (SM1-RXS-T1P-502), “Reactor Vessel Vibration Test Preoperational Test Procedure,” March 1, 2017 (attached to this report in EDMS).

13. Westinghouse Document, APP-CVAP-T2I-001, Rev. 0, “AP1000 Comprehensive Vibration Assessment Program (CVAP) Pre and Post Hot Functional Test (HFT) Visual Inspection Instructions,” November 4, 2016.

14. ASME Boiler & Pressure Vessel Code, Section III, 1998 Edition with Addenda through 2000.

15. ASME OM-S/G-2007, “Standards And Guides For Operation And Maintenance Of Nuclear Power Plants.”

16. Westinghouse Document, SM1-MI01-GNR-4165, Rev. 0, “Field Deviation Notice for Failed Lower Internals CVAP sensors,” November 15, 2016.

APPENDIX A TRANSDUCER LOCATIONS

This appendix provides figures to illustrate transducer locations for the Vibration Measurement Program. In these figures, sensors are labeled by instrument ID as shown in Table 3-1. The figures are summarized in Table A-1.

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Table A-1 Illustrations of Transducer Locations

Figure No.	Description

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APPENDIX B RMS UNCERTAINTY ON MEASUREMENT

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APPENDIX C MEASUREMENT ACCEPTANCE CRITERIA

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Southern Nuclear Operating Company

ND-18-0826

Enclosure 10

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

APP-GW-GLR-182, Revision 0, Comprehensive Vibration Assessment Program (CVAP) Final Report for the Sanmen 1 AP1000 Plant (Non-Proprietary)

(LAR-18-019)

(This Enclosure consists of 233 pages, including this cover page)

WESTINGHOUSE PROPRIETARY CLASS 2

DOCUMENT REVIEW SHEET

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ALTERNATE DOC. NO. N/A September 2017

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WORK BREAKDOWN STRUCTURE NUMBER: CVAP

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Comprehensive Vibration Assessment Program (CVAP) Final Report for the Sanmen 1 AP1000 Plant

RECORD OF REVISIONS

Rev.	Date	Revision Description
0	8/24/2017	Original Issue
1	See EDMS	The document was revised for editorial corrections and to provide additional clarifications, so change bars are not used. These changes are administrative per Appendix I of APP-GW-GAP-147, Rev. 7, thus a new licensing impact determination is not required. The previous licensing impact determination remains applicable and is attached in EDMS to SM1-CVAP-T2R-300, Rev. 0.

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ACRONYMS AND TRADEMARKS

ADS	Automatic Depressurization System
ASME	American Society of Mechanical Engineers
CB	Core Barrel
CBVMS	Core Barrel Vibration Monitoring System
[	] a,c
CS	Core Shroud
CSS	Core Support Structures
CVAP	Comprehensive Vibration Assessment Program
DAQ	Data Acquisition System
DCD	Design Control Document
DMIMS	Digital Metal Impact Monitoring System
ECNC	Eddy Current Non-Contact
FEA	Finite Element Analysis
FEM	Finite Element Model
FFR	Full-Flow Restrictors
FFT	Fast Fourier Transform
FIV	Flow-Induced Vibration
HFP	Hot Full Power
HFT	Hot Functional Test
IGA	Instrumentation Grid Assembly
IHP	Integrated Head Package
IITA	Incore Instrument Thimble Assembly
LCSP	Lower Core Support Plate
LGT	Lower Guide Tube
LRS	Lower Radial Support
MDF	Mechanical Design Flow
MMF	Minimum Measured Flow
NRC	Nuclear Regulatory Commission
PSD	Power Spectral Density
PWR	Pressurized Water Reactor
QIN	Quickloc Instrument Nozzle
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RMS	Root Mean Square
RV	Reactor Vessel
RVCH	Reactor Vessel Closure Head
RVI	Reactor Vessel Internals
SCSS	Secondary Core Support Structure
SQOR	Startup Quality Observation Report
SRSS	Square Root Sum of the Squares
UGT	Upper Guide Tube
U.S.	United States
USA	Upper Support Assembly
USC	Upper Support Column

USP	Upper Support Plate
VSE	VADE Satellite Enclosures
VSP	Vortex Suppression Plate
VADE	Vibration and Diagnosis Expansion

1 BACKGROUND

The United States (U.S.) Nuclear Regulatory Commission (NRC) Regulatory Guide 1.20, Revision 2, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing" (Ref. 1), provides guidance for the comprehensive vibration assessment program (CVAP) for nuclear power plants during preoperational and initial startup testing. The program demonstrates that the reactor vessel internals (RVI) are adequately designed to withstand flow-induced vibration (FIV) forces during normal and anticipated transient plant operating conditions for the design life of the plant.

In accordance with the commitment to U.S. Reg. Guide 1.20, Revision 2, in the **AP1000**® Design Control Document (DCD) (Ref. 2), the first constructed AP1000 plant RVI assembly at Sanmen 1 is classified as a Prototype, as defined in Ref. 1. The CVAP for a Prototype RVI configuration includes the following elements:

- **Vibration Analysis Program**

  The analysis program (Ref. 3) consists of a vibration analysis for steady-state and anticipated transient conditions corresponding to preoperational and initial startup test and normal operating conditions. This includes creating structural and hydraulic models, determining natural frequencies and associated mode shapes, and estimating random and deterministic forcing functions. The analysis program also calculates expected and acceptable responses for selected vibration measurement program sensor locations and develops acceptance criteria for the vibration measurement program, including permissible deviations.

- **Vibration Measurement Program**

  The in-plant vibration measurement program (Ref. 4) verifies the structural integrity of the RVI for FIV, determines the margin of safety associated with steady-state and anticipated transient conditions for normal operation, and verifies the results of the analysis program. Margin of safety in a particular component is established by comparing the limiting measured response in the component to the maximum allowable response at the measurement location. Appropriate transducers are placed throughout the RVI to monitor significant lateral, vertical, and torsional motions of major RVI components in all significant modes of vibration, and their hydraulic responses. These transducer data are recorded for all steady-state and bounding anticipated transient modes of operation (flow transients), including expected reactor coolant pump (RCP) speeds and combinations permissible during the hot functional test (HFT). The selected test duration with plant operation at normal operating modes ensures that each critical component experiences at least 10^6 cycles of vibration (computed at the lowest frequency for which the component has a significant structural response) prior to the final inspection.

- **Inspection Program**

  The inspection program (Ref. 4) consists of pre- and post-HFT inspections of the RVI. The inspection program includes a tabulation of all RVI components and local inspected areas, and a description of the inspection procedure including inspection method, documentation, access provisions, and any specialized equipment used during inspection.

- **Documentation of Results**

  A review and correlation of the results of the analysis, vibration measurement, and inspection program are conducted following completion of the inspection program to determine if the acceptance criteria are satisfied. This evaluation of results and a description of any modifications or actions necessary to demonstrate the structural adequacy of the RVI are documented in a preliminary report (Ref. 17) and final report (this document) as specified in U.S. Reg. Guide 1.20 (Ref. 1).

The analysis, measurement, and inspection programs for the CVAP for the AP1000 plant are provided in two documents. The Vibration Analysis Program (Ref. 3) includes a description of the AP1000 plant RVI, vibration analysis methodology, response predictions for the RVI components, and acceptance criteria for applicable sensor locations. The Measurement and Inspection Programs for the AP1000 plant CVAP are documented in the CVAP Measurement and Inspection Program Report (Ref. 4).

The Vibration Analysis Program and Measurement and Inspection Program satisfy the requirements of Ref. 1, as outlined in Table 2-2 of Ref. 3 and Table 2-1 of Ref. 4.

The Preliminary Report (Ref. 17) contains the evaluation of the Sanmen 1 CVAP Vibration Measurement and Inspection Program results with respect to the test acceptance criteria. Anomalous data that could affect the structural integrity of the RVI are identified and evaluated on a preliminary basis, including instrumentation failures that have occurred during the HFT. The Preliminary Report satisfies the requirements of Ref. 1, Section C.2.4.1.

This Final Report provides a detailed comparison of analytical predictions, test measurements, and inspections. Descriptions of any significant deviations, comparisons between measured and analytical responses, determination of high-cycle fatigue margins for component responses, and evaluation of unanticipated observations are included. The Final Report satisfies the requirements of Ref. 1, Section C.2.4.2.

1.1 FRAMEWORK FOR IMPLEMENTING U.S. NRC REGULATORY GUIDE 1.20

Guidance for the CVAP for prototype reactor internals is outlined in Section C.2 of Ref. 1. The framework established for the AP1000 CVAP to implement these regulatory guidelines is presented in Table 1-1. The regulatory guidelines are presented along with the corresponding elements of the AP1000 CVAP.

Table 1-1. Framework for AP1000 RVI CVAP Implementation of U.S. Reg. Guide 1.20

U.S. Reg. Guide 1.20, Part C.2 and C.3		CVAP Report(s)	
Section	**Guidelines**	**Section(s)**	**Program Elements**
CVAP Vibration Analysis Program (Ref. 3)			
2.1	Vibration Analysis Program	5	Description of vibration analysis program
		6, 7	Justification of the CVAP configuration and acceptance criteria
2.1.1	The theoretical structural and hydraulic models and analytical formulations or scaling laws and scale models used in the analysis	5.2, 5.3	Structural models
		5.4	Hydraulic models
		7.1.2	Scaling relationships
2.1.2	The structural and hydraulic system natural frequencies and associated mode shapes which may be excited during steady state and anticipated transient operation	5.3	Structural modes and frequencies
2.1.3	The estimated random and deterministic forcing functions, including any very-low-frequency components, for steady state and anticipated transient operation	5.4	Forcing function development
2.1.4	The calculated structural and hydraulic responses for steady state and anticipated transient operation	5.5	Predictions are provided of RVI component structural responses and subsequent limiting locations relative to normal operating and test related plant operating conditions
2.1.5	A comparison of the calculated structural and hydraulic responses for preoperational and initial startup testing with those for normal operation	6	General analysis methodologies are described, including the approach for extrapolating preoperational test results to normal operating conditions
2.1.6	The anticipated structural or hydraulic vibratory response (defined in terms of frequency, amplitude, and modal contributions) that is appropriate to each sensor location for steady-state and anticipated transient pre-operational and startup conditions	5.4	Component structural evaluations provide predictions of anticipated structural responses at CVAP sensor locations during CVAP testing
2.1.7	The test acceptance criteria with permissible deviations and the basis for the criteria	7, 8	Acceptance criteria include consideration of predictive analysis and measurement uncertainties

Table 1-1. Framework for AP1000 RVI CVAP Implementation of U.S. Reg. Guide 1.20

U.S. Reg. Guide 1.20, Part C.2 and C.3		CVAP Report(s)	
Section	**Guidelines**	**Section(s)**	**Program Elements**
CVAP Measurement Program (Ref. 4)			
2.2	Vibration Measurement Program	4	Description of the Measurement Program
2.2.1	Description of data acquisition and reduction system	4.2, 4.3	Transducer types, specifications, frequency/amplitude ranges
		4.1, 4.2	Transducer locations, descriptions
		4.3	Precautions during design, installation
		4.3	Transducer redundancy
		4.3	Transducer testing
		4.4	Data acquisition system and analysis, including frequency and modal content, precautions during data collection, signal conditioning, real-time frequency and time-domain analysis
		4.4	Discussion of data analysis
2.2.2	Test operating conditions	5.1, 5.2, 5.3	CVAP steady-state and transient test conditions, test data collection points, required duration of testing, description and justification of test vs. normal operating conditions and configurations, disposition of fuel assemblies
		3.1, 3.2	Design configuration (normal operating), test configuration
CVAP Inspection Program (Ref. 4)			
2.3	Inspection Program	6	Description of the Inspection Program
2.3.1	Tabulation of RVI components and areas to be inspected	6.1	Detailed tabulation of inspection locations, type of inspections performed, and inspection methods
2.3.2	A tabulation of specific inspection areas that can be used to verify segments of the vibration analysis and measurement program	6.2	Areas to be inspected, basis for inspections
2.3.3	A description of the inspection procedure	6.1, 6.3	Detailed inspection methods noted for each of the tabulated inspection locations
CVAP Preliminary Report (Ref. 17)			
2.4, 2.4.1	The preliminary report should summarize an evaluation of the raw and, as necessary limited processed data and the results of the inspection program with respect to the test acceptance criteria.	3.6	Component measured responses compared to the acceptance criteria
		4.2	Pre-HFT and post-HFT inspection results
		2, 5	Conclusions demonstrating acceptable measurement and inspection results

Table 1-1. Framework for AP1000 RVI CVAP Implementation of U.S. Reg. Guide 1.20			
U.S. Reg. Guide 1.20, Part C.2 and C.3		CVAP Report(s)	
Section	**Guidelines**	**Section(s)**	**Program Elements**
CVAP Final Report (this document)			
2.4, 2.4.2	The final report should include:		a,c
2.4.2.a	A description of any deviations from the specified measurement and inspection programs, including instrumentation reading and inspection anomalies, instrumentation malfunctions, and deviations from the specified operating conditions.		
2.4.2.b	A comparison between the measured and analytically determined modes of structural and hydraulic response (including those parameters from which the input forcing function is determined) for the purpose of establishing the validity of the analytical technique.	4.6	Comparison of predicted vs. measured responses, predicted vs. measured damping
		4.7	Discussion of forcing functions
2.4.2.c	A determination of the margin of safety associated with normal steady-state and anticipated transient operation.	4.6	Detailed evaluation of measured margins
		2, 6	Overall conclusions demonstrating successful analysis program and acceptable measurement and inspection results
2.4.2.d	An evaluation of measurements that exceeded acceptable limits not specified as test acceptance criteria or of observations that were unanticipated and the disposition of such deviations.	N/A	No acceptance limits were exceeded; no observations represented deviations in the measured responses
2.4.3	If necessary, include an evaluation and description of the modifications or actions planned in order to justify the structural adequacy of the reactor internals.	N/A	No modifications to the RVI were necessary to demonstrate the structural adequacy of the RVI
2.4.4	The collection, storage and maintenance of all records relevant to the analysis, measurement and inspection phases of the CVAP should be consistent with U.S. Reg. Guide 1.88.	Ref. 21, Section 5.9.3.2	Records related to the CVAP are stored in accordance with the CVAP Functional Specification (Ref. 21)

2 EXECUTIVE SUMMARY

The purpose of the CVAP is to verify the structural integrity of the RVI for FIV 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.

The RVI assembly is depicted in Figure 2-1. Components instrumented for vibration measurement during the HFT include [ ] a,c The instrumentation is described in detail in Section 4.2.

The evaluations documented in this report demonstrate the structural integrity of the Sanmen 1 AP1000 reactor vessel internals with respect to FIV for the 60-year design life of the plant, in accordance with U.S. Reg. Guide 1.20, Revision 2 (Ref. 1).

2.1 MEASUREMENT AND INSPECTION RESULTS

The Sanmen 1 HFT began in August 2016 and was completed in December 2016. Testing was performed at numerous steady-state and transient operating conditions during HFT, including the limiting conditions for the RVI shown in Table 4-2. The cumulative duration of testing at normal operating conditions was greater than 15 days, exposing the RVI to normal operating modes for greater than [ ] a,c cycles of vibration (see Section 4.4). This exceeds the Ref. 1 requirement for a minimum of 10^6 cycles of vibration.

During HFT, vibration data were acquired for the RVI components listed in Table 4-38 to compare with acceptance criteria [ ] a,c Transducer signals were conditioned and recorded for post-test reduction and analysis. Responses during testing were monitored in real-time and the signals were evaluated for spectral content.

Root mean square (RMS) values of the measured signals related to structural response were computed and compared to the acceptance criteria based on analysis as described in the Vibration Analysis Program (Ref. 3). The limiting measured response values and the associated acceptance criteria are compared in Table 4-38; in all cases, these results indicate positive measured margins for high-cycle fatigue.

To assess wear or fatigue of the components, the RVI were visually inspected before and after the HFT as specified in the Inspection Program (Ref. 4). The inspections included major load bearing surfaces, contact surfaces, welds, and maximum stress locations identified by analysis. Photographic records of the pre-HFT and post-HFT inspections were made. The inspections are documented in the Owner's Pre-HFT and Post-HFT Inspection Reports (Ref. 9 and Ref. 10). Comparisons of the visual inspection results before and after HFT were performed (see Section 5.2); based on these results, no signs of abnormal wear or contact for the RVI components were found.

a,c

[ ]

a,c

2.2 CORRELATION OF THE CVAP

The CVAP has demonstrated good overall agreement between predicted and measured component frequencies, the predicted RMS responses are consistent with or conservatively high compared to the measured responses, and analytical damping values used in the predictions are similar or conservatively low compared to the measured damping values. The results of the inspection program are consistent with the measurement program, i.e., no indications of unexpected vibratory wear or behavior were found.

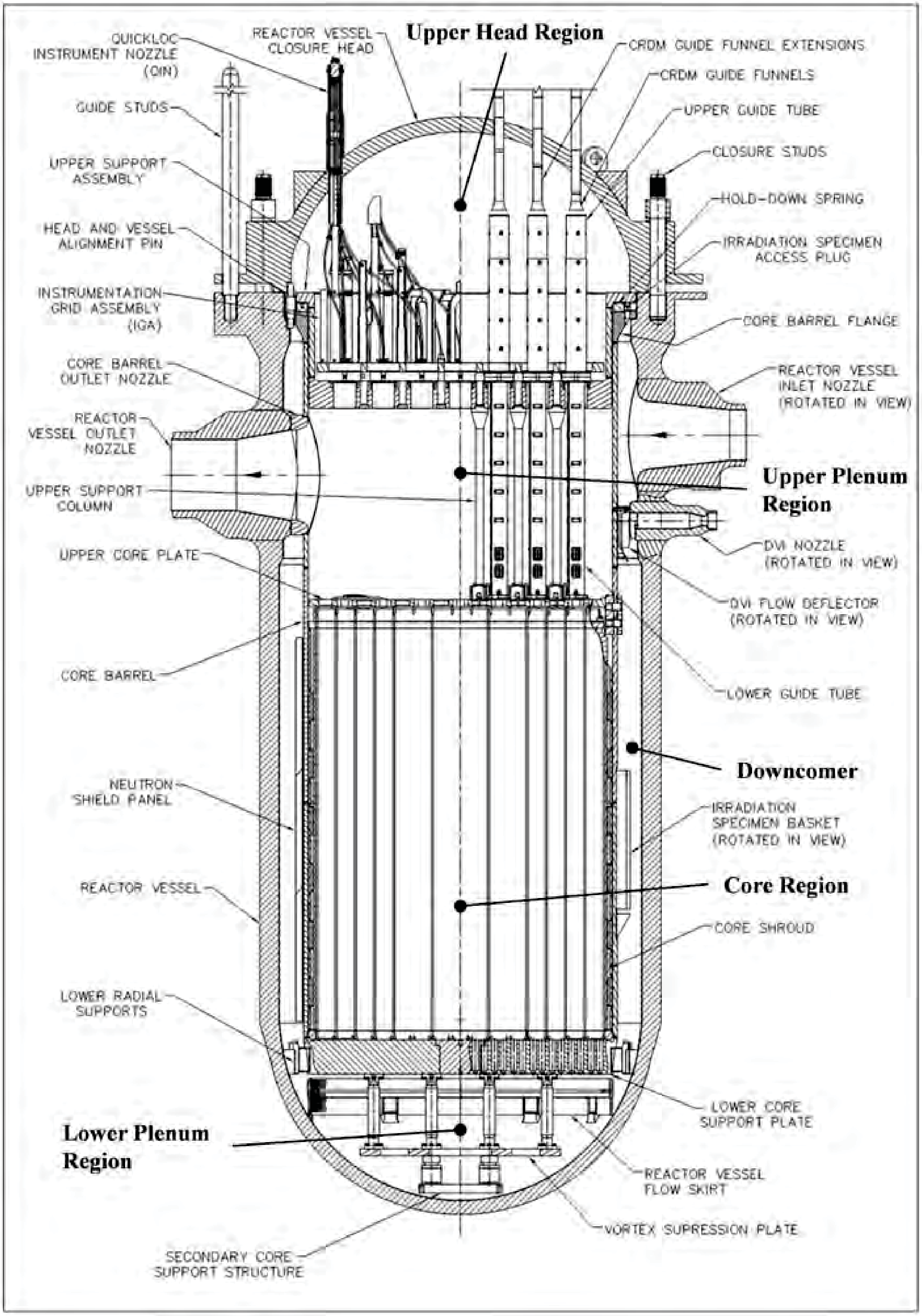

Figure 2-1. AP1000 Reactor Vessel Internals

3 ANALYSIS PROGRAM

3.1 OVERVIEW

The vibration analysis program provides predicted responses and corresponding acceptance criteria (derived from the material endurance limit) for selected RVI components. The vibration analysis program is documented in Ref. 3.

The overall conclusions from the vibration analysis program (Ref. 3) are summarized below.

- The RVI configuration and flow conditions during CVAP are confirmed to be appropriate representations of the RVI configuration and flow conditions during normal operation.
- Significant mode shapes and frequencies for the RVI are identified and confirmed by comparison to plant and scale-model test data.
- Positive margins are predicted for all components, which confirms the structural integrity of the AP1000 RVI for FIV during steady-state and anticipated transient conditions for HFT and normal operation.

The overall methodology for the predictive response analyses is summarized as follows (see Figure 3-1):

a,c

Using the methodology described above the RVI component structural response predictions were performed for the CVAP analyses conditions and are documented in Ref. 3.

a,c

3.2 SUMMARY OF ANALYSES

The following subsections provide an overview of the analysis program.

3.2.1 Structural Models

The dynamic analysis of the RVI is performed using two models – one for the overall RVI assembly and one for the IGA assembly. The mass and stiffness representation of the IGA is contained in the overall RVI model. This allows the overall RVI model to capture the correct dynamic response of the upper support plate and reactor vessel (RV) upper head, which are used as boundary conditions for the IGA analyses.

The overall RVI model is a fluid-structure model of the RV and RVI used for the structural FIV analysis of the RVI and RV system. The model consists of the major components of the system and accounts for fluid coupling and hydrodynamic mass effects. The model confirmation is performed by benchmarking the significant modes of the RVI component to available test data.

The IGA model is a finite element model (FEM) that includes all structural components critical to the structural response of the IGA. A modal test program was conducted to confirm the modal characteristics of the IGA. The modes of the model are benchmarked to the results of this program.

3.2.2 Forced Response Analysis

The RVI components are subjected to several categories of flow and acoustic related loads. The forcing functions can result in different responses from the components. The forcing functions generate loads on the RVI components in one of two ways: (1) random loads such as flow turbulence, or (2) deterministic loads such as RCP acoustic pressures.

[

] a,c The loads developed from these functions are applied to the structural models discussed in Section 3.2.1.

[

] a,c To justify the damping values used in the turbulence analysis, the frequency-dependent analytical damping values were compared to the damping values derived from test data over an equivalent range of frequencies. This comparison confirmed that the applied damping values used in the analysis are conservative.

3.2.3 Predicted Responses for the CVAP Measurement Locations

Due to uncertainty in the hydraulic resistance of the reactor coolant system and in the RCP operating characteristics, the volumetric flow rate could vary between Minimum Measured Flow (MMF) and Mechanical Design Flow (MDF). CVAP predictions were determined at both of these conditions to account for variability in the expected sensor response with respect to the actual RCS flow rate during the CVAP test. This was accomplished by scaling the MDF-based RMS predictions by biased forcing function extrapolation factors to provide MMF-based predictions.

[

] a,c

3.2.4 Measurement Acceptance Criteria

The ASME high-cycle fatigue evaluations consider the maximum stress locations of the RVI components. For practical reasons, CVAP sensors are placed at locations nearby, but not coincident with, the maximum stress locations. Therefore, the measurement acceptance criteria are established to ensure that measurements at the sensor locations are sufficiently low, such that stresses at the limiting locations do not exceed the material endurance limit. The acceptance criteria developed for the CVAP consist of limits on the RMS amplitudes of the test measurements. A transfer function is used to relate the measured RMS response at each sensor to the predicted stress at the limiting location. The acceptance criteria for the CVAP are summarized in Appendix C.

3.2.4.1 Extrapolation

Biased extrapolation factors are used to correlate measured CVAP conditions to limiting operating and test conditions. Random turbulence typically dominates the calculated extrapolation factors, and scales as a function of ratios of fluid density, fluid velocity, and the spectral shape of the forcing function. Exponents on the fluid density and fluid velocity ratios are dependent on the specific forcing function.

4 MEASUREMENT PROGRAM

4.1 INTRODUCTION

The objective of the measurement program is to obtain sufficient data to confirm predictions of the margin of safety at operating conditions for steady-state and transient normal operation. Hence, the RVI instrumentation includes transducers to measure the structural response.

4.2 INSTRUMENTATION

The CVAP instrumentation mounted on the RVI is described in detail in the Vibration Measurement Program (Ref. 4) and summarized in Table 4-1. This table summarizes the types and locations of the CVAP sensors, and includes comments recorded for these sensors during HFT. A total of [ ] a,c were monitored during testing. The sensor locations are illustrated in Appendix A.

4.2.1 Disposition of Failed Sensors

a,c

[ ]

Table 4-1. Transducer Locations for AP1000 CVAP Vibration Measurement Test

Instrumented Component	ID(6)	Transducer Type	Approximate Transducer Location(1)	Direction of Sensitivity	Comments / Status

a,c

Table 4-1. Transducer Locations for AP1000 CVAP Vibration Measurement Test

Instrumented Component	ID(6)	Transducer Type	Approximate Transducer Location(1)	Direction of Sensitivity	Comments / Status

a,c

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a,c

4.3 DATA ACQUISITION

4.3.1 Data Acquisition System

The CVAP data acquisition is performed using the Westinghouse Vibration and Diagnosis Expansion (VADE) system (Figure 4-1), which is described in detail in the Vibration Measurement Program (Ref. 4).

a,c

During the HFT, the DAQ performed as expected, i.e., there were no issues that affected the collection or quality of the measurement data.

a,c

4.3.2 Data Acquisition Method

The DAQ records electrical signals from the transducers mounted on the RVI These recorded time histories are post-processed and analyzed using Fast Fourier Transform (FFT) techniques. [ ] a,c Broadband RMS levels of selected channels were determined to establish that the response of the reactor vessel internals are within the acceptable design limits.

4.3.2.1 Signal Conditioning, Calibration, and Baseline Measurements

A detailed description of the signal conditioning equipment is given in the Vibration Measurement Program (Ref. 4). All sensors and signal conditioning equipment were calibrated before installation. Each transducer and its associated cables were uniquely identified. Confirmation of correct channel location and identification was performed during installation of the instrumentation.

After installation and field wiring of the transducers, field system verification checks were performed for each channel prior to initial data acquisition. Thereafter, baseline data were collected to determine the channel noise level.

4.3.2.2 Data Recording

Signals from [ ] a,c were recorded and monitored during testing. For transient testing [ ] a,c the transient event. Plant parameters were recorded in the control room during each test condition, [ ] a,c These parameters were used to confirm the applicable test conditions for each CVAP Test Point.

4.3.2.3 Data Monitoring

The DAQ was configured to monitor the incoming signals for “over range” and “no signal” conditions before and during data recording. [ ] a,c Measurements were compared to the established acceptance criteria for each location to confirm the structural response was within the acceptable limits. The acceptance criteria are based on the allowable vibratory stress limits for the instrumented components and the installed location of the transducers.

4.4 TEST CONDITIONS

Vibration data required for the CVAP were obtained during HFT at Sanmen 1, which began in August 2016 and completed in December 2016. Testing was performed at numerous steady-state and transient operating conditions during HFT (Ref. 12) given in Table 4-2. Limiting test conditions were identified by analysis in Ref. 4. The cumulative duration of testing at normal operating conditions was greater than 15 days (Ref. 12). Per Section 5.3 of Ref. 4, a duration of [ ] a,c generates over 10^6 cycles of vibration based on the lowest significant structural frequency of the RVI components [ ] a,c as required by U.S. Reg. Guide 1.20 (Ref. 1). Therefore, the actual duration of 15+ days exposed the RVI to normal operating modes for a minimum of [ ] a,c cycles of vibration, which exceeds the minimum regulatory requirement.

a,c

[ ]

Table 4-2. Baseline and Limiting Test Conditions for AP1000 CVAP (Ref. 12)							
CVAP Test Point[(1)]	Description of Test Point	Inlet Coolant Temp. (°F)	Number of Operating Pumps	Pump Speed (rpm)	Best Estimate RCS Flow Rate (gpm)	Test Run Number(s)	Comments/ Basis

a,c

Table 4-2. Baseline and Limiting Test Conditions for AP1000 CVAP (Ref. 12)							
CVAP Test Point(1)	Description of Test Point	Inlet Coolant Temp. (°F)	Number of Operating Pumps	Pump Speed (rpm)	Best Estimate RCS Flow Rate (gpm)	Test Run Number(s)	Comments/ Basis

a,c

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a,c

4.5 DATA REDUCTION AND ANALYSIS

During the HFT, data signal time histories were monitored in real-time for signal amplitude, electrical noise level, and frequency content. Broadband RMS levels of selected channels were compared to previously defined acceptance criteria to demonstrate acceptability of the RVI measured responses.

4.5.1 General Discussion

[ ] a,c Analyses were performed for each component to identify the amplitude of the FIV response, the frequency and mode shape of the resonance response peaks in the spectra, [

] a,c

Broadband measured responses, acceptance criteria, and CVAP margins are provided for components in the Lower Internals Assembly, Upper Internals Assembly, and IGA in Sections 4.6.1, 4.6.2, and 4.6.3, respectively. [
] a,c The specific frequency range for the RMS calculations is identified for each component. [

] a,c

4.5.2 Calculation Methods

The following concepts apply to the data reduction as used in the analysis of the CVAP test results reported in Section 4.6.

4.5.2.1 Power Spectral Density, Cross-Spectral Density, and Coherence Functions

Chapter 19 of Ref. 5 describes the PSD, cross-spectral density, and coherence functions as follows. The finite Fourier transform of a time sample of duration T (where $j = \sqrt{-1}$) is defined as:

$$X(f,T) = \int_0^T x(t)e^{-j2\pi ft}\,dt = \int_0^T x(t)\cos(2\pi ft)\,dt - j\int_0^T x(t)\sin(2\pi ft)\,dt \quad \text{(Ref. 5, Eq. 19.3)}$$

The PSD, also referred to as the autospectral density function, is the Fourier transform of the autocorrelation function of a dynamic signal.

$$G_{xx}(f) = \lim_{T\to\infty}\frac{2}{T}E[|X(f,T)|^2]\,, f > 0 \quad \text{(Ref. 5, Eq. 19.13)}$$

where E [ ] denotes the expected value of [ ], which implies an ensemble average (i.e., arithmetic average). Given two stationary random vibrations, *x(t)* and *y(t)*, the cross-spectral density function is defined as:

$$G_{xy}(f) = \lim_{T\to\infty}\frac{2}{T}E[X(f,T)Y(f,T)]\,, f > 0 \quad \text{(Ref. 5, Eq. 19.17)}$$

Then, the coherence function between two random vibrations *x(t)* and *y(t)* is:

$$\gamma_{xy}{}^2(f) = {|G_{xy}(f)|^2}\Big/{G_{xx}(f)G_{yy}(f)},\ f > 0 \quad \text{(Ref. 5, Eq. 19.20)}$$

Physically, the coherence function is an indication of the degree of linear relationship between two signals. For frequencies where $\gamma_{xy}{}^2(f) = 0$ there is no linear relationship between *x(t)* and *y(t)* (vibrations are uncorrelated), whereas at frequencies where $\gamma_{xy}{}^2(f) = 1$ there is a perfect relationship between *x(t)* and *y(t)* (one vibration can be exactly predicted from the other).

a,c

4.5.2.2 Root Mean Square (RMS) Calculation Method

To determine the narrow band RMS result (equivalent RMS value of the time history result that is filtered to include only a specified frequency band), it is recognized that the mean square value over a specified frequency range is the sum of the autospectral density coefficients for that range (Ref. 5). The equivalent RMS value of the signal over that range is the square root of the mean square value for the range.

a,c

a,c

4.5.2.5 Spectra Plotting

The evaluations in Section 4 and the Appendices make use of various types of spectra plots. Figure 4-3 shows the legend nomenclature used for these plots throughout the document.

a,c

4.5.3 Measurement Uncertainty

Appendix B documents the uncertainty applicable to the measured responses, including uncertainty associated with data analysis and post-processing. Uncertainty in the acceptance criteria is addressed with conservative bias as part of the predictive analysis and acceptance criteria development (Ref. 3).

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4.5.4 Measured CVAP Margin

The measured CVAP margins are calculated by comparing the measured RMS response, including uncertainty, to the corresponding RMS acceptance criterion from Appendix C. Thus, margin is defined as:

$$Margin = \frac{Acceptance\ Criterion}{Measurement + Uncertainty} - 1$$

Where:

Acceptance Criterion	Maximum allowable sensor response (RMS) at the limiting condition (Appendix C)
Measurement	Measured sensor response (RMS) at the limiting condition
Uncertainty	Measurement uncertainty (Appendix B)

Margins are reported for the limiting condition for each component, that is, the condition that produces the smallest measured margin. For components with large measured margins, the component margin may be conservatively calculated using the highest overall measured response and the smallest overall acceptance criterion.

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4.5.6 Contact at Structural Interfaces

There are three guidance features in the RVI where small structural gaps that exist during normal operation may come into contact under certain conditions (see Figure 2-1):

- Between the lower radial support (LRS) keys and clevis inserts
- Between the CB and RV outlet nozzles
- At the alignment plates between the CB and CS and between the CB and UCP

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4.6 COMPONENT MEASURED RESPONSES

The measured responses are [

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For each component, predicted and measured responses, acceptance criteria, and calculated CVAP margins are presented in the subsections shown below. For information, a summary of the reactor vessel motion is provided in Appendix D.

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For the results presented in these sections, the units for strain are microstrain (με, 10^{-6} in/in), and the units for displacement are mils (10^{-3} in).

4.6.1 Lower Internals

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4.6.2 Upper Internals

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4.6.3 Instrumentation Grid Assembly (IGA)

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Table 4-38. Summary of RVI Measurements, Acceptance Criteria, and CVAP Margins

RVI Component(1)	Gage Identifier	Location	Total RMS Measured(2)	Total RMS Measured + Uncertainty(2)	Total RMS Acceptance Criterion(3)	CVAP Measured Margin(4)

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4.7 FORCING FUNCTIONS

As discussed in Section 3.2.2, the RVI components are forced by either random or deterministic loadings. The following subsections discuss the two different loading mechanisms applicable to the RVI.

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5 INSPECTION PROGRAM

5.1 BACKGROUND

The planned pre-HFT and post-HFT inspection methods and locations are provided on inspection drawings (Ref. 6 and Ref. 7, respectively), with inspection activities specified in Ref. 6.g, Ref. 7.g, and Ref. 13. The inspections are visual, checking for cracks, wear, and debris or loose parts. The RVI are removed from the reactor vessel for these inspections. Inspections were performed in accordance with the RVI installation manual (Ref. 8) and CVAP inspection procedure (Ref. 11).

The inspection locations of interest were selected in accordance with guidance given in Ref. 1. The components and areas of inspection include all major load-bearing elements that retain the position of the core support structures (CSS); lateral, vertical, and torsional restraints; locking and bolting components; contact surfaces; and the reactor vessel interior for debris and loose parts. The inspection locations and methods are described in Ref. 4 and presented in detail in Ref. 6, Ref. 7, and Ref. 10. A representative subset of the inspection locations is also shown in Figure N-1.

The results of the pre-HFT and post-HFT inspections at each location are recorded in the form of inspection notes and photographs, and are documented in the pre-HFT and post-HFT inspection reports (Ref. 9 and Ref. 10). The inspections consider the items listed in Ref. 6.g and Ref. 7.g, including:

- Identification of any broken or loose parts detected before or after the HFT.
- Examination of weld surfaces on the CB for abnormal markings or cracks.
- Examination of interface surfaces such as the contact surfaces of the lower radial support keys and clevis inserts, the CB and reactor vessel outlet nozzle contact surfaces, the alignment plate contact surfaces, the CB flange top and bottom surfaces, the upper support assembly flange top and bottom surfaces, and the hold-down spring interfaces with the CB and upper support assembly flanges.

5.2 INSPECTION RESULTS

5.2.1 Pre-HFT Inspection Results

The pre-HFT inspections of the reactor vessel internals were performed in accordance with the Inspection Program (Ref. 4) and inspection drawings (Ref. 6). The pre-HFT inspection results are documented in Ref. 9 and summarized in Table N-1. The pre-HFT inspections are summarized below:

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5.2.2 Post-HFT Inspection Results

The post-HFT inspections of the reactor vessel internals were performed in accordance with the Inspection Program (Ref. 4), inspection instructions (Ref. 13), and inspection drawings (Ref. 7). The inspection results are documented in Ref. 10 and summarized in Table N-1. The post-HFT inspections are summarized below:

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6 SUMMARY OF RESULTS AND CONCLUSIONS

The HFT at Sanmen 1 was conducted between August 2016 and December 2016. This period of operation resulted in greater than [ ]^{a,c} cycles of vibration at normal operating conditions accumulated by the core support structures, which exceeds the U.S. Reg. Guide 1.20 requirement for a minimum of 10^6 cycles of vibration (see Section 4.4).

Based on evaluations of the predicted and measured responses, comparison of measured responses to their respective acceptance criteria, and comparison of the pre-HFT and post-HFT inspection results as summarized below, this report demonstrates the structural integrity of the Sanmen 1 AP1000 reactor vessel internals with respect to FIV for the 60-year design life of the plant, in accordance with U.S. Reg. Guide 1.20, Revision 2.

6.1 SUMMARY OF MEASUREMENT RESULTS

An evaluation was conducted of the vibration measurements recorded during HFT [

]^{a,c} As documented in Section 4.6 and summarized in Table 4-38, the RVI component predicted responses are similar to or conservative with respect to the measurements, and the measured responses are below their respective acceptance criteria.

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6.2 SUMMARY OF INSPECTION RESULTS

Inspections of the RVI before and after HFT were compared to evaluate evidence of unusual contact, wear, or broken or damaged parts. The inspection results were also compared for consistency with the measured vibration results.

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6.3 CORRELATION OF PREDICTIONS, MEASUREMENTS, AND INSPECTIONS

The CVAP has demonstrated good overall agreement between predicted and measured component frequencies, the predicted RMS responses are consistent with or conservatively high compared to the measured responses, and analytical damping values used in the predictions are similar or conservatively low compared to the measured damping values. The results of the inspection program are consistent with the measurement program, i.e., no indications of unexpected vibratory wear or behavior were found.

7 REFERENCES

1. U.S. NRC Regulatory Guide 1.20, Rev. 2, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," May 1976.

2. Westinghouse Document, APP-GW-GL-700, Rev. 19, "AP1000 Design Control Document," June 17, 2011.

3. Westinghouse Document, APP-CVAP-GER-003 (WCAP-17984), Rev. 0, "Comprehensive Vibration Assessment Program (CVAP) Vibration Analysis Program for the AP1000 Plant," July 28, 2015 (as modified by APP-CVAP-GEF-024, Rev. 0, "Changes to IGA Strain Gage Locations and Routing").

4. 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.

5. A. G. Piersol and T. L. Paez, "Harris Shock and Vibration Handbook," 6th Edition, McGraw Hill, 2009.

6. Westinghouse Inspection Drawings for Pre-HFT Inspections (Westinghouse Proprietary) (as modified by APP-MI01-GEF-371, Rev. 0, "Updates to CVAP Inspections"):

   a. APP-MI01-T2I-001, Rev. 2, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Inspection Data."

   b. APP-MI01-T2I-002, Rev. 1, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Inspection Data."

   c. APP-MI01-T2I-003, Rev. 1, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Flow Skirt Inspection Data."

   d. APP-MI01-T2I-004, Rev. 1, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Instrumentation Grid Assembly Inspection (IGA Lowered Operating Elevation)."

   e. APP-MI01-T2I-005, Rev. 1, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Instrumentation Grid Assembly Inspection (IGA Raised Refueling Elevation)."

   f. APP-MI01-T2I-006, Rev. 1, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Instrumentation Grid Assembly Inspection."

   g. APP-MI01-T2I-007, Rev. 0, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Inspection Data."

7. Westinghouse Inspection Drawings for Post-HFT Inspections (Westinghouse Proprietary):

   a. APP-MI01-T2I-001, Rev. 3, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Inspection Data."

b. APP-MI01-T2I-002, Rev. 2, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Inspection Data."

c. APP-MI01-T2I-003, Rev. 2, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Flow Skirt Inspection Data."

d. APP-MI01-T2I-004, Rev. 2, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Instrumentation Grid Assembly Inspection (IGA Lowered Operating Elevation)."

e. APP-MI01-T2I-005, Rev. 2, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Instrumentation Grid Assembly Inspection (IGA Raised Refueling Elevation)."

f. APP-MI01-T2I-006, Rev. 2, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Instrumentation Grid Assembly Inspection."

g. APP-MI01-T2I-007, Rev. 1, "AP1000 Reactor Internals Vibrational Check-Out Functional Test Inspection Data."

8. Westinghouse Document, APP-MI01-Z0M-001, Rev. 4, "AP1000 RVI Installation Requirements Manual," (Westinghouse Proprietary), December 8, 2016.

9. Westinghouse Document, SM1-CVAP-T2R-100, Rev. 0 (SMRC-VT-RS-03), "Sanmen Unit 1 Pre-Hot Functional Test Visual Inspection Report of the AP1000 Reactor Vessel Internals," July 13, 2016.

10. Westinghouse Document, SM1-CVAP-T2R-400, Rev. 0 (SMNP-M-00557-SCPP), "Sanmen Unit 1 Post-Hot Functional Test Visual Inspection Report of the AP1000 Reactor Vessel Internals," March 10, 2017.

11. Westinghouse Document, SM1-RXS-T2R-50006, Rev. 0, "Pre and Post-Hot Functional Inspection Test of Reactor Vessel Internals Preoperational Test Procedure," June 14, 2017 (Executed test procedure SM1-RXS-T1P-501, attached to Ref. 20 in EDMS).

12. Westinghouse Document, SM1-RXS-T2R-50007, Rev. 0, "Reactor Vessel Internals Vibration Test Preoperational Test Procedure," June 14, 2017 (Executed test procedure SM1-RXS-T1P-502, attached to Ref. 20 in EDMS).

13. Westinghouse Document, APP-CVAP-T2I-001, Rev. 0, "AP1000 Comprehensive Vibration Assessment Program (CVAP) Pre and Post Hot Functional Test (HFT) Visual Inspection Instructions," November 4, 2016.

14. ASME Boiler & Pressure Vessel Code, Section III, 1998 Edition with Addenda through 2000.

15. ASME OM-S/G-2007, "Standards And Guides For Operation And Maintenance Of Nuclear Power Plants."

16. Westinghouse Document, SM1-MI01-GNR-4165, Rev. 0, "Field Deviation Notice for Failed Lower Internals CVAP sensors," November 15, 2016.

17. Westinghouse Document, SM1-CVAP-T2R-200, Rev. 0, "Comprehensive Vibration Assessment Program (CVAP) Preliminary Report for the Sanmen 1 AP1000 Plant," March 17, 2017 (Transmitted via project letter CPO_CSM_000532).

18. A. Trenty, "Operational Feedback on Internal Structure Vibration in 54 French PWRs During 300 Fuel Cycles," Progress in Nuclear Energy, Vol. 29, No. 3/4 pp. 347 to 356, 1995.

19. Westinghouse Document, SM1-MI01-GNR-4155, Rev. 0, "Field Deviation Notice for Discrepancies Found in Radial Support Key Bearing Surfaces (Location 25)," March 15, 2017.

20. Westinghouse Document, LTR-ARIDA-17-60, Rev. 0 "Sensor Calibrations, Data Acquisition System (DAQ) Connections, Executed Test Procedures, and Test Run Log for the SM1 Comprehensive Vibration Assessment Program (CVAP)," August 24, 2017.

21. Westinghouse Document, APP-CVAP-Z01-001, Rev. 2, "AP1000 Comprehensive Vibration Assessment Program Functional Specification," February 24, 2015.

APPENDIX A TRANSDUCER LOCATIONS

This appendix provides figures to illustrate transducer locations for the Vibration Measurement Program. In these figures, sensors are labeled by instrument ID as shown in Table 4-1. The figures are summarized in Table A-1.

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Table A-1. Illustrations of Transducer Locations

Figure No.	Description

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APPENDIX B RMS MEASUREMENT UNCERTAINTY

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APPENDIX C MEASUREMENT ACCEPTANCE CRITERIA

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APPENDIX D [ ] a,c

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APPENDIX E [ ]a,c

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APPENDIX F [ ]a,c

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APPENDIX N PRE-HFT AND POST-HFT INSPECTION RESULTS

The pre- and post-HFT inspection results are summarized in Table N-1. Note that the inspection drawings (Ref. 6) were updated after the pre-HFT inspections to incorporate lessons learned from those inspections, specifically to clarify instructions and level of detail for the inspections (see Ref. 7 and Ref. 13). The "features to be inspected" in Table N-1 and inspection locations shown in Figure N-1 are based on the updated inspection drawings (Ref. 7).

Table N-1. Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
1	Core Shroud Studs, Nuts, Locking Caps, and Dowel Pin Welds	1, 2	7, 11		
2	Hold-Down Spring Interface Surface Condition	6	9		
3	Upper Support Column Screw Locking Devices	1, 2, 13	7, 11		
4	Upper Core Plate Inserts	3, 6	9, 12		

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Table N-1. Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
5	Guide Tube Flange Bolts and Locking Devices	1, 2, 13	7, 11		
6	USP Skirt Longitudinal Weld (Inner and Outer Surface) (*For the pre-HFT inspection, Location 6 duplicated the Location 10 inspections and was omitted as redundant)*	N/A	6, 13		
7	Upper Support Skirt to Upper Support Plate Girth Weld	1	6		
8	Upper Support Skirt to Upper Support Flange Girth Weld	1	6		
9	Guide Tube Welds (*The upper guide tubes were not installed at core locations D-12, E-11 and F-10 during the Hot Functional Test. Special cover plates were installed at these three locations to accommodate the CVAP instrumentation.)*	1, 19	6, 16		

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Table N-1. Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
10	Upper Core Plate Insert Locking Devices and Welds	1, 2	7, 11		
11	Upper Core Barrel to Core Barrel Flange Girth Weld (Inner and Outer Surface)	1	6		
12	Upper Core Barrel to Mid Barrel Girth Weld (Inner and Outer Surface)	1	6		
13	Lower Core Barrel to Lower Core Support Plate Girth Weld	1	6		
14	Alignment Plate Interface Surfaces	9	9		
15	Core Barrel Outlet Nozzle Interface Surfaces	6	9		

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Table N-1. Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
16	Neutron Shield Panel Dowel Pin Welds	1	6		
17	Neutron Shield Panel Screw Locking Devices and Welds	1, 2	7, 11		
18	Interface Surfaces at the Spacer Pads Along the Top and Bottom Ends of the Neutron Panels	8	12		
19	Core Shroud C-Panel to W-Panel Welds	1	6		
20	Lower Core Support Plate Fuel Alignment Pin Lockwasher Welds	1	7		
21	Secondary Core Support Assembly to Base Plate Weld	1	6		

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Table N-1. Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
22	Locking Devices and Contact of the Butt Columns Where Attached to the Lower Core Support Plate and Vortex Suppression Plate and Welds	1, 2, 8	7, 11, 12		
23	Locking Devices and Contact of the Secondary Core Support Columns at the Lower Core Support Plate and at the Vortex Suppression Plate and Welds	1, 2, 8	7, 11, 12		
24	Radial Support Key Welds	1	6		
25	Radial Support Key Interface Surfaces	4	9		

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Table N-1. Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
26	Head And Vessel Alignment Pins, Contact Surfaces and Lock Bar Welds	1,4	7, 8		
27	Irradiation Specimen Basket Screw Locking Devices, Welds, and Dowel Pins	1, 2	7, 11		
28	Vessel Nozzle Interface Surface Condition	6	9		
29	Vessel Clevis Interface Surfaces, Locking Devices, and Dowel Pin Welds	1, 3, 9	7, 9, 12		

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Table N-1. Pre- and Post-Hot Functional Test Inspections

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Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
30	Reinforcement Pad Welds	1	6		
31	Core Barrel Longitudinal Welds (Inner and Outer) *(For the pre-HFT inspection, this location included the USP skirt longitudinal weld)* *(For the post-HFT inspection, the USP skirt longitudinal weld was deleted from Location 31 and renumbered as Location 6)*	1, 12	6, 13, 15		
32	Direct Vessel Injection (DVI) Deflector Stud Lock Bar Welds	1, 6	7		
33	Core Shroud Top Plate Inserts (*The radial inserts were not installed at any of the four locations during the Hot Functional Test to accommodate CVAP instrumentation fixtures. The CVAP instrumentation fixtures were replaced with the standard radial inserts after the Hot Functional Test)*	6	9, 12		
34	Core Shroud Top Plate Insert Locking Devices (*The radial inserts were not installed at any of the four locations during the Hot Functional Test)*	1, 2	7, 11		

Table N-1. Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
35	Alignment Plate Screws and Locking Devices (*The lock bars for the middle elevation screws at 225 and 315 degrees extend beyond the reinforcement pad surface as part of the special CVAP hardware and were replaced with the standard items after the Hot Functional Test)*	1	7		
36	Roto-Lock Inserts and Locking Tab Welds	1, 6	6, 9		
37	Head Cooling Spray Nozzle Weld	1	6		
38	Inside Diameter of the Quickloc Instrument Nozzle (QIN)	5, 9	9		
39	Direct Vessel Injection (DVI) Deflector Stud Tack Welds	1	7		
40	Reactor Vessel Direct Vessel Injection (DVI) Nozzle Interior Face	9	9		

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Table N-1. Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
41	DVI Deflector Outer Surface	9	9		
42	Upper Support Assembly Flange Top and Bottom Surface	6	9		
43	Core Barrel Flange Top and Bottom Surface	6	9		
44	Reactor Vessel Closure Head Bottom Surface	6	9		
45	Reactor Vessel Support Ledge Top Surface	6	9		

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Table N-1. Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
46	Head and Vessel Alignment Pin Keyways in Reactor Vessel Closure Head	4	9		
47	Head and Vessel Alignment Pin Keyways in Reactor Vessel	4	8, 9		
48	Core Barrel Outlet Nozzle Weld	1	6		
49	Mid to Lower Core Barrel Girth Weld	1	6		

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Table N-1. Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
50	Interface Surface of Upper Support Assy Skirt Outer Wall and Core Barrel Inner Wall	6	9		
51	Bottom Of Base Plate and Reactor Vessel	6	9		
52	Upper Core Plate Fuel Alignment Pin Nuts and Lock Welds	1, 13	7		
53	Lower Guide Tube Support Pin Locking Devices	1, 2, 13	11, 15		
54	Lower Core Support Plate Access Plug Cap Screw Lock Bar Welds and Lock Pin Welds	1	7		
55	Interior of Reactor Vessel and Outside of Flow Skirt	11	10		

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Table N-1. Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
56	Flow Skirt to Reactor Vessel Support Lug Welds	1	6		
57	Inner Surfaces of Flow Skirt	4	8		
58	Top Surface of Flow Skirt (Absence of Contact)	6	9		
59	Flow Skirt Tabs, Slots and Area Around Adjacent Holes	4	8		
60	Vertical Welds Between Flow Skirt and Flange Sections	1	6, 13		
61	Flow Skirt Shell to Flange Welds	1	6		
62	IITA Tubes at Compression Fitting Assemblies	9, 12	15, 17		
63	Welds at Compression Fitting Locations	1, 12	15, 17		
64	IITA Tubes as They Exit the Quickloc Upper Support Flange Assembly	9	17		

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Table N-1. Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
65	Threaded Structural Fasteners and Locking Cups at All Locations on the IGA Plate	1, 2, 9, 12	11, 17		
66	Locking Nut to Support Pin and Locking Nut to IGA Plate Welds	1	17		
67	Locking Nut to Locating Pin and Locking Nut to IGA Plate Welds	1	17		
68	Interface Surfaces of Locating Pins, Support Pins, and Corresponding Seating Surface and the Locating Holes and Seating Surface on the Upper Support Plate	9	9		
69	Joints Between The IGA Instrument Tube, IGA Split Collar Threaded Structural Fasteners and Locking Cups	1, 2, 9, 13	11, 17		
70	All Welds in the Quickloc Upper & Lower Support Assemblies	1	6		
71	IGA Instrument Tube Outside Surface at the IGA Spring Can Bottom Interface	7, 9, 13	14, 15		

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Table N-1. Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
72	IGA Spring Can Bottom Surface Interface with the IGA Instrument Tube Sleeve Top Surface	9, 13	9, 15		
73	Top Surface and the Top 2 Inches of the OD of the IGA Instrument Tube Sleeve	9, 13	9, 15		
74	OD Surfaces of the Quickloc Stalk Can and the IGA Quickloc Upper Support Flange Assembly	9	9		
75	IGA Guide Studs at the Interface Location of the Guide Bushings	9	17		
76	IGA IITA Tube Support Welds to IGA Plate	1, 12	6, 15		
77	IGA Guide Stud Attachment Bolts & Locking Devices	1, 2	7, 11		
78	Upper Support Column Nut to Upper Support Column Lock Welds	1, 2, 13	7		

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Table N-1. Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)
79	IGA Guide Bushing Attachment (Bolts, Locking Cups and Welds)	1, 2	11, 17		
80	Welds on Upper Support Column Instrumentation Adaptor	1	17		
81	Quickloc Stalk Alignment Screws and Hold-Down Screws	1, 2	17		
82	Contact Locations Between Full Flow Restrictors and Upper Core Plate and Fuel Alignment Pins	6	9		
83	IGA IITA Tube Support Welds to Quickloc Lower Support Assemblies	1, 12	6, 15		
84	SCSS guide post to energy absorber welds	1	6		
85	SCSS energy absorber to housing welds	1	6		

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Table N-1. Pre- and Post-Hot Functional Test Inspections

Location	Features To Be Inspected	Pre-HFT Inspection Notes(1)	Post-HFT Inspection Notes(2)	Pre-HFT Comments and Observations(3)	Post-HFT Comments and Observations(4)

Notes:
1. See Ref. 6.g for pre-HFT inspection notes.
2. See Ref. 7.g for post-HFT inspection notes.
3. Refer to the indicated page number(s) in Ref. 9.
4. Refer to the indicated page number(s) in Ref. 10.

[ ] a,c

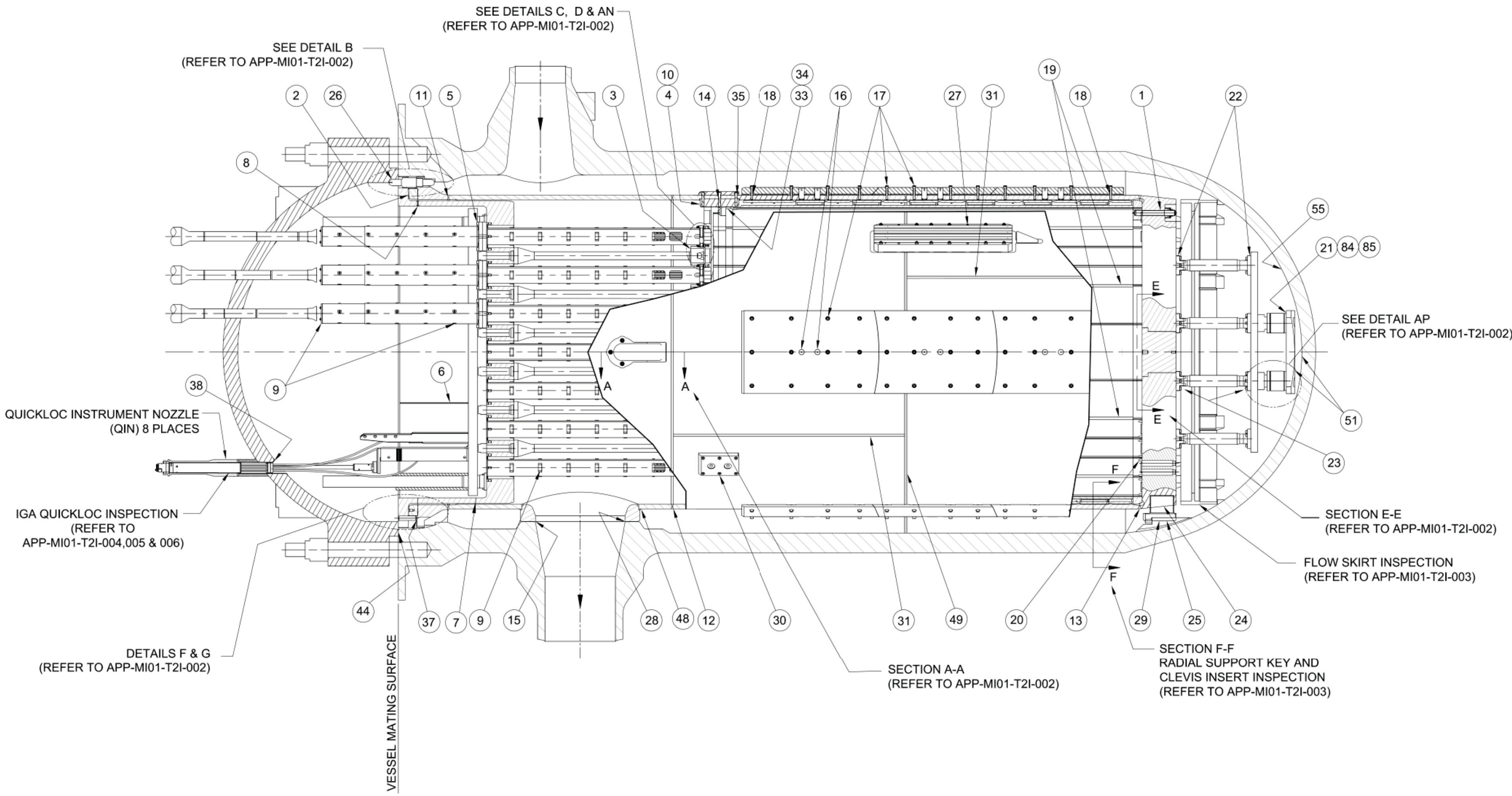

Figure N-1. Selected Pre-HFT and Post-HFT Inspection Locations

Southern Nuclear Operating Company

ND-18-0826

Enclosure 11

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Sanmen Nuclear Power Company Ltd. Affidavit supporting the proprietary nature of the information in enclosure 1

(LAR-18-019)

(This Enclosure consists of 3 pages, including this cover page)

AFFIDAVIT OF Wu Yuanming

I, Wu Yuanming, being duly sworn, hereby depose and say that I am the vice president, Sanmen Nuclear Power Company Ltd.; and that I am duly authorized to sign and file with the Nuclear Regulatory Commission ("NRC") this affidavit on behalf of Sanmen Nuclear Power Company Ltd. and state:

1. The proprietary and commercial information sought to be withheld from public disclosure in connection with Southern Nuclear Operating Company (SNC) letter ND-18-0826 for 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) submittal to the NRC is from Unit 1 and Unit 2 of Sanmen Nuclear Power Company Ltd. and I am authorized to execute this affidavit on behalf of Sanmen Nuclear Power Company Ltd.

2. This information provided is to support of SNC letter ND-18-0826 for 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). The proprietary and commercial information sought to be withheld in this submittal is the Sanmen first plant only and first three plant only test data which is appropriately marked in Enclosure 1, "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)" (Proprietary). This document includes proprietary and commercial information that should be held in confidence by the NRC pursuant to the policy reflected in 10 CFR § 2.390(a)(4) because:

   a. The information sought to be withheld from public disclosure is owned by Sanmen Nuclear Power Company Ltd. Sanmen Nuclear Power Company Ltd. holds this information in confidence.

   b. The information is of a type customarily held in confidence by Sanmen Nuclear Power Company Ltd. and is not customarily disclosed to the public. The information reveals the distinguishing aspects of a proprietary and confidential process with commercial value and technological advantage.

   c. The information is being transmitted to the NRC in confidence and, under the provisions of 10 CFR § 2.390, is to be received in confidence by the NRC.

   d. The information sought to be protected is not available in public sources and could not be gathered readily from other publicly available information.

   e. Public disclosure of this proprietary information is likely to cause substantial harm to Sanmen Nuclear Power Company Ltd. competitive position by disclosing non-public commercial information. The information requested to be withheld reveals commercially valuable and sensitive information and its disclosure could adversely affect Sanmen Nuclear Power Company Ltd. because it would enhance the ability of third parties, including competitors, to gain knowledge of our commercial strategies.

3. Accordingly, Sanmen Nuclear Power Company Ltd. requests that the designated document be withheld from public disclosure pursuant to 10 CFR § 2.390(a)(4).

I declare that the foregoing is true and correct to the best of my knowledge, information and belief.

Wu Yuanming ______________________ Executed on 2018. 7. 26

Name Date