February 5, 2018

MEMORANDUM TO: Samuel S. Lee, Chief
Licensing Branch 1
Division of New Reactor Licensing
Office of New Reactors

FROM: Marieliz Vera Amadiz, Project Manager ***/RA/***
Licensing Branch 1
Division of New Reactor Licensing
Office of New Reactors

SUBJECT: U. S. NUCLEAR REGULATORY COMMISSION STAFF REPORT OF REGULATORY AUDIT FOR NUSCALE POWER, LLC; NUSCALE REACTOR INTERNALS COMPREHENSIVE VIBRATION ASSESSMENT PROGRAM AND SEISMIC ANALYSIS

On January 6, 2017, NuScale Power, LLC (NuScale) submitted a design certification (DC) application, for a Small Modular Reactor, to the U.S. Nuclear Regulatory Commission (NRC) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17013A229). The NRC staff started its detailed technical review of NuScale's DC application on March 15, 2017.

The NRC staff conducted an audit of component design specifications associated with the NuScale DC application, Final Safety Analysis Report, Section 3.9.2. The audit was initiated on May 16, 2017, and ran through November 2, 2017, in accordance with the audit plan in ADAMS Accession No. ML17118A283.

The purpose of the audit was to: (1) gain a better understanding of the NuScale design; (2) verify information; (3) identify information that may require docketing to support the basis of the licensing or regulatory decision; and (4) review related documentation and non-docketed information to evaluate conformance with regulatory guidance and compliance with NRC regulations.

The audit was performed to gain a better understanding of the reactor internals comprehensive vibration assessment program (CVAP) and seismic analysis of the NuScale Standard Plant DC application are being performed in accordance with the methodology and criteria described in the NuScale FSAR.

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The NRC staff conducted the audit via access to NuScale's electronic reading room. The audit was conducted in accordance with the NRC Office of New Reactors (NRO) Office Instruction NRO-REG-108, "Regulatory Audits."

The publicly available version of the audit report and the audit attendee list are enclosed with this memorandum.

Document transmitted herewith contains sensitive unclassified information. When separated from the enclosure, this document is "DECONTROLLED."

Docket No. 52-048

Enclosures:
1. Audit Summary – (Non-Proprietary)
2. Attendee List
3. Audit Summary – (Proprietary)

cc w/encl.: DC NuScale Power, LLC Listserv (w/o Enclosure 3)

SUBJECT: U. S. NUCLEAR REGULATORY COMMISSION STAFF REPORT OF REGULATORY AUDIT FOR NUSCALE POWER, LLC; NUSCALE REACTOR INTERNALS COMPREHENSIVE VIBRATION ASSESSMENT PROGRAM AND SEISMIC ANALYSIS
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U.S. NUCLEAR REGULATORY COMMISSION

SUMMARY AUDIT REPORT OF NUSCALE POWER, LLC, REACTOR INTERNALS

COMPREHENSIVE VIBRATION ASSESSMENT PROGRAM

AND SEISMIC ANALYSIS

NRC Audit Team:

Yuken Wong, NRO Senior Mechanical Engineer (NRC), Audit Lead
Stephen Hambric, (NRC Consultant)
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1.0 BACKGROUND

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, Section 47, "Contents of applications; technical information," states that:

> *The application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. The information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the [U. S. Nuclear Regulatory Commission] NRC, and procurement specifications and construction and installation specifications by an applicant. The Commission will require, before design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if the information is necessary for the Commission to make its safety determination.*

On January 6, 2017, NuScale Power, LLC (NuScale) submitted a design certification (DC) application for a Small Modular Reactor to the U.S. Nuclear Regulatory Commission (NRC) Agencywide Documents Access and Management System Accession (ADAMS) No. ML17013A229. A Comprehensive Vibration Assessment Program (CVAP) report, TR-0716-50439-P, Rev. 0, "NuScale Comprehensive Vibration Assessment Program Technical Report," was submitted as part of the design certification document (DCD). In conducting the review of the NuScale DC application related to the reactor internals CVAP and the reactor internals analysis for the faulted condition (Service Level D) as specified in NRC NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, Components, and Equipment," the NRC staff requested that the applicant make available design documents, drawings, test plans, and test reports to: (1) gain a better understanding of the NuScale design; (2) verify information; (3) identify

information that may require docketing to support the basis of the licensing or regulatory decision; and (4) review related documentation and non-docketed information to evaluate conformance with regulatory guidance and compliance with NRC regulations. Specifically, the NRC staff reviewed the documents to confirm that the reactor internals are designed to withstand dynamic loads associated with the environmental conditions.

The NRC staff provided NuScale with the audit plan to facilitate the audit, as documented in ADAMS Accession No. ML17118A283. The NRC staff followed the NRO Office Instruction NRO-REG-108 (Revision 0), "Regulatory Audits," in performing the audit of the NuScale design specifications.

At the NRC office in Rockville, Maryland, from May 16, 2017, through November 2, 2017, staff members from the Mechanical Engineering Branch (MEB) of the Division of Engineering and Infrastructure in the NRC Office of New Reactors (NRO) conducted a regulatory audit of the dynamic testing and analysis of the NuScale reactor internals. The NRC staff reviewed the Nuscale design documents, drawings, test plans, and test reports related to the reactor internals CVAP and analysis for Service Level D (seismic in combination with pipe break events). The NRC staff's observations and findings are documented in the Audit Results section.

2.0 DOCUMENTS REVIEWED

General and Flow-Induced Vibration (FIV) Global Inputs

- ER-A010-2085, Rev. 1, "Reactor Module Flow Induced Vibration Methodology Development Plan," July 13, 2017, 59 pages.
- EC-A030-2713, Rev. 2, "Primary and Secondary Steady State Parameters," September 7, 2016, 146 pages.
- EC-A010-3204, Rev. 1, "RCS Loop CFD," July 11, 2017, 93 pages.
- ER-A010-2151, Rev. 1, "FIV Methodology Development for Damping Values," 14 July 14, 2017, 16 pages.
- ER-A010-2157, Rev. 0, "Methodology Development for Hydrodynamic Effect Evaluation for Reflector and Core Barrel," November 23, 2015, 30 pages.
- EC-A014-3306, Rev. 2, "Steam Generator Structural Model," August 11, 2017, 101 pages.

Methodology Development

- ER-A010-2158, Rev. 1, "Methodology Development for Turbulent Buffeting Analysis," July 14, 2017, 44 pages.
- EC-A010-2230, Rev. 1, "NuScale Power Module Flow Induced Vibration Methodology for Acoustic Resonance, Leakage Flow Instability, and Flutter/Gallop," July 14, 2017, 27 pages.
- ER-A010-2232, Rev. 1, "FIV Methodology Development – Turbulent Buffeting Degradation Mechanisms," July 14, 2017, 18 pages.

Analyses and Results

- EC-A023-2160, Rev. 1, “Vortex Shedding Evaluation for Reactor Vessel Internals and Steam Generator,” 67 pages.

- EC-A010-2779, Rev. 1, “Evaluation of Acoustic Resonance, Flutter/Gallop, and Leakage Flow Induced Vibration for Reactor Module,” August 30, 2017, 41 pages.

- EC-A014-2911, Rev. 2, “Fluid Elastic Instability Analysis of Steam Generator Tubes,” October 9, 2017, 36 pages.

- EC-A023-3535, Rev. 0, “Reactor Vessel Internals Turbulent Buffeting Degradation Evaluation,” November 18, 2016, 72 pages.

- EC-A014-3986, Rev. 0, “Steam Generator Turbulent Buffeting Degradation Evaluation,” October 21, 2016, 48 pages.

Testing

- ER-A010-4079, Rev. 1, “Comprehensive Vibration Assessment Program Measurement and Inspection Report,” October 8, 2017, 39 pages.

- TSD-T050-54312, Rev. 0, “Test Specification – Steam Generator Flow Induced Vibration,” June 29, 2017, 34 pages.

- TSD-1013-5210, Rev. 1, “SIET TF1 Final Test Specifications,” November 8, 2013, 11 pages.

- TSD-T050-3424, Rev. 2, “Test Specification - Steam Generator Flow Restriction Device Test,” February 17, 2016, 32 pages.

- Alden Research Labs test plan: TI-0416-48552, Rev. 1, “Steam Generator Flow Restrictor Testing Test Plan,” April 2016, 79 pages.

- TSD-T070-8245, Rev. 6, “Test Specification – Flow Induced Vibration of NuScale Control Rod Assembly,” May 11, 2017, 30 pages.

- NuScale Test Report, EP-0703-1417-F01, Rev. 1, “SIET TF1,” July 16, 2014.

- SIET Test Report, 02511 RA 15, “Helical Coil Steam Generator Test Program – Fluid-Heated Facility (SIET TF2) Final Data Report,” November 17, 2015

Level D Analysis

- EC-A010-2322, Rev. 2, “Reactor Module Seismic Model”

- EC-A010-3559, Rev. 3, “Reactor Module Seismic Calculation”

- EC-A011-2278, Rev. 0, “RPV Primary Stress Analysis”

- RPV Support Interface Loads (RPV_support_interface_loads.pdf)

- EC-A014-00003224, Rev. 0, “SG Tube Structural Integrity Performance Criterion Calculation”
- ER-A010-3616, Rev. 1, “RXM Blowdown Loading Specification,” March 31, 2016

Drawings

- Drawing NP12-01-A010-M-GA-1957-S01 through S06, Rev. 1, “Reactor Module Assembly”
- Drawing NP12-01-A011-M-GA-1932-S1 through S3, Rev. C, “Reactor Pressure Vessel Assembly”
- Drawing NP12-01-A011-M-GA-2689-S01 through S19, Rev. 0, “Steam Generator”
- Drawing NP12-01-A023-M-GA-2303-S01 through S12, Rev. 1, “Reactor Vessel Internals – Upper Riser”
- Drawing NP12-01-A023-M-GA-1958-S01 through S5, Rev. 2, “Reactor Vessel Internals”
- Drawing NP12-01-A023-M-GA-2304-S01 through S13, Rev. 1, “Reactor Vessel Internals – Lower Riser”

3.0 AUDIT RESULTS

3.1 Reactor Internals Comprehensive Vibration Assessment Program

Many of these documents were revised during the audit, and the testing documents are still under revision. The documents explain analysis methodologies, models, analysis results, and future testing plans based on the analysis results. NuScale relies heavily on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section III Nonmandatory Appendix N-1300, “Flow-induced vibration of tubes and tube banks,” for much of their analyses, along with Au-Yang’s, “Flow-induced vibration of power and process plant components: a practical workbook,” and S.S. Chen’s paper on fluid-elastic instability (FEI) and vortex shedding (VS) of helical steam generator tubing: “Tube Vibration in a Half-scale Sector Model of a Helical Steam Generator,” Journal of Sound and Vibration, 91(4), pp. 539-569, 1983.

3.1.1 Evaluation of General, Inputs, and Methodology Documents

ER-A010-2085, Rev. 1, is the basis for most of the submitted CVAP Report, defining reactor internals which are screened for FIV, along with short summaries of analysis approaches. The original version (Rev. 0) of the document also outlined planned testing, but this was removed from Rev. 1 to avoid duplication with ER-A010-4079, Rev. 1. Detailed descriptions of analysis methods are found in the individual documents. NuScale screens the following components for FIV:

- reactor vessel internals (RVI) and structures,
- steam generator (SG) components, and

- reactor coolant system piping up to and including the isolation valves.

All components are initially screened, and those deemed susceptible to any of the FIV mechanisms are evaluated in more detail. Based on the detailed analyses, selected components are identified for testing to confirm the analyses for these components.

The following FIV mechanisms are addressed, and are listed along with the corresponding NuScale analysis methodology documents:

- Turbulent Buffeting (TB)
    - ER-A010-2158
- Acoustic resonance (AR), Leakage Flow Instability (LFI), and Flutter and Galloping (F/G)
    - EC-A010-2230
- FEI and VS
    - There is no separate document for FEI and VS. Instead, the methodologies (largely based on ASME B&PV Code III Appendix N) are described in the analysis results documents for the SG and RVI

For TB, NuScale also assesses fatigue life associated with contact between adjacent components (ER-A010-2232). All other FIV mechanisms are only evaluated for their presence (since they are associated with "lock-in" of structural or acoustic motion with a flow-induced excitation mechanism). If the lock-in does not occur, the response is considered negligible. <u>The audit, however, revealed nonconservative assumptions in NuScale's approach, and that there is moderate to high risk that VS lock-in and FEI can occur. Therefore, requests for additional information (RAI) will be issued requesting fatigue life assessments of the SG tubing, incore instrument guide tube (ICIGT), and control rod drive shaft (CRDS) under VS and/or FEI conditions.</u>

Key parameters for FIV analyses are listed below, along with NuScale supporting internal document names:

- Flow velocities
    - EC-A010-3204
- Structural mode shapes and resonance frequencies
    - EC-A014-3306
- Damping ratios
    - ER-A010-2151
- TB forcing functions
    - ER-A010-2158

After initial screening, several RVI were identified for FIV analyses per Table 3-1 of the NuScale CVAP Technical Report (TR-0716-50439-P, Rev. 0).

3.1.1.1 Computational Fluid Dynamics Flow Analyses

EC-A010-3204, Rev. 1, "RCS Loop CFD"

A three-dimensional Computational Fluid Dynamics (CFD) Reynolds Averaged Navier Stokes (RANS) model was built of the NuScale reactor module and analyzed for several conditions using CFX (from ANSYS). The SG and reactor core are modeled as porous media with assumed loss coefficients and power density. The report also states,

> *… this CFD analysis is not suitable for estimating localized flow rates/speeds around tubes and core components.*

Flow velocity convergence was verified by CFD grid refinement studies. However, flow vorticity is not converged. Since vorticity is not used in NuScale's FIV analyses, the NRC staff finds that this lack of convergence is not important.

Several planes of flow velocity were extracted from the CFD solution for the highest flow conditions (case 10) and average and maximum velocities reported, along with contour plots of several planes throughout the reactor. Maximum velocities are generally twice those of the average velocities. The flow speeds are higher in the center of the riser [ ] and slower near the walls, indicating a typical pipe flow boundary layer. The flow decelerates slightly in the upper transition region to [ ], where the flow turns from axial to radial flow before moving down through the steam generator. The turning flow region is where the primary coolant travels transversely across the ICIGT and CRDS, potentially inducing VS forces. The flow is slower in the SG region [ ] due to the larger cross sectional area.

The NRC staff finds that the CFD analyses are reasonable. However, there is uncertainty regarding the assumed SG and reactor core loss coefficients and power densities. If these assumptions are incorrect, the actual flow velocities could be higher, reducing the small margins against vortex shedding-lock-in and fluid-elastic instability currently reported by NuScale.

3.1.1.2 Structural FE Modeling

EC-A014-3306, Rev. 2, "Steam Generator Structural Model"

The SG was modeled as a full assembly to assess global modes of vibration (and support seismic model development), and three individual tubes (columns 1, 11, and 21) were also modeled to estimate tube resonances. The models include the tube supports and cantilevers. External fluid loading is approximated as that of water mass displaced by the tube, and internal fluid loading is modeled as either all steam or all water. The fluid loading is modeled by increasing the structural density. The tubes are assumed to be welded to the inlet and outlet plenums, which is appropriate. The tube-support conditions restrict transverse motion, but allow "slip," or tangential motion at the supports.

The resonance frequencies for tubes filled with water are the most limiting (lowest frequencies). Also, the modes for column 21 (the largest column diameter) are the lowest, and therefore the

most limiting. The fundamental "breathing" mode resonates at [ ] at the outer column for a water-filled tube. However, the breathing mode corresponds to motion which is primarily tangential and not likely to couple with vortices which induce transverse forces. Modes with transverse motion start or "cut on" at [ ] (radial motion, which could couple to the unsteady lift induced by VS) and [ ] (vertical motion which could couple to the unsteady drag induced by VS). These resonances, along with others below [ ], are evaluated for coupling with VS and FEI in separate analysis documents.

Revision 2 of the report includes analyses of SG tube modes with assumed inactive tube supports near the top and bottom of the tubes. In-plane (radially) oriented modal frequencies are as low as [ ] and vertically oriented modal frequencies are as low as [ ]. NuScale discusses qualitatively the risk associated with inactive tubes in their VS/FEI SG tube evaluations, but claims that this possibility is low. This report discusses this issue in more detail in the "Analysis" section below.

The author of this NuScale report includes several comments and concerns in Appendix G about the design of the SG supports, cantilevers, and welds. These comments are listed below:

- [ ]
- [ ]
- [ ]
- [ ]
- [ ] and
- [ ]

These comments should be addressed by NuScale and an RAI will be issued to NuScale.

*3.*1.1.3 Structural Damping

ER-A010-2151, Rev. 1, "FIV Methodology Development for Damping Values"

The RVI are assigned damping of 0.3 percent for TB analysis. Damping assumed for RVI evaluated for VS is not specified in this document, but is assumed to be 0.5 percent for ICIGT in EC-A023-2160. No damping is assumed for CRDS VS since NuScale states that frequency based screening is satisfied.

The SG tubes are assigned a damping value of 0.5 percent for TB analysis, but a higher value of 1.5 percent for VS and FEI analyses, claiming that in the event VS and FEI occurs interactions with the SG supports will provide higher damping than for lower amplitude

response. Damping greater than 1 percent should be verified by testing per RG 1.20. NuScale claims this assumption will be validated by separate effects testing. The staff is concerned, however, that damping cannot be accurately measured in the presence of instrumentation and wiring. This concern will be part of an RAI to be issued to NuScale.

3.1.1.4   Turbulent Buffeting Analysis

ER-A010-2158, Rev. 1, “Methodology Development for Turbulent Buffeting Analysis”

The TB analyses are the most complicated of the NuScale FIV analyses. Whereas VS, AR, FEI, and F/G are assessed by comparing resonance and flow-excitation frequencies and determining margin of safety against these mechanisms, actual vibration and alternating stresses must be computed for TB. TB calculations require knowledge of forcing functions induced by complex turbulent flow fields, structural mode shapes, and three dimensional random analysis methods. ER-A010-2158 defines NuScale’s forcing functions and analysis approach (Rev. 1 of this document was heavily revised, with many comments removed from Rev. 0 regarding the need for testing to determine actual forcing functions for complex flows). A second document, ER-A010-2232, describes methods for assessing any degradation of RVI due to TB-induced vibration and stress.

The RVI are evaluated for root mean square (RMS) and peak displacements, which are compared to clearances to assess the possibility of contact. Established methods are cited for assessing impact and wear fatigue. Fatigue is also assessed for alternating stresses. Forcing functions are defined for parallel, cross, and axial flows, and are comprised of the usual pressure power spectral densities (PSDs), convection velocities, and correlation lengths. Unlike with previous boiling water reactor (BWR) and pressurized water reactor (PWR) applicants, NuScale does not provide measurements of any of these forcing components. Instead, empirical models from the literature (mostly from Au-Yang’s workbook) are applied.

The NRC staff finds that in some simple cases, the empirical models (provided bounding assumptions are used) are acceptable. However, the flow in some regions, particularly in the lower reactor, does not resemble those of the empirical models for simple flows. In particular Revision 0 of ER-A010-2158 states:

> *The lower riser region represents the most complex flow path in the reactor coolant flow circuit and some components in this region are exposed to cross flow. None of the open source PSDs evaluated are representative of this geometry. Testing of CRAGT and the interfacing components is planned to validate that the analytical approach selected to predict the RMS response is sufficiently bounding.*

CRAGT forcing function testing was originally planned, but has since been removed from NuScale’s planning. The NRC staff will issue an RAI related to this concern.

Similar comments are made regarding flow-induced forces within the SG tubes (at least in Rev. 0 of this document; they are largely removed from Rev. 1). Testing in the SIET Test Facility (TF)-1 facility in Placenza, Italy, is cited, and was to be used to generate internal TB forces. The SIET TF1 testing (described later in this document) revealed strong pressure pulsations at frequencies between [   ]. NuScale has not accounted for these test data and instead assumes a non-bounding forcing function for the SG secondary coolant flow. The staff

is concerned that these unexplained high pressures are neglected, and believes that test data should be used to define forcing functions internal to the tubes.

The standard random analysis method, where forcing functions built from PSDs, correlation lengths, and convection velocities, is not always used by NuScale. Instead, it appears that a simplified [ ] It is also unclear whether NuScale uses cross-modal coupling in their analyses. While acceptable for some applications, the absence of cross-modal terms, particularly for heavily fluid-loaded structures, is generally non-conservative. Convection velocities are assumed to range from 50-100 percent of flow velocity (it is unclear if NuScale uses the appropriate flow velocity from CFD analyses). Coherence integral length scales are fractions of annulus widths or structural dimensions (cylinder hydraulic radii, for example), with multiple analyses for lower and upper bounds to capture worst-case conditions.

Since NuScale may not be using standard full random analysis of structural response to TB forces, benchmarking of the implementation of the [ ] against known solutions to canonical problems is highly recommended. The [ ] with NuScale assumptions, if indeed used by NuScale, should be shown to be a bounding methodology if used in structural response and degradation assessments of components with small margin against lock-in and/or structural degradation.

ER-A010-2232 describes methods for assessing impact, fatigue (due to impact or alternating stresses), wear (due to impact), and fretting (due to impact and sliding). Revision 1 is a significant revision, removing the lists of components to be evaluated for impact and fatigue.

The RMS deflections are estimated and multiplied by 5 to capture 99.9999 percent of all deflection peaks. An empirical estimate from Au-Yang is used to approximate the stresses induced by surface contact.

For fatigue life analyses, NuScale assumes a Rayleigh distribution of “zero-peak level” alternating stresses. The distribution is integrated over a multiple of RMS value to compute cumulative usage factors. Next, the number of fatigue cycles is computed for the “averaged crossing frequency,” calculated as the ratio of integrated velocity to displacement PSDs. While commonly used, the averaged crossing frequency results in the TB analysis document are much lower than component fundamental resonance frequencies, leading to non-conservative fatigue life estimates. It is not credible that the average crossing frequency can be lower than the fundamental resonance frequency.

For wear analyses, empirical equations and coefficients for tubes in holes from Connors (cited in Au-Yang’s book) are assumed.

The NuScale evaluation does not appear to fully account for static deformation of components affected by TB. Deflections due to gravity, buoyancy, thermal, and fluid flow (including SG tube internal flow inertial loads) effects may cause smaller than expected clearances between neighboring components, and increase the frequency and magnitude of fretting and wear. These effects may need to be addressed in an updated degradation assessment for VS forces (the very small TB forces are unlikely to cause significant degradation). These effects will be questioned in RAIs requesting updated VS evaluations.

Alternating stresses are only estimated using simplified beam and plate theory. However, peak stresses generally occur at structural joints, where weld factors often need to be considered.

Many of the RVI are welded structures, some with partial penetration welds. Alternating stress assessments at joints and welds should be discussed in VS degradation assessments (the very small TB forces are unlikely to cause significant alternating stresses). RAIs will be issued to request appropriate VS alternating stress assessments.

3.1.1.5 Acoustic Resonance, Leakage Flow Instability, and Flutter/Gallop Analysis

EC-A010-2230, Rev. 1, "NuScale Power Module Flow Induced Vibration Methodology for Acoustic Resonance, Leakage Flow Instability, and Flutter/Gallop"

The AR is evaluated for the Decay Heat Removal System (DHRS) branch lines. The DHRS system activates when heat removal is necessary, using the SG piping system with natural circulation (no pumps). Primary shear layer instability mode frequencies are compared to those of acoustic resonances in the side branches (associated with the usual closed-open standing waves) to assess the possibility of lock-in. This comparison is done via the Strouhal Number of the primary shear layer instability using values established by Ziada. The steam nozzles and steam plenum are also evaluated for AR. The main steam isolation valves (MSIVs) specified by NuScale have no side branches, and are therefore not susceptible to AR.

Gallop (lift/plunge instability) and flutter (coupled lift/torsional instability), of the lower SG support bars, are evaluated using standard methods. Critical velocities are established and compared to the actual flow velocities estimated from CFD analysis. For the support bars, the limiting critical velocity U/fD is [ ] (if dimensionless flow velocity is less than [ ], torsional galloping is avoided). Flutter is avoided by ensuring ratios of torsional to bending mode frequencies are not 1, 1/3, or 3.

The LFI was originally discussed in Revision 0 of EC-A010-2230, but the discussions were removed from Revision 1, with NuScale stating that the pressure differences along any possible leakage paths are small, such that significant leakage flow cannot occur. The only component screened for LFI is the SG inlet flow restrictor, which is discussed in the testing overview planning document. The risk of LFI for the SG inlet flow restrictor is unknown since no analytical or computational assessments have been presented.

3.1.2 Evaluation of Analysis Results Documents

The analysis results are spread among several reports listed below:

- RVI TB and SG degradation
    - EC-A023-3535, Rev. 0, "Reactor Vessel Internals Turbulent Buffeting Degradation Evaluation"
    - EC-A014-3986, Rev. 0, "Steam Generator Turbulent Buffeting Degradation Evaluation"
- RVI and SG AR, LFI, and F/G
    - EC-A010-2779, Rev. 1, "Evaluation of Acoustic Resonance, Flutter/Gallop, and Leakage Flow Induced Vibration for Reactor Module"

- RVI and SG VS
    - EC-A023-2160, Rev. 1, “Vortex Shedding Evaluation for Reactor Vessel Internals and Steam Generator”
- SG FEI
    - EC-A014-2911, Rev. 2, “Fluid Elastic Instability Analysis of Steam Generator Tubes”

3.1.2.1 RVI and SG TB and TB degradation

The EC-A023-3535 (RVI) and EC-A014-3986 (SG) discuss TB evaluations and degradation estimates for components that are exposed to turbulent flow. Neither the RVI nor SG TB report has been updated to reflect revisions to structural modeling in Rev. 2 of EC-A014-3306 or CFD flow solutions in Revision 1 of EC-A010-3204. This may be due to very low TB degradation estimates (with one exception – the CRAGT wear estimate). NuScale follows the procedures discussed in the TB analysis and degradation methodology documents. Structural FE models are shown, along with assumed boundary conditions. The meshes are quite coarse, and boundary conditions are all highly idealized. All components are modeled individually (unlike the assemblies modeled for seismic evaluations). ICIGT and CRDS supports are assumed to be horizontally rigid at the support plates, which is nonconservative since there are clearances between these components and the support plates (subsequent impact and wear assessments acknowledge that the CRDS and ICIGT will impact and slide along the supports). Fluid-loading (hydrodynamic) mass is assumed to be that of the volume displaced by a given structure, which is a reasonable approximation.

Forcing functions are applied per the methodology documents, and assuming averaged CFD flow velocities for RVI and maximum estimated gap velocities for the SG tubing. Since the forcing functions are defined based on free-stream or core flow velocities of simplified channel or pipe flows, it is non-conservative to assume averaged velocities for the RVI TB analyses. Instead, peak velocities near the center of channel or pipe flow should be applied to be consistent with the assumed forcing function models. NuScale does assume convective velocities are equal to 100 percent of the averaged velocities, though, which partially corrects for the inappropriate use of averaged velocities (convective velocities are usually 60-80 percent of the free stream velocity). Also, the CFD calculations have been revised, leading to slightly increased flow velocities which are not accounted for in the TB analyses. Finally, NuScale acknowledges that the CRAGT forcing functions are unknown, and simple channel flow is assumed. As in other documents, NuScale calls for separate effects testing to establish a more appropriate forcing function. However, NuScale has since cancelled this testing.

The coherences of the forcing functions are assumed to be one-dimensional over slender structures so that only one length scale is used. While this is true for some structures, it is not for others where two-dimensional modeling is usually necessary, and length scales are different in the flow and cross-flow directions. Using a single length scale effectively filters some structural resonances out of forced response calculations, leading to non-conservative calculated vibrations. However, based on the information provided in the report it is unclear how important two-dimensional effects are for NuScale structures.

The following components are evaluated:

- Core barrel, outer diameter
- CRA card
- CRAGT, inner and outer diameters, and CRAGT support
- CRDS and CRDS support
- Flow diverter
- Upper and lower ICIGT
- Injection line: riser, downcomer, and downcomer interior
- Lower core plate
- Lower riser, inner and outer diameter
- Reflector
- Spray nozzle
- Upper core plate and upper core plate support
- Upper riser, inner and outer diameter and hanger
- SG tubing
- SG inlet flow restrictor inserts

<u>The SG tube support cantilevers do not appear to be evaluated in either report. The NRC staff will request this analysis in an RAI.</u>

The RMS deflections and averaged crossing frequencies are calculated for each component. The crossing frequencies seem low for most components (much less than the fundamental resonance frequency), which will lead to lower and non-conservative fatigue usage estimates since fewer cycles are counted. No mesh convergence studies are performed, nor are the uncertainties associated with the idealized boundary conditions assessed. Resonance frequencies and mode shapes for some components are inconsistent with those in revised analysis documents which evaluate the effects of vortex shedding and FEI.

Alternating stresses are estimated using simple beam and plate theory, and are much lower than allowable fatigue limits. However, no estimates of higher stresses at welds or other stress concentration locations are performed. The TB deflections are compared to 1/5 the clearances to neighboring components (this bounds 99.9999 percent of possible deflections). Only the CRDS to CRDS support, ICIGT to ICIGT support, and CRAGT to CRAGT support have high enough deflections to warrant impact fatigue, wear, and fretting analysis. Impact stresses are calculated, and while higher than those due to simple alternating stresses, are still so low that negligible fatigue usage will occur. Even increasing the seemingly low crossing frequencies to

the fundamental resonance frequencies does not increase fatigue usage to even 1 percent of allowable fatigue life. Fretting assessments also show benign damage.

Wear estimates are more significant, however. The CRDS and ICIGT may experience up to [,] tube wall thickness wear depth. These values, while not negligible, are still small over the life of the plant. The CRAGT estimated wear, however, is very large [ ] of wall thickness. While NuScale has used a conservative assumed deflection of 5 times RMS, there is significant unresolved uncertainty in their assumed forcing function, which does not resemble anything close to the complex flow through the CRAGT. In EC-A023-3535, NuScale calls for testing to determine a more appropriate forcing function. NuScale has since cancelled this testing in other documents. The staff will issue an RAI recommending that CRAGT FIV testing be reinstated to address the significant wear risk.

The SG tube TB degradation assessments assume 0.5 percent damping, a maximum flow velocity of [ ], a spanwise correlation length of slightly less than one tube diameter (reasonable), and allow up to [ ] wear through the tube thickness. The external (primary coolant) forcing function is assumed per Au-Yang for cross-flow over a cylinder. The internal flow forcing function, however, ignores the strong forcing peaks observed in the SIET TF1 testing. These peaks, however, are at higher frequencies, and may not cause significant deflection or stress. Nevertheless, the measured internal forces should be applied to SG structural models to assess possible deformation and damage, and an RAI will be issued to request this assessment. Mode shapes from EC-A014-3306, Rev. 0 are used in the deflection analysis. Only vertically oriented modes are considered for primary flow excitation, and both radial and vertically oriented modes are included in analyses for secondary flow (internal) excitation.

The SG tube deflection estimates are small, less than 1/5 of the clearances with the supports. However, impact stresses are still estimated (since NuScale states in other documents that they expect thermal expansion and flow loading to press the tubes against the supports, this seems appropriate). NuScale specifies an assumed impact stress intensity of [ ]. For a 60 year life, this leads to a [ ] fatigue usage factor, which is quite low. Since the tubes will be inspected during refueling outages throughout the plant life, abnormal wear will be detected and tubes will be plugged before reaching the plugging limit of [ ] reduction in tube thickness. Wear is also estimated even though the deflections are assumed to be small (again, this is appropriate given NuScale's assertions that static deflections will cause contact). NuScale estimated [ ] wear over the 60 year reactor life, which is less than the [ ] limit established by NuScale.

While NuScale's TB analyses are highly simplified (coarse meshes with highly idealized boundary conditions), use of non-conservative assumptions regarding flow velocities and forcing functions, and calculated low averaged crossing frequencies, the TB degradation is so small (with the exception of the CRAGT) that there is no need to issue RAIs to correct these aspects of their analyses. The small TB effects are primarily due to the very low flow velocities in the NuScale reactor, which are an order of magnitude lower than those in operating PWRs. There are two/three exceptions however: the unknown CRAGT forcing functions and the possibility of significant wear through the wall thickness, and the unknown alternating stresses at welds and other stress concentration locations. RAIs will be issued to address these concerns. Also, TB degradation of the SG support cantilever does not appear to be analyzed, and an RAI will be issued to request those analyses.

3.1.2.2 RVI and SG AR, LFI, and F/G

EC-A010-2779 addresses various FIV issues: AR in valve and other standpipes, LFI, and F/G. The only mechanism with significant concern for the NuScale plant is AR in the DHRS piping.

The only component screened for LFI is the SG inlet flow restrictor inserts, which will be evaluated via separate effects testing in the near future.

The only component evaluated for F/G is the SG lower support cantilever. Standard equations for assessing F/G are used, and although NuScale's assumed flow velocity is likely biased low, there is significant margin against F/G.

The following components were evaluated for AR:

- containment system and steam generator system steam piping including nozzles and the MSIVs,
- SG steam plenum,
- DHRS steam piping from containment tee to DHRS actuation valves,
- DHRS condensate piping from SG system feedwater piping tee to DHRS condenser,
- reactor recirculation valve ports, and
- instrument ports.

Reasonable properties are assumed for steam and water, and flow rates are calculated based on mass flow through the piping. The NRC staff is unsure if maximum or average velocities were used. Standard closed side branch formulas, including open-end corrections, are used to estimate resonance frequencies. The Strouhal Number (fD/U) is computed using the side branch opening diameters, not including any fillets (which does not appear to be defined yet). This is conservative since not including the fillets leads to smaller Strouhal Numbers. The Strouhal number for a given configuration is compared to the range specified by Ziada of 0.35-0.62 where the shear layers couple with the primary side branch acoustic resonance (the AR condition). If the Strouhal number is outside this range, then AR cannot occur. The closed side branch from the steam system tee to the DHRS actuation valve is at highest risk for AR, with an estimated Strouhal Number of [ ] above the upper 0.62 limit). The DHRS condensate line from the SG system feedwater tee to the DHRS passive condenser is at little risk for AR due to primary shear layer loading, with an estimated Strouhal Number between [ ], well above the limit.

NuScale only considers the primary shear layer instability in their AR screening. However, the secondary shear layer has been observed to lock-in with standpipe resonances in main steam piping in operating BWRs. It has become standard practice for licensees requesting extended power uprates (EPUs) or those proposing new plant designs to confirm, through scale model testing (SMT) or in-plant measurements, that both first and second order shear layer instabilities do not lock-on to side-branch acoustic modes and damage valves or generate acoustic loads which could damage other components including reactor internals. The DHRS condensate line may be susceptible to second order shear layer AR with a Strouhal Number of [ ] compared to the AR for second order shear layers of 1.24 (twice the 0.62 value for primary shear layers).

The NRC staff recommends that separate effects testing be performed on the DHRS lines and side branches to screen for primary and secondary shear layer lock-in with standpipe acoustic resonances (these tests have been requested in a follow-up questions to RAI 8884). Previous SMT acceptable to the NRC has employed air as the working fluid, and polyvinyl chloride (PVC) or metal for the piping (provided the internal geometries are properly represented). An RAI will be issued as a follow-up to RAI 8884, Question 03.09.02-7 which requests analysis of possibility of second order shear layer AR occurring, as well as separate effects testing to confirm margin of safety against this mechanism.

3.1.2.3 RVI and SG VS

In EC-A023-2160, both RVI and SG response to VS are addressed. The following components are evaluated for possible lock-in of VS and structural modes:

- SG lower tube support cantilever,
- SG helical tubing,
- upper support block,
- core support block,
- lower and upper core plates,
- CRAGT,
- ICIGT,
- CVS injection RVI,
- CRDS,
- CRDS support,
- CRAGT support plate,
- RCS thermowells,
- secondary side thermowells,
- upper riser hanger brace, and
- SG – vertical vs. radial modes (VS unsteady lift vs. drag).

No forced responses are calculated – only the possibility of VS lock-in.

Assumed flow velocities are based on CFD analyses described in EC-A010-3204. In that reference, several planes of flow distributions are provided at various locations in the reactor and SG regions. Although NuScale has stated to the NRC on numerous occasions that their simplified FIV analyses are bounding due to their selection of conservative modeling inputs, the velocities used for their VS simulations are not bounding. Rather than use the maximum

velocities reported in EC-A010-3204, NuScale uses the average velocities instead. The average velocities are roughly half that of the maximum ones. NuScale should update all VS calculations and reported margins of safety using maximum bounding velocities.

The SG tubing structural modes are described in EC-A014-3306. Structural FE models of non-SG tubing are similar to those described in the TB document, but have been updated. Again, highly simplified models and boundary conditions are applied. Water mass loading is estimated as the amount of water displaced by a structural volume (a reasonable low frequency approximation). The most conservative (highest) water mass density is assumed. The CRDS, ICIGT, and SG tubing are assumed to be simply supported along the walls of nearby holes or tabs. However, TB degradation evaluations all assume a finite clearance between these structures and their nearby supports. It is non-conservative to assume that these nearby walls/supports provide any support to these structures since estimated resonance frequencies are increased by these boundaries. SG tubing damping is assumed to be 1.5 percent (higher than the RG 1.20 guidance of 1 percent). Internal secondary coolant mass density is assumed to be that of water for conservative structural resonance frequency estimates. Three columns of SG tubing are modeled, with the outermost column having the lowest resonance frequencies.

The CRDS and ICIGT are the components most susceptible to VS lock-in. Both sets of structures extend through the region above the upper riser (above the top CRDS support plate and below the pressurizer baffle plate) where the flow turns radially into the SG region. The radial flow crosses over the cylindrical cross sections of the structures, generating vortex shedding.

Characteristic dimensions are established for all components. For circular cross sections, the characteristic dimension is simply the diameter. For other bluff bodies, the smallest dimension is assumed for conservatism (this increases the shedding frequency closer to the lowest structural resonance frequency of a given component).

The ASME B&PV Code Subsection N-1324.1 specifies four groups of criteria for avoidance of VS lock-in with structural resonances. NuScale applies the simplest criterion (a) for initial screening: that the so-called "reduced velocity," the actual flow velocity normalized by the worst-case vortex shedding velocity, is less than one. This is essentially the same as ensuring that the vortex shedding and structural resonance frequencies do not align. All components except for the CRDS, ICIGT, and SG tubing have significant (more than [ ]) margin.

The CRDS and ICIGT were reevaluated using ASME N-1324.1 criterion (d), which requires 30 percent difference between structural resonance frequencies and shedding frequencies for both lift and drag (twice the lift frequency). NuScale calculates that the CRDS has [ ] margin against lock-in with vortex shedding drag loading. However, the staff questions the appropriateness of NuScale's boundary condition assumptions at the CRDS support plate holes. If those boundary conditions are removed, the CRDS structural resonance frequencies will decrease, reducing that margin.

The ICIGT was further evaluated using ASME N-1324.1 criteria b and c, which combine reduced damping with reduced velocities. 0.5 percent damping is assumed, which is conservative, and modal masses are estimated for the mode shapes, including fluid mass loading. The ICIGT modes do not pass criterion b, which requires very high reduced damping of greater than 64. However, the modes have small margin against criterion c, which requires reduced damping greater than 1.2 and reduced velocities less than 3.3. The lowest ICIGT

modes have margin of [ ] against the reduced velocity part of criterion c. This is small, given the non-conservatism of assuming horizontal constraint boundary conditions at the CRDS support plates. There are also no mesh convergence studies of the ICIGT model, or of the CRDS model, adding possible bias to the results (resonance frequencies generally decrease with increasing mesh density).

The SG tubing near the bottom of the SG is also evaluated using ASME N-1324.1 criteria b and c. Since NuScale assumes 1.5 percent damping (which is not substantiated), the reduced damping is quite large, and exceeds the ASME allowable of 64 with a lowest margin of [ ]. It is unclear what modes NuScale is evaluating in Revision 1 of EC-A023-2160. The resonance frequencies cited do not seem to match those in EC-A014-3306, Rev. 2. In EC-A014-3306, Rev. 2, SG tubing modes are also provided assuming single inactive supports. The resonance frequencies for those modes are much lower than those with active supports, and NuScale states that VS analyses show no margin against lock-in with inactive supports. The quantitative results for the inactive supports are not shown, however, and NuScale claims that such conditions are implausible, and that thermal and hydraulic loads will press the tubes into the supports. This is inconsistent with the TB degradation analyses, which assumes clearances between the tubes and supports. NuScale needs to (a) clarify what the actual boundary conditions are expected to be based on quantitative, not qualitative assessments, and (b) ensure consistency between their analysis methods. There are also no mesh convergence studies for the SG tubing. Also, the NRC staff has not been able to confirm from the information provided that the SG tubing and support assembly is properly modeled given the complexity of that system, which adds uncertainty to the estimated resonance frequencies. NuScale is planning separate effects tests to confirm the resonance frequency estimates and damping (see testing document section).

There is significant uncertainty in NuScale's highly simplified VS lock-in analyses, and the NRC staff believe that there is moderate to high risk that the SG tubing, CRDS, and ICIGT will experience VS-lock-in with low order structural modes. Not only is expanded separate effects and perhaps initial startup testing needed to address these risks, but there is a need for forced response analyses, including contact and stresses (including at joints/welds). These analyses should be based on worst-case, not averaged, velocities, as well as worst-case structural boundary condition assumptions.

3.1.2.4 SG FEI

The EC-A014-2911, Rev. 2 describes the most damaging FIV mechanism – Fluid Elastic Instability (FEI), where the flow-induced vibration of arrays of closely spaced structures is amplified significantly. The only component subject to FEI in the NuScale design is the SG tubing near the bottom of the SG. The small streamwise spacing between the rest of the SG tubes precludes strong vortex shedding, and therefore FEI.

NuScale uses the well-known Connors equation for evaluating the possibility of FEI in their SG tubing, where the dimensionless critical velocity (inverse of the Strouhal Number) is proportional to the mass-damping ratio normalized by fluid density and the square of the cylinder diameter:

$$\frac{U_{crit}}{fD} = C\left(\frac{m_t(2\pi\xi)}{\rho D^2}\right)^a$$

If the SG gap flow velocity is lower than critical, FEI will not occur. NuScale also conservatively requires [ ] margin against the critical velocity. NuScale computes the mass/length (*m'*) for individual mode shapes and assumes 1.5 percent damping (ξ) for all modes. Connors specified the parameters C = 2.7 and α = 0.21 to bound 90 percent of measured data for straight and U-bend SG tubing. NuScale suggests that the data and parameters by Chen (Journal of Sound and Vibration, 91 (4), 539-569, 1983) for helical tubing arrays are more appropriate, where C = [ ] and α = [ ]. However, unlike the Connors coefficients which bound 90 percent of measured data, the Chen coefficients used by NuScale fit the mean of the measured data, such that 50 percent of FEI conditions are at lower critical velocities. Chen's earlier publication (Journal of Sound and Vibration 78(3) 355-381, 1981) specifies a lower C = 2.31 to bound all measured data. It is therefore inappropriate to use the higher coefficient of 4.51 in the SG FEI analysis. Either the lower value of 2.31 should be used, or alternative coefficients suggested which, like the Connors coefficients, bound 90 percent of measured data.

NuScale uses an estimated maximum gap velocity of [ ] for their FEI assessments for SG tubing with active supports. However, they reduce their estimated flow velocity to the average flow velocity of [ ] for their assessments with inactive tube supports. Their justifications for this reduction are not substantiated quantitatively. Note that Chen and ASME B&PV Code defines flow velocity as gap velocity, not averaged velocity. NuScale further adjusts the assumed flow velocity by mode shape motion in the direction perpendicular to the flow, filtering out modes with little motion in the transverse directions (such as the fundamental "breathing" modes of the tubing).

Mass-damping is presented for the lowest 20 modes of each SG column previously analyzed for VS lock-in. All supports are assumed to provide fixed boundary conditions in the transverse directions, but allow sliding. There is no quantitative confirmation of this assumption, and NuScale states that this assumption should be confirmed by testing. The resonance frequencies in EC-A014-2911, Rev. 2 seem inconsistent with those in the FE modeling document EC-A014-3306, Rev. 2 (this is also the case for the VS evaluations). NuScale computes a minimum reduced critical velocity of [ ] over all modes spanning the three columns using Connors' coefficients and a higher reduced critical velocity of [ ] using Chen's coefficients. Chen's 1983 data show, however, that helical tubes can become unstable for much lower reduced critical velocities of [ ]. NuScale uses the reduced critical velocity value of [ ] based on Connors coefficients for their evaluations, but based on Chen's 1983 data a lower threshold of [ ] may be more appropriate (particularly if the actual damping is lower than 1.5 percent).

Reduced velocities are presented for the lowest 20 modes of the three columns for light and heavy conditions (with different assumed fluid mass densities for different operating temperatures). Not surprisingly, the largest diameter column with heavy water is most limiting due to its lower resonance frequencies, with only [ ] margin against FEI. When a support is inactive, this margin disappears (although NuScale's unsubstantiated reduction of gap velocity to average velocity restores that margin).

Since there is no confirmation that all SG tubing supports will be active, nor that higher than allowable damping of 1.5 percent is realistic, the NRC staff finds that there is moderate to high risk of FEI in the SG lower tubing. NuScale is in the process of documenting the results of significant separate effects SG tubing testing in the SIET facility, with additional tests planned in the near future. Staff review of the SIET TF2 measured tube vibration data revealed several strong resonance peaks which may be due to FEI or VS. The upcoming testing should be

rigorously planned to ensure that critical parameters like modal frequencies, damping, and the presence of any VS/lock-in and FEI at prototypic flow conditions are identified. NRC staff should be closely involved in this test planning to ensure all concerns are resolved.

3.1.3 Evaluation of Testing and Inspection Documents

Several component tests have already been performed by NuScale which include FIV related measurements. Additional tests are being planned. The following is a list of some of the reports reviewed during the audit:

- ER-A010-4079, Rev. 1, “Comprehensive Vibration Assessment Program Measurement and Inspection Report.”
- TSD-T050-54312, Rev. 0, “Test Specification – Steam Generator Flow Induced Vibration.”
- TSD-T050-3424, Rev. 2, “Test Specification - Steam Generator Flow Restriction Device Test.”
- TSD-T070-8245, Rev. 6, “Test Specification – Flow Induced Vibration of NuScale Control Rod Assembly.”
- EC-A014-2354, Rev. 1, “Frequency Analysis of TF-1 Dynamic Pressure Data, Matthew Snyder.”

The following SIET testing reports were made available to the NRC staff at the NuScale Rockville, Maryland office:

- EP-0703-1417-F01, Rev. 1, “SIET TF1.”
- 02511 RA 15, “SIET Helical Coil Steam Generator Test Program – Fluid-Heated Facility (SIET TF2) Final Data Report.”

The SIET TF2 test data were also made available for review.

The ER-A010-4079, Rev. 1 is the most up to date description of NuScale’s current testing plans. Several of the other reports are out of date, and some have been superseded. In ER-A010-4079, Rev. 1, NuScale summarizes the FIV analysis results and lists any components with low margin against AR, VS, and FEI:

- SG tubing: [ ] margin against FEI, [ ] margin against VS;
- ICIGT: [ ] margin against VS;
- CRDS: [ ] margin against VS; and
- DHRS piping: [ ] margin against AR.

To address these low margins NuScale plans the following testing:

- Separate effects testing for SG tubing to measure damping, mode shapes, resonance frequencies, and vibration amplitude.

- Separate effects testing for the SG inlet flow restrictor to confirm there is no LFI. Although not identified as a risk by analysis (no analysis was performed), LFI is a possibility for the restrictor inserts due to high flow rates and pressure drops.

- Factory testing of the lowest resonance frequencies of the ICIGT and CRDS in air. However, there is no testing planned for in water, or in the presence of reactor flow. Also, this testing will not be performed in time to support DC review.

- Initial startup testing of the vibration of the DHRS piping to assess the presence of AR.

Test plans, pre-test predictions, and testing needs documents for separate effects testing are planned for release in the second quarter of 2018. These tests are very limited compared to those performed by previous DC holders, do not address all low-margin components, and do not measure sufficient quantities in appropriate operating environments. Much of the testing is also after DC, leaving many concerns to be addressed by the COL applicant/holder. In particular:

- The CRAGT wall thickness is projected to degrade by [ ] under TB loading. NuScale acknowledges in several documents that actual flow-induced forcing is not understood for this component. Early revisions of documents recommended testing to establish appropriate forcing function estimates, but NuScale has recently cancelled those plans. This testing should be reestablished and plans coordinated with the NRC staff to ensure all FIV related concerns associated with the CRAGT are met.

- ICIGT and CRDS analyses make non-conservative assumptions, including restricting transverse motion at CRDS support plates which are designed to have significant clearances, and using lower averaged flow velocities instead of peak velocities near the center of the riser. Also, measurements are limited to in-air resonance frequencies and deferred to factory testing after DC. The measurements should be expanded to include in-water testing, and if possible in the presence of cross-flow similar to that above the upper riser. The testing should be performed much earlier, as separate effects testing, in time to support DC review by the NRC.

- DHRS AR testing will not be performed until initial startup, when it is much too late to change the design to eliminate AR should it occur. AR measurements should be performed as separate effects testing in time to support DC review. Both primary and secondary shear layer modes should be evaluated.

- SG tube follow-up separate effects testing is planned. However, discussions with NuScale have revealed that this testing may not be performed with appropriate operating conditions (empty tubes, for example), and that planned structural damping measurements may be affected by the presence of instrumentation and wiring.

The TSD-T050-54312 describes the current SG tubing separate effects test plan, which will be updated in mid-2018. The current plan will use empty tubes so that instrumentation wiring may be routed inside the tubes. In-air and in-water modal analyses will be performed. Operational tests will be performed for unheated primary coolant flow at flow rates up to 120 percent of

nominal (75 percent, 100 percent, 110 percent, and 120 percent), with measurements of external pressure fluctuations, vibrations, and tube and tube support stresses and strains. Currently, no additional tests are planned to establish critical FEI flow speeds (unless FEI occurs at one of the speeds tested). Tubing and vessel instrumentation is specified. However, there is no discussion on how the instrumentation and wiring might bias measurements of modal frequencies (will be biased low by instrumentation mass, which is conservative) or damping (will be biased high by wiring and mounting, which is non-conservative). There is also no discussion of how prototypic tube support conditions will be established (NuScale suggests that thermal expansion in the actual reactor will ensure strong contact, but these tests are unheated). Finally, if this test will not include internal flowing secondary coolant at boiling conditions, NuScale should not use this test as true confirmation that VS/lock-in and/or FEI will not occur (or that [ ] internal loading will not occur). Therefore, VS/lock-in and FEI of SG tubes will need to be evaluated with in-plant instrumentation during initial startup testing.

TSD-T050-3424 describes flow restrictor testing plans, but the plans seem limited to the original three design concepts. The final flow restrictor design differs slightly from those tested, and a new test is planned (but not described in TSD-T050-3424). An updated plan is expected in mid-2018. Until the test plan and final restrictor design is provided to the NRC, the risk of LFI for the SG inlet flow restrictors remains unknown, and cannot therefore be considered low risk.

TSD-T070-8245, Rev. 6 is a revised test plan for the CRA which originally included the CRAGT. Rev. 6 removes previously planned CRAGT FIV testing. However, the CRAGT is part of the test hardware, and prototypic flows up to [ ] of nominal will be generated throughout the assembly for three CRA heights. Flow rates will be measured, but flow through the CRAGT holes seems to be an optional measurement. Vibration of the CRA rodlets will be measured. Since there is already significant instrumentation planned for this test, vibration and pressure measurements should be made on and around the CRAGT. The staff recommends that NuScale reinstate CRAGT testing given the significant uncertainty in forcing function definition, and the currently estimated [ ] wall thickness reduction due to TB forces.

The SIET TF1 and TF2 testing documents were evaluated on-site at the NuScale Rockville office. Two NuScale SG testing document sets were examined: one on SIET-TF1 testing (electrically heated individual tubes) and the other on SIET-TF2 (flow-heated tube bundles) testing. The SG separate effects tests were performed by SIET labs.

The NuScale SG is comprised of several columns of helical tubes in the outer annulus between the riser and pressure vessel wall. The primary coolant flows downward over the tubes, and the secondary coolant flows upward inside the tubes. The secondary coolant enters the SG tubes near the bottom of the reactor as subcooled water and exits near the top as superheated steam. This means that the intermediate regions are filled with two-phase flow with continuously varying void fractions. The TF1 tests measured internal wall pressures due to internal secondary flow, and the TF2 tests measured tube vibration due to external primary flow.

Data plots are available in the TF1 test report. Raw data in the form of Microsoft Excel files were provided for the TF2 tests, since no final data report was written by SIET. NuScale has also not yet provided a TF2 data report. The TF1 testing simulated secondary flow within three prototypic SG tubes which were electrically heated through externally applied current. The inlet fluid was preheated to a specified quality and forced upward through the SG tubes with (diabatic testing) and without (adiabatic testing) external heating. Pressures and temperatures were measured along the tubes to assess thermal-hydraulic and fluid-dynamic behavior. Mass flow

rates in individual tubes were also monitored to assess density wave oscillation (DWO) phenomena.

The TF2 testing was more complex, with pre-heated primary fluid forced past five SG tube columns. Various primary flow rates, temperatures and pressures were evaluated, along with varying secondary flow conditions (air, water, boiling water, quiescent, and flowing). Tube vibrations were measured through sets of strain gages installed at the inlets and discharges. Although NuScale does not advertise the SIET TF1 and TF2 tests as suitable for benchmarking, there is significant good quality data in both tests that may be used to assess the bias and uncertainty of various components of NuScale's FIV assessment procedure. NuScale should be encouraged to expand the use of these data to support the appropriateness of their FIV evaluation tools.

3.1.3.1 TF1 tests: NuScale EP-0703-1417-F01, Rev. 1, 1232 pages, Christhian Galvez

This report describes the TF1 (Electrically Heated Facility – EHF) testing of three SG tubes filled with secondary fluid at various operating conditions. The report includes an internal memo (ME-0516-49448 by C. Galvez) discussing, "Apparent discrepancies in SIET TF1 data and documentation," based on evaluation of CR-1115-19000. These discrepancies were associated with the pre-heating system and do not appear to affect the conclusions or FIV data. Analyzed data are plotted for all test conditions in the report.

Along with a few preliminary system characterization tests, the following tests were conducted, which include pressure fluctuation measurements at several locations along the tubes:

- EHF Adiabatic tests (no electrical tube heating): TA0001-TA0077.
- EHF Diabatic tests (with electrical tube heating): TD0001-TD0084.
- EHF DWO tests with variable input powers (several per test): TO0001_Initial_power – TO0036_Initial_power.

For adiabatic tests, fluid conditions are (mostly) constant along the tubes, so that void fraction may be easily estimated from various test data. There is some energy loss due to friction and conduction to the exterior. Adiabatic testing is primarily meant to measure pressure drops in the secondary fluid along tubes at various operating conditions. The adiabatic test temperature drop along tube is small – about [ ] (thermodynamic quality and enthalpy presumably have similar drops). Pressure drops are higher (most likely due to friction losses and gravity) – [ ] for the highest mass flows (pressure drops are smaller for lower mass flows, indicating most of the losses are due to drag). [.] The higher the quality, the more heated the inlet flow. Negative quality indicates subcooled inlet water. There is no formal definition of 'thermodynamic quality' to explain negative values (most of the two-phase flow literature defines quality in terms of percent vapor, ranging between 0 and 100 percent).

For diabatic testing, enthalpy and quality increase as the flow moves upward along the tubes which is more prototypic of actual operating conditions. Inlet fluid temperature is set, and temperature along the tube is increased to specified levels via electrical current. Temperature

settings seem constant along the tubes, but enthalpy increases gradually as heat transfer occurs. Pressure drops are similar to those for adiabatic testing.
Two types of dynamic pressure measurements were made – one with single pressure taps and the other with differential pressure (DP) taps.

There are three strong spectral peaks in the pressure spectra measured in both the adiabatic and diabatic tests. Very low frequency (0-2 Hz) peaks are due to a hydraulic circuit frequency in the test facility and are not relevant to loads in the actual SG. The second and third peaks are claimed to be due to boiling water (two-phase flow), and only appear for cases with saturated water/steam. The frequencies of the second peaks range between [ ], and appear in the DP sensors. The frequencies of the third peaks range from about [ ], and appear only in the P sensors. In general, higher enthalpy leads to stronger [ ] pressures. It is unclear why the frequencies of the boiling water peak pressure fluctuations are different in the P and DP sensors.

For the adiabatic testing the [ ] peak "dampens" along the tube due to slight thermodynamic quality decreasing caused by heat losses to the environment (also head loss due to drag). For diabatic testing the peaks are relatively constant along the tubes since enthalpy increases from the electric heating. The peak frequency seems to shift by about [ ] from sensor to sensor. This may be due to sensors at different relative locations around the tube circumference with respect to the flow profile. Diabatic test condition TD017 is meant to be most representative of operating conditions. Inlet quality is [ ] and rises to [ ] at the end of the tube due to the electric heating. Peak pressure fluctuations are about [ ]. Based on the tube diameter and wall thickness, lobar shell modes (modes with significant deformation around the circumference) initiate, or "cut on" above [ ], so that the [ ] pressure peaks should only excite beam-like bending modes of the SG tubes. Since the first bending modes appear in the SG FE calculations at around [ ], there will be many bending modes near the [ ] peak pressures that will be driven by these loads and should be accounted for. Generally, however, high order modes do not experience significant deformation and strain. Nevertheless, NuScale should perform this evaluation and include the resulting deformations and strains in its degradation evaluations.

Special tests investigated DWO between parallel tubes (2 and 3) at low pressure (about half nominal) since, "DWO are more likely to occur with the reactor depressurized during the decay heat removal phase." DWO tests were therefore not at prototypic conditions. Tests were conducted at very low flow, half flow, nominal flow, and about twice nominal flow rates over a wide range of inlet qualities, with tube heating activated. The flow rate data are uncertain due to high sensitivity of the needle valve used to restrict inlet flow (data quality is also affected due to signal to noise issues).

The DWO mass flow fluctuations occur for all flow rates at certain tube heating levels and are extremely low in frequency [ ]. The tests at a given flow/pressure condition were stopped once DWO was observed (heating was increased until DWO initiated). The oscillations are coupled between tubes, where flow rate increases in one tube while it decreases in the other. Flow rate can become slightly negative in one tube, while increasing to 2 times or higher in the other tube. SIET states that the DWO tests may have been affected by an anomaly in TF1 where the vertical discharge pipes downstream of the steam plenum were not fully insulated, allowing condensation and water buildup that intermittently discharged. However, this should not affect the clean sinusoidal nature of the observed DWO. The NRC staff concurs with NuScale's recent assertions that tube-to-tube DWO need not be considered in NuScale's CVAP

since the oscillation frequencies are so low. TF2 testing further confirms this assertion. Also, NuScale claims that the inlet flow restrictors have been redesigned to mitigate DWO.

3.1.3.2 TF2 tests: SIET Helical Coil Steam Generator Test Program – Fluid-Heated Facility Final Data Report, 02511 RA 15, R. Ferri, 58 pages

TF2 testing primary objectives were thermal hydraulic in nature. Therefore, the FEI and DWO tests may have been non-prototypic, mainly due to non-prototypic tolerances on tube supports (see below). However, many acceptable FEI and DWO tests were run spanning a range of operating conditions and flow speeds.

A tube bundle of 252 helical coils simulating the 5 (out of 21 in a full SG) innermost columns with multiple tubes in a column of the SG was tested. The five tube columns are in an annulus formed by two cylindrical barrels and supported at 90 degree increments. Each tube column is fed by a feedwater heater. Outgoing steam is collected in a steam header and driven outside the vessel top nozzle by pipes. The five columns can be fed individually or together. The tubes are provided with inlet orifices to damp DWO.

A “deviations list” is included in the short TF2 test report, including a discussion that tube support fabrication tolerances were not held to manufacturing specifications for TF2 tests, perhaps leading to longer unsupported tube sections. Also, there were physical limitations on the flow, temperature, and pressure.

Unlike TF1 tests, there was no data analysis by SIET for TF2. Instead, the raw data were provided, with NuScale engineers responsible for subsequent analysis. No analysis report has been provided yet. NuScale claims an analysis report, discussing whether TF2 tests are useful for FIV, will be provided in 2018.

Unlike the TF1 test, the TF2 test was a “fluid-heated” test, with forced flow via pumps. Differential pressure sensors, thermocouples, and a few strain gages (12) were installed on three tubes

Several TF2 tests were performed:

- Adiabatic – no flow through the SG, intended to measure primary flow pressure drops over SG array.

- Diabatic – includes primary flow and secondary tube flow, measured heat transfer.

- Transients – ramp up, ramp down

- Density Wave Oscillation – identification of conditions leading to oscillations in parallel channels caused by local density variations at tube outlets. Sensors DP2301-2306 best reveal DWO, which occurs for many test conditions. Time histories are coarsely sampled ([ ] increment), but show the oscillations clearly. In TF2, the oscillation rates are almost half that of those in TF1 [ ], further confirming that tube-to-tube DWO should not be an FIV-related issue.

- FEI with diabatic (with) and adiabatic (without) tube flow. Tested with tubes empty, full of cold water, full of hot water, and full of boiling water. Strain gage signals are the only data acquired at high sampling rate [ ].

The FEI tests included several conditions. Several Microsoft/Excel strain gage spectra files were reviewed to identify low frequency tube resonances excited by the primary and/or secondary flow. Several strong response peaks are evident in the tube strain spectra in all tubes and locations at frequencies between [ ], particularly for boiling water flow at high flow rates. These peak response frequencies are similar to resonance frequencies estimated for the actual SG design by NuScale structural finite element models. In contrast, the maximum primary flow with empty tube conditions showed only one weakly excited resonance. It is clear from the SIET TF2 data that it is not reasonable to perform any future SG FIV tests with empty tubes. Future SG tube testing therefore should include secondary boiling water flow.

A future "SG-FIV" test is planned with more extensive FIV-related instrumentation on tubes and supports. Testing is targeted for late 2018. Only primary flow tests are planned, with empty tubes. Filled tubes are being considered, but NuScale is concerned that some of the instrumentation is invasive, and could lead to gas pockets within the tubes. The test specification has been updated (but is high level only). Pre-test predictions with more detail are expected in mid 2018, which should include a justification of empty tube testing. NuScale is also planning to use damping measurements to justify their use of 1.5 percent damping in their SG tube FEI and VS analyses. However, the presence of instrumentation and wiring will increase damping and affect these measurements, along with any VS and FEI assessments (which depend strongly on damping). NuScale should address how they will account for instrumentation and wiring effects.

3.1.4 Summary of major concerns

The NRC staff will focus RAIs associated with FIV on the major issues discovered during the audit:

- moderate to high risk of SG tubing VS/lock-in and FEI,
- moderate to high risk of ICIGT and CRDS VS/lock-in,
- moderate risk of AR in DHRS side branches,
- unknown risk of FIV mechanisms on CRAGT, along with NuScale's estimate that [,] of the wall thickness will be worn away over the life of the plant,
- inadequate docketing of the evaluation of these mechanisms in the current CVAP report, and
- inadequate separate effects and initial startup testing to address these risk areas.

To reduce the risk of NuScale's planned testing not resolving staff concerns, the NRC staff recommends continued and frequent interaction between the staff and NuScale engineers during their ongoing test planning. This may include site visits where necessary.

3.2 Reactor Internals Service Level D Analysis

3.2.1 Assumption in the Analysis Requiring Later Verification

Various types of assumption are made in the analyses. For assumptions requiring later verification, tracking numbers are assigned by the applicant. Some of these assumptions are listed below, and the NRC staff may confirm the closure of these assumptions in follow-up audits.

EC-A010-2322, Rev. 2, "Reactor Module Seismic Model"		
Assumption	Description	Tracking ID
SG Secondary Side Liquid	It is assumed that 13.67 percent of secondary side volume is liquid and the remaining is vapor in operating condition.	ODI-5-0210
Belleville Washers at Core Support	Four Belleville washers are assumed to exist at bottom of core support with spring constant of [ ] for each washer. The washers shift the seismic response frequency of the core.	ODI-17-0002

EC-A010-3559, Rev. 3, "Reactor Module Seismic Calculation"		
Assumption	Description	Tracking ID
Bounding Case for NPM Stiffness Adjustment	The NPM stiffness is adjusted to 77 percent of nominal stiffness of Soil Type 7 and cracked concrete condition. It is assumed that this analysis case bounds all other cases with respect to NPM stiffness adjustment.	ODI-17-0001

EC-A011-2278, Rev. 0, "RPV Primary Stress Calculation"		
Assumption	Description	Tracking ID
Nozzle Loads (excluding RPV 83)	The RPV nozzle loads (exclude RPV nozzle 83) are not finalized. Preliminary loads are used in the calculation.	ODI-16-0221
RVV (Reactor Ventilation Valve) Nozzle (RPV 83)	The RPV 83 nozzle loads (the third reactor vent valve on the reactor top head) are not generated yet. It is assumed that these loads will be bounded by other two reactor vent nozzle loads (RPV 16 and RPV 17).	ODI-16-1081
SG Blowdown and SSE Loads	Loads are currently not provided for a SG tube at the tube sheet in the blowdown and SSE loads specifications. A calculation using assumed blowdown and SSE loads is performed for a SG tube resulting a Level D primary stress of 0.90 of the allowable stress. The assumed loads will be compared to the blowdown and SSE loads for bounding. Reevaluation is necessary if the assumed loads are smaller than the blowdown and SSE loads.	ODI-16-1082
Stress Classification Lines	The determination of appropriate stress classification lines for evaluation of linearized stress results is based on engineering judgement. The stress classification lines will be verified.	ODI-16-1087

3.2.2 RPV Primary Stress Calculation

The load combination of RPV Level D primary stress calculation involves safe-shutdown earthquake (SSE) load that is combined with the largest load among design basis pipe break (DBPB), main steam pipe break (MSPB), and feedwater pipe break (FWPB) using the square root of the sum of squares (SRSS) method. The NRC staff noticed that the loads of MSPB and FWPB are missing in the load combination. The applicant stated during the audit that blowdown analysis of MSPB and FWPB has not been performed. The analysis will be performed in the second quarter of 2018. When the loads are available, a confirmatory analysis will be performed to demonstrate that the DBPB load bounds the MSPB and FWPB loads and that the results of current calculation are valid. If not, the RPV primary stress calculation will need to be reevaluated. The applicant also stated that when the loads are available they will perform another RPV Level D stress calculation in accordance with the load combination as stated in DCD Tier 2, Table 3.9-3, "Required Load Combinations for Reactor Pressure Vessel American Society of Mechanical Engineers Stress Analysis," without the SSE load involved.

The interface and support loads at various RPV cross sections and supports are generated for both certified seismic design response spectra (CSDRS) and Generic High Frequency Hard-Rock Response Spectra (GHFHRRS). The larger seismic loads between the two cases are combined with the DBPB load. The NRC staff determined that the approach is conservative because the analyses considered both the CSDRS and GHFHRRS inputs. The NRC staff verified that the SSE loads are combined with the DBPB loads using SRSS method.

The RPV interface and support loads are based on the ER-A010-2867, Rev. 2, "Reactor Module (RXM) Seismic Load Specification," ER-A010-3616, Rev. 1, "RXM Blowdown Load Specification," and ER-A010-3625, Rev. A, "RXM Nozzle Load Specification." These three load specifications specify all interface and support loads for NPM component stress calculation. The RXM Nozzle Load Specification is still in Rev. A stage and is not available for staff's review during the audit. The RXM Seismic Load Specification is being update and is also not available for staff's review. These specifications will need to be completed before the safety finding can be made, and the staff may audit these documents during the later phase of the DC review.

The RPV_support_interface_loads.pdf file contains the RPV interface and support loads from the inputs of CSDRS and GHFHRRS (equivalent to the term CSDRS-HF {CSDRS-High Frequency} used in TR-0916-51502-P, Rev. 0, "NuScale Power Module Seismic Analysis"). The NRC staff noticed that in some cases, the GHFHRRS interface loads are higher than those of CSDRS. TR-0916-51502-P, Rev. 0 only contains NPM interface and support loads from the CSDRS input. The report may need to be updated to include the NPM interface and support loads from the GHFHRRS (CSDRS-HF) input.

The following load combination is used in the Level D stress calculation:

P + DW + B + EXT +/- SRSS(SSE + DBPB)

It is not clear what loads are associated with the EXT loads, locations of the loads, and values of the loads. The staff will issue an RAI to request the information.

3.2.3 RVI Components and Core Support Structures

The fundamental frequencies of the RVI components and core support structures are shown in Table 1. Most of the RVI components and core support structures have relatively high fundamental frequency [ ]. The CRDS, Lower and Upper ICIGT, and upper riser have fundamental frequency below [ ] Without considering the fluid gap, the core barrel and reflector have fundamental frequency of [ ], respectively. With fluid gap considered in the analysis, the frequencies of the first five modes of the core barrel-fluid gap-reflector coupled system are [ ] (see Note 2 in Table 1). The NRC staff believes that the fluid gap between the core barrel and reflector should be considered in the seismic analysis of the core barrel and reflector. The NRC staff will issue an RAI for this issue.

Table1. Fundamental Frequencies of RVI Components and Core Support Structures

Component	Fundamental Frequency (Hz) (1)	Mode Shape
Core barrel	[ ]	[ ]
CRA card	[ ]	[ ]
CRAGT	[ ]	[ ]
CRAGT support grid	[ ]	[ ]
CRDS (control rod draft shaft)	[ ]	[ ]
CRDS support	[ ]	[ ]
Flow diverter	[ ]	[ ]
Lower ICIGT	[ ]	[ ]
Upper ICIGT	[ ]	[ ]
Injection line, riser	[ ]	[ ]
Injection line down comer	[ ]	[ ]
Lower core plate	[ ]	[ ]
Upper core plate	[ ]	[ ]
Lower riser	[ ]	[ ]
Spray nozzle	[ ]	[ ]
Reflector	[ ]	[ ]
Upper core support	[ ]	[ ]
Upper riser hanger support	[ ]	[ ]
Upper riser	[ ]	[ ]

(1) Data from EC-A023-3535, Rev. 0, "RVI Turbulent Buffeting Degradation Evaluation" except as noted.

(2) Does not consider the narrow fluid gap [ ] between the core barrel and reflector. If the fluid gap is considered, the frequencies of the first five modes of the core barrel-fluid gap-reflector coupled system are [ ] based on ER-A010-2157, Rev.0, "Methodology Development for Hydrodynamic Effect Evaluation for Reflector and Core Barrel."

(3) Data from EC-A023-2160, Rev. 1, "RVI and SG vortex shedding evaluations."

The vertical spectral acceleration at high frequency end could reach to [ ] at core support plate. The staff determines that uplift may be a concern for the RVI components not restrained in the vertical direction, and will issue an RAI.

3.2.4 Diametric Gaps of Lower ICIGT, Upper ICIGT, CRDS, and CRAGT

The diametric gaps of lower ICIGT, upper ICIGT, CRDS, and CRAGT are shown in Table 2. The lower ICIGT and upper ICIGT have diametric gap size of [ ]. CRDS and CRAGT have diametric gap size of [ ]. The assumption of fixed boundary condition in seismic analysis may need verification.

Table 2. Diametric Gap of Lower ICIGT, Upper ICIGT, CRDS, and CRAGT

	Lower ICIGT	Upper ICIGT	CRDS	CRAGT
Diametric Gap (in) (1)	[ ]	[ ]	[ ]	[ ]

(1) Data from Table 4-2 of ER-A010-4079, Rev. 1, "Comprehensive Vibration Assessment Program Measurement and Inspection Report."

3.2.5 Reactor module Seismic Calculation

In EC-A010-3559, it is assumed (ODI-17-0001) that adjustment of NPM stiffness to 77 percent of nominal stiffness will bound all other cases with respect to NPM stiffness adjustment in the 3D NPM full model. The applicant will verify the assumption later. If not bounded, all the seismic interface and support loads generated from the 3D NPM full model need to be recalculated and the RPV primary stress calculation needs to be reevaluated.

3.2.6 SG Structural Model

EC-A014-3306, Rev. 2, "Steam Generator Structural Model," indicates that the major modes of the SG/riser/vessel model are [

]. The report also indicates that a single helical SG tube has frequencies of the [ ] depending on location of the tube. Both cases indicate that the SG assembly is not flexible. However, the stiffness of the steam generator is not considered in the NPM full model. The applicant may need to perform a confirmative analysis to demonstrate that ignoring SG stiffness has no significant impact on the results presented in TR-0916-51502-P, Rev.0. Alternatively, the applicant may need to consider incorporating a simplified SG model in the NPM full model, similar to the simplified fuel assembly beam-spring model incorporated in the NPM full model. The NRC staff will issue an RAI.

U.S. NUCLEAR REGULATORY COMMISSION

SUMMARY AUDIT REPORT OF NUSCALE POWER, LLC, REACTOR INTERNALS

COMPREHENSIVE VIBRATION ASSESSMENT PROGRAM

AND SEISMIC ANALYSIS

LIST OF ATTENDEES

May 16, 2017 – November 2, 2017

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