July 31, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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11555 Rockville Pike
Rockville, MD 20852-2738

**SUBJECT:** NuScale Power, LLC Submittal of "NuScale Comprehensive Vibration Assessment Program Analysis Technical Report," TR-0716-50439, Revision 2

**REFERENCES:**

1. Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, "Submittal of Technical Reports Supporting the NuScale Design Certification Application," dated December 30, 2016 (ML17005A112)
2. NuScale Technical Report, "NuScale Comprehensive Vibration Assessment Program," Revision 0, TR-0716-50439, dated December 2016 (ML17005A122)
3. Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, Submittal of "NuScale Comprehensive Vibration Assessment Program Technical Report," Revision 1, TR-0716-50439, dated January 20, 2018 (ML18022A221)

NuScale Power, LLC (NuScale) hereby submits Revision 2 of the "NuScale Comprehensive Vibration Assessment Program Analysis Technical Report" (TR-0716-50439).

Enclosure 1 contains the proprietary version of the report titled "NuScale Comprehensive Vibration Assessment Program Analysis Technical Report," TR-0716-50439, Revision 2. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 1 has also been determined to contain Export Controlled Information. This information must be protected from disclosure per the requirements of 10 CFR § 810. Enclosure 2 contains the nonproprietary version of the report titled "NuScale Comprehensive Vibration Assessment Program Analysis Technical Report," TR-0716-50439, Revision 2.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Samuel Lee, NRC, OWFN-8H12
Gregory Cranston, NRC, OWFN-8H12
Marieliz Vera, NRC, OWFN-8H12

Enclosure 1: "NuScale Comprehensive Vibration Assessment Program Analysis Technical Report," TR-0716-50439-P, Revision 2 proprietary version
Enclosure 2: "NuScale Comprehensive Vibration Assessment Program Analysis Technical Report," TR-0716-50439-NP, Revision 2 nonproprietary version
Enclosure 3: Affidavit of Zackary W. Rad, AF-0719-66160

**Enclosure 1:**

“NuScale Comprehensive Vibration Assessment Program Analysis Technical Report,” TR-0716-50439-P, Revision 2 proprietary version

**Enclosure 2:**

"NuScale Comprehensive Vibration Assessment Program Analysis Technical Report," TR-0716-50439-NP, Revision 2 nonproprietary version

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Licensing Technical Report

# NuScale Comprehensive Vibration Assessment Program Analysis Technical Report

July 2019

Revision 2

Docket: 52-048

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**Department of Energy Acknowledgement and Disclaimer**

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**Abstract**

This report describes the Comprehensive Vibration Assessment Program (CVAP) for the NuScale Power Module (NPM) that verifies the structural integrity of the internals for flow induced vibration. The CVAP conforms to the guidance of NRC Regulatory Guide 1.20 (Reference 8.1.1). The content of this licensing technical report provides additional information to substantiate the statements made in the NuScale Design Control Document, thereby facilitating a comprehensive review by the NRC of the NPM design.

## Executive Summary

A Comprehensive Vibration Assessment Program (CVAP) for the NuScale Power Module (NPM) is established in accordance with the NRC Regulatory Guide (RG) 1.20 (Reference 8.1.1).

The CVAP ensures that the structural components of the NPM exposed to fluid flow are precluded from the detrimental effects of flow induced vibration (FIV).

The NPM represents a unique design in its size, arrangement, and operating conditions, although its technology is based on well-proven light water reactor designs with long operational experience. Accordingly, the first operational NPM is classified as a prototype in accordance with RG 1.20. After the first NPM is qualified as a valid prototype, subsequent NPMs will be classified as non-prototype Category I.

Given its prototype classification, the NuScale CVAP addresses the applicable criteria of RG 1.20, Section 2.

The NPM differs from other light water reactor designs in that it is a small modular reactor that is an integral, self-contained, movable nuclear steam supply system that can be installed individually or in a series of up to 12 units at a power station. The NPM design is passive, with primary coolant driven by natural circulation flow. Natural circulation flow velocities are very low, thereby decreasing the propensity for detrimental FIV effects. The CVAP establishes the scope of analyses, testing, and inspections required to ensure that components of the NPM are not subject to unacceptable vibratory degradation. When completed, the NuScale CVAP provides the requisite assurance that the NPM components are not subject to detrimental FIV.

# 1.0 Introduction

## 1.1 Purpose and Scope

This report describes the CVAP for the NPM to verify the structural integrity of the components to FIV. The CVAP conforms to the guidance of NRC RG 1.20 (Reference 8.1.1). This program is required for all new nuclear power plants. The CVAP includes the collection of analysis, testing, and inspection that demonstrates a sufficient margin of safety and structural integrity against the detrimental effects of FIV for components in the NPM.

In defining the scope of the CVAP, the NRC references the definition of reactor core support and internal structures inside the reactor vessel provided by the Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NG, of the American Society of Mechanical Engineers (ASME) (Reference 8.1.2). The NuScale reactor vessel internals (RVI), including the steam generator (SG) tube supports, are designed to Subsection NG. Components that make up the primary and secondary coolant pressure boundaries of the NPM are designed to Section III, Division 1, Subsection NB and are included in the scope of the CVAP, because they are exposed to primary and secondary coolant flows.

Based on the integral design of the NuScale NPM, the SG components are located within the fluid volume of the reactor pressure vessel (RPV), along with the RVI and the pressurizer. Likewise, RG 1.20 includes references to evaluation of steam dryers, steam system components, and SG internal components as part of a CVAP. Regulatory Guide 1.20 does not discuss the need to evaluate the vibration characteristics of the fuel components, which include both the fuel bundles and the control rod assemblies (CRAs). Based on these considerations, there are three focus areas within the NPM that are included within the scope of the CVAP: reactor vessel internals, steam generators, and piping.

To finalize the CVAP, two additional technical reports are developed and provided to the Nuclear Regulatory Commission. The first report (Reference 8.1.14) contains the measurement program details for each prototype test, including test operating conditions, test durations, instrument types and locations, applicable testing hold points, and pre-test predictions of the expected and allowable experimental results, considering bias errors and random uncertainties. The second report provides the post-test evaluation of the testing completed to support the measurement program. In the second report, the differences between the expected and measured experimental results are either resolved or confirmed to be in the analytically-predicted allowable ranges. The second report also documents the inspection program results.

## 1.2 Abbreviations

Table 1-1 Abbreviations

| Term | Definition |
|---|---|
| AR | acoustic resonance |
| ASME | American Society of Mechanical Engineers |
| CFD | computational fluid dynamics |
| CNTS | containment system |
| CNV | containment vessel |
| CRA | control rod assembly |
| CRAGT | control rod assembly guide tube |
| CRD | control rod drive |
| CVAP | Comprehensive Vibration Assessment Program |
| CVCS | chemical and volume control system |
| DHRS | decay heat removal system |
| FEI | fluid elastic instability |
| F/G | flutter/gallop |
| FIV | flow-induced vibration |
| FW | feedwater |
| ICIGT | in-core instrument guide tube |
| LFI | leakage flow instability |
| MS | main steam |
| MSIV | main steam isolation valve |
| NPM | NuScale Power Module |
| PSD | power spectral density |
| PWR | pressurized water reactor |
| RCS | reactor coolant system |
| RG | Regulatory Guide |
| RMS | root mean square |
| RPV | reactor pressure vessel |
| RRV | reactor recirculation valve |
| RVI | reactor vessel internals |
| RVV | reactor vent valve |
| RXC | reactor core |
| SG | steam generator |
| SGS | steam generator system |
| TB | turbulent buffeting |

| Term | Definition |
|---|---|
| TH | thermal-hydraulic |
| VS | vortex shedding |

Table 1-2 Definitions

| Term | Definition |
|---|---|
| Acoustic resonance | A phenomenon where an acoustic wave is generated at a frequency that coincides with the natural frequency of a confining structure. |
| Initial startup testing | Testing conducted on the prototype after fuel loading. |
| Fixed boundary condition | For the single SG tube “fixed” boundary condition modal analysis, the local UX, UY, UZ, ROTY, and ROTZ are constrained at the support, but twisting about the tube axis is left free. |
| Fluid elastic instability | Instability that arises when, during one vibration cycle, the energy absorbed from the fluid exceeds the energy dissipated by damping. This phenomenon is associated with arrays of closely packed circular cylinders. |
| Flutter/gallop | The phenomenon in which drag and lift forces due to fluid flow act on a bluff body with a non-circular cross section. If the bluff body vibrates, the draft and lift forces change due to the change in flow angle, which can increase the vibration amplitude. In this phenomenon, the amount of energy dissipated by damping is less than the energy imparted on the structure by the fluid flow. |
| Leakage flow instability | A condition in which fluid flow through a thin space with at least one flexible structural boundary results in vibration of the flexible boundary due to the negative fluid damping becoming larger than the total fluid damping. |
| Prototype | A configuration of RVI that, because of its arrangement, design, size, or operating conditions, represents a unique design for which no valid example exists. |
| Prototype testing | Testing that is used to validate analysis inputs, methods, and margins of safety for components susceptible to FIV phenomena. This testing consists of separate effects and initial startup testing. Testing is required to be performed using a full-scale, prototypic arrangement of the region of interest. Separate effects testing is performed at a test facility. Initial startup testing is performed on the first NPM after fuel loading. |
| Safety margin | For strongly-coupled FIV mechanisms, the percentage difference between the analytically predicted value and the acceptance criteria that represents the predicted onset of the mechanism for a component. For turbulent buffeting, safety margins are evaluated for the analytically-predicted fatigue against the limits that are acceptable over the component design life. |
| Separate effects testing | Testing performed on a prototypic portion of the NPM in a test facility. |
| Sliding boundary condition | For the single SG tube “sliding” boundary condition modal analysis, the local UY, UZ, ROTY, and ROTZ are constrained, but twisting about the tube axis is left free, as is displacement along the tube axis. |
| Turbulent buffeting | A weak coupling between a structure and the random pressure fluctuations induced by turbulent flow. The effects of low amplitude structural vibrations induced from turbulence are evaluated to assess impact and fatigue. |

| Term | Definition |
|---|---|
| Valid prototype | A configuration of RVI that has successfully completed a comprehensive vibration assessment program for prototype RVI and has experienced no adverse in-service vibration phenomena. |
| Vortex shedding | A phenomenon due to flow separation on the surface of a bluff body located in the flow field, leading to the shedding of vortices at locations of flow separation. Due to interaction between the vortices, time varying forces are generated that act on the body. |

## 2.0 NuScale Power Module Design Overview for Flow Induced Vibration

The NPM is an integral, self-contained, movable nuclear steam supply system. It can be used individually at a power facility to generate 50 MWe or installed in a series of up to 12 NPMs to generate up to 600 MWe. The NPM includes various hydraulic systems, components, and structures that are relevant to the CVAP. These include the containment system (CNTS), the reactor coolant system (RCS), the control rod drive system, the reactor core (RXC), the steam generator system (SGS), the in-core instrumentation system, the decay heat removal system (DHRS), and the emergency core cooling system. The NPM includes the containment vessel (CNV) and an integral RPV, which includes the pressurizer and the SGs, and is located inside the CNV. During normal operation, the NPM is located in an operating bay in the Reactor Building (RXB) and is partially immersed in the reactor pool.

Figure 2-1 depicts the NPMs located in respective operating bays. The major NPM components are depicted in Figure 2-2 and described in the following subsections.

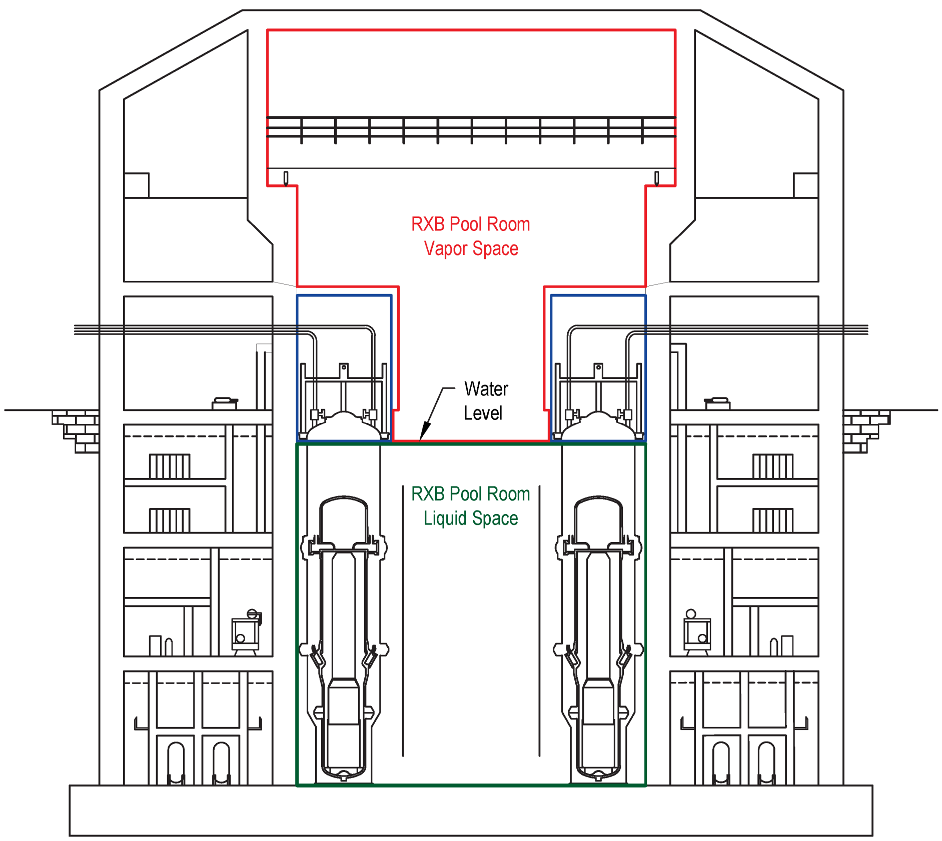


Figure 2-1 NuScale Power Modules located in respective operating bays

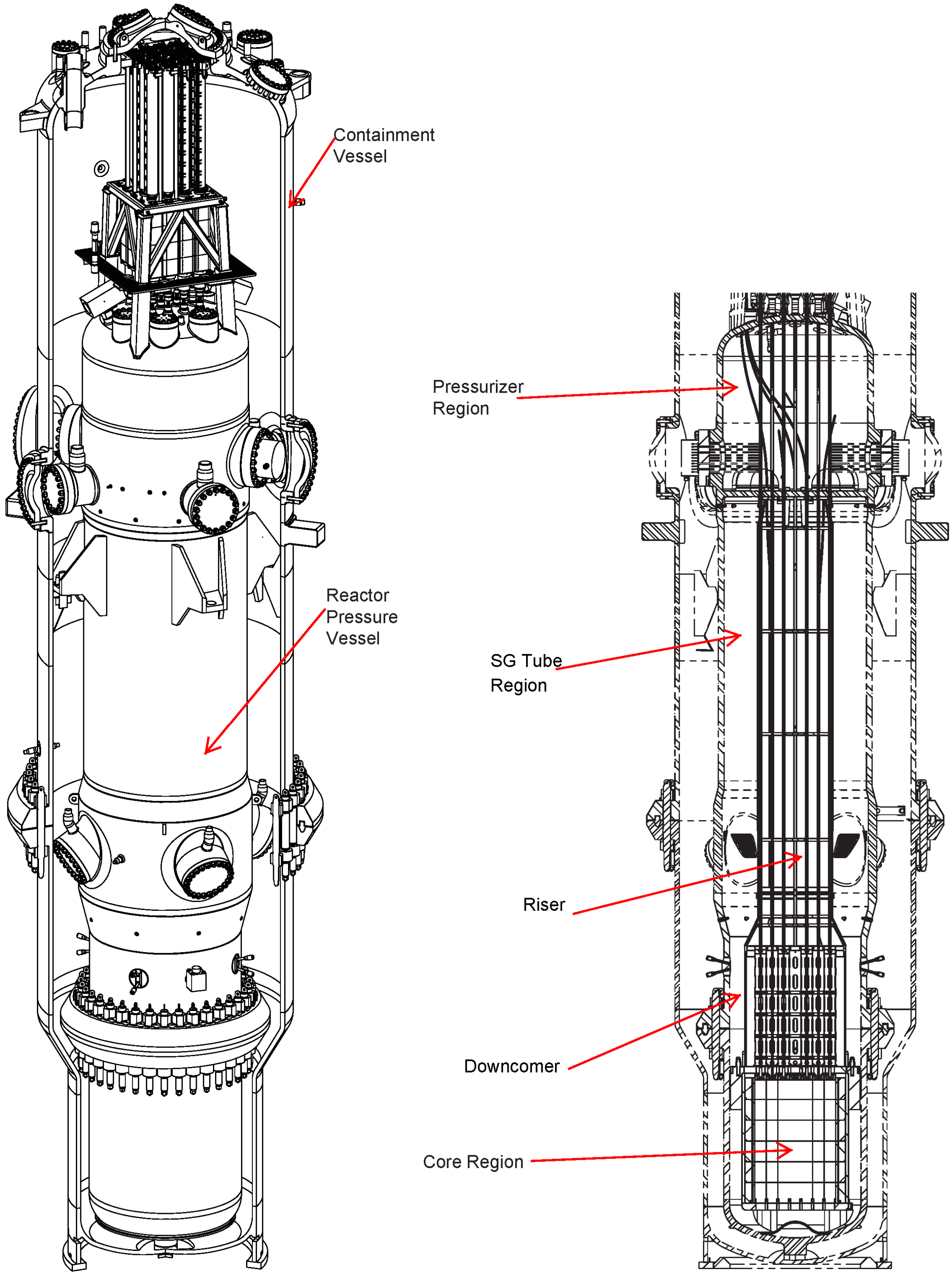


Figure 2-2 NuScale Power Module general arrangement

## 2.1 Primary Coolant Flow Conditions

The RCS is a passive system driven by natural circulation flow that relies on interfacing active systems for operational control. During operation at power, the reactor coolant flows upward through the RXC where it removes heat from the fuel assemblies. The heated reactor coolant exits the RXC and continues to flow upward through the central riser (composed of the lower and upper riser assemblies). At the top of the upper riser, the flow is turned by the pressurizer baffle plate to flow downward through the annular space between the upper riser and the RPV. This annular space between the upper riser and the RPV contains the SG helical tube bundles. As the reactor coolant flows downward across the SG helical tube bundles, it transfers heat to the secondary side coolant. The colder reactor coolant leaving the SG helical tube bundles continues to flow downward through the annular space between the core barrel and the RPV. As flow passes the bottom of the core barrel, the flow is turned upward by the RPV lower head and flow diverter, and is returned to the RXC. The motive force for the reactor coolant flow during operation at power is natural convection, driven by the difference in coolant density between the hot coolant leaving the RXC and the colder coolant leaving the SGs, and the elevation difference between the RXC (heat source) and the SGs (heat sink).

The NPM reactor coolant flow has the following operational characteristics:

- Given that there are no RCS pumps, primary coolant flow is by natural circulation. Flow is limited by thermal driving head, with no pump overspeed conditions, no excess flow in transients, and no modulating pressure excitation due to vane passing frequencies.
- Primary and secondary flows are mostly axial, with no RPV hot and cold leg nozzles impinging flow on the core support assembly. The low-velocity axial flows result in low turbulent sources for FIV.
- Primary single-phase flow is on the outside of the SG tubes with low velocity. Flow velocity is approximately an order of magnitude less than pressurized water reactor (PWR) designs with secondary-side two-phase flow on the inside of tubes.

A schematic of the primary coolant flow path is shown in Figure 2-3. Table 2-1 summarizes the NuScale reactor coolant flows in relation to other PWR applications. In the SG tube region, only a simplified depiction of the flow is provided. The primary-side flow occurs in the downward direction, across the columns of helical SG tubes. The secondary-side flow travels in the upward direction inside the helical SG tubes.

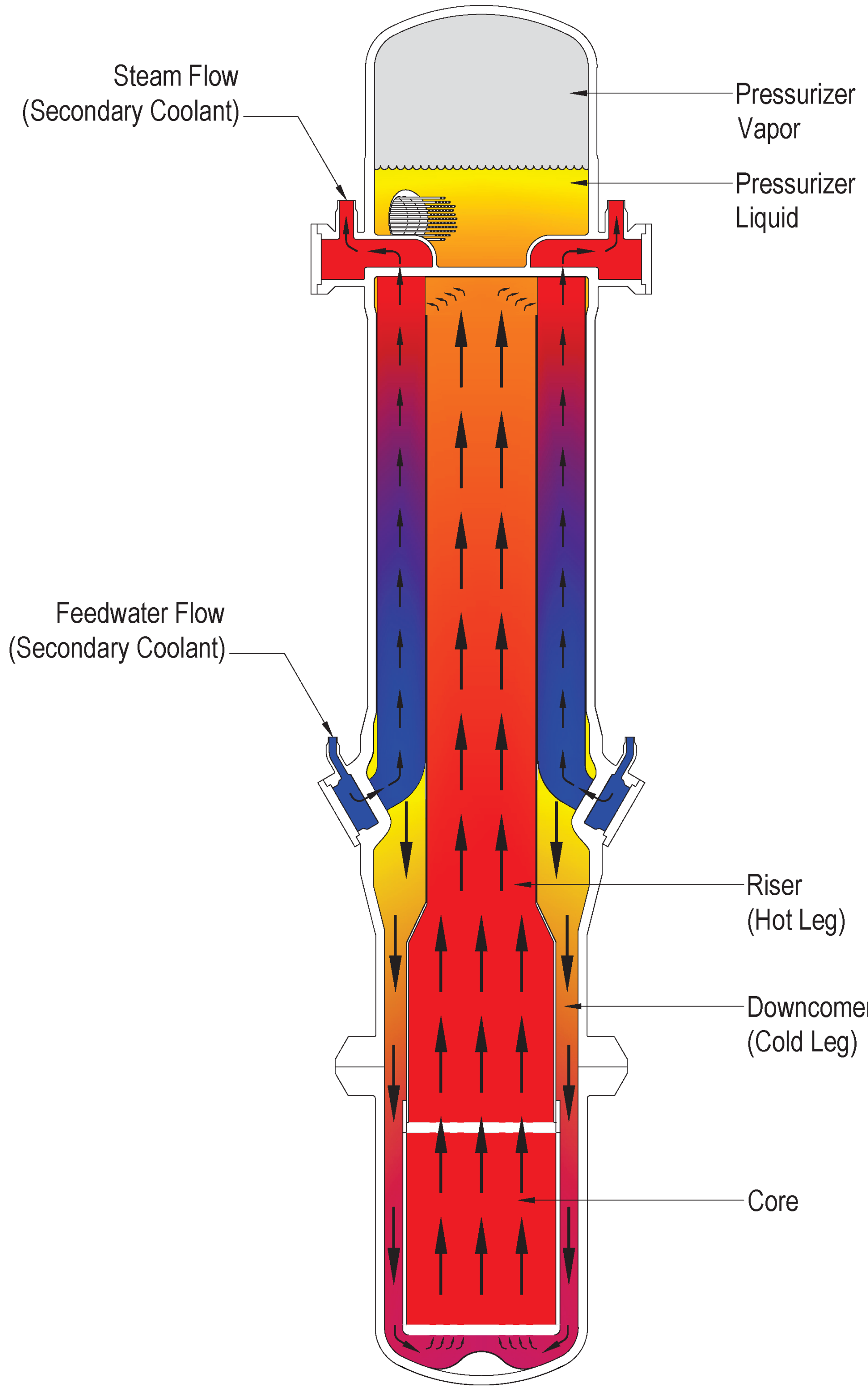


Figure 2-3 NuScale Power Module primary and secondary flow schematic

Table 2-1 Pressurized water reactor flow velocity comparison

| Design (Note 1) | Average Velocity (ft/s) | | | | Maximum Design Flow Rate ($lb_m$/s) | Primary Coolant Loop Transit Time (seconds) |
|---|---|---|---|---|---|---|
| | Steam Generator Gap | Downcomer | Core | Upper Internals Cross Flow | | |
| NuScale | 1.4 | 1.7 | 3.6 | 1.5 | 1,456 | 60.8 |
| EPR | | 24 | 16 | 30 | 55,000 | 9.9 |
| AP1000 | | 19 | 16 | 40 | 34,800 | 10.3 |
| US-APWR | | 23 | 14 | 30 | 54,092 | 12.6 |
| SONGS | 18 | - | - | - | - | - |

Notes to Table 2-1: (1) Velocities, maximum design flow rates, and loop transit times for the ERP, US-APWR and AP1000 PWR designs are per References 8.1.6, 8.1.7, 8.1.8, and 8.1.9.
(2) SONGS steam generator gap velocity is per Reference 8.1.10.

Within the NPM, the RCS water level is normally maintained in the pressurizer region above the elevation of the pressurizer heaters. The water in the pressurizer region is heated by the pressurizer heaters to a temperature greater than the temperature of the coolant leaving the RXC, in order to maintain a saturated water-steam interface in the pressurizer region.

## 2.2 Secondary Coolant Flow Conditions

Under normal operation, the feedwater is pumped to the SG through the SGS feedwater piping. The feedwater flows into the feedwater plenum and is distributed to the SG helical tubing. At the entrance to each tube, an SG tube flow restrictor restricts the secondary side flow to provide flow stability. The subcooled feedwater is heated to superheated steam in the SG helical tubes. The flow from the tubes combines in the steam plenums and exits the plenums at the steam supply nozzles. The SGS steam piping then supplies steam to the CNTS steam lines. Secondary flow is controlled as a function of reactor power and the highest flow rates occur at full-power conditions. The steam piping velocity at full-power operating conditions is below 150 feet per second.

The DHRS connects to the CNTS steam piping outside of containment and the SGS feedwater piping inside containment. Upon DHRS actuation, a two-phase natural circulation loop is established. Heat transfer through the SG tubes causes liquid water to boil, rise, and travel up the steam piping. The steam is redirected to the DHRS steam piping and then to the condenser due to the closed main steam isolation valve (MSIV) and open DHRS actuation valves. Heat transfer to the reactor pool water through the condenser tubes causes the steam to condense. The liquid returns to the SG by gravity through the DHRS condensate line. The flow rate through the loop is restricted by an orifice in the DHRS steam piping.

## 2.3 Component Screening for Flow-Induced Vibration

Table 2-2 provides a summary of the NPM components that meet the FIV screening criteria and are classified as susceptible to FIV.

The following subsections discuss in more detail the components that are screened for FIV and the components that are found to be susceptible to FIV based on the screening criteria. Components that are classified as susceptible to FIV require analysis, measurement, and inspection to meet the intent of the CVAP. Flow-induced vibration mechanisms and screening criteria, which are derived from References 8.1.3 and 8.1.4, are summarized in Table 2-3.

Table 2-2 NuScale Power Module components screened for susceptibility to flow induced vibration mechanisms

| NPM Region or Category | Component | Section Number |
|---|---|---|
| Components exposed to secondary coolant flow | Steam piping, nozzle, MSIVs, MSIV upstream and downstream bypass lines, CNTS MS drain valve branch | 2.3.1.1 |
| | SG steam plenum Note 1 | 2.3.1.2 |
| | DHRS steam piping | 2.3.1.3 |
| | DHRS condensate piping | 2.3.1.3 |
| | Helical SG tubing Note 1 | 2.3.1.4 |
| | SG tube inlet flow restrictors | 2.3.1.5 |
| | SGS pressure relief valve branch, CNTS FW drain valve branch | 2.3.1.6 |
| SG tube supports exposed to primary coolant flow | SG tube supports | 2.3.2.1 |
| | Lower SG support | 2.3.2.2 |
| Upper riser assembly exposed to primary coolant flow | Upper riser section | 2.3.3.1 |
| | Riser section slip joint | 2.3.3.2 |
| | In-core instrument guide tube (ICIGT) | 2.3.3.3, 2.3.8 |
| | Control rod drive (CRD) shaft | 2.3.3.4, 2.3.8 |
| | CRD shaft support | 2.3.3.5 |
| | Upper riser hanger brace | 2.3.3.6 |
| | CRD shaft sleeve | 2.3.8 |
| Lower riser assembly exposed to primary coolant flow | Lower riser section | 2.3.4.1 |
| | Control rod assembly guide tube (CRAGT) assembly | 2.3.4.2, 2.3.8 |
| | CRAGT support plate | 2.3.4.3 |
| | Upper core plate | 2.3.4.4 |
| Core support assembly exposed to primary coolant flow | Core barrel | 2.3.5.1 |
| | Upper support block | 2.3.5.2 |
| | Core support block | 2.3.5.3 |
| | Reflector block | 2.3.5.4 |
| | Lower core plate | 2.3.5.5 |
| | Fuel pin interface | 2.3.5.6 |

| NPM Region or Category | Component | Section Number |
|---|---|---|
| Other RVI exposed to primary coolant flow | Pressurizer spray RVI | 2.3.6.1 |
| | Chemical and volume control system (CVCS) injection RVI | 2.3.6.2 |
| | Flow diverter | 2.3.6.3 |
| | Thermowells Note 2 | 2.3.6.4 |
| | Component and instrument ports | 2.3.6.5 |
| Primary coolant piping | RCS Injection to reactor vent valve (RVV) and reactor recirculation valve (RRV) reset lines | 2.3.7.1 |
| | CNTS CVC drain valve branches | 2.3.7.2 |

Notes to Table 2-2:
1. Component is exposed to primary and secondary coolant flow.
2. Thermowells also evaluated in NPM piping exposed to secondary coolant flow.

Table 2-3 Flow-induced vibration screening criteria

| Phenomenon | Screening Criteria |
|---|---|
| Fluid elastic instability (FEI) | • array of cylinders (minimum one row), i.e., geometry<br>• array pitch/diameter < 2.0; array must sufficiently confine fluid to allow feedback between adjacent cylinders |
| Vortex shedding (VS) | • bluff body (or edge of a cavity in line with flow) , i.e., geometry<br>• subject to cross-flow<br>• absence of downstream structures to disrupt vortices |
| Turbulent buffeting (TB) | • subject to turbulent flow (axial, cross-flow or combination)<br>• component interface that is in load path of one or more components subject to turbulent flow |
| Acoustic resonance (AR) | • suitable geometry to generate an AR, typically a hollow or cavity<br>• single phase environment within hollow/cavity |
| Leakage flow instability (LFI) | Conditions 1 and 2 are met:<br>1. narrow annular flow path exists, i.e., geometry<br>2. flexible structure in annulus, bounded by fixed surface<br>AND<br>either Condition 3 or Condition 4 is satisfied:<br>3. flow conditions to generate sufficient flow velocity and pressure differential through annular flow path<br>4. annular flow velocity greater than the critical flow velocity for LFI (see Section 2.3.8) |

| Phenomenon | Screening Criteria |
|---|---|
| Galloping/flutter | • non-circular cross section, i.e., geometry<br>• aspect ratio (length/width) in prevailing direction of flow is less than 4.0 (for tall rectangular structure) and less than 2.0 (for low, long rectangular structure) |

### 2.3.1 Components Exposed to Secondary Flow

The components exposed to secondary flow are contained in the SGs, SGS piping, and DHRS. The SGS transfers heat from the reactor coolant to produce superheated steam, while providing a leak-tight pressure boundary between the primary reactor coolant and the secondary-side coolant. Additionally, the SGs remove residual and decay heat from the RXC in conjunction with the DHRS following DHRS actuation.

The SGs consist of two independent, but intertwined, helical tube bundles. Each SG has a pair of feedwater plenums and a pair of steam plenums. The SGs are once-through helical coils with primary-side reactor coolant outside the tubes and secondary-side fluid inside the tubes. On the secondary side, preheated feedwater enters the SGs through the SGS feedwater piping and the feedwater supply nozzles and feed plenums. Feedwater flows up the helical tubes where it is heated, boiled, superheated, and exits the SGs through the steam plenums and main steam supply nozzles to the SGS steam lines. The SGS steam piping then supplies steam to the CNTS steam lines. The components exposed to secondary-side flow that screen for FIV are identified in the following subsections.

#### 2.3.1.1 Steam Piping, Plenum Exit Nozzle, Main Steam Isolation Valve, MSIV Upstream and Downstream Bypass Lines, and CNTS MS Drain Valve Branch

The SGS piping includes the steam piping inside containment (see Figure 2-4). The SGS steam lines begin at the steam supply nozzle safe ends on the steam plenums and terminate inside the CNV at the CNV penetration nozzle safe ends. Outside the CNV, the steam lines are termed CNTS steam piping through the MSIV to the NPM disconnect flange. The CNTS steam lines have three regions of side branches that are closed during normal operation: the tees to the DHRS steam lines, connections upstream and downstream of the MSIV body to the bypass MSIV, and branch lines to the CNTS MS drain valves.

These components meet the screening criteria for AR. Vortices can potentially form off the leading edges of transitions within these components. Similar shedding could potentially occur due to shedding off the main steam valve bodies. When shedding frequencies become close or equal to the acoustic frequencies of the downstream piping and valve or nozzle bodies, AR can occur. No other FIV phenomena are credible for these regions.

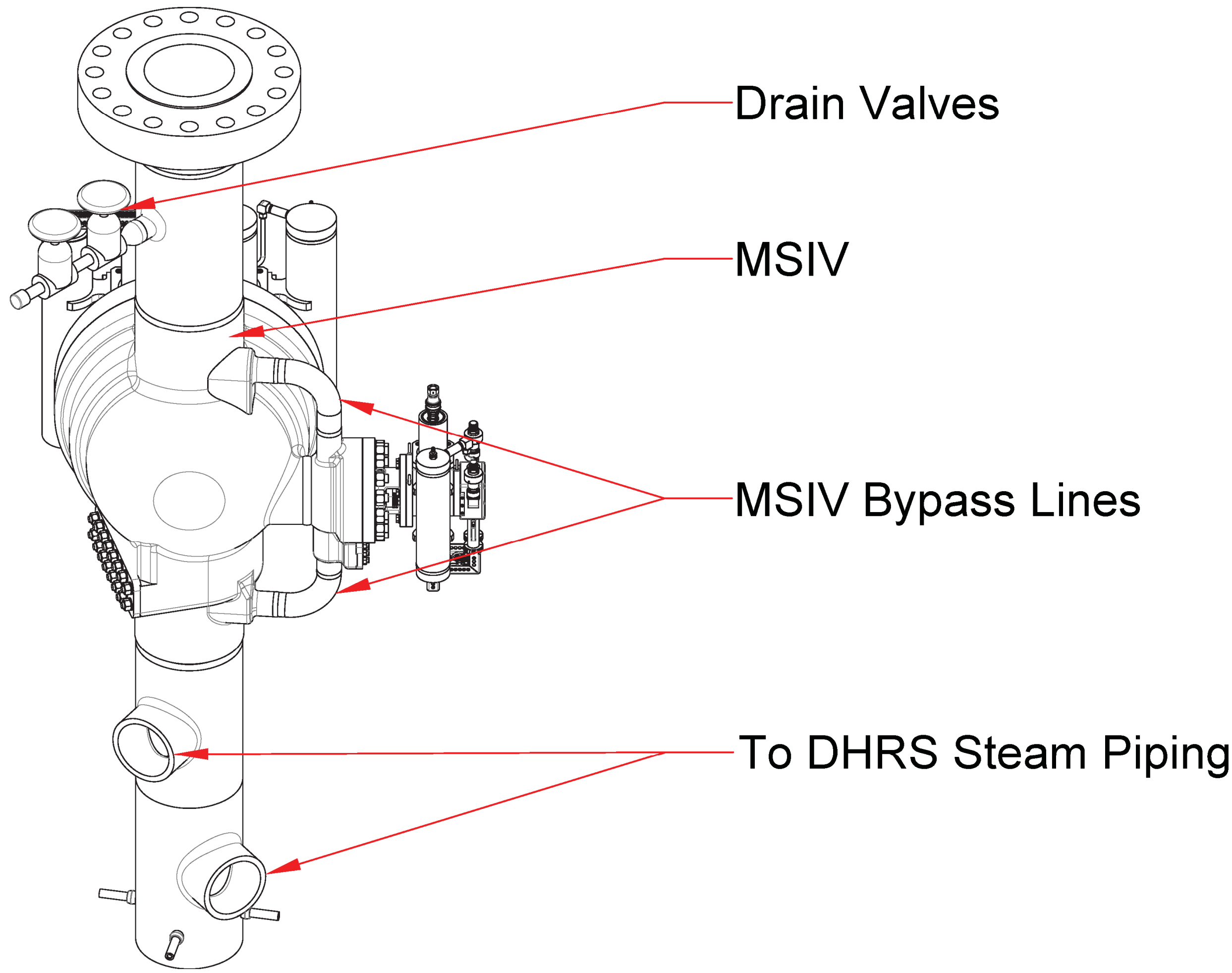


Figure 2-4    Steam piping downstream of steam nozzles

#### 2.3.1.2 Steam Plenum

As shown in Figure 2-5, the SG steam plenums are located above the SG tube bundle and the pressurizer baffle plate. The plenum tube sheet region provides the termination point for the helical SG tubes and the plenum provides the flow path from the SG tubes to the steam nozzle located on the outside of the RPV.

The plenum itself is most comparable to a chamber composed of the primary head and SG tube sheet in a conventional recirculating SG, as it is effectively an independent pressure vessel chamber. Although the outside of the plenum is subject to pressurizer coolant flow, based on the size and thickness of the plenum and low pressurizer flow rates, no significant FIV response is plausible. The steam plenum is a bluff body; however, based on the SG tubes connecting on the downstream side, VS would be disrupted. Based on lack of any potential leakage flow paths and roughly cylindrical geometry, leakage flow and galloping/flutter are excluded. The sole potential FIV phenomenon to which the steam plenums could potentially be subject is internal AR.

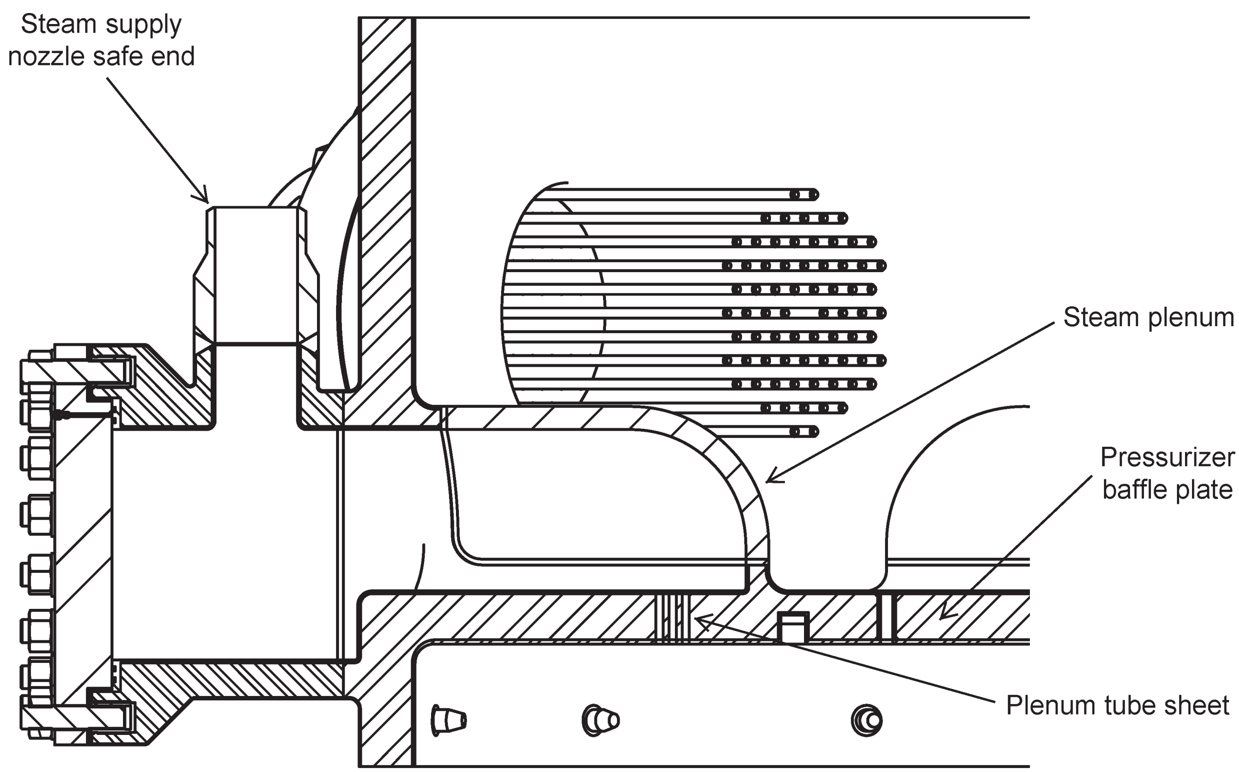


Figure 2-5 Steam plenum region

#### 2.3.1.3 Decay Heat Removal System Steam and Condensate Piping

During normal operation, the DHRS is not used, and the DHRS actuation valves are closed. During off-normal operations, the SGS may function in conjunction with the DHRS to remove decay heat and bring the RCS to a safe shutdown temperature. Upon DHRS actuation, the SGS receives feedwater from the DHRS condensate lines. Steam generated by the SGs is routed to the CNTS steam lines as during normal operations. Figure 2-6 illustrates the DHRS piping from the actuation valves at the CNTS main steam lines to the SGS feedwater tee inside containment.

The DHRS steam and condensate lines represent branch lines in which there is normally no flow during operation. Acoustic resonance is possible for these lines. Within the DHRS condenser, AR in the header assemblies is unlikely to be a concern as the transmitted pressure waves will lose energy as they pass through the series of 1-in. tee junctions (at the condenser tube entrances and exits), eliminating the potential for AR. No other FIV phenomena are credible for these regions.

{{

}}2(a),(c),ECI

Figure 2-6 Decay heat removal system lines 1 and 2 from actuation valves to steam generator system feedwater tee

### 2.3.1.4 Helical Steam Generator Tubing

The SG has two independent helical coil tube bundles. The helical SG tubes span the distance between the feedwater and steam plenums shown in Figure 2-7. The two tube bundles are coiled in opposite directions as shown in Figure 2-7. Each tube bundle is composed of 21 columns of tubes and the tubes from each plenum are intertwined or “stacked” on top of each other (the alternation of tubes from one plenum to another is shown in Figure 2-7 as alternating colors). Each bundle has two feedwater plenums at the bottom and two steam plenums at the top (eight plenums total). The SG tubes have an outside diameter of 0.625 in. and a nominal wall thickness of 0.050 in.

Secondary-feed flow enters the tubes at the feed plenums and boils producing superheated steam internally along the length of the tubes. As such, the random pressure fluctuations associated with boiling inside the tubes provides a secondary source of turbulent energy to the tubes in addition to the turbulent forces due to reactor coolant cross flow outside the tubes.

Like typical U-tube or straight-tube bundles, the potential for FEI of the tube bundle exists. Vortex shedding is assumed to be possible for all span locations where there are no tubes directly downstream. Vortex shedding from SG tubes has not been demonstrated to occur in tests of closely-packed tube arrays since the presence of tubes directly downstream disrupts coherent vortex formation; however, the VS mechanism is considered for the SG tubes for completeness. Vibration due to incoherent vortices may exist for the interior SG tubes. These vibrations are accounted for in the turbulent buffeting loads. Based on the tube geometry and the lack of leakage flow paths, leakage and galloping/flutter are excluded.

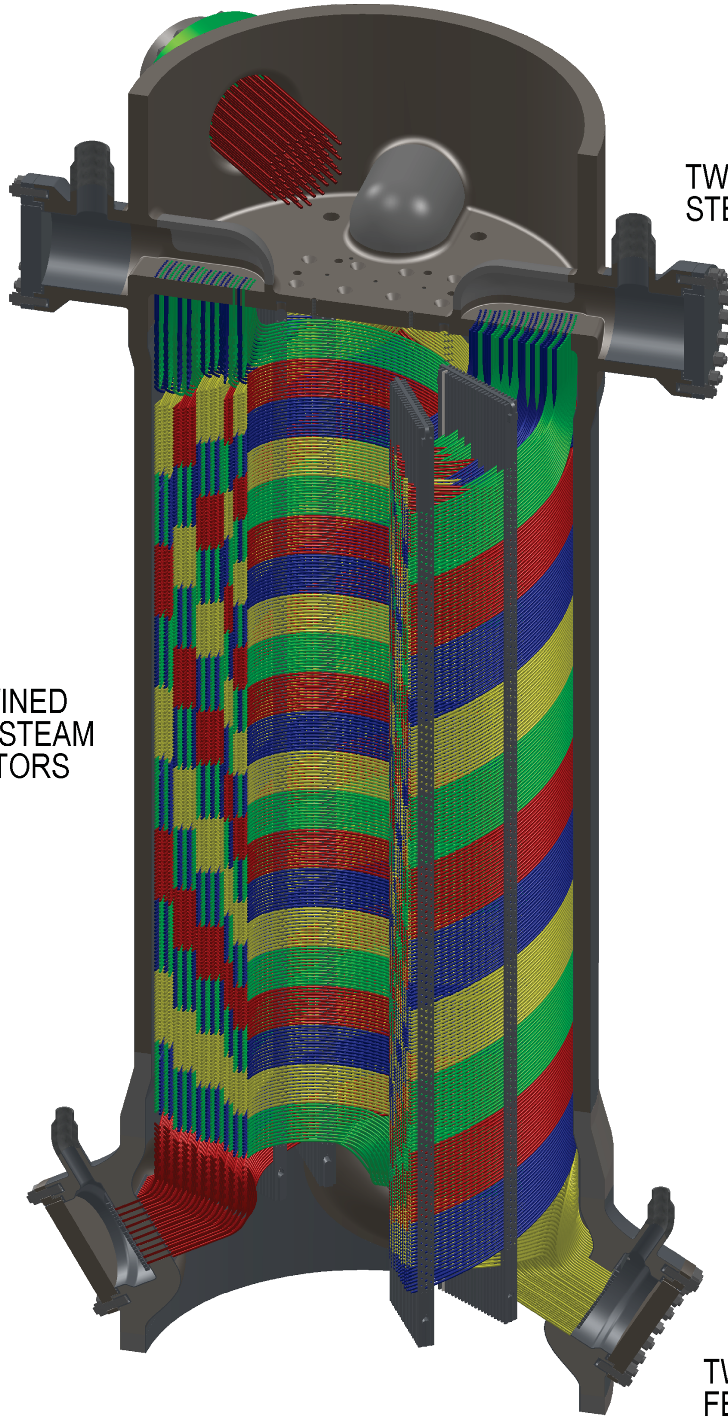


Figure 2-7 Helical steam generator tube bundle

### 2.3.1.5 Steam Generator Tube Inlet Flow Restrictors

Each individual SG tube requires an inlet flow restriction device for the purpose of flow stability. Figure 2-8 provides a representation of the flow restrictor concept. The flow restrictor fits into the tube inlet and is designed to provide flow stability by restricting the volume of the secondary-side flow through the tube. The flow restriction is created by a series of narrow annular gaps between the restrictor and the tube inner diameter.To hold the flow restrictors in position, mounting hardware (plate) is required within the feed plenum. The plate is removable and held in place with fasteners. The flow restrictors are attached to this mounting plate. The flow restrictors and mounting hardware are anchored at a series of points with individual fasteners rather than with extended seams (e.g., welds). Based on the narrow annular gaps between the SG tube and the flow restrictor and the relatively large pressure loss in this region (Table 2-4), the flow restrictor is susceptible to leakage flow induced vibration. The mounting plate is stiffer than the flow restrictor and provides larger flow area. Therefore, leakage flow induced vibration is not a concern for the mounting plate. Similarly, the turbulent buffeting vibrations of the flow restrictor will bound those of the mounting plate due to the increased stiffness and reduced convective velocity of the mounting plate compared to the flow restrictor.

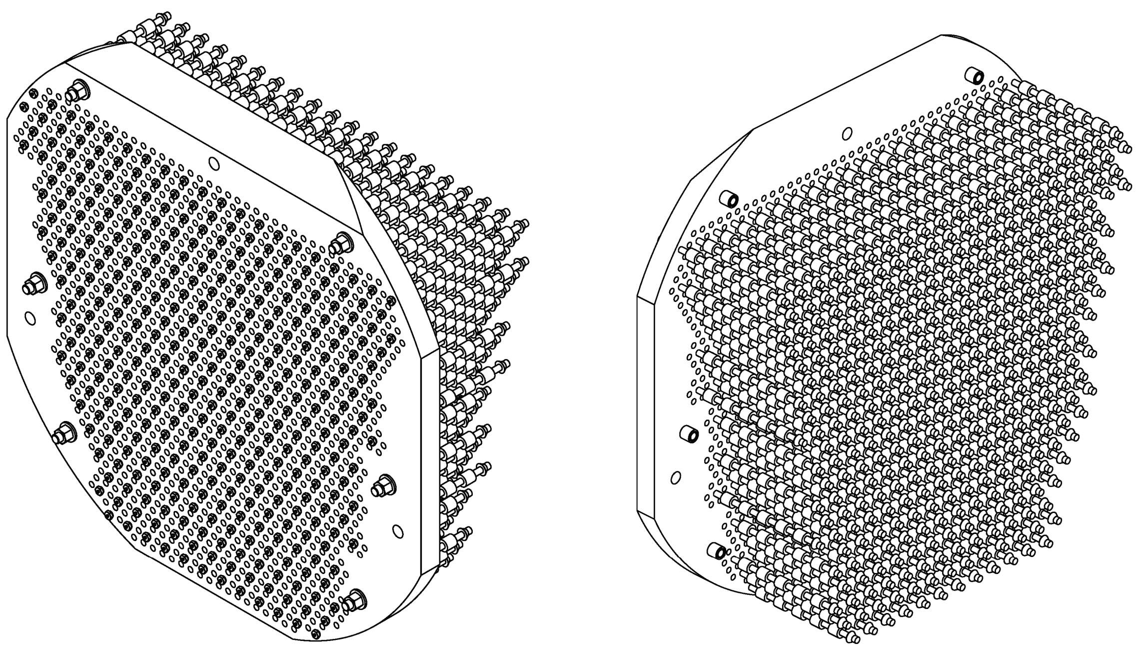

Figure 2-8 Tube inlet flow restrictor and mounting plate

#### 2.3.1.6 SGS Pressure Relief Valve Branch and CNTS FW Drain Valve Branch

The SGS piping includes the feedwater piping inside containment. The SGS feedwater lines initiate inside the CNV at the CNV penetration nozzle safe ends and terminate at the feedwater supply nozzle safe ends on the feedwater plenums. A pressure relief valve is located on each of the two feedwater lines inside containment to provide thermal relief for the SGS.

The CNTS FW piping is located outside containment between the CNV penetration nozzle safe ends and the NPM disconnect flange. A drain valve is connected to each FW line.

The pressure relief valves and drain valves are connected to the main piping with a short branch of small diameter piping and weldolet. The fluid in these piping regions is single phase. The valves are normally closed, and the branch piping represents a flow occluded region connected to the main piping. Vortices may be generated as flow passes the discontinuity in the piping created by the branch. If the vortices generate an acoustic wave coincident with a structural mode, acoustic resonance could occur in the branch piping lines. No other FIV phenomena are credible for these regions.

### 2.3.2 Steam Generator Supports

The SG steam and feedwater plenums are integral parts of the RPV. The SG helical tubing is provided with supports and lower SG support members, which are welded to the RPV inner wall. In addition to considering these supports in the evaluation of the helical tubing for FIV, the supports themselves are assessed to ensure the designs are acceptable to prevent detrimental FIV.

#### 2.3.2.1 Steam Generator Tube Supports

Tube supports, shown in Figure 2-9 and Figure 2-10, span the full height of the helical tube bundle and are anchored at the attachment of the upper SG support to the bottom of the integral steam plenum. Based on their form (effectively a solid bar), the tube supports are not subject to leakage flow or AR. Additionally, based on the confinement of the supports within the tube bundle, where the tortuous flow path creates turbulence, formation of coherent vortices will not occur. The axial alignment of the tube supports provides an aspect ratio greater than 4.0 and an angle of attack of effectively 0.0 thus precluding galloping and flutter.

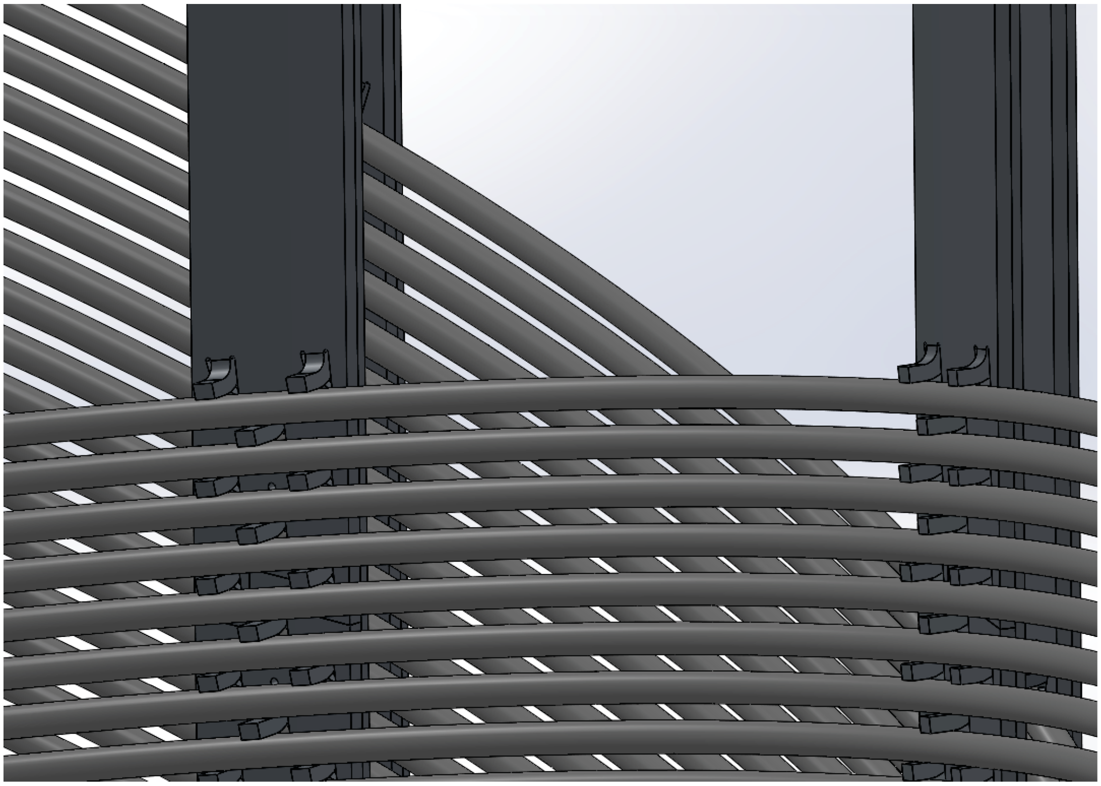

Figure 2-9 Steam generator support design showing support tabs

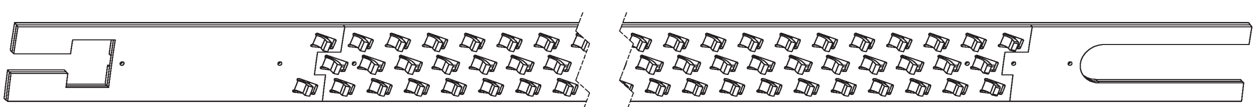

Figure 2-10 Steam generator tube support assembly (shown in a horizontal installation)

Figure 2-10 shows the tube supports interface with the SG tubes. Clearances are provided between the tube and tube support tabs and between the tube and back of the adjacent tube support. Other clearances in the assembly include the tube support top section to upper SG support clearance, the tube support bottom section to lower SG support clearance, and the tube support middle tab to groove clearance. The remainder of the tube support contains welded connections. The tube to tube support tab clearances are provided for manufacturability of the tube supports and for ease of assembly of the steam generator as a whole. The tube support top section to upper SG support clearances are provided so that the cutouts fit over the upper SG supports during steam generator assembly. The tube support bottom section has a larger vertical clearance to the lower SG support to allow for thermal expansion of the steam generator assembly. The middle of the three columns of tabs on the tube support extends into the groove in the back of the adjacent tube support. This contact allows adjacent tube supports to support each other in the circumferential direction.

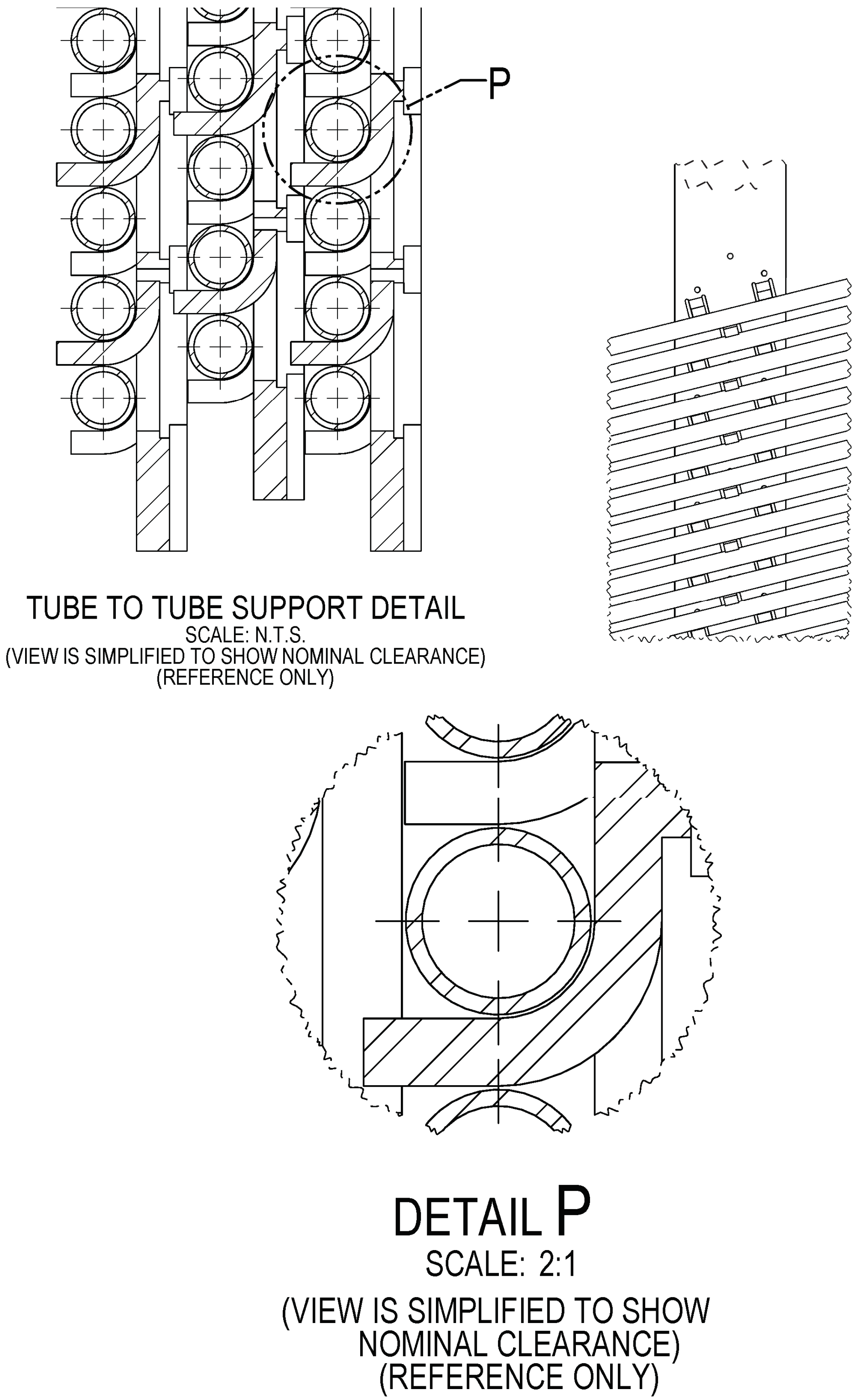


Figure 2-11 Steam generator tube to tube support interface

#### 2.3.2.2 Lower SG Supports

Figure 2-12 shows the lower SG supports. The upper SG supports are welded to the RPV and the integral pressurizer baffle. The lower SG supports are welded to the RPV shell below the SG tube bundle. Individual tube supports are not joined to the lower SG supports. Below the tube bundle, vortex shedding from the lower SG supports and turbulent buffeting are possible. Based on its form (solid bar) and an aspect ratio of about 3.5 as a low, long rectangle, it is not subject to leakage flow, AR, or plunge galloping. Torsional gallop should be evaluated. The angle of attack for flow impacting the lower SG supports is likely close to 0.0, which would exclude flutter; however, flow velocities in this region have not yet been rigorously developed. Due to the overall support structure stiffness, flutter is most likely excluded; however, analysis is performed to substantiate this conclusion.

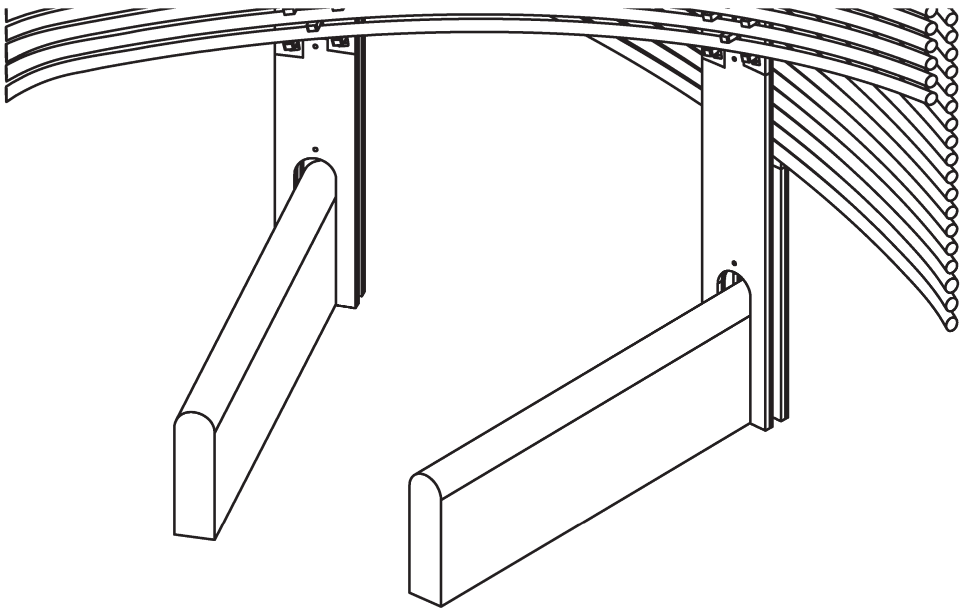

Figure 2-12 Lower SG supports with inner steam generator columns removed for clarity

### 2.3.3 Upper Riser Assembly

The upper riser assembly includes the upper riser section slip joint, the upper riser section, a series of supports for the CRD shafts and ICIGTs, and the upper riser hanger assembly. The portions of the upper riser assembly that screen for FIV are identified in the following subsections.

#### 2.3.3.1 Upper Riser Section

The upper riser section is shown in Figure 2-13. The upper riser section is supported by the upper riser hanger in the vertical direction. Horizontally, it is primarily supported by the SG tube supports in the radial direction. The upper riser section itself is an open cylinder. Fluid enters the upper riser section at the transition and turns 180 degrees over the upper edge of the riser.

The upper riser section is susceptible to parallel flow TB due to the upward flow inside the riser and the downward flow on the riser exterior. The upper riser section is not susceptible to FEI, AR, gallop, or flutter as it is an open cylinder. The upper riser section directs the fluid flow and does not cross the flow path, precluding it from a VS susceptibility. Flow in the region above the riser is fully turbulent. This region does not represent a cavity, where AR could develop, because it contains the CRD shaft sleeves, ICIGTs, hot leg thermowells, hanger braces, as well as the SG tubes. Therefore, FIV due to mechanisms other than TB are not credible for the upper riser section.

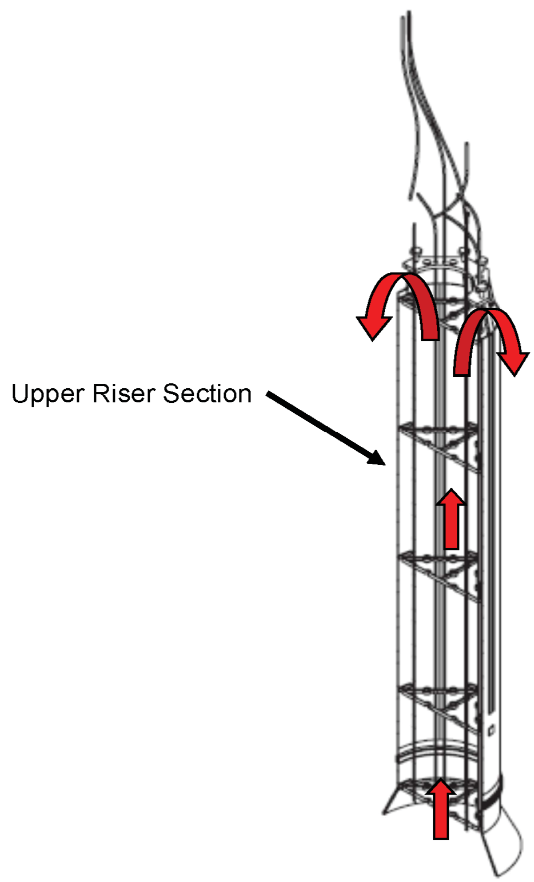


Figure 2-13 Upper riser section with characteristic flow direction noted

### 2.3.3.2 Riser Section Slip Joint

A friction fit joint is located at the junction between the upper riser assembly and the lower riser assembly as shown in Figure 2-14. This joint is required to allow NPM disassembly during the refueling process. At cold conditions the hold-down force on the joint from the bellows installed in the upper riser assembly plus deadweight of the bellows and upper riser transition is approximately {{ }}2(a),(c) lbf. The force is higher at operating conditions when thermal expansion further compresses the bellows {{ }}2(a),(c) This region does not screen for LFI because of the large hold-down force and the very small pressure difference of {{ }}2(a),(c) psid (Table 2-4) between the hot and cold legs of the primary coolant loop due to the natural circulation primary coolant flow. The lifting force on the upper riser transition from buoyancy and the pressure difference is approximately {{ }}2(a),(c) lbf. Therefore, LFI is screened out because fluid forces at the slip joint that act to open the leak channel are much lower than the opposing forces of deadweight and upper riser bellows compression. The slip joint itself is not susceptible to FEI, AR, gallop, or flutter as it is an open cylinder. The slip joint directs the fluid flow and does not cross the flow path, precluding it from VS susceptibility. The portions of the slip joint in contact with the hot and cold legs are susceptible to TB based on the flow conditions.

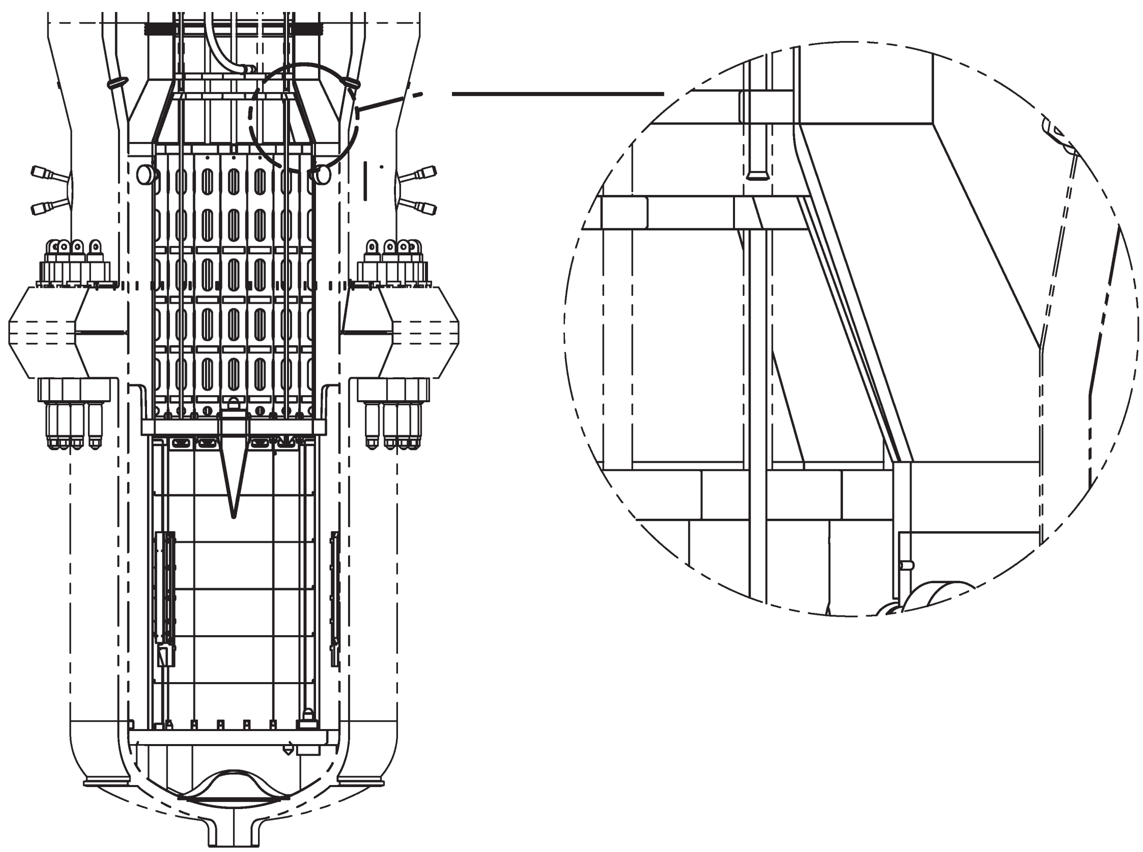

Figure 2-14 Riser section slip joint

#### 2.3.3.3 In-core Instrument Guide Tube

The ICIGTs extend from the upper RPV head to the top of the fuel assemblies. On the interior of the ICIGTs reside the in-core instruments which are routed through the pressure boundary at the RPV head and down into the core. The ICIGTs interface with the upper RPV head, pressurizer baffle plate, upper riser hanger ring, CRD shaft supports, lower riser assembly ICIGT support, and the upper core plate. Each ICIGT is divided into three regions: tube sections within the pressurizer, upper riser, and lower riser. Each tube section is welded to at least one support location which fixes tube translation and rotation. The remainder of the ICIGT support interfaces provide lateral support while allowing small vertical displacements to accommodate differential thermal expansion movement.

The geometry of the ICIGTs is constructed in a manner that they are not susceptible to FEI, acoustic resonance, gallop, or flutter. Although small gaps exist between the ICIGTs and CRD shaft supports and the lower ICIGT support, screening evaluations show that the gap velocity is negligible compared to the critical velocity for leakage flow instability; therefore, LFI is not credible (Section 2.3.8). The ICIGTs are exposed to turbulent flow and are susceptible to TB. Above the upper riser section and below the pressurizer baffle plate, the ICIGTs are subject to crossflow; therefore, VS is also applicable for this component.

#### 2.3.3.4 Control Rod Drive Shaft

The CRD shafts pass through the CRD shaft supports as they are routed to the fuel assemblies. The CRD shaft support openings are one of the CRD shaft alignment features and the clearance between the two components is small. Similar to the ICIGT, although the clearance between the component and support is small, the gap velocity is sufficiently low compared to the critical velocity that LFI is not credible (Section 2.3.8). The CRD shafts also pass through the pressurizer baffle plate. During steady state operation, there is negligible pressure difference between the riser outlet and the pressurizer. Due to the momentum of the flow as it exits the riser, it is possible that some flow passes through the annular flow regions between the CRD shaft and the pressurizer baffle plate. This flow is expected to be very low, based on the low driving force. Leakage flow instability screening for the CRD shaft interface with the pressurizer baffle plate and upper riser hanger ring has determined that the interface is not susceptible to LFI, as shown in Section 2.3.8.

Above the uppermost CRD shaft support, the fluid changes direction as it turns to the SG tube region. The CRD shaft is protected from cross flow by sleeves. Using the screening criteria, this interface is not susceptible to the FIV phenomena other than TB.

#### 2.3.3.5 Control Rod Drive Shaft Support

The CRD shaft support is attached to the upper riser section and is normal to the flow direction, as shown in Figure 2-15. As the primary fluid moves around the support beams, VS and TB may occur. Using the screening criteria, this component is not susceptible to the other FIV mechanisms.

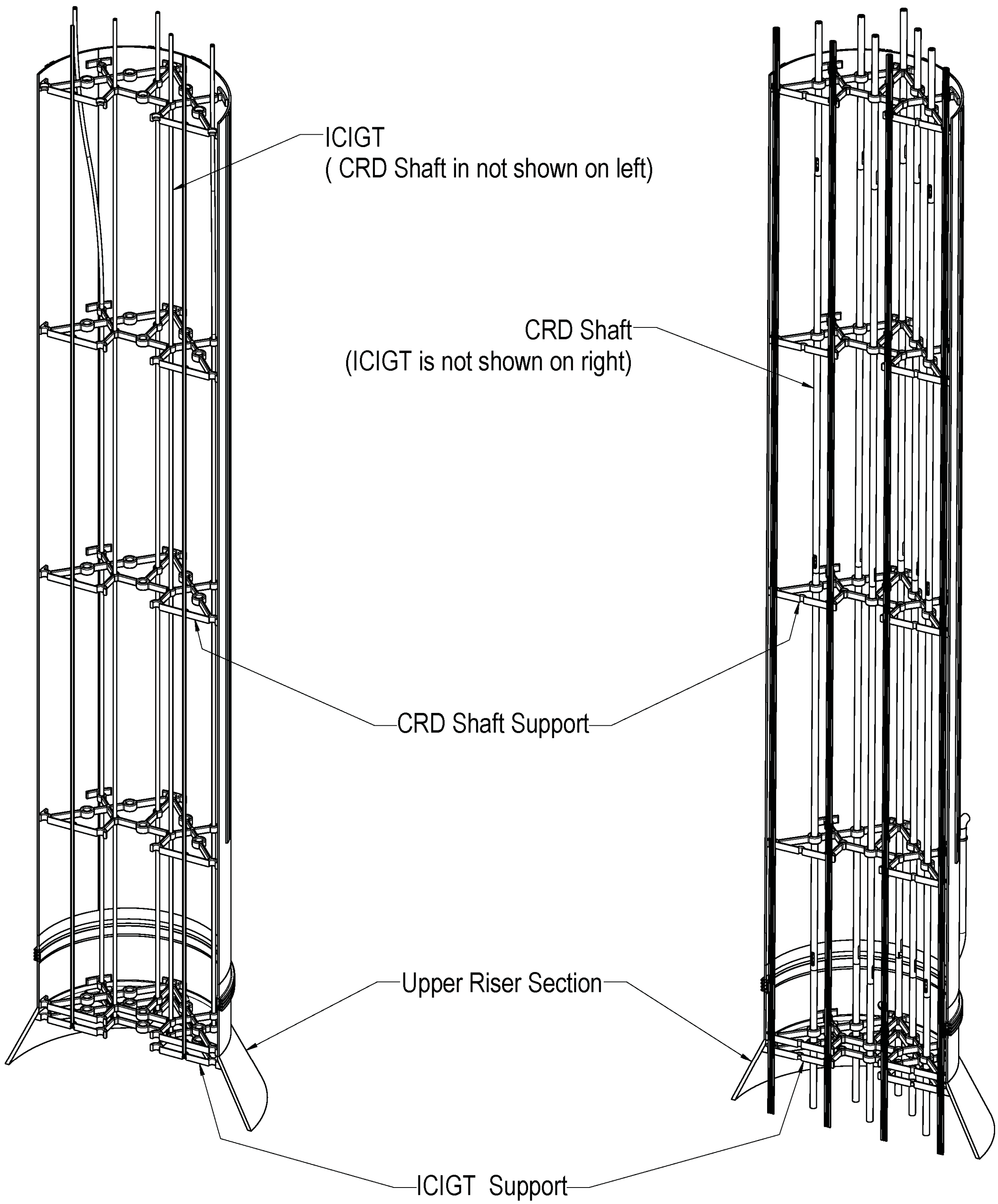


Figure 2-15 Control rod drive shaft, in-core instrument guide tube, and supports

#### 2.3.3.6 Upper Riser Hanger Brace

As shown in Figure 2-16, the upper riser hanger assembly connects the upper riser section to the pressurizer baffle plate. Fasteners are used to attach the hanger ring to the baffle plate, such that there is no flow past or above this part. However, similar to the ICIGT and CRD shaft, the upper riser hanger braces experience cross flow as the hot leg fluid turns from the upper riser into the SG region. Based on this, the upper riser hanger braces are susceptible to TB and VS. No other FIV mechanisms are credible for this region.

#### 2.3.3.7 CRD Shaft Sleeve

As shown in Figure 2-16, there are 16 sleeves that protect the CRD shaft from cross flow above the top of the upper riser. The sleeves are welded to the hanger ring. As the flow area above the upper riser and below the integrated steam plenum is not a cavity, generation of an AR condition due to the vortices shed from the CRD shaft sleeves is not possible. Based on this, the CRD shaft sleeves are susceptible to TB and VS. No other FIV mechanisms are credible for this region.

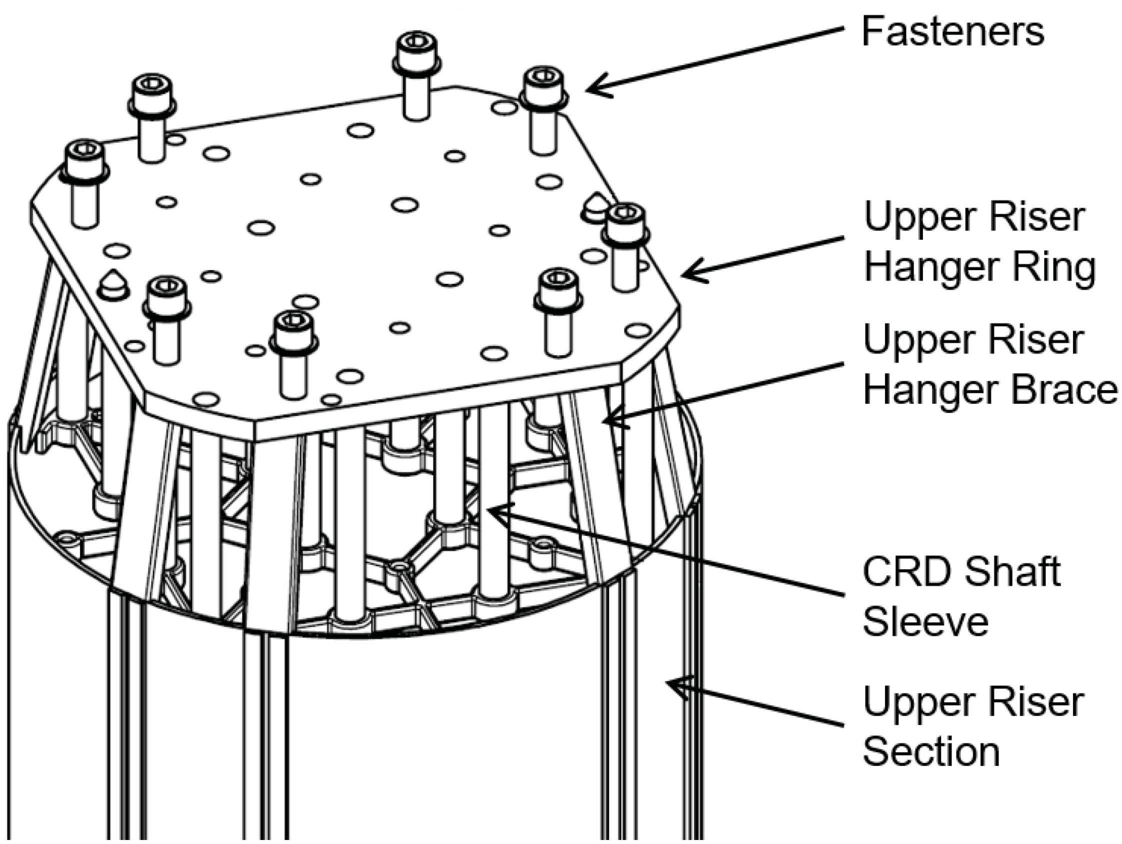


Figure 2-16 Upper riser hanger assembly

### 2.3.4 Lower Riser Assembly

The lower riser assembly includes the lower riser section, the upper core plate, CRA guide tubes, CRA guide tube support plate, and CRD shaft supports. The portions of the lower riser assembly that screen for FIV are identified in the following subsections.

#### 2.3.4.1 Lower Riser Section

The lower riser section is the cylindrical section in the lower riser assembly, as shown in Figure 2-17. This section transfers the loads from the slip joint and the guide tube support plate to the lower core plate. It also separates the up-flowing fluid above the core from the down-flowing fluid in the downcomer. The lower riser section is susceptible to TB due to parallel flow and vortices generated by the feed plenums. The open cylindrical shape precludes the lower riser section from being susceptible to FEI, AR, leakage flow, gallop, or flutter. The lower riser section is not susceptible to VS because no part of the component is opposing the flow path. Therefore, the lower riser section is only susceptible to TB.

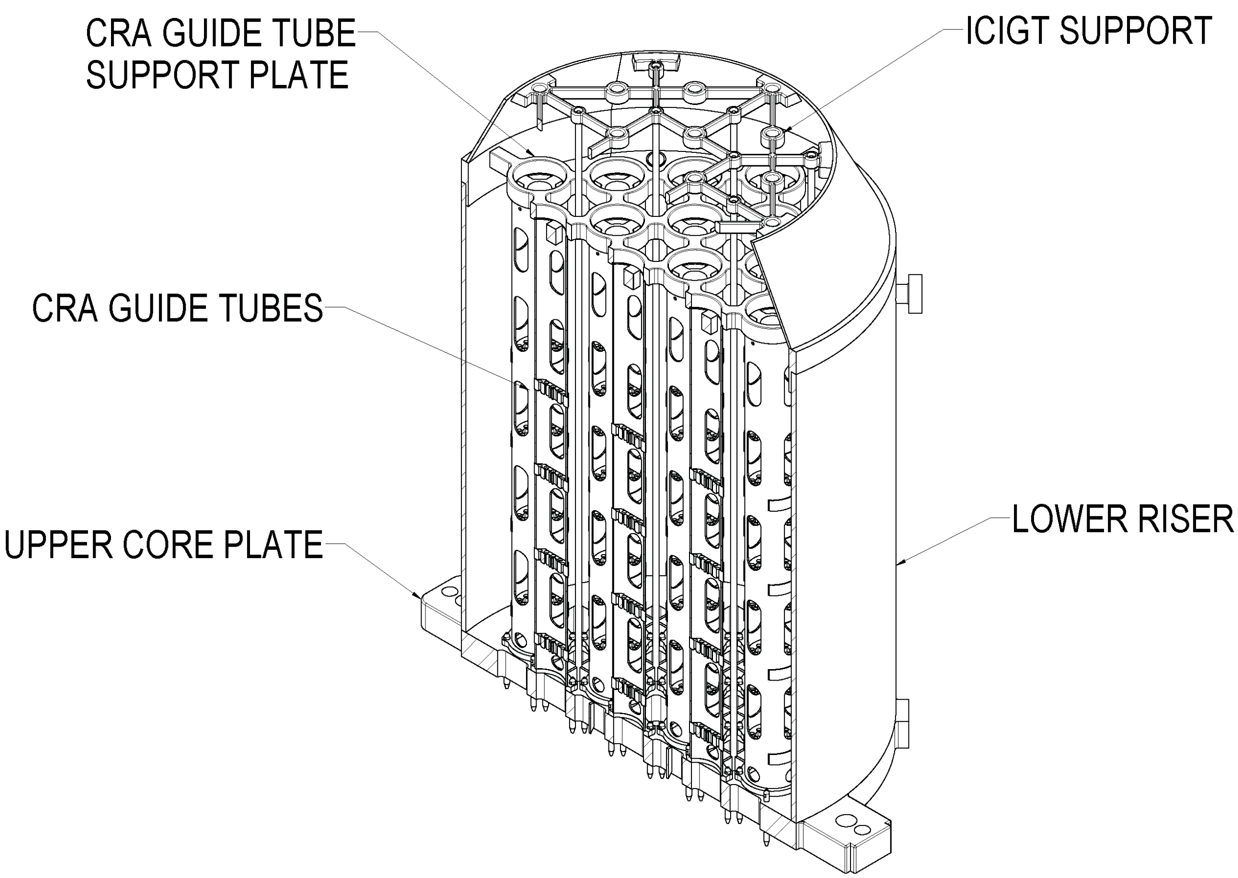


Figure 2-17 Lower riser assembly

### 2.3.4.2 Control Rod Assembly Guide Tube Assembly

The CRAGT supports the CRAs at varying amounts of control rod insertion, as shown in Figure 2-18. The CRAGT assembly includes four CRA cards, the CRA lower flange, the CRA guide tube, and the CRA alignment cone. The CRA cards, lower flange, and alignment cone are welded to the CRAGT guide tube to form the CRAGT assembly. The CRAGT assemblies are supported by the upper core plate and the guide tube support plate (Section 2.3.4.3).

The CRAGT components have many sharp edges to cause VS and TB. The CRAGT assembly is not susceptible to leakage flow because the annular gap velocity at the CRAGT support, driven by a small pressure difference, is well below the critical velocity that screens this component for LFI. There is no cavity region in the CRAGT assembly where AR could form. The CRAGT assembly is designed to allow flow to pass in and out of the guide tube. There are no flow-occluded regions and any vortices that form are dissipated by the turbulent flow. Using the screening criteria, the CRAGT is not susceptible to the FIV phenomena, other than VS and TB.

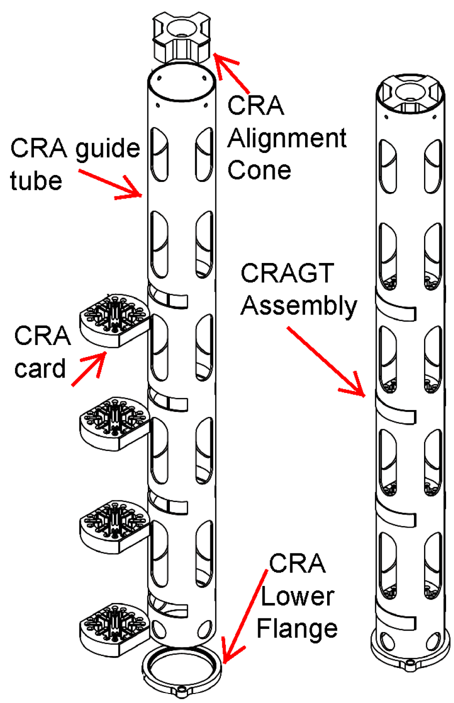


Figure 2-18 Control rod assembly guide tube assembly

### 2.3.4.3 Control Rod Assembly Guide Tube Support Plate

The CRAGT support plates are located above the CRAGT assembly, as depicted in Figure 2-17. Similar to the CRAGT assembly, the support plate is subject to turbulent flow and also represents a bluff body subject to cross flow. Further, there is no cavity region downstream of the CRAGT support plate where AR could form. As shown in Figure 2-15, the downstream region contains the control rod drive shaft, control rod drive shaft supports and ICIGT. Therefore, TB and VS are applicable mechanisms. Other mechanisms are not considered credible.

#### 2.3.4.4 Upper Core Plate

The upper core plate functions in conjunction with the core support assembly to align and support the reactor core system. The upper core plate is attached to the bottom of the lower riser by a socket head cap screw and alignment dowel. The upper core plate is subject to turbulent flow and also represents a bluff body subject to cross flow. Therefore, TB and VS are applicable mechanisms. Other mechanisms are not considered credible.

### 2.3.5 Core Support Assembly

The core support assembly includes the core barrel, upper support blocks, lower core plate, fuel pin interface, reflector blocks, and the RPV surveillance specimen capsule holder and capsules. All surveillance specimens are not shown in Figure 2-19, but consist of four capsule holders attached to the outer surface of the core barrel at the mid height of the core support assembly. While many of the components are exposed to turbulent flow and some are exposed to cross flow, vibration of these components or components in the load path are not expected based on the component weights and stiffness. Based on this, representative components in the core support assembly are screened for susceptibility to FIV.

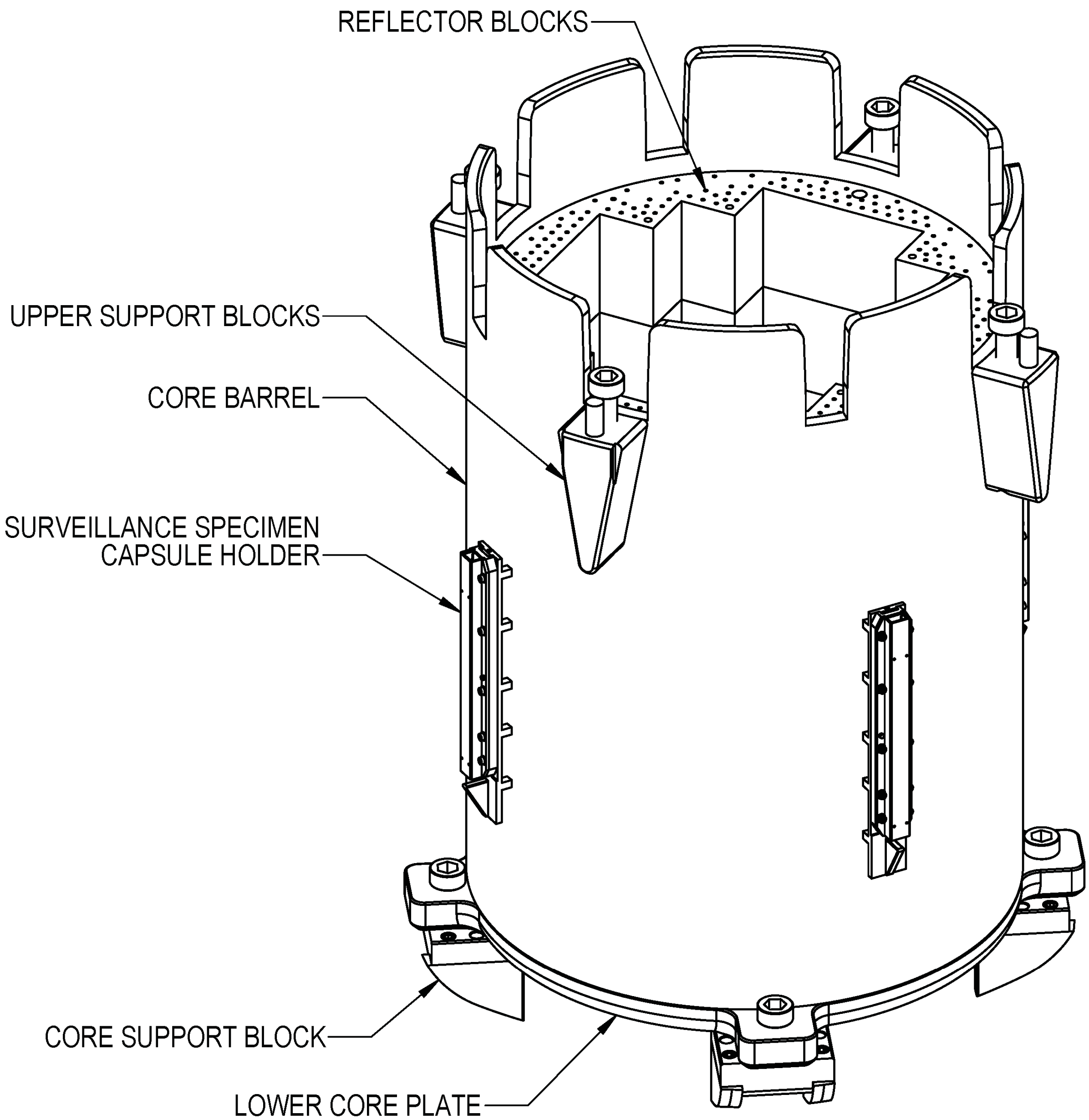


Figure 2-19 Core support assembly

#### 2.3.5.1 Core Barrel

The core barrel is a large cylinder designed to carry the core support loads and separate the down-flowing fluid from the fuel, as denoted in Figure 2-19. The core barrel is susceptible to TB using the screening criteria. Using the screening criteria, the core barrel is not susceptible to the other FIV phenomena.

#### 2.3.5.2 Upper Support Block

The upper support block is attached to the core barrel and opposes the fluid as it travels through the downcomer, as shown in Figure 2-19. There is no cavity region downstream of the upper support block where AR could form. This block is susceptible to VS and TB phenomena. Using the screening criteria, the upper support block is not susceptible to the other FIV phenomena.

#### 2.3.5.3 Core Support Block

The core support blocks are located in the downcomer. This feature transfers the dead weight and accident loads from the lower core plate to the lower head of the RPV, as denoted in Figure 2-23. There is no cavity region downstream of the core support blocks where AR could form. This block opposes the fluid flowing through the downcomer into the lower plenum and is susceptible to VS and TB phenomena. Using the screening criteria, the core support block is not susceptible to the other FIV mechanisms.

#### 2.3.5.4 Reflector Block

The reflector blocks are aligned by pins and stacked on the lower core plate inside the core barrel, as denoted in Figure 2-19. Primary coolant flows through small channels in the reflector and the inner surface of the reflector is subject to turbulent core flow. The only mechanism applicable to this component is TB. Other mechanisms are not credible due to the flow conditions and component geometry.

#### 2.3.5.5 Lower Core Plate

The lower core plate is located below the fuel assemblies, as depicted in Figure 2-19. The lower core plate is subject to turbulent flow and also represents a bluff body subject to cross flow. There is no cavity region downstream of the lower core plate where AR could form. The structures downstream of the lower core plate are primarily narrow flow channels composed of fuel assemblies. Therefore, TB and VS are applicable mechanisms. Other mechanisms are not considered credible including AR, which cannot form in the highly turbulent conditions downstream of the lower core plate. Similar to the upper core plate, the lower core plate is significantly thicker and stiffer than the CRAGT support plate and experience similar flow velocities; therefore, the response of the lower core plate to these mechanisms is expected to be bounded.

#### 2.3.5.6 Fuel Pin Interface

The fuel top and bottom nozzles interface with fuel pins that are installed in the upper and lower core plates, as shown in Figure 2-20. This interface is a location for impact primarily driven by the TB of the fuel assembly. The fuel pin is not directly subject to other FIV mechanisms based on component geometry and flow conditions.

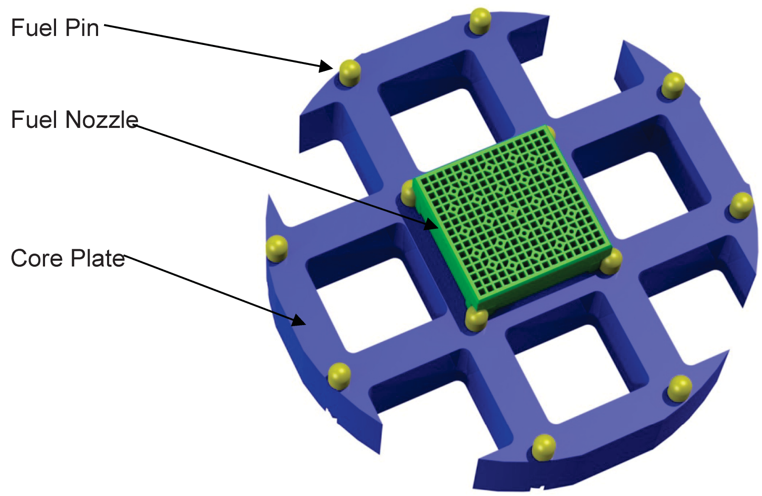


Figure 2-20 Typical fuel nozzle-to-fuel pin interface

### 2.3.6 Other Reactor Vessel Internals

There are other design features located in the primary and secondary coolant flow paths that require FIV screening. These consist of instrumentation and system connections that support interfacing systems. For the NPM design, the majority of the RPV connections are located in the steam space of the pressurizer. This region is always steam and is not exposed to high average flow rates, even under transient and accident conditions, due to the very large cross sectional area on the region. Therefore, for the majority of the instrument and interfacing system connections in the NPM design, degradation due to FIV is not credible. The instrumentation and system connections that screen for FIV are identified in the following subsections.

#### 2.3.6.1 Pressurizer Spray Reactor Vessel Internals

The pressurizer spray lines and nozzles are attached to a nozzle in the upper RPV head and extend downward into the steam region of the pressurizer (Figure 2-21). Fluid is pumped through these components to provide the pressurizer spray. Jet flow fluctuations do not occur at the pressurizer spray nozzle because the chemical and volume control system (CVCS) is a forced flow system. The flow is subsonic and does not undergo phase change upon exiting the nozzle. Therefore, flow vortices are not generated. These lines are susceptible to TB as there is a large pressure loss across the nozzle and flow rates through the spray line piping are turbulent when spray flow is used. Using the screening criteria, the pressurizer spray lines and nozzles are not susceptible to the other FIV mechanisms.

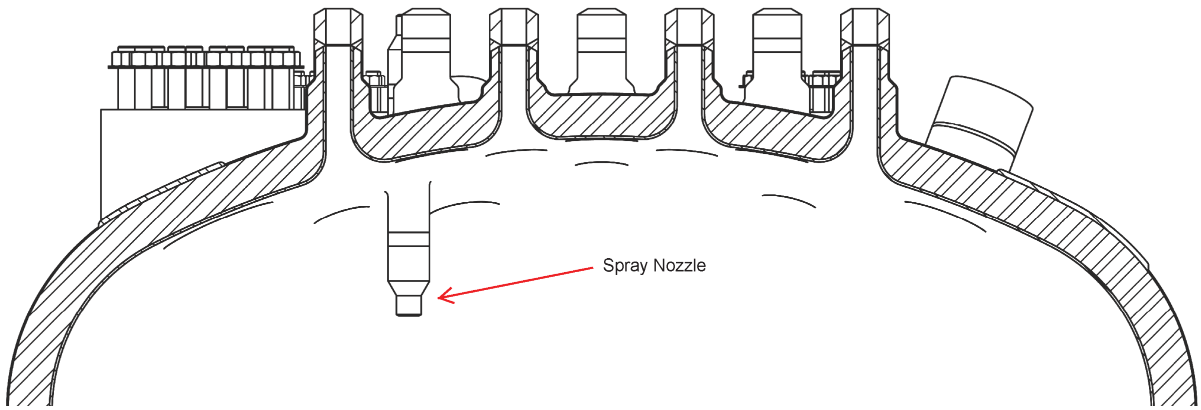


Figure 2-21 Pressurizer spray reactor vessel internals

### 2.3.6.2 Chemical and Volume Control System Injection Reactor Vessel Internals

The CVCS inlet line is routed from a flange in the RPV wall into the downcomer (Figure 2-22). The line continues through the upper riser section into the up-flowing region above the core. The line is susceptible to VS and TB in both regions. Using the screening criteria, the CVCS injection RVI is not susceptible to other FIV phenomena.

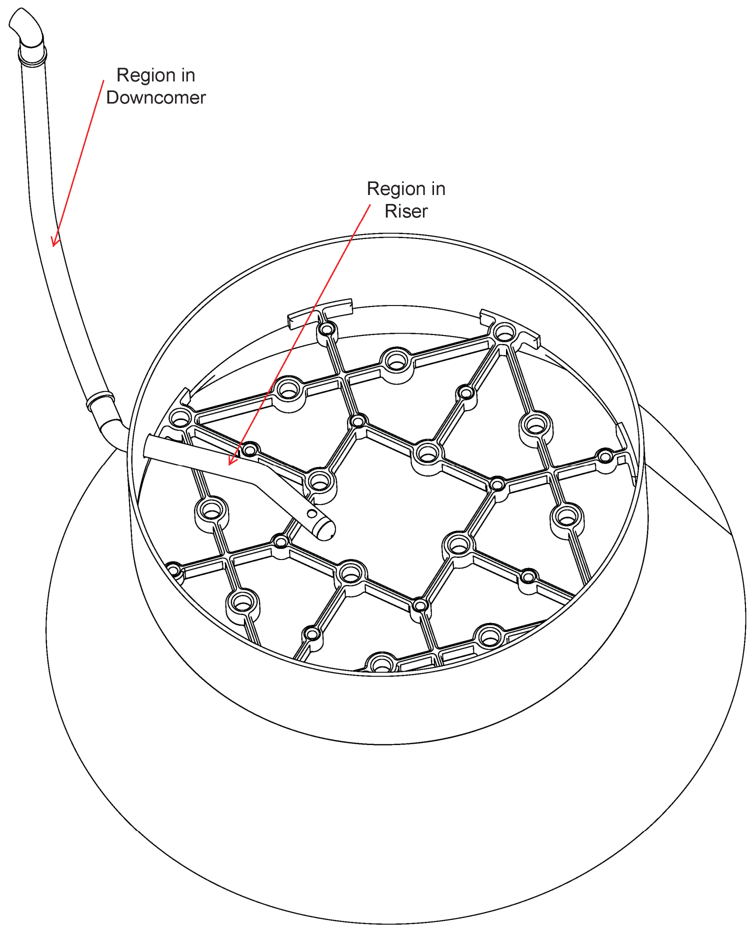


Figure 2-22 Chemical and volume control system injection reactor vessel internals

#### 2.3.6.3 Flow Diverter

A flow diverter is attached to the RPV bottom head under the core support assembly (Figure 2-23). This flow diverter smooths the turning of the reactor coolant flow from the downward flow outside the core barrel to upward flow through the fuel assemblies. The flow diverter reduces flow turbulence and recirculation, and minimizes flow-related pressure loss in this region. Because flow in this region is turbulent, the flow diverter is susceptible to TB. Other FIV mechanisms are not applicable to this component.

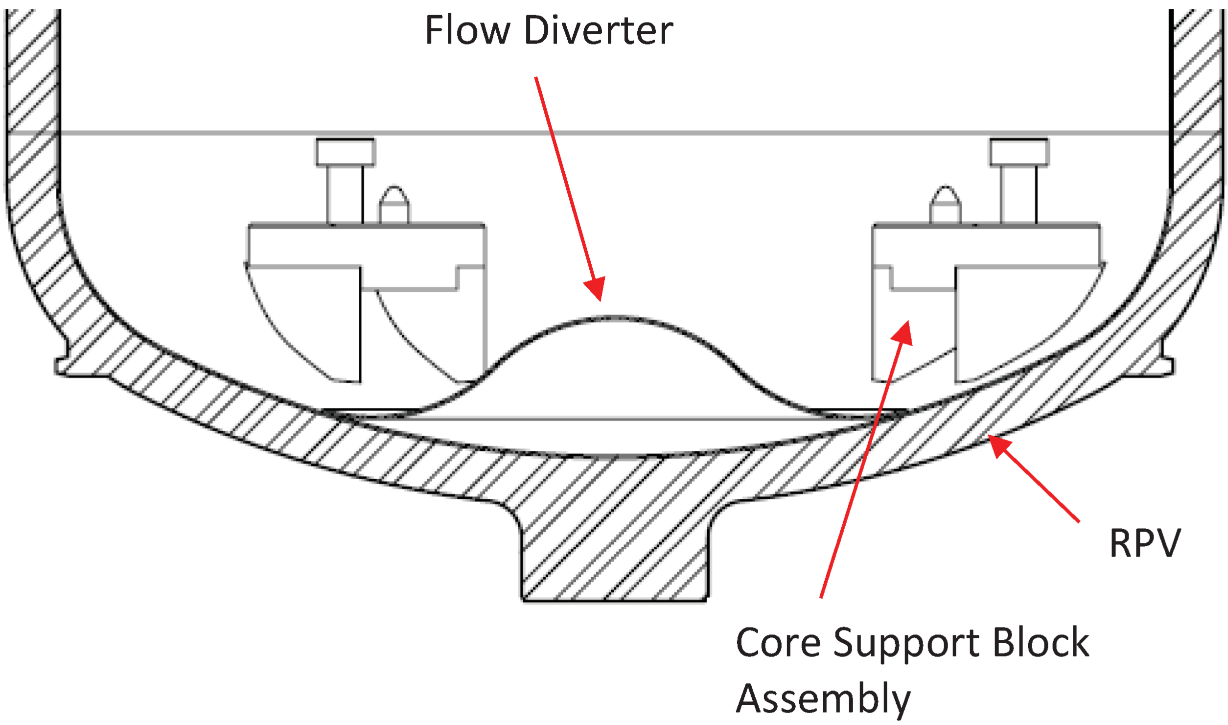


Figure 2-23 Flow diverter

#### 2.3.6.4 Thermowells

Within the primary coolant flow path, RCS temperature instruments are installed in thermowells located at the entrance to the SG tubes and in the downcomer. Thermowells are welded to the RPV. Thermowells are also used to measure temperature in secondary side piping. In locations where they are used, thermowells extend into the flow path and are exposed to turbulent cross-flow conditions. Therefore, they are susceptible to VS and TB. Other FIV mechanisms are not applicable to these components.

#### 2.3.6.5 Component and Instrument Ports

Acoustic resonances due to the generation of vortices at closed branch lines are evaluated. Penetrations that create a hollow cavity and that are located in regions with adjacent flow are susceptible to AR. Cavities form acoustic standing waves which may be excited due to the presence of vortex shedding in the vicinity. Due to the integral design of the NPM, the RPV contains very few penetrations along the primary coolant flow path. For the NPM design, the only components that meet this criterion are the primary coolant flow sensors and RRVs, which are both located in the downcomer. Due to the flow conditions and geometry in these regions, no FIV mechanisms other than AR are credible for component and instrument ports.

#### 2.3.6.6 ECCS Valves

During an accident the Emergency Core Cooling System (ECCS) may be actuated, allowing flow through the ECCS valves. The ECCS valves are an angle globe valve design, wherein flow makes a 90 degree turn and passes around the valve disc when the valve is open (See Figure 2-24). The valve disc is not in direct cross flow, and downstream structures (the valve 90 degree turn) are present to disrupt any potential vortices generated by the valve internals. The valve body is designed for reaction loads of valve discharge and seismic loads, and is therefore thick-walled relative to schedule 160 piping. It is not a bounding component for turbulent buffeting analysis. Due to the geometry in the ECCS valves, no FIV mechanisms are credible for through-valve flow.

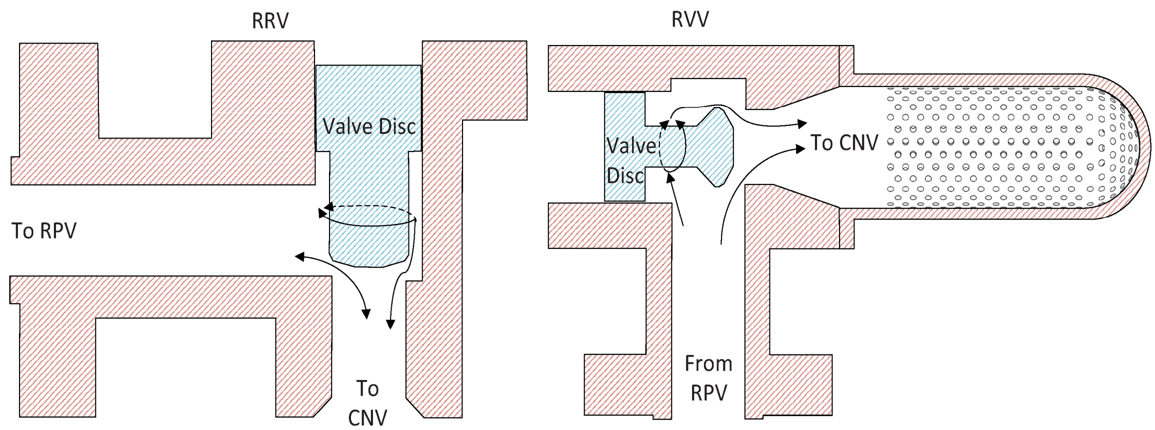


Figure 2-24 ECCS Valve Internal Flow Diagram

### 2.3.7 Primary Coolant Piping

#### 2.3.7.1 RCS Injection to RVV and RRV Reset Lines

Inside containment, the RCS injection line contains two tee locations that connect to the emergency core cooling system reset valves. During normal operation, there is no flow in the valve reset lines. One tee connection is provided in the upper region of the injection line and one is provided in the lower portion of the injection line. Vortices could form at the tee locations, and the small diameter lines leading to the reset valves represents a flow occluded region. Therefore, these tee locations are susceptible to acoustic resonance. Due to the flow conditions and geometry in these regions, no FIV mechanisms other than AR are credible for these locations.

#### 2.3.7.2 CNTS CVC Drain Valves

Each of the CNTS CVC piping lines: injection, discharge, pressurizer spray, and RPV high point degasification, contain short branch connections to a small diameter drain valve. These drain valves are closed during normal operation, and the short branch represents a flow occluded region. Therefore, these tee locations are susceptible to acoustic resonance. Due to the flow conditions and geometry in these regions, no FIV mechanisms other than AR are credible for these locations.

### 2.3.8 Leakage Flow Instability Screening Using Critical Gap Velocity

References 8.1.15 and 8.1.16 define a methodology for evaluating the hydrodynamic added mass, damping, and stiffness due to the fluid dynamic forces caused by the coupled motion of the walls of a tapered passage. Reference 8.1.16 applies to a tapered one-dimensional passage coupled to walls with one rotational and one translational degree of freedom. Reference 8.1.15 applies to a tapered annular passage with a wall having a single translational degree of freedom. Theoretical values obtained using the methodology of References 8.1.15 and 8.1.16 correspond well to those obtained from experiments (Reference 8.1.17). Therefore, the methodology of Reference 8.1.16 is a valid approximation to quantitatively assess the potential for LFI at annular passages adjacent to beam or tube type structures such as the CRAGT, ICIGT, CRD shaft sleeve, and CRD shaft supports.

Critical velocity evaluations are performed to screen reactor vessel internals components 1 – 10 in Table 2-4. The pressure differences shown in Table 2-4 are estimated from CFD analyses and loss coefficients used in the reactor coolant system thermal-hydraulic model. The inlet gap velocity is calculated using a formula from Reference 8.1.18 and the critical gap velocity is calculated using a methodology from References 8.1.15 and 8.1.16.

In Table 2-4, the critical velocity is defined as the velocity at which the hydrodynamic damping (with zero structural damping included) becomes negative. If positive structural damping is added to hydrodynamic damping, the critical flow velocity is higher.

It is noted that the annular gaps surrounding the CRD shaft, CRD shaft sleeve, ICIGT, and CRAGT are of uniform width, i.e., none are tapered. Nonetheless, the critical flow velocity shown in Table 2-4 for these components is calculated assuming an exit annular gap 25% greater than the inlet annular gap, which is less stable and thus more conservative.

Table 2-4 Reactor vessel internals components screened for LFI

| Reactor Module Components | | | Pressure Difference (psi) | Inlet Gap Velocity (in/sec) | Critical Gap Velocity in/sec | Notes |
|---|---|---|---|---|---|---|
| # | Interior | Exterior | | | | |
| 1 | CRD shaft | CRD shaft supports within riser | {{ | | | upflow |
| 2 | CRD shaft | top CRD shaft support | | | | upflow |
| | | | | | | downflow |
| 3 | CRD shaft sleeve | top CRD shaft support | | | }}2(a),(c) | Upflow |

| Reactor Module Components | | | Pressure Difference (psi) | Inlet Gap Velocity (in/sec) | Critical Gap Velocity in/sec | Notes |
|---|---|---|---|---|---|---|
| # | Interior | Exterior | | | | |
| 4 | CRD shaft | CRD shaft sleeve | {{ | | | upflow |
| | | | | | | downflow |
| 5 | CRD shaft | pressurizer baffle plate | | | | upflow |
| | | | | | | downflow |
| 6 | CRD shaft | upper riser hanger ring | | | | upflow |
| | | | | | | downflow |
| 7 | ICIGT | ICIGT supports within riser | | | | upflow |
| 8 | ICIGT | lower ICIGT support | | | | upflow |
| 9 | CRAGT | CRAGT support plate | | | | upflow |
| 10 | CRD shaft | CRD shaft alignment cone | | | }}2(a),(c) | upflow |
| 11 | upper riser assembly at slip joint | lower riser assembly at slip joint | {{ }}2(a),(c) | Slip joint is maintained in a closed condition (Section 2.3.3.2). | | |
| 12 | SG inlet flow restrictor | SG tube | {{ }}2(a),(c) | A separate effects test is performed to validate that LFI is not a concern (Sections 2.3.1.5 and 4.1.1). | | |

Note(s) for Table 2-4:

1. This velocity is calculated based on a pressure drop for inflow to the pressurizer during a reactor safety valve actuation, which is bounding. Gap velocities during normal steady-state operation are significantly lower (Section 2.3.3.4)

The screening evaluations indicate that LFI is not a concern for the leakage paths around the control rod drive shaft, ICIGT, CRAGT, and control rod drive shaft sleeve. The calculated critical velocity for LFI exceeds the actual gap velocity in each instance. No additional testing or analyses are recommended for these components. The riser slip joint and SG inlet flow restrictor are screened using alternate approaches as discussed in the sections cited in Table 2-4.

## 2.4 Regulatory Requirements

Consistent with RG 1.20, Section 2, the prototype CVAP for the NPM is composed of three sub-programs. The program includes

- a vibration and stress analysis program
- a vibration measurement program

- an inspection program

The analysis program uses theoretical analysis to predict the natural frequencies, mode shapes, and structural responses of the NPM components to various sources of flow excitations.

The measurement program consists of prototype testing that is used to validate the analysis program inputs, results, and margins of safety. Prototype testing consists of separate effects and initial startup tests. The measurement program verifies the structural integrity of the NPM components. If discrepancies are identified between the analysis and the measurement programs, reconciliation is performed.

The inspection program consists of inspections of the applicable NPM components before and after initial startup testing in order to confirm that the vibratory behavior of the susceptible components is acceptable. Inspection is generally performed outside the NPM, but if the components are not separable, then an in situ inspection process can be specified. Inspections consist of visual examinations.

To finalize the CVAP, two additional technical reports are developed. The first report contains the measurement program details for each prototype test, including test operating conditions, test durations, instrument types and locations, applicable testing hold points, and pre-test predictions of the expected and allowable experimental results, considering bias errors and random uncertainties. The second report provides the post-test evaluation of the testing completed to support the measurement program. In this report, the differences between the expected and measured experimental results are dispositioned and all results are confirmed to be in the analytically predicted allowable ranges. The second report also documents the inspection program results.

## 2.5 Classification of NuScale Power Module

Regulatory Guide 1.20 provides guidance to verify the structural integrity of the NPM internals susceptible to FIV. The verification measures depend upon the classification of the internals.

The NPM represents a unique design in its size, arrangement, and operating conditions, although its technology is based on well-proven light water reactor designs with long operational experience. Accordingly, the first operational NPM is classified as a prototype in accordance with RG 1.20. After the first NPM is qualified as a valid prototype, subsequent NPMs will be classified as non-prototype Category I.

Given its prototype classification, the NuScale CVAP addresses the applicable criteria of RG 1.20, Section 2.

## 3.0 Vibration Analysis Program

The analysis program begins with a list of FIV phenomena and a list of components that could be subjected to these phenomena. Screening criteria for each FIV phenomena are developed from literature, References 8.1.3 and 8.1.4, as discussed in Section 2.3.

Due to the very low primary coolant flow rates and passive safety designs, many regions of the NPM are not susceptible to FIV and do not meet FIV screening criteria. For example, the annular region between the outside of the RPV and the inside of the CNV is similar in geometry to the core barrel and reactor vessel of conventional PWRs, which is a location typically susceptible to FIV. For the NuScale design, this region only contains liquid when the NPM is filled with reactor pool water in preparation for refueling, and during accident scenarios when primary or secondary coolant condenses on the CNV wall and accumulates in the annular space. In both of these scenarios, the bulk flow rates in the annular region are laminar and eventually settle into static, pool conditions. Therefore, while this region appears similar to a region that would typically screen for FIV in conventional reactor designs, FIV is not a concern in this region for the NuScale design.

For NPM components or structures that meet the screening criteria for a phenomenon, analysis is performed to confirm whether the structure or component is susceptible to the FIV phenomena. For the NPM components that are evaluated for TB, the response of these structures, in terms of vibrational amplitude and stress, to this source of flow excitation is determined.

There are six FIV phenomena that are evaluated for NPM components:

- fluid elastic instability
- vortex shedding
- turbulent buffeting
- acoustic resonance
- leakage flow instability
- flutter and gallop (F/G)

The analysis program provides methodologies to analyze the screened components for each type of FIV mechanism. This analysis work can be divided into two categories:

- developing FIV inputs that are common to each of the analyses or components
- developing specific analyses to determine the susceptibility and response of the components to the various sources of flow excitation

For all phenomena with the exception of TB, the FIV mechanisms are characteristic of a strong fluid-structure coupling system. The NPM components are designed so there is a sufficient margin of safety to the potential onset of these FIV phenomena. Turbulent buffeting occurs when a component is subject to turbulent flow, which is the dominant flow condition of the primary and secondary coolant. For TB, the fluid-structure coupling is weak and results in low amplitudes of vibration. Provided the impact stresses and fatigue usage are not detrimental to the component or structure over the design life, the acceptance criteria for this source of flow excitation are met.

Per Section 2.2 of RG 1.20, the purpose of the measurement program is:

*"… to verify the structural integrity of the reactor internals, determine the margin of safety associated with steady-state and anticipated transient conditions for normal operation, and confirm the results of the vibration analysis."*

The results of the measurement program are used to validate FIV analysis inputs, results and the margins of safety. Due to the first of a kind NPM design, component screening analysis errs on the side of including potentially susceptible components, even when they could be excluded based on engineering judgment or precedent. This is a conservative engineering approach that minimizes the risk of failing to analyze a significant component. Compared to the existing PWR and boiling water reactor designs, the natural circulation design of the NPM is inherently less susceptible to FIV due to the lower primary coolant velocities. Based on these two factors, FIV analysis results demonstrate that many components have very large margins of safety. The margin of safety is the means by which structural integrity is assured. Therefore, when a margin of safety is sufficiently large, validation by testing is not necessary.

The following subsections provide an overview of the analysis program. The scope of the measurement program is determined based on the results of the analysis program. This report also identifies the prototype components that require inspection before and after the measurement program. Because the measurement program focuses on the limiting components, inspection is used to confirm the assumptions regarding which components are limiting. For the components that are instrumented to support the measurement program, inspection provides a secondary confirmation to the FIV performance and integrity of these structures.

The applicability of the FIV phenomena to various components in the NPM is summarized in Table 3-1 and the components are identified in Figure 3-1. In this table, the triangle symbol indicates that a component meets the screening criteria for the FIV mechanism and requires evaluation in the analysis program. Dashes indicate that the mechanism is not credible for the component. The locations of the components are discussed in detail in Section 2.3.

Table 3-1 NuScale Power Module components and their susceptibility to flow-induced vibration mechanisms

| NPM Component Category | Component | FEI | VS | TB | AR | LFI | F/G |
|---|---|---|---|---|---|---|---|
| Components exposed to secondary side flow | Steam piping, nozzle, MSIVs, MSIV upstream and downstream bypass lines, CNTS MS drain valve branches | - | - | - | ▲ | - | - |
| | SG steam plenum [Note 1] | - | - | - | ▲ | - | - |
| | DHRS steam piping | - | - | - | ▲ | - | - |
| | DHRS condensate piping | - | - | - | ▲ | - | - |
| | SGS pressure relief valve branch and CNTS FW drain valve branches | - | - | - | ▲ | - | - |
| | Helical SG tubing [Note 1] | ▲ | ▲ | ▲ | - | - | - |
| | SG tube inlet flow restrictors | - | - | ▲ | - | ▲ | - |
| SG tube supports | SG tube supports | - | - | ▲ | - | - | - |
| | Lower SG supports | - | ▲ | ▲ | - | - | ▲ |
| Upper riser assembly | Upper riser section | - | - | ▲ | - | - | - |
| | Riser section slip joint | - | - | ▲ | - | | - |
| | ICIGT | - | ▲ | ▲ | - | | - |
| | CRD shaft | - | - | ▲ | - | | - |
| | CRD shaft support | - | ▲ | ▲ | - | - | - |
| | CRDS shaft sleeve | - | ▲ | ▲ | - | - | - |
| | Hanger brace | - | ▲ | ▲ | - | - | - |
| Other RVI | Pressurizer spray RVI | - | - | ▲ | - | - | - |
| | CVCS injection RVI | - | ▲ | ▲ | - | - | - |
| | RRV port | - | - | - | ▲ | - | - |
| | Thermowells [Note 2] | - | ▲ | ▲ | - | - | - |
| | Instrument ports | - | - | - | ▲ | - | - |
| | Flow diverter | - | - | ▲ | - | - | - |
| Core support assembly | Core barrel | - | - | ▲ | - | - | - |
| | Upper support block | - | ▲ | ▲ | - | - | - |
| | Fuel pin interface | - | - | ▲ | - | - | - |
| | Core support block | - | ▲ | ▲ | - | - | - |
| | Reflector | - | - | ▲ | - | - | - |
| | Lower core plate | - | ▲ | ▲ | - | - | - |

| NPM Component Category | Component | FEI | VS | TB | AR | LFI | F/G |
|---|---|---|---|---|---|---|---|
| Lower riser assembly | Lower riser section | - | - | ▲ | - | - | - |
| | CRAGT support plate | - | ▲ | ▲ | - | - | - |
| | CRAGT assembly | - | ▲ | ▲ | - | - | - |
| | Upper core plate | - | ▲ | ▲ | - | - | - |
| Primary coolant piping | RCS injection to RVV/RRV reset lines | - | - | - | ▲ | - | - |
| | CNTS CVC drain valve branches | - | - | - | ▲ | - | - |

Note(s) for Table 3-1:

1. Component is exposed to primary and secondary coolant flow.
2. Thermowells also evaluated in NPM piping exposed to secondary coolant flow.

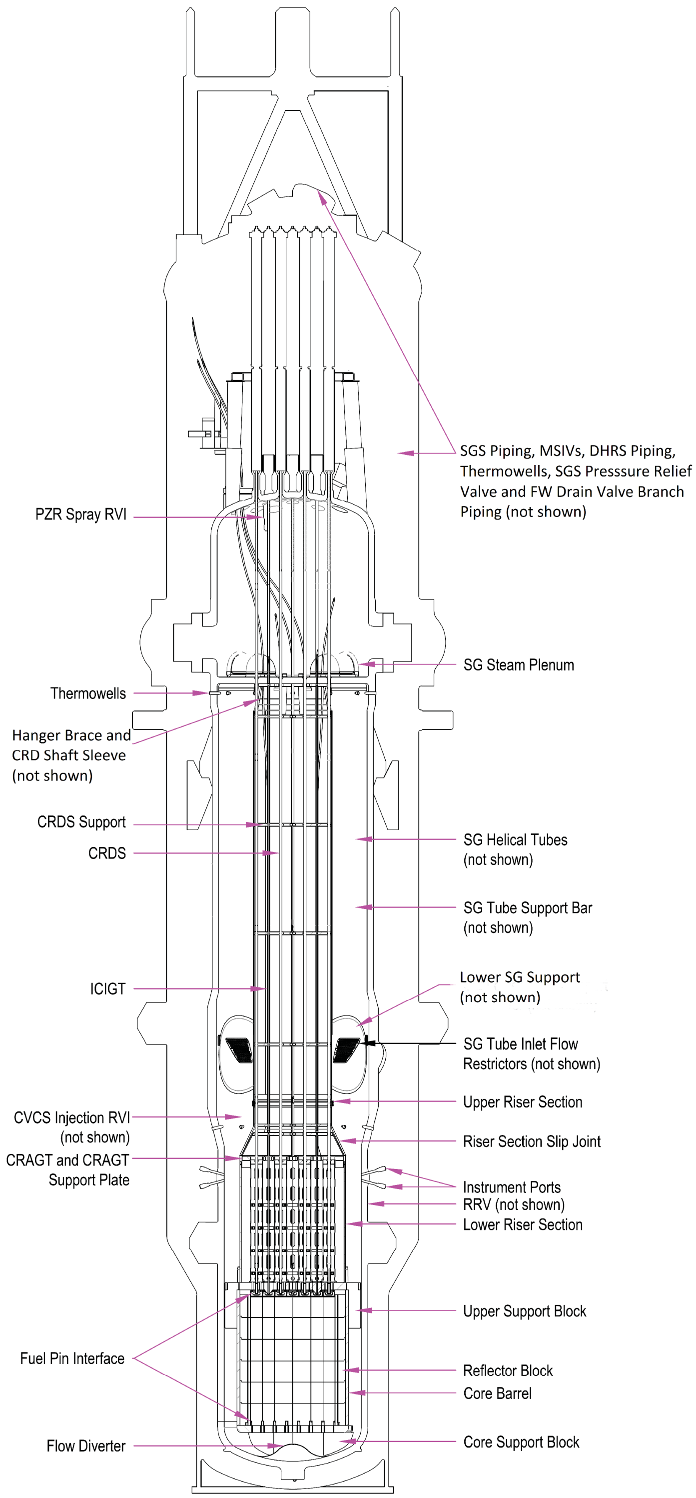


Figure 3-1 NuScale Power Module components and regions that meet flow-induced vibration screening criteria

## 3.1 Analysis Program Inputs

For components that screen for FIV, analysis is performed to determine component susceptibility. The general criterion used to determine if testing is required to verify an analytical input, method, or result is the calculated margin of safety.

The margin of safety is the means by which structural integrity against FIV degradation is assured. For strongly-coupled FIV mechanisms, the percentage difference between the analytically predicted value and the acceptance criteria that represents the predicted onset of the mechanism for a component is the margin of safety. For TB, safety margins are evaluated for the analytically predicted fatigue against the limits that are acceptable over the component design life.

Due to the low primary coolant flow rates in the NuScale design, the safety margin for most mechanisms and components is greater than 100 percent. For safety margins of these magnitudes, uncertainty and error in the inputs and other analytical simplifications can be tolerated without impacting the acceptability of the safety margin. There are few NPM components that are determined to have less than a 100 percent margin of safety. To ensure that testing is performed to validate a sufficient breadth of the analysis program, a safety margin of 100 percent is specified as the limit below which analytical inputs, methods, and safety margins are validated. It is typical that the differences between analytical and tested results are within 20 percent, so the recommended threshold for requiring testing is considered highly bounding.

The following sections discuss the analysis program inputs, many of which are common among the different FIV analyses, as identified in Table 3-2. Analysis program methods and results are provided in Section 3.2.

Table 3-2 Selected common inputs for flow-induced vibration analysis

| **Phenomena** | **Structural Natural Frequency ($f_n$)** Section 3.1.1 | **Structural Mode Shape ($\varphi$)** Section 3.1.1 | **Flow Velocity (U)** Section 3.1.2 | **Damping Ratio ($\zeta$)** Section 3.1.3 |
|---|---|---|---|---|
| FEI | ▲ | ▲ | ▲ | ▲ |
| VS | ▲ | ▲ Note(1) | ▲ | ▲ Note(1) |
| TB | ▲ | ▲ | ▲ | ▲ |
| AR | - | - | ▲ | - |
| LFI | - | - | - | - |
| F/G | ▲ | - | ▲ | - |

Note(s) for Table 3-2:

1. Mode shape and damping ratio are used in the SG tube and ICIGT VS evaluations. These inputs are not required for VS evaluation of other components.

### 3.1.1 Structural Natural Frequency and Mode Shapes

The natural frequencies and mode shapes of the structures are required for FEI, VS, TB and F/G analyses. Using the finite element analysis software ANSYS (Reference 8.1.11), complex shapes, such as a helical SG tube with transition bends can be modeled. Once appropriate finite element models are created, developing physically accurate boundary conditions, such as the interface between the tubes and tube supports or the hydrodynamic interaction between a structure and the surrounding fluid, require sensitivity studies and potentially confirmatory testing. Based on these considerations, the two analytical focus areas in the FIV structural model development are the definition of boundary conditions and modeling of hydrodynamic mass effects. For inputs where these two items have greater uncertainty and the uncertainty has a significant effect on the margin of safety, verification of frequency and mode shape results by testing is specified.

Boundary conditions for geometries where small gaps may exist, such as between a CRD shaft and its support, may increase uncertainties. These clearances may lead to non-linear behavior and the assumptions used in establishing the boundary conditions at these locations may affect structural frequencies. A CRD shaft support may be "inactive" if a sufficiently large clearance exists between the CRD shaft and the support. At locations where there is the potential for an inactive support condition, the frequency and mode shape evaluation accounts for the limiting expected support arrangement in order to determine bounding structural frequencies and mode shapes. The SG tubes are evaluated with sliding and fixed boundary conditions to represent variability in the contact at the tube supports.

The effect of the surrounding fluid (e.g., hydrodynamic mass) also affects structural frequencies. Typically, simplified "lumped mass" approaches are employed to account for the effect of surrounding fluid on structural frequencies; the mass of the surrounding fluid is included in determining a total mass of the structure for analytical purposes.

In general, analytical models consider the normal design conditions and do not consider all possible uncertainties and biases, such as those associated with manufacturing tolerance and material property allowable ranges. Where possible, values are selected to provide bounding frequency and mode shape results. For TB analysis, the natural frequencies and mode shapes have a smaller effect on the results due to the relatively weak coupling of the fluid and structure and the broad frequency spectrum that is analyzed. With regards to FEI and VS, the frequencies and for the SG tubes the mode shapes have a higher importance for accurately predicting the margin of safety. The selection of mesh size for the SG structural model is justified through mesh sensitivity studies that indicate that the selected mesh size is appropriate for the frequency range of interest. As part of the pre-verified status of the ANSYS software, software test cases for modal analysis were validated against known solutions for common geometries, using a variety of element types and boundary conditions. The performance of mesh sensitivity studies and test cases ensures that ANSYS is providing reasonable outputs for the given inputs. The methodology for selecting the mesh densities, boundary conditions, and fluid loading for the SG tubes is to be validated against the TF-3 test results. The boundary conditions used in the current design analysis have been informed by preliminary testing at the TF-3 facility to provide the most prototypic results possible at this time.

Table 3-3 identifies if the analytical inputs require verification, which is based on the safety margin identified in Sections 3.2.1, 3.2.2, and 3.2.3, and the testing that will be performed to provide the required analytical input verification.

If a margin of safety greater than 100 percent is determined, it is not necessary to verify the component frequencies and mode shapes by testing. For some components, verification is not required for each mechanism due to differences in the safety margins. However, if the test data is available for a component, it is used for post-test evaluation of all mechanisms for that component, to the extent practical.

Table 3-3 Structural natural frequencies and mode shapes input summary

| Component | Analysis Category | Safety Margin |
|---|---|---|
| Helical SG tubing | FEI | {{ }}2(b),(c),ECI |
| | TB | {{ }}2(b),(c),ECI |
| | VS Note(1), | {{ }}2(b),(c),ECI |
| ICIGT | VS Note(1) | {{ }}2(b),(c),ECI |
| | TB(2) | {{ }}2(b),(c),ECI |

Note(s) for Table 3-3:

1. Mode shape and damping ratio are used in the SG tube and ICIGT VS evaluations. These inputs are not required for VS evaluation of other components.
2. Component has very low fatigue usage and prototype testing will not be performed. ICIGT will be instrumented during initial startup testing.

### 3.1.2 Flow Velocity

#### 3.1.2.1 Steady State Velocity Analysis

Appropriate representation of the flow conditions is essential to performing the analysis of all FIV phenomena. The flow velocities in the axial/parallel and crossflow directions relative to the normal axes of each component are needed as analytical inputs. Due to the importance of generating appropriate flow rates, RCS and secondary-flow rates for the various regions of the NPM are determined and verified using three different analytical methods: hand calculation, thermal-hydraulic (TH) analysis, and computational fluid dynamics (CFD) analysis. Additionally, separate effects tests have been performed to validate the significant inputs in the TH modeling, such as core and SG form losses and SG performance, which is a significant contributor to the natural circulation buoyant force. The TH analysis provides validated maximum design flow rate results based on testing. The CFD average flow velocities (Reference 8.1.12) and hand calculations provide additional assurance of the validated flow rates. Maximum design flow rates (or higher) are used in all FIV evaluations. Maximum design flow represents the highest expected primary coolant flow rate used in design and safety analyses and is the licensing basis flow rate; operation above this flow rate is not permitted without evaluation against safety analysis limits.

Several biases are applied in the TH analysis to ensure the calculation of a bounding maximum design flow rate. The steam generator and fuel assembly loss coefficients are biased based on the uncertainty in the loss coefficient correlations. Other pressure losses are decreased to be bounding. Factors such as core bypass flow, and steam generator heat transfer are biased to provide a bounding primary coolant flow rate.

- Steam generator loss coefficient bias: The best-estimate loss coefficient for the steam generator is decreased by {{ }}$^{2(b),(c),ECI}$ for the maximum design flow calculation. This accounts both for error in the loss coefficient correlation, as determined by comparison to test data, and test data measurement uncertainty.

- Fuel assembly loss coefficient bias: {{ }}$^{2(b),(c),ECI}$ is subtracted from the overall fuel assembly loss coefficient to bound loss coefficient correlation confidence and experimental uncertainty. The bias value is based on pressure drop testing of a full-size prototype of the NuScale fuel assembly.

- Reactor coolant system form loss coefficient bias: For the regions in the RCS loop except for the core and steam generator, the best-estimate form loss coefficients are reduced by {{ }}$^{2(b),(c),ECI}$ for calculation of the maximum design flow. Form loss coefficients for most of the components in the RCS loop are based on empirical correlations. Using a comparison of test data to an empirical correlation, a {{ }}$^{2(b),(c),ECI}$ reduction is appropriate for regions outside the SG and core. The sum of the SG and core form losses is greater than {{ }}$^{2(b),(c),ECI}$ of the total pressure loss in the RCS loop, making the choice of form loss coefficient elsewhere of less significance.

- Core bypass flow bias: To address uncertainty in the fraction of the total RCS flow rate that bypasses the core, {{ }}$^{2(b),(c),ECI}$ of the total RCS flow is added to the best estimate core bypass flow for the maximum design flow conditions.

- Steam generator heat transfer bias: Thermal conductivity of the steam generator tube wall material is decreased by a factor of {{ }}$^{2(b),(c),ECI}$ for the maximum design flow calculation. Changing the tube wall conductivity in this manner has a significant effect on the value of the overall heat transfer coefficient. The equivalent secondary side film coefficient variation produced by the change in wall conductivity bounds the average inaccuracies in the TH analysis heat transfer predictions as compared to the steam generator test data.

For most FIV mechanisms, the free-stream velocity is the required input. Free-stream velocity is the incident velocity before it impacts a susceptible target. However, in the FIV analyses it is typically assumed to be the flow velocity as it passes the target or constriction. The flow area upstream of the constriction is larger than the flow area associated with the constriction; therefore, the average free-stream velocity is lower than the average velocity at the target. While there may be local variations in the free-stream velocity, it is not credible that these variations would occur uniformly at the target face at a velocity higher than the average velocity through the constricted area.

#### 3.1.2.2 Transient Velocity Analysis

Transient analysis is performed to determine pressures, temperatures, flow velocities, and static quality for the various NPM regions to support ASME stress analysis. The NPM typically operates at steady-state, full-power conditions. At lower reactor power levels, natural circulation flow decreases due to the lower heat addition from the core. Based on a review of time-history analyses, there are no normal operating transient responses that result in higher flow rates than are achieved at maximum design full-power conditions. Normal operating transients, such as load following, ramp decreases in reactor power, and reactor trips, result in decreases in flow velocities relative to full-power operating conditions due to the lower heat addition from the core. For conservatism, flow velocities 5% or greater than the maximum design flow value should be used for all FIV mechanisms except for turbulent buffeting. For turbulent buffeting, maximum design flow velocities are acceptable.

#### 3.1.2.3 SG Tube Gap Velocity for FEI and VS Evaluations

In accordance with ASME N-1331.1, the flow velocity in the gaps between the tubes is calculated based on the approach flow velocity that would occur if the tubes were not present, multiplied by the ratio of the tube pitch and divided by the pitch minus the diameter. The flow through the SG assembly is dominated by the vertical flow, therefore the radial pitch (distance between two columns) of {{ }}$^{2(b),(c),ECI}$ is more appropriate to use than the axial pitch {{ }}$^{2(b),(c),ECI}$ (vertical distance between the centerlines of two adjacent tubes in the same column). Using radial pitch is conservative compared to the actual distance between tubes in each column, which would be the diagonal of the radial and axial pitches, {{ }}$^{2(b),(c),ECI}$. The ratio of the radial pitch divided by the difference between the radial pitch minus the tube outer diameter is equal to 2.204.

The approach/mean velocity is based on the maximum design flow rate of 660.5 kg/s at 100% reactor power and the flow area between the RPV and the riser, minus the area of the tube supports and backing strips. FEI is applicable to the helical portions of the tube bundle, and primary coolant density increases from the top of the bundle (steam plenum side) to the bottom (feedwater plenum side). Based on the cold leg temperature of 507.8ºF, the density at the bottom of the tube bundle is 49.11 lb/ft$^3$. Using these parameters, the free stream velocity is 8.87 in/s. The gap velocity is equal to (8.87 in/s)(2.204)= 19.56 in/s. Based on the hot leg temperature of 590.0ºF, the density at the top of the tube bundle is 43.59 lb/ft$^3$. Using these parameters, the free stream velocity is 10 in/s. The gap velocity is equal to (10 in/s)(2.204) = 22.04 in/s.

A 5% increase in the velocity is accounted for per Section 3.1.2.2 to bound transient changes in velocity that may be experienced during operating conditions. This results in a gap flow velocity of 23.14 in/s near the steam plenum side of the bundle and 20.54 in/s near the feedwater plenum side of the bundle, per ASME N-1331.1 for the FEI evaluation.

VS is applicable to the transition bend portion of the tube near the FW plenum. Based on an increased inner diameter of the RPV in the transition bend region ({{ }}2(b),(c),ECI in the helical region), and the decreased density of the SG tubes due to the termination of each subsequent column in the FW plenum, the gap velocity used in the VS evaluation is 19.38 in/s.

### 3.1.2.4 Velocities Used in Flow-Induced Vibration Analysis

To determine the crossflow velocities for components subject to a combination of crossflow and axial flow, since bounding estimates of cross flow velocities can be assumed while still showing acceptable FIV performance, it can be conservatively assumed that the crossflow on any component is equal to the calculated total flow velocity for that component. This method overestimates the forcing function of the vibration. Based on the low reactor coolant flow rate for the NuScale design, this approach is a conservative estimate that produces acceptable margin.

Table 3-4 identifies the methods of obtaining the velocities that are used in the FIV analyses and which flow rate analysis derives them. Bounding flow conditions are used in all analyses. Testing to validate the velocities is not necessary because these FIV inputs have already been validated with separate effects testing, and one or more analytical method and bounding maximum design flow velocities are used.

Table 3-4 Flow conditions input summary

| Analysis Category | Assumed Conditions | Analysis Method |
|---|---|---|
| FEI | Maximum design flow – average velocity | TH |
| VS | Maximum design flow – average velocity | TH [Note 2] |
| AR | Maximum design flow – average velocity | CFD/TH [Note 2] |
| | Maximum CVCS flow – average velocity | TH [Note 3] |
| F/G | Maximum design flow – average velocity | CFD |
| LFI | None [Note 1] | None |
| TB | Maximum design flow – average velocity | CFD |

Note(s) for Table 3-4:

1. LFI confirmation is by prototype testing only for components that are screened as potentially susceptible to LFI.
2. For the evaluation of AR and VS mechanisms for components exposed to secondary coolant flows, the TH flow is used. CFD analysis is not performed to characterize secondary side flow.
3. For the evaluation of AR mechanisms for components within piping exposed to primary coolant, the maximum CVCS flow is used.

Table 3-5 lists velocities used in the analyses of components subject to FIV. The analysis methods that produce these velocities are identified in Table 3-4, except as noted otherwise below.

Table 3-5 Velocities used in FIV analyses

| Analysis Category | Component | Velocity (in/s) |
|---|---|---|
| FEI Note 1 | Helical SG tubing | {{ |
| VS Note 1 | Helical SG tubing | |
| | Lower SG support | |
| | RCS hot region thermowell | |
| | RCS cold region thermowell | |
| | CNTS steam thermowell | |
| | CNTS feedwater thermowell | |
| | Control rod drive shaft sleeve | |
| | CRD shaft support | |
| | Control rod assembly guide tubes | |
| | CRAGT support | |
| | Upper riser hanger brace | |
| | CVCS Injection RVI (in downcomer) | |
| | In-core instrument guide tubes | |
| AR Note 4, Note 5 | DHRS steam line tee | |
| | DHRS condensate line tee | |
| | Reactor recirculation valve nozzle | |
| | Flowmeter port | |
| | FW drain valve | |
| | MS drain valves | |
| | MSIV upstream and downstream bypass lines, | |
| | SGS pressure relief valve | |
| | CNTS CVC drain valves: Injection line | |
| | CNTS CVC drain valves: Discharge line | |
| | CNTS CVC drain valves: Pressurizer Spray line | |
| | CNTS CVC drain valves: Degasification line | |
| | CNTS CVC drain valves: Degasification line with N2 | |
| | RCS Injection to RRV and RVV reset lines | }}2(b),(c),ECI |

| Analysis Category | Component | Velocity (in/s) |
|---|---|---|
| F/G Note 1 | Lower SG support | {{ }}2(b),(c),ECI |
| LFI | None, LFI confirmation is by prototype testing only for components that are screened as potentially susceptible to LFI. | |
| TB | Helical SG tubing, primary flow | {{ |
| | Helical SG tubing, secondary flow (steam) | |
| | Helical SG tubing, secondary flow (liquid) | |
| | SG inlet flow restrictor | |
| | Lower SG support | |
| | CRAGT inner diameter | |
| | CRAGT outer diameter | |
| | CRD shaft | |
| | Lower ICIGT | |
| | Upper ICIGT | |
| | RVI assembly | }}2(b),(c),ECI |

Notes for Table 3-5:
1. 5% margin is included in these values for transient velocity changes
2. Primary side gap velocity based on ASME N-1331.1
3. Component of this velocity perpendicular to the tubes is used in the analysis
4. 5% is added to these values in the analysis to account for transient velocity changes
5. Velocity is from TH analysis

### 3.1.3 Damping Ratios

Damping can be created from various sources, such as material, fluid viscosity, or structural interactions. Damping reduces a structural response. The damping ratios for structures have historically been determined through testing. ASME Boiler and Pressure Vessel Code, Appendix N-1300 (Reference 8.1.2) provides recommendations for damping ratios of SG tubes.

- Analysis for FEI of SG tubes: damping due to viscous effects of the primary fluid is not credited. Damping created by other sources (material and structural interaction) is expected to be 1.5 percent based on the guidance in Paragraph N-1331.3 of Appendix N. The damping ratio has a significant influence on the stability ratio that is compared to the acceptance criteria, which represents the margin to the onset of FEI for the SG tubes. Additionally, RG 1.20 states that any attempt to specify structural damping coefficients greater than 1 percent for frequencies greater than seismic frequencies should be supported by experimental measurements. Therefore, prototype testing is required to confirm that the damping ratio of 1.5% that is credited in the FEI analysis for the SG tube is appropriate.
- Analysis for VS of SG tubes and ICIGTs: a damping ratio of 1.0 percent is used. Prototype testing is planned to confirm the damping ratio of 1.0% that is credited in the VS analysis for the SG tube is appropriate because the safety margin for the SG tube is less than 100%.
- Analysis for TB: a damping ratio of 0.5 percent is used for the SG Alloy 690 tubes and the stainless steel type 304 ICIGT and 0.3 percent is used for all other RVI stainless steel structures. These damping ratios are representative of hysteresis (material damping) and are less than 1%. They conservatively neglect damping due to structural interactions and viscosity. Compared with the FEI and VS analyses, a smaller damping ratio is assumed for the SG tubes because lower amplitudes of vibration with less tube-to-tube support interactions are expected with this source of flow excitation. This guidance is consistent with Appendix N-1300. Because the damping values used in TB analysis are based only on material damping and are less than 1%, they are considered to be sufficiently bounding. It is not credible that this input could have a non-conservative effect on the calculated margin of safety. Therefore, testing is not required to verify the damping values used in TB analyses.
- Analysis for LFI: because components undergo prototype testing if the possibility of LFI is indicated by screening evaluations, further analysis is not recommended and a damping value is not provided.

As summarized in Table 3-6, the only damping values that require verification are the SG tube damping values used in the VS and FEI analyses. The basis for verifying these analytical inputs is the margin of safety, as identified in Sections 3.2.1, 3.2.2, and 3.2.3.

Table 3-6 Damping ratios input summary

| Component | Analysis Category | Safety Margin | Damping Value Used | Verification Method and Testing Phase | Test |
|---|---|---|---|---|---|
| Helical SG tubing | FEI | {{ }}2(b),(c),ECI | 0.015 | Separate Effects | SG FIV (Section 4.1.2) |
| | VS | {{ }}2(b),(c),ECI | 0.01 | | |

## 3.2 Analysis Program Methods and Results

### 3.2.1 Fluid Elastic Instability

The helical SG tubing is the only component that is susceptible to FEI.

Fluid elastic instability is a phenomenon that has no vibration effects on tubing unless a critical threshold velocity of crossflow is exceeded. The analysis for FEI is performed by calculating a critical velocity and a mode shape-weighted mean-pitch velocity for each mode. The velocities are compared to determine the stability ratio of the tube.

An FEI analysis of the inner, middle and outer SG tube columns is performed as these tube locations provide bounding frequencies and mode shapes. Two tubes within each column, and both fixed and sliding boundary condition frequencies and mode shapes are evaluated. The reduced pitch velocities are computed using the crossflow component of the primary flow normal to the tube and the translational mode shapes per Equation 82 of N-1331.2 of Reference 8.1.2, recognizing that mode shapes for the lower frequencies are dominated by axial and torsional motions. Bounding Connors' equation values are used to provide conservative values for the critical velocity (i.e., lowest velocity) based on experimental results in Chen (Reference 8.1.23), and Appendix N-1330 of the ASME code.

The FEI analysis results show the maximum reduced pitch velocity, a dimensionless number determined for comparison against the critical velocity, is {{ }}2(b),(c),ECI for column 21 and {{ }}2(b),(c),ECI for column 1. The reduced pitch velocity is higher for Column 21 because of the longer spans of tubing between supports and the correspondingly lower natural frequencies. For both columns, the minimum reduced critical velocity is {{ }}2(b),(c),ECI for the steam region, and {{ }}2(b),(c),ECI for the FW region.

Even for the most limiting SG tube location, margin to the FEI acceptance criteria and the predicted onset of FEI is provided. However, due to the prototype classification of the design, the significant analysis inputs and margin of safety are verified by testing because the margin of safety is less than 100 percent for the outermost columns in the SG. The FEI reduced velocity results provided in Table 3-7 are plotted in Figure 3-6.

Table 3-7 FEI reduced critical velocities and safety margins for Connor's Constants of C=1.9 a=0.05

| Column, Boundary Condition and Tube | Limiting Frequency (Hz) | $U_{i,r}$ Steam Region | Safety Margin Steam Region | $U_{i,r}$ FW Region | Safety Margin FW Region |
|---|---|---|---|---|---|
| Column 21 Fixed Tube A | {{ | | | | |
| Column 21 Fixed Tube B | | | | | |
| Column 21 Sliding Tube A | | | | | |
| Column 21 Sliding Tube B | | | | | |
| Column 11 Fixed Tube A | | | | | |
| Column 11 Fixed Tube B | | | | | |
| Column 11 Sliding Tube A | | | | | |
| Column 11 Sliding Tube B | | | | | |
| Column 1 Fixed Tube A | | | | | |
| Column 1 Fixed Tube B | | | | | |
| Column 1 Sliding Tube A | | | | | |
| Column 1 Sliding Tube B | | | | | }}2(b),(c),ECI |

### 3.2.2 Vortex Shedding

The VS analysis consists of determining component structural natural frequencies and determining the minimum component characteristic dimension that is necessary to avoid the generation of vortices at the fundamental frequency. The inherent assumption with this method is that the crossflow velocity is uniform and over the full length of the structure. Subsection N-1324.1 of Reference 8.1.2 provides four different design methods to prevent the lock-in conditions associated with VS. These are discussed in the paragraphs below as methods (a) through (d). Method (a) is the simplest method and is used for the majority of the components evaluated for VS lock-in.

Based on Method (a) of subsection N-1324, VS lock-in is avoided if the characteristic dimension of the component is greater than the free-stream velocity divided by the component fundamental frequency. Over 100 percent margin is demonstrated for the lower SG support, CVCS injection RVI, CRAGT support plate, CRD shaft support, CRAGTs, thermowells, hanger brace, upper support block, core support block and upper and lower core plates.

The SG tubes and ICIGTs are shown to avoid VS lock-in using a combination of methods (a), (b), (c), and (d) of Paragraph N-1324. All relevant modes of these components are evaluated. For most modes, more than one acceptance criterion is met.

The FIV inputs and analysis in Reference 8.1.2 are based on conservative assumptions and industry-accepted methods for estimating VS lock-in conditions. For components with greater than 100 percent margin, testing is not required to verify analysis inputs or results. Table 3-8 and Table 3-9 provide results for components evaluated with N-1324.1 Method A.

Table 3-8 VS Results for Components with Simple Geometries with N-1324.1 Method A

| Component | Frequency | Velocity | Characteristic Dimension | Safety Margin |
|---|---|---|---|---|
| SG lower support | {{ | | | |
| RCS Thermowell – hot region | | | | |
| CNTS Thermowell – steam | | | | |
| CNTS Thermowell – FW | | | | |
| Hanger brace | | | | }}2(b),(c),ECI |

Table 3-9 VS Results for Components Modeled with ANSYS with N-1324.1 Method A

| Component | Frequency – Nominal Mesh | Frequency – Coarse Mesh | Velocity | Characteristic Dimension | Safety Margin |
|---|---|---|---|---|---|
| CRAGT Assembly | {{ | | | | |
| CRAGT Support | | | | | |
| CRD shaft support | | | | | |
| CVCS injection RVI | | | | | |
| ICIGT | | | | | |
| CRD shaft sleeve | | | | | }}2(b),(c),ECI |

Notes: for Table 3-9:

1. Natural frequency is determined using a closed-from solution.

The results for the first 5 modes plus mode 9 are provided in Table 3-10 for the ICIGT. The reported safety margins are the maximum of either method A, B or C. Modes 1, 2 and 5 do not contain any mode shape in the upper riser region where cross flow occurs, so these modes cannot be excited by vortex shedding. Modes 3 and 4 are the first bending modes in the cross flow region, in the X and Z directions. Mode 9, which is the second bending mode, is also provided in Table 3-10. After mode 5, the frequencies are high enough that each subsequent mode passes using method A.

Table 3-10 VS ICIGT Assessment with N-1324.1 Methods A, B and C

| Mode | Frequency | Reduced Velocity | Reduced Damping | Passing Criteria | Safety Margin |
|---|---|---|---|---|---|
| 1 | {{ | | | | |
| 2 | | | | | |
| 3 | | | | | |
| 4 | | | | | |
| 5 | | | | | |
| 9 | | | | | }}2(b),(c),ECI |

Table 3-11 provides the results for the limiting mode of each column, tube, and boundary condition evaluated for VS in the SG design. Out of all columns, tubes and boundary conditions that are assessed for the SG tubes, there are four results with safety margins less than 100%. These modes are shown in Figure 3-2, Figure 3-3, Figure 3-4 and Figure 3-5. The lowest two safety margin values are for the fundamental frequency of Column 1 tube B, which is a bending mode of the FW transition span. The frequencies of this mode are approximately 1 Hz different due to the different boundary conditions that are assumed.

Table 3-11 VS SG Tube Results

| # | Column | Tube | Boundary Condition | Frequency | Reduced Velocity | Reduced Damping | Passing Criteria | Safety Margin |
|---|---|---|---|---|---|---|---|---|
| 1 | 1 | B | Fixed | {{ | | | | |
| 2 | 1 | B | Sliding | | | | | |
| 3 | 1 | A | Sliding | | | | | |
| 4 | 1 | A | Fixed | | | | | |
| 5 | 11 | B | Sliding | | | | | |
| 6 | 11 | B | Fixed | | | | | |
| 7 | 21 | B | Sliding | | | | | |
| 8 | 21 | B | Fixed | | | | | }}2(b),(c),ECI |

{{

}}2(b),(c),ECI

Figure 3-2 Column 1 Tube B Fixed BC, Mode 1 (lowest safety margin)

{{

}}2(b),(c),ECI

Figure 3-3 Column 1 Tube B Sliding BC, Mode 1 (2nd lowest safety margin)

{{

}}2(b),(c),ECI

Figure 3-4 Column 21 Tube B Sliding BC, Mode 1 (4th lowest safety margin)

{{

}}2(b),(c),ECI

Figure 3-5 Column 11 Tube B Sliding BC, Mode 1 (4th lowest safety margin)

### 3.2.3 Turbulent Buffeting

Although the random vibration due to TB is of much smaller amplitude than that experienced in FEI or VS, it is important because it exists whenever there is turbulent flow over a susceptible component.

Turbulent buffeting vibration can be induced in parallel flow, axial flow, or crossflow. Turbulence-induced vibration occurs due to a fluctuating pressure in the flow. The fluctuating pressure is quantified by a forcing function, which is characterized by its power spectral density (PSD) and correlation function. Three significant parameters are needed to characterize the PSD and correlation function: convective velocity, correlation length, and the PSD.

The convective velocity as implemented in the coherence function, describes the phase relationship of the forcing function at two different points on the surface of the structure. From a physical standpoint, the convective velocity is the velocity at which eddies are swept downstream, thus causing a phase shift in the fluctuating pressure between two points. Based on literature review, convective velocity can be a maximum of the free-steam velocity and a minimum of 0.5 times the free-stream velocity for parallel or axial flow. The limiting convective velocity within the range is selected when evaluating the coherence function.

The correlation length is equivalent to the scale of the largest turbulent eddy, which is a measure of the longest distance over which the velocity at two points of the flow field is correlated. For each component, the flow is characterized as parallel, cross, two phase, or axial, and the characteristic length is determined. A bounding correlation length range is determined based on literature review. Similar to the convective velocity, the range of correlation lengths is evaluated to ensure a bounding value is selected for each component forcing function.

The PSD is determined to support characterization of the forcing function. The PSD as applied to flow-induced turbulent response refers to the energy distribution of a variable as a function of frequency. Many components can be approximated as cylinders in cross flow or annular flow. For these idealized arrangements, bounding PSDs based on literature are used. Table 3-12 summarizes the PSD types that are applied to components susceptible to TB. The overall analytical uncertainty for the SG tubes is judged to be the highest relative to other susceptible components; therefore, prototype testing is performed to verify the adequacy of the PSDs used for the SG tubes.

In the TB evaluations, the vibration amplitude is not calculated for components with a first mode frequency greater than 200 Hz because vibration amplitudes are insignificant with high structural frequencies. Components with lower first mode frequencies experience higher root mean square (RMS) vibration amplitudes, which translate into more limiting fatigue results. Vibration amplitude is also not determined for components that based on flexibility and flow conditions can be shown to be bounded by the analyzed components. By design, supports are thicker and less flexible than the components they support; hence, the component is analyzed for turbulent buffeting vibrations. The results generally bound the performance of the support. Therefore, no

PSD is specified in Table 3-12 for certain components identified as susceptible to TB in Table 3-1.

Table 3-12 Turbulent buffeting power spectral density inputs used in analysis

| Flow Type and Component Shape | Applicable Components | Literature PSD Used |
|---|---|---|
| Single phase axial flow for tubes | {{ | Equation 3-2 |
| Two phase axial flow for tubes | | Equation 3-3, Equation 3-4, Equation 3-5, Equation 3-6 |
| Tube bundle cross-flow | | Equation 3-7 |
| Bounding annular flow | }}2(a),(c),ECI | Equation 3-1 |

Note(s) for Table 3-12:

1. The RVI assembly is a collection of structures that responds to turbulent buffeting forces as a group. It includes the lower and upper riser, reflector, core barrel, core support blocks, CRAGT supports, lower and upper core plates, and the CRD shaft supports.

Equation 3-1 is from Reference 8.1.3. It provides a PSD for components with annular flow velocities. It is applied to some components that experience crossflow, and this simplification is bounding based on the frequencies and characteristic lengths of the analyzed components. Due to the relatively low flow velocities, the reduced frequency for some components is larger than five. For those cases, the PSD is typically evaluated with a reduced frequency of five to provide bounding results.

$$G_p(f) = \rho^2 v_f^3 R_h 0.155 e^{-3.0F}, 0 < F < 1$$

$$G_p(f) = \rho^2 v_f^3 R_h 0.027 e^{-1.26F}, 1 < F < 5$$

Equation 3-1

where:

| | | |
|---|---|---|
| $G_P(f)$ | = | PSD of the turbulent pressure as a function of modal frequency (psi$^2$/Hz), |
| $\rho$ | = | Fluid density (lbf-s$^2$/in$^4$), |

| | | |
|---|---|---|
| $v_f$ | = | Free-stream velocity (in/s), |
| $R_h$ | = | Hydraulic radius (in), and |
| $F$ | = | Reduced frequency (-). |

The test data from Chen (Reference 8.1.13) is used to characterize a PSD for axial flow over tubes. The relationship is applicable to low Strouhal numbers.

$$G_p(f) = 0.000002720\rho^2 v_f^3 2R_h S^{-0.25}, 0 < S \le 5$$

$$G_p(f) = 0.0002275\rho^2 v_f^3 2R_h S^{-3}, 5 < S \le 10$$

Equation 3-2

where:

| | | |
|---|---|---|
| $G_P(f)$ | = | PSD of the turbulent pressure as a function of modal frequency ($psi^2$/Hz), |
| $\rho$ | = | Fluid density (lbf-$s^2$/$in^4$), |
| $v_f$ | = | Free stream velocity (in/s), |
| $R_h$ | = | Hydraulic radius (in), and |
| $S$ | = | Strouhal number (-). |

For the two-phase region, the correlations by Giraudeau from Reference 8.1.21 is implemented. These correlations are force PSDs of two-phase fluids around 90° elbows. Giraudeau found that the force PSD on an elbow is caused by the changes in the mass flux through the tube due to turbulent two-phase flow. The helical tube will experience similar behavior in the two phase region.

The Giraudeau correlations are based on Equations 17, 19, 20, and 21 from Reference 8.1.21, repeated below in Equation 3-3, Equation 3-4, Equation 3-5, and Equation 3-6. The five empirical constants that go into the Giraudeau PSD are defined in Table 1 of the reference and Table 3-13 below. The constants are defined for four void fractions. When the actual void fraction in the tube is between the tabulated values, linear interpolation is used. As a two-phase PSD, the Giraudeau correlation is only used when the void fraction is between 0.05 and 0.95. These limits are selected based on the range of void fractions used in the test data that supports the Giraudeau correlation. Note that there is a typo in Table 1 of Reference 8.1.21 for $k_2$ at a void fraction of 0.95. The typo is corrected in Table 3-13 and in the associated calculations.

The Giraudeau PSD is a force PSD on a 90° elbow. To convert the force PSD to a pressure PSD, it is divided by the area of the element squared. Also, each element only sweeps out a portion of a 90° bend. Therefore, the PSD is also multiplied by the square root of one minus the cosine of the angle swept out by the element. This factor ensures that the sum of all force vectors on elements that make up a 90° bend sum to the appropriate value. The Giraudeau PSD is for flow around an elbow and is therefore, only applied radially on the tube. The single phase PSD is also applied in the two-phase region. The single phase PSD is insignificant compared to the Giraudeau PSD in the radial direction, but provides excitation in the vertical direction.

$$\begin{aligned} \bar{f} &< \bar{f}_0 & \bar{\Phi} &= \mathrm{k}_1 \bar{f}^{m_1} \\ \bar{f} &\geq \bar{f}_0 & \bar{\Phi} &= \mathrm{k}_2 \bar{f}^{m_2} \end{aligned}$$ Equation 3-3

$$\bar{\Phi} = \frac{\Phi}{\left(\rho_l j^2 D_h{}^2\right)^2} \frac{j}{D_h} We^{0.8}$$ Equation 3-4

$$\bar{f} = \frac{f D_h}{j}$$ Equation 3-5

$$We = \frac{\rho_l j^2 D_h}{\sigma}$$ Equation 3-6

where:

| | | |
|---|---|---|
| $\bar{f}$ | = | Reduced frequency (-) |
| $\bar{f}_0$ | = | Reduced frequency of maximum force |
| $\bar{\Phi}$ | = | Normalized force PSD (-) |
| $\Phi$ | = | Force PSD (lbf$^2$/Hz) |
| $f$ | = | Frequency (Hz) |
| $D_h$ | = | Hydraulic diameter (in) |
| $\rho_l$ | = | Liquid density (lbf-s$^2$/in$^4$) |
| $j$ | = | Mixture velocity (in/s) |
| $We$ | = | Weber number (-) |
| $\sigma$ | = | Surface tension (lbf/in) |
| $\mathrm{k}_1$ and $\mathrm{k}_2$ | = | Correlation factor (-) |
| $m_1$ and $m_2$ | = | Correlation exponent (-) |

Table 3-13 Giraudeau PSD correlation empirical constant

| Void Fraction | $\bar{f}_0$ | $k_1$ | $k_2$ | $m_1$ | $m_2$ |
|---|---|---|---|---|---|
| 0.25 | 0.050 | 3.644e4 | 2.234 | 1.39 | -1.84 |
| 0.50 | 0.058 | 3.466e7 | 0.460 | 3.31 | -3.06 |
| 0.75 | 0.040 | 8.866e8 | 0.492 | 3.43 | -3.19 |
| 0.95 | 0.018 | 7.105e9 | 0.004[1] | 3.58 | -3.44 |

1. Reference 8.1.21 contains a typo in Table 3-13. The value of 0.040 is corrected to 0.004 here. The correct value can be verified by comparing the calculated PSD using the $k_1$ and $k_2$ functions at $\overline{f_0}$.

Lastly, a PSD to represent crossflow over a tube bundle geometry is specified in Equation 3-7. This equation is from Reference 8.1.3 and is expected to provide bounding results for the frequency range, flow velocities, and SG tube diameter of the NuScale SG tube bundle. As mentioned previously, this assumption is verified by testing.

$$G_p(f) = 0.01, F < 0.1$$

$$G_p(f) = 0.2, 0.1 \le F \le 0.4 \qquad \text{Equation 3-7}$$

$$G_p(f) = 5.3x10^{-4} F^{-3.5}, F > 0.4$$

where:

| | | |
|---|---|---|
| $G_P(f)$ | = | PSD of the turbulent pressure as a function of modal frequency (psi$^2$/Hz) and |
| $F$ | = | Reduced frequency (-). |

The structural response due to the turbulence is calculated using the inputs that have been discussed. Equations to determine the RMS response are assigned based on the direction of the flow and the dimension of the structure, using the appropriate PSDs, damping ratios, modal analysis results, and flow characteristics for the analyzed components. Equations 8.45, 8.46, and 8.47 of Reference 8.1.3 are the basis for the mean square response calculation. The RVI assembly is evaluated with a random vibration spectral analysis without using correlation lengths, by applying fully correlated PSDs on panels that are larger than the correlation length. Each panel is uncorrelated with the other panels. This method overestimates the correlation of the turbulent buffeting forces, producing bounding vibration estimates. Using the RMS response, degradation mechanisms associated with impact and vibration fatigue are evaluated. For the structures where the support (RVI assembly) and the structure may move independently, the relative motion is calculated by adding the two vibrations together with a square root of the sum of the squares method. Components with separation of less than five times the RMS vibration are evaluated for impact. Components with fundamental frequencies less than 200 Hz and those whose response cannot be bounded by nearby components exposed to similar turbulent conditions are evaluated for fatigue.

Table 3-12 lists the components analyzed for turbulent buffeting. Each has a fundamental frequency below 200 Hz. After the RMS response is calculated as discussed above, components undergo further separation screening: the acceptance criterion for a component remaining separated from an adjacent component is that the clearance is greater than 5σ RMS deflection. For components that do not meet the criterion, the surface stress is calculated and an impact fatigue assessment is performed.

Surface stress is calculated using a semi-empirical approximation from Reference 8.1.3 for a vibrating tube in a loose hole (Equation 3-8). The crossing frequency is used as the impact frequency.

$$S_{rms} = c\left(\frac{E^4 M_e f_i^2 y_{\max}^2}{D^3}\right)^{1/5} \qquad \text{Equation 3-8}$$

where:

| | | |
|---|---|---|
| $S_{rms}$ | = | Surface stress due to impact (psi) |
| $c$ | = | Contact stress parameter, page 357 of Reference 8.1.3 (-) |
| $E$ | = | Elastic modulus (psi) |
| $M_e$ | = | Effective mass, for a tube usually taken as 2/3 of the mass of the two adjacent spans (lbf-s$^2$/in) |
| $f_i$ | = | Impact frequency (Hz) |
| $y_{\max}$ | = | Maximum RMS vibration amplitude (in) |
| $D$ | = | Tube outer diameter (in) |

An alternating impact stress is calculated as one-half of the surface stress, and used with a fatigue curve based on RMS stress (similar to Figure 11.6 of Reference 8.1.3) to obtain a usage factor. In addition to the impact fatigue, TB can cause fatigue due to vibration stresses. However, due to the very low vibration amplitudes and alternating stresses, the vibration stresses do not result in fatigue usage for any component susceptible to TB. Out of the components analyzed for TB, four (not including the RVI assembly, which acts as the support for the CRD shaft and ICIGT) components are shown to impact their adjacent supports and are discussed below.

Impact between the SG tubes and SG tube support is predicted to occur, and the fatigue usage due to vibration and impact is calculated to be {{ }}2(b),(c),ECI which provides a margin of safety of {{ }}2(b),(c),ECI This result demonstrates adequate SG tube performance for the component design life, subject to verification testing of analytical inputs and safety margin as identified in Table 3-14.

Impact between the ICIGT and the CRD shaft supports (which move as part of the RVI assembly), between the CRD shaft and the CRD shaft supports, and between the CRAGT and the guide tube support is predicted to occur. For all three impact pairs, the results show that the RMS vibrations do not result in fatigue usage due to the displacements themselves or due to impact. These results are due to the very low alternating stresses generated from the TB, which can be primarily attributed to the low-flow velocities.

Table 3-14 provides an overview of the analysis results and required testing.

Table 3-14 Turbulent buffeting results summary

| Component | Contact Occurs? | Fatigue Margin (%) Note 1 | Items to Verify | Verification Method and Testing Phase | Test | Frequency (Hz) |
|---|---|---|---|---|---|---|
| SG helical tubing | yes | {{ }}2(b),(c),ECI | Frequencies mode shapes vibration amplitude | Separate effects | SG FIV (Section 4.1.2) | Various |
| ICIGT Note 2 | Yes | {{ }}2(b),(c),ECI | N/A | N/A | N/A | {{ |
| CRD shaft | Yes | 100 | N/A | N/A | N/A | |
| RVI assembly | Yes | 100 | N/A | N/A | N/A | |
| CRAGT | Yes | 100 | N/A | N/A | N/A | }}2(b),(c),ECI |

Note(s) for Table 3-14:

1. Safety margin is reported based on the margin to the allowable fatigue usage based on the predicted fatigue usage due to vibration and/or impact.
2. The lower ICIGT is reported because it has the lowest frequency and the highest response of the upper and lower ICIGTs.

### 3.2.4 Acoustic Resonance

Acoustic resonance is evaluated for the steam plenums and nozzles, main steam isolation valves, MSIV bypass lines, SGS piping, CNTS piping, DHRS piping, and at valve and instruments ports. It was determined that AR is not possible at the steam plenums and nozzles. The flow through these nozzles prevents the formation of shear waves and AR in these cavities. The MSIVs are an unlikely source of pressure fluctuations associated with AR because the MSIVs are directly mounted on the steam piping with no standpipe. The locations that flow excitation due to AR may be possible are at the branch lines and cavities at the following locations:

- the closed side branches from the CNTS MS piping with connections to:
  - DHRS steam line and actuation valves
  - MS drain valves

  - Upstream line to bypass MSIV
  - Downstream line to bypass MSIV
- the closed side branches from the SGS feedwater piping with connections to:
  - DHRS condensate line to the DHRS condenser
  - pressure relief valves
- the closed side branches in the CNTS FW piping to the FW drain valves
- the closed side branches in the CNTS CVC piping to the CVC drain valves
- the closed side branches in the RCS injection line to the RRV and RVV reset lines
- RCS instrument and valve ports

The SG inlet flow restrictors are designed to limit density wave oscillations (DWO) inside the SG tubes. Minor DWO are a possible source of AR at lower power. However, DWO are not expected to affect the steam plenum or piping because frequencies are less than 0.5 Hz, which is well below the component and piping AR frequencies.

Acoustic resonances due to the generation of shear waves at closed branch lines are evaluated with the following methodology. To determine if there is a concern for AR, the piping locations where this source of flow excitation is possible are identified and the Strouhal number is calculated for each location. To determine the margin to AR, the calculated Strouhal number is compared to the critical Strouhal numbers that could lead to the onset of AR. As documented in Reference 8.1.19, Figure 17, experiments show that the critical Strouhal number at the onset of resonance is dependent on the local geometric parameters including the ratio of the branch and main pipe diameters (d/D) as well as the ratio of the nearest upstream flow disturbance to the main pipe diameter, (x/D). See Table 3-15 for NuScale critical Strouhal numbers.

Table 3-15 NuScale Critical Strouhal Numbers

| Location of Component | x/D | d/D | Critical Strouhal Number |
|---|---|---|---|
| CNTS Steam Piping to DHRS Steam Line | {{ | | |
| DHRS Condensate Condenser to SGS FW Line | | | |
| Ultrasonic Flowmeter Cavity | | | |
| RRV Cavity | | | |
| CNTS MS Drain Valve Branch | | | |
| MSIV Upstream Bypass Line | | | |
| MSIV Downstream Bypass Line | | | |
| RCS injection line to RRV line | | | |
| RCS injection line to RVV line | | | |
| CNTS CVC Drain Valve Piping – Degasification line with N2 | | | |
| SGS Pressure Relief Valve Branches | | | |
| CNTS FW Drain Valve Branch | | | }}2(b),(c),ECI |

Note: Second order critical Strouhal numbers are determined by doubling the first order critical Strouhal number listed in this table.

A delay of onset to acoustic resonance occurs when the upstream edge of the cavity is rounded. Figure 8 in Reference 8.1.20 shows a delay in velocities when comparing the first order sharp edge acoustic pressure to the round edge acoustic pressure. There is a 105 m/s flow velocity at the sharp edge which is delayed to 135 m/s with a round edge. Because of its relationship to velocity, this has an effect on the critical Strouhal number. It delays its onset by multiplying it by the ratio 105/135.

The Strouhal number for the DHRS steam and condensate lines provide approximately {{ }}2(b),(c),ECI and more than 100 percent margin, respectively to each region's first order critical Strouhal number for susceptibility to AR. More than 100 percent margin is also demonstrated for the RRV cavity and instrument cavities in the RCS downcomer region, the CNTS CVC and FW drain valve branches, the SGS pressure relief valve branches, and the RRV and RVV reset line tees. The MSIV bypass lines have {{ }}2(a),(c),ECI margin and the CNTS MS drain valve branches have {{ }}2(a),(c),ECI margin.

The safety margin is determined by comparing the critical Strouhal number to a minimum component Strouhal number calculated at maximum flow conditions for each region. Positive margin to either the first or second order critical Strouhal number indicates that the region would not be exposed to an onset of AR at an intermediate flow condition. All locations have positive margin to first order shear layer mode excitation, however a few locations are predicted to experience second order shear layer mode excitation at lower flow rate conditions. These locations are the branches of the CNTS

MS drain valves, the tees connecting the CNTS steam piping and DHRS steam lines, and the MSIV upstream and downstream bypass lines. These locations will be tested in order to detect any acoustic excitation and resulting pressure amplifications or vibrations. The relevant reactor power levels at which second order shear layer instabilities could be present at these locations are between 65% and 70% for the CNTS steam piping to DHRS steam line, between 90% and 95% for the CNTS MS drain valve branches, and between 60% and 65% for the MSIV upstream and downstream bypass lines.

Safety margin results for first order shear layer mode excitation and testing information are summarized in Table 3-16. Results provided below show safety margin for the MSIV upstream bypass line which is slightly lower than for the downstream line.

Table 3-16 Acoustic resonance results summary

| Component | Safety Margin | Items to Verify | Verification Method and Testing Phase | Test |
|---|---|---|---|---|
| DHRS steam piping | {{ }}2(b),(c),ECI | Vibration amplitude | Initial startup testing | Flow testing (Section 4.2) |
| MSIV bypass lines | {{ }}2(a),(c),ECI | Vibration amplitude | Initial startup testing | Flow testing (Section 4.2) |
| CNTS MS drain valve branch | {{ }}2(a),(c),ECI | Vibration amplitude | Initial startup testing | Flow testing (Section 4.2) |
| DHRS condenser to SG FW line | {{ }}2(a),(c),ECI | N/A | N/A | N/A |
| Ultrasonic Flowmeter cavity | {{ }}2(a),(c),ECI | N/A | N/A | N/A |

### 3.2.5 Leakage Flow Instability

Leakage flow instability is sensitive to flow and geometry conditions. For NPM components that meet the screening criteria for LFI, testing is required to determine susceptibility to LFI.

The major parameters that have been shown to lead to LFI are large pressure differences across small annular gaps, component flexibility, and small diffusion angles. Due to the natural circulation design of the NPM, most regions are not susceptible to LFI because pressure differences across these interfaces, and thus gap velocities, are very small under all operating conditions. One exception to this is on the secondary coolant side at the entrance to the SG tubes, where a flow restrictor upstream of each SG tube is provided. The SG tube flow restrictor is designed to provide flow stability by restricting the volume of secondary side flow through the tube. The flow restriction is created by narrow annular gaps between the flow restrictor and the tube inner diameter. A separate effects test is performed to validate that LFI is not a concern for the SG tube flow restrictor, per Table 3-17.

Table 3-17 Leakage flow instability results summary

| Component | Safety Margin | Items to Verify | Verification Method and Testing Phase | Test |
|---|---|---|---|---|
| SG tube inlet flow restrictors | Need to verify | Vibration amplitude | Separate effects testing | SG tube inlet flow restrictor test (Section 4.1.1) |

### 3.2.6 Gallop and Flutter

The lower SG support is the only NPM structure that requires evaluation for flow excitation created by gallop and flutter.

Flow tests of rectangular cross sections have been performed to investigate the influence of the VS frequency and the response of the structure to torsional gallop considering both smooth and turbulent flow conditions. The results of the flow test summarized in Reference 8.1.5 are applicable to rectangular cross sections whose height-to-width ratio is between 0.2 and 5.0. For the lower SG support, this ratio is {{ }}$^{2(a),(c),ECI}$, and the results of these flow tests are applicable. The testing predicts that the critical velocity to the onset of torsional galloping is 149 ft/sec which is much greater than the primary flow velocity in this region of approximately {{ }}$^{2(a),(c),ECI}$. Further, the design of the lower SG support prevents plunge galloping from becoming a source of flow excitation because the lower SG support has a sufficiently large height-to-width ratio.

Flutter is also precluded for the lower SG support. Because significant margin exists between the bending and torsional frequency of the SG lower tube support ({{ }}$^{2(b),(c),ECI}$), it is not possible for plunge gallop and torsional gallop to become coupled such that the SG lower tube support will experience flutter. Due to the large margins to the onset of these phenomena, no confirmatory testing is required to verify the flutter and gallop analyses.

### 3.2.7 Comparison to San Onofre Nuclear Generating Station Replacement Steam Generator Issues

The NuScale SG differs from traditional recirculating and once-through SG designs, such as the replacement San Onofre Nuclear Generating Station (SONGS) SGs. Accordingly, the failures experienced with the SONGS replacement steam generators and the consequent lessons learned were considered but are not directly applicable to the NuScale design.

The NuScale SG is a first of a kind design. SG tubes are supported by 21 sets of 8 tube supports. The tube supports provide full-circumferential support of the tubes. Circumferential spacing of the supports is optimized to provide the minimum possible tube free span lengths given the fit constraints with the steam and feed plena and tube transition regions. Shorter tube free span lengths ensure that SG tube modal frequencies are sufficiently high to preclude unacceptable damage due to flow induced vibration.

For traditional (and the SONGS replacement) SG design, industry U-bend design practice applied flat bar (anti-vibration bar) supports because of advantages such as decreased tube-to-anti-vibration bar wear rates. However, experience with the SONGS replacement SGs showed that flat bar supports do not adequately restrict in-plane tube motions. The NuScale SG tube supports are designed to prevent out of plane tube-to-tube wear by separating the tube columns radially, and tube support tabs prevent in-plane tube-to-tube wear by separating the tubes axially. The NuScale SG design also requires small clearances between the SG tubes and tube support column tabs at three points of contact, ensuring that there are contact forces to prevent inactive supports and unacceptable tube-to-tube support wear.

Additionally, the NuScale SG tube wall thickness is thicker than existing SG designs, and the use of Alloy 690 SG tubing mitigates SG corrosion compared to Alloy 600 used in other SG designs. The reactor coolant flowrates in the NuScale NPM are an order of magnitude lower than flowrates across the SG in PWR recirculating steam generators, such as the SONGS design. This low flow rate reduces the turbulent flow energy available to cause FIV degradation of the SG tubes.

These NuScale SG features were developed based on the SONGS lessons learned and previous industry operating experience, and ensure structural integrity against damage and degradation due to FIV, such as those issues observed with the SONGs replacement SGs.

FEI is a phenomena that affects the outside of the SG tubes. Gap velocities for SONGS were approximately an order of magnitude higher than in the NuScale design, based on the higher secondary coolant steam flow velocities compared to the NuScale natural circulation primary coolant velocities. Also, based on thermal hydraulic principles, uncertainties in the steam conditions and the resulting impact on the SONGS flow velocities are significantly higher than NuScale's uncertainties. The uncertainties in the NuScale hot and cold leg temperatures and flow rates are bounded based on the capabilities of the natural circulation design. Further, the effect of temperature on calculated velocities is low for a subcooled liquid compared to steam conditions.

All NuScale SG tube vibrational modes that see a component of cross flow have been evaluated. The SONGS analysis did not consider modes parallel to the AVBs even though they experience cross flow. Ultimately these were shown to cause FEI. NuScale analyzed all tube frequencies where there is a component of the mode shape in the direction perpendicular to the flow. SONGS bending modes are estimated at approximately 3.7 Hz (Reference 8.1.22) versus approximately 30 Hz for column 21 of the NuScale SG.

In addition, the SONGS design used a higher damping value of 3%. The NuScale analysis uses a more conservative value of 1.5% and this parameter will be validated based on prototypic modal testing prior to initial startup testing.

NuScale has followed ASME BPVC Subsection N-1330 guidelines for FEI analysis. Results show positive safety margin to the onset of FEI at limiting operating conditions. Figure 3-6 demonstrates that the most limiting frequencies and mode shapes of each column, tube and assumed boundary condition in the tube bundle is stable, with a reduced velocity below the critical value based on a range of Connors' constants that are evaluated. Also included on this figure are the reduced velocity and mass damping for the SONGS replacement steam generator, row 142 and 100, calculated using the same ASME BPVC Subsection N-1330 equations, but with a less conservative damping value of 3%. The SONGS mass damping and reduced velocity values are calculated in Reference 8.1.22. Depending on the Connor's constants assumed, the predicted safety margin is on the order of negative 830% [(8.12-75.67)/8.12] for row 142, using the methodologies of ASME Appendix N.

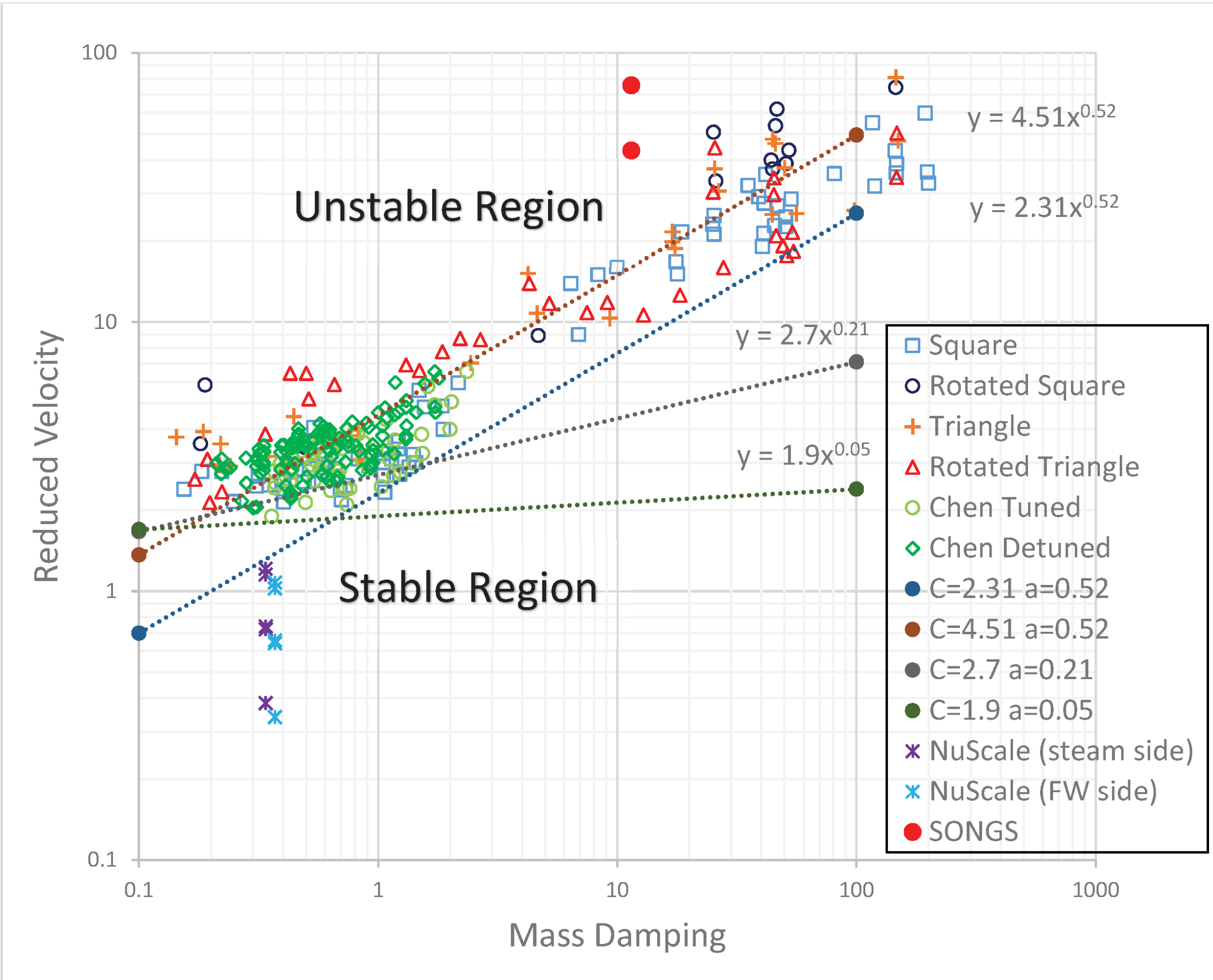


Figure 3-6 Stability Diagram Results of ASME Appendix N and Chen (Reference 8.1.23) FEI Test Data with Overlay of NuScale and SONGS Reduced Velocity Values

## 4.0 Vibration Measurement Program

To validate the FIV inputs, analytical results, and the margins of safety determined in the analysis program, a combination of separate effects, and initial startup testing are performed. Separate effects testing is performed on a prototypic portion of the design. Initial startup testing is performed under full-power normal operating conditions. The results of all three testing types are used to validate the prototype NPM design.

Compared with existing PWR and boiling water reactor designs, the NPM components are less susceptible to FIV due to the lower primary coolant flow velocities. Accordingly, analysis demonstrates that many of the NPM components exhibit a significant margin to the potential onset of the unstable type FIV. Additionally, many of the NPM components are of similar geometry (size, shape, and support) and exposed to similar flow conditions such that the test results for one component can be used to bound similar components. Based on these considerations, components with the lowest margin of safety are selected for the measurement program to validate FIV analysis inputs and results. The following criteria are used to determine if the susceptible component requires validation.

- Components with more than 100 percent margin to the onset of fluid elastic instability, vortex shedding, acoustic resonance and flutter, gallop do not require measurement validation.
- Components with less than 100 percent margin to the onset of fluid elastic instability, vortex shedding, acoustic resonance and flutter, gallop require validation by measurement.
- Predictive analyses of NPM components that are susceptible to LFI are not performed. To ensure this source of flow excitation is not active with the design, separate effects testing is performed to demonstrate an adequate margin of safety.
- For components susceptible to TB, components that experience fatigue usage due to alternating stresses caused by TB vibrations or due to impact with an adjacent support structure require validation by measurement.

For components with more than a 100 percent margin of safety, large errors in the inputs and errors associated with analytical simplifications can be tolerated without impacting the acceptability of the FIV margin. Further, conservative inputs and established analytical methods have been used to predict the margin of safety. Therefore, instrumentation and measurement of the response of these components is not warranted.

For components with less than a 100 percent margin of safety, prototype testing is performed. For fluid elastic instability, vortex shedding, and acoustic resonance, measuring the component vibration amplitude or dynamic pressure is expected to only provide a binary indication of component performance and may not allow for complete validation of the predicted analytical margin. When possible, testing is performed to validate relevant input parameters because they can be quantitatively used to sufficiently validate predicted analytical margin.

Because components susceptible to TB experience vibration when exposed to turbulent flow, it is possible to validate the TB analysis during natural circulation operating conditions. The analysis of the NPM components for TB currently considers PSDs that have been published in open literature and used by the industry. Based upon the computed response of the NPM components considering these FIV inputs, the ICIGT is the only component with less than 100% margin for TB. The fatigue usage is very low and the safety margin is at {{ }}2(b),(c),ECI based on a total allowable fatigue usage of 1.0. Separate effects testing will not be performed, however, the ICIGT will be instrumented during initial startup testing.

Pre-test predictions for all prototype tests that have an associated design analysis methodology are performed to ensure that the overall experiment design, including test conditions, number and location of sensors, and sensor accuracy are sufficient to validate the analysis program. Pre-test predictions provide the expected test result ranges considering uncertainties due to operating conditions, manufacturing tolerances, instrument error, and other sources of experimental biases and uncertainties. Pre-test predictions demonstrate the range of acceptable experimental results that can be used to validate analysis inputs, results, and margins of safety. Post-test analysis verifies the results fall within the pre-test prediction acceptable range, and justifies technically relevant differences between the predicted and actual test results.

Section 2.2 of RG 1.20 suggests that steam, feedwater and condensate piping should be instrumented for vibration measurement during initial startup testing. With the exception of the DHRS steam piping, these components either do not screen for FIV or have been shown to have a margin of safety greater than 100 percent. Only components with less than 100 percent safety margin are tested in the prototype measurement program, consistent with the overall measurement program objectives of validating relevant analytical inputs, results, and margins of safety.

Table 4-1 summarizes the testing and inspections to be performed to verify the FIV analysis program for the prototype NPM. The components for the overall planned validation testing scope are summarized below:

- CNTS MS line branch connections: Testing to validate the AR safety margin is performed during initial startup testing for the DHRS steam piping, MSIV bypass lines and CNTS MS drain valve branches. See Section 4.2 for additional details.
- SG helical tubing: Testing to validate the safety margin for fluid elastic instability, vortex shedding, and turbulent buffeting is performed as a separate effects test. See Section 4.1.2 for additional details.
- SG tube inlet flow restrictors: Testing to validate that LFI is precluded is performed in a separate effects test. See Section 4.1.1 for additional details.

Note that flow testing to provide an assessment of the CRAGT fingers and rodlets is performed to demonstrate acceptable vibration performance of components that are part of the reactor core system and are not within the scope of the CVAP. Testing results will be reviewed to ensure vibration levels are acceptable for the CRAGT; however, detailed validation testing of the CRAGT is not required due to the predicted safety margins of greater than 100%.

Table 4-1 Analysis program verification testing and inspections

| NPM Component Category | Component | Susceptible Mechanisms | Mechanisms with less than 100% Safety Margin | Prototype Testing | |
|---|---|---|---|---|---|
| | | | | Separate Effects | Initial Startup |
| Components exposed to secondary side flow | SGS piping, nozzle, MSIVs | AR | - | - | - |
| | SG steam plenum Note 2 | AR | - | - | - |
| | CNTS MS line branch connections | AR | AR | - | CNTS MS line testing |
| | DHRS condensate piping | AR | - | - | - |
| | SGS pressure relief valve branches, FW drain valve branches | AR | - | - | - |
| | SG helical tubing Note 2 | FEI, VS, TB | FEI, VS, TB | SG FIV testing | |
| | SG tube inlet flow restrictors | LFI, TB | LFI | SG flow restrictor FIV testing | - |
| SG tube supports | SG tube supports | TB Note 1 | - | - | - |
| | Lower SG support | VS, TB, F/G Note 1 | - | - | - |
| Upper riser assembly | Upper riser section | TB | - | - | - |
| | Riser section slip joint | TB | - | - | - |
| | ICIGT | VS, TB | TB | - | ICIGT instrumenta tion |
| | CRD shaft | TB | - | - | - |
| | CRD shaft support | VS, TB | - | - | - |
| | CRD shaft sleeve | VS, TB | - | - | - |
| | Hanger brace | VS, TB | - | - | - |
| Other RVI | PZR spray RVI | TB | - | - | - |
| | CVCS injection RVI | VS, TB | - | - | - |
| | RRV port | AR | - | - | - |
| | Thermowells Note 3 | VS, TB | - | - | - |
| | Instrument ports | AR | - | - | - |
| | Flow diverter | TB | - | - | - |

| NPM Component Category | Component | Susceptible Mechanisms | Mechanisms with less than 100% Safety Margin | Prototype Testing | |
|---|---|---|---|---|---|
| | | | | Separate Effects | Initial Startup |
| Core support assembly | Core barrel | TB | - | - | - |
| | Upper support block | VS, TB | - | - | - |
| | Fuel pin interface | TB | - | - | - |
| | Core support block | VS, TB | - | - | - |
| | Reflector | TB | - | - | - |
| | Lower core plate | VS, TB | - | - | - |
| Lower riser assembly | Lower riser section | TB | - | - | - |
| | CRAGT support plate | VS, TB | - | - | - |
| | CRAGT assembly | VS, TB | - | - | - |
| | Upper core plate | VS, TB | - | - | - |
| Primary coolant piping | RVV and RRV reset line tees | AR | - | - | - |
| | CNTS CVC drain valve branches | AR | - | - | - |

Note(s) for Table 4-1:
1. Mechanism does not require verification due to predicted safety margin; however, test results will be available due to other required testing and will be used to validate inputs, methods and safety margin to the extent practical.
2. Component is exposed to primary and secondary coolant flow.
3. Thermowells are located in the RCS and in NPM piping exposed to secondary coolant flow.

## 4.1 Separate Effects Testing

Separate effects tests are planned for components that are judged to have the highest susceptibility to FIV based on the analysis program results. Performing separate effects testing, which is isolated full-scale mockup testing of the NPM components of interest, is advantageous because it provides the most accurate method to verify the FIV performance of these components before the prototype NPM is fabricated. This plan allows design changes prior to fabrication, if necessary. Separate effects testing for the SG tube inlet flow restrictor and SG tube bundle are performed. A summary of the testing scope and objectives are summarized in the following sections. The specific test details, such as operating conditions, test durations, instrument types and locations, applicable testing hold points, and pre-test predictions of the expected and allowable experimental results, considering bias errors and random uncertainties, will be provided in the CVAP Measurement Program Report.

### 4.1.1 Steam Generator Tube Inlet Flow Restrictor Test

This separate effects test provides an assessment of the vibration performance of the SG tube inlet flow restrictors. The test results are used to verify acceptable performance against LFI. Although verification for TB is not required because impact is not predicted to occur, the testing results may be used to verify TB analysis inputs and methods for this component, to the extent practical. This test is described further in Section 5.3 of the

NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report (Reference 8.1.14).

### 4.1.2 Steam Generator Flow Induced Vibration Test

The full-scale mockup of the SG tube bundle has five prototypic helical columns and supports. This separate effects test provides an assessment of the vibration performance of the SG tubes and tube supports to aid in demonstrating that FEI and VS are not active sources of flow excitation at the equivalent full-power normal operating conditions. The SG tube bundle testing may not achieve the TH conditions corresponding to the predicted onset of the FEI and VS phenomena. However, the testing will provide validation of analytical inputs such as frequency and mode shape. The damping ratio associated with the tube-to-tube support interaction with amplitudes of vibration equivalent to those at full-power normal operating conditions will also be determined with this test to allow the verification of this FIV input used in the analyses. The response of the tube bundle to flow excitation due to TB will be measured, as well as the primary-side flow PSD to verify this input and the analytical results for the tube.

The tests include in-air frequency and damping measurements, in-water frequency and damping measurements, and flow testing of the full-scale five column model.

The following simplifications are adapted into the design of the SG tube bundle mockup facility. While these represent deviations from full-power normal operating conditions, these differences are judged to either not affect the vibration results or corrections can be performed analytically to account for these differences.

- Because the objective of this test is to characterize FIV resulting from single-phase primary flow, testing with a fluid at room temperature is sufficient to define the modal frequencies, the damping ratio, and the PSDs. A correction to these FIV inputs to account for the effect of the fluid properties at a higher temperature is performed analytically.
- Tests are performed at low pressure. Pressure is expected to have an insignificant effect on tube frequencies and the other FIV inputs.
- The tests are performed using a tube bundle with five helical coil columns as compared to 21 columns in the NPM design. Five columns are chosen as a reasonable number of tubes to enable the fluid-structure interaction with adjacent tubes. Further, five columns of tubes are sufficient for the development of a representative level of turbulence in the primary fluid to characterize the PSD for the analysis of TB.
- Tests are run with air inside the tubes. The actual frequencies would be affected by the added mass and flow of water and steam within the tubes. This change in frequency can be predicted analytically.
- Two phase PSD is significant for turbulent buffeting. The tube response due to interior flow is bounded analytically based on the TF-1 and TF-2 test results.

- The SG tubes are fabricated from Type 304 stainless steel in lieu of Alloy 690 material. Corrections to the damping ratio to account for differences associated with these material differences will be made to provide justification for the damping ratio that will be used in the FIV analyses.

In addition to the testing described above, flow testing results are required to validate the secondary-side PSD applied to the inside of the SG tubes in the TB evaluation. This is planned as a separate effects test included within the steam generator FIV testing scope; however, the specific testing details required to validate the secondary-side PSD are not identified in this section and are provided in the measurement program technical report.

This test is described further in Section 5.1 of the NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report (Reference 8.1.14).

## 4.2 Lead Unit Initial Startup Testing

Initial startup testing is performed on the first NPM after the first fuel load. Due to the natural circulation design of the NPM, it is not possible to obtain the limiting TH conditions that are necessary to verify the FIV inputs and results until the NPM is operating near full-power conditions. Initial startup testing will be performed for a sufficient duration to ensure one million vibration cycles for the component with the lowest structural natural frequency. It is expected to take less than 2.5 days to obtain one million cycles of vibration. This is a conservative estimate because the lowest natural frequency of any component evaluated in the CVAP is approximately {{

}}2(b),(c),ECI.

The initial startup test will be performed with online vibration monitoring of the DHRS steam piping, MSIV bypass lines, and CNTS MS drain valve branches. During the initial startup power ascension, these areas will be monitored for indication of acoustic resonance due to excitation of a shear layer mode at any partial or full power flow rate. Testing of this piping section is performed in accordance with the requirements of Part 3 of ASME OM-2012, Division 2 (OM Standards). In the event that an unacceptable vibration response develops any time during initial startup testing, the test conditions will be adjusted to stop the vibration and the reason for the vibration anomaly will be investigated prior to continuing with the planned testing. Vibration amplitudes in the DHRS steam lines, MSIV bypass lines, and CNTS MS drain valve branches are measured to confirm the AR analysis results.

Initial startup testing is described further in Section 5.2 and 6.0 of the NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report (Reference 8.1.14).

## 5.0 Vibration Inspection Program

Prior to and following initial startup testing, components are inspected for mechanical wear and signs of vibration-induced damage. Initial startup testing provides a sufficient duration for the limiting NPM component to experience a minimum of one million cycles of vibration. All components that are evaluated in the analysis program undergo inspection. For the components validated in the measurement program by testing, the inspection provides a secondary confirmation of the FIV integrity of the NPM components. For components that do not require testing due to large safety margins, the inspection confirms that the testing performed on more limiting components sufficiently bounds the performance of the non-tested components.

The components that are considered the most susceptible to FIV are examined in limiting and representative locations to demonstrate acceptable performance. For example, only limiting locations of the exterior to the SG helical tubing are inspected. Inspection is performed on all major load-bearing components, restraints, locking or bolting features, and contact surfaces, in accordance with the guidance of Section 2.3 of RG 1.20. The interior of the RPV is also inspected for loose parts in credible regions. Components may be removed and inspected outside the pressure vessel, but many NPM components cannot be removed from their installed locations. For those components or when practical, an in situ inspection is performed.

The NPM components are inspected following the guidelines and requirements provided in the 2013 edition of the ASME Section III, Paragraph NG-5111, Paragraph NB-5111 and using the methods defined in the ASME Section V, Article 9. The visual inspections are performed using “VT-1” and “VT-3”, as defined by ASME Section XI, Subarticle IWB-2500, Tables IWB-2500-1 B-N-1, B-N-2 and B-N-3. The acceptance criteria for these nondestructive surface examinations are provided in Table 5-1 and will be used to inspect the surfaces and welds of the components identified for inspection. The examination methods are defined in Table 5-1.

Table 5-1 NuScale Power Module inspection plan

| Parts Examined | | Examination Method | Acceptance Criteria | Extent |
|---|---|---|---|---|
| CRD shaft<br>CRAGT assembly<br>CRD shaft support<br>Upper riser section<br>Lower riser section<br>Core support block<br>CRAGT support plate<br>PZR spray RVI<br>Hanger braces<br>CVCS injection RVI | SG steam plenum and nozzle<br>SGS piping<br>MSIVs<br>SG tube supports<br>SG tube inlet flow restrictors<br>Lower SG support<br>Helical SG tubing<br>Lower core plate<br>Upper core plate<br>ICIGT<br>Thermowells<br>Instrument and RRV ports<br>Lower fuel pin | Visual, VT-3 | IWB-3520.2 | Surface |
| Upper fuel pin<br>Upper support block<br>Core barrel<br>CNTS MS line branch connections<br>DHRS condensate piping | | Visual, VT-1 | IWB-3520.1 | Welds |
| Flow diverter and reflector | | General visual | No evidence of loose parts | Surface |

# 6.0 Measurement and Inspection Plan for Non-Prototype Category I NuScale Power Modules

After the CVAP is completed for the first prototype, future NPMs meet the classification requirements of non-prototype Category I designs. There are no differences between NPMs other than allowable variations in manufacturing tolerances, which will be bounded by the validated safety margins for the prototype NPM.

Either a limited vibration measurement program or the prototype inspection program is performed to verify that the vibration analysis and inspection results are consistent with those observed in the prototype NPM. Similar to the approach taken for determining the scope of the prototype measurement program, the results of the prototype measurement and inspection programs will be used to inform the required measurement or inspection options for non-prototype Category I NPMs. To ensure that sufficiently bounding and relevant measurements and inspections for non-prototype Category I NPMs are specified, this topic will be assessed after the prototype measurement and inspection programs are completed in a separate CVAP report for non-prototype Category I NPMs.

# 7.0 Summary and Conclusions

To provide assurance that the NPM components will not experience adverse effects of FIV, NuScale has performed analysis of NPM components that are susceptible to various FIV mechanisms. The analysis is validated using separate effects, and initial startup testing, and component inspections following initial startup testing. This report provides the technical justification for which analysis methods and inputs are required to be verified, and outlines the testing and inspections that will be performed to provide the verification data.

Analysis demonstrates that for all NPM components, FIV is either not predicted to occur or the effects of FIV are shown to be acceptable for the component design life. This result is attributed to the low flow velocities that are inherent in the natural circulation design of the NPM. For components having a safety margin greater than 100 percent for a particular FIV mechanism, testing is not performed. For components with less than 100 percent margin for a particular FIV mechanism, prototype testing is performed to validate key analytical inputs, results and safety margins.

Inspection is performed on all components that are susceptible to FIV. For components that are tested, the inspection provides a secondary confirmation to the FIV integrity of the NPM components. For components that do not require testing due to large safety margins, the inspection confirms that the testing performed on more limiting components sufficiently bounds the performance of the non-tested components.

To finalize the CVAP, two additional technical reports are provided to the Nuclear Regulatory Commission. The first report contains the measurement program details for each prototype test, including test operating conditions, test durations, instrument types and locations, applicable testing hold points, and pre-test predictions of the expected and allowable experimental results, considering bias errors and random uncertainties. The second report provides the post-test evaluation of the testing completed to support the measurement program. In this report, the differences between the expected and measured experimental results are either resolved or confirmed to be in the analytically predicted allowable ranges. The second report contains the inspection program results.

Either a limited vibration measurement program or the prototype inspection program is performed for the non-prototype Category I NPMs to verify the vibration analysis and inspection results are consistent with those observed in the prototype NPM. Similar to the approach taken for determining the scope of the prototype measurement program, the results of the prototype measurement and inspection programs will be used to inform the required measurement or inspection options for non-prototype Category I NPMs. To ensure that sufficiently bounding and relevant measurements and/or inspections for non-prototype Category I NPMs are specified, this topic will be assessed after the prototype measurement and inspection programs are completed in a separate CVAP report for non-prototype Category I NPMs.

This report outlines the scope of the CVAP measurement and inspection program that is used to validate the CVAP analysis program.

The analysis, measurement, and inspection programs specified in this report provide assurance that NPM components do not experience adverse effects of FIV during operation, in accordance with the guidance of RG 1.20.

## 8.0 Referenced Documents

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## Enclosure 3:

Affidavit of Zackary W. Rad, AF-0716-66160

**NuScale Power, LLC**
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

**NuScale Power, LLC**

AFFIDAVIT of Zackary W. Rad

I, Zackary W. Rad, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor’s expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale’s competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying report reveals distinguishing aspects about the method by which NuScale has developed its comprehensive vibration assessment program technical report.

NuScale has performed significant research and evaluation to develop a basis for this CVAP methodology and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed report titled “NuScale Comprehensive Vibration Assessment Program Analysis Technical Report,” TR-0716-50439-P, Revision 2. The enclosure contains the designation “Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{  }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR § 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided 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 is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 31, 2019.

Zackary W. Rad