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# DRAFT REGULATORY GUIDE

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## DRAFT REGULATORY GUIDE DG-1323

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## COMPREHENSIVE VIBRATION ASSESSMENT PROGRAM FOR REACTOR INTERNALS DURING PREOPERATIONAL AND STARTUP TESTING

## A. INTRODUCTION

**Purpose**

This regulatory guide (RG) describes methods and procedures that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable when developing a comprehensive vibration assessment program (CVAP) for reactor internals (including steam dryers in boiling water reactor [BWR] nuclear power plants) during preoperational and startup testing.

**Applicability**

The NRC staff considers this methodology acceptable to support its review of applications for (1) nuclear reactor construction permits (CPs) or operating licenses (OLs) under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," (Ref. 1), (2) design certifications (DCs) and combined licenses (COLs) under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 2), and (3) license amendment requests for extended power uprates (EPUs) at operating reactors. The staff also considers this methodology acceptable for use by licensees of operating plants planning significant plant modifications that might induce potential adverse flow effects on structures, systems, and components (SSCs) within the scope of this regulatory guide.

**Applicable Rules and Regulations**

- U.S. Code of Federal Regulations, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities," (10 CFR 50).

- 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 1, "Quality Standards and Records," requires that reactor internals be designed

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This regulatory guide is being issued in draft form to involve the public in the development of regulatory guidance in this area. It has not received final staff review or approval and does not represent an official NRC final staff position. Public comments are being solicited on this draft guide and its associated regulatory analysis. Comments should be accompanied by appropriate supporting data. Comments may be submitted through the Federal-rulemaking Web site, http://www.regulations.gov, by searching for Docket ID NRC-2015-0161. Alternatively, comments may be submitted to the Rules, Announcements, and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Comments must be submitted by the date indicated in the *Federal Register* notice.

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and tested to quality standards commensurate with the importance of the safety function to be performed.

- 10 CFR 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases," requires reactor internals to be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operations, maintenance and testing.

- 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants," requires reactor internals to be designed and tested according to appropriate quality standards.

- 10 CFR 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," governs the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities licensed under Section 103 of the Atomic Energy Act of 1954, as amended (68 Stat. 919), and Title II of the Energy Reorganization Act of 1974 (88 Stat. 1233).

**Related Guidance**

- NUREG-0800, "Standard Review Plan" (SRP), Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components," (Ref. 3) provides guidance to the NRC staff in reviewing dynamic testing and analysis of piping systems, mechanical equipment, reactor internals, and their supports under vibratory loadings.

**Purpose of Regulatory Guides**

The NRC issues regulatory guides to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated accidents, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in regulatory guides will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

**Paperwork Reduction Act**

This regulatory guide contains information collection requirements covered by 10 CFR Parts 50 and 52 that the Office of Management and Budget (OMB) approved under OMB control number 3150-0011 and 3150-0151, respectively. The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information collection request or requirement unless the requesting document displays a currently valid OMB control number.

# B. DISCUSSION

**Reason for Revision**

This revision of RG 1.20 (Revision 4) expands the guidance related to flow-induced vibration (FIV), acoustic resonance (AR), acoustic-induced vibration (AIV), and mechanical-induced vibration (MIV) for BWRs and pressurized water reactors (PWRs). This revision also expands the scope to include small modular reactor (SMR) nuclear power plants, including guidance for the control rod drive system (CRDS) and control rod drive mechanisms (CRDMs) which might be contained in an integral reactor vessel. The expanded guidance in Revision 4 is based in part on lessons learned from the review of recent applications, including both new plant applications and EPU applications. In addition, Revision 4 re-defines and clarifies the prototype, limited prototype, and non-prototype classifications of reactor internal configurations.

**Background**

Nuclear power plant operation can lead to adverse flow induced and mechanically induced vibrations and resonances in plant systems and their components. Some plant components, such as the steam dryer in a BWR nuclear power plant, perform no safety function, but must retain their structural integrity to avoid the generation of loose parts that might impair the capability of other plant equipment to perform safety functions. A CVAP is necessary to limit potential adverse flow effects at BWR and PWR nuclear power plants, including SMRs, during design, construction, and operation, including situations when power uprates or major plant modifications are proposed. This program includes the analytical methodologies, assumptions, computer programs, and code and code edition for the evaluation of the plant components, including the method of determining the load definition and the uncertainties and bias errors of analytical and measurement procedures. The program also includes a comparison of component stresses against code allowable limits. Finally, the program includes testing methods, instrumentation, and measurements.

Adverse effects in reactors caused by FIV, AR, AIV and MIV can be sensitive to minor changes in arrangement, design, size, and operating conditions. For two nominally identical nuclear power plants, one might experience significant adverse flow effects, such as valve and steam dryer failures, while the other does not. Relatively small changes in operating conditions can cause a previously small adverse flow effect to be magnified, leading to structural failures. For example, severe acoustic excitation occurred in the steam system of one BWR nuclear power plant when flow was increased by 16 percent for EPU operation. Also, a steam dryer in another BWR plant experienced fatigue cracking caused by the reactor pump excitation at its vane passing frequency (VPF). Specific guidance for these assessments, both predictive and measurement-based, is provided in this regulatory guide. In developing a suitable measurement program, it is essential that the selected locations for vibration and acoustic monitoring instrumentation be evaluated for potential effects on the component and system dynamic response.

Operating experience has revealed failures of steam dryers and main steam system components (including relief valves) in BWR nuclear power plants following EPU implementation. These failures have demonstrated the importance of detailed analysis of potential adverse flow effects. Studies of those failures have determined that flow-excited acoustic resonances (where instabilities in the fluid flow excite acoustic modes) within the valve standpipes and branch lines in main steam lines (MSLs) can produce mid- to high-frequency pressure fluctuations in the standpipes of the MSL valves, causing their damage . These pressure fluctuations within the standpipes of the valves might also excite the acoustic modes of the steam columns in the MSLs, causing extremely high sound radiation and damaging the steam dryer, and possibly other reactor internals and steam system components. In those failures, the instabilities of the separated flow (shear layer) over the standpipe openings “locked-in” to the acoustic resonance of the fluid column within the standpipe. “Lock-in” refers to feedback between the flow instability (i.e., shear layer oscillation) and the acoustic mode over a certain range of flow velocity, leading to strong amplification in the fluctuating pressures of the flow instability and acoustic mode. In addition, hydrodynamic loading acting directly on the steam dryer and other reactor internals and steam dryer components can produce FIV, causing excessive vibratory stresses. Variations in the reactor recirculation pump (RRP)[1] speed can lead to changes in pump excitation frequencies and might affect its pulsation amplitude and transmitted mechanical vibration. As a result, nuclear power plant licensees have developed scale model testing (SMT) and structural and acoustic models to evaluate potential adverse effects.

For SMRs, components such as CRDMs and steam generators (SGs) might be within or directly connected to the reactor pressure vessel (RPV). Consolidating reactor and reactor coolant system components into a single integral reactor module creates the potential for increased FIV and MIV. For FIV, these include primary coolant flow over the control rod tubes and drive mechanisms; flow through RRPs; flow through and around the SG tube assemblies; flow through valves; and turbulent steam flow passing over valve standpipes attached to the MSLs. For MIV, the RRPs and the valves could also generate mechanical excitation tones in connected piping and other structures. If exciting frequencies coincide with the natural frequencies of the SSCs or acoustic resonance frequencies, unacceptable vibration levels could occur. Excessive vibration could lead to (a) fatigue failure of various reactor internals, (b) loose parts causing erosion or wear in reactor internal parts, and (c) interference with the operation of the CRDS. This guide discusses methods to assess the vibratory loading on reactor internals induced by various sources, including those from pumps and valves. However, the vibration of pumps, valves, and any other non-internal components is not within the scope of this regulatory guide.

A reliable evaluation of potential adverse effects of FIV, AR, AIV and MIV on nuclear power plant components includes the proper consideration of bias errors and uncertainties in the predictive analysis and in the measurement program. Bias errors might result from the under-prediction of pressure loading, stress, strain, or acceleration when modeling SSCs and acoustic volumes. They might also result from errors in the measurement of data used to benchmark prediction methods. Uncertainties might result from the random error associated with measurement of plant parameters. Guidance is provided herein for assessing end-to-end bias and uncertainty to encompass individual bias and uncertainty values. End-to-end benchmarking encompasses all bias errors and uncertainties associated with simulations (for example, combining bias errors and uncertainties for assumed inputs, simulations of loading, simulations of structural response, and calculations of resulting alternating stresses) as well as measurements (for example, locations and sensitivities of measurement transducers; and signal processing parameters, such as frequency resolution). Rather than benchmarking individual components of the simulation and measurement procedures and combining them (such as using the square root of the sum of the squares method), end-to-end benchmarking compares only the final simulated and measured results, resulting in the end-to-end bias errors and uncertainties.

Finally, the overall CVAP is implemented over several phases. A description of the implementation is provided in the initial application submittal. Testing and inspection activities might occur after the NRC’s approval, with monitoring by the licensee and oversight by NRC staff during construction, power ascension testing, and refueling outage inspections. These CVAP elements are defined through applicable final safety analysis report (FSAR) provisions and/or license conditions, and confirmed through inspections, tests, analyses, and acceptance criteria (ITAAC) for nuclear power plants

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[1] This term generically refers to reactor coolant pumps and reactor recirculation pumps. This regulatory guide refers to reactor recirculation pumps from this point forward.

licensed under 10 CFR Part 52. For example, ITAAC for reactor internals in a BWR nuclear power plant may include the following:

- Reactor internal structures meet American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section III, "Nuclear Power Plant Components," Subsection NG-3000 (Ref. 4), as incorporated by reference in 10 CFR 50.55a. [The applicant may propose relaxed weld quality factors for secondary welds in BWR steam dryers where justified.]

- The set of as-built BWR steam dryer instrumentation includes an adequate number of pressure sensors, strain gages, and accelerometers to apply the steam dryer analysis program based on startup test data.

- The design and construction of MSL branch lines and safety/relief valve and safety valve standpipe geometry in BWR nuclear power plants will preclude first and second shear layer wave acoustic resonances.

- Maximum calculated alternating stress intensity for a BWR steam dryer at the detailed design stage provides a margin of safety compared to the ASME BPV Code limit in Subsection NG. The acceptable margin of safety will be established as part of the application review based on the design and analysis details.
  In addition, the NRC staff has imposed license conditions to address BWR steam dryer monitoring during and after plant startup.

"Reactor internals," as used in this regulatory guide, consist of core support structures and adjoining internal structures inside the RPV. Specifically, those core support and internal structures are defined in Article NG-1120 of Section III, of the ASME BPV Code. For example, for BWR nuclear power plants, reactor internals might include the following components:

- chimney and partitions
- chimney head and steam separator assembly
- steam dryer assembly
- feedwater spargers
- standby liquid control header and spargers and piping
- RPV vent assembly
- core plate
- top guide
- control rod drive housing and guide tube
- orificed fuel support
- jet pump and support
- shroud and shroud support
- core plate and reactor pump differential pressure lines
- in-core monitoring housing system/in-core guide tubes and stabilizers

For PWR nuclear power plants, reactor internals might include the following components:

- core barrel
- upper core support assembly
- lower core support assembly
- control rod guide assembly

- in-core instrumentation guide tubes
- flow distribution device
- heavy reflector
- irradiation specimen baskets

For SMRs, reactor internals might include the following components because of their location inside the integral RPV module, even though some components might not be traditionally classified as reactor internals.

- reactor coolant/recirculation pumps
- riser
- SGs
- pressurizer
- CRDMs and supports
- feedwater lines

Although this regulatory guide applies to reactor internals, it provides guidance that could be helpful for the evaluation of potential adverse effects on SG internals and tubes in PWRs.

This regulatory guide discusses activities separate from inservice inspection and inservice testing programs established in compliance with 10 CFR 50.55a, "Codes and Standards."

**History of Revisions**

The original guidelines of RG 1.20 (dated December 1971) served as the basis for testing many prototype and "similar-to-prototype" (referred to in this guide as "limited prototype" or "non-prototype") reactor internals. However, operating experience and the tendency for the design of subsequent reactor internals to differ from that of the initially designated prototypes made the basic prototype and non-prototype classifications difficult to apply, resulting in the need for time-consuming case-by-case resolution of reactor internal classifications and corresponding vibration assessment programs.

Consequently, the original guidelines were refined in Revision 1 of RG 1.20 (dated June 1975) to incorporate items that would expedite NRC staff review of the applicant or licensee's CVAP. Generally, this was accomplished by increasing the specificity of the guidelines for the vibration analysis, measurement, and inspection programs, as well as including guidelines for scheduling significant phases of the CVAP.

In particular, Revision 1 of RG 1.20 expanded the previous classifications and outlined an appropriate CVAP for each class. In general, under certain conditions, the expanded classifications and corresponding programs allowed reactor internals to be used as "limited prototypes" with specific provisions. The expanded classifications made using this guide compatible and consistent with design and operating experience, at that time.

Revision 2 of RG 1.20 (dated May 1976) retained the expanded reactor internal classifications. However, Revision 2 included various changes in the corresponding vibration assessment programs and the reporting of results, which were made because of substantive public comments and additional staff review.

Revision 3 of RG 1.20 (dated March 2007) modified the overall vibration assessment program for reactor internals, and summarized expectations regarding the evaluation of potential adverse flow effects.

Revision 3 also included changes to address COL applications or applications that do not reference a certified reactor design. Finally, based on operating experience, Revision 3 provided new guidance for steam dryers in BWR plants and information that could be applied to monitoring programs for plant components outside the reactor vessel.

**Harmonization with International Standards**

The NRC staff reviewed guidance from the International Atomic Energy Agency (IAEA) and did not identify any standards related to the subject of RG 1.20 that provided useful guidance to NRC staff, applicants, or licensees.

# C. STAFF REGULATORY GUIDANCE

The NRC staff considers the guidance in this regulatory guide to provide an acceptable method for developing and implementing a CVAP for reactor internals at nuclear power plants. In particular, the applicant or licensee should apply the classifications identified in Section C.1 to categorize reactor internals according to design, operating parameters, and operating experience. The applicant or licensee should establish an appropriate CVAP using the guidance in Sections C.2, C.3, and C.4, as they relate to the specific classifications of the applicable reactor internals.

This regulatory guide describes acceptable methods for evaluating the potential adverse effects from FIV, AR, AIV, and MIV. Section C.2.0 of this regulatory guide provides detailed information for these vibration mechanisms. In general, these vibration excitation mechanisms need to be assessed for reactor internals in BWR, PWR, and SMR nuclear power plants.

Consistent with RG 1.68, "Initial Test Program for Water-Cooled Nuclear Power Plant," (Ref. 5) the term "preoperational testing," as used in this guide, consists of testing before fuel loading, and "startup testing" refers to testing after fuel loading.

Reactor internals important to safety are designed to accommodate steady-state and transient vibratory loads throughout the service life of the reactor. The overall program includes predictive analysis, a measurement program, and an inspection program. The term "comprehensive" appears in the title of the overall program to emphasize that these individual elements are used together to verify structural integrity of the reactor internals (including BWR steam dryers):

- The predictive analysis provides theoretical verification of structural integrity and the basis for the choice of components and specific locations to be monitored in the measurement and inspection programs.
- The measurement program confirms the results of the predictive analyses. However, the measurements (i.e., data acquisition, reduction, and interpretation processes) need to be sufficiently flexible and sensitive to permit identification and definition of any significant vibratory modes that are present but were not evaluated in the predictive program.
- The inspection program addresses both quantitative and qualitative verification of the predictive analysis and measurement program results.

## 1. CLASSIFICATION OF REACTOR INTERNALS

For the purpose of specifying an acceptable CVAP, the applicant should classify its reactor internal configurations (including the steam dryer in BWR nuclear power plants) as "prototype" "limited prototype," or "non-prototype."

### 1.1 Prototype

A "prototype" is a configuration of reactor internals that, because of its arrangement, design, size, or operating conditions, represents a first-of-a-kind or unique design for which no previous "valid prototype" can be referenced.

After the NRC staff accepts the prototype design and the CVAP is completed with no adverse inservice vibration phenomena experienced during an adequate period of operation, it may be proposed as a "valid prototype." A valid prototype may be referenced in subsequent applications, as appropriate, subject to the restrictions and provisions defined below. The adequacy of operational experience needed to validate a prototype is assessed on a case-by-case basis, depending on the breadth and quality of the operational experience data. For example, to be proposed as a valid prototype, a BWR steam dryer will need to complete, as a minimum, the CVAP with no adverse inservice vibration phenomena and at least one inspection after a full operating cycle with the identification of no adverse indications. Reactor internals other than a BWR steam dryer will be evaluated on a case-by-case basis for justification as a valid prototype. For example, to be proposed as a valid prototype, natural circulation reactor internals (i.e., reactor internals in a natural circulation reactor) that will not undergo a preoperational test will need to complete the CVAP with no adverse inservice vibration phenomena and at least one inspection following initial startup testing with the identification of no adverse indications.

See Section C.1.4 of this regulatory guide for special considerations for classifying reactor internals in multi-unit plants and standard reactor designs.

An acceptable CVAP for prototype reactor internals is delineated below in Section C.2. The applicant should assess and document the applicability or non-applicability of all potential sources of vibratory excitation that are described below in Section C.2.

### 1.2 Limited Prototype

A "limited prototype" is reactor internal component that is similar to a "prototype" reactor internal component but cannot be justified as a "non-prototype" reactor internal component. Differences between a limited prototype and a prototype might include, for example, scaling (similar shape but different size) or more than nominal changes to wall thicknesses, welding and connections, boundary conditions, operational parameters (e.g., flow rates, temperature, and pressure), and operating experience. To classify reactor internals as a limited prototype, the applicant/licensee needs to demonstrate by test or analysis that the differences in arrangement, design, size, and operating conditions between the limited prototype reactor internals and the referenced valid prototype reactor internals have no significant effect on the excitation mechanisms and the vibratory response. If this determination cannot be justified, the NRC staff will classify the reactor internals as a prototype as described in this regulatory guide.

See Section C.1.4 of this regulatory guide for special considerations for classifying reactor internals in multi-unit plants and standard reactor designs.

An acceptable CVAP for limited prototypes is delineated below in Section C.3. The applicability needs to be assessed and documented for all potential sources of vibratory excitation that are described below in Section C.2.

The applicant or licensee may propose that a design previously classified as a "limited prototype" be reclassified as a "valid prototype" for reference in a new application after the limited prototype completes its CVAP and has adequate operating experience. The NRC staff will evaluate this proposal as part of the application.

### 1.3 Non-Prototype

A “non-prototype” is a reactor internal component that has substantially the same arrangement, design, size, and operating conditions as a “valid prototype.” Any nominal differences in arrangement, design, size, and operating conditions have been quantitatively shown (by test or analysis) to have no significant effect on the vibratory response and excitation previously determined for a valid prototype. If this determination cannot be justified, the NRC staff will classify the reactor internal component as a prototype or limited prototype as described in this regulatory guide. Based on current operating experience, the applicant or licensee for a BWR nuclear power plant should provide significant justification where it proposes to classify a BWR steam dryer as a “non-prototype” reactor internal component.

An acceptable program for non-prototype reactor internals is delineated below in Section C.4. The applicability or non-applicability of all potential sources of vibratory excitation that are described below in Section C.2 needs to be assessed and documented.

See Section C.1.4 of this regulatory guide for special considerations for classifying reactor internals in multi-unit plants and standard reactor designs.

### 1.4 Special Considerations for Classifying Reactor Internals in Multi-Unit Plants and Standard Reactor Designs

For a multi-unit nuclear power plant on a single site or a standard reactor design being constructed at the same time for units at multiple sites, the applicant/licensee might not have operating experience with the reactor internals for those multiple reactor units when applying for a design certification, COL, or OL; or proposing an EPU license amendment. The acceptability of classifying reactor internals as prototype, limited prototype, or non-prototype components in individual units of a multi-unit nuclear power plant or for multiple units of a standard reactor design will be assessed on a case-by-case basis depending on the breadth and quality of previously submitted information for similar SSCs.

For example, the steam dryer in one BWR unit may be classified as a “prototype” and the steam dryers in the other on-site BWR units may be classified as “limited prototype” where justified. The applicant or licensee may demonstrate that a BWR steam dryer classified as a “limited prototype” in a reactor unit at a multi-unit plant performs in a manner consistent with the steam dryer in the unit classified as a “prototype” by using a reduced amount of instrumentation on the “limited prototype” steam dryer or, where justified, using MSL instrumentation rather than on-dryer instrumentation for the “limited prototype” steam dryer. Using MSL instrumentation for monitoring limited prototypes is acceptable when (a) prototype benchmarking shows a high margin of safety in the dryer alternating stresses, and (b) prototype MSL measurements do not reveal anomalies in data quality that would add significant uncertainty in monitoring the dryer. See section C.2.2.1 for more guidance.

For reactor internals other than BWR steam dryers, the non-dryer reactor internals in one reactor unit may be classified as a “prototype” and the non-dryer reactor internals in the other reactor units may be classified as either “limited prototype” or “non-prototype” where justified. One example is that natural circulation reactor internals in one unit may be classified as a “prototype” and those reactor internals in the other units may be classified as “limited prototype” before the “prototype” reactor internals are inspected following initial startup testing. The natural circulation reactor internals in the other units may be classified as a “non-prototype” after the inspection of the “prototype” reactor internals following initial startup testing. The applicant/licensee should follow the applicable guidance in this regulatory guide for “prototype,” “limited prototype,” and “non-prototype” reactor internals for the individual reactor units.

## 2. CVAP FOR PROTOTYPE REACTOR INTERNALS

The CVAP for prototype reactor internals consists of a vibration and stress analysis, vibration measurement, inspection, and correlation of predicted and measured results to demonstrate the acceptable performance of reactor internals important to safety for the full range of flow, temperature, and pressure conditions associated with normal steady-state and anticipated transient operation of the nuclear power plant.

As part of the CVAP, applicants designing or proposing to construct and operate a new nuclear power plant, licensees planning to request an EPU for an existing nuclear power plant, and licensees planning a major modification to an existing nuclear power plant should analyze the effects of potential vibration mechanisms that can affect the reactor internals. The following list comprises the main excitation mechanisms that need to be addressed:

a. **FIV and AR produced by fluid flow across or parallel to structural components.** These mechanisms include vibration induced by flow turbulence (turbulent buffeting), acoustic resonance excitation by separated flow instabilities, vibration induced by vortex shedding excitation, and fluid-elastic instability (FEI) (Ref. 6). FIV occurs not only in the lift (or normal to the flow) direction, but also in the drag (or streamwise) direction. Up to this time, predictive analysis and testing of FIV of plant components caused by vortex shedding excitation and FEI focused primarily on structural vibration in the lift direction. However, experiences from nuclear power plants revealed thermowell failure because of streamwise vibration excited by vortex shedding from the thermowell. Therefore, the applicant/licensee should develop the CVAP to include possible streamwise (or in-plane) vibration because of FEI and vortex shedding excitation. As previously noted, studies of past failures have determined that flow-excited acoustic resonances within isolation valves, in standpipes of safety relief valves (SRVs), and dead-end branch lines in the MSLs of BWRs can produce mid- to high-frequency pressure fluctuations and vibration that can damage MSL valves, the steam dryer, and other reactor internals and steam system components. In addition, hydrodynamic loading acting directly on a steam dryer (caused by flow turbulence and boiling water rumbling) can result in undesirable dynamic stresses.

b. **AIV caused by reactor pump pressure pulsation or pressure waves emanating from acoustic resonators such as the standpipes of SRVs in MSLs**. RRPs generate pressure pulsations at multiple frequencies, including the pump shaft speed, the impeller VPF, and their harmonics. These pressure pulsations are caused by hydrodynamic forces induced by the rotating impeller interacting with distorted in-flow. The hydrodynamic forces act as acoustic dipole sources within the working fluid. These pressure pulsations could excite the acoustic modes of the water/steam system inside the RPV, causing significant acoustic loads on reactor internals. Depending on the number and arrangement of the pumps and the relative phase between their respective drive frequencies and resulting forcing functions, local pressure pulsations could reach several times those of a single pump. For RRPs that are driven with a variable frequency drive, the excitation frequencies vary as the drive frequency varies, leading to potential interactions with structural and acoustic resonances in the plant. The enhancement and propagation of resonant acoustic waves can exert substantial acoustic loading on structures such as BWR steam dryers.

c. **MIV of reactor internals and other structural components caused by structure-borne vibration transmission from RRPs and other machinery.** As stated in (b) above, RRPs generate dynamic forces at the pump shaft speed, impeller VPF, and their various harmonics. These forces act directly as acoustic dipoles on upstream and downstream fluid, but also act on the pump mounting structures and can be mechanically transmitted through the reactor wall to other components connected to the feedwater and steam piping, or to other components inside the reactor. MIV is exacerbated when pumps are mounted directly to the reactor vessel instead of connected to the RPV through external piping. As previously noted, studies of past failures in nuclear power plants have determined that the steam dryer in a BWR plant experienced fatigue failure caused by vibration transmission from the reactor pumps at the VPF.

The CRDS in some SMRs are not part of the pressure boundary, and therefore the areas of review are different than those for conventional light water reactors. In some SMRs, all internal CRDS components, including the CRDMs, are exposed to primary coolant flow, and corresponding temperature and flow-induced loads. Therefore, all components in a fully immersed CRDS need to be evaluated for FIV, AIV, MIV, and the potential generation of loose parts. Also, in some SMRs that consolidate all major reactor components into a single modular system, additional dynamic excitation might be imparted on the CRDS components. Dynamic excitation because of fluid flow, flow-excited acoustic resonances, and mechanical sources should be addressed in the CRDS design.

## 2.1 Vibration and Stress Analysis Program

The applicant or licensee should perform a vibration and stress analysis for those steady-state and anticipated transient conditions that correspond to preoperational, startup test, and normal operating conditions. The vibration and stress analysis needs to address the structural, hydraulic and acoustic models, analytical and computational formulations, and scaling laws and scale models used in the analysis, including all bias errors and uncertainties for reactor internals that might be adversely affected by FIV, AR, AIV and MIV. Based on operating experience, the analyses need to address the following aspects:

a. Identify significant vibration and acoustic resonances caused by various vibration excitation mechanisms that have the potential to damage reactor components, including BWR steam dryers.

b. Determine the pressure and force fluctuations and vibration in the applicable plant systems under flow conditions up to and including the full operating power level. Such pressure fluctuations and vibration can result from various excitation mechanisms, such as FIV, AR, AIV and MIV, and need to be assessed for the full range of the plant system fluid flow conditions. In particular, any excitation that is reinforced by structural or acoustic vibration feedback needs to be assessed, and, if necessary, mitigated by design modifications.

c. Justify and benchmark the methods used for computing resultant vibration and alternating stress in plant systems.

The applicant or licensee should compare stress at locations susceptible to fatigue cracking with the ASME BPV Code fatigue limits to validate the end-to-end analysis. If necessary, the applicant/licensee should perform modifications to the structure or other components to demonstrate design margin to Code allowable limits. The BWR applicant or licensee should perform a rigorous assessment of stress in steam dryers.

### 2.1.1 *Structural, Hydraulic, and Acoustic Modeling*

The vibration and stress analysis program in the CVAP should address the following aspects related to structural, hydraulic, and acoustic modeling:

a. the structural models used to compute the vibration response of reactor internals.

b. other models of steam or water volumes coupled to the structure.

c. natural frequencies and associated mode shapes that might be excited during steady-state and anticipated transient operation for reactor internals.

d. frequency response functions (FRFs) between key drive and response locations, along with the assumed damping used in the calculations, expressed as vibration or stress normalized by input force.

Acceptable methods are summarized below.

Modes of Vibration

The applicant or licensee should develop tables of significant structural natural frequencies and accompanying figures of corresponding mode shapes. Benchmarking the analytic mode shapes and natural frequencies involves comparison of measured and simulated data. The differences between measured and simulated natural frequencies are used to establish uncertainty ranges for FRFs and final response calculations. If benchmarking reveals that the simulated natural frequencies are within +/-10 percent of those measured, FRFs and final responses are computed over that range of uncertainty. The forcing function time histories are expanded and compressed by +/-10 percent and applied repeatedly to the structural dynamic model, with the worst case values retained and applied in the analysis. Several analysis increments are used, depending on the damping assumed. Previous acceptable analyses assuming 1 percent damping have used 2.5 percent frequency shifting increments in their analyses (i.e., loads are shifted by -10 percent, -7.5 percent, -5.0 percent, and so on up to +7.5 percent, +10 percent).

Structural Damping

The applicant or licensee should substantiate assumed structural damping. In some instances, the NRC's damping guidance for very-low-frequency seismic analyses has been incorrectly applied as justification for using high damping factors for mid- to high-frequency analyses. Damping factors used in structural dynamic modeling need to be based on mid- to high-frequency measurements on structures representative of the reactor internals being modeled. Measurement techniques for damping are available in the standards promulgated by the American National Standards Institute (ANSI) and other organizations. The applicant or licensee should describe using measurement techniques for damping in the modeled environment (including air, steam, or water environments). Based on past experiences, specification of structural damping coefficients greater than 1 percent needs to be substantiated with measurements.

Some BWR licensees requesting EPU license amendments applied additional hydrodynamic damping to the dynamic response of perforated plates in steam dryer banks. Using similar hydrodynamic damping in future applications is acceptable where the estimated damping value is based on a sound technical foundation and is demonstrated to be conservative for the particular application.

<u>Frequency Response Functions</u>

The applicant or licensee should document the modeling approach as well as modeling details, including assumed boundary conditions and material properties, used to compute mode shapes and FRFs. The applicant or licensee should specify the uncertainties and bias errors associated with the approach and specific model, along with their bases. In many cases, bias errors in numerical models (such as for stress) are associated with insufficient mesh refinement of stress concentration regions and improper modeling of structural joints. These errors may be accounted for with stress concentration factors (SCFs). The uncertainties are often associated with differences between numerical models and as-manufactured structures, such as differences in material properties, connections (e.g., bolts, welds, and rivets), and geometries (e.g., plate and piping thicknesses). Bias and uncertainties associated with these differences may be estimated based on comparisons of simulations and measurements of structures similar in construction to the reactor internals being modeled. When benchmarking the structural dynamic modeling of components using dynamic testing, a sufficient number of drive and response points are applied to characterize all of the important modes of vibration and FRFs.

The applicant or licensee should update the models to reflect reactor operating conditions, including material properties at operating temperatures, as well as water and steam fluid loading on reactor internals, including uncertainties in the natural frequencies and FRFs.

Some structures might have variable or ill-defined boundary conditions during normal reactor operating conditions. Structural models need to address possible variations in the boundary conditions. For example, steam dryers rest on mounting pads attached to the RPV with variable amounts of contact area and supported load during normal operating conditions, such that there is no simple means for representing the boundary conditions. Another example is that the control rod guide tubes and rods for some SMRs are effectively long beam structures, with structural resonance frequencies depending on the spacing of support structures (the hanger plates). The resonance frequencies vary with rod height. Therefore, evaluation of multiple CRDM/control rod heights is necessary to address all operating and transient conditions of the reactor. If the guide tubes or rods vibrate excessively, they might contact each other during rod movement and interfere with their function.

### 2.1.2 *Forcing Functions*

The applicant or licensee should determine the design load definition for all reactor internals, including BWR steam dryers, up to the full licensed power level, and validate the methods used to determine the load definitions based on justified SMT or data acquired from other plants. Instrumentation for BWR steam dryers needs to measure pressure loading, strain, and acceleration to confirm the SMT, plant data, and analysis results. For EPU requests, BWR plant data at current licensed power conditions should be used to confirm the results of the SMT and analysis for the steam dryer load definition.

The applicant or licensee should describe the estimated random and deterministic forcing functions, including very low-frequency components, for steady-state and anticipated transient operation for reactor internals that might be adversely affected by FIV, AR, AIV and MIV.

Acceptable methods are summarized below. Applicants for new designs, particularly SMRS, should consider additional relevant forcing functions which are not described here.

The applicant or licensee should determine the excitation mechanisms that are relevant to the various structural components and evaluate the forcing functions for each component. The applicant or licensee should justify the methodologies for determining pressure fluctuations and vibration in plant systems. Experience indicates that computational fluid dynamics (CFD) analyses might not provide sufficient quantitative information regarding unsteady pressure loading. If applicable, bounding analyses may be used where justified. Full-scale testing, along with SMT and analysis, may also be used to evaluate potential adverse effects of flow and mechanically excited resonances and verify the alternating loads predicted by CFD or bounding analyses. Testing and analysis may also be used to ensure appropriate bias error and uncertainties are properly addressed. Proper application of bias and uncertainties ensures that evaluation approaches lead to verified bounding simulations of vibration and alternating stress. When using SMT, the applicant or licensee should quantify the effects on the measurements because of (1) reduced Reynolds Number, (2) differences between working fluids and structural components (materials and dimensions), and (3) other differences between the SMT model and full-scale conditions. For example, of particular concern for BWR steam dryers are the following flow-induced forcing functions:

a. separated, impinging and reattached flows in the reactor dome, including low-frequency hydrodynamic loading on BWR steam dryers.

b. boiling water excitation of the immersed lower portion (skirt) of BWR steam dryer.

c. flow turbulence and narrow-band excitation in MSLs in BWRs.

Flow Excitation with Feedback and Lock-In Mechanisms

The applicant or licensee should evaluate forcing functions that might be amplified by lock-in with an acoustic or structural resonance (referred to as self-excitation mechanisms). A lock-in of a forcing function with a resonance strengthens the resonance amplitude. The resulting amplitudes of the forcing function and resonance response can therefore be significantly higher than the amplitudes associated with non-lock-in conditions.

The applicant or licensee should address all potential flow-excited acoustic or structural resonances that lead to feedback and loading amplification. Tables may be used to specify expected flow rates and resonance frequencies, along with the possible ranges of lock-in and potential loading amplifications. The applicant or licensee should define the uncertainties in any of the lock-in parameters (such as the characteristic Strouhal numbers of the flow-excitation sources).

The applicant/licensee should compare flow instability frequencies to those of acoustic and structural modes in the reactor dome, MSLs, any connected valves, and reactor internals, as applicable. The amplitude of any identified self-excitation or lock-in should not be analyzed by simply applying linear extrapolation techniques. Finite element simulations or measurements may be used to determine the resonance frequencies.

The applicant or licensee should evaluate reactor internals that could be excited by fluid flow which contracts and accelerates between the gaps and annuli among reactor internals. For example, for the CRDS, flow excitation of the control rods, CRDMs, tie rods and plates, upper flange webs, and around the power cables should be assessed. In addition, the applicant or licensee should evaluate the potential coincidence of the drive and CRDS component natural frequencies because excessive vibration could lead to interference with the operation of the control rods.

If potential self-excitation or lock-in is identified, the applicant/licensee should establish specific mitigation procedures where the lock-in leads to vibration or stress that exceeds allowable limits. For example, the following forcing functions need to be addressed for lock-in susceptibility:

a. Flow (or shear layer) instabilities over openings in the MSLs, such as control and safety valve standpipes and dead-end flanges, can lead to strong narrow-band excitation because of lock-in with acoustic or structural resonances. For example, acoustic resonances in standpipes can be excited by various shear layer oscillation modes (also known as hydrodynamic modes). Acoustic resonances excited by the first shear layer mode (the lowest frequency of shear layer oscillation) might be significant and, therefore, need to be mitigated by suitable design modifications (e.g., acoustic side-branches attached to standpipes) or operating condition changes. Information on acoustic resonances can be found in ASME PVP 2007-26658, "Identification of Quad Cities Main Steam Line Acoustic Sources and Vibration Reduction" (Ref. 7). On the other hand, the excitation by the second (and higher) shear layer mode generally produces less significant resonances and, therefore, excitation by the higher shear layer modes might be acceptable if the resulting vibration and alternating stresses in relevant components are demonstrated to meet the acceptable limits. Acceptable assessment of acoustic resonance of standpipes addresses the effect of the following parameters:

(1) Strouhal number analysis to evaluate critical flow rates (including uncertainties in the Strouhal number),
(2) effects of the ratio between the standpipe and main pipe diameters,
(3) effects of edge radii at the inlet of the standpipes,
(4) effects of upstream elbows,
(5) distance between standpipes, and
(6) relative length of standpipes.

b. Shear layer excitation of "trapped acoustic modes" associated with shallow cavities in isolation gate valves attached to MSLs in BWRs.

Several methods may be used to quantify forcing functions, including SMT, CFD, and Acoustic Inference Methods. Guidance for these methods is provided below:

Scale Model Testing

If SMT is used to support the analysis, the following aspects need to be addressed:

a. SMT facilities generally involve a lower Reynolds number than that present in actual nuclear power plants because of the smaller scale and lower static pressure of the SMT. Because self-excitation mechanisms (such as flow-excited acoustic resonance) are generally dependent on Reynolds number, the applicant or licensee should demonstrate that the SMT results are not influenced by further increases in the Reynolds number of the SMT.

b. When examining flow-excited acoustic resonance mechanisms, the differences between the model parameters and those of the full scale installation need to be evaluated. These include, but are not limited to, acoustic attenuation of sound waves and reflection coefficient at the model boundaries. Acoustic attenuation is affected by component size (e.g., pipe diameter), static pressure, and void fraction of the medium (e.g., the wetness degree of steam or air). Scale models built to study acoustic resonance excitation are generally designed with reflective boundary conditions to enhance the quality factor of the resonant modes and, thereby, obtain conservative data.

c. SMT to examine fluid-structure interaction mechanisms needs to be conducted on dynamically similar scale models based on all relevant dimensionless parameters of the full-scale installation. Scale models based on conservative assumptions are recommended for which the fluid-elastic parameter is higher, and the vibration damping coefficient is lower, than in the full scale installation. The fluid-elastic parameter relates the dynamic fluid force to the structural elastic force.

d. When SMTs are performed at transient conditions of pressure, temperature or flow velocity, repeated test runs of the same test conditions are necessary to obtain reliable averaged test data with reasonable uncertainties.

e. The model geometry needs to replicate the details of the full-scale geometry accurately, particularly at critical locations that are sensitive to flow excitation mechanisms, such as the locations of flow separation. For example, when modeling the geometry of a closed side-branch representing an SRV standpipe, the edge radius of the standpipe inlet can strongly influence the onset flow velocity and the sound intensity of acoustic resonances in the standpipe. Therefore, the size of the scale model needs to be sufficiently large to allow the evaluation of small relevant geometrical details.

f. SMTs performed under transient test conditions are often associated with fast temperature variations during the tests. For example, fast depressurization of a limited capacity tank results in fast temperature variations with time. The resulting changes in the speed of sound and acoustic resonance frequencies can affect the acoustic response during transient tests (e.g., producing artificially wide acoustic resonance peaks in the pressure power spectral densities (PSDs)). Sensitivity analyses of the effect of the sample length of the pressure signal on the PSDs need to be performed to ensure bounding estimate of the loading functions.

g. When evaluating the growth rates of resonance peaks as a function of power level, the root mean square (RMS) amplitude within a relatively narrow frequency band centered around the resonance peak needs to be evaluated and not the total RMS amplitude of the entire frequency range. The boundaries of the resonant frequency band can be placed at the interface between the resonance peak and the neighboring background representing the broadband response.

h. When conducting SMTs, particularly under transient test conditions, the uncertainties associated with determining the flow velocity (or Mach number) during the tests need to be analyzed. SMTs need to be performed up to slightly higher power levels than the maximum expected power in the full scale installation to compensate for the uncertainties in determining the flow velocity during the SMT.

i. For self-excited vibration and acoustic resonance mechanisms with lock-in phenomena, the absolute value of flow excitation level (e.g., forcing function or amplitudes of vibration and pressure pulsation) measured from the SMT cannot be scaled up to full-scale plant conditions. This is because the measured variable in the SMT is influenced by the associated system response, which is different from the plant response. However, ratios of excitation levels measured from the SMT, if performed adequately, can be used to scale in-plant measured excitation levels to higher power levels. In previous EPU applications, this ratio is referred to as a "bump-up-factor" (BUF). Using BUFs obtained from SMTs should be compared to BUFs determined from in-plant measurements from prototype plants to demonstrate that the results are bounding. BUFs represent the ratios between the dynamic loadings at a power level to that measured at a lower power level. In addition, BUFs are generally location- and frequency-dependent and their minimum value at any frequency cannot be lower than the square of the ratio between the characteristic flow velocities of both power levels.

j. Plant-specific data are needed to demonstrate the acceptability of SMT test data and confirm the vibration analysis results.

<u>CFD Modeling</u>

If CFD models are used to develop unsteady forcing functions or compute the distribution of flow velocity to develop the forcing functions, applicable items from the following list need to be addressed:

a. The dynamic fluid forces acting on reactor internals and other SSCs exposed to fluid flow are often estimated from the local flow velocity and density, which are computed by means of CFD codes. All computational codes used in the vibration analysis to determine local flow velocities need to be validated on systems which are similar to the plant components in both geometrical complexity and flow regime.

b. When constructing fluid domain models to be used in CFD and thermal-hydraulic codes, an accurate representation of the fluid domain geometry details within the reactor vessel is needed, including proper definitions and representations of the smallest flow areas, and additional features inside an SMR reactor vessel (such as internal CRDM mechanisms).

c. The flow distribution inside the RPV might not be uniformly distributed–for example, if the design includes RRPs or multiple piping inlets in a natural circulation mode. When performing the vibration analysis of the reactor internals, the worst scenario of reactor flow distribution caused by all feasible combinations of pump operation patterns or natural circulation flow patterns needs to be evaluated. For example, for a particular BWR design, plant operation at 100 percent reactor flow with only 8 pumps running is likely to produce local flow velocities higher than those produced when 10 pumps are operating at the same reactor load. Consideration of such possibilities ensures that flow excitation mechanisms caused by the maximum possible flow velocity inside the reactor are taken into account in the vibration analysis.

d. Grid size sensitivity tests need to be performed to demonstrate the independence of results from grid size.

e. Steam needs to be modeled as a real gas.

f. For unsteady flow simulations, acoustic/vibration coupling (if sufficiently significant to affect the flow behavior) needs to be included to simulate enhancement of flow instabilities.

g. For unsteady flow simulations, the time step size needs to be demonstrated as not influencing the results (i.e., perform time step sensitivity tests).

h. For unsteady simulations of high frequency flow oscillation, using large eddy simulation (LES) or direct numerical simulation (DNS) at high Reynolds number flow is acceptable. If used, LES, DNS, or other methods need to be demonstrated to be bounding for a representative test case. If applicable, compressibility effects need to be included to model any coupling of the flow and the acoustic waves in the fluid (for self-excitation and lock-in effects).

i. When estimating upper bounds of dynamic forcing functions on reactor internals and other structural components, conservative simplifications and approximations need to be used if they are needed to complete the analysis. This includes, for example, estimating the correlation length and phase distributions of the fluid dynamic forces on structures exposed to fluid flow.

j. Past review of EPU applications indicated that variability in reactor operating parameters can affect flow rates, working fluid mass density, pressures, and other quantities. These variations need to be addressed when assessing reactor internals and other safety-related components at worst-case operating conditions.

Force Inference Approaches

In some BWR EPU requests, licensees have employed inverse acoustic models to estimate fluctuating pressures within the RPV and on BWR steam dryers. These pressures are inferred from measurements of fluctuating pressures either (a) on dryer surfaces, or (b) within the MSLs connected to the RPV. Benchmarking of these procedures on plants and systems similar to the plant being designed or licensed is acceptable. All uncertainties and bias errors associated with the on-dryer or MSL pressure measurements and modeling parameters need to be clearly defined. The bases for the uncertainties and bias errors, such as experimental evaluation of modeling software, need to be described. There are many approaches for measuring MSL pressures and computing fluctuating pressures within the RPV and the MSLs. Although some approaches reduce bias and uncertainty, all approaches have a finite bias and uncertainty that need to be addressed. In particular, it is challenging to fully quantify all alternating loads acting on SSCs (including steam dryers) within complex environments (surrounded by moving steam with varying wetness, partially immersed in boiling water), especially using remote measurements and inference techniques. Based on experience, the following guidance provides approaches that minimize uncertainty and bias error (MSL pressure measurement details are provided in Section C.2.2 of this regulatory guide):

a. At least two measurement locations need to be used on each MSL. However, using three measurement locations on each MSL improves input data to acoustic propagation models, particularly if the locations are spaced logarithmically. This configuration will reduce the uncertainty in describing the waves exiting and entering the RPV. Acoustic sources should not exist between any of the measurement locations, unless specifically justified.

b. Acoustic modeling parameters, such as the speed of sound, reflection coefficients from boundaries between steam and water, and sound attenuation (damping), may be adjusted when developing and benchmarking models against measurements, but should not deviate significantly from those based on theory and measurement.

c. All acoustic wavelengths over all frequencies with significant loading need to be resolved by discretizing with at least six subdivisions.

d. Circular regions of acoustic models, such as MSL inlets in RPVs, need to be represented in a manner that properly encompasses the actual area of the circular cross section. Linearly subdividing a circular region in a numerical model can artificially reduce the effective cross sectional area.

Once specified and benchmarked, the same speed of sound, attenuation coefficient, and reflection coefficient may be used in similar plants. However, different flow conditions (temperature, pressure, quality factor) might dictate adjustments of these parameters within reasonable expectations.

Mechanical and Acoustic Forces from RRPs

Where applicable, the vibration analysis needs to examine the effects of RRPs by pulsation on reactor internals. Analysis of AIV and MIV involves knowledge of the RRP forcing functions. Operating RRPs generate various exciting forces at multiples of their drive frequency, including those induced by electromagnetic oscillations within motor cores, by imbalance and misalignment (which can be caused by steady hydraulic side forces), and by hydrodynamic forces at multiples of impeller VPFs. The hydrodynamic forces are induced by the impeller vanes rotating through non-uniform in-flow. These forces act on both the acoustic waves within the piping, as well as on the mechanical bearing systems, and therefore on the piping structures. For reactors with multiple pumps, the forces can be amplified when synchronized. For worst-case conditions, the pump forces are perfectly synchronized, and the total force is the product of a single pump force and the number of pumps. When not synchronized, excitation tones with time varying amplitude commonly known as a "beating phenomenon," can also occur as discussed below. Any of the tones, when aligned with a structural or acoustic resonance, can lead to strong vibrations of reactor components. These pump sources are combined with computational acoustic and structural models of the reactor internal (water) domain to determine the excitation forces acting on the internal structures, such as control rod drive housing, control rod guide tube, differential pressure lines, and the housing system, guide tubes, and stabilizers for in-core monitors. Acoustic and mechanical pulsations generated by RRPs and their effects on reactor internals might be more intense in some SMRs because of the close proximity of the pumps to reactor internals.

The applicant for new BWRs, PWRs, and SMRs should address the following issues, as applicable for the specific design:

a. The acoustic and mechanical forcing functions of individual pumps need to be based on data obtained from full-scale experiments performed on pump test stands (e.g., at the pump supplier facility). Tests of sub-scale pumps may be acceptable if full-scale test data are not available, but the conservatism of the scaling rules needs to be demonstrated.

b. If the pump excitation frequencies (e.g., rotor speed, VPF and their harmonics) spanning all expected operating conditions of the reactor are within 10 percent of the frequency of a structural or acoustic resonance mode of reactor internals, the vibration analysis needs to assume coincidence between the pump excitation frequency and the resonance frequency.

c. The spatial distribution of the combined forcing function from all simultaneously operating pumps depends not only on the number and arrangement of the operating pumps, but also on the phase between the forcing functions of individual pumps. Multiple sources of pump pulsation tones can lead to a "beating" phenomenon, which can magnify the pressure pulsation. At the beating peaks, the pressure pulsations could conceivably be several times those of a single pump. Also, the effects of one or more pumps being out of service on the combined forcing function applied to the reactor internals need to be assessed. Therefore, the vibration analysis of reactor internals needs to address various scenarios of phase difference between the pump forcing functions. For example, if a reactor is operated by 10 pumps, 1 scenario to be analyzed would consider all forcing functions of the 10 pumps to be in phase while another scenario would assume the forcing functions of 5 adjacent pumps to be in phase, but out of phase with those of the other 5 pumps. These two scenarios are not expected to materialize for long periods. However, they would provide indications of the maximum pump loading functions that might occur for short periods of time.

d. When computing the system response to RRP excitations, conservative boundary conditions need to be applied at the boundaries of the modeled domain.

### 2.1.3 *Computing and Benchmarking Structural and Acoustic Operational Response*

The applicant or licensee should summarize the calculated structural and acoustic responses for operation under steady-state and anticipated transient conditions. This summary needs to identify the random, deterministic, and overall integrated maximum response, very low-frequency components of response, and the level of cumulative fatigue damage (if any).

Acceptable methods are summarized below.

Dynamic responses (defined in terms of frequency, amplitude, modal content, and vibratory stresses) need to be determined at critical locations, including where vibration sensors might need to be mounted on the reactor internals. The calculated responses need to include vibrations for components that have maximum vibration level criteria, as well as stresses for components that have maximum stress criteria (such as the fatigue stress limits specified in Section III of the ASME BPV Code). The margins to the criteria need to be evaluated.

Based on the uncertainties and bias errors identified for individual analysis components, the applicant or licensee should determine the end-to-end uncertainties and bias errors, and describe the method used in combining the individual uncertainties and bias errors. For vibration analyses, frequency-dependent bias errors and uncertainties need to be determined.

When computing vibratory stresses in SSCs, it is appropriate to provide spectra of dominant stress components (usually associated with maximum tensile stress directions), as well as cumulative stress spectrum plots. The cumulative plots show the integration of a stress spectrum over frequency, and identify dominant frequency peaks.

The FRFs described in Section C.2.1.1 and forcing functions described in Section C.2.1.2 have a computational uncertainty associated with the frequencies of the response peaks attributable to resonant modes, the vibration and stress calculations need to address those uncertainties by shifting either the FRFs or the forcing functions in frequency to span the uncertainty in the response peak frequencies. An acceptable approach to resolving the uncertainty associated with natural frequencies is to align any forcing function peaks with all modal peaks within the range of frequency uncertainty, and to determine the worst-case vibration and stress. All uncertainty and bias associated with natural frequencies is eliminated with this approach. The uncertainty and bias associated with the FRF amplitudes are not eliminated by aligning all forcing function and modal peaks. An alternative approach is to perform several analyses in which the FRFs or forcing functions are shifted by increments within the frequency uncertainty range. Once again, the worst-case vibration or stress needs to be identified because the frequency uncertainty might lead to a negative (non-conservative) bias in the vibration and stress when modal peaks are misaligned with forcing function peaks.

Forced response calculations may be performed using time-domain or frequency domain approaches, provided they are verified to be accurate against computational benchmarks. When using a time-domain structural analysis approach over a limited subset of time history data acquired to infer SSC loading, it is possible that the peak loading conditions might not be included in the analysis. In this case, additional frequency-dependent bias errors and uncertainties need to be determined by comparing the time increment subset used to the total time history dataset. Linear structural response may be assumed, so that statistical assessments of the measured time histories may be related to corresponding statistical assessments of the resulting structural vibrations and stresses. The bias errors and uncertainties need to be included in the analysis results from the limited time history subset.

When using a structural dynamic analysis approach with the Rayleigh damping method, peak responses at various frequencies need to be evaluated to ensure nonconservative damping has not been applied. In particular, below and above the "anchor frequencies" where damping is specified, structural damping might be higher and artificially reduce the resonant peak response. Deviation from the accepted 1 percent damping ratio needs to be justified in determining final peak vibration and stress levels.

Benchmarking of Overall (End-to-End) Computed Response

Dynamic benchmarking of flow-, acoustically, or mechanically- induced structural response simulation procedures and structural vibration monitoring is preferable using end-to-end measurements, such as alternating surface strains on the structure. End-to-end benchmarking encompasses bias errors and uncertainties associated with loading estimates (including unknown loading mechanisms), mapping of surface loading models to structural dynamic models, and structural dynamic modeling. When multiple simulations are performed spanning a range of frequency-shifted loads (for example, +/-10 percent in increments of 2.5 percent), the upper bound of the simulations needs to be compared to the corresponding measurements. Any differences need to be addressed by frequency-dependent bias errors and uncertainties, which are applied to all subsequent dynamic analyses. It is also necessary to validate the simulation of intermediate quantities, such as loads acting on the structure. In such cases, any revisions to simulation procedures need to adhere to previously validated assumptions and measurements. For example, artificially adjusting the speed of sound or damping to achieve agreement with measured data is inappropriate. Best engineering estimates of modeling parameters need to be used, and any errors evaluated in the final end-to-end bias errors and uncertainties.

Benchmarking of FIV or MIV methodologies will produce statistical estimates of bias errors and uncertainties. The estimates need to be based on comparisons of measured and simulated acoustic and structural responses at sufficient locations to reasonably characterize all critical regions of an SSC. Standard practice is to specify uncertainties based on two standard deviations of a dataset. When a single standard deviation is used, the methodology outputs need to bound those at all measurement locations at critical peak frequency regions. If the methodology results, combined with bias errors and uncertainties, are not bounding, the discrepancies need to be used to estimate the effect of the underprediction on fatigue life such that SSC replacement will be planned before failure. In some cases where structures are subdivided into clearly divisible sections, spatially-dependent bias errors and uncertainties may be computed over those regions.

Frequency-dependent end-to-end bias errors and uncertainties computed from experimental benchmarking of simulation procedures are not universally applicable to a class of structures. For example, a steam dryer of different size, geometry, location, orientation, or construction from the prototype will be driven by flow- or mechanically induced forces that are shifted in frequency or amplitude from those of the prototype. Therefore, frequency-dependent negative bias errors should not be applied to different-sized structures or structures driven by different flow fields. In these cases, loads (or structural response functions) need to be shifted in frequency during the analysis to ensure bounding worst-case interactions between loads and response are identified.

The benchmarking of vibrations or surface strains does not ensure that peak stresses are properly calculated in a numerical structural model. Peak stresses usually occur near corners and welds and other stress concentration locations. Separate convergence studies may be necessary for these locations to ensure peak stresses are properly determined.

Stress Convergence and High-Cycle Fatigue Evaluation

For BWR and PWR reactor internals, and in particular for BWR steam dryers, the potential for high-cycle fatigue cracking at fillet-welded joints and other locations of stress concentration needs to be evaluated in detail and eliminated at the design stage. To account for uncertainty, an alternating peak stress design limit should be selected that provides a design margin compared to the ASME BPV Code allowable alternating peak stress limit. Pressure, acceleration, and strain data collected during power ascension testing are used to confirm or adjust the analytical peak stress predictions. After adjustment based on measured data during power ascension testing, the revised alternating peak stress intensity prediction is limited to the ASME BPV Code allowable alternating peak stress intensity at the material endurance limit.

Developing conservative predictions of peak stress involves three elements. First, the structural model needs to be an accurate representation of the actual structure, in terms of geometry, material properties, and boundary conditions, with sufficient model refinement to respond to the applied dynamic loads and to provide appropriate stress output for the fatigue analysis. Second, the applied dynamic loads need to be known and properly applied to the structural model. Third, the relationship between the model stress output and a conservative prediction of the peak stress needs to be known.

a. Structural Model Development

To conduct the structural analysis, a widely-used, well-verified finite element computer code (e.g., ANSYS) needs to be applied. In the current context, the analysis is linear elastic.

Modeling of reactor internals (including BWR steam dryers) typically entails using solid, plate/shell, and beam elements. Limited use of other element types may also be appropriate. Connecting plate/shell elements and beam elements to solid elements involves special modeling techniques to ensure rotational compatibility and moment transfer. Various techniques have been developed and successfully applied. It is the applicant's responsibility to verify that such connections have been appropriately modeled. This is significant because two-sided (or double) fillet welds are often used to provide a connection between a thin plate type sub-component and a heavy section, as in a BWR steam dryer. The predicted stress in the thin plate element at the connection is used in the fatigue evaluation of the connecting double fillet weld. The implemented connection modeling technique is not allowed to result in a reduction or distortion of stress in the plate at the connection.

The next step is developing a suitable finite element mesh, consistent with the loading, the expected structural response, and the intended use of the stress analysis output. The finite element mesh for the dynamic model needs to be sufficiently refined to (1) capture the spatial variation of the applied dynamic pressure loading; and (2) accurately respond up to the highest frequency contained in the dynamic pressure loading. To ensure item (2), a mesh sensitivity study needs to be conducted. For local areas of the stress model, where stress output for fatigue analysis will be extracted, it is necessary to check stress convergence by systematically reducing the local element size. Before applying appropriate SCFs to the model stress output, it needs to be established that the desired stress output has converged, or that a reasonably accurate projection of the converged stress can be made from the results of the successive mesh refinement analyses.

The final step in the model development is specification of the dynamic analysis parameters. This will depend on the selected method of solution; i.e., time domain or frequency domain. If time domain is selected, then the solution time increment should be no larger than 0.125 times the shortest period of interest. For example, if 250 Hz is the highest frequency of interest, then the solution time increment should be no larger than 0.125/250 = 0.0005. Rayleigh damping of 1 percent is acceptable for BWR steam dryers. To ensure that the solution is not over-damped in the frequency range of interest, 1 percent should be specified at 0 Hz and at the highest frequency of interest. This will be conservative at intermediate frequencies. Alternate Rayleigh damping anchor frequencies need to be quantitatively justified. For components other than BWR steam dryers, higher values of damping may be appropriate if justified.

If a frequency domain analysis procedure is selected, then FRFs at sufficient locations, for a sufficient number of frequencies, up to the highest frequency of interest, need to be calculated. It is necessary to ensure that the FRF data are sufficiently complete to achieve solution accuracy. Comparison to a time domain solution is an acceptable method to address this aspect. Modal damping of 1 percent is acceptable for BWR steam dryers. For components other than BWR steam dryers, higher values of damping may be appropriate if justified.

b. Applying Dynamic Loads to the Structural Model

The excitation mechanisms that might cause dynamic loading are described in the previous section of this regulatory guide. Application of the dynamic loading to the structural model is within the scope of the structural analysis. The loading file for dynamic analysis of rapidly changing surface pressure, with time-varying spatial distribution comprises a very large data set. The input time increment needs to be sufficiently small to capture the highest frequency input pressure fluctuations of interest. Typically, the input time increment should be no larger than 0.25 times the shortest period of interest. For example, if 250 Hz is the highest frequency of interest, then the input time increment should be no larger than 0.25/250 = 0.001 sec.

Also, the mesh in the model used for load generation usually does not coincide precisely with the mesh in the model used for structural analysis. Interpolation procedures are used to define the load at nodes in the structural model. In such cases, checks on localized loading distributions, particularly in areas of expected high stress, need to be conducted to ensure conservative load mapping has occurred.

c. Fatigue Analysis of Two-Sided Fillet Welds

Before the use of general purpose finite element structural analysis computer codes, structural analysts and experimenters developed semi-empirical methods to ensure the structural reliability of fillet-welded connections subject to cyclic loading. The methods rely on the calculation of a “nominal” average stress plus linear bending stress through the thickness of the plate-type member at the fillet-welded connection, using hand calculations or simple computer models. A fatigue strength reduction factor (effective stress concentration factor) of 4 is then applied, based on extensive cyclic load testing. The resulting “peak” stress is used in the fatigue evaluation for cyclic loading.

Currently, structural analysts rely on mathematical simulation to solve complex problems. However, because of the unknown geometric condition at the root of a fillet weld, it is not practical to directly predict the stress field using mathematical simulation, regardless of the refinement of the local model. Using the stress results from a finite element analysis in the fatigue evaluation of fillet welds depends on the local geometric complexity and the level of model refinement at the double fillet weld connection. Two methods have been evaluated by the NRC staff, and found to be acceptable, with certain limitations on their application. The methods may be used, in conjunction with using appropriate multipliers, to achieve acceptable predictions of peak stress for use in a fatigue evaluation. The intent is to ensure a level of conservatism consistent with the traditional semi-empirical method that has been successfully employed for many years.

The first method is analogous to the traditional method. The finite element analysis results from a global model of the structure, subjected to the applied dynamic loading, are used to calculate a nominal average stress plus linear bending stress at the location of the double fillet weld. The fillet weld is a local detail and is not included in the global model. Solution convergence with mesh refinement needs to be established before proceeding to the next step. The worst case nominal stress distribution at the double fillet weld location is multiplied by a factor of 4, to obtain the peak stress estimate for use in the fatigue evaluation. The most conservative approach is to assume that the calculated peak stress occurs in both the positive and negative directions, producing a peak stress range equal to 2 times the calculated peak stress. The alternating peak stress, which equals half of the peak stress range, is then equal to the calculated peak stress. At the initial design stage, a safety factor (e.g., 2) may be applied to account for uncertainty, or another factor justified. After the calculated peak stress is adjusted based on measurements taken during power ascension testing, the safety factor is not necessary. The adjusted peak stress needs to be less than the endurance limit.

The second method follows the first method, through post processing of the results of the global model analysis, as described above. This establishes the loading and location for a detailed submodel analysis. In the detailed submodel analysis, idealized fillet welds are explicitly modeled using an array of solid elements. The linearized stress distribution through the throat of the fillet weld is calculated from the solid element stress output. The adequacy of the solid element mesh in the submodel needs to be verified by a stress convergence study. The converged, linearized stress prediction at the root of the fillet weld is multiplied by a factor of 3, to obtain the peak stress estimate for use in the fatigue evaluation. From this point, the evaluation follows the first method. Because the second method involves isolation of a local region of the global model, it is necessary to verify that (1) the local model is sufficiently large to preclude boundary effects on the response of interest; (2) the boundary conditions applied to the local model properly simulate the behavior of the local region in the global model; and (3) the pressure loading is properly applied to the local model. An acceptable method to verify this is to create an intermediate local submodel, with grids and elements identical to the global model, and analyze the intermediate local submodel with the appropriate boundary conditions extracted from the global model analysis, before making any mesh refinement and local geometry changes using solid elements. The intermediate local submodel results will match the global model results if items (1), (2), and (3) have been properly implemented. The intermediate local submodel, after further mesh refinement and addition of the fillet weld solid elements, will become the final local submodel.

Operational Vibration and Stress Limits

Computed vibrations and stresses need to be compared to allowable levels, such as the fatigue criteria specified by ASME, or other criteria substantiated by testing and analysis. For steam dryers, minimum factors of safety below allowable levels need to be specified and justified. The ASME stress limits are to be used to establish operational limits on monitoring instrumentation to be applied to the structure for in-plant testing (see Section C.2.2).

Steam dryer vibration and alternating stress simulation calculations might need additional factors of safety on allowable limits. Additional factors are necessary when partial or no benchmarking of valid prototypes has been performed, or when remote assessment methods are used. An example is using circumferentially oriented strain gages on BWR MSLs to infer acoustic wave amplitudes within the MSLs. The wave amplitudes are commonly used to develop approximate fluctuating pressure loads on steam dryers. However, there might be additional localized forcing functions on a reactor internal which are not detectable at the remote monitoring locations. The additional factor of safety ensures that all loading mechanisms are bounded. There is no set value for this additional factor of safety, and it is assessed on a case-by-case basis depending on the breadth and quality of previously submitted information for similar structures. When on-dryer testing is used to benchmark the design analysis results or monitor vibration and stress, the additional factor of safety is no longer necessary, provided the measurements capture all key vibration and stress peaks.

#### 2.1.4 *Preoperational and Testing Analysis*

The applicant or licensee should summarize the calculated structural and hydro-acoustic responses for preoperational and startup testing conditions, compared to those for normal operation. This summary should address the adequacy of the test simulation to normal operating conditions. Also, the variability in reactor operating parameters can affect flow rates, working fluid mass density, pressures, and other quantities. The applicant/licensee should account for these variations and assess reactor internals and other safety-related components at worst-case operating conditions.

As-built components often differ from original designs. Welds, plating thicknesses, and other design parameters, when changed, will affect the vibration and alternating stress response of a structure. Such changes need to be captured in updated vibration and alternating stress calculations, and checked against acceptance limits.

To ensure optimal choice of sensor locations and orientations when instrumenting reactor internals or other structures for monitoring during pre-operational and start-up tests, the instrument locations need to be based on the results of structural modeling and vibration analysis using as-built specifications. To minimize measurement uncertainty, accelerometers need to be placed at or near predicted peak response locations, and strain gages need to be placed at or near locations with predicted minimum gradients in strain. Coherence needs to be maximized between sensor and critical response locations, such as welds and other stress concentration locations. The anticipated structural or hydro-acoustic vibratory response that is appropriate to each of the sensor locations for steady-state and anticipated transient preoperational and startup test conditions needs to be determined.

### 2.2 Vibration and Stress Measurement Program

The applicant or licensee should develop and implement a vibration measurement program to verify the structural integrity of 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. For SSCs with no prior history of adverse effects because of FIV, AR, AIV, or MIV; and which have been shown by analysis (using acceptable methods from Section C.2.1) to have an acceptable margin of safety against such effects, no instrumentation on the SSC or associated measurements are necessary. Measurements need to be performed, however, on systems and components that have been adversely affected by FIV, AR, AIV and MIV in the past (such as BWR steam dryers as described in Section C.2.2.1 below), and on those SSCs for which the analysis has shown less margin of safety against such effects. Instrumentation will be needed for new components that have no operating experience. The measurement program is summarized below.

### 2.2.1 *Specific Guidance for BWR Steam Dryers*

The plant startup testing to evaluate potential adverse flow effects on BWR plant reactor internals needs to include the steam dryer. For plant startup, plant data from instrumentation mounted directly on the steam dryer need to be collected at significant locations (including the outer hood and skirt, and other potential high-stress locations) to confirm that the alternating stress on individual steam dryer components will be within allowable limits during plant operation. The locations of sensors directly mounted on the dryer need to be based on the dryer structural modeling and vibration analysis. The sensors need to provide sufficient information to confirm the acceptability of the stress analysis of the entire steam dryer, and need to include pressure sensors, strain gauges, and accelerometers.

Limits (peak or RMS levels, and/or limit curves over frequency) for the steam dryer sensors need to provide assurance that the alternating stresses in the individual steam dryer components will not exceed the ASME BPV Code fatigue limits. The acceptance limits, while including the bias errors and uncertainties from the end-to-end vibration and stress analyses, need to also include errors and uncertainties associated with the vibration and stress measurement program (in particular, those associated with the data acquisition systems and instrumentation).

The MSLs may be instrumented to collect data to estimate steam pressure fluctuations and to identify the presence of flow-excited acoustic resonances that might adversely affect MSL valves and steam dryers. The direct steam dryer measured data need to be used to calibrate the MSL instrumentation and data analysis for dryer forcing function estimation before removal or failure of an excessive number of the steam dryer sensors that precludes a reliable analysis. EPU experience has shown that the application of MSL pressure measurements is complex.

Strain gage arrays may be used to relate the hoop strain in an MSL to the internal pressure. However, although accurate individual strain measurements on a flat surface are straightforward, measuring a summed set of signals across a circumferentially oriented array of strain gages in a highly pressurized pipe with curved surfaces is more challenging. The net hoop strain is often a small fraction of the total strain at a given sensor location, and the total strain can include significant bending and ovaling of components. System errors or background noise in the array installation can therefore have much larger effects on a summed array measurement than on a single sensor measurement. Sensor and weld integrity, non-uniform circumferential wall thickness distributions, wiring, and data acquisition issues (such as erroneous gain and/or calibration factors) have caused difficulties in past applications. Measurement guidance is provided below:

a. One means to help verify in-situ calibration of a strain gage array to pressure is to perform a static pressurization calibration where measured hoop strain is compared directly to plant pressure.

b. Four strain gages, evenly spaced around a pipe circumference, are necessary to filter non-breathing signals from a measurement. Based on experience, additional gages are needed to provide redundancy because of the frequent failure of gages under plant operating conditions.

c. If MSL measurements are repeated in the future with replaced gages, these measurements need to be compared with previous data to ensure reasonable consistency.

d. Instrumentation wiring needs to be properly labeled. MSL strain gage spectra need to be compared for similar locations to ensure reasonable consistency and that mislabeling has not inadvertently occurred.

e. MSL strain measurements acquired at different times, such as during the application process and during power ascension, might differ because of aging of the gages, gage welding, wiring, and data acquisition. In some cases, individual sensors might fail leading to changes in the summed array signal used to infer pressures. In these cases, comparisons to previous MSL data and any derived limits need to be made with consistent sensor sets. For example, any sensors in an MSL circumferential array that fail need to be removed from the array summation in both new and previous datasets, as well as limits and limit curves.

### 2.2.2 *In-plant Measurement Issues*

In-plant measurements of surface pressures, vibration (via accelerometers), and strain (via strain gages) can be affected by several mechanisms. Guidance for the in-plant measurement program is provided below:

a. All in-plant instrumentation needs to be qualified for the environment in which it will operate.

b. Bench testing of sensors in representative environments and on structures similar to those in the plant is necessary to establish guidelines for installation and data acquisition. Sensitivities need to be confirmed via the bench testing.

c. Strain gages that are welded to a base structure need to be installed according to vendor guidelines as modified by lessons learned from bench testing, by trained welders using appropriate welding procedures to ensure the functionality of the sensors. Past installations of strain gages to reactor internals and MSLs have not always been consistent with vendor guidelines or have been of poor quality. In particular, strain gages mounted to pressurized MSLs have not always produced measurements consistent with those on other nominally identical MSLs and locations. In some cases, additional guidance beyond nominal vendor guidelines has been necessary based on specialized laboratory tests on structures of similar size and materials.

d. Strain gage signals might be affected by static preloading when mounted to pressurized surfaces, and can be significantly affected if the welds on the gage edges crack. Periodic shunt calibration is recommended to confirm strain gage sensitivities and performance.

e. Surface pressure measurements can be affected by strong sensor vibration and result in “measuring” vibration rather than surface pressure.

f. All sensor signals can be affected by electrical noise, particularly when inadequately shielded sensor wiring passes close to strong electromagnetic fields. Tonal harmonics of electrical supply frequencies are commonly observed, but broad-band signal corruption has also been observed in previous plant data. All instrumentation wiring needs to be properly shielded and, if possible, routed through reactor penetrations that do not include electrical supply lines.

g. Data acquisition systems need to be calibrated to ensure that signals are not altered by data acquisition cards, cabling, or other mechanisms.

In-plant measurements will include data noise (for example, from power cables) that can affect benchmarking. It is acceptable to use noise reduction techniques, provided they do not lead to excessive signal reduction and unreasonable data. Electrical signals may be removed using narrow-band filtering, provided the notch filters are not wider than necessary, removing actual signals near the electrical frequencies. When notch filtering is used, the subsequent structural dynamic analyses need to include frequency shifting of the loading to ensure that reasonable loads are applied to all structural resonances. It is also acceptable to add broad-band noise to the notch-filtered frequency bands to avoid non-physical “dips” in the spectral representation of the time signals. If used, the added noise needs to be comparable to signals at adjacent frequencies. If broad-band noise filtering techniques are applied, the applicant/licensee needs to illustrate that excessive signal reduction does not occur. Coherence processing and wavelet noise reduction are both reasonable filtering techniques of broad-band noise.

### 2.2.3 *Vibration Measurement Program Documentation*

The vibration measurement program should include a description of the following systems, plans, and acceptance criteria, addressing the measurement and data issues discussed in Sections C.2.2.1 and C.2.2.2:

a. <u>Guidance on instrumentation and data acquisition and reduction system:</u> The instrumentation and data acquisition and reduction system needs to include the following:

   (1) transducer types and their specifications, including applicable frequency and amplitude ranges;

   (2) transducer positions to monitor significant lateral, vertical, and torsional structural motions of major reactor internals in shell, beam, and rigid body modes of vibration, as well as significant hydraulic responses which can be used to confirm the input forcing functions;

   (3) methods to ensure data quality (e.g., optimization of signal-to-noise ratio, relationship of recording times to data reduction provisions, and choice of instrumentation system);

(4) types and locations of transducers to provide redundancy in case some sensors fail;

(5) online data evaluation system to provide immediate verification of general data quality;

(6) procedures for determining frequency, modal content, and maximum values of response; and

(7) all bias errors (such as model underprediction) and random uncertainties (such as instrumentation error) associated with the instrumentation and data acquisition systems.

b. <u>Guidance on power ascension plan:</u> The power ascension plan needs to include the test operating conditions and provisions to compare measurements and any accompanying analyses with acceptance limits before ascension to higher power levels. Also, projected vibration and stress levels at higher power levels need to be estimated based on trending of lower power data, and shown to be below acceptance limits. In particular, the power ascension program needs to include the following, as applicable:

(1) power levels at which data should be acquired and analyzed;

(2) activities to be accomplished during data analysis;

(3) plant parameters to be monitored in comparison with applicable acceptance limits;

(4) inspections to be conducted for steam, feedwater, and condensate systems and components during the specified power levels;

(5) methods to be used to trend plant parameters (see additional guidance below for details);

(6) acceptance criteria for monitoring and trending plant parameters, and for conducting inspections (see item c below);

(7) actions to be taken if acceptance criteria are not satisfied;

(8) provisions for providing information to the NRC staff on plant data, evaluations, inspections, and procedures before and during power ascension.

c. <u>Guidance on acceptance criteria for measurements:</u> The measurement program needs to include acceptance criteria for measurements during power ascension with permissible deviations, and the bases for the criteria. The criteria need to be established in terms of maximum allowable response levels in the structure, and presented in terms of maximum allowable response levels at sensor locations. For example, typical provisions, which may be made explicit in a license condition for BWR steam dryers depending on the application, are as follows.

(1) Power ascension limits for BWR steam dryers commonly are set for two sets of criteria – Level 1 and Level 2. Exceeding Level 2 criteria typically triggers additional engineering assessments, but does not specify a plant power reduction. Exceeding Level 1 criteria dictates (a) a reduction of plant power such that the limits are satisfied, and (b) completion of engineering assessments to demonstrate satisfactory structural integrity.

(2) Power ascension limits might need to be reestablished immediately before reactor power ascension, particularly if they originally were developed earlier in the application process. If plant conditions or SSC designs change from those used to establish original limits, or sensors used to establish the limits fail before power ascension, the limits will need to be reestablished. Previous limits and accompanying benchmarking need to be updated at current plant conditions and with currently valid sensors. The updated limits and benchmarking need to be compared to previous results, but with the previous results also updated using only currently valid sensors. Any significant differences between previous and updated limits and benchmarking need to be resolved.

(3) Power ascension for BWR steam dryer needs to be in small increments when approaching full power, with data taken at each increment (e.g., 5 percent). The approach to power ascension testing can vary depending on the purpose (e.g., new plant startup or EPU power ascension with existing, new, or modified steam dryer). The applicant or licensee should work with the NRC to develop an acceptable program that establishes power levels and durations where power level should be maintained at a suitable percentage and for a suitable period of time to allow for the acquisition of data. In addition, the applicant or licensee should establish specific power levels where power will be held for an additional time period after making the startup data available to the NRC project manager such that the NRC staff may evaluate the data. In the past, the NRC staff has accepted time periods of 72 to 96 hours; and power levels of approximately 5 percent above the original licensed thermal power for EPU power ascensions at BWR plants, and 75 percent, 85 percent, and 95 percent of licensed thermal power for new BWR startup plans. After the data are made available to the NRC staff, no further NRC action is necessary to authorize ascension to the next power level when the applicable time duration is reached; enforcement action (i.e., an order) would be necessary to halt power ascension if the NRC staff's evaluation determined it was warranted.

(4) Power ascension acceptance limit checks may be of in-plant instrumentation frequency spectra (often called "limit curves"), or of peak or RMS quantities. However, if peak or RMS limits are used, (a) a sufficient number and types of sensors and locations are needed so that all critical peak stress regions on a structure are monitored, and (b) accompanying confirmations are performed to ensure that all important resonance and forcing function frequency peaks at the peak stress locations are bounded by simulation methods.

(5) If an instrumentation limit is exceeded during power ascension, triggering a reanalysis of structural alternating stress, the reanalysis may be performed using approximate methods that have been shown to be reasonable and conservative in previous benchmarking. The final structural analysis after the completion of power ascension needs to be conducted using the full analysis procedures.

(6) During power ascension, structurally mounted instrumentation may be used to re-benchmark design analyses. As discussed previously, this instrumentation needs to be chosen and located so that all key peak vibration and stress locations are properly monitored. Any re-benchmarking needs to use data from a power level that is sufficiently high for unsteady loading and dynamic response to be well above any noise floors of the measurements, and representative of the forcing functions that will occur at full power. Generally, power levels of at least 75 percent of full power are specified.

(7) As a reactor ascends in power, limits on spectra (limit curves) or peak/RMS values need to be continuously updated based on the most recent data acquired. Also, corresponding estimates of full-power levels need to be continuously updated using trending analysis. The trending needs to be based on a reasonable number of data points over variable power levels. In the event that a flow, mechanically, or acoustically coupled resonance appears, the trending needs to apply conservative functions to ensure worst-case response at higher power levels is adequately bounded. When using a polynomial trending, a minimum of a squared power law with respect to power should be used. Coupled resonance polynomial orders should be higher than a squared power. If a projected limit is exceeded, more detailed analyses will need to be conducted.

(8) End-to-end bias and uncertainties for the overall BWR steam dryer analysis procedure are updated by comparing predicted and measured strains or accelerations at each power level to confirm the conservatism of the predicted stress and vibration. Predicted responses need to be updated using the frequency-dependent end-to-end bias errors and uncertainty values. If the measured sensor data exceed the adjusted predictions, then the bias errors, uncertainty values, and limit curves need to be adjusted to ensure measured sensor responses do not exceed the adjusted predictions.

d. Guidance on test duration: The applicant or licensee should specify the planned duration of all testing in normal operating modes to ensure that the testing will subject each critical component to at least $10^6$ cycles of vibration (i.e., computed at the lowest frequency for which the component has a significant structural response) before the final inspection of the reactor internals.

e. Guidance on disposition of fuel assemblies: The applicant or licensee should address the disposition of fuel assemblies. Preoperational testing should be performed with the reactor internals important to safety and the fuel assemblies (or dummy assemblies that provide equivalent dynamic mass and flow characteristics) in position. The testing may be conducted without real or dummy fuel assemblies if it is justified (by analytical or experimental means) that such conditions will yield conservative results.

## 2.3 Inspection Program

The applicant or licensee should describe the inspection program for inspections of the reactor internals both before and after operation in modes consistent with those tested and analyzed for the design. The reactor internals should be removed from the reactor vessel for these inspections if feasible. If removal is not feasible, the inspections need to be performed using examination equipment appropriate for in situ inspection. The inspection program documentation should include the following information:

a. A tabulation of all reactor internals and local areas to be inspected, including the following details:

   (1) all major load-bearing elements of the reactor internals that are relied upon to retain the core support structure in position;

   (2) the lateral, vertical, and torsional restraints provided within the vessel;

   (3) those locking and bolting components whose failure could adversely affect the structural integrity of the reactor internals;

   (4) those surfaces that are known to be or might become contact surfaces during operation;

   (5) those critical locations on the reactor internals as identified by the vibration analysis, such as the steam dryers in BWRs; and

   (6) the interior of the reactor vessel for evidence of loose parts or foreign material.

b. A tabulation of specific inspection areas to verify segments of the vibration analysis and measurement program.

c. A description of the inspection procedure, including the method of examination (e.g., visual and nondestructive surface examinations), method of documentation, provisions for access to the reactor internals, and specialized equipment to be employed during the inspections to detect and quantify indications of vibration.

## 2.4 Documentation of Results

The applicant or licensee should provide for the review of the results of the vibration and stress analysis, measurement, and inspection programs to determine whether the measurement and inspection acceptance criteria are satisfied. A summary of the results should be prepared in the form of initial, preliminary and final reports as follows:

a. The initial report, prepared during the design approval process, should summarize the analysis procedures, design and analysis results, margins of safety for vibration and stress, test plan and acceptance criteria, and any alternatives or anomalies.

b. The preliminary report should summarize the evaluation of the initial and, as necessary, limited processed data and the results of the initial inspection program with respect to the test acceptance criteria. Any changes made to the analysis procedure and simulated results that occurred subsequent to the initial report based on updated benchmarking or in-process changes need to be identified and justified. Anomalous data that could bear on the structural integrity of the reactor internals need to be identified, as well as the method to be used for evaluating such data.

c. If the results of the CVAP are acceptable, the final report should be prepared after completion of vibrating testing, and needs to include the following information:

    (1) description of any deviations from the specified measurement and inspection programs, including instrumentation reading and inspection anomalies, instrumentation malfunctions, and deviations from the specified operating conditions;

    (2) comparison between measured and analytically determined structural response (including natural frequencies, mode shapes, and damping factors, if measurable) and hydro-acoustic vibration, strain, and pressure response (including those parameters from which the input forcing function is determined) for the purpose of establishing the conservatism of the predictive analysis techniques;

    (3) updates to modeling procedures and/or bias errors and uncertainties based on results from (2);

    (4) determination of the margins of safety associated with operation under normal steady-state and anticipated transient conditions, including the margins of safety associated with any flow-excited acoustic or structural resonances; and

    (5) evaluation of unanticipated observations or measurements that exceeded acceptable limits not specified as test acceptance criteria, as well as the disposition of such deviations.

d. If (a) an inspection of the reactor internals reveals defects, evidence of unacceptable motion, or excessive or undue wear; (b) the results from the measurement program fail to satisfy the specified test acceptance criteria; or (c) the results from the analysis, measurement, and inspection programs are inconsistent, the final report needs to include an evaluation and description of the modifications or actions planned to justify the structural adequacy of the reactor internals and an evaluation that identified the deficiencies in the initial analysis methods that yielded unpredictable results.

## 2.5 Schedule

The applicant or licensee should establish a schedule for the vibration assessment program to be provided to the NRC (1) during the CP or OL review for new nuclear reactor applications under 10 CFR Part 50, (2) during the review of DC applications under 10 CFR Part 52, (3) during the review of COL applications under 10 CFR Part 52, (4) during the review of EPU applications, or (5) before major plant modifications. The schedule needs to address the following considerations:

a. For CP applications under 10 CFR Part 50, the reactor internals design needs to be classified in the preliminary safety analysis report (PSAR) as a prototype, limited prototype or a non-prototype. Experimental or analytical justification of the non-prototype classification needs to be presented during the CP review under 10 CFR Part 50. For OL applications, the classification may be revised in the FSAR if schedule changes with respect to the previously designated reference reactor make such reclassification appropriate.

For applications submitted under 10 CFR Part 52, the issues related to justification of the non-prototype classification need to be resolved during the review of the DC or COL application. If the justification is insufficient to meet the guidelines provided in this regulatory guide, the applicant will need to develop a test plan to obtain additional data as a prototype as discussed in this regulatory guide. The reactor internals design needs to be classified as a prototype, limited prototype or a non-prototype category in the application. If the internals are classified as non-prototype, the applicable prototype reactor internals need to be identified.

b. During the staff's review of the CP, OL, DC, or COL application, as appropriate, the scope of the CVAP needs to be established.

c. The preoperational test procedures, power ascension program, and CVAP report need to be made available to the NRC in a timely manner for staff review and resolution of comments.

d. The preliminary and final reports, which together summarize the results of the vibration analysis, measurement, and inspection programs, should be submitted to the NRC within 60 and 180 days, respectively, following the completion of vibration testing or earlier if the analysis reveals operational issues. As applicable, a full steam dryer stress analysis report and evaluation should be submitted to the NRC within 90 days of first reaching 100 percent thermal power.

## 3. CVAP FOR LIMITED PROTOTYPE REACTOR INTERNALS

If the operating conditions for the limited and the applicable valid prototype reactor internals are the same, the CVAP for limited prototype reactor internals important to safety needs to be performed at all significant flow, temperature, and pressure conditions associated with normal steady-state and anticipated transient operation under the same test conditions imposed on the valid prototype. However, if there are differences in the operating conditions, the effect of these differences from the operating conditions of the valid prototype on the structural integrity of the limited prototype reactor internals needs to be evaluated based on the results of a CVAP.

Because of similarities to a valid prototype, the assessment of a limited prototype might not involve a vibration measurement program as comprehensive as the measurement program applicable to a prototype. One example is the possible use of only MSL pressure measurements in BWR plants to infer the fluctuating loading, and subsequently the alternating stress state for a steam dryer that is similar in design and operation to a valid prototype, which has previously been benchmarked using comprehensive on-dryer vibration and strain measurements. The applicant or licensee may justify a limited prototype assessment through use of a limited subset of on-dryer measurements during the start-up vibration measurement program. In cases where large safety margins on vibration and alternating stress exist, the applicant or licensee might justify using remote MSL-based monitoring alone. See Section C.1.4 of this regulatory guide for special considerations for classifying reactor internals in multi-unit plants and standard reactor designs.

Inspection, documentation, and schedules for limited prototype reactor internals should be in accordance with the program guidelines delineated in Sections C.2.3, C.2.4, and C.2.5 of this regulatory guide.

### 3.1 Vibration and Stress Analysis Program

In preparing the vibration and stress analysis program for the limited prototype, the applicant or licensee should specify the valid prototype, and justify its use to support the limited prototype classification. If the valid prototype CVAP was conducted on a reactor outside the United States, the details and the results of the program need to be included in the limited prototype description and need to meet the criteria in this regulatory guide.

The vibration and stress analysis for the applicable valid prototype, which includes a summary of the anticipated structural and hydraulic response and test acceptance criteria, should be updated to account for the differences between the valid prototype and the limited prototype reactor internals. The vibration and stress analysis related to the differences needs to be consistent with the general guidelines delineated in this regulatory guide for prototype reactor internals.

The applicant or licensee should be aware that minor design or operating changes from the valid prototype might increase substantially the vibration and alternating stresses, not only for the components with modified design, but also for other components that might appear unrelated to the design modifications. As an example, a small increase in the length of the SRV standpipes can trigger acoustic resonance in the MSLs and thereby substantially increase the alternating loading on the steam dryer of BWRs. Change in the RRP speed is another example, where the VPF may shift to align with a resonance. If these issues exist, the applicant or licensee should address them in the CVAP.

### 3.2 Vibration Measurement Program

The applicant or licensee should develop and perform a vibration measurement program for limited prototype reactor internals during preoperational and startup testing. Generally, the vibration measurements program would be confined to the limited prototype internals, but if the vibration analysis indicates possible initiation of vibration feedback excitation mechanisms because of design modifications of the limited prototypes, or if the applicable valid prototype is operating at conditions close to those at which vibration feedback excitation mechanisms can be initiated, the vibration measurement program needs to be expanded to include other reactor internals that might be adversely affected by any possible self-excited vibration mechanisms.

Sufficient and appropriate instrumentation needs to be applied to verify that the vibratory response of the limited prototype reactor internals is consistent with the vibration analysis results, test acceptance criteria, and vibratory response observed in the valid prototype. The vibration measurement program for a limited prototype should follow the general guidelines for the vibration measurement program delineated in Section C.2.2 of this regulatory guide for prototype reactor internals.

If the measured responses are found to be significantly higher than the anticipated responses for specific components (i.e., above acceptance limits), those components should be removed from the reactor vessel and visually examined, if feasible. Components for which removal is not feasible will need to be examined in situ using appropriate inspection equipment. The interior of the reactor vessel needs to be visually checked for loose parts and foreign material. In addition, the cause for the higher responses needs to be identified and adequately resolved by re-evaluating the vibration analysis and/or the measurement program. If further evaluation identifies a fundamental difference in response between the referenced valid prototype and the limited prototype, then it is necessary to implement a CVAP for prototype reactor internals. Classification as a limited prototype is no longer valid.

For an applicant or licensee planning to use remote monitoring measurements to qualify a limited prototype, the bias errors and uncertainties of the instrumentation, measurements, and measurement system need to be factored into the acceptance criteria. For example, when planning to use limits on MSL pressure fluctuations measured by strain gage arrays to qualify the alternating stress state of a BWR steam dryer, the expected variability and bias of the measurements need to be quantified and compared to those made on the valid prototype. Use of a limited number of sensors on the steam dryer may provide an acceptable means to assess the variability of the vibration measurement procedure. Instrumentation and data acquisition for the limited prototype should be similar to that used in the prototype plant. However, when the results of various measurements are being compared, whether they are obtained from different plants or from different power levels of the same plant, the same noise filtering technique should be used in all measurements being compared.

## 4. CVAP FOR NON-PROTOTYPE REACTOR INTERNALS

During the preoperational and startup test program, non-prototype reactor internals important to safety need to be evaluated for all significant flow, temperature, and pressure conditions associated with normal steady-state and anticipated transient operation under the same test conditions imposed on the applicable prototype. Evaluation of the effects of such operation on the structural integrity of the non-prototype reactor internals needs to be based on the results of a CVAP.

Based on current operating experience, the applicant/licensee for a BWR nuclear power plant should provide detailed justification if it proposes to classify a BWR steam dryer as a “non-prototype” reactor internal component.

See Section C.1.4 of this regulatory guide for special considerations for classifying reactor internals in multi-unit plants and standard reactor designs.

Inspection, documentation, and schedules for non-prototype reactor internals should be in accordance with the guidelines delineated in Sections C.2.3, C.2.4, and C.2.5 of this regulatory guide.

### 4.1 Vibration and Stress Analysis Program

When developing the vibration and stress analysis program for non-prototype reactor internals, the applicant or licensee should provide sufficient justification to support the non-prototype classification. If the valid prototype CVAP was conducted on a reactor outside the United States, the details and the results of the program need to be included in the application related to a non-prototype, and meet the criteria in this regulatory guide.

The vibration and stress analysis for the non-prototype, which includes a summary of the anticipated structural and hydraulic response and test acceptance criteria, should be updated to account for any nominal differences that might exist between the valid prototype and the non-prototype reactor internals. The vibration and stress analysis update related to any nominal differences needs to be conducted in a manner consistent with the general guidelines delineated in this regulatory guide for prototype reactor internals.

As noted above, the applicant or licensee should be aware that minor design or operating changes from the valid prototype might substantially increase the vibration and alternating stresses, not only for the components with modified design, but also for other components that might appear unrelated to the design modifications. As an example, a small increase in the length of the SRV standpipes can trigger acoustic resonance in the MSLs and thereby substantially increase the alternating loading on the steam dryer of BWRs. Change in RRP speed is another example, where the VPF may shift to align with a resonance. If these issues exist, the applicant or licensee should address them in the CVAP.

### 4.2 Vibration Measurement Program

The vibration measurement program for non-prototype reactor internals justified by a valid prototype may be reduced if an inspection program for the non-prototype reactor internals is implemented. For example, where reactor internals other than steam dryers are justified as "non-prototype," the vibration measurement program may be omitted where the inspection program is implemented.

If a vibration measurement program is proposed in lieu of an inspection program for non-prototype reactor internals, the vibration measurement program needs to have sufficient and appropriate instrumentation to verify that the vibratory response of the measured internals is consistent with the vibration analysis results, test acceptance criteria, and vibratory response observed in the valid prototype. The vibration measurement program should follow the general guidelines for the vibration measurement program delineated in Section C.2.2 of this regulatory guide for prototype reactor internals.

If the measured responses are found to be significantly higher than the anticipated responses for specific components (i.e., above acceptance limits), those components should be removed from the reactor vessel and visually examined, if feasible. Components for which removal is not feasible need to be examined in situ using appropriate inspection equipment. The interior of the reactor vessel needs to be checked for loose parts and foreign material. In addition, the cause for the higher responses needs to be identified and adequately resolved by re-evaluating the vibration analysis and/or the measurement program. If further evaluation identifies a fundamental difference in response between the referenced valid prototype and the non-prototype, then it is necessary to implement a CVAP for prototype or limited prototype reactor internals.

# D. IMPLEMENTATION

The purpose of this section is to provide information on how applicants and licensees[2] may use this guide and information regarding the NRC's plans for using this regulatory guide. In addition, it describes how the NRC staff complies with 10 CFR 50.109, "Backfitting," and any applicable finality provisions in 10 CFR Part 52 "Licenses, Certifications, and Approvals for Nuclear Power Plants."

**Use by Applicants and Licensees**

Applicants and licensees may voluntarily[3] use the guidance in this document to demonstrate compliance with the underlying NRC regulations. Methods or solutions that differ from those described in this regulatory guide may be deemed acceptable if they provide sufficient basis and information for the NRC staff to verify that the proposed alternative demonstrates compliance with the appropriate NRC regulations. Current licensees may continue to use guidance the NRC found acceptable for complying with the identified regulations as long as their current licensing basis remains unchanged**.**

Licensees may use the information in this regulatory guide for actions which do not require NRC review and approval such as changes to a facility design under 10 CFR 50.59, "Changes, Tests, and Experiments." Licensees may use the information in this regulatory guide or applicable parts to resolve regulatory or inspection issues.

**Use by NRC Staff**

The NRC staff does not intend or approve any imposition or backfitting of the guidance in this regulatory guide. The NRC staff does not expect any existing licensee to use or commit to using the guidance in this regulatory guide, unless the licensee makes a change to its licensing basis. The NRC staff does not expect or plan to request licensees to voluntarily adopt this regulatory guide to resolve a generic regulatory issue. The NRC staff does not expect or plan to initiate NRC regulatory action which would require the use of this regulatory guide without further backfit consideration. Examples of such NRC regulatory actions that the NRC does not expect or plan to take include issuance of an order requiring the use of the regulatory guide, requests for information under 10 CFR 50.54(f) as to whether a licensee intends to commit to use of this regulatory guide, generic communication, or promulgation of a rule requiring the use of this regulatory guide.

During regulatory discussions on plant specific operational issues, the staff may discuss with licensees various actions consistent with staff positions in this regulatory guide as one acceptable means of meeting the underlying NRC regulatory requirement. Such discussions would not ordinarily be considered backfitting even if prior versions of this regulatory guide are part of the licensing basis of the facility. However, unless this regulatory guide is part of the licensing basis for a facility, the staff may not represent to the licensee that the licensee's failure to comply with the positions in this regulatory guide constitutes a violation.

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[2] In this section, the term "licensees" refers to licensees of nuclear power plants under 10 CFR Parts 50 and 52, and the term "applicants" refers to applicants for licenses and permits for (or relating to) nuclear power plants under 10 CFR Parts 50 and 52, and applicants for standard design approvals and standard design certifications under 10 CFR Part 52.

[3] In this section, "voluntary" and "voluntarily" means that the licensee is seeking the action of its own accord, without the force of a legally binding requirement or an NRC representation of further licensing or enforcement action.

If an existing licensee voluntarily seeks a license amendment or change and (1) the NRC staff's consideration of the request involves a regulatory issue directly relevant to this new or revised regulatory guide and (2) the specific subject matter of this regulatory guide is an essential consideration in the staff's determination of the acceptability of the licensee's request, then the staff may request that the licensee either follow the guidance in this regulatory guide or provide an equivalent alternative process that demonstrates compliance with the underlying NRC regulatory requirements. This is not considered backfitting as defined in 10 CFR 50.109(a)(1) or a violation of any of the issue finality provisions in 10 CFR Part 52.

Additionally, an existing applicant may be required to comply with new rules, orders, or guidance if 10 CFR 50.109(a)(3) applies.

If a licensee believes that the NRC is either using this regulatory guide or requesting or requiring the licensee to implement the methods or processes in this regulatory guide in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfit appeal with the NRC in accordance with the guidance in NRC Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection" (Ref. 8) and NUREG-1409, "Backfitting Guidelines," (Ref. 8) .

List of Acronyms

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| AIV | acoustic-induced vibration |
| ANSI | American National Standards Institute |
| AR | acoustic resonance |
| ASME | American Society of Mechanical Engineers |
| AVB | anti-vibration bar |
| BUF | bump up factor |
| BWR | boiling water reactor |
| CFD | computational fluid dynamics |
| COL | combined license |
| CRDM | control rod drive mechanism |
| CRDS | control rod drive system |
| CVAP | comprehensive vibration assessment program |
| DC | design certification |
| DNS | direct numerical simulation |
| EPU | extended power uprate |
| FEI | fluid-elastic instability |
| FIV | flow-induced vibration |
| FRF | frequency response function |
| ITAAC | inspections, tests, analyses, and acceptance criteria |
| LES | large eddy simulation |
| MASR | minimum alternating stress ratio |
| MIV | mechanical-induced vibration |
| MSL | main steam line |
| OMB | Office of Management and Budget |
| PSD | power spectral density |
| PWR | pressurized water reactor |
| RMS | root mean square |
| RPV | reactor pressure vessel |
| RRP | reactor recirculation pump |
| SG | steam generator |
| SMR | small modular reactor |
| SMT | scale model testing |
| SCF | stress concentration factor |
| SSC | systems, structures and components |
| VPF | vane passing frequency |

# REFERENCES[4]

1. *U.S. Code of Federal Regulations*, "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter 1, Title 10, "Energy" (10 CFR 50).

2. *U.S. Code of Federal Regulations*, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Part 52, Chapter 1, Title 10, "Energy" (10 CFR 52).

3. U.S. Nuclear Regulatory Commission (NRC), "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," NUREG-0800, Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components.

4. American Society of Mechanical Engineers (ASME), "Rules for Construction of Nuclear Power Plant Components," Boiler and Presssure Vessel Code, Section III, Division I, New York, NY.[5]

5. NRC, "Initial Test Program for Water-Cooled Nuclear Power Plant," Regulatory Guide 1.68.

6. Weaver, D.S., et. al., "Flow-induced vibrations in power and process plants: Progress and prospects," Journal of Pressure Vessel Technology, 122, pp. 339-347.

7. DeBoo, et. al., "Identification of Quad Cities Main Steam Line Acoustic Sources and Vibration Reduction," ASME PVP2007-26658, Proceedings of ASME PVP 2007 conference, San Antonio, Texas, July 2007.

8. NRC, "Management of Facility-Specific Backfitting and Information Collection," Management Directive 8.4.

9. NRC, "Backfitting Guidelines," NUREG-1409.

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4 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

5 Copies of American Society of Mechanical Engineers (ASME) standards may be purchased from ASME, Two Park Avenue, New York, New York 10016-5990; telephone (800) 843-2763. Purchase information is available through the ASME Web-based store at http://www.asme.org/Codes/Publications/.