U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH

February 2017
Revision 4

REGULATORY GUIDE

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REGULATORY GUIDE 1.20

(Draft was issued as DG-1323, dated July 2015)

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 during preoperational and startup testing. The scope of this RG covers boiling water reactor (BWR) and pressurized water reactor (PWR) reactor internals, and small modular reactor (SMR) reactor internals.

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 RG.

Applicable Regulations

- CFR, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities," (10 CFR 50).
    - 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 1, "Quality Standards and Records," requires that

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reactor internals be designed, fabricated, and tested to quality standards commensurate with the importance of the safety functions to be performed.

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

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 RGs in order to (1) provide guidance to applicants and licensees, (2) explain techniques that the staff uses in evaluating specific problems or postulated events, and (3) inform the public about methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations. Regulatory Guides are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs 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 RG contains and references information collections covered by 10 CFR Parts 50 and 52 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget (OMB), control numbers 3150-0011 and 3150-0151.

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TABLE OF CONTENTS

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 the reactor internals of BWRs and PWRs. This revision also expands the scope to include reactor internals of SMRs, with guidance for the control rod drive system (CRDS) and control rod drive mechanisms (CRDMs) which might be contained within an SMR integral reactor vessel.

Background

Nuclear power plant operating experience has demonstrated that flow, acoustic, and mechanical induced vibrations and resonances can adversely affect plant systems and their components, and reactor internals in particular. Some plant internal 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. An effective CVAP can limit potential adverse flow effects on reactor internals for BWR and PWR nuclear power plants, including SMRs, during design, construction, and operation, including situations when power uprates or major plant modifications are proposed.

The expanded guidance in Revision 4 is based in part on lessons learned from the review of recent new plant applications, EPU applications, and BWR steam dryer operating experience. Much of the revised guidance uses steam dryer lessons learned as examples (see Appendix A to this RG). Although the examples are specific to steam dryers, the generic guidance is applicable to all reactor internals. In addition, Revision 4 simplifies and clarifies the prototype, limited prototype, and non-prototype classifications of reactor internal configurations.

The consequences of FIV, AR, AIV, and MIV have been found to 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, as a result of these phenomena, 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 which may have been partially caused by the reactor pump mechanical excitation at its vane passing frequency (VPF).

Operating experience has revealed failures of steam dryers and main steam system components (including relief valves) in BWR nuclear power plants following EPU implementation. Studies of those failures have determined that flow-excited acoustic resonances 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 might also excite the acoustic modes of the steam columns in the MSLs, causing extremely high sound pulsations 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 (e.g., 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 vibration. 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. Some nuclear power plant licensees have developed scale model testing (SMT) and structural and acoustic models to evaluate potential adverse flow 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 (RCS) components into a single compact integral reactor module may affect FIV and MIV, even though the flow rates are expected to be lower than those in traditional reactors. The small size and compactness of SMRs result in small flow areas and therefore the flow velocities in SMRs with RRPs can be comparable to those in BWRs and PWRs. SMRs with natural-circulation RCS flow might have flow distribution affected by the phenomena described in this RG. In addition, the close proximity of SMR internals and external components like pumps and valves could lead to unexpected and undesirable interactions between forcing functions and resonances. For FIV in a specific SMR design, these effects may result from 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, RRPs and the valves could also generate mechanical excitation in connected piping and other structures. If exciting frequencies coincide with the natural frequencies of the SSCs or acoustic resonance frequencies, unacceptable vibration levels or flow disruptions 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, or (c) interference with the operation of the CRDMs.

"Reactor internals," as used in this RG, 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 American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (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, 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

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

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

- RRPs
- riser
- SGs
- pressurizer
- CRDMs and supports
- feedwater lines

Information That May be Useful for Other Plant Components

Although the scope of this RG is applicable to reactor internals in BWR, PWR and SMR nuclear power plants, the information in this RG may be useful in evaluating the design and operating performance of nuclear power plant fluid systems to avoid adverse effects of vibration phenomena on plant SSCs other than reactor internals. For example, this RG provides information that might be helpful in the evaluation of potential adverse flow effects on SG internals and tubes in PWRs. In addition, this RG describes phenomena that can cause adverse flow effects (such as damaging vibration) to components (such as valves, pipe supports, and snubbers) in steam systems as a result of the flow of steam over a branch line that generates an acoustic resonance. The phenomena described in this RG can also occur in liquid fluid systems such as flow of reactor coolant over a branch line in the RCS that results in damaging vibration of components in the branch line. Nuclear power plant designers, applicants, and licensees might find this information helpful in avoiding adverse flow effects as a result of the phenomena described in this RG for SSCs in a wide range of fluid systems and applications in a nuclear power plant.

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's 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. Revision 2 included various changes in the corresponding vibration assessment programs and to 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 RG 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 RG describes acceptable methods for evaluating the potential adverse effects from FIV, AR, AIV, and MIV. Section C.2 of this RG provides detailed information for these vibration mechanisms. 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. 4), 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 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 excitation mechanisms and vibration 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 as prototype, limited prototype, or non-prototype. The individual components comprising the reactor internals in a nuclear power plant should be evaluated together in determining whether the reactor internals are to be classified as a prototype, limited prototype, or non-prototype. In some cases, a single component may be considered separately. The classification of BWR steam dryers is discussed in Appendix A to this RG.

Figure 1 and Table 1 summarize the guidance provided in Sections C.2-C.4 for the different prototype classifications.

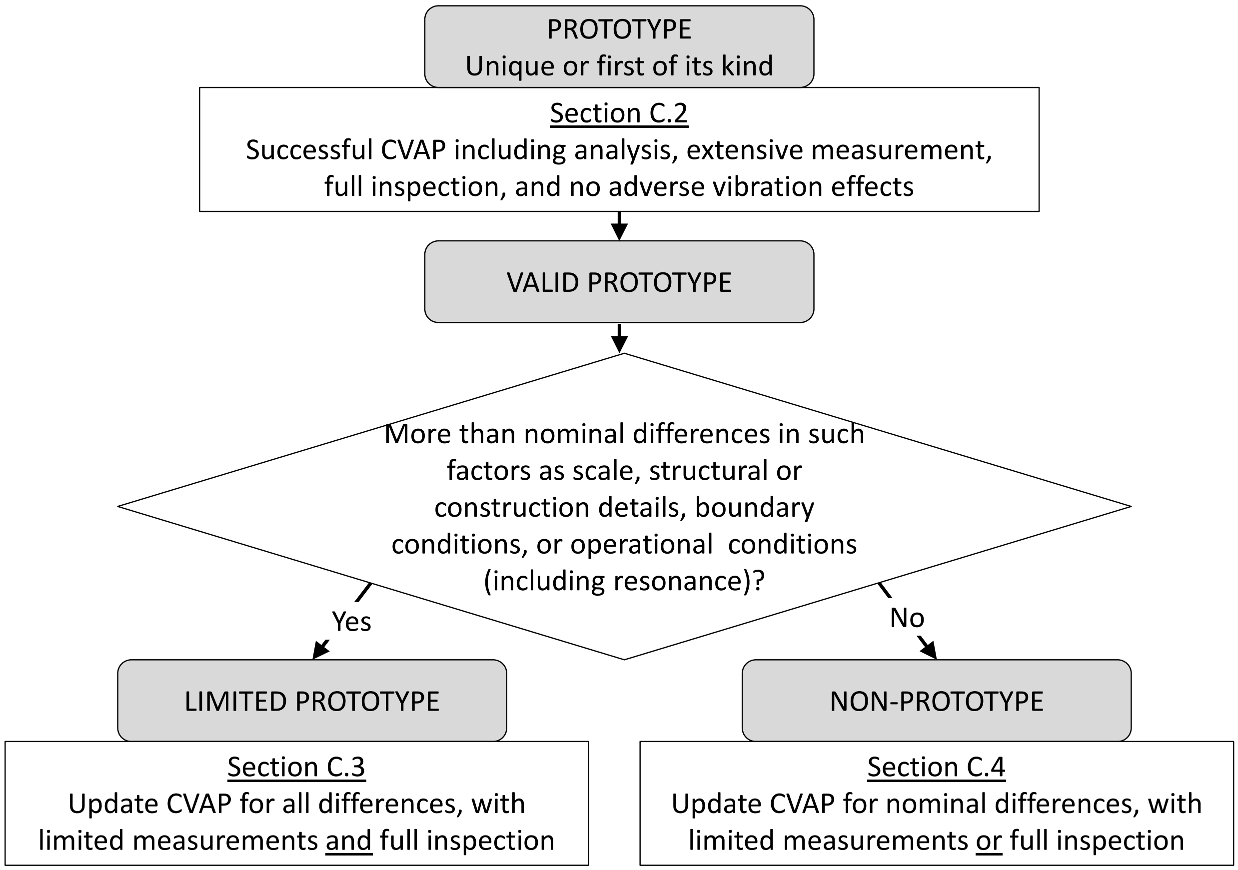

Figure 1. Flowchart for prototype classification (See Appendix A for Steam Dryers)

	Vibration and Stress Analysis Program	**Vibration Measurement Program**	**Inspection Program**
Prototype	Comprehensive	Extensive	Full inspection. Acceptance criteria include no evidence of: loose parts, cracking, unacceptable motion, and significant wear
Limited Prototype	CVAP update for all differences	Limited measurements	Same as Prototype
Non-Prototype	CVAP update for all differences	Optional[1] (may be waived in favor of **Inspection Program**) If performed, limited measurements	Optional[1] (may be waived in favor of **Vibration Measurement Program**) If performed, same as Prototype
[1] Except for BWR Steam Dryers; See Appendix A.			

Table 1. Summary of prototype classification guidance (See Appendix A for Steam Dryers)

1.1 Prototype

A “prototype” is a configuration of reactor internals, or a single component that, because of the 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 vibration or excessive loading, it may be classified as a “valid prototype.” A valid prototype may be referenced in subsequent applications, as appropriate, subject to the restrictions and provisions defined below. The duration of the CVAP needed to validate a prototype is assessed on a case-by-case basis. The results of this CVAP should quantitatively demonstrate that there were no adverse vibration phenomena or excessive loading. Reactor internals should be evaluated on a case-by-case basis for justification as a valid prototype. For example, to be a valid prototype, natural circulation reactor internals (i.e., reactor internals in a natural circulation reactor) will need to complete the CVAP with no adverse inservice vibration phenomena and at least one inspection following initial startup testing with no adverse indications (if a preoperational test and inspection is not performed).

See Section C.1.4 of this RG 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 in Section C.2.

1.2 Limited Prototype

Reactor internals may be classified as a “limited prototype” where they are similar to “prototype” reactor internals but cannot be justified as “non-prototype” reactor internals. 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 or licensee should demonstrate that the differences in arrangement, design, size, and operating conditions (including RPV and MSL and safety relief valve (SRV) flow-induced pressures in BWR nuclear power plants) 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 RG.

See Section C.1.4 of this RG 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 or non-applicability of all potential sources of vibratory excitation that are described below in Section C.2 needs to be assessed and documented.

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 with no adverse vibration phenomena or excessive loading. The NRC staff will evaluate this proposal as part of the application.

1.3 Non-Prototype

Reactor internals may be classified as a “non-prototype” where they have substantially the same arrangement, design, size, and operating conditions as a “valid prototype.” Any nominal differences in arrangement, design, size, and operating conditions (including RPV and MSL and SRV flow-induced pressures in BWR nuclear power plants) have been quantitatively shown to have no significant effect on the excitation mechanisms and vibratory response previously determined for a valid prototype. If this determination cannot be justified, the NRC staff will classify the reactor internals as a prototype or limited prototype as described in this RG.

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 RG 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 or 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 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 the reactor internals in the other units on the same site or standard reactor design units at multiple sites.

For example, an internal component in one unit may be classified as a prototype and the same components in the other units may be classified as limited prototype or non-prototype where justified. The applicant or licensee may demonstrate that an internal component classified as a limited prototype in a reactor unit at a multi-unit plant performs in a manner consistent with the component in the unit classified as a prototype by using a reduced amount of instrumentation on the limited prototype component or, where justified, using remote instrumentation rather than on-structure instrumentation for the limited prototype component. Using remote instrumentation for monitoring limited prototypes is acceptable when (a) prototype benchmarking shows a high margin of safety in the component alternating stresses (for example, in the past the NRC has found a margin of safety of 2.0 to be acceptable for remote monitoring of steam dryers), and (b) prototype remote measurements do not reveal anomalies in data quality that would add significant uncertainty in monitoring the component. See Appendix A to this RG for more guidance on applying the above procedure to BWR steam dryers.

Another 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 preoperational or 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 preoperational or initial

startup testing. The applicant or licensee should follow the applicable guidance in this RG for prototype, limited prototype, and non-prototype reactor internals for the individual reactor units.

1.5 Special Considerations for Replacement/Modification of Individual Reactor Internal Components

The definitions of prototype, limited prototype, and non-prototype also apply in cases where an individual internal component is being replaced or modified. When it can be shown that the changes to a replaced or modified component have negligible effect on the FIV, MIV, and AIV of other reactor internals, only the replaced or modified internal component needs to satisfy the guidance in this RG. However, if it cannot be shown with reasonable assurance that the replaced or modified component has a negligible effect on the FIV, MIV, and AIV of other reactor internal components, then the other affected internal components should also be evaluated in accordance with the guidance in this RG.

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 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 and licensees should analyze the effects of potential excitation mechanisms that can affect the reactor internals. It is important to note that in many cases, the excitation mechanisms originate outside the reactor pressure vessel. 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), AR excitation by separated flow instabilities, vibration induced by vortex shedding excitation, and fluid-elastic instability (FEI). FIV of structures in cross-flow can occur either in a direction perpendicular to the flow direction (also called the lift direction) or parallel to the flow direction (also called 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 has 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 or licensee should develop the CVAP to include possible streamwise (or in-plane) vibration because of FEI and vortex shedding excitation. Other examples of FIV and AR are provided from BWR EPU experiences. As previously noted, studies of past failures have determined that flow-excited AR within isolation valves, in standpipes of 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 steam flow turbulence, boiling water adjacent to the skirt, and other internal dynamic forces) can result in undesirable dynamic stresses. While these examples are not applicable to all reactor internals, they serve as guidance for an applicant to consider when evaluating all potential FIV and AR issues in the CVAP.

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 inflow. 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.

The CRDS in some SMRs is 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, AR, 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 AR, and mechanical sources should be addressed in the CRDS design.

The evaluation of potential adverse effects of FIV, AR, AIV and MIV on nuclear power plant reactor internals includes consideration of bias errors and uncertainties in the predictive analysis and in the measurement program. Bias errors might result from the underprediction 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. Note that it is also acceptable to use bounding

assumptions for bias errors and uncertainties (provided they are substantiated) instead of rigorous benchmarking. However, bounding assumptions can often lead to overly conservative vibration and stress estimates that may necessitate further evaluation.

The overall CVAP for reactor internals is implemented over several phases. A description of the implementation along with as-designed predictive results demonstrating acceptable structural integrity 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 licensed under 10 CFR Part 52.

2.1 Vibration and Stress Analysis Program

To provide confidence in the structural integrity of the prototype reactor internals prior to the preoperational testing and plant startup, 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. Based on operating experience, the analyses need to address the following aspects:

a. Identify significant vibration and acoustic resonances caused by various excitation mechanisms that have the potential to damage reactor internal components.

b. Determine the pressure and force fluctuations and vibration in the applicable plant systems. Such pressure fluctuations and vibration can result from various excitation mechanisms, such as FIV, AR, AIV and MIV. 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 analysis. If necessary, the applicant or licensee should perform modifications to the structure or other components to demonstrate design margin to Code allowable limits. A BWR applicant or licensee should perform a rigorous assessment of stress in steam dryers as specified in Appendix A.

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; and

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.

2.1.1.1 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 necessitates comparing measured and simulated data. The differences between measured and simulated natural frequencies for identical 'matched' mode shapes[2] are used to establish uncertainty ranges for FRFs and final response calculations. For example, if benchmarking reveals that the simulated natural frequencies are within +/-10 percent of those measured (for matched simulated and measured mode shapes), 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 responses retained and applied in the analysis. Several analysis increments are used, depending on the damping assumed. For example, previous acceptable analyses assuming 1 percent damping have used 2.5 percent frequency shifting increments (i.e., loads are shifted by -10 percent, -7.5 percent, -5.0 percent, and so on up to +7.5 percent, +10 percent). If higher damping is justified, fewer shifting increments may be allowed. It should be noted that the +/-10 percent discussion above is an example. In some cases, simulated and measured resonance frequencies may be closer (+/-5 percent for example), or further separated (+/-20 percent).

2.1.1.2 Vibration Damping

The applicant or licensee should substantiate assumed vibration damping. Note that vibration damping in operating nuclear power plant internals (including steam dryers) is induced by many mechanisms, including but not limited to:

(a) structural damping due to internal hysteretic losses,
(b) frictional damping at structural joints and interfaces,
(c) radiation damping caused by interaction between vibrating structures and surrounding steam or water; and
(d) viscous damping from aerodynamic/hydrodynamic oscillating flow around edges or through holes.

In some instances, the NRC's damping guidance for very-low-frequency seismic analyses has been incorrectly cited 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 at typical vibration amplitudes 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 its measurement techniques for damping in the modeled environment (including air, steam, or water environments). Based on past experiences, specification of vibration damping

[2] A standard approach for matching measured and simulated mode shapes is using the Modal Assurance Criterion, described in Allemang, 2003, Reference 11.

coefficients greater than 1 percent needs to be substantiated with measurements. When using a structural dynamic analysis approach with the Rayleigh damping method, maximum responses at various frequencies need to be evaluated to justify that conservative damping has been applied. In particular, below and above the "anchor frequencies" where the target damping is specified, modeled vibration damping is higher and may reduce the response. Deviation from the accepted 1 percent damping ratio needs to be justified in determining final vibration and stress levels.

2.1.1.3 Frequency Response Functions

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. 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 errors 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. The effects of the variability in boundary conditions need to be evaluated and either shown to be negligibly small or accounted for in the vibration analysis. For example, boundary condition variability could lead to coincidence of resonance frequencies and tonal excitation, causing a strong increase in structural response. Large uncertainty in boundary condition impedance (mass and/or stiffness) can also affect broad-band structural response. In general, if boundary condition uncertainty leads to more than a 10 percent increase in structural response, it should be addressed, with the worst-case response reported.

Examples of uncertain boundary conditions include steam dryers resting on mounting pads attached to the RPV with variable amounts of contact area and supported load during normal operating 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 loads for all reactor internals, up to the full licensed power level, and validate the methods used to determine the load definitions, based on validated SMT and/or data acquired from other plants. Section C.2.1.3 presents additional guidance on benchmarking.

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 as determined to be applicable.

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 loads in plant systems. In recent years, progress has been achieved in the field of computational fluid dynamics (CFD). Although this technique may be suitable for use in the future to estimate flow-induced forcing functions, its current capability is most likely not adequate to compute pressure fluctuation spectra acting on large complex structures subject to high Reynolds number turbulent flow. Currently, CFD-based load computations need to be validated by in-plant measurements and/or SMT results. 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 errors 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.

2.1.2.1 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.[3] Lock-in can occur over a wide range of operating conditions (i.e., over a large range of flow rate). The most effective method to avoid lock-in is to separate resonance frequencies from the flow excitation frequency by about 15 to 20 percent. If this is not possible, the maximum possible vibration level within the plant lock-in operating range may not exceed allowable limits.

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

[3] The lock-in phenomenon can occur during steam or liquid flow for nuclear power plant components in addition to reactor internals.

measurements may be used to estimate the resonance frequencies, subject to the guidance in Section C.2.1.1.

The applicant or licensee should evaluate reactor internals that could be excited by fluid flow which converges and accelerates between the gaps and annuli among reactor internals. For example, for any internal 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 or licensee should establish specific mitigation procedures where the lock-in leads to vibration or stress that exceeds allowable limits.

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

2.1.2.2 Scale Model Testing

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

a. SMT generally involves 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 AR) are generally dependent on Reynolds number, the applicant or licensee should demonstrate that the SMT results are not sensitive to further increases in the Reynolds number of the SMT.

b. When examining flow-excited AR 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 coefficients 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 amount of water carry over in the steam). Scale models built to study AR 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 (such as blow-down testing), 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 AR frequencies can affect the acoustic response during transient tests (e.g., producing artificially wide AR 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 AR 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. Because transient SMTs may have more uncertainty than steady-state tests, BUFs derived from transient tests need to be based on a wider velocity range in the SMT than the corresponding in-plant range. In deriving a BUF from transient SMTs for EPU applications, for example, 103 percent of the EPU power level has been used to define the upper bound.

k. Plant-specific data should be used to demonstrate the acceptability of SMT test data and confirm the vibration analysis results. When plant-specific data are not available, the acceptability of the SMT results needs to be demonstrated by a different method. For example, a slightly modified scale model may be tested to demonstrate the SMT

acceptability against available data of another plant, or the SMT results may be confirmed during the startup tests of the specific plant.

2.1.2.3 Computational Fluid Dynamics Modeling

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, whether for single or two-phase flows, 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 small flow passages, and additional features inside an SMR reactor vessel (such as internal CRDM mechanisms and flow passages between the tubes of the steam generator).

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. The effects of reverse flow through non-operating pumps or bypass flow passages need to be shown to be either negligibly small or accounted for in formulating the forcing functions. Consideration of such possibilities ensures that flow excitation mechanisms caused by high flow velocities 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), Detached Eddy Simulation (DES), direct numerical simulation (DNS) or other hybrid simulation methods at high Reynolds number flow is acceptable. If used, the simulation results 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 should be used if they are needed to complete the analysis. This includes, for example, estimating the correlation lengths and phase distributions of the fluid dynamic forces on structures exposed to fluid flow.

j. Past review of EPU applications indicate 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.

2.1.2.4 Force Inference Approaches

In some cases, inverse predictive methods may be used to infer fluctuating loads acting on reactor internals. These methods typically combine in-plant measurements with simulated transfer functions between the loaded surfaces and suspected source locations. Such methods need to be benchmarked. Benchmarking of force inference procedures on plants and systems similar to the plant being designed or licensed is acceptable. All uncertainties and bias errors associated with the 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 pressures and/or strains and inferring fluctuating loads. 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 reactors, especially using remote measurements and inference techniques.

Once specified and benchmarked, the same assumed material properties, speeds of sound, attenuation coefficients, and/or reflection coefficients may be used in similar plants. However, different flow conditions (such as temperature, pressure, and quality factor) might dictate reasonable adjustments of these parameters.

2.1.2.5 Mechanical and Acoustic Forces from RRPs

Where applicable, the vibration analysis needs to examine the effects of RRPs on reactor internals. 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 inflow and discharge 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. Estimates of these pump sources are combined with predictive 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 tubes, 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 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. In multi-loop plants, all pumps might not be operating, such that reverse flow occurs through the non-operating pumps. 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, one 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[4]

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).

[4] A variety of factors (such as weld quality and residual stresses) can affect the acceptability of the structural integrity of nuclear power plant components. NRC Standard Review Plan Section 4.5.2, "Reactor Internal and Core Support Structure Materials," provides additional guidance for structural integrity considerations.

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. In general, frequency-dependent bias errors and uncertainties need to be determined. For cases where a single frequency dominates the response, RMS or peak value bias errors or uncertainties are acceptable. Alternatively, direct end-to-end benchmarking may be used instead of combining individual bias errors and uncertainties, as described in the next section. It is also acceptable, where substantiated, to use bounding assumptions to determine individual or end-to-end bias errors and uncertainties.

When computing vibratory stresses in SSCs, it is appropriate to provide the following for dominant stress components (usually associated with maximum tensile stress directions):

- RMS stresses
- Frequency spectra
- Cumulative alternating stress plots, where the integration of the stress spectrum with increasing frequency is shown. In these plots, alternating stress is 0 at 0 frequency, and increases with frequency until the integrated total stress (the RMS level) is reached.
- Frequencies of the highest stresses (generally any frequency band which contributes 10 percent or more of the RMS level)

In cases where the frequencies of the applied forces are well below the first resonance frequency of a loaded structure, a single value based on equivalent static stress may be reported in lieu of the frequency-dependent data listed above.

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. Therefore, 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 conservative 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. 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 reactor internal 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 in the analysis 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 of such statistical assessments need to be included in the analysis results.

When using a structural dynamic analysis approach with the Rayleigh damping method, maximum 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 response. Deviation from the previously accepted 1 percent damping ratio above and below the anchor frequencies needs to be justified in determining final maximum vibration and stress levels.

2.1.3.1 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 vibrations and/or strains on the structure. End-to-end benchmarking encompasses bias errors and uncertainties associated with loading estimates (including unknown or difficult-to-quantify loading mechanisms, such as excitation by boiling water surrounding a steam dryer skirt or excitation by water droplets in wet steam environment), 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. Supplemental measurements to validate the simulation of intermediate quantities, such as loads acting on the structure, are also helpful, but not always necessary when end-to-end benchmarking is used. 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 beyond reasonable ranges 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 the differences between measured and simulated acoustic and/or structural responses averaged over sufficient locations to reasonably characterize all critical regions of a reactor internal. The average of the differences is the bias. It is acceptable to specify uncertainties based on two standard deviations of the differences. When a single standard deviation is used, the methodology outputs, combined with bias and uncertainty, need to bound those at all measurement locations at critical peak frequency regions. If the methodology results are not bounding, the discrepancies need to be used to estimate the effect of the underprediction on fatigue life such that reactor internal 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 structure 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.

It is important to note that the benchmarking of vibrations or surface strains does not ensure that maximum stresses are properly calculated in a structural model. Maximum stresses usually occur near corners and welds and other stress concentration locations. Separate convergence studies may be necessary for these locations to ensure maximum stresses are properly determined.

2.1.3.2 Stress Convergence and High-Cycle Fatigue Evaluation

Vibratory loading on structures raises a concern about the potential for cyclic fatigue failure at locations of high stress. The most susceptible locations are at structural discontinuities, and especially at fillet-welded joints between structural members.

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.

For all reactor internals, the final adjusted alternating stress intensity is limited to the ASME BPV Code allowable alternating stress intensity. For high cycle fatigue, this is the material endurance limit (i.e., no crack initiation for infinite cycles), as defined in the ASME BPV Code.

To develop conservative predictions of total stress (primary + secondary + peak stress) for use in a fatigue evaluation in accordance with ASME BPV Code, Section III, Subsection NG, there are three important elements. (a) The structural model needs to accurately represent 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 evaluation. (b) The dynamic loads on the structure need to be known and properly applied to the structural model. (c) The relationship between the structural model stress output and a conservative prediction of the total stress needs to be known.

a. Structural Model Development

To conduct the structural analysis, a sufficiently validated and 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 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 membrane and linear bending 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 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 membrane and linear bending stress output will be extracted for fatigue evaluation, the analysis includes confirmation of stress convergence by systematically reducing the local element size, before applying appropriate stress concentration factors to the structural model stress output.

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. One-percent damping is acceptable for structures surrounded by steam. For reactor internals immersed in water, higher values of damping may be used if justified with measurements. When using Rayleigh damping in a direct integration time history analysis, analysis has confirmed that the solution is not over-damped in the frequency range of interest. One acceptable approach is to specify the target damping value at the lowest frequency of interest and at the highest frequency of interest. This will be conservative at intermediate frequencies. The use of alternate Rayleigh damping anchor frequencies needs to be quantitatively justified, and will be reviewed by the staff on a case-by-case basis.

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 structures surrounded by steam such as BWR steam dryers. For components immersed in water, higher values of modal damping may be appropriate if justified by measurement.

b. Applying Dynamic Loads to the Structural Model

The excitation mechanisms that might cause dynamic loading are described in the previous section of this RG. 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. The input time increment may be no larger than 0.25 times the shortest period of interest. Using 0.125 times the shortest period of interest has been generally acceptable.

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" membrane and 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 maximum local 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. The appropriate use of stress results from a finite element analysis in the fatigue evaluation depends on the local geometric complexity and the type of model refinement at the double fillet weld connection. For example, for the fatigue evaluation of double fillet welds, the staff found two methods to be acceptable, with certain qualifications on their application. The methods may be used, in conjunction with appropriate multipliers, to achieve acceptable predictions of total stress for use in a fatigue evaluation. Both methods provide a level of conservatism consistent with the guidance in ASME BPV Code Section III, Subsection NG.

The first method is analogous to the traditional method mentioned above. The finite element analysis results from a global model of the structure, subjected to the applied dynamic loading, are used to calculate a nominal membrane and linear bending stress at the location of the double fillet weld. The geometry of 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 total stress estimate for use in the fatigue evaluation. A conservative approach is to assume that the calculated total stress occurs in both the positive and negative directions, producing a total stress range equal to twice the calculated total stress. The alternating total stress, which equals one half of the total stress range, is then equal to the calculated total stress.

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 total 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 to analyze the intermediate local submodel with the appropriate boundary conditions extracted from the global model analysis. 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 mesh refinement and addition of the fillet weld solid elements, becomes the final local submodel for this method.

2.1.3.3 <u>Operational Vibration and Stress Acceptance Criteria</u>

Computed vibrations and stresses need to be compared to allowable levels, such as the fatigue acceptance criteria specified by ASME, or other criteria substantiated by testing and analysis. Any 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.2.2.3).

2.1.4 Preoperational and Startup 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, variability in reactor operating parameters during a fuel cycle can affect flow rates, working fluid mass density, pressures, and other quantities. The applicant or 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 prototype reactor internals evaluated by the vibration and stress analysis program, 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 reactor internals with no prior history of adverse effects caused by 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 is necessary. Measurements need to be performed, however, for 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 Appendix A). The need for instrumentation is addressed for new components that have no operating experience. The measurement program is summarized below.

2.2.1 In-plant Measurement Issues

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

a. All in-plant instrumentation needs to be qualified for the environment (temperature, pressure, wetness) in which it will operate. Once the sensors are qualified, small changes in the test conditions should not affect their dynamic signals.

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 proper 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 erroneous unintended measurements of vibration rather than surface pressure. It is therefore appropriate to use vibration compensated pressure sensors.

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 vessel and containment 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 or 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.

2.2.2 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 underprediction) and random uncertainties (such as instrumentation error) associated with the instrumentation and data acquisition systems.

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

(1) power levels and flow rates 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, if applicable, 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 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. These criteria are based on analysis results and allowable vibration and material fatigue limits (see Section C.2.1.3). Depending on the component specific acceptance criteria, some of the following provisions may be applicable to reactor internals.

(1) Power ascension limits may consist of 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 Level 1 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 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). 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 some situations, the applicant or licensee may need to 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 plants. 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 maximum 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. However, 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. The sensor locations should be selected such that the measurements are adequately correlated with the maximum vibration and stress locations on the structure. 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.

End-to-end bias and uncertainties for the overall dynamic and stress 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 end-to-end bias errors and uncertainty values. If the measured sensor data exceed the adjusted predictions, then the bias errors, uncertainty values, and acceptance limits need to be adjusted to ensure measured sensor responses do not exceed the adjusted predictions.

d. <u>Guidance on test duration:</u> 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. <u>Guidance on test configuration and test conditions:</u> The test conditions should be representative of the plant normal operating and transient conditions. The applicant or licensee should address the disposition of fuel assemblies. Preoperational testing should be performed with the reactor internals 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 reasonable results.

2.3 Inspection Program

The applicant or licensee should describe the inspection program for inspections of the prototype 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 for the prototype reactor internals 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 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 RG, the applicant will need to develop a test plan to obtain additional data as a prototype as discussed in this RG. 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.

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 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 effects 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.

See Section C.1.4 of this RG 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 RG.

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 RG.

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 RG 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 (lock-in) 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 RG for prototype reactor internals. 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. Either a subset of the measurements made for the prototype may be used, or a remote monitoring method may be applied with alternative instrumentation. However, such remote monitoring will be demonstrated and benchmarked during the prototype vibration measurement program. See Appendix A for an example of a remote monitoring process for BWR steam dryers.

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. Therefore, it is important for the applicant or licensee to be confident in the decision to classify reactor internals as a limited prototype to avoid the need to supplement its vibration monitoring program if the classification is subsequently determined not to be 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. 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

Non-prototype reactor internals need to be evaluated for all significant flow, temperature, and pressure conditions associated with normal steady-state and anticipated transient operation evaluated for the applicable valid 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.

See Section C.1.4 of this RG 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 RG.

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 RG.

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 RG for prototype reactor internals.

The applicant or licensee should be aware that minor design or operating changes from the valid prototype may 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

For reactor internals other than a BWR steam dryer, a vibration measurement program may be proposed in lieu of an inspection program for non-prototype reactor internals. In such instances, 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 RG for prototype reactor internals. Because of similarities to a valid prototype, the assessment of a non-prototype might not involve a vibration measurement program as comprehensive as the measurement program applicable to a prototype. Either a subset of the measurements made for the prototype may be used, or a remote monitoring method may be applied with alternative instrumentation. However, such remote

monitoring will be demonstrated and benchmarked during the prototype vibration measurement program.

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. Therefore, it is important for the applicant or licensee to be confident in the decision to classify reactor internals as a non-prototype to avoid the need to supplement its vibration monitoring program if the classification is subsequently determined not to be valid.

D. IMPLEMENTATION

The purpose of this section is to provide information on how applicants and licensees[5] may use this guide and information regarding the NRC's plans for using this RG. 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[6] use the guidance in this document to demonstrate compliance with the underlying NRC regulations. Methods or solutions that differ from those described in this RG 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 RG 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 RG 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 RG. The NRC staff does not expect any existing licensee to use or commit to using the guidance in this RG, 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 RG 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 RG 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 RG, requests for information under 10 CFR 50.54(f) as to whether a licensee intends to commit to use of this RG, generic communication, or promulgation of a rule requiring the use of this RG.

During regulatory discussions on plant specific operational issues, the staff may discuss with licensees various actions consistent with staff positions in this RG 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 RG are part of the licensing basis of the facility. However, unless this RG 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 RG constitutes a violation.

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 RG and (2) the specific subject matter of this RG 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

[5] 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.

[6] 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.

guidance in this RG 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 RG or requesting or requiring the licensee to implement the methods or processes in this RG 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. 7) and NUREG-1409, "Backfitting Guidelines," (Ref. 8).

List of Acronyms

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
FSAR	final safety analysis report
ITAAC	inspections, tests, analyses, and acceptance criteria
LES	large eddy simulation
MIV	mechanical-induced vibration
MSL	main steam line
OMB	Office of Management and Budget
PSD	power spectral density
PWR	pressurized water reactor
RCS	reactor coolant system
RMS	root mean square
RPV	reactor pressure vessel
RRP	reactor recirculation pump
RSD	replacement steam dryer
SG	steam generator
SMR	small modular reactor
SMT	scale model testing
SRV	safety relief valve
SSC	structures, systems and components
VPF	vane passing frequency

REFERENCES[7]

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. CFR, "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. NRC, Regulatory Guide (RG) 1.68, "Initial Test Program for Water-Cooled Nuclear Power Plant."

5. American Society of Mechanical Engineers (ASME), "Rules for Construction of Nuclear Power Plant Components," *Boiler and Pressure Vessel Code*, Section III, Division I, New York, NY.[8]

6. 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.

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

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

9. S.A. Hambric, et al., "Flow-Induced Vibration Effects on Nuclear Power Plant Components Due to Main Steam Line Valve Singing," NUREG/CP-0152, Volume 6, Proceedings of the Ninth NRC/ASME Symposium on Valves, Pumps and Inservice Testing, 2006.

10. S. Ziada, "Flow-Excited Acoustic Resonance in Industry," Institute of Thermomechanics, Flow Induced Vibration, Zolotarev & Horacek, eds., Prague, 2008.

11. Allemang, R.J., "The Modal Assurance Criterion – Twenty Years of Use and Abuse," Sound and Vibration, August 2003.

7 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.

8 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/.

APPENDIX A: GUIDANCE AND EXAMPLES FOR A COMPREHENSIVE VIBRATION ASSESSMENT PROGRAM FOR A BOILING WATER REACTOR STEAM DRYER

Operating experience has revealed failures of steam dryers and main steam system components (including relief valves) in boiling water reactor (BWR) nuclear power plants following implementation of extended power uprates (EPUs). Studies of those failures have determined that flow-excited acoustic resonances (where instabilities in the fluid flow excited acoustic modes) within the valve standpipes and branch lines in main steam lines (MSLs) can produce strong mid- to high-frequency pressure fluctuations in the standpipes of the MSL valves, causing damage. These pressure fluctuations within the standpipes of the valves might also excite the acoustic modes of the steam in the MSLs, causing extremely high sound radiation and damaging the steam dryer, and possibly other reactor internals and steam system components.

This appendix to Regulatory Guide (RG) 1.20 provides specific guidance for the development and implementation of a comprehensive vibration assessment program (CVAP) for steam dryers in BWR nuclear power plants, with examples based on experiences with steam dryer EPU and new plant applications. The appendix follows the organizational structure, including section numbering and titles, of the main text of the RG.

1. CLASSIFICATION OF REACTOR INTERNALS

New steam dryer designs, such as those proposed for the Economic Simplified Boiling Water Reactor (ESBWR) nuclear power plant, or by existing licensees as replacement steam dryers (RSDs), have usually been classified as prototypes. To be reclassified as a valid prototype, a BWR steam dryer should have completed a successful CVAP, with acceptable results from an inspection of the steam dryer following one full operating cycle, and the establishment of an appropriate long-term inspection program.[9]

Existing or new steam dryers may be classified as limited prototypes if they are substantially similar to a valid prototype, per the guidance provided in RG Section C.1.

Based on current operating experience, the applicant or licensee for a BWR nuclear power plant should provide significant justification if it proposes to classify a BWR steam dryer as a non-prototype. For example, the applicant or licensee should demonstrate that the loading on a non-prototype steam dryer will be less significant or the same as the loading on the valid prototype steam dryer. Also, the vibration response should be the same as or lower than that of the valid prototype steam dryer.

As described in this RG, a prototype steam dryer would receive:

(1) a comprehensive vibration and stress analysis program,
(2) an extensive vibration measurement program, and
(3) a full inspection program.

If there is no evidence of loose parts, cracking, unacceptable motion, or significant wear following the inspection program, the prototype steam dryer may be proposed as a valid prototype. Acceptance of the valid prototype classification will be evaluated on a case-by-case basis.

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[9] For a BWR, the steam dryer initial startup testing is performed after fuel load and its inspection is typically performed after the first cycle of operation.

A limited prototype steam dryer would receive:

(1) an update to the vibration and stress analysis program for any differences from the prototype steam dryer,
(2) a reduced vibration measurement program based on instrumentation either directly on the steam dryer or on MSL lines to quantify steam dryer alternating loading and alternating stresses, and
(3) a full inspection program.

A non-prototype steam dryer would receive:

(1) an update to the vibration and stress analysis program for any differences from the prototype steam dryer,
(2) a reduced vibration measurement program to confirm that steam dryer loads do not exceed those on the valid prototype, and
(3) a full inspection program.

Note that the guidance for non-prototype steam dryers deviates from the general guidance in the regulatory guide: Information That May be Useful for Other Plant Components neither the steam dryer measurement program nor inspection program may be waived in favor of the other program. However, for a non-prototype steam dryer, the applicant or licensee may limit the measurement program to only verifying that dryer loads, and in particular acoustic resonance and RRP VPF loads, will not exceed those in the valid prototype. The instrumentation required for this demonstration will be sufficiently sensitive to monitor these loads, but need not be as extensive as that used on limited prototypes.

For a multi-unit BWR nuclear power plant, where multiple units will be constructed and begin operation almost simultaneously, the steam dryer for one unit may be designated as the prototype and the steam dryers in the other units may be designated as limited prototypes. The prototype steam dryer may not be classified as a valid prototype because the inspection part of the CVAP for the prototype steam dryer will not have been completed prior to initial operation of the remaining units. The applicant or licensee should recognize that the reduced vibration measurement program for a limited prototype steam dryer might need to be upgraded based on the results of the startup testing of the prototype steam dryer.

2. CVAP FOR A PROTOTYPE BWR STEAM DRYER

In developing the CVAP for a prototype BWR steam dryer, it is important to recognize that 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 can produce strong 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 steam flow turbulence, boiling water adjacent to the skirt, and other internal dynamic forces) can result in undesirable dynamic stresses. Also, studies of past failures in nuclear power plants suggest that the steam dryer in a BWR nuclear power plant experienced fatigue failure caused in part by structure-borne loads emanating from the reactor recirculation pumps (RRPs) at the vane passing frequency (VPF).

All prototype steam dryers should complete a full CVAP as discussed under the main body of this RG. Along with the CVAP, the NRC staff has imposed license conditions to address BWR steam dryer

monitoring during and after plant startup. These conditions are consistent with those described in the main body of RG 1.20.

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for reactor internals in a BWR nuclear power plant licensed under Part 52 in Title 10 of the *Code of Federal Regulations* (10 CFR Part 52) may address the following design commitments:

- Reactor internal structures meet American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code), Section III, "Nuclear Power Plant Components," Subsection NG-3000 (Ref. 5), as incorporated by reference in 10 CFR 50.55a.

- The set of BWR steam dryer instrumentation includes an adequate number of pressure sensors, strain gages, and accelerometers to confirm acceptance limits are met, and to apply to the steam dryer analysis program using startup test data.

- The design and construction of MSL branch lines and SRV standpipe geometry in BWR nuclear power plants will preclude first shear layer wave acoustic resonance.[10]

2.1 Vibration and Stress Analysis Program

Vibration and stress analysis programs for new and replacement BWR steam dryers classified as prototypes have been rigorous, combining detailed forcing function modeling with detailed structural finite element (FE) models. Along with confirming the structural integrity of the prototype steam dryer design, the strain and stress predictions should be used to establish the locations and allowable limits for instrumentation used for power ascension monitoring.

2.1.1 Structural, Hydraulic, and Acoustic Modeling

2.1.1.1 <u>Modes of Vibration</u>

Simulated modes of vibration using structural FE models have been compared to those measured on steam dryers prior to installation and after installation in the reactor pressure vessel (RPV). In most cases, comparisons of the simulations and measurements show that the resonance frequencies are within +/-10 percent.

2.1.1.2 <u>Vibration Damping</u>

All accepted steam dryer applications have applied vibration damping of 1 percent of critical damping. As stated in the RG, the 1 percent damping includes multiple damping mechanisms, including structural, frictional, radiation, and other forms of damping, including the effects of water droplets on the steam dryer surfaces. Some BWR licensees requesting EPU license amendments have 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.

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[10] Several studies have been conducted of the potential for flow-excited acoustic resonance effects on steam dryers and other MSL components in BWR nuclear power plants that may be applied in the design and construction of MSLs. See references 6, 9, and 10.

2.1.1.3 Frequency Response Functions

Frequency response functions have been computed and compared to those measured on steam dryers prior to installation in the RPV with dynamically free boundary conditions. However, it is more appropriate to evaluate final alternating strain and vibration calculations against measured data on an inservice steam dryer with actual boundary conditions. Note that there have been cases where uncertain boundary conditions have led to additional analyses to ensure bounding stresses are calculated. In particular, steam dryers resting on mounting pads attached to the RPV have variable amounts of contact area and supported load during normal operating conditions.

2.1.2 Forcing Functions

Of particular importance in evaluating 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. flow turbulence and narrow-band excitation in the SRVs and deadlegs in MSLs; and

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

Items a and b have been directly accounted for in all steam dryer applications. The effects in item c are covered by bias factors based on end-to-end benchmarking, as discussed below.

2.1.2.1 Flow Excitation with Feedback and Lock-In Mechanisms

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 SRV 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. 6 in RG 1.20). 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, some of which are associated with the well-known amplification effects of multiple standpipes in close proximity:

(1) Strouhal number (dimensionless product of frequency and pipe diameter divided by flow velocity) analysis to evaluate critical flow rates (including uncertainties in the Strouhal number),
(2) effects of the ratios between the standpipes and main pipe diameters,
(3) effects of edge radii at the inlets of the standpipes,
(4) effects of upstream elbows,

(5) distances between standpipes, and
(6) relative lengths of standpipes.

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

2.1.2.2 Scale Model Testing

The NRC staff has not accepted Scale Model Testing (SMT) data for directly computing steam dryer loading or response due to differences between the operating fluid, scaled dimensions, and operating conditions. However, the staff has accepted SMT data for computing Bump Up Factors (BUFs) between loads at EPU and current licensed thermal power (CLTP) conditions. All EPU applications have implemented a methodology to estimate the steam dryer loading at EPU conditions from that computed at CLTP conditions. Some applicants/licensees used SMT-based BUFs and others used BUFs based on plant measurements. More information related to the SMT can be found in the main body of RG 1.20 (Section C.2.1.2.2). The staff has evaluated all SMT submissions on a case-by-case basis; no established methodology has been accepted for general use.

2.1.2.3 Computational Fluid Dynamics Modeling

Computational Fluid Dynamics (CFD) modeling has not been used to quantitatively determine steam dryer loading in BWR applications submitted to the NRC. However, CFD models have been used to demonstrate that the steam flow within an RPV is qualitatively similar to that in the RPV of a prototype steam dryer.

2.1.2.4 Force Inference Approaches

In some BWR EPU requests, licensees have employed inverse predictive models using in-plant measurements as inputs to estimate fluctuating pressures within the RPV and on BWR steam dryers. These pressure loads are inferred from localized measurements of fluctuating pressures and/or strains either (a) on steam dryer surfaces, or (b) within the MSLs connected to the RPV. The MSL-based methods are based on the well-known Two Microphone Method, which computes the fluctuating pressures and particle velocities within a pipe given pressure measurements at two locations. The method is used to compute particle velocities and pressures at the MSL inlets, and subsequently loads on the dryer surfaces using a three-dimensional acoustic model of the steam within the RPV.

Based on experience, the following guidance is provided for MSL-based loading inference approaches that minimize uncertainty and bias error for steam dryer applications (MSL pressure measurement details are provided in Section 2.2 of this appendix).

a. At least two measurement locations need to be used on each MSL. However, using three measurement locations on each MSL improves the input data to the 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.

b. Acoustic modeling parameters in the MSL and RPV steam models, such as the speed of sound, reflection coefficients from boundaries between steam and water, and sound attenuation (damping), are affected by the amount of water carry-over in the steam, especially within steam dryers. The values of these parameters 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.

e. When using structural models of a steam dryer as part of a force inference procedure, additional bias errors and uncertainties need to be addressed. For example, the modes of vibration in a structural predictive model are not expected to exactly match those in an actual structure. A range of frequencies therefore needs to be evaluated when inferring structural loading, with worst case loads used in subsequent analyses. Any bias errors in structural amplitudes also need to be addressed.

2.1.2.5 Mechanical and Acoustic Forces from RRPs

Forces induced by pulsations at the VPF of RRPs need to be accounted for in steam dryer alternating stress analyses. Forces may be estimated based on controlled tests by the pump supplier, and combined with dynamic models of the structural components between the RRP and steam dryer. However, these forces may also be accounted for using on-dryer strain and/or vibration measurements. The relative contribution of the stresses at VPF may be determined and applied to replacement or new steam dryer designs. However, such assessments should be demonstrated to be conservative, and be based on upper-bound estimates.

2.1.3 Computing and Benchmarking Structural and Acoustic Operational Response

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

The guidance for BWR steam dryers is consistent with that in the main body of RG 1.20. Previous applicants have chosen to compute an overall bias error and uncertainty based on comparing measured and simulated strains and vibrations over regions of the steam dryer, most commonly subdividing the steam dryer into upper and lower sections. For example, several strain gages and/or accelerometers might be installed at various locations on the upper dryer and lower dryer regions. In this approach, the simulation bias is calculated as the mean of the differences between the simulated and measured strains and/or vibrations averaged over the sensor locations, and the uncertainty is calculated as the variance over the sensor locations. Using this approach, however, the applicant or licensee should also demonstrate that the corrected simulated strain and/or vibration (with bias error and uncertainty added to the simulations) bounds measured levels at all measurement locations.

2.1.3.2 Stress Convergence and High-Cycle Fatigue Evaluation

The guidance for BWR steam dryers is consistent with that in the main body of RG 1.20. To account for uncertainty in predictive analysis results for BWR steam dryers, the allowable alternating stress intensity limit used for design should be selected to provide a margin as compared to the ASME BPV Code allowable alternating stress intensity limit. Based on data collected during power ascension testing, the analytical predictions are either confirmed or appropriately adjusted to account for any bias

errors and uncertainties. After confirmation or adjustment of the analytical predictions, the added design margin is no longer necessary.

2.1.3.3 Operational Vibration and Stress Acceptance Criteria

Acceptance limits for alternating stress intensity are found in ASME BPV Code, Section III, Subsection NG. For austenitic stainless steels at temperatures up to 425 °C (850 °F), typical of BWR steam dryers, the high cycle fatigue limit at 10^{11} cycles is 93.7 Megapascals (MPa) (13.6 kips per square inch (ksi)) per Figure I-9.2 and Table I-9.2. It is noted that earlier versions of the ASME fatigue limits included multiple curves (A, B, and C), to account for mean stress from various sources, such as gravity loading or residual stresses caused during manufacturing. The multiple fatigue curves have been eliminated; current practice is to use the single value of 93.7 MPa (13.6 ksi), independent of mean stress.

2.1.4 Preoperational and Startup Testing Analysis

The guidance for BWR steam dryers is consistent with that in the main body of RG 1.20.

2.2 Vibration and Stress Measurement Program

The plant startup testing program to evaluate potential adverse flow effects on a prototype BWR steam dryer should obtain plant data from instrumentation mounted directly on the steam dryer 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 should be based on the steam dryer structural modeling and vibration analysis. The sensors should provide sufficient information to confirm the acceptability of the stress analysis of the entire steam dryer, and should include pressure sensors, strain gauges, and accelerometers.

Surface pressure instrumentation on BWR steam dryers may be used to measure and confirm pressure loading estimates from SMT or numerical analyses. Alternatively, surface strain and vibration measurements may be used to confirm end-to-end calculations that are based on the loading estimates. In this case, surface pressure measurements may also be used to supplement the justification of the loading estimates. 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.

Limits (peak or root mean square (RMS) levels, and/or limit curves over frequency) for the steam dryer sensors should 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, should 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 the steam dryer. The direct steam dryer measured data should be used to calibrate the MSL instrumentation and data analysis for steam dryer forcing function estimation before removal or failure of an excessive number of the steam dryer sensors that precludes a reliable analysis.

Strain gage arrays may be used to relate the hoop strain in an MSL to the internal ‘breathing’ pressure (the pressure pulsations associated with acoustic plane waves with no circumferential or radial variability). 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 the MSL piping. 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. At least four strain gages, evenly spaced around a pipe circumference, are necessary to filter signals not associated with hoop strain (for example, pipe bending, and at higher frequencies, ovaling) from a measurement. Based on experience, additional gages are important 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 should be compared with previous data to ensure reasonable consistency.

d. Instrumentation wiring should be properly labeled. MSL strain gage spectra should 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 should be made with consistent sensor sets. For example, any sensors in an MSL circumferential array that fail should be removed from the array summation in both new and previous datasets, as well as limits and limit curves.

2.2.1 In-plant Measurement Issues

The guidance for BWR steam dryers is consistent with that in the main body of RG 1.20.

2.2.2 Vibration Measurement Program Documentation

The guidance for BWR steam dryers is consistent with that in the main body of RG 1.20.

2.3 Inspection Program

The inspection program for a prototype BWR steam dryer should follow the guidance in the main body of RG 1.20. The licensee should evaluate the results of the steam dryer inspections to determine the need for repair prior to plant restart, or justify leave-as-is with re-inspection during the next refueling outage. The licensee should develop a long-term inspection program for the steam dryer based on the inspection results with adjustments to the program as appropriate based on findings during future inspections.

2.4 Documentation of Results

The guidance for BWR steam dryers is consistent with that in the main body of RG 1.20.

2.5 Schedule

A full BWR steam dryer stress analysis report and evaluation should be submitted to the NRC within 90 days of first reaching 100 percent licensed thermal power.

3. CVAP FOR A LIMITED PROTOTYPE BWR STEAM DRYER

Limited prototype BWR steam dryers should receive (1) an update to the vibration and stress analysis program accounting for any differences with the prototype steam dryer, (2) a reduced measurement program based on instrumentation either directly on the steam dryer or on the MSL lines to quantify steam dryer alternating loading and alternating stresses, and (3) a full inspection program.

An applicant for a design certification or combined license for a BWR nuclear power plant under 10 CFR Part 52 should establish ITAAC as indicated in Section 2 of this appendix with adjustments consistent with the limited prototype steam dryer measurement program.

3.1 Vibration and Stress Analysis Program

Any differences between the following elements of the prototype and limited prototype BWR steam dryer vibration and stress analysis program should be quantitatively determined, including the following:

(a) plant measured inputs to the load definition methods, such as on-dryer and/or MSL strain gage measurements;
(b) load definition software and methodology, and/or models used to map inputs to steam dryer loading;
(c) FE modeling software, modeling procedures (element discretization, element type), material properties, or boundary conditions; and
(d) any BUFs determined from SMT.

Any other differences should also be accounted for, typically in the form of additional bias corrections and uncertainties. The final limited prototype steam dryer analysis results, including loading, vibration response, and final alternating stresses, should be compared to those for the prototype steam dryer, and acceptable margin to fatigue limits should be demonstrated.

3.2 Vibration Measurement Program

A vibration measurement program should be implemented for limited prototype BWR steam dryers. However, the measurements need not be as rigorous as those applied to the valid prototype steam dryer. Instead, either a subset of the on-dryer vibration and strain instrumentation may be applied and monitored during power ascension, or a remote monitoring method may be used. An example of remote monitoring is using circumferentially oriented strain gages on BWR MSLs to infer acoustic wave amplitudes within the MSLs, which are then used to develop approximate fluctuating pressure loads on the steam dryer. The loads may be applied to the steam dryer FE model to compute the alternating stress

for a steam dryer that is similar in design and operation to a valid prototype steam dryer, which has previously been benchmarked using comprehensive on-dryer vibration and strain measurements.

Remote monitoring via MSL strain gage arrays may also be used to compare to allowable RMS limits or frequency-dependent limit curves. However, the expected variability and bias of the MSL strain gage measurements should be quantified and compared to those made on the valid prototype steam dryer. Use of a limited number of sensors on the steam dryer may provide an acceptable means to assess this variability. If the variability cannot be established, then additional factors of safety may be applied to the allowable limits.

For limited prototype steam dryers, the use of remote instrumentation for monitoring is acceptable when (a) vibration and stress analysis shows a high margin of safety in the component alternating stresses (for example, in the past the NRC has found a margin of safety of 2.0 to be acceptable for remote monitoring of steam dryers), and (b) prototype remote measurements do not reveal anomalies in data quality that would add significant uncertainty in monitoring the component.

3.3 Inspection Program

The inspection program for a limited prototype BWR steam dryer should follow the guidance in Section 2.3 of this appendix.

4. CVAP FOR A NON-PROTOTYPE BWR STEAM DRYER

Based on current operating experience, the applicant or licensee for a BWR nuclear power plant should provide significant justification if it proposes to classify a BWR steam dryer as a “non-prototype” reactor internal component. The applicant or licensee should develop an appropriate vibration and stress analysis program, vibration measurement program, and inspection program based on its justification for classifying the steam dryer as a non-prototype. Note that for BWR non-prototype steam dryers, both vibration measurement program and inspection program are necessary.

An applicant for a design certification or combined license for a BWR nuclear power plant under 10 CFR Part 52 should establish ITAAC as indicated in Section 2 of this appendix with adjustments consistent with the non-prototype steam dryer measurement program.

4.1 Vibration and Stress Analysis Program

The guidance for non-prototype BWR steam dryers is the same as that for limited prototypes provided in Section 3.1 of this appendix.

4.2 Vibration Measurement Program

The applicant or licensee may justify implementation of a reduced vibration measurement program for a non-prototype BWR steam dryer based on the establishment of an inspection program for a non-prototype steam dryer. For example, the measurement program may be limited to only verifying that steam dryer loads, and in particular acoustic resonance and RRP VPF loads, will not exceed those in the valid prototype. The instrumentation required for this demonstration will be sufficiently sensitive to monitor these loads, but need not be as extensive as that used on limited prototypes.

4.3 Inspection Program

The inspection program for a non-prototype BWR steam dryer should follow the guidance in Section 2.3 of this appendix.