U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

3.9.2 DYNAMIC TESTING AND ANALYSIS OF SYSTEMS, STRUCTURES, AND COMPONENTS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for mechanical engineering reviews

Secondary - None

I. AREAS OF REVIEW

This Standard Review Plan (SRP) section addresses the criteria, testing procedures, and dynamic analyses employed to ensure the structural and functional integrity of piping systems, mechanical equipment, reactor internals, and their supports (including supports for conduit and cable trays, and ventilation ducts) under vibratory loadings, including those due to flow-induced excitations (and loading caused by flow instabilities over standoff pipes and branch lines in the steam system) and postulated seismic events. Compliance with the specific criteria guidance in ~~subsection~~Subsection II of this SRP section will provide reasonable assurance of appropriate dynamic testing and analysis of structures, systems, and components (SSCs) within the scope of

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USNRC STANDARD REVIEW PLAN

this SRP section in conformance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a~~;~~, “General Provisions,” 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix A, “General Design Criteria (GDC) for Nuclear Plants,” GDC 1, “Quality Standards and Records,” GDC 2, “Design Bases for Protection against Natural Phenomena,” GDC 4, “Environmental and Dynamic Effects Design Bases,” GDC 14, ~~and~~ “Reactor Coolant Pressure Boundary,” and GDC 15~~;~~, “Reactor Coolant System Design,” 10 CFR Part 50, Appendix B~~; and~~ , “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants”; and 10 CFR 52.47(b) and 10 CFR 52.80(a).

The specific areas of review are as follows:

1. Piping vibration, safety relief valve (SRV) vibration, thermal expansion, and dynamic effects testing should be conducted during startup testing. The systems to be monitored should include:

   A. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPV Code) Class 1, 2, and 3 systems

   B. other high-energy piping systems inside seismic Category I structures (“seismic Category I” is defined in Regulatory Guide (RG) 1.29~~)~~), “Seismic Design Classification.”

   C. high-energy portions of systems whose failure could reduce the functioning of any seismic Category I SSC to an unacceptable safety level

   D. seismic Category I portions of moderate-energy piping systems located outside containment

The supports and restraints necessary for operation during the life of the nuclear power plant are considered to be part of the piping system.

The purpose of these tests is to confirm that these piping systems, restraints, components, and supports have been adequately designed to withstand flow-induced dynamic loadings under the steady-state and operational transient conditions anticipated during service and to confirm that normal thermal motion is not restrained. Particular attention should be given to any potential adverse flow conditions that could lead to abnormally strong pressure pulsations and vibrations, such as those caused by flow over valve openings. The test program description should specify the different flow modes, selected locations for visual inspections and other measurements, the acceptance criteria, and corrective actions if excessive vibration or indications of thermal motion restraint occur.

2. The following areas related to the seismic analysis of seismic Category I mechanical equipment described in the applicant's safety analysis report (SAR) are reviewed. For the methods and criteria for seismic qualification testing of seismic Category I mechanical equipment, refer to SRP Section 3.10. For the design of nuclear air and gas treatment systems and components, acceptable methods and criteria are described in ASME Code AG-1-2012.

A. Seismic Analysis Method. For all seismic Category I systems, components, equipment and their supports (including supports for conduit and cable trays, and ventilation ducts), and for certain non-seismic Category I items that are to be designed to seismic criteria, the applicable seismic analysis methods (response spectra, time history, equivalent static load) are reviewed. The manner in which the dynamic system analysis method is performed is reviewed. The method chosen for selection of significant modes and an adequate number of masses or degrees of freedom is reviewed. The manner in which maximum relative displacements between supports is evaluated in the seismic dynamic analysis is reviewed. In addition, other significant effects that are addressed in the dynamic seismic analysis such as hydrodynamic effects and nonlinear response are reviewed.

B. Determination of Number of Earthquake Cycles. The number of earthquake cycles during one seismic event, the maximum number of cycles for which systems and components are designed, and the criteria and procedures used by the applicant to establish these parameters are reviewed by the staff for consistency with the methods described in SRP Section 3.7.3.

C. Basis for Selection of Frequencies. As applicable, criteria or procedures used to separate fundamental frequencies of components and equipment from the forcing frequencies of the support structure are reviewed.

D. Three Components of Earthquake Motion. The procedures by which the three components of earthquake motion are considered in determining the seismic response of systems, and components are reviewed.

E. Combination of Modal Responses. When a response spectra method is used, the description of the procedure for combining modal responses (shears, moments, stresses, deflections, and accelerations) is reviewed, including that for modes with closely spaced frequencies. For example, when a response spectrum approach is used for calculating the seismic response of systems or components, the phase relationship between various modes is lost. Only the maximum responses for each mode can be determined. The maximum responses for modes do not generally occur at the same time, and these responses need to be combined according to an appropriate procedure selected to approximate or bound the response of the system.

F. Multiple-supported Equipment and Components with Distinct Inputs. The criteria and procedures for seismic analysis of equipment and components supported at different elevations within a building and between buildings with distinct inputs are reviewed.

G. Use of Constant Vertical Static Factors. Where applicable, the justification provided for using constant static factors rather than a vertical seismic system dynamic analysis to compute vertical response loads for design of affected systems, components, equipment and their supports is reviewed.

H. Criteria Used for Damping. The criteria to account for damping in systems, components, equipment and their supports are reviewed.

3. Analyses of dynamic responses of structural components within the reactor vessel caused by steady-state and operational flow transient conditions, including all fluctuating loading induced by flow, acoustic, or mechanical sources, are reviewed to confirm that they are consistent with the approach described in RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing," Revision 4.[1] For example, reviews are conducted of analyses required for prototype, limited prototype, and nonprototype plants and components (such as steam dryers in boiling water reactor [BWR] nuclear power plants), and for operating plants requesting a power uprate. The breadth of analyses needed for prototype, limited prototype, and nonprototype plants and components depends on the type and complexity of the reactor internals, and their classification. For guidance on classification of SSCs as prototypes, limited prototypes, and nonprototypes, see Section C.1 of RG 1.20, Revision 4.

The following structures in BWRs should be included in the dynamic analysis and reviewed by the staff. Those marked with an asterisk, such as the steam dryer, are non-safety-related components; they do not perform a safety function but must retain their structural integrity to avoid the generation of loose parts that might adversely impact the capability of other plant equipment to perform their safety functions.

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

For pressurized water reactors (PWRs), reactor internals might include the following components:

- core barrel
- upper core support assembly

[1] This revision to the regulatory guide is currently under development and is expected to be issued for public comment in parallel with this Standard Review Plan section.

- lower core support assembly
- control rod guide assembly
- in-core instrumentation guide tubes
- flow distribution device
- heavy reflector
- irradiation specimen baskets

For small modular reactors (SMRs), the Comprehensive Vibration Assessment Program (CVAP) might address 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:

- core support structures
- reactor coolant/recirculation pumps
- riser
- steam generator
- pressurizer
- control rod drive mechanisms and supports
- in-core instrumentation guide tubes
- feedwater lines

The purpose of the analyses is to assess the vibration behavior of the components, including the definition of the input-forcing functions, and estimation of the consequent vibration and alternating stress levels. The analyses should verify the structural integrity of reactor internals for flow-induced vibration (FIV), acoustic resonance (AR), acoustic-induced vibration (AIV), and mechanical-induced vibration (MIV). Flow excitation mechanisms such as vortex-induced vibration, flow-excited acoustic resonance, fluid-elastic instability, and turbulence buffeting as well as other flow excitations of flow separation, reattachment, and impinging flow instabilities should be considered. These mechanisms are often nonlinear and their adverse effects cannot be predicted by linear extrapolation of existing plant data. In some cases, the instabilities in these flow fields can couple with acoustic or structural resonances, causing high dynamic loads throughout the steam system and RPV. These self-excited loads can be orders of magnitude higher than those which do not couple to acoustic or structural resonances. A complete assessment of the likelihood of any potential self-excitation mechanisms that lead to adverse flow effects at all expected reactor operating conditions should be evaluated. AIV and MIV due to pressure pulsation and vibration of the reactor recirculation pump (RRP) should also be reviewed for all SSCs, including reactor internals.

The following areas related to the dynamic response analysis are reviewed along with their bias errors and uncertainties. For more details on U.S. Nuclear Regulatory Commission (NRC) guidance to applicants, refer to RG 1.20, Revision 4, Section C.2.

A. Structural, Hydraulic, and Acoustic Modeling. All analytic and numerical modeling procedures, models, and calculations are reviewed, including models of fluid hydraulic and acoustic behavior and their coupling to structural models. In particular, modes of vibration, structural damping, frequency response functions (FRFs), and the effects of variable or uncertain boundary conditions should be assessed.

(i) The resonance frequencies and mode shapes from simulations and measurements should be compared to establish the accuracy bounds of the models.

(ii) Damping assumptions should be reviewed, and any claims of greater than 1~~%~~ percent damping should be assessed for adequacy.

(iii) Simulated and measured (from modal or shaker testing) FRFs should be examined, along with resulting bias errors and uncertainties caused by differences between modeled and as-built conditions.

For further details, refer to RG 1.20, Revision 4, Section C.2.1.1.

B. Transient and Steady-State Flow-Induced Forcing Functions. Forcing functions within the reactor vessel and the feedwater and steam systems are reviewed, including unsteady excitations induced by flow around the sampling probes in the feedwater piping and over the standoff pipes of safety valves in the main steam lines (MSLs). The methods for specifying the forcing functions (e.g., analytic or numerical techniques such as computational fluid dynamics (CFD) models, test-analysis combination methods such as scale model testing (SMT), and response deduction methods) should be assessed. Any forcing functions caused by such adverse flow effects as flow instabilities over standoff pipe openings are reviewed for adequacy.

(i) If SMT is used to support the dynamic analysis, the review process should include the following:

(1) Any deviations between the model and the plant parameters should be assessed in detail. Such parameters include Reynolds number, acoustic attenuation, fluid-elastic parameter and small details of the model geometry.

(2) The effects of transient test conditions on the speed of sound and the accuracy of pressure and flow velocity measurements.

(3) The procedure used to determine the amplitudes of acoustic and mechanical resonances and their expected amplification with power level (referred to as a bump-up factor [BUF]).

(ii) If CFD models are used to develop unsteady forcing functions, or compute the distribution of flow velocity which is used to develop the forcing functions, the following items should be reviewed as applicable:

(1) The procedure for validating the CFD code used to estimate the forcing functions.

(2) The model grid size and the computation time step.

(3) The appropriateness of the simulated flow conditions.

(4) All parameters and simplifying assumptions that might affect the CFD results and the computed forcing functions, including fluid properties), correlation length and phase of fluid forces, fluid-acoustic or fluid-structure interaction, and accuracy of the fluid domain model.

(iii) In recent BWR extended power uprate (EPU) requests, licensees have employed inverse acoustic models to estimate the forcing function on the steam dryer. This approach is aided by measurements of fluctuating pressures either (a) on dryer surfaces, or (b) within the MSLs connected to the RPV. If the dynamic analysis of the steam dryer is based on such an approach, the review should include the following:

(1) The benchmarking procedure together with all modeling parameters, bias errors and uncertainties.

(2) The location, selection, and mounting procedure of sensors (in particular, strain gages, pressure transducers, and accelerometers) together with the measures taken to provide sufficient redundancy.

(3) The acoustic modelling parameters (e.g., speed of sound, sound attenuation, and reflection coefficient) that are determined from the benchmarking procedure.

(4) The mesh size of the acoustic model and its conformity to the plant geometry.

(iv) Acoustic and vibration forcing functions generated by RRPs on SSCs should be reviewed. This includes pressure pulsation at the vane passing frequency (VPF) of the pump and its higher harmonics and mechanical vibration at the pump rotor rotation speed. When the forcing functions of the RRPs are reviewed, the following items should be addressed:

(1) The data used to formulate the RRP forcing functions.

(2) The frequency range of pump excitations, in comparison to the structural and acoustic resonance frequencies of reactor and piping components.

(3) The number and distribution of the operating pumps which are used to develop the forcing functions.

For further details, refer to RG 1.20, Revision 4, Section C.2.1.2.

C. Results of Vibration and Stress Calculations. The calculated vibration and stress levels of reactor internal structures are reviewed with their safety margins and acceptance criteria. The procedures for combining the vibration and stress response models (Item 3.A above) with the forcing functions (Item 3.B above) to compute overall vibration and stress response are also reviewed. The results for structures and components with a history of failures from adverse flow effects (such as steam dryers excited by flow instabilities over the openings of valve standoff pipes) are evaluated in greater detail. In particular, the following items are reviewed:

(i) Chosen locations for vibration and stress calculations, and any locations where instrumentation will be installed for subsequent power ascension testing; along with the choice of analysis approach (time or frequency domain) and the acceptability of damping modeling (such as the Rayleigh damping method used for time domain calculations).

(ii) Benchmarking of analysis methodologies, particularly end-to-end benchmarking using previous testing on valid prototypes.

(iii) Stress convergence and high-cycle fatigue evaluation, particularly structural model development, application of dynamic loads to the structural model, and fatigue analysis of two-sided fillet welds.

(iv) Vibration and stress limits and their justification.

For further details, refer to RG 1.20, Revision 4, Section C.2.1.3.

D. Preoperational and Initial Testing Analysis. Calculated structural and hydroacoustic responses for preoperational and initial startup testing conditions as well as for normal operation are reviewed. Any adjustments to the calculated response during the ~~ITAAC~~inspections, tests, analyses, and acceptance criteria (ITAAC) process to address differences between as-designed and as-built conditions are also reviewed. Further, the rationale for the selection of the testing locations and measurement types (strain, vibration, pressure) is reviewed.

4. The NRC staff will review the plans for vibration monitoring of reactor internals conducted during the preoperational and startup test program. General guidance for preoperational and initial startup testing for prototypes, limited prototypes, and nonprototypes is identical to that for the analyses described in Section I.3. Also, components that should be considered for instrumentation in preoperational and initial startup testing are identical to those listed in Section I.3.

For components with no prior history of adverse effects due to FIV, AR, AIV, or MIV; and which have been shown by analysis to have a high margin of safety against such effects, no on-structure instrumentation or measurements are necessary. Measurements need to be performed, however, on systems and components that have been adversely affected by FIV, AR, AIV and MIV in the past (such as BWR steam dryers), and on those reactor internals for which analysis has not shown a high margin of safety against such effects. Instrumentation will also be needed for new components that have no operating experience. Less instrumentation and measurements are needed for limited prototypes. Additional guidance for limited prototypes is found in RG 1.20, Revision 4, Section C.3.2, and for nonprototypes in Section C.4.2,

The purpose of vibration monitoring is to confirm and complement the analyses to demonstrate that adverse flow, mechanical, and acoustic excitation mechanisms expected during all potential plant operating conditions will not cause excessive vibrations or structural damage. The NRC staff reviews the test program description to ensure it includes all potential plant operating conditions (such as pressures, flow rates, and temperatures), potential flow-induced resonance behavior, mechanical and acoustic sources, sensor types and locations (and how the locations are connected to key response regions identified by the analyses), a description of test procedures and methods to be used to process and interpret the measured data including bias errors and uncertainties, a description of the visual inspections to be made, planned comparisons of the test results with the analytical predictions along with acceptability criteria, and plans for updating and correcting the analyses based on the monitoring results.

The following areas related to the dynamic response analysis are reviewed along with their bias errors and uncertainties. For more details, refer to RG 1.20, Revision 4, Section C.2.

A. Pre-Testing Documentation. The pre-testing documentation and test plan should provide assurance that accurate and reliable measurements will be made, and that the provisions listed in RG 1.20, Revision 4, Section C.2.2.3 are met. They include the items below.

(i) Instrumentation types, locations, anticipated life, and planned use (validation of analyses, limits on vibration or stress). The end-to-end bias errors and uncertainties associated with the overall measurement procedures should be reviewed. RG 1.20, Revision 4, Section C.2.2.2 contains additional guidance related to in-plant measurement issues.

(ii) The power ascension plan, including planned operating conditions, measurement conditions and power levels, limits, and plans for extrapolating to higher power levels prior to further power ascension.

(iii) Acceptance criteria and allowable limits, including actions to be taken if any limits are exceeded, and plans for updates to benchmarking bias errors and uncertainties.

(iv) Duration of testing, with justification that the duration is sufficient to ensure adequate fatigue life assessment.

(v) Plans for whether actual or dummy fuel elements will be used during the preoperational testing.

B. Power Ascension and Post-Testing Provisions. The staff reviews the plans developed by the applicant for power ascension monitoring and post-testing evaluation for reasonableness and consistency with the original analysis results. In particular, the following items should be reviewed:

(i) Vibration, strain, and pressure transducer data, including peak and root-mean-square (RMS) values and, where applicable, frequency spectra.

(ii) Comparisons of vibration, strain, and pressure transducer data to acceptance limits, as well as previously simulated analysis results.

(iii) Updates to analysis model benchmarking based on comparisons of measurements and simulations, as well as to acceptance limits.

(iv) Estimates, through data extrapolation, of anticipated vibration, strain, and pressure levels at subsequent power levels, and comparisons to acceptance limits.

(v) Final data analysis, comparisons to simulations and limits, and benchmarking, to be included in the final report provided to the staff following power ascension.

C. Special Guidance for Steam Dryer Monitoring. The staff reviews the applicant's plans to obtain and evaluate BWR steam dryer vibration and stress measurements. Key review elements include:

(i) Data from hood and skirt instrumentation, including pressure sensors and strain gages.

(ii) RMS and frequency-dependent acceptance limits on the instrumentation.

(iii) Cumulative alternating stress analyses for the locations showing the highest stress levels.

(iv) Remote testing and monitoring of dryer vibration through MSL strain gage arrays, and whether the projected dryer loading and alternating stresses are acceptable for remote monitoring (rather than on-dryer instrumentation).

Further details are provided in RG 1.20, Revision 4, Section C.2.2.1.

5. Dynamic system analyses should confirm the structural design adequacy and ability, with no loss of function, of the reactor internals to withstand the loads from a loss-of-coolant accident (LOCA) in combination with the safe-shutdown earthquake (SSE). The staff review addresses the methods of analysis, the considerations in defining the mathematical models, the descriptions of the forcing functions, the computational scheme, the acceptance criteria, and the interpretation of analytical results.

6. The testing methods should include a description of their use to correlate results from the reactor internals vibration and stress tests with the analytical results from dynamic analyses of the reactor internals under steady-state and operational flow transient conditions and under any significant loading induced by adverse flow conditions, or by mechanical or acoustic resonance effects. Where applicable, the methods should also include a description of how they will be used to correlate any SMT results with those of analytical simulations or in-plant measurements.

   Plans for benchmarking analytic results should be reviewed. In particular, methods for determining conservative bias errors and uncertainties should be assessed. It is preferable for applicants to apply end-to-end benchmarking procedures that encompass all individual bias errors and uncertainties. This is best accomplished with measurements of SSC operational vibration, surface pressures, and structural strains.

   In addition, test results from plants of similar characteristics may be used to verify the mathematical models for the loading condition of LOCAs in combination with the SSE by comparing such dynamic characteristics as the natural frequencies. The staff review addresses the methods for comparison of test and analytical results and for verification and validation of the analytical models. However, any differences between the plant under review and previous similar plants leading to the appearance of flow-excited acoustic or structural resonances should be reviewed in detail.

7. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with SSCs related to this SRP section in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff reviews the ITAAC to verify that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

   Examples of ITAAC related to FIV of reactor internals specified for the Economic Simplified Boiling Water Reactor (ESBWR) DC and AP1000 DC are provided in Appendix A and Appendix B, respectively, to this SRP section. Similar ITAAC, or a subset, may be appropriate for a given application depending on the details of the reactor internals FIV review.

8 COL Action Items and Certification Requirements and Restrictions. For a DC application, the staff reviewer verifies that the DC documents include the appropriate COL action items, and requirements and restrictions (e.g., interface requirements and site parameters).

For a COL application referencing a DC, the staff reviewer determines whether the COL applicant has adequately addressed the COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, the staff reviewer also verifies that the COL applicant adequately addressed the requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC).

Review Interfaces

Other SRP sections interface with this section as follows:

1. Some of the computer programs used in the analyses addressed in this SRP section are reviewed under SRP Section 3.9.1. Computer programs and modeling approaches used to calculate dynamic and stress responses of structures and systems at frequencies above those of seismic events are reviewed according to the acceptance criteria described in ~~subsection~~Subsection II.3 of this SRP section.

2. The designs of ASME Code Class 1, 2, and 3 components, component supports, and core support structures related to load combinations and stress limits are reviewed under SRP Section 3.9.3.

3. The design of reactor vessel internal components is reviewed under SRP Section 3.9.5.

4. The seismic qualification testing of seismic Category I mechanical equipment is reviewed under SRP Section 3.10.

5. Verification that (i) the various flow modes to be used to conduct the vibration test of the reactor internals represent the steady-state and operational transient conditions anticipated for the reactor during its service, and (ii) an acceptable hydraulic analysis has determined the loads acting on the reactor coolant system (RCS) piping and the reactor internals is performed under SRP Section 4.4.

6. Review of applications that propose to eliminate consideration of design loads of the dynamic effects of pipe rupture is performed under SRP Section 3.6.3.

7. Review of the applicant's determination of the number of earthquake cycles to be considered in seismic Category I subsystem and component design, as well as the seismic system analysis is performed under SRP Sections 3.7.2 and 3.7.3.

8. Review of piping system analyses including seismic and LOCA analyses is performed under SRP Section 3.12.

Additional details for the review of the applicant's vibration analysis and testing program are contained in RG 1.20, Revision 4. RG 1.20 is an integral supplement to this SRP section.

The specific acceptance criteria and review procedures are contained in the referenced SRP sections.

II. ~~II.~~ ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. 10 CFR 50.55a and GDC 1 to 10 CFR Part 50, Appendix A, as they relate to the testing of systems and components to quality standards commensurate with the importance of the safety function to be performed.

2. GDC 2 and 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," as they relate to SSCs important to safety designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena.

3. GDC 4~~,~~ as it relates to SSCs important to safety appropriately designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.

4. GDC 14~~,~~ as it relates to designing SSCs of the reactor coolant pressure boundary (RCPB) to have an extremely low probability of rapidly propagating failure and of gross rupture.

5. GDC 15~~,~~ as it relates to designing the RCS with sufficient margin to assure that the RCPB is not exceeded during normal operating conditions, including anticipated operational occurrences.

6. Appendix B to 10 CFR Part 50, as it relates to quality assurance in the dynamic testing and analysis of SSCs.

7. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act, and the NRC's regulations;

8. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the COL, the provisions of the Atomic Energy Act, and the NRC's regulations.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for the review described in this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

1 Relevant requirements of GDCs 1, 2, 4, 14, and 15 are met if vibration, thermal expansion, and dynamic effects testing are conducted during startup functional testing for specified high- and moderate-energy piping and their supports and restraints. The purposes of these tests are to confirm that the piping, components, restraints, and supports have been designed to withstand the dynamic loadings and operational transient conditions encountered during service as required by the ASME BPV Code and to confirm that no unacceptable restraint of normal thermal motion occurs. Test specifications should be in accordance with ASME OM-S/G-1990, "Standards and Guides For Operation of Nuclear Power Plants," Part 3, "Requirements for Preoperational and Initial Start Up Vibration Testing of Nuclear Power Plant Piping Systems," and Part 7, "Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems."

An acceptable test program to confirm the adequacy of the designs should include the following:

A. A list of systems to be monitored.

B. A list of the flow modes of operation and transients, such as pump trips, valve closures, to which the components will be subjected during the test. (For additional guidance, see RG 1.68). For example, the transients of the RCS heatup tests should include, but are not limited to, the following:

(i) Reactor coolant pump start.

(ii) Reactor coolant pump trip.

(iii) Operation of pressure-relieving valves.

(iv) Closure of a turbine stop valve.

C. A list of all potential flow-induced resonance conditions and whether testing reveals their presence at typical plant operating conditions. In particular, the lock-in of flow instability modes with acoustic resonances due to pipe flow over side branches needs to be assessed.

D. A list of selected locations in the piping system at which visual inspections and measurements (as needed) will be performed during the tests. For each of these selected locations, the deflection (peak-to-peak), pressure, or other appropriate criteria to demonstrate that the stress and fatigue limits are within the design levels should be provided.

E. A list of snubbers on systems that experience sufficient thermal movement to measure snubber travel from cold to hot position.

F. A description of the thermal motion monitoring program (i.e., verification of snubber movement, adequate clearances and gaps, and acceptance criteria and methods for measuring motion).

G. If vibration is noted beyond the acceptance levels set by the criteria of Item II.1.D above, corrective restraints should be designed, incorporated in the piping system analysis, and installed. If during the test piping system restraints are determined to be inadequate or are damaged, corrective restraints should be installed and another test should determine whether the vibrations have been reduced to an acceptable level. If no snubber piston travel is identified, the corrective action to be taken to ensure that the snubber is operable should be described.

2 To meet the requirements of GDC 2, acceptance criteria for the areas of review identified in ~~subsection~~Subsection I.2 of this SRP section are described below. Other approaches that are justified as equivalent to or more conservative than the stated acceptance criteria may be used to confirm the ability of all seismic Category I systems and components and their supports to function as needed during and after an earthquake.

A. Seismic Analysis Methods. The seismic analysis of all seismic Category I systems, components, equipment, and their supports (including supports for conduit and cable trays and ventilation ducts) should utilize either a suitable dynamic analysis method or an equivalent static load method, if justified.

(i) Dynamic Analysis Method. A dynamic analysis (e.g., response spectrum method or time history method) should be performed when the use of the equivalent static load method cannot be justified. To be acceptable, such analyses should address the following items:

(1) Use of either the time history or the response spectrum method.

(2) Use of an adequate number of mass degrees of freedom in dynamic modeling. SRP Section 3.7.2, Subsection II.1.A.iv provides detailed acceptance criteria for selecting an adequate number of discrete mass degrees of freedom in dynamic modeling, to determine the response of all seismic Category I and applicable non-seismic Category I systems and plant equipment.

(3) Investigation of a sufficient number of modes to ensure participation of all significant modes. When using either the

response spectrum method or the modal superposition time history method, all modes with frequency f ≤ zero period acceleration (ZPA) frequency should be explicitly included in the analysis. As identified in SRP Section 3.7.2, Subsection II.1.A.v, the additional response associated with high frequency modes (i.e., f > ZPA frequency) should be included in the total dynamic solution using the guidance and methods described in RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Sections C.1.4 and C.1.5.

(4) Consideration of maximum relative displacements among supports of seismic Category I systems and components.

(5) Inclusion of such significant effects as piping interactions, externally-applied structural restraints, hydrodynamic (both mass and stiffness effects) loads, and nonlinear responses.

(ii) Equivalent Static Load Method. An equivalent static load method is acceptable if:

(1) There is justification that the system can be realistically represented by a simple model and the method produces conservative results in responses. Typical examples or published results for similar systems may be provided in support of the use of the simplified method.

(2) The design and simplified analysis account for the relative motion between all points of support.

(3) To obtain an equivalent static load of equipment or components that can be represented by a simple model, a factor of 1.5 is applied to the peak acceleration of the applicable floor response spectrum. A factor of less than 1.5 may be used where justified.

(4) In addition, for equipment that can be modeled adequately as a one-degree-of-freedom system, the use of a static load equivalent to the peak of the floor response spectra is acceptable. For piping supported at only two points, the use of a static load equivalent to the peak of the floor response spectra is also acceptable.

B. Determination of Number of Earthquake Cycles. The number of earthquake cycles during one seismic event, the maximum number of cycles for which applicable systems and components are designed, and the criteria and the applicant's procedures to establish these parameters are reviewed by the staff in accordance with the guidance of SRP Section 3.7.3.

C. Basis for Selection of Frequencies. To avoid resonance, the fundamental frequencies of components and equipment selected should be less than 2 or more than twice the dominant frequencies of the support structure. Use of equipment frequencies within this range is acceptable if the equipment is adequately designed for the applicable loads.

D. Three Components of Earthquake Motion. Depending upon what basic methods are used in the seismic analysis (i.e., response spectra or time history method), the following two approaches are acceptable for the combination of three-dimensional earthquake effects:

(i) Response Spectra Method. When the response spectra method is applied for seismic analysis, the maximum structural responses due to each of the three components of earthquake motion should be combined by taking the square root of the sum of the squares of the maximum co-directional responses caused by each of the three components of earthquake motion at a particular point of the structure or the mathematical model.

(ii) Time History Analysis Method. When the time history analysis method is applied for seismic analysis, two methods for combining the responses are acceptable as follows:

(1) The method for combining the three-dimensional effects is identical to that described in Item (i) except that the maximum responses are calculated by the time history method instead of the response spectrum method, or

(2) Obtain time history responses from each of the three components of the earthquake motion and combine them at each time step algebraically. The maximum response is obtained from the combined time solution. When this method is used, the earthquake motions specified in the three different directions should be statistically independent.

E. Combination of Modal Responses. SRP Section 3.7.2 and RG 1.92 present criteria and guidance for modal response combination methods acceptable to the staff.

F. Multiple-Supported Equipment and Components ~~With~~with Distinct Inputs. Equipment and components in some cases are supported at several points by either a single structure or two separate structures. The motions of the primary structure or structures at each of the support points may be significantly different.

A conservative and acceptable approach for equipment items supported at two or more locations is to use an upper-bound envelope of the individual response spectra for these locations to calculate maximum inertial responses of multiple-supported items. In addition, the relative displacements at the support points should be considered. Conventional static analysis procedures are acceptable for this purpose. The maximum relative support displacements can be obtained from the structural response calculations or, as a conservative approximation, from the floor response spectra. For the latter option, the maximum displacement of each support (S_d) is predicted by:

$$\cancel{S_d = S_a g / \omega^2}$$

$$\underline{S_d = S_a g / \omega^2}$$

where S_a is the spectral acceleration in the unit of "g" at the high frequency end of the spectrum curve (which, in turn, is equal to the maximum floor acceleration), g is the gravity constant, and ω is the fundamental frequency of the primary support structure in radians per second. The support displacements can then be imposed on the supported item in the worst case combination. The responses due to the inertia effect and relative displacements should be combined by the absolute sum method.

In the case of multiple supports located in a single structure, an alternate acceptable method using the floor response spectra determines dynamic responses due to the worst case single floor response spectrum selected from a set of floor response spectra at various floors and applied identically to all the floors provided there is no significant shift in frequencies of the spectra peaks. In addition, the support displacements should be imposed on the supported item in the most unfavorable combination by static analysis procedures. Further criteria and methods for the evaluation of multiple support arrangement analysis issues are described in SRP Sections 3.7.2 and 3.7.3.

These methods can result in overestimation of seismic responses. Acceptable alternate response spectrum analysis methods that provide more realistic estimation of seismic responses are discussed in ~~subsection~~Subsection II.9 of SRP Section 3.7.3.

In lieu of the response spectrum approach, time histories of support motions may be used as excitations to the systems. Because of the increased analytical effort compared to the response spectrum techniques, usually only a major equipment system would warrant a time history approach. The time history approach does, however, provide more realistic results in some cases as compared to the response spectrum envelope method for multiple-supported systems.

G. Use of Constant Vertical Static Factors. The use of constant vertical load factors as vertical response loads for the seismic design of all seismic Category I systems, components, equipment, and their supports in lieu of a vertical seismic system dynamic analysis is acceptable only if the structure is rigid in the vertical direction, up to the highest frequency of interest.

H. Criteria Used for Damping. RG 1.61,"Damping Values for Seismic Design of Nuclear Power Plants," provides acceptable values that may be applied in damping analyses. The methods for analysis of damping should be consistent with those described in SRP Section 3.7.2.

3. To meet the requirements of GDCs 1 and 4, the following guidelines, in addition to RG 1.20, apply to the methodologies used to predict the vibration of reactor internals. If the reactor internals are a non-prototype or limited prototype design, the applicant should specify the results of tests and analyses for the valid prototype reactor and provide a summary of the results, including an assessment of the potential for adverse flow effects. Applicants should provide sufficient detail to support the classification of nonprototype or limited-prototype, even when data are obtained from nuclear power plants in other countries. Applicants are expected to provide all necessary data, even when rights to these data are held by other companies or licensees. The acceptability of any limited prototype or nonprototype designation is to be assessed on a case-by-case basis depending on the breadth and quality of the available information.

In their submission, applicants/licensees should address the considerations for selecting the mathematical models and computer software, the interpretation of analytical and numerical results and corresponding bias errors and uncertainties, the acceptance criteria, and the methods for verifying simulations by means of tests.

Dynamic responses of reactor internals to self-excited flow oscillations should be estimated. The analysis should clearly identify whether each mechanism will be excited during the planned operating range of the power plant. Full dynamic analysis is requested for mechanisms expected to generate adverse flow effects, including estimation of vibration and stress amplitudes at the critical locations and, in particular, where vibration sensors will be mounted on the reactor internals. RG 1.20, Revision 4, Section C.2.1.2 provides more guidance on self-excited flow instabilities.

The dependence of the dynamic response on mechanical excitation forces like reactor recirculation pump frequencies and the flow path configuration should also be evaluated. Any frequency coincidence between the pump blade passing frequency and the natural frequencies of the internal structures should be identified and supplemented with bias error and uncertainty analysis.

A. An acceptable summary of structural, hydraulic, and acoustic modeling consists of the following items. Additional detail is available in RG 1.20, Revision 4, Section C.2.1.1

(i) Simulated and measured (from modal or shaker testing) resonance frequencies and mode shapes should be examined to establish the appropriateness and accuracy of the analytic model(s). Simulated and measured mode shapes should be similar, and corresponding resonance frequencies should be used to establish the bounds of modeling accuracy over frequency. Subsequent response calculations should include forcing functions conservatively shifted upward and downward in frequency to span the accuracy range of the models. Many recent applications have adopted an uncertainty range of ±10% over frequency.

(ii) The damping factors for different modes should be properly identified and substantiated. In prior submissions, applicants/licensees have referenced NRC damping guidance for very low frequency seismic analyses as justification for high damping factors for mid-to-high frequency analyses. RG 1.20 corrects this guidance and states that damping factors used in structural dynamic modeling needs to be based on mid- to high-frequency measurements or rigorous analyses conducted on structures typical of the reactor internal structure modeled. In general, any assumed damping greater than 1% of critical needs to be substantiated with measurements.

(iii) The uncertainties and bias errors of the amplitudes of the FRFs also should be provided, and based on specific comparison of simulated and measured data. The uncertainties and bias errors may be estimated from comparisons of simulations to measurements made on structures similar in construction to the reactor internal being modeled. The results of dynamic vibration response tests with known force inputs are expected for replacement steam dryer analyses.

B. The forcing functions should account for the effects of transient flow conditions and the corresponding frequency content. Any potential amplification of a forcing function caused by self-excitation or lock-in of a flow instability with a structural or acoustic resonance should be clearly quantified (see RG 1.20, Revision 4, Section C.2.1.2 for more guidance on self-excited flow instabilities). Acoustic and mechanical forcing functions should also be described.

The suitability of the approach used to define forcing functions should be assessed with expected bias errors and uncertainties of the selected approach. In addition to direct measurements in nuclear power plants, the following approaches may be used to formulate the forcing functions.

(i) SMTs may be used to assess the susceptibility of plant components to FIV and fluid-structure interaction mechanisms as well as to help develop the forcing functions and/or establish amplification factors (sometimes called BUFs) between loads at higher plant power levels (see Section C.2.1.2 – Scale Model Testing of RG 1.20, Revision 4, for additional details). The use of SMT results in the vibration and stress analysis is acceptable where shown to be conservative. Geometric and dynamic similarity between SMT and full-scale plants needs to be established, with critical geometric details sufficiently modeled. Reynolds and Mach number effects need to be quantified, along with any differences in vibration damping and sound attenuation. Multiple tests are needed to establish uncertainties, and need to span a wide range of power and flow conditions to ensure worst-case conditions are measured, particularly when transient tests, such as blowdown testing, is conducted. Growth rates of resonance peaks should be measured over narrow frequency bands. Any SMT-based BUFs should be shown to be conservative when compared to previously observed ratios from similar full-scale plants.

(ii) CFD simulations can be used to develop unsteady forcing functions, or to compute the distribution of flow velocity which is used to develop the forcing functions (see Section C.2.1.2 of RG 1.20, Revision 4 for additional details). Any CFD codes used to determine the forcing functions need to be validated and shown to provide bounding excitation limits. The computation model needs to reflect the details of the fluid domain, and the effects of grid size and time step need to be addressed. The simulated flow cases should include, but not be limited to, the worst-case scenario causing the strongest mal-distribution of reactor flow. In complex flow situations, only conservative assumptions should be made to determine the forcing functions (e.g., correlation length and phase distribution of fluid forces).

(iii) Force inference methods have been used in previous EPU applications to aid the vibration analysis of (replacement) steam dryers in BWRs. These methods use inverse acoustic models to estimate fluctuating pressures within the RPV and on BWR steam dryers. The dryer pressures are inferred from alternating strain or pressure measurements on the dryer surface or on the MSLs connected to the RPV (see Section C.2.1.2of RG 1.20, Revision 4, for additional details). Force inference methods should be benchmarked on plants and systems similar to the plant being designed or licensed. All uncertainties and bias errors resulting from the methodology benchmarking need to be established. The selection of sensor locations and the use of reliable methods to filter extraneous noise from sensor signals need to be addressed. The geometric and mesh size adequacy of the acoustic model used to compute the pressure acting on the steam dryer need to be demonstrated, and the acoustic parameters (such as the speed of sound, reflection coefficient and sound attenuation) need to be substantiated.

(iv) Analysis of AIV and MIV involves knowledge of the RRP forcing functions. These include several tonal forces at multiples of pump drive frequency, and those induced by electromagnetic oscillations within motor cores, by imbalance and misalignment, and by hydrodynamic forces at multiples of impeller VPFs (see Section C.2.1.2 of RG 1.20, Revision 4, for additional details). The method used to determine the forcing functions of individual pumps needs to be substantiated. For reactors with several pumps, the effect of various scenarios of pump operation modes (including number, arrangement and phase) together with the possibility of local amplification and beat of pump tones needs to be addressed. The frequency range of pump excitations (hydrodynamic and mechanical), its relation to resonance frequencies of reactor internals (mechanical and acoustic), and the possibility of resonance of internal components by the pump excitations need to be adequately addressed for the entire range of reactor operating conditions.

Alternatively, the applicant/licensee may use other approaches than those listed above to formulate conservative bounding forcing functions. However, sufficient supporting justification should be provided to demonstrate that the selected approach is technically sound and realistically predicts the forcing function, or an upper bound of the forcing function. In addition, an assessment of bias errors and uncertainties should be provided.

C. Acceptable methods of obtaining dynamic responses, margins of safety, and acceptance criteria for subsequent startup testing for vibration and stress predictions are as follows. Additional detail for each subtopic is available in RG-_1.20, Revision 4, Section C.2.1.3.

(i) Chosen locations for vibration and stress calculations should span all critical regions of the structure, along with locations where instrumentation will be installed for subsequent power ascension testing. Calculations should be conducted over multiple conditions where forcing function time histories are expanded and compressed to span the uncertainty range of simulated resonance frequencies. For each location, the maximum response, the modal contribution to the total response (in case of cyclic or resonant behavior), and the response causing the maximum stress amplitude should be calculated. The choice of analysis approach (time or frequency domain) should be substantiated, confirming that conservative analysis results are obtained, along with the acceptability of damping modeling (such as the Rayleigh damping method used for time domain calculations). If Rayleigh damping is used, the 'anchor frequencies' should be chosen appropriately to ensure that conservative damping is applied over all important frequencies.

(ii) Benchmarking of analysis methodologies, particularly end-to-end benchmarking using previous testing on prototypes, should be substantiated. If individual bias errors and uncertainties of the various modeling components (including structural and forcing function) are combined into approximate end-to-end values, they should be checked to ensure that the predictions bound the measurements in the end-to-end comparisons. Any non-conservative differences should be quantified for their potential impact on structural integrity and fatigue life.

(iii) Structural Evaluation

(1) Structural Model Development and Stress Convergence

For structural analysis, a verified finite element computer code needs to be employed. Modeling of reactor internal structures (e.g., steam dryer) typically entails the use of solid, plate/shell, and beam elements. 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. The connection modeling technique that is implemented is not allowed to result in a reduction or distortion of stress in the plate at the connection.

A suitable finite element mesh, consistent with the loading, the expected structural response, and the intended use of the stress analysis output, needs to be used. The finite element mesh 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. A mesh sensitivity study needs to confirm the adequacy of the mesh. The method of applying the dynamic loading to the structural model and the dynamic analysis parameters, which depend on the method of solution, needs to be substantiated. For local areas of the model, where stress output for fatigue analysis will be extracted, it is necessary to check stress convergence by systematically reducing the local element size. The acceptance criterion is that the desired stress output has converged, or that a reasonable projection of the converged stress can be made from the results of the successive mesh refinement analyses.

More detailed acceptance criteria, including the specification of the dynamic analysis parameters and the application of the dynamic pressure loading, are described in RG 1.20, Revision 4, Section C.2.1.3.

(2) High-Cycle Fatigue Evaluation of Two-Sided Fillet Welds

Fillet welded connections are common in reactor internals, especially in BWR steam dryers. Stress results from a finite element analysis are typically used in the fatigue evaluation of fillet welded connections. Acceptable approaches depend on the local geometric complexity and the level of model refinement at the fillet welded connection. Two acceptable methods for determining the peak stress for use in a fatigue evaluation are described in RG 1.20, Revision 4, Section C.2.1.3. The applicant/licensee should demonstrate the conservatism of the method employed to determine the peak stress at the weld, for use in the fatigue evaluation.

(iv) Vibration and stress limits should be documented and substantiated. Acceptance criteria should be established for allowable responses and for the location of vibration sensors during startup testing. Such criteria relate to the code-allowable stresses, strains, and limits of deflection established to preclude loss of function of the reactor internals. Any deviations from commonly accepted standards, such as ASME BPV Code on fatigue limits, should be justified. Additional factors of safety imposed on fatigue limits should be consistent with previously accepted levels. For example, remote monitoring of steam dryer alternating stress via MSL strain gage array measurements commonly needs an added factor of safety of 2.

D. Calculated structural and hydroacoustic responses for preoperational and initial startup testing conditions as well as for normal operation must be made available, along with plans for subsequent comparisons to test data during power ascension. Any adjustments to calculated response during the ITAAC process to address differences between as-designed and as-built conditions should be documented, particularly changes in welds, materials, wall thicknesses, and any other parameter which affects structural response significantly. Choice of testing locations must be based on correlation to regions of predicted maximum structural vibration and stress, as well as minimized sensitivity to uncertainty in sensor orientation and location. For example, strain gages should be installed and oriented near low strain gradient locations, and accelerometers installed near maximum amplitude locations.

4 For requirements of GDCs 1 and 4, the preoperational and initial startup vibration and stress test program for the reactor internals should conform to the provisions specified in RG 1.20, Revision 4. The testing needs to include vibration and strain prediction, vibration and strain monitoring, adverse flow effects (flow-induced acoustic and structural resonances), data reduction, bias errors and uncertainty analysis, and surface inspections.

The vibration testing should be conducted with the fuel elements in the core or with dummy elements with equivalent dynamic effects and flow characteristics. Testing without fuel elements in the core may be acceptable if testing in this mode is demonstrably conservative. The planned duration of the test for the normal operation modes to ensure that all critical components are subjected to at least 106 cycles of vibration should be provided. For instance, if the lowest response frequency of the core internal structures is 10 Hz, a total test duration of 1.2 days or more is acceptable. Vibration predictions, test acceptance criteria and bases, and permissible deviations from the criteria should be made available before the test. The methods and procedures to process the test data for meaningful interpretation of the vibration and stress behavior of various components should be provided. Vibration interpretation should include the amplitude, frequency content, stress state, and possible effects on safety functions. There should be a detailed analysis performed of bias errors and uncertainties of instrumentation and data acquisition systems. A list of in-plant measurement issues that should be addressed by the applicant to ensure high quality data is in RG 1.20, Revision 4, Section C.2.2.2. Additional guidance for acceptable elements of the test program is below.

A. Pretesting documentation provisions.

(i) The vibration monitoring instrumentation should be described, including instrument types and specifications (including useful frequency and amplitude ranges) and diagrams of locations, including those with the most severe vibratory motions or the most effect on safety functions. Instrumentation should be capable of functioning in typical reactor conditions, and survive the expected duration of the testing. The data acquisition (DAQ) system should also be assessed, including connectivity of the instrumentation to the DAQ system. Routing of the instrumentation cabling should be checked to ensure that no significant electromagnetic interference occurs that could corrupt the measured electrical signals. Additional detail is available in RG 1.20, Revision 4, Section C.2.2.3 item a.

(ii) The power ascension test plan should include normal operation and upset transients. The power ascension program for startup testing should include specific power level plateaus with sufficiently long duration to allow data recording and reduction, comparisons with predetermined limit loading, and inspections and walkdowns for steam, feedwater, and condensate systems. The test program also should include details of actions to be taken if acceptance criteria are not satisfied. Plans should be made available for walkdown inspections and visual and nondestructive surface inspections after completion of the vibration tests. The inspection program description should include the areas subject to inspection, the methods of inspection, the design access provisions to the reactor internals, and the equipment to be used for such inspections, which preferably should follow the removal of the internals from the reactor vessel. Where removal is not feasible, the inspections should be by means of equipment appropriate for in-situ inspection. The areas inspected should include all load-bearing interfaces, core restraint devices, high-stress locations, and locations critical to safety functions. MSL valves also should be inspected if adverse flow effects (flow-induced acoustic and structural resonances) are observed during the startup test. Further information on test procedures is addressed in RG 1.20, Revision 4, Section C.2.2.3 item b.

(iii) Limits on vibration and strain, for peak, RMS, and/or spectra should be provided, along with details on procedures for checking measurements against limits. Plans for benchmarking updates should be provided in the event previous simulations do not bound measured data. More details on limits are provided in RG 1.20, Revision 4, Section C.2.2.3 item c, along with a detailed approach for BWR steam dryers.

B. Power ascension and post-testing provisions. The following data should be provided and compared to limits, and used to rebenchmark simulations as necessary.

(i) Vibration, strain, and pressure transducer data, including peak and RMS values, and where applicable frequency spectra, should be below acceptance limits. If not, rebenchmarking and reanalysis should be conducted to confirm structural integrity, and if necessary update acceptance limits.

(ii) Estimates, via data extrapolation, of anticipated vibration, strain, and pressure levels at subsequent power levels should be made, and compared to acceptance limits. If violations of limits are projected, reanalysis should be conducted to confirm structural integrity at higher power levels. Any flow-excited resonance phenomenon should be extrapolated conservatively based on narrow frequency band growth over increasing reactor power levels. Linear or quadratic extrapolation is generally non-conservative for flow-excited resonance.

(iii) Final data analysis, comparisons to simulations and limits, and benchmarking, should be submitted in a final report to the staff. Any differences between as-built and as-designed structures should be clearly described. Final margins of safety should be submitted, and shown to be greater than 1.0.

C. Special guidance for BWR steam dryers. Monitoring to evaluate potential adverse flow effects on reactor internal components should include the steam dryer. The instrumentation directly mounted on the steam dryer should include pressure sensors, strain gages, and accelerometers. If necessary, the MSLs also should be instrumented to collect data to determine steam pressure fluctuations in MSLs and identify the presence of flow-excited acoustic resonances. These pressure pulsations are used in some inverse analysis methods to define the dryer loading function, the dryer dynamic response and the resulting alternating stresses. Alternating stresses caused by mechanical loads from RRP VPF tones should be quantified based on test data. Additional guidance on direct, as well as remote (via MSL strain gage arrays) dryer stress monitoring is available in RG 1.20, Revision 4, Section C.2.2.1.

5 For requirements of GDCs 2, 4, 14, and 15, dynamic system analyses should confirm the structural design adequacy of the reactor internals to withstand the dynamic loadings of the design-basis LOCA in combination with the SSE. Where a substantial separation between the forcing frequencies of the LOCA (or SSE) loading and the natural frequencies of the internal structures can be demonstrated, the analysis may treat the loadings statically.

Evaluations performed under SRP Section 3.6.3 address review of applications that propose to eliminate consideration of design loads of the dynamic effects of pipe rupture. Evaluation in this Section should interface with the evaluation in Section 3.6.3.

The most severe dynamic effects from LOCA loadings generally result from a postulated double-ended rupture of a primary coolant loop near a reactor vessel inlet or outlet nozzle with the reactor in the most critical normal operating mode. However, all other postulated break locations should be evaluated and the location producing the controlling effects should be identified.

Mathematical models used for dynamic system analysis for LOCAs in combination with SSE effects should include the following:

A. Modeling should include reactor internals and dynamically-related piping, pipe supports, components, and fluid-structure interaction effects when applicable. Typical diagrams and the modeling basis should be developed and described.

B. Mathematical models should typify system such structural characteristics as flexibility, mass inertia effect, geometric configuration, and damping (including possible coexistence of viscous and Coulomb damping).

C. Any system structural partitioning and directional decoupling in the dynamic system modeling should be justified.

D. The effects of flow upon the mass and flexibility properties of the system should be addressed.

Typical diagrams and the basis for postulating the LOCA-induced forcing function should be provided, including a description of the governing hydrodynamic equations and the assumptions for mathematically tractable flow path geometries, tests for determining flow coefficients, and any semi-empirical formulations and scaled model flow testing for determining pressure differentials or velocity distributions. The acceptability of the hydraulic analysis, as reviewed on request, is based on established engineering practice and generic topical reviews by the staff.

The methods and procedures for dynamic system analyses should be described, including the governing equations of motion and the computational scheme for deriving results. Time domain forced-response computation is acceptable for both LOCA and SSE analyses. The response spectrum modal method may be used for SSE analysis.

The stability of such elements in compression as the core barrel and the control rod guide tubes under outlet pipe rupture loadings should be evaluated.

Either response spectra or time histories may be used for specifying seismic input motions of the SSE at the reactor core supports.

The criteria for acceptance of the analytical results are described in SRP Sections 3.9.3 and 3.9.5. For PWRs, the criteria and review methods for verifying whether the applicant has appropriately addressed asymmetric blowdown loadings on reactor internals are described in SRP Section 3.9.5.

6 For requirements of GDC 1, as to the correlation of tests and analyses of reactor internals, the applicant should address the following items to ensure the adequacy and sufficiency of the test and analysis results. Vibration predictions should be verified by test results. This procedure should consider all sources of bias errors and uncertainties, and preferably be based on end-to-end benchmarking, where measured structural vibration and/or strains are compared to simulations. If the test results differ substantially from the predicted response behavior, the vibration analysis should be modified appropriately to achieve agreement with test results and validation of the analytical method and input forcing functions as appropriate for predicting responses of the prototype unit as well as of other units where confirmatory tests are conducted. An acceptable correlation analysis should include the following comparisons:

A. Comparison of the measured response frequencies with the analytically obtained natural frequencies of the reactor internals for validation of the mathematical models used in the analysis.

B. Comparison of the response amplitude time variation and the frequency content from test and analysis.

C. Comparison of the measured amplitudes, frequencies, and time variations of loads with those assumed during design calculations for validation of the predicted flow, mechanical, and acoustic forcing functions.

D. Comparison of the maximum responses from test and analysis for verification of alternating stress levels.

E. Comparison of the mathematical model for dynamic system analysis under operational flow transients and under combined LOCA and SSE loadings for similarities.

F. Comparison of measurements and predictions of any adverse flow phenomena (e.g., flow excited acoustic and/or structural resonances), in order to validate the models and methodologies used in the predictive analysis of vibration induced by the adverse flow phenomena. If test data indicate the presence of significant fluid-structure interaction (FSI) behavior, where structural vibration feeds back into the flow-induced loading and causes response much higher than originally predicted, the staff will assess the applicant's/licensee's methodology for quantifying the FSI on a case by case basis. An acceptable FSI methodology needs to ensure that conservative, bounding vibration and alternating stress are determined.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this SRP section is discussed in the following paragraphs:

1. GDC 1 requires that systems and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions performed.

Vibration, thermal expansion, and dynamic effects tests are described in this SRP section for startup functional testing of specified high-energy and moderate-energy piping and their supports and restraints. Guidance is provided herein and in RG 1.20 for analysis of vibration of reactor internals. These vibration analyses are confirmed by prototype testing. Dynamic analyses methods are described in this SRP section for all seismic Category 1 systems, components, equipment, and their supports (including supports for conduit and cable trays, and ventilation ducts).

Compliance with the requirements of GDC 1 provides assurance that systems and components within the scope of this SRP section are capable of performing their intended safety functions.

2. GDC 2 requires that SSCs important to safety be designed to withstand the effects of expected natural phenomena combined with effects of normal and accident conditions without loss of capability to perform their safety functions.

Vibration testing, dynamic analyses, and suitable comparisons are described in this SRP section for SSCs important to safety. The tests, analyses, and comparisons are in accordance with sound engineering practices and provide assurance that these SSCs are designed to withstand natural phenomena in combination with normal and accident conditions.

Compliance with the requirements of GDC 2 provides assurance that SSCs within the scope of this SRP section are capable of performing their intended safety functions.

3. GDC 4 requires that the SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions of normal operation, maintenance, testing, and postulated accidents, including LOCAs.

Staff positions on design of SSCs to withstand the dynamic effects of LOCAs in combination with other normal and design basis loads are described in SRP Section 3.9.2. Testing to verify the ability of SSCs to withstand anticipated loads is also described.

Compliance with the requirements of GDC 4 provides assurance that SSCs within the scope of this SRP section are capable of performing their intended safety functions.

4. GDC 14 requires that the RCPB be designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture.

Staff positions described in SRP Section 3.9.2 address dynamic testing of components of the RCPB to ensure that they will withstand the applicable dynamic loads associated with normal operation and transient conditions without leakage, rapidly propagating failure, or gross rupture.

Compliance with the requirements of GDC 14 provides assurance that the RCPB will have an extremely low probability of leakage or failure.

5. GDC 15 requires that the RCS be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Staff positions are described in SRP Section 3.9.2 on dynamic testing of components of the RCPB to resist the appropriate dynamic loads associated with normal operation and transient conditions. Vibration, thermal expansion, and dynamic effects testing are described to verify the design.

Compliance with the requirements of GDC 15 provides assurance that the RCPB will remain intact, thus preventing the spread of radioactive contamination.

III. REVIEW PROCEDURES

The NRC reviewer will select material from the procedures described below, as may be appropriate for a particular application.

These review procedures are based on the identified SRP acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's justification that the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1 Test specifications should be in accordance with ASME OM-S/G-1990, Part 3 and Part 7.

The staff reviews the treatment of dynamic responses of safety-related piping systems by the following procedures:

During the construction permit (CP) stage, the preliminary safety analysis report (PSAR) is reviewed for whether the applicant has specified a provision to conduct a piping steady-state vibration, thermal expansion, and operational transient test program. The applicant's program description should be sufficiently comprehensive to contain the elements of an acceptable program as described in ~~subsection~~Subsection II.1 (Acceptance Criteria) of this SRP section.

During the operating license (OL) stage, the final safety analysis report (FSAR) is reviewed to ensure that the applicant's PSAR provision is fulfilled and the program is developed in sufficient detail. In this and other review procedures, the level of information reviewed for DC and COL applications submitted under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," should be consistent with that of an FSAR submitted in an OL application, except where as-built information is needed. Verification that the as-built facility conforms to the approved design is performed through the ITAAC process.

The reviewer should verify that the applicant's program as described in Sections 3.9.2 and 14.0 of the FSAR is sufficiently developed to:

A. Establish the rationale and bases for the acceptance criteria and selection of locations for monitoring pipe motions.

B. Provide the displacement or other appropriate limits at locations monitored.

C. Describe the techniques and instruments (as applicable) for monitoring or measuring pipe motions.

D. Ensure that corrective actions based on the test results and the effectiveness of the corrective actions will be documented.

2 For seismic system analysis review, the following review procedures are implemented:

A. Seismic Analysis Methods. For all seismic Category I systems, components, equipment, and their supports (including supports for conduit and cable trays and ventilation ducts), the applicable methods of seismic analysis (response spectra, time history, equivalent static load) are reviewed to verify that the techniques are in accordance with the acceptance criteria in ~~subsection~~Subsection II.2.A of this SRP section.

Common industry practice is to assume rigid and fixed attachments between the seismic subsystems (i.e., equipment and piping) and the supporting seismic systems (i.e., structures). In some cases, particularly for heavy equipment, this assumption potentially can cause underestimation of seismic loadings. The reviewer should verify that appropriate assumptions have been made in the seismic analyses as to the stiffness of the seismic subsystem anchorage.

B. Determination of Number of Earthquake Cycles. The number of earthquake cycles during one seismic event, the maximum number of cycles for which applicable systems and components are designed, and the applicant's criteria and procedures to establish these parameters are reviewed by the staff in accordance with the guidance in SRP Section 3.7.3.

C. Basis for Selection of Frequencies. As applicable, criteria or procedures to separate fundamental frequencies of components and equipment from the forcing frequencies of the support structure are reviewed for compliance with the acceptance criteria of ~~subsection~~Subsection II.2.C of this SRP section.

D. Three Components of Earthquake Motion. The procedures by which the three components of earthquake motion are considered in the determination of the seismic response of systems are reviewed for compliance with the acceptance criteria of ~~subsection~~Subsection II.2.D of this SRP section.

E. Combination of Modal Responses. The procedures for combining modal responses are reviewed for compliance with the acceptance criteria of ~~subsection~~Subsection II.2.E of this SRP section when a response spectrum modal analysis method is used.

F. Multiple-Supported Equipment and Components ~~With~~with Distinct Inputs. The criteria for the seismic analysis of multiple-supported components and equipment with distinct inputs are reviewed for accordance with the acceptance criteria of ~~subsection~~Subsection II.2.F of this SRP section.

G. Use of Constant Vertical Static Factors. Use of constant static factors as response loads in the vertical direction for the seismic design of any seismic Category I systems in lieu of a detailed dynamic method is reviewed for whether constant static factors are used only if the structure is rigid in the vertical direction based on the definition for rigidity in ~~subsection~~Subsection II.2.G of this SRP section.

H. Criteria for Damping. The criteria for accounting for damping in systems, components, equipment, and their supports are reviewed in accordance with the criteria in ~~subsection~~Subsection II.2.H of this SRP section.

3 For a CP application review, the reviewer should verify that the applicant describes the analysis of the vibration of such reactor internal structures as those listed in ~~subsection~~Subsection I.3 of this SRP section if designated as a prototype, limited prototype, or non-prototype design. The vibration analysis should consider adverse effects from possible vibration excitation mechanisms (FIV, AR, AIV and MIV). The methods and procedures for the analysis should be described.

For an OL application review, the reviewer should verify that that the application includes a detailed dynamic analysis for a prototype, limited prototype, or non-prototype design for vibration prediction prior to the performance of preoperational and initial startup vibration tests. Acceptance of the analysis is based on the technical soundness of the analytical method and procedures and the degree of compliance with the acceptance criteria in Section II.3 of this SRP. In addition, the analysis should be verified by correlation with the test results.

4 For a CP application, the reviewer should verify that the program for preoperational and initial startup vibration testing of reactor internals for FIV includes the following:

A. The applicant should specify the intention to perform preoperational and initial startup vibration monitoring based on categorization of each specific component as a prototype, limited prototype, or non-prototype.

B. The applicant should describe the preoperational and initial startup vibration test program. The staff review will be based on compliance of this program with the requirements of ~~subsection~~Subsection II.4 (Acceptance Criteria) of this SRP section.

C. If the component is a non-prototype, the applicant should specify the valid prototype component in a plant of similar design. The staff reviews the appropriateness of the designated valid prototype, including any differences in the flow conditions or the design of reactor internal structures, from the valid prototype to verify whether any design modifications substantially alter the behavior of the flow transients and the response of the reactor internals. Additional detailed analysis, SMTs, or installation of some instrumentation during the confirmatory test may be necessary to complete the review. In addition, the applicant will need to implement a prototype program if timely adequate test results are not obtained for the designated valid prototype.

For an OL application, the reviewer should verify the following:

A. The applicant has developed a detailed preoperational and initial startup vibration test program and the schedule for its implementation. If elements of the program differ substantially from the guidelines specified in RG 1.20, the applicant has justified the differences.

B. The applicant has addressed the acceptability of vibration prediction, the visual surface inspection procedures, the details of instrumentation for vibration monitoring, the methods and procedures for processing the test results, and such supplementary tests as component vibration tests, flow tests, and scaled model tests.

C. For a nonprototype plant, the applicant has addressed the applicability of the designated valid prototype, including the design and operating condition similarities of the reactor internal structures to those of the prototype. Additional detailed analysis, SMTs, or vibration monitoring in the confirmatory tests may be needed to complete the review.

5. In the CP stage review of the dynamic analysis of the reactor internals under faulted condition loadings, the applicant should make available this analysis or identify the applicable document with the required information. The scope and methods of analysis should be described.

In the OL review, the staff reviews the detailed information for whether an adequate analysis has been made of the capability of reactor internal structures to withstand dynamic loads from the most severe LOCA in combination with the SSE. The staff review includes the analytical methods and procedures, the basis of the forcing functions, the mathematical models to represent the dynamic system, and the stability investigations for the core barrel and essential compressive elements. Acceptance of the analysis is based on (1) the technical soundness of the analytical methods, (2) the degree of compliance with the acceptance criteria listed, and (3) verification that stresses under the combined loads are within allowable limits of the applicable code and deformations are within the limits set to ensure the ability of reactor internal structures to perform needed safety functions. The reviewer verifies that an acceptable hydraulic analysis has been performed.

6. The reviewer will evaluate the applicant's program to implement the preoperational and initial startup test procedure to correlate the test measurements with the analytically/computationally predicted flow-induced dynamic response of the reactor internals. The applicant's provisions in this area should specify submittal of a timely report of the monitoring results. The applicant's provisions should ensure that the report will summarize the analyses and test results for review of the compatibility of the results from tests and analyses, the consistency between mathematical models for different loadings, and the validity of the interpretation of the test and analysis results.

7. For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the FSAR meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also determine whether the identified COL action items are adequate and sufficient. Where the reviewer identifies additional COL action items, these COL action items should be included in the DC FSAR or DCD.

For review of a COL application, the scope of the review is dependent on whether the COL applicant references a certified design, an early site permit (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report). The NRC staff reviewer should determine whether the COL applicant has adequately responded to each COL action item, and supplemented the DC application or other referenced NRC approvals to address any plant-specific design aspects.

For review of submittals for replacement components or power uprates for operating reactors, the reviewer should follow the above procedures to verify that the design meets the acceptance criteria.

For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC.

IV. EVALUATION FINDINGS

The NRC staff reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

1. The applicant has met the relevant requirements of GDCs 14 and 15 for the design and testing of the reactor coolant pressure boundary to ensure a low probability of rapidly propagating failure and of gross rupture and to ensure that design conditions are not exceeded during normal operation, including anticipated operational occurrences, by an acceptable vibration, thermal expansion, and dynamic effects test program to be conducted during startup and initial operation on specified high- and moderate-energy piping and its systems, restraints, and supports. The tests provide adequate assurance that the piping and piping restraints of the system are designed to withstand vibrational dynamic effects of valve closures, pump trips, and other operating modes of design-basis flow conditions. In addition, the tests provide assurance of adequate clearances and free movement of snubbers for unrestrained thermal movement of piping and supports during normal system heatup and cooldown operations. The planned tests will develop loads similar to those experienced during reactor operation.

2. The applicant has met the relevant requirements of GDC 2 for demonstrating design adequacy of all seismic Category I systems, components, equipment, and their supports to withstand earthquakes by meeting the relevant acceptance criteria of SRP Sections 3.7.2 and 3.7.3, including the applicable regulatory positions of RGs 1.61 and 1.92 and by providing acceptable seismic systems analysis procedures and criteria. The scope of review of the seismic system analysis included the seismic analysis methods of all seismic Category I systems, components, equipment, and their supports and procedures for seismic analysis of multiple-supported equipment and components with distinct inputs, justification for the use of constant vertical static factors, and determination of composite damping.

   The system analyses are performed by the applicant on an elastic basis. Modal response spectrum, multi-degree of freedom, and time history methods form the bases for the analyses of all major seismic Category I systems, components, equipment, and their supports. Modal response parameters are combined in accordance with the appropriate acceptable methods described in SRP Section 3.7.2 and/or RG 1.92. The square root of the sum of the squares of the maximum codirectional responses is used in accounting for three components of the earthquake motion for both the time history and response spectrum methods. Floor spectra inputs to be used for design and test verifications of systems, components, equipment, and their supports are generated from the time history method, taking into account variation of parameters by peak widening. There will be a vertical seismic system dynamic analysis for all systems, components, equipment, and their supports where analyses show significant structural amplification in the vertical direction.

3 The applicant has met the requirements of GDCs 1 and 4 for design and testing of reactor internals with the potential to generate loose parts to quality standards commensurate with the importance of the safety functions performed with appropriate protection against dynamic effects. The applicant has met the regulatory positions of RG 1.20 for the conduct of preoperational and initial startup vibration tests by a preoperational and initial startup vibration program planned for the reactor internals providing an acceptable basis for design adequacy of these internals under test loading conditions comparable to those experienced during operation. The combination of tests, predictive analysis, and post-test inspection provides adequate assurance that the reactor internals will, during their service lifetime, withstand the FIV of reactor operation without loss of structural integrity. The integrity of the reactor internals in service is essential to proper positioning of reactor fuel assemblies and unimpaired operation of the control rod assemblies for safe reactor operation and shutdown.

4 The applicant has met the relevant requirements of GDCs 2 and 4 for design of systems and components important to safety to withstand the effects of earthquakes and appropriate combinations of the effects of normal and postulated accident conditions with the effects of the SSE by a dynamic system analysis which provides an acceptable basis for the structural design adequacy of the reactor internals to withstand the combined dynamic loads of postulated LOCA and SSE and (for a BWR) the combined loads of a postulated MSL rupture and SSE. The analysis provides adequate assurance that the combined stresses and strains in the reactor internals will not exceed the allowable design stress and strain limits for the materials of construction and that the consequent deflections or displacements at any structural elements of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The methods for component analysis have been found compatible with those for the systems analysis. The proposed combinations of component and system analyses are, therefore, acceptable. The assurance of structural integrity of the reactor internals under LOCA conditions for the most adverse postulated loading event adds confidence that the design will withstand a spectrum of lesser pipe breaks and seismic loading events.

5. The applicant has met the relevant requirements of GDC 1 for systems and components designed and tested to quality standards commensurate with the importance of the safety functions performed by the proposed program to correlate the test measurements with the analysis results. The program provides an acceptable basis for demonstrating

the compatibility of the results from tests and analyses, the consistency between mathematical models used for different loadings, and the validity of the interpretation of the test and analysis results.

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this SRP section.

In addition, to the extent that the review is not discussed in other SER sections, the findings will summarize the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications submitted six months or more after the date of issuance of this SRP section, unless superseded by a later revision.

VI. REFERENCES

~~1. Code of Federal Regulations, Title 10 (10 CFR Part 50).~~

~~2. Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," Revision 4.~~

~~3. Regulatory Guide 1.29, "Seismic Design Classification."~~

~~4. Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."~~

~~5. Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."~~

~~6. Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis."~~

~~7. NRC Inspection Manual Chapter IMC-2504, "Construction Inspection Program - Non-ITAAC Inspections," issued April 25, 2006.~~

~~8.~~1 American National Standards Institute, ANSI S2.31-1979 (R2004), "Methods for the Experimental Determination of Mechanical Mobility, Part 1: Basic Definitions and Transducers," ~~American National Standards Institute~~New York, NY.

~~9.~~2 American National Standards Institute, ANSI S2.32-1982 (R2004), “Methods for the Experimental Determination of Mechanical Mobility, Part 2: Measurements Using Single-Point Translational Excitation,” ~~American National Standards Institute.~~New York, NY

~~10.~~3 American Society of Mechanical Engineers, ASME AG-1-2012, “Code on Nuclear Air and Gas Treatment~~.”~~,” New York, NY.

~~11.~~4 American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, “Nuclear Power Plant Components,” ~~American Society of Mechanical Engineers~~New York, NY.

~~12.~~5 American Society of Mechanical Engineers, ASME OM-S/G-2013, “Standards and Guides ~~For~~for Operation of Nuclear Power Plants,” Part 3, “Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems,” including the addenda, and Part 7, “Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems,” ~~ASME.~~ New York, NY.

6 Chu, S. L., M. Amin, and S. Singh, “Spectral Treatment of Actions of Three Earthquake Components on Structures,” Nuclear Engineering and Design, Volume 21, pp. 126-136 (1972).

7 Ewins, D.J., “Modal Testing: Theory, Practice and Application,” 2nd Edition, Taylor and Francis Group, 2000.

~~13.~~8 International Standards Organization, ISO 7626-5:1994, “Vibration and shock - Experimental determination of mechanical mobility - Part 5: Measurements using impact excitation with an exciter which is not attached to the structure.”

~~14.~~1 ~~S. L. Chu, M. Amin, and S. Singh, “Spectral Treatment of Actions of Three Earthquake Components on Structures,” Nuclear Engineering and Design, Volume 21, pp. 126-136 (1972).~~

~~15. D.J. Ewins, “Modal Testing: Theory, Practice and Application,” 2nd Edition, Taylor and Francis Group, 2000.~~

~~16.~~9 ~~M.~~ Hetenyi, M., “Beams on Elastic Foundation,” The University of Michigan Press (1946).

~~17.~~10 ~~R. P.~~ Kassawara, R. P., and D. A. Peck, “Dynamic Analysis of Structural Systems Excited at Multiple Support Locations,” 2nd ASCE Specialty Conference on Structural Design of Nuclear Plant Facilities, Chicago, Dec. 17-18, 1973.

~~18.~~11 ~~N. M.~~ Newmark, N. M., “Earthquake Response Analysis of Reactor Structures,” Nuclear Engineering and Design, Volume 20, pp. 303-322 (1972).

~~19.~~12 ~~N. M.~~ Newmark, N. M., J. A. Blume, and K. K. Kapur, “Design Response Spectra for Nuclear Power Plants,” Journal of the Power Division, American Society of Civil Engineers, pp.~~-~~ 287-303, November 1973.

~~20.~~13 ~~N. M.~~ Newmark, N. M., and E. Rosenblueth, “Fundamentals of Earthquake Engineering,” Prentice Hall (1971).

14 *U.S. Code of Federal Regulations*, “Domestic Licensing of Production and Utilization Facilities,” Part 50, Title 10, “Energy.”

15 U.S. Nuclear Regulatory Commission, “Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing,” Regulatory Guide 1.20, Revision 4.

16 U.S. Nuclear Regulatory Commission, “Seismic Design Classification,” Regulatory Guide 1.29.

17 U.S. Nuclear Regulatory Commission, “Damping Values for Seismic Design of Nuclear Power Plants,” Regulatory Guide 1.61.

18 U.S. Nuclear Regulatory Commission, “Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors,” Regulatory Guide 1.68.

19 U.S. Nuclear Regulatory Commission, “Combining Modal Responses and Spatial Components in Seismic Response Analysis,” Regulatory Guide 1.92.

20 U.S. Nuclear Regulatory Commission, “Construction Inspection Program - Non-ITAAC Inspections,” NRC Inspection Manual Chapter IMC-2504, issued April 25, 2006.

PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the Standard Review Plan are covered by the requirements of 10 CFR Part 50 and 10 CFR Part 52, and were approved by the Office of Management and Budget, approval number 3150-0011 and 3150-0151.

PUBLIC PROTECTION NOTIFICATION

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

Appendix A
ESBWR ITAAC Related to Flow-Induced Vibration of Reactor Internals
Design Control Document, Tier 1, Revision 10

	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
ESBWR DCD Tier 1 Table 2.1.1-3 8a.	The RPV internal structures listed in Table 2.1.1-1 (chimney and partitions, chimney head and steam separators assembly, and steam dryer assembly) must meet the limited provisions of ASME Code Section III regarding certification that these components maintain structural integrity so as not to adversely affect RPV core support structure.	Inspections will be conducted of the as built internal structures as documented in the ASME Code design reports.	The RPV internal structures listed in Table 2.1.1-1 (chimney and partitions, chimney head and steam separators assembly, and steam dryer assembly) meet the limited provisions of ASME Code Section III, NG-1122 (c), regarding certification that these components maintain structural integrity so as not to adversely affect RPV core support structure.
8b.	The RPV internal structures listed in Table 2.1.1-1 (chimney and partitions, chimney head and steam separators assembly, and steam dryer assembly) meet the requirements of ASME B&PV Code, Subsection NG-3000, except for the weld quality and fatigue factors for secondary structural non-load bearing welds.	Inspections will be conducted of the as built internal structures as documented in the ASME Code design reports.	The RPV internal structures listed in Table 2.1.1-1 (chimney and partitions, chimney head and steam separators assembly, and steam dryer assembly) meet the requirements of ASME B&PV Code, Subsection NG-3000, except for the weld quality and fatigue factors for secondary structural non-load bearing welds.
12.	The number and	An analysis of the	The number and locations of

	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	locations of pressure sensors installed on the steam dryer for startup testing ensure accurate pressure predictions at critical locations.	number and locations of pressure sensors installed on the steam dryer for startup testing will be performed.	pressure sensors installed on the steam dryer for startup testing ensure accurate pressure predictions at critical locations.
13.	The number and locations of strain gages and accelerometers installed on the steam dryer for startup testing are capable of monitoring the most highly stressed components, considering accessibility and avoiding discontinuities in the components.	An analysis of the number and locations of strain gages and accelerometers installed on the steam dryer for startup testing will be performed.	The number and locations of strain gages and accelerometers installed on the steam dryer for startup testing are capable of monitoring the most highly stressed components, considering accessibility and avoiding discontinuities in the components.
14.	The number and locations of accelerometers installed on the steam dryer for startup testing are capable of identifying potential rocking and of measuring the accelerations resulting from support and vessel movements.	An analysis of the number and locations of accelerometers installed on the steam dryer for startup testing will be performed.	The number and locations of accelerometers installed on the steam dryer for startup testing are capable of identifying potential rocking of and measuring the accelerations resulting from support and vessel movements.

	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
16.	The as-built steam dryer predicted peak stress is below the fatigue limitation.	Analyses using NRC-approved methodologies are performed.	A report of the fatigue analyses of the as built steam dryer exists and demonstrates that the maximum calculated alternating stress intensity provides at least a Minimum Alternating Stress Ratio of 2.0 to the allowable alternating stress intensity of 93.7 MPa (13,600 psi).
ESBWR DCD Tier 1 Table 2.1.2-3 36.	The main steam line and SRV/SV branch piping geometry precludes first and second shear layer wave acoustic resonance conditions from occurring and avoids pressure loads on the steam dryer at plant normal operating conditions.	Analysis of the as-built piping system and equipment analysis, for acoustic resonance at plant normal operating conditions, will be performed.	The main steam line and SRV/SV branch piping geometry precludes first and second shear layer wave acoustic resonance conditions from occurring and results in no significant pressure loads on the steam dryer at plant normal operating conditions.

Appendix B
AP1000 ITAAC Related to Flow-Induced Vibration of Reactor Internals
Design Control Document, Tier 1, Revision 19

	Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
AP1000 DCD Tier 1 Table 2.1.3-2 ITAAC 7	The reactor internals will withstand the effects of flow induced vibration.	i) A vibration type test will be conducted on the (first unit) reactor internals representative of AP1000. ii) A pre-test inspection, a flow test and a post-test inspection will be conducted on the as-built reactor internals.	i) A report exists and concludes that the (first unit) reactor internals have no observable damage or loose parts as a result of the vibration type test. ii) The as-built reactor internals have no observable damage or loose parts.

Standard Review Plan Section 3.9.2
Description of Changes

Section 3.9.2, "DYNAMIC TESTING AND ANALYSIS OF SYSTEMS, STRUCTURES, AND COMPONENTS"

In addition to the changes itemized below, editorial changes were made throughout for clarity, consistency, and applicability. Changes incorporated into Revision 4 include:

I. AREAS OF REVIEW

- The review of piping seismic analysis was deleted. The piping analysis review is addressed in SRP Section 3.12.
- The general discussion of dynamic analysis was updated to remove specific references to steam generator internals. RG 1.20 provides information that can be useful to the staff in applying the guidance in this SRP section to steam generator internals where necessary. Future guidance updates to address review of steam generator design and performance may address this topic in further detail.
- Small modular reactors were added to the reactor internals dynamic analysis review.
- A list of pressurized water reactor and small modular reactor internal components were added.
- The review area of reactor internals dynamic analysis and testing was updated for prototype, limited prototype and nonprototype reactors to be consistent with RG 1.20.
- Examples of ITAAC related to dynamic analysis and testing of reactor internals specified for the ESBWR and AP1000 design certifications were added as Appendices A and B, respectively.
- A pointer was added to RG 1.206.
- The review interface with SRP Section 3.9.3 was edited for clarity.
- A review interface was added to SRP Section 3.12.

II. ACCEPTANCE CRITERIA

- General Design Criterion 4 requirement was clarified to be consistent with the review scope of this SRP.
- A list of potential flow-induced resonance conditions was added to the piping dynamic test program.

- Referenced SRP Section 3.7.2 for the acceptance criteria for the number of discrete mass degrees of freedom in dynamic modeling.

- Clarified the sufficient number of modes in a dynamic analysis and referenced SRP Section 3.7.2.

- Updated the reactor internals dynamic analysis and testing for prototype, limited prototype and nonprototype reactors to be consistent with RG 1.20.

III. REVIEW PROCEDURES

- The review of piping seismic analysis was deleted.

- Clarified the review of the COL action items.

- Added the reviewed of submittals for replacement components or power uprates for operating reactors.

IV. IMPLEMENTATION

- None.

V. REFERENCES

- References were updated in concert with changes referenced above.