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 fluid flow (and especially loading caused by adverse flow conditions, such as flow instabilities over standoff pipes and branch lines in the steam system) and postulated seismic events. Compliance with the specific criteria guidance in subsection II of this SRP section will provide reasonable assurance of appropriate dynamic testing and analysis of systems, components, and equipment within the scope of this SRP section in conformance with 10 CFR 50.55a; 10 CFR Part 50 Appendix A, General Design Criteria (GDCs) 1, 2, 4, 14, and 15; 10 CFR 50 Appendix B; and 10 CFR 52.47(b) and 10 CFR 52.80(a). The specific areas of review are as follow:

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

   A. all American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 systems,

   B. other high-energy piping systems inside Seismic Category I structures (the term, "Seismic Category I," is defined in Regulatory Guide (RG) 1.29),

   C. high-energy portions of systems whose failure could reduce the functioning of any Seismic Category I plant feature to an unacceptable safety level, and

   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 plant are considered to be parts 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. The test program description should include a list of different flow modes, a list of selected locations for visual inspections and other measurements, the acceptance criteria, and possible 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, the acceptable methods and criteria are provided in ASME Code AG-1-1997.

   A. <u>Seismic Analysis Method</u>. For all Category I systems, components, equipment and their supports (including supports for conduit and cable trays, and ventilation ducts), and for certain non-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 consideration is given in the seismic dynamic analysis to maximum relative displacements between supports is reviewed. In addition, other significant effects that are accounted for 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 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 in general occur at the same time and these responses have to be combined according to some procedure selected to approximate or bound the response of the system. 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.

F. Analytical Procedures for Piping Systems. The analytical procedures applicable to seismic analysis of piping systems, including methods used to consider differential piping support movements at different support points located within a structure and between structures, are reviewed.

G. Multiply-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.

H. 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.

I. Torsional Effects of Eccentric Masses. The criteria and procedures that are used to consider the torsional effects of eccentric masses (e.g., valve operators) in seismic system analyses are reviewed.

J. Category I Buried Piping Systems. For Category I buried piping, the seismic criteria and methods which consider the effect of fill settlement, including pipe profile and pipe stresses, the movements at support points, penetrations, and anchors, are reviewed.

K. Interaction of Other Piping With Category I Piping. The seismic analysis procedures to account for the seismic motion of non-Category I piping systems in the seismic design of Category I piping are reviewed.

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

3. Dynamic responses of structural components within the reactor vessel caused by steady-state and operational flow transient conditions should be analyzed for prototype (first of a design) reactors. Generally, this analysis is also required from licensees requesting a power uprate for an existing power plant or steam generator replacement in a pressurized-water reactor (PWR). However, it is not required for non-prototypes except that segments of an analysis (in particular, assessments of any potential adverse flow effects) may be necessary if there are deviations from the prototype internals design or operating conditions, or if the non-prototype is based on a conditional prototype which has experienced problems in the past due to adverse flow effects. If the reactor internal structures are a non-prototype design, reference should be made to the results of tests and analyses for the prototype reactor and a brief summary of the results should be given. A more detailed summary of results associated with assessing the potential of any adverse flow effects should also be given.

Plant components, such as the steam dryer in a boiling water reactor (BWR) nuclear power plant, perform no safety functions 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. Therefore, the following structures in BWRs should also be included in the dynamic analysis :

- Chimney head* and steam separator assembly*
- Steam dryers assembly*
- Feedwater spargers*
- Standby liquid control header, spargers, and piping
- Reactor pressure vessel (RPV) vent assembly
- Sampling probes in feedwater, steam, and condensate systems

(* denotes non-safety related components.)

Similarly, for PWR nuclear power plants, the internal components of steam generators also must be included in the dynamic analysis.

The purpose of this analysis is to assess the vibration behavior of the components, including the definition of the input-forcing functions and estimation of the consequent vibration and stress levels. Before conducting the analyses, applicants/licensees should address the specific locations for calculated responses, the considerations for selecting the mathematical models and computer software, the interpretation of analytical and numerical results and concomitant bias errors and uncertainties, the acceptance criteria, and the methods for verifying predictions by means of tests.

The analyses should consider such various flow excitation mechanisms 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. 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 and/or structural resonances, causing high dynamic loads throughout the steam system and RPV. These "self-excited" loads are 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 which lead to adverse flow effects at all expected reactor operating conditions should be conducted by the applicant/licensee.

The following areas related to the dynamic response analysis are reviewed along with their bias errors and uncertainties.

A. <u>Results of vibration and stress calculations</u>. The calculated vibration and stress levels in reactor internal structures and in main steam line (MSL) valves are reviewed with their safety margins. The results for structures and components with a history of failures from adverse flow effects (like steam dryers and safety relief valves excited by flow instabilities over the openings of valve standoff pipes) are given greater scrutiny.

   The dynamic properties of internal structures, including natural frequencies, mode shapes relevant to the vibration and stress response, damping factors, and frequency response functions (FRFs) are reviewed.

   Any potential adverse flow conditions which lead to self-excited response (where structural and/or acoustic vibration couples to the forcing function, increasing its amplitude) are reviewed with greater scrutiny, particularly for conditions which have led to failures in the past.

B. <u>Transient and steady-state flow-induced forcing functions</u>. Forcing functions within the reactor vessel and within the feedwater and steam systems (*e.g.*, those induced by flow around the sampling probes in the feedwater piping and over the standoff pipes of valves in the MSLs are reviewed along with the method(s) for specifying the forcing functions (analytic or numerical tools like computational flow dynamics (CFD) models, test-analysis combination methods like Scale Model Testing, and response deduction methods). Any forcing functions caused by such adverse flow effects as flow instabilities over standoff pipe openings are reviewed with greater scrutiny.

C. <u>Methods for obtaining vibration and stress predictions</u>. 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 reviewed.

D. <u>Verification of predictions by comparison to test results</u>. The comparisons of predictions and test results are reviewed along with any use of the comparisons to substantiate uncertainties and bias errors in individual analysis components or the end-to-end analysis procedure.

4. Flow-induced vibration and acoustic resonance testing of reactor internals should be conducted during the preoperational and startup test program. Generally, this analysis is also required from licensees requesting a power uprate for an existing plant or steam generator replacement in a PWR. However, it is not required for non-prototypes except testing of some reactor internals may be necessary if there are substantial deviations from the prototype internals design or operating conditions. If the reactor internal structures are a non-prototype design, the applicant should refer to the results of tests and analyses for the prototype reactor and give a brief summary of the results.

   Plant components, such as the steam dryer in a BWR nuclear power plant, perform no safety functions 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. Therefore, for example, the following structures in BWRs also should be included in these tests:

   - Chimney head* and steam separator assembly*
   - Steam dryers assembly*
   - Feedwater spargers*
   - Standby liquid control header, spargers, and piping
   - RPV vent assembly
   - Sampling probes in feedwater, steam, and condensate systems

   (* denotes non-safety related components.)

   Similarly, for PWR nuclear power plants, the internal components of steam generators also must be tested against adverse flow effects (flow-induced vibration and acoustic resonance).

   The tests should consider such various flow excitation mechanisms as vortex-induced vibration, flow-excited acoustic resonance, fluid-elastic instability, turbulence buffeting as well as other flow excitations from flow separation, reattachment, and impinging flow instabilities. These mechanisms are often nonlinear and their adverse effects cannot be predicted by linear extrapolation of existing plant data.

   The purpose of this test is to demonstrate that adverse flow effects (caused by such mechanisms) similar to those expected during operation will not cause unanticipated flow-induced vibrations of significant magnitude or structural damage. The test program description should include a list of flow modes, a list of sensor types and locations, 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, and a comparison of the test results with the analytical predictions.

5. Dynamic system analyses should confirm the structural design adequacy and ability, with no loss of function, of the reactor internals and unbroken loops of the reactor coolant piping to withstand the loads from a loss-of-coolant accident (LOCA) in combination with the safe-shutdown earthquake (SSE). The staff review covers the methods of analysis, the considerations in defining the mathematical models, the descriptions of the forcing

functions, the calculational scheme, the acceptance criteria, and the interpretation of analytical results.

6. The methods should be described by an explanation of how they will be used 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 in particular under any adverse flow conditions. The methods should also be described by an explanation of how they will be used to correlate the results of scale model tests with those of analytical simulations or in-plant measurements.

   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 covers 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 any flow-excited acoustic or structural resonances should be reviewed carefully.

7. Inspection, Test, Analysis, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the applicant's proposed information on the ITAAC associated with the systems, structures, and components (SSCs) related to this SRP section is reviewed in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria - Design Certification." The staff recognizes that the review of ITAAC is performed after review of the rest of this portion of the application against acceptance criteria contained in this SRP section. Furthermore, the ITAAC are reviewed to assure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

8. COL Action Items and Certification Requirements and Restrictions. COL action items may be identified in the NRC staff's final safety evaluation report (FSER) for each certified design to identify information that COL applicants must address in the application. Additionally, DCs contain requirements and restrictions (*e.g.*, interface requirements) that COL applicants must address in the application. For COL applications referencing a DC, the review performed under this SRP section includes information provided in response to COL action items and certification requirements and restrictions pertaining to this SRP section as identified in the FSER for the referenced certified design.

Review Interfaces

The listed SRP sections interface with this section as follows:

1. Section 3.9.1: some of the computer programs used in the analyses addressed in this SRP section are reviewed. 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 II.3 of this SRP section.

2. Section 3.9.3: the designs of ASME Code Classes 1, 2, and 3 components, component supports, and core support structures are reviewed.

3. Section 3.9.5: the design of reactor vessel internal components is reviewed.

4. Section 3.10: the seismic qualification testing of Seismic Category I mechanical equipment is reviewed.

6. In addition, other evaluations that interface with the overall review of this SRP section are coordinated as follows:

   A. Section 4.4: Verification on request 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 that (ii) an acceptable hydraulic analysis has determined the loads acting on the reactor coolant system piping and the reactor internals.

   B. Section 3.6.3: review of applications that propose to eliminate consideration of design loads of the dynamic effects of pipe rupture.

   C. Sections 3.7.2 and 3.7.3: review of the applicant's determination of the number of earthquake cycles to be considered in Category I subsystem and component design, as well as the seismic system analysis.

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

II. ACCEPTANCE CRITERIA

Requirements

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

1. 10 CFR CFR 50.55a and General Design Criterion (GDC) 1 to 10 CFR 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 50, Appendix S, as they relate to systems, structures, and components 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 systems, structures, and components important to safety appropriately protected against the dynamic effects of discharging fluids.

4. GDC 14, as it relates to designing systems, structures, and components of the reactor coolant pressure boundary to have an extremely low probability of rapidly propagating failure and of gross rupture.

5. GDC 15, as it relates to designing the reactor coolant system with sufficient margin to assure that the reactor coolant pressure boundary 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 systems, structures, and components.

7. 10 CFR 52.47(a)(1)(vi), as it relates to ITAAC (for design certification) sufficient to assure that the SSCs in this area of review will operate in accordance with the certification.

8. 10 CFR 52.97(b)(1), as it relates to ITAAC (for combined licenses) sufficient to assure that the SSCs in this area of review have been constructed and will be operated in conformity with the license and the Commission'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 review described in Subsection I of 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 code and to confirm that no unacceptable restraint of normal thermal motion occurs.

   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 like pump trips, valve closures, etc. to which the components will be subjected during the test. (For additional guidance see RG 1.68). For example, the transients of the reactor coolant system heatup tests should include but not necessarily be limited to:

      (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 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 show that the stress and fatigue limits are within the design levels should be provided.

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

E. A description of the thermal motion monitoring program (*i.e.*, verification of snubber movement, adequate clearances and gaps, including acceptance criteria and how motion will be measured).

F. If vibration is noted beyond the acceptance levels set by the criteria of Item II.1.C 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 measured at those stations indicated in Item II.1.D of the acceptance criteria, 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 described in subsection I.2 of this SRP section are given below. Other approaches which can be 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. <u>Seismic Analysis Methods</u>. The seismic analysis of all 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) <u>Dynamic Analysis Method</u>. A dynamic analysis (*e.g.*, response spectrum method, time history method, etc.) should be used when the use of the equivalent static load method cannot be justified. To be acceptable such analyses should consider the following items:

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

(2) Use of an adequate number of masses or degrees of freedom in dynamic modeling to determine the response of all Category I and applicable non-Category I systems and plant equipment. The number is adequate when additional degrees of freedom do not result in more than a 10-percent increase in responses. Alternately, the number of degrees of freedom may be taken as equal to twice the number of modes with frequencies less than 33 Hz.

(3) Investigation of a sufficient number of modes to ensure participation of all significant modes. The criterion for sufficiency is that the inclusion of additional modes does not result in more than a 10-percent increase in responses.

(4) Consideration of maximum relative displacements among supports of 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 submitted 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 which 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 with adequate justification.

In addition, for equipment which 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 preferably should be less than ½ 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 adopted 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 codirectional responses caused by each of the three components of earthquake motion at a particular point of the structure or of the mathematical model.

(ii) Time History Analysis Method. When the time history analysis method is employed for seismic analysis, two types of analysis are generally performed depending on the complexity of the problem. (1) to obtain maximum responses to each of the three components of the earthquake motion: in this case 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 spectrum method. (2) To 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 in this case can be obtained from the combined time solution. When this method is used, to be acceptable 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,"Combining Modal Responses and Spatial Components in Seismic Response Analysis," present criteria and guidance for modal response combination methods acceptable to the staff.

F. Analytical Procedures for Piping Systems. The seismic analysis of Category I piping may use either a dynamic analysis or an equivalent static load method. The acceptance criteria for the dynamic analysis or equivalent static load methods are described in subsection II.2.A of this SRP section.

G. Multiply-Supported Equipment and Components 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 quite different.

A conservative and acceptable approach for equipment items supported at two or more locations is to use an upper-bound envelope of all the individual response spectra for these locations to calculate maximum inertial responses of multiply-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:

$$S_d = S_a \, g \, / \, \omega^2$$

where S_a is the spectral acceleration in "g's" 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 most unfavorable 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 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 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 multiply-supported systems.

H. <u>Use of Constant Vertical Static Factors</u>. The use of constant vertical load factors as vertical response loads for the seismic design of all Category I systems, components, equipment, and their supports in lieu of a vertical seismic system dynamic analysis is acceptable only if the structure is demonstrably rigid in the vertical direction. The criterion for rigidity is that the lowest frequency in the vertical direction be more than 33 Hz.

I. Torsional Effects of Eccentric Masses. For Seismic Category I systems, if the torsional effect of an eccentric mass like a valve operator in a piping system is judged to be significant, the eccentric mass and its eccentricity should be included in the mathematical model. The criteria for significance will have to be determined case by case.

J. Category I Buried Piping Systems. For Category I buried piping systems, the following items should be considered in the analysis:

(i) The inertial effects due to an earthquake upon buried piping systems should be adequately considered in the analysis. Use of the procedures described in the references is acceptable.

(ii) The effects of static resistance of the surrounding soil on piping deformations or displacements, differential movements of piping anchors, bent geometry and curvature changes, etc., should be adequately considered. Use of the procedures described in the references is acceptable.

(iii) When applicable, the effects of local soil settlements, soil arching, etc., also should be considered in the analysis.

K. Interaction of Other Piping with Category I Piping. To be acceptable, each non-Category I piping system should be designed to be isolated from any Category I piping system by either a constraint or barrier or should be located remotely from the seismic Category I piping system. If isolation of the Category I piping system is not feasible or practical, adjacent non-Category I piping should be analyzed according to the same seismic criteria applicable to the Category I piping system. For non-Category I piping systems attached to Category I piping systems, the dynamic effects of the non-Category I piping should be simulated in the modeling of the Category I piping. The attached non-Category I piping, up to the first anchor beyond the interface, also should be designed not to cause a failure of the Category I piping during an earthquake of SSE intensity.

L. Criteria Used for Damping. RG 1.61,"Damping Values for Seismic Design of Nuclear Power Plants," provides acceptable values which may be used. 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 DG-1163 (Revision 3 of RG 1.20) “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing”, apply to the analytical solutions to predict vibrations of reactor internals for prototype plants. Generally, this analysis is required only for prototype designs and power uprate of existing plants; However, it is not required for non-prototypes except that segments of an analysis (in particular, assessments of any potential adverse flow effects) may be necessary if there are deviations from the prototype internals design or operating conditions or if the non-prototype is based on a conditional prototype which has experienced problems from

adverse flow effects. If the reactor internal structures are a non-prototype design, the applicant should refer to the results of tests and analyses for the prototype reactor and give a brief summary of the results. A more detailed summary of results of assessment of the potential of any adverse flow effects also should be given.

A. The results of vibration and stress calculations should consist of the following:

(i) Dynamic responses to operating transients at critical locations of the internal structures should be determined and, in particular, at the locations where vibration sensors will be mounted on the reactor internals. For each location, the maximum response, the modal contribution to the total response, and the response causing the maximum stress amplitude should be calculated.

(ii) The damping factors for different modes should be properly selected and substantiated. In prior submissions, utilities have cited NRC damping guidance for very low frequency seismic analyses as justification for high damping factors for mid-to-high frequency analyses. DG-1163 corrects this guidance and requires that damping factors used in structural dynamic modeling be based on mid- to high-frequency measurements or rigorous analyses conducted on structures typical of the reactor internal structure modeled.

(iii) The dynamic properties of internal structures, including the natural frequencies and shapes of the dominant modes, should be characterized. In analyses of a component structural element basis, the presence of dynamic coupling among component structure elements should be investigated. Upper bounds on the uncertainties of all natural frequencies of the relevant resonance modes should be provided. The uncertainties and bias errors of the amplitudes of the frequency response functions (FRFs) also should be provided. 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 performance of hammer tests would be expected for replacement steam dryers.

(iv) Dynamic responses of reactor internals to self-excited flow oscillations should be estimated. The applicants/licensees should analyze in detail adverse flow effects generated by various excitation mechanisms like vortex-induced vibration flow-excited acoustic resonance, fluid-elastic instability, and other flow instabilities (*e.g.*, separated and impinging flow instabilities). These mechanisms may be assessed by theoretical, numerical, or experimental techniques, including scale model testing. 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. DG-1163, Section C.2.1.3 provides more guidance on self-excited flow instabilities.

(v) The dependance of the dynamic response on hydrodynamic excitation forces like coolant recirculation pump frequencies and the flow path configuration should 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 error and uncertainty analysis.

(vi) Acceptance criteria should be established for allowable responses and for the location of vibration sensors. Such criteria relate to the code-allowable stresses, strains, and limits of deflection established to preclude loss of function of the reactor core structures and fuel assemblies.

B. The forcing functions should account for the effects of transient flow conditions and the 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 DG-1163, Section C.2.1.3 for more guidance on self-excited flow instabilities). Acceptable methods for formulating forcing functions for vibration prediction include the following:

(i) Analytical method: based on standard hydrodynamic theory, the governing differential equations for vibratory motions should be developed and solutions obtained with appropriate boundary conditions and parameters. This method is acceptable where the geometry along the fluid flow paths is mathematically tractable.

(ii) Test-analysis combination method: based on data obtained from plant or scale model tests (*e.g.*, velocity or pressure distribution data), forcing functions should be formulated to include the effects of complex flow path configurations and wide variations of pressure distributions. The suitability of any 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.

(1) Scale Model Tests (SMTs): If SMTs are used to develop forcing functions, the following areas should be considered.

(a) The scale model should be dynamically similar to the prototype. The dynamic similarity should cover all fluid, structural (such as piping dimensions and elbow locations), and acoustic parameters relevant to the phenomenon considered. If some distortions in the dimension-less parameters of the scale model should be made, the applicants/licensees should show that these distortions are conservative. As an example, sound attenuation in scale models is normally substantially higher than that of the prototype due to viscous heat conduction and other losses higher in small-size models tested at low pressures, leading to the requirement that the scale model size and its test pressure be sufficiently large to ensure the re-production of such specific flow phenomena as flow-induced vibration and acoustic resonance present in the prototype.

(b) The effects of structural damping and sound attenuation (in the test medium) on the loading function measured in the scale model should be considered carefully. Any non-conservative deviations in these parameters from those of the prototype reactor should be corrected when the loading function is scaled to that of a full-size reactor pressure vessel (RPV).

(c) The conservative simulation of boundary conditions in the scale model.

(d) Whether the size of the scale model is sufficiently large to allow investigation of small relevant details in geometry (*e.g.*, branch line openings).

(e) Validation of the SMT results by measurements in nuclear power plants.

(2) CFD: If CFD simulations are used to develop unsteady forcing functions, the following areas should be considered.

(a) Include acoustic/vibration coupling to simulate enhancement of flow instabilities (if any).

(b) Grid size sensitivity tests.

(c) The Courant number requirement should be met.

(d) There should be unsteady simulations using Large Eddy Simulation (LES) or Direct Numerical Simulation (DNS) at high Reynolds number flow and including compressibility effects to model any coupling of the flow with the acoustic waves in the fluid (self-excitation or lock-in effects).

(e) Real gas simulation should be used (*i.e.*, use state equation of steam as real gas).

(f) The simulation procedures should be validated on similar (*i.e.*, complex and high Reynolds number) flow situations.

(3) Acoustic Modeling of Steam System: If an acoustic model of the steam system (the steam within the MSLs and the RPV) computes

fluctuating pressures within the RPV and on BWR steam dryers inferred from measurements of fluctuating pressures within the MSLs connected to the RPV, the following areas should be considered.

(a) There should be at least two measurement locations on each MSL in a BWR; however, three measurement locations on the MSLs improve input data to an acoustic model, particularly if the locations are spaced logarithmically, reducing uncertainty in describing the waves coming from and going into the RPV. With two or three measurement locations, there should be no acoustic sources between the measurement locations, unless justified.

(b) Strain gages (at least four gages circumferentially oriented and placed at equal distance along the circumference) may be used to relate the hoop strain in the MSL to the internal pressure. Strain gages should be calibrated according to the MSL dimensions (diameter, thickness, and static pressure). Alternatively, pressure measurements made with transducers flush-mounted against the MSL internal surface may be used. The effects of flow turbulence on any direct pressure measurements should be considered, however.

(c) The speed of sound in any acoustic models should not be changed from plant to plant but rather be a function of temperature and steam quality.

(d) Reflection coefficients at any boundary between steam and water should be based on rigorous modeling or on direct measurement. The uncertainty of the reflection coefficients should be clearly defined.

(e) Any sound attenuation coefficients should be a function of steam quality (variable between the chimney and reactor dome) rather than constant throughout a steam volume (like the volume within the RPV).

(f) Once validated, the same speed of sound, attenuation coefficient, and reflection coefficient should be used in other plants; however, different flow conditions (temperature, pressure, quality factor) may require adjustments of these parameters.

(4) Response-deduction method: based on a derivation of response characteristics from plant or SMT data, forcing functions should be formulated; however, as such functions may not be unique and are also expected to depend on material properties and loss factors, the

computational procedures and the basis for selection of the representative forcing functions should be described together with all bias errors and uncertainties (see subsection II.3.B.(ii)(1) of this SRP section, “Scale Model Tests,” for guidelines on inferring forcing functions from plant or scale model testing data).

C. Acceptable methods of obtaining dynamic responses for vibration and stress predictions are as follows:

(i) If a numerical model is used to compute mode shapes and FRFs, the modeling approach should be documented along with the model itself. Uncertainties and bias errors for both the approach and the specific model should be provided along with their bases. Additional guidance on numerical uncertainties and bias errors can be found in DG-1163.

(ii) Force-response computations are acceptable if the characteristics of the forcing functions are predetermined conservatively and the mathematical model of the reactor internals is appropriately typical of the design.

(ii) If the forcing functions are not predetermined, either a special analysis of response signals measured from reactor internals of similar design may predict amplitude and modal contributions or parameter studies useful for extrapolating the results from tests of internals or components of similar designs based on composite statistics may be used. The latter approach should be used only when the expectation that flow-induced vibration or acoustic resonance will not occur for the operation conditions covering the extrapolated range of the forcing functions is shown beyond doubt.

D. Vibration predictions should be verified by RPV, steam, feed water and condensate piping, and safety relief valve test results. This procedure should consider all sources of bias errors and uncertainties. If the test results differ substantially from the predicted response behavior, the vibration analysis should be modified appropriately for more 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.

4. For requirements of GDCs 1 and 4, the preoperational vibration and stress test program for the internals of a prototype reactor, for existing reactors under consideration for power uprate, and for non-prototype reactors whose valid or conditional prototypes have experienced structural failures due to adverse flow effects in any plant (*e.g.*, steam dryer cracking and valve failures) should conform to the requirements for a prototype test as specified in DG-1163, including vibration prediction, vibration monitoring, adverse flow effects (flow-induced acoustic and structural resonances, data reduction, bias errors and uncertainty analysis, and walkdown and surface inspections. The test program to demonstrate design adequacy of the reactor internals should include, but not necessarily be limited to, the following:

A. 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.

B. The vibration monitoring instrumentation should be described briefly, 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.

C. Testing to evaluate potential adverse flow effects on reactor internal components should include the steam dryer and MSL valves. The instrumentation directly mounted on the steam dryer should include pressure sensors, strain gages, and accelerometers. The MSLs also should be instrumented to collect data to determine steam pressure fluctuations to identify the presence of flow-excited acoustic resonances and to allow the analysis of those pressure fluctuations to calculate MSL valve loading and vibration and steam dryer loading and stress. Accelerometers should be mounted on the main steam valves to record the presence and the level of any flow-excited acoustic resonance or vibration.

D. The planned duration of the test for the normal operation modes to ensure that all critical components are subjected to at least 10^6 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.

E. Testing should include all of the flow modes of normal operation and upset transients. The proposed set of flow modes is acceptable if it provides a conservative basis for determining the dynamic response of the tested components and is reviewed on request. The power ascension program for startup testing should include specific hold points 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. Further information on test procedure is addressed in DG-1163.

F. The methods and procedures to process the test data for meaningful interpretation of the vibration 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 detailed analysis of bias errors and uncertainties of instrumentation, data acquisition systems, and models to estimate loading functions from the measured data.

G. Vibration predictions, test acceptance criteria and bases, and permissible deviations from the criteria should be provided before the test.

H. The applicant/licensee is expected to provide a summary evaluation of plant startup and power ascension to the staff within 90 days of plant startup. If full licensed power is not achieved in that time period, the applicant/licensee is expected to provide a supplemental report within 30 days after achieving full licensed power.

I. There should be walkdown inspections during 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.

For later reactor internals with the same design, size, configuration, and operating conditions as the prototype, the vibration test program should comply with the requirements of the appropriate non-prototype program as specified in DG-1163.

5. For requirements of GDCs 2, 4, 14, and 15 dynamic system analyses should confirm the structural design adequacy of the reactor internals and the reactor coolant piping (unbroken loops) to withstand the dynamic loadings of the most severe 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.

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 investigated.

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.

   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. Comparison of the measured and predicted damping factors as a function of natural frequencies for validation of the damping assumed in the analysis.

   B. Comparison of the analytically obtained mode shapes with the shape of measured motion for identification of the modal combination or verification of a specific mode.

   C. Comparison of the response amplitude time variation and the frequency content from test and analysis for verification of the postulated forcing function.

   D. Comparison of the measured amplitudes, frequencies, and time variations of loads with those predicted by test-analysis combination method for validation of the predicted forcing function.

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

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

G. Comparison of measurements and predictions of any adverse flow phenomena (*e.g.*, flow-excited acoustic and/or structural resonances) for validation of the model(s) predicting the loading induced by the phenomena.

7. For new applications, 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."

Technical Rationale

The technical rationale for application of these requirements to reviewing 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 DG-1163 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 provide 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 systems and components 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 systems and components important to safety. The tests, analyses, and comparisons are in accordance with sound engineering practices and provide assurance that these systems and components are designed to withstand natural phenomena in combination with normal and accident conditions.

   Compliance with the requirements of GDC 2 provide assurance that systems and components within the scope of this SRP section are capable of performing their intended safety functions.

3. GDC 4 requires that the nuclear power plant systems and components 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 systems and components 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 components and systems to withstand anticipated loads is also described.

   Compliance with the requirements of GDC 4 provide assurance that systems and components within the scope of this SRP section are capable of performing their intended safety functions.

4. GDC 14 requires that the reactor coolant pressure boundary 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 reactor coolant pressure boundary to ensure that they will withstand the applicable design-basis seismic and dynamic loads in combination with other environmental and natural phenomena loads without leakage, rapidly propagating failure, or gross rupture.

   Compliance with the requirements of GDC 14 provide assurance that the reactor coolant pressure boundary will have an extremely low probability of leakage or failure.

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

   Staff positions are described in SRP Section 3.9.2 on design of the reactor coolant pressure boundary to resist seismic, LOCA, and other appropriate environmental loads individually and in combination. Dynamic analyses are described to confirm the structural design adequacy of the reactor coolant pressure boundary. Vibration, thermal expansion, and dynamic effects testing are also described to verify the design.

   Compliance with the requirements of GDC 15 provide assurance that the reactor coolant pressure boundary will remain intact, thus preventing the spread of radioactive contamination.

III. <u>REVIEW PROCEDURES</u>

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review specified in subsection I of this SRP section, the review procedure is identified below. These review procedures are based on the identified SRP acceptance criteria. For deviations from these specific acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the relevant NRC requirements identified in subsection II.

GDCs 1, 2, 4, 14, and 15 state that all SSCs important to safety should be designed and tested to perform safety functions in operational transients, earthquakes, and LOCA loadings.

For new applications, 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."

Under these GDCs, the staff reviews the treatment of dynamic responses of safety-related piping systems and reactor internal structures by the following procedures:

1. During the construction permit (CP) stage, the preliminary safety analysis report (PSAR) is reviewed for whether the applicant has made a commitment 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 all the elements of an acceptable program as described in 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 commitment is fulfilled and the program is developed in sufficient detail. The reviewer should be assured 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 needed) for monitoring or measuring pipe motions.

   D. Ensure that the staff will be provided documentation of any corrective action from the test and confirmation by additional testing to substantiate the effectiveness of the corrective action.

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

   A. <u>Seismic Analysis Methods</u>. For all 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 for whether the techniques are in accordance with the acceptance criteria in 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). This assumption allows neglect of the influence of the anchorage system stiffness on the dynamic response. In some cases, particularly for heavy equipment, this assumption potentially can cause under-estimation of seismic loadings. For new applications, the reviewer should verify whether 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 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 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 II.2.E of this SRP section when a response spectrum modal analysis method is used.

F. Analytical Procedures for Piping Systems. For all Category I piping and applicable non-Category I piping, the methods of seismic analysis (response spectra, time history, equivalent static load) are reviewed for techniques in accordance with the acceptance criteria of subsection II.2.F of this SRP section. Typical mathematical models are reviewed to judge whether all significant degrees of freedom have been included.

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

H. Use of Constant Vertical Static Factors. Use of constant static factors as response loads in the vertical direction for the seismic design of any 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 II.2.H of this SRP section.

I. Torsional Effects of Eccentric Masses. The procedures for seismic analysis of Category I piping systems are reviewed for compliance with the acceptance criteria of subsection II.2.I of this SRP section.

J. Category I Buried Piping Systems. The analysis procedures for Category I buried piping are reviewed for accordance with the acceptance criteria of subsection II.2.J of this SRP section, including review of the procedures for considering the effect of fill settlement, including pipe profile and pipe stresses, and the differential movements at support points, penetrations, and anchors. For any procedures not adequately justified additional justification is requested from the applicant.

K. Interaction of Other Piping with Category I Piping. The criteria for design of the interfaces between Category I and non-Category I piping are reviewed for compliance with the acceptance criteria of subsection II.2.K of this SRP section.

L. 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 II.2.L of this SRP section.

3. At the CP stage, the applicant should commit to an analysis of the vibration of such reactor internal structures as those listed in subsection I.3 of this SRP section if designated as a prototype design. The vibration analysis should consider adverse flow effects from possible flow-induced vibrations and acoustic resonances. The methods and procedures for the analysis should be described.

At the OL stage, there should be a detailed dynamic analysis for a prototype design for vibration prediction prior to the performance of preoperational 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 addition, the analysis is verified by correlation with the test results when available.

For both CP and OL stages, applicants for extended power uprate of existing nuclear power plants should commit to a vibration analysis for the requested power uprate.

For both CP and OL stages, for reactor internal structures of non-prototype design, the applicant should refer to the reactor prototypical of the reactor reviewed with a brief summary of test and analysis results for the prototype. Alternatively, the information may be in another document (*e.g.*, a topical report) to which the applicant should refer.

4. At the CP stage, review of the program for preoperational vibration testing of reactor internals for flow-induced vibrations includes the following:

A. The applicant should clarify the intention to perform either a prototype or non-prototype test.

B. If the plant is designated as a prototype, or the plant is reviewed for extended power uprate, there should be a brief description of the preoperational vibration test program. The staff review will be based on compliance of this program with the requirements of subsection II.4 (Acceptance Criteria) of this SRP section.

C. If the plant is a non-prototype, the applicant should refer to the prototype plant of similar design. The staff reviews the validity of the designated prototype, including any differences in the flow conditions or the design of reactor internal structures, from the prototype plant 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 required to complete the review. In addition, the applicant should commit to the prototype test if timely adequate test results are not obtained for the designated prototype.

At the OL stage, the staff review includes the following procedures:

A. A detailed preoperational vibration test program and the tentative schedule for the test are reviewed. If elements of the program differ substantially from the guidelines specified in Regulatory Guide 1.20, there should be discussion of the need and justification for the differences. On request, the reviewer verifies whether the flow modes are acceptable.

B. For a prototype plant and plants reviewed for extended power uprate, the review includes 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 non-prototype plant, the staff verifies the applicability of the designated prototype, including the design and operating condition similarities of the reactor internal structures to those of the prototype. Additional detailed analysis, scaled model tests, 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 and unbroken loops of the reactor coolant piping under faulted condition loadings, the applicant should commit to this analysis or identify the applicable document, usually a topical report, with the required information. The scope and methods of analysis should be described briefly.

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 and unbroken loops 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 and piping to perform needed safety functions. On request from primary reviewer, the reviewer verifies whether an acceptable hydraulic analysis has been used.

6. The program that the applicant has committed to implement as part of the preoperational test procedure is reviewed principally to correlate the test measurements with the analytically predicted flow-induced dynamic response of the reactor internals. The applicant's statements in this area are reviewed for a commitment to submit a timely

report. The report should 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 reviews of DC and COL applications under 10 CFR Part 52, the reviewer should follow the above procedures to verify that the design set forth in the safety analysis report, and if applicable, site interface requirements meet the acceptance criteria. For DC applications, the reviewer should identify necessary COL action items. With respect to COL applications, the scope of the review is dependent on whether the COL applicant references a DC, an ESP, or other NRC-approved material, applications, and/or reports.

   After this review, SRP Section 14.3 should be followed for the review of Tier I information for the design, including the postulated site parameters, interface criteria, and ITAAC.

IV. EVALUATION FINDINGS

The 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 (SER). 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 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 Category I systems, components, equipment, and their supports and procedures for modeling, inclusion of torsional effects, seismic analysis of Category I piping systems, seismic analysis of multiply-supported equipment and components with distinct inputs, justification for the use of constant vertical static factors, and determination of composite damping. The review has included design criteria and procedures for evaluation of the interaction of non-Category I with Category I piping. The review has also included criteria and seismic analysis procedures for reactor internals and Category I buried piping outside containment.

The system analyses are performed by the applicant on a elastic basis. Modal response spectrum, multi-degree of freedom, and time history methods form the bases for the analyses of all major 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 DG-1163 for the conduct of preoperational vibration tests by a preoperational 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 provide adequate assurance that the reactor internals will, during their service lifetime, withstand the flow-induced vibrations 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 and unbroken piping loops to withstand the combined dynamic loads of postulated LOCA and SSE and (for a BWR) the combined loads of a postulated main steam line rupture and SSE. The analysis provides adequate assurance that the combined stresses and strains in the components of the reactor coolant system and 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 the FSAR, the review should justify a similar finding with the phrase "will be implemented" modified as "has been implemented."

For DC and COL reviews, the findings will also summarize (to the extent that the review is not discussed in other SER sections) the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable, and interface requirements and combined license action items relevant to this SRP section.

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 docketed six months or more after the date of issuance of this SRP section, unless superceded by a later revision.

VI. REFERENCES

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

2. RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing."

3. RG 1.29, “Seismic Design Classification.”

4. RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."

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

6. RG 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. ANSI S2.31-1979 (R2004), Methods for the Experimental Determination of Mechanical Mobility, Part 1: Basic Definitions and Transducers.

9. ANSI S2.32-1982 (R2004), Methods for the Experimental Determination of Mechanical Mobility, Part 2: Measurements Using Single-Point Translational Excitation.

10. ASME AG-1-1997, “Code on Nuclear Air and Gas treatment.”

11. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

12. ASME OM-S/G-2000, "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," including the addenda and Part 7, "Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems," ASME.

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

16. M. Hetenyi, "Beams on Elastic Foundation," The University of Michigan Press (1946).

17. R. P. Kassawara, 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. N. M. Newmark, "Earthquake Response Analysis of Reactor Structures," Nuclear Engineering and Design, Volume 20, pp. 303-322 (1972).

19. N. M. Newmark, 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. N. M. Newmark and E. Rosenblueth, "Fundamentals of Earthquake Engineering," Prentice Hall, (1971).

SRP Section 3.9.2
Description of Changes

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in [Draft] Revision 3 dated June 1996, of this SRP section. See ADAMS accession number ML052070336

In addition, this SRP section was administratively updated in accordance with NRR Office Instruction LIC-200, Revision 1, “Standard Review Plan (SRP) Process.” The revision also adds standard paragraphs to extend application of this updated SRP section to prospective applicant submissions pursuant to 10 CFR Part 52.

The technical changes are incorporated in Revision 3, [Month] 2007.

Review Responsibilities - Reflects changes in review branches resulting from reorganization and branch consolidation. Change is reflected throughout the SRP.

I. AREAS OF REVIEW

1. Added a note to clarify that supports for conduit and cable trays, and ventilation ducts are included in the areas of review.

2. Added 10 CFR 52.47(a)(1)(vi) and 52.97(b)(1) to the list of regulations to be complied to.

3. Added a note in subsection I.2 that the methods and criteria for seismic qualification testing of seismic Category I mechanical equipment are provided in SRP Section 3.10.

4. Provided a reference for the design of nuclear air and gas treatment systems and components.

5. Added a note in subsection I.2.A to reflect that all items so evaluated are Category I items and non-Category items required to be designed to seismic criteria.

6. Essentially, the text of subsection I.2.B is already in accordance with the suggested resolution of Integrated Impact No. 211 (e.g., refer to SRP Section 3.7.3 for guidance). Modified the wording of the subsection for clarification. Deleted the phrase “Category I”“ in the sentence because it is too restrictive. There may be some non-Category I items or interfacing items (see RG 1.29) that should be included.

7. Added subsection I.7 on Inspection, Test, Analysis, and Acceptance Criteria (ITAAC) for design certification and combined license reviews.

8. Added “Review Interface”“ to describe the interfacing SRP sections and the review responsibility of interfacing review branches.

9. Added a Review Interface with SRP Section 3.6.3 to address the review of leak-before-break in excluding consideration of dynamic effects of pipe rupture from the design basis (reference Integrated Impact 214)

10. Added a Review Interface with SRP Sections 3.7.2 and 3.7.3 which are cited throughout SRP Section 3.9.2 with regard to the methods and criteria of seismic system and subsystem analyses.

II. ACCEPTANCE CRITERIA

1. Added discussion of 10 CFR 52.47(a)(1)(vi) and 10 CFR 52.97(b)(1) as they relate to ITAAC for the review of design certification and combined license applications.

2. Added a footnote to state that SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to evaluate how any proposed alternatives to the SRP acceptance criteria provide an acceptable method of complying with the NRC regulations.

3. Deleting existing text in subsection II.2.B and referred to SRP subsection 3.7.3, as suggested in Integrated Impact No. 211. The revised text was copied from the revised text of subsection I.2.B.

4. Deleted existing text in subsection II.2.E that provides information that duplicates the information that appears in RG 1.92, but may be in conflict because of somewhat different wording. Direct citation of RG 1.92 is appropriate, as noted in the subsection and in SRP Sections 3.7.2 and 3.7.3.

5. Added a phrase at the end of subsection II.2.E to indicate that RG 1.92 guidance is acceptable to the staff. Also added reference to SRP Sections 3.7.2.

6. Added a note in subsection II.2.G to reflect additional methods (e.g., NUREG-1061 methods described as acceptable in SRP Section 3.7.3) for evaluation of multiple support arrangements in SRP Sections 3.7.2 and 3.7.3 (ref. Integrated Impact No. 211).

7. Added a note in subsection II.2.G to reflect acceptable alternatives described in SRP Section 3.7.3.

8. Deleted the reference to ASME Code Case N-411, and noted that the methods for analysis of damping should be consistent with those described in SRP Section 3.7.2.

9. Made editorial changes to correct interface branch names in subsections II.4.D and II.5, and throughout this SRP section.

10. Since the recommendation of Integrated Impact No. 214 was not implemented in detail in this SRP subsection of II.5, added reference to SRP Section 3.9.5 to reflect criteria and reviews related to asymmetric blowdown loadings on PWR reactor internals.

11. Added “”Technical Rationale”“ to describe the bases for application of the acceptance criteria of this SRP section to reviewing the dynamic testing and analysis of systems, components, and equipment. Added technical rationales for compliance with each of the regulatory requirements.

III. REVIEW PROCEDURES

1. Added a paragraph in subsection III to correlate review procedures with areas of review in subsection I and acceptance criteria in subsection II. Made a note on the staff review procedure for alternative criteria.

2. Added a paragraph in subsection III to cite Parts 3 and 7 of ASME OM-S/G-1990 (Reference Integrated Impact No. 209).

3. Revised review procedures to agree with updates in subsection II.

4. Added a paragraph in subsection III.2.A to reflect the importance of assumptions made with regard to the stiffness of the seismic subsystem anchorage in the seismic analyses.

5. Added subsection III.7 for performing the review under 10 CFR Part 52.

IV. EVALUATION FINDINGS

1. Revised pertinent portions of evaluating findings to agree with updates in subsections II and III.

2. Added a paragraph at the end of subsection IV to reflect reviews for the design certification and combined license applications.

V. IMPLEMENTATION

1. Made editorial changes to capture applicability of this SRP section to 10 CFR Part 52 and time frame in which SRP update goes to effect.

VI. REFERENCES

1. Renumbered per SRP update format.

2. Added new references as cited in the updated SRP.

3. At the end of the SRP section, added “Paperwork Reduction Statement” and “Public Protection Notification.