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

A. INTRODUCTION

General Design Criterion (GDC) 1, "Quality Standards and Records," as set forth in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR Part 50), "Domestic Licensing of Production and Utilization Facilities" (Ref. 1), requires that nuclear power plant structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Toward that end, 10 CFR 50.34, "Contents of applications; technical information," requires applicants to provide a preliminary analysis and evaluation of the design and performance of the facility's SSCs, including the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of SSCs provided to prevent accidents and mitigate their consequences.

This guide describes a methodology that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in implementing the above requirements, as they relate to the internals of light-water-cooled nuclear power reactors during preoperational and initial startup testing.

As such, the staff considers this methodology acceptable to support reviews of applications that the agency expects to receive for new nuclear reactor construction permits or operating licenses under 10 CFR Part 50, as well as design certifications and combined licenses (COLs) that do not reference a standard design under 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants" (Ref. 2).

"Reactor internals," as used in this regulatory guide, comprise core support structures and adjoining internal structures. Specifically, those core support and internal structures are defined in Article NG-1120 of Section III, "Nuclear Power Plant Components," of the Boiler and Pressure Vessel Code (Ref. 3) promulgated by the American Society of Mechanical Engineers (ASME). For example, in boiling-water reactor (BWR) nuclear power plants, reactor internals might include the following components (* denotes non-safety-related components):

- chimney* and partitions*
- chimney head* and steam separator assembly*
- steam dryer assembly*
- feedwater spargers*
- standby liquid control (SLC) header and spargers and piping
- reactor pressure vessel (RPV) vent assembly
- core plate
- top guide
- control rod drive housing and guide tube
- orificed fuel support
- jet pump and support
- shroud and shroud support

In addition to guidance for vibration assessment of reactor internals, this regulatory guide provides helpful information on methods for evaluating the potential adverse effects from pressure fluctuations and vibrations in piping systems for both BWR and pressurized-water reactor (PWR) nuclear power plants. For example, in PWRs, potential adverse effects should be evaluated for the steam generator internals.

Consistent with Regulatory Guide 1.68.1, "Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling-Water Reactor Power Plants" (Ref. 4), "preoperational testing," as used in this guide, consists of those tests conducted prior to fuel loading, and "initial startup testing" refers to those tests conducted after fuel loading.

This guide does not cover inservice inspections and inservice monitoring programs to verify that the reactor internals have not been subjected to structural degradation as a result of vibration during normal reactor operation.

Although this regulatory guide is directed to new nuclear power plants, current licensees planning to propose a power uprate might also find the guidance herein to be helpful in establishing a power ascension testing program.

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

B. DISCUSSION

Reactor internals important to safety are designed to accommodate steady-state and transient vibratory loads throughout the service life of the reactor. This guide presents a comprehensive vibration assessment program that the NRC staff considers acceptable for use in verifying the structural integrity of reactor internals for flow-induced vibrations prior to commercial operation. The overall program includes individual analytical, measurement, and inspection programs. The term “comprehensive” appears in the title of the overall program to emphasize that the individual analytical, measurement, and inspection programs should be used cooperatively to verify structural integrity and to establish the margin of safety. For example, the analytical program should be used to provide theoretical verification of structural integrity, and should also be the basis for the choice of components and areas to be monitored in the measurement and inspection programs. Similarly, the measurement program should be used to confirm the analysis, but the program (i.e., data acquisition, reduction, and interpretation processes) should be sufficiently flexible to permit definition of any significant vibratory modes that are present but were not included in the analysis. In addition, the inspection program should be considered and used as a powerful tool for both quantitative (e.g., indicative of maximum total relative motion) and qualitative (e.g., establishing boundary conditions by inspection evidence at component interfaces) verification of the analytical and measurement program results.

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

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

In particular, Revision 1 of Regulatory Guide 1.20 expanded on the previous classifications (as defined in Regulatory Position 1 in Section C of this regulatory guide) and outlined an appropriate comprehensive vibration assessment program for each class. In general, under certain conditions, the expanded classifications and corresponding programs allow prototype reactor internals to be used as “limited prototypes” after they have experienced some adverse inservice vibration phenomena, and reactor internals may be used as “limited non-prototypes” if they are (in some respects) structurally dissimilar from the designated prototypes. Thus, the expanded classifications make the use of this guide compatible and consistent with design and operating experience.

Revision 2 of Regulatory Guide 1.20 retained the expanded classifications of Revision 1. However, Revision 2 included various changes in the corresponding vibration assessment programs and the reporting of results, which were made as a result of substantive public comments and additional staff review.

Revision 3 of Regulatory Guide 1.20 modifies the overall vibration assessment program for reactor internals, and summarizes expectations regarding the evaluation of potential adverse flow effects. Flow-induced excitation and pump-induced vibration should be addressed by the vibration assessment program described in this regulatory guide. This revision also includes changes to address COL applications or applications that do not reference a certified reactor design. This revision 3 does not modify the guidance in Revision 2, but does provide new guidance for steam dryers in BWR plants and helpful information for monitoring programs for plant components outside the reactor vessel.

Since adverse flow effects in reactors caused by flow-excited acoustic and structural resonances are sensitive to minor changes in arrangement, design, size, and operating conditions, even applications submitted for non-prototypes should include rigorous assessments of the potential for such adverse effects to appear. For any two nearly identical nuclear power plants, one may experience significant adverse flow effects, such as valve and steam dryer failures, while the other does not. Also, small changes in operating condition can cause a small adverse flow effect to magnify substantially, leading to structural failures. For example, severe acoustic excitation occurred in the steam system of one BWR nuclear power plant when flow was increased by 16 percent for extended power uprate (EPU) operation. Applicants and licensees may determine the need for detailed evaluations based on analysis results and available industry experience. Specific guidance for these assessments, both analytic and measurement-based, is also offered in this guide. The location of instrumentation should be evaluated for its potential impact on component and system performance.

Operating experience has revealed failures of steam dryers and main steam system components (including relief valves) in BWR nuclear power plants following implementation of EPUs. These failures have demonstrated the importance of detailed analysis of potential adverse flow effects on the RPV internal components, including the steam dryer, and main steam system components, such as safety relief valves. Studies of those failures have determined that flow-excited acoustic resonances (where instabilities in the fluid flow excite acoustic modes) within the valve stand pipes and branch lines in main steam lines (MSLs) can play a significant role in producing mid- to high-frequency pressure fluctuations and vibration. These can damage MSL valves, the steam dryer, and possibly other RPV internals and steam system components. In those failures, the instabilities of the separated shear flow over the standpipe openings actually "lock in" to the acoustic resonance of the fluid column within the standpipe. ("Lock-in" refers to a constructive feedback between the flow instability and the acoustic mode over the certain range of flow velocity, leading to strong amplification of the fluctuating pressures in the flow instability and acoustic mode.) In addition, hydrodynamic loading acting directly on the steam dryer and other RPV internals and steam dryer components can also produce flow-induced vibration, causing undesirable stresses that should be addressed. As a result, nuclear power plant licensees have sponsored scale model testing (SMT) and development of analytical models to evaluate potential adverse flow effects in support of power uprate operation.

A reliable evaluation of potential adverse flow effects on nuclear power plant components includes the proper consideration of bias errors and random uncertainties in the analysis. For example, bias errors may include the under-prediction of pressure loading, stress, strain, or acceleration when modeling plant performance. Random uncertainties may include the random error associated with measurement of plant parameters.

C. REGULATORY POSITION

In general, the NRC staff recommends that applicants use the classifications identified in Regulatory Position 1 (below) to categorize reactor internals according to design, operating parameters, and operating experience with potential prototypes. Applicants should then establish an appropriate comprehensive vibration assessment program using the guidelines specified in the succeeding regulatory positions, as they relate to the given classification(s). The comprehensive vibration assessment programs outlined in this guide are summarized in Figure 1.

1. Classification of Reactor Internals Relative to the Comprehensive Vibration Assessment Program

1.1 Prototype

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

1.2 Valid Prototype

A "valid prototype" is a configuration of reactor internals that has successfully completed a comprehensive vibration assessment program for "prototype" reactor internals and has experienced no adverse inservice vibration phenomena. A valid prototype that is subsequently modified in design (e.g., as in item 1.3 below) remains a valid prototype relative to its original design. If the prototype comprehensive vibration assessment program was conducted on a reactor outside the United States, the detailed results of the program should be included in any application related to a non-prototype submitted to the NRC for review, and should address all of the provisions in this regulatory guide.

1.3 Conditional Prototype

A "conditional prototype" is a "valid prototype" that has subsequently experienced adverse inservice vibration phenomena and, consequently, has been modified in arrangement, design, size, or operating conditions. Upon satisfying conditions described elsewhere in this guide, the conditional prototype serves as the reference design for non-prototype, Category III and IV reactor internals configurations. For any applications submitted for non-prototype, Category III and IV configurations, the detailed results of past comprehensive vibration assessment programs conducted on the conditional prototype should be submitted, and should focus on any specific adverse inservice vibration phenomena, and how they were mitigated.

1.4 Non-Prototype, Category I

"Non-prototype, Category I" reactor internals are those configurations that have substantially the same arrangement, design, size, and operating conditions as a specified "valid prototype," for which nominal differences in arrangement, design, size, and operating conditions have been shown (by test or analysis) to have no significant effect on the vibratory response and excitation of those reactor internals important to safety.

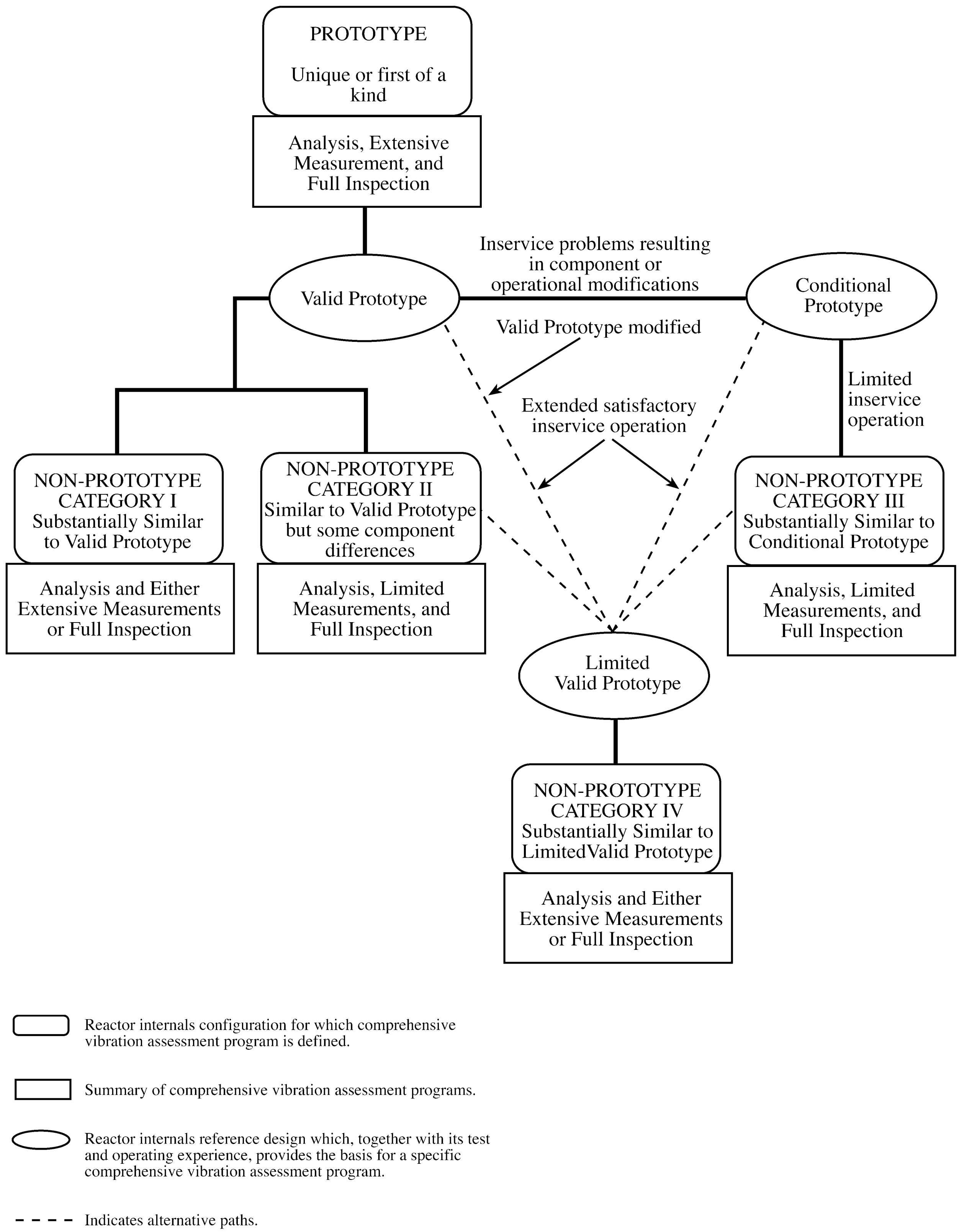

Figure 1. Summary of Comprehensive Vibration Assessment Programs

1.5 Non-Prototype, Category II

"Non-prototype, Category II" reactor internals are those configurations that have substantially the same size and operating conditions as a specified "valid prototype," but have some differences in component arrangement or design, and those differences have been shown (by test or analysis) to have no significant effect on the vibratory response and excitation of those unmodified reactor internals important to safety.

1.6 Limited Valid Prototype

A "limited valid prototype" is a non-prototype, Category II or III reactor internals configuration that has successfully completed the appropriate comprehensive vibration assessment program without experiencing any adverse inservice vibration phenomena. An operating "valid prototype" that has demonstrated extended satisfactory inservice operation subsequent to a design modification may be considered a "limited valid prototype" relative to the modified reactor internals configuration. Similarly, a "conditional prototype" that has demonstrated extended satisfactory inservice operation may also be considered a "limited valid prototype."

1.7 Non-Prototype, Category III

"Non-prototype, Category III" reactor internals are those configurations that have substantially the same arrangement, design, size, and operating conditions as a specified "conditional prototype," but have insufficient operating history to justify being designated as a "limited valid prototype," as defined below. Differences in arrangement, design, size, and operating conditions should be shown (by test or analysis) to have no significant effect on the vibratory response and excitation of those reactor internals important to safety.

1.8 Non-Prototype, Category IV

"Non-prototype, Category IV" reactor internals are those configurations that have substantially the same arrangement, design, size, and operating conditions as a specified "limited valid prototype," for which nominal differences in arrangement, design, size, and operating conditions have been shown (by test or analysis) to have no significant effect on the vibratory response and excitation of those reactor internals important to safety.

2. Comprehensive Vibration Assessment Program for Prototype Reactor Internals

Associated with the prototype and Category I, II, III, and IV non-prototype classifications are the comprehensive vibration assessment programs delineated in Regulatory Positions 2 and 3 and summarized in Figure 1. These five classifications are defined relative to the three prototype *reference design classifications* (i.e., valid prototype, conditional prototype, and limited valid prototype).

The comprehensive vibration assessment program should be implemented in conjunction with preoperational and initial startup testing. In addition, the comprehensive program should consist of a vibration and fatigue analysis, a vibration measurement program, an inspection program, and a correlation of their results.

Applicants proposing to construct and operate a new nuclear power plant, or licensees planning to request a power uprate for an existing power plant, should perform a detailed analysis of potential adverse flow effects (both flow-excited acoustic resonances and flow-induced vibrations) that can severely impact RPV internal components (including the steam dryer in BWRs) and other main steam system components, as applicable. The guidelines provided below for evaluating the potential adverse effects from pressure fluctuations and vibrations in piping systems should be considered for both PWRs and BWRs. For example, in PWRs these potential adverse effects should be evaluated for the steam generator internals. The applicant or licensee should determine applicable areas of the guidelines for its specific plant.

As previously noted, studies of past failures have determined that flow-excited acoustic resonances within the valves, stand-off pipes, and branch lines in the MSLs of BWRs can play a significant role in producing mid- to high-frequency pressure fluctuations and vibration that can damage MSL valves, the steam dryer, and other RPV internals and steam system components. In addition, hydrodynamic loading (flow-induced vibration) acting directly on the steam dryer and other RPV internals and steam dryer components can cause undesirable stresses that should be addressed.

The applicant/licensee should also evaluate the potential adverse effects from pressure fluctuations and vibration on piping and components in the applicable plant systems, including the reactor coolant, steam, feedwater, and condensate systems, up to full proposed operating conditions. Among others, these plant components include safety relief valves, power-operated valves, and sampling probes. Based on past experience, the applicant/licensee should pay particular attention to cantilevered piping and components. The applicant/licensee should make modifications to plant piping or components based on the results of this evaluation, as necessary, to reduce the pressure fluctuation and vibration levels such that all acceptance criteria are met.

2.1 Vibration and Stress Analysis Program

The applicant/licensee should perform a vibration and stress analysis for those steady-state and anticipated transient conditions that correspond to preoperational, initial startup test, and normal operating conditions. The vibration and stress analysis submittal should include the following items:

(1) Describe the theoretical structural and hydraulic models and analytical formulations or scaling laws and scale models used in the analysis, including all bias errors and uncertainties for reactor internals that, based on past experience, are not adversely affected by the flow-excited acoustic resonances and flow-induced vibrations. Additional analyses should be performed on those systems and components, such as steam dryers and main steam system components in BWRs and steam generator internals in PWRs, that may potentially be adversely affected by the flow-excited acoustic resonances and flow-induced vibrations. These additional analyses are summarized below:

   (a) Determine the pressure fluctuations and vibration in the applicable plant systems under flow conditions up to and including the full operating power level. Such pressure fluctuations and vibration can result from hydrodynamic effects and acoustic resonances under the plant system fluid flow conditions.

   (b) Justify the method for determining pressure fluctuations, vibration, and resultant cyclic stress in plant systems. Based on past experience, computational fluid dynamics (CFD) analyses might not provide sufficient quantitative information regarding high-frequency pressure loading without supplemental analyses. Scale testing can be applied for the high-frequency acoustic pressure loading and for verifying the pressure loading results from CFD analyses and the supplemental analyses, where the bias error and random uncertainties are properly addressed.

(c) Address significant acoustic resonances that have the potential to damage plant piping and components including steam dryers, and perform modifications to reduce those acoustic resonances, as necessary, based on the analysis.

When following this regulatory guide, licensees of operating nuclear power plants should obtain plant-specific data to confirm the scale testing and analysis results for pressure fluctuations and vibration prior to submitting a power uprate request.

If scale model testing is used to support the applicant's submission, the following areas should be considered:

(a) the effects of sound attenuation in the model (or effects of pressure, size, and medium) on the generation of any self-excitation mechanism (flow-excited acoustic or structural resonances)

(b) the effects of sound attenuation on the acoustic pressures within the RPV and on all reactor internals

(c) the conservatism of the 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 geometrical details (such as branch line openings)

If the applicant uses CFD models, all associated uncertainties and bias errors should be presented:

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

(b) Show that grid size does not affect the results (i.e., perform grid size sensitivity tests).

(c) Meet requirement of Courant number.

(d) Perform unsteady simulations using large eddy simulation or direct numerical simulation at high Reynold's number flow, and include compressibility effects to model any coupling of the flow and the acoustic waves in the fluid (self-excitation, or lock-in effects).

(e) Use real gas simulation (i.e., use state equation of steam as real gas).

(f) Validate the simulation procedures on similar (i.e., complex) flow situations.

(2) Describe the structural and hydraulic system natural frequencies and associated mode shapes that may be excited during steady-state and anticipated transient operation, for reactor internals that, based on past experience, are not adversely affected by the flow-excited acoustic resonances and flow-induced vibrations. Additional analyses should be performed on those systems and components, such as steam dryers and main steam system components in BWRs and steam generator internals in PWRs, that may potentially be adversely affected by the flow-excited acoustic resonances and flow-induced vibrations. These additional analyses are summarized below.

Determine the damping of the excited mode shapes, and the frequency response functions (FRFs, i.e., vibration induced by unit loads or pressures, and stresses induced by unit loads or pressures), including all bias errors and uncertainties.

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 associated with both the approach and the specific model should be provided, along with their bases. In many cases, bias errors in numerical models (for stresses, in particular) are associated with insufficient resolution of stress concentration regions. These errors may sometimes be accounted for with stress concentration factor corrections. Uncertainties are often associated with differences between ideal models and as-manufactured structures, such as differences in material properties, connections (bolts, welds, and rivets), and geometries (plate thicknesses). The uncertainty (and perhaps bias) associated with these differences may be estimated based on comparisons of simulations and measurements of structures similar in construction to the reactor internal being modeled.

Upper bounds on the uncertainties associated with all significant natural frequencies of the mode shapes, which may be excited during steady-state and transient operation, should be provided, along with the uncertainties and bias errors associated with the amplitudes of the FRFs. The uncertainties associated with modeling the fluid loading (by water and/or steam) on reactor internal structures should also be reported (specifically, how they relate to uncertainties in the natural frequencies and FRFs). The FRF amplitude uncertainties are generally associated with the construction differences described above. FRF bias errors are associated with the construction differences, as well as the damping assumed for the modes.

Away from resonance frequencies, the bias error is dominated by geometry and material differences between the modeled and actual structures. At resonance frequencies, the bias error is dominated by the assumed damping. In many prior submissions, licensees have cited NRC damping guidance for very-low-frequency seismic analyses as justification for using high damping factors for mid- to high-frequency analyses. Revision 3 of this regulatory guide corrects this misconception, and specifies that damping factors used in structural dynamic modeling should be based on mid- to high-frequency measurements or rigorous analyses conducted on structures representative of the reactor internal structure being modeled. Acceptable measurement techniques for damping are available in the standards promulgated by the American National Standards Institute (ANSI) and the International Organization for Standardization (ISO), and may be applied in air, steam, or water environments, as applicable. If the applicant believes that inservice vibrations will be sufficiently high to lead to non-linear structural behavior (thereby increasing damping), the measurements may be made using applied forces consistent with those anticipated inservice vibration levels. However, the applicant should be prepared to also use non-linear dynamic analyses to compute vibration and stress should this be the case. Based on past experiences, any attempt to specify structural damping coefficients greater than 1 percent for frequencies greater than seismic frequencies should be strongly substantiated with measurements.

(3) Describe the estimated random and deterministic forcing functions, including any very-low-frequency components, for steady-state and anticipated transient operation for reactor internals that, based on past experience, are not adversely affected by the flow-excited acoustic resonances and flow-induced vibrations. Additional analyses should be performed on those systems and components, such as steam dryers and main steam system components in BWRs and steam generator internals in PWRs, that may potentially be adversely affected by the flow-excited acoustic resonances and flow-induced vibrations. These additional analyses are summarized below.

Evaluate any forcing functions that may be amplified by lock-in with an acoustic and/or structural resonance (sometimes called self-excitation mechanisms). A lock-in of a forcing function with a resonance strengthens the resonance amplitude. The resulting amplitudes of the forcing function and resonance response can therefore be significantly higher than the amplitudes associated with non-lock-in conditions.

All potential flow-excited acoustic or structural resonances that lead to feedback and loading amplification (commonly termed lock-in) should be addressed. Tables of expected flow rates and resonance frequencies, along with the possible ranges of lock-in and potential loading amplifications, should be provided. Uncertainties in any of the lock-in parameters (such as the characteristic Strouhal numbers of the flow-excitation sources) should be clearly defined.

If any potential self-excitation or lock-in is identified, the applicant should provide specific mitigation procedures that would be employed if the lock-in leads to vibration and/or stresses that exceed allowable limits. The following list enumerates some of the forcing functions that should be considered:

(a) flow instabilities over openings in the MSLs, like control and safety valve stand pipes, blind flanges, and others that lead to strong narrow-band excitation, which can lock-in to acoustic and/or structural resonances, considering the following parameters:

  (i) Strouhal number analysis to check critical flow rates (including any uncertainties in Strouhal number)

  (ii) effects of diameter ratio

  (iii) effects of upstream elbows

  (iv) distance between stand pipes

  (v) relative length of stand pipes

  Flow instability frequencies should be compared to those of acoustic modes in the reactor dome and structural modes in the MSLs, any connected valves, and reactor internal structures. Finite element (FE) simulations or measurements may be used to determine the resonance frequencies.

  Any identified self-excitation or lock-in should not be analyzed by simply using linear extrapolation techniques.

(b) separated, impinging and reattached flows in the reactor dome, including low-frequency hydrodynamic loading on the steam dryer in BWRs

(c) flow turbulence and narrowband excitation in the steam ring of MSLs in BWRs

The applicant/licensee should determine the design load definition for all reactor internals, including the steam dryer in BWRs up to the full licensed power level, and should validate the method used to determine the load definitions based on scale model or plant data. BWR applicants should include instrumentation on the steam dryer to measure pressure loading, strain, and acceleration to confirm the scale model testing and analysis results. BWR licensees should obtain plant data at current licensed power conditions for use in confirming the results of the scale model testing and analysis for the steam dryer load definition prior to submitting a power uprate request.

In recent BWR EPU requests, some licensees have employed a model to compute fluctuating pressures within the RPV and on BWR steam dryers that are inferred from measurements of fluctuating pressures within the MSLs connected to the RPV. Applicants should clearly define all uncertainties and bias errors associated with the MSL pressure measurements and modeling parameters. The bases for the uncertainties and bias errors, such as any experimental evaluation of modeling software, should be clearly presented. There are many approaches for measuring MSL pressures and computing fluctuating pressures within the RPV and the MSLs. Although some approaches reduce bias and uncertainty, they still have a finite bias and uncertainty, which should be reported. Based on historical experience, the following guidance is offered regarding approaches that minimize uncertainty and bias error:

(a) At least two measurement locations should be employed on each MSL in a BWR. However, using three measurement locations on each MSL improves input data to the model, particularly if the locations are spaced logarithmically. This will reduce the uncertainty in describing the waves coming out of and going into the RPV. Regardless of whether two or three measurement locations are used, no acoustic sources should exist between any of the measurement locations, unless justified.

(b) Strain gages (at least four gages, circumferentially spaced and oriented) 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 accounted for in a bias error and uncertainty estimate.

(c) The speed of sound used in any acoustic models should not be changed from plant to plant, but rather should 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 direct measurement. The uncertainty of the reflection coefficients should be clearly defined. Note that simply assuming 100-percent reflection coefficient is not necessarily conservative.

(e) Any sound attenuation coefficients should be a function of steam quality (variable between the steam dryer and reactor dome), rather than constant throughout a steam volume (such as 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 dictate adjustments of these parameters.

(4) Summarize the calculated structural and hydraulic responses for operation under steady-state and anticipated transient conditions for reactor internals that, based on past experience, are not adversely affected by the flow-excited acoustic resonances and flow-induced vibrations. This summary should identify the random, deterministic, and overall integrated maximum response, any very-low-frequency components of response, and the level of cumulative fatigue damage.

Additional analyses should be performed on those systems and components, such as steam dryers and main steam system components in BWRs and steam generator internals in PWRs, that may potentially be adversely affected by the flow-excited acoustic resonances and flow-induced vibrations. These additional analyses are summarized below.

The calculated responses should include vibrations for components that have maximum vibration limits, as well as stresses for components that have maximum stress criteria (such as the fatigue stress limits specified in Section III of the ASME Boiler and Pressure Vessel Code, Ref. 3). The margins against violating the criteria should be reported. Based on the uncertainties and bias errors identified in items 1–3 above, an end-to-end uncertainty and bias error should be reported, along with a clear explanation of how the individual uncertainties and bias errors have been combined.

Since the transfer functions (or FRFs) described in item 2, and forcing functions described in item 3, have an uncertainty associated with the frequencies of the response peaks attributable to resonant modes, the vibration and stress calculations should address those uncertainties by shifting either the FRFs or forcing functions in frequency to span the uncertainty in the response peak frequencies. An optimal approach to resolving the uncertainty associated with natural frequencies is to align any forcing function peaks with all modal peaks within the range of frequency uncertainty, and to report the worst-case vibration and/or stress. All uncertainty and bias associated with natural frequencies is eliminated with this approach. Note that the uncertainty and bias associated with the FRF amplitudes are not eliminated by aligning all forcing function and modal peaks. An alternative, less-optimal approach is to perform several analyses in which the FRFs or forcing functions are shifted by increments within the frequency uncertainty range. Once again, the worst case vibration or stress should be reported, since the frequency uncertainty leads to a negative (non-conservative) bias in the vibration and stress when any modal peaks are misaligned with any forcing function peaks.

(5) Summarize the calculated structural and hydraulic responses for preoperational and initial startup testing conditions, compared to those for normal operation. This summary should address the adequacy of the test simulation to normal operating conditions.

(6) Identify the anticipated structural or hydraulic vibratory response [defined in terms of frequency, amplitude (displacement, acceleration, and/or strain), and modal contributions] that is appropriate to each of the sensor locations for steady-state and anticipated transient preoperational and startup test conditions.

(7) Specify the test acceptance criteria with permissible deviations and the bases for the criteria. The criteria should be established in terms of maximum allowable response levels in the structure, and presented in terms of maximum allowable response levels at sensor locations.

Additional analyses should be performed on those systems and components, such as steam dryers and main steam system components in BWRs and steam generator internals in PWRs, that may potentially be adversely affected by the flow-excited acoustic resonances and flow-induced vibrations. These additional analyses are summarized below.

Based on steam dryer and MSL valve failures that have occurred in BWR plants operating at EPU conditions, the following test acceptance criteria are provided for future BWR license applications. After developing a steam dryer load definition, an applicant for the construction and operation of a BWR nuclear power plant (or a licensee using this regulatory guide in planning a power uprate for an operating BWR nuclear power plant) should apply the load definitions to vibration and stress models to determine the vibrations of the valves and stresses within the steam dryer, with justified damping assumptions and applicable weld factors and stress intensities. After including applicable bias errors and random uncertainties, the applicant/licensee should compare valve vibrations against applicable limits, and peak stresses at critical steam dryer locations to the fatigue limits in the ASME Boiler & Pressure Vessel Code (Ref. 3).

The applicant/licensee should also compare stresses, at any locations that might have experienced fatigue cracking, with the ASME Code fatigue limits to validate the stress model. The applicant/licensee should also compare the primary and secondary stresses that the steam dryer may experience as a result of plant transients to the applicable ASME Code service level limits. The BWR applicant/licensee should also implement modifications to the BWR steam dryer based on the design stress margin or to any components responsible for high excitation to reduce that excitation, so that none of the resulting stresses exceed the Code allowable limits.

The BWR applicant/licensee should also develop a vibration limit curve for valves and a stress limit curve for the steam dryer for power ascension to provide assurance that the valve vibrations and stress in the individual steam dryer components will not exceed the ASME Code fatigue limits. The limit curves, while including the bias errors and uncertainties from the end-to-end vibration and stress analyses, should also include those associated with the vibration and stress measurement program (in particular, those associated with the data acquisition systems and instrumentation). The BWR applicant/licensee should also develop a method for collecting plant data during power ascension and full licensed power conditions that can be used to calculate the valve vibrations steam dryer stress, including appropriate bias errors and random uncertainties. As the steam dryer is not a Code component, the applicant/licensee may justify different stress acceptance criteria.

The PWR applicant/licensee should evaluate the stress and design margin for internal components (such as the steam dryers internal to the steam generators) in the steam generators for the planned operating conditions. Past operating experience and analysis may be used to support the determination of adequate design margin for the stress on PWR steam generator internal components.

2.2 Vibration and Stress Measurement Program

The applicant/licensee should develop and implement a vibration measurement program to verify the structural integrity of reactor internals, determine the margin of safety associated with steady-state and anticipated transient conditions for normal operation, and confirm the results of the vibration analysis.

Additional measurements should be performed on those systems and components, such as steam dryers and main steam system components in BWRs and steam generator internals in PWRs, that may potentially be adversely affected by the flow-excited acoustic resonances and flow-induced vibrations. These additional analyses are summarized below.

The initial plant startup testing to evaluate potential adverse flow effects on BWR plant reactor internals components should include the steam dryer and MSL valves. For initial plant startup, the applicant/licensee should collect plant data from instrumentation mounted directly on the steam dryer at significant locations (including the outer hood and skirt, and other potential high-stress locations) to verify that the stress on individual steam dryer components is within allowable limits during plant operation. The instrumentation directly mounted on the steam dryer should provide sufficient information for a stress analysis of the entire steam dryer, and should include pressure sensors, strain gauges, and accelerometers. The MSLs should also be instrumented to collect data to determine steam pressure fluctuations in order to identify the presence of flow-excited acoustic resonances and allow analysis of those pressure fluctuations to calculate MSL valve loading and vibration, and steam dryer loading and stress. The direct steam dryer data should be used to calibrate the MSL instrumentation and data analysis prior to removal or failure of the steam dryer instrumentation. BWR licensees planning a power uprate may use plant instrumentation to evaluate steam dryer pressure loading and stress, rather than installing steam dryer instrumentation where justified.

As part of the startup and power ascension program for BWR and PWR plants, the steam, feedwater, and condensate lines and associated components, including safety relief valves and power-operated valves and their actuators, should be instrumented to measure vibration during plant operation to verify that qualification limits will not be exceeded for the piping and individual components.

The vibration measurement program submittal should include a description of the following systems and conditions:

(1) the data acquisition and reduction system, including the following details:

- (a) transducer types and their specifications, including useful frequency and amplitude ranges
- (b) transducer positions, which should be sufficient to monitor significant lateral, vertical, and torsional structural motions of major reactor internal components in shell, beam, and rigid body modes of vibration, as well as significant hydraulic responses and those parameters that can be used to confirm the input forcing function
- (c) precautions being taken to ensure acquisition of quality data (e.g., optimization of signal-to-noise ratio, relationship of recording times to data reduction requirements, choice of instrumentation system)
- (d) online data evaluation system to provide immediate verification of general data quality
- (e) procedures for determining frequency, modal content, and maximum values of response
- (f) all bias errors (such as model underprediction) and random uncertainties (such as instrumentation error) associated with the instrumentation and data acquisition systems

(2) test operating conditions, including the following details:

(a) For all steady-state and transient modes of operation, additional analyses should be performed on those systems and components, such as steam dryers and main steam system components in BWRs and steam generator internals in PWRs, that may potentially be adversely affected by the flow-excited acoustic resonances and flow-induced vibrations. In particular, the applicant/licensee should establish a power ascension program, which includes, as applicable, (i) specific hold points and their durations during power ascension; (ii) activities to be accomplished during the specified hold points; (iii) plant parameters to be monitored in comparison with applicable limit curves; (iv) inspections and walkdowns to be conducted for steam, feedwater, and condensate systems and components during the specified hold points; (v) methods to be used to trend plant parameters; (vi) acceptance criteria for monitoring and trending plant parameters, and for conducting walkdowns and inspections; (vii) actions to be taken if acceptance criteria are not satisfied; and (viii) provisions for providing information to the NRC staff on plant data, evaluations, walkdowns, inspections, and procedures prior to and during power ascension, including interactions during hold points and any instance in which acceptance criteria are not satisfied, and resolution of safety concerns identified during the staff's review of that information prior to further power ascension or continued full-power operation.

For a BWR plant with an instrumented steam dryer, the applicant/licensee should determine the steam dryer stress from the direct instrumentation, and compare that stress to the applicable limit curves considering bias errors and random uncertainties, as applicable.

For an operating BWR plant without an instrumented steam dryer, the applicant/licensee should calculate the steam dryer stress using data from steam system instrumentation, and considering appropriate bias errors and random uncertainties.

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

(b) The applicant/licensee should specify the planned duration of all testing in normal operating modes to ensure that the testing will subject each critical component to at least 10^6 cycles of vibration (i.e., computed at the lowest frequency for which the component has a significant structural response) prior to the final inspection of the reactor internals. The duration of testing for non-prototype reactor internals should be no less than that for the applicable reference design classifications of reactor internals (i.e., valid, conditional, or limited valid prototype).

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

Plant operating experience, such as from the Quad Cities and Dresden nuclear power stations, has shown that adverse flow effects might not appear for an extended period of time following initial startup or power ascension. Therefore, it would be beneficial to maintain the program for monitoring potential adverse flow effects (such as flow-excited acoustic or structural resonances) on plant systems and components for a sufficient time period to verify that adverse flow effects are not occurring at new nuclear power plants or those implementing a power uprate. This program should include monitoring of plant data, performance of walkdowns, and inspection of components during power ascension and operation under full licensed power conditions. The program should also include inspections and walkdowns that will be performed during refueling outages and extended shutdowns with "as low as is reasonably achievable" (ALARA) consideration. The extent and duration of this program following startup and power ascension should be determined by the licensee based on the review of operating experience at its plant and other nuclear power plants.

2.3 Inspection Program

The inspection program should provide for inspections of the reactor internals prior to and following operation in those steady-state and transient modes consistent with the test conditions discussed in this regulatory guide. The reactor internals should be removed from the reactor vessel for these inspections. If removal is not feasible, the inspections should be performed using examination equipment appropriate for in situ inspection. The inspection program submittal should include the following information:

(1) tabulation of all reactor internal components and local areas to be inspected, including the following details:
   - (a) all major load-bearing elements of the reactor internals that are relied upon to retain the core support structure in position
   - (b) the lateral, vertical, and torsional restraints provided within the vessel
   - (c) those locking and bolting components whose failure could adversely affect the structural integrity of the reactor internals
   - (d) those surfaces that are known to be or may become contact surfaces during operation
   - (e) those critical locations on the reactor internal components as identified by the vibration analysis, such as the steam dryers in BWRs
   - (f) the interior of the reactor vessel for evidence of loose parts or foreign material

(2) tabulation of specific inspection areas that can be used to verify segments of the vibration analysis and measurement program.

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

2.4 Documentation of Results

The results of the vibration and stress analysis, measurement, and inspection programs should be reviewed and correlated to determine the extent to which the test acceptance criteria are satisfied. A summary of the results should be submitted to the NRC in the form of preliminary and final reports:

(1) The preliminary report should summarize an evaluation of the raw and, as necessary, limited processed data and the results of the inspection program with respect to the test acceptance criteria. Anomalous data that could bear on the structural integrity of the reactor internals should be identified, as should the method to be used for evaluating such data.

(2) If the results of the comprehensive vibration assessment program are acceptable, the final report should include the following information:

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

    (b) comparison between measured and analytically determined modes of structural response (including damping factors) and hydraulic response (including those parameters from which the input forcing function is determined) for the purpose of establishing the validity of the analytical technique

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

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

(3) If (a) inspection of the reactor internals reveals defects, evidence of unacceptable motion, and/or excessive or undue wear; (b) the results from the measurement program fail to satisfy the specified test acceptance criteria; or (c) the results from the analysis, measurement, and inspection programs are inconsistent, the final report should also include an evaluation and description of the modifications or actions planned in order to justify the structural adequacy of the reactor internals.

2.5 Schedule

A schedule for the vibration assessment program should be established and submitted to the NRC (1) during the construction permit review for new nuclear reactor applications under 10 CFR Part 50, or (2) during the review of the design certification document (DCD) for standard design certification applications under 10 CFR Part 52, or (3) as part of the application for COL applications under 10 CFR Part 52. The schedule should provide for the following:

(1) The reactor internals design should be classified in the preliminary safety analysis report (PSAR) as a prototype or a specific non-prototype category, for applications requesting construction permits under 10 CFR Part 50. The classification may be revised in the final safety analysis report (FSAR) if schedule changes with respect to the previously designated reference reactor make such reclassification appropriate.

In the case of design certification applications under 10 CFR Part 52, and COL applications under 10 CFR Part 52 that do not reference a standard design, the reactor internals design should be classified as a prototype or a specific non-prototype category in the DCD or COL application, respectively. (If the internals are classified as non-prototype, the applicant should identify the applicable prototype reactor internals in the PSAR, DCD, or COL application, as appropriate.

Experimental or analytical justification of the non-prototype classification should be presented during the construction permit review for applications submitted under 10 CFR Part 50.

In the case of applications submitted under 10 CFR Part 52, the issues related to justification of the non-prototype classification should be resolved during the review of the DCD or COL application. If the test data proves insufficient to meet the guidelines provided in this regulatory guide, the COL applicant should develop a test plan to ensure that any additional data are obtained and submitted to the staff.

(2) During the staff's review of the construction permit, DCD, or COL application, as appropriate, a commitment should be established regarding the scope of the comprehensive vibration assessment program.

(3) A description of the vibration measurement and inspection phases of the comprehensive vibration assessment program should be submitted to the NRC in sufficient time to permit utilization of the staff's related recommendations. (For scheduling purposes, the applicant should allow 90 days for the staff's review and comment period.)

(4) A summary of the vibration analysis program should be submitted to the NRC at least 60 days prior to submission of the description of the vibration measurement and inspection programs. As an alternative, the information in this report may be submitted to the NRC in conjunction with the report (discussed above). In this case, for scheduling purposes, the applicant should allow 120 days for the staff's review and comment period for the composite report.)

(5) The preliminary and final reports, which together summarize the results of the vibration analysis, measurement, and inspection programs, should be submitted to the NRC within 60 and 180 days, respectively, following the completion of vibration testing.

3. Comprehensive Vibration Assessment Program for Non-Prototype Reactor Internals

During the preoperational and initial startup test program, non-prototype reactor internals important to safety should be subjected to all significant flow modes associated with normal steady-state and anticipated transient operation under the same test conditions imposed on the applicable prototype. Evaluation of the effects of such operation on the structural integrity of the non-prototype reactor internals should be based on the results of a comprehensive vibration assessment program developed for the specific non-prototype classification (i.e., Category I, II, III, or IV).

The following sections address the comprehensive vibration assessment programs for the specific classifications of non-prototype reactor internals. These programs should be documented and scheduled in accordance with the program guidelines delineated in this regulatory guide for prototype reactor internals.

3.1 Non-Prototype, Category I

3.1.1 *Vibration and Stress Analysis Program*

The valid prototype should be specified, and sufficient evidence should be provided to support the non-prototype, Category I classification. If the valid prototype comprehensive vibration assessment program was conducted on a reactor outside the United States, the detailed results of the program should be included in any application related to a non-prototype submitted to the NRC for review, and should meet all of the criteria in this regulatory guide.

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

3.1.2 *Vibration Measurement Program*

The vibration measurement program may be omitted if the inspection program is implemented. However the vibration measurement program related to the evaluation of the potential adverse flow effect from pressure fluctuations and vibrations in piping systems for both PWRs and BWRs, should not be omitted.

If a vibration measurement program is implemented in lieu of an inspection program, the applicant/licensee should incorporate sufficient and appropriate instrumentation to verify that the vibratory response of the non-prototype, Category I reactor internals is consistent with the vibration analysis results, test acceptance criteria, and vibratory response observed in the valid prototype. The vibration measurement program should include a description of the data acquisition and reduction systems, as well as test operating conditions, consistent with the general guidelines for the vibration measurement program delineated in this regulatory guide for prototype reactor internals.

3.1.3 *Inspection Program*

If an inspection program is implemented in lieu of a vibration measurement program, the applicant/licensee should follow the inspection program guidelines delineated in this regulatory guide for prototype reactor internals.

The inspection program may be omitted if the vibration measurement program is implemented. However, if significant discrepancies exist between anticipated and measured responses for specific components, the applicant/licensee should remove those components from the reactor vessel and perform a visual examination. Components for which removal is not feasible should be examined in situ using appropriate inspection equipment. In any case, the applicant/licensee should visually check the interior of the reactor vessel for loose parts and foreign material.

3.2 Non-Prototype, Category II

3.2.1 *Vibration and Stress Analysis Program*

The valid prototype should be specified, and sufficient evidence should be provided to support the non-prototype, Category II classification, which involves demonstrating that the structural differences between the valid prototype and the non-prototype, Category II reactor internals have no significant effect on the vibratory response and excitation of those unmodified non-prototype, Category II components. If the valid prototype comprehensive vibration assessment program was conducted on a reactor outside the United States, the detailed results of the program should be included in any application related to a non-prototype submitted to the NRC for review, and should conform to the guidance set forth in this regulatory guide.

The vibration and stress analysis for the valid prototype, which includes a summary of the anticipated structural and hydraulic response and test acceptance criteria, should be modified to account for the nominal differences that may exist between the valid prototype and the non-prototype, Category II reactor internals.

The applicant/licensee should specifically establish test acceptance criteria for those non-prototype, Category II reactor internals that have structural differences relative to the valid prototype. The vibration and stress analysis, as well as the test acceptance criteria related to any structural differences, should be consistent with the general guidelines delineated in this regulatory guide for prototype reactor internals.

3.2.2 *Vibration Measurement Program*

The applicant/licensee should implement a vibration measurement program for non-prototype, Category II reactor internals during preoperational and initial startup testing.

The vibration measurement program should include a description of the data acquisition and reduction systems, as well as test operating conditions, consistent with the general guidelines for the vibration measurement program delineated in this regulatory guide for prototype reactor internals. This program should incorporate sufficient and appropriate instrumentation to define the vibratory and stress response (i.e., frequency, amplitude, modal content) of those reactor internal components important to safety that *have* been modified relative to the valid prototype, in order to establish the margin of safety and demonstrate that the test acceptance criteria have been satisfied.

In addition, the vibration measurement program should incorporate sufficient and appropriate instrumentation to monitor those reactor internal components important to safety that *have not* been modified relative to the valid prototype. This instrumentation can confirm that the vibratory response of such components complies with the guidelines for non-prototype, Category II reactor internals and is consistent with the results obtained for similar components in the measurement program for the valid prototype.

3.2.3 *Inspection Program*

The applicant/licensee should implement an inspection program that follows the related guidelines delineated in this regulatory guide for prototype reactor internals.

3.3 Non-Prototype, Category III

3.3.1 *Vibration and Stress Analysis Program*

The conditional prototype should be specified, and sufficient analytical or experimental evidence should be provided to support the non-prototype, Category III classification and demonstrate the applicability of data from the vibration measurement program on the prototype to the conditional prototype. The following should be demonstrated:

(1) The conditional prototype is substantially similar in arrangement, design, size, and operating conditions to the non-prototype, Category III reactor internals.

(2) Response modes attributable to inservice vibration problems and the ensuing component or operational modifications do not significantly affect the applicability of results from the vibration measurement program on the prototype to the conditional prototype, or the effects are limited to structural components and response modes that permit clear separation of these effects from other results of the vibration measurement program.

The applicant/licensee should provide details concerning the adverse vibration experience of the conditional prototype, as well as experimental or analytical information to demonstrate that the vibration problems associated with the conditional prototype have been corrected for both the conditional prototype and the applicable non-prototype, Category III reactor internals. The vibration and stress analysis, as well as the test acceptance criteria related to any component or operational modifications, should be consistent with the general guidelines delineated in this regulatory guide for prototype reactor internals.

The vibration and stress analysis on the prototype to the conditional prototype, which includes a summary of the anticipated structural and hydraulic response and test acceptance criteria for the measurement program on the prototype, should be modified to account for the component or operational modifications applicable to the conditional prototype and non-prototype, Category III reactor internals. The vibration and stress analysis, as well as the test acceptance criteria related to any component or operational modifications, should be consistent with the general guidelines delineated in this regulatory guide for prototype reactor internals.

Test acceptance criteria, with permissible deviations, should be specified for reactor internal components important to safety. Each component should be categorized according to whether the results from the vibration measurement program on the prototype to the conditional prototype are applicable.

3.3.2 *Vibration Measurement Program*

The applicant/licensee should implement a vibration measurement program for non-prototype, Category III reactor internals during preoperational and initial startup testing.

The vibration measurement program should incorporate sufficient and appropriate instrumentation to define the vibratory response of those reactor components important to safety that, because of structural or operational modifications relative to the original design of the conditional prototype, are expected to have response characteristics substantially different from those measured for the given components during the vibration measurement program on the prototype to the conditional prototype.

The vibration measurement program should also incorporate sufficient and appropriate instrumentation to monitor all other components and confirm that their measured responses are substantially similar to those obtained for the given components during the vibration measurement program on the prototype to the conditional prototype.

In addition, the vibration measurement program should satisfy the general guidelines delineated in this regulatory guide for a prototype vibration measurement program.

3.3.3 *Inspection Program*

The applicant/licensee should implement an inspection program that satisfies the related guidelines delineated in this regulatory guide for prototype reactor internals.

3.4 <u>Non-Prototype, Category IV</u>

3.4.1 *Vibration and Stress Analysis Program*

The limited valid prototype should be specified, and sufficient evidence should be provided to support the non-prototype, Category IV classification.

3.4.2 *Vibration Measurement Program*

The vibration measurement program may be omitted if the inspection program is implemented.

If a vibration measurement program is implemented in lieu of an inspection program, the applicant/licensee should incorporate sufficient and appropriate instrumentation to verify that the vibratory response of the non-prototype, Category IV reactor internals is consistent with the vibration analysis results, test acceptance criteria, and vibratory response observed for the referenced limited valid prototype.

The vibration measurement program should be consistent with the guidelines delineated in this regulatory guide for non-prototype, Category I reactor internals.

3.4.3 *Inspection Program*

If an inspection program is implemented in lieu of a vibration measurement program, the applicant/licensee should follow the inspection program guidelines delineated in this regulatory guide for non-prototype, Category I reactor internals.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide. No backfitting is intended or approved in connection with the issuance of this guide. Although this regulatory guide is directed to new nuclear power plants, current licensees planning to propose a power uprate might also find the guidance herein to be helpful in establishing a power ascension testing program.

Except in those cases in which a licensee proposes or has previously established an acceptable alternative method for complying with specified portions of the NRC's regulations, the NRC staff will use the methods described in this guide to evaluate (1) submittals in connection with applications for construction permits, standard plant design certifications, operating licenses, early site permits, and combined licenses, and (2) submittals from operating reactor licensees who voluntarily propose to initiate system modifications if there is a clear nexus with the subject for which guidance is provided herein.

REGULATORY ANALYSIS / BACKFIT ANALYSIS

The regulatory analysis and backfit analysis for this regulatory guide is available in Draft Regulatory Guide DG-1163, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing" (Ref. 5). The NRC issued DG-1163 in November 2006 to solicit public comment on the draft of this Revision 3 of Regulatory Guide 1.20.

REFERENCES

1. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," U.S. Nuclear Regulatory Commission, Washington, DC.[1]

2. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.[2]

3. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," Division I, American Society of Mechanical Engineers, New York, NY.[2]

4. Regulatory Guide 1.68.1, "Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling-Water Reactor Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.[3]

5. Draft Regulatory Guide DG-1163, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," U.S. Nuclear Regulatory Commission, Washington, DC.[4]

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[1] All NRC regulations listed herein are available electronically through the Public Electronic Reading Room on the NRC's public Web site, at http://www.nrc.gov/reading-rm/doc-collections/cfr/. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301) 415-4737 or (800) 397-4209; fax (301) 415-3548; email PDR@nrc.gov.

[2] Copies may be purchased from the American Society of Mechanical Engineers, Three Park Avenue, NewYork, NY 10016-5990; phone (212) 591-8500; fax (212) 591-8501; www.asme.org.

[3] Regulatory Guide 1.68.1 is available electronically through the Electronic Reading Room on the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/, and under Accession #ML062750162 in the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. Single copies may also be obtained free of charge by writing the Reproduction and Distribution Services Section, ADM, USNRC, Washington, DC 20555-0001, by fax to (301) 415-2289, or by email to DISTRIBUTION@nrc.gov. Active guides may also be purchased from the National Technical Information Service (NTIS). Details may be obtained by contacting NTIS at 5285 Port Royal Road, Springfield, Virginia 22161, online at http://www.ntis.gov, by telephone at (800) 553-NTIS (6847) or (703) 605-6000, or by fax to (703) 605-6900. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville, Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555-0001. The PDR can also be reached by telephone at (301) 415-4737 or (800) 397-4209, by fax at (301) 415-3548, and by email to PDR@nrc.gov.

[4] Draft Regulatory Guide DG-1163 is available electronically under Accession #ML062750162 in the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville, Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555-0001. The PDR can also be reached by telephone at (301) 415-4737 or (800) 397-4209, by fax at (301) 415-3548, and by email to PDR@nrc.gov.