Review of Reactor Internals Vibration Programs

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Abstract

In the review of the reactor internals vibration assessment program described in the Design Control Document (DCD) for a Design Certification application, the U.S. Nuclear Regulatory Commission (NRC) staff focuses on system and component design and accessibility for performance, preoperational and startup testing activities, and the general description of related programs using the guidance in NRC Standard Review Plan (SRP) Sections 3.9.2 and 3.9.5, and Revision 3 to Regulatory Guide (RG) 1.20. In the past, the DCD provided significant flexibility for COL applicants to develop startup testing and operational programs. As a result, Design Centers have different approaches in fully describing these programs. All Design Centers should address startup and operational programs to enable the NRC to make its safety determination for the Design Certification application. The COL application including the Final Safety Analysis Report (FSAR) needs to support an NRC decision that startup testing and operational programs will provide reasonable assurance of safe plant operation. COL applicants may rely on a combination of DCD and FSAR information to fully describe their preoperational and startup programs. This paper discusses the various regulations that require, before design certification, information normally contained in certain procurement specifications and construction and installation specifications to be completed and made available for audit if the information is necessary for the Commission to make its safety determination. In addition, the regulations require Design Certification applications to address operating experience, and COL applications to include information necessary to demonstrate how operating experience is incorporated into plant design. The paper discusses Commission Paper SECY-05-0197, "Review of Operational Programs in a Combined License Application and General Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria [ITAAC]." The paper also discusses SRP Sections 3.9.2 and 3.9.5 and RG 1.20 for guidance on preoperational and startup test programs for reactor internals. This paper discusses how the programs can be clearly and sufficiently described in terms of scope and level of detail to allow a reasonable assurance finding of acceptability.

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Introduction

The nuclear industry is developing new reactor designs, and is planning to construct and operate new nuclear power plants. The U.S. Nuclear Regulatory Commission (NRC) has certified several new reactor designs and is evaluating applications for the certification of additional reactor designs. The NRC is also evaluating applications from nuclear industry organizations for licenses to construct and operate new nuclear power plants. This paper discusses the review of reactor internals vibration assessment programs by the NRC for new nuclear power plants.

NRC Regulations

In Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," of Title 10, "Energy," in the *Code of Federal Regulations* (10 CFR Part 52), the NRC specifies rules for the certification of reactor designs, and for the issuance of combined licenses to construct and operate nuclear power plants. In appendices to 10 CFR Part 52, the NRC has certified the design of the U.S. Advanced Boiling Water Reactor (ABWR), System 80+ Reactor, AP600 Reactor, and AP1000 Reactor. The NRC is currently reviewing Design Certification applications for the Economic Simplified Boiling Water Reactor (ESBWR), U.S. Evolutionary Power Reactor (U.S EPR), and U.S. Advanced Pressurized Water Reactor (US-APWR). The NRC is reviewing Combined License (COL) applications for numerous nuclear power plants referencing certified reactor designs or Design Certification applications.

As part of Design Certification applications, the NRC regulations in 10 CFR 52.47 require that information normally contained in certain procurement specifications and construction and installation specifications be completed and made available for audit if the information is necessary for the Commission to make its safety determination. The application must contain a Final Safety Analysis Report (FSAR) that describes the facility, presents the design bases and limits on its operation, and presents a safety analysis of the structures, systems, and components (SSCs) and of the facility as a whole. For example, this includes the design of the facility such as the principal design criteria for the facility described in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and a description of the quality assurance (QA) program as indicated in Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. In addition, the regulations require Design Certification applications to address operating experience from current nuclear power plants.

As part of COL applications, the NRC regulations in 10 CFR 52.79 require that the application contain an FSAR that provides information at a level sufficient to enable the Commission to reach a final conclusion on all safety matters that must be resolved before COL issuance. For example, the FSAR must describe the design of the facility such as the principal design criteria for the facility specified in 10 CFR Part 50, Appendix A, and must provide a description of the QA program as indicated in 10 CFR Part 50, Appendix B. In 10 CFR Part 52, the NRC regulations also require COL applications to include (a) a description of the programs and their implementation necessary to ensure that the systems and components meet the requirements of the ASME *Boiler and Pressure Vessel Code* (BPV Code) and the ASME *Code for Operation and Maintenance of Nuclear Power Plants*; (b) plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of SSCs; and (c) information necessary to demonstrate how nuclear power plant operating experience is incorporated into the plant design. The NRC regulations also require that the COL application provide an evaluation of the

facility against the Standard Review Plan (SRP) revision in effect 6 months before the docket date of the application.

Regulatory Guidance

The NRC has developed regulatory guides and SRP sections, following the broad guidance in Commission papers, to provide more specific guidance for Design Certification and COL applicants, and the NRC staff technical reviewers, regarding compliance with the NRC regulations for the development and implementation of operational programs for new nuclear power plants.

SRP Section 3.9.2 (Revision 3), "Dynamic Testing and Analysis of Systems, Structures, and Components," describes the NRC staff review of the criteria, testing procedures, and dynamic analyses used to ensure the structural and functional integrity of piping systems, mechanical equipment, reactor internals, and their supports 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. For example, the NRC staff will review the evaluation of dynamic responses of structural components within the reactor vessel caused by steady-state and operational flow transient conditions for prototype reactors. If the reactor internal structures consist of a non-prototype design, reference should be made to the results of test and analyses for the prototype reactor and a summary of the results should be provided. Some 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. Similarly, for pressurized water reactor (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. SRP Section 3.9.2 also specifies that flow-induced vibration (FIV) and acoustic resonance testing of reactor internals should be conducted during the preoperational and startup test program. The purpose of this testing is to demonstrate that adverse flow effects will not cause unanticipated FIV of significant magnitude that can result in structural damage.

SRP Section 3.9.5 (Revision 3), "Reactor Pressure Vessel Internals," describes the NRC staff review of (1) the physical or design arrangements of all reactor internal structures, components, assemblies, and systems; (2) the basis for the design of the reactor internals, loading conditions of normal operation, anticipated operational occurrences, potential adverse flow effects of flow-excited vibrations and acoustic resonances, postulated accidents, and seismic events; (3) validation of modeling procedures for any computational methods used to determine stresses in the reactor internal components and structures; (4) the design bases for the mechanical design of the reactor vessel internals; (5) combination of design and service loadings, and stress intensity or deformation limits; (6) potential adverse flow effects on the reactor pressure vessel internals including BWR steam dryers; (7) inspections, tests, analyses, and acceptance criteria (ITAAC) for new reactor review under 10 CFR Part 52; and (8) COL action items specified in Design Certification applications and addressed in COL applications.

Appendix A, "NRC Review of Potential Adverse Flow Effects in Nuclear Power Plant Systems," to SRP Section 3.9.5 discusses the regulatory and technical evaluation of new reactor applications in response to lessons learned from adverse flow effects at operating nuclear power plants. For example, SRP Section 3.9.5, Appendix A discusses the acceptance criteria in

10 CFR Part 50, Appendix A, for the NRC review of the consideration of potential adverse flow effects. The SRP appendix also discusses the technical evaluation of the consideration of potential adverse flow effects including pressure fluctuation and vibration in plant systems, the design load definition for BWR steam dryers, BWR steam dryer stress and limit curves, PWR steam generator stress and design margin, the evaluation of other plant components, collection and evaluation of power ascension data, and monitoring of potential adverse flow effects. NRC staff members conducting reviews in accordance with SRP Section 3.9.5 review reactor internals design, but not steam generator, valve, piping, or other component design. Accordingly, actions to address potential adverse flow effects for these other components should be addressed in the license application sections in which these components are discussed, rather than in the section dealing with reactor internals. Interim Staff Guidance (ISG) for review and evaluation to address adverse flow effects in equipment other than reactor internals is provided in DC/COL-ISG-010, "Review of Evaluation to Address Adverse Flow Effects in Equipment other than Reactor Internals" (ADAMS Accession# ML092890285).

Revision 3 to Regulatory Guide (RG) 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing," describes a methodology that the NRC staff considers acceptable for use in implementing the NRC regulations as they relate to the internals of nuclear power reactors during preoperational and initial startup testing. RG 1.20 specifies reactor internals as comprising core support structures and adjoining internal structures that are defined in Section III of the ASME BPV Code. Reactor internals in BWR nuclear power plants include, for example, the core plate, top guide, feedwater spargers, steam dryer assembly, fuel support, jet pump and support, and shroud and shroud support.

The NRC staff has prepared several revisions to RG 1.20 to update the guidance for the development of vibration assessment programs for reactor internals. Revision 1 to RG 1.20 expanded the classifications of reactor internals and outlined a vibration assessment program for each class. Revision 2 to RG 1.20 updated the vibration assessment programs for reactor internal classes in response to public comments and further NRC staff review. Revision 3 to RG 1.20 modifies the overall vibration assessment program for reactor internals, and summarizes expectations regarding the evaluation of potential adverse flow effects. Future updates of the applicable SRP sections and RG 1.20 will incorporate the changes adopted as a result of the final version of DC/COL-ISG-010.

The Commission's Staff Requirements Memorandum (SRM) (dated September 11, 2002) for Commission Paper SECY-02-0067, "Inspections, Tests, Analyses, and Acceptance Criteria for Operational Programs (Programmatic ITAAC)," stated that ITAAC for an operational program (with the exception of emergency planning programs) are unnecessary if the program and its implementation are fully described in the COL application and found to be acceptable by the NRC. Preservice and inservice inspection and testing programs, QA program, and equipment qualification program are among the identified operational programs. The Commission also stated that the burden is on the COL applicant to provide the necessary and sufficient programmatic information for approval of the COL without ITAAC. In its SRM for SECY-04-0032 (May 14, 2004), the Commission defined "fully described" as when the program is clearly and sufficiently described in terms of the scope and level of detail to allow a reasonable assurance finding of acceptability. The Commission also noted that required programs should always be described at a functional level and at an increasing level of detail where implementation choices could materially and negatively affect the program effectiveness and acceptability. Commission Paper SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," summarizes the NRC position regarding the full description of

operational programs to be provided by COL applicants. The Commission approved the use of license conditions for implementation milestones for operational programs that are fully described or referenced in the FSAR as discussed in the SRM for SECY-05-0197, dated February 22, 2006.

RG 1.206, "Combined License Applications for Nuclear Power Plants," provides guidance for a COL applicant in preparing and submitting its COL application in accordance with the NRC regulations. For example, Section C.I.3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment," in RG 1.206 states that, for site-specific design features not included in the referenced certified design, the COL applicant should provide 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 attributable to FIV, acoustic resonance, postulated pipe breaks, and seismic events. Section C.I.3.9.5.4, "BWR Reactor Pressure Vessel Internals Including Steam Dryer," in RG 1.206 states that the COL applicant should present a detailed analysis of potential adverse flow effects (e.g., FIV and acoustic resonances) that can severely impact BWR reactor pressure vessel internals (including the steam dryer) and other main steam system components not covered in the referenced certified design. Further, Section C.IV.4, "Operational Programs," in RG 1.206 discusses the requirement in 10 CFR 52.79(a) for descriptions of operational programs that need to be included in the FSAR for a COL application to allow a reasonable assurance finding of acceptability. RG 1.206 follows the guidance provided in SECY-05-0197 for fully describing operational programs in support of COL applications.

NRC Review Approach

The focus of the NRC review of Design Certification and COL applications differ with respect to operational programs. In particular, the NRC review of the Design Certification application focuses on design aspects related to those programs while the COL review evaluates whether those programs are sufficiently described to support issuance of a license to construct and operate the facility. Whether the Design Certification has been previously issued (such as for the AP1000 reactor and ABWR) or is currently under review (such as for the ESBWR, U.S. EPR, and US-APWR), the Design Certification holder or applicant and the COL applicant should work together to ensure that the description of the operational programs in the COL FSAR and applicable portions of the Design Certification Design Control Document (DCD) satisfy the NRC regulations.

Reactor internals important to safety are designed to accommodate steady-state and transient vibratory loads throughout the service life of the reactor. In the review of the reactor internals vibration assessment program described in the DCD for a Design Certification application, the NRC staff evaluates the system and component design and accessibility for performance, preoperational and startup testing activities, and the general description of related programs using the guidance in SRP Sections 3.9.2 and 3.9.5. The DCD may provide significant flexibility for COL applicants to develop startup testing and operational programs. The NRC review of the reactor internals vibration assessment program described in the Design Certification application confirms that the program is adequate with respect to the system and component design, and accessibility for inspection and testing programs. The NRC review also confirms that the general description of the program is consistent with the NRC regulations and Commission guidance.

The NRC staff will review the Design Certification application using the guidance in SRP Sections 3.9.2 and 3.9.5 with respect to the design aspects of vibration assessment for reactor internals. For example, the NRC staff will review the evaluation of dynamic responses of structural components within the reactor vessel caused by steady-state and operational flow transient conditions for prototype reactors. The staff will also address potential adverse flow effects of flow-excited vibrations and acoustic resonances, and validation of modeling procedures for computational methods used to determine stresses in the reactor internal components and structures. The staff will review the ITAAC for new reactors and COL action items specified in the Design Certification application. As discussed in SRP Section 3.9.5, Appendix A, the staff will evaluate the consideration of pressure fluctuation and vibration in plant systems, the design load definition for BWR steam dryers, BWR steam dryer stress and limit curves, and the plans for collection and evaluation for power ascension data and monitoring of potential adverse flow effects. The staff follows Interim Staff Guidance DC/COL-ISG-010 regarding the review of potential adverse flow effects in equipment other than reactors internals.

COL applicants may incorporate by reference provisions in the DCD provided with a Design Certification application as part of its description of the preoperational and startup testing programs in support of the COL application. The NRC staff will evaluate the COL application and portions of the Design Certification DCD incorporated by reference based on the Commission guidance in SECY-05-0197 that the operational program be fully described to support COL issuance. RGs 1.20 and 1.206 provide guidance for COL applicants with respect to the development and implementation of reactor internals vibration programs for new nuclear power plants. The NRC staff evaluates the COL application for compliance with the NRC regulations applicable to reactor internals vibration programs using the guidance in SRP Sections 3.9.2 and 3.9.5.

Operating experience at current BWR nuclear power plants has revealed failures of steam dryers and main steam system components (including relief valves) following implementation of extended power uprates (EPUs). These failures have demonstrated the importance of detailed analysis of potential adverse flow effects on the reactor 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 reactor internals and steam system components. In those failures, the instabilities of the separated shear flow over the standpipe openings “lock in” to the acoustic resonance of the fluid column within the standpipe, such that feedback occurs between the flow instability and the acoustic mode over a 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 reactor internals and steam dryer components can also produce FIV causing undesirable stresses that should be addressed. 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 conditions can cause a small adverse flow effect to magnify substantially, leading to structural failures. 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 more detailed information on the acoustic resonance phenomenon, see NUREG/CP-0152, Volume 6 (July 2006), "Proceedings of the Ninth NRC/ASME Symposium on Valves, Pumps and Inservice Testing," pp. 3B:49-69.

In RG 1.206, Section C.I.3.9.2.3, "Dynamic Response Analysis of Reactor Internals under Operational Flow Transients and Steady-State Conditions," the NRC staff recommends that COL applicants provide analytical methods and procedures to predict vibrations of PWR and BWR pressure vessel internals (including the BWR steam dryer and other main steam system components and PWR steam generator internals) that the referenced certified design does not address. The analysis should determine the dynamic responses to operational transients and hydrodynamic and acoustic loadings at locations where sensors would be mounted on the reactor internals (including steam dryers and main steam system components). Section C.I.3.9.2.4, "Preoperational Flow-Induced Vibration Testing of Reactor Internals," states that the COL applicant should provide a detailed analysis of potential adverse flow effects (e.g., FIV and acoustic resonances) that can severely impact PWR and BWR reactor pressure vessel internals (including the BWR steam dryer and other main steam system components and PWR steam generator internals) that are not covered in the referenced certified design. Acoustic and computational fluid dynamic analyses and scale model testing should supplement the analysis. The COL applicant should describe the utilization of instruments on vulnerable components (including pressure, strain, and acceleration sensors on the steam dryer) to obtain direct loading data to ensure structural adequacy of the components against the potential adverse flow effects. Section C.I.3.9.2.6, "Correlation of Reactor Internals Vibration Tests with the Analytical Results," of RG 1.206 states that the COL applicant should provide details of the test program to correlate the test measurements with the analytically predicted flow-induced dynamic response of the BWR and PWR reactor internals (including BWR steam dryers and other main steam system components and PWR steam generator internals) not addressed in the referenced certified design. Section C.I.3.9.5.4, "BWR Reactor Pressure Vessel Internals Including Steam Dryer," states that the COL applicant should present a detailed analysis of potential adverse flow effects (e.g., FIV and acoustic resonances) that can severely impact BWR reactor pressure vessel internals (including the steam dryer) and other main steam system components not covered in the referenced certified design. For a prototype reactor, if the applicant has not completed the FIV testing of reactor internals at the time it files the COL application, the applicant should describe the implementation program, including milestones and completion dates.

Revision 3 to RG 1.20 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 that 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 provide 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 should be sufficiently flexible to permit the definition of any significant vibratory modes that are present but were not included in the analysis. In addition, the inspection program should be used for both quantitative and qualitative verification of the analytical and measurement program results.

The NRC currently identifies approximately 4,000 codes and standards in regulations, regulatory guides, branch technical positions, the standard review plan, inspection procedures and NUREG documents. According to SECY 99-029, approximately 20 voluntary consensus

standards are mandated in NRC regulation. A number of such consensus standards applicable to the comprehensive vibration assessment programs for reactor internals are as follows:

(1) ASME OM-S/G-Part 3, “Requirements for Preoperational and Initial Start-up Vibration Testing of Nuclear Power Plant Piping Systems” (Including Addenda).

This Part establishes the requirements for preservice and initial startup testing to assess the vibration of certain piping systems used in light-water reactor (LWR) power plants. This Part may serve as a guide to assess vibration levels of applicable piping system during plant operation. The piping covered is that required to perform a specific function in shutting down a reactor to the safe shutdown condition, in maintaining the safe shutdown condition, or in mitigating the consequences of an accident. This Part establishes test methods, test intervals, parameters to be measured and evaluated, acceptance criteria, corrective actions, and records requirements.

(2) ASME OM-S/G-Part 7, “Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems”

This Part provides guidance for preservice and inservice testing to assess the thermal expansion of certain piping systems used in LWR power plants. The piping covered is that required to perform a specific function in shutting down a reactor to the safe shutdown condition, in maintaining the safe shutdown condition, or in mitigating the consequences of an accident. This Part establishes test methods, test intervals parameters to be measured and evaluated, acceptance criteria, corrective actions, and records requirements.

(3) BWR Vessel and Internal Project (BWRVIP Report 182) – “Guidelines for Demonstration of Steam Dryer integrity for Power Uprate”- September 2007

The purpose of this document is to provide guidance that can be followed by BWR utilities applying for a power uprate, but the information contained therein may be applicable in demonstrating the structural integrity of the steam dryer up to the highest planned power level. The loading on the steam dryer and associated stresses which need to be defined at appropriate power levels by the applicant are discussed in this document. An overall approach for demonstrating steam dryer structural integrity with the aid of scale model tests and analytical models is defined in detail. The requirements relating to the technical basis for and benchmarking of any analytical or testing methodologies utilized in demonstrating steam dryer integrity and submission to NRC for review have been provided. Specific acceptance criteria and values for key parameters to be used in the evaluation of steam dryers are defined. The document is intended to comply with the guidance provided in RG 1.20. While this report addresses only the steam dryer, the pressure fluctuations inside the MSLs may also have a detrimental effect on MSL instrumentation and other components, such as relief valve operators. Potential detrimental effects on such components as a result of MSL vibrations at power uprate conditions shall also be addressed as part of a power uprate submittal. Techniques for conducting such an assessment are not addressed in this report.

In RG 1.20, the NRC staff recommends that nuclear power plant applicants use specific classifications to categorize reactor internals according to design, operating parameters, and operating experience with potential prototypes. Applicants would then establish an appropriate comprehensive vibration assessment program using the guidelines as they relate to each classification. The classifications specified in RG 1.20 include various types of prototype (valid or conditional) and non-prototype (Categories I to IV) reactor internals.

With respect to prototype reactor internals, the comprehensive vibration assessment program described in RG 1.20 should be implemented in conjunction with preoperational and initial startup testing. 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 should perform a detailed analysis of potential adverse flow effects that can severely impact reactor internal components (including BWR steam dryers) and other main steam components. The applicant 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. The applicant 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 applicant 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 BWR steam dryers and main steam system components and PWR steam generator internals, that might be adversely affected by the flow-excited acoustic resonances and flow-induced vibrations. For initial plant startup, the nuclear power plant licensee should collect plant data from instrumentation mounted directly on the BWR steam dryer at significant locations to verify that the stress on individual steam dryer components is within the allowable limits during plant operation. 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 data acquisition and reduction system, and the test operating conditions. Plant operating experience has shown that adverse flow effects might not appear for an extended period of time following initial startup and power ascension. Therefore, it would be beneficial to maintain the program for monitoring potential adverse flow effects on plant systems and components for a sufficient time period to verify that adverse flow effects are not occurring at new nuclear power plants.

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. 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 tabulation of reactor internal components and local areas to be inspected, tabulation of specific inspection areas that can be used to verify the vibration analysis and measurement program, and a description of the inspection procedure, including the examination method, documentation, provisions for reactor internal access, and special equipment to be employed during the inspections to detect and quantify vibration effects.

The COL holder should review and correlate the results of the vibration and stress analysis, measurement, and inspection programs to determine the extent to which the test acceptance criteria are satisfied. A summary of the results should be submitted to the NRC. A schedule for

the vibration assessment program should be established and submitted to the NRC during the review of Design Certification and COL applications.

With respect to non-prototype reactor internals, the comprehensive vibration assessment program should be developed using the guidance in RG 1.20 for the four categories of non-prototype reactor internals. Where justified, information obtained from comprehensive vibration assessment programs for prototype reactor internals may be used to support the programs 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. A vibration and stress analysis program, vibration measurement program, and inspection program should be developed for non-prototype Category I through IV reactor internals using the guidance in RG 1.20.

Where vibration assessment test data obtained from foreign prototype reactor internals have been used to support the programs for domestic non-prototype reactor internals, it is incumbent upon the applicant to validate and submit the applicable foreign prototype test data for NRC review and approval.

Conclusion

The NRC has certified several new reactor designs and is evaluating applications for the certification of additional reactor designs. The NRC is also evaluating applications from nuclear industry organizations for licenses to construct and operate new nuclear power plants. In the review of the reactor internals vibration assessment program described in the DCD for a Design Certification application, the NRC staff focuses on system and component design and accessibility for performance, preoperational and startup testing activities, and the general description of related programs. As part of the review of a COL application, the NRC will determine whether the descriptions of the startup testing and operational programs provide reasonable assurance of safe plant operation. COL applicants may rely on a combination of DCD and FSAR information to fully describe their preoperational and startup programs to support their COL applications. The NRC will review Design Certification and COL applications for their compliance with the NRC regulations for reactor internals vibration assessment programs using guidance in applicable SRP sections and regulatory guides.