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Testing Programs Related to Potential Adverse Flow Effects In Nuclear Power Plants

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Abstract – *Potential adverse flow effects on nuclear power plant Structures, Systems and Components, (SSCs), can result from various causes. For example, reactor vessel and main steam system piping and components (including the steam dryer, safety relief and power-operated valves, and pipe supports) in BWR nuclear power plants can be damaged by pressure fluctuations and vibration resulting from acoustic resonances occurring in the main steam system or reactor vessel. The acoustic resonance phenomenon can also occur in pressurized water reactor (PWR) nuclear power plants with resulting damage to plant piping and components. Sampling probes in feedwater and condensate systems in nuclear power plants are also susceptible to adverse flow effects.*

The NRC staff reviews the evaluation performed by applicants for the construction and operation of nuclear power plants under 10 CFR Part 50 or Part 52 of the potential for adverse flow effects on plant SSCs. This proposed paper will discuss the various aspects of the programs established by applicants and licensees for monitoring plant data to verify that adverse flow effects are not occurring. These aspects of the test programs include the following: conducting walkdowns, and inspecting components during power ascension and operation at full licensed power conditions, dynamic response analysis of reactor internals under operational flow transients and steady-state conditions; preoperational flow-induced vibration testing of reactor internals; correlation of the test results with the analytical results.

Enclosure 2

I. INTRODUCTION

In March 2007, the NRC issued Revision 3 to the Regulatory Guide (RG) 1.20 “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing,” to incorporate lessons learned from nuclear power plant operating experience regarding the effects of flow-induced vibration. In addition to reactor internals, the NRC staff included information on methods for evaluating potential adverse effects from pressure fluctuations and vibrations in piping systems for BWR and PWR nuclear power plants. This paper summarizes guidance provided in Revision 3 to RG 1.20.

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. 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.

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 (e.g., MSL branch-lines with closed safety relieve valves). 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.

II. 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 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.

Licensees of operating nuclear power plants should obtain plant-specific data to confirm the scale testing and analysis results for pressure fluctuations and vibration when planning a power uprate.

If scale model testing is used, 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 CFD models are used, 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 considered. 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 considered, 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 evaluated (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. The 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 determine 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 determined. 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 effort 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, The margins against violating the criteria should be reported. Based on the uncertainties and bias errors, an end-to-end uncertainty and bias error should be determined, along with a clear evaluation of how the individual uncertainties and bias errors are combined.

Since the transfer functions (or FRFs) and forcing functions 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 use 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 requested, 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 effort 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 suggested 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 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 and Pressure Vessel Code.

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.

III 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.

(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 cycles of vibration (i.e., computed at the lowest 6 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).

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.

IV. INSPECTION PROGRAM

The inspection program should provide for inspections of the reactor internals prior to and following operation in steady-state and transient modes. 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 should

include the following:

(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; and

(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

V. 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 documentation of the results is as follows:

(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.

VI. CONCLUSION

Pressure fluctuations and flow induced vibration occurring in the main steam system of a power plant can cause significant damage to safety and non-safety related Structures, Systems and Components. NRC RG 1.20 presents a comprehensive vibration assessment program and describes an implementing methodology to verify the structural integrity of reactor internals and

piping systems for flow-induced vibrations prior to commercial operation. The overall program includes individual analytical, measurement, and inspection programs.

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. This revision incorporates information obtained from operating experience. This revision also provides new guidance for steam dryers in BWR plants and helpful information for monitoring programs for plant components outside the reactor vessel.

REFERENCES

1. Regulatory Guide 1.20 Revision 3, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," U.S. Nuclear Regulatory Commission, Washington, DC.