NUREG-0800

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

3.9.5 REACTOR PRESSURE VESSEL INTERNALS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for mechanical engineering reviews

Secondary - None

I. AREAS OF REVIEW

Reactor pressure vessel (RPV) internals consist of all structural and mechanical elements inside the reactor vessel. General Design Criteria (GDCs) 1, 2, 4, and 10, 10 CFR 50.55a, and 10 CFR Part 52 require that structures and components important to safety be constructed and tested to quality standards commensurate with the importance of the safety functions performed and designed with appropriate margins to withstand effects of anticipated operational occurrences and normal operation, natural phenomena like earthquakes, postulated accidents including loss-of-coolant accidents (LOCA), and events and conditions outside the nuclear power unit.

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 safety functions.

For the purpose of this standard review plan (SRP) section, the term "reactor internals" includes core support and other internal structures and refers to all structural and mechanical elements inside the RPV with the exception of the following:

1. Reactor fuel elements and the reactivity control elements out to the coupling interfaces with the drive units (the fuel system design is covered in SRP Section 4.2, but the structural aspects of reactor fuel assemblies are reviewed with the reactor internals in this SRP section).

2. Control rod drive elements (the drive elements inside the guide tubes are covered in SRP Section 3.9.4, but the guide tubes are reviewed with the reactor internals in this SRP section).

3. In-core instrumentation (in-core instrumentation support structures are reviewed with the reactor internals in this SRP section).

The specific areas of review are as follow:

1. The physical or design arrangements of all reactor internals structures, components, assemblies, and systems, including the positioning and securing of such items within the RPV, the provision for axial and lateral retention and support of the internals assemblies and components, and the accommodation of dimensional changes due to thermal and other effects.

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. All combinations of design and service loadings (*e.g.*, operating differential pressure and thermal effects, potential adverse flow effects (flow-excited vibrations and acoustic resonances), seismic loads, and transient pressure loads of postulated loss-of-coolant accidents (LOCAs)) accounted for in design of the reactor internals should be listed. The distribution of the design and service loadings acting on the internal components and structures should be described. The analytical or experimental methods for determining the loading conditions and their validation should be described along with their random uncertainties and bias errors. Draft Regulatory Guide DG-1163 (Revision 3 of RG 1.20) provides further details on the determination of loading conditions caused by adverse flow effects.

3. If computational methods (*e.g.*, the finite element method) are used to determine stresses in the reactor internal components and structures, validation of the modeling procedures for the analyses should be presented. The validation may include comparisons of simulated natural frequencies, mode shapes, and frequency response functions with experimental results.

4. The design bases for the mechanical design of the reactor vessel internals, including such allowable limits as maximum allowable stresses; stability under dynamic loads; deflection, cycling, and fatigue limits; and core mechanical and thermal restraints (positioning and holddown). Details of dynamic analyses,

input forcing functions, and response to loadings (including those due to adverse flow effects) are addressed in SRP Section 3.9.2. Justification for the structural damping value(s) for the dynamic analyses should be provided. The bias errors and random uncertainties of dynamic analysis should be listed.

5. Each combination of design and service loadings, categorized by allowable design or service limits (defined in the American Society of Mechanical Engineers (ASME) Code (Reference 10) and SRP Section 3.9.3), and stress intensity or deformation limits. Design or service loadings should include safe shutdown earthquake and operating basis earthquake loads as appropriate.

6. Potential adverse flow effects on the reactor pressure vessel internals including the steam dryer in BWRs (See the Appendix A of this SRP Section, SRP Section 3.9.2, and Draft RG DG-1163).

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

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

Review Interfaces

The listed SRP sections interface with this section as follows:

1. Section 3.6.2: evaluation of rupture locations, rupture loads, and dynamic effects of postulated rupture of piping.

2. Section 3.9.1: evaluation of the adequacy of analysis methods for seismic Category I RPV internals and system dynamic analysis, identification of design transients and of service lifetime transient cyclic loadings to be reflected in the design and fatigue analyses of RPV internals.

3. Section 3.9.2: evaluation of the adequacy of dynamic analyses under steady state and operational flow transient conditions and the proposed program for preoperational and startup testing of flow-induced vibration and acoustic resonance for RPV internals.

4. Section 3.9.3: evaluation of the adequacy of the structural integrity design of the RPV internals, including adequacy the of design fatigue curves for reactor internals materials to account for cumulative reactor service-related environmental and usage factor effects and consideration of each combination of design, service, and postulated event loadings.

5. Section 3.9.4: evaluation of the adequacy of the mechanical design of the control rod drive system, including the control rod drive elements.

6. In addition, other evaluations that interface with the overall review of the reactor internals are coordinated as follows:

   A. Section 3.13: review of the adequacy of programs for assurance of integrity of bolting and threaded fasteners.

   B. Section 3.6.3: evaluation of the adequacy of analyses justifying exclusion of certain postulated pipe ruptures from design bases.

   C. Section 4.2: verification of fuel system design, including fuel behavior effects on reactor core design under various normal and accident operating conditions.

   D. Section 4.5.2: review of material aspects of reactor internals.

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

II. ACCEPTANCE CRITERIA

Requirements

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

1. GDC 1 and 10 CFR 50.55a require that reactor internals be designed to quality standards commensurate with the importance of the safety functions performed.

2. GDC 2 requires that reactor internals be designed to withstand the effects of natural phenomena such as earthquakes without loss of capability to perform safety functions.

3. GDC 4 requires that reactor internals be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated pipe ruptures, including LOCAs. Dynamic effects associated with postulated pipe ruptures may be excluded from the design basis when analyses demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for piping.

4. GDC 10 requires that reactor internals be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

5. 10 CFR 52.47(a)(1)(vi), as it relates to ITAAC (for design certification) sufficient to assure that the SSCs in this area of review will operate in accordance with the certification.

6. 10 CFR 52.97(b)(1), as it relates to ITAAC (for combined licenses) sufficient to assure that the SSCs in this area of review have been constructed and will be operated in conformity with the license and the Commission's regulations.

SRP Acceptance Criteria

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

1. Requirements for loads, loading combinations, and limits applicable to those portions of reactor internals constructed to Subsection NG of the ASME Code are presented in SRP Section 3.9.3.

2. The design and construction of the core support structures should comply with the requirements of Subsection NG, "Core Support Structures," of the ASME Code and SRP Section 3.9.3.

3. The design criteria, loading conditions, and analyses that provide the bases for the design of reactor internals other than the core support structures should meet the guidelines of NG-3000 and be constructed not to affect the integrity of the core support structures adversely (NG-1122). If other guidelines (*e.g.*, manufacturer standards or empirical methods based on field experience and testing) are the bases for the stress, deformation, and fatigue criteria, those guidelines should be identified and their use justified.

4. Deformation limits for reactor internals should be established by the applicant and presented in the safety analysis report. The basis for these limits should be included. The stresses of these displacements should not exceed the specified limits. The requirements for dynamic analysis of these components are addressed in SRP Section 3.9.2.

5. The reactor internals should be designed to accommodate asymmetric blowdown loads from postulated pipe ruptures. The applicant's evaluation of such loads should demonstrate that they do not exceed the limits imposed by the applicable codes and standards. Where double-ended guillotine break of reactor coolant piping is postulated, criteria for evaluating loading transients and structural components are specified in NUREG-0609.

6. Potential adverse flow effects of flow-induced vibration (FIV) and acoustic resonances on reactor internals (including the steam dryer in BWRs) should be adequately addressed in accordance with relevant criteria stated in the Appendix to this SRP Section.

<u>Technical Rationale</u>

The technical rationale for application of these requirements to reviewing this SRP section is discussed in the following paragraphs:

1. GDC 1 and 10 CFR 50.55a require that SSCs important to safety be designed to quality standards commensurate with the importance of the safety functions performed. The reactor internals include SSCs performing safety functions and SSCs whose failure can affect the performance of other SSC safety functions, including reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the primary reactor coolant system). Application of this requirement to the reactor internals provides assurance that established design practices of proven or demonstrated effectiveness achieve a high likelihood that these safety functions will be performed.

2. GDC 2, in relevant part, requires that SSCs important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. The reactor internals perform or may (through their failure) affect the performance of safety functions like core cooling and fission product confinement. Application of GDC 2 to the reactor internals provides assurance that they will withstand earthquakes without damaging fuel cladding or interfering with core cooling.

3. GDC 4, in relevant part, requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with environmental conditions of normal operations, maintenance, testing, and postulated accidents, including LOCAs. The reactor internals perform or (through their failure) may affect the performance of safety functions like reactivity monitoring and control, core cooling, and fission product confinement. Application of GDC 4 to the reactor internals provides assurance that the effects of environmental conditions to which they are exposed over their installed life will not diminish the likelihood of performance of these safety functions under all operating conditions, including accidents. This provides assurance that failures of the reactor internals from environmental service conditions that could cause loss of capability to monitor reactivity, fuel damage from loss of reactivity control, structural damage to fuel cladding, or interference with core cooling are not likely to occur.

   NUREG-0609 evaluates certain postulated pipe ruptures (*e.g.,* double-ended guillotine breaks of primary reactor coolant loop piping) known to cause asymmetric blowdown loadings on the reactor internals. GDC 4 allows such dynamic effects of postulated pipe ruptures to be excluded from the design basis when analyses approved by the staff demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. Application of GDC 4 to the reactor internals provides assurance that asymmetric loading effects of postulated pipe ruptures are either accommodated in the design (with assurance of the functionality and integrity of reactor internals) or demonstrated to be extremely unlikely to occur and that overstress failures of the reactor internals that could cause loss of capability to monitor reactivity, fuel damage resulting from loss of reactivity control, structural damage to fuel cladding, or interference with core cooling are unlikely to occur.

4. GDC 10 requires that the reactor core and its coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. The reactor internals perform or may (through their failure) affect the performance of safety functions like reactivity control and core cooling essential for assurance that specified acceptable fuel design limits are not exceeded. Application of GDC 10 to the reactor internals provides assurance of design with sufficient margin to ensure functionality and integrity during any condition of normal operation, including the effects of anticipated operational occurrences, to achieve a high likelihood of performance of these safety functions. Assured performance of these safety functions in turn assures that specified acceptable fuel design limits for reactivity control and core cooling are not exceeded, thus assuring the integrity of the fuel and its cladding.

III. REVIEW PROCEDURES

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

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

1. The configuration and general arrangement of all mechanical and structural internal elements covered by this SRP section are reviewed and compared to those of previously licensed similar plants. Any significant changes in design or operating conditions are noted and the applicant is asked to confirm that such changes do not affect the FIV test results required by SRP Section 3.9.2.

2. As to the design and analysis of reactor internals, a statement by the applicant that they are designed in accordance with Subsection NG of the ASME Code and SRP Section 3.9.3, "Core Support Structures," is acceptable. In lieu of such a commitment, the reviewer must determine whether the design and analysis of these components are consistent with the requirements addressed in Subsection II of this SRP section by requiring the applicant to describe the procedures and criteria for the design of these components, including a list of the design and service stress limits for all of the

applicable loading conditions. The stresses of adverse flow conditions (flow-excited vibrations and acoustic resonances) should be treated as primary stresses while satisfying the ASME Code service stress limits.

3. The reviewer verifies whether the asymmetric blowdown loadings upon reactor internals from pipe ruptures (at postulated locations not excluded in leak-before-break analyses) have been evaluated by the applicant and are accommodated in the design consistently with criteria in Subsection II.5 (Acceptance Criteria) of this SRP section.

4. The deformation limits specified for these components are reviewed to verify whether the applicant has stated that these deflections will not interfere with the function of related components (*e.g.*, control rods and standby cooling systems) and whether the stresses of these displacements are less than the specified limits for the core support structures.

5. At the operating license stage, the calculated stresses and deformations are reviewed for whether they exceed the specified limits.

6. The staff reviews the consideration of potential adverse flow effects on plant systems (including the steam dryer in BWRs) for operation up to full licensed power conditions by an applicant for the construction and operation of a nuclear power plant or by a licensee of an operating nuclear power plant proposing a power uprate license amendment or a major plant modification in accordance with criteria and procedures stated in the Appendix to this SRP Section. The staff determines whether the applicant/licensee has provided reasonable assurance that the flow-induced effects on plant systems, piping, components, and reactor internals in particular will continue to meet the GDC requirements during power ascension and long-term plant operation.

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

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

IV. EVALUATION FINDINGS

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

The staff concludes that the design of reactor internals is acceptable and meets the requirements of GDCs 1, 2, 4, and 10, 10 CFR 50.55a, and/or 10 CFR Part 52. This conclusion is based on the following findings:

1. The applicant has met the requirements of GDC 1, 10 CFR 50.55a, and/or 10 CFR Part 52 by designing the reactor internals to quality standards commensurate with the importance of the safety functions performed. The design procedures and criteria for the reactor internals are in compliance with the requirements of Subsection NG of the ASME Code, Section III. The applicant has adequately evaluated the potential adverse flow effects on the reactor internals, including steam dryer in a BWR, up to full licensed power conditions.

2. The applicant has met the requirements of GDCs 2, 4, and 10 by designing components important to safety to withstand the effects of earthquakes and of normal operation, maintenance, testing, and postulated accidents (including LOCAs) with sufficient margin to maintain their capability to perform safety functions. The applicant also has designed the reactor internals with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

   The specified design transients, design and service loadings, and combination of loadings as applied to the design of the reactor internals structures and components provide reasonable assurance that in an earthquake or a system transient during normal plant operation the consequent deflections and stresses imposed on these structures and components would not exceed allowable stresses and deformation limits for the materials of construction. Limitation of stresses and deformations under such loading combinations is an acceptable basis for the design of these structures and components to withstand the most adverse loading events postulated to occur during service lifetime without loss of structural integrity or impairment of function.

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

V. IMPLEMENTATION

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

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

VI. REFERENCES

1. 10 CFR 50.55a, "Codes and Standards."

2. 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records."

3. 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection Against Natural Phenomena."

4. 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases."

5. 10 CFR Part 50, Appendix A, GDC 10, "Reactor Design."

6. 10 CFR Part 52, "Early Site Permit; Standard Design Certification; and Combined Licenses for Nuclear Power Plants."

7. NUREG-0609; "Asymmetric Blowdown Loads on PWR Primary Systems: Resolution of Generic Task Action Plan A-2;" Hosford, S.B.; Mattu, R.; Meyer, R.O.; Division of Safety Technology; January, 1981.

8. Draft RG DG-1163, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing."

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

10. ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components," ASME.

VII. APPENDIX

"NRC Review of Potential Adverse Flow Effects in Nuclear Power Plant Systems."

APPENDIX A
SRP Section 3.9.5

NRC REVIEW OF POTENTIAL ADVERSE FLOW EFFECTS IN NUCLEAR POWER PLANT SYSTEMS

Regulatory Evaluation

Nuclear power plant operation can encounter adverse flow effects of flow-excited vibrations and acoustic resonances on plant systems and their components. Some plant components, like the steam dryer in a boiling water reactor (BWR) nuclear power plant, perform no safety function but must retain their structural integrity to avoid the generation of loose parts that might adversely impact the capability of other plant equipment to perform safety functions. Adverse flow may also affect the safety function of components like safety relief valves on the BWR main steam lines. The staff reviews the consideration of potential adverse flow effects at BWR and pressurized-water reactor (PWR) nuclear power plants under construction, requesting power uprates, or proposing major plant modifications (*e.g.*, steam generator replacement). The staff's review includes consideration of the design input parameters and the design-basis loads and load combinations for plant components for normal operation, upset, emergency, and faulted conditions. The review also covers the analytical methodologies, assumptions, computer programs, and code and code edition for the evaluation of the plant components, including the method of determining load definition and the uncertainties and bias errors of analytical and measurement procedures. The review also includes a comparison of the stresses against code allowable limits. NRC acceptance criteria are based on (1) requirements of general design criteria (GDC) 1 in Appendix A to 10 CFR Part 50 that structure, system, and component (SSC) essential to the prevention of accidents that can could affect the public health and safety or to mitigation of their consequences be designed, fabricated, erected, tested, and inspected to quality standards commensurate with the importance of the safety functions performed; (2) requirements of GDC 2 that systems and components essential to the prevention of accidents that can could affect the public health and safety or to mitigation of their consequences be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; and (3) requirements of GDCs 40 and 42 for protection of engineered safety features against dynamic effects and missiles that might ensue from plant equipment failures as well as the effects of a loss of coolant accident.

Technical Evaluation

Potential adverse flow effects on nuclear power plant SSCs can ensue from various flow excitation mechanisms like fluid-elastic instability, vortex-induced vibration, flow-excited acoustic resonance, and turbulent buffeting. 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 from flow-excited acoustic resonances within the main steam system 1or reactor vessel. The flow-excited acoustic resonance phenomenon also can occur in PWR nuclear power plants with 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 staff reviews evaluations of the potential for adverse flow effects on plant SSCs by applicants for the construction and operation of nuclear power plants under 10 CFR Part 50 or Part 52. The staff reviews the evaluation by operating nuclear power plant licensees of the potential for adverse flow effects on plant SSCs of proposed changes in licensed operating conditions (*e.g.*, in support of a power uprate license amendment request) or planned major plant modifications (*e.g.*, replacement of a steam generator at a PWR nuclear power plant). The staff reviews the program established by applicants and licensees for analyzing potential adverse flow effects on plant components and the flow-induced vibration (FIV) startup test program, including monitoring of plant data, conducting walkdowns, and inspecting components during power ascension and operation at full licensed power conditions to verify that adverse flow effects do not occur.

As adverse flow effects in nuclear power plants caused by flow-excited acoustic and structural resonances are sensitive to minor changes in arrangement, design, size, and operating conditions, even applications submitted for minor modifications in non-prototype plants must include rigorous assessments of the potential for such adverse effects. A nuclear power plant nearly identical to another can experience significant such adverse flow effects as valve and steam dryer failures while the other does not. Small changes in operating condition can magnify a small adverse flow effect substantially, leading to structural failures. Draft Regulatory Guide (RG) DG-1163 offers specific guidance for these assessments from both analyses and measurements.

The following is a summary of the approach to be used by the staff in reviewing the consideration by applicants and licensees of potential adverse flow effects on nuclear power plant systems and components depending on the applicable BWR or PWR design. Additional details of the evaluation procedure and acceptance criteria are outlined in standard review plan (SRP) Section 3.9.2 and DG-1163.

1. <u>Pressure Fluctuations and Vibration in Plant Systems</u>. An applicant to construct and operate a nuclear power plant and a licensee of an operating nuclear power plant proposing a power uprate license amendment or a major plant modification (*e.g.*, a PWR steam generator replacement) is expected to determine the pressure fluctuations and vibration in the applicable plant systems under flow conditions up to and including the full operating power level. The pressure fluctuations and vibration can come from various excitation mechanisms like instabilities of separated flows, FIVs, and flow-excited acoustic resonances under the plant system fluid flow conditions. The applicant/licensee is expected to justify the method for determining pressure fluctuations and vibration in plant systems. Experience indicates that computational fluid dynamics (CFD) analyses might not provide sufficient quantitative information about unsteady pressure loading without supplemental analyses. Scale model testing and analysis can evaluate potential adverse flow effects, including hydrodynamic and acoustic pressure loading, and verify the pressure loading results from CFD analyses and the supplemental analyses where the bias error and random uncertainties are properly addressed. Subsections II.3 and II.4 of SRP Section 3.9.2 state specific acceptance criteria and review procedures for methods for evaluating applicable plant systems and components, including dynamic analysis methodology, CFD simulations, scale model testing, and plant startup testing. The applicant/licensee is expected to address possible flow-excited acoustic resonances and FIVs with the potential to damage plant piping and components (like the steam dryer) by modifications, if necessary, to reduce

the amplitudes of the acoustic resonances and vibrations. Licensees of operating nuclear power plants are expected to obtain plant-specific data to confirm the scale model testing, structural and acoustic analysis, and CFD results (as applicable) for pressure fluctuations and vibration prior to submitting a power uprate request or proposing a major plant modification. For a power uprate, the extent of the uprate can be considered in determining the detailed analysis and tests necessary to address potential adverse flow effects.

2. Design Load Definition for BWR Steam Dryers. An applicant for the construction and operation of a BWR nuclear power plant and a licensee proposing a power uprate for an operating BWR nuclear power plant are expected to determine the design load definition for the steam dryer up to the full licensed power level. The applicant/licensee uses this design load definition to analyze the BWR steam dryer for American Society of Mechanical Engineers (ASME) Code load combinations for limiting transient and accident events, including normal, upset, emergency, and faulted conditions. The stresses due to adverse flow conditions (flow-induced vibrations and acoustic resonances) should be treated as primary stresses and included in ASME Code load combinations. As addressed in SRP Section 3.9.2, the applicant/licensee is expected to validate the method used to determine the steam dryer load definition based on scale model, CFD, or plant data. The BWR applicant is expected to include instrumentation on the steam dryer to measure pressure loading, strain, and vibrations to confirm any scale model testing and/or the results of dynamic and acoustic analysis. The BWR licensee might not need to instrument its original steam dryer for a power uprate, but will be expected to obtain additional necessary plant data at current licensed power conditions (*e.g.*, pressure fluctuations in main steam lines) for use in confirmation of the results of any scale model testing and/or analysis for the steam dryer load definition prior to the power uprate request.

3. BWR Steam Dryer Stress and Limit Curves. After developing a steam dryer load definition, an applicant for the construction and operation of a BWR nuclear power plant and a licensee proposing a power uprate for an operating BWR nuclear power plant apply the steam dryer load definition to a stress model (usually a finite element model) of the steam dryer to determine the stress on the steam dryer with justified damping assumptions and applicable weld factors and stress intensities. Use of damping coefficients greater than 1 percent should be substantiated with measurements. After including applicable bias errors and random uncertainties, the applicant/licensee compares the peak stresses for individual steam dryer locations to the fatigue limits in the ASME Boiler & Pressure Vessel Code. The applicant/licensee also compares the primary and secondary stresses on the steam dryer from plant transients to the applicable ASME Code Service Level limits. The stresses from adverse flow effects (flow-excited vibrations and acoustic resonances) should be treated as primary stresses. The BWR applicant/licensee is expected to implement any necessary modifications to the BWR steam dryer based on the design stress margins. The BWR applicant/licensee also is expected to develop a stress limit curve for power ascension for assurance that the stress in the individual steam dryer components will not exceed the ASME Code limits. The BWR applicant/licensee develops a method that can be used for calculating the steam dryer stress, including appropriate bias errors and random uncertainties, for collecting plant data during power ascension and full licensed

power conditions. As the steam dryer is not a Code component, the applicant/licensee may justify different stress acceptance criteria.

4. PWR Steam Generator Stress and Design Margin. An applicant for the construction and operation of a PWR nuclear power plant and a licensee proposing a power uprate or planning a major plant modification at an operating PWR nuclear power plant is expected to evaluate the dynamic response, stress, and design margin of the internal components in the steam generators. This evaluation is expected to address potential adverse flow effects caused by flow-excited vibrations and acoustic resonances on the steam generator tube bundle, moisture separator, steam dryer, main steam piping, and safety valves. Past operating experience and analysis may support the determination of adequate design margin for the stress on PWR steam generator internal components.

5. Evaluation of Other Plant Components. An applicant for the construction and operation of a nuclear power plant and a licensee of an operating BWR or PWR nuclear power plant proposing a power uprate license amendment or to implement a major plant modification are expected to evaluate potential adverse effects from pressure fluctuations and vibration on piping and components of plant systems, including the reactor coolant, steam, feedwater, and condensate systems, up to full proposed operating conditions. These plant components include safety relief valves, power-operated valves, and sampling probes. Experience indicates that particular attention should be paid to cantilevered piping and components. For example, steam flow over the opening of a safety valve standpipe and feedwater flow around the sampling probes may excite acoustic resonances or cause flow-induced vibration that can damage associated components (valves, steam dryer, or pipes). The applicant/licensee is expected to make any necessary modifications to plant piping or components based on the results of this evaluation, as necessary, to increase the structural capability of that equipment, or to reduce the pressure fluctuation and vibration levels.

6. Power Ascension Data. An applicant for the construction and operation of a BWR or PWR nuclear power plant and a licensee of an operating BWR or PWR nuclear power plant proposing a power uprate license amendment or a major plant modification is expected to develop a program to confirm its evaluation of the potential adverse flow effects on plant systems and components during power ascension and operation at full licensed power conditions. For example, the applicant/licensee is expected to establish a power ascension program that includes, as applicable, (a) specific hold points and their durations during power ascension; (b) activities accomplished during the hold points; (c) plant parameters monitored and applicable limit curves; (d) inspections and walkdowns of steam, feedwater, and condensate systems and components during the hold points; (e) methods for trending plant parameters; (f) acceptance criteria for monitoring and trending plant parameters and conducting walkdowns and inspections; (g) actions taken if acceptance criteria are not satisfied; and (h) provisions for informing the staff about plant data, evaluations, walkdowns, inspections, and procedures prior to and during power ascension, including interactions during hold points and any instance where acceptance criteria are not satisfied, and resolution of safety concerns during the staff review of that information prior to further power ascension or continued full power operation. For a BWR plant with an instrumented steam dryer, the steam dryer stresses are determined from the direct instrumentation and compared to the applicable limit

curves with appropriate bias errors and random uncertainties. For an operating BWR plant without an instrumented steam dryer, the steam dryer stresses are calculated from data from steam system instrumentation fully accounting for appropriate bias errors and random uncertainties. The applicant/licensee is expected to provide a summary of its evaluation of plant startup and power ascension to the staff within 90 days of reactor criticality. If full licensed power is not achieved in that time period, the applicant/licensee is expected to provide a supplemental report within 30 days of achieving full licensed power. The extent of NRC staff interaction will depend on the evaluation results and plant experience.

7. Long-term Monitoring of Potential Adverse Flow Effects. An applicant for the construction and operation of a BWR or PWR nuclear power plant and a licensee of an operating BWR or PWR nuclear power plant proposing a power uprate license amendment or a major plant modification is expected to develop a program for monitoring potential adverse flow effects on plant systems and components over the long term. This program is expected to include monitoring plant data with vibration sensors at critical locations, conducting walkdowns, and inspecting components during power ascension and operation at full licensed power conditions to verify whether adverse flow effects occur. The program also is expected to include inspections and walkdowns during refueling outages and extended shutdowns with appropriate ALARA consideration. This program may be adjusted based on operating experience. The BWR Owners' Group BWR Vessel Internals Project has developed guidance for inspections of steam dryers at BWR nuclear power plants that may be applied to address NRC comments. The shutdown inspection methods are expected to be qualified for detecting fatigue and intergranular stress-corrosion cracks in reactor internal components.

Conclusion

The staff reviews the consideration of potential adverse flow effects on plant systems (including the steam dryer in BWR nuclear power plants) for power ascension up to operation at full licensed power conditions by an applicant for the construction and operation of a BWR or PWR nuclear power plant and a licensee of an operating BWR or PWR nuclear power plant proposing a power uprate license amendment or a major plant modification. From the review procedure and the acceptance criteria outlined in SRP Section 3.9.2 and DG-1136, the staff determines whether the applicant/licensee has provided reasonable assurance that the adverse flow effects on plant systems, piping, and components will continue to meet the GDC requirements during power ascension and long-term plant operation. Therefore, the staff concludes, if appropriate, that operation of the nuclear power plant up to full licensed power conditions is acceptable with respect to potential adverse flow effects.

SRP Section 3.9.5
Description of Changes

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

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

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

1. This version of updated SRP Section 3.9.5 is a revision from SRP Section 3.9.5, Draft Rev. 3 - April 1996 with all the superscript numbers and the Attachments A&B deleted.

2. Added an Appendix, “NRC Review of Potential Adverse Effects in Nuclear Power Plant Systems,” at the end of this SRP.

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

Editorial changes were made to reflect the primary review branch for SRP Section 3.9.5.

I. AREAS OF REVIEW

1. Added regulatory requirement 10 CFR Part 52.

2. Added a paragraph to include, in the areas of review, the consideration of structure integrity of the steam dryer in a BWR.

3. Revised Item I.2 and added Items I.3 and I.6 to include the review of potential adverse flow effects on the reactor internals including steam dryer in a BWR.

4. Revised “Review Interfaces” to describe the review responsibility of interfacing review organizations.

II. ACCEPTANCE CRITERIA

Added acceptance criteria Item II.6 for the review of potential adverse flow effects on the reactor internals including steam dryer in a BWR.

III. REVIEW PROCEDURES

Revised the review procedures to cover the updates in Subsection II (Acceptance Criteria) and some editorial changes.

IV. EVALUATION FINDINGS

Revised to add regulatory requirement, 10 CFR Part 52, and a statement related to the review of potential adverse flow effects on the reactor internals including steam dryer in a BWR.

V. IMPLEMENTATION

No change.

VI. REFERENCES

Added new reference as cited in the updated SRP Section 3.9.5.

VII APPENDIX

Added Appendix, "NRC Review of Potential Adverse Flow Effects in Nuclear Power Plant Systems."