U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

3.9.5 REACTOR PRESSURE VESSEL INTERNALS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for mechanical engineering reviews

Secondary - None

I. AREAS OF REVIEW

The reactor pressure vessel (RPV) internals consist of structural and mechanical elements inside the reactor vessel. Under this DSRS for the mPower™ design, the reactor internal and core support structures reviewed include all structures and components as defined in NG-1120 of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code ("the Code"). In addition, any structure or component within the reactor vessel that is either a reactor coolant pressure boundary, or has emergency core cooling or safe shutdown capability, shall be treated according to their safety function, and not as a reactor internal. This includes, but is not limited to the steam generator (including tubes and tubesheet), the riser, and the control rod drives. Any supporting structures for these components, for example, control rod guide frames, tie rods, or the lattice plate, may be considered reactor internals/core supports. 10 CFR Part 50, Appendix A, 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. They must also be 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.

Nuclear power plant operation can encounter adverse flow effects of flow-excited vibrations and acoustic resonances on plant systems and their components. Since the reactor internal and core support structure is part of a system that is safety related, all reactor internals, including plant components important to safety but that perform no safety functions, 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. Nuclear power plant operation can encounter adverse flow effects of flow-excited vibrations and acoustic resonances on plant systems and their components.

The mPower™ reactor pressure vessel (RPV) internals include the following classifications of equipment:

1. Safety-related and risk-significant equipment

2. Safety-related and nonrisk-significant equipment

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3. Nonsafety-related and risk-significant Regulatory Treatment of Nonsafety Systems (RTNSS) equipment

4. Nonsafety-related nonrisk-significant equipment.

The mPower^{TM} application will include the classification of systems, structures, and components (SSCs); a list of risk significant SSCs; and a list of RTNSS equipment. In accordance with DSRS Section 3.2 and Standard Review Plan (SRP) Sections 17.4 and 19.3, the staff will review this information and confirm the determination of safety-related and risk-significant SSCs.

The specific areas of this DSRS review are as follows:

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, high neutron flux, 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 LOCA) 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. Regulatory Guide (RG) 1.20 provides further details on the determination of loading conditions caused by adverse flow effects (e.g., flow-induced excitation and pump-induced vibration).

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 ASME Code 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 (See Appendix A of this DSRS Section, SRP Section 3.9.2, and RG 1.20).

7. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the SSCs related to this DSRS section in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this DSRS section. Furthermore, the staff reviews the ITAAC to ensure 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. For a design certification (DC) application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

   For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

Other SRP and DSRS sections interface with this section as follows:

1. Evaluation of rupture locations, rupture loads, and dynamic effects of postulated rupture of piping is performed under DSRS Sections 3.6.2.

2. 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 is performed under DSRS Section 3.9.1.

3. 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 is performed under SRP Section 3.9.2.

4. 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 is performed under 3.9.3.

5. Evaluation of the adequacy of the mechanical design of the control rod drive system, including the control rod drive elements is performed under DSRS Section 3.9.4.

6. Review of the adequacy of programs for assurance of integrity of bolting and threaded fasteners is performed under DSRS Section 3.13.

7. Verification of fuel system design, including fuel behavior effects on reactor core design under various normal and accident operating conditions is performed under DSRS Section 4.2.

8. Review of material aspects of reactor internals is performed under DSRS Section 4.5.2.

9. Review of the Probabilistic Risk Assessment is performed under SRP Section 19 for potential risk significance of SSCs.

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(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Atomic Energy Act (AEA), and the U.S. Nuclear Regulatory Commission's (NRC's) regulations.

6. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the AEA, and the NRC's regulations.

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for review described in this DSRS section. The DSRS is not a substitute for the NRC's regulations, and compliance with it is not required. Identifying the differences between this DSRS section and the design features, analytical techniques, and procedural measures proposed for the facility, and discussing how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria, is sufficient to meet the intent of 10 CFR 52.47(a)(9), "Contents of applications; technical information."

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), acoustic resonances, and acoustical tones on reactor internals should be adequately addressed in accordance with relevant criteria stated in the Appendix to this DSRS Section.

7. Since important to safety components are part of the reactor vessel internals for the mPower™ reactor, a leak tight boundary is required to separate the primary coolant from the secondary coolant. To accomplish this leak tightness, the ASME rules for pressure boundary construction may be used.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this DSRS 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 main steam or feedwater 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

These review procedures are based on the identified DSRS acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1. Programmatic Requirements and Guidance - In accordance with the guidance in NUREG – 0800 "Introduction," Part 2 as applied to this DSRS Section, the staff will review the programs proposed by the applicant to satisfy the following programmatic requirements . If any of the proposed programs satisfies the acceptance criteria described in Subsection II, it can be used to augment or replace some of the review procedures. It should be noted that the wording of "to augment or replace" applies to nonsafety-related risk-significant SSCs, but "to replace" applies to nonsafety-related nonrisk-significant SSCs according to the "graded approach" discussion in NUREG-0800 "Introduction," Part 2. Commission regulations and policy mandate programs applicable to SSCs. Examples of those programs and associated guidance follows:

    - Maintenance Rule SRP Section 17.6 (DSRS Section 13.4, Table 13.4, Item 17, RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." and RG 1.182; "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants".

    - Quality Assurance Program SRP Sections 17.3 and 17.5 (DSRS Section 13.4, Table 13.4, Item 16).

    - Technical Specifications (DSRS Section 16.0 and SRP Section 16.1) – including brackets value for DC and COL. Brackets are used to identify information or characteristics that are plant specific or are based on preliminary design information.

    - Reliability Assurance Program (SRP Section 17.4).

    - Initial Plant Test Program (RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants, "DSRS Section 14.2, and DSRS Section 13.4, Table 13.4, Item 19).

    - ITAAC (DSRS Chapter 14).

2. In accordance with 10 CFR 52.47(a)(8),(21), and (22), and 10 CFR 52.79(a)(17) and (20), for new reactor license applications submitted under Part 52, the applicant is required to (1) address the proposed technical resolution of unresolved safety issues and medium- and high-priority generic safety issues which are identified in the version of NUREG-0933 current on the date up to six months before the docket date of the application and which are technically relevant to the design; (2) demonstrate how the operating experience insights have been incorporated into the plant design; and, (3) provide information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v). These cross-cutting review areas should be addressed by the reviewer for each technical subsection and relevant conclusions documented in the corresponding safety evaluation report (SER) section.

3. The configuration and general arrangement of all mechanical and structural internal elements covered by this DSRS 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 DSRS 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 DSRS 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 COL 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 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 DSRS 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 review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

   For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

   For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.

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. 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.

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 the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this DSRS section.

In addition, to the extent that the review is not discussed in other SER sections, the findings will summarize the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

V. IMPLEMENTATION

The staff will use this DSRS section in performing safety evaluations of mPower^{TM}-specific design certification (DC), or COL, applications submitted by applicants pursuant to 10 CFR Part 52. The staff will use the method described herein to evaluate conformance with Commission regulations.

Because of the numerous design differences between the mPower^{TM} and large light-water nuclear reactor power plants, and in accordance with the direction given by the Commission in SRM- COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010 (ML102510405), to develop risk-informed licensing review plans for each of the small modular reactor reviews including the associated pre-application activities, the staff has developed the content of this DSRS section as an alternative method for mPower^{TM} -specific DC, or COL submitted pursuant to 10 CFR Part 52 to comply with 10 CFR 52.47(a)(9), "Contents of applications; technical information."

This regulation states, in part, that the application must contain "an evaluation of the standard plant design against the SRP revision in effect 6 months before the docket date of the application." The content of this DSRS section has been accepted as an alternative method for complying with 10 CFR 52.47(a)(9) as long as the mPower^{TM} DCD FSAR does not deviate significantly from the design assumptions made by the NRC staff while preparing this DSRS section. The application must identify and describe all differences between the standard plant design and this DSRS section, and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria. If the design assumptions in the DC application deviate significantly from the DSRS, the staff will use the SRP as specified in 10 CFR 52.47(a)(9). Alternatively, the staff may supplement the DSRS section by adding appropriate criteria in order to address new design assumptions. The same approach may be used to meet the requirements of 10 CFR 52.79(a)(41) for COL applications.

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. RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," Revision 3, March 2007.

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

10. RG 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants.”

11. RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.”

12. RG 1.182, “Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.”

13. RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition).”

14. RG 1.215, “Guidance for ITAAC Closure Under 10 CFR Part 52.”

VII. <u>APPENDIX</u>

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

APPENDIX A
Design-Specific Review Standard Section 3.9.5

U.S. NUCLEAR REGULATORY COMMISSION
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 internal components perform no safety function, but must retain their structural integrities to avoid the generation of loose parts that might adversely impact the capabilities of other plant equipment to perform safety functions. Adverse flow may also affect the safety function of components like safety relief valves on the boiling-water reactor (BWR) main steam lines. The staff reviews the consideration of potential adverse flow effects at pressurized-water reactor and BWR 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. The U.S. Nuclear Regulatory commission (NRC) acceptance criteria are based on (1) requirements of General Design Criterion (GDC) 1 in Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 that structure, system, and components (SSCs) 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 safety related components can be damaged by pressure fluctuations and vibration from flow-excited acoustic resonances within the main steam system or reactor vessel. The flow-excited acoustic resonance phenomenon also can occur in mPower™ 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 mPower™ 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 reactor internal components failures while the other does not. Small changes in operating condition can magnify a small adverse flow effect substantially, leading to structural failures. Regulatory Guide (RG) 1.20 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. Additional details of the evaluation procedure and acceptance criteria are outlined in Standard Review Plan (SRP) Section 3.9.2 and RG 1.20.

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 new design, power uprate license amendment or a major plant modification 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 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.

2. Steam Generator Stress and Design Margin. An applicant for the construction and operation of a nuclear power plant and a licensee proposing a power uprate, planning a major plant modification, or a new design for a 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, main steam piping, and safety valves. Past operating experience and analysis may support the determination of adequate design margin for the stress on steam generator internal components.

3. Evaluation of Other Plant Components. An applicant for the construction and operation of a mPower™ nuclear power plant and a licensee of an operating mPower™ nuclear power plant proposing a power uprate license amendment or 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, reactor internal components, 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.

4. Power Ascension Data. An applicant for the construction and operation of a mPower™ nuclear power plant and a licensee of an operating mPower™ 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.

5. Monitoring of Potential Adverse Flow Effects. Plant operating experience, such as from the Quad Cities and Dresden nuclear power plants, has shown that adverse flow effects might not appear for an extended period of time following initial startup or power

ascension. Therefore, an applicant for the construction and operation of mPower™ nuclear power plant and a licensee of an operating mPower™ nuclear power plant proposing a power uprate or a major modification is expected to maintain its 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. 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 as low as is reasonably achievable consideration. The extent and duration of this program following startup and power ascension would be determined by the licensee based on the review of operating experience at its plant and other nuclear power plants. 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 for power ascension up to operation at full licensed power conditions by an applicant for the construction and operation of a mPower™ nuclear power plant and a licensee of an operating mPower™ nuclear power plant proposing a power uprate license amendment or a major plant modification. From the review procedure and the acceptance criteria outlined in Design-Specific Review Standard Section 3.9.5 and SRP Section 3.9.2 and RG 1.20, 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.