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 in nuclear power plants. The U.S. Nuclear Regulatory Commission (NRC) regulations in Title 10 of the *Code of Federal Regulations* (10 CFR~~-~~) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, General Design Criteria (GDCs) for Nuclear Plants," GDC 1, "Quality Standards and Records," GDC 2, "Design Bases for Protection against Natural Phenomena," GDC 4, ~~and 10;~~"Environmental and Dynamic Effects Design Bases," and GDC 10, "Reactor Design"; 10 CFR 50.55a~~;~~, "General Provisions"; and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants"; require that operation, natural phenomena such as earthquakes, postulated accidents including loss-of-coolant accidents (LOCAs), and events and conditions outside the nuclear power unit.

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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 Some plant components, such as 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. For the purpose of this ~~standard review plan~~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 of BWRs, pressurizer water reactors (PWRs) and small modular reactors (SMRs) with the exception of the following:

1 ~~1.~~ Reactor fuel elements and the reactivity control elements out to the coupling interfaces with the drive units (the fuel system design is addressed in SRP Section 4.2).

2 ~~2.~~ Control rod drive elements (the drive elements inside the guide tubes, as well as additional portions of the control rod drive system that may be internal to the reactor vessel assembly of SMRs, are addressed in SRP Section 3.9.4, but the guide tubes are reviewed with the reactor internals in this SRP section).

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

Section 3.9.5, “Reactor Pressure Vessel Internals,” of the Design-Specific Review Standard (DSRS) for the Babcock and Wilcox (B&W) mPower iPWR Design (issued for interim use and comment and available at Agencywide Documents Access and Management System (ADAMS) Accession No. ML12272A077) provides additional guidance for the review of the mPower iPWR Design reactor internals.

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 addresses the analytical methodologies, assumptions, the code, and code edition for the evaluation of the plant components. The review also includes a comparison of stresses against code allowable limits. The specific areas of review are as follows:

1 ~~1.~~ The physical or design arrangements of all reactor internals structures, components, assemblies, and systems, are evaluated 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 ~~2.~~ The basis for the reactor internals’ design loading conditions for normal operation, anticipated operational occurrences, potential adverse flow effects, postulated accidents, and seismic events are evaluated. All combinations of design and service loadings (e.g., operating differential pressure and thermal effects, adverse flow effects, seismic loads, and transient pressure loads of postulated LOCAs) addressed in the 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. The determination of loading conditions caused by flow-induced vibration (FIV) and acoustic resonance (AR), acoustic-induced vibration (AIV), and mechanically-induced vibration (MIV) is reviewed as part of the application review using the guidance in SRP Section 3.9.2. The adequacy of the listed load combinations used in the design is reviewed as part of the application review using the guidance in SRP Section 3.9.3.

3 ~~3.~~ If computational methods (e.g., finite element methods) are used to determine stresses in the reactor internal components and structures, validation of the modeling procedures for the analyses should be presented along with bias errors and uncertainties. The validation may include comparisons of simulated natural frequencies, mode shapes, and frequency response functions with experimental results. The NRC staff reviewer will use the provisions in Regulatory Guide (RG) 1.20 in the evaluation of the applicant's development and validation of computational models for structural dynamic analysis, and the review of structural models. The staff reviewer will also use the provisions in RG 1.20 in evaluating the applicant's determination of appropriate bias errors and random uncertainties associated with computational models. As part of this review, the staff reviewer will evaluate the applicant's demonstration of stress convergence near critical regions, such as welds and other joints.

4 ~~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 FIV, AR, AIV, and MIV) are addressed in SRP Section 3.9.2.

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

6 ~~6.~~ 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 structures, systems, and components (SSCs) related to this SRP 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 the application has been reviewed against the other acceptance criteria contained in this SRP 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.

7 ~~7.~~ COL Action Items and Certification Requirements and Restrictions. For a DC application, the NRC staff reviewer will verify that the DC documents include appropriate

COL action items, and requirements and restrictions (e.g., interface requirements and site parameters).

For a COL application referencing a DC, the NRC staff reviewer will determine whether the COL applicant has adequately addressed the COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, the staff reviewer will also verify that the COL applicant adequately addressed the requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC). Refer to RG 1.206 for additional information that should be included in a COL application.

8 ~~8.~~ Tier 2* Information. Based on the NRC review of the DC application for the ESBWR nuclear power plant design, the DC applicant for a BWR nuclear power plant design might provide the preliminary design of the steam dryer as part of the DC application with the design details to be finalized as part of the as-built analysis and frequency response testing when demonstrating compliance with the applicable ITAAC. Therefore, when reviewing a DC application for a BWR nuclear power plant design, the NRC staff reviewer should verify that the applicant's FSAR or DCD specifies as Tier 2* information critical aspects of the methodology for completing the final design of the steam dryer, such that it may not be modified without NRC staff review and approval.

Review Interfaces

Other SRP sections interface with this section as follows:

1. ~~1.~~ Evaluation of rupture locations, rupture loads, and dynamic effects of postulated rupture of piping is performed under SRP Section 3.6.2.

2. ~~2.~~ Evaluation of the adequacy of analyses justifying exclusion of certain postulated pipe ruptures from design bases is performed under SRP Section 3.6.3.

3. ~~3.~~ 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 SRP Section 3.9.1.

4. ~~4.~~ 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 RPV internals against various vibration excitation mechanisms (such as FIV, AR, AIV, and MIV) is performed under SRP Section 3.9.2.

~~5.~~
5. Evaluation of the adequacy of the structural integrity design of the RPV internals, including adequacy of the design fatigue curves for the reactor internals materials to account for cumulative service-related and environmental usage factor effects, and consideration of each combination of design, service, and postulated event loadings, is performed under SRP Section 3.9.3.

6. ~~6.~~ Evaluation of the adequacy of the mechanical, hydraulic, and electrical components of the control rod drive system, including the control rod drive elements, is performed under SRP Section 3.9.4.

7. ~~7.~~ Evaluation of the functional design and qualification of pumps, valves, and dynamic restraints, including the capability to withstand various vibration excitation mechanisms, is performed under SRP Section 3.9.6.

8. ~~8.~~ Review of the adequacy of programs for assurance of integrity of bolting and threaded fasteners is performed under SRP Section 3.13.

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

10. Review of material aspects of reactor internals is performed under SRP Section 4.5.2.

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

2. ~~2.~~ GDC 2 requires that reactor internals be designed to withstand the effects of natural phenomena, such as earthquakes, combined with the effects of normal or accident conditions, without loss of capability to perform safety functions.

3. ~~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. ~~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. ~~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 plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act, and the NRC's regulations;

6. ~~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 COL, the provisions of the Atomic Energy Act, and the NRC'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 the review described in 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. ~~1.~~ Provisions for loads, loading combinations, and limits applicable to those portions of reactor internals constructed to Subsection NG, "Core Support Structures," of the ASME BPV Code as incorporated by reference in 10 CFR 50.55a are discussed in SRP Section 3.9.3.

2. ~~2.~~ The design and construction of the core support structures should comply with the provisions of Subsection NG of the ASME BPV Code as incorporated by reference in 10 CFR 50.55a. In the Federal Register notice providing the final rule to amend the regulations to incorporate by reference the ASME BPV Codes and new and revised ASME BPV Code cases dated June 21, 2011 (76 FRN 36323), the NRC clarified that Subsection NG is incorporated by reference in 10 CFR 50.55a, but is not mandated. Therefore, Subsection NG is not an NRC requirement. Rather, the NRC considers Subsection NG to be approved for use by applicants and licensees of nuclear power plants by virtue of the NRC's overall approval of Section III, Division 1 rules without condition. In this manner, approval of Subsection NG is similar to regulatory guidance provided in RGs in that it provides an acceptable method for meeting NRC requirements and, in this particular case, 10 CFR Part 50, Appendix A, GDC 1. An applicant may

propose means other than those specified by the provisions in Subsection NG for meeting the applicable regulations.

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. ~~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 induced by these displacements should not exceed the specified limits. The provisions for dynamic analysis of these components are addressed in SRP Section 3.9.2.

5. ~~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. SRP Section 3.6.2 provides additional guidance on transient and dynamic loading effects from postulated pipe ruptures.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this SRP section is discussed in the following paragraphs:

1. ~~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. ~~2.~~ GDC 2, in part, requires that SSCs important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. Reactor internals perform safety functions, or might (through their failure) adversely impact the performance of safety functions (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. ~~3.~~ GDC 4, in 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. Reactor internals perform safety functions, or might (through their failure) adversely impact the performance of safety functions (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) that might cause asymmetric blowdown loadings on the reactor internals. SRP Section 3.6.2 provides additional guidance on blowdown loading. GDC 4 allows such dynamic effects of postulated pipe ruptures to be excluded from the design basis when analyses accepted 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. ~~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. Reactor internals perform safety functions, or might (through their failure) adversely impact the performance of safety functions, such as 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 an acceptable 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 provides confidence 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 NRC staff reviewer will select material from the procedures described below, as may be appropriate for a particular case.

These review procedures are based on the identified SRP acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's justification that the proposed alternatives provide an acceptable method for 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. The applicant should demonstrate that any significant changes in design or operating conditions do not affect the comprehensive vibration analysis described in RG 1.20 and SRP Section 3.9.2.

2. As to the design and analysis of reactor internals, the applicant should demonstrate that they are designed and analyzed in accordance with Subsection NG of the ASME BPV Code as incorporated by reference in 10 CFR 50.55a, and as discussed in SRP Section 3.9.3. In lieu of such a demonstration, 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 reviewing the applicant's description of the procedures and criteria for the design of these components, including the design and service stress limits for all of the applicable loading conditions. The stresses at adverse flow and operating conditions should be treated as primary stresses while satisfying the ASME BPV Code service stress limits.

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

4. The deformation limits specified for these components are reviewed as part of a DC, COL, or operating license application to verify whether the applicant has demonstrated that these deflections will not interfere with the function of related components (e.g., control rods and standby cooling systems) and whether the stresses induced by these displacements are less than the specified limits for the core support structures.

5. For the review of a DC application, the NRC staff reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), specified in the final safety analysis report (FSAR) submitted with the DC application, meet the SRP acceptance criteria. Some DC applicants have referred to their FSAR as the design control document (DCD). The reviewer should also determine whether the identified COL action items are adequate and sufficient. Where the reviewer identifies additional COL action items, these COL action items should be included in the DC FSAR or DCD.

   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 (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report). The NRC staff reviewer should determine whether the COL applicant has adequately responded to each COL action item, and supplemented the DC application or other referenced NRC approvals to address any plant-specific design aspects.

For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC.

For review of a construction permit (CP) application under 10 CFR Part 50, the reviewer should verify that the design described in the preliminary SAR will meet the acceptance criteria.

For the review of an operating license (OL) application under 10 CFR Part 50, the reviewer should follow the above procedures to verify that the design described in the FSAR meets the acceptance criteria.

For review of submittals for replacement components or extended power uprate (EPU) for operating reactors, the reviewer should follow the above procedures to verify that the design meets the acceptance criteria.

6. Tier 2* Information

Based on the NRC review of the DC application for the ESBWR nuclear power plant design, the DC applicant for a BWR nuclear power plant design might provide the preliminary design of the steam dryer as part of the DC application with the design details to be finalized as part of the as-built analysis and frequency response testing when demonstrating compliance with the applicable ITAAC. Therefore, for a DC application for a BWR nuclear power plant design, the NRC staff reviewer should verify that the applicant's FSAR or DCD specifies as Tier 2* information critical aspects of the detailed methodology for completing the final design of the steam dryer, such that it may not be modified without NRC staff review and approval.

IV. EVALUATION FINDINGS

The NRC staff reviewer verifies that the applicant has provided sufficient information and that the review and calculations (as applicable) support the following conclusions to be included in the staff's safety evaluation report (SER) in accordance with the bases for those conclusions described in the SER.

The NRC staff concludes that the design of reactor internals is acceptable and meets the requirements of 10 CFR Part 50, Appendix A, GDCs 1, 2, 4, and 10; 10 CFR 50.55a; and 10 CFR Part 52 [as applicable]. This conclusion is based on the following findings:

1. The applicant has met the requirements of GDC 1, 10 CFR 50.55a, and 10 CFR Part 52 [as applicable] 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 BPV Code, Section III, as incorporated by reference in 10 CFR 50.55a.

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 SRP 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 NRC 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 the review of applications submitted six months or more after the date of issuance of this SRP section, unless superseded by a later revision.

VI. REFERENCES

1. ~~10 CFR 50.55a,~~ American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components," New York, NY.

~~1.~~2. *U.S. Code of Federal Regulations*, "Codes and Standards." §50.55a, Title 10, "Energy."

~~2.~~3. ~~10 CFR~~ *U.S. Code of Federal Regulations*, Domestic Licensing of Production and Utilization Facilities," Part 50, Title 10, "Energy," Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants," GDC 1, "Quality Standards and Records."

~~3.~~4. ~~10 CFR~~ *U.S. Code of Federal Regulations*, Domestic Licensing of Production and Utilization Facilities," Part 50, Title 10, "Energy," Appendix A, "General Design Criteria

(GDC) for Nuclear Power Plants," GDC 2, "Design Bases for Protection ~~Against~~against Natural Phenomena."

~~4.~~5. ~~10 CFR~~ *U.S. Code of Federal Regulations*, Domestic Licensing of Production and Utilization Facilities," Part 50, Title 10, "Energy." Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants," GDC 4, "Environmental and Dynamic Effects Design Bases."

~~5.~~6. ~~10 CFR~~ *U.S. Code of Federal Regulations*, Domestic Licensing of Production and Utilization Facilities," Part 50, Title 10, "Energy." Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants," GDC 10, "Reactor Design."

~~6.~~7. ~~10 CFR Part 52, "Early Site Permit; Standard Design Certification; and Combined~~ *U.S. Code of Federal Regulations*, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Part 52, Title 10, "Energy."

~~7.~~8. ~~NUREG-0609;~~U.S. Nuclear Regulatory Commission, "Asymmetric Blowdown Loads on PWR Primary Systems: Resolution of Generic Task Action Plan A-2~~;" Hosford, S.B.; Mattu, R.; Meyer, R.O.;~~," NUREG-0609, Division of Safety Technology~~;~~, Office of Nuclear Reactor Regulation, January, 1981.

~~8.~~9. U.S. Nuclear Regulatory ~~Guide 1.20~~Commission, "Comprehensive Vibration Assessment Program for Reactor Internals ~~During~~during Preoperational and Initial Startup Testing~~."~~," Regulatory Guide 1.20.

10. ~~ASME Boiler and~~U.S. Nuclear Regulatory Commission, "Combined License Applications for Nuclear Power Plants," Regulatory Guide 1.206.

~~9.~~11. U.S. Nuclear Regulatory Commission, "Reactor Pressure Vessel ~~Code, Section III, Division 1, "Nuclear Power Plant Components," ASME.~~Internals,"

~~10.~~ Design-Specific Review Standard for B&W mPower iPWR Design Section 3.9.5, ~~"Reactor Pressure Vessel Internals"~~ADAMS Accession No. ML12272A077.

~~Regulatory Guide 1.206, "Combined License Applications For Nuclear Power Plants."~~

~~11.~~

PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the Standard Review Plan are covered by the requirements of 10 CFR Part 50 and 10 CFR Part 52, and were approved by the Office of Management and Budget, approval number 3150-0011 and 3150-0151.

PUBLIC PROTECTION NOTIFICATION

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

Standard Review Plan Section 3.9.5
Description of Changes

Section 3.9.5, "REACTOR PRESSURE VESSEL INTERNALS"

In addition to the changes itemized below, editorial changes were made throughout for clarity, consistency, and applicability. Changes incorporated into Revision 4 include:

I. AREAS OF REVIEW

- Changes were made to clarify reactor pressure vessel internals definition, including references to SMRs and Section 3.9.5 of the DSRS for B&W mPower IPWR Design.

- Appendix A: NRC Review of Potential Adverse Flow Effects was deleted as a result of consolidation of the review of potential adverse flow effects in SRP Section 3.9.2, and a pointer to Regulatory Guide 1.20 and SRP Section 3.9.2 for the determination of loading conditions caused by flow-induced vibration, acoustic resonance, acoustic-induced vibration, and mechanically-induced vibration was added as a result.

- The review interface with SRP Section 3.9.6 was added.

II. ACCEPTANCE CRITERIA

- Added the words, "fabricated, erected, tested, and inspected" to the requirements for GDC 1 and 10 CFR 50.55a.

- Updated design and construction compliance requirements for CSS with references to a Federal Register notice and ASME BPV Code cases.

III. REVIEW PROCEDURES

- Removed the review procedure for potential adverse flow effects in concert with the change in areas of review as mentioned above.

IV. IMPLEMENTATION

- No significant changes.

V. REFERENCES

- References were updated by adding the DSRS for Section 3.9.5 and the deletion of Appendix A: NRC Review of Potential Adverse Flow Effects in Nuclear Power Plants Systems in concert with the above changes.