U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER^{TM} iPWR DESIGN

3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for mechanical engineering reviews

Secondary - None

I. AREAS OF REVIEW

The mPower^{TM} integral pressurized water reactor (iPWR) designed by Babcock & Wilcox (B&W) incorporates the reactor core, control rod drive mechanisms, reactor coolant pumps, a single once through steam generator, and the pressurizer inside the reactor vessel. This limits the size of piping connected to the reactor vessel and the number of mechanical components outside the reactor vessel. Connections to the mPower^{TM} reactor vessel include the steam system, the feedwater system, the reactor inventory control and purification system, safety relief valves, the passive emergency core cooling system, and the component cooling water system via the reactor support loop.

This Design Specific Review Standard (DSRS) section addresses information in the safety analysis report (SAR) on methods of analysis for seismic Category I components and supports, including both those designated as American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV code), Section III (hereafter "the Code") Class 1, 2, 3, or core support and those not covered by the Code. Certain aspects of dynamic system analysis methods are addressed in Standard Review Plan (SRP) Section 3.9.2 as well as this DSRS section. Also reviewed is information on design transients for ASME Code Class 1 components, component supports, reactor core support structures and reactor vessel internal components.

The specific areas of review are as follow:

1. Transients used in the design and fatigue analyses of all ASME Code Class 1 components, component supports, reactor core support structures and reactor vessel internal components.

2. Computer programs to be used in analyses of seismic Category I, ASME B&PV Code and non-Code components listed in this DSRS section.

3. Experimental stress analyses to be used in lieu of theoretical stress analyses.

4. The elastic-plastic stress analysis methods to be performed in the design of any components.

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5. The environmental conditions to which all safety-related components will be exposed over the life of the plant.

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 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.

7. COL Action Items and Certification Requirements and Restrictions. For a 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. The review of the acceptability of the listed transients and the number of cycles and events expected over the service lifetime of the plant is performed under DSRS Section 15.0.

2. The review of programs for ensuring bolting and threaded fastener adequacy and integrity are reviewed is performed under DSRS Section 3.13.

3. The review of seismic cyclic ground input loading is performed under DSRS Sections 3.7.2 and 3.7.3 where the methods for determining the seismic cyclic loading to be used for fatigue analysis of appropriate components are provided.

4. The review of the consideration given to minimize degradation of materials due to corrosion based upon the environmental conditions to which equipment will be exposed is performed under DSRS Section 6.1.1.

5. The design of ASME Code Class 1, 2, and 3 components, component supports, and core support structures are reviewed under SRP Section 3.9.3. The design of reactor vessel internal components are reviewed under DSRS Sections 3.9.4 and 3.9.5.

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

II. ACCEPTANCE CRITERIA

Requirements

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

1. 10 CFR 50, Appendix A, General Design Criterion (GDC) 1, as to the requirement that SSCs be designed, fabricated, erected, and, tested to quality standards commensurate with the importance of the safety functions to be performed.

2. GDC 2, as to the requirement that SSCs be designed to withstand seismic events without loss of capability to perform their safety functions.

3. GDC 14, as to the requirement that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

4. GDC 15, as to the requirement that the reactor coolant system and associated auxiliary, control and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

5. 10 CFR Part 50, Appendix B, Section III, as it relates to quality of design control using the quality assurance criteria provided.

6. 10 CFR Part 50, Appendix S, as it relates to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics.

7. 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;

8. 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.

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. To meet the requirements of GDCs 1, 2, 14, 15, and 10 CFR Part 50, Appendix S, the applicant should provide a complete list of transients to be used in the design and fatigue analysis of all Code Class 1 and core support components, supports, and reactor internals within the reactor coolant pressure boundary. The number of events for each transient and the number of load and stress cycles per event and for events in combination should be included. All transients, such as startup and shutdown operations, power level changes, emergency and recovery conditions (including, for new applications, natural convection cooldown), switching operations (i.e., startup or shutdown of one or more coolant loops), control system or other system malfunctions, component malfunctions, transients from single operator errors, inservice hydrostatic tests, seismic events as determined from the criteria specified in Appendix S to 10 CFR Part 50, and design-basis events contained in the Code-required "Design Specifications" for the components of the reactor coolant pressure boundary, should be specified.

   The section of the applicant's SAR on transients will be acceptable if the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure conditions caused by those transients. To a large extent the selection of these specific transient conditions is based upon engineering judgment and experience. Some guidance on the selection of these transients and combinations can be found in SRP Section 3.9.3. Transients and consequent loads and load combinations with appropriate specified design and service limits should provide a complete basis for design of the reactor coolant pressure boundary for all conditions and events expected over the service lifetime of the plant.

   The staff should consider the number of transients appropriate for the design life of the plant. Also, environmental conditions to which equipment important to safety will be exposed (e.g., chemistry of the coolant water) should be considered to minimize the degradation of materials due to corrosion.

2. To meet the requirements of 10 CFR Part 50, Appendix B, and GDC 1, a list of computer programs used in dynamic and static analyses to determine the structural and functional integrity of seismic Category I components, ASME B&PV Code and non-Code components should be provided. For each program, as a minimum, the following information should be provided to demonstrate its applicability and validity:

   A. The author, program source, dated version, and facility.

   B. A description and the extent and limitation of its application.

   C. The computer program solutions to a series of test problems demonstrated to be substantially compatible to solutions obtained from any one of sources (i) through (iv) within the acceptable margin using benchmark problems acceptable to the staff (e.g., NUREG/CR-1677, “Piping Benchmark Problems.” Vols. I and II):

      (i) Hand calculations

(ii) Analytical results published in relevant engineering literature

(iii) Acceptable experimental tests

(iv) A similar computer program previously accepted by NRC or acceptable to the staff (i.e., commercial computer program)

A summary comparison of the solution obtained from sources (i) through (iv) should be provided in either graphical or numerical form. For source (v), the complete computer printout of the input and the solution should be submitted for every benchmark problem. These solutions may be referenced, and need not be resubmitted, in subsequent license applications, provided the information submitted under Items A and B remains unchanged.

3. To meet the requirements of GDCs 1, 14, and 15, if experimental stress analysis methods are used in lieu of analytical methods for any seismic Category I components, ASME B&PV Code or non-Code components and supports, the section of the SAR addressing the experimental stress analysis methods is acceptable if the information meets the provisions of Appendix II to ASME B&PV Code, Section III, Division 1 and, as in the case of analytical methods, if the information is sufficiently detailed to show the design meeting the provisions of the Code-required "Design Specifications."

4. To meet the requirements of GDCs 1, 14, and 15 when Service Level D limits are specified by the applicant for ASME B&PV Code Class 1 components, component supports, core support structure components, and reactor vessel internal components, and other non-Code items, the methods of analysis to calculate the stresses and deformations should conform to the methods outlined in Appendix F to ASME Code, Section III, Division 1, subject to the conditions addressed in Subsection III.4 of this DSRS section.

<u>Technical Rationale</u>

The technical rationale for application of these criteria to reviews performed under this DSRS section is discussed in the following paragraphs:

1. GDC 1 requires in part, that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. 10 CFR Part 50, Appendix B, sets forth, in part, provisions to assure that appropriate standards are specified and included in design documents (design methods and computer programs for the design and analysis of seismic Category I Code Class 1, 2, 3, and core support structures and non-Code structures) and that deviation from such standards are controlled.

   Special topics for mechanical components encompass items related to design transients like component supports, core supports, and reactor internals designated as Class 1, 2, and 3 under ASME Code, Section III, and those not covered by the Code. The applicability and validity of these criteria are demonstrated by requirements that the design methods and computer programs in design and analysis be within current state-of-the-art limits and design control measures acceptable to the staff.

The requirements of GDC 1 provide reasonable assurance that the regulatory requirements for design methodology and quality assurance are satisfied so that SSCs important to safety are capable of performing their intended functions.

2. GDC 2 requires, in part, that SSCs important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The related requirements in Appendix S to 10 CFR Part 50 specify that applicants include seismic events in the design basis and in their postulated design transients.

   Pursuant to GDC 2, the reviewer evaluates whether mechanical components are designed to withstand the loads generated by natural phenomena. The reviewer also verifies whether the applicant has provided a list of postulated design transients with consideration of seismic events.

   The requirements of GDC 2 provide reasonable assurance that SSCs important to safety have the capability to withstand the effects of natural phenomena and to perform their intended functions.

3. GDC 14 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and of gross rupture.

   GDC 15 requires that the reactor coolant system and its auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

   GDC 14 and GDC 15 apply to this review because SSCs important to safety are exposed to postulated transients anticipated during the design life of the plant. If SSCs are to perform their design functions there must be adequate assurance that mechanical components will remain functional under all postulated combinations of normal operating conditions, anticipated operational occurrences, postulated pipe breaks, and seismic events.

   Compliance with the requirements of GDC 14 and GDC 15 provides reasonable assurance that the design transients and consequent loads and load combinations with the appropriate specific design and service limits for ASME Code Class 1 and core support components, supports, and reactor internals form a complete basis for the design of the reactor coolant pressure boundary for all anticipated conditions and extremely low-probability events expected during the service life of the plant.

III. REVIEW PROCEDURES

The 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 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 (SRP Section 13.4, Table 13.4, Item 17, Regulatory Guides (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 SRP Section 13.4, Table 13.4, Item 19).

    - ITAAC (SRP/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 6 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 list of transients, the number of events estimated for each transient presented in the applicant's SAR, and the method for determining this number are compared to the same information on similar and previously licensed applications and to the acceptance criteria outlined in subsection II of this DSRS section. Any deviations from previous accepted practice are noted and the applicant should justify them. For Code Class 1 and core

support components and supports, the reviewer verifies whether for each transient loading condition or combination an acceptable Code service limit is specified (i.e., Design, Level A, Level B, Level C, or Level D as specified in ASME Code, Section III, Division 1).

4. The information on computer programs presented in the applicant's SAR is reviewed as follows:

   A. The list of programs is evaluated to determine whether the applicant adequately describes each program with respect to the type of analysis performed and the specific components to which the program is applied.

   B. The submitted computer test problem solutions recommended in Subsection II.2.C of this DSRS section are reviewed and compared to the test solutions. Satisfactory agreement of computer and test solutions, usually within a $\pm$5 percent error band, verifies the quality and adequacy of the computer programs for the functions for which they were designed.

5. If the applicant elects to use experimental stress analysis techniques in lieu of theoretical stress analyses, sufficient information should be presented in the SAR to demonstrate that the requirements of Appendix II to ASME Code, Section III, Division 1, applicable to the conditions in the "Design Specifications" have been met.

6. If the applicant employs an elastic or an elastic-plastic method of analysis to evaluate the design of safety-related Code or non-Code items for which Service Level D limits have been specified (NB-3225 and Appendix F to ASME Code, Section III, Division 1) the review considers the following points:

   A. The applicant should demonstrate that the stress-strain relationship for component materials to be used in the analysis is valid. The ultimate strength values at service temperature should be justified.

   B. The analytical procedures to be used in the analysis are reviewed to determine the validity of the analysis. Any computer program used should meet the applicable requirements of Subsection II.2.C of this DSRS section.

   C. If elastic system analysis is used, its application may require detailed review and justification if applied to the analysis of systems which contain active components with close tolerances or systems in which the sequence of load application could significantly affect the actual stress distribution.

   D. If elastic, elastic-plastic or limit analysis methods are used for components with elastic or elastic-plastic system analyses, the bases for these procedures are reviewed. The applicant should provide assurance that the calculated item or item support deformations and displacements do not violate the corresponding limits and assumptions on which the methods for the system analysis are based.

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.

1. The applicant has met the relevant requirements of 10 CFR Part 50, Appendix B, and GDC 1 by submitting information that demonstrates the applicability and validity of the design methods and computer programs for the design and analysis of seismic Category I Code Class 1, 2, 3, and core support structures, and non-Code structures within the present state-of-the-art limits and by acceptable design control measures to ensure the quality of the computer programs.

2. The applicant has met the relevant requirements of GDC 2 and 10 CFR Part 50, Appendix S, by including in design transients seismic events as part of the design basis for withstanding the effects of natural phenomena.

3. The applicant has met the relevant requirements of GDC 14 and 15 by demonstrating that the design transients and consequent loads and load combinations with appropriately specified design and service limits for Code Class 1 and core support components, supports, and reactor internals provide a complete basis for design of the reactor coolant pressure boundary for all conditions and events expected over the service lifetime of the plant.

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 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™ 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™ -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™ 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 Part 50, Appendix A, GDC 1, "Quality Standards and Reports."

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

3. 10 CFR Part 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary."

4. 10 CFR Part 50, Appendix A, GDC 15, "Reactor Coolant System Design."

5. 10 CFR Part 50, Appendix B, "Quality Assurance Requirements for Nuclear Power Plants and Fuel Reprocessing Plants."

6. 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants."

7. ASME Boiler and Pressure Vessel Code, Section III, Division I, "Nuclear Power Plant Components," American Society of Mechanical Engineers."

8. SRP Section 3.9.3, "ASME Code Class 1, 2, 3 Components, Component Supports, and Core Support Structures."

9. Report NUREG/CR-1677, "Piping Benchmark Problems." Vols. I and II.