Proposed - For Interim Use and Comment

U.S. NUCLEAR REGULATORY COMMISSION

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

3.9.4 CONTROL ROD DRIVE SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of component performance and testing

Secondary - None

I. AREAS OF REVIEW

The control rod drive system (CRDS) consists of the reactor core reactivity control rods and the related mechanical components which provide the means for their mechanical movement. Title 10 of *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criteria (GDC) 26 and 27 require that the CRDS provide one of the two independent reactivity control systems. The rods and the drive mechanisms shall be capable of reliably controlling reactivity changes under conditions of normal operation, including anticipated operational occurrences, and under postulated accident conditions. A positive means for inserting the rods shall always be maintained to ensure appropriate margin for malfunction, such as stuck rods, and specified acceptable fuel design limits are not exceeded. Since the CRDS is important to safety and the entire CRDS is an integral part of the reactor vessel internals, GDC 1, 2, 4, 14 and 29 and 10 CFR 50.55a require that the system be designed, fabricated, and tested to quality standards commensurate with the safety functions to be performed so as to assure an extremely high probability of accomplishing the safety functions in the event of anticipated operational occurrences, postulated accidents and natural phenomena such as earthquakes.

The specific areas of review are as follows:

1. The descriptive information, including design criteria, testing programs, drawings, and a summary of the method of operation of the control rod drives, is reviewed to permit an evaluation of the adequacy of the system to perform its mechanical function properly.

2. A review is performed of design codes, standards, specifications, and standard practices, as well as GDC, regulatory guides, and branch positions that apply to the design, fabrication, construction, and operation of the CRDS.

   The various criteria, described in general terms above, should be supplied along with the names of the apparatus to which the criteria apply. Various components of the CRDS are reviewed to determine the acceptability of design margins for allowable values of stress, deformation, and fatigue used in the analyses. If an experimental testing program is used in lieu of analysis, the program is reviewed to determine whether it adequately covers stress, deformation, and fatigue.

ADAMS Accession No. ML12272A020

3. A review of applicable design loads and their appropriate combinations, the corresponding design stress limits, and the corresponding allowable deformations is performed. The deformations are of interest in the present context only in those instances where a failure of movement due to excessive deformation could be postulated and such movement would be necessary for a safety-related function.

   If the applicant selects an experimental testing option in lieu of establishing a set of stress and deformation allowables, a detailed description of the testing program must be provided for review. The load combinations, design stress limits and allowable deformations criteria should be provided for review. The final safety evaluation report (FSER) for design certification (DC) applications, the actual design should be compared with the design criteria and limits to demonstrate that the criteria and limits have not been exceeded.

   Loadings imposed during normal plant operation and startup and shutdown transients include but are not limited to pressure, deadweight, temperature effects, and anticipated operational occurrences. Loadings associated with specific seismic and other dynamic events (i.e., flow-induced vibration, acoustic resonance) are then combined with the above plant-type loads. The response to each set of combined loads has a selected stress or deformation limit. The selection of a specific limit is influenced by the probability of the postulated event and the need to assure operation during and after the event. Regulatory Guide (RG) 1.20 provides further details on the determination of loading conditions caused by adverse flow effects.

   The review of the design includes the electrical penetrations for power supply to the control rod drive mechanism (CRDM) DC motors and the mechanical penetrations for the fluid line connections to the hydraulic latching assemblies. In addition, the equipment qualification (i.e., seismic and environmental) of CRDS components is included.

4. A review of the portion of the safety analysis report (SAR) or design control document (DCD) that describes plans for the conduct of an operability assurance program or that references previous test programs or standard industry procedures for similar apparatus is performed. For example, the life cycle test program for the CRDS is reviewed. The operability assurance program is reviewed to ascertain coverage of the following:

   A. Life cycle test program

   B. Proper service environment imposed during test, including appropriate conditions for normal operation, anticipated operational occurrences, seismic or other dynamic events (i.e., flow-induced vibration, acoustic resonance), and postulated accident conditions

   C. Mechanism functional tests

   D. Program results

5. <u>Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)</u>. For 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 Standard Review Plan (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.

6. 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. Verification that CRDS components are acceptably classified and that corresponding appropriate quality standards are applied under DSRS Sections 3.2.2 and/or 5.2.1.1.

2. The adequacy of the fuel system design, including effects of the CRDS on fuel behavior in meeting the requirements of the reactor core design under various normal operating and accident conditions, is reviewed under DSRS Section 4.2.

3. The functional design of reactivity control systems, including the CRDS and its design for protection against the effects of postulated piping and equipment failures, is reviewed under DSRS Section 4.6.

4. The adequacy of programs for assuring the integrity of bolting and threaded fasteners is reviewed under DSRS Section 3.13.

5. The material aspects of CRDS are reviewed under DSRS Section 4.5.1.

6. The adequacy of specified seismic, environment, and service conditions for equipment qualification and of the overall demonstration that components of the CRDS are qualified to perform their functions is reviewed under SRP Section 3.10 and DSRS Section 3.11.

7. The structural integrity of Code Class 1, 2, and 3 components, component supports, and core support structures is reviewed under SRP Section 3.9.3.

8. Evaluation of the adequacy of analysis methods for seismic Category I CRDS components and assemblies dynamic analysis, identification of design transients and of service lifetime transient cyclic loadings to be reflected in the design and fatigue analyses of CRDS components and assemblies is performed under DSRS Section 3.9.1.

9. Evaluation of the functional and design criteria for the CRDS, which may have an active function during and after a faulted plant condition, against the requirements related to component functionality assurance and seismic qualification programs is performed under SRP Section 3.10.

10. 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 for CRDS components and assemblies is performed under SRP Section 3.9.2.

11. Evaluation of the adequacy of dynamic analyses under steady state and operational flow transient conditions for acoustic resonance for CRDS components and assemblies is performed under DSRS Section 3.9.5.

12. Evaluation of the adequacy of the structural integrity design of the CRDS, 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 SRP 3.9.3.

13. The hydraulic supply to the CRDM latching mechanism by the RCIPS is reviewed under DSRS Section 9.3.6.

14. 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, as they relate to the CRDS, require that the CRDS be designed to quality standards commensurate with the importance of the safety functions (e.g., rod insertion and withdrawal, scram operation and timing) to be performed.

2. GDC 2, as it relates to CRDS, requires that the CRDS be designed to withstand the effects of an earthquake without loss of capability to perform its safety functions.

3. GDC 4, as it relates to CRDS, requires that the CRDS components be designed to accommodate the effects of and to be compatible with the environmental conditions during normal plant operation as well as postulated accidents.

4 GDC 14, as it relates to CRDS, requires that the portions of the CRDS be designed, constructed, and tested for the extremely low probability of leakage or gross rupture.

5. GDC 26, as it relates to CRDS, requires that the CRDS be one of the independent reactivity control systems that is designed with appropriate margin to assure its reactivity control function under conditions of normal operation, including anticipated operational occurrences.

6. GDC 27, as it relates to CRDS, requires that the CRDS be designed with appropriate margin, and in conjunction with the emergency core cooling system, be capable of controlling reactivity and cooling the core under postulated accident conditions.

7. GDC 29, as its relates to CRDS, requires that the CRDS, in conjunction with reactor protection systems, be designed to assure an extremely high probability of accomplishing its safety functions in the event of anticipated operational occurrences.

8. 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.

9. 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. The descriptive information is determined to be sufficient provided the minimum requirements for such information meet the design and functional requirements of the CRDS components.

2. Construction (as defined in NCA-1110 of Section III of the ASME Code) should meet the following codes and standards utilized by the nuclear industry which have been reviewed and found acceptable:

    A. For pressurized portions of equipment classified as Quality Group A, B, C (RG 1.26):

        Section III of the ASME Code, Class 1, 2, or 3 as appropriate.

    B. For pressurized portions of equipment classified as Quality Group D (RG 1.26):

        i. Section VIII, Division 1, of the ASME Code for vessels and pump casings.

        ii. For piping systems (American National Standards Institute, ANSI):(1)

(1)This list may be extended if staff reviews and accepts other ANSI and Manufacturers Standardization Society (MSS) standards in the piping system area after this DSRS is issued.

B16.5	Steel Pipe Flanges and Flanged Fittings
B16.9	Steel Butt Welding Fittings
B16.11	Steel Socket Welding Fittings
B16.25	Butt Welding Ends
B16.34	Steel Valves with Flanged and Butt Welding Ends
B31.1	Power Piping
MSS-SP-25	Marking for Valves, Fittings, Flanges, and Unions

C. For nonpressurized equipment:

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

The design and construction of the nonpressurized portions of the CRDS should comply with the requirements of Subsection NG, "Core Support Structures," of the ASME Code and SRP Section 3.9.3.

The design criteria, loading conditions, and analyses that provide the bases for the design of the nonpressurized portions of the CRDS 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.

Deformation limits for the nonpressurized portions of the CRDS 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 3.9.2.

Potential adverse flow effects of flow-induced vibration (FIV) and acoustic resonances on the nonpressurized portions of the CRDS should be adequately addressed in accordance with relevant criteria stated in the Appendix of DSRS 3.9.5.

3. For the various design and service conditions defined in NB-3113 of Section III of the ASME Code, load combination sets are as given in SRP Section 3.9.3.

   The stress limits applicable to pressurized and nonpressurized portions of the control rod drive systems should be as given in SRP Section 3.9.3 for the response to each loading set.

4. The operability assurance program will be acceptable provided the observed performance as to wear, functioning times, latching, and ability to overcome a stuck rod meet system design requirements.

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 establish requirements regarding the quality standards to be applied to the CRDS. Specifically, 10 CFR 50.55a identifies the ASME Code requirements, Code editions, and addenda that must be applied to pressure-retaining portions of the CRDS that are of the highest importance to safety. RG 1.26 identifies acceptable standards to be applied for pressure-retaining portions of the CRDS that are less important to safety but which may contain radioactive material. The CRDS is an independent reactivity control system designed to ensure the capability to control reactivity changes in the reactor under normal operating and accident conditions. The fuel cladding is protected by CRDS safety functions, including insertion of adequate negative reactivity to preserve these fission product barriers under specified conditions. In addition, the CRDS provides a barrier to the release of fission products. The application of GDC 1 and 10 CFR 50.55a requirements to the design, fabrication, installation, and testing ensures the CRDS meets quality standards that are adequate to provide assurance that these safety functions will be performed.

2. GDC 2 establishes requirements regarding the ability of the CRDS to withstand the effects of an earthquake. The CRDS must satisfy Seismic Category I requirements and be capable of controlling reactivity when subjected to a seismic or dynamic disturbance thereby ensuring that the fission process can be rapidly terminated. Consequently, plant protection and safety is augmented by the capability of the CRDS to perform its safety function under earthquake or other dynamic conditions.

3. GDC 4 requires that SSCs be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and be appropriately protected against dynamic effects (e.g., flow-induced vibration and other thermo-hydraulic loading conditions). The CRDS provides the capability to safely shut down the reactor during normal operations and anticipated operational occurrences and either prevents or mitigates the consequences associated with postulated accident scenarios. The design of the CRDS must ensure that the ability to perform these safety-related functions is not compromised by adverse environmental conditions. Compliance with GDC 4 ensures that the CRDS will remain functional under adverse postulated environmental conditions and provide essential reactor shutdown capabilities.

4. GDC 14 establishes requirements regarding the RCPB portion of the CRDS. The CRDM is relied on, in part, to provide a barrier to the release of fission products to the containment through proper design of the control rod drive housing and components that are part of the RCPB. Application of the GDC 14 criteria to the CRDM components functioning as a RCPB enhances safety by ensuring that the RCPB will have an extremely low probability of failure.

5. GDC 26 establishes requirements regarding the reactivity control systems' redundancy and capability. The CRDS is one of the reactivity control systems relied on during normal operating and anticipated operational occurrences to control reactivity changes and ensure that the fuel design limits are not exceeded. Application of GDC 26 criteria

to the CRDS improves safety by providing protection for the fuel rods and cladding, which is the primary barrier to the release of fission products.

6. GDC 27 establishes requirements regarding the combined reactivity control system capability. The CRDS is one of the reactivity control systems relied on to control reactivity changes and ensure that the capability to cool the core is maintained during postulated accident conditions. Requiring compliance with GDC 27 for the CRDS augments the protection provided for the primary fission product barrier by providing one means to ensure that the core will be maintained in a coolable geometry under postulated accident conditions.

7. GDC 29 establishes requirements regarding the capability of the CRDS to accomplish its safety functions in the event of anticipated operational occurrences. In order to provide protection for the fuel rods and cladding, which is the primary barrier to the release of fission products, the CRDS must have a high probability of accomplishing its safety function during anticipated operational occurrences. Application of this requirement augments plant protection and safety by requiring a highly reliable fast-acting control rod drive mechanism capable of operation during anticipated operational occurrences.

8. The specified codes and standards establish requirements for construction of the applicable portions of the CRDS. The individual components of the CRDS must be designed, fabricated, installed, and tested to quality standards commensurate with the importance of the safety function to be performed by that component. The individual codes and standards each provide a set of applicable limits that the design must meet in order to ensure that the applicable component can carry out its designated safety function.

III. <u>REVIEW PROCEDURES</u>

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, 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 objectives of the review are to determine that design, fabrication, and construction of the CRDM provide structural adequacy and that suitable life cycle testing programs have been utilized to prove operability under service conditions.

   In the DCD review, it should be determined that the design criteria utilize proper load combinations, stress and deformation limits, and that operability assurance is provided by reference to a previously accepted testing program, or a commitment is made to perform a testing program that includes the essential elements listed below. In the operating license (OL) review, the results of any testing program not previously reviewed should be evaluated.

4. The design criteria presented should be evaluated for the internal pressure-containing portions and other portions of the CRDS.

   Of particular interest are any new and unique features that have not been used in the past. Pressure-containing components are checked to ensure that they meet the design requirements of the codes and criteria that have been accepted and are identified in DSRS Section 3.2.2. The review of the functional design of reactivity control systems, including CRDS, is performed as part of the review of DSRS Section 4.6. The loading combinations for the various plant operating conditions are reviewed as part of the review of SRP Section 3.9.3; given these loading combinations, the stress limits of the appropriate code should not be exceeded, or the limits in SRP Section 3.9.3 should not be exceeded if limits are not specified in the listed design code.

The choice of structural materials of construction for the CRDS is reviewed in DSRS Section 4.5.1.

5. Loading combinations are defined as those loadings associated with plant operations that are expected to occur one or more times during the lifetime of the plant and include, but are not limited to, loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power, combined with loadings caused by natural or accident events including water hammer. The load combinations that are postulated to occur are specified for each of the design and service conditions as defined in Paragraph NB-3113 and NG-3000 of the ASME Code. These load combinations are defined in SRP Section 3.9.3 and are part of the review.

   The design stress limits, including fatigue limits and deformation limits appropriate to the components of the CRDM, are compared to the limits of specified codes, previously designed and successfully operating systems, or the results of scale model and prototype testing programs.

6. The CRDM should be subjected to a life cycle test program to determine the ability of the drive components to function during and after normal operation, anticipated operational occurrences, seismic events, and postulated accident conditions over the full range of temperatures, pressures, loadings, and misalignment expected in service. The tests should include functional tests to determine insertion and withdrawal times, latching operation, scram operation and time, system valve operation and scram accumulator leakage for hydraulic latching mechanism, ability to overcome a stuck rod condition, misalignment testing, and wear. Rod travel and number of operational trips and test trips expected during the mechanism operational life should be duplicated in the tests.

   The reviewer checks the elements of the test program to be sure all required parameters have been included, and finally reviews the test results to determine acceptability. Excessive wear, malfunction of components, operating times beyond determined limits, leakage, etc.; all would be cause for retesting.

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 FSAR meets the acceptance criteria. DCs have referred to the FSAR as the 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 requirement of GDC 1 and 10 CFR 50.55a, with respect to designing components important to safety to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for the control rod drive system are in conformance with the requirements of appropriate ANSI and ASME Codes. The applicant has adequately evaluated the potential adverse flow effects on CRDS up to full licensed power conditions.

2. The applicant has met the requirements of GDC 2, 4, 14, and 26 with respect to designing the control rod drive system to withstand effects of earthquakes and conditions of normal operation, including maintenance, testing, anticipated operational occurrences, and postulated accidents (including LOCAs) with adequate margins to assure the system's reactivity control function and with extremely low probability of leakage or gross rupture of the reactor coolant pressure boundary. The specified design transients, design and service loadings, combination of loads, and resulting stresses and deformations under such loading combinations are reviewed within SRP Section 3.9.3.

3. The applicant has met the requirements of GDC 27 and 29 with respect to designing the CRDS to assure its capability of controlling reactivity and cooling the reactor core with appropriate margin in conjunction with either the emergency core cooling system or the reactor protection system. The operability assurance program is acceptable with respect to meeting system design requirements in observed performance as to wear, functioning times, latching, and overcoming a stuck rod.

Accordingly, the staff concludes that the design of the CRDS is acceptable and meets the requirements of GDC 1, 2, 4, 26, 27, and 29, and 10 CFR 50.55a.

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™-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 50.55a, “Codes and Standards.”

2. 10 CFR Part 50, Appendix A, General Design Criterion 1, “Quality Standards and Records.”

3. 10 CFR Part 50, Appendix A, General Design Criterion 2, “Design Bases for Protection Against Natural Phenomena.”

4. 10 CFR Part 50, Appendix A, General Design Criterion 4, “ Environmental and Dynamic Effects Design Bases.”

5. 10 CFR Part 50, Appendix A, General Design Criterion 14, “Reactor Coolant Pressure Boundary.”

6. 10 CFR Part 50, Appendix A, General Design Criterion 26, “Reactivity Control System Redundancy and Capability.”

7. 10 CFR Part 50, Appendix A, General Design Criterion 27, “Combined Reactivity Control Systems Capability.”

8. 10 CFR Part 50, Appendix A, General Design Criterion 29, “Protection Against Anticipated Operational Occurrences.”

9. RG 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing."

10. RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants.”

11. RG 1.29, “Seismic Design Classification.”

12. ASME Boiler and Pressure Vessel Code, Section III, “Nuclear Power Plant Components,” and Section VIII, Division 1, “Pressure Vessels,” American Society of Mechanical Engineers.

13. ANSI B 16.5, “Steel Pipe Flanges and Flanged Fittings,” American National Standard Institute.

14. ANSI B 16.9, “Wrought Steel Butt Welding Fittings,” American National Standard Institute.

15. ANSI B 16.11, “Forged Steel Fittings, Socket-Welding and Threaded ,” American National Standard Institute.

16. ANSI B 16.25, “Butt Welding Ends - Pipe, Valves, Flanges, and Fittings,” American National Standard Institute.

17. ANSI B 16.34, ”Steel Valves with Flanged and Butt Welding Ends,” American Society of Mechanical Engineers.

18. ANSI B 31.1, “Power Piping,” American National Standard Institute.

19. MSS-SP-25, “Marking for Valves, Fittings, Flanges, and Unions,” Manufacturers Standardization Society.

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

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

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

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

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