U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

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 control rods and the related mechanical components which provide the means for mechanical movement. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants," GDC 26, "Reactivity Control System Redundancy and Capability," and GDC 27, "Combined Reactivity Control Systems Capability," require that the CRDS provide one of the independent reactivity control systems. The rods and the control rod drive mechanisms (CRDMs) 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. Since the CRDS is important to safety and portions of the CRDS are often a part of the reactor coolant pressure boundary (RCPB), GDC 1, "Quality Standards and Records," GDC 2, "Design Bases for Protection against Natural Phenomena," GDC 14, "Reactor Coolant Pressure Boundary," and

GDC 29, “Protection against Anticipated Operational Occurrences,” and 10 CFR 50.55a, “Codes and Standards,” 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.

Information in the areas noted below is provided in the applicant’s safety analysis report and is reviewed by the primary review branch in accordance with this Standard Review Plan (SRP) section. This information pertains to the CRDS, which is considered to extend to the coupling interface with the reactivity control elements in the reactor pressure vessel. For electromagnetic systems, the review under this SRP section is limited to just the CRDM portion of the CRDS. For hydraulic systems, the review covers the CRDM and also the hydraulic control unit, the condensate supply system, and the scram discharge volume. For both types of systems, the CRDM housing should be treated as part of the RCPB; the relevant mechanical engineering information is presented either in this SRP section or by reference to the sections on the RCPB.

CRDS components internal to the reactor vessel assembly of small modular reactors (SMRs) are not part of the pressure boundary, and therefore the areas of review are somewhat different than those for conventional pressurized water reactors (PWRs). However, any cables or cable bundles connected to motors or actuators of the CRDMs that enter the reactor vessel through penetration(s) must meet appropriate acceptance criteria. CRDS components internal to a vessel assembly, including the CRDMs, are exposed to primary coolant flow, and corresponding temperature- and flow-induced loads. Section 3.9.4, “Control Rod Drive Systems,” 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. ML12272A020) provides additional guidance for review of the mPower iPWR Design CRDS. Also, mechanical and acoustic pulsations from other major reactor components, including valves and reactor recirculation pumps (RRPs), should be considered in the CRDS design. Evaluation of flow-induced loads is performed under SRP Section 3.9.2 and Regulatory Guide (RG) 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing,” Revision 4.

If other types of CRDS are proposed or if new features that are not specifically mentioned here are incorporated into the CRDS of current types, the information supplied for the new systems or new features should be similar to the information described below.

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 close proximity of all major plant components in small modular reactor (SMR) designs and the immersion of the CRDS within some SMR concepts could lead to alternating loading not previously considered. Therefore, a review is also performed of the overall reactor operation, and a matrix of all potential alternating flow, acoustic, and mechanical loads on the CRDS is

developed. Details of such potential loads are provided in SRP Section 3.9.2 and RG 1.20 Revision 4.

The various criteria, described in general terms above, should be supplied along with the names of the apparatus to which the criteria apply. Pressurized portions of the system that are a part of the RCPB are reviewed to determine the extent to which the applicant complies with the Class 1 requirements of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPV Code). Those portions that are not part of the RCPB are reviewed for compliance with other specified parts of Section III, or other sections of the ASME BPV Code. The non-pressurized portions of the CRDS are reviewed to determine the acceptability of design margins for allowable values of stress, deformation, and fatigue used in the analyses. For CRDS internal to the vessel assembly, the review includes the mechanical components of the CRDS, as well as the power cables and their connections to any direct current motors and electromagnetic or hydraulic actuators in the CRDM assemblies. If an experimental testing program is used in lieu of analysis, the program is reviewed to determine whether it adequately addresses stress, deformation, and fatigue. Whether analysis or testing is used to support the CRDS design, it should be based on the limiting plant operating conditions with worst-case static and alternating loads.

3. A review of applicable design loads (static and alternating) and their appropriate combinations, the corresponding design stress and fatigue 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 static or oscillating deformation could be postulated and such movement would be necessary for a safety-related function. If applicable, fatigue assessments, primarily of connections between CRDS components, should be performed to ensure the assembly maintains integrity throughout the design life of the plant.

   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 (which for CRDS internal to the vessel assembly include both static and all postulated alternating flow, acoustic, and mechanical loads, including those from other components throughout the reactor), design stress limits and allowable deformations criteria should be provided for review in the preliminary safety analysis report (PSAR). In the final safety analysis report (FSAR) for an operating license, or 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 (e.g., flow-induced vibration, flow-excited acoustic resonance, and mechanical tones from other components such as RRPs) are then combined with the above plant-type loads. For boiling water reactors (BWRs) only, the CRDS is reviewed to verify that the system is capable of withstanding adverse dynamic loads such as water hammer. 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. Details on the determination of loading conditions caused by adverse flow effects are provided in SRP Section 3.9.2 and RG 1.20, Revision 4.

4. A review of the portion of the SAR 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, including acceptance criteria for testing.

   B. Proper service environment imposed during testing, including appropriate conditions for normal operation, anticipated operational occurrences, seismic or other dynamic events (e.g., flow-induced vibration, flow-excited acoustic resonance, and any mechanical tones), and postulated accident conditions. For embedded CRDS designs, estimated flow rates over all critical components should be evaluated, including flow over the CRDS at variable latch heights.

   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 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 after the rest of this portion of the application has been reviewed against 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.

6. <u>COL Action Items and Certification Requirements and Restrictions</u>. 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. Refer to RG 1.206, “Combined License Applications for Nuclear Power Plants,” for additional information that should be included in a COL application.

Review Interfaces

Other SRP sections interface with this section as follows:

1. Verification that CRDS pressure-retaining components are acceptably classified and that corresponding appropriate quality standards are applied is performed under SRP Sections 3.2.2 and/or 5.2.1.1.

2. Evaluation of BWR CRDS piping with respect to locations and effects of postulated piping failures is performed under SRP Section 3.6.2.

3. 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 SRP Section 4.2.

4. 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 SRP Section 4.6.

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

6. The material aspects of CRDS are reviewed under SRP Section 4.5.1.

7. The adequacy of specified seismic, environmental, 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 Sections 3.10 and 3.11.

8. 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.

9. Evaluation of the adequacy of (1) the dynamic analysis methods used for seismic Category I CRDS components and assemblies, and (2) the design transients and service lifetime transient cyclic loadings used in the design and fatigue analyses of CRDS components and assemblies is performed under SRP Section 3.9.1.

10. Evaluation of the seismic and dynamic qualification of 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.

11. 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, acoustically, and mechanically induced vibration for CRDS components and assemblies internal to a vessel assembly is performed under SRP Section 3.9.2 and RG 1.20, Revision 4.

12. The adequacy of any electrical supply lines to the CRDM latching mechanism is reviewed under SRP Section 8.3.1 or 8.3.2, as applicable.

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

II. ACCEPTANCE CRITERIA

Requirements

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

1. GDC 1 and 10 CFR 50.55a, as they relate to the CRDS, require that the CRDS be designed to quality standards commensurate with the importance of the safety functions to be performed.

2. GDC 2, as it relates to the 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 14, as it relates to the CRDS, requires that the RCPB portion of the CRDS be designed, constructed, and tested for the extremely low probability of leakage or gross rupture.

4. GDC 26, as it relates to the 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.

5. GDC 27, as it relates to the 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.

6. GDC 29, as it relates to the 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.

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 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.

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 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. Construction (as defined in NCA-1110 of Section III of the ASME BPV 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 BPV 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 BPV Code for vessels and pump casings.

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

B16.5	Steel Pipe Flanges and Flanged Fittings
B16.9	Wrought Steel Butt Welding Fittings
B16.11	Forged Steel Fittings, Socket-Welding and Threaded
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 (Non-ASME BPV Code):

      Design margins presented for allowable stress, deformation, and fatigue should be equal to or greater than margins for other plants of similar design with successful operating experience. A justification of any decreases in design margins should be provided.

(1) This list can be extended by a staff review and acceptance of other ANSI and Manufacturers Standardization Society (MSS) standards in the piping system area.

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

   The stress limits applicable to pressurized and nonpressurized portions of the CRDS should be as given in SRP Section 3.9.3 for the response to each loading set. For BWRs, the CRDS design should adequately consider water hammer loads to assure that system safety functions can be achieved.

3. 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.

<u>Technical Rationale</u>

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. 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 of lower safety significance 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 and RCPB are protected by CRDS safety functions, including insertion of adequate negative reactivity to preserve these fission product barriers under specified conditions. In addition, CRDS designs may comprise a portion of the RCPB and 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 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 conditions.

3. GDC 14 establishes requirements regarding the RCPB portion of CRDS designs. 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 part of the RCPB enhances safety by ensuring that the RCPB will have an extremely low probability of failure.

4. GDC 26 establishes requirements regarding the redundancy and capability of the reactivity control systems. 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.

5. 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.

6. 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.

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

   In the DC or construction permit (CP) 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 provisions are specified 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.

2. The design criteria presented should be evaluated for the internal pressure-containing portions and other portions of the CRDS, including the CRDM housing, the hydraulic control unit, the condensate supply system and scram discharge volume, and portions such as the cylinder, tube, piston, and collet assembly.

   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 SRP Section 3.2.2. In the case of CRDS designs that are internal to a vessel assembly, design criteria should be presented which account for alternating loads induced by the flow around and within the CRDS components, including the power cables to the electromagnetic and/or hydraulic actuators. Design criteria accounting for any mechanically induced forces on the CRDS components, such as those from the RRPs, should also be provided. Finally, criteria that address any flow-induced acoustic resonances elsewhere in the reactor, or in downstream steam lines, should be specified.

   The review of the functional design of reactivity control systems, including CRDS, is performed as part of the review of SRP 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 displacement and 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.

   For a BWR CRDS that includes a scram discharge volume system, the reviewer verifies that the system piping design meets or exceeds the acceptable owner's group classifications and criteria discussed in the enclosure to Generic Letter 86-01 to ensure that breaks and through wall cracks in the piping need not be postulated. The reviewer also verifies that acceptable commitments are made regarding associated inspections, periodic visual verification of the scram system piping integrity, and actions in response to detected leakage to adequately address prevention and mitigation of the effects of leakage associated with potential failures of this piping.

   For a BWR CRDS that includes a control rod drive return line, the reviewer verifies acceptable provisions for the return line design and its implementation in accordance with Generic Letter 80-95 and Part II, Section 8 of NUREG-0619.

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

3. 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 loads for BWRs. The load combinations that are postulated to occur are specified for each of the design and service conditions as defined in Paragraph NB-3113 of the ASME BPV 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.

4. The CRDM of a new design or configuration 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. All-important flow, acoustic, and mechanical excitation mechanisms should be present during the testing (see SRP Section 3.9.2 and RG 1.20, Revision 4, for a discussion of potential excitation mechanisms). This may require the presence of RRPs and potentially isolation valves in the life cycle tests. 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 CRDS, ability to overcome a stuck rod condition, 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, scram accumulator leakage, etc., could be cause for retesting.

5. 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 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 included in 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 (ESP) 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.

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 SER. The reviewer also states the bases for those conclusions.

1. The applicant has met the requirements 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 CRDS are in conformance with the requirements of appropriate ANSI and ASME Codes.

2. The applicant has met the requirements of GDC 2, 14, and 26 with respect to designing the CRDS to withstand effects of earthquakes and conditions of normal operation, including anticipated operational occurrences, 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, 14, 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 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 staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

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

VI. REFERENCES

1. American National Standard Institute, ANSI B16.5, "Steel Pipe Flanges and Flanged Fittings," New York, NY.

2. American National Standard Institute, ANSI B16.9, "Wrought Steel Butt Welding Fittings," New York, NY.

3. American National Standard Institute, ANSI B16.11, "Forged Steel Fittings, Socket-Welding and Threaded," New York, NY.

4. American National Standard Institute, ANSI B16.25, "Butt Welding Ends - Pipe, Valves, Flanges, and Fittings," New York, NY.

5. American National Standard Institute, ANSI B16.34, “Steel Valves with Flanged and Butt Welding Ends,” New York, NY.

6. American National Standard Institute ANSI B31.1, “Power Piping,” New York, NY.

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

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

9. U.S. Code of Federal Regulations, “Codes and Standards,” §50.55a, Title 10 “Energy.”

10. U.S. Code of Federal Regulations, “Domestic Licensing of Production and Utilization Facilities,” Part 50, Title 10 “Energy.” Appendix A, GDC 1, “Quality Standards and Records.”

11. U.S. Code of Federal Regulations, “Domestic Licensing of Production and Utilization Facilities,” Part 50, Title 10 “Energy,” Appendix A, GDC 2, “Design Bases for Protection Against Natural Phenomena.”

12. U.S. Code of Federal Regulations, Domestic Licensing of Production and Utilization Facilities,” Part 50, Title 10 “Energy,” Appendix A, GDC 14, “Reactor Coolant Pressure Boundary.”

13. U.S. Code of Federal Regulations, Domestic Licensing of Production and Utilization Facilities,” Part 50, Title 10 “Energy,” Appendix A, GDC 26, “Reactivity Control System Redundancy and Capability.”

14. U.S. Code of Federal Regulations, Domestic Licensing of Production and Utilization Facilities,” Part 50, Title 10 “Energy,” Appendix A, GDC 27, “Combined Reactivity Control Systems Capability.”

15. U.S. Code of Federal Regulations, Domestic Licensing of Production and Utilization Facilities,” Part 50, Title 10 “Energy,” Appendix A, GDC 29, “Protection against Anticipated Operational Occurrences.”

16. U.S. Nuclear Regulatory Commission, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” Regulatory Guide 1.26,

17. U.S. Nuclear Regulatory Commission, “Seismic Design Classification,” Regulatory Guide 1.29, Revision 4, March 2007

18. U.S. Nuclear Regulatory Commission “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking,” NUREG-0619, November 1980.

19. U.S. Nuclear Regulatory Commission, NRC Letter to BWR Applicants and Licensees, “Final Edition of NUREG-0619, BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking,” Generic Letter No. 80-95, November 13, 1980.

20. U.S. Nuclear Regulatory Commission, NRC Letter to BWR Applicants and Licensees, “Safety Concerns Associated with Pipe Breaks in the BWR Scram System,” Generic Letter No. 86-01, January 3, 1986.

21. U.S. Nuclear Regulatory Commission, “Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing,” Regulatory Guide 1.20.

22. U.S. Nuclear Regulatory Commission, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants,” Regulatory Guide 1.70.

23. U.S. Nuclear Regulatory Commission, “Combined License Applications for Nuclear Power Plants,” Regulatory Guide 1.206.

Standard Review Plan Section 3.9.4
Description of Changes
Section 3.9.4, "Control Rod Drive Systems"

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

- Added CRDS description for SMRs and pointer to SRP Section 3.9.2 and RG 1.20 for evaluation of flow-induced loads.

- Added review of fatigue limits assessment for 60 year design life of the plant.

- Updated life cycle test program to include acceptance criteria for testing and proper service environment imposed during testing to include dynamic events, including guidance for embedded CRDS designs.

II. SRP ACCEPTANCE CRITERIA

- Fixed reference to RG 1.70 for Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants.

III. REVIEW PROCEDURES

- Added discussion on flow, acoustic, and mechanical excitation mechanisms and pointer to SRP Section 3.9.2 and RG 1.20 for potential excitation mechanisms.

IV. IMPLEMENTATION

- No significant changes

V. REFERENCES

- References were updated in concert with changes referenced above, including RG 1.20, RG 1.70, and RG 1.206.