U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

10.4.7 CONDENSATE AND FEEDWATER SYSTEM

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of power conversion systems

Secondary - None

I. AREAS OF REVIEW

The condensate and feedwater system (CFS) provides feedwater at the required temperature, pressure, and flow rate to the steam generators (SGs). Condensate is pumped from the main condenser hotwell by the condensate pumps, passes through the low-pressure feedwater heaters to the feedwater pumps, and then is pumped through the high-pressure feedwater heaters to the nuclear steam supply system.

The primary reviewer reviews the CFS from the condenser outlet up to the nuclear steam supply system to ensure conformance to General Design Criteria (GDC) 2, 4, 5, 44, 45, and 46. There are also interfaces with the secondary water makeup system and condensate cleanup system. The CFS is used for normal shutdown. The only part of the CFS classified as safety-related, i.e., required for safe-shutdown or in the event of postulated accidents, is the feedwater piping from the SG inlet nozzles up to and including the outermost containment isolation valve. This portion of the system must be designed to ensure feedwater system isolation in accident situations, such as a feedwater line break, and containment isolation in cases in which the feedwater system could potentially become a containment bypass pathway (e.g., steam generator tube rupture).

The specific areas of review are as follows:

1. Review of the characteristics of the CFS with respect to the capability to supply adequate feedwater to the nuclear steam supply system as required for normal operation, and shutdown.

2. Determination that an acceptable design has been established for:

   A. The interfaces of the CFS with secondary makeup and the condensate cleanup system with regard to functional design requirements and seismic design classification.

   B. The feedwater system with regard to possible fluid flow instabilities (e.g., water hammer) during normal plant operation as well as during upset or accident conditions.

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C. The detection of major system leaks that could affect the functional performance of safety-related equipment.

3. 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 design-specific review standard (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 DSRS Section 14.3 and appropriate subsections.

4. 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 DSRS and SRP sections interface with this section as follows:

1. Review for flood protection is performed under DSRS Section 3.4.1.

2. Review of the protection against internally generated missiles is performed under DSRS Section 3.5.1.1.

3. Review of protection against missiles generated by natural phenomena, including tornados, is performed under DSRS Section 3.5.1.4.

4. Review of the SSCs to be protected against externally generated missiles is performed under DSRS Section 3.5.2.

5. Review of high- and moderate-energy pipe breaks is performed under SRP Section 3.6.1.

6. Review of the fire protection program is performed under SRP Section 9.5.1.

7. Review of the environmental qualification of mechanical and electrical equipment is performed under DSRS Section 3.11.

8. Determination that transients resulting from feedwater flow control malfunctions will not violate the primary system pressure boundary integrity criterion are performed under DSRS Sections 15.1.1 through 15.1.4.

9. Determination that the loss of normal feedwater flow will not violate the fuel damage criterion or the system pressure boundary integrity criterion is performed under DSRS Section 15.2.7.

10. Evaluation of the system power sources with respect to their capability to perform safety-related functions during normal, transient, and accident conditions is performed under DSRS Section 8.3.1.

11. Review of the acceptability of design analyses, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe-shutdown earthquake (SSE), the probable maximum flood (PMF), and tornado missiles are performed under DSRS Sections 3.3.1, 3.3.2, 3.5.3, 3.7.1 through 3.7.4, 3.8.4, and 3.8.5.

12. Determination that the components, piping, and structures are designed in accordance with applicable codes and standards is performed under DSRS Sections 3.9.1 through 3.9.3. The analysis includes a determination of the acceptability of design analyses, procedures, and criteria used to establish the adequacy of devices or restraints as they may relate to significant water hammers in system piping, and a review of test programs of components that may be affected by water hammers.

13. Determination of the acceptability of seismic and quality group classifications for system components is performed under DSRS Section 3.2.1.

14. Review of the adequacy of the inservice testing program of pumps and valves is performed under DSRS Section 3.9.6.

15. Review of the adequacy of the containment isolation system and the acceptability of the containment leakage testing program, is performed under DSRS Sections 6.2.4 and 6.2.6.

16. Verification that preservice inspection requirements are met for system components is performed under DSRS Section 6.6.

17. Evaluation of feedwater system materials, including their selection and fabrication, fracture toughness of Class 2 and 3 components, and erosion/corrosion is performed under DSRS Section 10.3.6.

18. Review of technical specifications is performed under DSRS Chapter 16.0.

19. Review of quality assurance programs is performed under SRP Chapter 17.0.

20. Review of the seismic qualification of Category I instrumentation and electrical equipment is performed under DSRS Section 3.10.

21. Review of the instrumentation and controls associated with the steam generator level control system is performed under DSRS Chapter 7 upon request of the primary reviewer.

22. Review of the probabilistic risk assessment performed under SRP Chapter 19.0 for potential risk significance of CFS elements.

II. ACCEPTANCE CRITERIA

Requirements

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

1. GDC 2, “Design Bases for Protection Against Natural Phenomena,” as related to the system being capable of withstanding the effects of earthquakes.

2. GDC 4, “Environmental and Dynamic Effects Design Bases,” as related to the dynamic effects associated with possible fluid flow instabilities (e.g., water hammers) during normal plant operation, as well as during upset or accident conditions.

   GDC 5, “Sharing of Structures, Systems, and Components,” as related to the capability of shared systems and components important to safety to perform required safety functions.

4. GDC 44, "Cooling Water," as it relates to:

   A. The capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions.

   B. Redundancy of components so that under accident conditions the safety function can be performed assuming a single active component failure. (This may be coincident with the loss of offsite power for certain events.)

   C. The capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained.

5. GDC 45, "Inspection of Cooling Water System," as related to design provisions to permit periodic inservice inspection of system components and equipment.

6. GDC 46, "Testing of Cooling Water System," as related to design provisions to permit appropriate functional testing of the system and components to ensure structural integrity and leak-tightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, shutdown, and accident conditions.

7. Title 10 of the *Code of Federal Regulations* (CFR), Section 20.1406, as it relates to the detection and isolation of radioactive material in the CFS so as to minimize contamination of the associated systems, facility, and the environment; and also eventual decommissioning.

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 plant that incorporates the DC is built and will operate in accordance with the DC, 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 COL, 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 are set forth below. 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.” The same approach may be used to meet the requirements of 10 CFR 52.79(a)(41) for COL applications.

1. Seismic Events. The requirements of GDC 2 are met by demonstrating that SSCs important to safety will be designed to withstand the effects of natural phenomena such as earthquakes. Acceptance is based on meeting the guidance of Regulatory Guide (RG) 1.29, Position C.1 for safety-related portions and Position C.2 for nonsafety-related portions.

2. Dynamic Effects. The requirements of GDC 4 as related to protecting structures, systems and components against the dynamic effects associated with possible fluid flow instabilities (e.g., water hammers) during normal plant operation, as well as during abnormal and accident conditions are met by meeting the guidance related to feedwater-control-induced water hammer. Guidance for water hammer prevention and mitigation is found in NUREG-0927, Revision 1.

   Failure of high-energy piping, such as feedwater piping can result in complex challenges to operating staff and the plant because of potential system interactions of high energy steam. Accordingly, material standards and inspection programs, shall incorporate adequate considerations to avoid erosion and corrosion. Guidance for acceptable inspection programs is found in Generic Letter (GL) 89-08 and in Electric Power Research Institute (EPRI) NP-3944, "Erosion/Corrosion in Nuclear Plant Steam Piping: Causes and Inspection Guidelines."

3. Sharing of Structures, Systems, and Components. The requirements of GDC 5 are met by demonstrating the capability of important to safety components in the CFS, which are shared by multiple units to perform their required safety functions.

4. Heat Removal Capability. The requirements of GDC 44, as related to the capability to transfer heat from SSCs important to safety to an ultimate heat sink are met by demonstrating that the CFS is capable of providing heat removal under both normal

operating and accident conditions. Sufficient redundancy of components is demonstrated so that under accident conditions the safety function can be performed assuming a single active component failure (which may be coincident with the loss of offsite power for certain events.) The system demonstrates capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained.

5. Inspection. The requirements of GDC 45 are met by demonstrating that the design contains provisions to permit periodic inservice inspection of system components and equipment.

6. Testing. The requirements of GDC 46 are met by demonstrating that the design contains provisions to permit appropriate functional testing of the system and components to ensure structural integrity and leak-tightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, shutdown, and accident conditions.

7. Flow Accelerated Corrosion. Piping system designs, including material standards and inspection programs, shall incorporate adequate considerations to avoid erosion and corrosion. Guidance for acceptable inspection programs is found in GL 89-08 and in EPRI NP-3944, "Erosion/Corrosion in Nuclear Plant Steam Piping: Causes and Inspection Guidelines."

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 2 requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes.

   This criterion applies to DSRS Section 10.4.7 because the review identifies safety-related and risk-significant CFS components and determines that they are designed to withstand the effects of earthquakes and other natural phenomena. CFS is isolated in the event of certain accidents. RG 1.29, Positions C.1 and C.2, provide guidance for determining compliance with this criterion.

   Meeting the requirements of this criterion provides a level of assurance that the capability to shutdown the reactor safety will be maintained during the most severe expected earthquake or other natural phenomena.

2. GDC 4 requires that SSCs important to safety shall be appropriately protected against dynamic effects that may result from equipment failures and from events and conditions outside the nuclear power unit.

   GDC 4 applies to DSRS Section 10.4.7 because the review verifies that safety-related and risk-significant CFS components are protected against the effects of high-energy pipe ruptures. CFS is isolated in the event of certain accidents. This review also considers the dynamic consequences of flow instabilities (specifically, water hammer) resulting from normal operation and during anticipated operational occurrences.

Meeting the requirements of this criterion provides further assurance that the integrity of the feedwater piping will be maintained, thereby minimizing the likelihood of a loss-of-coolant accident (LOCA) that could cause fuel damage.

3. GDC 5 requires that SSCs important to safety shall not be shared by nuclear power units, unless it can be shown that such sharing will not significantly impair the ability to perform safety functions, including an orderly shutdown and cooldown of remaining units in the event of an accident in one unit.

   GDC 5 applies to DSRS Section 10.4.7 because the review determines whether safety-related and risk-significant CFS components are shared and, if so, evaluates the impact of that sharing on safety functions.

   Meeting the requirements of this criterion provides further assurance that all reactors at a multiple-unit site will be capable of completing normal shutdown in the event of a component failure in one reactor.

4. GDC 44 requires that a system be provided to transfer heat from SSCs important to safety to an ultimate heat sink. The safety function of this system shall be to transfer the specified combined heat load under normal operating and accident conditions. Suitable redundancy in components and features, as well as suitable interconnections, leak detection, and isolation capabilities, shall be provided to ensure that the system safety function can be accomplished for loss of either onsite or offsite power assuming a single failure.

   GDC 44 applies to this DSRS Section 10.4.7 because the review establishes that the CFS is capable of providing heat removal from the reactor system during normal conditions. Meeting the requirements of this criterion provides a level of assurance that the capability for heat removal from the reactor will be retained during normal and accident conditions, thus protecting fuel cladding from elevated temperatures.

5. GDC 45 requires that the cooling water system shall be designed to permit appropriate periodic inspection of important components (e.g., heat exchangers and piping) to ensure the integrity and capability of the system.

   GDC 45 applies to SRP Section 10.4.7 because the CFS provides cooling water to the reactor or steam generators and because the CFS is isolated in the event of certain accidents. This review verifies that the feedwater system design facilitates inspection.

   Meeting the requirements of this criterion provides a level of assurance that the CFS will be able to perform its safety function in the event of an accident.

6. GDC 46 requires that the cooling water system shall be designed to facilitate periodic pressure and functional testing that will ensure (a) the structural and leaktight integrity of cooling water system components, (b) the operability and the periodic performance of the system's active components, and (c) the operability of the system as a whole. The criterion further requires that the testing ensure, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for LOCAs, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

GDC 46 applies to SRP Section 10.4.7 because the CFS provides the proper cooling water inventory for PWR steam generators during normal operation. The CFS is isolated after a loss-of-feedwater accident has occurred. During such conditions, the CFS feedwater piping inside the containment is used as the conduit for feedwater flow from alternate systems. This review determines that the CFS is designed to accommodate testing the system and its components.

Meeting the requirements of this criterion provides a level of assurance that the CFS will be able to perform reliably under normal operating conditions and will perform its safety function in the event of an accident.

7. The requirements of 10 CFR 20.1406 are met when the interconnections between the CFS and other plant systems are designed to preclude CFS contamination of connecting systems, or the contamination of CFS by connections with interfacing radioactive systems.

III. REVIEW PROCEDURES

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

By following these review procedures, the acceptability of the CFS design may be determined based on review of the corresponding information in the final safety analysis report (FSAR) and identification of programmatic requirements that provide reasonable assurance CFS design, fabrication, installation, testing, and operation will satisfy the acceptance criteria in Subsection II.

Upon request from the reviewer, the interface reviewers will provide input for the areas of review stated in Subsection I. The reviewer obtains and uses such input as required to ensure that this review procedure is complete.

The FSAR is reviewed to determine that the system description and diagrams delineate the function of the condensate and feedwater system under normal and abnormal conditions. The reviewer verifies the following:

1. Programmatic Requirements – 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 that include:

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

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

C. 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.

D. Reliability Assurance Program (SRP Section 17.4).

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

F. ITAAC (DSRS Chapter 14).

2. In accordance with 10 CFR 52.47(a)(8),(21), and (22), 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 that are identified in the version of NUREG-0933 current on the date 6 months before application and that 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 safey evaluation report (SER) section.

3. The system has been designed to function as required for all modes of operation. The results of failure modes and effects analyses presented in the safey analysis report (SAR), if any, are used in making this determination.

4. The system piping is designed to preclude hydraulic instabilities from occurring in the piping for all modes of operation. As appropriate, the reviewer evaluates the results of model tests and analyses that are relied on to verify that water hammer will not occur, or proposed tests of the installed system that are intended to verify design adequacy.

   The feedwater control valve and controller design shall be verified to be stable and to be compatible with system(s) under imposed operating conditions (e.g., control functions required, range of control and pressure drop characteristics, valve stroke, trim, etc.). Test data or operating experience data shall be used where available. In addition, the applicant has committed to review plant operating and maintenance procedures to ensure that precautions for avoidance of steam/water hammer and water hammer occurrences have been provided.

   Guidance for water hammer prevention and mitigation is found in NUREG-0927.

5. The outermost containment isolation valves and all downstream piping to the nuclear steam supply system are designed in accordance with seismic Category I requirements. The review for seismic design and the review for seismic and quality group classification are performed as indicated in Subsection I of this DSRS section.

6. The CFS design, or other plant systems, provide the capability to detect and control leakage from the system.

7. The essential portion of the system has been designed so that system function will be maintained as required in the event of adverse environmental phenomena or loss of offsite power. The review for protection against natural phenomena is performed in the Chapter 3 DSRS sections. The reviewer evaluates the system, using engineering judgment and the results of failure modes and effects analyses, to determine that the failure of nonessential portions of the system or of other systems not designed to seismic Category I standards (and located close to essential portions of the system), will not preclude operation of the essential portions of the CFS. The reviewer shall also ensure that failure of nonseismic Category I structures that house, support, or are close to essential portions of the CFS will not preclude operation of the essential portions of the CFS.

8. Piping system designs, including material standards and inspection programs, incorporate adequate considerations to avoid erosion and corrosion. Guidance for acceptable inspection programs is found in GL 89-08 and in EPRI NP-3944, "Erosion/Corrosion in Nuclear Plant Steam Piping: Causes and Inspection Guidelines."

9. For multiple-unit sites, sharing of any CFS safety-related and risk-significant SSCs will not impair its ability to perform its intended safety function.

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 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 (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

For review of both DC and COL applications, DSRS 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 staff's technical review and analysis, as augmented by the application of programmatic requirements in accordance with the staff's technical review approach in the NUREG - 0800 "Introduction," Part 2 support conclusions of the following type to be included in the staff's SER. The reviewer also states the bases for those conclusions.

The condensate and feedwater system includes all components and equipment from the condenser outlet to the connection with the nuclear steam supply system, heater, secondary makeup and cleanup system. Based on the review of the applicants proposed design criteria, the design bases, and safety classification for the safety-related portions of the condensate and feedwater system and the requirements for system performance for all conditions of plant operation, the staff concludes that the design of the condensate and feedwater system and supporting systems is in conformance with the Commission regulations as set forth in GDCs 2, 4, 5, 44, 45, and 46. This conclusion is based on the following and the fact that programmatic requirements will provide assurance that the CFS will be designed, installed, and tested as described in the FSAR:

1. The applicant has met the requirements of GDC 2 with respect to safety-related portions of the system being capable of withstanding the effects of earthquakes by meeting RG 1.29, Position C.1 for the safety-related portions and Position C.2 for the nonsafety-related portions.

2. The applicant has met the requirements of GDC 4 with respect to the dynamic effects associated with possible fluid flow instabilities (e.g., water hammers) by having the feedwater system designed in accordance with the guidance contained in NUREG-0927 and thereby eliminating or reducing the possibility of water hammers in steam generators

   The applicant has adequately addressed feedwater control valve and controller designs with respect to water hammer potential and the applicant has committed to review operating and maintenance procedures to ensure that precautions taken will minimize, or avoid, water hammers.

3. The applicant has met the requirements of GDC 5 with respect to the capability of shared safety-related or risk significant systems and components perform required functions. The interconnections of the CFS between each unit are designed so that the capability to mitigate the consequences of an accident in either unit and to achieve safe-shutdown in that unit is retained without reducing the capability of the other unit to achieve safe-shutdown.

4. The applicant has met the requirements of GDC 44 with respect to cooling water by providing a redundant and isolable system capable of transferring heat loads from the reactor system to a heat sink under both normal operating and accident conditions. The applicant has demonstrated that the condensate and feedwater system can provide sufficient cooling water to transfer the heat load of the reactor system under normal operating conditions. The applicant has also demonstrated that portions of the system can be isolated during accidents that occur concurrently with loss of onsite or offsite power and a single failure so that the safety function of the system will not be compromised.

5. The applicant has met the requirements of GDC 45 with respect to inspection of cooling water systems by providing a feedwater system design that permits inservice inspection of safety-related components and equipment, including inspection of piping systems for erosion and corrosion, and inspection of feedwater nozzles for fatigue.

6. The applicant has met the requirements of GDC 46 with respect to testing of cooling water systems by providing a feedwater system design that permits operational functional testing of the system and its components. Functional testing ensures structural integrity and leaktightness, operability, and performance of active components during normal, shutdown, and accident conditions.

The staff concludes that the design of the CFS conforms to all applicable GDCs and positions of the RG cited and is, therefore, acceptable. 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 DSRS 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^{TM} and large light-water nuclear reactor power plants, and in accordance with the direction given by the Commission in SRM-COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010 (Agencywide Documents Access and Management System Accession No. ML102510405), to develop risk-informed licensing review plans for each of the small modular reactor reviews, including the associated pre-application activities, the staff has developed the content of this DSRS section as an alternative method for mPower^{TM} -specific DC, COL, or ESP applications 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 Standard Review Plan (SRP) revision in effect 6 months before the docket date of the application." The content of this DSRS section has been accepted as an alternative method for complying with 10 CFR 52.47(a)(9), as long as the mPower^{TM} DCD FSAR does not deviate significantly from the design assumptions made by the NRC staff while preparing this DSRS section. The application must identify and describe all differences between the standard plant design and this DSRS section, and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria. If the design assumptions in the DC application deviate significantly from the DSRS, the staff will use the SRP as specified in 10 CFR 52.47 (a)(9). Alternatively, the staff may supplement the DSRS section by adding the 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 20.1406, 'Minimization of Contamination."

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

3. 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases."

4. 10 CFR Part 50, Appendix A, GDC 5, "Sharing of Structures, Systems, and Components."

5. 10 CFR Part 50, Appendix A, GDC 44, "Cooling Water."

6. 10 CFR Part 50, Appendix A, GDC 45, "Inspection of Cooling Water System."

7. 10 CFR Part 50, Appendix A, GDC 46, "Testing of Cooling Water System."

8. RG 1.29, "Seismic Design Classification."

9. RG 1.215, “Guidance for ITACC Closure Under 10 CFR Part 52.”

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

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

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

13. GL 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning."

14. NUREG-0927, Revision 1, "Evaluation of Water Hammer Occurrences in Nuclear Power Plants," March 1984.

15. EPRI NP-3944, "Erosion/Corrosion in Nuclear Plant Steam Piping: Causes and Inspection Guidelines."

16. 10 CFR 52.47, “Contents of applications.”

17. 10 CFR 52.80(a), “Issuance of combined licenses.”