U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

15.3.1-15.3.2 LOSS OF FORCED REACTOR COOLANT FLOW - TRIPS OF ONE OR MORE PUMP MOTORS AND FLOW CONTROLLER MALFUNCTIONS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for review of transient and accident analyses

Secondary - None

I. <u>AREAS OF REVIEW</u>

The mPower™ reactor coolant system comprises a reactor vessel with a single internal steam generator (SG) and multiple internal reactor coolant pumps (RCPs). After passing upward through the reactor core, primary coolant flows up through a riser and down through the steam generator tubes, returning to the RCPs. Feedwater and steam lines penetrate the vessel providing the secondary flow to the tube-side of the single steam generator to remove the heat generated in the core.

The forced reactor coolant flow in mPower™ may be reduced as a result of trips of one or more of its reactor coolant pump motors or by flow controller malfunctions. The reactor response to the initiating event depends on the assumed extent of flow reduction. For a relatively small reduction in flow, for example a single pump trip (which may not lead to a reactor or turbine trip), it is possible that the reactor will continue full-power operation under forced circulation conditions. In the extreme case of a complete loss of reactor coolant flow in mPower™ the reactor would trip with subsequent transition to natural circulation primary system flow. Because they are non-safety related systems, It is assumed that the balance-of-plant (BOP) system and the reactor coolant inventory and purification system (RCIPS) are not available for rejection of core power. Under this complete loss of flow accident, the emergency core cooling system (ECCS) is activated.

A decrease in reactor coolant flow while a plant is at power could result in degraded core heat transfer. An increase in fuel temperature and accompanying fuel damage then could result if specified acceptable fuel design limits (SAFDLs) are exceeded during the transient. This Design-Specific Review Standard (DSRS) section covers a number of transients expected to occur with moderate frequency that decrease forced reactor coolant flow rate. Each transient should be addressed in individual sections of the applicant's technical submittal as specified in Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" and RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

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The specific areas of review for mPower^{TM} are as follows:

1. Core thermal and hydraulic transients with partial and complete loss of reactor coolant flow are evaluated. The potential for flow rate reduction resulting from spurious actuation of flow controllers is reviewed.

2. A partial loss of coolant flow may be caused by a mechanical or electrical failure in a pump motor, a fault in the power supply to the pump motor, a pump motor trip caused by such anomalies as over-current or phase imbalance. A complete loss of forced coolant flow may be the result of the simultaneous loss of electrical power to all pump motors.

3. The review includes the postulated initial core and reactor conditions pertinent to the loss of flow transient; the methods of thermal and hydraulic analysis; the postulated sequence of events, including time delays before and after protective system actuation; assumed reactions of reactor system components; the functional and operational characteristics of the reactor protection system affecting the sequence of events; and all operator actions required to secure and maintain the reactor in a safe condition.

4. Results of the applicant's analyses are reviewed for whether values of pertinent system parameters are within expected ranges for the type and class of reactor under review. The system parameters evaluated include core flow and flow distribution, channel heat flux (average and hot), minimum critical heat flux ratio (or minimum critical power ratio), departure from nucleate boiling ratio (DNBR), vessel water level, thermal power and vessel pressure. Results of the applicant's fuel damage analysis are reviewed by the methods described in DSRS Section 4.2.

5. The sequence of events described in the applicant's technical submittal is reviewed by the organization responsible for the review of reactor systems and coordinated with the organization responsible for instrumentation and controls. The reactor systems review concentrates on the need for the reactor protection system, the engineered safety system, and operator action to secure and maintain the reactor in a safe condition.

6. Analytical methods are reviewed for whether the mathematical modeling and computer codes have been reviewed and accepted by the staff. If a referenced analytical method has not been reviewed, the reactor systems reviewer initiates a generic evaluation of the new analytical model.

7. The values of all parameters in a new analytical model, including initial core and system conditions, are reviewed. The reactor systems reviewer is responsible for the use of appropriate physics and fuel data in any staff calculations.

8. <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.

Review Interfaces

Other DSRS sections interface with this section as follows:

1. General information on transient and accident analyses is provided in DSRS Section15.0.

2. The design bases, the design, the test programs and the proposed technical specifications for the ECCS are reviewed under DSRS 6.3.

3. The design of the overpressure protection system is reviewed under DSRS 5.2.2 to gain familiarity with the design and operation of the pressure relief system.

4. Design basis radiological consequence analyses associated with design basis accidents are reviewed under DSRS Section 15.0.3.

5. Instrumentation and controls aspects of the sequence described in the applicant's technical submittal are reviewed to confirm that reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis under DSRS Sections 7.2 through 7.5.

6. Generic reviews of the thermal-hydraulic computer models used for this transient and, as appropriate, additional analyses related to these accidents for selected reactor types are reviewed under DSRS Section 4.4.

7. Preoperational tests are reviewed under DSRS Section 14.2. The primary reviewer of this section confirms with the lead reviewer of 14.2 that a commitment has been made in the applicant's technical submittal to conduct preoperational tests to verify flow coastdown calculations.

8. The determination of the risk significance of SSCs relied upon to meet required functions during the accidents are based on the review of the probabilistic risk analysis under SRP Chapter 19.

II. ACCEPTANCE CRITERIA

Requirements

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

1. 10 CFR 50, Appendix A, General Design Criteria (GDCs) 10 as to design of the reactor core and associated coolant, control, and protection systems with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).

2. 10 CFR 50, Appendix A, GDC 13 as to the providing of instrumentation to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variable and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

3. 10 CFR 50, Appendix A, GDC 15 as to design of the reactor coolant system and associatedauxiliary, control and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs.

4. 10 CFR 50, Appendix A, GDC 17 as to onsite and offsite electric power systems being provided so structures, systems, and components (SSCs) important to safety function during normal operation, including AOOs. The safety function for each power system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) SAFDLs and design conditions of the reactor coolant pressure boundary are not exceeded as a result of AOOs.

5. 10 CFR 50, Appendix A, GDC 20 as to design of the protection system (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that SAFDLs are not exceeded as a result of AOOs and (2) to sense accident conditions and to initiate the operation of SSCs important to safety.

6. 10 CFR 50, Appendix A, GDC 26 as to the reliable control of reactivity changes to assure that under conditions of normal operation, including AOOs, and with appropriate margin for malfunctions such as stuck rods, specific acceptable fuel design limits are not exceeded.

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above 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 Title 10 of the *Code of Federal Regulations* (10 CFR) Part 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.

The basic objectives of the review of loss of forced reactor coolant flow transients are to identify the most limiting transients and to verify whether, for the most limiting transients, the plant response to the loss of flow transients satisfies fuel damage and system pressure criteria.

The following specific criteria are necessary to meet the regulatory requirements for incidents of moderate frequency:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

2. Fuel-cladding integrity must be maintained by the minimum DNBR remaining above the 95 percent probability/95 percent confidence DNBR limit based on acceptable correlations (see DSRS Section 4.4).

3. An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently.

4. The requirements stated in RG 1.105, "Instrument Spans and Setpoints," are evaluated for their impact on the plant response to AOOs addressed in this DSRS section.

5. Onsite and offsite electric power systems must be maintained so safety-related SSCs function during normal operation and AOOs.

6. The most limiting plant system single failure, as defined in the "Definitions and Explanations" of 10 CFR 50, Appendix A, must be assumed in the analysis and should follow the guidance of RG 1.53.

7. The performance of nonsafety-related systems during transients and accidents and of single failures of active and passive systems (especially the performance of check valves in passive systems), must be evaluated and verified by the guidance of SECY 77-439, SECY 94-084 and RG 1.206.

8. The applicant's analysis of the most limiting AOOs should use an acceptable model. Unapproved analytical methods proposed by the applicant are evaluated by the staff for acceptability.

9. Parameter values in the analytical model should be suitably conservative. The following values are acceptable:

    A. Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed operating plus an allowance of 2 percent to account for power measurement uncertainty unless (i) a lower number can be justified through the measurement uncertainty methodology and evaluation or (ii) the uncertainty is accounted for otherwise (see DSRS 4.4). The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.

B. Conservative scram characteristics are assumed (e.g., maximum time delay with the most reactive rod held out of the core unless (i) a different conservatism factor can be justified through the uncertainty methodology and evaluation or (ii) the uncertainty is accounted for otherwise (see DSRS Section 4.4).

C. The core burn-up is selected to yield the most limiting combination of moderator temperature reactivity feedback, void reactivity feedback, Doppler reactivity feedback, axial power profile, and radial power distribution.

D. Mitigating systems should be assumed as actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with RG 1.105 and as determined by the organization responsible for instrumentation and controls.

10. Programmatic Requirements: The NRC regulations require that each operating license contain a technical specification (TS) that define "…the limits, operating conditions, and other requirements imposed upon facility operation for the protection of public health and safety…" The licensee's analysis of DSRS 15.3.1-15.3.2 must be consistent with the information presented in the licensee's TS.

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 10 requires design of the reactor core and its coolant, control, and protection systems with appropriate margin so SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.

   GDC 10 applies to this section because the reviewer evaluates the consequences of loss of forced reactor coolant flow, including pump motor trips and flow controller malfunctions, AOOs that may cause SAFDLs to be exceeded because a transient reduction in reactor coolant flow causes a corresponding rise in fuel-cladding temperature.

   GDC 10 requirements assure that SAFDLs are not exceeded and that fuel-cladding integrity is maintained for AOOs involving loss of forced-reactor coolant flow.

2. GDC 13 requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to assure adequate safety, and of controls that can maintain these variables and systems within prescribed operating ranges.

   GDC 13 applies to this section because the reviewer evaluates the sequence of events, including automatic actuations of projection systems, and manual actions, and determines whether the sequence of events is justified, based upon the expected values of the relevant monitored parameters and instrument indications.

3. GDC 15 requires design of the reactor coolant system and its auxiliary, control, and protection systems with sufficient margin so design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs.

   GDC 15 applies to this section because the reviewer analyzes AOOs involving loss of forced reactor coolant flow. In these transients, a reduction in reactor coolant flow can cause the reactor coolant system pressure to increase above normal levels; therefore, for loss-of-flow transients under this DSRS section, the reactor coolant pressure should be analyzed to satisfy the pressure acceptance criterion.

   GDC 15 requirements assure that the design conditions of the reactor coolant pressure boundary are not exceeded for AOOs of loss of forced reactor coolant flow evaluated in this DSRS section.

4. GDC 17 requires onsite and offsite electrical power systems so safety-related SSCs perform intended functions. Each power system (assuming the other system is not functioning) must provide sufficient capacity and capability so SAFDLs and design conditions of the reactor coolant pressure boundary are not exceeded in AOOs.

   GDC 17 applies because this DSRS section reviews the analysis of a group of abnormal operating occurrences to which GDC 17 must be applied.

   GDC 17 requirements assure that SAFDLs and design conditions of the reactor coolant pressure boundary are not exceeded in initiating events that decrease flow in the reactor coolant system concurrent with a loss of offsite power (LOOP).

5. GDC 20 requires design of the protection system (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that SAFDLs are not exceeded as a result of AOOs and (2) to sense accident conditions and to initiate the operation of SSCs important to safety.

   GDC 20 applies to this DSRS section because the reviewer evaluates the consequences of AOOs of loss of forced reactor coolant flow, including pump motor trips or flow controller malfunctions. This section, DSRS Sections 4.2 through 4.4 and 7.2 through 7.5, and RGs 1.53 and 1.105 provide guidance for reactor coolant system design with appropriate margin; thus, when the reactor protection system senses an accident condition, it initiates the operation of safety-related SSCs so SAFDLs are not exceeded.

   GDC 20 requirements assure that SAFDLs are not exceeded during any AOO of loss of forced reactor coolant flow, including pump motor trips or flow controller malfunctions.

6. GDC 26 requires that one of the reactivity control systems at nuclear power plants include control rods that can control reactivity changes so SAFDLs are not exceeded under conditions of normal operation, including AOOs. The design for this system must have an appropriate margin to accommodate malfunctions like stuck rods.

GDC 26 applies to this DSRS section because the reviewer analyzes AOOs involving loss of forced reactor coolant flow. The transients analyzed in this section may involve the movement of control rods in response to the transient. Rod misalignment, including stuck rods, can aggravate thermal-hydraulic conditions. GDC 26 requires a thermal margin sufficient to accommodate these conditions. Review under this DSRS section examines this margin for whether SAFDLs are exceeded.

GDC 26 requirements assure inclusion of appropriate margins to accommodate malfunctions (e.g., stuck rods) of the reactivity control system, assurance that SAFDLs are not exceeded.

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.

The procedures below are used during the construction permit (CP), COL, and operating license (OL) reviews. During the CP review, the values of system parameters and setpoints in the analysis are preliminary and subject to change. At the OL or COL review stage, final values should be in the analysis, and the reviewer should compare these to the limiting safety system settings in the proposed technical specifications.

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 Standard Review Plan (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 (USIs) and medium- and high-priority generic safety issues (GSIs) 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). Reference: 10 CFR 52.47(a)(21), 10 CFR 52.47(a)(22) , and 10 CFR 52.47(a)(8), respectively. 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 section.

3. The applicant's technical submittal description of each loss of reactor coolant flow transient is reviewed for occurrences leading to the initiating event. The sequence of events from initiation until stabilization is reviewed to ascertain:

A. The extent to which normally operating plant instrumentation and controls are assumed to function.

B. The extent to which plant and reactor protection systems are required to function.

C. The extent to which credit is taken for the functioning of normally operating plant systems.

D. The extent to which the operation of engineered safety systems is required.

E. The extent to which operator actions are required.

F. Whether the description accounts for appropriate margin for malfunctions (e.g., stuck rods).

G. Whether the description accounts for instrumentation uncertainties of system and operating parameters appropriately.

4. If the applicant's technical submittal states that a particular loss of flow transient is not as limiting as some other similar transients, the reviewer evaluates the applicant's justification. The reviewer confirms whether all types of flow loss transients are

considered (e.g., pump trips during two-, three-, and four-loop operations). The applicant's technical submittal must present a quantitative analysis of the most limiting loss of flow transient. For this transient, the reactor systems reviewer, in coordination with the instrumentation and controls reviewer, reviews the timing of the initiation of protection, engineered-safety, and other systems needed to limit the consequences of the loss of flow adequately. The reviewer compares the predicted variation of system parameters to various trip and system initiation setpoints and evaluates the effects of system and component single, active failures which may alter the course of the transient. For new applications, LOOP should not be considered a single failure; each loss of flow transient should be analyzed with and without a LOOP in combination with a single active failure. The instrumentation and controls review of applicant's technical submittal Chapter 7 confirms whether the instrumentation and control design is consistent with the requirements for safety system actions for these events.

5. The applicant's mathematical models to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed for whether these models have been reviewed and accepted by the staff. If not, the reviewer initiates a generic review of the applicant's proposed model.

6. The values of system parameters and initial core and system conditions as input to the model are reviewed. Of particular importance are the reactivity feedbacks and control rod worths in the applicant's analysis and the variations of moderator temperature, void, and Doppler reactivity feedbacks with core life. The reviewer evaluates both the justification showing that the applicant has selected the core burn-up yielding the minimum margins and the values of the reactivity parameters in the applicant's analyses.

7. The results of the analysis are reviewed and compared to the acceptance criteria of subsection II of this DSRS section for the maximum pressure in the reactor coolant and main steam systems as well as minimum DNBR. Time-related variations of the following parameters should be reviewed for consistency:

    A. reactor power;

    B. heat fluxes (average and maximum);

    C. reactor coolant system pressure;

    D. core coolant flow rates;

    E. coolant conditions (inlet temperature, core average temperature, average exit and hot channel exit temperatures, and steam fractions);

    F. pressure relief valve flow rate; and

    G. flow rate from the reactor coolant system to the containment system (if applicable).

Values of the more important of these parameters for the core flow increase AOOs are compared to those predicted for other similar plants for whether they are within the expected range. The reactor systems organization reviews the SAFDLs. The organization responsible for emergency preparedness and radiation protection is notified of the extent of fuel failures predicted by the analysis if SAFDLs are exceeded. The quality assurance and maintenance review confirms whether the applicant's technical submittal commits to conduct pre-operational tests to verify flow coast-down calculations.

8. 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 applicant's technical submittal meets the acceptance criteria. DCs have referred to the applicant's technical submittal 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 applicant's technical submittal.

   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).

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 DSRS Introduction, 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.

Several types of plant occurrences can result in an unplanned decrease in reactor coolant flow rate. Those expected during the life of the plant result in reactor coolant (or recirculation) pump trips or flow controller malfunctions. All these postulated transients have been reviewed. The transient was found the most limiting for core thermal margins and pressure within the reactor coolant and main steam systems. The applicant analyzed this transient using a mathematical model reviewed and accepted by the staff. The values of the input parameters to this model were reviewed and found suitably conservative.

The staff concludes that the plant design for transients expected to occur during plant life and result in a loss or decrease in forced reactor coolant flow is acceptable and meets the requirements of GDCs 10, 13, 15, 17, 20 and 26. This conclusion is based on the following findings:

1. The applicant meets the requirements of GDCs 10, 17, 20, and 26 by demonstrating that SAFDLs are not exceeded in this event. This requirement is met as the results of the analysis show that the thermal margin limit (minimum DNBR) is satisfied as indicated by DSRS Section 4.4.

2. The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation was available, and that actuations of protection systems, automatic and manual, occurred at values of monitored parameters that were within the instruments' prescribed operating ranges.

3. The applicant meets the requirements of GDCs 15 and 17 by demonstrating that the reactor coolant pressure boundary limits are exceeded in this event. This requirement is met as the analysis shows that the maximum pressure of the reactor coolant and main steam systems does not exceed 110 percent of the design pressure.

4. The applicant meets GDC 26 requirements for the capability of the reactivity control system to control reactivity adequately during this event with appropriate margin for stuck rods because the SAFDLs are not exceeded.

5. The applicant meets the positions of RG 1.53, SECY 77-439, SECY 94-084, and RG 1.206 on the single-failure criterion and RG 1.105 on instrument actuations of safety-related SSCs.

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.

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 (SMR) 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 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™ DCD final safety analysis report 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), and COL applications.

VI. REFERENCES

1. 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities."

2. 10 CFR 50, Appendix A, GDC 10, "Reactor Design."

3. 10 CFR 50, Appendix A, GDC 13, "Instrumentation and Control."

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

5. 10 CFR 50, Appendix A, GDC 17, "Electric Power Systems."

6. 10 CFR 50, Appendix A, GDC 20, "Protection System Functions."

7. 10 CFR 50, Appendix A, GDC 26, "Reactivity Control System Redundancy and Capability."

8. 10 CFR 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

9. RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems."

10. RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

11. RG 1.105, "Instrument Spans and Setpoints."

12. RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

13. NUREG-0737, "Clarification of TMI Action Plan Requirements."

14. NUREG-0933, "A Prioritization of Generic Safety Issues."

15. NUREG-1801, "Generic Aging Lessons Learned Report," Revision 1, volumes 1-2.

16. SECY-77-439, "Single Failure Criterion."

17. SECY-94-084, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs."

18. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

19. ANSI/ANS-51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," (replaced ANSI N18.2-1974, reaffirmed 1988, withdrawn 1998).