U.S. NUCLEAR REGULATORY COMMISSION

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

15.2.1-15.2.5 LOSS OF EXTERNAL LOAD; TURBINE TRIP; LOSS OF CONDENSER VACUUM; CLOSURE OF MAIN STEAM ISOLATION VALVE; AND STEAM PRESSURE REGULATOR FAILURE (CLOSED)

REVIEW RESPONSIBILITIES

Primary - Organization responsible for review of transient and accident analyses for pressured-water reactors and boiling-water reactors

Secondary - None

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

A number of initiating events that occur with moderate frequency results in unplanned decreases in heat removal by the secondary system. Each event covered in this design-specific review standard (DSRS) section should be addressed in individual sections of the safety analysis report (SAR) or design control document (DCD) 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)."

The mPower^{TM} emergency core cooling system (ECCS) is a safety-related system designed to provide core cooling with water stored inside containment for a minimum of 72 hours. The safety function is accomplished passively without alternating current (AC) power and assuming a single failure.

The specific areas of review are as follows:

1. <u>Loss of External Load</u>: In a loss of external load event, an electrical disturbance causes loss of a significant portion of the generator load. This loss of load situation is different from the loss of AC power condition considered in DSRS Section 15.2.6 in that offsite AC power remains available to operate the station auxiliaries (e.g., reactor coolant pumps). Onsite emergency diesel generators are therefore not required for the loss of external load event. Immediate fast closure of the turbine control valves (TCVs) is initiated for a loss of generator load. Following the loss of load without operation of the turbine bypass or the main steam pressure relief (PORVs), atmospheric dump or safety relief valves, there is a sudden reduction in steam flow causing the pressure and temperature in the shell side of the once-through steam generator to increase. The latter effect, in turn, results in an increase of reactor coolant temperature, a decrease in coolant density, an increase of water volume in the pressurizer, and an increase in reactor coolant pressure. The reactor and the reactor coolant pumps trip, and the ECCS is initiated. The primary system is depressurized and decay heat is transferred to the ultimate heat sink (UHS).

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2. <u>Turbine Trip</u>: In a turbine trip event, a malfunction of a turbine or reactor system causes the turbine to trip off the line by abruptly stopping steam flow to the turbine. Without operation of the turbine bypass or the main steam pressure relief (PORVs), atmospheric dump or safety relief valves there is a sudden reduction in steam flow causing the pressure and temperature in the shell side of the once-through steam generator to increase. The latter effect, in turn, results in an increase of reactor coolant temperature, a decrease in coolant density, an increase of water volume in the pressurizer, and an increase in reactor coolant pressure. The reactor and the reactor coolant pumps trip, and the ECCS is initiated. Decay heat is transferred to the UHS. This event may be different from the loss of external load conditions as a result of differences in the steam flow reduction time scale.

3. <u>Loss of Condenser Vacuum</u>: A loss of condenser vacuum event is a malfunction that can result in a turbine trip; thus, the remarks in Paragraph 2 apply to this event. In addition, due to system interaction, the loss of condenser vacuum event also causes the condensate and feedwater pumps to trip due to low suction pressure. The corresponding peak pressure in the primary and secondary systems requires separate analysis because the initial conditions that lead to peak pressure are different for the primary and secondary systems.

4. <u>Main Steam Isolation Valve (MSIV) Closure</u>: The effect of MSIV closure is limited steam flow to the turbine. The results are similar to those addressed in Paragraph 1.

5. <u>Steam Pressure Regulator Failure</u>: Steam pressure regulator failure in a closed position yields a transient similar to those previously addressed. Generally, the rate of change of system parameters is slower for a steam pressure regulator failure, and less severe transient results.

6. Review of these five described transients includes the sequence of events, the analytical models, the values of parameters in the analytical models, and the predicted consequences of the transients.

    A. The sequence of events described in the analysis is reviewed with concentration on the assumptions for the reactor protection system, the engineered safety systems, and required operator actions to secure and maintain the reactor in a safe condition.

    B. The reactor systems review includes the analytical methods and considers whether all mathematical models and computer codes have been reviewed and accepted by the staff. If a referenced analytical method or code has not been reviewed, then a generic evaluation of the new analytical model or code needs to be performed.

    C. The results of the analyses are reviewed for whether predicted values of pertinent system parameters are within expected ranges for the type and class of reactor under review. The predicted results of the transient analyses then are reviewed for whether the consequences meet the acceptance criteria of Subsection II of this DSRS section.

D. A review is performed of all parameters in the analytical models, including the initial conditions of the core and systems. In addition, the review includes core physics, fuel design, and core thermal-hydraulics data in the SAR (or DCD) analysis as part of the review of DSRS Sections 4.2 through 4.4. Finally, Section 5.2.2 is reviewed for adequacy of the overpressure protection of the reactor coolant pressure boundary (RCPB).

7. Combined Operating License (COL) Action Items and Certification Requirements and Restrictions. For a design certification (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 Section 15.0.

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

3. The sequences of events are reviewed with respect to the reactor system and its interfaces with instrumentation and control systems. Aspects of the transient sequences described in the SAR (or DCD) are evaluated to determine whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems under DSRS Sections 7.2 and 7.3.

4. Potential bypass modes and the possibility of manual control by the operator are reviewed under DSRS Sections 7.2 through 7.5.

5. Technical specifications (TSs) are reviewed under DSRS Section 16.0.

6. The determination of the safety-related and risk-significant of SSCs relied upon to meet required functions during the accidents are based on the review of the probabilistic risk analysis under Standard Review Plan (SRP) Chapter 19.

II. ACCEPTANCE CRITERIA

Requirements

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

1. General Design Criterion (GDC) 10, as to reactor coolant system design with appropriate margin so specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations, including anticipated operational occurrences (AOOs).

2. GDC 13, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, as to design of the reactor coolant system and its auxiliaries with appropriate margin so the pressure boundary is not breached during normal operations, including AOOs.

4. GDC 17, as to onsite and offsite electric power systems so safety-related structures, systems, and components (SSCs) function during normal operation, including AOOs. The safety function for each power system (assuming the other system is not functioning) is to provide sufficient capacity and capability so SAFDLs and RCPB design conditions are not exceeded during AOOs.

5. GDC 26, as to the control of reactivity changes so SAFDLs are not exceeded during AOOs. This control is accomplished by provisions for appropriate margin for malfunctions (*e.g.*, stuck rods).

<u>DSRS Acceptance Criteria</u>

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 of the *Code of Federal Regulations* (CFR), Section 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. The basic objectives of the review of the initiating events listed in Subsection I of this DSRS section:

    A. To identify which moderate-frequency event that results in an unplanned decrease in secondary system heat removal is the most limiting, in particular as to primary pressure, secondary pressure, minimum departure from nucleate boiling ratio (DNBR) and long-term decay heat removal.

    B. To verify whether the predicted plant response for the most limiting event satisfies the specific criteria for fuel damage and system pressure.

    C. To verify whether the plant protection systems setpoints assumed in the transients analyses are selected with adequate allowance for measurement inaccuracies as delineated in RG 1.105.

D. To verify whether the event evaluation considers single failures, operator errors, and performance of nonsafety-related systems consistent with the RG 1.206 regulatory guidelines.

2. With the American Nuclear Society (ANS) standards as guidance, specific criteria meet the relevant requirements of GDCs 10, 13, 15, 17, and 26 for events of moderate frequency.

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

B. Fuel cladding integrity must be maintained by the minimum DNBR remaining above the 95/95 DNBR limit based on acceptable correlations (see SAR (or DCD) Section 4.4) and by satisfaction of any other SAFDL applicable to the particular reactor design.

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

D. The requirements in RG 1.105, "Instrument Spans and Setpoints," are used for their impact on the plant response to the type of AOOs addressed in this DSRS section.

E. The most limiting plant system single failure, as defined in "Definitions and Explanations," 10 CFR Part 50, Appendix A, must be assumed in the analysis according to the guidance of RG 1.53 and GDC 17.

F. Performance of nonsafety-related systems during transients and accidents and single failures of active and passive systems (especially as to the performance of check valves in passive systems) must be evaluated and verified according to the guidance of SECY-77-439, SECY-94-084, SECY-95-132, and RG 1.206

3. The applicant should analyze these events using an acceptable analytical model. Any other analytical method proposed by the applicant is evaluated by the staff for acceptability. Staff performs an evaluation of new generic methods.

The values of the parameters in the analytical model should be suitably conservative. The following values are acceptable:

A. The reactor is initially at 102 percent of the rated (licensed) core thermal power (to account for a 2 percent power measurement uncertainty unless a lower number can be justified through measurement uncertainty methodology and evaluation or unless the uncertainty otherwise is accounted for (see SAR (or DCD) Section 4.4)), and primary loop flow is at the nominal design flow less the flow measurement uncertainty.

B. Conservative scram characteristics are assumed (maximum time delay with the most reactive rod held out of the core) unless (1) a different conservatism factor can be justified through the uncertainty methodology and evaluation or (2) the uncertainty is otherwise accounted for (see SAR (or DCD) 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 to be actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with RG 1.105.

Programmatic Requirements: The U.S. Nuclear Regulatory Commission (NRC) regulations require that each operating license contain a 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 15.2.1-15.2.5 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 reactor core and its coolant, control, and protection systems with appropriate margin so SAFDLs are not exceeded during any conditions of normal operation, including the effects of AOOs.

   GDC 10 applies to this section because the reviewer evaluates the consequences of AOOs that could decrease heat removal by the secondary system and result in the fuel cladding thermal design criteria to be exceeded. RG 1.105 provides guidance for keeping instrument setpoints within TS limits.

   GDC 10 requirements provide assurance that SAFDLs are not exceeded for initiating events that decrease heat removal by the secondary system.

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 sequences of events, including automatic actuations of protection systems, and manual actions, and determines whether the sequence of events is justified, based upon the expected values of the relevant monitored parameters and instruments indications.

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

   GDC 15 applies to this section because the reviewer evaluates the consequences of AOOs that could decrease heat removal by the secondary system and lead to an increase in the reactor coolant temperature and pressure.

   GDC 15 requirements provide assurance that RCPB design conditions are not exceeded for initiating events that decrease heat removal by the secondary system.

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

   GDC 17 applies to this DSRS section because it governs review of the analysis of abnormal operating occurrences to which it must be applied.

   GDC 17 requirements provide assurance that SAFDLs and RCPB design conditions are not exceeded in initiating events that decrease heat removal by the secondary system, concurrent with a loss of offsite power (LOOP).

5. GDC 26 requires two independent reactivity control systems with different design principles to control reactivity changes so acceptable fuel design limits are not exceeded.

   GDC 26 applies to this section because the reviewer evaluates the consequences of AOOs that could decrease heat removal by the secondary system and lead to reactivity changes within the core causing the fuel cladding thermal design criteria to be exceeded.

   GDC 26 requires reactivity control systems to control reactivity changes reliably with appropriate margin for malfunctions (*i.e.*, stuck control rods) so that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. Where applicable, the reviewer examines these margins for whether thermal criteria are satisfied.

   GDC 26 requirements provide assurance that SAFDLs are not exceeded, ensuring an appropriate margin for malfunctions of the reactivity control system.

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 are used for the DC application review, the construction permit (CP), operating license (OL), and COL reviews. During below 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 TSs.

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 of this DSRS, 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 structures, systems, and components (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 1.18, “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. TSs (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). 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 (SER) section.

3. The SAR (or DCD) description of these transients is reviewed for the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed for:

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 operation of engineered safety systems is required.

E. The extent to which operator actions are required.

F. Appropriate margin for malfunctions (*e.g.*, stuck rods).

G. Appropriate accounting for instrumentation uncertainties of system and operating parameters.

4. If the SAR (or DCD) states that one of these transients is not as limiting as other similar transients, the reviewer evaluates the applicant's justification. The SAR (or DCD) must present a quantitative analysis of the most limiting reduction-of-heat-removal transient. For this transient, the reactor systems reviewer, in consultation 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 transient adequately to an acceptable level. The reactor systems reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The instrumentation and controls reviewer consults on automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR (or DCD) states that operator action is needed or expected.

5. To the extent deemed necessary, the reviewer evaluates the effect of single active system or component failures that may affect the course of the transient. For new applications, LOOP should not be considered a single failure; each of the reduction-of-heat-removal transients should be analyzed with and without a LOOP in combination with a single active failure. This phase of the review uses the system review procedures described in the DSRSs for SAR (or DCD) Chapters 5, 6, 7, and 8.

6. The applicant's mathematical models to evaluate core performance and to predict system pressure in the reactor coolant system and main steam line 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.

7. The values of system parameters and initial core and system conditions as input to the model are reviewed. Of particular importance are (1) the reactivity feedback and control rod worths in the applicant's analysis and (2) the variations of moderator temperature, void, and Doppler reactivity feedback with core life. The reviewer evaluates the applicant's justification showing that the core burn-up selected yields the minimum safety margins.

8. The results of the analysis are reviewed and compared to the acceptance criteria of Subsection II of this DSRS section for fuel integrity, the possibility of the event becoming more serious, and the maximum pressure in the reactor coolant and main steam systems. The following parameters are reviewed:

A. reactor power;
B. heat fluxes (average and maximum);
C. reactor coolant system pressure;
D. minimum DNBR;
E. core flow rate;
F. coolant conditions (inlet temperature, core average temperature, average exit and hot channel exit temperatures, and steam fractions);
G. steam line pressure;

H. containment pressures and temperatures;
I. maximum pressurizer water volume;
J. pressure safety and relief valve flow rates; and
K. flow rate from the reactor coolant system to the containment system (if applicable).
L. ECCS heat removal rate.

The reviewer provides a judgment as to whether the calculation results are within the expected range. If analyses have previously been published for similar plants, the more important parameters for the limiting transient are compared to predictions for those plants.

9. For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the DCD. The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

   For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

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

The staff concludes that the plant design is acceptable as to transients resulting in unplanned decreases in heat removal by the secondary system, transients expected with moderate frequency, and transients where the predicted response meets the requirements of GDCs 10, 13, 15, 17, and 26. This conclusion is based on the following findings:

1. The applicant meets the requirements of GDCs 10 and 26 by demonstrating that SAFDLs are not exceeded for this event. The applicant also meets GDC 15 requirements by preventing plant transients from resulting in unplanned decreases in heat removal by the secondary system and demonstrating reactor coolant pressure limits not exceeded by these events and resultant leakage within acceptable limits.

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

3. The transient initiating events that might occur with moderate frequency are:

   A. turbine trip,
   B. loss of external load,
   C. steam pressure regulator malfunctions,
   D. main steam isolation valve closure,
   E. loss of condenser vacuum,
   F. loss of nonemergency AC power to the station auxiliaries,
   G. loss of normal feedwater flow.[1]

4. In a review of the transients that could result from these postulated events, it was found that the most limiting in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the ____________ transient. This transient was evaluated by the applicant using a mathematical model that had been previously reviewed and found to be acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative and in accordance with the recommendation of RG 1.105. The results of the analysis of the transient showed that cladding integrity was maintained by ensuring that the minimum departure from nucleate boiling ratio did not decrease below _________ and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of their design pressures.

5. The applicant meets the requirements of GDCs 17 and 26 by demonstrating that SAFDLs are not exceeded for this event. In addition, the applicant meets GDC 15 requirements by demonstrating that the reactor coolant pressure limits are not exceeded by this event and that resultant leakage is within acceptable limits.

6. The applicant meets the positions of RG 1.53, SECY-77-439, SECY-94-084, SECY-95-132 and RG 1.206 on the single-failure criterion and RG 1.105 on instrument actuations of safety-related systems and components.

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 (Agencywide Documents Access and Management System Accession No. ML102510405), to develop risk-informed licensing

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[1] The SER should present one statement for moderate frequency transients involving unplanned decrease in heat removal by the secondary system; thus, the results of reviews under DSRS Sections 15.2.6 and 15.2.7 are included in this statement.

review plans for each of the small modular reactor reviews, including the associated pre-application activities, the staff has developed the content of this DSRS section as an alternative method for mPower™ -specific DC, or COL submitted pursuant to 10 CFR Part 52 to comply with 10 CFR 52.47(a)(9), “Contents of applications; technical information.”

This regulation states, in part, that the application must contain “an evaluation of the standard plant design against the 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 FSAR does not deviate significantly from the design assumptions made by the NRC staff while preparing this DSRS section. The application must identify and describe all differences between the standard plant design and this DSRS section, and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria. If the design assumptions in the DC application deviate significantly from the DSRS, the staff will use the SRP as specified in 10 CFR 52.47(a)(9). Alternatively, the staff may supplement the DSRS section by adding appropriate criteria in order to address new design assumptions. The same approach may be used to meet the requirements of 10 CFR 52.79(a)(41), and COL applications.

VI. REFERENCES

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

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

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

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

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

6. 10 CFR Part 50, Appendix A, GDC 26, "Reactivity Control System Redundancy and Capability."

7. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

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

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

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

11. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses."

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

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

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

15. SECY-95-132, “Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs."

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

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

18. 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).