U.S. NUCLEAR REGULATORY COMMISSION

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

15.2.7 LOSS OF NORMAL FEEDWATER FLOW (mPower^{TM} iPWR)

REVIEW RESPONSIBILITIES

Primary - Organization responsible for review of transient and accident analyses for iPWRs

Secondary - None

I. AREAS OF REVIEW

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a loss of offsite power (LOOP). Each event covered in this Design-Specific Review Standard (DSRS) section should be discussed in individual sections of the applicant's technical submittal, as specified in Regulatory Guide (RG) 1.70 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 ac power and assuming a single failure.

The specific areas of review are as follows:

1. The sequence of events described in the applicant's technical submittal is reviewed. The reviewer concentrates on the need for the reactor protection system, the ECCS, and operator action to secure and maintain the reactor in a safe condition.

2. The analytical methods are reviewed to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, a generic evaluation of the analytical model is performed.

3. The predicted results of the transient are reviewed to ascertain that the values of pertinent system parameters are within expected ranges for the type and class of reactor under review. Further, the predicted results of the transient are reviewed to ensure that the consequences meet the acceptance criteria given in subsection II, below.

4. 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

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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. 1.

3. Values of the parameters in the analytical models of the reactor core are reviewed for compliance with plant design and specified operating conditions, acceptance criteria for fuel cladding damage limits are determined, and the core physics, fuel design, and core thermal-hydraulics data in the safety analysis report (SAR) analysis are reviewed under DSRS Sections 4.2, 4.3, and 4.4.

4. Technical specifications are reviewed under DSRS Section 16.0.

5. Operational assumptions as factored into related instrumentation and controls for the ECCS used in the analysis are reviewed for appropriateness, under DSRS Sections 7.2 through 7.5.

   Instrumentation and controls aspects of the sequence described in the SAR are reviewed to confirm that 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, interlocks with auxiliary or shared systems and compliance with RG 1.105 under DSRS Sections 7.2 through 7.5.

6. The determination of the safety-related and risk significance of structures, systems, and components (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 it relates to the reactor coolant system being designed with appropriate margin to ensure that 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 it relates to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations including AOOs.

4. GDC 17, as it relates to providing onsite and offsite electric power systems to ensure that SSCs important to safety will function during normal operation, including anticipated operational occurrences. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded during an AOO.

5. GDC 26, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded, including AOOs. This is accomplished by assuring that appropriate margin for malfunctions, such as stuck rods, are accounted for.

<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 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. The basic objective in the review of the loss of normal feedwater transient is to confirm that the following criteria are met:

    A. The plant responds to the loss of feedwater transient in such a way that the criteria regarding fuel damage and system pressure are met.

    B. There is sufficient capacity for long term decay heat removal for the plant to reach a stabilized condition.

    C. The plant protection systems setpoints assumed in the transient analyses are selected with adequate allowance for measurement uncertainties as delineated in RG 1.105.

    D. The event evaluation takes into consideration single failures, operator errors, and performance of non-safety related systems that are consistent with regulatory guidelines set forth in RG 1.206.

2. Using the ANS standards as guidance, specific criteria have been developed to meet the relevant requirements of GDC 10, 13, 15, 17, and 26 for events of moderate frequency and they are as follows:

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

   B. Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit based on acceptable correlations (see SAR (or design control document (DCD)) Section 4.4), as well as by satisfaction of any other SAFDL that may be applicable to the particular reactor design.

   C. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

   D. To meet the requirements of GDCs 10 and 15, the positions of RG 1.105, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of transient addressed in this DSRS section.

   E. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of RG 1.53 and GDC 17.

   F. The guidance provided in SECY 77-439, SECY 94-084 and RG 1.206 with respect to the consideration of the performance of non-safety related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems) must be evaluated and verified.

3. The applicant's analysis of the loss of normal feedwater transient should be performed using an acceptable analytical model. If the applicant proposes to use analytical methods which have not been approved, these methods are evaluated by the staff for acceptability. The reviewer performs an evaluation of new generic methods.

   The value of parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model.

   A. The initial power level is taken as the licensed core thermal plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The operating conditions 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, i.e., the maximum time delay with the most reactive rod held out of the core, unless (a) a different conservatism factor can be justified through the uncertainty methodology and evaluation, or (b) the uncertainty has otherwise been accounted for (see SAR (or DCD) Section 4.4).

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

D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with RG 1.105.

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 15.2.7 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. Compliance with GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

   GDC 10 is applicable to DSRS Section 15.2.7 because this section evaluates the loss of normal feedwater flow transient. A part of the evaluation relates to the reactor coolant system being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs. RG 1.105 provides guidance for ensuring that instrument setpoints are initially within and remain within the technical specification limits.

   Meeting the requirements of GDC 10 provides assurance that specified acceptable fuel design limits are not exceeded for the initiating events evaluated in this DSRS section involving a decrease in 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 instrument indications.

3. Compliance with GDC 15 requires that the reactor coolant system and associated auxiliary, control and protection systems shall be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs.

   GDC 15 is applicable to DSRS Section 15.2.7 because this section evaluates the consequences of the events of a loss of normal feedwater flow transient that result in a

decrease in heat removal by the secondary system with the potential for causing the reactor coolant system pressure to change in response to the increase in reactor coolant temperature.

Meeting the requirements of GDC 15 provides assurance that the design conditions of the reactor coolant pressure boundary are not exceeded for the initiating events evaluated in this DSRS section involving a decrease in heat removal by the secondary system.

4. Compliance with GDC 17 requires (in part) that an onsite and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that SAFDLs and design conditions of the reactor coolant pressure boundary are not exceeded as a result of AOOs. GDC 17 is applicable to DSRS Section 15.2.7 because the loss of normal feedwater flow transient is an AOO.

   Meeting the requirements of GDC 17 provides assurance that SAFDLs and design conditions of the reactor coolant pressure boundary are not exceeded as a result of a loss of normal feedwater.

5. Compliance with GDC 26 requires that two independent reactivity control systems be provided capable of reliably controlling reactivity changes to ensure that acceptable fuel design limits are not exceeded.

   GDC 26 is applicable to DSRS Section 15.2.7 because this section evaluates the consequences of the events of a loss of normal feedwater flow that result in a decrease in heat removal by the secondary system with the potential for causing changes in reactivity within the core that could cause the thermal design criteria for the fuel cladding to be exceeded. DSRS 15.2.7 ensures that the thermal margin be sufficient to accommodate these conditions and ensures that the appropriate margins for malfunctions of reactivity controls such as stuck rods are accounted for.

   Meeting the requirements of GDC 26 provides assurance that SAFDLs are not exceeded by ensuring that there is appropriate margin for malfunctions of the reactivity control system.

III. REVIEW PROCEDURES

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

1. Programmatic Requirements - 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 (TS) (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. Inspections, tests, analyses, and acceptance criteria (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 SER section.

The procedures below are used for the design certification (DC) application review, the construction permit (CP), operating license (OL), and combined license (COL) applications. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The description of the loss of normal feedwater flow transient presented by the applicant in the SAR (or DCD) is reviewed regarding the occurrences leading to the initiating event.

The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

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

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

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

6. That appropriate margin for malfunctions, such as stuck rods is accounted for.

7. The operation of auxiliary systems that is required.

8. That instrumentation uncertainties of system and operating parameters are appropriately accounted for.

If the SAR (or DCD) states that the loss of feedwater transient is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. If a quantitative analysis of the loss of feedwater transient is presented in the SAR (or DCD), the reviewer assesses the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the loss of feedwater transient to an acceptable level. The reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The review of Chapter 7 of the SAR (or DCD) confirms that the reactor design is consistent with the requirements for safety systems actions for these events.

To the extent deemed necessary, the reviewer evaluates the effect of single active failures of systems and components which may alter the course of the transient. The LOOP should not be considered a single failure; loss of feedwater should be analyzed with and without a LOOP in combination with a single active failure. This part of the review uses the procedures described in the DSRS sections for Chapters 4, 5, 6, 7, 8, and 9 of the SAR (or DCD).

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam line are reviewed to determine if these models have been previously reviewed and found acceptable by the staff. If not, a generic review of the model proposed by the applicant is initiated.

The values of system parameters and initial core and system conditions used as input to the model are reviewed. Of particular importance are the reactivity feedbacks and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void, and Doppler reactivity feedback with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated.

The results of the analysis are reviewed, including the effects of the LOOP and the possibility of the event developing into a more serious event (e.g., a stuck open power operated relief valves on the pressurizer that could lead to a small-break loss-of-coolant accidents if not isolated), and

compared with the acceptance criteria presented in Subsection II of this DSRS section regarding maximum pressure in the reactor coolant and main steam systems. The parameters reviewed are:

1. variations with time during the transient of the neutron power,

2. heat fluxes (average and maximum),

3. reactor coolant system pressure,

4. minimum DNBR,

5. core coolant flow rate,

6. decay heat removal rate,

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

8. steamline pressure,

9. containment pressure and temperature,

10. pressure relief valve flow rate and quality,

11. pressurizer water volume, and

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

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.

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. <u>EVALUATION FINDINGS</u>

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.

The staff has reviewed the analyses of the loss of normal feedwater flow event and concludes that the analyses have adequately accounted for the operation of the plant and were performed using acceptable analytical models.

The staff further concludes that the analyses have demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the reactor coolant pressure boundary pressure limits will not be exceeded as a result of the loss of normal feedwater flow.

The staff concludes that the plant design is acceptable with regard to transients resulting from loss of normal feedwater that are expected to occur with moderate frequency and that the predicted response meets the requirements of GDC 10, 13, 15, 17, and 26. This conclusion is based on the following:

The applicant has met the requirements of GDC 10 and 26 with respect to demonstrating that SAFDLs are not exceeded for this event and has met the requirements of GDC 15 with respect to demonstrating that the reactor coolant pressure limits have not been exceeded by this event.

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. The parameters used as input to the analytical 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.

Thus, the applicant has met the requirements of GDC 17, and 26 with respect to demonstrating that SAFDLs are not exceeded for this event.

The applicant has met the positions of RG 1.53, SECY 77-439, SECY 94-084 and RG 1.206 as related to the single-failure criterion and RG 1.105 as related to instrument actuations of systems and components important to safety.

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

This regulation states, in part, that the application must contain “an evaluation of the standard plant design against the SRP revision in effect 6 months before the docket date of the application.” The content of this DSRS section has been accepted as an alternative method for complying with 10 CFR 52.47(a)(9) as long as the mPower™ DCD FSAR does not deviate significantly from the design assumptions made by the NRC staff while preparing this DSRS section. The application must identify and describe all differences between the standard plant design and this DSRS section, and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria. If the design assumptions in the DC application deviate significantly from the DSRS, the staff will use the SRP as specified in 10 CFR 52.47(a)(9). Alternatively, the staff may supplement the DSRS section by adding appropriate criteria in order to address new design assumptions. The same approach may be used to meet the requirements of 10 CFR 52.79(a)(41), and COL applications.

VI. REFERENCES

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

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

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

4. 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities."

5. 10 CFR 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

6. 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).

7. GDC 10, "Reactor Design."

8. GDC 13, “Instrumentation and Control.”

9. GDC 15, "Reactor Coolant System Design."

10. GDC 17, "Electric Power Systems."

11. GDC 26, "Reactivity Control System Redundancy and Capability."

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

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

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

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

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."