U.S. NUCLEAR REGULATORY COMMISSION

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

15.2.6 LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES (mPower^{TM} iPWR)

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

The loss of nonemergency alternating current (AC) power is assumed to result in the loss of all power to the station auxiliaries. This situation, which could be the result of a complete loss of either the external (offsite) grid or the onsite AC distribution system, is different from the loss of load condition considered in Design-Specific Review Standard (DSRS) Section 15.2.1-15.2.5 because, in the latter case, AC power remains available to operate the station auxiliaries. The major difference is that in the loss-of-ac-power transient, all the reactor coolant circulation pumps are tripped simultaneously by the initiating event, resulting in a flow coast-down as well as a decrease in heat removal by the secondary system.

The mPower^{TM} emergency core coding systems (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.

Within a few seconds of the loss of nonemergency AC power, the turbine trips, the reactor coolant system is isolated, and the pressure and temperature of the coolant increase. A reactor trip is initiated. The ECCS is actuated automatically and decay heat is transferred to the ultimate heat sink.

The loss of AC power has the following effects: (i) immediate load rejection with fast closure of the turbine control valves, (ii) loss of power to the condensate and feedwater pumps, resulting in loss of feedwater, and (iii) the reactor is isolated after loss of main condenser vacuum. Therefore, the review of the loss-of-ac-power transient includes the sequence of events, the analytical model, the values of parameters in the analytical model, and the predicted consequences of the transient. The specific areas of review are as follows:

1. The sequence of events described in the applicant's safety analysis report (SAR) is reviewed with concentration on the need for the reactor protection system, the engineered safety systems, 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 reviewed previously and accepted by the staff. If a

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referenced analytical method has not been reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by appropriate reviewers.

3. The predicted results of the transient analysis are reviewed for whether the consequences meet the acceptance criteria of subsection II of this DSRS session. The results of the analysis are reviewed for whether the values of pertinent system parameters are within expected ranges for the type and class of reactor under review.

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 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 sequence described in the SAR are reviewed for whether the reactor and plant protection and safeguards controls and instrumentation systems function as assumed in the safety analysis for automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems under DSRS Chapter 7.

4. Technical specifications are reviewed under DSRS Section 16.0.

5. 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 SAR analysis are reviewed under DSRS Sections 4.2, 4.3, and 4.4.

6. The determination of the safety-related and risk significance of SSCs relied upon to meet required functions during the accidents are based on the review of the probabilistic risk analysis under Standard Review Plant (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 (RCS) design with appropriate margin so specified acceptable fuel design limits are not exceeded during normal operation including anticipated operational occurrences.

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 RCS and its auxiliaries with appropriate margin so the pressure boundary is not breached during normal operation including anticipated operational occurrences.

4. GDC 26, as to reliable control of reactivity changes so specified acceptable fuel design limits are not exceeded in anticipated operational occurrences. This control is accomplished by appropriate margin for malfunctions like stuck rods.

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the U.S. Nuclear Regulatory Commission's (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.

Specific criteria necessary to meet the relevant requirements of GDCs 10, 13, 15, and 26 for events of moderate frequency (see definitions of design and plant process conditions in are as follows:

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

2. Fuel cladding integrity should be maintained by keeping the minimum departure from nucleate boiling ratio (DNBR) above the 95/95 DNBR limit based on acceptable correlations (see DSRS Section 4.4).

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

4. For the requirements of GDCs 10 and 15, the positions of Regulatory Guide (RG) 1.105, "Instrument Setpoints for Safety-Related Systems," have impact on the plant response to the type of transient addressed in this DSRS section.

5. The most limiting plant system single failure, as defined in the "Definitions and Explanations" of 10 CFR Part 50, Appendix A, must be assumed in the analysis and must satisfy the positions of RG 1.53.

   The applicant's analysis of the loss of AC power transient should be based on an acceptable and NRC-approved model. If the applicant proposes analytical methods not approved, these are evaluated by the staff for acceptability and approval. For new generic methods, the reviewer requests an appropriate evaluation.

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

   A. The initial power level is taken as the licensed core thermal power plus an allowance of 2 percent to account for power measurement uncertainties unless the applicant can justify a lower power level.

   B. Conservative scram characteristics are assumed (i.e., for the maximum time delay with the most reactive rod held out of the core).

   C. The core burn-up 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. Compliance with RG 1.105 is determined.

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 15.2.6 must be consistent with the information presented in the licensee's TS.

<u>Technical Rationale</u>

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 that the reactor core and its coolant, control, and protection systems be designed with appropriate margin so specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

   GDC 10 applies to this section because the reviewer evaluates the consequences of the loss of nonemergency AC power to the station auxiliaries. This anticipated operational occurrence creates a potential for specified acceptable fuel design limits to be exceeded. Within seconds after the loss of power, the turbine and the reactor both trip, and the pressure and temperature of the reactor coolant increase. RG 1.53 provides guidance for application of the single-failure criterion to the design and analysis of nuclear power plant protection systems. RG 1.105 describes a method acceptable to the staff for keeping instrument setpoints within the technical specification limits.

GDC 10 provides meeting the requirements of assurance that specified acceptable fuel design limits are exceeded and that fuel cladding integrity is maintained in loss of nonemergency AC power to the station auxiliaries.

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 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. GDC 15 requires that the RCS and its auxiliary, control, and protection systems be designed with sufficient margin so design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. GDC 15 applies to this section because the reviewer evaluates the consequences of the loss of nonemergency AC power to the station auxiliaries. This transient is an anticipated operational occurrence, and the reactor coolant pressure must be analyzed to confirm whether the pressure acceptance criterion is satisfied.

   GDC 15 requirements provide assurance that the design conditions of the reactor coolant pressure boundary are not exceeded in the loss of nonemergency AC power to the station auxiliaries.

4. GDC 26 requires that one of the reactivity control systems consist of control rods capable of reliably controlling reactivity changes with appropriate margin for malfunctions like stuck rods so that specified acceptable fuel design limits are not exceeded under conditions of normal operation, including anticipated operational occurrences.

   GDC 26 applies because the transient analyzed in this section involves the movement of control rods in response to the loss of nonemergency AC power and because rod misalignment, including stuck rods, can produce thermal-hydraulic conditions more severe than otherwise. GDC 26 requires a thermal margin sufficient to accommodate these conditions. Under DSRS Section 15.2.6 these margins are examined where applicable for whether the thermal criteria remain satisfied.

   GDC 26 requirements provide assurance by appropriate margin for malfunctions of the reactivity control system, including stuck rods, that specified acceptable fuel design limits 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 for the review of DC application review and COL applications.

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 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, Regulatory Guides 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. 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. 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 (TMI) 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 description of the loss of AC power transient presented by the applicant in the SAR is reviewed from transient initiation until condition stabilization 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 credit taken for the functioning of normally operating plant systems.

   D. The operation of engineered safety systems required.

   E. The extent to which operator actions are required.

   F. The operation of standby diesel generators required.

   G. Whether the description accounts for appropriate margin for malfunctions like stuck rods (per subsection II.B of this DSRS section).

4. If the SAR states that the loss of AC power transient is not as limiting as some other similar transient, the reviewer evaluates the applicant's justification. If the applicant's technical submittal presents a quantitative analysis of the loss of AC power transient, the timing of the initiation of those protection, engineered safety, and other systems needed to limit transient consequences acceptably level is reviewed. The reviewer compares the predicted variation of system parameters to various trip and system initiation setpoints. The review of SAR Chapter 7 confirms whether the instrumentation and control systems design is consistent with the requirements for safety system actions for these events. To the extent necessary, the reviewer evaluates the effects of single, active system and component failures which may affect the course of the transient. This aspect of the review uses the procedures described in DSRS sections for SAR Chapters 4, 5, 6, 7, 8, and 9.

5. The applicant's mathematical models for evaluating core performance and predicting system pressure in the RCS and main steam lines are reviewed by 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. System parameter values and initial core and system conditions as input to the model are reviewed. Of particular importance are the reactivity and control rod worths in the applicant's analysis and the variations of moderator temperature, void and Doppler reactivity feedback with core life. The applicant's justification to show that it has selected the core burn-up that yields minimum margins is evaluated. The values of the reactivity parameters in the applicant's analysis are reviewed.

7. The results of the analysis are reviewed and compared to the acceptance criteria of subsection II of this DSRS section as to the maximum pressure in the reactor coolant and main steam systems. The variations during the transient of neutron power, heat fluxes (average and maximum), RCS pressure, minimum DNBR, coolant conditions (inlet temperature, core average temperature, average exit and hot channel exit temperatures), steam line pressure, containment pressure, pressure relief valve flow

rate, and flow rate from the RCS to the containment system (if applicable) are reviewed. Time-related variations of the following parameters are reviewed:

A. reactor power;

B. heat fluxes (average and maximum);

C. RCS pressure;

D. minimum DNBR;

E. core flow rates;

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

G. steam line pressure;

H. containment pressure;

I. pressure relief valve flow rate;

J. flow rate from the RCS to the containment system (if applicable); and

K. decay heat removal rate.

The reviewer provides a judgment as to whether the calculation results are within the expected range.

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 final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

   For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit 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 The staff concludes that the plant design as to transients expected to occur with moderate frequency and to result in the loss of all power to the station auxiliaries is acceptable and meets the relevant requirements of GDCs 10, 13, 15, and 26 and the applicable TMI Action Plan items.

This conclusion is based on the following findings:

1. The applicant meets the requirements of GDCs 10 and 26 by demonstrating that resultant fuel integrity is maintained because the specified acceptable fuel design limits were not exceeded for the event.

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 meet GDC 15 requirements by demonstrating that the reactor coolant pressure boundary limits were not exceeded by this event and that resultant leakage is within acceptable limits. This requirement is met because the maximum pressure within the reactor coolant and main steam systems did 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 specified acceptable fuel design limits were not exceeded.

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 six 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. GDC 10, "Reactor Design."

2. GDC 13, “Instrumentation and Control.”

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

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

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

6. RG 1.105, "Instrument Spans and Setpoints for Safety-Related Systems."

7. NUREG-0660, "NRC Action Plans Developed as a Result of the TMI-2 Accident,” August 1980.

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