U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

15.4.1 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER STARTUP CONDITION

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of transient and accident analyses for iPWR.

Secondary - None

I. AREAS OF REVIEW

A malfunction of the control rod drive system may cause an uncontrolled withdrawal of an individual control rod drive mechanism (CRDM) or a bank of CRDMs from the reactor under subcritical or low-power startup conditions. During the inadvertent withdrawal of the individual CRDM or bank of CRDMs and the subsequent rapid insertion of positive reactivity, the reactor power level and fuel and coolant temperatures may rapidly increase as a result of the mismatch between power generation and heat removal. Depending on the magnitude of the positive reactivity insertion and the resulting response of the reactor systems, the resulting power surge may lead to overheating of the fuel with resulting fuel damage and radiological release.

Reactor trips, including the high neutron flux trip, overpower and over temperature delta T trips, and pressurizer high-pressure and pressurizer water-level trips, provide plant protection. The increase in fuel temperature would result in an insertion of negative reactivity, which would counter the insertion from the withdrawal event. The reactor becomes subcritical, fuel rod heat fluxes decrease and temperatures stabilize as the reactor transitions to decay heat power level and decay heat removal. The objective of the analysis is to demonstrate that the margins to the departure from nucleate boiling are adequate and that acceptable fuel design limits are not exceeded.

The specific areas of review are as follows:

1. The NRC reviewer evaluates the effects and consequences of an uncontrolled control rod assembly bank from a subcritical or low-power (e.g., startup-range) condition to assure conformance with the requirements of General Design Criteria (GDC) 10, 17, 20, and 25 under this design-specific review standard (DSRS) section. The review under this DSRS section covers the description of the causes of the transient and the transient itself, the initial conditions, the reactor parameters used in the analysis, the analytical methods and computer codes used, and the consequences of the transient as compared with the acceptance criteria.

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2. Combined 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. Reactivity feedback parameters and control rod worths are reviewed under DSRS Section 4.3.

II. ACCEPTANCE CRITERIA

Requirements

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

The following GDC apply:

1. GDC 10, which requires that specified acceptable fuel design limits (SAFDLs) are not to be exceeded during normal operation, including the effects of anticipated operational occurrences (AOOs).

2. GDC 13, which requires that 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 17, which requires provision of an onsite electric power system and an offsite electric power system to permit functioning of structures, systems, and components important to safety.

4. GDC 20, which requires that the protection system initiate automatically appropriate systems to assure that SAFDLs are not exceeded as a result of AOOs.

5. GDC 25, which requires that the reactor protection system be designed to assure that SAFDLs are not exceeded in the event of a single malfunction of the reactivity control system.

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 Title 10, *Code of Federal Regulations* (10 CFR) 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.

The requirements of GDC 10, 20, and 25 concerning the SAFDLs are assumed to be met for this event when:

1. The departure from nucleate boiling (DNBR) thermal margin limits as specified in DSRS Section 4.4 are met.

2. Fuel centerline temperatures as specified in DSRS Section 4.2 do not exceed the melting point.

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 that the reactor core and associated coolant, control, and protection systems be designed such that SAFDLs are not exceeded during normal operation, including the effects of AOOs. Control rod withdrawal is an AOO. The fuel cladding is the first barrier of protection against radioactive release. Meeting GDC 10 ensures that the fuel cladding integrity is not challenged during this AOO.

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 justified, based upon the expected values of the relevant monitored parameters and instrument indications.

3. GDC 17 requires that an onsite electric power system and an offsite electric power system 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. This section reviews an AOO. Meeting GDC 17 ensures that the fuel cladding integrity is not challenged during an uncontrolled control rod assembly withdrawal in conjunction with a loss of onsite or of offsite power.

4. GDC 20 requires that the protective system automatically initiate the operation of the reactivity control system to ensure that fuel design limits are not exceeded as a result of AOOs. The withdrawal of a control assembly significantly impacts local fuel pin power, and could lead to cladding failure. Measures are required to ensure that an abnormal rod withdrawal is detected and automatically terminated prior to fuel design safety limits being violated. Meeting GDC 20 ensures that cladding integrity is not challenged during this AOO.

5. GDC 25 requires that the reactor protection system be designed to ensure that SAFDLs are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal of control rods. A failure of the reactivity control system that would create an unmitigated withdrawal of a control assembly could lead to cladding failure. Meeting GDC 25 ensures that a power transient fostered from a reactivity addition due to a single failure of the reactivity control system will be detected and terminated prior to a challenge of the fuel cladding integrity.

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.

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 Standard Review Plan (SRP) Section 17.6 (DSRS Section 13.4, Table 13.4, Item 17, Regulatory Guides (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. 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 six 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. Peak conditions for the transient are maximized by low initial power; thus, the power level of the reactor should be at the lowest possible value compatible with the control rod configuration used for the accident. The postulated initial reactor coolant flow, pressure and inlet temperature (i.e., the extremes of postulated conditions) should be consistent with the rod and power configuration to give minimum DNBR.

4. Peak conditions for the transient are maximized by large reactivity addition rates near prompt critical; thus, the control rod configurations for the assumed withdrawal must be examined to confirm that such a maximized state has been included in the calculations.

5. The exact analysis of the transient would ideally involve a three dimensional, coupled neutron kinetics-thermal hydraulics calculation. However, acceptable results may be obtained with a neutron point-kinetics analysis and a coupled or separate hot fuel rod thermal analysis, if conservative input data are used. The reviewer determines whether the applicant's analytical methods are acceptable by using one or more of the following procedures:

A. Determine whether the method has been reviewed and approved previously, by considering past safety evaluation reports and reports prepared in response to technical assistance requests.

B. Perform a de novo review of the method (usually described in a separate licensing topical report, and frequently handled outside the scope of the review for a particular facility).

C. Perform auditing-type calculations with methods available to the staff.

D. Require additional, bounding calculations by the applicant to cover portions of the applicant's analytical methods that have not been fully reviewed or approved.

6. The input to the neutron kinetics analysis model should be examined to assure that the input is appropriately conservative both for the state of the reactor and for the particular way it is used in the analysis. The power distribution or peaking factors used in the neutron kinetics and hot pin thermal calculations must provide a conservative representation of the control rod configuration under consideration. The Doppler feedback parameters should be related conservatively to the values accepted in the review under DSRS Section 4.3, considering the time in cycle and temperature conditions of the fuel. Non-weighting of the parameters is conservative, but weighting factors for the particular flux distribution shapes involved in the transients may be used if fully explored and justified. The moderator reactivity parameters used should also be conservatively related to the values accepted in the review under DSRS Section 4.3. The parameters that lead to most positive or least negative reactivity values should be used and for a pressurized-water reactor this occurs at beginning of life. If the reactivity feedback is negative, it may be conservatively taken as zero.

7. The analysis should consider the relationships between the particular spatial flux shapes for the transient and the nuclear instrument response to assure that scrams occur at the times used in the analysis, that valid scram power levels are assumed, and that conservative scram delays and reactivity functions are used.

8. The significant results of the analysis should be presented and should include maximum power levels reached for the reactor and the peak fuel rod, reactor temperatures and pressures, maximum heat flux levels, and the related fuel duty (operating conditions and performance). The latter are compared with the acceptance criteria in subsection II of this DSRS.

9. The analysis must consider a loss of offsite power in conjunction with the limiting single active failure when assessing the consequences of the AOO.

10. 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 submittal meets the acceptance criteria. . 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 DC application.

    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.

1. The possibilities for single failures of the reactor control system which could result in uncontrolled withdrawal of control rods under low power startup conditions have been

reviewed. The scope of the review has included investigations of initial conditions and control rod reactivity worth, the course of the resulting transients or steady-state conditions, and the instrument response to the transient or power maldistribution. The methods used to determine the peak fuel rod response, and the input into that analysis, such as power distributions and reactivity feedback effects due to moderator and fuel temperature changes, have been examined. If audit calculations have been done, they should be summarized.

2. The staff concludes that the requirements of GDC 10, 13, 17, 20, and 25 have been met. This conclusion is based on the following:

   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 applicant has met the requirements of GDC 10 that the SAFDLs are not exceeded, GDC 20 that the reactivity control systems are automatically initiated so that SAFDLs are not exceeded, and GDC 25 that single malfunctions in the reactivity control system will not cause the SAFDLs to be exceeded with and without offsite electrical power availability in accordance with the requirements of GDC 17.

   These requirements have been met by comparing the resulting extreme operating conditions and response for the fuel (i.e., fuel duty) with the acceptance criteria for fuel damage (e.g., critical heat flux, fuel temperatures, and clad strain limits should not be exceeded), to assure that fuel rod failure will be precluded for this event. The basis for acceptance in the staff review is that the applicant's analyses of the maximum AOOs for single error control rod withdrawal from a subcritical or low-power condition have been confirmed, that the analytical methods and input data are reasonably conservative and that SAFDLs will not be 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 (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 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 Part 50, Appendix A, GDC 10, ‘Reactor design.

2. 10 CFR Part 50, Appendix A, GDC 13,”Instrumentation and Control.”

3. 10 CFR Part 50, Appendix A, GDC 17, "Electric power systems."

4. 10 CFR Part 50, Appendix A, GDC 20, "Protection system functions."

5. 10 CFR Part 50, Appendix A, GDC 25, "Protection system requirements for reactivity control malfunctions."