U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

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

REVIEW RESPONSIBILITIES:

Primary - Organization responsible for review of transient and accident analyses for PWRs/BWRs

Secondary - None

I. AREAS OF REVIEW

The specific areas of review are as follows:

1. Reactor Systems reviewer evaluates the effects and consequences of an uncontrolled control rod assembly withdrawal (a bank for a pressurized water reactor; and a single rod, with current control modes, for a boiling water reactor) from a subcritical or low-power (e.g., startup-range) condition to assure conformance with the requirements of General Design Criteria 10, 17, 20, and 25 under this SRP section. The review under this SRP 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. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

   For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

Other SRP sections interface with this section as follows:

1. General information on transient and accident analyses is provided in SRP Section 15.0.

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

3. Reactivity coefficients and control rod worths are reviewed under SRP Section 4.3

The specific acceptance criteria and review procedures are contained in the referenced SRP sections.

II. ACCEPTANCE CRITERIA

Requirements

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

1. The following General Design Criteria apply:

   A. Criterion 10, which requires that specified acceptable fuel design limits are not to be exceeded during normal operation, including the effects of anticipated operational occurrences.

   B. Criterion 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.

   C. Criterion 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.

   D. Criterion 20, which requires that the protection system initiate automatically appropriate systems to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences.

E. Criterion 25, which requires that the reactor protection system be designed to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control system.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for the review described in this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

1. The requirements of GDC 10, 20, and 25 concerning the specified acceptable fuel design limits are assumed to be met for this event when:

    A. The thermal margin limits (DNBR for PWRs and MCPR for BWRs) as specified in SRP Section 4.4 are met.

    B. Fuel centerline temperatures (for PWRs) as specified in SRP Section 4.2 do not exceed the melting point.

    C. Uniform cladding strain (for BWRs) as specified in SRP Section 4.2 does not exceed 1%.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this SRP 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 specified acceptable fuel design limits are not exceeded during normal operation, including the effects of anticipated operational occurrences. Control rod withdrawal is an anticipated operational occurrence. 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 anticipated operational occurrence.

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 specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences. This section reviews an anticipated operational occurrence. 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 anticipated operational occurrences. 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 anticipated operational occurrence.

5. GDC 25 requires that the reactor protection system be designed to ensure that specified acceptable fuel design limits 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

The reviewer will select material from the procedures described below, as may be appropriate for a particular case.

These review procedures are based on the identified SRP 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. 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, or CPR conditions.

2. 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. For a PWR, control bank withdrawal should be used. For a BWR, with the present control rod withdrawal procedures, a single rod of maximum worth available in a normal configuration should be used. In many cases this will be a rod near the 50% rod density configuration. (More recent modes of BWR control such as group withdrawal may require that other configurations be examined.)

3. 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.

4. 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 coefficient should be related conservatively to the values accepted in the review under SRP Section 4.3, considering the time in cycle and temperature conditions of the fuel. Non-weighting of the coefficients 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 coefficients used should also be conservatively related to the values accepted in the review under SRP Section 4.3. The most positive or least negative values should be used and for a PWR this occurs at beginning of life. If the coefficient is negative, it may be conservatively taken as zero.

5. 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.

6. 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 SRP.

7. For new application reviews, the analysis must consider a loss of offsite power in conjunction with the limiting single active failure when assessing the consequences of the anticipated operational occurrence. (This position is based upon interpretation of

GDC 17, as documented in the Final Safety Evaluation Report for the ABB-CE System 80+ design certification.)

8. For boiling water reactor applicants, the evaluation should also include the effects and consequences of a control rod removal error during refueling operations.

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 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 (ESP) 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 review and calculations (if applicable) 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 General Design Criteria 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 specified acceptable fuel design limits are not exceeded, GDC 20 that the reactivity control systems are automatically initiated so that specified acceptable fuel design limits are not exceeded, and GDC 25 that single malfunctions in the reactivity control system will not cause the specified acceptable fuel design limits 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 transients 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 specified acceptable fuel design limits will not be exceeded.

For BWR/6 Designs

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

1. 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.

2. The applicant has met the requirements of GDC 10 that the specified acceptable fuel design limits are not exceeded, GDC 20 that the reactivity control systems are automatically initiated so that specified acceptable fuel design limits are not exceeded, and GDC 25 that single malfunctions in the reactivity control system will not cause the specified acceptable fuel design limits to be exceeded with and without offsite electrical power availability in accordance with the requirements of GDC 17. These requirements have been met since the system design contains a Rod Pattern Control System. This system has been reviewed and found acceptable because single failures in the reactor control system which could result in uncontrolled withdrawal of control rods under low-power conditions have been precluded. The scope of the review has included the design features, eighteen of which act to prevent such withdrawals. This review has shown that no single failure will permit an uncontrolled rod withdrawal that could lead to reactivity insertions greater than those routinely encountered during operation.

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 SRP section.

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications submitted six months or more after the date of issuance of this SRP section, unless superseded by a later revision.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor design.

2. 10 CFR Part 50, Appendix A, General Design Criterion 13, “Instrumentation and Control.”

3. 10 CFR Part 50, Appendix A, General Design Criterion 17, "Electric power systems."

4. 10 CFR Part 50, Appendix A, General Design Criterion 20, "Protection system functions."

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