NUREG-0800

U.S. NUCLEAR REGULATORY COMMISSION

# STANDARD REVIEW PLAN

## 15.4.2 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL AT POWER

**REVIEW RESPONSIBILITIES**

**Primary -** Organization responsible for the review of reactor systems

**Secondary -** None

I. AREAS OF REVIEW

The specific areas of reviews are as follows:

1. The organization responsible for reactor systems evaluates the effects and consequences of an uncontrolled control rod assembly withdrawal (a bank for a pressurized water reactor (PWR); a single rod with current control modes for a boiling water reactor (BWR)) at power to ensure conformance with the requirements of General Design Criteria 10, 17, 20, and 25 under this Standard Review Plan (SRP) section. The review under this SRP section covers the description of the causes of the anticipated operational occurrence (AOO) and of the AOO itself, the initial conditions, the reactor parameters used in the analysis, the analytical methods and computer codes used, and the consequences of the AOOs as compared with the acceptance criteria.

2. COL action items may be identified in the NRC staff's final safety evaluation report (FSER) for each certified design to identify information that COL applicants must address in the application. Additionally, DCs contain requirements and restrictions (e.g., interface requirements) that COL applicants must address in the application. For COL applications referencing a DC, the review performed under this SRP section considers

information provided in response to COL action items and certification requirements and restrictions pertaining to this SRP section, as identified in the FSER for the referenced certified design.

Review Interfaces

The listed SRP sections interface with this section as follows:

1. Fuel centerline temperatures (for PWRs) are reviewed by the organizations responsible for reactor systems under SRP Section 4.2, subsections II.A.2(a) and (b).

2. Uniform cladding strain (for BWRs) are reviewed by the organization responsible for reactor systems under SRP Section 4.2, subsection II.A.2(b).

3. The reactivity coefficients and control rod assembly worths involved are reviewed by the organization responsible for reactor systems under SRP Section 4.3.

4. The thermal margin limits (departure from nucleate boiling ratio (DNBR) for PWRs and minimum critical power ratio (MCPR) for BWRs) are reviewed by the organization responsible for reactor systems under SRP Section 4.4, subsection II.1.

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. General Design Criterion 10 (GDC 10), which requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not to be exceeded during any condition of normal operation, including the effects of AOOs.

2. General Design Criterion 17 (GDC 17), which requires, in part, 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.

3. General Design Criterion 20 (GDC 20), which requires, in part, that the protection system shall be designed to initiate automatically the operation of appropriate systems to ensure that specified acceptable fuel design limits are not exceeded as a result of AOOs.

4. General Design Criterion 25 (GDC 25), which requires that the reactor protection system be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for review described in Subsection I of 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 General Design Criteria 10, 17, 20, and 25 concerning the specified acceptable fuel design limits are assumed to be met for this event when:

   A. The thermal margin limits departure from nucleate boiling ratio for PWRs and maximum critical power ratio for BWRs as specified in SRP Section 4.4, subsection II.1, are met.

   B. Fuel centerline temperatures (for PWRs) as specified in SRP Section 4.2, subsection II.A.2(a) and (b), do not exceed the melting point.

   C. Uniform cladding strain (for BWRs) as specified in SRP Section 4.2, subsection II.A.2(b), does not exceed 1%.

<u>Technical Rationale</u>

The technical rationale for application of these regulatory requirements to review under this SRP 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 this section because the reviewer evaluates the effects and consequences of an uncontrolled control rod assembly withdrawal (a bank for a PWR; a single rod with current control modes for a BWR) at power to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of AOOs. SRP Section 15.4.2 as well as SRP Sections 4.2, 4.3, and 4.4 provide guidance for ensuring that instrument setpoints are initially within and remain within the technical specification limits.

   Meeting the requirements of GDC 10 provides reasonable assurance that specified acceptable fuel design limits are not exceeded for AOOs caused by an uncontrolled control rod assembly withdrawal at power.

2. Compliance with GDC 17 requires (in part) that an onsite 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 (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of AOOs and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

   GDC 17 is applicable to SRP Section 15.4.2 because this section reviews an anticipated operational occurrence to which the GDC is applicable.

Meeting the requirements of GDC 17 provides reasonable assurance that an uncontrolled control rod assembly withdrawal at power, in combination with a LOOP, will not result in a reactor transient that could cause the reactor coolant pressure boundary design conditions or the fuel design limits to be exceeded.

3. Compliance with GDC 20 requires that the protection system be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

   GDC 20 is applicable to this section because the reviewer evaluates the effects and consequences of an uncontrolled control rod assembly withdrawal at power (i.e., a bank for a PWR; a single rod with current control modes for a BWR). The reactor protection system (RPS) automatically initiates the operation of appropriate systems, including the reactivity control system (RCS), to terminate the AOOs analyzed in this SRP section. The AOOs are terminated in a timely manner so that acceptable specified fuel design limits are not exceeded for either a PWR or a BWR. SRP Section 15.4.2 as well as SRP Sections 4.2, 4.3, and 4.4 provide guidance for ensuring that specified acceptable fuel design limits are not exceeded as a result of AOOs.

   Meeting the requirements of GDC 20 provides reasonable assurance that specified acceptable fuel design limits are not exceeded by ensuring that the RPS initiates the operation of appropriate systems in a timely manner to terminate AOOs caused by an uncontrolled control rod assembly withdrawal at power.

4. Compliance with GDC 25 requires that the protection system be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

   GDC 25 is applicable to this section because, as discussed in Section II above, the reviewer evaluates the effects and consequences of an uncontrolled control rod assembly withdrawal at power (i.e., a bank for a PWR; a single rod with current control modes for a BWR). One criterion specifies that the RPS be designed to ensure that acceptable fuel design limits are not exceeded for either a PWR or a BWR during normal operation or AOOs, including in the event of a single malfunction of the RCS. The RPS operates in a timely manner to initiate automatic termination of the AOOs analyzed in this SRP section. SRP Section 15.4.2 as well as SRP Sections 4.2, 4.3 and 4.4 provide guidance for ensuring that specified acceptable fuel design limits are not exceeded as a result of operation or AOOs.

   Meeting the requirements of GDC 25 provides reasonable assurance that a single malfunction of the reactivity control system, together with AOOs caused by the initiating event of an uncontrolled control rod assembly withdrawal at power, will not cause specified acceptable fuel design limits to be exceeded.

## III. <u>REVIEW PROCEDURES</u>

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

For each area of review specified in subsection I of this SRP section, the review procedure is identified below. These review procedures are based on the identified SRP acceptance criteria.

For deviations from these specific acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the relevant NRC requirements identified in subsection II.

1. The review process and the areas examined differ somewhat, depending on whether a BWR or PWR is being reviewed. For both systems, the review considers the entire power range from low to full power and the allowed extreme range of reactor conditions during the operating (fuel) cycle, including rod configurations, power distribution, and associated reactivity feedback components. The continuous withdrawal of normal configurations of rods should be assumed for the initial conditions in the AOO calculation. For a PWR, this is one or two control banks; for a BWR with current modes of control, it is a single control rod (future addresses under consideration may change this to group movement). The review considers a full range of rod or bank withdrawals, up to maximum rod or bank worths and rates of reactivity addition.

   The exact analysis of the AOO would ideally involve a three-dimensional, coupled neutron kinetics, thermal-hydraulics calculation. However, acceptable results may be obtained with suitable approximate calculations. The problem examined and the approximations used differ for a PWR and a BWR.

2. For a BWR, past analyses and reviews have shown that at maximum rod worths and rates of reactivity addition, the reactor power increases slowly and the total increase is relatively small, so that the AOO may be approximated by steady-state analyses. Because of changes in local power distribution attributable to rod motion and strong void feedback effects on the power distribution, three-dimensional, steady-state, coupled neutron distribution, thermal-hydraulics calculations that take account of these effects are required. The AOO is halted by action of a rod block system, which should block rod withdrawal before fuel safety limits are reached.

   The review process for a BWR, discussed above, is concentrated on the steady-state aspects of the AOO to ensure that initial and subsequent power distributions are maximized, that the reactor conditions produce MCPR and that the response of the rod block system is conservatively calculated considering minimum operation of the associated local power range monitoring system.

3. A PWR analysis, on the other hand, generally involves larger power changes and requires AOO calculations. Because power distributions in the course of the AOO can frequently be predicted conservatively using design-limit peaking factors, point kinetics may be used for the nuclear AOO. The nuclear AOO is coupled, however, to core and system thermal-hydraulic response to the power changes (fuel and moderator thermal feedback and system instrumentation response).

   For a PWR, the reviewer ascertains that a full range of AOO conditions are analyzed; the AOO calculation models are adequate; and that scram response of the flux, temperature, or pressure instrumentation is correctly calculated. The range of parameters to be considered includes:

   A. Initial power levels from low to full power.

   B. Reactivity insertion rates from very low to maximum possible for the control system, including allowance for uncertainties.

   C. Fuel and moderator feedback reactivity coefficients covering the range expected throughout the cycle, including allowance for uncertainties.

D. Power peaking factors at design limits for the initial power level conditions.

4. For both types of reactors, the reviewer determines whether the applicant's analytical methods and models are acceptable, including steady-state, AOO, system response, and fuel response models. This may be done 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 (SERs) and reports prepared in response to technical assistance requests.

   B. Perform an independent review of the method (usually described in a separate licensing topical report and frequently completed, on a generic basis, outside the scope of the review for a particular facility).

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

   D. Request additional bounding calculations from the applicant to confirm the validity of those portions of the applicant's analytical methods that are not fully reviewed or approved.

5. 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, NUREG-1462, Volume 2, August 1994).

6. The results of the analysis should be presented and should include maximum power levels reached for the reactor and the peak fuel rod, scram or rod block actions that occur, 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 section.

7. For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in draft SRP Section 14.3, to verify that the design set forth in the standard safety analysis report, including inspections, tests, analyses, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 draft contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.

9. For reviews of DC and COL applications under 10 CFR Part 52, the reviewer should follow the above procedures to verify that the design set forth in the safety analysis report, and if applicable, site interface requirements meet the acceptance criteria. For DC applications, the reviewer should identify necessary COL action items. With respect to COL applications, the scope of the review is dependent on whether the COL applicant references a DC, an ESP or other NRC-approved application and/or reports.

   After this review, SRP Section 14.3 should be followed for the review of Tier I information for the design, including the postulated site parameters, interface criteria, and ITAAC.

## IV. EVALUATION FINDINGS

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. If on completion of the review the staff finds the applicant's analysis acceptable, conclusions of the following type should be included in the staff's safety evaluation report:

1. The possibilities for single failures of the reactor control system which could result in uncontrolled withdrawal of control rods beyond normal limits under power operation conditions have been reviewed.

2. The scope of the review has included investigations of possible initial conditions and the range of reactivity insertions, the course of each resulting AOO, and the instrumentation response to the AOO.

3. The methods used to determine the peak fuel rod response, and the input into that analysis, such as power distributions, rod reactivities, and reactivity feedback effects of moderator and fuel temperature changes, have been examined.

   If audit calculations have been done, they should be summarized.

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

The applicant has met the following requirements:

   A. GDC 10 and GDC 17, ensuring that the specified acceptable fuel design limits are not exceeded;

   B. GDC 20, ensuring that the reactivity control systems are automatically initiated so that specified acceptable fuel design limits are not exceeded; and

   C. GDC 25, ensuring that single malfunctions in the reactivity control system will not cause the specified acceptable fuel design limits to be exceeded.

   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 ensure that fuel rod failure will be precluded for this event. The basis for acceptance in the staff review is that the applicant's analysis of maximum AOOs for single error control rod malfunctions 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 DC and COL reviews, the findings will also summarize (to the extent that the review is not discussed in other SER sections) the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable, interface requirements and combined license 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 docketed 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 17, "Electric Power Systems."

3. 10 CFR Part 50, Appendix A, General Design Criterion 20, "Protection System Functions."

4. 10 CFR Part 50, Appendix A, General Design Criterion 25, "Protection System Requirements for Reactivity Control Malfunctions."

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**SRP Section 15.4.2**
Description of Changes

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in (Draft) Revision 1, dated April, 1996 of this SRP. See ADAMS accession number ML062580049.

In addition this SRP section was administratively updated in accordance with NRR Office Instruction, LIC-200, Revision 1, "Standard Review Plan (SRP) Process." The revision also adds standard paragraphs to extend application of the updated SRP section to prospective submittals by applicants pursuant to 10 CFR Part 52.

The technical changes are incorporated in Revision 2, dated [Month] 2007.

Review Responsibilities - Reflects changes in review branches resulting from reorganization and branch consolidation. Change is reflected throughout the SRP.

I. AREAS OF REVIEW

Reformatted the section with new numbering system. Incorporated reference to 10 CFR Part 52 from draft revision 1 - April 1996.

II. ACCEPTANCE CRITERIA

Reformatted the section with new numbering system. Incorporated reference to 10 CFR Part 52 from draft revision 1 - April 1996.

III. REVIEW PROCEDURES

Reformatted the section with new numbering system. Incorporated reference to 10 CFR Part 52 from draft revision 1 - April 1996.

IV. EVALUATION FINDINGS

None.

V. IMPLEMENTATION

None.

VI. REFERENCES

None.