NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

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

REVIEW RESPONSIBILITIES

Primary - ~~Core Performance Branch (CPB)~~Reactor Systems Branch (SRXB)[1]

Secondary - None

I. AREAS OF REVIEW

The ~~CPB~~SRXB[2] 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,[3] 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.

Review Interfaces[4]

The SRXB also reviews[5] ~~T~~the reactivity coefficients and control rod worths utilized in this review ~~are also evaluated by the CPB~~ under SRP Section 4.3.

II. ACCEPTANCE CRITERIA

1. The following General Design Criteria ~~(Ref. 1)~~[6] 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 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.[7]

c~~b~~. 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.

d~~c~~. 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.

2. 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~~, subsection II.1~~[8] are met.

   b. Fuel centerline temperatures (for PWRs) as specified in SRP Section 4.2~~, subsection II.A.2(a) and (b)~~[9] do not exceed the melting point.

   c. Uniform cladding strain (for BWRs) as specified in SRP Section 4.2~~, subsection II.A.2(b)~~[10] does not exceed 1%.

<u>Technical Rationale:</u>[11]

The technical rationale for application of the above acceptance criteria to the control rod withdrawal from low power transient 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 assures that the fuel cladding integrity is not challenged during this anticipated operational occurrence.

2. 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 assure that (1) 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 assures that the fuel cladding integrity is not challenged during and uncontrolled control rod assembly withdrawl in conjunction with a loss of onsite or of offsite power.

3. GDC 20 requires that the protective system automatically initiate the operation of the reactivity control system to assure 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 assure that an abnormal rod withdrawal be detected and automatically terminated prior to fuel design safety limits being violated. Meeting GDC 20 assures that cladding integrity is not challenged during this anticipated operational occurrence.

4. GDC 25 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 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 assures 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, in determining whether the acceptance criteria are met, considers the following:

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[12]. 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.)[13]

8. For boiling water reactor applicants, the evaluation should also include the effects and consequences of a control rod removal error during refueling operations.[14]

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[15]

IV. EVALUATION FINDINGS

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

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 worths, 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.)

The staff concludes that the requirements of General Design Criteria 10, 17,[16] 20, and 25 have been met. This conclusion is based on the following:

The applicant has met the requirement 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. 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, 17,[17] 20, and 25 have been met. This conclusion is based on the following:

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. 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~~,~~[18] 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 design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP Section.[19]

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52.[20] Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of 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.[21]

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design~~," General Design Criterion 20, "Protection System Functions," and General Design Criterion 25, "Protection System Requirements for Reactivity Control Malfunctions~~."

2. 10 CFR Part 50, Appendix A, General Design Criterion 17, "Electric Power Systems."[22]

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."[23]

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Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 15.4.1.
2.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 15.4.1.
3.	**Integrated Impact 941**	Added GDC 17 to the list of requirements reviewed in this section.
4.	SRP-UDP format item, Reformat Areas of Review.	Added "Review Interfaces" subsection to Areas of Review. Reformatted existing description in order to more clearly elucidate how the SRXB reviews other SRP sections, related to the topic of control assembly withdrawal events.
5.	SRP-UDP format item, Reformat Areas of Review.	Added a leader for the new "Review Interfaces" subsection.
6.	SRP-UDP format Item, Reference Citations.	Deleted the parenthetical reference for GDCs.
7.	**Integrated Impact 941**	Added GDC 17 to the Acceptance Criteria and renumbered remaining AC accordingly.
8.	Editorial, Reference Verification.	Deleted subsection portion of the citation in that such specific references are not necessary for the reviewer to locate appropriate specific criteria. Further, the paragraph designations are not yet finalized for the revised sections 4.2 and 4.4.
9.	Editorial, Reference Verification.	Deleted subsection portion of the citation in that such specific references are not necessary for the reviewer to locate appropriate specific criteria. Further, the paragraph designations are not yet finalized for the revised sections 4.2 and 4.4.
10.	Editorial, Reference Verification.	Deleted subsection portion of the citation in that such specific references are not necessary for the reviewer to locate appropriate specific criteria. Further, the paragraph designations are not yet finalized for the revised sections 4.2 and 4.4.

Item	Source	Description
11.	SRP-UDP format item, Develop Technical Rationales.	Technical Rationale were developed for the Acceptance Criteria in accordance with the format requirements for the SRP-UDP.
12.	Editorial	Spelled out an acronym used in the text.
13.	**Integrated Impact 941**	Add review requirements to assure that for CE System 80+ submittals, the applicant considered a loss of offsite power in conjunction with the limiting single active failure when assessing the consequences of the anticipated operational occurrence.
14.	**Integrated Impact 948**	Clarified that the analysis should consider a control rod removal error during refueling operations, or supply cogent technical rationale why such an analysis does not need to be considered.
15.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
16.	**Integrated Impact 941**	Added GDC 17 to the list of requirements met.
17.	**Integrated Impact 941**	Added GDC 17 to the list of requirements met.
18.	Editorial.	Removed inappropriate comma from the text.
19.	10 CFR 52 implementation.	Added paragraph to address evaluation findings for review of applications under 10 CFR 52.
20.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard sentence to address application of the SRP section to reviews of applications filed under 10 CFR Part 52, as well as Part 50.
21.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
22.	**Integrated Impact 941**	Added GDC 17 to references list.
23.	Editorial.	Separated multiple GDC references into separate numbered references.

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Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
941	Modify the Review Procedure applicable to CE System 80+ applications to include the assumption of the loss of offsite power (LOOP) in addition to the limiting single failure event for the analysis of this anticipated operational occurrence.	III.7, Review Procedure
948	Modify the Review Procedure to assure that applicants presenting BWR submittals have considered the control rod assembly withdrawal during refueling or control blade removal error during refueling operations while assessing the impact of the uncontrolled control rod assembly withdrawal anticipated operational occurrence.	I, II, III, IV, and V