



## U.S. NUCLEAR REGULATORY COMMISSION

# STANDARD REVIEW PLAN

### 15.4.3 CONTROL ROD MISOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR)

#### REVIEW RESPONSIBILITIES

**Primary** - Organization responsible for the review of Reactor Systems

**Secondary** - None

#### I. AREAS OF REVIEW

The specific areas of review are as follows:

1. The types of control rod misoperations that are assumed to occur. For a pressurized water reactor (PWR), this may include one or more rods moving or displaced from normal or allowed control bank positions (such as dropped rods and rods left behind when inserting or withdrawing banks, or single rod withdrawal) and may include the automatic control system attempting to maintain full power.
2. Descriptions of rod position, flux, pressure, and temperature indication systems, and those actions initiated by these systems (e.g., turbine runback, rod withdrawal prohibit, rod block) which can mitigate the effects or prevent the occurrence of various misoperations.

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### USNRC STANDARD REVIEW PLAN

This Standard Review Plan, NUREG-0800, has been prepared to establish criteria that the U.S. Nuclear Regulatory Commission staff responsible for the review of applications to construct and operate nuclear power plants intends to use in evaluating whether an applicant/licensee meets the NRC's regulations. The Standard Review Plan 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 an acceptable method of complying with the NRC regulations.

The standard review plan sections are numbered in accordance with corresponding sections in the Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)." Not all sections of the standard format have a corresponding review plan section. The SRP sections applicable to a combined license application for a new light-water reactor (LWR) will be based on Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," until the SRP itself is updated.

These documents are made available to the public as part of the NRC's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Individual sections of NUREG-0800 will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience. Comments may be submitted electronically by email to [NRR\\_SRP@nrc.gov](mailto:NRR_SRP@nrc.gov).

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3. Descriptions of the sequence of events occurring during each anticipated operational occurrence (AOO), e.g., rod drop followed by automatic return to full power with possible power overshoot, including the effect of important feedback mechanisms and trips.
4. Descriptions of the calculational models used and justification of their validity and adequacy.
5. The input to the calculations, including rod worths, power distributions, and feedback coefficients and evidence of the conservatism of the input.
6. Results of the analyses, including, for each of the AOOs considered, plots of the time history of reactor power, reactor vessel pressure, critical heat flux for the limiting fuel rod, and maximum fuel centerline temperature or linear heat generation rate.
7. 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 includes 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. The organization responsible for reactor systems also performs the following reviews under the SRP sections indicated:
  - A. Review of the limiting anomalies for a boiling water reactor (BWR) with current modes of control rod operation is performed under Standard Review Plan (SRP) Sections 15.4.1 and 15.4.2, and no additional areas are considered under SRP Section 15.4.3.
  - B. Review of uniform cladding strain and fuel centerline temperatures are performed under SRP Section 4.2.
  - C. Review of Doppler and void coefficients are performed under SRP Section 4.3.
  - D. Review of thermal margin limits are performed under SRP Section 4.4.
2. The organization responsible for reactor systems will coordinate other branches' evaluations that interface with the overall review of the system, as follows:

The organization responsible for instrumentation and controls reviews those safety systems required to prevent misoperations, as required by General Design Criterion 25, as well as the control rod system. The purpose of the review is to determine the events that are to be included as single error malfunctions (e.g., single rod withdrawal). The organization responsible for instrumentation and controls performs these reviews as part of its primary review responsibility under SRP Sections 7.2 and 7.7.

For those areas of review identified above as part of the primary responsibility of other branches, the acceptance criteria and methods of application are contained in the referenced SRP section.

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 anticipated operational occurrences.
2. 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 anticipated operational occurrences.
3. 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.

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

1. The thermal margin limits (departure from nucleate boiling ratio for PWRs) as specified in SRP Section 4.4, subsection II.1, are met.
2. Fuel centerline temperatures as specified in SRP Section 4.2, subsection II.A.2(a) and (b), do not exceed the melting point.
3. Uniform cladding strain as specified in SRP Section 4.2, subsection II.A.2(b), does not exceed 1%.

## Technical Rationale

The technical rationale for application of these requirements to reviewing 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 a control rod misoperation due to system malfunction or operator error for a BWR or a PWR to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of AOOs. This SRP section and SRP Sections 4.2, 4.3, 4.4, 7.2, 7.7, 15.4.1, and 15.4.2 provide guidance for ensuring that instrument setpoints are initially within and remain within the technical specification limits, thereby ensuring that specified acceptable fuel design limits are not exceeded.

Meeting the requirements of GDC 10 provides reasonable assurance that specified acceptable fuel design limits are not exceeded during either normal operations or AOOs (e.g., control rod misoperation) due to system malfunction or operator error.

2. Compliance with GDC 20 requires that each reactor protection system be designed (1) to initiate the automatic 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 a control rod misoperation due to system malfunction or operator error for a PWR or a BWR to ensure that acceptable fuel design limits are not exceeded as a result of AOOs. The reactor protection system automatically initiates the operation of appropriate systems, including the reactivity control system (RCS), to terminate the AOOs analyzed in this SRP section. AOOs such as those caused by a control rod misoperation are terminated in a timely manner so that acceptable specified fuel design limits are not exceeded for either a PWR or a BWR. This SRP section and SRP Sections 4.2, 4.3, 4.4, 7.2, 7.7, 15.4.1, and 15.4.2 provide guidance for ensuring that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences.

Meeting the requirements of GDC 20 provides reasonable assurance that specified acceptable fuel design limits will not be exceeded when the reactor protection system initiates operation of appropriate systems to terminate AOOs caused by control rod misoperations due to system malfunction or operator error.

3. Compliance with 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 systems, such as accidental withdrawal (not ejection or dropout) of control rods.

GDC 25 is applicable to this section because the reviewer evaluates the effects and consequences of a control rod misoperation due to system malfunction or operator error for either a PWR or a BWR at power. One criterion specifies that the reactor protection

system be designed to ensure that specific acceptable fuel design limits are not exceeded for either a PWR or a BWR during normal operations or during an AOO, including the event of a single malfunction of the RCS. The reactor protection system operates in a manner that automatically terminates the AOOs analyzed in this SRP section. This SRP section and SRP Sections 4.2, 4.3, 4.4, 7.2, 7.7, 15.4.1, and 15.4.2 provide guidance for ensuring that specified acceptable fuel design limits are not exceeded as a result of AOOs.

Meeting the requirements of GDC 25 provides reasonable assurance that a single malfunction of the reactivity control system, together with anticipated operational occurrences caused by the initiating event of a control rod misoperation due to system malfunction or operator error during either normal operation or an AOO, will not cause specified acceptable fuel design limits to be exceeded.

### III. REVIEW PROCEDURES

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.

The reviewer, in determining whether the criteria are met, should determine the AOOs to be considered for this event. Generally, the list of errors should include: (1) inadvertently withdrawing one or several rods, (2) leaving one or several rods behind during bank withdrawal, and (3) inserting one or several rods with power compensation in other portions of the core. In addition to these events, the reviewer must also decide, by postulating single failures in equipment or errors in operation, whether additional single rod malfunctions can be created. Once the list of AOOs has been established, the reviewer must determine acceptability in accordance with the criteria of subsection II of this SRP section.

1. For each failure event analyzed, the cases which result in a limiting fuel rod condition should be presented. Initial conditions and parameter values selected for these cases should be justified with a sensitivity analysis or discussion. Conditions of first-order importance for any time in cycle are initial power level and distribution, initial rod configuration, reactivity addition rate, moderator temperature, fuel temperature, and void reactivity coefficients.
2. For each event, the analytical methods used by the applicant are reviewed. Those steady-state and AOO methods that are primarily based on reactor physics considerations are the responsibility of the organization responsible for reactor systems. Where thermal-hydraulic methods are involved, review assistance may be requested as described in SRP Section 4.4. In either case, the reviewer should determine whether the applicant's evaluation methods are acceptable. 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 specific technical assistance requests.

- B. Perform an independent review of the method (usually described in a separate licensing topical report and often completed, on a generic basis, outside the scope of the review for a particular facility).
  - C. Perform auditing-type calculations with 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 method that have not already been fully reviewed and approved.
3. For each event, the results are evaluated. In addition to verifying conformance to the acceptance criteria of subsection II above, the reviewer determines that:
- A. Input conditions (e.g., pressure, temperature, flow rate) are at the adverse end of the range of values specified as the operating range.
  - B. Initial power is 102% of licensed core thermal power, unless a lower power level is justified by the applicant.
  - C. Output signals (power, temperature, flux perturbation) provide adequate alarm or scram signals.
  - D. Nuclear conditions that interact with this event (e.g., Doppler coefficient, void coefficient) have been calculated as described in SRP Section 4.3.)
4. 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.

1. The possibilities for single failures of the reactor control system which could result in a movement or misposition of control rods beyond normal limits have been reviewed. The scope of the review has included investigations of possible rod misposition configurations, the course of the resulting AOOs or steady-state conditions, and the instrumentation response to the AOO or power maldistribution. The methods used to determine the peak fuel rod response, and the input to that analysis, such as power distribution changes, rod reactivities, 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, 20, and 25 have been met. This conclusion is based on the following:

The applicant has met the following requirements:

- A. GDC 10, 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.
- 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 maximum configurations and AOs for single error control rod malfunctions have been analyzed, that the analysis 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, and interface requirements and combined license action items relevant to this SRP section.

#### V. IMPLEMENTATION

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. 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, 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 20, "Protection System Functions"
- 3. 10 CFR Part 50, Appendix A, General Design Criterion 25, "Protection System Requirements for Reactivity Control Malfunctions."

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#### **PAPERWORK REDUCTION ACT STATEMENT**

The information collections contained in the draft Standard Review Plan are covered by the requirements of 10 CFR Part 50 and 10 CFR Part 52, and were approved by the Office of Management and Budget, approval number 3150-0011 and 3150-0151.

#### **PUBLIC PROTECTION NOTIFICATION**

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

**SRP Section 15.4.3**  
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 ML0061990652.

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.