U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.4.3 CONTROL ROD MISOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR)

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - None

I. AREAS OF REVIEW

CPB reviews the following:

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. For a boiling water reactor (BWR) with current modes of control rod operation, limiting anomalies are reviewed under SRP Sections 15.4.1 and 15.4.2, and no additional areas are considered here.

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.

   Those safety systems required to prevent misoperations, as required by General Design Criterion 25, as well as the control rod system are reviewed in SRP Sections 7.2 and 7.7. The purpose of the review is to determine what events are to be included as single error malfunctions (e.g., examine single rod withdrawal).

3. Descriptions of the sequence of events occurring during each transient, 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 transients 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.

II. ACCEPTANCE CRITERIA

1. The following General Design Criteria (Ref. 1) apply:

   a. Criterion 1C, 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 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.

   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) as specified in SRP Section 4.4, subsection II.1 are met.

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

   c. Uniform cladding strain as specified in SRP Section 4.2, subsection II.A.2(b) do not exceed 1%.

III. REVIEW PROCEDURES

The reviewer, in determining whether the criteria are met, must determine the transients that should be considered for this event. Generally, the list of errors should include: inadvertently withdrawing one or several rods; leaving one or several rods behind during bank withdrawal; and 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 transients has been established, the reviewer must determine acceptability in accordance with the criteria of subsection II of this SRP.

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 transient methods that are primarily based on reactor physics considerations are the responsibility of CPB. Where thermal-hydraulic methods are involved, review assistance may be requested as described in SRP Section 4.4. In either case, the reviewer must 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 a de novo review of the method (usually described in a separate licensing topical report, and often 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 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) provided 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.

IV. EVALUATION FINDINGS

If the reviewer's evaluation shows that the applicant's analyses are acceptable, the following kinds of statements should be included in the staff's safety evaluation report:

> The possibilities for single failures of the reactor control system which could result in a movement or malposition of control rods beyond normal limits have been reviewed. The scope of the review has included investigations of possible rod malposition configurations, the course of the resulting transients or steady-state conditions, and the instrumentation response to the transient 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 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 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 maximum configurations and transients 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.

V. IMPLEMENTATION

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

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.

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."