U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.4.1

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

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Core Performance Branch (CPB)

I. AREAS OF REVIEW

The effects and consequences of an uncontrolled control rod assembly withdrawal (a bank for a pressurized water reactor; a single rod, with current control modes, for a boiling water reactor) from a subcritical or low power (e.g., startup range) condition are evaluated by RSB. CPB reviews the reactivity coefficients and control rod worths under Standard Review Plan 4.3. The review under this plan 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.

II. ACCEPTANCE CRITERIA

1. The following general design criteria (Ref. 1) apply:

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

   b. Criterion 20, which requires that the protection system action be initiated automatically.

2. The following fuel design limits serve as the acceptance criteria for this event:

   a. Critical heat flux should not be exceeded. Examples of limits used previously to satisfy this criterion are:

      (1) In boiling water reactors (BWR's), the minimum critical heat flux ratio (MCHFR) calculated with the Hench-Levy correlation (Ref. 2) should exceed 1.0 at all times. The value of the minimum critical power ratio (CPR) calculated with the GETAB analysis (Ref. 3) will vary for different plants and product lines. Typically, the value will exceed 1.06.

(2) In pressurized water reactors (PWR's), the minimum departure from nucleate boiling ratio (DNBR) should not be less than 1.30 using the W-3 correlation (Ref. 4), or less than 1.32, using the B&W-2 correlation (Ref. 5).

If the application under review does not use one of these limits, then the correlation used by the applicant is reviewed by RSB under Standard Review Plan 4.4 and a criterion for critical heat flux is established that is acceptable here.

b. Fuel temperature and clad strain limits consistent with the acceptance criteria of Standard Review Plan (SRP) 4.2 should not be exceeded. For steady-state or nearly steady-state conditions, this can be expressed in terms of a linear heat generation rate (usually expressed in kW/ft). Examples of this criterion are:

(1) For BWR's, a linear heat generation rate of 24-28 kW/ft, which would result in limited UO_2 melting, but is not sufficient to cause 1% clad strain and potential clad failure.

(2) For PWR's, a linear heat generation rate of 20-22 kW/ft, which would result in a centerline fuel temperature equal to or less than the melting point of UO_2.

The specific value of the linear heat generation rate for this criterion is established during each review in a manner consistent with the acceptance criteria of SRP 4.2. For non-equilibrium states, the calculated transient temperatures and strains corresponding to these steady-state limits should not be exceeded.

III. <u>REVIEW PROCEDURES</u>

The procedures below are used for both the construction permit (CP) and operating license (OL) reviews. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values are used in the analysis and the reviewer should compare these to the limiting safety settings included in the proposed technical specifications.

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 minimum core pressure and maximum temperature (i.e., the extremes of postulated conditions) should be consistent with the rod and power configuration to give minimum DNBR or MCHFR 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 at the 50 percent rod density configuration. (More recent modes of BWR control such as group withdrawal may require that other configurations be examined.) The withdrawal rate should be the maximum available to the system.

3. The exact analysis of the transient would normally 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 (TARs).

   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 calculatiors 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 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 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 4.3, considering the time in cycle and temperature conditions of the fuel. The use of beginning of lifetime (BOL) Doppler coefficients is the most conservative; they should be used unless other values are specifically justified. 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 4.3. The most positive or least negative values should be used and for a PWR this occurs at BOL. If the coefficient is negative, it may be conservatively taken as zero, as is generally done in BWR analyses.

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 temperatures, DNBR or MCHFR, and maximum clad strain. The latter are compared to the acceptance criteria.

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 check calculations have been done, they should be summarized).

"The resulting extreme conditions of fuel power, temperature, and departure from nucleate boiling have been compared to acceptance criteria for fuel integrity, which for this reactor are (insert criteria from SRP 4.2 and 4.4). The analyses have shown that these limits are not exceeded.

"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 fuel damage limits are not exceeded. The staff concludes that the calculations contain sufficient conservatism, with respect to both assumptions and models, to assure that fuel damage will not result from such control rod assembly accidents."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants".

2. J. M. Healzer, J. E. Hench, E. Janssen, and S. Levy, "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors," APED-5186 (proprietary) and APED-5286 (nonproprietary), General Electric Company (1966).

3. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," NEDO-10958, General Electric Company (1973).

4. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," Jour. Nuclear Energy, Vol. 21, 241-248 (1967).

5. J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, and L. J. Stanek, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," in "Two-Phase Flow and Heat Transfer in Rod Bundles," American Society of Mechanical Engineers, New York (1969).

6. Standard Review Plan 4.2, "Fuel System Design."

7. Standard Review Plan 4.3, "Nuclear Design."

8. Standard Review Plan 4.4, "Thermal and Hydraulic Design."

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SRP 15.4.2