U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.4.2 UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL AT POWER

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) at power are evaluated by RSB. CPB reviews the reactivity coefficients and control rod assembly worths involved under Standard Review Plan (SRP) 4.3. The review under this plan covers the description of the causes of the transient and of 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 transients 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) should not be less than 1.0 using the Hench-Levy correlation (Ref. 2), or typically the minimum critical power ratio (MCPR) should not be less than 1.06 using GETAB analysis (Ref. 3).

(2) In pressurized water reactors (PWR's), the minimum departure from nucleate boiling ratio (DNBR) should not be less than 1.3, using the W-3 correlation (Ref. 4), or less than 1.32, using the B&W-2 correlation (Ref. 5). The use of these correlations is limited to 14 x 14 and 15 x 15 rod arrays.

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

b. Fuel temperature and fuel clad strain limits consistent with the acceptance criteria of SRP 4.2 (Ref. 6) 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 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. REVIEW PROCEDURES

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 covers 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 transient 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 modifications under consideration may change this to group movement). The review covers 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 transient would normally 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

is relatively small, so that the transient 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 transient is halted by action of the rod block monitor (RBM) system, which should block rod withdrawal before fuel safety limits are reached.

The review process for a BWR, while recognizing the inherent transient nature of the problem, is concentrated on the steady-state aspects of the transient to assure that initial and subsequent power distributions are maximized, that the reactor conditions produce minimum CHFR, and that the response of the RBM 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 transient calculations. Because power distributions in the course of the transient can frequently be predicted conservatively using design-limit peaking factors, point kinetics may be used for the nuclear transient. The nuclear transient 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 transient conditions are explored, that the transient 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, transient, 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 (SER's) and reports prepared in response to technical assistance requests (TAR's).

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 are not fully reviewed or approved.

5. The significant 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 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 beyond normal limits under power operation conditions have been reviewed. The scope of the review has included investigations of possible initial conditions and the range of reactivity insertions, the course of the resulting transient and the instrumentation response to the transient. The methods used to determine the peak fuel rod response, and the input that analysis, such as power distributions, rod reactivities, and reactivity feedback effects of 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 acceptance criteria from SRP's 4.2 and 4.4, as appropriate). The analyses have shown that these limits are not exceeded.

"The basis for acceptance in the staff review is that the applicant's analysis of maximum transients for single error control rod malfunctions 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 input assumptions and models to assure that fuel damage will not result from control rod withdrawal transients."

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-5286, General Electric Company, July 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 Plans 4.2, "Fuel System Design," and 4.4, "Thermal and Hydraulic Design."

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SRP 15.4.3