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(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.4.8 SPECTRUM OF ROD EJECTION ACCIDENTS (PWR)

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - Accident Evaluation Branch (AEB)

I. AREAS OF REVIEW

The CPB evaluates the consequences of a control rod ejection accident in the area of physics. The review covers the possible initial conditions, rod patterns and worths, scram worth as a function of time, adequacy of the various reactivity coefficients, adequacy of the calculational methods, and any core parameters which affect the peak reactor pressure or the probability of fuel rod failure.

The relevant thermal-hydraulic analyses are reviewed under SRP Section 4.4.

The AEB reviews, as part of its secondary review responsibility, described in the appendix to this SRP section, the radiological consequences of a rod ejection accident by using a source term for dose calculations based on the amount of failed fuel as obtained by CPB from the reactor core analyses. The evaluation finding provided is as indicated in the attached Appendix.

The applicant's determination of the reactor trip delay time, i.e., the time elapsed between the instant the sensed parameter reaches the level for which protective action is required and the onset of negative reactivity insertion, is reviewed under SRP Sections 7.2 and 7.3.

II. ACCEPTANCE CRITERIA

CPB acceptance criteria are based on meeting the requirements of General Design Criterion 28 (Ref. 1) as it relates to the effects of postulated reactivity accidents neither resulting in damage to the reactor coolant pressure boundary greater than limited local yielding, nor causing sufficient damage to impair significantly the capacity to cool the core.

Regulatory Positions and specific guidelines necessary to meet the relevant requirements of GDC 28 are in Regulatory Guide 1.77.

Regulatory Guide 1.77 (Ref. 2) identifies acceptable analytical methods and assumptions that may be used in evaluating the consequences of a control rod ejection accident. Specific criteria used by CPB in evaluating the control rod ejection accident are:

a. Reactivity excursions should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel rod.

b. The maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed the "Service Limit C" as defined in the ASME Code (Ref. 3).

c. The fission product inventory in the fuel rods calculated to experience a departure from nucleate boiling (DNB) condition is an input to the radiological evaluation by AEB. The radiological criteria used in the evaluation of control rod ejection accidents (PWRs) are given in Appendix B of Regulatory Guide 1.77 (Ref. 2).

III. REVIEW PROCEDURES

1. Review of the applicant's analyses, showing that the first of the acceptance criteria above is met, proceeds as follows:

   a. A spectrum of initial conditions is considered, which must include both zero-power and full-power conditions, at beginning and end of a reactor fuel cycle (BOC and EOC), to assure examination of upper bounds on possible fuel damage. Initial full-power conditions should include the uncertainties in the calorimetric measurement of power.

   b. From the initial conditions of (a) and from control rod patterns, the limiting rod worth is determined. Where confirmation is considered necessary the reviewer may calculate, as an audit, the worth of limiting rods.

   c. Reactivity coefficient values corresponding to the limiting initial conditions must be used at the beginning of the transient. The reviewer checks the reactivity coefficient curves used by the applicant with those reviewed by the CPB under SRP Section 4.3. The two coefficients of most interest are the Doppler and moderator coefficients. If no three-dimensional space-time calculation is performed, the reactivity feedback must be conservatively weighted to account for the variation in the missing dimension(s).

   d. The reviewer inspects the control rod insertion assumptions which include: trip parameters, trip delay time, rod velocity curve, and differential rod worth. Trip parameters and delay time are reviewed under SRP Section 7.2. Control rod worth is checked by the reviewer for consistency with the review performed under SRP Section 4.3.

   e. The applicant's analytical methods are reviewed. The reviewer may use the results of previous case work, if the analytical methods have been previously reviewed and approved by the staff. Otherwise he must perform a de novo review on this case. Alternatively an audit of several calculations, using methods considered acceptable to the staff, may be done by the reviewer (or consultants to the staff). The primary concern of the reviewer is how well the elements of the analytical model represent the true three-dimensional problem. Other items checked by the reviewer include feedback mechanisms, number of delayed neutron groups, two-dimensional representation of fuel element distribution, primary flow treatment, and scram input.

f. Results of the calculations done by procedures described in steps a-e are expressed as values of the radially-averaged fuel rod enthalpy (in units of cal/gm). The reviewer determines that the maximum value does not exceed 280 cal/gm.

2. Verification of compliance with the second acceptance criterion is accomplished as follows:

   a. The same procedures considered in steps a-f above are followed.

   b. For each accident, the maximum primary system pressure should be calculated by an analytical method acceptable to the staff or, as before, an independent audit calculation is made by the staff. The reviewer checks the results (as obtained by the applicant or the staff) for compliance with the second criterion.

3. The number of fuel rods experiencing clad failure is determined (for use in evaluating the radiological consequences) by the following procedure:

   a. The reviewer determines that an acceptable procedure for calculating a departure from nucleate boiling condition during the reactivity excursion has been used. This may be done by referring to previous cases for the same nuclear steam supply system (NSSS) vendor. If no approved technique is available, as might be the case for the first project using a new or substantially revised model, the reviewer must perform a separate detailed review (which may be documented separately in a topical report).

   b. The reviewer must determine that the number of rods used in the radiological evaluation is the number of rods calculated to have a departure from nucleate boiling. Departure from nucleate boiling must be calculated in accordance with the criteria reviewed and accepted under SRP Section 4.4. Typically, the criterion defines a departure from nucleate boiling ratio (DNBR) less than 1.30 when DNB correlations such as W-3 (Ref. 4) or BAW-2 (Ref. 5) are used.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and his review supports conclusions of the following type, to be included in the staff's safety evaluation report:

> The staff concludes that the analysis of the rod ejection accidents is acceptable and meets the requirements of General Design Criterion 28. This conclusion is based on the following:

The applicant met the requirements of GDC 28 with respect to preventing postulated reactivity accidents that could result in damage to the reactor coolant pressure boundary greater than limited local yielding, or cause sufficient damage that would significantly impair the capability to cool the core. The requirements have been met by demonstrating that the regulatory positions of Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for PWR's" are complied with. The staff has evaluated the applicant's analysis of the assumed control rod ejection accident and finds the assumptions, calculation techniques, and consequences acceptable. Since the calculations resulted in peak fuel enthalpies less than 280 cal/gm, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten UO_2 was assumed not to occur. The pressure surge was, therefore, calculated on the basis of conventional heat transfer from the fuel and resulted in a pressure increase below "Service Limit C" (as defined in Section III, "Nuclear Power Plant Components," of the ASME Boiler and Pressure Vessel Code) for the maximum control rod worths assumed. The staff believes that the calculations contain sufficient conservatism, both in the initial assumptions and in the analytical models, to ensure that primary system integrity will be maintained.

V. IMPLEMENTATION

The following section 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.

Implementation schedules for conformance to parts of the method described herein are contained in the referenced regulatory guide.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 28, "Reactivity Limits."

2. Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."

3. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

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). (See also BAW-10000 and BAW-10036.)