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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.4.8 SPECTRUM OF ROD EJECTION ACCIDENTS (PWR)

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - Reactor Systems Branch (RSB)
Accident Analysis Branch (AAB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

I. AREAS OF REVIEW

CPB evaluates the consequences of a control rod ejection accident in the area of physics. RSB, under Standard Review Plan (SRP) 4.4, reviews the relevant thermal-hydraulic analyses. The CPB 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 pin failure.

AAB reviews 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 physics and thermal-hydraulic analyses.

EICSB in SRP 7.2 and 7.3 reviews 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.

II. ACCEPTANCE CRITERIA

Regulatory Guide 1.77 (Ref. 1) identifies acceptable analytical methods and assumptions that may be used in evaluating the consequences of a rod ejection accident. Two criteria are used by CPB in evaluating the rod ejection accident:

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

2. The maximum reactor pressure during any portion of the assumed transient should be less than the value that will cause stresses to exceed the emergency condition stress limits as defined in the ASME Code (Ref. 2).

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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 AAB. The radiological criteria used in the evaluation of rod ejection accidents (PWR's) are given in Appendix B of Regulatory Guide 1.77 (Ref. 1).

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 fuel lifetime (BOL and EOL), to assure examination of upper bounds on possible fuel damage.

   b. From the initial conditions of (a) and from control rod patterns (Ref. 3) 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 under SRP 4.3 (Ref. 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.

   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 covered under SRP 7.2 by EICSB. Rod worth is checked by the reviewer for consistency with SRP 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 complete 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 analytical model elements 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 transient primary system pressure should have been 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 is usually 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 ratio (DNBR) less than 1.30 when a DNB correlation such as W-3 (Ref. 4) is used, or 1.32 when a DNB correlation such as B&W-2 (Ref. 5) is 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 has evaluated the applicant's analysis of the assumed 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 the emergency condition stress limit (as defined in Section III, "Nuclear Power Plant Components," of the ASME Boiler and Pressure Vessel Code) for the maximum 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.

"The consequences of the rod ejection accident have been evaluated, and the design of the plant has been found to assure that the recovery from the accident is sufficiently rapid and effective to limit the activity releases. The evaluation of radiological consequences has been performed using the recommendation of Regulatory Guide 1.77, the computer code ____________, and a conservative description of the plant response to the accident. The calculated doses are presented in Table ____________. Technical specification limits on primary-secondary coolant leakage assure that the potential doses are well within 10 CFR Part 100 exposure guidelines."

V. <u>REFERENCES</u>

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

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

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

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).

APPENDIX

STANDARD REVIEW PLAN 15.4.8

RADIOLOGICAL CONSEQUENCES OF CONTROL ROD EJECTION ACCIDENT (PWR)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Core Performance Branch (CPB)
Site Analysis Branch (SAB)

I. AREAS OF REVIEW

The AAB review under this appendix covers the following areas:

1. The plant response to a control rod ejection accident.

2. The calculation of whole body and thyroid doses at the exclusion area boundary and low population zone outer boundary due to the releases resulting from a rod ejection accident.

The purposes of the review are to assure that the plant procedures for recovery from a rod ejection accident and the plant technical specifications are properly taken into account in computing the whole body and thyroid doses at the nearest exclusion area boundary and low population zone (LPZ) outer boundary, and to compare the calculated doses against the appropriate guidelines.

The physics and thermal-hydraulic aspects of the accident are reviewed by CPB. Verification of the applicant's calculations of the number of fuel pins experiencing departure from nucleate boiling (DNB) and the amount of fuel reaching the clad melting temperature is obtained from the CPB.

II. ACCEPTANCE CRITERIA

The plant is considered adequately designed against a rod ejection accident at the construction permit stage, and the primary-secondary system leakage appropriately limited, if calculations show that the resulting doses at the nearest exclusion area boundary are on the order of 150 rem to the thyroid and 20 rem to the whole body, or less, for the first two hours after the accident, and 150 rem thyroid and 20 rem whole body, or less, for the course of accident at the LPZ outer boundary. Higher doses may be acceptable at the operating license (OL) review stage, up to the guidelines of 10 CFR Part 100. Technical specifications should be set to assure that the doses resulting from a rod ejection accident are limited to the guideline values.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this appendix as may be appropriate for a particular case. The judgment on areas given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

The detailed review of the radiological consequences of a rod ejection accident is done at OL stage when system parameters and accident analysis results are fully developed. At the CP stage, the reviewer estimates the doses from the rod ejection accident based on results from similar plants that have been recently reviewed.

The AAB review of the rod ejection accident at the OL stage covers the following topics:

1. Release of the radioisotopes to the environment via the containment building.

2. Release of radioisotopes to the environment through the secondary system.

3. Calculation of resulting doses.

Physical plant parameters, such as the steam generator steaming rates, are reviewed to ascertain their conservatism.

Regulatory Guide 1.77 (Ref. 2) should be used as a guide in the analysis of the accident. The release of radioisotopes through the secondary system should be analyzed independently by means of a digital computer code. Computer codes are currently under development within NRC. Documentation will be published in a NUREG report. In the analysis of this accident, a loss of offsite power is assumed. It is also assumed that nuclides released to the primary coolant due to any fuel failures or melting (this information is obtained from the CPB) are instantaneously and uniformly mixed in the coolant at the time of the accident. For releases via the containment building, Regulatory Guide 1.77 recommends that 100% of the noble gases and 25% of the iodines contained in the fuel which is estimated to reach initiation of melting be available for release from the containment. For releases through the secondary system, 100% of the noble gases and 50% of the iodines contained in the fuel which is estimated to reach initiation of melting are assumed to be released to the primary coolant.

The SAB provides the reviewer with the distance to the nearest boundary of the exclusion area, the accident (5 percentile) wind speed and X/Q, and the 0-8 hr and 8-24 hr X/Q values at the outer boundary of the LPZ. These X/Q values are used to estimate the consequences of releases from the containment and the consequences at the LPZ outer boundary of releases from the secondary system. The X/Q value for calculating the two-hour dose at the nearest exclusion area boundary from the releases through the secondary system is obtained from Regulatory Guide 1.5 (Ref. 3) and corrected for wind speeds differing from 1 m/sec (inverse ratio). A breathing rate of 3.47×10^{-4} m^3/sec is used in calculating the thyroid doses for the first 8 hours after the accident; from 8 to 24 hours, a breathing rate of 1.75×10^{-4} m^3/sec is used.

Although the resulting doses in case of an actual accident would be a composite of the doses computed for releases via the containment building and through the secondary system, both doses should be presented. If either dose approaches the limit, calculation of representative composite cases should be considered (the AAB branch chief should be consulted).

If the doses resulting from the releases through the secondary system exceed the limits specified in Section II above, the technical specification limit on primary-secondary system leakage is reduced accordingly. If the doses resulting from the potential releases from the primary containment exceed the specified limits, the pressure setpoint for actuation of the containment sprays may have to be reduced to obtain credit for spray removal of the fission products.

The physics and thermal-hydraulic aspects of the accident are reviewed by the CPB. Verification of the applicant's calculations of the number of fuel pins reaching DNB and the amount of fuel reaching the fuel melting temperature are obtained (and documented by buckslip) from the CPB. It is important to note that the fuel melting temperature criterion used for release of large fractions of fission gases corresponds to the initiation of melting as opposed to the 280 cal/gm used as a criterion by the CPB for core disruption.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that the review and calculations support conclusions such as the following, to be included with the CPB findings in the staff's safety evaluation report at the operating license stage:

> "The consequences of the rod ejection accident have been evaluated, and the design of the plant has been found to assure that the recovery from the accident is sufficiently rapid and effective to limit the activity releases. The evaluation of radiological consequences has been performed using the recommendations of Regulatory Guide 1.77, the computer code ________, and a conservative description of the plant response to the accident. The calculated doses are presented in Table ____________. Technical specification limits on primary-secondary coolant leakage assure that the potential doses are well within 10 CFR Part 100 exposure guidelines."

At the construction permit stage the following paragraph is included with the CPB findings in the staff's safety evaluation report:

> "On the basis of our experience with the evaluation of the control rod ejection accident for PWR plants, we have estiamted the doses from this accident to be on the order of ___________rem to the thyroid and ____________rem to the whole body at the nearest boundary of the exclusion area. If a reevaluation of this accident at the operating license stage results in dose estimates that exceed the 10 CFR Part 100 guidelines, appropriate limits on primary-secondary leakage and the setpoint for containment spray actuation will be set."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

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

3. Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors."

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SRP 15.4.9