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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.4.9 SPECTRUM OF ROD DROP ACCIDENTS (BWR)

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 drop accident in a boiling water reactor (BWR) in the area of physics. The CPB review covers the applicant's description of the occurrences that lead to the accident, safety features designed to limit the amount of reactivity available and the rate at which reactivity can be added to the core, and methods used to analyze the accident. A general reference on control rod drop accident analysis is noted in Reference 1.

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

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

The applicant's determination of the reactor trip delay time, or the amount of time which elapses between the instant the sensed parameter (e.g., pressure or neutron flux) 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. 2) 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.

Specific criteria necessary to meet the relevant requirements of GDC 28 are as follows:

1. Reactivity excursions should not result in radially averaged fuel rod 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 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).

3. The number of fuel rods predicted to reach assumed fuel failure thresholds and associated parameters such as the amount of fuel reaching melting conditions will be an input to a radiological evaluation. The assumed failure thresholds are a radially averaged fuel rod enthalpy greater than 170 cal/gm at any axial location for zero or low power initial conditions, and fuel cladding dryout for rated power initial conditions.

III. REVIEW PROCEDURES

1. Review of the applicant's analyses showing compliance with the first of the above criteria is carried out as follows:

   a. The reviewer verifies that the applicant has considered a spectrum of initial conditions for this event that covers the range of time-in-cycle and initial power levels.

   b. The reviewer verifies that the maximum expected individual control rod worths are used. In developing control rod worth criteria, the nominal control rod withdrawal pattern must be considered, as well as those abnormal patterns that are not precluded by an instrumentation system accepted under the review of SRP Section 7.

   c. The reviewer determines that an acceptable and conservative function is used to describe the control rod worth as a function of control rod position and that the control rod position as a function of time is suitably conservative.

   d. The reviewer determines that conservative reactivity coefficients, notably the Doppler coefficient, are used and that they are compatible with those described in SRP Section 4.3.

   e. The reviewer assures that the scram action is conservatively represented in the use of the integral scram worth curve (SRP Section 4.3) and in the use of the scram delay time.

   f. The reviewer checks the analytical methods or assures that they have been reviewed and approved previously. The reviewer may also perform an independent audit calculation using methods acceptable to the staff. The applicant's methods should account conservatively for all major reactivity feedback mechanisms.

2. The reviewer inspects the results of the calculation of maximum reactor pressure to determine compliance with the second criterion listed in subsection II of this SRP (the reviewer may do an audit calculation when appropriate).

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

   a. The reviewer determines that the transient critical power ratio (CPR) has been computed by an acceptable technique (either previously reviewed or reviewed de novo during this review) for analyses using full power conditions.

   b. The reviewer determines that the number of rods with enthalpy exceeding 170 cal/gm has been computed by an acceptable method.

   c. The reviewer determines that the amount of fuel exceeding melting conditions has been computed by an acceptable method.

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 drop accident is acceptable and meets the requirements of General Design Criterion 28. This conclusion is based on the following:
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> 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 since the staff has evaluated the applicant's analysis of the assumed control rod drop accident and finds the assumptions, calculational techniques, and consequences acceptable. Since the calculations predict peak fuel enthalpies less than 280 cal/gm, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molted 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 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.

VI. REFERENCES

1. "Rod Drop Accident Analysis for Large Boiling Water Reactors," NEDO-10527, General Electric Company, March 1972; Supplement 1 to NEDO-10527, July 1972; and Supplement 2 to NEDO-10527, January 1973.

2. 10 CFR 50, Appendix A, General Design Criterion 28, "Reactivity Limits."

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

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15.4.9 RADIOLOGICAL CONSEQUENCES OF CONTROL ROD DROP ACCIDENT (BWR) APPENDIX A

REVIEW RESPONSIBILITIES

Primary - Accident Evaluation Branch (AEB)

Secondary - Core Performance Branch (CPB)

I. AREAS OF REVIEW

The AEB review under this appendix to SRP Section 15.4.9 includes the following aspects of the postulated control rod drop accident for a boiling water reactor facility:

1. an examination of the plant response to the accident;

2. the release of fission products from the core to the environment via the turbine and condensers, as a result of the accident; and

3. the calculation of whole-body and thyroid doses at the exclusion area boundary (EAB) and the low population zone (LPZ) boundary due to the releases from the accident.

A secondary review is performed by the CPB and the results are used by AEB in the overall evaluation of the accident analysis. The core response aspects of the accident are reviewed by the CPB. Verification of the applicant's calculation of the number of fuel rod failures and the amount of fuel reaching the melting temperature is provided by the CPB.

II. ACCEPTANCE CRITERIA

The acceptance criteria are based on the requirements of 10 CFR Part 100 as related to mitigating the radiological consequences of an accident. The plant site and dose mitigating engineered safety features are acceptable with respect to the radiological consequences of a postulated control rod drop accident if the calculated whole-body and thyroid doses at the exclusion area boundaries (EAB) and at the low population zone (LPZ) boundaries are well within th exposure guideline values in 10 CFR Part 100, paragraph 11 (Ref. 1). "Well within" is

defined as 25% of the 10 CFR Part 100 exposure guideline values or 75 rem for the thyroid and 6 rem for whole-body doses.

The fission product source term used in the dose analysis is acceptable if it meets the guidelines of Regulatory Guide 1.77 (Ref. 2).

III. REVIEW PROCEDURES

The reviewer selects and emphasizes specific aspects of this appendix to Standard Review Plan Section 15.4.9 as appropriate for the particular plant. The judgment of which areas need to be given attention and emphasis is based on the similarity of the information presented in the SAR or other licensing submittals.

Based on past reviews by the staff, a control rod drop accident is expected to result in radiological consequences less than 10% of the Part 100 guideline values even with conservative assumptions. The reviewer should examine the site meteorology, plant features, and fuel damage as a result of the accident for the plant in question and compare these with the corresponding features and resulting doses for previously reviewed plants to ascertain whether a specific calculation of the radiological consequences should be performed. The reviewer should examine the applicant's description of the control rod drop accident, in particular, the sequence of events following the accident to assure that the most severe case from the standpoint of release of fission products to the environment is analyzed. Unless unusual plant or site features are present or the applicant's calculation shows an unusually large amount of fuel damage, a specific calculation of the radiological consequences is not necessary. In this case a comparison of the pertinent plant and site features is sufficient to conclude that the consequences of this event meet the acceptance criteria given in subsection II. However, a specific evaluation of this accident should be performed for the first application involving a particular standardized design to establish a reference point for comparison of future applications incorporating the design.

Where a specific calculation of the radiological consequences is to be performed, the core response aspects of the accident are reviewed by the CPB. Verification of the applicant's calculation of the number of fuel rod failures and the amount of fuel reaching the fuel melting temperature is obtained from the CPB. The following assumptions regarding the plant condition and release and transport of radioactivity are used in the independent AEB calculations:

1. A coincident loss of offsite power is assumed at the time of the accident.

2. The integrity of the turbine and condensers is unaffected by the rod drop accident.

3. The combination of reactor operating mode, control rod positions, core burnup, etc., that results in the largest source term, is selected for evaluation.

4. No allowance is made for activity decay prior to accident initiation, regardless of the reactor status for the selected case.

5. The amount of activity accumulated in the fuel-clad gap is assumed to be the same as that in Regulatory Guide 1.77 (Ref. 2).

6. The nuclide inventory of the fraction of the fuel which reaches or exceeds the initiation temperature of fuel melting (typically 2842°C) at any time during the course of the accident is calculated and 100% of the noble gases and 50% of the iodines contained in this fraction are assumed released to the reactor coolant. CPB should be requested to review analyses which propose that fuel melting is not likely to result in significant releases prior to MSIV closure.

7. Those fuel rods presumed to fail are assumed to have operated at power levels 1.5 times that of the average power level of the core.

8. Any nuclides released to the reactor coolant from fuel cladding failures or fuel melting are instantaneously and uniformly mixed in the reactor coolant in the pressure vessel at the time of the accident.

9. For conservative analysis it is assumed that 10% of the iodines and 100% of the noble gases released in the pressure vessel reach the turbine and condensers. A more realistic analysis may be performed as needed on a case-by-case basis. Such analysis accounts for the quantity of contaminated steam carried from the pressure vessel to the turbine and condensers based on a review of the minimum transport time from the pressure vessel to the first main steam isolation valve (MSIV) and considers the MSIV closure time.

10. All noble gases remain in a gaseous state and are available for leakage from the turbine and condensers.

11. Of those iodines which reach the turbine and condensers, 90% are removed by partitioning and plateout in the turbine and condensers leaving 10% airborne and available for leakage.

12. The turbine and condensers leak to the atmosphere at a rate of 1% per day for a period of 24 hours, at which time the leakage is assumed to terminate. Condenser leakage rates lower than 1% per day and shorter in duration than 24 hours will be reviewed on a case-by-case basis. Credit for condenser vacuum discharge isolation on high activity level in the steam, or credit for filtration of the condenser vacuum discharge, will also be reviewed on a case-by-case basis.

13. The effects of radiological decay during holdup in the turbine and condensers are taken into account.

14. The atmospheric dispersion factors (X/Q values), breathing rates, and dose conversion factors are the same as those used in the calculation of doses from a loss-of-coolant accident (Ref. 3).

The above assumptions are used in conjunction with a branch-approved computer code such as TACT to compute the radiological consequences. The whole-body and thyroid doses presented by the applicant in the SAR and those calculated independently by the staff are compared with the acceptance criteria in subsection II.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided by the applicant and that the applicant's analysis and the staff's independent

evaluation support conclusions of the following type, to be included in the AEB staff's safety evaluation report:

Where the radiological consequences have not been specifically calculated, the findings may be in the following form:

> The staff concludes that the distances to the exclusion area and to the low population zone boundaries for the (INSERT PLANT NAME) site, in conjunction with the operation of the dose mitigating ESF systems, are sufficient to provide reasonable assurance that the calculated radiological consequences of a postulated control rod drop accident are well within the exposure guidelines as set forth in 10 CFR Part 100, paragraph 11.
>
> The staff conclusion is based upon (1) its review of the applicant's analysis of the accident and radiological consequences and (2) the staff review of the same accident for similar plants at a number of sites using the source term assumptions of Regulatory Guide 1.77 and upon the similarity of those plant features for the (INSERT PLANT NAME) which affect the radiological consequences of the rod drop accident.

Where the radiological consequences have been calculated, the findings may be of the following form:

> The staff concludes that the distances to the exclusion area and to the low population zone boundaries for the (INSERT PLANT NAME) site, in conjunction with the operation of the dose mitigating ESF systems, are sufficient to provide reasonable assurance that the calculated radiological consequences of a postulated control rod drop accident are well within the exposure guidelines as set forth in 10 CFR Part 100, paragraph 11.
>
> The staff conclusion is based upon (1) its review of the applicant's analysis of the accident and radiological consequences and (2) an independent dose calculation by the staff using the source term assumptions contained in Regulatory Guide 1.77, the atmospheric dispersion factors as discussed in SRP Section 2.0, and other conservative assumptions.

V. IMPLEMENTATION

The following provides guidance to applicants and licensees regarding the staff's plans for using this appendix to SRP Section 15.4.9.

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 discussed herein are contained in the reference regulatory guides.

VI. REFERENCES

1. 10 CFR Part 100, Paragraph 11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

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

3. Appendix A, SRP Section 15.6.5, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident (Containment Leakage Contribution)."