U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

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

REVIEW RESPONSIBILITIES

Primary - ~~Accident Evaluation Branch (AEB)~~Emergency Preparedness and Radiation Protection Branch (PERB)[1]

Secondary - ~~Core Performance Branch (CPB)~~Reactor Systems Branch (SRXB)[2]

I. AREAS OF REVIEW

The ~~AEB~~PERB[3] review under this appendix to Standard Review Plan (SRP)[4] 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.

The standard design certification applicant may make reasonable assumptions regarding certain site parameters such as χ/Q and exclusion area and low population zone characteristics.[5]

Review Interfaces

The PERB will coordinate other branch evaluations that interface with the overall review of the radiological consequences of a control rod drop accident as follows:[6]

A secondary review is performed by the ~~CPB~~SRXB[7] and the results are used by ~~AEB~~PERB[8] in the overall evaluation of the accident analysis. The core response aspects of the accident are reviewed by the ~~CPB~~SRXB as part of its primary review responsibility for SRP Section 15.4.9.[9] 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~~SRXB based on the SRP Section 15.4.9 review.[10]

The parameters in the atmospheric diffusion and dose calculation models are reviewed by PERB as part of its primary review responsibility for SRP Section 15.6.5, Appendix A.[11]

The exclusion area and the low population zone are reviewed by the Civil Engineering and Geosciences Branch (ECGB) as part of its primary review responsibility under SRP Sections 2.1.1, 2.1.2, and 2.1.3.[12]

The site characteristics assumed by the design certification applicant will be reviewed by the ECGB as part of its primary review responsibility for SRP Section 2.1.1.[13]

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 the exposure guideline values in 10 CFR Part 100, paragraph 11 ~~(Ref. 1)~~.[14] "Well within" is defined as 25% of the 10 CFR Part 100 exposure guideline values or 750 mSv (75 rem)[15] for the thyroid and 60 mSv (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)~~.[16]

Technical Rationale[17]

The technical rationale for application of these acceptance criteria to reviewing the radiological consequences of a control rod drop accident is discussed in the following paragraphs:

> Compliance with 10 CFR 100.11 requires that the exclusion area and the low population zone be defined based on assurances that specified limits will not be exceeded for radiation doses to individuals at the outer boundaries of those regions for postulated fission product releases.

This requirement applies to this appendix because rod drop accidents are included among the potential accidents for which fission product releases are postulated. Review under SRP Section 15.4.9 focuses on the extent of fuel damage to allow a determination of the source term to be used by the Appendix A reviewer in evaluating compliance with 10 CFR 100.11. Guidance for determining the acceptability of the source term and dose calculations is found in Regulatory Guide 1.77.

Meeting this requirement provides assurance that offsite radiation doses from a BWR rod drop accident are well within the guideline doses specified in 10 CFR 100.11.[18]

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 safety analysis report (SAR)[19] 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~~SRXB.[20] 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~~SRXB.[21] The following assumptions regarding the plant condition and release and transport of radioactivity are used in the independent ~~AEB~~PERB[22] 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)~~.[23]

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~~SRXB[24] should be requested to review analyses which propose that fuel melting is not likely to result in significant releases prior to ~~MSIV~~ main steam isolation valve (MSIV)[25] 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~~)~~[26] 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 (χ/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)~~under Appendix A of SRP Section 15.6.5.[27]

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.

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[28]

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~~PERB[29] staff's safety evaluation report (SER):[30]

> 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~~ engineered safety feature[31] 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.

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.[32]

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.

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52.[33] 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.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section.[34]

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)."~~[35]

SRP Draft Section 15.4.9

Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	Current PRB name and abbreviation	Changed the primary review branch name to Emergency Preparedness and Radiation Protection Branch and the abbreviation to PERB.
2.	Current secondary review branch name and designation	Changed the secondary review branch name to Reactor Systems Branch and the abbreviation to SRXB.
3.	Current PRB designation	Changed the PRB designation to PERB.
4.	Editorial	Provided "SRP" as an abbreviation for "Standard Review Plan."
5.	SRP-UDP format item	Added review area item applicable to standard design certification applicant.
6.	SRP-UDP format item	Added a subsection titled Review Interfaces and inserted the standard introductory paragraph.
7.	Current SRB abbreviation	Changed SRB abbreviation to SRXB.
8.	Current PRB designation	Changed the PRB designation to PERB.
9.	SRP-UDP format item	Added identification of the SRP Section guiding the interfacing review. Updated the review branch abbreviation.
10.	SRP-UDP format item	Added identification of the SRP Section guiding the interfacing review. Updated the review branch abbreviation.
11.	SRP-UDP format item	Added review interface for item included in Review Procedures as number 14.
12.	SRP-UDP format item	Added interface for review of exclusion area and low population zone boundaries.
13.	SRP-UDP format item	Added an interface for the review of the site parameter of the standard design certification applicant.
14.	SRP-UDP format item	Eliminated obvious reference.
15.	SRP-UDP format item	Converted rems to metric equivalent.
16.	SRP-UDP format item	Eliminated obvious reference.
17.	SRP-UDP format item	Added new section heading, Technical Rationale. Inserted the standard introductory paragraph.
18.	SRP-UDP format item	Added the technical rationale for 10 CFR 100.11.
19.	Editorial	Defined "SAR" as "safety analysis report."
20.	Updated review branch initials	Updated SRB abbreviation to SRXB.

Item	Source	Description
21.	Updated review branch initials	Updated SRB abbreviation to SRXB.
22.	Current PRB designation	Changed the PRB designation to PERB.
23.	SRP-UDP format item	Eliminated obvious reference.
24.	Updated review branch initials	Updated SRB abbreviation to SRXB.
25.	Editorial	Spelled out "MSIV" as "main steam isolation valve."
26.	Editorial	Defined MSIV in note 25 above.
27.	Editorial	Inserted the SRP Section number in the text. The citation has been eliminated from the list of References since it is not usual to include the document in hand in a list of reference documents.
28.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
29.	Current PRB designation	Changed the PRB designation to PERB.
30.	Editorial	Provided "SER" as initialism for "safety evaluation report."
31.	Editorial	Spelled out "EFS" as "engineered safety feature."
32.	SRP-UDP Format Item, Implement 10 CFR 52 Related Changes	To address design certification reviews a new paragraph was added to the end of the Evaluation Findings. This paragraph addresses design certification specific items including ITAAC, DAC, site interface requirements, and combined license action items.
33.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard sentence to address application of the SRP section to reviews of applications filed under 10 CFR Part 52, as well as Part 50.
34.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
35.	Editorial	Eliminated the Reference to another section of the SRP.

SRP Draft Section 15.4.9

Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
	No Integrated Impacts were incorporated in this SRP Section.	