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(Formerly NUREG-75/087)

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)~~Reactor Systems Branch (SRXB)[1]

Secondary - ~~Accident Evaluation Branch (AEB)~~Emergency Preparedness and Radiation Protection Branch (PERB)[2]

I. AREAS OF REVIEW

The ~~CPB~~SRXB[3] evaluates the consequences of a control rod drop accident in a boiling water reactor (BWR) in the area of physics. The ~~CPB~~SRXB[4] 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.

An evaluation for the advanced boiling water reactor (ABWR) is described in Reference 4. The evaluation was conducted by the staff because it was the first application involving an ABWR standard design with hypothetical site boundaries, thereby establishing a reference for comparing future applications.[5]

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

Review Interfaces

SRXB will coordinate other branch evaluations with the overall review of this Standard Review Plan (SRP) section, as follows:[6]

1.[7] The ~~AEB~~PERB,[8] 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~~SRXB[9] from the reactor core analyses as the source for dose calculations. The evaluation finding provided is as indicated in the attached appendix.

2.[10] 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~~ by the Instrumentation & Controls Branch (HICB) as part of its primary review responsibility for[11] SRP Sections 7.2 and 7.3.

II. ACCEPTANCE CRITERIA

~~CPB~~SRXB[12] acceptance criteria are based on meeting the requirements of General Design Criterion 28 (GDC 28)[13] ~~(Ref. 2)~~[14] 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/g~~m~~[15] 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 (Reference 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/g~~m~~ at any axial location for zero or low power initial conditions, and fuel cladding dryout for rated power initial conditions.

Technical Rationale

The technical rationale for application of acceptance criteria for rod drop accidents is discussed in the following paragraphs:[16]

Compliance with GDC 28 requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase. Such a design ensures that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to cause serious impairment of core cooling capability.

GDC 28 is applicable to SRP Section 15.4.9 because this section involves the review of reactivity excursions in BWRs resulting from control rod drop accidents. The review provides an assessment of the transient fuel enthalpy to determine whether cladding failure or fuel rupture is predicted to occur and, if so, to what extent. This review establishes both the coolability of the core after the transient and the source term used to evaluate the radiological consequences of the accident. It also determines the maximum reactor coolant pressure to ensure that stress limits for the reactor pressure vessel are not exceeded.

Meeting the requirements of GDC 28 provides a level of assurance that fuel damage and reactor vessel pressure will not be excessive in the event of a control rod drop accident in a BWR.[17]

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~~ Chapter[18] 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~~ ensures[19] 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~~ ensures 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/g~~m~~ 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.

4. For ABWR reviews, the reviewer compares the applicant's safety analysis report (SAR) with assumptions used by the staff to compute rod drop accident doses and with the resultant radiological consequences. Thus, the reviewer either confirms that the applicant's design would produce similar results or notes significant differences.[20]

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.[21]

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided, and ~~his~~the[22] 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:
>
> 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/g~~m~~, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molte~~d~~n[23] $U0_2$ 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.

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.[24]

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.

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.[25] 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.[26]

VI. REFERENCES

1. "Rod Drop Accident Analysis for Large Boiling Water Reactors," NED0-10527, General Electric Company, March 1972; Supplement 1 to NED0-10527, July 1972; and Supplement 2 to NED0-10527, 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.

4. NUREG-1503, ABWR Final Safety Evaluation Report, Section 15.4.1, "Control Rod Drop Accidents," July 1994.[27]

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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 primary review branch name and designation	Changed PRB to SRXB.
2.	Current secondary review branch name and designation	Changed SRB to PERB.
3.	Current primary review branch designation	Changed PRB to SRXB.
4.	Current primary review branch designation	Changed PRB to SRXB.
5.	**Integrated Impact 1357**	Added paragraph discussing the evaluation process for BWR control rod drop.
6.	SRP-UDP format item	Added "Review Interfaces" and lead-in paragraph to AREAS OF REVIEW.
7.	SRP-UDP format item	Created separate paragraph "1" under "Review Interfaces."
8.	Current secondary review branch designation	Changed SRB to PERB.
9.	Current primary review branch designation	Changed PRB to SRXB.
10.	SRP-UDP format item	Created separate paragraph "2" under "Review Interfaces."
11.	Current review branch name and designation	Added name and designation of review interface branch (HICB) for SRP Sections 7.2 and 7.3.
12.	Current primary review branch designation	Changed PRB to SRXB.
13.	Editorial modification	Provided initialism for "General Criterion 28."
14.	SRP-UDP format item	Deleted unnecessary citation of "Ref. 2."
15.	Editorial correction	Corrected abbreviation for gram (global for this SRP section).
16.	SRP-UDP format item	Added "Technical Rationale" and lead-in paragraph to ACCEPTANCE CRITERIA.
17.	SRP-UDP format item	Added technical rationale related to GDC 28.
18.	Editorial correction	Changed "SRP Section 7" to "SRP Chapter 7."
19.	Editorial correction	Changed "assures" to "ensures" (global for this SRP section).
20.	**Integrated Impact 1357**	Added review procedure for ABWR rod drop accidents.

Item	Source	Description
21.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
22.	Editorial change	Replaced "his" with "the."
23.	Editorial correction	Corrected spelling by changing "molted" to "molten."
24.	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.
25.	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.
26.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
27.	**Integrated Impact 1357**	Added ABWR FSER Section 15.4.1 to REFERENCES.

SRP Draft Section 15.4.9
Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
1357	Add the results of the staff's safety review of the ABWR rod drop accident to establish a reference for future applications.	I. AREAS OF REVIEW, second paragraph. III. REVIEW PROCEDURES, paragraph 4. VI. REFERENCES, Reference 4.