U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

15.4.9 SPECTRUM OF ROD DROP ACCIDENTS (BWR)

REVIEW RESPONSIBILITIES

Primary - Organization responsible for review of transient and accident analyses for PWRs/BWRs

Secondary - None

I. AREAS OF REVIEW

The specific areas of review are as follow:

1. The reviewer evaluates the consequences of a control rod drop accident in a boiling water reactor (BWR) in the area of physics. The reviewer covers the applicant's description of the occurrences that lead to the accident, initial conditions, rod patterns and worth, safety features designed to limit the amount of reactivity, the rate at which reactivity can be added to the core, and methods for analyzing the accident. Reference 5 is on control rod drop accident analysis generally.

   The review also examines potential fission product releases from a control rod drop accident. These releases contribute to the source term in analyses for determining compliance with dose limits specified in either 10 CFR 100.11 or 10 CFR 50.67. The staff conducted an evaluation of the advanced boiling water reactor (ABWR) described in Reference 2 because it was the first application for an ABWR standard design with hypothetical site boundaries establishing a reference for future applications.

   This review applies to BWRs only.

Revision 3 - March 2007

2. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

   For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

Other SRP sections interface with this section as follows:

1. General information on transient and accident analyses is provided in SRP Section 15.0.

2. Design basis radiological consequence analyses associated with design basis accidents are reviewed under SRP Section 15.0.3.

3. Reactivity coefficients and control rod worths are reviewed under SRP Section 4.3.

4. Relevant thermal-hydraulic analyses are reviewed under SRP Section 4.4.

5. The applicant's determination of the reactor trip delay time (i.e., the time elapsed between when 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.

The specific acceptance criteria and review procedures are contained in the reference SRP sections.

II. ACCEPTANCE CRITERIA

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations are as follows for the review described in this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

1. General Design Criterion (GDC) 13, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

2. Acceptance criteria are based on GDC 28 requirements as to the effects of postulated reactivity accidents that result in neither damage to the reactor coolant pressure boundary greater than limited local yielding nor result in sufficient damage to impair significantly core cooling capacity.

Regulatory positions and specific guidelines necessary to meet the relevant GDC 28 requirements are in SRP Section 4.2.

The maximum reactor pressure during any portion of the assumed excursion should be less than the value that causes stress to exceed the "Service Limit C" as defined in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

3. 10 CFR 100.11 and 10 CFR 50.67 establish radiation dose limits for individuals at the boundary of the exclusion area and at the outer boundary of the low population zone. The fission product inventory released from all failed fuel rods is an input to the radiological evaluation to SRP section 15.0.3. SRP Section 4.2 describes fuel rod failure mechanisms. Guidance for calculating radiological consequences is in Regulatory Guides 1.183 and 1.195.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this SRP section is discussed in the following paragraphs:

1. GDC 13 requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to assure adequate safety, and of controls that can maintain these variables and systems within prescribed operating ranges.

   GDC 13 applies to this section because the reviewer evaluates the sequence of events, including automatic actuations of protection systems, and manual actions, and determines whether the sequence of events is justified, based upon the expected values of the relevant monitored parameters and instrument indications.

2. GDC 28 requires reactivity control system design with appropriate limits on the potential reactivity amount and rate increases so the effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding nor disturb the core, its support structures, or other reactor pressure vessel internals sufficiently for serious impairment of core cooling capability.

3. GDC 28 applies to this section because it is for review of BWR reactivity excursions caused by control rod drop accidents. The review assesses the transient fuel enthalpy for whether cladding failure or fuel rupture is predicted and to what extent, establishes both the coolability of the core after the transient and the source term for evaluation of radiological consequences, and determines the maximum reactor coolant pressure so stress limits for the reactor pressure vessel are not exceeded.

4. GDC 28 requirements provide assurance that fuel damage and reactor vessel pressure will not be excessive in a BWR control rod drop accident.

III. REVIEW PROCEDURES

The reviewer will select material from the procedures described below, as may be appropriate for a particular case.

These review procedures are based on the identified SRP acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

The reviewer should be guided by the general considerations in the introductory section to this SRP chapter. Initial conditions and assumptions defining the limiting scenario for each of the subsection II acceptance criteria (e.g. fuel rod failure, peak pressure, long-term cooling) and maximizing fuel rod failures due to each failure mechanism (e.g. pallet/cladding mechanical interaction versus dry-out) may differ.

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

   A. The reviewer verifies whether the applicant considers for this event a spectrum of initial conditions that cover the range of time-in-cycle and initial power levels.

   B. The reviewer verifies the use of maximum expected individual control rod worths. The nominal control rod withdrawal pattern and abnormal patterns not precluded by an instrumentation system accepted under SRP Chapter 7 review must be considered in the development of control rod worth criteria.

   C. The reviewer determines whether an acceptable and conservative function describes the control rod worth as a function of control rod position and whether the control rod position as a function of time is suitably conservative.

   D. The reviewer determines whether conservative reactivity coefficients, notably the Doppler, are compatible with those described in SRP Section 4.3.

   E. The reviewer ensures that the scram action is represented conservatively in the integral scram worth curve (SRP Section 4.3) and in the scram delay time.

   F. The reviewer checks the analytical methods for previous review and approval. The reviewer also may perform an independent calculation. 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 for compliance with the acceptance criterion in subsection II of this SRP section (the reviewer may do an audit calculation when appropriate).

3. For the second acceptance criterion the number of fuel rods with clad failure is determined (for the SRP section 15.0.3 reviewer evaluating the radiological consequences of the control rod drop accident) by the following procedure:

   A. The reviewer determines whether the transient critical power ratio is computed by an acceptable technique (reviewed either previously or *de novo* during this review) for analyses using at-power conditions.

B. The reviewer must determine the number of failed rods for the radiological evaluation. The number of fuel rod failures for each of the failure mechanisms addressed in SRP Section 4.2 must be combined.

C. The reviewer determines the acceptability of the time-dependent steaming and activity releases from each potential release path (condenser, etc.). Each scenario should be investigated in combination and separately for the most severe release path.

4. For ABWR reviews, the reviewer compares the applicant's safety analysis report to staff assumptions for computing rod drop accident doses and to the radiological consequences. Thus, the reviewer confirms that the applicant's design would produce similar results or notes significant differences. The review also must evaluate the applicant's ability to satisfy the coolability criteria (See SRP Section 4.2).

5. For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

   For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

The staff concludes that the rod drop accident analysis is acceptable and meets GDC 13 and GDC 28 requirements. This conclusion is based on the following findings:

The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation was available, and that actuations of protection systems, automatic and manual, occurred at values of monitored parameters that were within the instruments' prescribed operating ranges.

The applicant meets GDC 28 requirements as to prevention of postulated reactivity accidents that could result in damage to the reactor coolant pressure boundary greater than limited local yielding or damage that impairs core cooling capability significantly. The staff has evaluated the applicant's analysis of the assumed control rod drop accident and finds the assumptions, calculation techniques, and consequences acceptable. As the calculations predict peak fuel temperatures below melting conditions, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten $U0_2$ presumably did not occur. The pressure surge results 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 are sufficiently conservative in both the initial assumptions and analytical models to maintain primary system integrity.

For DC and COL reviews, the findings will also summarize (to the extent that the review is not discussed in other SER sections) the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable, and interface requirements and combined license action items relevant to this SRP section.

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications submitted six months or more after the date of issuance of this SRP section, unless superseded by a later revision.

VI. REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, "Instrumentation and Control."

2. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 28, "Reactivity Limits."

3. NUREG-1503, ABWR Final Safety Evaluation Report, Section 15.4.1, "Control Rod Drop Accidents," July 1994.

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

5. "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.
