U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

15.4.8 SPECTRUM OF ROD EJECTION ACCIDENTS (PWR)

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the 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 ejection accident for the potential damage to the reactor coolant pressure boundary and for whether the fuel damage from such an accident could impair cooling water flow. 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 available, the rate at which reactivity can be added to the core, and methods for analyzing the accident.

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The review also examines potential fission product releases from a rod ejection 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. This review applies to pressurized water reactors only.

2. <u>COL Action Items and Certification Requirements and Restrictions</u>. COL action items may be identified in the NRC staff's final safety evaluation report (FSER) for each certified design to identify information that COL applicants must address in the application. Additionally, DCs contain requirements and restrictions (e.g., interface requirements) that COL applicants must address in the application. For COL applications referencing a DC, the review performed under this SRP section includes information provided in response to COL action items and certification requirements and restrictions pertaining to this SRP section, as identified in the FSER for the referenced certified design.

<u>Review Interfaces</u>

The listed SRP sections interface with this section as follows:

1. Section 4.3: review of the reactivity coefficient and control rod worths.

2. Section 4.4: review of the relevant thermal-hydraulic analyses.

3. Sections 7.2 and 7.3: review of 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).

4. For COL reviews of operational programs, the review of the applicant's implementation plan is performed under SRP Section 13.4, "Operational Programs."

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

II. <u>ACCEPTANCE CRITERIA</u>

<u>Requirements</u>

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. 10 CFR 52.47(a)(1)(vi), as it relates to ITAAC (for design certification) sufficient to assure that the SSCs in this area of review will operate in accordance with the certification.

2. 10 CFR 52.97(b)(1), as it relates to ITAAC (for combined licenses) sufficient to assure that the SSCs in this area of review have been constructed and will be operated in conformity with the license and the Commission's regulations.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for review described in Subsection I of 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 meeting 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 sufficient damage to impair significantly core cooling capacity.

   Regulatory positions and specific guidelines necessary to meet the relevant requirements of GDC 28 are in Regulatory Guide 1.77 and SRP Section 4.2.

   The maximum reactor pressure during any portion of the assumed excursion should be less than the value that result in stresses that 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 under SRP Section 15.4.8, Appendix A. 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 requirements and/or SRP acceptance criteria to the areas of review addressed by this SRP section ids 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 systems designed with appropriate limits on potential reactivity increases so the effects of a rod ejection accident can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to impair the core cooling capability.

   GDC 28 requirements apply to this section because the reviewer evaluates the maximum reactor pressure during any portion of the transient corresponding to a rod ejection. ASME Codes provide guidance for the acceptability of anticipated accident pressure. The review also examines the extent of fuel damage from a rod ejection accident. Regulatory Guide 1.77 and SRP Section 4.2 provide guidance for acceptability of anticipated core damage.

   This criterion provides assurance that the capability to bring the reactor to a safe shutdown condition will not be impaired by a control rod ejection accident.

3. 10 CFR 100.11 or 10 CFR 50.67 requires that the exclusion area and the low population zone be defined by assurances that specified limits for postulated fission product releases will not be exceeded in radiation doses to individuals at the outer boundaries of those regions.

   These requirements apply to this section because rod ejections are included among the potential accidents for which fission product releases are postulated. Review under this SRP section determines the source term used by the reviewer for Appendix A.

   These requirements provide assurance that offsite radiation doses from pressurized water reactor rod ejection accident will not exceed guideline doses specified in 10 CFR 100.11 or 10 CFR 50.67.

III. <u>REVIEW PROCEDURES</u>

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

For each area of review specified in subsection I of this SRP section, the review procedure is identified below. These review procedures are based on the identified SRP acceptance criteria. For deviations from these specific acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the relevant NRC requirements identified in subsection II.

Initial conditions and assumptions for the limiting scenario with respect to each Section II criteria (e.g. fuel failure, pressure, coolability) may be different. Further, limiting conditions for each fuel failure mechanism (e.g. DNB or PCM1) may be different. The review needs to consider these effects.

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

A. A spectrum of initial conditions, which must include zero, intermediate, and full-power, is considered at the beginning and end of a reactor fuel cycle for examination of upper bounds on possible fuel damage. At-power conditions should include the uncertainties in the calorimetric measurement.

B. From the initial conditions, considering all possible control rod patterns allowed by technical specification/core operating limit report power-dependent insertion limits, the limiting rod worths are determined. Where confirmation is necessary, the reviewer may calculate, as an audit, the worth of limiting rods.

C. Reactivity coefficient values of 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 Section 4.3. The Doppler and moderator coefficients are the two of most interest. If there is no three-dimensional space-time calculation, the reactivity feedback must be weighted conservatively to account for the variation in the missing dimension(s).

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 reviewed under SRP Section 7.2. Control rod worth is checked by the reviewer for consistency with the review under SRP Section 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 reviewed and approved by the staff. Otherwise, he must do a *de novo* review. Alternatively, the reviewer may audit several calculations, using methods acceptable to the staff (or staff consultants). The reviewer's primary concern is how well the elements of the analytical model represent the true three-dimensional problem. The reviewer also checks feedback mechanisms, number of delayed neutron groups, two-dimensional representation of fuel element distribution, primary flow treatment, and scram input.

2. For acceptance criterion II.2, the number of fuel rods with clad failure is determined (for use by the Appendix A reviewer in evaluating radiological consequences of the rod ejection accident) by the following procedure:

A. The reviewer determines whether an acceptable procedure for calculating a departure from nucleate boiling (DNB) condition during the reactivity excursion is used. This determination may be done by reference to previous cases for the same nuclear steam supply system vendor. If no approved technique is available (*e.g.*, the first project using a new or substantially revised model), the reviewer must perform a separate detailed review which may be documented separately in a topical report. DNB must be calculated in accordance with the criteria reviewed and accepted under SRP Section 4.4. Typically, the criteria define a departure from nucleate boiling ratio (DNBR) less than 1.30 when DNB correlations like W-3 (Reference 5) or BAW-2 (Reference 7) are used.

B. The reviewer must determine the total number of failed rods used in the radiological evaluation. The number of fuel rod failures due to each failure mechanism addressed in SRP Section 4.2 must be combined.

C. The reviewer determines the acceptability of the time-dependent activity releases from both containment leakage and plant cool-down (steaming/release via atmospheric dump valves). Each scenario should be investigated in combination and separately for the most severe release path.

3. For reviews of DC and COL applications under 10 CFR Part 52, the reviewer should follow the above procedures to verify that the design set forth in the safety analysis report, and if applicable, site interface requirements meet the acceptance criteria. For DC applications, the reviewer should identify necessary COL action items. With respect to COL applications, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit, or other NRC-approved material, applications, and/or reports.

   After this review, SRP Section 14.3 should be followed for the review of Tier I information for the design, including the postulated site parameters, interface criteria, and ITAAC.

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 analysis of the rod ejection accidents is acceptable and meets 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 for prevention of postulated reactivity accidents that could result in damage to the reactor coolant pressure boundary greater than limited local yielding or result in sufficient damage to impair the core cooling capability significantly. The requirements are met by a demonstration of compliance with the regulatory positions of Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors." The staff has evaluated the applicant's analysis of the assumed control rod ejection accident and finds the assumptions, calculation techniques, and consequences acceptable. As the calculations demonstrate 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, "Nuclear Power Plant Components," of the ASME Boiler and Pressure Vessel Code) for the maximum control rod worths assumed. The staff believes that the calculations are sufficiently conservative, both in 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 docketed six months or more after the date of issuance of this SRP section, unless superceded by a later revision.

VI. REFERENCES

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

2. 10 CFR Part 50, Appendix A, GDC 28, "Reactivity Limits."

3. 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

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

5. NRC Inspection Manual Chapter IMC-2504, "Construction Inspection Program - Non-ITAAC Inspections," issued April 25, 2006.

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

7. 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). (See also BAW-10000 and BAW-10036.)

8. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," Jour. Nuclear Energy, Volume 21, 241-248 (1967).

PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the draft Standard Review Plan are covered by the requirements of 10 CFR Part 50 and10 CFR Part 52, which were approved by the Office of Management and Budget, approval number 3150-0011 and 3150-0151.

PUBLIC PROTECTION NOTIFICATION

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

SRP Section 15.4.8
Description of Changes

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in Draft Revision 3, dated April 1996, of this SRP section. See ADAMS accession number ML052070721.

In addition, this SRP section was administratively updated in accordance with NRR Office Instruction LIC-200, Revision 1, "Standard Review Plan (SRP) Process." The revision also adds standard paragraphs to extend application of this updated SRP section to prospective applicant submissions pursuant to 10 CFR Part 52.

The technical changes are incorporated in Revision 3, dated [Month] 2007:

Review Responsibilities - Reflects changes in review branches resulting from reorganization and branch consolidation. Change is reflected throughout the SRP.

I. AREAS OF REVIEW

1. The following new review areas are added: safety features designed to limit the amount of reactivity available, the rate at which reactivity can be added to the core, and methods for analyzing the accident.

2. The following review areas are deleted: 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 rod failure.

3. The option for compliance with source term per 10 CFR 50.67 is added.

4. COL Action Items and Certification Requirements and Restrictions are added.

5. The radiological review is eliminated as a review interface.

6. An COL interface with SRP 13.4, "Operational Programs" is added.

II. ACCEPTANCE CRITERIA

1. Regulatory Guides 1.183 and 1.195 are added.

2. Criteria related to 10 CFR 52.47(a)(1)(vi), for ITAAC (for design certification) sufficient to assure that the SSCs in this area of review will operate in accordance with the certification is added.

3. Criteria related to 10 CFR 52.97(b)(1), for ITAAC (for combined licenses) are added.

4. Compliance with 10 CFR 50.67 added as an option.

III. REVIEW PROCEDURES

1. Initial conditions include possible control rod patterns allowed by technical specification/core operating limit report power-dependent insertion limits.

2. The following specific guidance is eliminated: For each accident, the maximum primary system pressure should be 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. A review procedure for COL applications under 10 CFR Part 52, including ITAAC, is added.

IV. EVALUATION FINDINGS

1 A requirement that the reviewer states the bases for those safety evaluation conclusions is added.

2. Findings for DC and COL are added.

V. IMPLEMENTATION

1. The following is deleted: Implementation schedules for conformance to parts of the method described herein are contained in the referenced regulatory guide.

VI. REFERENCES

1. The following reference is added: NRC Inspection Manual Chapter IMC-2504, “Construction Inspection Program - Non-ITAAC Inspections,” issued April 25, 2006.