U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.3.3 - 15.3.4 REACTOR COOLANT PUMP ROTOR SEIZURE AND REACTOR COOLANT PUMP SHAFT BREAK

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - None

I. AREAS OF REVIEW

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a reactor coolant pump in a pressurized water reactor (PWR) or recirculation pump in a boiling water reactor (BWR). Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. This SRP section is intended to cover both of these accidents.

The review is conducted with radioactivity releases and with the postulated initial and long-term core and reactor conditions that are pertinent to the rotor seizure or broken shaft events, the methods of thermal and hydraulic analysis, the postulated sequence of events including time delays prior to and after protective system actuation, the assumed reactions of reactor system components, the functional and operational characteristics of the reactor protection system in terms of how it affects the sequence of events, and all operator actions required to secure and maintain the reactor in a safe condition.

The results of the applicant's analyses are reviewed to assess fuel damage and to ensure that values of pertinent system parameters are within expected ranges for the type and class of reactor under review. Fuel damage is assessed by the methods described in SRP Section 4.2 (Ref. 13). The system parameters that are evaluated include: core flow and flow distribution (including hydraulic instabilities), channel heat flux (average and hot), minimum critical heat flux ratio

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

(or minimum critical power ratio), departure from nucleate boiling ratio, vessel water level, thermal power, vessel pressure, steam line pressure (BWR), main steam flow (BWR), and feedwater flow (BWR).

The sequence of events described in the SAR is reviewed by RSB. This review is coordinated with Instrumentation and Control Systems Branch (ICSB). The RSB review concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the RSB reviewer requests initiation of a generic evaluation of the new analytical model by CPB.

The values of all parameters used in a new analytical model, including the initial conditions of the core and system, are reviewed. It is the responsibility of the RSB reviewer to contact his counterpart in the Core Performance Branch (CPB) to ensure that the appropriate physics and fuel data have been used in any staff calculations.

In addition, the RSB will coordinate other branches' evaluations that interface with the overall review of the system as follows: The ICSB reviews the instrumentation and controls aspects of the sequence described in the SAR to confirm that reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis as part of its primary review responsibility for SRP Sections 7.2 through 7.5 (Refs. 14 through 17). The CPB performs generic reviews of the thermal-hydraulic computer models used for this transient and also performs, upon request, additional analyses related to these accidents for selected reactor types as part of its primary review responsibility for SRP Section 4.4 (Ref. 12). The Accident Evaluation Branch (AEB) is notified regarding the extent of the fuel failures that are predicted by the analysis. AEB then evaluates the radiological consequences of the event.

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding review branch.

II. ACCEPTANCE CRITERIA

The RSB acceptance criteria for maintaining control rod insertability and core coolability during a RCP rotor seisure or broken shaft event are based on meeting the relevant requirements of the following regulations:

A. General Design Criteria 27 and 28, as they relate to the reactor coolant system being designed with appropriate margin to assure that the capability to cool the core is maintained.

B. General Design Criterion 31, as it relates to the reactor coolant system being designed with sufficient margin to assure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized.

C. 10 CFR Part 100, as it relates to the calculated doses at the site boundary.

The basic objectives of the review of the accident resulting from a rotor seizure or shaft break in a reactor coolant pump are:

1. To identify which of these accidents is the more limiting.

2. To verify that, for the accident, the plant responds in such a way that the criteria regarding fuel damage, radiological consequences, and system pressure are met.

The specific criteria necessary to meet the relevant requirements of GDC 27, 28, and 31 and 10 CFR Part 100 for the rotor seizure and shaft break event are:

1. Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.

2. The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit for PWRs and the CPR remains above the MCPR safety limit for BWRs based on acceptable correlations (see SRP Section 4.4). If the DNBR or CPR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

3. Any activity release must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines.

4. The integrity of the reactor coolant pumps should be maintained, such that loss of a-c power and containment isolation will not result in pump seal damage.

5. The auxiliary feedwater system must be safety grade and, when required, automatically initiated.

6. Tripping of the reactor coolant pumps should be consistent with the resolution to Action Item II.K.3.5 of NUREG-0718 and NUREG-0737.

7. A rotor seizure or shaft break in a reactor coolant pump should not, by itself, generate a more serious condition or result in a loss of function of the reactor coolant system or containment barriers.

8. Only safety-grade equipment should be used to mitigate the consequences of the event. Safety functions should be accomplished assuming the worst single failure of a safety system active component (see Refs. 5 and 6).

9. The ability to achieve long-term coolability of the core should be verified.

10. This event should be analyzed assuming turbine trip and coincident loss of offsite power and coastdown of undamaged pumps.

The applicant's analysis should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 7 through 11 are acceptable. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation. There are certain assumptions regarding important parameters used to describe the initial plant conditions and postulated system failures which should be used. These are listed below:

i. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.

ii. Conservative scram characteristics are assumed, i.e., for a PWR maximum time delay with the most reactive rod held out of the core, and for a BWR a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate.

iii. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The applicants' analyses of the rotor seizure and shaft break events are reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events, from initiation until a stabilized condition is reached, is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent and time at which operator actions are required.

6. That appropriate margin for malfunctions, such as stuck rods (see II.3.b), are accounted for.

If the SAR states that one of the accidents is not as limiting as the other, the reviewer evaluates the justification presented by the applicant. The applicant is to present a quantitative analysis in the SAR of the accident that is determined to be more limiting. For the accident that is found more limiting, the reviewer confirms that the effects of the accident are determined for each mode of operation (e.g., one-, two-, three-, or four-loop) allowed by the technical specifications. Either a separate analysis should be presented or each mode of operation or the effects of each mode should be referenced to the limiting case.

For the more limiting accident, the RSB reviewer, with the aid of the ICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the accident to acceptable levels. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The ICSB review of Chapter 7 of the SAR confirms that the instrumentation and control systems design is consistent with the requirements for safety systems actions for these events.

To the extent deemed necessary, the RSB reviewer evaluates the effect of single active failures of safety systems and components which may alter the course of the accident. This phase of the review uses the system review procedures described in the SRP sections for Chapters 5, 6, 7 and 8 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, CPB is requested to initiate a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by the RSB. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the applicant's analysis are reviewed and compared to the acceptance criteria presented in subsection II regarding the maximum pressure in the reactor coolant and main steam systems. The temporal changes of the neutron power, heat fluxes (average and maximum), reactor coolant system pressure, minimum DNBR (PWR) or CPR (BWR); core and recirculation loop coolant flow rates (BWR), coolant conditions (inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam fractions), steamline pressure, containment pressure, pressure relief valve flow rate, and flow rate from the reactor coolant system to the containment systems (if applicable) during the transient are reviewed. The more important of these parameters (as listed in subsection I of this SRP section) are compared to those predicted for other similar plants to confim that they are within the expected range. The percentage of fuel rods that experience failure is reviewed and AEB is notified regarding the extent of fuel failures predicted by the analysis.

CPB is consulted regarding the acceptance criteria for fuel rod failure and core coolability.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions which should be included in the staff's safety evaluation report (SER).

The staff concludes that the consequences of postulated rotor seizure or broken shaft events meet the requirements set forth in the General Design Criteria 27, 28, and 31 regarding control rod insertability and core coolability, 10 CFR Part 100 guidelines regarding radiological dose at the site boundary, and applicable TMI Action Plan items. This conclusion is based upon the following:

(a) The applicant has demonstrated that the resultant fuel damage was limited such that control rod insertability would be maintained, and that no loss of core cooling capability resulted. The minimum departure from nucleate boiling ratio (DNBR) or critical power ratio (CPR) experienced by any fuel rod was ____, resulting in ____% of the rods experiencing cladding perforation.

(b) The applicant met the requirements of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the postulated accident.

(c) The analyses and effects of pump rotor seizure and shaft breaks, during various modes of operation and with and without offsite power, have been reviewed.

(d) The accidents analyzed were evaluated using a methematical model that has been previously reviewed and found acceptable by the staff.

(e) The parameters used as input to this model were reviewed and found to be suitably conservative.

(f) The radioactivity release has been evaluated using the computer code SARA and a conservative description of the plant response to the accident. A decontamination factor of _____ between the water and steam phases and a X/Q value of ____ sec/m^3 has been used in our evaluation of radiological consequences. The calculated doses are presented in Table ____. Technical specification limits on primary and secondary coolant activities will limit potential doses to a small fraction of the 10 CFR Part 100 exposure guidelines. The potential doses are within 10 CFR Part 100 exposure guidelines even if the accident should occur coincident with an iodine spike.

V. IMPLEMENTATION

The following 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 proposed 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 referenced NUREGs.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 27, "Combined Reactivity Control Systems Capability."

2. 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactivity Limits."

3. ASME Boilure and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

4. "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from the Director, NRR, to NRR Staff" (Issue No. 1), U.S. Nuclear Regulatory Commission, NUREG-0318, November 1976.

5. "Staff Discussion of Twelve Additional Technical Issues Raised by Responses to November 3, 1976 Memorandum from the Director, NRR, to NRR Staff" (Issue No. 22), U.S. Nuclear Regulatory Commission, NUREG-0158, December 1976.

6. F. M. Bordelon, "Calculation of Flow Coastdown after Loss of Reactor Coolant Pump," WCAP-7973, Westinghouse Electric Corporation, August 1970.

7. C. D. Morgan, H. C. Cheatwood, and J. R. Glandermans, "RADAR - Reactor Thermal and Hydraulic Analysis During Reactor Flow Coastdown," BAW-10069, Babcock and Wilcox Company, July 1973.

8. R. H. Stoudt and J. E. Busby, "CADD - Computer Applications to Direct Simulation of Transient Events on Water Reactors," BAW-10080 (nonproprietary) and BAW-10076 (proprietary), Babcock and Wilcox Company, July 1973.

9. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973.

10. R. Linford, "Analytical Methods of Transient Evaluations in the General Electric Boiling Water Reactor," NEDP-10802, General Electric Company, April 1973.

11. Standard Review Plan Section 4.4, "Thermal and Hydraulic Safety."

12. Standard Review Plan Section 4.2, "Fuel System Design."

13. Standard Review Plan Section 7.2, "Reactor Trip System."

14. Standard Review Plan Section 7.3, "Engineered Safety Features System."

15. Standard Review Plan Section 7.4, "Systems Required for Safe Shutdown."

16. Standard Review Plan Section 7.5, "Safety-Related Display Instrumentation."

17. 10 CFR Part 5, Appendix A, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundry."

18. 10 CFR Part 50, Appendix A, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

19. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License."

20. NUREG-0737, "Clarification of TMI Action Plan Requirements."

21. 10 CFR Part 50, Appendix A, General Design Criterion 28, "Reactivity Limits."