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(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.4.4-15.4.5 STARTUP OF AN INACTIVE LOOP OR RECIRCULATION LOOP AT AN INCORRECT TEMPERATURE, AND FLOW CONTROLLER MALFUNCTION CAUSING AN INCREASE IN BWR CORE FLOW RATE

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch

Secondary - None

I. AREAS OF REVIEW

A number of transients that may occur with moderate frequency cause either increased core flow or introduction of cooler or de-borated water into the core. These transients result in an increase in core reactivity due to decreased moderator temperature, moderator boron concentration, or core void fraction. This SRP section is intended to be applicable to all such transients.* Each of these transients should be discussed in individual sections of the applicant's safety analysis report (SAR), as required by the Standard Format (Reference 1).

The specific transients (Table 15-1 of Reference 1) evaluated are:

1. Boiling water reactor (BWR): startup of an idle recirculation pump.
2. BWR: flow controller malfunction causing increased recirculation flow.
3. Pressurized water reactor (PWR) with loop isolation valves; startup of a pump in an initially isolated inactive reactor coolant loop where the rate of flow increase is limited by the rate at which the isolation valves open.
4. PWR without loop isolation valves: startup of a pump in an inactive loop.

The review of the core flow increase transients considers the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transients. The RSB reviewer concentrates on

*Continuous boron dilution is considered in another section of the SRP.

the need for the reactor protection system 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 reviewer initiates a generic evaluation of the new analytical model. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

The predicted results of the transients are reviewed to assure that the consequences meet the acceptance criteria given in subsection II, below. Further, the results of the transients are reviewed to ascertain that the values of pertinent system parameters are within ranges expected for the type and class of reactor under review.

In addition, the RSB will coordinate other branches' evaluations that interface with the overall review of the system as follows: The Instrument and Control Branch (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. The Core Performance Branch (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 Sections 4.2 through 4.4. 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 branch.

II. ACCEPTANCE CRITERIA

The RSB acceptance criteria are based on meeting the relevant requirements of the following regulations:

A. General Design Criteria (GDC) 10 and 20 as it relates to the reactor coolant system being designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences.

B. General Design Criteria 15 and 28 as it relates to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to assure that the pressure boundary will not be breached during normal operations including anticipated operational occurrences.

C. General Design Criterion 26 as it relates to the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences. This is accomplished by assuring that appropriate margin for malfunctions, such as stuck rods, is accounted for.

The basic objectives of the review of the transients described above are:

1. To identify which of the transients are the most limiting.

2. To verify that, for the most limiting transient, the plant responds in such a way that the criteria regarding fuel damage and system pressure are met.

The specific criteria necessary to meet the relevant requirements of the regulations identified above for incidents of moderate frequency are as follows:

(a) Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (Ref. 2).

(b) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remaining 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).

(c) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

(d) An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, fuel failure must be assumed for all rods for which the DNBR or CPR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage mode (see SRP Section 4.2) that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding.

(e) The requirements stated in Regulatory Guide 1.105, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of transients addressed in this SRP section.

(f) The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and should satisfy the guidance stated in Regulatory Guide 1.53.

The applicant's analysis of the most limiting transients should be performed using an acceptable model. If analytical methods which have not been approved are proposed by the applicant, they are evaluated by the staff for acceptability.

The values of parameters used in the analytical model are to be suitably conservative. The following values are considered acceptable:

a. Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainty. An analysis to determine the effects of a flow increase must be made for each allowed mode of operation (i.e., one, two, or three loops initially operating) or the effects referenced to a limiting case.

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

c. 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.

d. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with Regulatory Guide 1.105 as determined by ICSB.

The reviewer shall verify that the protection system (1) initiates automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded for this event, and (2) senses the plant conditions and initiates the operation of systems and components important to safety.

For BWR plants where flow control is part of the reactivity control system, GDC 25, 26, and 28 must be satisfied for this event; otherwise, GDC 25, 26, and 28 are not applicable. Where applicable, GDC 25, 26, and 28 are satisfied if compliance with GDC 10 and 15 is demonstrated.

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 description of the core flow increase transients presented in the SAR is 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 to which operator actions are required.

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

If the SAR states that a particular core flow transient is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. The applicant should present a quantitative analysis in the SAR of the increase in flow transient that is determined to be most limiting. For this transient, the RSB reviewer, with the aid of the ICSB reviewer, reviews the timing of the initiation of protection, engineered safety feature, and other systems needed to limit the consequences of the core flow increase transient to acceptable levels. The RSB reviewer compares the predicted variation of system parameters with various trip setpoints. The ICSB review of Chapter 7 of the SAR confirms that the instrumentation and control system 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 transient. 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 reviewer considers and evaluates the possibility of a single failure that would permit the loop isolation valves to open prior to startup of a pump in an idle loop (for those plants with loop isolation valves). If this could occur, the core flow rate increase would not be limited by the rate at which the valve opens, and the resulting rate of reactivity insertion could be greater than for other transients of this group.

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, a generic review of the model proposed by the applicant is initiated.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by 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 the selected core burnup yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters and fuel data used in the applicant's analysis.

The results of the analysis are reviewed and compared to the acceptance criteria presented in subsection II of this SRP section regarding the maximum pressure in the reactor coolant and main steam systems. The variations with time during the transient 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 system (if applicable) are reviewed. The values of the more important of these parameters for the core flow increase transients are compared to those predicted for other similar plants to see that they are within the range expected.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contain 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):

A number of plant transients can result in a core flow increase. Those that might be expected to occur with moderate frequency are the startup of an idle recirculation pump (BWR); flow controller malfunction causing increasing core flow (BWR); startup of a pump in an inactive reactor coolant loop (PWR); and startup of a pump in an initially isolated inactive reactor coolant pump.* All these postulated transients have been reviewed. It was found that the most limiting with regard to core thermal margins and pressure within the reactor coolant and main steam systems was the ________ transient. This transient was evaluated by the applicant using a mathematical model that has been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative.

The staff concludes that the plant design with regard to transients that result in an increase in coolant flow through the reactor core is acceptable and meets the relevant requirements of General Design Criteria 10, 15, 20, 26, and 28. This conclusion is based on the following:

1. The applicant has met the requirements of General Design Criteria 10, 20, and 26 with respect to demonstrating that the specified acceptable fuel design limits are not exceeded for this event.

2. The applicant has met the requirements of General Design Criteria 15 and 28 with respect to assuring that the design conditions of the reactor coolant pressure boundary are not exceeded because the protection system operates to maintain the maximum pressure within the reactor coolant and main steam system pressures below 110% of the design values.

3. The applicant has met the positions of Regulatory Guide 1.53 as related to the single-failure criterion and Regulatory Guide 1.105 as related to instrument actuations of systems and components important to safety.

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 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.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides.

*The SER should present one statement for all similar transients.

VI. REFERENCES

1. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

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

3. General Design Criterion 10, "Reactor Design."

4. General Design Criterion 15, "Reactor Coolant System Design."

5. General Design Criterion 20, "Protection System Functions."

6. General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

7. General Design Criterion 28, "Reactivity Limits."

8. Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems."

9. Regulatory Guide 1.105, "Instrument Spans and Setpoints."