NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.1.1 15.1.2 15.1.3 15.1.4 DECREASE IN FEEDWATER TEMPERATURE, INCREASE IN FEEDWATER FLOW, INCREASE IN STEAM FLOW, AND INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - None

I. AREAS OF REVIEW

A number of transients which are expected to occur with moderate frequency, and which involve an unplanned increase in heat removal by the secondary system, are covered by this SRP section. Excessive heat removal, i.e., a heat removal rate in excess of the heat generation rate in the core, causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. The power level increase will lead to a reactor trip. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure.

Each of the transients covered by this SRP section should be discussed in individual sections of the safety analysis report (SAR), as required by the Standard Format (Ref. 1). The transients to be evaluated include:

1. Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs)

   a. Feedwater system malfunctions that result in a decrease in feedwater temperature.

   b. Feedwater system malfunctions that result in an increase in feedwater flow.

   c. Steam pressure regulator malfunctions or failures that result in increased steam flow.

2. PWRs Only

a. Inadvertent opening of a steam generator relief or safety valve.

The topics covered in the primary review include: postulated initial core and reactor conditions which are pertinent to feedwater system malfunctions, pressure regulator or pressure relief valve malfunctions, methods of thermal and hydraulic analysis, postulated sequence of events including time delays prior to and after protective system actuation, assumed reactions of reactor system components, 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 transient analysis are reviewed to ensure that the values of pertinent system parameters are within the ranges expected for the type and class of reactor under review. The parameters include: core flow and flow distribution, channel heat flux (average and hot), minimum critical power ratio (MCPR), departure from nucleate boiling ratio (DNBR), vessel water level, thermal power, vessel pressure, steam line pressure (for BWRs), steam line flow (for BWRs), feedwater flow (for BWRs), and reactivity.

The sequence of events described in the SAR for these transients is reviewed by RSB. The RSB reviewer 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 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 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 RSB will coordinate other branch evaluations that interface with the overall review of the transient analyses as follows: The Instrumentation and Control Systems Branch (ICSB) reviews the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems wil function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems as part of its primary review responsibility for SRP Sections 7.2. through 7.5. The Core Performance Branch (CPB) upon request from RSB, reviews the values of the parameters used in the analytical models which relate to the reactor core for conformance to plant design and specified operating conditions; determines the acceptance criteria for fuel cladding damage limits; and reviews the core physics, fuel design, and core thermal-hydraulics data used in the SAR analysis as part of its primary review responsibility for SRP Sections 4.2 through 4.4. The Accident Evaluation Branch (AEB) using fuel damage results provided by RSB evaluates the radiological consequences associated with the fuel failure. The review of the Technical Specifications is coordinated and performed by the Licensing Guidance Branch (LGB) as part of its primary review responsibility for SRP Section 16.0.

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 are based on meeting the relevant requirements of the following regulations:

A. General Design Criterion 10, 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 Criterion 15, 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 breeched 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 are accounted for.

D. TMI Action Plan items II.E.5.1 and II.E.5.2 of NUREG-0718 as they relate to assuring that any design modifications that result from the resolution of these Action Plan items are properly accounted for in the analyses.

The basic objectives of the review of the transients which result from an increase in heat removal are:

1. To identify which of the moderate-frequency* transients that result in increased heat removal are the most limiting.

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

The specific criteria necessary to meet the requirements of GDC 10, 15, and 26 for incidents of moderate frequency are:

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

2. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs and the CPR remains above the the MCPR safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).

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

4. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered and is

*The term "moderate-frequency" is used in this SRP section in the same sense as in the descriptions of design and plant process conditions in References 9 and 10.

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 model (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.

5. To meet the requirements of General Design Criteria 10, 15 and 26 the positions of Regulatory Guide 1.105, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of transient addressed in this SRP section.

6. 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 shall satisfy the positions of Regulatory Guide 1.53 (Ref. 12).

The applicant's analysis of transients caused by excessive heat removal should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 5 through 8 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 initiates an evaluation.

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

a. 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.

b. 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.

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. Compliance with Regulatory Guide 1.105 is determined by ICSB.

III. REVIEW PROCEDURES

The procedures below are used for 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 are used in the analysis and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

RSB reviews the applicant's description of the transients caused by excessive heat removal with specific attention to the occurrences that lead 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 (per II.3.b) are accounted for.

If the SAR states that a particular transient involving an increase in heat removal is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. The applicant is to present a quantitative analysis in the SAR of the increase-in-heat-removal 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 those protection, engineered safety, and other systems needed to limit the consequences of the transient to an acceptable level. 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 systems and components which may affect the course of the transient. This phase of the review uses the system review procedures described in the SRP sections for Chapters 4, 5, 6, 7, 8 and 9 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, a generic review of the models 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 values of reactivity coefficients and control rod worths used by the applicant in his analysis, and the variations of moderator temperature, void, and Doppler coefficients of reactivity with core life. The reviewer evaluates the justification provided by the applicant to show that the core burnup selected yields the minimum margins. CPB is consulted regarding the values of the reactivity parameters 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, as listed in subsection I of this SRP section, are compared to those predicted for other similar plants to see that they are within the range expected.

The NRC has undertaken a program to reduce the sensitivity of B&W plants to feedwater transients (Items II.E.5.1 and II.E.5.2, NUREG-0660 and 0718). When this program is complete, the RSB reviewer, with the aid of other branches as appropriate, should incorporate the program results into the review of this SRP section.

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:

> A number of plant transients can result in an unplanned increase in heat removal by the secondary system. Those that might be expected to occur with moderate frequency can be caused by feedwater system or pressure regulator malfunctions or the inadvertent opening of a steam generator safety or relief valve (PWR only). All of these postulated transients have been reviewed. It was found that the most limiting in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the _________ transient.
>
> The staff concludes that the analysis of transients resulting in an unplanned increase in heat removal by the secondary system that are expected to occur with moderate frequency is acceptable and meets the requirements of General Design Criteria 10, 15, and 26 and TMI Action Plan items II.E.5.1 and II.E.5.2.
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> 1. In meeting GDC 10, 15, and 26 as indicated below we have determined that the applicant's analysis was performed using a mathematical model that had 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. In addition, we have further determined that the positions of Regulatory Guide 1.53 as related to the single failure criterion and Regulatory Guide 1.105 for instruments have also been satisfied.
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> 2. The applicant has met the requirements of GDC 10 and 26 with respect to demonstrating that resultant fuel damage is maintained since the specified acceptable fuel design limits were not exceeded for this event.

3. The applicant has met the requirements of GDC 15 with respect to demonstrating that the reactor coolant pressure boundary limits have not been exceeded by this event and that resultant leakage will be within acceptable limits. This requirement has been met since the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

4. The applicant has met the requirements of GDC 26 with respect to the capability of the reactivity control system to provide adequate control of reactivity during this event while including appropriate margins for stuck rods since the specified acceptable fuel design limits were not exceeded.

5. The applicant has met the requirements of II.E.5.1 and II.E.5.2 by properly accounting for all design modifications in the analysis that has been made as a result of resolution of this item.

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 and NUREGS.

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

4. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

5. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973.

6. "Reference Safety Analysis Report - RESAR-3,"Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report - RESAR-41," Westinghouse Nuclear Energy Systems, December 1973. "Reference Safety Analysis Report - RESAR-3S," Westinghouse Nuclear Energy Systems, July 1975; and "Reference Safety Analysis Report - RESAR-414," Westinghouse Nuclear Energy Systems, October 1976.

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

8. "Standard Nuclear Steam System B-SAR-205," Babcock & Wilcox Company, February 1974.

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

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

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

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

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

14. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident."

15. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses."