NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.6.1 INADVERTENT OPENING OF A PWR PRESSURIZER PRESSURE RELIEF VALVE OR A BWR PRESSURE RELIEF VALVE

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - None

I. AREAS OF REVIEW

The inadvertent opening of a pressure relief valve results in a reactor coolant inventory decrease and a decrease in reactor coolant system pressure. A pressure relief valve, as defined in ANSI B95.1-1972 (Ref. 1), is a pressure relief device which is designed to reclose and prevent further fluid flow after normal conditions have been restored. The effect of the pressure decrease is to decrease the neutron flux (via moderator density feedback). In a pressurized water reactor (PWR), a reactor trip occurs due to low reactor coolant system (RCS) pressure. In a boiling water reactor (BWR), the pressure relief valve discharges into the suppression pool. Normally there is no reactor trip in a BWR. The pressure regulator senses the RCS pressure decrease and partially closes the turbine control valves (TCV) to stabilize the reactor at a lower pressure. The reactor power settles out at nearly the initial power level. The coolant inventory is maintained by the feedwater control system using water from the condensate storage tank via the condenser hotwell.

The review of these transients should consider the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transient.

The sequence of events described in the applicant's safety analysis report (SAR) 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 the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced method has not been previously reviewed, the reviewer initiates a generic evaluation of the new analytical model. 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 transient are reviewed to assure that the consequences meet the acceptance criteria given in subsection II of this SRP section. The analysis results are reviewed to ascertain that pertinent system parameter valves are within ranges expected for the type and class of reactor under review.

The RSB will coordinate other branch evaluations that interface with the overall review of this SRP section as follows: The Instrumentation and Controls Systems Branch (ICSB) reviews the instrumentation and controls aspects of the sequence described to confirm that the 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. Upon request, ICSB will verify the sequence described by the applicant for the safety analysis with regard to automatic actuation, remote sensing, indication, control, interlocks with auxiliary or shared systems, potential bypass modes and the possibility of manual control by the operator. The Power Systems Branch (PSB) upon request from RSB, will verify that the control systems power sources needed to function to mitigate the event are available as required by the applicant's description of the event. The Equipment Qualification Branch (EQB) upon the request of RSB, will verify that the equipment necessary to mitigate the event is qualified for the transient and post-transient environments. The EQB will also identifiy, if requested, equipment whose failure as a result of the initiating event could adversely affect the consequences. The Core Performance Branch (CPB) upon request from RSB, will verify that the core physic data used by the applicant, or by the staff in independent analyses, is the appropriate data to be used as part of its primary review responsibility for SRP Sections 4.2 through 4.4. The CPB will also verify, if requested, that acceptance criteria 1 of SRP Section 4.4 is satisfied throughout the transient.

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 (GDC) 10 (Reference 2), 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 (Reference 3), 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 (Reference 4) as it relates to the reactivity control system to provide adequate control of reactivity changes during manual operations and anticipated transients to assure that the acceptable fuel design limits are not exceeded.

D. TMI Action Plan requirements items No's. II.K.2.8, II.K.3.1, II.K.3.5, II.K.3.16, II.K.3.25 and II.K.3.40 of NUREGs-0718 and -0737 (Ref. 11 and 12).

The general objective in the review of inadvertent primary pressure relief valve opening events is to confirm that the following criteria are met:

1. The consequences of the transient are less severe than the consequences of another transient that results in a decrease of reactor coolant inventory and has the same anticipated frequency classification.

2. The plant responds to the pressure relief valve opening transient 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:

a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (Ref. 5).

b. 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 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 failures 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.

e. To meet the requirements of General Design Criteria 10, 15 and 26, the positions of Regulatory Guide 1.105 (Ref. 9), "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.

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 shall satisfy the positions of Regulatory Guide 1.53 (Ref. 10).

The applicant's analysis of this transient should be performed using an acceptable analytical model. If the applicant proposes to use other analytical methods, which have not been previously reviewed and approved by the staff, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer initiates an evaluation of the new analytical model.

*The term "moderate frequency" is used in this SRP section in the same sense as in the description of design and plant process conditions in References 7 and 8.

The values of the parameters used in the analytical model are to 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 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, 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 applicant's description of the inadvertent pressure relief valve opening transient 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. The following TMI Action Plan (Ref. 11 and 12) items are reviewed to assume compliance with the acceptance criteria:

   a. II.K.2.8. For Babcock and Wilcox designs, the reviewer evaluates the auxiliary feedwater system upgrade and automatic auxiliary feedwater initiation as they relate to determining the auxiliary feedwater performance requirements for this event, if the applicant's evaluation of this transient indicate that the system will be required to function.

b. II.K.3.1. If, as a result of the evaluations performed as required by II.K.3.2, or if the applicant has in the design, an automatic power-operated relief valve isolation system, the reviewer confirms it has been properly accounted for in the analyses.

c. II.K.3.5. The reviewer evaluates the assumption made regarding reactor coolant pump trip to assure that they are consistent and conservatively modeled with respect to the final pump trip criteria which results from resolution of Task Action Plan item II.K.3.5.

d. II.K.3.16. For Boiling Water Reactor designs, the reviewer confirms that the results of the applicant's feasibility study, and, if required, system modifications to reduce the number of challenges to and the number of failure of relief valves, have been properly included in the evaluation of the event.

e. II.K.3.25 and II.K.3.40. If, as a result of the transient, or as a result of loss of A/C power, containment isolation is indicated to occur, the reactor coolant pump component cooling water may be lost. The reviewer evaluates the applicant's submittal to determine that the reactor coolant pump seal integrity is not lost. If it cannot be established that seal integrity is assured, the reviewer assures that the evaluation of this event correctly accounts for seal failure.

If the SAR states that the inadvertent pressure relief valve opening transient is not as limiting as some other similar transient, the reviewer evaluates the justificaton presented by the applicant. If a quantitative analysis of the transient is presented in the SAR, 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 consequence of the transient to acceptable levels. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints.

To the extent deemed necessary, the RSB reviewer evaluates the effects of single active failures of systems and components which may alter the course of the transient. In this phase of the review the system reviews are performed as described in the SRP sections for Chapters 5, 6, 7 and 8 of the SAR. The reviewer considers possible single failures in systems that replenish or maintain the reactor coolant inventory.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam line are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, the reviewer initiates 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 RSB. Of particular importance are the reactivity coefficients and control rod worths used by the applicant in his 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.

The results of the 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 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 temeperature (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. Values of the more important of these parameters for the transient caused by the inadvertent pressure refief valve opening are compared to those predicted for other similar plants to confirm that they are within the expected range.

Upon request from the RSB reviewer, other branches will provide input for the areas of review stated in subsection I of this SRP section. The RSB reviewer obtains and uses the input requested as required to assure that the review procedure is complete.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and that the 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 decrease of reactor coolant inventory. Those that might be expected to occur with moderate frequency are pressure relief valve openings, minor primary pipe breaks, and (in BWRs) loss of feedwater.* 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. This transient was evaluated by the applicant 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. The results of the analysis of the ________ transient showed that the specified acceptable fuel design limits were maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR)** did not decrease below ______ and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.
>
> The staff concludes that the plant design in regards to transients that are expected to occur with moderate frequency and result in a decreased primary coolant inventory is acceptable and meets the relevant requirements of General Design Criteria (GDC) 10, 15 and 26 and the applicable TMI Action Plan items. This conclusion is based on the following:

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

**Minimum critical power ratio (MCPR) for a BWR.

1. 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 the event.

2. 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 pressure.

3. 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 margin for stuck rods since the specific acceptable fuel design limits were not exceeded.

4. In meeting GDC 10, 15 and 26 the staff has determined that the 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 position of Regulatory Guide 1.53 with respect to single failure criterion and Regulatory Guide 1.105 for instruments have also been satisfied.

5. The applicant has met Task Action Plan item [identify item No.] by [describe means used by the applicant to implement the action plan 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.

V. REFERENCES

1. ANSI B95.1-1972, "Terminology for Pressure Relief Devices," American Society of Mechanical Engineers.

2. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."

3. 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant Pressure Boundary."

4. 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

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

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

7. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

8. ANS Trial Use Standard N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

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

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

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

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