U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

15.2.7 LOSS OF NORMAL FEEDWATER FLOW

REVIEW RESPONSIBILITIES

Primary - Organization responsible for review of transient and accident analyses for PWRs/BWRs

Secondary - None

I. AREAS OF REVIEW

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a loss of offsite power (LOOP). The sequence of events for the loss of feedwater transient differs between a boiling water reactor (BWR) and a pressurized water reactor (PWR). A PWR has a backup (auxiliary or emergency) feedwater system while a BWR relies on the emergency core cooling system (ECCS) and reactor core isolation cooling (RCIC) system for backup core cooling. In either case, loss of feedwater flow results in an increase in reactor coolant temperature and pressure which eventually requires a reactor trip to prevent fuel damage. Each event covered in this Standard Review Plan (SRP) section should be discussed in individual sections of the Safety Analysis Report (SAR) or the Design Control Document (DCD), as specified in Regulatory Guide 1.70 and Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

For both PWRs and BWRs, fission product decay heat must be transferred from the reactor coolant system following a loss of normal feedwater flow. This can be accomplished by actuation of one or several of the following systems: steam relief system, steam bypass to the

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condenser, reactor core isolation cooling system (BWR), isolation condenser system (ESBWR), emergency core cooling system (BWR) and auxiliary or emergency feedwater system (PWR).

The specific areas of review are as follows:

1. the sequence of events,

2. the analytical model,

3. the values of parameters used in the analytical model, and

4. the predicted consequences of the transient.

The sequence of events described in the applicant's SAR or DCD is reviewed by both the organization responsible for reactor systems and the organization for the instrumentation and control systems. The reactor systems reviewer concentrates on:

1. the need for the reactor protection system,

2. the engineered safety systems,

3. operator action to secure and maintain the reactor in a safe condition,

4. the role and performance of auxiliary feedwater and the related operator actions, and

5. the role and performance of RCIC and the related operator actions.

The analytical methods are reviewed by the organization responsible for reactor systems 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 reactor systems reviewer requests initiation of a generic evaluation of the new analytical model by the organization responsible for methods and code review.

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

COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

Other SRP sections interface with this section as follows:

1. General information on transient and accident analyses is provided in SRP Section 15.0.

2. Design basis radiological consequence analyses associated with design basis accidents are reviewed under SRP Section 15.0.3. 1.

3. Values of the parameters in the analytical models of the reactor core are reviewed for compliance with plant design and specified operating conditions, acceptance criteria for fuel cladding damage limits are determined, and the core physics, fuel design, and core thermal-hydraulics data in the SAR analysis are reviewed under SRP Sections 4.2, 4.3, and 4.4.

4. Technical specifications are reviewed under SRP Section 16.0.

5. Reliability of the auxiliary feedwater system is reviewed to ensure compliance with the requirements of 10 CFR 50.34(f)(1)(ii), 10 CFR 50.34(f)(2)(xii), and the guidance of TMI Action Item II.K.2.19 of NUREG-0737 under SRP Section 10.4.9.

6. Operational assumptions as factored into related instrumentation and controls for the auxiliary feedwater system used in the analysis are reviewed for appropriateness, under SRP Sections 7.2 through 7.5.

7. Instrumentation and controls aspects of the sequence described in the SAR is reviewed to confirm that reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, interlocks with auxiliary or shared systems and compliance with Regulatory Guide 1.105 under SRP Sections 7.2 through 7.5. Design of the auxiliary feedwater system is also evaluated to determine that: i. the requirements and guidance of 10 CFR 50.34(f)(2)(xii) are met, and ii. the appropriate delay time for auxiliary feedwater initiation is assumed in the analysis.

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

II. ACCEPTANCE CRITERIA

Requirements

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

1. General Design Criterion (GDC) 10, as it relates to the reactor coolant system being designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations including anticipated operational occurrences (AOOs).

2. 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.

3. GDC 15, as it relates to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations including AOOs.

4. GDC 17, as it relates to providing onsite and offsite electric power systems to ensure that SSCs important to safety will function during normal operation, including anticipated operational occurrences. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded during an AOO.

5. GDC 26, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded, including AOOs. This is accomplished by assuring that appropriate margin for malfunctions, such as stuck rods, are accounted for.

6. TMI Action Plan item II.K.2.19 of NUREG-073 and 10 CFR 50.34(f)(1)(ii)[1] and 10 CFR 50.34(f)(2)(xii) as they relate to the performance requirements of the auxiliary feedwater system for the loss of normal feedwater flow event.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for the review described in 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. The basic objective in the review of the loss of normal feedwater transient is to confirm that the following criteria are met:

   A. The plant responds to the loss of feedwater transient in such a way that the criteria regarding fuel damage and system pressure are met.

   B. There is sufficient capacity for long term decay heat removal for the plant to reach a stabilized condition.

   C. The plant protection systems setpoints assumed in the transient analyses are selected with adequate allowance for measurement uncertainties as delineated in Regulatory Guide 1.105.

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(1)For Part 50 applicants not listed in 10 CFR 50.34(f), the applicable provisions of 10 CFR 50.34(f) will be made a requirement during the licensing process.

D. The event evaluation takes into consideration single failures, operator errors, and performance of non-safety related systems that are consistent with regulatory guidelines set forth in RG 1.206.

2. Using the ANS standards as guidance, specific criteria have been developed to meet the relevant requirements of GDCs 10, 13, 15, 17, and 26 for events of moderate frequency and they are as follows:

   A. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values.

   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 SAR (or DCD) Section 4.4), as well as by satisfaction of any other SAFDL that may be applicable to the particular reactor design.

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

   D. To meet the requirements of GDCs 10 and 15, 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.

   E. 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 and GDC 17.

   F. The guidance provided in SECY 77-439, SECY 94-084 and RG 1.206 with respect to the consideration of the performance of non-safety related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems) must be evaluated and verified.

3. The applicant's analysis of the loss of normal feedwater transient should be performed using an acceptable analytical model. If the applicant proposes to use analytical methods which have not been approved, these methods are evaluated by the staff for acceptability. For new generic methods the reviewer requests an evaluation by the appropriate organization for reactor systems.

   The value of 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, unless (a) a different conservatism factor can be justified through the uncertainty methodology and evaluation, or (b) the uncertainty has otherwise been accounted for (see SAR (or DCD) Section 4.4).

C. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, 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.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this SRP section is discussed in the following paragraphs:

1. Compliance with GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

   GDC 10 is applicable to SRP Section 15.2.7 because this section evaluates the loss of normal feedwater flow transient. A part of the evaluation relates to the reactor coolant system being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs. Regulatory Guide 1.105 provides guidance for ensuring that instrument setpoints are initially within and remain within the technical specification limits.

   Meeting the requirements of GDC 10 provides assurance that specified acceptable fuel design limits are not exceeded for the initiating events evaluated in this SRP section involving a decrease in heat removal by the secondary system.

2. 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 sequences 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.

3. Compliance with GDC 15 requires that the reactor coolant system and associated auxiliary, control and protection systems shall be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs.

GDC 15 is applicable to SRP Section 15.2.7 because this section evaluates the consequences of the events of a loss of normal feedwater flow transient that result in a decrease in heat removal by the secondary system with the potential for causing the reactor coolant system pressure to change in response to the increase in reactor coolant temperature.

Meeting the requirements of GDC 15 provides assurance that the design conditions of the reactor coolant pressure boundary are not exceeded for the initiating events evaluated in this SRP section involving a decrease in heat removal by the secondary system.

4. Compliance with GDC 17 requires (in part) that an onsite and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that SAFDLs and design conditions of the reactor coolant pressure boundary are not exceeded as a result of AOOs. GDC 17 is applicable to SRP Section 15.2.7 because the loss of normal feedwater flow transient is an AOO.

   Meeting the requirements of GDC 17 provides assurance that SAFDLs and design conditions of the reactor coolant pressure boundary are not exceeded as a result of a loss of normal feedwater.

5. Compliance with GDC 26 requires that two independent reactivity control systems be provided capable of reliably controlling reactivity changes to ensure that acceptable fuel design limits are not exceeded.

   GDC 26 is applicable to SRP Section 15.2.7 because this section evaluates the consequences of the events of a loss of normal feedwater flow that result in a decrease in heat removal by the secondary system with the potential for causing changes in reactivity within the core that could cause the thermal design criteria for the fuel cladding to be exceeded. SRP 15.2.7 ensures that the thermal margin be sufficient to accommodate these conditions and ensures that the appropriate margins for malfunctions of reactivity controls such as stuck rods are accounted for.

   Meeting the requirements of GDC 26 provides assurance that SAFDLs are not exceeded by ensuring that there is appropriate margin for malfunctions of the reactivity control system.

III. REVIEW PROCEDURES

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

These review procedures are based on the identified SRP acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant’s evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

The procedures below are used for the design certification (DC) application review, the construction permit (CP), operating license (OL), and combined license (COL) applications.

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 loss of normal feedwater flow transient presented by the applicant in the SAR (or DCD) is reviewed by the organization responsible for reactor systems 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 are accounted for.

7. The operation of auxiliary systems that is required.

8. That instrumentation uncertainties of system and operating parameters are appropriately accounted for.

If the SAR (or DCD) states that the loss of feedwater transient is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. If a quantitative analysis of the loss of feedwater transient is presented in the SAR (or DCD), the reactor systems reviewer, with the aid of the instrumentation and control systems reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the loss of feedwater transient to an acceptable level. The reactor systems reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The review of Chapter 7 of the SAR (or DCD) by the organization responsible for instrumentation ans control systems confirms that their design is consistent with the requirements for safety systems actions for these events.

To the extent deemed necessary, the reactor systems reviewer evaluates the effect of single active failures of systems and components which may alter the course of the transient. For new applications, LOOP should not be considered a single failure; loss of feedwater should be analyzed with and without a LOOP in combination with a single active failure. This part of the review uses the procedures described in the SRP sections for Chapters 4, 5, 6, 7, 8, and 9 of the SAR (or DCD).

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 the organization responsible for reactor systems 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 the organization responsible for reactor systems. 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.

The results of the analysis are reviewed, including the effects of the LOOP and the possibility of the event developing into a more serious event (e.g., a stuck open PORV on the pressurizer that could lead to a SBLOCA if not isolated), and compared with the acceptance criteria presented in subsection II of this SRP section regarding maximum pressure in the reactor coolant and main steam systems. The parameters reviewed are:

1. variations with time during the transient of the neutron power,

2. heat fluxes (average and maximum),

3. reactor coolant system pressure,

4. minimum DNBR (PWR) or CPR (BWR),

5. core and recirculation loop coolant flow rates (BWR),

6. coolant conditions (inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam fractions),

7. steamline pressure,

8. containment and suppression pool (if applicable) pressures and temperatures ,

9. pressure relief valve flow rate and quality,

10. pressurizer water volume (in PWRs), and

11. flow rate from the reactor coolant system to the containment system (if applicable).

The more important of these parameters for the loss of normal feedwater transient are compared with those predicted for other similar plants to see that they are within the range expected.

For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

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 has reviewed the analyses of the loss of normal feedwater flow event and concludes that the analyses have adequately accounted for the operation of the plant and were performed using acceptable analytical models.

The staff further concludes that the analyses have demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of the loss of normal feedwater flow.

The staff concludes that the plant design is acceptable with regard to transients resulting from loss of normal feedwater that are expected to occur with moderate frequency and that the predicted response meets the requirements of GDC 10, 13, 15, 17, and 26. This conclusion is based on the following:

The applicant has met the requirements of GDC 10 and 26 with respect to demonstrating that SAFDLs are not exceeded for this event and has met the requirements of GDC 15 with respect to demonstrating that the reactor coolant pressure limits have not been exceeded by this event.

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 parameters used as input to the analytical model were reviewed and found to be suitably conservative and in accordance with the recommendation of Regulatory Guide 1.105. The results of the analysis of the transient showed that cladding integrity was maintained by ensuring that the minimum departure from nucleate boiling ratio (or minimum critical power ratio for a BWR) did not decrease below____and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of their design pressures.

Thus, the applicant has met the requirements of GDC 17, and 26 with respect to demonstrating that SAFDLs are not exceeded for this event.

The applicant has met the positions of Regulatory Guide 1.53, SECY 77-439, SECY 94-084 and RG 1.206 as related to the single-failure criterion and Regulatory Guide 1.105 as related to instrument actuations of systems and components important to safety.

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL 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 submitted six months or more after the date of issuance of this SRP section, unless superseded by a later revision.

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

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. Regulatory Guide RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

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

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

5. 10 CFR 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

6. ANSI/ANS 51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," (replaced ANSI N18.2-1974; reaffirmed 1988; withdrawn 1998).

7. ANSI/ANS 52.1-1983, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," (replaced ANS Trial Use Standard N212-1974; reaffirmed 1988; withdrawn 1998).

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

9. General Design Criterion 13, "Instrumentation and Control."

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

11. General Design Criterion 17, "Electric Power Systems."

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

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

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

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

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

17. SECY-77-439, "Single Failure Criterion."

18. SECY-94-084, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs."