U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

15.3.1-15.3.2 LOSS OF FORCED REACTOR COOLANT FLOW INCLUDING TRIP OF PUMP MOTOR AND FLOW CONTROLLER MALFUNCTIONS

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

A decrease in reactor coolant flow while a plant is at power could result in degraded core heat transfer. An increase in fuel temperature and accompanying fuel damage then could result if specified acceptable fuel design limits (SAFDLs) are exceeded during the transient. This Standard Review Plan (SRP) section covers a number of transients expected to occur with moderate frequency that decrease forced reactor coolant flow rate. Each transient should be addressed in individual sections of the applicant's safety analysis report (SAR) or design control document (DCD), as specified in Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" and Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)." The specific areas of review are as follow:

1. Core thermal and hydraulic transients with partial and complete loss of reactor coolant flow are evaluated, including:

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A. For boiling water reactors (BWRs), partial and complete recirculation pump trips and malfunctions of the recirculation flow controller causing decreasing flow.

B. For pressurized water reactors (PWRs), partial and complete reactor coolant pump trips.

2. A partial loss of coolant flow may be caused by a mechanical or electrical failure in a pump motor, a fault in the power supply to the pump motor, a pump motor trip caused by such anomalies as over-current or phase imbalance, or a failure within the recirculation flow control network (BWR) resulting in decreased flow (if applicable). A complete loss of forced coolant flow may be the result of the simultaneous loss of electrical power to all pump motors.

3. The review includes the postulated initial core and reactor conditions pertinent to the loss of flow transient; the methods of thermal and hydraulic analysis; the postulated sequence of events, including time delays before and after protective system actuation; assumed reactions of reactor system components; the functional and operational characteristics of the reactor protection system affecting the sequence of events; and all operator actions required to secure and maintain the reactor in a safe condition.

4. Results of the applicant's analyses are reviewed for whether values of pertinent system parameters are within expected ranges for the type and class of reactor under review. The system parameters evaluated include core flow and flow distribution, channel heat flux (average and hot), minimum critical heat flux ratio (or minimum critical power ratio), departure from nucleate boiling ratio (DNBR), vessel water level, thermal power, vessel pressure, steam line pressure (BWR), main steam flow (BWR), and feedwater flow (BWR). Results of the applicant's fuel damage analysis are reviewed by the methods described in SRP Section 4.2.

5. The sequence of events described in the SAR or DCD is reviewed by the organization responsible for the review of reactor systems and coordinated with the organization responsible for instrumentation and controls. The reactor systems review concentrates on the need for the reactor protection system, the engineered safety system, and operator action to secure and maintain the reactor in a safe condition.

6. Analytical methods are reviewed for whether the mathematical modeling and computer codes have been reviewed and accepted by the staff. If a referenced analytical method has not been reviewed, the reactor systems reviewer initiates a generic evaluation of the new analytical model.

7. The values of all parameters in a new analytical model, including initial core and system conditions, are reviewed. The reactor systems reviewer is responsible for the use of appropriate physics and fuel data in any staff calculations.

8. <u>COL Action Items and Certification Requirements and Restrictions</u>. 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.

3. 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 under SRP Sections 7.2 through 7.5.

4. Generic reviews of the thermal-hydraulic computer models used for this transient and, as appropriate, additional analyses related to these accidents for selected reactor types are reviewed under SRP Section 4.4.

5. Preoperational tests are reviewed under SRP Section 14.2. The primary reviewer of this section confirms with the lead reviewer of 14.2 that a committment has been made in the SAR to conduct preoperational tests to verify flow coastdown calculations.

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

II. ACCEPTANCE CRITERIA

Requirements

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

1. 10 CFR 50, Appendix A, General Design Criteria (GDCs) 10 and 20 as to design of the reactor coolant system with appropriate margin so SAFDLs are not exceeded during normal operations, including anticipated operational occurrences (AOOs).

2. 10 CFR 50, Appendix A, 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. 10 CFR 50, Appendix A, GDC 15 as to design of the reactor coolant system and its auxiliaries appropriate margin so the pressure boundary is not breached during normal operations, including AOOs.

4. 10 CFR 50, Appendix A, GDC 17 as to onsite and offsite electric power systems so structures, systems, and components (SSCs) important to safety function during normal operation, including AOOs. The safety function for each power system (assuming the other system is not functioning) must be to provide sufficient capacity and capability so SAFDLs and design conditions of the reactor coolant pressure boundary are not exceeded during AOOs.

5. 10 CFR 50, Appendix A, GDC 26 as to the reliable control of reactivity changes so SAFDLs are not exceeded, including during AOOs. This control is accomplished by accounting for appropriate margin for malfunctions (e.g., stuck rods).

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.

The basic objectives of the review of loss of forced reactor coolant flow transients are to identify the most limiting transients and to verify whether, for the most limiting transients, the plant response to the loss of flow transients satisfies fuel damage and system pressure criteria.

The following specific criteria are necessary to meet the regulatory requirements for incidents of moderate frequency:

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

2. Fuel-cladding integrity must be maintained by the minimum DNBR remaining above the 95 percent probability/95 percent confidence DNBR limit for PWRs and the critical power ratio (CPR) remaining above the minimum critical power ratio (MCPR) safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).

3. An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently.

4. The requirements stated in RG 1.105, "Instrument Spans and Setpoints," are evaluated for their impact on the plant response to AOOs addressed in this SRP section.

5. Onsite and offsite electric power systems must be maintained so safety-related SSCs function during normal operation and AOOs.

6. The most limiting plant system single failure, as defined in the "Definitions and Explanations" of 10 CFR 50, Appendix A, must be assumed in the analysis and should follow the guidance of RG 1.53.

7. The performance of nonsafety-related systems during transients and accidents and of single failures of active and passive systems (especially the performance of check valves in passive systems), must be evaluated and verified by the guidance of SECY 77-439, SECY 94-084 and RG 1.206.

8. The applicant's analysis of the most limiting AOOs should use an acceptable model. Unapproved analytical methods proposed by the applicant are evaluated by the staff for acceptability.

9. Parameter values in the analytical model should be suitably conservative. The following values are acceptable:

   A. Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed operating plus an allowance of 2 percent to account for power measurement uncertainty unless (i) a lower number can be justified through the measurement uncertainty methodology and evaluation or (ii) the uncertainty is accounted for otherwise (see SRP 4.4). 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 (e.g., maximum time delay with the most reactive rod held out of the core for a PWR, a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate for a BWR), unless (i) a different conservatism factor can be justified through the uncertainty methodology and evaluation or (ii) the uncertainty is accounted for otherwise (see SRP Section 4.4).

   C. The core burn-up 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 as actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with RG 1.105 and as determined by the organization responsible for instrumentation and controls.

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. GDC 10 requires design of the reactor core and its coolant, control, and protection systems with appropriate margin so SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.

   GDC 10 applies to this section because the reviewer evaluates the consequences of loss of forced reactor coolant flow, including pump motor trips and flow controller malfunctions, AOOs that may cause SAFDLs to be exceeded because a transient reduction in reactor coolant flow causes a corresponding rise in fuel-cladding temperature.

GDC 10 requirements assure that SAFDLs are not exceeded and that fuel-cladding integrity is maintained for AOOs involving loss of forced-reactor coolant flow.

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 sequence of events, including automatic actuations of projection 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. GDC 15 requires design of the reactor coolant system and its auxiliary, control, and protection systems with sufficient margin so design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs.

   GDC 15 applies to this section because the reviewer analyzes AOOs involving loss of forced reactor coolant flow. In these transients, a reduction in reactor coolant flow can cause the reactor coolant system pressure to increase above normal levels; therefore, for loss-of-flow transients under this SRP section, the reactor coolant pressure should be analyzed to satisfy the pressure acceptance criterion.

   GDC 15 requirements assure that the design conditions of the reactor coolant pressure boundary are not exceeded for AOOs of loss of forced reactor coolant flow evaluated in this SRP section.

4. GDC 17 requires onsite and offsite electrical power systems so safety-related SSCs perform intended functions. Each power system (assuming the other system is not functioning) must provide sufficient capacity and capability so SAFDLs and design conditions of the reactor coolant pressure boundary are not exceeded in AOOs.

   GDC 17 applies because this SRP section reviews the analysis of a group of abnormal operating occurrences to which GDC 17 must be applied.

   GDC 17 requirements assure that SAFDLs and design conditions of the reactor coolant pressure boundary are not exceeded in initiating events that decrease flow in the reactor coolant system concurrent with a loss of offsite power (LOOP).

5. GDC 20 requires design of the protection system (A) to initiate automatically the operation of appropriate systems, including the reactivity control systems, so SAFDLs are not exceeded in AOOs and (B) to sense accident conditions and initiate the operation of safety-related SSCs.

   GDC 20 applies to this SRP section because the reviewer evaluates the consequences of AOOs of loss of forced reactor coolant flow, including pump motor trips or flow controller malfunctions. This section, SRP Sections 4.2 through 4.4 and 7.2 through 7.5, and RGs 1.53 and 1.105 provide guidance for reactor coolant system design with appropriate margin; thus, when the reactor protection system senses an accident condition, it initiates the operation of safety-related SSCs so SAFDLs are not exceeded.

GDC 20 requirements assure that SAFDLs are not exceeded during any AOO of loss of forced reactor coolant flow, including pump motor trips or flow controller malfunctions.

6. GDC 26 requires that one of the reactivity control systems at nuclear power plants include control rods that can control reactivity changes so SAFDLs are not exceeded under conditions of normal operation, including AOOs. The design for this system must have an appropriate margin to accommodate malfunctions like stuck rods.

   GDC 26 applies to this SRP section because the reviewer analyzes AOOs involving loss of forced reactor coolant flow. The transients analyzed in this section may involve the movement of control rods in response to the transient. Rod misalignment, including stuck rods, can aggravate thermal-hydraulic conditions. GDC 26 requires a thermal margin sufficient to accommodate these conditions. Review under this SRP section examines this margin for whether SAFDLs are exceeded.

   GDC 26 requirements assure inclusion of appropriate margins to accommodate malfunctions (e.g., stuck rods) of the reactivity control system, assurance that SAFDLs are not exceeded.

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

1. The SAR or DCD description of each loss of reactor coolant flow transient is reviewed for occurrences leading to the initiating event. The sequence of events from initiation until stabilization is reviewed to ascertain:

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

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

C. The extent to which credit is taken for the functioning of normally operating plant systems.

D. The extent to which the operation of engineered safety systems is required.

E. The extent to which operator actions are required.

F. Whether the description accounts for appropriate margin for malfunctions (e.g., stuck rods).

G. Whether the description accounts for instrumentation uncertainties of system and operating parameters appropriately.

2. If the SAR or DCD states that a particular loss of flow transient is not as limiting as some other similar transients, the reviewer evaluates the applicant's justification. The reviewer confirms whether all types of flow loss transients are considered (e.g., pump trips during two-, three-, and four-loop operations). The SAR or DCD must present a quantitative analysis of the most limiting loss of flow transient. For this transient, the reactor systems reviewer, in coordination with the instrumentation and controls reviewer, reviews the timing of the initiation of protection, engineered-safety, and other systems needed to limit the consequences of the loss of flow adequately. The reviewer compares the predicted variation of system parameters to various trip and system initiation setpoints and evaluates the effects of system and component single, active failures which may alter the course of the transient. For new applications, LOOP should not be considered a single failure; each loss of flow transient should be analyzed with and without a LOOP in combination with a single active failure. The instrumentation and controls review of SAR Chapter 7 or the corresponding DCD chapter confirms whether the instrumentation and control design is consistent with the requirements for safety system actions for these events.

3. The applicant's mathematical models to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed for whether these models have been reviewed and accepted by the staff. If not, the reviewer initiates a generic review of the applicant's proposed model.

4. The values of system parameters and initial core and system conditions as input to the model are reviewed. Of particular importance are the reactivity coefficients and control rod worths in the applicant's analysis and the variations of moderator temperature, void, and Doppler coefficients of reactivity with core life. The reviewer evaluates both the justification showing that the applicant has selected the core burn-up yielding the minimum margins and the values of the reactivity parameters in the applicant's analyses.

5. The results of the analysis are reviewed and compared to the acceptance criteria of subsection II of this SRP section for the maximum pressure in the reactor coolant and main steam systems as well as minimum DNBR (PWR) or MCPR (BWR if applicable). Time-related variations of the following parameters should be reviewed for consistency:

- reactor power;
- heat fluxes (average and maximum);
- reactor coolant system pressure;
- core and recirculation loop coolant flow rates (BWR), if applicable);
- coolant conditions (inlet temperature, core average temperature PWR core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam fractions);
- steam line pressure (BWR):
- pressure relief valve flow rate; and
- flow rate from the reactor coolant system to the containment system (if applicable).

Values of the more important of these parameters for the core flow increase AOOs are compared to those predicted for other similar plants for whether they are within the expected range. The reactor systems organization reviews the SAFDLs. The organization responsible for emergency preparedness and radiation protection is notified of the extent of fuel failures predicted by the analysis if SAFDLs are exceeded. The quality assurance and maintenance review confirms whether the SAR or DCD commits to conduct pre-operational tests to verify flow coast-down calculations.

6. 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.

Several types of plant occurrences can result in an unplanned decrease in reactor coolant flow rate. Those expected during the life of the plant result in reactor coolant (or recirculation) pump trips or flow controller malfunctions. All these postulated transients have been reviewed. The _____ transient was found the most limiting for core thermal margins and pressure within the reactor coolant and main steam systems. The applicant analyzed this transient using a mathematical model reviewed and accepted by the staff. The values of the input parameters to this model were reviewed and found suitably conservative.

The staff concludes that the plant design for transients expected to occur during plant life and result in a loss or decrease in forced reactor coolant flow is acceptable and meets the requirements of GDCs 10, 13, 15, 17, 20 and 26. This conclusion is based on the following findings:

1. The applicant meets the requirements of GDCs 10, 17, 20, and 26 by demonstrating that SAFDLs are not exceeded in this event. This requirement is met as the results of the analysis show that the thermal margin limits (minimum DNBR for PWRs, MCPR for BWRs) are satisfied as indicated by SRP Section 4.4.

2. 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.

3. The applicant meets the requirements of GDCs 15 and 17 by demonstrating that the reactor coolant pressure boundary limits are exceeded in this event. This requirement is met as the analysis shows that the maximum pressure of the reactor coolant and main steam systems does not exceed 110 percent of the design pressure.

4. The applicant meets GDC 26 requirements for the capability of the reactivity control system to control reactivity adequately during this event with appropriate margin for stuck rods because the SAFDLs are not exceeded.

5. The applicant meets the positions of RG 1.53, SECY 77-439, SECY 94-084, and RG 1.206 on the single-failure criterion and RG 1.105 on instrument actuations of safety-related SSCs.

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 following is intended as guidance to applicants and licensees on the staff's plans for this SRP section.

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.

VI. REFERENCES

1. 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities."

2. 10 CFR 50, Appendix A, GDC 10, "Reactor Design."

3. 10 CFR 50, Appendix A, GDC 13, “Instrumentation and Control.”

4. 10 CFR 50, Appendix A, GDC 15, "Reactor Coolant System Design."

5. 10 CFR 50, Appendix A, GDC 17, "Electric Power Systems."

6. 10 CFR 50, Appendix A, GDC 20, "Protection System Functions."

7. 10 CFR 50, Appendix A, GDC 26, "Reactivity Control System Redundancy and Capability."

8. 10 CFR 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

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

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

11. RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

12. RG 1.105, "Instrument Spans and Setpoints."

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

14. NUREG-0933, "A Prioritization of Generic Safety Issues."

15. NUREG-1801, "Generic Aging Lessons Learned Report," Revision 1, volumes 1-2.

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

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

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

19. 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).

20. 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).

PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the Standard Review Plan are covered by the requirements of 10 CFR Part 50 and 10 CFR Part 52, and were approved by the Office of Management and Budget, approval number 3150-0011 and 3150-0151.

PUBLIC PROTECTION NOTIFICATION

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.