U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

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

Secondary - None

I. AREAS OF REVIEW

1. A number of events 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 Standard Review Plan (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 initiating events covered by this 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).

Revision 2 - March 2007

The specific areas of review are as follows:

A. Pressurized Water Reactors (PWRs)

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

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

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

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

v. For AP1000 plants, Inadvertent actuation of the passive RHR (PRHR).

The inadvertent actuation of the PRHR system may be caused by operator action or a false actuation signal that opens the valves that normally isolate the PRHR heat exchanger from the RCS. This moderate-frequency event causes an injection of relatively cold water into the RCS and results in the addition of positive reactivity in the presence of a negative MTC.

B. Boiling Water Reactors (BWRs)

i. Loss-of a Feedwater Heater

ii. Shutdown Cooling (RHR) Malfunction

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

iv. For ESBWR, inadvertent actuation of isolation condenser.

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 (for BWRs), thermal power, vessel pressure, steam line pressure (for BWRs), steam line flow (for BWRs), feedwater flow (for BWRs), and reactivity.

The staff reviews the sequence of events described in the SAR for these transients. The 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 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 reactor systems 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 reviewer 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.

2. 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.

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. For B&W plants, the applicant's design criterion for the allowable number of actuation cycles of the emergency core cooling system and the reactor protection system consistent with the expected occurrence rates of severe overcooling events, considering both anticipated transients and accidents is reviewed.

4. Technical specifications are reviewed under SRP Section 16.0.

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 10 (GDC 10), as it relates to the reactor coolant system being designed with appropriate margin to ensure that specified acceptable fuel design limits

are not exceeded during normal operations including anticipated operational occurrences.

2. General Deisgn Criterion 13 (GDC 13), as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of approprite controls to maintain these variables and systems within prescribed operating ranges.

3. General Design Criterion 15 (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 anticipated operational occurrences.

4. General Design Criterion 20 (GDC 20), as it relates the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences.

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

<u>SRP Acceptance Criteria</u>

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 the transients which result from an increase in heat removal are:

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

2. To verify that, for the most limiting initiating events, 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 General Design Criteria 10, 15, 20, 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.

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 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. To meet the requirements of General Design Criteria 10, 13, 15, 20, 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.

5. 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.

The applicant's analysis of transients caused by excessive heat removal should be performed using an acceptable analytical model, the NRC approved methodologies and the computer codes. 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 performs an evaluation based on SRP section 15.0.2, "Transient and Accident Analysis methods."

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:

1. 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.

2. 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.

3. 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.

4. 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 the Instrumentation and Control Systems.

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 this section because the reviewer evaluates the consequences of anticipated operational occurrences that have the potential to exceed allowable thermal design criteria for fuel cladding integrity. These anticipated operational occurrences involve the transient increase in heat removal by the secondary system, which in turn causes reactor power to increase in response to the resultant lowering of the temperature of the reactor coolant. Regulatory Guide 1.53 provides guidance with respect to the application of the single failure criterion to the design and analysis of nuclear power plant protection systems. 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 anticipated operational occurrences evaluated in this SRP section involving excessive 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 sequence 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 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 anticipated operational occurrences.

   GDC 15 is applicable to this section because these overcooling events cause the reactor coolant system pressure to change in response to the drop in reactor coolant temperature. Although most of these events cause the reactor coolant pressure to decrease, some cause reactor coolant pressure to increase, depending on the worst single failure assumed. For example, for the ABWR the most severe initiating event in this group is a feedwater controller failure during maximum demand (run out of two feedwater pumps). This results in an increase in reactor pressure, but the increase is well within the ASME Code limit. Therefore, for these overcooling transients of SRP Section 15.1.1, the reactor coolant pressure needs to be analyzed to ensure that the pressure acceptance criterion is satisfied.

   Meeting the requirements of GDC 15 provides assurance that the design conditions of the reactor coolant pressure boundary are not exceeded for the anticipated operational occurrences evaluated in this SRP section involving excessive heat removal by the secondary system.

4. Compliance with GDC 20 requires that the reactor protection system be designed to initiate the operation of appropriate systems automatically, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during normal operation, including anticipated operational occurrences.

   GDC 20 is applicable to this section because the reviewer evaluates the reactor protection system that operates to shut down the reactor automatically to terminate the events (anticipated operational occurrences) analyzed in this SRP section. The events are terminated by the reactor protection system in a timely manner such that fuel cladding integrity is maintained. For a BWR, this means that the minimum value of the critical power ratio reached during the transient should be such that 99.9% of the fuel rods in the core would not be expected to experience boiling transition during core-wide transients. This limiting value of the minimum critical power ratio (MCPR) is called a safety limit. For example, the ABWR certified design MCPR is 1.07. For a PWR, this means that the minimum value of the departure from nucleate boiling ratio (DNBR) reached during the transient must remain above the 95/95 DNBR limit for the applicable DNBR correlation. For example, the System 80+ certified design, the DNBR is 1.24.

Meeting the requirements of GDC 20 provides assurance that specified acceptable fuel design limits are not exceeded by ensuring that the reactor protection system acts in a timely manner to terminate reactor operation prior to reaching a safety limit.

5. Compliance with GDC 26 requires that one of the reactivity control systems be control rods capable of reliably controlling reactivity changes to ensure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded.

   GDC 26 is applicable to this section because the reviewer evaluates these overcooling events analyzed in this section that may involve the movement of control rods in response to the initiating event, and rod misalignment, including stuck rods, can produce more severe thermal-hydraulic conditions than would otherwise exist. GDC 26 requires that the thermal margin be sufficient to accommodate these conditions. SRP Section 15.1.1 examines these margins where applicable to ensure that the thermal criteria limits are not exceeded.

   Meeting the requirements of GDC 26 provides assurance that specified acceptable fuel design limits are not exceeded by ensuring that there is appropriate margin for malfunctions of the reactivity control system, including stuck rods.

III. REVIEW PROCEDURES

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

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

1. Reactor systems 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:

   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 credit taken for the functioning of normally operating plant systems.

D. The operation of engineered safety systems that is required.

E. The extent to which operator actions are required.

F. That appropriate margin for malfunctions, such as stuck rods are accounted for.

2. If the SAR states that a particular initiating event involving an increase in heat removal is not as limiting as some other similar event, 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 event that is determined to be most limiting. For this event, 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 transient to an acceptable level. The reactor systems reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The instrumentation and control systems 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.

3. To the extent deemed necessary, the reactor systems 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.

4. 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 the reactor systems 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.

5. The values of system parameters and initial core and system conditions used as input to the model are reviewed by the reactor systems. Of particular importance are the values of reactivity coefficients and control rod worths used by the applicant in this 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 burn-up selected yields the minimum margins. The reactor systems reviews the values of the reactivity parameters used in the applicant's analysis.

6. The results of the analysis are reviewed and compared with the acceptance criteria presented in subsection II of this SRP section regarding the maximum pressure in the reactor coolant and main steam systems. Time-related variations of the following parameters are reviewed:

   – reactor 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);
   – steam line pressure;

- containment pressure;
- pressure relief valve flow rate; and
- flow rate from the reactor coolant system to the containment system (if applicable).

The values of the more important of these parameters, as listed in subsection I of this SRP section, are compared with those predicted for other similar plants to see that they are within the range expected.

The NRC has completed a program to reduce the sensitivity of B&W plants to feedwater transients, with emphasis on overcooling events that have occurred at B&W plants (Items II.E.5.1 and II.E.5.2, NUREG-0737). This sensitivity is attributed to a number of design features including the small secondary water inventory in the once-through steam generators and a relatively small pressurizer.

Concerns regarding steam generator overcooling are related to the potential for loss of natural circulation due to bubble formation and the high frequency of high-pressure safety injection actuation during the transients. These transients may produce undesirable pressure/temperature conditions (pressurized thermal shock) that may cause excessive cycles of safety-related equipment, such as thermal cycling of safety injection nozzles and operation of primary system safety relief valves. A related concern is the possible overfilling of the steam generators by which water may be introduced in the steam lines, producing loads beyond the design basis.

The resolution of these concerns consists of design modifications that provide for automatic auxiliary feedwater flow and steam generator level control, and main feedwater overfill protection. The resolution of Items II.E.5.1 and II.E.5.2 is contained in References 13 and 14.

7. 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.

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 transient 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, 13, 15, 20, and 26.

1. In meeting General Design Criteria 10, 13, 15, 20, 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 and consistent with the plant design. 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.

2. The applicant has met the requirements of General Design Criteria 10, 20, and 26 with respect to demonstrating that resultant fuel integrity is maintained since the specified acceptable fuel design limits were not exceeded for this event.

3. 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.

4. 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.

5. The applicant has met the requirements of General Design Criteria 20 and 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.

6. 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 for B & W plants.

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 to provide guidance to applicants and licensees regarding the NRC staff's plans for using 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.

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

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

6. 10 CFR Part 50, Appendix A, General Design Criterion 13, Instrumentation and Control.”

7. 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant System Design."

8. 10 CFR Part 50, Appendix A, General Design Criterion 20, "Protection System Functions."

9. 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

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

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

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

13. NRC Memorandum dated March 15, 1983, Mattson to Dircks, "Closeout of NUREG-0660 Item II.E.5.1 Design Sensitivity of B&W Plants for Operating Plants."

14. NRC Memorandum dated September 28, 1984, Denton to Dircks, "Closeout of TMI Action Plan Task II.E.5.2, Transient Response of B&W-Designed Reactors."

15. NUREG-0793, Midland Safety Evaluation Report and Supplement 1, Section 5.5, "Design Sensitivity of B&W Reactors."
