U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

15.5.1 - 15.5.2 INADVERTENT OPERATION OF ECCS AND CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

Certain anticipated operational occurrences (AOOs) can cause an unplanned increase in reactor coolant system (RCS) inventory. Depending on the boron concentration and temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or overpressurization of the RCS. Alternatively, a power level decrease and depressurization may result. The reactor will trip from high water level, high flux, high pressure, low pressure, or from a safety injection signal.

If the AOO that causes an unplanned increase in RCS inventory is a spurious actuation of the emergency core cooling system (ECCS), then the reactor is automatically tripped by the ECCS actuation signal. The ECCS, once started, is not stopped by receipt of an automatic signal. Manual action, taken according to Emergency Operating Procedures, is required to stop the ECCS.

Revision 2 - March 2007

USNRC STANDARD REVIEW PLAN

This Standard Review Plan, NUREG-0800, has been prepared to establish criteria that the U.S. Nuclear Regulatory Commission staff responsible for the review of applications to construct and operate nuclear power plants intends to use in evaluating whether an applicant/licensee meets the NRC's regulations. The Standard Review Plan 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 an acceptable method of complying with the NRC regulations.

The standard review plan sections are numbered in accordance with corresponding sections in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)." Not all sections of Regulatory Guide 1.70 have a corresponding review plan section. The SRP sections applicable to a combined license application for a new light-water reactor (LWR) are based on Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

These documents are made available to the public as part of the NRC's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Individual sections of NUREG-0800 will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience. Comments may be submitted electronically by email to NRR_SRP@nrc.gov.

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This section of the Standard Review Plan (SRP) addresses the class of AOOs that increase RCS inventory [2]; but have no effect on core boron concentration. AOOs that decrease core boron concentration are addressed in SRP Section 15.4.6.

AOOs that can increase RCS inventory are:

1. Boiling water reactors (BWRs) — Inadvertent operation of the high pressure core spray, high pressure coolant injection, or reactor core isolation cooling system and standby liquid control system.

   Other BWR transients that can result in an increase in reactor coolant inventory include feedwater system malfunctions (increasing flow), steam pressure regulator malfunctions (decreasing flow), loss of electrical load, turbine trip, main steam isolation valve (MSIV) closure, and loss of condenser vacuum. Although these transients are the subject of other SRP sections that consider their effects on system parameters other than coolant inventory, the potential impact of these transients on RCS inventory is considered herein.

2. Pressurized Water Reactors (PWRs) — Inadvertent operation of ECCS, or a malfunction of the chemical and volume control system (CVCS). It is important to consider the inadvertent operation of ECCS in PWRs in which the charging pumps are started and aligned to the ECCS upon receipt of a safety injection (SI) signal. High pressure safety injection pumps are not normally capable of delivering flow to the RCS when the RCS is at nominal pressure.

   The review of events leading to an increase in reactor coolant inventory considers the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transient.

   The reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and especially operator action to secure and maintain the reactor in a safe condition.

   The analytical methods are reviewed to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff, and have been applied in accordance with any limitations that may have been specified in the staff's acceptance. If a referenced analytical method has not been previously reviewed, the 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 predicted results of those transients analyzed are reviewed to ensure that the consequences meet the acceptance criteria given in subsection II, below.

   Further, the results of the analysis are reviewed to ascertain that the values of pertinent system parameters are within ranges expected for the type and class of reactor under review.

3. <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.

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

2 GDC 13, which requires, in part that the effect of instrumentation shall be provided to monitor variables and systems over their anticipated ranges for anticipated operational occurrences to assure adequate safety. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, which requires that the reactor coolant system and its associated auxiliary control and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operations, including anticipated operational occurrences.

4. GDC 26, which requires, in part, the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded under conditions of normal operation, including anticipated operational occurrences, with appropriate margin for malfunctions, such as stuck rods.

5. For plants with licensing bases that incorporate RG 1.70, ANS 51.1 (for PWRs), or ANSI/ANS-52.1-1978 (for BWRs), there are acceptance criteria, in SRP 15.0, “Condition II” events, or events of moderate frequency, “Condition III events, or infrequent events, and “Condition IV” events, or postulated accidents of low probability. Acceptance criteria are also defined for Condition II, III, and IV events. Regulatory Issue Summary (RIS) 2005-29, which relates to the escalation of a Condition II event into a Condition III or IV event, is also applicable to these plants.

The basic objectives in reviewing the events leading to an increase in reactor coolant inventory are:

1. To identify which of the AOOs leading to an RCS inventory increase are the most limiting.

2. To verify that, for the most limiting transients, the plant responds to the RCS inventory increase in such a way that the criteria regarding fuel damage, RCS pressure, and escalation to a more serious event are met.

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. This event is an AOO, as defined in 10 CFR 50, Appendix A. Acceptance criteria for AOOs are specified in SRP 15.0.

The specific acceptance criteria derived from GDC 10, 13, 15, and 26, and from the aforementioned ANS standards, are:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values in accordance with the ASME Boiler and Pressure Vessel Code.

2. Fuel cladding integrity should be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR) safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).

3. An AOO should not generate a more serious plant condition without other faults occurring independently.

The applicant's analysis of events leading to an increase of reactor coolant inventory should be performed using an acceptable analytical model. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer performs an evaluation of the new method as part of its review under this SRP section.

The values of 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.

3. 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.

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 fuel design limits are not exceeded during any condition of normal operation, including AOOs.

   The requirements of GDC 10 apply to the CVCS malfunction events, since power level could conceivably increase as water is being added to the RCS, until a reactor trip condition is reached.

   Meeting this criterion provides reasonable assurance that AOOs will not result in fuel damage and subsequent fission product release.

2. Compliance with GDC 13 requires the provision of instrumentation that is capable of monitor variables and systems over their anticipated ranges for normal operation, for AOOs, and for accident conditions as appropriate, to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to 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 assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs.

   The requirements of GDC 15 apply to mass addition events since the additional RCS inventory could pressurize the RCS. Without a significant addition of heat to the RCS water, the RCS would not pressurize to levels exceeding the shutoff head of the ECCS pumps, and therefore would not be expected to violate the RCS pressure safety limits. Similarly, without power generation (i.e., after reactor trip) the main steam system would not be likely to pressurize beyond the steam line safety valve setpoint levels.

Meeting this criterion provides reasonable assurance that AOOs will not result in damage to the reactor coolant pressure boundary and subsequent fission product release.

4. Compliance with GDC 26 requires cite regulation verbation. The requirements of GDC 26 apply to this section since the appropriate mitigation for an AOO is a reactor trip (and in this case, manual action to end the mass addition to the RCS). Once shut down, the reactor should remain in a shutdown condition.

   Meeting this criterion provides reasonable assurance that anticipated operational occurrences will not result in fuel damage and subsequent fission product release.

5. By meeting the ANS design requirement that states, “by itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV type without other incidents occurring independently,” or by satisfying the corresponding criterion of SRP Section 15.0 for AOOs (i.e., an AOO cannot generate a postulated accident without other incidents occurring independently). Such compliance limits the probability of initiating any of the more safety significant events at a relatively high frequency (i.e., one or more incidents prevent the event from developing into a more serious event occurring during the lifetime of the plant).

The mass addition events, the inadvertent operation of ECCS (SRP Section 15.5.1), and the CVCS malfunction that increases reactor coolant inventory and (SRP Section 15.5.2), are more likely to challenge this criterion than other AOOs. The inadvertent operation of ECCS is a concern only in plants that are equipped with ECCS pumps (or charging pumps that are used in the ECCS mode) that can deliver flow to the RCS when the RCS is at nominal pressure.

The inadvertent operation of an ECCS is postulated to occur as the result of spurious SI signal. The SI signal trips the reactor and actuates the ECCS. Therefore, since the reactor is shut down throughout the transient, violation of the DNBR safety limit is not likely to be a concern.

The inadvertent operation of an ECCS, that actuates charging pumps in an SI mode, is assumed to operate the charging system at its peak performance level (i.e., no failures are assumed). The shutoff head of the charging system is necessarily greater than the nominal RCS pressure, and possibly high enough to lift the pressurizer safety valves; but not high enough to pressurize the RCS to 110 percent of its design pressure (e.g., 2750 psia). Therefore, overpressurization of the RCS is not likely to be a concern.

Because the inadvertent operation of ECCS causes an immediate reactor trip, there is no power/steam flow mismatch. Consequently, there is little or no effect upon the main steam system. Overpressurization of the main steam system, therefore, is not likely to be a concern.

If the inadvertent operation of the ECCS causes one or more pressurizer power-operated relief valves (PORVs) to open while the pressurizer is water-solid, then the PORV is generally assumed to fail open (i.e., PORVs are assumed to fail in the open position after having relieved water, if they are not (1) safety-related equipment and (2) qualified for water relief). The inadvertent operation of the ECCS, therefore, can lead to a LOCA, which may be considered the an AOO, a postulated accident, a Condition II event, or a Condition IV event, depending upon break size and the event categorization scheme in the licensing basis (see SRP Section 15.0).

Typically, design basis accident analyses show that an AOO cannot become a more serious event, by demonstrating that the pressurizer does not become water-solid at any time during the transient, and therefore, a PORV cannot ever relieve water [7]. The event ends when the charging flow is terminated by the operator. The analysis objective is to show that the pressurizer does not become water-solid before the operator can terminate the transient, usually at about ten minutes (or longer) after the event begins. If the plant is equipped with PORVs that are (1) safety-related equipment and (2) qualified for water relief, then they may be assumed to reseat properly after having relieved water. The pressurizer safety valves, too, may be assumed to reseat properly after having relieved water; but only if such valves have been qualified for water relief.

It is conservative to assume that PORVs open and relieve steam in order to limit the RCS pressurization, and thereby increase the charging flow rate (and the resulting pressurizer fill rate). This shortens the time available to the operator to terminate the charging flow before the pressurizer fills.

Unlike the inadvertent operation of the ECCS, the CVCS malfunction that increases reactor coolant inventory (see SRP Section 15.5.2), a related AOO, does not lead directly to a reactor trip. The reactor is tripped automatically, from a signal that is generated during the transient, e.g., high pressurizer pressure or level. Since power is being generated prior to reactor trip, the event could cause DNB to occur. However, this is not likely, since (1) core pressure increases, and (2) the reactor protection system automatically trips the reactor when it senses a reduction in thermal margin. Like the inadvertent operation of ECCS, the CVCS malfunction is not expected to pose a concern with respect to RCS and main steam system overpressurization.

The CVCS malfunction event should meet the same acceptance criteria as the inadvertent operation of ECCS. The event is mitigated (i.e., terminated) when the operator shuts off the charging flow. The CVCS malfunction event is expected to be less limiting (i.e., to fill the pressurizer more slowly) than the inadvertent operation of ECCS event. This is due to some coolant shrinkage that occurs when the reactor is tripped, and to the lower charging flow rate that is delivered when the charging pumps are not operating as part of the ECCS.

III. REVIEW PROCEDURE

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 applicant's description of events leading to an increase in reactor coolant inventory is reviewed with respect to the occurrences leading to the initiating event. The sequence of events, from initiation until a stabilized condition is reached, is reviewed to determine the following:

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. Credit taken for the functioning of normally operating plant systems;

4. Operation of required engineered safety systems;

5. The extent to which operator actions are required; (Note: an operator action to shut off the charging pump flow is normally required to terminate these AOOs.

6. That appropriate margin for malfunctions, such as stuck rods (see subsection II.3.b), is taken into consideration.

The applicant should present a quantitative analysis in the SAR of the most limiting events that lead to an increase in reactor coolant inventory. Such an analysis should demonstrates that AOOs could not develop into more serious events. The reviewer examines the timing of the initiation of those protection and engineered safety systems, and operator actions needed to limit the consequences of the event to acceptable levels. The reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints.

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 to determine if these models have been previously reviewed and found acceptable by the staff. If not, a generic review of the models is initiated.

The values of system parameters and initial core and system conditions used as input to the model are reviewed. 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 the selected core burnup yields the minimum margins.

The results of the applicant's analysis are reviewed in accordance with the acceptance criteria presented in subsection II regarding maximum pressure in the reactor coolant and main steam systems, the minimum critical heat flux ratio (MCHFR) DNBR and the possibility of escalation to a more serious event. The variations with time during the transient of the neutron power, heat fluxes (average and maximum), reactor coolant system pressure, minimum DNBR (PWR) or CPR (BWR); core and recirculation loop coolant flow rates (BWR), coolant conditions (inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam fractions), steam line pressure, containment pressure, pressure relief valve flow rate, and flow rate from the reactor coolant system to the containment system are reviewed, as applicable. The review will also compare values of the more important of these parameters for the events leading to an increase in reactor coolant inventory with those predicted for other similar plants to confirm that they are within the expected range.

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 concludes that the analysis of a transient resulting in an increase in reactor coolant inventory is acceptable and meets the requirements of General Design Criteria 10, 15, and 26, and the guidance of ANS standards. This conclusion is based on the following:

1. In meeting General Design Criteria 10,13, 15, and 26 as discussed below, the staff has determined that the applicant's analysis was performed using a mathematical model that has 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 staff has further determined that the positions of Regulatory Guide 1.53 for the single-failure criterion and Regulatory Guide 1.105 for instruments have also been satisfied.

2. The applicant has met the requirements of GDC 10, and 26 with respect to demonstrating that resultant fuel damage is maintained because the specified acceptable fuel design limits were not exceeded for this event.

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

6. The applicant has satisfied the ANS design criteria that prohibits the escalation of an AOO to a more serious incident without other incidents, occurring independently.

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 guide, no reformed NUREG.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criteria.

2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

3. ANS 51.1, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor [PWR] Plants," 1983 (replaces ANSI N18.2-1973) (withdrawn in 1998).

4. ANSI/ANS-52.1-1983, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," (withdrawn in 1998).

5. Regulatory Issue Summary 2005-29, "Anticipated Transients that could Develop into More Serious Events," December 14, 2005 (ADAMS Accession No. ML051890212).

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

7. RS-001, Revision 0, "Review Standard for Extended Power Uprates," December 2003, (Note 8 of Matrix 8 of Section 2.1).

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

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