**U.S. NUCLEAR REGULATORY COMMISSION**

# STANDARD REVIEW PLAN

## 15.1.5 STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT (PWR)

**REVIEW RESPONSIBILITIES**

**Primary -** Organization responsible for the review of transient and accident analyses for PWRs

**Secondary -** None

I. AREAS OF REVIEW

1. The steam release resulting from a rupture of a main steam pipe will cause an increase in steam flow which decreases with time as the steam pressure decreases. The increased steam flow causes increased energy removal from the reactor coolant system and results in a reduction of coolant temperature and pressure. The negative moderator temperature coefficient and the cooldown of the reactor system causes an increase in core reactivity. The core reactivity increase may cause a loss of reactor core shutdown margin and a resulting increase in reactor power. If the plant is at power, the reactor is automatically tripped and the main steam and feedwater line isolation valves are automatically closed. Decay heat is removed as necessary through the unaffected steam generators by venting steam from the secondary system safety and relief valves. The auxiliary feedwater system (AFWS) supplies makeup water to the unaffected steam generator(s). For AP1000 the passive RHR (PRHR) provides the safety related means of decay heat removal.

Analysis of the transient following a steam line break is sensitive to the fluid discharge rate at the break so that a range of break sizes must be considered both inside and outside containment to determine the acceptability of the system response. The course that the transient takes and its ultimate effects also depend on the assumed initial power level and mode of operation (e.g., hot shutdown; full power; one-, two-, or three-loop operation). Evaluation with various assumed initial conditions is required to verify that the condition leading to the severest consequences has been identified.

The specific areas of review are as follows:

– Postulated initial core and reactor conditions pertinent to the steam line break accident;

– Methods of thermal and hydraulic analyses, including the effects of hydraulic instabilities;

– Postulated sequence of events, including analyses to determine the time of reactor trip and time delays prior to and subsequent to initiation of the reactor protection system;

– Assumed responses of the reactor coolant and auxiliary systems;

– Functional and operational characteristics of the reactor protection system in terms of its effects on the sequence of events;

– Operator actions required to secure and maintain the reactor in a safe shutdown condition;

– Core power excursion due to power demand created by excessive steam flow out the break; and

– Variables influencing neutronics.

The results of the analyses are reviewed to ensure that pertinent system parameters are within expected ranges. The parameters of importance for these transients include:

– Reactor coolant system (RCS) pressure,
– Steam generator pressure,
– Fluid temperatures,
– Clad temperatures,
– Discharge flow rates,
– Steam line and feedwater flow rates,
– Safety and relief valve flow rates,
– Pressurizer and steam generator water levels,
– Reactor power,
– Total core reactivity,
– Hot and average channel heat flux, and
– Minimum departure from nucleate boiling ratio (DNBR).

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both Reactor Systems and Instrumentation and Control. The Reactor Systems reviewer concentrates on the capability of the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition.

The analytical methods are reviewed by 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, Reactor Systems initiates an evaluation of the new analytical model.

2. <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.

<u>Review Interfaces</u>

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. 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. The auxiliary feedwater system is reviewed to verify its ability to function following a steam line break given a single active component failure with either onsite or offsite power under SRP Section 10.4.9.

5. Effects of blow-down loads, including jet propulsion piping and component supports and the design bases for safety and relief valves are reviewed under SRP Sections 3.6.2 and 3.9.1 through 3.9.3. Design bases for safety and relief valves is also reviewed under SRP Section 3.9.3.

6. Fracture toughness properties of the reactor coolant pressure boundary and reactor vessel are reviewed under Sections 5.2.3 and 5.3.1.

7. The response of the containment to ruptures of steam lines with regard to the effects of pressure and temperature on the containment functional capabilities is reviewed under SRP Section 6.2.1. Analytical methods for deriving mass energy releases exiting the postulated break are reviewed under SRP Section 6.2.1.3.

8. Aspects of the sequence described in the SAR are reviewed to evaluate whether the 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, and interlocks with auxiliary or shared systems. This review includes the instruments and controls required to ensure automatic and manual auxiliary feedwater system initiation and flow indication in the control room and is performed under SRP Sections 7.1 through 7.7. The potential bypass modes and the possibility of manual control by the operator are also reviewed under SRP Sections 7.1 through 7.7.

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:

The general objective of the review of steam line rupture events is to verify that short-term and long-term ability to cool the core has been achieved by confirming that the primary reactor coolant system is maintained in a safe status for a break equivalent in area to the double-ended rupture of the largest steam line. The acceptance criteria are based on meeting the relevant requirements of the following regulations:

1. General Design Criterion 10 (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.

2. General Design Criterion 17 (GDC 17), as it relates to the requirement that an onsite and offsite electric power system be provided to permit the 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 the acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded during an anticipated operational occurrence and that core cooling, containment integrity, and other vital functions are maintained in the event of an accident.

3. General Design Criteria 27 (GDC 27) and 28 (GDC 28), as they relate to the reactor coolant system being designed with appropriate margin to ensure that acceptable fuel design limits are not exceeded and that the capability to cool the core is maintained.

4. General Design Criterion 31(GDC 31), as it relates to the reactor coolant system being designed with sufficient margin to ensure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized.

5. General Design Criterion 35 (GDC 35), as it relates to the reactor cooling system and associated auxiliaries being designed to provide abundant emergency core cooling.

   Requirements for ensuring adequate decay heat removal and reactor coolant pump Integrity and operation are specified in 10 CFR 50.34(f)(2)(xii)[1] and 10 CFR 50.34(f)(1)(iii), respectively.

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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.

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.

Specific criteria necessary to meet the relevant requirements of the above regulations are as follows:

1. Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.

2. The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit for PWRs based on acceptable correlations (see SRP Section 4.4). If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

3. The radiological criteria used in the evaluation of steam system pipe break accidents (PWRs only) appear in SRP section 15.0.3.

4. The integrity of the reactor coolant pumps should be maintained such that loss of ac power and containment isolation will not result in pump seal damage.

5. The auxiliary feedwater system or other means of decay heat removal must be safety related and, when required, automatically initiated. In the case of AP1000 the PRHR provides the safety related means of decay heat removal.

6. Tripping of the reactor coolant pumps should be consistent with the resolution to Task Action Plan item II.K.3.5.

There are certain assumptions regarding important parameters used to describe the initial plant conditions and postulated system failures which should be used. These are listed below:

1. The reactor power level and number of operating loops assumed at the initiation of the transient should correspond to the operating condition which maximizes the consequences of the accident. These assumed initial conditions will vary with the particular NSSS design, and sensitivity studies will be required to determine the most conservative combination of power level and plant operating mode. These sensitivity studies may be presented in a generic report and referenced in the SAR.

2. Assumptions as to the loss of offsite power and the time of loss should be made to study their effects on the consequences of the accident. A loss of offsite power may occur simultaneously with the pipe break or during the accident, or offsite power may not be lost. Analyses should be made to determine the most conservative assumption appropriate to the particular plant design. The reviewer should note that the assumption that offsite power is not lost may maximize heat removal from the core and reactor

system and thereby maximize containment pressure and reactivity feedback within the core. The analyses should take account of the effect that loss of offsite power has on reactor coolant pump and main feedwater pump trips and on the initiation of auxiliary feedwater flow, and the effects on the sequence of events for these accidents. For new applications, loss of offsite power should be considered in addition to any limiting single active failure. (This position is based upon interpretation of GDC 17, as documented in the Final Safety Analysis Report for the ABB-CE System 80+ design certification.)

3. The effects (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) of postulated steam line breaks on other systems should be considered in a manner consistent with the intent of Branch Technical Position (BTP) 3-3 and BTP 3-4.

4. The worst single active component failure should be assumed to occur. For new applications, loss of offsite power should not be considered as a single failure, (see assumption b above). The assumed single failure may cause more than one steam generator to blow down, failure of main feedwater to isolate, or may be in any of the systems required to control the transient.

5. The maximum-worth rod should be assumed to be held in the fully withdrawn position. An appropriate rod reactivity worth versus rod position curve should be used. Local power peaking at the location of the stuck out control rod should be considered. Local power peaking will affect the DNBR analysis in the initial period as the safety rods are entering the core and during any subsequent return to power resulting from reactivity addition to the core from the cooldown.

6. The core burnup (time in core life) should be selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

7. The initial core flow assumed for the analysis of the steam line rupture accident should be chosen conservatively. If the minimum core flow allowed by the technical specifications is assumed, the minimum DNBR margin results; however, for the analysis of steam line break accidents, this may not be the most conservative assumption. For example, maximum initial core flow results in increased reactor coolant system cooldown and depressurization, decreased shutdown margin, and an increased possibility that the core will become critical and return to power. Since it is not clear what initial core flow is most conservative, the assumed value should be justified.

8. Failure of a steam line at a plant with multiple coolant loops will cause asymmetric temperatures within the reactor core. Asymmetric core temperatures will affect the local power distribution and the DNBR analysis. Assumptions for mixing in the downcomer and the reactor vessel lower plenum will affect the predicted core temperature distributions, reactivity feedback and local power. Assumptions for mixing should be chosen so as to be conservative for predicting maximum local core power and DNBR.

9. For postulated pipe failure in nonseismically qualified portions of the main steam line (outside containment and downstream of the main steam isolation valves, (MSIVs) due to a seismically initiated event, only safety related equipment should be assumed operative to mitigate the consequences of the break.

10. For postulated instantaneous pipe failures in seismically qualified portions of the main steam line (inside containment and upstream of the MSIVs), only safety related equipment should be assumed operative. If, in addition, a single malfunction or failure

of an active component is postulated, credit may be taken for the use of a backup nonsafety-related component to mitigate the consequences of the break.

11. During the initial 10 minutes of the transient, should credit for operator action be required (e.g., reactor coolant pump trip), an assessment for the limiting consequence must be performed in order to account for operator delay and/or error.

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 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.

2. Compliance with GDC 17 requires (in part) that onsite and offsite electrical power systems be provided to ensure the functioning of structures, systems, and components important to safety. The safety function for each power system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that (a) the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (b) the core is cooled and containment integrity and other vital functions are maintained.

   GDC 17 is applicable to this section because it requires that the loss of offsite power be considered — not as a single failure event, but assumed in the analyses for each event without changing the event category. Thus, the applicant is required to consider a loss of offsite power concurrent with a single failure in the analysis of steam system piping failures.

3. Compliance with GDC 27 requires that reactivity control systems be designed to have a combined capability (in conjunction with poison added by the emergency core cooling system) of reliably controlling reactivity changes, thereby ensuring that the capability for core cooling is maintained under postulated accident conditions and with appropriate margin for stuck rods.

   Compliance with GDC 28 requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (a) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (b) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor temperature and pressure, and addition of cold water.

GDC 27 and GDC 28 are applicable to this section because the reviewer evaluates steam system piping failures, both inside and outside containment, that could cause transient conditions affecting reactor coolant temperature and pressure, including complex changes in core reactivity. The applicant's analyses of these transients in the SAR must demonstrate that reactivity, pressure, and temperature changes will not be severe enough to cause an unacceptable impact on the reactor coolant pressure boundary or on the capability for cooling the core. These analyses must be independently reviewed by the staff in accordance with this SRP section.

4. Compliance with GDC 31 requires that, under the stress of operation, maintenance, testing, and postulated accidents, the reactor pressure boundary shall be designed with sufficient margin to ensure that (a) the boundary behaves in a non-brittle manner and (b) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other boundary material variables under a full range of conditions. The design will also address such issues as the uncertainties of determining material properties; the effects of irradiation on material properties; residual, steady state, and transient stresses; and the sizes of flaws.

   GDC 31 is applicable to this section because the reviewer evaluates steam system piping failures, both inside and outside containment, that could cause transient conditions with a potentially harmful effect on the reactor coolant pressure boundary. A steam system piping break can result in a rapid decrease in reactor coolant temperature and steam generator pressure, placing undue stress on the reactor coolant pressure boundary. This potential problem could be aggravated by a pressurization of the primary system when the emergency core cooling system is activated. The amount of stress to the reactor coolant pressure boundary depends on the severity of the transient. The severity of the transient is assessed by the applicant in the SAR and is reviewed by the staff in accordance with this SRP section.

5. Compliance with GDC 35 requires a system that will provide abundant emergency core cooling. The system safety function shall be to transfer heat from the reactor core after any loss of reactor coolant at a rate ensuring that (a) fuel and clad damage interfering with continued effective core cooling is prevented and (b) clad metal-water reaction is limited to negligible amounts.

   GDC 35 is applicable to this section because the reviewer evaluates steam system piping failures, both inside and outside containment, that could cause transient conditions with the potential to challenge the emergency core cooling system. During a steam system piping break, excessive steam loss will result in a rapid reduction of reactor coolant temperature and steam generator pressure. A subsequent reactor trip can further reduce the primary system pressure, producing a void within the pressure vessel and creating the need for emergency core cooling.

## 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), 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 or COL 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.

1. The reviewer determines the acceptability of the analytical models and assumptions, as follows:

   A. The values of system parameters and initial core and system conditions used as input to any analytical model are reviewed by Reactor Systems. Of particular importance are (1) the reactivity coefficients and control rod worths used in the analysis and (2) the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The reviewer will evaluate the justification provided by the applicant to show that the core burnup yielding the minimum margins has been selected. Core Performance reviews core-related parameters such as DNB correlations and the values of the reactivity parameters used in the analysis. The reviewer confirms that the amount of secondary coolant expelled from the system (for breaks outside containment) has been calculated conservatively by evaluating the methods and assumptions, by comparing these results with those of an acceptable analysis performed on another plant of similar design, or by comparing the results with staff calculations.

   B. The acceptability of the methods equations, sensitivity studies, and models proposed by the applicant are evaluated.

   C. Analytical models should be sufficiently detailed to simulate the reactor coolant (primary), steam generator (secondary), and auxiliary systems. The reviewer evaluates the following functional requirements:

      i. Reactor trip signal: credit taken for any reactor trip signal is reviewed by Instrument and Control to confirm that, under accident conditions, the instrumentation and control systems are capable of the assumed response.

      ii. Emergency core cooling system (ECCS): credit taken for actuation of the ECCS is reviewed by Instrument and Control to verify the ability of the instrumentation and control systems to respond as assumed.

      iii. Auxiliary feedwater system: the availability of the auxiliary feedwater system to supply adequate auxiliary feedwater flow to the intact steam generators during the accident and the subsequent shutdown condition is evaluated. This is done by Plant Systems as to availability of the system and by Reactor Systems as to capability to effect an orderly shutdown. Since auxiliary feedwater system designs are diverse and may require both automatic and manual actuation, preoperational tests should be specified to identify any necessary operator actions and to establish times required for their completion. In the case of AP1000 the safety related function of decay heat removal is performed by the PRHR.

D. Time-related variations of the following parameters are reviewed:

- reactor power;
- heat fluxes (average and maximum);
- total core reactivity;
- reactor coolant system pressure;
- minimum DNBR;
- coolant conditions (inlet temperature core average temperature and average exit and hot channel exit temperatures;
- fuel rod conditions (maximum fuel center–line temperature, maximum clad temperature, or maximum fuel enthalpy);
- steam generator pressure;
- containment pressure;
- relief and/or safety valve flow rates;
- discharge flow rate;
- steam line and feedwater flow rates; and
  pressurizer and steam generator water levels.

The values of the more important of these parameters for the steam line break accident (as listed in subsection I) are compared with those predicted for other similar plants to see that they are within the range expected.

1. To the extent deemed necessary, the reviewer evaluates the effect of single active failures of systems and components that may affect the course of the accident. For new applications, loss of offsite power should not be treated as a single active failure, as discussed under subsection II, assumptions b and d. This phase of the review is done using the system review procedures described in the SRP sections for Chapters 5, 6, 7, 8, and 10 of the SAR. The reviewer also considers single failures that may cause more than one steam generator to blow down or failure of main feedwater to isolate, thus increasing the reactivity addition to the core.

2. The reviewer confirms that a commitment has been made in the SAR to conduct preoperational tests for verifying that valve discharge rates and response times (including, for example, opening and closing times for main feedwater, auxiliary feedwater, turbine and main steam isolation valves, and steam generator and pressurizer relief and safety valves) have been conservatively modeled in the accident analyses. In addition, preoperational testing should include verification of reactor trip delay times, startup delay times for auxiliary feedwater system actuation, safety injection signal delay time, and delay times for delivery of any high concentration boron solution required to bring the plant to a safe shutdown condition. The reviewer should note that for AP1000 the safety related means of decay heat removal is the PRHR rather than auxiliary feedwater.

3. Based on the above information, Plant Systems evaluates the radiological consequences of the design basis steam line break accident as described in the SRP section 15.0.3.

4. Upon request from the primary reviewer, other secondary reviewers will provide input for the areas of review stated in subsection I of this SRP section. The primary reviewer obtains and uses such input as required to ensure that this review procedure is complete.

5. The reliability and operability of the auxiliary feedwater systems are reviewed to ensure compliance with the requirements of 10 CFR 50.34(f)(1)(ii) (TMI issue ii E 1.1) and 10 CFR 50.34(f)(2)(xii) (TMI issue ii E 1.2) as they relate to auxiliary feedwater system performance requirements for steam system piping failures. The reviewer should see Chapter 20 of the NRC FSAR for AP1000 to see how those post TMI requirements are met by the PRHR and the non-safety related start-up feedwater system SUFWS of AP1000.

6. The reliability and integrity of the reactor coolant pump seals during loss of alternating-current power and loss of coolant to the seals (e.g. resulting from containment isolation) are reviewed to ensure compliance with the requirements of 10 CFR 50.34(f)(1)(iii).

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.

The staff concludes that the consequences of postulated steam line breaks meet the relevant requirements set forth in the General Design Criteria 13, 17, 27, 28, 31, and 35 regarding (1) the ability to insert the control rods and to cool the core and (2) TMI Action Plan items. This conclusion is based upon the following:

A. 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.

B. The applicant has met the requirements of GDC 27 and GDC 28 by demonstrating that the resultant fuel damage was limited such that the ability to insert control rods would be maintained and that no loss of core cooling capability resulted. The minimum departure from nucleate boiling ratio (DNBR) experienced by any fuel rod was _________, resulting in __% of the rods experiencing cladding perforation.

C. The applicant has met the requirements of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the postulated accident.

D. The applicant has met the requirements of GDC 35 with respect to demonstrating the adequacy of the emergency cooling systems to provide abundant core cooling and reactivity control (via boron injection).

E. The analyses and effects of steam line break accidents inside and outside containment, during various modes of operation with and without offsite power (as required by GDC 17), have been reviewed and were evaluated using a mathematical model that has been previously reviewed and found acceptable by the staff.

F. The parameters used as input to this model were reviewed and found to be suitably conservative.

G. The radioactivity release has been evaluated using the computer code SARA and a conservative description of the plant response to the accident. A decontamination factor of _______ between the water and steam phases and a χ/Q value of _____ sec/m3 has been used in our evaluation of radiological consequences. The calculated doses are presented in Table _____. Technical specification limits on primary and secondary coolant activities will limit potential doses to a small fraction of the 10 CFR Part 100 exposure guidelines. The potential doses are within 10 CFR Part 100 exposure guidelines even if the accident should occur coincident with an iodine spike.

H. The applicant has met the requirements of 10 CFR 50.34(f)(1)(ii) and 10 CFR 50.34(f)(2)(xii) with respect to demonstrating the adequacy of the design of auxiliary feedwater or other qualified systems to remove decay heat following steam system piping failures.

I. The applicant has met the requirements of 10 CFR 50.34(f)(1)(iii) with respect to demonstrating the integrity and operation of the reactor coolant pumps to withstand the postulated accident.

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.

## VI. REFERENCES


1. 10 CFR Part 50, General Design Criterion 13, “Instrumentation and Control.”

2. 10 CFR Part 50, General Design Criterion 17, "Electric Power Systems."

3. 10 CFR Part 50, General Design Criterion 27, "Combined Reactivity Control Systems Capability."

4. 10 CFR Part 50, General Design Criterion 28, "Reactivity Limits."

5. 10 CFR Part 50, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

6. 10 CFR Part 50, General Design Criterion 35, "Emergency Core Cooling."

7. Branch Technical Position 3-3, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."

8. Branch Technical Position 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment."