NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.6.5 LOSS-OF-COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Accident Evaluation Branch (AEB)

I. AREAS OF REVIEW

Loss-of-coolant accidents (LOCA) are postulated accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the normal reactor coolant makeup system, from piping breaks in the reactor coolant pressure boundary. The piping breaks are postulated to occur at various locations and include a spectrum of break sizes, up to a maximum pipe break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant pressure boundary. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished.

General Design Criteria 35 (Ref. 1) requires each pressurized water reactor (PWR) and boiling water reactor (BWR) to be equipped with an emergency core cooling system (ECCS) that refills the vessel in a timely manner to satisfy the requirements of the regulations for ECCS given in 10 CFR Part 50, §50.46 and Appendix K to 10 CFR Part 50 (Ref. 2) and the applicable general design requirements discussed in SRP Section 6.3 (Ref. 3). The analysis of ECCS performance has an impact on the design of the piping and support structures for the reactor coolant system, the design of the steam generators, the containment design, and the possible need for pump overspeed protection.

The review of the applicant's analysis of the spectrum of postulated loss-of-coolant accidents is closely associated with the review of the ECCS, as described in SRP Section 6.3. As a portion of the review effort described in this SRP section and in SRP Section 6.3, RSB evaluates whether the entire break spectrum (break size and location) has been addressed; whether the appropriate break locations, break sizes, and initial conditions were selected in a manner that conservatively predicts the consequences of the LOCA for evaluating ECCS performance; and whether an adequate analysis of possible failure modes of ECCS equipment and the effects of the failure modes on the ECCS performance have been provided. For postulated break sizes and locations, the RSB review includes the postulated initial reactor core and reactor system conditions, the postulated sequence of events including time delays prior to and after emergency power actuation, the calculation of the power, pressure, flow and temperature transients, the functional and operational characteristics of the reactor protective and ECCS systems in terms of how they affect the sequence of events, and operator actions required to mitigate the consequences of the accident.

The calculational framework used for the evaluation of the ECCS system in terms of core behavior is called an evaluation model. It includes one or more computer programs, the mathematical models used, the assumptions and correlations included in the program, the procedure for selecting and treating the program input and output information, the specification of those portions of the analysis not included in computer programs, the values of parameters, and all other information necessary to specify the calculational procedure. The evaluation model used by the applicant must comply with the acceptance criteria for ECCS given in 10 CFR Part 50, §50.46 and Appendix K to 10 CFR Part 50. The evaluation model must have been previously documented and reviewed and approved by the staff. Should the LOCA blowdown calculations be modified for the purpose of studying structural behavior (for example, core support structure design, control rod guide structure design, steam generator design, reactor coolant system piping and support structure design), all differences should be identified and described by the applicant. RSB reviews these modifications, including analytical techniques, computer programs, values of input parameters, break size, type, and location, and all other pertinent information, and makes recommendations regarding their acceptability to other branches as required. RSB initiates a generic computer program review as required.

RSB is also responsible for the review of the failure mode analysis of the ECCS in conjunction with the effort described in SRP Section 6.3. The RSB reviews the analytical techniques and computer programs used by the applicant for the blowdown, refill, and reflood portions of the loss-of-coolant transient. The RSB also reviews the analytical techniques and computer programs used by the applicant for the cladding temperature, cladding rupture and swelling calculations. RSB will perform independent audit blowdown, refill, reflood and cladding calculations as required to verify the applicant's conclusions.

AEB as part of its secondary review responsibility provides an evaluation of fission product releases and radiological consequences. This effort is described in the appendices to this SRP section and their results are included in the SER writeup.

The RSB will coordinate, as required and by request, other branch evaluations that interface with the overall review of this SRP section as follows: The Auxiliary Systems Branch (ASB) review of Chapters 9 and 10 of the applicant's SAR includes an evaluation of auxiliary systems (e.g., service water system, component cooling system, ultimate heat sink, condensate storage facility) to confirm that these systems can supply all the functions required to support the ECCS in performing its function during and following a loss-of-coolant accident. Upon request, ASB will verify that the auxiliary system described by the applicant for the safety analysis supply all the functions required. ASB also, upon request from RSB, reviews the failure modes analysis of the ECCS to verify that an adequate analysis of possible failure modes of ECCS equipment and the effect of the failure modes on the ECCS performance has been provided. The Containment Systems Branch (CSB) review of SRP Section 6.2.1 includes an evaluation of the functional capability of the containment for the spectrum of loss-of-coolant events. CSB verifies, upon request from RSB, that the assumptions used for the containment response analysis have been selected in a conservative manner for the LOCA analysis performed. Upon request from RSB, CSB reviews the containment pressure calculations utilized by the applicant, or by the staff in an audit analysis, for the reflood portion of the ECCS performance analyses. The Core Performance Branch (CPB), upon request from RSB, verifies that the core physics data used by the applicant, or by the staff in independent audit analyses, are the appropriate data to be used. CPB also, upon request from RSB, reviews the power transient calculations (including moderator temperature, void and fuel temperature feedback effects, and decay heat) and the cladding rupture and swelling calculations. The Instrumentation and Controls System Branch (ICSB) review of SRP Sections 7.2 and 7.3 includes a review of the reactor protection system and associated ECCS controls and instrumentation with regard to automatic actuation, remote sensing and indications, remote control, and redundancy. Upon request from RSB, ICSB verifies that the reactor protection system and associated ECCS controls and instrumentation will function as described in the applicant's sequence of events for the safety analyses performed. ICSB also, upon request from RSB, reviews the failure modes analysis of the ECCS to verify that an adequate analysis of possible failure modes of ECCS equipment and the effect of the failure modes on the ECCS performance has been provided. The Power Systems Branch (PSB) review of 8.3.1 and 8.3.2 includes the emergency onsite power functional capabilities. The PSB, upon request from RSB, will verify that the control systems power sources needed to function to mitigate the event are available as required by the applicant's description of the event. PSB also, upon request from RSB, reviews the failure modes analysis of the ECCS to verify that an adequate analysis of possible failure modes of ECCS equipment and the effect of the failure modes on the ECCS performance has been provided. The Mechanical Engineering Branch (MEB) review of SRP Sections 3.6 and 3.9 includes a review of the effects of the blowdown loads on core support structures and on control rod guide structures. MEB verifies, upon request from RSB, that the core remains in a coolable geometry following a loss-of-coolant accident and that the control rods can also be inserted. MEB also evaluates the effects of blowdown loads on the piping of the reactor coolant system and on the support structures of the components of the reactor coolant system. Upon request from RSB, MEB verifies that acceptable criteria (Ref. 5) have been employed in the design of the reactor coolant system and its supports to prevent failures in the reactor coolant pressure boundary on in engineered safety feature equipment in the event of a LOCA.

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding primary branch.

II. ACCEPTANCE CRITERIA

RSB acceptance criteria are based on meeting the relevant requirements of the following regulations:

a. 10 CFR Part 50, §50.46 and Appendix K as it relates to ECCS equipment being provided that refills the vessel in a timely manner for a loss-of-coolant accident resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary.

b. General Design Criterion 35 as it relates to redundant ECCS components being provided to adequately cool the core during a loss-of-coolant accident.

c, 10 CFR Part 100 (Ref. 4) as it relates to mitigating the radiological consequences of an accident.

Specific criteria necessary to meet the relevant requirements of the regulations identified above and necessary to meet task action plan items of NUREG-0718 and -0737 (Ref. 6 and 7) are as follows:

1. An evaluation of ECCS performance has been performed by the applicant in accordance with an approved evaluation model that satisfies the requirements of 10 CFR Part 50, §50.46 and Appendix K to 10 CFR Part 50. For the full spectrum of reactor coolant pipe breaks, the results of the evaluation must show that the specific requirements of the acceptance criteria for ECCS are satisfied as given below:

   a. The calculated maximum fuel element cladding temperature does not exceed 2200°F.

   b. The calculated total oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

   c. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

   d. Calculated changes in core geometry are such that the core remains amenable to cooling.

   e. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

2. The radiological consequences of the most severe LOCA are within the guidelines of 10 CFR Part 100. Appendices A, B and D to this SRP section provides the results of the LOCA analysis.

3. The TMI Action Plan (Ref. 6 and 7) requirements for II.E.2.3, II.K.2.8, II.K.3.5, II.K.3.25, II.K.3.30, II.K.3.31, and II.K.3.40 have been met.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) 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, final values should be used in the analysis and the reviewer compares these to the limiting safety system settings included in the proposed technical specifications.

For the review of the ECCS performance analysis, as presented in the applicant's safety analysis report (SAR), the reviewer verifies the following:

1. The calculations were performed using an approved evaluation model. The application should clearly state this and properly reference the evaluation model. If the analysis is done with a new evaluation model, a generic review of the new model is required.

2. An adequate failure mode analysis has been performed to justify the selection of the most limiting single active failure. This analysis is reviewed in part under SRP Section 6.3. If the design has been changed from that presented in previous applications, changes in the reactor coolant system, reactor core, and ECCS are reviewed with respect to the most limiting single failure.

3. A variety of break locations and the complete spectrum of break sizes were analyzed. If part of the evaluation is done by referencing earlier work, design differences (ECCS, reactor coolant system, reactor core, etc.) between the facilities in question are reviewed. If there are significant differences, sensitivity studies on the important parameters should have been made by the applicant. If such sensitivity studies are not presented in the SAR, the reviewer requests that they be made.

4. The parameters and assumptions used for the calculations conform to those of the approved evaluation model and were conservatively chosen, including the following points:

    a. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.

    b. The maximum linear heat generation rate used should be based on 102% of the proposed licensed core thermal power and the technical specification limit on peaking factors, or on the technical specification limits on maximum linear heat generation rate.

    c. All permitted axial power shapes, as given in Section 4.3 of the SAR, should be addressed by the analyses. Normally, the evaluation model will identify the least favorable axial shape as a function of break size. If the evaluation model did not discuss axial shapes, or the discussion is not applicable to a given case, sensitivity studies are requested.

    d. The initial stored energy was conservatively calculated by the applicant. The value used is checked against the applicant's steady-state temperatures, as given in SAR Section 4.4, similiar calculations performed by the staff, or calculations done for similar plants by previous applicants.

    e. Appropriate analyses are presented to support any credit taken for control rod insertion.

    f. The applicant's analysis conservatively addresses the operation of the reactor coolant pump.

5. Reactor protection system actions and safety injection actuation and delivery are consistent with the set points and the associated uncertainties and delay times listed in the SAR (OL review). The ECCS flow rates should be checked against the applicant's data on head-flow characteristics of the ECCS pumps given in Section 6.3 of the SAR and against typical safety injection tank discharge curves used for the analysis. The Regional Offices under the Office of Inspection and Enforcement may be requested to provide data of this type from the startup tests for new designs and from periodic tests on duplicate designs.

6. The results of the applicant's calculations are consistent with those of staff calculations for typical plants and also with the results of calculations performed for similar systems by previous applicants. The following variables should be reviewed on a generic basis and spot-checked thereafter: power transients for various breaks; pressure transients at various system locations; flow transients near the break, in the core, and in the downcomer; reactor coolant temperature and quality at core inlet, core outlet, and in-core; cladding temperature transients (core average, hot assembly, hot pin); heat transfer coefficients during blowdown, refill, and reflood; heat flux transients from piping and vessel walls; primary-secondary heat transfer (PWRs only); timing of clad rupture (if the peak clad temperature could be appreciably higher when perforation occurs at a different but equally probable time, calculations with modified assumptions are requested); peak clad temperature as a function of break size (if it is uncertain whether the peak value has been found, additional calculations are requested); predicted "end-of-bypass" time compared to calculated downcomer flow and to staff calculations for typical plants; pump speed transients; containment pressure transients (if staff calculations are not available, these are requested from CSB); and carryover fraction (if it is not an input to the calculations).

7. The calculated peak clad temperature, maximum local oxide thickness, and core average zirconium-water reaction meet the acceptance criteria for ECCS given in 10 CFR Part 50, §50.46 and Appendix K to 10 CFR Part 50.

8. The applicant's analysis addresses the full LOCA sequence of events to the point where the plant is in the long-term cooling mode and removal of decay heat has been well established for both large and small breaks. The reviewer checks the assumed sources of coolant water, redundancy of delivery routes, alignment of valves, control of boron concentration (PWR) and all required operator actions.

9. The following TMI Action Plans (Ref. 6 and 7) items are reviewed to assure compliance with the acceptance criteria:

   a. II.E.2.3. The reviewer evaluates the uncertainty analyses performed by the applicant to meet item II.E.2.3 to assure that the modeling assumptions and phenomena for small-break LOCA calculations are properly accounted for to determine the acceptability of the ECCS performance pursuant to Appendix K of 10 CFR Part 50.

   b. II.K.2.8. For Babcock and Wilcox designs, the reviewer confirms that the auxiliary feedwater system upgrade and automatic auxiliary feedwater initiation performed under this TMI action plan item have been properly accounted for in the LOCA analyses.

c. II.K.3.5. The reviewer evaluates the assumptions made regarding reactor coolant pump trip to assure that they are consistent and conservatively modeled with respect to the final pump trip criteria which result from resolution of TMI action plan item II.K.3.5.

d. II.K.3.25 and II.K.3.40. If, as a result of a LOCA, or as a result of loss of A/C power, containment isolation is indicated to occur, the reactor coolant pump component cooling water may be lost. The reviewer evaluates the applicant's submittal to determine that the reactor coolant pump seal integrity is not lost. If it cannot be established that seal integrity is assured, the reviewer assures that the evaluation of this event correctly accounts for seal failure.

e. II.K.3.30 and II.K.3.31. The reviewer evaluates the small-break LOCA model verification performed by the applicant and assures that any modifications required are incorporated into the specific plant analyses.

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

The review of fission product releases and radiological consequences of design basis (most severe) LOCA is performed by AEB as described in the appendix to this SRP section.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and that the review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report:

The staff concludes that the loss-of-coolant analysis resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary is acceptable and meets the relevant requirements of 10 CFR Part 50, §50.46 and Appendix K, GDC 35, and 10 CFR Part 100. This conclusion is based on the following:

The applicant has performed analyses of the performance of the emergency core cooling system (ECCS) in accordance with the Commission's regulations (10 CFR Part 50, §50.46 and Appendix K to 10 CFR Part 50). The analyses considered a spectrum of postulated break sizes and locations and were performed with an evaluation model which had been previously reviewed and approved by the staff as described in ________. The results of the analyses show that the ECCS satisfy the following criteria:

1. The calculated maximum fuel rod cladding temperature does not exceed 2200°F.

2. The calculated maximum local oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of

the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. Calculated changes in core geometry are such that the core remains amenable to cooling.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

6. The applicant has met the requirements of TMI Action Plan items II.E.2.3, II.K.2.8 (B&W), II.K.3.5, II.K.3.25 (BWR, W, or CE), II.K.3.30, II.K.3.31, and II.K.3.40 (B&W).

The radiological consequences meets 10 CFR Part 100 requirements for the postulated spectrum of loss-of-coolant accidents (LOCA) which were evaluated from the viewpoint of site acceptability. For the purposes of this analysis, large fractions of the fission products were assumed to be released from the core even though these releases would be precluded by the performance of the ECCS.

The evaluation findings of the AEB resulting from the reviews detailed in Appendices A, B, and D, as applicable, should be inserted in the safety evaluation report draft at this point. See Appendices A - D for typical findings and conclusions.

The staff concludes that the calculated performance of the emergency core cooling system following a postulated loss-of-coolant accident and the conservatively calculated radiological consequences of such an accident conform to the Commission's regulations and to applicable regulatory guides and staff technical positions and, accordingly, the ECCS is considered acceptable.

V. IMPLEMENTATION

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

Except in those cases in which the applicant proposes an acceptable alternative method for complying with the specific portions of the Commission's regulations, the methods described herein will be used by the staff in its evaluation of conformance with Commission regulations.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 35, "Emergency Core Cooling."

2. 10 CFR Part 50, §50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," and Appendix K to 10 CFR Part 50, "ECCS Evaluation Models."

3. Standard Review Plan Section 6.3, "Emergency Core Cooling System."

4. 10 CFR Part 100, "Reactor Site Criteria."

5. NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems: Resolution of Generic Task Action Plan A-2."

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

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