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)~~(SRXB)[1]

Secondary - ~~Accident Evaluation Branch (AEB)~~Emergency Preparedness and Radiation Protection Branch (PERB)[2]

I. AREAS OF REVIEW

Loss-of-coolant accidents (LOCAs) 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 Criteri~~a~~on 35 ~~(Ref. 1)~~[3] 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 50.46 and Appendix K to 10 CFR Part 50 ~~(Ref. 2)~~[4] and the applicable general design requirements discussed in Standard Review Plan (SRP) Section 6.3 ~~(Ref. 3)~~.[5] 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~~SRXB[6] 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~~SRXB[7] 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.~~[8] 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~~SRXB[9] 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~~SRXB[10] initiates a generic computer program review as required.

The SRXB review of this SRP section covers the following areas:[11]

1. The ~~RSB~~SRXB is ~~also~~[12] responsible for the review of the failure mode 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[13] in conjunction with the effort described in SRP Section 6.3.

2. The ~~RSB~~SRXB[14] reviews the analytical techniques and computer programs used by the applicant for the blowdown, refill, and reflood portions of the loss-of-coolant transient.

3. The ~~RSB~~SRXB[15] also reviews the analytical techniques and computer programs used by the applicant for power transient calculations (including moderator temperature, void and fuel temperature feedback effects, and decay heat) and[16] for the cladding temperature, cladding rupture and swelling calculations.

4. The ~~RSB~~SRXB ~~will~~[17] performs independent audit blowdown, refill, reflood and cladding calculations as required to verify the applicant's conclusions.

5. For a small break loss-of-coolant accident, the SRXB reviews the potential for the addition of un-borated water into the core from reactor coolant pump seals, and the potential for additional core damage caused by reactivity transients from the un-borated water.[18]

6. The SRXB 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.[19]

The PERB review of this SRP section covers the following area:[20]

~~AEB~~PERB[21] 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 safety evaluation report (SER) writeup.

<u>Review Interfaces</u>[22]

The SRXB reviews fuel failure modes and burst correlations for compliance with 10 CFR 50.46 as part of its fuel design review in SRP Section 4.2.[23]

The ~~RSB~~SRXB[24] 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 applicants 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.~~[25] ~~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.~~[26]

1. The Containment Systems and Severe Accident Branch ~~(CSB)~~(SCSB)[27] 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~~SCSB[28] verifies, upon request from ~~RSB~~SRXB,[29] 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~~SRXB,[30] ~~CSB~~SCSB[31] 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.~~[32]

2. The Instrumentation and Controls ~~System~~ Branch ~~(ICSB)~~(HICB)[33] 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~~SRXB,[34] ~~ICSB~~HICB[35] 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.

3. The ~~ICSB~~HICB[36] also, upon request from ~~RSB~~SRXB,[37] reviews the failure modes analysis of the ECCS to verify that an adequate analysis of possible failure modes of ECCS instrumentation and controls[38] equipment and the effect of the failure modes of that equipment[39] on the ECCS performance has been provided.

4. The ~~Power Systems Branch (PSB)~~Electrical Engineering Branch (EELB)[40] review of 8.3.1 and 8.3.2 includes the emergency onsite power functional capabilities. The ~~PSB~~EELB,[41] upon request from ~~RSB~~SRXB,[42] 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.

5. The ~~PSB~~EELB[43] also, upon request from ~~RSB~~SRXB,[44] 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.

6. The Plant Systems Branch (SPLB) upon request from SRXB reviews Chapters 9 and 10 of the applicants SAR including 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.[45] The SPLB also reviews the integrity of the reactor coolant pump seals as part of its primary review responsibility for SRP Section 9.2.2.[46]

7. The Mechanical Engineering Branch ~~(MEB)~~(EMEB)[47] review of SRP Sections ~~3.6 and 3.9~~3.6.2, 3.9.2, 3.9.3, 3.9.4, and 3.9.5[48] includes a review of the effects of the blowdown loads on core support structures and on control rod guide structures. The ~~MEB~~EMEB[49] verifies, upon request from ~~RSB~~SRXB,[50] that the core remains in a coolable geometry following a loss-of-coolant accident and that the control rods can also be inserted.

8. The ~~MEB~~EMEB[51] 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~~SRXB,[52] ~~MEB~~EMEB[53] verifies that acceptable criteria (Ref. 5) have been employed in the design of the reactor coolant system and its supports to prevent failures of the reactor coolant pressure boundary and engineered safety feature equipment in the event of a LOCA.

9. The Human Factors Assessment Branch (HHFB) evaluates plant operating procedures as a part of its review of SRP Section 13.5.2. Upon request from SRXB, HHFB verifies that the plant operating procedures include actions relative to reactor coolant pump trip following small break loss-of-coolant accidents that are based on plant-specific safety evaluations.[54]

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~~Acceptance is[55] based on meeting the relevant requirements of the following regulations:

~~a~~A. 10 CFR ~~Part 50, §~~[56]50.46~~ and Appendix K~~[57] 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~~B. General Design Criterion (GDC)[58] 35 as it relates to redundant ECCS components being provided to adequately cool the core during a loss-of-coolant accident.

~~c~~C. 10 CFR Part 100~~ (Ref. 4)~~[59] 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 the TMI Action Plan requirements~~ task action plan items of NUREG-0718 and -0737~~[60] (Ref. 6 and 7) are as follows:

1. An evaluation of ECCS performance has been performed by the applicant in accordance with an~~ approved~~[61] evaluation model that satisfies the requirements of 10 CFR ~~Part 50, §~~[62]50.46. Regulatory Guide 1.157 and Section I of Appendix K to 10 CFR Part 50 provide guidance on acceptable evaluation models.[63] For the full spectrum of reactor coolant pipe breaks, and taking into consideration requirements for reactor coolant pump operation during a small break loss-of-coolant accident, (Refs. 10, 11, and 12)[64] 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 1200 °C (2200 °F~~,~~[65].

    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.

<u>Technical Rationale</u>[66]

The technical rationale for application of these acceptance criteria to the analysis of the consequences of loss-of-coolant accidents is discussed in the following paragraphs:[67]

1. Compliance with 10 CFR 50.46 requires that light water cooled nuclear power reactors be equipped with an emergency core cooling system designed so that core performance following postulated loss-of-coolant accidents conforms to specified criteria related to limiting core damage.

   The requirements specified in 10 CFR 50.46 provide an acceptable and conservative means of calculation of the consequences of loss-of-coolant accidents from a spectrum of pipe break sizes and locations that have been subject to careful review and experimental verification. If the calculations of the performance of the emergency core cooling system are conducted in accordance with these methods, there is a high level of probability that the acceptance criteria on core performance will not be exceeded and damage to the core and offsite consequences will be minimized. Regulatory Guide 1.157, "Best Estimate Calculations of Emergency Core Cooling System Performance," and Appendix K to 10 CFR Part 50, provide guidance on evaluation models needed to demonstrate compliance with the acceptance criteria. Appendix K also specified documentation required for evaluation models.

   Meeting the requirements outlined in the references provides assurance that following a loss-of-coolant accident the reactor core will remain in a coolable geometry and offsite consequences will be within the guidelines specified in 10 CFR Part 100.[68]

2. Compliance with GDC 35 requires that a means of providing abundant emergency core cooling be provided that will transfer heat from the reactor core in the event of a loss-of-

coolant accident, and that suitable redundancy of components and features is provided so that the safety function can be accomplished assuming a single failure.

GDC 35 specifies that an emergency core cooling system be installed in all nuclear power reactors. SRP Section 15.6.5 specifies the analytical procedures that are to be followed to establish that the ECCS will function to meet acceptance criteria specified in 10 CFR 50.46. 10 CFR part 50, Appendix K and Regulatory Guide 1.157, "Best Estimate Calculations of Emergency Core Cooling System Performance," provide guidance on calculational procedures needed to demonstrate compliance with the acceptance criteria.

Meeting the requirements of GDC 35 will provide assurance that following a loss-of-coolant accident that the reactor core will remain in a coolable geometry and offsite consequences will be within the guidelines specified in 10 CFR Part 100.[69]

3. Compliance with 10 CFR 100, Reactor Site Criteria, describes criteria which guide the Commission in its evaluation of the suitability of proposed sites for nuclear power and testing reactors. Part 100 specifies radiation dose guidelines that should not be exceeded in the event of postulated accidents including loss-of-coolant accidents.

In order to satisfy the requirements of 10 CFR 100, the applicant must demonstrate that the offsite doses resulting from various accidents presented in the SAR are within the guideline values. Meeting the guideline doses is achieved by a combination of engineered safety features installed in the nuclear facility, an effective emergency core cooling system, and siting the nuclear plant in an area that does not exceed population density requirements.

Meeting the nuclear power plant siting criteria provides a level of assurance that the plant will pose no undue risk to the public as a result of the consequences of loss-of-coolant accidents.[70]

III. REVIEW PROCEDURES

The procedures below are used during ~~both~~ the construction permit (CP), ~~and~~ standard design certification, combined license (COL), and[71] operating license (OL) reviews. During the CP review, the values of system parameters setpoints used in the analysis will be preliminary in nature and subject to change. At the OL, COL, or the standard design certification[72] 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~~[73] evaluation model as specified in 10 CFR 50.46 following the guidance of Appendix K, Section I, or Regulatory Guide 1.157.[74] 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~~[75] 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, similar 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 including requirements for reactor coolant pump trip during small break loss-of-coolant accidents as required by Generic Letters 85-12, 86-05, and 86-06.[76]

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, COL, or standard design certification review[77]). 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~~[78] 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~~SCSB[79]); 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,~~[80] 50.46(b)[81] ~~and Appendix K to 10 CFR Part 50~~.[82]

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 steps shall be included in PWR emergency operating procedures as a condition for reactor coolant pump startup after a small break loss-of-coolant accident:

a. Verify adequate single phase natural circulation,

b. If single phase natural circulation cannot be established, verify adequate two phase natural circulation,

c. Determine if reactor coolant pump restart is needed and desired, and

d. Verify that all reactor coolant pump restart criteria are met.[83]

10. 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 10 CFR 50.46 and Regulatory Guide 1.157[84] or 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. Generic Letters 85-012, 86-005, and 86-006 provide guidance on implementation of TMI Action Item II.K.3.5.[85]

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~~PERB[86] as described in the appendix to this SRP section.

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[87]

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,~~[88] 50.46~~ and Appendix K~~,[89] 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 __________~~.that follows the guidance contained in Regulatory Guide 1.157 or Section I of Appendix K to Part 50 and meets the requirements of 10 CFR 50.46.[90] The results of the analyses show that the ECCS satisfy the following criteria:

1. The calculated maximum fuel rod cladding temperature does not exceed 1200 ^{o}C (2200 ^{o}F).[91]

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 (Generic Letters 85-012, 86-005, and 86-006 provide

guidance on implementation of TMI Action Item II.K.3.5),[92] 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 meet 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~~PERB[93] 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, B, and 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.

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.[94]

V. IMPLEMENTATION

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

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52.[95] 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.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section.[96]

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."

8. Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance."[97]

9. Generic Letter 85-012, "Implementation of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" for Westinghouse Designed Nuclear Steam Supply Systems."[98]

10. Generic Letter 86-005, Implementation of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" for Babcox and Wilcox Designed Nuclear Steam Supply Systems."[99]

11. Generic Letter 86-006, "Implementation of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" for Combustion Engineering Designed Nuclear Steam Supply Systems."[100]

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SRP Draft Section 15.6.5

Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	Current PRB abbreviation	Change made to reflect current PRB abbreviation, SRXB.
2.	Current SRB name and abbreviation	Change made to reflect current SRB name, Emergency Preparedness and Radiation Protection Branch, and abbreviation, PERB.
3.	SRP-UDP format item	Changed "Criteria" to "Criterion" and deleted (Ref. 1).
4.	SRP-UDP format item	Deleted (Ref. 2).
5.	SRP-UDP format item	Defined SRP and deleted (Ref. 3).
6.	Current PRB abbreviation	Change made to reflect current PRB abbreviation, SRXB.
7.	Current PRB abbreviation	Change made to reflect current PRB abbreviation, SRXB.
8.	Integrated Impact No. 578	The September 16, 1988 revision to 10 CFR 50.46 requires the use of an acceptable evaluation model, and not necessarily one that has been previously documented and reviewed and approved by the staff.
9.	Current PRB abbreviation	Change made to reflect current PRB abbreviation, SRXB.
10.	Current PRB abbreviation	Change made to reflect current PRB abbreviation, SRXB.
11.	SRP-UDP format item	Added lead-in sentence to paragraphs describing SRXB review responsibilities for this SRP section and put in numbered paragraph form.
12.	Current PRB abbreviation	Change made to reflect current PRB abbreviation, SRXB.
13.	SRP-UDP format item	Amplified objective of failure mode analysis to capture the SRXB review requirement that was previously the responsibility of ASB and deleted from "Review Interfaces".
14.	Current PRB abbreviation	Change made to reflect current PRB abbreviation, SRXB.
15.	Current PRB abbreviation	Change made to reflect current PRB abbreviation, SRXB.
16.	SRP-UDP format item	Added this item as a review responsibility for SRXB which was deleted from "Review Interfaces".
17.	Current PRB abbreviation	Change made to reflect current PRB abbreviation, SRXB.

SRP Draft Section 15.6.5
Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
18.	Integrated Impact No. 577	Added reference to the potential for addition of un-borated water into a PWR core during a small break loss-of-coolant accident.
19.	SRP-UDP format item	Added this item as a review responsibility for SRXB which was deleted from "Review Interfaces".
20.	SRP-UDP format item	Added lead-in sentence to paragraph describing PERB review responsibility for this SRP section.
21.	Current SRB abbreviation	Change made to reflect current SRB abbreviation, PERB.
22.	SRP-UDP format item	Added "Review Interfaces" to "Areas of Review" and put in numbered paragraph form to describe how other branches support the SRXB review of the LOCA.
23.	**Integrated Impact 578**	Added a Review Interface to SRP 4.2 for review of fuel failure modes and burst correlations for compliance with 10 CFR 50.46.
24.	Current PRB abbreviation	Change made to reflect current PRB abbreviation, SRXB.
25.	SRP-UDP format item	The SPLB has review responsibility for Chapters 9 and 10 of the SAR. Sentence moved to Item 6 of Review Interfaces.
26.	SRP-UDP format item	The SRXB now has review responsibility for these sections.
27.	Current review branch name and abbreviation	Change made to reflect current review branch name, Containment Systems and Severe Accident Branch, and abbreviation, SCSB.
28.	Current review branch abbreviation	Change made to reflect current review branch abbreviation, SCSB.
29.	Current review branch abbreviation	Change made to reflect current review branch abbreviation, SRXB.
30.	Current review branch abbreviation	Change made to reflect current PRB abbreviation, SRXB.
31.	Current review branch abbreviation	Change made to reflect current PRB abbreviation, SCSB.
32.	SRP-UDP format item	The SRXB now has review responsibility for these sections.
33.	Current review branch name and abbreviation	Change made to reflect current review branch name, Instrumentation and Controls Branch, and abbreviation, HICB.
34.	Current PRB abbreviation	Change made to reflect current PRB abbreviation, SRXB.

SRP Draft Section 15.6.5
Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
35.	Current review branch abbreviation	Change made to reflect current review branch abbreviation, HICB.
36.	Current review branch abbreviation	Change made to reflect current review branch abbreviation, HICB.
37.	Current PRB abbreviation	Change made to reflect current PRB abbreviation, SRXB.
38.	Editorial	Addition made to specify review responsibility of HICB.
39.	Editorial	Added phrase to clarify review responsibility of HICB.
40.	Current review branch name and abbreviation	Change made to reflect current review branch name, Electrical Engineering Branch, and abbreviation, EELB.
41.	Current review branch abbreviation	Change made to reflect current review branch abbreviation, EELB.
42.	Current PRB abbreviation	Change made to reflect current PRB abbreviation, SRXB.
43.	Current review branch abbreviation	Change made to reflect current review branch abbreviation, EELB.
44.	Current PRB abbreviation	Change made to reflect current PRB abbreviation, SRXB.
45.	SRP-UDP format item	Change made to reflect SPLB responsibility for SAR Chapters 9 and 10.
46.	Editorial	Added a review interface with SRP Section 9.2.2. SRP Section 15.6.5 contains Acceptance Criteria, Review Procedures and Evaluation Findings with regard to reactor coolant pump seal integrity issues associated with TMI Action Items II.K.3.25 and II.K.3.40 (superseded by II.K.2.16). Reactor coolant pump seal integrity and conformance with these TMI Action Items are reviewed in SRP Section 9.2.2.
47.	Current review branch abbreviation	Change made to reflect current review branch abbreviation, EMEB.
48.	Editorial	Corrected references to SRP sections.
49.	Current review branch abbreviation	Change made to reflect current review branch abbreviation, EMEB.
50.	Current PRB abbreviation	Change made to reflect current PRB abbreviation, SRXB.
51.	Current review branch abbreviation	Change made to reflect current review branch abbreviation, EMEB.
52.	Current PRB abbreviation	Change made to reflect current PRB abbreviation, SRXB.

Item	Source	Description
53.	Current review branch abbreviation	Change made to reflect current review branch abbreviation, EMEB.
54.	Integrated Impact No. 579	Included in Areas of Review the requirement that SRXB request HHFB to review operation procedures relative to reactor coolant pump trip following a loss-of-coolant accident based on plant-specific safety evaluations.
55.	Editorial	Changed to standard lead-in sentence.
56.	Editorial	Revised designation to 10 CFR 50.46.
57.	Integrated Impact No. 578	Deleted "and Appendix K to 10 CFR Part 50" because the September 16, 1988 revision to 10 CFR 50.46 allows evaluation models that realistically describe the behavior of the reactor system during loss-of-coolant accidents in addition to those specified in Appendix K.
58.	Editorial	Added abbreviation for General Design Criteria (GDC).
59.	SRP-UDP format item	Deleted (Ref. 4).
60.	Editorial	Revised designation of TMI Action Plan requirements.
61.	Integrated Impact No. 578	Deleted "approved" because the September 16, 1988 revision to 10 CFR 50.46 allows evaluation models that realistically describe the behavior of the reactor system during loss-of-coolant accidents. These evaluation models need not have been approved prior to the licensing action under consideration.
62.	Editorial	Revised designation to 10 CFR 50.46.
63.	Integrated Impact No. 578	Revised specific criteria to indicate that RG 1.157 and Section I of Appendix K to 10 CFR Part 50 both represent guidance for acceptable evaluation models, consistent with the September 16, 1988 revision to 10 CFR 50.46 which allows evaluation models that realistically describe the behavior of the reactor system during loss-of-coolant accidents in addition to those specified in Appendix K.
64.	Integrated Impact No. 579	Added consideration of requirements for reactor coolant pump operation during a small break loss-of-coolant accident.
65.	SRP-UDP format item	Conversion to SI units.
66.	SRP-UDP format item	"Technical Rationale" added to "Acceptance Criteria" subsection and put in numbered paragraph form to describe the bases for referencing the regulations and the GDC.
67.	SRP-UDP format item	Added lead-in sentence to "Technical Rationale."

Item	Source	Description
68.	SRP-UDP format item	Added technical rational for 10 CFR 50.46 and 10 CFR Part 50, Appendix K.
69.	SRP-UDP format item	Added technical rationale for GDC 35.
70.	SRP-UDP format item	Added technical rationale for 10 CFR Part 100.
71.	SRP-UDP format item	Added reference to combined license and standard design certification reviews per 10 CFR Part 52.
72.	SRP-UDP format item	Added reference to combined license and standard design certification reviews per 10 CFR Part 52.
73.	Integrated Impact No. 578	Deleted "approved" because the September 16, 1988 revision to 10 CFR 50.46 allows evaluation models that realistically describe the behavior of the reactor system during loss-of-coolant accidents. These evaluation models need not have been approved prior to the licensing action under consideration.
74.	Integrated Impact No. 578	Added reference to the acceptable evaluation models identified in 10 CFR 50.46.
75.	Integrated Impact No. 578	Deleted reference to an approved evaluation model because the September 16, 1988 revision to 10 CFR 50.46 allows evaluation models that realistically describe the behavior of the reactor system during loss-of-coolant accidents. These evaluation models need not have been approved prior to the licensing action under consideration.
76.	Integrated Impact No. 579	Added reference to requirements for reactor coolant pump trip during loss-of-coolant accidents.
77.	SRP-UDP format item	Added reference to COL and standard design certification reviews per 10 CFR Part 52.
78.	Editorial	Deleted reference to Office of Inspection and Enforcement.
79.	Current review branch abbreviation	Change made to reflect current review branch abbreviation, SCSB.
80.	Editorial	Revised designation of 10 CFR 50.46.
81.	Integrated Impact No. 578	Added (b) because the acceptance criteria are in 10 CFR 50.46(b).
82.	Integrated Impact No. 578	Deleted "and Appendix K to 10 CFR Part 50" because the September 16, 1988 revision to 10 CFR 50.46 allows evaluation models that realistically describe the behavior of the reactor system during loss-of-coolant accidents in addition to those specified in Appendix K.

Item	Source	Description
83.	Integrated Impact No. 577	Added steps to be taken prior to PWR reactor coolant pump restart after a small break loss-of-coolant accident.
84.	Integrated Impact No. 578	Added Regulatory Guide 1.157 which provides modeling procedures acceptable to the NRC staff for meeting ECCS performance criteria in addition to that specified in 10 CFR 50.46.
85.	Integrated Impact No. 579	Added reference to Generic Letters 85-012, 86-005, and 86-006 which describe implementation of TMI Task Action II.K.3.5.
86.	Current SRB abbreviation	Change made to reflect current SRB abbreviation, PERB.
87.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
88.	Editorial	Revised designation of 10 CFR 50.46.
89.	Integrated Impact No. 578	Deleted "and Appendix K" because the September 16, 1988 revision to 10 CFR 50.46 allows evaluation models that realistically describe the behavior of the reactor system during loss-of-coolant accidents in addition to those specified in Appendix K.
90.	Integrated Impact No. 578	Revised statement to reflect meeting the guidance of Regulatory Guide 1.157 or Appendix K Section I and the requirements of 10 CFR 50.64.
91.	SRP-UDP format item	Conversion to SI units.
92.	Integrated Impact No. 579	Added reference to Generic Letters 85-012, 86-005, and 86-006 which describe implementation of TMI Task Action II.K.3.5.
93.	Current SRB abbreviation	Change made to reflect current SRB abbreviation, PERB.
94.	SRP-UDP Format Item, Implement 10 CFR 52 Related Changes	To address design certification reviews a new paragraph was added to the end of the Evaluation Findings. This paragraph addresses design certification specific items including ITAAC, DAC, site interface requirements, and combined license action items.
95.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard sentence to address application of the SRP section to reviews of applications filed under 10 CFR Part 52, as well as Part 50.
96.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
97.	Integrated Impact No. 578	Added Regulatory Guide 1.157 to references.

Item	Source	Description
98.	Integrated Impact No. 579	Added Generic Letter 85-012 to references.
99.	Integrated Impact No. 579	Added Generic Letter 86-005 to references.
100.	Integrated Impact No. 579	Added Generic Letter 86-006 to references.

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SRP Draft Section 15.6.5
Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
577	Address restrictions on reactor coolant pump restart after loss-of-coolant accidents to avoid the potential for core damage caused power surges resulting from boron dilution.	Section I, AREAS OF REVIEW, eighth paragraph. Section III, REVIEW PROCEDURES, second paragraph, subparagraph number 9.
578	Address the option to use an acceptable best estimate evaluation model in lieu of the model specified 10 CFR Part 50, Appendix K.	Section I, AREAS OF REVIEW, fourth paragraph. Section II, ACCEPTANCE CRITERIA, first paragraph, subparagraph number 1. Section II, ACCEPTANCE CRITERIA, second paragraph, subparagraph number 1. Section III, REVIEW PROCEDURES, second paragraph, subparagraph number 1. Section III, REVIEW PROCEDURES, second paragraph, subparagraph number 4. Section III, REVIEW PROCEDURES, second paragraph, subparagraph number 7. Section III, REVIEW PROCEDURES, second paragraph, subparagraph number 10.a. Section IV, EVALUATION FINDINGS, first paragraph, first subparagraph. Section IV, EVALUATION FINDINGS, first paragraph, second subparagraph. Section IV, EVALUATION FINDINGS, first paragraph, second subparagraph, paragraph number 6. Section VI, REFERENCES, reference 8.

Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
579	Address staff guidance relative to reactor coolant pump trip criteria during a loss-of-coolant accident.	Section I, AREAS OF REVIEW, Review Interfaces, subparagraph number 9. Section II, ACCEPTANCE CRITERIA, second paragraph, subparagraph number 1. Section, III REVIEW PROCEDURES, second paragraph, subparagraph number 4.f. Section, III REVIEW PROCEDURES, second paragraph, subparagraph number 10.c. Section IV, EVALUATION FINDINGS, first paragraph, second subparagraph, paragraph number 6. Section VI, REFERENCES, reference 9. Section VI, REFERENCES, reference 10. Section VI, REFERENCES, reference 11.