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(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.2.8 FEEDWATER SYSTEM PIPE BREAKS INSIDE AND OUTSIDE CONTAINMENT (PWR)

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch ~~(RSB)~~(SRXB)[1]

Secondary - ~~Accident Evaluation Branch (AEB)~~Emergency Preparedness and Radiation Protection Branch (PERB)[2]
Instrumentation and Controls Branch (HICB)[3]

I. AREAS OF REVIEW

The ~~transient~~anticipated operational occurrence (AOO)[4] that results from a postulated feedwater line break is sensitive to the break discharge rate; consequently, a range of break sizes should be evaluated both inside and outside containment to determine the acceptability of the response. Depending upon the size and location of the break and the plant operating conditions at the time of the break, the break could cause either a reactor coolant system (RCS)[5] cooldown (by excessive energy discharge through the break or a ~~reactor coolant system~~RCS heatup (by reducing feedwater flow to the affected steam generator). Therefore, analyses of various postulated break sizes and locations are needed to identify the particular situation that is most limiting with respect to system effects.

If a feedwater line rupture causes the water in the steam generator to be discharged through the break, the water will not be available for decay heat removal after reactor scram. The break location and size may be such to prevent addition of any feedwater to the affected steam generator. An auxiliary feedwater system (AFWS)[6] is therefore provided to ~~assure~~ensure[7] that feedwater is available to provide decay heat removal.

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

The review includes:

1. evaluation of the applicant's postulated initial core and reactor conditions pertinent to the feedwater line break,

2. the methods of thermal and hydraulic analysis, the postulated sequence of events, including analyses to determine the time of reactor trip and time delays prior and subsequent to initiation of reactor protection system (RPS)[8] actions,

3.[9] the assumed response of the reactor coolant and auxiliary systems, the functional and operational characteristics of the ~~reactor protection system~~(RPS) in terms of its effects on the sequence of events, and all operator actions required to secure and maintain the reactor in a safe shutdown condition.

The results of the analyses are reviewed to ensure that the values of pertinent system parameters, discussed in subsection II below, are within expected ranges. The parameters of importance for these transients include:

1. ~~reactor coolant system~~RCS pressure,

2. steam generator pressure,

3. fluid temperatures,

4. fuel and clad temperatures,

5. break discharge flow rate,

6. steamline and feedwater flow rates,

7. safety and relief valve flow rates,

8. pressurizer and steam generator water levels,

9. mass and energy transfer within the containment (for breaks inside containment),

10. reactor power,

11. total core reactivity,

12. hot and average channel heat flux, and

13.[10] minimum departure from nucleate boiling ratio (DNBR).

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both ~~RSB~~SRXB and ~~ICSB~~HICB.[11] The ~~RSB~~SRXB and HICB[12] reviewers concentrates on the need for the ~~reactor protection system~~(RPS), the engineered safety systems (ESS),[13] and operator action to secure and maintain the reactor in a safe conditions.

The analytical methods are reviewed by ~~RSB~~SRXB 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, the reviewer requests initiation of a generic evaluation of the new analytical model. ~~RSB~~SRXB ~~also~~ reviews the values of all the parameters used in the analytical model. ~~CPB~~SRXB[14] also[15] reviews the initial conditions of the core and all nuclear design parameters. This includes:

1. power level,
2. power distribution,
3. Doppler coefficients,
4. moderator temperature coefficients,
5. void coefficients,
6. reactor kinetics,
7. DNB correlations, and
8.[16] control rod worth.

In addition, SRXB reviews the auxiliary feedwater system to confirm that the flow provided is acceptable for controlling the transient following a feedwater line break.[17]

~~A secondary review is performed by the Accident Evaluation Branch (AEB) and the results are used by ASB to complete the overall evaluation of the break analysis. The AEB evaluates the fission product release assumptions used in determining any offsite releases and verifies that the radiological consequences resulting from a feedwater pipe break are within acceptable limits as part of its primary review responsibility for SRP Section 15.6.5. The result of AEB's analysis is transmitted to RSB for use in the SER writeup. The ICSB reviewer concentrates on the instrumentation and control aspects of the sequence described in the SAR 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. ICSB also evaluates potential bypass modes and the possibility of manual control by the operator as part of its primary reactor responsibility for SRP Sections 7.1 through 7.7.~~

~~In addition the RSB will coordinate other branches' evaluations that interface with the overall review of feedwater system pipe breaks as follows: The Auxiliary Systems Branch (ASB) reviews the auxiliary feedwater system to verify that it can function following a feedwater line break, given a single active component failure and with either onsite or offsite power as part of its primary review responsibility for SRP Section 10.4.9. RSB reviews the auxiliary feedwater~~

~~system to confirm that the flow provided is acceptable for controlling the transient following a feedwater line break. The Mechanical Engineering Branch (MEB) evaluates potential water-hammer effects on safety valve integrity as part of its primary review responsibility for SRP Section 3.9 series. The Containment Systems Branch (CSB) reviews the methodology which evaluates the response of the containment to breaks of feedwater lines with regard to the effects of pressure and temperature on the containment functional capabilities as part of its primary review responsibility for SRP Section 6.2.1. The ICSB reviewer concentrates on the instrumentation and control aspects of the sequence described in the SAR 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. ICSB also evaluates potential bypass modes and the possibility of manual control by the operator as part of its primary reactor responsibility for SRP Sections 7.1 through 7.7.~~

Review Interfaces[18]

1. The SRXB also performs the following reviews under the Standard Review Plan (SRP)[19] sections indicated:

   a. SRP Section 4.4 for the evaluation of potential for core damage on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit for PWRs based on acceptable correlations.

   b. SRP Section 4.2, for an acceptable fuel damage model.

   c. SRP Section 10.4.9, AFWS to confirm that the flow provided is acceptable for controlling the transient following the feedwater line break.

   d. SRP Sections for Chapters 5, 6, 7, 8, and 10 to evaluate the effect of single active failures of systems and components that may alter the course of the accident, to the extent considered necessary, using the system review procedures described in the standard review plan sections.[20]

2. In addition the ~~RSB~~SRXB will coordinate other branches' evaluations that interface with the overall review of feedwater system pipe breaks as follows:

   a. The ~~Auxiliary Systems Branch (ASB)~~Plant Systems Branch (SPLB)[21] reviews the ~~auxiliary feedwater system~~ AFWS to verify that it can function following a feedwater line break, given a single active component failure and with either onsite or offsite power as part of its primary review responsibility for SRP Section 10.4.9. The SPLB evaluates the ability of the AFWS to supply adequate feedwater flow to the unaffected steam generators during the accident and subsequent shutdown.[22] SPLB also reviews, under SRP Section 10.4.9, how the applicant has met the requirements of 10 CFR 50.34(f)(1)(ii).[23] 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.[24]

b. The Mechanical Engineering Branch ~~(MEB)~~(EMEB)[25] evaluates potential water-hammer effects on safety valve integrity as part of its primary review responsibility for SRP Section 3.9 series.

c. The Containment Systems and Severe Accidents Branch ~~(CSB)~~ (SCSB)[26] reviews the ~~auxiliary feedwater system~~AFWS to confirm that the flow provided is acceptable for controlling the transient following a feedwater line break. SCSB reviews the methodology which evaluates the response of the containment to breaks of feedwater lines with regard to the effects of pressure and temperature on the containment functional capabilities as part of its primary review responsibility for SRP Section 6.2.1.

d. A secondary review is performed by the ~~Accident Evaluation Branch (AEB)~~PERB and the results are used by ~~ASB~~SPLB to complete the overall evaluation of the break analysis.

e. The ~~AEB~~PERB evaluates the fission product release assumptions used in determining any offsite releases and verifies that the radiological consequences resulting from a feedwater pipe break are within acceptable limits as part of its ~~primary~~secondary[27] review responsibility for SRP Section 15.6.5. The result of ~~AEB's~~PERB's[28] analysis is transmitted to ~~RSB~~SRXB for use in the SER writeup.

f. The ~~ICSB~~HICB reviewer concentrates on the instrumentation and control aspects of the sequence described in the SAR 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. The HICB review includes an evaluation of the instrumentation and controls required to ensure automatic and manual auxiliary feedwater system initiation and flow indication in the control room in accordance with the requirements of 10 CFR 50.34(f)(2)(xii), as part of its primary review responsibility for SRP Section 7.1.[29] ~~ICSB~~HICB ~~also~~ evaluates potential bypass modes and the possibility of manual control by the operator as part of its primary reactor responsibility for SRP Sections 7.1 through 7.7. HICB also reviews the credit taken for a reactor trip signal or for actuation of ESS to determine the ability of the instrumentation and control systems to respond as assumed under accident conditions.[30]

g. The Materials and Chemical Engineering Branch (EMCB) reviews the fracture toughness properties of the reactor coolant pressure boundary and reactor vessel as part of its primary review responsibility for SRP Sections 5.2.3 and 5.3.1.[31]

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 sections of the corresponding primary branch.

II. ACCEPTANCE CRITERIA

The basic objective of the review of feedwater system pipe break events is to confirm that the reactor primary system is maintained in a safe status for a range of feedwater line breaks, up to and including a break equivalent in area to the double-ended rupture of the largest feedwater line.

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

A. General Design Criterion 17 (GDC 17) as it relates to providing onsite and offsite electric power systems to ensure that structures, systems, and components important to safety will function. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that design conditions of the reactor coolant pressure boundary are not exceeded and the core is cooled in the event of postulated accidents.[32]

B. General Design Criteria 27 (GDC 27) and 28 (GDC 28),[33] as they relate to the reactor coolant system being designed with appropriate margin to ~~assure~~ensure that acceptable fuel design limits are not exceeded, and that the capability to cool the core is maintained.

C. General Design Criterion 31 (GDC 31),[34] as it relates to the ~~reactor coolant system~~RCS being designed with sufficient margin to ~~assure~~ensure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized.

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

E. 10 CFR Part 100, as it relates to the calculated doses at the site boundary.

~~In addition, task action plan items that are necessary to meet the requirements to maintain adequate decay heat removal and reactor coolant pump (RCP) integrity and operation are items II.E.1, II.K.2.1, II.E.1.2, II.K.2.8, II.K.3.5, II.K.2.16, and II.K.3.25, and II.K.3.40 of NUREG-0718 and NUREG-0737 (Refs. 6 and 7).~~ Requirements related to maintaining adequate decay heat removal, and reactor coolant pump integrity and operation are contained in 10 CFR 50.34(f)(1)(ii), 10 CFR 50.34(f)(2)(xii), and 10 CFR 50.34(f)(1)(iii). In addition, Task Action Plan item II.K.3.5 of NUREG-0737 addresses automatic trip of reactor coolant pumps during a loss-of-coolant accident. The resolution of this issue is contained within Generic Letters 83-10A through 83-10F, 85-12, 86-05, and 86-06.[36] Specific criteria necessary to meet the relevant requirements of these regulations are as follows:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design pressures ~~(Ref. 3)~~(ASME Boiler and Pressure Vessel Code, Section III)[37] for low probability events and below 120% of the design pressures for very low probability events such as double-ended guillotine breaks.

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. Any activity release must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines.

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

5. The ~~auxiliary feedwater system~~AFWS must be safety grade and automatically initiated when required.

6. Tripping of the ~~reactor coolant pumps~~RCPs should be consistent with the resolution to TMI Action Plan Item II.K.3.5.

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

a. The power level assumed and number of loops operating 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 nuclear steam supply system 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 if considered applicable.

b. The assumptions as to whether offsite power is lost and the time of loss should be made conservatively. Offsite power may be lost simultaneously with the occurrence of the pipe break, the loss may occur during the accident, or offsite power may not be lost. A study should be made to determine the most conservative assumption appropriate to the plant design being reviewed. The study should take account of the effects that loss of offsite power has on reactor coolant and main feedwater pump trips and on the initiation of auxiliary feedwater, and the resulting modification of the sequence of events.

c. The effects of the postulated feedwater line breaks on other systems (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) should be considered in a manner consistent with the intent of Branch Technical Positions ASB 3-1 and MEB 3-1 ~~(Ref. 5)~~.[38]

d. The worst single active component failure should be assumed to have occurred in the systems required to control the transient. For new applications, loss of offsite power

(LOOP) should not be considered a single failure; feedwater pipe breaks should be analyzed with and without LOOP, as discussed in assumption b above, in combination with a single active failure. (This position is based upon interpretation of GDC 17, as documented in the Final Safety Evaluation Report for the ABB-CE System 80+ design certification.)[39]

e. The maximum rod worth should be assumed to be held in the fully withdrawn position, per GDC 25. An appropriate rod reactivity worth versus rod position curve should be assumed.

f. 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.

g. The initial core flow assumed for the analysis of the feedwater line rupture accident should be chosen conservatively. If the minimum core flow allowed by the technical specifications is assumed, the minimum DNBR margin results for the case of a feedwater line rupture inside containment. However, this may not be the most conservative assumption. For example, maximum initial core flow results in increased ~~reactor coolant system~~RCS 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 applicant's assumption should be justified by appropriate sensitivity studies.

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

Technical Rationale[40]

The technical rationale for application of these acceptance criteria to reviewing the feedwater system pipe breaks inside and outside containment (PWR) is discussed in the following paragraphs:[41]

1. The requirements of 10 CFR Part 100 specify how the exclusion area, low population zone, and population center distance should be determined. Further, radiation exposure criteria stipulated in 10 CFR Part 100 provide reference values to be used in the site suitability determination based on postulated fission product releases associated with accidental events.

   10 CFR Part 100 is applicable to this section because it specifies the methodology for calculating radiation exposures at the site boundary for postulated accidents or events such as loss of a reactor coolant pump. For transients having a moderate frequency of occurrence, any release of radioactive material must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines. A small fraction is interpreted to be less than 10 percent of the 10 CFR Part 100 reference values. For the purpose of this review, the radiological consequences of any feedwater system

pipe break must include consideration of the containment, confinement, and filtering systems. The applicant's source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing should reflect NRC-approved source terms and methodologies.

Meeting this requirement provides assurance that, in the event of a feedwater system pipe break, radiation exposures at the site boundary will not exceed a small fraction of the reference values specified in 10 CFR Part 100.[42]

2. Compliance with GDC 17 requires (in part) that an onsite and an offsite electric power system be provided to permit 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 assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment and other vital functions are maintained in the event of postulated accidents.

   GDC 17 is applicable to SRP Section 15.2.8 because this section reviews feedwater system pipe breaks which can be classed as AOOs or accidents, depending upon severity.

   Meeting the requirements of GDC 17 provides assurance that design conditions of the reactor coolant pressure boundary are not exceeded as a result of feedwater system pipe breaks inside and outside containment concurrent with LOOP, the core is cooled and containment and other vital functions are maintained.[43]

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

   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 (PWR) (unless prevented by positive means), rod drop (BWR), steam line rupture, changes in reactor temperature and pressure and cold water addition.

   GDC 27 and GDC 28 are applicable to SRP Section 15.2.8 because this section involves the review of feedwater system pipe breaks inside and outside containment that can result in transient conditions affecting reactor coolant temperature and pressure, with resultant 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 core cooling. The analyses must be independently reviewed by the staff in accordance with this SRP section.

Meeting the requirements of GDC 27 and GDC 28 provides assurance that an AOO initiated by feedwater system pipe breaks will not result in (a) an unacceptable stress on the reactor coolant pressure boundary, (b) acceptable fuel design limits being exceeded, or (c) the incapability of the core cooling systems to cool the core or the reactivity control systems to perform their design safety functions.[44]

4. Compliance with GDC 31 requires that the reactor pressure boundary be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions (a) the boundary behaves in a nonbrittle manner and (b) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (a) material properties, (b) the effects of irradiation on material properties, (c) residual, steady state, and transient stresses, and (d) size of flaws.

   GDC 31 is applicable to SRP Section 15.2.8 because this section involves the review of feedwater system pipe breaks inside and outside containment that could result in transient reactor coolant temperature and pressure conditions having the potential for adversely affecting the reactor coolant pressure boundary. A feedwater system pipe break could cause either a reactor coolant system cooldown by excessive energy discharge through the break or a reactor coolant system heatup by reducing feedwater flow to the affected steam generator. Heat-up of the reactor coolant by reducing feedwater flow to the affected steam generator and subsequently by the addition of decay heat, could cause undue stress on the reactor coolant system pressure boundary. The amount of stress to which the reactor coolant pressure boundary is subjected is dependent upon the severity of the AOO. The severity of the AOO is assessed by the applicant in their SAR and reviewed by the staff in accordance with this SRP section.

   Meeting the requirements of GDC 31 provides assurance that (a) the reactor coolant pressure boundary behaves in a nonbrittle manner and (b) the probability of rapidly propagating fracture is minimized.[45]

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

   GDC 35 is applicable to SRP Section 15.2.8 because this section involves the review of feedwater system pipe breaks both inside and outside containment that could result in transient reactor coolant temperature conditions having the potential for challenging the emergency core cooling system (ECCS). A feedwater system pipe break could cause either a reactor coolant system cooldown by excessive energy discharge through the break or a reactor coolant system heatup by reducing feedwater flow to the affected steam generator. Heat-up of the reactor coolant by reducing feedwater flow to the affected steam generator and subsequently by the addition of decay heat, could potentially initiate the ECCS to reduce the core coolant temperature to an acceptable level so that fuel and clad damage that could interfere with continued effective core cooling is prevented and that fuel clad metal-water reaction is limited to negligible amounts. The severity of this AOO is assessed by the applicant in their SAR and reviewed by the staff in accordance with this SRP section.

Meeting the requirements of GDC 35 provides assurance that a system will be available for providing abundant emergency core cooling, thus preventing fuel and clad damage that could interfere with continued effective core cooling and limiting any clad metal-water reaction to negligible amounts in the event of an AOO initiated by a feedwater system pipe break.[46]

III. REVIEW PROCEDURES

The procedures below are used during reviews of ~~both~~ construction permit (CP), ~~and~~ operating license (OL), and combined license (COL)[47] applications. 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.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by ~~RSB~~SRXB and are compared to the initial conditions listed in subsection II of this SRP section. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. ~~CPB is consulted regarding~~SRXB also reviews the values of reactivity parameters used in the applicant's analysis.[48]

Analytical models should be of sufficient detail to simulate the reactor coolant (primary), steam generator (secondary), and auxiliary systems. The equations, sensitivity studies, ~~and~~ models proposed by the applicant, and justification that the methods used are conservative in comparison with the semi-scale test data from NUREG/CR-4945[49] are reviewed by ~~RSB~~SRXB.

Credit taken for a reactor trip signal or for actuation of ~~engineered safety features~~ESS should be reviewed by ~~ICSB~~HICB to determine the ability of the instrumentation and control systems to respond as assumed under accident conditions.

The ability of the ~~auxiliary feedwater system~~AFWS to supply adequate feedwater flow to the unaffected steam generators during the accident and subsequent shutdown is evaluated by ~~ASB~~SPLB[50] as to availability and by ~~RSB~~SRXB as to capability to effect an orderly shutdown. Since ~~auxiliary feedwater system~~AFWS designs are diverse and may require both automatic and manual actuation, preoperational tests should be specified to identify any necessary operator actions and to determine the maximum times permitted for their completion.

To the extent considered necessary, the ~~RSB~~SRXB reviewer evaluates the effect of single active failures of systems and components that may alter the course of the accident. For new applications, the LOOP is not considered a single active failure, but considered in addition to a single active failure as discussed in subsection II, assumption d.[51] This phase of the review uses the system review procedures described in the standard review plan sections for Chapters 5, 6, 7, 8, and 10 of the SAR. The variations with time during the transient of parameters listed in Sections 15.X.X.3(C) and 15.X.X.4(C) of the Standard Format, ~~(Ref. 2)~~Regulatory Guide 1.70,[52] are reviewed. The more important of these parameters for the feedwater line break accident (as listed in subsection I of this SRP section) are compared to those predicted for other similar plants to see that they are within the expected range.

The reviewer confirms that the amount of secondary coolant expelled from the system has been calculated conservatively by evaluating the applicant's methods and assumptions, by comparison with an acceptable analysis performed on another plant of similar design, or by comparison with staff calculations for typical plants which will be available from ~~RSB~~SRXB on request.

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

Using the information developed in the review, the ~~AEB~~PERB reviewer evaluates the radiological consequences of the design basis feedwater line break. This evaluation based on a qualitative comparison with the results of the design basis steam line break, or on a detailed analysis using the approach described in the appendix to SRP Section 15.1.5.

The reliability and operability of the ~~auxiliary feedwater systems (~~AFWS~~)~~ are reviewed to ~~assure~~ensure ~~conformance to the following TMI Action Plan Items (References 6 and 7)~~compliance with the requirements of 10 CFR 50.34(f)(1)(ii) and 10 CFR 50.34(f)(2)(xii) as they relate to ~~auxiliary feedwater system~~AFWS performance requirements following feedwater piping failures.

~~(a) Items II.E.1 and II.K.2.1~~
~~(b) Items II.E.1.2 and II.K.2.8~~[53]

The influence of ~~reactor coolant pump~~RCP trip during ECCS initiation is reviewed to ~~assure conformance to the~~compliance with the requirements of TMI Action Item II.K.3.5, and the resolution thereto contained in Generic Letters 83-10A through 83-10F, 85-12, 86-05, and 86-06.~~(References 6 and 7)~~[54] Should tripping of the ~~reactor coolant pumps~~RCPs require manual action, delays in operator action must be assessed.

The reliability and integrity of the ~~reactor coolant pump~~RCP seals during loss of alternating-current power and loss of coolant to the seals (~~i.e.~~e.g.,[55] resulting from containment isolation) are reviewed to ~~assure~~ensure ~~conformance to the TMI Action Items II.K.2.16, and II.K.3.25, and II.K.3.40 (References 6 and 7)~~compliance with the requirements of 10 CFR 50.34(f)(1)(iii).[56]

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.[57]

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and ~~his~~ that the[58] 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 consequences of postulated feedwater line breaks meet the requirements set forth in the General Design Criteria 17,[59] 27, 28, 31, and 35 regarding ability to insert control rods ~~insertability~~ and ability to cool the core ~~coolability~~,[60] 10 CFR Part 100 guidelines regarding radiological dose at the site boundary, and applicable TMI Action Plan Items. This conclusion is based upon the following:
>
> (a) The applicant has met the requirements of GDC 27 and 28 by demonstrating that the resultant fuel damage was minimal, ability to insert the control rod ~~insertability~~[61] would be maintained and that no loss of core cooling capability resulted. The minimum departure from nucleate boiling ratio (MDNBR) experienced by any fuel rod was _______, resulting in ____% of the rods experiencing clad perforation.
>
> (b) 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.
>
> (c) 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).
>
> (d) The analyses and effects of feedwater line break accidents inside and outside containment, during various modes of operation and with and without offsite power, have been reviewed and evaluated using a mathematical model that has been previously reviewed and found acceptable by the staff.
>
> (e) The parameters used as input to this model were reviewed and found to be suitably conservative.

(f) 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 X/Q value of _______ sec/m^3 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 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.

(g) The applicant met the requirements of TMI Action Plan Items ~~II.E.1, II.K.2.1, and II.E.1.2, and II.K.2.8~~10 CFR 50.34(f)(1)(ii) and 10 CFR 50.34(f)(2)(xii)[62] with respect to demonstrating the adequacy of the auxiliary feedwater design to remove decay heat following feedwater piping failures.

(h) The applicant met the requirements of ~~TMI Action Plan Items II.K.2.16, and II.K.3.25 and II.K.3.40~~ 10 CFR 50.34(f)(1)(III)[63] with respect to demonstrating the integrity and operation of the ~~reactor coolant pumps~~RCPs to withstand the postulated accident.

(i) The applicant met the requirements of TMI Action Plan Item II.K.3.5 with respect to the operation and tripping of the ~~reactor coolant pumps~~RCPs. The assumptions used are conservative and consistent with the generic resolution to Item II.K.3.5 contained in Generic Letters 83-10A through 83-10F, 86-005, 86-006, and 85-12.[64]

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.[65]

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans 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.[66] Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method 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.[67]

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides and NUREGs, except for the position in Subsection II assumption d and in Subsection III regarding loss of offsite power and assumed single failures. This new position will be applied to new applications (for a Construction Permit, a manufacturing license, or design certification).[68]

VI. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

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

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

4. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

   a. General Design Criterion 17, "Electric Power Systems."[69]

   ~~a.~~b. General Design Criterion 27, "Combined Reactivity Control System Capability."

   ~~b.~~c. General Design Criterion 28, "Reactivity Limits."

   ~~c.~~d. General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

   ~~b.~~e. General Design Criterion 35, "Emergency Core Cooling."

5. Branch Technical Positions ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to SRP Section 3.6.1; and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to SRP Section 3.6.2.

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

7. NUREG-073~~6~~7,[70] "Clarification of TMI Action Plan Requirements."

8. NUREG/CR-4945, "Summary of the Semi Scale Program 1985 - 1986."[71]

9. Generic Letter 83-10A - Resolution of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all Licensees with Combustion Engineering (CE) Designed Nuclear Steam Supply Systems (NSSSs), February 8, 1983.

10. Generic Letter 83-10B - Resolution of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all Licensees with Combustion Engineering (CE) Designed Nuclear Steam Supply Systems (NSSSs), February 8, 1983.

11. Generic Letter 83-10C - Resolution of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all Licensees with Westinghouse Designed Nuclear Steam Supply Systems (NSSSs), February 8, 1983.

12. Generic Letter 83-10D - Resolution of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all Licensees with Westinghouse Designed Nuclear Steam Supply Systems (NSSSs), February 8, 1983.

13. Generic Letter 83-10E - Resolution of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all Licensees with Babcock & Wilcox (B&W) Designed Nuclear Steam Supply Systems (NSSSs), February 8, 1983.

14. Generic Letter 83-10F - Resolution of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all Licensees with Babcock & Wilcox (B&W) Designed Nuclear Steam Supply Systems (NSSSs), February 8, 1983.

15. Generic Letter 86-005 - Implementation of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all Applicants and Licensees with Combustion Engineering (CE) Designed Nuclear Steam Supply Systems (NSSSs), May 25, 1985.

16. Generic Letter 86-006 - Resolution of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all Licensees with Babcock & Wilcox (B&W) Designed Nuclear Steam Supply Systems (NSSSs), May 29, 1986.

17. Generic Letter 85-12 - Implementation of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all applicants and licensees with Westinghouse (W) Designed Nuclear Steam Supply Systems (NSSSs), July 28 1985.[72]

SRP Draft Section 15.2.8

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	Changed PRB from RSB to SRXB (global change for this section).
2.	Current SRB name and abbreviation	Changed SRB to Emergency Preparedness and Radiation Protection Branch (PERB).
3.	Current SRB name and abbreviation	Changed SRB to Instrumentation and Controls Branch (HICB).
4.	SRP-UDP format item	Changed transient to anticipated operational occurrence (AOO) to accommodate Generic Issue B-3, global change.
5.	Editorial	Added acronym (RCS) (global change for this section).
6.	Editorial	Added acronym (AFWS) (global change for this section).
7.	Editorial	Changed "assure" to "insure" (global change for this section).
8.	Editorial	Added acronym (RPS) (global change for this section).
9.	Editorial	Reformatted paragraph for clarity.
10.	Editorial	Reformatted paragraph for clarity.
11.	Current SRB abbreviation	Changed SRB to HICB (global change for this section).
12.	Editorial	HICB also reviews the RPS, ESS, and operator action to secure and maintain the reactor in a safe conditions.
13.	Editorial	Added acronym ESS two places.
14.	SRP-UDP format item	CPB was absorbed by SRXB.
15.	Editorial	Moved also from previous sentence for clarity.
16.	Editorial	Reformatted paragraph for clarity.
17.	SRP-UDP format item	Relocated from the second paragraph of the lined out paragraphs.
18.	SRP-UDP format item	Added "Review Interfaces" to AREAS OF REVIEW and formatted in numbered paragraphs to describe how SRXB reviews aspects of the feedwater system pipe breaks inside and outside containment under other SRP sections and how other branches support this review. Wording was preserved.
19.	Editorial	Defined "SRP" as "Standard Review Plan."
20.	SRP-UDP format item	Excerpted from REVIEW PROCEDURES.

SRP Draft Section 15.2.8
Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
21.	Current SRB name and abbreviation	Changed SRB to Plant Systems Branch (SPLB).
22.	SRP-UDP format item	Excerpted from REVIEW PROCEDURES.
23.	Integrated Impact No. 1048	Added discussion concerning how SPLB reviews for TMI II.E.1.1 under SRP 10.4.9 via 10 CFR 50.34 (f)(1)(ii).
24.	Editorial	Added a review interface with SRP Section 9.2.2. SRP Section 15.2.8 contains Acceptance Criteria, Review Procedures and Evaluation Findings with regard to reactor coolant pump seal integrity issues associated with TMI Action Items II.K.2.16 and II.K.3.25. Conformance with these TMI Action Items is reviewed in SRP Section 9.2.2.
25.	Current SRB abbreviation	Changed SRB to EMEB.
26.	Current SRB abbreviation	Changed SRB to SCSB and added full name of the branch.
27.	SRP-UDP format item	Stated that PERB is secondary reviewer for SRP 15.6.5.
28.	Current SRB abbreviation	Change SRB to PERB.
29.	Integrated Impact No. 1057	Added a paragraph addressing the primary review responsibility of the HICB for 10 CFR 50.34 (f)(2)(xii), TMI item II.E.1.2, under SRP Section 7.1.
30.	SRP-UDP format item	Excerpted from REVIEW PROCEDURES.
31.	Editorial	SRP Section 15.2.8 contains Acceptance Criteria (GDC 31) and Evaluation Findings regarding the integrity of the reactor coolant pressure boundary (RCPB). GDC 31 establishes the general requirements for fracture toughness of the RCPB, which are implemented by 10 CFR 50.60 and associated Appendices G and H. Conformance with the requirements of 10 CFR 50.60, Appendix G and Appendix H is reviewed in SRP Section 5.2.3 for the RCPB (other than the reactor vessel) and SRP Section 5.3.1 for the reactor vessel.
32.	Integrated Impact 1353	Added requirement for GDC 17.
33.	Editorial	Added "GDC 27" and "GDC 28" as acronyms for General Design Criteria 27 and 28.
34.	Editorial	Added "GDC 31" as acronym for General Design Criterion 31.
35.	Editorial	Added "GDC 35" as acronym for General Design Criterion 35.

SRP Draft Section 15.2.8
Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
36.	Integrated Impact Nos.1029, 1044, 1048, 1057, 1117, 1121, 1127	Deleted reference to TMI item II.K.2.1. Deleted reference to TMI item II.K.2.8. Deleted reference to TMI item II.E.1 which became part of II.E.1.1 and replaced by 10 CFR 50.34 (f)(1)(ii). Deleted reference to TMI item II.E.1.2 which is replaced by 10 CFR 50.34 (f)(2)(xii) and reviewed under SRP Sections 7.1, 7.3, and 7.5. Revised TMI item II.K.3.5 to include staff issued resolutions described in GLs 83-10A through 83-10F, 85-12, 86-05, and 86-06 Replaced to TMI items II.K.3.25 and II.K.16 with cites to 10 CFR 50.34 (f)(1)(iii). Deleted reference to TMI item II.K.3.40 which was superseded by item II.K.2.16 (see Integrated Impact No. 1121/1080 above).
37.	SRP-UDP format item	Deleted unnecessary reference citation and added title and section of an ASME Code.
38.	SRP-UDP format item	Deleted unnecessary reference citation.
39.	Integrated Impact 1353	Added discussion of LOOP in combination with a single failure for new plant applicants.
40.	SRP-UDP format item, develop technical rationale	Added "Technical Rationale" to ACCEPTANCE CRITERIA and formatted in numbered paragraphs describing the bases for referencing the General Design Criteria and other regulations.
41.	SRP-UDP format item, develop technical rationale	Added lead-in sentence for "Technical Rationale."
42.	SRP-UDP format item, develop technical rationale	Added technical rationale for 10 CFR Part 100.
43.	SRP-UDP format item, develop technical rationale	Added technical rationale for GDC 17.
44.	SRP-UDP format item, develop technical rationale	Added technical rationale for GDC 27 and GDC 28.
45.	SRP-UDP format item, develop technical rationale	Added technical rationale for GDC 31.
46.	SRP-UDP format item, develop technical rationale	Added technical rationale for GDC 35.
47.	SRP-UDP format item	Added reference to combined license (COL) in two places.
48.	SRP-UDP format item	CPB became part of SRXB.
49.	Integrated Impact No. 502	Added justification for conservatism requirement clause.
50.	Current SRB abbreviation	Changed SRB to SPLB.
51.	Integrated Impact 1353	Added clarification for LOOP.

Item	Source	Description
52.	SRP-UDP format item	Deleted reference citation and added title of reference title.
53.	Integrated Impact No. 1048 Integrated Impact No. 1029 Integrated Impact No. 1057 Integrated Impact No. 1044	Deleted reference to TMI III.E.1 because it was absorbed under III.E.1.1 and replaced by 10 CFR 50.24 (f)(1)(ii) the requirements of which are reviewed under SRP Section 10.4.9 (see "Review Interface"). Replaced reference to TMI II.K.2.1 with 10 CFR 50.34 (f)(2)(xii). Deleted reference to TMI II.E.1.2. Deleted reference to TMI II.K.2.8.
54.	Integrated Impact No. 1117	Revised sentence to include compliance with the requirements Task Action Plan II.K.3.5 and the staff resolution of the issue contained in GLs 83-10A through 83-10F, 85-12, 86-05, and 86-06 for guidance.
55.	Editorial	Replaced "i.e." with "e.g."
56.	Integrated Impact No. 1121/1080 Integrated Impact No. 1127	Replaced references to TMI items II.K.2.16 and II.K.3.25 with 10 CFR 50.34 (f)(1)(iii). Deleted reference to II.K.3.40.
57.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
58.	Editorial	Modified to eliminate a gender-specific reference.
59.	Integrated Impact No. 1353	Added reference to GDC 17.
60.	Editorial	Changed to eliminate use of "insertability" and "coolability," which are not proper words.
61.	Editorial	Changed to eliminate "insertability," which is not a proper word.
62.	Integrated Impact Nos.1048 1029 1057 1044	Replaced reference to TMI Action Item II.E.1 with reference to 10 CFR 50.34 (f)(1)(ii); deleted TMI item II.K.2.1; replaced reference to item II.E.1.2 with reference to 10 CFR 50.34 (f)(2)(xii); and deleted TMI item II.K.2.8.
63.	Integrated Impact Nos.1121/ 1080 1027	Replaced references to TMI Action Item II.K.2.16 and II.K.3.25 with reference to 10 CFR 50.34 (f)(1)(iii); and replaced reference to TMI item II.K.3.40 with reference to 10 CFR 50.34 (f)(1)(iii).
64.	Integrated Impact No. 1117	Added Generic Letters 83-10A through 83-10F, 86-005, 86-006, and 85-12.
65.	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.

Item	Source	Description
66.	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.
67.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
68.	Integrated Impact 1353	Discussed implementation of the new staff position regarding loss of offsite power and single failures.
69.	Integrated Impact 1353	Added reference for GDC 17 and renumbered.
70.	Editorial	Corrected an incorrect reference to NUREG-0737.
71.	SRP-UDP format item	Added Reference 8, which was cited in the text.
72.	Integrated Impact No. 1117	Added References 9 through 17.

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SRP Draft Section 15.2.8
Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
502	Reference NUREG/CR-4945 in REVIEW PROCEDURES.	Subsections III and VI
1029	Delete current citation to TMI item II.K.2.1.	Subsections II, III, and IV
1044	Delete current citation to TMI item II.K.2.8.	Subsections II, III, and IV
1048	Delete cite to TMI item II.E.1, reflect it in the review of SRP Section 10.4.9 replaced with the requirements of 10 CFR 50.34(f)(1)(ii).	Subsections I, II, III, and IV
1057	Delete cite to TMI item II.E.1.2, reflect it in the HICB review of Chapter 7 replaced with the requirements of 10 CFR 50.34(f)(2)(xii)	Subsections I, II, III, and IV
1117	Revise Review Procedures to incorporate guidance for the resolution of TMI Action Item II.K.3.5.	Subsection II, III, IV, and VI
1121	Replace cites to TMI items II.K.3.16 and II.K.3.25 with 10 CFR 50.34(f)(1)(ii).	Subsection II, III, and IV
1127	Deleted TMI item II.K.3.40.	Subsection II, III, and IV
1353	Added requirements of GDC 17 to appropriate sections.	Subsection II, III, IV, V, and VI