NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.2.7 LOSS OF NORMAL FEEDWATER FLOW

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a loss of offsite power (LOOP).[2] The sequence of events for the loss of feedwater transient differs between a boiling water reactor (BWR) and a pressurized water reactor (PWR). A PWR has a backup (auxiliary or emergency) feedwater system while a BWR relies on the emergency core cooling system (ECCS) and reactor core isolation cooling (RCIC) system for backup core cooling. In either case, loss of feedwater flow results in an increase in reactor coolant temperature and pressure which eventually requires a reactor trip to prevent fuel damage.

For both PWRs and BWRs, fission product decay heat must be transferred from the reactor coolant system following a loss of normal feedwater flow. This can be accomplished by actuation of one or several of the following systems: steam relief system, steam bypass to the condenser, reactor core isolation cooling system (BWR), emergency core cooling system (BWR) and auxiliary or emergency feedwater system (PWR).

The review of the loss of feedwater transient includes:

1. the sequence of events,

2. the analytical model,

3. the values of parameters used in the analytical model, and

4.[3] the predicted consequences of the transient.

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both ~~RSB~~SRXB and the Instrumentation and Control Systems Branch ~~(ICSB)~~HICB.[4] The ~~RSB~~SRXB reviewer concentrates on:

1. the need for the reactor protection system,

2. the engineered safety systems, and

3.[5] operator action to secure and maintain the reactor in a safe condition.

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 by ~~RSB~~SRXB.~~or the CPB as appropriate.~~[6]

The predicted results of the transient are reviewed to ~~assure~~ensure[7] that the consequences meet the acceptance criteria given in subsection II, below. The predicted results of the transient are reviewed to ascertain that the values of pertinent system parameters are within expected ranges for the type and class of reactor under review.

~~The RSB will coordinate other branch evaluations that interface with the overall review of the transient analysis as follows: The ICSB reviews the instrumentation and controls 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 ICSB evaluates the design of the auxiliary feedwater system to determine that the requirements and guidance of II.E.1.2 of NUREG-0737 are met. The RSB reviewer consults with the ICSB reviewer to assure that the appropriate delay time for auxiliary feedwater initiation is assumed in the analysis. The reliability of the auxiliary feedwater system is reviewed by the Auxiliary Systems Branch (ASB) in accordance with SRP Section 10.4 and in accordance with the requirements and guidance of II.E.1.1 of NUREG-0737 and II.K.2(1) (item 1 of Table C.2) in NUREG-0660. The RSB reviewer consults with the ASB reviewer to assure that the operational assumptions for the auxiliary feedwater system in the analysis is appropriate as part of its primary review responsibility for SRP Sections 7.2 through 7.5. The Core Performance Branch (CPB), upon request from RSB, reviews the values of the parameters used in the analytical models which relate to the reactor core for conformance to plant design and specified operating conditions; determines the acceptance criteria for fuel cladding damage~~

~~limits; and reviews the core physics, fuel design, and core thermal-hydraulics data used in the SAR analysis as part of its primary review responsibility for SRP Sections 4.2 through 4.4. The Accident Evaluation Branch (AEB), using fuel damage results provided by RSB, evaluates the radiological consequences associated with fuel failure. The review of the technical specifications is coordinated and performed by the Licensing Guidance Branch (LGB) as part of its primary review responsibility for SRP Section 16.0.~~[8]

Review Interfaces[9]

1. SRXB also performs the following reviews under the Standard Review Plan (SRP)[10] sections indicated:

   a. SRP Sections 4.2 through 4.4 for:~~The Core Performance Branch (CPB), upon request from RSB, reviews~~[11]

      i. the review of the values of the parameters used in the analytical models which relate to the reactor core for conformance to plant design and specified operating conditions,

      ii. determining the acceptance criteria for fuel cladding damage limits, and

      iii. reviewing the core physics, fuel design, and core thermal-hydraulics data used in the SAR analysis as part of ~~its~~SRXB's primary review responsibility.[12]

2. The ~~RSB~~SRXB will coordinate other branch evaluations that interface with the overall review of the transient analysis as follows:

   a. The ~~ICSB~~HICB reviews the instrumentation and controls 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 and compliance with Regulatory Guide 1.105.~~is determined by ICSB~~[13]

      The ~~ICSB~~HICB evaluates the design of the auxiliary feedwater system to determine that:

      1. the requirements and guidance of ~~II.E.1.2 of NUREG-0737~~10 CFR 50.34(f)(2)(xii) are met~~.~~, and[14]

      2. the appropriate delay time for auxiliary feedwater initiation is assumed in the analysis.

   b. The ~~Auxiliary Systems Branch (ASB)~~Plant Systems Branch (SPLB)[15] reviews the reliability of the auxiliary feedwater system in accordance with SRP

Section 10.4.9 ~~and in accordance with the requirements and guidance of II.E.1.1 of NUREG-0737 and II.K.2(1) (item 1 of Table C.2) in NUREG-0660.~~to ensure compliance with the requirements of 10 CFR 50.34(f)(1)(ii), 10 CFR 50.34(f)(2)(xii),[16] and the guidance of TMI Action Item II.K.2.19 of NUREG-0737.[17]

SPLB consults with SRXB to ensure that the operational assumptions for the auxiliary feedwater system in the analysis is appropriate, as part of its primary review responsibility for SRP Sections 7.2 through 7.5.

c. The ~~Accident Evaluation Branch (AEB)~~Emergency Preparedness and Radiation Protection Branch (PERB)[18] evaluates the radiological consequences associated with fuel failure using fuel damage results provided by ~~RSB~~SRXB.

d. The ~~Licensing Guidance Branch (LGB)~~Technical Specification Branch (TSB)[19] coordinates and performs reviews of the technical specifications as part of its primary review responsibility for SRP Section 16.0.

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 review branch.

II. ACCEPTANCE CRITERIA

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

A. General Design Criterion 10, as it relates to the reactor coolant system being designed with appropriate margin to ~~assure~~ensure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences.

B. General Design Criterion 15, as it relates to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to ~~assure~~ensure that the pressure boundary will not be breached during normal operations including anticipated operational occurrences.

C. General Design Criterion 17, as it relates to providing onsite and offsite electric power systems to ensure that structures, systems, and components important to safety will function during normal operation, including anticipated operational occurrences. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded during an anticipated operational occurrence.[20]

D. General Design Criterion 26, as it relates to the reliable control of reactivity changes to ~~assure~~ensure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences. This is accomplished by assuring that appropriate margin for malfunctions, such as stuck rods, are accounted for.

E. TMI Action Plan item II.K.2.19,[21] ~~II.E.1.1, II.E.1.2, and II.K.2.1~~ of NUREG-~~0718 and -~~0737and 10 CFR 50.34(f)(1)(ii) and 10 CFR 50.34(f)(2)(xii)[22] as they relate to the performance requirements of the auxiliary feedwater system for the loss of normal feedwater flow event.

1. The basic objective in the review of the loss of normal feedwater transient is to confirm that one of the following criteria are met:

   a. The consequences of the transient are less severe than the consequences of another transient that results in a decrease of heat removal by the secondary system, and has the same anticipated frequency classification.

   b. The plant responds to the loss of feedwater transient in such a way that the criteria regarding fuel damage and system pressure are met.

2. Specific criteria necessary to meet the relevant requirements of GDC 10, 15, 17,[23] and 26 for events of moderate frequency[1] are as follows:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (Ref. 1).

   b. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs and the CPR remains above the MCPR safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).

   c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

   d. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, fuel failure must be assumed for all rods for which the DNBR or CPR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding.

   e. To meet the requirements of General Design Criteria 10 and 15, the positions of Regulatory Guide 1.105, "Instrument Spans and Setpoints, are used with regard to

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[1]The term "moderate frequency" is used in this SRP section in the same sense as in the definitions of design and plant process conditions in References 9 and 10.

their impact on the plant response to the type of transient addressed in this SRP section.

f. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of Regulatory Guide 1.53 and GDC 17.[24]

3. The applicant's analysis of the loss of normal feedwater transient should be performed using an acceptable analytical model. Models which have been approved by the NRC are identified in References 2 through 8. References 20 through 24 are acceptable computer codes for non-LOCA transient analyses for system 80+ applications.[25] References 25 and 26 are acceptable transient analyses computer codes for the design of the ABWR.[26] If the applicant proposes to use analytical methods which have not been approved, these methods are evaluated by the staff for acceptability. For new generic methods the reviewer requests an evaluation by the appropriate branch.

The value of parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model.

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. Conservative scram characteristics are assumed, i.e., for a PWR – maximum time delay with the most reactive rod held out of the core and for a BWR – a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate.

c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, power profile and radial power distribution.

d. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with Regulatory Guide 1.105. ~~Compliance with Regulatory Guide 1.105 is determined by ICSB.~~[27]

Technical Rationale[28]

The technical rationale for application of the above acceptance criteria to reviewing analyses of transients initiated by steam system piping failures is discussed in the following paragraphs:[29]

1. Compliance with GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to ensure that specified

acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

GDC 10 is applicable to SRP Section 15.2.7 because this section evaluates the loss of normal feedwater flow transient. A part of the evaluation relates to the reactor coolant system being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations including AOOs. Regulatory Guide 1.105 provides guidance for ensuring that instrument setpoints are initially within and remain within the technical specification limits.

Meeting the requirements of GDC 10 provides assurance that specified acceptable fuel design limits are not exceeded for the initiating events evaluated in this SRP section involving a decrease in heat removal by the secondary system.[30]

2. Compliance with GDC 15 requires that the reactor coolant system and associated auxiliary, control and protection systems shall be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

   GDC 15 is applicable to SRP Section 15.2.7 because this section evaluates the consequences of the events of a loss of normal feedwater flow transient that result in a decrease in heat removal by the secondary system with the potential for causing the reactor coolant system pressure to change in response to the increase in reactor coolant temperature.

   Meeting the requirements of GDC 15 provides assurance that the design conditions of the reactor coolant pressure boundary are not exceeded for the initiating events evaluated in this SRP section involving a decrease in heat removal by the secondary system.[31]

3. Compliance with GDC 17 requires (in part) that an onsite and an offsite electric power system shall 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 ensure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of AOOs. GDC 17 is applicable to SRP Section 15.2.7 because the loss of normal feedwater flow transient is an AOO.

   Meeting the requirements of GDC 17 provides assurance that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of a loss of normal feedwater.[32]

4. Compliance with GDC 26 requires that two independent reactivity control systems be provided capable of reliably controlling reactivity changes to ensure that acceptable fuel design limits are not exceeded.

GDC 26 is applicable to SRP Section 15.2.7 because this section evaluates the consequences of the events of a loss of normal feedwater flow that result in a decrease in heat removal by the secondary system with the potential for causing changes in reactivity within the core that could cause the thermal design criteria for the fuel cladding to be exceeded. SRP 15.2.7 ensures that the thermal margin be sufficient to accommodate these conditions and ensures that the appropriate margins for malfunctions of reactivity controls such as stuck rods are accounted for.

Meeting the requirements of GDC 26 provides assurance that specified acceptable fuel design limits are not exceeded by ensuring that there is appropriate margin for malfunctions of the reactivity control system.[33]

III. REVIEW PROCEDURES

The procedures below are used during the reviews of both construction permit (CP) and operating license (OL) 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 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 description of the loss of normal feedwater flow transient presented by the applicant in the SAR is reviewed by ~~RSB~~SRXB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

6. That appropriate margin for malfunctions, such as stuck rods (per II.3.b above) are accounted for.

7. The operation of auxiliary systems that is required.

If the SAR states that the loss of feedwater transient is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. If a quantitative analysis of the loss of feedwater transient is presented in the SAR, the ~~RSB~~SRXB reviewer, with the aid of ~~ICSB~~HICB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the loss of feedwater transient to an acceptable level. The ~~RSB~~SRXB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The ~~ICSB~~HICB review of

Chapter 7 of the SAR confirms that the instrumentation and control systems design is consistent with the requirements for safety systems actions for these events.

To the extent deemed necessary, the ~~RSB~~SRXB reviewer evaluates the effect of single active failures of systems and components which may alter the course of the transient. For new applications, loss of offsite power (LOOP) should not be considered a single failure; loss of feedwater should be analyzed with and without a LOOP 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.)[34] This part of the review uses the procedures described in the SRP sections for Chapters 4, 5, 6, 7, 8, and 9 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam line are reviewed by ~~RSB~~SRXB to determine if these models have been previously reviewed and found acceptable by the staff. If not, a generic review of the model proposed by the applicant is initiated.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by ~~RSB~~SRXB. 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 the values of the reactivity parameters used in the applicant's analysis.~~[35]

The results of the analysis are reviewed, including the effects of the LOOP,[36] and compared ~~to~~ with[37] the acceptance criteria presented in subsection II of this SRP section regarding maximum pressure in the reactor coolant and main steam systems. The parameters reviewed are:[38]

1. ~~The~~ variations with time during the transient of the neutron power,

2. heat fluxes (average and maximum),

3. reactor coolant system pressure,

4. minimum DNBR (PWR) or CPR (BWR),

5. core and recirculation loop coolant flow rates (BWR),

6. coolant conditions (inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam fractions),

7. steamline pressure,

8. containment pressure,

9. pressure relief valve flow rate, and

10. flow rate from the reactor coolant system to the containment system (if applicable).

The more important of these parameters for the loss of normal feedwater transient are compared ~~to~~ with those predicted for other similar plants to see that they are within the range expected.

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.[39]

IV. EVALUATION FINDINGS

The evaluation findings under this SRP section are incorporated in a statement covering all transients of moderate frequency involving an unplanned decrease in heat removal by the secondary system. SRP Section 15.2.1-15.2.5 contains a general statement covering the findings of this SRP section.

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.[40]

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.[41] 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.[42]

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides and NUREGs.

VI. REFERENCES

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

2. NUREG-0151, Safety Evaluation Report, "GESSAR-251, Nuclear Steam Supply System Standard Design," General Electric Company, March 1977.

3. NUREG-0152, Safety Evaluation Report, "GESSAR-238, Nuclear Steam Supply System Standard Design," General Electric Company, March 1977.

4. NUREG-75/103, Safety Evaluation Report, "RESAR-41 Standard Reference System," Westinghouse Electric Corporation, December 1975.

5. NUREG-0104, Safety Evaluation Report, "RESAR-35, Standard Reference System," Westinghouse Electric Corporation, December 1976.

6. NUREG-0491, Safety Evaluation Report, "RESAR-414, Standard Reference System," Westinghouse Electric Corporation, November 1978.

7. NUREG-75/112, Safety Evaluation Report, "CESSAR System 80, Standard Reference System," Combustion Engineering Incorporated, December 1975.

8. NUREG-0433, Safety Evaluation Report, "B-SAR-205, Nuclear System Supply System," Babcock & Wilcox Company, May 1978.

9. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

10. ANS Trial Use Standard N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

11. General Design Criterion 10, "Reactor Design."

12. General Design Criterion 15, "Reactor Coolant System Design."

13. General Design Criterion 17, "Electric Power Systems."[43]

~~13.~~14. General Design Criterion 21, "Protection System Reliability and Testability."

~~14.~~15. General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

~~15.~~16. Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems."

~~16.~~17. Regulatory Guide 1.105, "Instrument Spans and Setpoints."

~~17.~~18. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses."

~~18.~~19. NUREG-0737, "Clarification of TMI Action Plan Requirements."

20. CESEC-III (CENPD-107; LD-82-001). (Calculates system parameters such as core power, flow, pressure, temperature, and valve actions during a transient.)

21. TORC (CENPD-161) and CETOP (CENPD-206-P-A). (TORC is used to simulate the three-dimensional fluid conditions within the reactor core. Results from TORC include the core radial distribution of the relative axial flow that is used to calibrate CETOP. TORC or CETOP is used for DNBR calculations using the CE-1 critical heat flux correlation.)

22. HERMITE (CENPD-188-A). (HERMITE is used to determine short-term response of the reactor core during the postulated reactor coolant pump rotor-seizure event and total loss-of-flow event.)

23. COAST (SSAR; CENPD-98). (Calculates the time-dependent reactor coolant mass flow rate in each loop during reactor coolant pump coastdown transients.)

24. STRIKEN-II (CENPD-133; CENPD-135 Supps. 2 and 4). (Calculates the cladding and fuel temperatures for an average or hot fuel rod.)[44]

25. General Electric Company, ODYNA – One Dimensional Dynamic Model (propriety computer software for use in ABWR transient analysis to simulate pressurization events).

26. General Electric Company, REDYNA (propriety computer software for use in ABWR transient analysis to simulate other than pressurization events).[45]

SRP Draft Section 15.2.7
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 RSB to SRXB (global change for this section).
2.	Editorial	Added acronym "LOOP."
3.	Editorial	Reformatted paragraph for clarity.
4.	Current PRB abbreviation	Changed (ICSB) to (HICB) (global change for this section).
5.	Editorial	Reformatted paragraph for clarity.
6.	SRP-UDP format item	CPB became part of SRXB.
7.	Editorial	Changed "assure" to "ensure" (global change for this section).
8.	SRP-UDP format item	Relocated and reformatted into the "Review Interface."
9.	SRP-UPD format item	"Review Interfaces" added to "AREAS OF REVIEW" and formatted in numbered paragraphs to describe how SRXB reviews aspects of the loss of normal feedwater flow under other SRP sections and how other branches support this review.
10.	Editorial	Defined "SRP" as "Standard Review Plan."
11.	SRP-UDP format item	The CPB became a part of SRXB.
12.	Editorial	Reorganized sentence for easier reading in new location.
13.	SRP-UDP format item	Relocated from subsection II.3.d.
14.	Integrated Impact No. 1047	Replaced TMI Action Plan II.E.1.2 with 10 CFR 50.34(f)(2)(xii).
15.	Current PRB abbreviation	Editorial change made to reflect current PRB abbreviation, (SPLB) for the review of Section 10.4.
16.	Integrated Impact No. 1054 Integrated Impact No. 1052	Replaced TMI Action Plan II.E.1.1 with 10 CFR 50.34(f)(1)(ii) Deleted TMI Action Plan II.K.2.1.
17.	Integrated Impact No. 1120	Added requirement for TMI Action Plan II.K.2.19.
18.	Current SRB name and abbreviation	Editorial change made to reflect current SRB name Emergency Preparedness and Radiation Protection Branch (PERB).
19.	Current PRB review responsibility	TSB has primary review responsibility for SRP Section 16.0.

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Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
20.	Integrated Impact #509	Added Acceptance Criteria to include General Design Criterion 17 to address the requirements for loss of off site power.
21.	Integrated Impact No. 1120	Added TMI Action Item II.K.2.19.
22.	Integrated Impact No. 1054 Integrated Impact No. 1047 Integrated Impact No. 1052	Replaced TMI Action Plan item II.E.1.1 with 10 CFR 50.34(f)(1)(ii). Replaced TMI Action Plan item II.E.1.2 with 10 CFR 50.34(f)(2)(xii). Deleted TMI Action Plan item II.K.2.1.
23.	Integrated Impact No. 509	Added GDC 17 requirement.
24.	Integrated Impact No. 509	Added GDC 17 requirement.
25.	Integrated Impact No. 1443	Added citation to References 20 through 24 in Acceptance Criteria for analytical models for the System 80+ plants.
26.	Integrated Impact No. 1444	Added citation to References 25 and 26 in Acceptance Criteria for analytical models for the ABWR plants.
27.	SRP-UDP format item	Relocated to "Review Interface" subsection.
28.	SRP-UPD format item, develop technical rationale	"Technical Rationale" added to ACCEPTANCE CRITERIA and formatted in numbered paragraphs describing the bases for referencing the General Design Criteria and other regulations.
29.	SRP-UDP format item, develop technical rationale	Added lead-in sentence for "Technical Rationale."
30.	SRP-UDP format item	Added technical rationale for GDC 10.
31.	SRP-UDP format item, develop technical rationale	Added technical rationale for GDC 15.
32.	SRP-UDP format item, develop technical rationale	Added technical rationale for GDC 17.
33.	SRP-UDP format item, develop technical rationale	Added technical rationale for GDC 26.
34.	**Integrated Impact 509**	Added new staff position relative to LOOP and single failures.
35.	SRP-UDP format item	CPB was absorbed by SRXB.
36.	Integrated Impact No. 509	Added requirement for LOOP.
37.	Editorial	Corrected use of "compared to" to "compared with" (global change for this section).
38.	Editorial	Reformatted paragraph into numbered items for ease of reading.

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Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
39.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
40.	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.
41.	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.
42.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
43.	Integrated Impact No. 509	Added GDC 17 as a new reference 13 and renumbered remaining references.
44.	Integrated Impact No. 1443	Added new references 20 through 24 to References.
45.	Integrated Impact No. 1444	Added new reference 25 and 26 to References.

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SRP Draft Section 15.2.7

Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
509	Include GDC 17; Acceptance Criteria and Review Procedures to incorporate staff guidance for the assumption of the LOOP.	ACCEPTANCE CRITERIA, C and 2.f Review Procedures and Reference 13.
510	ANSI NIB .2 and ANS N212	No action taken.
1047	Delete the citation of II.E.1.2.	"Review Interfaces," 1; ACCEPTANCE CRITERIA, E
1052	Delete citations related to TMI Action Plan item II.K.2.1.	"Review Interfaces," 2.b; ACCEPTANCE CRITERIA, E
1054	Delete the citations to TMI Action Plan item II.E.1.1.	"Review Interfaces," 2.b; ACCEPTANCE CRITERIA, E.
1120	Cite II.K.2.19 of NUREG-0737 in SRP 15.2.7.	"Review Interfaces," 2.b; ACCEPTANCE CRITERIA, E
1444	Revised Acceptance Criteria to add currently approved transient accident analysis models for CE 80+ plants.	ACCEPTANCE CRITERIA, VI; References 20-24
1443	Revised Acceptance Criteria to add currently approved transient accident analysis models for ABWR plants.	ACCEPTANCE CRITERIA, VI; References 25 and 26