NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.2.6 LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

The loss of nonemergency ac power is assumed to result in the loss of all power to the station auxiliaries. This situation could result either from a complete loss of the external grid (offsite) or a loss of the onsite ac distribution system. It is different from the loss of load condition considered in Standard Review Plan (SRP) Section 15.2.2 because, in the latter case, ac power remains available to operate the station auxiliaries. The major difference is that in the loss of ac power transient all the reactor coolant circulation pumps are simultaneously tripped by the initiating event. This causes a flow coastdown as well as a decrease in heat removal by the secondary system.

Within a few seconds the turbine trips and the reactor coolant system is isolated, causing the pressure and temperature of the coolant to increase. A reactor trip is initiated. The diesel generators are automatically started and provide electric power to the vital loads. The sensible and decay heat loads are handled by actuation of the steam relief system, steam bypass to the condenser, reactor core isolation cooling system in a boiling water reactor (BWR), emergency core cooling system (BWR), and auxiliary feedwater system in a pressurized water reactor (PWR).

The review of the loss of ac power transient includes the sequence of events, the analytical model, the values of parameters used in the analytical model, and 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[2] and the Instrumentation and Controls ~~Systems~~ Branch (~~ICSB~~HICB[3]). The ~~RSB~~SRXB[4] reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition.

The analytical methods are reviewed by ~~RSB~~SRXB[5] 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[6] ~~or the Core Performance Branch (CPB) as appropriate~~.[7]

The predicted results of the transient analysis are reviewed to ~~assure~~ ensure[8] that the consequences meet the acceptance criteria given in subsection II, below. The results of the analysis are reviewed to ascertain that the values of pertinent system parameters are within expected ranges for the type and class of reactor under review.

Review Interfaces[9]

The ~~RSB~~SRXB[10] will coordinate other branch evaluations that interface with the overall review of the transient analysis as follows:

1.[11] The ~~ICSB~~HICB[12] 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.

2. The ~~ICSB~~HICB[13] evaluates the design of the auxiliary feedwater system to determine that the requirements and guidance of II.E.1.2 of NUREG-0737 and 10 CFR 50.34(f)(2)(xii)[14] are met. The ~~RSB~~SRXB[15] reviewer consults with the ~~ICSB~~HICB[16] reviewer to ~~assure~~ ensure that the appropriate delay time for auxiliary feedwater initiation is assumed in the analysis.

3. The reliability of the auxiliary feedwater system is reviewed by the ~~ASB~~SPLB[17] in accordance with SRP Section 10.4.9[18] ~~and in accordance~~, including conformance[19] with the requirements and guidance of II.E.1.1 of NUREG-0737 and ~~II.K.2.(1) (item 1 of Table C.2) of NUREG-0660~~10 CFR 50.34(f)(1)(ii).[20] The ~~RSB~~SRXB[21] reviewer consults with the ~~ASB~~SPLB[22] reviewer to ~~assure~~ ensure that the operational assumptions for the auxiliary feedwater system in the analysis is appropriate.

4. ~~As part of its primary review responsibility for SRP Sections 7.2 through 7.5, the Core Performance Branch (CPB), upon request from RSB,~~ SRXB[23] 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.

5. The ~~Accident Evaluation Branch (AEB)~~Emergency Preparedness and Radiation Protection Branch (PERB),[24] using fuel damage results provided by ~~RSB~~SRXB,[25] evaluates the radiological consequences associated with fuel failure.

6. The review of the technical specifications is coordinated and performed by the ~~Licensing Guidance Branch (LGB)~~Technical Specifications Branch (TSB)[26] as part of its primary review responsibility for SRP Section 16.0.

For those areas of review identified above 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 review branch.

II. ACCEPTANCE CRITERIA

The ~~RSB~~SRXB[27] acceptance criteria are based on meeting the relevant requirements of the following regulations:

A. General Design Criterion 10 (GDC 10)[28] 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 operation including anticipated operational occurrences.

B. General Design Criterion 15 (GDC 15)[29] 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 operation including anticipated operational occurrences.

C. General Design Criterion 26 (GDC 26)[30] 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~~ ensuring that appropriate margin for malfunctions, such as stuck rods, are accounted for.

~~D. TMI Action Plan items II.E.1.1, II.E.1.2, and II.K.2(1) of NUREGs-0718 and -0737 as they relate to the performance requirements of the auxiliary feedwater system for the loss of nonemergency ac power event.~~[31]

Specific criteria necessary to meet the relevant requirements of GDC 10, 15, and ~~16~~26[32] for events of moderate frequency* are as follows:

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*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.

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

2. Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR)[33] remains above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR)[34] remains above the minimum critical power ratio (MCPR)[35] safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).

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

4. 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 failures 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.

5. To meet the requirements of General Design Criteria 10 and 15, the positions of Regulatory Guide 1.105, "Instrument~~ Spans and~~ Setpoints for Safety-Related Systems,"[36] are used with regard to their impact on the plant response to the type of transient addressed in this SRP section.

6. 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~~ (Ref. 14)~~.[37]

The applicant's analysis of the loss of ac power transient should be based on an acceptable model. Models which have been approved by the NRC are identified in References 2 through 8. References 19 through 23 are acceptable computer codes for non-LOCA transient analysis for CE80+ applications.[38] References 24 and 25 are acceptable transient analysis computer codes for Advanced Boiling Water Reactor (ABWR) applications.[39] If the applicant proposes analytical methods which have not been approved, these 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~~HICB.[40]

Technical Rationale

The technical rationale for application of these acceptance criteria to reviewing the loss of nonemergency ac power to the station auxiliaries is discussed in the following paragraphs:[41]

(a) 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 this section because the reviewer evaluates the consequences of the loss of nonemergency ac power to the station auxiliaries. This is an anticipated operational occurrence that creates a potential for specified acceptable fuel design limits to be exceeded. Within seconds after the loss of power, the turbine and the reactor both trip, and the pressure and temperature of the reactor coolant increase. Regulatory Guide 1.53 provides guidance with respect to the application of the single failure criterion to the design and analysis of nuclear power plant protection systems. Regulatory Guide 1.105 describes a method acceptable to the staff 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 will not be exceeded and that fuel cladding integrity is maintained for the loss of nonemergency ac power to the station auxiliaries.[42]

(b) Compliance with GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection systems 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 this section because the reviewer evaluates the consequences of the loss of nonemergency ac power to the station auxiliaries. This is an anticipated operational occurrence, and the reactor coolant pressure needs to be analyzed to ensure that the pressure acceptance criterion is satisfied.

Meeting the requirements of GDC 15 provides assurance that the design conditions of the reactor coolant pressure boundary are not exceeded for the loss of nonemergency ac power to the station auxiliaries.[43]

(c) Compliance with GDC 26 requires that one of the reactivity control systems consist of control rods capable of reliably controlling reactivity changes to ensure that – under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods – specified acceptable fuel design limits are not exceeded.

GDC 26 is applicable because the transient analyzed in this section will involve the movement of control rods in response to the transient, and because rod misalignment, including stuck rods, can produce more severe thermal-hydraulic conditions than would otherwise exist. GDC 26 requires that the thermal margin be sufficient to accommodate these conditions. SRP Section 15.2.6 examines these margins, where applicable, to ensure that the thermal criteria remain satisfied.

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, including stuck rods.[44]

III. REVIEW PROCEDURES

The procedures below are used during the review of ~~both~~ construction permit (CP), combined license (COL),[45] 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 COL or[46] 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 ac power transient presented by the applicant in the SAR is reviewed by ~~RSB~~SRXB[47] 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. The operation of standby diesel generators that is required.

7. That appropriate margin for malfunctions, such as stuck rods (per ~~II.3.b~~II.b[48] above) are accounted for.

If the SAR states that the loss of ac power 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 ac power transient is presented in the SAR, the ~~RSB~~SRXB[49] reviewer, with the aid of the ~~ICSB~~HICB[50] reviewer, reviews the timing of the initiation of those protection, engineered safety, standby diesel generator, and other systems needed to limit the consequences of the transient to an acceptable level. The ~~RSB~~SRXB[51] reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The ~~ICSB~~HICB[52] 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[53] reviewer evaluates the effects of single active failures of systems and components which may affect the course of the transient. This aspect of the review uses the procedures described in 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 lines are reviewed by ~~RSB~~SRXB[54] 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.[55] 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~~it[56] has selected the core burnup that yields the minimum margins is evaluated. ~~CPB is consulted regarding~~The SRXB[57] reviews the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared ~~to~~ with[58] the acceptance criteria presented in subsection II of this SRP section regarding the maximum pressure in the reactor coolant and main steam systems. ~~The variations with time during the transient of neutron power, heat fluxes (average and maximum), reactor coolant system pressure, minimum DNBR (PWR) or CPR (BWR), core and recirculation loop coolant flow rates (BWR), coolant conditions (inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam fractions), steam line pressure, containment pressure, pressure relief valve flow rate, and flow rate from the reactor coolant system to the containment system (if applicable) are reviewed.~~ Time-related variations of the following parameters are reviewed:

- reactor power;
- heat fluxes (average and maximum);
- reactor coolant system pressure;
- minimum DNBR (PWR) or CPR (BWR);
- core and recirculation loop coolant flow rates (BWR);

- coolant conditions (inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam fractions);
- steam line pressure;
- containment pressure;
- pressure relief valve flow rate; and
- flow rate from the reactor coolant system to the containment system (if applicable).[59]

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

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.[60]

IV. EVALUATION FINDINGS

The evaluation findings under this SRP section are incorporated in a statement covering all transients of moderate frequency involving a decrease in heat removal by the secondary system. See the findings statement in SRP Section ~~15.2.1-5~~15.2.1–15.2.5[61] for a typical statement.

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.[62]

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.[63] 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.[64]

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 Steam 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 21, "Protection System Reliability and Testability."

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

15. Regulatory Guide 1.105, "Instrument ~~Spans and~~ Setpoints for Safety-Related Systems."[65]

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

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

18. General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

19. CESEC-III (CENPD-107; LD-82-001). (Calculates system parameters such as core power, flow, pressure, temperature, and valve actions during a transient.)[66]

20. 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 channel axial flow that is used to calibrate CETOP. TORC or CETOP is used for DNBR calculations using the CE-1 critical heat flux correlation.)[67]

21. 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.)[68]

22. COAST (SSAR; CENPD-98). (Calculates the time-dependent reactor coolant mass flow rate in each loop during reactor coolant pump coastdown transients.)[69]

23. STRIKIN-II (CENPD-133; CENPD-135 Supps. 2 and 4). (Calculates the cladding and fuel temperatures for an average or hot fuel rod.)[70]

24. General Electric Company, ODYNA - One Dimensional Dynamic Model (proprietary computer software for use in ABWR transient analysis to simulate pressurization events).[71]

25. General Electric Company, REDYA - (proprietary computer software for use in ABWR transient analysis to simulate other than pressurization events).[72]

SRP Draft Section 15.2.6

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 primary review branch designation	Changed PRB to SRXB.
2.	Current primary review branch designation	Changed PRB to SRXB.
3.	Current review branch name and designation	Changed the review interface branch name and designation to Instrumentation and Controls Branch (HICB).
4.	Current primary review branch designation	Changed PRB to SRXB.
5.	Current primary review branch designation	Changed PRB to SRXB.
6.	Current primary review branch designation	Changed PRB to SRXB.
7.	SRP-UDP update item	Deleted the reference to the Core Performance Branch (CPB) since this branch no longer exists.
8.	Editorial	Changed "assure" to "ensure" (global change for this section).
9.	SRP-UDP format item	Added "Review Interfaces" under AREAS OF REVIEW.
10.	Current primary review branch designation	Changed PRB to SRXB.
11.	SRP-UDP format item	Changed one long paragraph describing review interfaces into numbered paragraphs, one for each review interface activity.
12.	Current review branch designation	Changed review interface branch to HICB.
13.	Current review branch designation	Changed review interface branch to HICB.
14.	Integrated Impact No. 1039	Added a citation to 10 CFR 50.34(f)(2)(xii) next to the citation to TMI Action Plan item II.E.1.2 for completeness.
15.	Current primary review branch designation	Changed PRB to SRXB.
16.	Current review branch designation	Changed review interface branch to HICB.
17.	Current review branch designation	Changed review interface branch to SPLB.
18.	Editorial	Added the specific SRP section number for the review of the auxiliary feedwater system, SRP Section 10.4.9.

SRP Draft Section 15.2.6
Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
19.	Editorial	Changed the wording to "including conformance" as more appropriate since SRP Section 10.4.9 addresses the detailed criteria and guidance associated with TMI Action Plan Item II.E 1.1 and 10 CFR 50.34(f)(1)(ii).
20.	Integrated Impact Nos. 1055 and 1092	Added a reference to 10 CFR 50.34(f)(1)(ii) for Integrated Impact No. 1055. Deleted the reference to Item II.K.2.(1) of NUREG-0660 for Integrated Impact No. 1092.
21.	Current primary review branch designation	Changed PRB to SRXB.
22.	Current review branch designation	Changed review interface branch to SPLB.
23.	Current review branch designation	The past review interface branch, CPB, is now the same as the primary review branch, SRXB. The references to SRP Sections 7.2 through 7.5 appear to be an error, since HICB has the primary review responsibility for these sections and they do not involve the subjects described.
24.	Current review branch designation	Changed review interface branch to Emergency Preparedness and Radiation Protection Branch (PERB).
25.	Current primary review branch designation	Changed PRB to SRXB.
26.	Current review branch designation	Changed review branch interface to Technical Specifications Branch (TSB).
27.	Current primary review branch designation	Changed PRB to SRXB.
28.	Editorial	Provided "GDC 10" as initialism for "General Design Criterion 10.
29.	Editorial	Provided "GDC 15" as initialism for "General Design Criterion 15.
30.	Editorial	Provided "GDC 26" as initialism for "General Design Criterion 26.
31.	Integrated Impact Nos. 1039, 1055, and 1092	Deleted a reference to TMI Action Plan Item II.E.1.2 in paragraph D of subsection II, Acceptance Criteria (Integrated Impact No. 1039). Deleted reference to TMI Action Plan Item II.E.1.1 (Integrated Impact No. 1055). Also deleted reference to II.K.2.(1) (Integrated Impact No. 1092), thus causing the entire paragraph to be deleted.
32.	Editorial	Corrected an apparent typographical error: deleted "16" and substituted "26" therefor.

Item	Source	Description
33.	Editorial	Spelled out "departure from nucleate boiling ratio" to introduce the acronym "DNBR" for first use in this SRP section.
34.	Editorial	Spelled out "critical power ratio" to introduce the acronym "CPR" for first use in this SRP section.
35.	Editorial	Spelled out "minimum critical power ratio" to introduce the acronym "MCPR" for first use in this SRP section.
36.	Editorial	Corrected the title of Regulatory Guide 1.105 to "Instrument Setpoints for Safety-Related Systems."
37.	SRP-UDP format item	Deleted unnecessary callout for "(Ref. 14)."
38.	Integrated Impact No. 1370	Added a citation to References 19 through 23 for the NRC-approved analytical models for CE80+.
39.	Integrated Impact No. 1371	Added a citation to References 24 and 25 for the NRC-approved analytical models for the ABWR.
40.	Current review branch designation	Changed review interface branch to HICB.
41.	SRP-UDP format item	Added "Technical Rationale" to ACCEPTANCE CRITERIA and standard lead-in sentence.
42.	SRP-UDP format item	Added technical rationale for GDC 10.
43.	SRP-UDP format item	Added technical rationale for GDC 15.
44.	SRP-UDP format item	Added technical rationale for GDC 26.
45.	SRP-UDP format item	Added text for combined license (COL).
46.	SRP-UDP format item	Added text for combined license (COL).
47.	Current primary review branch designation	Changed PRB to SRXB.
48.	Editorial	Corrected the referenced paragraph to "II.b" (there is no paragraph "II.3.b").
49.	Current primary review branch designation	Changed PRB to SRXB.
50.	Current review branch designation	Changed review interface branch to HICB.
51.	Current primary review branch designation	Changed PRB to SRXB.
52.	Current review branch designation	Changed review interface branch to HICB.
53.	Current primary review branch designation	Changed PRB to SRXB.
54.	Current primary review branch designation	Changed PRB to SRXB.

Item	Source	Description
55.	Current primary review branch designation	Changed PRB to SRXB.
56.	SRP-UDP format item	Changed "he" to "it" to remove gender.
57.	Editorial	Deleted the reference to the "CPB" and substituted "SRXB" therefor, since the SRXB has now assumed the review responsibilities of the CPB.
58.	Editorial	Changed "compared to" to "compared with" to accommodate scientific usage (global change for this section).
59.	Editorial	Simplified a complex sentence for readability and clarity.
60.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
61.	Editorial	Changed the reference to SRP Section 15.2.1–15.2.5 to agree with the table of contents manner of designating this section of the SRP for clarity.
62.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Evaluation Findings in design certification reviews.
63.	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.
64.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
65.	Editorial	Corrected the title of Regulatory Guide 1.105 to "Instrument Setpoints for Safety-Related Systems."
66.	Integrated Impact No. 1370	Added CESEC-III code as Reference 19.
67.	Integrated Impact No. 1370	Added TORC and CETOP codes as Reference 20.
68.	Integrated Impact No. 1370	Added HERMITE code as Reference 21.
69.	Integrated Impact No. 1370	Added COAST code as Reference 22.
70.	Integrated Impact No. 1370	Added STRIKIN code as Reference 23.
71.	Integrated Impact No. 1371	Added ODYNA code as Reference 24.
72.	Integrated Impact No. 1371	Added REDYA code as Reference 25.

SRP Draft Section 15.2.6

Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
417	ANS Trial Use Standard N212-1974 and ANSI N18.2-1974 have been superseded by ANS 52.1-1983 and ANS 51.1-1983, respectively.	No changes have been made to SRP Section 15.2.6 for Integrated Impact No. 417
1039	Revise SRP Section 15.2.6 to delete the reference to TMI Action Plan Item II.E.1.2 in the Acceptance Criteria and add a reference to 10 CFR 50.34(f)(2)(xii) in the Review Interface subheading for completeness.	I, Review Interface paragraph 2; II, ACCEPTANCE CRITERIA paragraph D
1055	Revise SRP Section 15.2.6 to delete the reference to TMI Action Plan Item II.E.1.1 in the Acceptance Criteria and add a reference to 10 CFR 50.34(f)(1)(ii) in the Review Interface subheading for completeness.	I, Review Interface paragraph 3; II, ACCEPTANCE CRITERIA paragraph D
1092	Delete the citation to TMI Action Plan item II.K.2(1) of NUREG-0660 in (1) the subheading "Review Interfaces" under Areas of Review, and (2) in the Acceptance Criteria item D.	I, Review Interface paragraph 3; II, ACCEPTANCE CRITERIA paragraph D
1370	Revise the acceptance criteria and references to include the currently-approved analytical methods and computer codes applicable to ABB-CE CE80+ plants.	II, ACCEPTANCE CRITERIA; VI, REFERENCES 19-23
1371	Revise the acceptance criteria and references to include the ODYNA and REDYA computer codes as being acceptable to the NRC staff for transient analysis of the ABWR.	II, ACCEPTANCE CRITERIA; VI, REFERENCES 24 and 25