**NUREG-0800**
**(Formerly NUREG-75/087)**

**U.S. NUCLEAR REGULATORY COMMISSION**

# STANDARD REVIEW PLAN

**OFFICE OF NUCLEAR REACTOR REGULATION**

## 15.4.4 – 15.4.5 STARTUP OF AN INACTIVE LOOP OR RECIRCULATION LOOP AT AN INCORRECT TEMPERATURE, AND FLOW CONTROLLER MALFUNCTION CAUSING AN INCREASE IN BWR CORE FLOW RATE

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (SRXB)[1]

Secondary - None

## I. AREAS OF REVIEW

A number of ~~transients~~anticipated operational occurrences (AOOs)[2] that may occur with moderate frequency cause either increased core flow or introduction of cooler or de-borated water into the core. These ~~transients~~AOOs result in an increase in core reactivity due to decreased moderator temperature, moderator boron concentration, or core void fraction.[1] This Standard Review Plan (SRP)[3] section is intended to be applicable to all such ~~transients~~AOOs. Each of these ~~transients~~AOOs should be discussed in individual sections of the applicant's safety analysis report (SAR), as required by Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." ~~the Standard Format (Reference 1).~~[4]

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[1] Continuous boron dilution is considered in another section of the Standard Review Plan~~SRP~~.[3]

The specific ~~transients~~AOOs (Table 15-1, "Representative Initiating Events to be Analyzed in Sections 15.X.X of the SAR,"[5] ~~of Reference 1~~in Regulatory Guide 1.70[6]) evaluated are:

1. Boiling water reactor (BWR): startup of an idle recirculation pump.

2. BWR: flow controller malfunction causing increased recirculation flow.

3. Pressurized water reactor (PWR) with loop isolation valves: startup of a pump in an initially isolated inactive reactor coolant loop where the rate of flow increase is limited by the rate at which the isolation valves open.

4. PWR without loop isolation valves: startup of a pump in an inactive loop.

The review of the core flow increase ~~transients~~AOOs considers the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the ~~transients~~AOOs. The ~~RSB~~SRXB[7] reviewer concentrates on the need for the reactor protection system and 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 initiates a generic evaluation of the new analytical model. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

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

~~In addition, the RSB will coordinate other branches' evaluations that interface with the overall review of the system as follows: The Instrument and Control Branch (ICSB) reviews the instrumentation and controls aspects of the sequence described in the SAR to confirm that reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis as part of its primary review responsibility for SRP Sections 7.2 through 7.5. The Core Performance Branch (CPB) performs generic reviews of the thermal-hydraulic computer models used for this transient and also performs, upon request, additional analyses related to these accidents for selected reactor types as part of its primary review responsibility for SRP Sections 4.2 through 4.4. The Accident Evaluation Branch (AEB) is notified regarding the extent of the fuel failures that are predicted by the analysis. AEB then evaluates the radiological consequences of the event.~~[9]

Review Interfaces[10]

1. SRXB also performs the following reviews under the SRP sections indicated:

   SRP Sections 4.2 through 4.4. The ~~Core Performance Branch (CPB)~~Reactor Systems Branch (SRXB)[11] performs generic reviews of the thermal-hydraulic computer models used for this ~~transient~~AOO and also performs, upon request, additional analyses related to these accidents for selected reactor types as part of its primary review responsibility.

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

   a. The Instrument and Control Branch ~~(ICSB)~~(HICB)[12] reviews the instrumentation and controls aspects of the sequence described in the SAR to confirm that reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis as part of its primary review responsibility for SRP Sections 7.2 through 7.5.

   b. ~~The Accident Evaluation Branch (AEB)~~Emergency Preparedness and Radiation Protection Branch (PERB)[13] is notified regarding the extent of the fuel failures that are predicted by the analysis. ~~AEB~~PERB[14] then evaluates the radiological consequences of the event.

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 branch~~.[15]

## II. ACCEPTANCE CRITERIA

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

A. General Design Criteria ~~(GDC)~~ 10 (GDC 10) and 20 (GDC 20) as ~~it~~they relate~~s~~[16] 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~~AOOs.

B. General Design Criteria 15 (GDC 15) and 28 (GDC 28) as ~~it~~they relate~~s~~[17] 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~~AOOs.

C. General Design Criterion 26 (GDC 26)[18] 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~~AOOs. This is accomplished by

~~assuring~~ensuring that appropriate margin for malfunctions, such as stuck rods, is accounted for.

The basic objectives of the review of the ~~transients~~AOOs described above are:

1. To identify which of the ~~transients~~AOOs are the most limiting.

2. To verify that, for the most limiting ~~transient~~AOOs, the plant responds in such a way that the criteria regarding fuel damage and system pressure are met.

The specific criteria necessary to meet the relevant requirements of the regulations identified above for incidents of moderate frequency are as follows:

(a) Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (~~Ref. 2~~ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000[19]).

(b) Fuel-cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR)[20] remains above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR)[21] remains above the minimum critical power ratio (MCPR)[22] 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 mode (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) The requirements stated in Regulatory Guide 1.105, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of ~~transients~~AOOs 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 should satisfy the guidance stated in Regulatory Guide 1.53.

The applicant's analysis of the most limiting ~~transients~~AOOs should be performed using an acceptable model. If analytical methods which have not been approved are proposed by the applicant, they are evaluated by the staff for acceptability.

The values of parameters used in the analytical model are to be suitably conservative. The following values are considered acceptable:

a. Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed to be operating, plus an allowance of 2% to account for power measurement uncertainty. An analysis to determine the effects of a flow increase must be made for each allowed mode of operation (i.e., one, two, or three loops initially operating) or the effects referenced to a limiting case.

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

c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial 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 as determined by ~~ICSB~~HICB.[23]

The reviewer shall verify that the protection system (1) initiates automatically the operation of appropriate systems, including the reactivity control systems, to ~~assure~~ensure that specified acceptable fuel design limits are not exceeded for this event and (2) senses the plant conditions and initiates the operation of systems and components important to safety.

For BWR plants where flow control is part of the reactivity control system, General Design Criteria~~GDC~~ 25, 26, and 28 must be satisfied for this event; otherwise, General Design Criteria~~GDC~~ 25, 26, and 28 are not applicable. Where applicable, General Design Criteria~~GDC~~ 25, 26, and 28 are satisfied if compliance with General Design Criteria~~GDC~~[24] 10 and 15 is demonstrated.

Technical Rationale[25]

The technical rationale for application of these acceptance criteria is discussed in the following paragraphs:[26]

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 this SRP section because the reviewer evaluates the consequences of events associated with startup of an inactive loop or recirculation loop at an incorrect temperature and with a flow controller malfunction causing an increase in BWR core flow rate. This section, SRP Sections 4.2 through 4.4 and 7.2 through 7.5, and Regulatory Guides 1.53 and 1.105 provide guidance for ensuring that the reactor core, coolant, and control and protection systems are designed with appropriate margin.

Meeting the requirements of GDC 10 provides assurance that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs.[27]

2. 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 SRP section because the reviewer evaluates the consequences of events associated with startup of an inactive loop or recirculation loop at an incorrect temperature and with a flow controller malfunction causing an increase in BWR core flow rate. This section, SRP Sections 4.2 through 4.4 and 7.2 through 7.5, and Regulatory Guides 1.53 and 1.105 provide guidance ensuring that the reactor coolant system and associated auxiliary, control, and protection systems are designed with appropriate margin.

   Meeting the requirements of GDC 15 provides assurance that the reactor coolant pressure boundary will not be breached during any condition of normal operation, including the effects of AOOs.[28]

3. Compliance with GDC 20 requires that the protection system be designed (a) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (b) to sense accident conditions and to initiate the operation of systems and components important to safety.

   GDC 20 is applicable to this section because the reviewer evaluates the consequences of the events associated with startup of an inactive loop or a recirculation loop at an incorrect temperature and with a flow controller malfunction causing an increase in BWR core flow rate. This section, SRP Sections 4.2 through 4.4 and 7.2 through 7.5, and Regulatory Guides 1.53 and 1.105 provide guidance for ensuring that the reactor coolant system is designed with appropriate margin. Thus, when the reactor protection system senses an accident condition, it will initiate the operation of systems and components important to safety to ensure that acceptable fuel design limits are not exceeded.

   Meeting the requirements of GDC 20 provides assurance that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs.[29]

4. Compliance with GDC 26 requires that one of the reactivity control systems shall use 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 to this section because the reviewer evaluates the consequences of the events associated with startup of an inactive loop or a recirculation loop at an incorrect temperature and with a flow controller malfunction causing an increase in BWR core flow rate. This section, SRP Sections 15.4.4, 4.2 through 4.4 and 7.2 through 7.5, and Regulatory Guides 1.53 and 1.105 provide guidance for ensuring that the reactivity control system (control rods) is capable of reliably controlling reactivity changes with appropriate margin for malfunctions such as stuck rods.

Meeting the requirements of GDC 26 provides assurance that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs.[30]

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

   GDC 28 is applicable to this section because the reviewer evaluates the consequences of the events associated with startup of an inactive loop or a recirculation loop at an incorrect temperature and with a flow controller malfunction causing an increase in BWR core flow rate. This section, SRP Sections 4.2 through 4.4 and 7.2 through 7.5, and Regulatory Guides 1.53 and 1.105 provide guidance for ensuring that the reactor coolant system and associated auxiliary, control, and protection systems are designed with appropriate margin to ensure that the reactor coolant pressure boundary will not be breached.

   Meeting the requirements of GDC 28 provides assurance that the reactor coolant pressure boundary will not be breached during any condition of normal operation, including the effects of AOOs.[31]

## III. REVIEW PROCEDURES

The procedures below are used during ~~both the~~ construction permit (CP), ~~and~~ operating license (OL), and combined license (COL) reviews. During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL or COL[32] 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 core flow increase ~~transients~~(AOOs) presented 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 (see III.3.b), is accounted for.

If the SAR states that a particular core flow ~~transient~~AOO is not as limiting as some other similar ~~transient~~AOO, the reviewer evaluates the justification presented by the applicant. The applicant should present a quantitative analysis in the SAR of the increase in flow ~~transient~~AOO that is determined to be most limiting. For this ~~transient~~AOO, the ~~RSB~~SRXB reviewer, with the aid of the ~~ICSB~~HICB[33] reviewer, reviews the timing of the initiation of protection, engineered safety feature, and other systems needed to limit the consequences of the core flow increase ~~transient~~AOO to acceptable levels. The ~~RSB~~SRXB reviewer compares the predicted variation of system parameters with various trip setpoints. The ~~ICSB~~HICB[34] review of Chapter 7 of the SAR confirms that the instrumentation and control system design is consistent with the requirements for safety system actions for these events.

To the extent deemed necessary, the ~~RSB~~SRXB reviewer evaluates the effect of single active failures of safety systems and components which may alter the course of the ~~transient~~AOO. This phase of the review uses the system review procedures described in the SRP sections for Chapters 5, 6, 7, and 8 of the SAR. The reviewer considers and evaluates the possibility of a single failure that would permit the loop isolation valves to open prior to startup of a pump in an idle loop (for those plants with loop isolation valves). If this could occur, the core flow rate increase would not be limited by the rate at which the valve opens, and the resulting rate of reactivity insertion could be greater than for other ~~transients~~AOOs of this group.

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 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 conditions, including fuel data,[35] 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 the selected core burnup yields the minimum margins is evaluated. ~~CPB is consulted regarding the values of the reactivity parameters and fuel data used in the applicant's analysis.~~[36]

The results of the analysis are reviewed and compared with 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 the 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 CPS (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).[37]

The values of the more important of these parameters for the core flow increase ~~transients~~AOOs are compared 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.[38]

## IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and that the~~his~~[39] review supports the following kinds of statements and conclusions which should be included in the staff's safety evaluation report (SER):

A number of plant ~~transients~~AOOs can result in a core flow increase.  Those that might be expected to occur with moderate frequency are the startup of an idle recirculation pump (BWR), flow controller malfunction causing increasing core flow (BWR), startup of a pump in an inactive reactor coolant loop (PWR), and startup of a pump in an initially isolated inactive reactor coolant pump loop.[2]  All these postulated ~~transients~~AOOs have been reviewed.  It was found that the most limiting with regard to core thermal margins and pressure within the reactor coolant and main steam systems was the _________ ~~transient~~AOO.  This ~~transient~~AOO was evaluated by the applicant using a mathematical model that has been previously reviewed and found acceptable by the staff.  The parameters used as input to this model were reviewed and found to be suitably conservative.

> The staff concludes that the plant design with regard to ~~transients~~AOOs that result in an increase in coolant flow through the reactor core is acceptable and meets the relevant requirements of General Design Criteria 10, 15, 20, 26, and 28.  This conclusion is based on the following:
>
> 1. The applicant has met the requirements of General Design Criteria 10, 20, and 26 with respect to demonstrating that the specified acceptable fuel design limits are not exceeded for this event.
>
> 2. The applicant has met the requirements of General Design Criteria 15 and 28 with respect to ensuring~~assuring~~ that the design conditions of the reactor coolant pressure boundary are not exceeded because the protection system operates to maintain the maximum pressure within the reactor coolant and main steam system pressures below 110% of the design values.
>
> 3. The applicant has met the positions of Regulatory Guide 1.53 as related to the single-failure criterion and Regulatory Guide 1.105 as related to instrument actuations of systems and components important to safety.

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.

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[2] The SER should present one statement for all similar transients.

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.

## VI. REFERENCES


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

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

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

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

5. General Design Criterion 20, "Protection System Functions."

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

7. General Design Criterion 28, "Reactivity Limits."

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

9. Regulatory Guide 1.105, "Instrument Spans and Setpoints."

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**SRP Draft Section 15.4.4**
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 to SRXB. |
| 2. | SRP-UDP format item | Changed "transients" to "anticipated operational occurrences (AOOs)" throughout this section to accommodate Generic Issue B-3. |
| 3. | Editorial | Defined "SRP" as "Standard Review Plan" in text of AREAS OF REVIEW and in footnote 1. |
| 4. | SRP-UDP format item | Added Regulatory Guide 1.70 and its title and deleted unnecessary reference callout, "(Reference 1)." |
| 5. | SRP-UDP format item | Added table description and title. |
| 6. | SRP-UDP format item | Added reference title in lieu of "Ref. 1." |
| 7. | Current PRB abbreviation | Changed PRB to SRXB (global change for this section). |
| 8. | Editorial | Changed "assure" to "ensure" (global change for this section). |
| 9. | SRP-UDP format item | Relocated under "Review Interfaces" and formatted into numbered paragraphs for clarity. |
| 10. | SRP-UDP format item | "Review Interfaces" added to AREAS OF REVIEW and formatted in numbered paragraphs to describe how SRXB reviews aspects of the "Startup of as Inactive Loop or Recirculation Loop at an Incorrect Temperature" under other SRP sections and how other branches support the review of the event. Wording was preserved. |
| 11. | Current SRB name and abbreviation | Changed SRB to Reactor Systems Branch (SRXB). |
| 12. | Current SRB abbreviation | Changed SRB to HICB. |
| 13. | Current SRB abbreviation | Changed SRB to Emergency Preparedness and Radiation Protection Branch (PERB). |
| 14. | Current SRB abbreviation | Changed SRB to PERB. |
| 15. | Editorial | Simplified text to provide clarity and readability. |
| 16. | Editorial | Provided "GDC 10" and "GDC 20" as initialisms for "General Design Criteria 10 and 20," and changed pronoun from singular to plural form to provide number agreement. |
| 17. | Editorial | Provided "GDC 15" and "GDC 28" as initialisms for "General Design Criteria 15 and 28," and changed pronoun from singular to plural form to provide number agreement. |

## SRP Draft Section 15.4.4

Attachment A - Proposed Changes in Order of Occurrence

| Item | Source | Description |
|---|---|---|
| 18. | Editorial | Provided "GDC 26" as initialism for "General Design Criterion 26." |
| 19. | SRP-UDP format item | Added document title in lieu of "Ref. 2." |
| 20. | Editorial | Defined "DNBR" as "departure from nucleate boiling ratio." |
| 21. | Editorial | Defined "CPR" as "critical power ratio." |
| 22. | Editorial | Defined "MCPR" as "minimum critical power ratio." |
| 23. | Current SRB abbreviation | Changed SRB to HICB. |
| 24. | Editorial | Changed "GDC" to "General Design Criteria" to accommodate plural usage. |
| 25. | SRP-UDP format item | "Technical Rationale" added to ACCEPTANCE CRITERIA and formatted in numbered paragraphs to describe the bases for referencing the General Design Criteria. |
| 26. | SRP-UDP format item | Added lead-in sentence to "Technical Rationale." |
| 27. | SRP-UDP format item | Added technical rationale for GDC 10. |
| 28. | SRP-UDP format item | Added technical rationale for GDC 15. |
| 29. | SRP-UDP format item | Added technical rationale for GDC 20. |
| 30. | SRP-UDP format item | Added technical rationale for GDC 26. |
| 31. | SRP-UDP format item | Added technical rationale for GDC 28. |
| 32. | SRP-UDP format item | Added reference to combined license (COL) review per 10 CFR Part 52, and made minor editorial adjustments to accommodate the change. |
| 33. | Current SRB abbreviation | Changed SRB to HICB. |
| 34. | Current SRB abbreviation | Changed SRB to HICB. |
| 35. | SRP-UDP format item | Relocated from the item 36 line-out.  Changed PRB to SRXB. |
| 36. | SRP-UDP format item | CPB became the PRB, SRXB, and therefore cannot be consulted. |
| 37. | Editorial | Revised an extremely complex sentence to improve clarity. |
| 38. | SRP-UDP Guidance, Implementation of 10 CFR 52 | Added standard paragraph to address application of Review Procedures in design certification reviews. |
| 39. | Editorial | Changed to provide noun-verb agreement and to eliminate gender-specific reference. |

| Item | Source | Description |
|---|---|---|
| 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. |

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## SRP Draft Section 15.4.4

Attachment B - Cross Reference of Integrated Impacts

| Integrated Impact No. | Issue | SRP Subsections Affected |
|---|---|---|
| | No Integrated Impacts were incorporated in this SRP Section. | |