NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.3.1 – 15.3.2 LOSS OF FORCED REACTOR COOLANT FLOW INCLUDING TRIP OF PUMP MOTOR AND FLOW CONTROLLER MALFUNCTIONS

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. An ~~resulting~~[2] increase in fuel temperature and accompanying fuel damage could then result if specified acceptable fuel damage limits are exceeded during the transient. A number of transients that are expected to occur with moderate frequency and that result in a decrease in forced reactor coolant flow rate are covered by this Standard Review Plan (SRP)[3] section. Each of these transients should be discussed in individual sections of the applicant's safety analysis report (SAR) as suggested by the Standard Format (Ref. 4).

Core thermal and hydraulic transients associated with partial and complete loss of reactor coolant flow are evaluated. These include:

1. For boiling water reactors (BWRs), partial and complete recirculation pump trips and malfunctions of the recirculation flow controller to cause decreasing flow.

2. For pressurized water reactors (PWRs), partial and complete reactor coolant pump trips.

A partial loss of coolant flow may be caused by a mechanical or electrical failure in a pump motor, a fault in the power supply to the pump motor, a pump motor trip caused by such anomalies as over-current or phase imbalance, or a failure within the recirculation flow control network (BWR) resulting in decreasing flow. A complete loss of forced coolant flow may result from the simultaneous loss of electrical power to all pump motors.

The review includes the postulated initial core and reactor conditions which are pertinent to the loss of flow transient; the methods of thermal and hydraulic analysis; the postulated sequence of events, including time delays prior to and after protective system actuation; assumed reactions of reactor systems components; the functional and operational characteristics of the reactor protection system in terms of how it affects the sequence of events; and all operator actions required to secure and maintain the reactor in a safe condition.

The results of the applicant's analyses are reviewed to ensure that values of pertinent system parameters are within expected ranges for the type and class of reactor under review. The system parameters that are evaluated include core flow and flow distribution, channel heat flux (average and hot), minimum critical heat flux ratio (or minimum critical power ratio), departure from nucleate boiling ratio, vessel water level, thermal power, vessel pressure, steam line pressure (BWR), main steam flow (BWR), and feedwater flow (BWR). The results of the applicant's fuel damage analysis are reviewed by the methods described in SRP Section 4.2~~ (Ref. 13)~~.[4]

The sequence of events described in the SAR is reviewed by ~~RSB~~SRXB.[5] This review is coordinated with the[6] Instrumentation and Controls~~ Systems~~ Branch ~~(ICSB)~~(HICB).[7] The ~~RSB~~SRXB[8] review concentrates on the need for the reactor protection system, the engineered safety system, and operator action to secure and maintain the reactor in a safe condition.

The analytical methods are reviewed by ~~RSB~~SRXB[9] 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 ~~RSB~~SRXB[10] reviewer ~~requests~~ initiat~~ion of~~es a generic evaluation of the new analytical model~~ by the Core Performance Branch (CPB)~~.[11]

The values of all parameters used in a new analytical model, including the initial conditions of the core and system, are reviewed. It is the responsibility of the ~~RSB~~SRXB[12] reviewer to~~ contact his counterpart in CPB to~~[13] ensure that the appropriate physics and fuel data have been used in any staff calculations.

<u>Review Interfaces</u>

~~In addition, t~~The~~ RSB~~ SRXB[14] will coordinate other branch~~es'~~ evaluations that interface with the overall review of~~ the~~ loss of reactor coolant flow transients~~system~~,[15] as follows:

1.[16] The ~~ICSB~~HICB[17] reviews the instrumentation and control 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 system analysis as part of its primary review responsibility for SRP Sections 7.2 through 7.5~~ (Ref. 15 through 18)~~.[18]

2. The ~~CPB~~SRXB,[19] as part of its primary review responsibility for SRP Section 4.4 ~~(Ref. 14)~~,[20] performs generic reviews of the thermal-hydraulic computer models used for this transient and also performs, ~~upon request~~as appropriate,[21] additional analyses related to these accidents for selected reactor types.

3. The ~~Procedures Test Review Branch (PTRB)~~Quality Assurance and Maintenance Branch (HQMB)[22] review confirms that a commitment has been made in the SAR to conduct preoperational tests to verify flow coastdown calculations.

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[23] acceptance criteria are based on meeting the relevant requirements of the following regulations:

A. General Design Criterion 10 (GDC 10)[24] ~~(Ref. 1)~~,[25] as it relates to the reactor coolant system being designed with appropriate margin to ~~assure~~ ensure[26] that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences.

B. General Design Criterion 15 (GDC 15)[27] ~~(Ref. 2)~~,[28] 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 (GDC 17) as it relates to providing onsite and offsite electric power systems to ensure that structures, systems, and components important to safety will function 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.[29]

~~C~~D. General Design Criterion 26 (GDC 26)[30] ~~(Ref. 33)~~,[31] 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.

The basic objectives of the review of loss of forced reactor coolant flow transients are:

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

2. To verify that, for the most limiting transients, the plant responds to the loss of flow transients 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 ~~GDC~~ General Design Criteria[32] 10, 15, and 26 for incidents of moderate frequency[a] are:

a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design ~~valves~~values[33] (Ref. 5).

b. Fuel-cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR)[34] 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, the number of 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.

The applicant's analysis of the loss of reactor coolant flow transients should use an acceptable analytical model. The equations, sensitivity studies, and models described in References 7 through 10 are acceptable. References 19 through 23 are acceptable computer codes for transient analyses of CE80+ applications that do not involve loss-of-coolant accidents (LOCAs).[35] References 24 and 25 are acceptable transient analysis computer codes for design analysis of the ABWR.[36] If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer ~~requests~~initiates an evaluation ~~by~~for ~~CPB~~SRXB.[37]

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[a] The term "moderate frequency" is used in this SRP section in the same sense as in the descriptions of design and plant process conditions in References 11 and 12.

The values of parameters used in the analytical model ~~would~~should[38] be suitably conservative. The following values are considered acceptable for use in the model:

a. The reactor is initially at rated output (licensed core thermal power) for the number of loops assumed operating, plus 2% to account for power measurement uncertainty, 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, axial power profile, and radial power distribution.

<u>Technical Rationale</u>

The technical rationale for application of these acceptance criteria to transients involving loss of flow is discussed in the following paragraphs:[39]

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 loss of forced reactor coolant flow, including a trip of pump motors and flow controller malfunctions. These are anticipated operational occurrences that create a potential to exceed specified acceptable fuel design limits because a transient reduction in reactor coolant flow causes a corresponding rise in fuel-cladding temperature.

   Meeting the requirements of GDC 10 provides a level of assurance that specified acceptable fuel design limits are not exceeded and that fuel-cladding integrity is maintained for anticipated operational occurrences involving loss of forced reactor coolant flow.[40]

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 analyzes anticipated operational occurrences involving loss of forced reactor coolant flow. In these transients, a reduction in reactor coolant flow can cause the reactor coolant system pressure to increase above normal levels. Therefore, for loss-of-flow transients covered by SRP

Section 15.3.1 – 15.3.2, the reactor coolant pressure needs to be analyzed to ensure that the pressure acceptance criterion is satisfied.

Meeting the requirements of GDC 15 provides a level of assurance that the design conditions of the reactor coolant pressure boundary are not exceeded for the anticipated occurrences involving loss of forced reactor coolant flow evaluated in this SRP section.[41]

c. Compliance with GDC 17 requires that onsite and offsite electrical power systems be provided to ensure that structures, systems, and components important to safety will perform their intended function. Each power system (assuming the other system is not functioning) shall 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 anticipated operational occurrences.

GDC 17 is applicable to SRP Section 15.3.1-15.3.2 because this section reviews the analysis of a group of abnormal operating occurrences to which the GDC must be applied.

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 initiating events involving a decrease in flow in the reactor coolant system, concurrent with a loss of offsite power (LOOP).[42]

d. Compliance with GDC 26 requires that one of the reactivity control systems at nuclear power plants include control rods with the capability to control reactivity changes, thereby ensuring that specified acceptable fuel design limits are not exceeded under conditions of normal operation, including anticipated operational occurrences. The design for this system must have an appropriate margin to accommodate malfunctions such as stuck rods.

GDC 26 is applicable to this section because the reviewer analyzes anticipated operational occurrences involving loss of forced reactor coolant flow. The transients analyzed in this section may involve the movement of control rods in response to the transient. In such cases, 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 15.3.1 – 15.3.2 examines this margin to ensure that specified acceptable fuel design limits are not exceeded.

Meeting the requirements of GDC 26 provides a level of assurance that appropriate margins are included to accommodate malfunctions (including stuck rods) of the reactivity control system, thereby minimizing the possibility that specified acceptable fuel design limits are not exceeded.[43]

III. REVIEW PROCEDURES

The procedures below are used during ~~both~~ the construction permit (CP), combined license (COL),[44] and operating license (OL) 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[45] review ~~state~~stage,[46] 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 each of the loss of reactor coolant flow transients 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 extent to which credit is[48] taken for the functioning of normally operating plant systems.

4. The extent to which the operation of engineered safety systems ~~that are~~ is[49] required.

5. The extent to which operator actions are required.

6. That appropriate margin for malfunctions, such as stuck rods, are accounted for.

If the SAR states that a particular loss of flow transient is not as limiting as some other similar transients, the reviewer evaluates the justification presented by the applicant. The reviewer confirms that all types of flow loss transients are considered, e.g., pump trips during two-, three-, and four-loop operation. The applicant is to present a quantitative analysis in the SAR of the loss of flow transient that is determined to be most limiting. For this transient, the ~~RSB~~SRXB[50] reviewer, in ~~coordinating~~ coordination[51] with the ~~ICSB~~HICB[52] reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to adequately limit the consequences of the loss of flow. The ~~RSB~~SRXB[53] reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints and evaluates the effects 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; each loss of flow transient 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.)[54] The ~~ICSB~~HICB[55] review of Chapter 7 of the SAR confirms that the instrumentation and control design is consistent with the requirements for safety systems actions for these events.

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 the ~~RSB~~SRXB[56] to determine if these models have been previously reviewed and found acceptable by the staff. If not, ~~CPB is requested to~~the SRXB should[57] initiate a generic review of the applicant's proposed model.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by the ~~RSB~~SRXB.[58] Of particular importance are the reactivity coefficients and control rod worths used by the applicant's ~~in his~~[59] 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 applicant ~~he~~[60] has selected the core burnup that yields the minimum margins is evaluated. ~~CPB is consulted regarding~~ SRXB reviews[61] the values of the reactivity parameters used in the applicant's analysis.

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 temporal changes 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 systems (if applicable) during the transient are reviewed. The important parameters for the loss of reactor coolant flow transients are compared to those predicted for other similar plants to verify that they are within the expected range.

Note: In the Final Safety Evaluation Report for the Advanced Boiling Water Reactor (ABWR)(Reference 26), the staff allowed an exception to Acceptance Criterion II.A for the postulated trip of all reactor internal pumps (RIPs) with offsite power available. Normally, such transients are treated as anticipated operational occurrences, which must not result in specified acceptable fuel design limits being exceeded. For this special case, the transient is not expected to occur during the lifetime of the plant and is not classified as an anticipated operational occurrence, but rather as an anticipated transient involving a common-mode software failure. Accordingly the following criterion for the radiological dose calculation was established: fuel failure need not be assumed in dose calculations for fuel rods that are at or below a temperature of approximately 600 ^{o}C (1111 ^{o}F) for less than 60 seconds for fuel that has achieved a burnup of 20 gigawatt-days per metric ton or less. (For fuel beyond this burnup, the dose calculations must assume fuel failure for all fuel rods that achieve transition boiling because the test data do not go beyond 20 gigawatt-days per metric ton.) The resulting dose should not exceed 10 percent of 10 CFR Part 100, which is considered appropriate for an event of such frequency because of the unique design features of ABWR instrumentation and control systems.[62]

~~CPB is consulted regarding~~The SRXB reviews[63] the specified acceptable fuel design limits (SAFDLs). ~~AEB~~The PERB[64] is notified regarding the extent of fuel failures predicted by the analysis if SAFDLs are exceeded.

The ~~PTRB~~HQMB[65] review confirms that a commitment has been made in the SAR to conduct preoperational tests to verify flow coastdown calculations.

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.[66]

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and ~~his~~ that the[67] review supports the following kinds of statements and conclusions which should be included in the staff's safety evaluation report (SER):

> Several types of plant occurrences can result in an unplanned decrease in reactor coolant flow rate. The ones expected during the life of the plant are those caused by reactor coolant (or recirculation) pump trips or a flow controller malfunction.[b] All these postulated transients have been reviewed. It was found that the most limiting in regard to core thermal margins and pressure within the reactor coolant and main steam systems was the _______ transient. This transient was evaluated by the applicant using a mathematical model that has been reviewed and found acceptable by the staff. The values of 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 that are expected to occur during plant life and result in a loss or decrease in forced reactor coolant flow is acceptable and meets the relevant requirements of General Design Criteria ~~(GDC)~~[68] 10, 15, 17,[69] and 26. This conclusion is based on the following:
>
> 1. The applicant has met the requirements of ~~GDC~~ General Design Criteria[70] 10, 17[71] and 26 with respect to demonstrating that the specified acceptable fuel design limits are not exceeded for this event. This requirement has been met since the results of the analysis showed that the thermal margin limits (~~MDNBR~~minimum departure from nucleate boiling ratio for ~~PWRs~~ pressurized water reactors and ~~MCPR~~ minimum critical power ratio for ~~BWRs~~ boiling water reactors)[72] are satisfied as indicated by SRP Section 4.4.
>
> 2. The applicant has met the requirements of General Design Criterion ~~GDC~~[73] 15 and 17[74] with respect to demonstrating that the reactor coolant pressure boundary limits have not been exceeded for this event. This requirement has been met

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

since the analysis showed that the maximum pressure of the reactor coolant and main steam systems did not exceed 110% of the design pressure.

3. The applicant has met the requirements of General Design Criterion ~~GDC~~[75] 26 with respect to the capability of the reactivity control system to provide adequate control of reactivity during this event while including appropriate margin for stuck rods since the specific acceptable fuel design limits were not exceeded.

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.[76]

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.[77] 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.[78]

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."

2. 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant."

3. 10 CFR Part 50, Appendix A, General Design Criterion 17, "Electric Power Systems."[79]

~~3~~4. 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

~~4~~5. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

~~5~~6. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

~~6~~7. "Standard Safety Analysis Report - BWR/6," General Electric Company, April 1973.

~~7~~8. "Reference Safety Analysis Report - RESAR-3," Westinghouse Nuclear Energy Systems, November 1973. "Reference Safety Analysis Report - RESAR-41," Westinghouse Nuclear Energy Systems, October 1976.

~~8~~9. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973.

~~9~~10. "Standard Nuclear Steam System B-SAR-205," Babcock & Wilcox Company, February 1976.

11~~0~~. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

12~~1~~. ANS Trial Use Standard N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

13~~2~~. Standard Review Plan Section 4.2, "Fuel System Design."

14~~3~~. Standard Review Plan Section 4.4, "Thermal and Hydraulic Safety."

15~~4~~. Standard Review Plan Section 7.2, "Reactor Trip System."

16~~5~~. Standard Review Plan Section 7.3, "Engineered Safety Features System."

17~~6~~. Standard Review Plan Section 7.4, "Systems Required for Safe Shutdown."

18~~7~~. Standard Review Plan Section 7.5, "Safety-Related Display Instrumentation."

19. CESEC-III (CENPD-107; LD-82-001). (Calculates system parameters such as core power, flow, pressure, temperature, and valve actions during a transient.)[80]

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 calculations for DNBR use the CE-1 critical heat flux correlation.)[81]

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.)[82]

22. COAST (SSAR; CENPD-98). (Calculates the time-dependent reactor coolant mass flow rate in each loop during reactor coolant pump coastdown transients.)[83]

23. STRIKIN-II (CENPD-133; CENPD-135 Supps. 2 and 4). (Calculates the cladding and fuel temperatures for an average or hot fuel rod.)[84]

24. General Electric Company, ODYNA - One Dimensional Dynamic Model (proprietary computer software for use in ABWR transient analysis to simulate pressurization events).[c] [85]

25. General Electric Company, REDYA (proprietary computer software for use in ABWR transient analysis to simulate other than pressurization events).[86]

26. NUREG-1503, Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design, July 1994, Section 15.2, "Trip of All Reactor Internal Pumps and Pressure Regulator Down-Scale Failure."[87]

[c] In Generic Letter 81-08, dated January 29, 1981, all BWR licensees and applicants were informed that transient analyses performed by the General Electric Company (GE) to support reload submittal received after February 1, 1981, must contain appropriate ODYN analyses in place of those previously performed with REDY for the limiting transients. These codes have since been modified by GE for use in the analysis of limiting transients on the standard design Advanced Boiling Water Reactor (ABWR). These modified codes, ODYNA and REDYA, were reviewed by the NRC staff and have been approved for design analysis of the ABWR.

SRP Draft Section 15.3.1
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 Reactor Systems Branch (SRXB).
2.	Editorial	Deleted redundant use of "result."
3.	Editorial	Defined "SRP" as "Standard Review Plan."
4.	SRP-UDP format item	Deleted unnecessary in-text callout for Reference 13.
5.	Current primary review branch designation	Changed PRB to SRXB.
6.	Editorial	Added "the" before branch name.
7.	Current review branch name and designation	Changed review interface branch to Instrumentation Controls Branch (HICB).
8.	Current primary review branch designation	Changed PRB to SRXB.
9.	Current primary review branch designation	Changed PRB to SRXB.
10.	Current primary review branch designation	Changed PRB to SRXB.
11.	SRP-UDP update item	Responsibility for reviewing analytical models for reactor transients has been assumed by SRXB.
12.	Current primary review branch designation	Changed PRB to SRXB.
13.	SRP-UDP update item	Responsibility for reviewing analytical models for reactor transients has been assumed by SRXB.
14.	Current primary review branch designation	Changed PRB to SRXB.
15.	SRP-UDP format item	Added "Review Interfaces" and modified lead-in paragraph for AREAS OF REVIEW.
16.	SRP-UDP format item	Divided paragraph describing review interfaces into numbered subparagraphs, one for each review interface.
17.	Current review branch designation	Changed review interface branch to HICB.
18.	SRP-UDP format item	Deleted unnecessary in-text citations for References 15 through 18.
19.	Current primary review branch designation	Changed PRB to SRXB.
20.	SRP-UDP format item	Deleted unnecessary in-text callout for Reference 14.

Item	Source	Description
21.	Editorial	Responsibility for reviewing analytical models for reactor transients has been assumed by SRXB.
22.	Current review branch name and designation	Changed review interface branch to Quality Assurance and Maintenance Branch (HQMB).
23.	Current primary review branch designation	Changed PRB to SRXB.
24.	Editorial	Provided "GDC 10" as an acronym for "General Design Criterion 10."
25.	SRP-UDP format item	Deleted unnecessary in-text callout for Reference 1.
26.	Editorial	Changed "assure" to "ensure" (global change for this section).
27.	Editorial	Provided "GDC 15" as acronym for "General Design Criterion 15."
28.	SRP-UDP format item	Deleted unnecessary in-text callout for Reference 2.
29.	**Integrated Impact 1512**	Added GDC 17 as a new acceptance criterion, item C and renumbered next criterion accordingly.
30.	Editorial	Provided "GDC 26 as acronym for "General Design Criterion 26."
31.	SRP-UDP format item	Deleted "(Ref. 33)" in accordance with standard practice to delete unnecessary references.
32.	Editorial	Changed "GDC" to "General Design Criteria" to accommodate plural usage.
33.	Editorial	Changed "valves" to "values" to correct an error.
34.	Editorial	Defined "DNBR," "CPR," and "MCPR" as "departure from nucleate boiling ratio," "critical power ratio," and "minimum critical power ratio," respectively.
35.	Integrated Impact No. 939	Added references to acceptable computer codes for non-LOCA analysis for CE80+ applications.
36.	Integrated Impact No. 940	Added references to acceptable computer codes for design analysis of the ABWR.
37.	SRP-UDP update item	Responsibility for reviewing analytical models for reactor transients has been assumed by SRXB.
38.	Editorial	Changed "would" to "should" to correct an error.
39.	SRP-UDP format item	Added "Technical Rationale" and lead-in paragraph to ACCEPTANCE CRITERIA.
40.	SRP-UDP format item	Added technical rationale for GDC 10.
41.	SRP-UDP format item	Added technical rationale for GDC 15.

Item	Source	Description
42.	**Integrated Impact 1512**	Added Technical Rationale for GDC 17.
43.	SRP-UDP format item	Added technical rationale for GDC 26.
44.	SRP-UDP format item	Added "combined license (COL)" to REVIEW PROCEDURES.
45.	SRP-UDP format item	Added "COL" to REVIEW PROCEDURES.
46.	Editorial	Substituted "stage" for "state" to correct an error.
47.	Current primary review branch designation	Changed PRB to SRXB.
48.	Editorial	Changed sentence to provide parallelism and improve clarity.
49.	Editorial	Changed sentence to provide parallelism and improve clarity.
50.	Current primary review branch designation	Changed PRB to SRXB.
51.	Editorial	Changed "coordinating" to "coordination" to correct an apparent typographical error.
52.	Current review branch designation	Changed review interface branch to HICB.
53.	Current primary review branch designation	Changed PRB to SRXB.
54.	**Integrated Impact 1512**	Added the new staff position from the CE 80+ FSER that indicates that LOOP may not be considered a single failure.
55.	Current review branch designation	Changed review interface branch to HICB.
56.	Current primary review branch designation	Changed PRB to SRXB.
57.	SRP-UDP update item	Responsibility for reviewing analytical models for reactor transients has been assumed by SRXB.
58.	Current primary review branch designation	Changed PRB to SRXB.
59.	SRP-UDP format item	Revised to eliminate gender-specific reference.
60.	SRP-UDP format item	Revised to eliminate gender-specific reference.
61.	SRP-UDP update item	Responsibility for reviewing analytical models for reactor transients has been assumed by SRXB.
62.	**Integrated Impact 1354**	Added a paragraph discussing the special criteria applied to a transient unique to the ABWR.
63.	SRP-UDP update item	Responsibility for reviewing analytical models for reactor transients has been assumed by SRXB.

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Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
64.	Current review branch designation	Changed review interface branch to PERB.
65.	Current review branch designation	Changed review interface branch to HQMB.
66.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
67.	SRP-UDP format item	Revised to eliminate gender-specific reference.
68.	Editorial	Deleted "(GDC)" as an acronym for "General Design Criteria." GDC is appropriately used as an acronym for General Design Criterion.
69.	**Integrated Impact 1512**	Added GDC 17 to the list of acceptance criteria addressed in sample evaluation findings.
70.	Editorial	Changed "GDC" to "General Design Criteria" to accommodate plural usage.
71.	**Integrated Impact 1512**	Added GDC 17 to the list of acceptance criteria addressed in sample evaluation findings.
72.	Editorial	Defined acronym used in sample SER section.
73.	Editorial	Defined GDC.
74.	**Integrated Impact 1512**	Added GDC 17 to the list of acceptance criteria addressed in sample evaluation findings.
75.	Editorial	Defined GDC.
76.	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.
77.	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.
78.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
79.	**Integrated Impact 1512**	Added GDC 17 to the list of references and renumbered other references accordingly.
80.	Integrated Impact No. 939	Added the CESEC-III code as Reference 19.
81.	Integrated Impact No. 939	Added the TORC code as Reference 20.
82.	Integrated Impact No. 939	Added the HERMITE code as Reference 21.
83.	Integrated Impact No. 939	Added the COAST code as Reference 22.
84.	Integrated Impact No. 939	Added the STRIKIN-II code as Reference 23.

Item	Source	Description
85.	Integrated Impact No. 940	Added the ODYNA code as Reference 24.
86.	Integrated Impact No. 940	Added the REDYA code as Reference 25.
87.	Integrated Impact No. 1354	Added Section 15.2 of the ABWR FSER as Reference 26.

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Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
685	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 15.3.1 for Integrated Impact No. 685.
939	Add NRC staff-approved codes for transient analysis of the CE80+ plant to SRP Section 15.3.1 – 15.3.2.	II, VI.18, VI.19, VI.20, VI.21, & VI.22
940	Add NRC staff-approved codes for transient analysis of the ABWR plant to SRP Section 15.3.1 – 15.3.2.	II, VI.23, & VI.24.
1354	Add the review procedures note for the ABWR to SRP Section 15.3.1 – 15.3.2 for the postulated trip of all of the reactor internal pumps with offsite power available for the ABWR.	III, & VI.25
1512	Added GDC 17 as Acceptance Criteria and incorporate staff positions from ABB-CE 80+ into the Review Procedures	II, III, IV and Reference 3.