NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.3.3 – 15.3.4 REACTOR COOLANT PUMP ROTOR SEIZURE AND REACTOR COOLANT PUMP SHAFT BREAK

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a reactor coolant pump (RCP)[2] in a pressurized water reactor (PWR) or recirculation pump in a boiling water reactor (BWR). Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. This Standard Review Plan (SRP)[3] section is intended to cover both of these accidents.

The review ~~is conducted with radioactivity releases and with the~~ includes evaluation of the applicant's postulated initial and long-term core and reactor conditions ~~that are~~ pertinent to the rotor seizure or broken shaft events; the methods of thermal and hydraulic analysis; the postulated sequence of events, including time delays prior to and after protective system actuation; the assumed reactions of reactor system 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.[4]

The results of the applicant's analyses are reviewed to assess fuel damage and to ensure that values of pertinent system parameters are within expected ranges for the type and class of reactor under review. Fuel damage is assessed by the methods described in SRP Section 4.2~~ (Ref. 13)~~.[5] ~~The system parameters that are evaluated include: core flow and flow distribution (including hydraulic instabilities), 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).~~ System parameters to be reviewed include the following:

- core flow and flow distribution (including hydraulic instabilities),
- 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).[6]

The sequence of events described in the safety analysis report (SAR)[7] is reviewed by ~~RSB~~RSXB. ~~This review is coordinated with Instrumentation and Control Systems Branch (ICSB).~~[8] The ~~RSB~~RSXB review 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 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 reviewer ~~requests initiation of~~ initiates a generic evaluation of the new analytical model by ~~CPB~~SRXB.[9]

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 reviewer to ~~contact his counterpart in the Core Performance Branch (CPB) to~~[10] ensure that the appropriate physics and fuel data have been used in any staff calculations.

The ~~CPB~~SRXB 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 Section 4.4~~ (Ref. 12)~~.[11]

Review Interfaces[12]

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

1. The ~~Instrumentation and Controls Branch ICSB~~HICB[13] 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~~ (Refs. 14 through 17)~~.[14] ~~The CPBSRXB 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 Section 4.4 (Ref. 12).~~[15]

2. The~~ Accident Evaluation Branch (AEB)~~ Emergency Preparedness and Radiation Protection Branch (PERB)[16] is notified regarding the extent of the fuel failures that are predicted by the analysis. ~~AEB~~PERB[17] then evaluates the radiological consequences of the event.

3. The Materials and Chemical Engineering Branch (EMCB) reviews the fracture toughness properties of the reactor coolant pressure boundary and reactor vessel as part of its primary review responsibility for SRP Sections 5.2.3 and 5.3.1.[18]

4. The SPLB reviews the integrity of the reactor coolant pump seals as part of its primary review responsibility for SRP Section 9.2.2.[19]

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding review branch.

II. ACCEPTANCE CRITERIA

The ~~RSB~~SRXB acceptance criteria for maintaining the ability to insert the control rods ~~insertability~~ and to cool the core~~ coolability~~[20] during a RCP rotor seizure or broken shaft event are based on meeting the relevant requirements of the following regulations:

A. General Design Criterion (GDC) 17, as it relates to providing onsite and offsite electric power systems to ensure that structures, systems, and components important to safety will function. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that design conditions of the reactor coolant pressure boundary are not exceeded and the core is cooled in the event of postulated accidents.[21]

~~A~~B. General Design Criteria 27 (GDC 27)[22] and 28 (GDC 28),[23] as they relate to the reactor coolant system being designed with appropriate margin to ~~assure~~ ensure[24] that the capability to cool the core is maintained.

~~B~~C. General Design Criterion 31 (GDC 31),[25] as it relates to the reactor coolant system being designed with sufficient margin to ~~assure~~ ensure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized.

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

The basic objectives of the review of the accident resulting from a rotor seizure or shaft break in a reactor coolant pump are:

1. To identify which of these accidents is the more limiting.

2. To verify that, for the accident, the plant responds in such a way that the criteria regarding fuel damage, radiological consequences, and system pressure are met.

The specific criteria necessary to meet the relevant requirements of ~~GDC~~ General Design Criteria[26] 27, 28, and 31 and 10 CFR Part 100 for the rotor seizure and shaft break event are:

1. Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.

2. The potential for core damage is evaluated on the basis that it is acceptable if the minimum departure from nucleate boiling ratio (DNBR)[27] remains above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR)[28] remains above the minimum critical power ratio (MCPR)[29] safety limit for BWRs based on acceptable correlations (see SRP Section 4.4). If the DNBR or CPR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

3. Any ~~activity~~ release of radioactive material[30] must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines.

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

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

~~6. Tripping of the reactor coolant pumps should be consistent with the resolution to Action Item II.K.3.5 of NUREG-0718 and NUREG-0737.~~[31]

~~7~~6.[32] A rotor seizure or shaft break in a reactor coolant pump should not, by itself, generate a more serious condition or result in a loss of function of the reactor coolant system or containment barriers.

~~8~~7. Only safety-grade equipment should be used to mitigate the consequences of the event. Safety functions should be accomplished assuming the worst single failure of a safety system active component (see Refs. 5 and 6). For new applications, loss of offsite power (LOOP) should not be considered a single failure; reactor coolant pump rotor seizures and shaft breaks should be analyzed with a LOOP (see item 9, below) 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.)[33]

~~9~~8. The ability to achieve and maintain long-term core cooling ~~coolability of the core~~[34] should be verified.

~~10~~9. This event should be analyzed assuming turbine trip and coincident loss of offsite power and coastdown of undamaged pumps.

The applicant's analysis should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 8 through 12 are acceptable. The NRC staff found References 13 and 14 to be acceptable transient analysis computer codes for design analysis of the Advanced Boiling Water Reactor (ABWR).[35] References 15 through 19 were found to be acceptable computer codes for transient analyses (i.e., except for loss-of-coolant accidents, or LOCAs) for the Combustion Engineering System 80+ final safety evaluation report staff review.[36] In addition, NUREG-1465 contains guidance on accident source terms for light-water nuclear power plants. When conducting transient analyses, the NUREG-1465 guidance is particularly important for reviewing fractions of relevant isotopes (noble gases, iodine, cesium, and rubidium) and chemical species of iodine assumed to exist within the gap between fuel pellets and cladding.[37] 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 an evaluation. There are certain assumptions regarding important parameters used to describe the initial plant conditions and postulated system failures which should be used. These are listed below:

i. 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.

ii. 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.

iii. 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.

Technical Rationale[38]

The technical rationale for application of these acceptance criteria to reviewing analyses of transients initiated by RCP rotor seizure and shaft break is discussed in the following paragraphs:[39]

1. 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) is to provide sufficient capacity and capability to ensure that: (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences; and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. GDC 17 is applicable to SRP Section 15.3.3-15.3.4 because this section reviews the analysis of events that are classified as abnormal operating occurrences or postulated accidents, depending on the severity of the results. Meeting the requirements of GDC 17 provides assurance that the design conditions of the reactor coolant pressure boundary are not exceeded as a result of reactor coolant pump rotor seizures and shaft breaks and that the core is cooled and containment and other vital functions are maintained.[40]

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

   Compliance with GDC 28 requires that reactivity control systems be designed with appropriate limits on the amount and rate of reactivity increase, thereby ensuring 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) disturb the core, its support structures, or other reactor pressure vessel internals sufficiently to impair the capability to cool the core. Postulated reactivity accidents to be considered shall include rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor temperature and pressure, and the addition of cold water.

   GDC 27 and GDC 28 are applicable to this section because the reviewer evaluates two events (i.e., RCP rotor seizure and shaft break) that will result in transient conditions having the potential to affect reactor coolant temperature and pressure, which in turn could result in complex changes in core reactivity. The applicant's analyses of these transients in the SAR must demonstrate that reactivity, pressure, and temperature changes will not be severe enough to cause an unacceptable impact on the reactor coolant pressure boundary or on the capability for core cooling. The analyses must be independently reviewed by the staff in accordance with this SRP section.

   Meeting the requirements of GDC 27 and GDC 28 provides a level of assurance that a transient initiated by an RCP rotor seizure or shaft break will not result in (a) unacceptable stress on the reactor coolant pressure boundary or (b) a reduction in the capability of the core cooling or reactivity control systems to perform their design safety functions.[41]

3. Compliance with GDC 31 requires that, under the stress of operating, maintenance, testing, and postulated accident conditions, the reactor pressure boundary shall be designed with sufficient margin to ensure that (a) the boundary behaves in a nonbrittle manner and (b) the probability of rapidly propagating fracture is minimized. The design

shall reflect consideration of service temperatures and other boundary material variables under operating, maintenance, testing, and postulated accident conditions. The design will also address such issues as the uncertainties of determining material properties; the effects of irradiation on material properties; residual, steady state, and transient stresses; and the sizes of flaws.

GDC 31 is applicable to this section because the reviewer evaluates two events (RCP rotor seizure and shaft break) that could result in transient conditions having the potential for adversely affecting the reactor coolant pressure boundary. Loss of a reactor cooling pump will cause a rapid reduction in coolant flow through the core and, consequently, an increase in temperature and pressure. The amount of stress to which the reactor coolant pressure boundary is subjected depends on the severity of the transient. The severity of the transient is assessed by the applicant in the SAR and reviewed by the staff in accordance with this SRP section.

Meeting the requirements of GDC 31 provides a level of assurance that a transient initiated by an RCP rotor seizure or shaft break will not result in an unacceptable stress on the reactor coolant pressure boundary or on the ability to cool the reactor core.[42]

4. To establish the suitability of a nuclear power plant site, 10 CFR Part 100 specifies how the exclusion area, low population zone, and population center distance should be determined. Further, radiation exposure criteria stipulated in 10 CFR Part 100 provide reference values to be used in the site suitability determination based on postulated fission product releases associated with accidental events.

   10 CFR Part 100 is applicable to this section because it specifies the methodology for calculating radiation exposures at the site boundary for postulated accidents or events such as loss of a reactor coolant pump. For transients having a moderate frequency of occurrence, any release of radioactive material must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines. A small fraction is interpreted to be less than 10 percent of the 10 CFR Part 100 reference values. For the purpose of this review, the radiological consequences of a RCP rotor seizure or shaft break must include consideration of the containment, confinement, and filtering systems. The applicant's source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing should reflect NRC-approved source terms and methodologies such as those contained in NUREG-1465.

   Meeting this requirement provides a level of assurance that, in the event of a transient initiated by a reactor coolant pump rotor seizure or shaft break, radiation exposures at the site boundary will not exceed a small fraction of the reference values specified in 10 CFR Part 100.[43]

III. REVIEW PROCEDURES

The procedures below are used during ~~both~~ the construction permit (CP), ~~and~~ operating license (OL), and combined license (COL)[44] 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 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 applicant's analyses of the rotor seizure and shaft break events are 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 extent to which credit is taken for the functioning of normally operating plant systems.[46]

4. The extent to which the operation of engineered safety systems ~~that~~ is required.[47]

5. The extent and time at which operator actions are required.

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

If the SAR states that one of the accidents is not as limiting as the other, the reviewer evaluates the justification presented by the applicant. The applicant is to present a quantitative analysis in the SAR of the accident that is determined to be more limiting. For the accident that is found more limiting, the reviewer confirms that the effects of the accident are determined for each mode of operation (e.g., one-, two-, three-, or four-loop) allowed by the technical specifications. Either a separate analysis should be presented or each mode of operation or the effects of each mode should be referenced to the limiting case.

For the more limiting accident, the ~~RSB~~SRXB reviewer, with the aid of the ~~ICSB~~HICB[48] reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the accident to acceptable levels. The ~~RSB~~SRXB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The ~~ICSB~~HICB review of Chapter 7 of the SAR confirms that the instrumentation and control systems design is consistent with the requirements for safety systems actions for these events.

To the extent deemed necessary, the ~~RSB~~SRXB reviewer evaluates the effect of single active failures of safety systems and components which may alter the course of the accident. For new applications, the LOOP is not considered a single active failure, but considered in addition to a single active failure as discussed in subsection II.7.[49] 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 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, ~~CPB is requested to initiate~~ SRXB initiates[50] a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by the ~~RSB~~SRXB. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. ~~CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.~~[51]

The results of the applicant's analysis are reviewed and compared to the acceptance criteria presented in subsection II 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.~~ 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).[52]

The more important of these parameters (as listed in subsection I of this SRP section) are compared to those predicted for other similar plants to confirm that they are within the expected range. The percentage of fuel rods that experience failure is reviewed and ~~AEB~~PERB[53] is notified regarding the extent of fuel failures predicted by the analysis.

~~CPB is consulted regarding the acceptance criteria for fuel rod failure and core coolability.~~[54]

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.[55]

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and that the[56] review supports the following kinds of statements and conclusions which should be included in the staff's safety evaluation report (SER).

The staff concludes that the consequences of postulated rotor seizure or broken shaft events meet the requirements set forth in the General Design Criteria 17,[57] 27, 28, and 31 regarding the ability to insert control rods ~~insertability~~ and to cool the core ~~coolability~~,[58] and 10 CFR Part 100 guidelines regarding radiological dose at the site boundary~~, and applicable TMI Action Plan items~~.[59] This conclusion is based upon the following:

(a) The applicant has demonstrated that the resultant fuel damage was limited such that the ability to insert control rods ~~insertability~~[60] would be maintained, and that no loss of core cooling capability resulted. The minimum departure from nucleate boiling ratio (DNBR) or critical power ratio (CPR) experienced by any fuel rod was ____, resulting in ____% of the rods experiencing cladding perforation.

(b) The applicant met the requirements of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the postulated accident.

(c) The analyses and effects of pump rotor seizure and shaft breaks, during various modes of operation and with and without offsite power, have been reviewed.

(d) The accidents analyzed were evaluated using a mathematical model that has been previously reviewed and found acceptable by the staff.

(e) The parameters used as input to this model were reviewed and found to be suitably conservative.

(f) The radioactivity release has been evaluated using the computer code SARA and a conservative description of the plant response to the accident. A decontamination factor of _____ between the water and steam phases and a X/Q value of ____ sec/m^3 has been used in our evaluation of radiological consequences. The calculated doses are presented in Table ____. Technical specification limits on primary and secondary coolant activities will limit potential doses to a small fraction of the 10 CFR Part 100 exposure guidelines. The potential doses are within 10 CFR Part 100 exposure guidelines even if the accident should occur coincident with an iodine spike.

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.[61]

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.[62] Except in those cases in which the applicant proposed 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.[63]

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced NUREGs, except for the position in Subsection II.7 and in Subsection III regarding loss of offsite power and assumed single failures. This new position will be applied to new applications (for a Construction Permit, a manufacturing license, or design certification).[64]

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 17, "Electric Power Systems."[65]

~~1~~2. 10 CFR Part 50, Appendix A, General Design Criterion 27, "Combined Reactivity Control Systems Capability."

~~2~~3. 10 CFR Part 50, Appendix A, General Design Criterion ~~15~~28,[66] "Reactivity Limits."

4. 10 CFR Part 50, Appendix A, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."[67]

~~4~~5. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

~~5~~6. "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from the Director, NRR, to NRR Staff" (Issue No. 1), U.S. Nuclear Regulatory Commission, NUREG-0318, November 1976.

~~6~~7. "Staff Discussion of Twelve Additional Technical Issues Raised by Responses to November 3, 1976 Memorandum from the Director, NRR, to NRR Staff" (Issue No. 22), U.S. Nuclear Regulatory Commission, NUREG-0158, December 1976.

~~7~~8. F. M. Bordelon, "Calculation of Flow Coastdown after Loss of Reactor Coolant Pump," WCAP-7973, Westinghouse Electric Corporation, August 1970.

~~8~~9. C. D. Morgan, H. C. Cheatwood, and J. R. Glandermans, "RADAR - Reactor Thermal and Hydraulic Analysis During Reactor Flow Coastdown," BAW-10069, Babcock and Wilcox Company, July 1973.

~~9~~10. R. H. Stoudt and J. E. Busby, "CADD - Computer Applications to Direct Simulation of Transient Events on Water Reactors," BAW-10080 (nonproprietary) and BAW-10076 (proprietary), Babcock and Wilcox Company, July 1973.

1~~1~~~~0~~. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973.

12~~1~~. R. Linford, "Analytical Methods of Transient Evaluations in the General Electric Boiling Water Reactor," NEDP-10802, General Electric Company, April 1973.

13. General Electric Co., ODYNA - One Dimensional Dynamic Model (proprietary computer software for use in ABWR transient analysis to simulate pressurization events).[1] [68]

14. General Electric Co., REDYA (proprietary computer software for use in ABWR transient analysis to simulate other than pressurization events).[69]

15. CESEC-III (CENPD-107; LD-82-001). (Calculates system parameters such as core power, flow, pressure, temperature, and valve actions during a transient.)[70]

16. 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 the DNBR use the CE-1 critical heat flux correlation.)

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[1] 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.

17. 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.)

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

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

20 NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995.[71]

21~~0~~. Standard Review Plan Section 4.4, "Thermal and Hydraulic Safety."

22~~1~~. Standard Review Plan Section 4.2, "Fuel System Design."

23~~2~~. Standard Review Plan Section 7.2, "Reactor Trip System."

24~~3~~. Standard Review Plan Section 7.3, "Engineered Safety Features System."

25~~4~~. Standard Review Plan Section 7.4, "Systems Required for Safe Shutdown."

26~~5~~. Standard Review Plan Section 7.5, "Safety-Related Display Instrumentation."

~~17. 10 CFR Part 5, Appendix A, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."~~[72]

~~18. 10 CFR Part 50, Appendix A, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."~~[73]

27~~6~~. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License."

28~~7~~. NUREG-0737, "Clarification of TMI Action Plan Requirements."

~~21. 10 CFR Part 50, Appendix A, General Design Criterion 28, "Reactivity Limits."~~[74]

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SRP Draft Section 15.3.3
Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	Current PRB abbreviation	Changed RSB to SRXB (global change for this section).
2.	SRP-UDP format item	Added acronym (RCP) used in ACCEPTANCE CRITERIA.
3.	Editorial	Defined SRP.
4.	Editorial	Revised first part of sentence to correct logic related to the scope of the review for this section (consistent with SRP Section 15.2.8).
5.	Editorial	Deleted reference identification.
6.	Editorial	Rearranged sentence for clarity.
7.	Editorial	Defined SAR.
8.	SRP-UDP format item	Deleted sentence. Coordination is covered under "Review Interfaces."
9.	SRP-UDP format item	Reworded sentence and changed CPB to SRXB. The Core Performance Branch has been incorporated into the SRXB.
10.	SRP-UDP format item	Revised sentence. The Core Performance Branch has been incorporated into the SRXB. Hence, there is no CPB counterpart for the SRXB reviewer to contact.
11.	SRP-UDP format item	Revised sentence and moved forward from "Review Interfaces" due to the fact that CPB has been combined with SRXB, eliminating an interface.
12.	SRP-UDP format item	"Review Interfaces" added to AREAS OF REVIEW and presented in numbered paragraph form to describe how SRXB coordinates the review of reactor temperature/pressure transients with other NRR branches.
13.	SRP-UDP format item	Changed HICB to Instrumentation and Control Branch (ICSB).
14.	Editorial	Deleted reference identification.
15.	SRP-UDP format item	Moved sentence forward due to the fact that CPB has been combined with SRXB, eliminating the interface.
16.	SRP-UDP format item	Changed Accident Evaluation Branch (AEB) to Emergency Preparedness and Radiation Protection Branch (TERB).
17.	SRP-UDP format item	Changed AEB to TERB.

Item	Source	Description
18.	Editorial	SRP Section 15.3.3-15.3.4 contains Acceptance Criteria (GDC 31) and Evaluation Findings regarding the integrity of the reactor coolant pressure boundary (RCPB). GDC 31 establishes the general requirements for fracture toughness of the RCPB, which are implemented by 10 CFR 50.60 and associated Appendices G and H. Conformance with the requirements of 10 CFR 50.60, Appendix G and Appendix H is reviewed in SRP Section 5.2.3 for the RCPB (other than the reactor vessel) and SRP Section 5.3.1 for the reactor vessel.
19.	Editorial	Added a review interface with SRP Section 9.2.2. SRP Section 15.3.3-15.3.4 contains Acceptance Criteria regarding reactor coolant pump seal integrity similar to that associated with TMI Action Items II.K.2.16 and II.K.3.25 as cited in other Chapter 15 Sections (e.g., 15.1.5 and 15.2.8). Reactor coolant pump seal integrity issues and conformance with these TMI Action Items are reviewed in SRP Section 9.2.2.
20.	Editorial	Replaced "insertability" and "core coolability" with "ability to insert" and "to cool the core" for clarity and precision, as well as to correct usage and eliminate jargon.
21.	**Integrated Impact 1513**	Added GDC 17 as acceptance criterion II.A and renumbered subsequent criteria to accommodate this addition.
22.	Editorial	Added abbreviation for General Design Criterion 27.
23.	Editorial	Added abbreviation for General Design Criterion 28.
24.	Editorial	Changed "assure" to "ensure" (global change for this section).
25.	Editorial	Added abbreviation for General Design Criterion 31.
26.	Editorial	Spelled out GDC.
27.	Editorial	Defined DNBR.
28.	Editorial	Defined CPR.
29.	Editorial	Defined MCPR.
30.	Editorial	Replaced "activity" with "of radioactive material" for clarity and precision.

SRP Draft Section 15.3.3
Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
31.	Integrated Impact No. 936	TMI Action Item II.K.3.5 was resolved in Generic Letters 83-10A through 83-10F. The issue deals with the need for tripping reactor coolant pumps (RCPs) after certain small-break loss-of-coolant accidents (LOCAs). This SRP section addresses transients initiated by RCP rotor seizure or shaft break and does not involve a LOCA. Hence, consideration of the TMI action item II.K.3.5 was inappropriate for this section.
32.	Editorial	Renumbered acceptance criteria because Criterion 6 was deleted.
33.	**Integrated Impact 1513**	Added discussion of LOOP in combination with a single failure.
34.	Editorial	Revised sentence for clarity and consistency with item 18 above.
35.	**Integrated Impact 1355**	Added reference to two proprietary computer codes that were found acceptable by the NRC staff for use in analyzing transients for the ABWR.
36.	Integrated Impact No. 927	Added a sentence referring to References 14 through 18. These are ABB-CE topical reports approved by NRC for non-LOCA transient and accident analysis of CE80+ plants.
37.	Integrated Impact No. 928	Added a sentence referring to Reference 19 (NUREG-1465). A draft of this document was used by the applicant and NRC staff for analyzing the radiological consequences of an RCP rotor seizure event as reported in the FSER for the System 80+ standard design.
38.	SRP-UDP format item	"Technical Rationale" added to ACCEPTANCE CRITERIA and presented in paragraph form.
39.	SRP-UDP format item	Added lead-in sentence for "Technical Rationale."
40.	**Integrated Impact 1513**	Added Technical Rationale for GDC 17.
41.	SRP-UDP format item	Added technical rationale for GDC 27 and GDC 28.
42.	SRP-UDP format item	Added technical rationale for GDC 31.
43.	SRP-UDP format item	Added technical rationale for 10 CFR Part 100.
44.	SRP-UDP format item	Added a reference to combined license (COL) reviews.
45.	SRP-UDP format item	Added a reference to COL review stage.
46.	Editorial	Revised sentence to achieve parallel construction.
47.	Editorial	Revised sentence to achieve parallel construction.
48.	SRP-UDP format item	Changed HICB to ICSB (global change for this section).

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Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
49.	**Integrated Impact 1513**	Added discussion of LOOP and single failure.
50.	SRP-UDP format item	Reworded sentence. The Core Performance Branch has been incorporated into the SRXB.
51.	SRP-UDP format item	Reworded sentence. The Core Performance Branch has been incorporated into the SRXB.
52.	Editorial	Revised a very complex sentence for clarity and consistency with the revised wording in SRP Section 15.2.1-15.2.5.
53.	SRP-UDP format item	Changed AEB to TERB.
54.	SRP-UDP format item	Deleted sentence. The Core Performance Branch has been incorporated into the SRXB.
55.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
56.	Editorial	Deleted gender-specific reference.
57.	**Integrated Impact 1513**	Added GDC 17 to the list of GDCs met by this review.
58.	Editorial	Replaced "insertability" and "core coolability" with "ability to insert" and "to cool the core" for clarity and precision.
59.	Integrated Impact No. 936	Deleted reference to TMI Action Plan items because the only action item was II.K.3.5, which had been inappropriately listed as a specific acceptance criterion in subsection II.
60.	Editorial	Replaced "insertability" with "ability to insert" for clarity and precision.
61.	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.
62.	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.
63.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
64.	**Integrated Impact 1513**	Discussed implementation of the new staff position regarding loss of offsite power and single failures.
65.	**Integrated Impact 1513**	Added GDC 17 to list of references and renumbered remaining references accordingly.
66.	Editorial	Corrected referenced GDC number.

Item	Source	Description
67.	SRP-UDP format item	Renumbered original Reference 19 to place GDC 31 immediately after GDC 28.
68.	Integrated Impact 1355	Added reference to a proprietary computer code approved by the NRC for use in analyzing transients for the ABWR.
69.	Integrated Impact No. 1355	Added reference to a proprietary computer code approved by the NRC for use in analyzing transients for the ABWR.
70.	Integrated Impact Nos. 927 & 928	Added References 15 through 19. These are ABB-CE topical reports approved by NRC for non-LOCA transient and accident analysis of CE80+ plants.
71.	Integrated Impact No. 928	Added a reference to NUREG-1465.
72.	Editorial	Deleted reference because it is incorrectly identified (Part 5) and is duplicated as reference 19.
73.	Editorial	Renumbered as Reference 4 to list GDC 31 immediately after GDC 28.
74.	Editorial	Deleted Reference 22 because GDC 28 has been listed (with an incorrect number) as Reference 3.

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SRP Draft Section 15.3.3

Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
927	Consider updating the ACCEPTANCE CRITERIA and related references to include staff approved analytical methods used to analyze the RCP seizure event for the System 80+ FSER.	Subsection II, ACCEPTANCE CRITERIA, fourth paragraph Subsection VI, REFERENCES, References 14 through 18
928	Consider updating the ACCEPTANCE CRITERIA and related references to reflect staff guidance contained in NUREG-1465 regarding gap fractions of relevant isotopes (noble gases, iodine, cesium, and rubidium) and chemical species of iodine to be assumed within the gap for analysis of the radiological consequences of transients initiated RCP rotor seizure or shaft break.	Subsection II, ACCEPTANCE CRITERIA, fourth paragraph Subsection VI, REFERENCES, Reference 19
936	Consider updating acceptance criterion II.6 and related references to reflect staff guidance provided in Generic Letters 83-10A through 10F, 85-12, 86-05, and 86-06. DECISION: Acceptance criterion II.6 was inappropriate for this SRP section. The subject is addressed in DRAFT Revision SRP 15.6.5.	Subsection II, ACCEPTANCE CRITERIA, deleted criterion 6
1355	Consider revising the specific acceptance criteria in subsection II and REFERENCES to include the ODYNA and REDYA computer codes as being acceptable to the staff.	Subsection II, ACCEPTANCE CRITERIA, fourth paragraph Subsection VI, REFERENCES, References 12 and 13
1513	Add GDC 17 as an Acceptance Criteria and incorporate staff positions from ABB-CE 80+ into the Review Procedures.	Add Acceptance Criteria step A. and 7 addressing GDC 17. Modify Review Procedures Model Evaluation Findings to add GDC 17. Add GDC 17 to References.