U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.2.1 – 15.2.5 LOSS OF EXTERNAL LOAD; TURBINE TRIP; LOSS OF CONDENSER VACUUM; CLOSURE OF MAIN STEAM ISOLATION VALVE (BWR); AND STEAM PRESSURE REGULATOR FAILURE (CLOSED)

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

A number of ~~transients~~ initiating events[2] which are expected to occur with moderate frequency result in unplanned decreases in heat removal by the secondary system. Each ~~transient~~ event[3] covered in this Standard Review Plan (SRP)[4] section should be discussed in individual sections of the safety analysis report (SAR), as ~~required by the Standard Format (Ref. 1)~~ specified in Regulatory Guide 1.70.[5] The ~~transients~~ initiating events and associated transients[6] to be evaluated are:

1. Loss of External Load

In a loss of external load event, an electrical disturbance causes loss of a significant portion of the generator load. This loss of load situation is different from the loss of ac power condition considered in ~~Standard Review Plan (~~SRP~~)~~[7] Section 15.2.6 in that offsite ac power remains available to operate the station auxiliaries (such as reactor coolant pumps). The onsite emergency diesel generators are therefore not required for the loss of external load ~~transient~~ event. Immediate fast closure of the turbine control valves (TCVs) and intercept valves is

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

initiated whenever a loss of generator load takes place. For a boiling water reactor (BWR), a fast TCV closure (0.150–0.2 sec) causes a sudden reduction in steam flow and results in a reactor pressure surge. For a BWR without select rod insert (SRI), reactor scram occurs. For a pressurized water reactor (PWR) there is also a sudden reduction in steam flow, and this causes the pressure and temperature in the shell side of the steam generator to increase. The latter effect, in turn, results in an increase in reactor coolant temperature, a decrease in coolant density, an increase in water volume in the pressurizer, and an increase in reactor coolant pressure. For a PWR with an integrated control system, reactor power can be run back to a lower level on TCV closure.

In all light-water-cooled reactors, sensible and decay heat can be removed through actuation of one or several of the following systems: steam relief system, steam bypass to the condenser, reactor core isolation cooling system (BWR), emergency core cooling systems, and auxiliary feedwater system (PWR).

2. Turbine Trip

In a turbine trip event, a malfunction of a turbine or reactor system causes the turbine to be tripped off the line by abruptly stopping steam flow to the turbine. This is different from the loss of electrical load condition described above in that fast closure of the turbine stop valves (TSVs) is initiated. The TSVs have faster (0.1 sec) closure times than the turbine control valves, resulting in more severe transients. For typical BWR and PWR plants, position switches on the TSVs sense the trip and initiate reactor scram. The remainder of this transient is similar to the previously discussed loss of electrical load.

3. Loss of Condenser Vacuum

A loss of condenser vacuum event is one of the malfunctions that can cause a turbine trip. The remarks in item 2, above, thus apply to this ~~transient~~ event.

4. Main Steam Isolation Valve Closure

The main steam isolation valve (MSIV) ~~transient~~ closure event[8] for BWRs can be initiated by various steam line or reactor system malfunctions and by various operator actions. As the MSIVs close, position switches initiate a reactor scram when the valves in three or more of the steam lines are less than 90% open, the reactor pressure is above ~~600 psi~~ 4140 kPa (600 psi),[9] and the reactor mode switch is in the RUN position. The effect of MSIV closure is to limit steam flow to the turbine. The results are similar to those discussed in item 1, above, but tend to be less severe since the MSIV closure time is much longer than that of the TCVs.

5. Steam Pressure Regulator Failure

Steam pressure regulator failure in a closed position yields a transient similar to the transients discussed above. Generally, because the rate of change of system parameters is slower for a steam pressure regulator failure, a less severe transient results.

The review of the transients described above includes the sequence of events, the analytical models, the values of parameters used in the analytical models, and the predicted consequences of the transients.

The sequence of events described in the SAR analysis is reviewed by ~~RSB~~SRXB in consultation with the Instrumentation and Control ~~Systems~~ Branch (~~ICSB~~HICB).[10] The ~~RSB~~SRXB reviewer concentrates on the assumptions used 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 that all mathematical models and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method or code has not been previously reviewed, the ~~RSB~~SRXB reviewer requests initiation of a generic evaluation of the new analytical model or code by ~~RSB~~SRXB.~~or the Core Performance Branch (CPB), as appropriate.~~[11]

The predicted results of the transient analyses are then reviewed to ~~assure~~ensure[12] that the consequences meet the acceptance criteria given in subsection II, below. Further, the results of the analyses are reviewed to ascertain that the predicted values of pertinent system parameters are within expected ranges for the type and class of reactor under review.

SRXB reviews the values of all parameters used in the analytical models, including the initial conditions of the core and system. SRXB also reviews core physics, fuel design, and core thermal-hydraulics data used in the SAR analysis as part of its primary review responsibility for SRP Sections 4.2 through 4.4.[13]

Review Interfaces[14]

The ~~RSB~~SRXB will coordinate other branches' evaluations that interface with the overall review of the transient analysis, as follows:

1. The ~~ICSB~~HICB[15] review of SRP Sections 7.2 and 7.3 is consulted on the instrumentation and controls aspects of the sequences described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems.

2. ~~ICSB~~HICB also reviews potential bypass modes and the possibility of manual control by the operator as part of its primary review responsibility for SRP Sections 7.2 through 7.5. ~~The Core Performance Branch (CPB) reviews the values of all the parameters used in the analytical models, including the initial conditions of the core and system. CPB also reviews the core physics, fuel design, and core thermal-hydraulics data used in the SAR analysis as part of its primary review responsibility for SRP Sections 4.2 through 4.4.~~[16]

3. The review of the technical specifications is coordinated with and performed by the ~~Licensing Guidance Branch~~ Technical Specifications Branch (TSB)[17] as part of its primary review responsibility for SRP Section 16.0.

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

II. ACCEPTANCE CRITERIA

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

A. General Design Criterion 10 (GDC 10)[18] as it relates to the reactor coolant system being designed with appropriate margin to ~~assure~~ensure that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences.

B. General Design Criterion 15 (GDC 15)[19] 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.[20]

~~C~~D.[21] General Design Criterion 26 (GDC 26)[22] 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, is accounted for.

Specific criteria necessary to meet the relevant requirements of ~~GDC~~ General Design Criteria 10, 15, 17,[23] and 26 are as follows:

1. The basic objectives of the review of the ~~transients~~ initiating events listed in subsection I[24] are:

   a. To identify which of the moderate-frequency ~~transients~~ events that result in an unplanned decrease in secondary system heat removal is the most limiting. (The term "moderate frequency" is used in this SRP section in the same sense as in the definitions of design and plant process conditions in References ~~8 and 9~~ 10 and 11.)[25]

   b. To verify that, for the most limiting ~~transient~~ event, the predicted plant response is such that the specific criteria given below regarding fuel damage and system pressure are satisfied.

c. To verify that the plant protection systems setpoints assumed in the transients analyses are selected with adequate allowance for measurement inaccuracies as delineated in Regulatory Guide 1.105~~ (Reference 3)~~.[26]

2. The criteria for incidents of moderate frequency are:

   a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (Ref. 2).

   b. Fuel cladding integrity shall be maintained by ensuring that 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).

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

3. The applicant should analyze these ~~transients~~ events using an acceptable analytical model. The equations, sensitivity studies, and models described in References 4 through 7 are acceptable. (Refs. 8 and 9 describe acceptable transient analysis computer codes used for design analysis of the Advanced Boiling Water Reactor, or ABWR.)[30] 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 by ~~RSB~~SRXB.

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

   a. The reactor is initially at 102% of the rated (licensed) core thermal power (to account for a 2% power measurement uncertainty), and primary loop ~~of~~ flow is at the nominal design flow,[31] less the flow measurement uncertainty.

   b. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core for PWRs and a 0.8 multiplier on the predicted reactivity insertion rate for BWRs.

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.

Technical Rationale[32]

The technical rationale for application of these acceptance criteria to reviewing analyses of reactor temperature and pressure transients at a nuclear power plant is discussed in the following paragraphs:[33]

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 section because the reviewer evaluates the consequences of the events specified in Regulatory Guide 1.70. Such events could result in a decrease in heat removal by the secondary system, potentially causing the thermal design criteria for the fuel cladding to be exceeded. Regulatory Guide 1.105 provides guidance for ensuring that instrument setpoints are initially within and remain within technical specification limits.

   Meeting the requirements of GDC 10 provides assurance that specified acceptable fuel design limits are not exceeded for initiating events involving a decrease in heat removal by the secondary system.[34]

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

   GDC 15 is applicable to this section because the reviewer evaluates the consequences of events specified in Regulatory Guide 1.70. Such events could result in a decrease in heat removal by the secondary system, leading to an increase in the reactor coolant temperature and pressure.

   Meeting the requirements of GDC 15 provides assurance that the design conditions of the reactor coolant pressure boundary are not exceeded for initiating events involving a decrease in heat removal by the secondary system.[35]

3. 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 15.1.5 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 initiating events involving a decrease in heat removal by the secondary system, concurrent with a loss of offsite power (LOOP).[36]

4. Compliance with GDC 26 requires that two independent reactivity control systems be provided to control reactivity changes, thereby ensuring that acceptable fuel design limits are not exceeded.

GDC 26 is applicable to this section because the reviewer evaluates the consequences of events specified in Regulatory Guide 1.70. Such events could result in a decrease in heat removal by the secondary system, potentially leading to changes in reactivity within the core that could cause the thermal design criteria for the fuel cladding to be exceeded. GDC 26 requires that the thermal margin be sufficient to accommodate these conditions. Where applicable, the reviewer examines these margins to ensure that the thermal criteria are being satisfied.

Meeting the requirements of GDC 26 provides assurance that specified acceptable fuel design limits are not exceeded, ensuring an appropriate margin for malfunctions of the reactivity control system.[37]

III. REVIEW PROCEDURES

The procedures below are used during ~~both~~ the construction permit (CP)~~, and~~ operating license (OL), and combined license (COL)[38] 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[39] 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 these transients presented by the applicant in the SAR is reviewed by ~~RSB~~SRXB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

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

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

3. The extent to which credit is taken for the functioning of normally operating plant systems.[40]

4. The extent to which the operation of engineered safety systems ~~that~~ is required.[41]

5. The extent to which operator actions are required.

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

If the SAR states that any one of these transients is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. The applicant is to present a quantitative analysis in the SAR of the reduction-of-heat-removal transient that is determined to be most limiting. For this transient, the ~~RSB~~SRXB reviewer, in consultation with the ~~ICSB~~HICB reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to adequately limit the consequences of the transient to an acceptable level. The ~~RSB~~SRXB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The ~~ICSB~~HICB reviewer provides consultation on automatic initiation, actuation delays, possible bypass modes, interlocks, and the feasibility of manual operation if the SAR states that operator action is needed or expected.

To the extent deemed necessary, the ~~RSB~~SRXB reviewer evaluates the effect of single active failures of systems and components which may affect the course of the transient. For new applications, loss of offsite power (LOOP) should not be considered a single failure; each of the reduction-of-heat-removal transients 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.)[42] This phase of the review uses the system review procedures described in the standard review plans 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 line are reviewed by ~~RSB~~SRXB to determine if these models have been previously reviewed and found acceptable by the staff. If not, ~~RSB~~SRXB initiates a generic review of the model proposed by the applicant. ~~CPB is consulted regarding the specified acceptable fuel design limits (SAFDLs).~~[43]

The values of system parameters and initial core and system conditions used as input to the model are reviewed by ~~RSB~~SRXB. Of particular importance are (1) the values of reactivity coefficients and control rod worths used ~~by~~ in the applicant's analysis and (2) the variations of moderator temperature, void, and Doppler coefficients of reactivity with core life.[44] The reviewer evaluates the justification provided by the applicant to show that the core burnup selected yields the minimum safety[45] margins. ~~CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.~~[46]

The results of the analysis are reviewed and compared ~~to~~ with[47] the acceptance criteria presented in subsection II of this SRP section regarding fuel integrity and 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 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).[48]

The more important of these parameters for the limiting transient are compared ~~to~~ with[49] those predicted for other similar plants to verify that they are within expected range.

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[50]

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and that the 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 plant design is acceptable with regard to transients resulting in unplanned decreases in heat removal by the secondary system that are expected to occur with moderate frequency and that the predicted response meets the requirements of General Design Criteria 10, 15, 17,[51] and 26. This conclusion is based on the following:
>
> The applicant has met the requirements of General Design Criteria 10 and 26 with respect to demonstrating that specified acceptable fuel design limits are not exceeded for this event and has met the requirements of General Design Criterion 15 with respect to demonstrating that the reactor coolant pressure limits have not been exceeded by this event, and that resultant leakage will be within

acceptable limits by ~~assuring~~ ensuring that plant transients do not result in an unplanned decrease in heat removal by the secondary system. ~~Those that might be expected to occur with moderate frequency are turbine trip, loss of external load, steam pressure regulator malfunctions, main steam isolation valve closure (in BWRs), loss of condenser vacuum, loss of nonemergency ac power to the station auxiliaries, and loss of normal feedwater flow.[†] 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 had 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 and in accordance with the recommendation of Regulatory Guide 1.105. The results of the analysis of the ____________ transient showed that cladding integrity was maintained by ensuring that the maximum departure from nucleate boiling ratio (or minimum critical heat ratio for a BWR) did not decrease below ______, and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of their design pressures.~~[52]

The transient initiating events that might be expected to occur with moderate frequency are:

- turbine trip,
- loss of external load,
- steam pressure regulator malfunctions,
- main steam isolation valve closure (in BWRs),
- loss of condenser vacuum,
- loss of nonemergency ac power to the station auxiliaries, and
- loss of normal feedwater flow.[2]

In a review of the transients that could result from these postulated events, 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 had been previously reviewed and found to be acceptable by the staff. The parameters used as input to this model were

---

[1] The SER should present one statement for moderate frequency transients involving an unplanned decrease in heat removal by the secondary system. Thus, the results of the reviews under SRP Sections 15.2.6 and 15.2.7 are included in this statement.

[2] The SER should present one statement for moderate frequency transients involving an unplanned decrease in heat removal by the secondary system. Thus, the results of reviews conducted under SRP Sections 15.2.6 and 15.2.7 are included in this statement.

reviewed and found to be suitably conservative and in accordance with the recommendation of Regulatory Guide 1.105. The results of the analysis of the ____________ transient showed that cladding integrity was maintained by ensuring that the maximum departure from nucleate boiling ratio (or minimum critical heat ratio for a BWR) did not decrease below ______ and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of their design pressures.

Thus, the applicant has met the requirements of General Design Criteria 10, 17, and 26 with respect to demonstrating that specified acceptable fuel design limits are not exceeded for this event. In addition, the applicant has met the requirements of General Design Criterion 15, demonstrating that the reactor coolant pressure limits have not been exceeded by this event and that resultant leakage will be within acceptable limits.[53]

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.[54]

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.[55] 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.[56]

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

4. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," General Electric Reports NEDO-24154 and NEDE-24154P Volumes I, II, and III, October 1978.

5. "Loftran Code Description," Westinghouse Electric Corporation Report WCAP-7907, October 1972 (in review).

6. "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107, April 1974 (in review).

7. "TRAP2-Fortran Program for Digital Simulation of the Transient Behavior of the Once-Through Steam Generator and Associated Reactor Coolant Systems," Babcock & Wilcox, BAW-10128, August 1976 (in review).

8. General Electric Company, ODYNA - One Dimensional Dynamic Model (proprietary computer software for use in ABWR transient analysis to simulate pressurization events).[3] [57]

9. General Electric Company, REDYA - (proprietary computer software for use in ABWR transient analysis to simulate other than pressurization events).[58]

~~8~~10. ~~ANSI N18.2,~~ ANS 51.1, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (~~1974~~1988).[59]

~~9~~11. ~~ANS Trial Use Standard N212,~~ ANS 52.1, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (~~1974~~1988).[60]

~~10~~12. 10 CFR Part 50, General Design Criterion 10, "Reactor Design."

~~11~~13. 10 CFR Part 50, General Design Criterion 15, "Reactor Coolant System Design."

14. 10 CFR Part 50 Appendix A, General Design Criterion 17, "Electric Power Systems".[61]

~~12~~15. 10 CFR Part 50, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

---

[3] 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) supporting 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 ABWR standard design. These modified codes, ODYNA and REDYA, have been reviewed by the NRC staff and approved for design analysis of the ABWR.

SRP Draft Section 15.2.1

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 PRB abbreviation	Changed PRB to SRXB (global change for this section).
2.	Editorial	Changed "transients" to "initiating events" (global change for this section). The "transients" listed in this subsection are frequently initiating events that can lead to pressure and temperature transients.
3.	Editorial	Corrected "transient" to "event," making the SRP more consistent with the language in the advanced reactor FSERs (global change for this section).
4.	Editorial	Defined SRP.
5.	Editorial	Changed "Standard Format" to "Regulatory Guide 1.70," changed "required by" to "specified in," and eliminated the unnecessary reference.
6.	Editorial	Changed "transients" to "initiating events and associated transients."
7.	Editorial	SRP previously defined (item 4 above).
8.	Editorial	Added the word "closure" because this transient is only initiated by "valve closure."
9.	SRP-UDP format item	Converted 600 psi to 4140 kPa.
10.	SRP-UDP format item	Updated name of Instrumentation and Control Systems Branch to Instrumentation and Control Branch (HICB).
11.	New PRB responsibility	Deleted reference to Core Performance Branch because models and codes used to analyze reactor transients are now the sole responsibility of the SRXB.
12.	Editorial	Changed "assure" to "ensure" (global change for this section).
13.	SRP-UDP format item	Moved forward "Review Interface" because review responsibilities previously assigned to CPB are now the responsibility of the PRB (SRXB).
14.	SRP-UDP format item	Added "Review Interfaces" 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.
15.	New PRB abbreviation	Changed ICSB to HICB (global change for this section).

SRP Draft Section 15.2.1
Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
16.	SRP-UDP format item	Moved forward "Review interface" because review responsibilities previously assigned to CPB are now the responsibility of the PRB (SRXB).
17.	Current PRB responsibility	Changed Licensing Guidance Branch to Technical Specifications Branch.
18.	Editorial	Provided acronym for General Design Criterion 10.
19.	Editorial	Provided initials for General Design Criterion 15.
20.	Integrated Impact No. 415	Added GDC 17 to ACCEPTANCE CRITERIA.
21.	Editorial	Relettered paragraph because GDC 17 was added to the acceptance criteria.
22.	Editorial	Provided acronym for General Design Criterion 26.
23.	Integrated Impact No. 415	Added GDC 17 to ACCEPTANCE CRITERIA. Also changed "GDC" to "General Design Criteria" to accommodate plural usage.
24.	Editorial	Corrected an omission of the subsection number.
25.	Editorial	Renumbering of references made necessary by the addition of two new references (8 and 9).
26.	Editorial	Deleted unnecessary citation for Reference 3.
27.	Editorial	Defined DNBR.
28.	Editorial	Defined CPR.
29.	Editorial	Defined MCPR.
30.	Integrated Impact No. 1352	Added reference to proprietary computer codes approved by the NRC for use in analyzing transients for the ABWR.
31.	Editorial	Deleted "of" and added comma to clarify sentence.
32.	SRP-UDP format item	Added "Technical Rationale" to ACCEPTANCE CRITERIA and presented in paragraph form.
33.	SRP-UDP format item	Added lead-in sentence for "Technical Rationale."
34.	SRP-UDP format item	Added technical rationale for GDC 10.
35.	SRP-UDP format item	Added technical rationale for GDC 15.
36.	Integrated Impact No. 415	Added technical rationale for GDC 17.
37.	SRP-UDP format item	Added technical rationale for GDC 26.
38.	SRP-UDP format item	Added a reference to combined license (COL) reviews.
39.	SRP-UDP format item	Added a reference to COL review stage.

Item	Source	Description
40.	Editorial	Revised sentence to achieve parallel construction.
41.	Editorial	Revised sentence to achieve parallel construction.
42.	Integrated Impact No. 415	Added information to address the GDC 17 requirement for assuming loss of offsite power.
43.	Editorial	Deleted sentence because CPB was combined with RSB to form SRXB (the PRB).
44.	Editorial	Broke up a complex sentence to improve clarity.
45.	Editorial	Added "safety" to clarify the intent of the sentence.
46.	Editorial	Deleted sentence because CPB was combined with RSB to form SRXB (the PRB).
47.	Editorial	Changed "compared to" to "compared with" to accommodate scientific usage.
48.	Editorial	Revised a very complex sentence to improve clarity.
49.	Editorial	Changed "compared to" to "compared with" to accommodate scientific usage.
50.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
51.	Integrated Impact No. 415	Added GDC 17 to acceptance criteria.
52.	Editorial	The last part of the first sentence of this paragraph, "by assuring that plant transients do not result in an unplanned decrease in heat removal by the secondary system," contradicts the first sentence of the previous paragraph. The paragraph was reorganized, and some minor wording changes were made to improve clarity.
53.	Editorial	The second paragraph of the model statement has been reorganized. Minor wording changes were made, and the contradictory statement was deleted.
54.	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.
55.	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.
56.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.

Item	Source	Description
57.	Integrated Impact No. 1352	Added reference to a proprietary computer code approved by the NRC for use in analyzing transients for the ABWR.
58.	Integrated Impact No. 1352	Added reference to a proprietary computer code approved by the NRC for use in analyzing transients for the ABWR.
59.	Integrated Impact No. 416	Revised reference (ANSI N18.2) to the current version (ANS 51.1).
60.	Integrated Impact No. 416	Revised reference (ANS Trial Use Standard N212-74) to the current version (ANS 52.1).
61.	Integrated Impact No. 415	Added GDC 17 to REFERENCES.

SRP Draft Section 15.2.1
Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
415	Modify Acceptance Criteria, Review Procedures, and Evaluation Findings to include General Design Criterion (GDC) 17, "Electric Power Systems."	Subsection II, ACCEPTANCE CRITERIA, first paragraph, criterion C second paragraph, lead sentence "Technical Rationale," item 3 Subsection III, REVIEW PROCEDURES, fourth paragraph. Subsection VI, REFERENCES, Reference 14
416	Update References 8 and 9. ANSI N18.2-74 and ANS Trial Use Standard N212-74 have been superseded by ANS 51.1 and ANS 52.1, respectively; the current versions, dated 1983, were reaffirmed in 1988.	Subsection VI, REFERENCES, References 10 and 11
1352	Add to ACCEPTANCE CRITERIA and REFERENCES the GE proprietary computer codes ODYNA AND REDYA, have been approved by NRR for use in analyzing transients for the ABWR.	Subsection II, ACCEPTANCE CRITERIA, item 3, first paragraph Subsection VI, REFERENCES, References 8 and 9