NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.6.1 INADVERTENT OPENING OF A PWR PRESSURIZER PRESSURE RELIEF VALVE OR A BWR PRESSURE RELIEF VALVE

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

The inadvertent opening of a pressure relief valve results in a reactor coolant inventory decrease and a decrease in reactor coolant system pressure. A pressure relief valve, as defined in ANSI B95.1-1972 ~~(Ref. 1)~~,[2] is a pressure relief device which is designed to reclose and prevent further fluid flow after normal conditions have been restored. The effect of the pressure decrease is to decrease the neutron flux (via moderator density feedback). In a pressurized water reactor (PWR), a reactor trip occurs due to low reactor coolant system (RCS) pressure. In a boiling water reactor (BWR), the pressure relief valve discharges into the suppression pool. Normally there is no reactor trip in a BWR. The pressure regulator senses the RCS pressure decrease and partially closes the turbine control valves (TCVs) to stabilize the reactor at a lower pressure. The reactor power settles out at nearly the initial power level. The coolant inventory is maintained by the feedwater control system using water from the condensate storage tank via the condenser hotwell.

The review of these transients should consider the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transient.

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by ~~RSB~~SRXB.[3] The ~~RSB~~SRXB[4] reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition.

The analytical methods are reviewed by ~~RSB~~SRXB[5] to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced method has not been previously reviewed, the reviewer initiates a generic evaluation of the new analytical model. The values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed. The predicted results of the transient are reviewed to ~~assure~~ ensure[6] that the consequences meet the acceptance criteria given in subsection II of this Standard Review Plan (SRP)[7] section. The analysis results are reviewed to ascertain that pertinent system parameter ~~valves~~ values[8] are within ranges expected for the type and class of reactor under review.

The ~~Core Performance Branch (CPB) upon request from RSB,~~SRXB[9] will verify that the core physics data used by the applicant, or by the staff in independent analyses, ~~is the~~ are appropriate ~~data to be used~~ as part of its primary review responsibility for SRP Sections 4.2 through 4.4. The ~~CPB~~SRXB[10] will also verify~~, if requested,~~[11] that acceptance ~~criteria~~ criterion[12] 1 of SRP Section 4.4 is satisfied throughout the transient.

<u>Review Interfaces</u>[13]

The ~~RSB~~SRXB[14] will coordinate other branch evaluations that interface with the overall review of this SRP section, as follows:

1.[15] The Instrumentation and Controls ~~Systems~~ Branch ~~(ICSB)~~(HICB)[16] reviews the instrumentation and controls aspects of the sequence described to confirm that the 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. Upon request, ~~ICSB~~HICB[17] will verify the sequence described by the applicant for the safety analysis with regard to automatic actuation, remote sensing, indication, control, interlocks with auxiliary or shared systems, potential bypass modes, and the possibility of manual control by the operator.

2. The ~~Power Systems Branch (PSB)~~Electrical Engineering Branch (EELB),[18] upon request from ~~RSB~~SRXB,[19] will verify that the control systems power sources needed to function to mitigate the event are available as required by the applicant's description of the event.

3. The ~~Equipment Qualification Branch (EQB)~~Plant Systems Branch (SPLB),[20] upon the request of ~~RSB~~SRXB,[21] will verify that the equipment necessary to mitigate the event is qualified for the transient and post-transient environments. The ~~EQB~~SPLB[22] will also identify, if requested, equipment whose failure as a result of the initiating event could adversely affect the consequences. The SPLB also reviews the integrity of the reactor coolant pump seals as part of its primary review responsibility for SRP Section 9.2.2.[23]

4. The Human Factors Assessment Branch (HHFB) evaluates plant operating procedures as part of its primary review responsibility for SRP Section 13.5.2. On request from SRXB, HHFB verifies that plant operating procedures include appropriate actions relative to a reactor coolant pump trip after the inadvertent opening of a PWR pressurizer relief valve as described in Generic Letters 85-12, 86-05, and 86-06.[24]

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

A. General Design Criterion ~~(GDC)~~ 10 ~~(Reference 2)~~ (GDC 10),[26] as it relates to the reactor coolant system being designed with appropriate margin to ~~assure~~ensure[27] that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences.

B. General Design Criterion 15 (GDC 15) ~~(Reference 3)~~,[28] as it relates to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to ~~assure~~ensure[29] that the pressure boundary will not be breached during normal operations, including anticipated operational occurrences.

C. General Design Criterion 26 (GDC 26) ~~(Reference 4)~~,[30] as it relates to the reactivity control system to provide adequate control of reactivity changes during manual operations and anticipated transients to ~~assure~~ensure[31] that the acceptable fuel design limits are not exceeded.

D. TMI Action Plan requirements items ~~No's.~~[32] II.K.2.8, II.K.3.1, II.K.3.5, II.K.3.16 (10 CFR 50.34(f)(1)(vi)), II.K.3.25 (10 CFR 50.34(f)(1)(iii)),[33] and II.K.3.40 of NUREGs-0718 and -0737 ~~(Refs. 11 and 12)~~.[34]

E. Acceptance criteria for implementation of TMI Action Plan item II.K.3.5 relative to a reactor coolant pump trip after the inadvertent opening of a PWR pressurizer pressure relief valve (a small-break loss-of-coolant accident) are contained in Generic Letters 85-12, 86-05, and 86-06.[35]

The general objective in the review of inadvertent primary pressure relief valve opening events is to confirm that the following criteria are met:

1. The consequences of the transient are less severe than the consequences of another transient that results in a decrease of reactor coolant inventory and has the same anticipated frequency classification.

2. The plant responds to the pressure relief valve opening transient in such a way that the criteria regarding fuel damage and system pressure are met.

The specific criteria necessary to meet the requirements of ~~GDC~~ General Design Criteria[36] 10, 15, and 26 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. 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)[37] safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).

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

d. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, fuel 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.

e. To meet the requirements of ~~GDC~~ General Design Criteria 10, 15, and 26, the positions of Regulatory Guide 1.105 ~~(Ref. 9)~~,[38] "Instrument ~~Spans and~~ Setpoints for Safety-Related Systems,"[39] are used with regard to their impact on the plant response to the type of transient addressed in this SRP section.

f. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of Regulatory Guide 1.53 ~~(Ref. 10)~~.[40]

The applicant's analysis of this transient should be performed using an acceptable analytical model. If the applicant proposes to use other analytical methods, which have not been previously reviewed and approved by the staff, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer initiates an evaluation of the new analytical model.

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[*] The term "moderate frequency" is used in this SRP section in the same sense as in the description of design and plant process conditions in References 7 and 8.

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

a. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating, plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.

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

c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

d. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with Regulatory Guide 1.105. Compliance with Regulatory Guide 1.105 is determined by ~~ICSB~~HICB.[41]

Technical Rationale

The technical rationale for application of acceptance criteria for the inadvertent opening of a PWR pressurizer pressure relief valve or a BWR pressure relief valve is discussed in the following paragraphs:[42]

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 an inadvertent opening of a PWR pressurizer pressure relief valve or a BWR pressure relief valve. These are anticipated operational occurrences with the potential to exceed allowable thermal design criteria for fuel cladding integrity. Regulatory Guide 1.53 provides guidance for applying the single failure criterion to the design and analysis of nuclear power plant protection systems. Regulatory Guide 1.105 provides guidance for ensuring that instrument setpoints are initially within and remain within the technical specification limits.

   Meeting the requirements of GDC 10 provides assurance that specified acceptable fuel design limits are not exceeded for the inadvertent opening of a PWR pressurizer pressure relief valve or a BWR pressure relief valve.[43]

2. Compliance with GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to ensure

that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

GDC 15 is applicable to this section because the reviewer evaluates the consequences of the inadvertent opening of a PWR pressurizer pressure relief valve or a BWR pressure relief valve (i.e., anticipated operational occurrences). Further, the reactor coolant pressure needs to be analyzed to ensure that the pressure acceptance criterion is satisfied.

Meeting the requirements of GDC 15 provides assurance that the design conditions of the reactor coolant pressure boundary are not exceeded for the inadvertent opening of a PWR pressurizer pressure relief valve or a BWR pressure relief valve.[44]

3. Compliance with GDC 26 requires that 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 transient analyzed by the reviewer may involve the movement of control rods in response to the transient. In such instances, 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 15.6.1 examines this margin to ensure that thermal criteria are satisfied.

Meeting the requirements of GDC 26 provides assurance that appropriate margins are included to accommodate malfunctions of the reactivity control system, including stuck rods, thereby minimizing the possibility that specified acceptable fuel design limits are not exceeded.[45]

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) 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 review, 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 description of the inadvertent pressure relief valve opening transient is reviewed by ~~RSB~~SRXB[46] regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

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

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

3. The credit taken for the functioning of normally operating plant systems.

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

5. The extent to which operator actions are required.

6. The following TMI Action Plan (Refs. 11 and 12) items are reviewed to ~~assume~~ensure[48] compliance with the acceptance criteria:

   a. II.K.2.8. For Babcock and Wilcox designs, the reviewer evaluates the auxiliary feedwater system upgrade and automatic auxiliary feedwater initiation as they relate to determining the auxiliary feedwater performance requirements for this event, if the applicant's evaluation of this transient indicate that the system will be required to function.

   b. II.K.3.1. If, as a result of the evaluations performed as required by II.K.3.2, or if the applicant has in the design an automatic power-operated relief valve isolation system, the reviewer confirms it has been properly accounted for in the analyses.

   c. II.K.3.5. The reviewer evaluates the assumption made regarding reactor coolant pump trip to ~~assure~~ ensure[49] that they are consistent and conservatively modeled with respect to the final pump trip criteria ~~which results from resolution of Task Action Plan item II.K.3.5~~ that are contained in Generic Letters 85-12, 86-05, and 86-06 (PWRs only).[50]

   d. II.K.3.16 (10 CFR 50.34(f)(1)(vi)).[51] For boiling water reactor designs, the reviewer confirms that the results of the applicant's feasibility study, and, if required, system modifications to reduce the number of challenges to and the number of failures of relief valves, have been properly included in the evaluation of the event.

   e. II.K.3.25 (10 CFR 50.34(f)(1)(iii))[52] and II.K.3.40. If, as a result of the transient, or as a result of loss of AC power, containment isolation is indicated to occur, the reactor coolant pump component cooling water may be lost. The reviewer evaluates the applicant's submittal to determine that the reactor coolant pump seal integrity is not lost. If it cannot be established that seal integrity is ~~assured~~ ensured,[53] the reviewer ~~assures~~ verifies[54] that the evaluation of this event correctly accounts for seal failure.

If the SAR states that the inadvertent pressure relief valve opening transient is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. If a quantitative analysis of the transient is presented in the SAR, the ~~RSB~~SRXB[55] reviewer, with the aid of the ~~ICSB~~HICB[56] reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequence of the transient to acceptable levels. The ~~RSB~~SRXB[57] reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints.

To the extent deemed necessary, the ~~RSB~~SRXB[58] reviewer evaluates the effects of single active failures of systems and components which may alter the course of the transient. In this phase of the review the system reviews are performed as described in the SRP sections for Chapters 5, 6, 7, and 8 of the SAR. The reviewer considers possible single failures in systems that replenish or maintain the reactor coolant inventory.

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[59] to determine if these models have been previously reviewed and found acceptable by the staff. If not, the reviewer initiates 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 ~~RSB~~SRXB.[60] Of particular importance are the reactivity coefficients and control rod worths used ~~by~~ in the applicant's ~~in his~~[61] 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~~ the selected ~~the~~ core burnup ~~that~~[62] yields the minimum margins is evaluated.

The results of the 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 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), steamline 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 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);
- steamline pressure;
- containment pressure;
- pressure relief valve flow rate; and
- flow rate from the reactor coolant system to the containment system (if applicable).[63]

Values of the more important of these parameters for the transient caused by the inadvertent pressure relief valve opening are compared ~~to~~ with[64] those predicted for other similar plants to confirm that they are within the expected range.

Upon request from the ~~RSB~~SRXB[65] reviewer, other branches will provide input for the areas of review stated in subsection I of this SRP section. The ~~RSB~~SRXB[66] reviewer obtains and uses the input requested as required to ~~assure~~ ensure[67] that the review procedure is complete.

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.[68]

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):

> A number of plant transients can result in a decrease of reactor coolant inventory. Those that might be expected to occur with moderate frequency are pressure relief valve openings, minor primary pipe breaks, and (in BWRs) loss of feedwater.* All of 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. The results of the analysis of the _____ transient showed that the specified acceptable fuel design limits were maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR)** did not decrease below ___ and that the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.
>
> The staff concludes that the plant design in regard to transients that are expected to occur with moderate frequency and result in a decreased primary coolant inventory is acceptable and meets the relevant requirements of General Design Criteria ~~(GDC)~~ 10, 15, and 26 and the applicable TMI Action Plan items. This conclusion is based on the following:
>
> 1. The applicant has met the requirements of ~~GDC~~ General Design Criteria 10 and 26 with respect to demonstrating that resultant fuel ~~damage~~integrity[69] is maintained since the specified acceptable fuel design limits were not exceeded for the event.

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*The SER draft should present one statement for all similar transients.

**Minimum critical power ratio (MCPR) for a BWR.

2. The applicant has met the requirements of ~~GDC~~ General Design Criterion 15 with respect to demonstrating that the reactor coolant pressure boundary limits have not been exceeded by this event and that resultant leakage will be within acceptable limits. This requirement has been met since the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressure.

3. The applicant has met the requirements of ~~GDC~~ General Design Criterion 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 specified acceptable fuel design limits were not exceeded.

4. In meeting ~~GDC~~ General Design Criteria 10, 15, and 26, the staff has determined that the analysis was performed 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. In addition, we have further determined that the positions of Regulatory Guide 1.53 with respect to single failure criterion and Regulatory Guide 1.105 for instrument~~s~~ setpoints[70] have also been satisfied.

5. The applicant has met Task Action Plan item [identify item number] by [describe means used by the applicant to implement the action plan item].

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.[71]

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.[72] 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.[73]

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides and NUREGs.

VI. REFERENCES

1. ANSI B95.1-1972, "Terminology for Pressure Relief Devices," American Society of Mechanical Engineers.

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

3. 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant Pressure Boundary."

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

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

6. Standard Review Plan Section 4.2, "Fuel System Design."

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

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

9. Regulatory Guide 1.105, "Instrument ~~Spans and~~ Setpoints for Safety-Related Systems."[74]

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

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

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

13. NRC Generic Letter 85-12, "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps,' for Westinghouse-Designed Nuclear Steam Supply Systems."[75]

14. NRC Generic Letter 86-05, "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps,' for Babcock and Wilcox-Designed Nuclear Steam Supply Systems."[76]

15. NRC Generic Letter 86-06, "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps,' for Combustion Engineering-Designed Nuclear Steam Supply Systems."[77]

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SRP Draft Section 15.6.1
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 modification	Deleted "(Ref. 1)" in accordance with standard practice to delete unnecessary callouts.
3.	Current primary review branch designation	Changed PRB to SRXB.
4.	Current primary review branch designation	Changed PRB to SRXB.
5.	Current primary review branch designation	Changed PRB to SRXB.
6.	Editorial modification	Changed "assure" to "ensure" to correct usage.
7.	Editorial modification	Defined "SRP" as "Standard Review Plan."
8.	Editorial modification	Deleted "valves" and substituted "values" to correct an apparent typographical error.
9.	Current primary review branch designation/ Editorial modification	The review responsibilities of the old CPB are now assigned to SRXB. Deleted the phrase "as requested" as no longer appropriate.
10.	Current primary review branch designation.	The review responsibilities of the old CPB are now assigned to SRXB.
11.	Editorial modification	Clarified wordy sentence containing a noun/verb disagreement.
12.	Editorial modification	Changed "criteria" to "criterion" to accommodate singular usage.
13.	SRP-UDP format item	Added "Review Interfaces" to AREAS OF REVIEW.
14.	Current primary review branch designation	Changed PRB to SRXB.
15.	SRP-UDP format item	Divided existing paragraph into subparagraphs 1 through 5 under "Review Interfaces." The existing text and order were preserved; branch names and designations have been updated where indicated. (Paragraph 5 is new; see note 23 below).
16.	Current review branch name and designation	Changed review interface branch to Instrumentation and Controls Branch (HICB).
17.	Current review branch designation	Changed review interface branch to HICB.
18.	Current review branch name and designation	Changed review interface branch to Electrical Engineering Branch (EELB).

Item	Source	Description
19.	Current primary review branch designation	Changed PRB to SRXB.
20.	Current review branch name and designation	Changed review interface branch to Plant Systems Branch (SPLB).
21.	Current primary review branch designation	Changed PRB to SRXB.
22.	Current review branch designation	Changed review interface branch to SPLB.
23.	Editorial	Added a review interface with SRP Section 9.2.2. SRP Section 15.6.1 contains Acceptance Criteria, Review Procedures and Evaluation Findings with regard to reactor coolant pump seal integrity issues associated with TMI Action Items II.K.3.25 and II.K.3.40 (superseded by II.K.2.16). Conformance with these TMI Action Items are reviewed in SRP Section 9.2.2.
24.	Integrated Impact No. 419	Added a new review interface branch, HHFB, to review plant operating procedures for RCP trip after the inadvertent opening of a PWR pressurizer pressure relief valve, as requested by SRXB.
25.	Current primary review branch designation	Changed PRB to SRXB.
26.	SRP-UDP format item	Deleted "(Reference 2)" in accordance with standard practice to delete unnecessary callouts. Introduced "GDC 10" as an initialism for "General Design Criterion 10."
27.	Editorial modification	Changed "assure" to "ensure" to correct usage.
28.	SRP-UDP format item	Deleted "(Reference 3)" in accordance with standard practice to delete unnecessary callouts. Defined "GDC 15" as "General Design Criterion 15."
29.	Editorial modification	Changed "assure" to "ensure" to correct usage.
30.	SRP-UDP format item	Deleted "(Reference 4)" in accordance with standard practice to delete unnecessary callouts. Defined "GDC 26" as "General Design Criterion 26."
31.	Editorial modification	Changed "assure" to "ensure" to correct usage.
32.	Editorial modification	Deleted "No's." as incorrect and unnecessary.
33.	SRP-UDP format item	Added references to 10 CFR 50.34(f) for two TMI Action Plan items.
34.	SRP-UDP format item	Deleted "(Refs. 11 and 12)" in accordance with standard practice to delete unnecessary callouts.
35.	Integrated Impact No. 419	Added a reference to Generic Letters 85-12, 86-05, and 86-06 to ACCEPTANCE CRITERIA.

Item	Source	Description
36.	Editorial modification	Changed "GDC" to "General Design Criteria" to accommodate plural usage (global change for this section).
37.	Editorial modification	Defined "DNBR," "CPR," and "MCPR" as "departure from nucleate boiling ratio," "critical power ratio," and "minimum critical power ratio," respectively.
38.	SRP-UDP format item	Deleted "(Ref. 9)" in accordance with standard practice to delete unnecessary callouts.
39.	Editorial modification	Corrected the title of Regulatory Guide 1.105.
40.	SRP-UDP format item	Deleted "(Ref. 10)" in accordance with standard practice to delete unnecessary callouts.
41.	Current review branch designation	Changed review interface branch to HICB.
42.	SRP-UDP format item	Added "Technical Rationale" and lead-in paragraph to ACCEPTANCE CRITERIA.
43.	SRP-UDP format item	Added technical rationale for GDC 10.
44.	SRP-UDP format item	Added technical rationale for GDC 15.
45.	SRP-UDP format item	Added technical rationale for GDC 26.
46.	Current primary review branch designation	Changed PRB to SRXB.
47.	Editorial modification	Revised sentence to provide parallel construction and to improve clarity.
48.	Editorial modification	Deleted "assume" and substituted "ensure" to correct an apparent typographical error.
49.	Editorial modification	Changed "assure" to "ensure" to correct usage.
50.	Integrated Impact No. 419	Added reference to Generic Letters 85-12, 86-05, and 86-06 to REVIEW PROCEDURES.
51.	SRP-UDP format item	Added a reference to 10 CFR 50.34(f) for this TMI Action Plan item.
52.	SRP-UDP format item	Added a reference to 10 CFR 50.34(f) for this TMI Action Plan item.
53.	Editorial modification	Changed "assured" to "ensured" to correct usage.
54.	Editorial modification	Changed "assures" to "verifies" to correct usage.
55.	Current primary review branch designation	Changed PRB to SRXB.
56.	Current review branch designation	Changed review interface branch to HICB.
57.	Current primary review branch designation	Changed PRB to SRXB.

Item	Source	Description
58.	Current primary review branch designation	Changed PRB to SRXB.
59.	Current primary review branch designation	Changed PRB to SRXB.
60.	Current primary review branch designation	Changed PRB to SRXB.
61.	SRP-UDP format item	Modified sentence to eliminate gender-specific pronoun.
62.	SRP-UDP format item	Modified sentence to eliminate gender-specific pronoun.
63.	Editorial modification	Reformatted a complex sentence into a list to improve clarity.
64.	Editorial modification	Changed "compared to" to "compared with" to correct usage.
65.	Current primary review branch designation	Changed PRB to SRXB.
66.	Current primary review branch designation	Changed PRB to SRXB.
67.	Editorial modification	Changed "assure" to "ensure" to correct usage.
68.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
69.	Editorial modification	Deleted "damage" and substituted "integrity."
70.	Editorial modification	The subject of Regulatory Guide 1.105 is instrument setpoints.
71.	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.
72.	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.
73.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
74.	Editorial modification	Corrected the title of Regulatory Guide 1.105.
75.	Integrated Impact No. 419	Added Generic Letter 85-12 as Reference 13.
76.	Integrated Impact No. 419	Added Generic Letter 86-05 as Reference 14.

Item	Source	Description
77.	Integrated Impact No. 419	Added Generic Letter 86-06 as Reference 15.

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Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
418	Consider removing references to ANSI B95.1. In addition, consider comparing the current and cited versions of ANSI N18.2 and ANSI Trial Use Standard N212 to allow SRP reviewers to use the more current versions of the standards.	No changes made
419	Revise Review Procedures to address staff positions related to evaluations of reactor coolant pump (RCP) operation for an inadvertent opening of a PWR pressurizer pressure relief valve (small-break loss-of-coolant accident).	I., AREAS OF REVIEW, paragraph 5 II., ACCEPTANCE CRITERIA, paragraph E III., REVIEW PROCEDURES, paragraph 6.c VI., REFERENCES, References 13, 14, & 15