NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

4.4 THERMAL AND HYDRAULIC DESIGN

REVIEW RESPONSIBILITIES

Primary - ~~Core Performance~~Reactor Systems Branch (~~CPB~~SRXB)[1]

Secondary - ~~Instrumentation and Control System Branch (ICSB)~~
~~Human Factors Engineering Branch (HFEB)~~
~~Procedures and Test Review Branch (PTRB)~~None[2]

I. AREAS OF REVIEW

The objectives of the review are to confirm that the thermal and hydraulic design of the core and the reactor coolant system (RCS) has been accomplished using acceptable analytical methods; is equivalent to or is a justified extrapolation from proven designs; provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and anticipated operational ~~transients~~ occurrences[3]; and is not susceptible to thermal-hydraulic instability. This SRP section describes the normal review of thermal and hydraulic design, i.e., that for a plant similar in core and primary coolant system design to previously reviewed plants. The review of new prototype plants, new Critical Heat Flux (CHF) or Critical Power Ratio (CPR)[4] correlations, and new analysis methods require that additional independent audit analyses be performed. The required analyses may be in the following form:

1. Independent computer calculations to substantiate reactor vendor analyses.

2. Reduction and correlations of experimental data to verify processes or phenomena which are applied to reactor design.

3. Independent comparisons and correlations are made of data from experimental programs. These reviews also include analyses of experimental techniques, test repeatability, and data reduction methods.

The review includes evaluation of the proposed technical specifications regarding safety limits and limiting safety system settings, to ascertain that these are consistent with the power-flow operating map for boiling water reactor (BWR) plants or the temperature-power operating map for pressurized water reactor (PWR) plants.

For new plant applicants, the review determines the acceptability of analyses and procedures related to thermal-hydraulic conditions under shutdown and low-power operations.[5]

The review also includes determination of the largest hydraulic loads on core and reactor coolant system components during normal operation and ~~postulated~~ design-basis[6] accident conditions. This information is used in the review of fuel holddown requirements.

The review also determines the acceptability of analyses and emergency procedures related to core thermal-hydraulic stability under conditions representative of anticipated transients without scram (ATWS) events.[7]

To accomplish the objectives, the reviewer examines features of core and RCS components, key process variables for the coolant system, calculated parameters characterizing thermal performance, data serving to support new correlations or changes in accepted correlations, and assumptions in the equations and solution techniques used in the analyses. The reviewer determines that the applicant has used approved analysis methods in the manner specified by topical reports describing the methods and by staff reports approving the methods. The analysis methods addressed are to include core thermal-hydraulic calculations to establish local coolant conditions, departure from nucleate boiling or boiling transition calculations, and thermal-hydraulic stability evaluation. If an applicant has used previously unapproved correlations or analysis methods, the reviewer initiates an evaluation, either generic or plant specific. Any changes to accepted codes, correlations, and analytical procedures, or the addition of new ones must be reviewed to determine that they are justified on theoretical or empirical grounds.

~~A secondary review is performed by ICSB, HFEB, and PTRB. ICSB~~SRXB[8] will review the functional performance and requirements for the Inadequate Core Cooling (ICC) monitoring system hardware. ~~Emergency procedure guidelines and the information display will be reviewed by PTRB and HFEB, respectively. The results of these reviews will be used by CPB to complete the overall evaluation of the thermal-hydraulic review and will be incorporated into the Safety Evaluation Report (SER).~~[9]

~~The review of power distribution assumption made for the core thermal and hydraulic analysis is coordinated with the review for core physics calculations, as described in the Standard Review Plan (SRP) Section 4.3, for consistency. The reviewer verifies that core monitoring techniques which rely on in-core or ex-core neutron sensor inputs are reviewed.~~[10]

Review Interfaces:[11]

SRXB also performs the following review under the SRP section indicated:[12]

The review of power distribution assumptions[13] made for the core thermal and hydraulic analysis is coordinated with the review for core physics calculations, as described in the Standard Review Plan (SRP) Section 4.3, for consistency. The reviewer verifies that core monitoring techniques which rely on in-core or ex-core neutron sensor inputs are reviewed.[14]

SRXB will coordinate other branches' evaluations that interface with the overall review of the system as follows:[15]

1. The Mechanical Engineering Branch (EMEB) reviews the adequacy of components and structures under accident loads, and the preoperational vibration test program as part of its responsibility for SRP Section 3.9.2.[16]

2. The Instrumentation and Controls Branch (HICB) will review the core protection and reactor protection hardware for compliance with requirements applicable to reactor trip systems as part of its primary review responsibility for SRP Section 7.2.[17] HICB will also review the Inadequate Core Cooling (ICC) monitoring system hardware for compliance with requirements applicable to information systems important to safety as part of its primary review responsibility for SRP Section 7.5.[18]

3. The Human Factors Assessment Branch (HHFB) will review the applicant's training for use of the loose parts monitor in accordance with its review responsibility for SRP Section 13.2.1.[19] HHFB will also review emergency procedure guidelines (EPGs) and associated programs for development of plant-specific emergency operating procedures (EOPs), including those for recognition of and response to inadequate core cooling conditions, as part of its primary review responsibility for SRP Section 13.5.2. The HHFB will also review human factors aspects of information displays[20] as part of its primary review responsibility for SRP Section 18.[21]

4. For new plant applicants, the Probabilistic Safety Assessment Branch (SPSB) coordinates and performs shutdown risk assessment reviews as part of its primary review responsibility for SRP Section 19.1 (Proposed).[22]

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

The results of these reviews will be used by SRXB to complete the overall evaluation of the thermal-hydraulic review and will be incorporated into the Safety Evaluation Report (SER).[23]

II. ACCEPTANCE CRITERIA

The thermal and hydraulic design of the reactor core and the RCS, as described in the applicant's safety analysis report (SAR), is acceptable if the design is in accordance with the following criteria:[24]

A. ~~The CPB acceptance criteria are based on meeting the relevant requirements of~~ [25]General Design Criterion 10 ~~(Ref. 1)~~[26], as it relates to the reactor core being designed, with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operation or anticipated operational occurrences (AOO).

B. General Design Criterion 12, as it relates to the reactor core and associated coolant, control, and protection systems being designed to assure that power oscillations, which can result in conditions exceeding specified acceptable fuel design limits, are not possible or can be reliably and readily detected and suppressed.[27]

Specific criteria necessary to meet the requirements of GDC 10 and GDC 12[28] are as follows:

1. SRP Section 4.2 specifies the acceptance criteria for evaluation of fuel design limits. One of the criteria provides assurance that there be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) or transition condition during normal operation or anticipated operational occurrence.

   Uncertainties in the values of process parameters, core design parameters, and calculational methods used in the assessment of thermal margin should be treated with at least a 95% probability at a 95% confidence level.

   Two examples of acceptable approaches to meet this criterion are:

   a. For departure from nucleate boiling ratio (DNBR), critical heat flux ratio (CHFR) or critical power ratio (CPR) correlations there should be a 95% probability at the 95% confidence level, that the hot rod in the core does not experience a departure from nucleate boiling or boiling transition condition during normal operation or anticipated operational occurrences; or

   b. For DNBR, CHFR or CPR correlations, the limiting (minimum) value of DNBR, CHFR, of CPR is to be established such that at least 99.9% of the fuel rods in the core would not be expected to experience departure from nucleate boiling or boiling transition during normal operation or anticipated operational occurrences.

   Correlations of critical heat flux are continually being revised as a result of additional experimental data, changes in fuel assembly design, and improved calculational techniques involving coolant mixing and the effect of axial power distributions. As guidance to the reviewer, the correlations listed below have been found acceptable for previously reviewed plants.

a. BWRs - The value of the minimum CPR calculated with the GETAB analysis (~~Ref. 2~~Reference 17)[29] will vary for different plants and/or fuel types. Typical values are 1.06 and 1.07.

b. PWRs - The value of the minimum DNBR calculated with due allowance for mixing grids (~~Refs. 3, 4, and 5~~References 18, 19 and 20)[30] is typically 1.30 using the BAW-2 correlation (~~Ref. 6~~Reference 21)[31] or the W-3 correlation (~~Ref. 7~~Reference 22)[32]. Much lower values, depending upon the test data base and fuel design, are acceptable for more recent correlations such as the WRB-1, CE-1, and BWC.

2. Problems affecting DNBR or CPR limits, such as fuel densification or rod bowing, are accounted for by an appropriate design penalty which is determined experimentally or analytically. Subchannel hydraulic analysis codes such as those described in References ~~8~~23 and ~~9~~24[33]~~,~~ should be used to calculate local fluid conditions within fuel assemblies for use in PWR DNB correlations. The acceptability of such codes must be demonstrated by measurements made in large lattice experiments or power reactor cores. The effects of radial pressure gradients in the core flow distribution should be evaluated. Calculations of BWR fluid conditions for use in CHF correlations have been in accordance with the models specified in References[34] ~~10~~25 and ~~11~~26.[35]

3. The design should address core oscillations and thermal hydraulic instabilities as follows:

a. The reactor core and associated systems should be ~~demonstrated~~designed to have sufficient margin to be free of undamped oscillations and other thermal-hydraulic instabilities for all conditions of steady-state operation (including part loop operation and extended cycle operation with reduced feedwater temperature, where these operating conditions are proposed)[36], and for anticipated operational occurrences.

b. If the possibility of oscillations cannot be eliminated, the capability shall exist to reliably and readily detect and suppress the oscillations should they occur.[37] Methodologies for resolving BWR core stability issues are presented in GE topical report NEDO-31960 along with its Supplement (References 29 and 30) and were approved by the NRC in Reference 15. These reports provide long-term solutions to BWR stability issues as well as methodologies that have been developed to support the design of systems needed to ensure that plants are in compliance with GDC 10 and 12.[38]

4. Methods for calculating single-phase and two-phase fluid flow in the reactor vessel and other components should include classical fluid mechanics relationships and appropriate empirical correlations. For components of unusual geometry, such as the following, these relationships should be confirmed empirically, using representative data bases from approved reports of the type listed below.

a. Reactor vessel (~~Ref. 12~~Reference 27)[39].

b. Jet pump (~~Ref. 13~~Reference 28)[40].

c. Core flow distribution (~~Refs. 12 and 14~~References 27 and 29).[41]

5. The proposed technical specifications should be established such that the plant can be safely operated at steady state conditions under all of the expected combinations of system parameters. The safety limits and limiting safety settings must be established for each parameter, or combinations of parameters, such that specific[42] acceptance criterion 1, above, is satisfied.

6. Preoperational and initial startup test programs should follow the recommendations of Regulatory Guide 1.68 ~~(Ref. 15)~~[43], as regards measurements, and confirmation of thermal hydraulic design aspects.

7. The design description and proposed procedures for use of the loose parts monitoring system should be consistent with the requirements of Regulatory Guide 1.133 ~~(Ref. 16)~~.[44]

8. The effects of crud should be accounted for in the thermal-hydraulic design by including it in the CHF calculations in the core or in the pressure drop throughout the RCS. Process monitoring provisions should assure the[45] capability for detection of a three percent ~~pressure~~[46] drop in the reactor coolant flow. The flow should be monitored every 24 hours.

9. Instrumentation provided for an unambiguous indication of ICC, such as primary coolant saturation meters in PWRs, reactor vessel measurement systems, and core exit thermocouples, should meet the design requirements ~~described in~~ of TMI Action item II.F.2 of ~~NUREG-0718 (Ref. 17) and~~ NUREG-0737 (~~Ref. 18~~Reference 8)[47] and for applicants subject to 10 CFR 50.34(f), paragraph 10 CFR 50.34(f)(2)(xviii).[48] Procedures for detection and recovery from conditions of ICC must be consistent with technical guidelines, including applicable EPGs developed pursuant to the TMI action plan,[49] that incorporate response predictions based on appropriate analyses.

10. Thermal-hydraulic stability performance of the core during an ATWS event is such that acceptable fuel design limits are not exceeded. An acceptable method for performing such an analysis for BWR cores has been performed by GE using the TRACG code (References 16, 33, and 34).[50]

Technical Rationale:[51]

The technical rationale for application of the above acceptance criteria to the reactor core thermal and hydraulic design is discussed in the following paragraphs.

1. General Design Criterion 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. Proper thermal-hydraulic design of the reactor core and associated systems is necessary to assure sufficient margin exists with regard to

maintaining adequate heat transfer from the fuel to the RCS. Failure to maintain sufficient margin can result in a transition from nucleate boiling to film boiling on the fuel cladding surface. Film boiling decreases the heat transfer coefficient at the clad surface and the surface temperature rises significantly, eventually leading to fuel failure and the release of fission products to the RCS. Compliance with GDC 10 provides assurance that the integrity of the fuel and cladding will be maintained, thus preventing the potential for release of fission products during normal operation or AOOs.

2. General Design Criterion 12 requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations which result in conditions exceeding specified acceptable fuel design limits are not possible, or can be reliably and readily detected and suppressed. Power oscillations within the reactor core may result from conditions such as improper fuel design or loading, improper reactivity control including control rod positioning, coolant flow instabilities, moderator void formation, and instabilities associated with nonhomogeneous reactor coolant density distributions. The occurrence of power oscillations can lead to excessive localized power peaking, cyclic thermal fatigue, and subsequent exceeding of fuel design limits eventually leading to fuel failure. Compliance with GDC 12 provides assurance that the thermal-hydraulic design of the reactor core and associated systems protects the reactor from the consequences of power oscillations that could challenge the integrity of the fuel and result in the release of fission products.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in section II of this SRP section. For operating license (OL) applications, the procedures are utilized to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report. The OL review also includes the proposed technical specifications, to assure that they are adequate in regard to safety limits, limiting safety system settings, and conditions of operation.

The reviewer must begin with an understanding of currently acceptable thermal and hydraulic design practice for the reactor type under review. This understanding can be most readily gained from topical reports describing CHF correlations, system hydraulic models and tests, and core subchannel analysis methods; from standard texts and other technical literature which establish the methodology and the nomenclature of this technology; and from documents which summarize current staff positions concerning acceptable design methods.

Much of the review described below is generic in nature and is not performed for each plant. That is, the ~~CPB~~SRXB[52] reviewer is to compare the core design and operating parameters to those of previously reviewed plants. He then devotes the major portion of his review effort to those areas where the application is not identical to previously reviewed plants.

The reviewer is to compare the information in the applicant's safety analysis report (SAR) to the documents referenced by the applicant or in this SRP section to determine conformance to the bounds established by such documents. The reviewer must confirm that void, pressure drop, and heat transfer correlations used to estimate fluid conditions (flow, pressure, quality) are within the ranges of applicability specified by their authors or in previous staff reviews, that the analysis methods are used in the manner specified by the developers or in previous staff reviews, that the reactor design falls within the ranges of applicability specified for accepted analysis methods, and that the design is within the criteria specified in II, above, and is not an unexplained or unwarranted extrapolation of other thermal-hydraulic designs.

The review does not routinely involve calculations by the staff. However, the reviewer should ensure that those applications based on statistical design methodologies include the coefficients required by the statistical model and define the parameter ranges for which the coefficients are applicable. Uncertainties in computer codes, correlations, design methods, and set point methodologies should be quantified and the method(s) of accounting for these uncertainties in the design procedures should be discussed. For example the sensitivity factors and their ranges of applicability must be reviewed for those plants utilizing the Westinghouse "Improved Thermal Design Procedure," (~~Ref. 19~~Reference 30)[53]. On occasion, e.g., if a new design or new design method is proposed, independent analyses are performed by the staff or by consultants under the direction of ~~CPB~~ SRXB[54]. These analyses verify the design or establish the range of applicability and associated accuracy of the new method and the reviewer ensures it is applied accordingly.

The reviewer should evaluate the design of software used in core protection systems. The review should establish the acceptability of the software by comparison with previously approved designs and assess any differences with regard to effects on the system performance and safety functions. Consistency of the core protection algorithms and logic functions with the thermal-hydraulic analyses should be verified along with the program for implementing the software. Confirmation of adequate software implementation should be based on documented testing that verifies the acceptability of the software calculational systems, the proper integration of software and hardware systems, and the acceptable static and dynamic operation of the integrated system when compared with the predictions of the thermal-hydraulic design analyses. The reviewer should consult with HICB concerning the design acceptability of the hardware portion of the core protection systems.[55]

The reviewer is to establish that the thermal-hydraulic design and its characterization by MCHFR or DNBR have been accomplished and are presented in a manner which accounts for all possible reactor operating states as determined from operating maps. In this regard, the reviewer must confirm that the power distribution assumptions of SAR Section 4.4 are a conservative (i.e. worst-case) accounting of the power distributions derived in SAR Section 4.3 from core physics analyses, and that the latter analyses include an acceptable calculation of local void fractions. He must also confirm that the mass flux used in these calculations takes into account the core flow distribution (including that for partial loop operation) and the worst case of core bypass flow. The reviewer confirms that the primary coolant flow range shown in the operating map will be verified by prestartup measurements.

The reviewer evaluates information in the applicant's safety analysis report with regard to thermal-hydraulic stability concerns during normal operations, anticipated operational occurrences, and ATWS events for conformance with specific criteria II.3.[56] Specifically for

BWR applications, the reviewer assesses the applicants design in accordance with the following:[57]

1. The reactor and associated systems include provisions to facilitate manual or automatic protective action to ensure that specified acceptable fuel design limits are not exceeded in the event of thermal hydraulic instabilities in accordance with the recommendations of topical reports NEDO-31960 and NEDO-31960 Supplement 1 as approved by the related NRC SER (Reference 15).[58]

2. Thermal-hydraulic behavior during ATWS events has been analyzed as discussed in specific acceptance criterion II.10. Procedures direct appropriate actions to suppress thermal-hydraulic instability-related power oscillations during an ATWS event, similar to the actions discussed in Reference 16.[59]

For PWR and BWR applicants proposing operation with one of the reactor coolant pumps out of operation (i.e., N-1 loop operation), the reviewer determines whether such a mode of operation is acceptable based on the applicant's safety analyses and proposed technical specifications (Reference 11). Plant specific aspects of the safety analyses, may identify safety questions that could affect decisions regarding the desirability of (N-1) loop operation. Considerations related to reactor thermal-hydraulics include effects on core flow and temperature distributions and the ability of instrumentation to accurately reflect in-core parameters related to specified limits of DNBR or MCHFR. For BWR applicants, the reviewer should verify that N-1 loop operation will not result in thermal-hydraulic instabilities associated with the corresponding reduction in RCS flow, or that adequate procedures, including supporting analyses and associated technical specifications exist to respond to potential instabilities. The reviewer should also verify that the possibility for jet pump vibration during N-1 operation has been addressed by the applicant. For ABWR applicants proposing to operate with less than the maximum number of 10 Reactor Internal Pumps (RIPs), the reviewer confirms that with fewer than 10 RIPs in operation, continued plant operation including any reactor power level restrictions, are compatible with the plant safety analyses, and flow test results demonstrate there are no significant differences in core flow patterns.[60]

The reviewer ensures that adequate account is taken of the effect of crud in the primary coolant system, such as in the calculation of CHF in the core, heat transfer in the steam generators, and pressure drop throughout the RCS.

The reviewer is to examine the calculation of hydraulic loads for normal operations, including anticipated ~~transients~~ operational occurrences,[61] to ensure they are properly estimated for the worst cases. Worst case hydraulic loads for normal operations are to be provided for use in the analysis of lifting force of the fuel (SRP Section 4.2). ~~CPB~~SRXB[62] will also provide ~~consultation to RSB upon request, regarding~~ [63]calculations for ~~postulated~~ design-basis[64] accident conditions. The EMEB[65] reviews the adequacy of components and structures under design-basis[66] accident loads (SRP Section 3.9.2) and ~~CPB~~SRXB[67] determines that a coolable core geometry is maintained (SRP Section 4.2).

The reviewer should ensure that an adequate loose parts monitoring system is provided. At the CP level, the design criteria for the system and the types, locations, and methods of mounting all intended sensors should be reviewed. The reviewer should compare the design to Regulatory Guide ~~1.33~~1.133[68] and to equipment used and application experience on comparable plants.

At the OL level; a more complete description of the loose parts monitoring[69] system including sensitivity specifications and operating procedures should be reviewed. The reviewer should ensure that operating procedures and training provisions are adequate to fully utilize the system potential for loose parts detection. The ~~Operator Licensing (OLB)~~HHFB[70] will provide consultation on staff training in accordance with ~~the~~[71] SRP Section 13.2.1[72].

The reviewer should review the vibration monitoring equipment and procedures to ensure that the monitoring provisions are adequate for the plant under review based on experience with comparable plants. The ~~CPB~~SRXB[73] will evaluate the application of neutron monitoring sensors for core vibration test analysis. The EMEB[74] is responsible for review of the preoperational vibration test program, as described in SRP Section 3.9.2, and provides technical consultation to ~~CPB~~SRXB[75] on the need for permanent vibration monitoring provisions for the plant under review.

The reviewer ensures that applicants have an acceptable program for incorporation of instrumentation and procedures for detection and recovery from conditions of inadequate core cooling (ICC) consistent with TMI Action Plan item II.F.2 of NUREG 0737 and for applicants subject to 10 CFR 50.34(f), paragraph 10 CFR 50.34(f)(2)(xviii), as follows:~~.~~[76]

1. ~~At the~~For CP ~~stage~~reviews, the reviewer verifies that the applicant ~~must~~has provided preliminary design information on selected instrumentation components and ~~must~~ specif~~y~~ied the design concept selected for the~~development~~ instrumentation in accordance with the ~~requirements~~guidance[77] of item II.F.2 of NUREG-0718 (Ref~~.~~erence ~~17~~7).[78]

2. ~~At the OL stage~~For other application reviews, the reviewer ensures that the applicant is in compliance with the documentation requirements and design requirements described in item II.F.2 of NUREG-0737 ~~(Ref. 18)~~.[79] PWR ICC instrumentation found acceptable to the staff is described in Generic Letter 82-28 (Reference 10). BWR ICC instrumentation found acceptable to the staff is described in Generic Letter 84-23 (Reference 11). In addition, the reviewer verifies that BWR applicants have addressed noncondensible gases that may become dissolved in the reference leg of BWR water level instrumentation consistent with the positions indicated in Generic Letter 92-04 (Reference 14).[80]

3.[81] The reviewer consults with ~~ICSB~~HICB[82] and ~~HFEB~~HHFB[83] concerning the design acceptability of the ICC[84] instrumentation and displays and with the ~~Reactor Systems Branch (RSB) and~~[85] ~~PTRB~~PERB[86] concerning the acceptability of guidelines and procedures for recognition and response to inadequate core cooling conditions.

For new plant applicants and those PWRs subject to reference 13, the reviewer verifies that analyses of the thermal-hydraulic conditions during shutdown and low-power operations have been completed. The analyses should supplement existing information and should encompass the thermodynamic and physical states, such as a rapid boron dilution event during shutdown conditions (see Reference 9), to which the plant can be subjected. The analysis should be of sufficient depth to provide a basis for shutdown procedures, instrumentation, equipment interaction, equipment response and operator response.[87]

The applicant's proposed preoperational and initial startup test programs are reviewed to determine that they are consistent with the intent of Regulatory Guide 1.68 ~~(Ref. 15)~~.[88] At the OL stage, the reviewer is to assure that sufficient information is provided by the applicant to identify clearly the test objectives, methods of testing, and acceptance criteria. ~~(See par. C.2.b of Reference 15.)~~[89]

The test scope should include verification of any safety analysis codes or methods which could affect the thermal-hydraulic evaluations and which have not been previously verified. The initial startup test should also include a description of plans for a signature analysis to determine alarm setting for the loose parts monitoring system, and a description of test programs for evaluation, qualification and calibration of ICC instrumentation.

The reviewer evaluates the proposed test programs to determine if they provide reasonable assurance that the core and reactor coolant system will satisfy functional requirements. As an alternative to this detailed evaluation, the reviewer may compare the core and reactor coolant system design to that of previously reviewed plants. If the design is essentially identical and if the proposed test programs are essentially the same as performed previously on other plants, the reviewer may conclude that the proposed test programs are adequate for the core and reactor coolant system.

If the core or the reactor coolant system differs significantly from that of previously reviewed designs, the impact of the proposed changes on the preoperational and initial startup testing programs are reviewed at the construction permit stage. This effort should particularly evaluate the need for any special design features required to perform acceptable test programs.

The proposed technical specifications that relate to the core and the reactor coolant system are evaluated. This evaluation is to cover all of the safety limits and bases that could affect the thermal and hydraulic performance of the core. The limiting safety system settings are reviewed to ascertain that acceptable margins exist between the values at which reactor trip occurs automatically for each parameter (or combinations of parameters) and the safety limits. The reviewer confirms that the limiting safety system settings and limiting conditions for operation, as they relate to the reactor coolant system, do not permit operation with any expected combination of parameters that would not satisfy specific acceptance[90] criterion 1 of section II. For example, the limiting condition of operation must assure that the reactor coolant pumps have adequate net positive suction head for all expected modes of operation.

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.[91]

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report. The following paragraph is applicable to ~~both~~ a CP, design certification (DC),[92] and OL:

> The thermal-hydraulic design of the core for the __________ plant was reviewed. The scope of review included the design criteria, preliminary core design, and the steady state analysis of the core thermal-hydraulic performance.* The review concentrated on the differences between the proposed core design (and criteria) and those designs and criteria that have been previously reviewed and found acceptable by the staff. It was found that all such differences were satisfactorily justified by the applicant. The applicant's thermal-hydraulic analyses were performed using analytical methods and correlations that have been previously reviewed by the staff and found acceptable.

For a CP, the following conclusions should be made:

> The staff concludes that the thermal-hydraulic design of the core meets the requirements of General Design ~~Criterion 10~~ Criteria 10 and 12~~,~~ of Appendix A to[93] 10 CFR Part 50. The core and associated coolant, control, and protection systems ~~has~~have[94] been designed with appropriate margin to assure that acceptable fuel design limits are not exceeded during steady-state operation or anticipated operational occurrences. In meeting this objective, the design provides assurance that thermal-hydraulic instabilities leading to power oscillations are not possible or can be reliably and readily detected and suppressed.[95] This conclusion is based on the applicant's analyses of the core thermal-hydraulic performance which was reviewed by the staff and found to be acceptable. The applicant will establish a preoperational and initial startup test program in accordance with Regulatory Guide 1.68 to measure and confirm the thermal-hydraulic design aspects. The loose parts and vibration monitoring system is designed consistent with the guidance ~~for compliance with the requirements~~ [96]of Regulatory Guide 1.133 and the instrumentation for the detection of inadequate core cooling is in compliance with the requirements of item II.F.2 of NUREG-0718.

For an OL application, the following types of conclusions should be supported.

> The staff concludes that the thermal-hydraulic design of the core meets the requirements of General Design ~~Criterion 10~~ Criteria 10 and 12~~,~~ of Appendix A to[97] 10 CFR Part 50 and is acceptable for final design approval. We also conclude that the reactor core and associated coolant, control, and protection systems have~~has~~ been designed[98] with appropriate margin to assure that acceptable fuel design limits are not exceeded during steady-state operation or anticipated operational occurrences. In meeting this objective, the design provides assurance that thermal-hydraulic instabilities leading to power

* For an OL review this sentence should be modified to include the implementation of the design criteria as represented by the final core design.

oscillations are not possible or can be reliably and readily detected and suppressed,[99] and that the reactor will perform its safety functions throughout its design lifetime under all modes of operation. This conclusion is based on the applicant's analyses of the core thermal-hydraulic performance which was reviewed by the staff and found to be acceptable. The applicant has committed to a preoperational and initial startup test program in accordance with Regulatory Guide 1.68 to measure and confirm the thermal-hydraulic design aspects. The staff has reviewed the applicant's preoperational and initial startup test program and has concluded that it is acceptable. We also conclude that the loose parts monitoring program is designed consistent with the guidance ~~for compliance with the requirements~~ [100]of Regulatory Guide 1.133, and is, therefore, acceptable. We have reviewed the instrumentation for the detection of inadequate core cooling and concluded that it is in compliance with the acquirements of Item II.F.2 of NUREG-0737 and is acceptable.

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.[101]

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plan 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.[102] Except in those cases 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.[103]

The provisions of specific criterion II.10 apply to reviews for all new BWR applications.[104]

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides ~~and~~, NUREGs, and Generic Letters[105].

VI. REFERENCES

1. 10 CFR 50.34(f), "Additional TMI-related requirements," paragraph 10 CFR 50.34(f)(2)(xviii).[106]

2. 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) events for light-water-cooled nuclear power plants."[107]

~~1~~3.[108] 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."

4. 10 CFR Part 50, Appendix A, General Design Criterion 12, "Suppression of Reactor Power Oscillations."[109]

~~15~~5. Regulatory Guide~~ No.~~ 1.68, "~~Preoperational and~~ Initial ~~Startup~~ Test Programs for Water-Cooled Nuclear Power~~ Reactors~~ Plants."[110]

~~16~~6. Regulatory Guide 1.133, "Loose Parts Detection Program for the Primary System of Light-Water-Cooled Reactors."

~~17~~7. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," Revision 1, June 1981.[111]

~~18~~8. NUREG-0737, "Clarification of TMI Action Plan Requirements~~.~~," November 1980.[112]

9. NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States." Final Report, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, September 1993.[113]

10. NRC Letter to All Licensees of Operating Westinghouse and CE PWRs (Except Arkansas Nuclear One-Unit 2 and San Onofre Units 2 and 3), "Inadequate Core Cooling Instrumentation System (Generic Letter No. 82-28)," December 10, 1982.[114]

11. NRC Letter to All Boiling Water Reactors (BWR) Licensees of Operating Reactors Except Lacrosse, Big Rock Point, Humboldt Bay and Dresden-1), "Reactor Vessel Water Level Instrumentation in BWRs (Generic Letter No. 84-23)," October 26, 1984.[115]

12. NRC Letter to All Licensees of Operating BWRs and PWRs and License Applicants, "Technical Resolution of Generic Issue No. B-59-(N-1) Loop Operation in BWRs and PWRs (Generic Letter No. 86-09)," March 31, 1986.[116]

13. NRC Letter to all Holders of Operating Licenses and Construction Permits for Pressurized Water Reactors (PWRs), “Loss of Decay Heat Removal (Generic Letter 88-17),” October 17, 1988.[117]

14. NRC Letter to All Boiling Water Reactor (BWR) Licensees of Operating Reactors, "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f) (Generic Letter No. 92-04)," August 19, 1992.[118]

15. A. C. Thadani, "Acceptance for Referencing of Topical Reports NEDO-31960 and NEDO-31960 Supplement 1, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology," U.S. Nuclear Regulatory Commission, July 12, 1993.[119]

16. A. C. Thadani, "Acceptance for Referencing of Topical Reports NEDO-32047 and NEDO-32164, Revision O, BWR Owners' Group Evaluation of ATWS Rule Issues and Mitigative Actions," U.S. Nuclear Regulatory Commission, February 5, 1994.[120]

~~2~~17. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," NEDO-10958, General Electric Company (1973).[121]

~~3~~18. F. F. Cadek, F. E. Motley, and D. P. Dominicis, "Effect of Axial Spacing on Interchannel Thermal Mixing with the R Mixing Vane Grid," WCAP-7941-L (proprietary), Westinghouse Electric Corporation, June 1972.[122]

~~4~~19. F. E. Motley and F. F. Cadek, "DNB Test Results for New Mixing Vane Grids (R)," WCAP-7695-L (proprietary), Westinghouse Electric Corporation, July 1972.[123]

~~5~~20. F. E. Motley and F. F. Cadek, "Application of Modified Spacer Factor to L Grid Typical and Cold Wall Cell DNB," WCAP-7988, Westinghouse Electric Corporation, October 1972. (See also WCAP-8030.)[124]

~~6~~21. J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, and L. J. Stanek, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," in "Two-Phase Flow and Heat Transfer in Rod Bundles," American Society of Mechanical Engineers, New York (1969). (See also BAW-10000 and BAW-10036.)[125]

~~7~~22. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," Journal of Nuclear Energy, Vol. 21, 241-248 (1967).

~~8~~23. "TEMP - Thermal Enthalpy Mixing Program," BAW-10021, Babcock and Wilcox Company, April 1970.[126]

~~9~~24. H. Chelemer, P. T. Chu, and L. E. Hochreiter, "THINC-IV-An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, Westinghouse Electric Corporation, June 1973. (See also WCAP-7359-L and WCAP-7838.)[127]

~~10~~25. B. C. Slifer and J. E. Hench, "Loss of Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," NEDO-10329, Appendix C, General Electric Company, April 1971.[128]

~~11~~26. J. Duncan and P. W. Marriott, "General Electric Company Analytical Model for Loss of Coolant Accident Analysis in Accordance with 10 CFR Part 50, Appendix K," NEDO-20566, General Electric Company, November 1975.

~~12~~27. B. S. Mullanax, R. J. Walker and B. A. Karrasch, "Reactor Vessel Model Flow Tests," BAW-10037 (non-proprietary version of BAW-10012), Revision 2, Babcock and Wilcox Company, September 1968.[129]

~~13~~28. "Design and Performance of General Electric Boiling Water Reactor Jet Pumps," APED-5460, General Electric Company, September 1968.[130]

~~14~~29. H. T. Kim, "Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello," NEDO-10299, General Electric Company, January 1971.[131]

~~19~~30. H. Chelemer, L. H. Boman, D. R. Sharp, "Improved Thermal Design Procedure," WCAP-8567(P)/8568(NP), Westinghouse Electric Corporation, July 1975.[132]

31. NEDO-31960, "BWR Owner's Group Long-Term Stability Solutions Licensing Methodology," General Electric Company, May 1991.[133]

32. NEDO-31960 Supplement 1, "BWR Owner's Group Long-Term Stability Solutions Licensing Methodology," General Electric Company, March 1992.[134]

33. NEDO-32047, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability," General Electric Company, February 1992.[135]

34. NEDO-32164, Revision 0, "Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS," General Electric Company, December 1992.[136]

~~APPENDIX~~[137]
~~STANDARD REVIEW PLAN 4.4~~

~~INDEPENDENT AUDIT ANALYSIS~~
~~(Appendix to SRP Section 4.4 has been deleted)~~

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SRP Draft Section 4.4

Attachment A - Proposed Changes in Order of Occurrence

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

Item	Source	Description
1.	Current PRB names and abbreviations.	Editorial change to reflect the current PRB name and responsibility for SRP Section 4.4.
2.	Current PRB names and abbreviations.	Editorial change to identify that there are currently no secondary PRBs for SRP Section 4.4.
3.	GSI B-3 Resolution.	Corrected "anticipated operational transients" to be "anticipated operational occurrences."
4.	Editorial	Added text to define the first use of the acronyms CHF and CPR.
5.	**Integrated Impact 1397**	This area of review describes the review for new plant applicants of the thermal-hydraulic design during shutdown and low-power conditions.
6.	GSI B-3 Resolution.	"postulated accident conditions" was revised to be "design-basis accident conditions."
7.	**Integrated Impact 221.**	Added Area of Review for determining the acceptability of the core thermal-hydraulic design under ATWS conditions.
8.	Current PRB names and abbreviations.	Editorial change to reflect the current PRB with responsibility for review of the Inadequate Core Cooling (ICC) monitoring system. Although this area was previously reviewed by the Instrumentation and Control System Branch (ICSB) [now the Instrumentation and Controls Branch (HICB)] as a secondary review responsibility for SRP Section 4.4, this secondary review responsibility no longer exists. Additionally, the HICB has stated in comments received on Integrated Impacts for SRP Section 7.1 that they are not responsible for reviews associated with TMI item II.F.2 which deals ICC systems. Therefore, the responsibility for the review was assigned to the primary review branch for SRP Section 4.4.

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Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
9.	Current PRB names and abbreviations, Editorial.	With the exception of the review of inadequate core cooling (ICC) monitoring hardware, the paragraph discussing secondary reviews was deleted. There are currently no secondary review branches responsible for SRP Section 4.4. The review responsibilities identified in the deleted paragraph are also contained in the Review Procedures subsection of SRP 4.4; therefore, the review responsibilities for emergency procedures are incorporated as a new Review Interface (interface item no. 3) to ensure consistency with the existing Review Procedures. The review of ICC hardware remains with the PRB responsible for SRP Section 4.4.
10.	SRP-UDP format item, Reformat Areas of Review	The paragraph was moved to review interface item 1 for the SRXB because of the association with SRP Section 4.3.
11.	SRP-UDP format item, Reformat Areas of Review	Added "Review Interfaces" heading to Areas of Review. Added review interfaces identified in the Review Procedures in numbered format to describe how other branches support the review of the thermal and hydraulic design of the reactor, and to identify the interfacing SRP Sections.
12.	SRP-UDP format item, Reformat Areas of Review	Added lead-in sentence for review interfaces involving the same PRB responsible for SRP Section 4.4.
13.	Editorial.	Pluralized the word "assumption."
14.	SRP-UDP format item, Reformat Areas of Review	To be consistent with other SRP Sections and SRP-UDP format guidance, the paragraph was moved from Areas of Review, to Review Interfaces because of the interface with SRP Section 4.3.
15.	Editorial	Added lead-in sentence to review interfaces with other branches consistent with other SRP Sections.
16.	Editorial.	Added review interface with EMEB. The interface was derived from the existing Review Procedures that describe the responsibilities of the EMEB with regard to review of components and structures as part of their review responsibility for SRP Section 3.9.2.
17.	**Integrated Impact 924.**	Added a review interface with the HICB for reviews of core and reactor protection hardware systems that interface with software systems reviewed in SRP Section 4.4.
18.	Current PRB names and abbreviations, Editorial.	The review interface was derived from the Review Procedures and the deleted secondary reviews.

Item	Source	Description
19.	Current PRB names and abbreviations, Editorial.	The review interface for the HHFB was derived from the Review Procedures which state that the "Operator Licensing Branch (OLB) will provide consultation on staff training in accordance with the SRP Section 13.2." SRP Section 13.2 has been superseded by SRP Sections 13.2.1 and 13.2.2 which address reactor operator training and training for non-licensed staff respectively. The context of the Review Procedure is training in the use of the loose parts monitor which, because it is a plant system, appears to fall under the category of reactor operator training and thus the guidelines of SRP Section 13.2.1. The current PRB responsible for SRP Section 13.2.1 is the HHFB.
20.	Editorial	Relocated the Review Interface covering EPGs to the appropriate PRB/SRP Section, added discussion of the review of associated procedure development programs, and revised to reflect that human factors aspects of several displays related to core thermal-hydraulic performance and/or inadequate core cooling may be reviewed under SRP Chapter/Section 18.
21.	Editorial	The new SRP section replacing existing SRP Chapter 18 sections is currently labeled as SRP Section 18.
22.	**Integrated Impact 1397.**	This review interface identifies reviews conducted to satisfy SECY 93-087, NUREG-1449 and Generic Letter 88-17 guidance on Shutdown and Low Power Operations. The staff requested that design certification applicants complete an assessment of shutdown and low-power risk. The shutdown and low-power risk assessment must identify design-specific vulnerabilities and weaknesses and document consideration and incorporation of design features that minimize such vulnerabilities. Thermal and hydraulic analysis issues related to shutdown condtions are part of the shutdown risk assessments. The shutdown and low-power risk assessment is the responsibility of the SPSB and will be included in the proposed SRP Section 19.1 on risk assessments.
23.	SRP-UDP format item, Reformat Areas of Review.	Added text to new review interfaces to be consistent with the content of other SRP Sections.
24.	Editorial.	Added lead-in sentence to acceptance criteria to be consistent with other SRP Sections and to facilitate the addition of new acceptance criteria, GDC 12.
25.	Editorial.	Enumerated the Acceptance Criteria, and revised the text describing GDC 10 to be consistent with the addition of GDC 12.
26.	SRP-UDP format item, Reformat reference citations	Parenthetical reference identification for GDC 10 was deleted in accordance with SRP-UDP guidance.

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Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
27.	**Integrated Impact 220.**	Incorporated GDC 12 as new Acceptance Criteria for SRP Section 4.4.
28.	**Integrated Impact 220.**	Added reference to GDC 12 in lead-in sentence to specific criteria.
29.	SRP-UDP format item, Reformat reference citations, Editorial	Revised parenthetical reference identification to be consistent with SRP-UDP guidance and renumbered the reference consistent with the changes to subsection VI.
30.	SRP-UDP format item, Reformat reference citations, Editorial	Revised parenthetical reference identification to be consistent with SRP-UDP guidance and renumbered references consistent with changes to subsection VI.
31.	SRP-UDP format item, Reformat reference citations, Editorial	Revised parenthetical reference identification to the existing citation of the BAW-2 correlation, and renumbered the reference consistent with the changes to subsection VI.
32.	SRP-UDP format item, Reformat reference citations, Editorial	Revised parenthetical reference identification to the existing citation of the W-3 correlation, and renumbered the reference consistent with changes to subsection VI.
33.	Editorial	Renumbered references to be consistent with changes to subsection VI. References.
34.	Editorial	Pluralized the word reference, because there are two references listed.
35.	Editorial	Renumbered references to be consistent with changes to subsection VI. References.
36.	**Integrated Impact 222.**	Added text to the acceptance criteria to include consideration of thermal-hydraulic stability during extended cycle operation with reduced feedwater temperature.
37.	**Integrated Impact 220.**	Incorporated elements of GDC 12 into existing specific criteria related to reactor core oscillations and instabilities.
38.	**Integrated Impact 220.**	Added NRC approved methodologies for resolving BWR core stability issues as presented in GE Topical Reports NEDO-31960 and its supplement.
39.	SRP-UDP format item, Reformat reference citations, Editorial	Revised parenthetical reference identification to be consistent with SRP-UDP guidance, and renumbered the reference consistent with changes to subsection VI.
40.	SRP-UDP format item, Reformat reference citations, Editorial	Revised parenthetical reference identification to be consistent with SRP-UDP guidance, and renumbered the reference consistent with changes to subsection VI.

Item	Source	Description
41.	SRP-UDP format item, Reformat reference citations, Editorial	Revised parenthetical reference identification to be consistent with SRP-UDP guidance, and renumbered the references consistent with changes to subsection VI.
42.	Editorial.	The text was modified to refer to "specific acceptance criteria 1."
43.	SRP-UDP format item, Reformat reference citations.	Deleted parenthetical identification of reference citation for Regulatory Guide 1.68 consistent with SRP-UDP format guidance.
44.	SRP-UDP format item, Reformat reference citations.	Deleted parenthetical identification of reference citation for Regulatory Guide 1.133 consistent with SRP-UDP format guidance.
45.	Editorial	"The" was added as a grammatical correction to the sentence.
46.	Editorial.	The word "pressure" is deleted so that the sentence requires detection of a 3 % drop in flow.
47.	SRP-UDP format item, Reformat reference citations, Editorial	Revised parenthetical reference identification to the existing citation of NUREG 0737, and renumbered the reference consistent with changes to subsection VI.
48.	**Integrated Impact 921.**	Clarified text with regard to TMI Action Plan item II.F.2 and added reference to 10 CFR 50.34(f)(2)(xviii). Reference to NUREG 0718 was deleted because it is a guidance document related to 10 CFR 50.34(f)(2)(xviii) and is cited in the Review Procedures.
49.	**Integrated Impact 921**	Added reference to EPGs as a source of criteria for ICC response procedures.
50.	**Integrated Impact 221.**	Added specific criteria related to thermal-hydraulic analysis of core performance under ATWS conditions.
51.	SRP-UDP format item, Develop Technical Rationale	Added Technical Rationale for GDCs 10 and 12. Technical Rationale is a new SRP-UDP format item.
52.	Current PRB names and abbreviations.	Editorial change to reflect the current PRB name and responsibility for SRP Section 4.4.
53.	SRP-UDP format item, Reformat reference citations.	Revised the parenthetical identification for the reference to the Westinghouse "Improved Thermal Design Procedure," to be consistent with SRP-UDP guidance.
54.	Current PRB names and abbreviations.	Editorial change to reflect the current PRB name and responsibility for SRP Section 4.4.
55.	**Integrated Impact 924.**	A Review Procedure is added to address the review and evaluation of core protection software with regard to verification of design and implementation as it relates to the thermal-hydraulic design analyses.

Item	Source	Description
56.	**Integrated Impact 223.**	Added generic text referring to the applicable acceptance criteria to ensure that thermal-hydraulic stability is considered for all appropriate operating conditions. The generic nature of the text allows application of the review procedure to both PWRs and BWRs.
57.	**Integrated Impacts 220 and 221.**	Added a paragraph for the reviewer to evaluate the thermal-hydraulic stability of BWR applicants. The paragraph provides a lead-in to specific items to be reviewed.
58.	**Integrated Impact 220.**	Added guidance to Review Procedures for long-term resolution of thermal-hydraulic concerns by incorporating the long term required actions from Generic Letter 94-02.
59.	**Integrated Impact 221.**	Added guidance to Review Procedures for resolution of thermal-hydraulic stability during ATWS events by reference to Topical Reports NEDO-32047 and -32164.
60.	**Integrated Impact 224.**	Added guidance to Review Procedures for (N-1) loop operation by incorporating information from Generic Letter 86-09. Similar information from the ABWR FSER were provided for inoperable Reactor Internal Pumps (RIPs).
61.	GSI B-3 Resolution	Corrected "anticipated transients" to be "anticipated operational occurrences."
62.	Current PRB names and abbreviations	Editorial change to reflect the current PRB name and responsibility for SRP Section 4.4.
63.	Current PRB names and abbreviations	Editorial change to delete the text stating the CPB will provide consultation to the RSB. The responsibilities of the previous CPB and RSB are combined under the current SRXB.
64.	GSI B-3 Resolution	Corrected "postulated accident" to "design-basis accident."
65.	Current PRB names and abbreviations	Editorial change to add "The" to the beginning of the sentence and to update the abbreviation for the Mechanical Engineering Branch from "MEB" to "EMEB."
66.	GSI B-3 Resolution	Added "design-basis" prior to the word accident.
67.	Current PRB names and abbreviations	Editorial change to reflect the current PRB name and responsibility for SRP Section 4.4.

Item	Source	Description
68.	Editorial	Corrected typographical error to change Regulatory Guide 1.33 to Regulatory Guide 1.133, which is the correct regulatory guide reference for loose parts monitoring.
69.	Editorial	Added "loose parts monitoring" to provide clarification regarding the "system" that is the subject of the review procedure.
70.	Current names and abbreviations	Editorial change to reflect the current PRB name and responsibility for SRP Section 13.2.1. Note that SRP Section 13.2 was superseded by SRP Sections 13.2.1 and 13.2.2. SRP Section 13.2.1 was judged to be the applicable section since it deals with the requirements for reactor operators and the context of the paragraph containing the change deals with system operation.
71.	Editorial	"The" was deleted as a grammatical correction.
72.	Editorial	SRP Section 13.2 was superseded by SRP Sections 13.2.1 and 13.2.2. SRP Section 13.2.1 was judged to be the applicable section since it deals with the requirements for reactor operators and the context of the paragraph containing the change deals with system operation.
73.	Current PRB names and abbreviations	Editorial change to reflect the current PRB name and responsibility for SRP Section 4.4.
74.	Current PRB names and abbreviations	Editorial change to update the abbreviation for the Mechanical Engineering Branch from "MEB" to "EMEB."
75.	Current PRB names and abbreviations	Editorial change to reflect the current PRB name and responsibility for SRP Section 4.4.
76.	**Integrated Impact 921.**	Text was added to identify the applicable requirements associated with inadequate core cooling and to provide a lead-in to subordinate review procedures.
77.	**Integrated Impact 921.**	Separated text from the 12th paragraph of the existing review procedures and revised the text to create a subordinate paragraph to the lead-in paragraph discussing compliance with the requirements of TMI Action Plan item II.F.2 and 10 CFR 50.34(f)(2)(xviii).
78.	SRP-UDP format item, Reformat reference citations	The parenthetical identification for NUREG 0718 was revised and renumbered in accordance with SRP-UDP guidance.
79.	SRP-UDP format item, Reformat reference citations	The parenthetical identification for NUREG 0737 was deleted, because it is only required on the first occurrence of the reference, which occurred in the Acceptance Criteria, paragraph II.9.

Item	Source	Description
80.	**Integrated Impact 921.**	Added a number to the 13th paragraph of the existing Review Procedures to create a subordinate paragraph to the lead-in paragraph discussing compliance with TMI Action Plan item II.F.2 and 10 CFR 50.34(f)(2)(xviii). Added related text and references to Generic Letters 82-28, 84-23, and 92-04, with regard to staff positions and guidance related to inadequate core cooling instrumentation.
81.	**Integrated Impact 921,** Editorial	Separated text from the 13th paragraph in the SRP to create a subordinate item under the lead-in paragraph discussing TMI Action Plan item II.F.2 and 10 CFR 50.34(f)(2)(xviii)compliance.
82.	Current PRB names and abbreviations	Editorial change to reflect the current PRB name for the Instrumentation and Controls Branch.
83.	Current PRB names and abbreviations	Editorial change to reflect the current PRB name for the Human Factors Assessment Branch.
84.	Editorial	Added "ICC" to identify the type of instrumentation to which the paragraph is referring.
85.	Current PRB names and abbreviations	Editorial change to delete reference to consultation with the Reactor Systems Branch since this Branch is now responsible for SRP Section 4.4.
86.	Current PRB names and abbreviations	Editorial change to reflect the current PRB name for the Emergency Preparedness and Radiation Protection Branch.
87.	Integrated Impact 1397.	This paragraph describes the type of thermal-hydraulic analyses required during shutdown conditions. Shutdown procedures, instrumentation, operator response, and equipment interaction and response will be dependent upon the results of anaylses to develop a bases for critical thermodynamic events such as rapid boron dilution and postulated times to core uncovery during a loss of shutdown decay heat removal.
88.	SRP-UDP format item, Reformat reference citations	The parenthetical identification for Regulatory Guide 1.68 was deleted in accordance with the SRP-UDP guidance.
89.	SRP-UDP format item, Reference Verification	Regulatory Guide 1.68 has been revised. There is no position C.2.b in the current version of the Regulatory Guide.
90.	Editorial.	The text was modified to refer to "specific acceptance criteria 1."
91.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.

Item	Source	Description
92.	10 CFR 52 applicability	Added "design certification" to the list of application types for which the general Evaluation Findings statement applies.
93.	**Integrated Impact 220.**	Incorporated GDC 12 into the Evaluation Findings.
94.	Editorial	The text was revised to include the entire scope of applicability of GDCs 10 and 12.
95.	**Integrated Impact 220.**	Incorporated requirements of GDC 12 into the Evaluation Findings for CP applicants.
96.	Editorial	The findings, as written, imply that Regulatory Guide 1.133 is a requirement. Regulatory Guides provide guidance and positions acceptable to the staff for implementing regulatory requirements, but are not requirements themselves. The text of the finding was modified to indicate that the design is found to be consistent with the Regulatory Guide.
97.	**Integrated Impact 220.**	Incorporated GDC 12 into the Evaluation Findings.
98.	Editorial	The text was revised to include the entire scope of applicability of GDCs 10 and 12, and corrected the verb tense.
99.	**Integrated Impact 220.**	Incorporated requirements of GDC 12 into the Evaluation Findings for OL applicants.
100.	Editorial	The findings, as written, imply that Regulatory Guide 1.133 is a requirement. Regulatory Guides provide guidance and positions acceptable to the staff for implementing regulatory requirements, but are not requirements themselves. The text of the finding was modified to indicate that the design is found to be consistent with the Regulatory Guide.
101.	10 CFR 52 applicability.	Added standard text addressing the use of the SRP review procedures for design certification reviews.
102.	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.
103.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
104.	**Integrated Impact 221.**	Added a statement to indicate that the requirement established in specific criteria II.10 is only applicable to new applications.
105.	Editorial	Revised to reflect the existence of implementation information and schedules in referenced Generic Letters.

Item	Source	Description
106.	**Integrated Impact 921.**	Added reference to 10 CFR 50.34(f)(2)(xviii) with regard to TMI Action Plan requirements.
107.	**Integrated Impact 221.**	Added reference to 10 CFR 50.62 for ATWS issues. The rule was added as specific criteria in subsection II of the SRP.
108.	Editorial	Relocated references for Regulatory Guides and NUREGs and renumbered references to accommodate the addition of GDC 12.
109.	**Integrated Impact 220.**	Added GDC to References subsection.
110.	Reference Verification	Updated the title of Regulatory Guide 1.68.
111.	Reference Verification	Updated title, revision number and publication date to reflect the latest revision of this reference..
112.	Reference Verification	Added publication date for this reference.
113.	Integrated Impact 1397.	Added a reference to NUREG-1449 which documents the NRC staff's evaluation and recommendations for shutdown and low-power operations.
114.	**Integrated Impact 921.**	Added reference to Generic Letter 82-28 regarding inadequate core cooling instrumentation systems.
115.	**Integrated Impact 921.**	Added reference to Generic Letter 84-23 regarding reactor vessel level indication in BWRs as related to review of inadequate core cooling instrumentation systems.
116.	**Integrated Impact 224.**	Added reference to Generic Letter 86-09 regarding N-1 loop operation.
117.	Integrated Impact 1397.	Added a reference to Generic Letter 88-17 to support the new review procedure paragraph covering containment analyses during shutdown conditions.
118.	**Integrated Impact 921.**	Added reference to Generic Letter 84-23 regarding reactor vessel level indication in BWRs as related to review of inadequate core cooling instrumentation systems.
119.	**Integrated Impact 220.**	Added the NRC SER for GE Topical Reports NEDO-31960 and NEDO-31960 Supplement 1 to the References subsection of the SRP.
120.	**Integrated Impact 221.**	Added the NRC SER for GE Topical Reports NEDO-32047 and NEDO-32164 to the References subsection of the SRP.
121.	Unverified reference	The reference is a vendor report and could not be verified.

Item	Source	Description
122.	Unverified reference	The reference is a vendor report and could not be verified.
123.	Unverified reference	The reference is a vendor report and could not be verified.
124.	Unverified reference	The reference is a vendor report and could not be verified.
125.	Unverified reference	The reference was not available and could not be verified.
126.	Unverified reference	The reference is a vendor report and could not be verified.
127.	Unverified reference	The reference is a vendor report and could not be verified.
128.	Unverified reference	The reference is a vendor report and could not be verified.
129.	Unverified reference	The reference is a vendor report and could not be verified.
130.	Unverified reference	The reference is a vendor report and could not be verified.
131.	Unverified reference	The reference is a vendor report and could not be verified.
132.	Unverified reference	The reference is a vendor report and could not be verified.
133.	**Integrated Impact 220.**	Added thermal-hydraulic stability long-term resolution guidance to References subsection.
134.	**Integrated Impact 220.**	Added thermal-hydraulic stability long-term resolution guidance to References subsection.
135.	**Integrated Impact 221.**	Added thermal-hydraulic stability during ATWS guidance to References subsection.
136.	**Integrated Impact 221.**	Added thermal-hydraulic stability during ATWS guidance to References subsection.
137.	Editorial	Deleted page for deleted appendix.

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SRP Draft Section 4.4

Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
220	Incorporate GDC 12 into the Acceptance Criteria, modifying the "specific criteria" to refer to NEDO-31960 and NEDO-31960 Supplement 1, and augment the Review Procedures to address the staff positions related to BWR stability including those contained in Generic Letter 94-02.	II, III, IV and VI
221	Add Review Procedures to incorporate the resolution of thermal-hydraulic instability concerns during ATWS events, as described in NEDO-32047, NEDO-32164, Revision O and the staff safety evaluation of these topic reports.	I, II, III, and VI
222	Modify the Review Procedures to address the potential for thermal-hydraulic instabilities during extended cycle operation.	II
223	Modify Review Procedures to address the verification by testing of thermal-hydraulic stability in new PWR designs.	III
224	Incorporate guidance into Review Procedures for (N-1) loop operation from Generic Letter 86-09 and for inoperable RIPs in the evolutionary BWRs.	III, VI
921	Modify the Review Procedures to address staff requirements and guidance on the adequacy of inadequate core cooling instrumentation in relation to conformance with TMI Action Item II.F.2.	II, III, VI
924	Modify the Review Procedures to address staff positions relative to the adequacy, verification and testing of core protection software.	I, III
1397	Modify Review Procedure to address thermal-hydraulic analyses that may be required to support shutdown operations.	I, III, VI