U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

4.4 THERMAL AND HYDRAULIC DESIGN

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of thermal and hydraulic design for Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs)

Secondary - Other organizations responsible for the review analyses related to thermal and hydraulic design

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) (1) uses acceptable analytical methods, (2) is equivalent to or is a justified extrapolation from proven designs, (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and anticipated operational occurrences (AOOs), and (4) is not susceptible to thermal-hydraulic instability. The specific areas of review are as follows:

1. This SRP section describes the normal review of thermal and hydraulic design (i.e., a design for a plant similar in core and RCS design to previously reviewed plants). The review of new prototype plants, new critical heat flux (CHF) or critical power ratio (CPR) correlations, and new analysis methods require additional independent audit analyses. The required analyses may be in the following form:

A. Independent computer calculations to substantiate reactor vendor analyses.

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

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

2. The review evaluates the proposed technical specifications regarding safety limits and limiting safety system settings to ascertain that they 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.

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

4. The review determines the largest hydraulic loads on core and RCS components during normal operation and design-basis accident conditions. This information is used in the review of fuel holddown requirements.

5. The review evaluates the uncertainty analysis methodology and the uncertainties of variables and correlations such as CHF and CPR. The review also evaluates the uncertainties associated with the combination of variables.

6. To accomplish the objectives, the reviewer examines core and RCS component features, 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 described in topical reports and applied in staff reports. The analysis methods to be addressed include core thermal-hydraulic calculations to establish local coolant conditions, departure from nucleate boiling (DNB) 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.

7. The reviewer will evaluate the functional performance and requirements for the inadequate core cooling (ICC) monitoring system hardware.

8. Inspection, Test, Analysis, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the applicant's proposed information on the ITAAC associated with the systems, structures, and components (SSCs) related to this SRP section is reviewed in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria - Design Certification." The staff recognizes that the review of ITAAC is performed after review of the rest of this portion of the application against acceptance criteria contained in this SRP section. Furthermore, the ITAAC are reviewed to assure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

9. <u>COL Action Items and Certification Requirements and Restrictions</u>. COL action items may be identified in the NRC staff's final safety evaluation report (FSER) for each certified design to identify information that COL applicants must address in the application. Additionally, DCs contain requirements and restrictions (e.g., interface requirements) that COL applicants must address in the application. For COL applications referencing a DC, the review performed under this SRP section includes information provided in response to COL action items and certification requirements and restrictions pertaining to this SRP section, as identified in the FSER for the referenced certified design.

<u>Review Interfaces</u>

The listed SRP sections interface with this section as follows:

1. The review of power distribution assumptions made for the core thermal and hydraulic analysis is coordinated with the review for core physics calculations under SRP Section 4.3. The reviewer verifies that the core monitoring techniques that rely on in-core or ex-core neutron sensor inputs are evaluated.

2. The review of anticipated transients without scram (ATWS) and coupled neutronic-thermal-hydraulic instabilities for BWRs is performed under SRP Sections 15.8 and 15.9, respectively.

3. The review of the adequacy of components and structures under accident loads and the preoperational vibration test program is performed under SRP Section 3.9.

4. The review of the core protection and reactor protection hardware to determine compliance with the requirements applicable to reactor trip systems is performed under SRP Section 7.2.

5. The review of ICC monitoring system hardware to determine compliance with the requirements applicable to information systems important to safety is performed under SRP Section 7.5.

6. The review of the applicant's training of loose parts monitor is performed under SRP Section13.2.

7. The review of emergency procedure guidelines (EPGs) and associated programs for development of plant-specific emergency operating procedures, including those associated with recognizing and responding to inadequate core cooling (ICC) conditions, is performed under SRP Section 13.5.

8. The review of the human factors aspects of information displays is performed under SRP Section 18.

9. For new plant applicants, the review of shutdown risk assessment is performed under SRP Section 19.1.

The primary review organizations will use the results of these reviews to complete the overall evaluation of the thermal-hydraulic review; the results will also be incorporated into the safety evaluation report (SER).

The specific acceptance criteria and review procedures are contained in the referenced SRP sections.

II. ACCEPTANCE CRITERIA

Requirements

The thermal and hydraulic design of the reactor core and the RCS, as described in the applicant's safety analysis report (SAR) or design certification document (DCD) for new plants, is acceptable if the design is in accordance with specific criteria. Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. General Design Criterion (GDC) 10, as it relates to whether the design of the reactor core includes appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation or AOOs.

2. GDC 12, as it relates to whether the design of the reactor core and associated coolant, control, and protection systems assures that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for review described in Subsection I of this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

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

1. SRP Section 4.2 specifies the acceptance criteria for the evaluation of fuel design limits. One criterion provides assurance that there be at least a 95-percent probability at the 95-percent confidence level that the hot fuel rod in the core does not experience a DNB or transition condition during normal operation or AOOs.

   Uncertainties in the values of process parameters (e.g., reactor power, coolant flow rate, core bypass flow, inlet temperature and pressure, nuclear and engineering hot channel factors), core design parameters, and calculational methods used in the assessment of thermal margin should be treated with at least a 95-percent probability at the 95-percent confidence level. The assessment of thermal margin should also consider the uncertainties in instrumentation. The origin of each uncertainty parameter, such as fabrication uncertainty, computational uncertainty, or measurement uncertainty e.g., reactor power, coolant temperature, flow), should be identified. Each uncertainty parameter should be identified as statistical or deterministic and should clearly describe the methodologies used to combine uncertainties.

   Core design and operating changes for extended power uprates (EPUs) should be performed in a manner that ensures adequate safety margin. At a minimum, there should be a 95-percent probability at the 95-percent confidence level that a hot fuel rod in the reactor core will not experience a DNB or a transition condition during normal operation or AOOs. Specifically, this safety criterion should be satisfied while accounting for changes in radial and bundle power distribution, including any changes in critical heat flux ratio (CHFR) and CPR. The reviewer should confirm the adequacy of the flow-based average power range monitor flux trip and safety limit minimum critical power ratio at the uprated conditions (Review Standard RS-001). The reviewer should also ensure that the correlations used in the EPU analysis do not exceed their validation range under uprated normal operation and AOO conditions.

The following are two examples of acceptable approaches to meeting this criterion:

A. For departure from nucleate boiling ratio (DNBR), CHFR or CPR correlations, there should be a 95-percent probability at the 95-percent confidence level that the hot rod in the core does not experience a DNB or boiling transition condition during normal operation or AOOs.

B. The limiting (minimum) value of DNBR, CHFR, or CPR correlations is to be established such that at least 99.9 percent of the fuel rods in the core will not experience a DNB or boiling transition during normal operation or AOOs.

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.

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 "TEMP-Thermal Enthalpy Mixing Program," BAW-10021, Babcock and Wilcox Company, April 1970 and "THINC-IC-An Improved Program for Thermal-Hydraulic Analysis Of Rod Bundle Cores," WCAP-7956, Westinghouse Electric Corporation, June 1973, 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 review should include the effects of radial pressure gradients in the core flow distribution. The reviewer should also confirm that calculations of BWR fluid conditions for use in CHF correlations have been made in accordance with the models specified in "Loss of Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," NEDO-10329, Appendix C, General Electric Company, April 1971 and "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.

3. The design should address core oscillations and thermal-hydraulic instabilities as described in SRP Section 15.9.

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 those listed below, these relationships should be confirmed empirically using representative databases from approved reports..

A. reactor vessel ("Reactor Vessel Model Flow Tests," BAW-10037 (nonproprietary version of BAW-10012), Rev. 2, Babcock and Wilcox Company, September 1968)

B. jet pump ("Design and Performance of General Electric Boilng Water Reactor Jet Pumps," APED-5460, General Electric Company, September 1968)

C. core flow distribution (BAW-10037 and "Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello," NEDO-10299, General Electric Company, January 1971, DRAFT Rev. 2, April 1996)

D. void fraction distribution for BWRs

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

6. Preoperational and initial startup test programs should follow the recommendations of Regulatory Guide 1.68, as it relates to measurements and the confirmation of thermal-hydraulic design aspects (Regulatory Guide 1.68).

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 (RG 1.133).

8. The thermal-hydraulic design should account for the effects of crud in the CHF calculations in the core or in the pressure drop throughout the RCS. Process monitoring provisions should assure the capability to detect a 3-percent 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 of TMI Action Plan Item II.F.2 of NUREG-0737. Applicants subject to 10 CFR 50.34(f) should meet the requirements of 10 CFR 50.34(f)(2)(xviii). 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, that incorporate response predictions based on appropriate analyses.

10. Thermal-hydraulic stability performance of the core during an ATWS event should not exceed acceptable fuel design limits. SRP Sections 15.8 and 15.9 describe an acceptable method for performing such an analysis for BWR and PWR cores.

Technical Rationale

The technical rationale for application of these requirements and/or SRP acceptance criteria to the areas of review addressed by this SRP section is discussed in the following paragraphs:

1. GDC 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 that 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. GDC 12 requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations that 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 exceedence 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 protect 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. <u>REVIEW PROCEDURES</u>

The procedures below are used during DC and COL reviews to assure that the design criteria and bases and the preliminary design detailed in the preliminary SAR meet the acceptance criteria given in Subsection II of this SRP section. For operating license (OL) and COL applications, the procedures verify that the final design, as detailed in the final safety analysis report, appropriately implements the initial design criteria and bases. The OL and COL reviews also encompass the proposed technical specifications to assure that they are adequate with regard to safety limits, limiting safety system settings, and conditions of operation.

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review specified in Subsection I of this SRP section, the review procedure is identified below. These review procedures are based on the identified SRP acceptance criteria. For deviations from these specific acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1. The reviewer must understand currently acceptable thermal and hydraulic design practice for the reactor type under review. This understanding can be most readily gained from (1) topical reports describing CHF correlations, system hydraulic models and tests, and core subchannel analysis methods, (2) standard texts and other technical literature which establish the methodology and the nomenclature of this technology, and (3) documents that summarize current staff positions concerning acceptable design methods.

2. Much of the review described below is generic in nature and is not performed for each plant. The reviewer compares the core design and operating parameters to those of previously reviewed plants and then devotes the major portion of the review effort to those areas in which the application is not identical to previously reviewed plants.

3. The reviewer compares the information in the applicant' s SAR or DCD for new plants to the documents referenced by the applicant or included in this SRP section to determine

conformance to the bounds established by such documents. The reviewer confirms that (1) the 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, (2) the analysis methods are used in the manner specified by their developers or in previous staff reviews, (3) the reactor design falls within the ranges of applicability specified for accepted analysis methods, and (4) the design is within the criteria specified in Subsection II, above, and is not an unexplained or unwarranted extrapolation of other thermal-hydraulic designs.

4. The reviewer evaluates the analytical methods used in the thermal-hydraulic analysis, including the applicability of the codes and correlations used and the uncertainty analysis methodologies implemented. For transient analysis, the setpoint limits and instrumentation uncertainty values used for establishing steady-state conditions preceding transient initiation should be evaluated to ensure appropriate conservatism. The review examines the method of employing peaking factors and hot channel factors in the thermal-hydraulic analysis. The basis for the input parameters used in the uncertainty evaluation and the resulting uncertainty in reactor thermal-hydraulic parameters should also be evaluated.

5. The reviewer does not routinely evaluate calculations. 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 setpoint 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 using the Westinghouse revised thermal design procedure ("Revsied Thermal Design Procedure," WCAP-11397-P-A, Westinghouse Electric Corporation, July 1975). On occasion (e.g., if a new design or new design method is proposed), the staff or consultants, under the direction of the primary review organization, perform independent analyses. These analyses verify the design or establish the range of applicability and associated accuracy of the new method; the reviewer ensures it is applied accordingly.

6. The reviewer evaluates the functional requirements for instrumentation used in monitoring those thermal-hydraulic parameters important to safety, such as in-core power distribution and coolant temperature measurements. Chapter 7 of the SAR or DCD for new plants and the review requirements in SRP Section 7 should detail the instrumentation design and logic.

7. The reviewer evaluates the design of software used in core protection systems and establishes its acceptability by comparing it with previously approved designs and assessing any differences with regard to system performance and safety functions effects. 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. The reviewer bases confirmation of adequate software implementation 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 to the predictions of the thermal-hydraulic design analyses. The reviewer should consult with the organization responsible for the review of the design acceptability of the hardware portion of the core protection systems.

8. The reviewer establishes that the thermal-hydraulic design and its characterization by minimum critical heat flux ratio (MCHFR) or DNBR have been accomplished and are presented in a manner that accounts for all possible reactor operating states as determined from operating maps. In this regard, the reviewer confirms that the power distribution assumptions of SAR or DCD Section 4.4 are a conservative (i.e., worst-case) accounting of the power distributions derived in SAR or DCD Section 4.3 from core physics analyses and that the latter analyses include an acceptable calculation of local void fractions. The reviewer also confirms that the mass flux used in these calculations accounts for the core flow distribution (including that for partial loop operation and natural circulation effects in the case of ESBWR) and the worst case of core bypass flow. The reviewer confirms that startup measurements will verify the primary coolant flow range shown in the operating map.

9. The applicable reviewer considers the design review areas of applicability associated with ATWS and thermal-hydraulic instability using the guidance found in the requirements of SRP Sections 15.8 and 15.9.

10. 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 the acceptability of such a mode of operation based on the applicant's safety analyses and proposed technical specifications (Generic Letter No. 82-28). Plant-specific aspects of the safety analyses may identify safety questions which 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. When performing review of thermal-hydraulics instabilities resulting from (N-1) loop and other operational circumstances, the reviewer should use the guidance found in the requirements of SRP Section 15.9. The reviewer should also verify that the applicant has addressed the possibility for jet pump vibration during (N-1) loop operation. For advanced BWR applicants proposing to operate with less than the maximum number of 10 reactor internal pumps (RIPs), the reviewer confirms that continued plant operation with fewer than 10 RIPs in operation including any reactor power level restrictions, is compatible with the plant safety analyses and flow test results do not demonstrate significant differences in core flow patterns.

11. 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.

12. The reviewer examines the calculation of hydraulic loads for normal operations, including AOOs, to ensure that 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). The reviewer will also provide calculations for design-basis accident conditions. The review of the adequacy of components and structures under design-basis accident loads is performed under SRP Section 3.9. The review to determine that a coolable core geometry is maintained is performed under SRP Section 4.2.

13. The reviewer should ensure that an adequate loose parts monitoring system is provided. For COL applications, the design criteria for the system and the types, locations, and methods of mounting for all intended sensors should be reviewed. The reviewer should

compare the design to Regulatory Guide 1.133, equipment used, and application experience on comparable plants.

OL and COL reviews consist of a more complete description of the loose parts monitoring system, including sensitivity specifications and operating procedures. The reviewer should ensure that operating procedures and training provisions are adequate to fully use the system potential for loose parts detection. The review of the adequacy of staff training is performed under SRP Section 13.2.

14. The reviewer should evaluate the vibration monitoring equipment and procedures to ensure that they are adequate for the plant under review based on the experience of comparable plants. The reviewer will evaluate the application of neutron monitoring sensors for core vibration test analysis. The organization responsible for review of system design examines the preoperational vibration test program under SRP Section 3.9 and provides technical consultation to the primary organization reviewer on the need for permanent vibration monitoring provisions for the plant under review.

15. The reviewer ensures that applicants have an acceptable program for incorporating instrumentation and procedures for detection and recovery from conditions of ICC that meets the requirements of TMI Action Plan Item II.F.2 of NUREG 0737 and applicants subject to 10 CFR 50.34(f) should meet the requirements of 10 CFR 50.34(f)(2)(xviii) (Ref. 1) as follows:

    A. The reviewer verifies that the applicant has provided preliminary design information on selected instrumentation components and specified the design concept selected for the instrumentation in accordance with the guidance of item II.F.2 of NUREG-0737.

    B. The reviewer ensures that the applicant complies with the documentation requirements and design requirements described in item II.F.2 of NUREG-0737. Generic Letter 82-28 describes acceptable PWR ICC instrumentation. Generic Letter 84-23 describes acceptable BWR ICC instrumentation. 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.

    C. The reviewer consults with the organization responsible for the review of the design acceptability of the ICC instrumentation and displays. The reviewer also consults with the organization responsible for the review of the acceptability of guidelines and procedures for recognition and response to ICC conditions.

16. For new plant applicants and those PWRs subject to Generic Letter No. 88-17, 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 thermodynamic and physical states, such as a rapid boron dilution event during shutdown conditions (NUREG-1449) 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.

Similarly, for ESBWR systems where core flow and flow distribution is directly related to core power, the thermal and hydraulic analysis should provide a basis for shutdown procedures, equipment response, and operator response.

17. The reviewer determines whether the applicant's proposed preoperational and initial startup test programs are consistent with the intent of Regulatory Guide 1.68. The reviewer assures that the applicant has provided sufficient information to clearly identify the test objectives, methods of testing, and acceptance criteria.

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

    The reviewer evaluates the proposed test programs to determine whether they provide reasonable assurance that the core and RCS will satisfy functional requirements. As an alternative to this detailed evaluation, the reviewer may compare the core and RCS design to that of previously reviewed plants. If the design is essentially identical and 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 RCS.

    If the core or the RCS 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 COL stage. This effort should particularly evaluate the need for any special design features required to perform acceptable test programs.

18. The reviewer evaluates the proposed technical specifications that relate to the core and the RCS. This evaluation covers all 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 RCS, do not permit operation with any expected combination of parameters that would not satisfy specific acceptance 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.

19. For reviews of DC and COL applications under 10 CFR Part 52, the reviewer should follow the above procedures to verify that the design set forth in the safety analysis report, and if applicable, site interface requirements meet the acceptance criteria. For DC applications, the reviewer should identify necessary COL action items. With respect to COL applications, the scope of the review is dependent on whether the COL applicant references a DC, an ESP or other NRC-approved material, applications, and/or reports.

    After this review, SRP Section 14.3 should be followed for the review of Tier I information for the design, including the postulated site parameters, interface criteria, and ITAAC.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information in the SAR and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

The following paragraph is applicable to DC:

> The thermal-hydraulic design of the core for the __________ plant was reviewed. The scope of review included the design criteria, preliminary core design, and 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 the applicant satisfactorily justified all such differences The applicant performed its thermal-hydraulic analyses using analytical methods and correlations that have been previously reviewed by the staff and found to be acceptable.

For OL and COL applications, 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 Criteria 10 and 12 of Appendix A to 10 CFR Part 50 and is acceptable for final design approval. The staff also concludes that the reactor core and associated coolant, control, and protection systems have 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 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. The staff also concludes that the design of the loose parts monitoring program is consistent with the guidance of Regulatory Guide 1.133 and is therefore, acceptable. The staff has reviewed the instrumentation for the detection of inadequate core cooling and concluded that it complies with the requirements of item II.F.2 of NUREG-0737 and is therefore acceptable.

For DC and COL reviews, the findings will also summarize (to the extent that the review is not discussed in other SER sections) the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable, and interface requirements and combined license action items relevant to this SRP section.

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate 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, unless superceded by a later revision.

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

VI. REFERENCES

1. 10 CFR 50.34(f), "Additional TMI-Related Requirements," paragraph 10 CFR 50.34(f)(2)(xviii).

2. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

3. 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."

5. Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," August 1978.

6. Regulatory Guide 1.133, Rev. 1, "Loose Parts Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981, ADAMS Accession No. ML003740137.

7. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

8. 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.

9. 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.

10. 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.

11. 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.

12. 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.

13. 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.

14. B.S. Mullanax, R.J. Walker, and B.A. Karrasch, "Reactor Vessel Model Flow Tests," BAW-10037 (nonproprietary version of BAW-10012), Rev. 2, Babcock and Wilcox Company, September 1968.

15. “Design and Performance of General Electric Boiling Water Reactor Jet Pumps,” APED-5460, General Electric Company, September 1968.

16. H.T. Kim, “Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello,” NEDO-10299, General Electric Company, January 1971. DRAFT Rev. 2, April 1996, 4.4-16.

17. H. Chelemer, L.H. Boman, and D.R. Sharp, “Revised Thermal Design Procedure,” WCAP-11397-P-A, Westinghouse Electric Corporation, July 1975.

18. Review Standard RS-001, “Review Standard for Extended Power Uprates,” RS-001, Revision 0, December 2003.

19. NRC Inspection Manual Chapter IMC-2504, “Construction Inspection Program—Non-ITAAC Inspections,” April 25, 2006.

20. “TEMP—Thermal Enthalpy Mixing Program,” BAW-10021, Babcock and Wilcox Company, April 1970.

21. 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.)

22. 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.

23. 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.

SRP Section 4.4
Description of Changes

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in (Draft) Revision 2, dated April 1996 of this SRP. See ADAMS accession number ML041690061.

In addition this SRP section was administratively updated in accordance with NRR Office Instruction, LIC-200, Revision 1, "Standard Review Plan (SRP) Process." The revision also adds standard paragraphs to extend application of the updated SRP section to prospective submittals by applicants pursuant to 10 CFR Part 52.

The technical changes are incorporated in Revision 2, dated [Month] 2007:

Review Responsibilities - Reflects changes in review branches resulting from reorganization and branch consolidation. Change is reflected throughout the SRP.

I. AREAS OF REVIEW

1. Added the area for review of uncertainty analysis methodology and uncertainties of variables and correlations relevant to the thermal and hydraulic design.

2. Deleted the area for review of the thermal-hydraulic design related to the reactor stability and ATWS. Separate SRPs exist for those conditions—SRP Section 15.8 for ATWS and SRP Section 15.9 for stability.

II. ACCEPTANCE CRITERIA

1. Explicitly described the acceptance criterion regarding uncertainty parameters and evaluation methodology.

2. Added the acceptance criterion for the thermal and hydraulic design following core design and operating changes resulting from extended power uprates.

3. Deleted examples for typical values of CPR (BWRs) and DNBR (PWRs).

4. Added void fraction distribution for BWRs for acceptance criterion 4 because of its significance on core power distribution, subchannel flow rate, and the like.

III. REVIEW PROCEDURES

1. Added another review procedure to ensure review of analytical models, uncertainty evaluation methodology, hot spots, hot channel factors, and the like.

2. Added a review procedure for evaluation of the functional requirements for instrumentation used in monitoring those thermal-hydraulic parameters important to safety, such as in-core power distribution and coolant temperature measurements.

3. Added the second paragraph to remind the reviewer that core flow is dictated by core power levels in ESBWR systems, and that thermal and hydraulic analysis should provide a basis for shutdown procedure, equipment response, and operator response.

4. Added a review procedure for DC and COL applications under 10 CFR Part 52.

IV. <u>EVALUATION FINDINGS</u>

None

V. <u>IMPLEMENTATION</u>

None

VI. <u>REFERENCES</u>

1. Removed references that are not explicitly referenced in this SRP section.

2. Added references for extended power uprates and NRC inspection manual.