U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

4.4 THERMAL AND HYDRAULIC DESIGN

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - Instrumentation and Control System Branch (ICSB)
Human Factors Engineering Branch (HFEB)
Procedures and Test Review Branch (PTRB)

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; 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 CHF or CPR 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.

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

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 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).

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.

II. ACCEPTANCE CRITERIA

The CPB acceptance criteria are based on meeting the relevant requirements of General Design Criterion 10 (Ref. 1), 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).

Specific criteria necessary to meet the requirements of GDC 10 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) 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) is typically 1.30 using the BAW-2 correlation (Ref. 6) or the W-3 correlation (Ref. 7). 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 and 9, 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 Reference 10 and 11.

3. The reactor should be demonstrated 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 for anticipated operational occurrences.

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).

b. Jet pump (Ref. 13).

c. Core flow distribution (Refs. 12 and 14).

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 acceptance criterion 1, above, is satisfied.

6. Preoperational and initial startup test programs should follow the recommendations of Regulatory Guide 1.68 (Ref. 15), 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).

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 capability for detection of a three percent pressure 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 item II.F.2 NUREG-0718 (Ref. 17) and NUREG-0737 (Ref. 18). Procedures for detection and recovery from conditions of ICC must be consistent with technical guidelines that incorporate response predictions based on appropriate analyses.

III. <u>REVIEW PROCEDURES</u>

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 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). 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. 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 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 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, 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 will also provide consultation to RSB upon request, regarding calculations for postulated accident conditions. MEB reviews the adequacy of components and structures under accident loads (SRP Section 3.9.2) and CPB 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 and to equipment used and application experience on comparable plants.

At the OL level; a more complete description of the 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 Branch (OLB) will provide consultation on staff training in accordance with the SRP Section 13.2.

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 will evaluate the application of neutron monitoring sensors for core vibration test analysis. The MEB is responsible for review of the preoperational vibration test program, as described in SRP Section 3.9.2, and provides technical consultation to CPB 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. At the CP stage, the applicant must provide preliminary design information on selected instrumentation components and must specify the design concept selected for development instrumentation in accordance with the requirements of item II.F.2 of NUREG-0718 (Ref. 17).

At the OL stage, 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). The reviewer consults with ICSB and HFEB concerning the design acceptability of the instrumentation and displays and with the Reactor Systems Branch (RSB) and PTRB concerning the acceptability of guidelines and procedures for recognition and response to inadequate core cooling conditions.

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). 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.)

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 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.

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 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, 10 CFR Part 50. The core has been designed with appropriate margin to assure that acceptable fuel design limits are not exceeded during steady-state operation or anticipated operational occurrences. 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

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

with Regulatory Guide 1.68 to measure and confirm the thermal-hydraulic design aspects. The loose parts and vibration monitoring system is designed for compliance with the requirements 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, 10 CFR Part 50 and is acceptable for final design appoval. We also conclude that the reactor core has been design with appropriate margin to assure that acceptable fuel design limits are not exceeded during steady-state operation or anticipated operational occurrences 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 for compliance with the requirements 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.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plan for using this SRP section.

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.

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

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."

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

3. 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.

4. 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.

5. 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.)

6. 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.)

7. 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. "TEMP - Thermal Enthalpy Mixing Program," BAW-10021, Babcock and Wilcox Company, April 1970.

9. 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.)

10. 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.

11. 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. 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.

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

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

15. Regulatory Guide No. 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

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

17. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits."

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

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

APPENDIX
STANDARD REVIEW PLAN 4.4

INDEPENDENT AUDIT ANALYSIS

(Appendix to SRP Section 4.4 has been deleted)