Proposed - For Interim Use and Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

4.4 THERMAL AND HYDRAULIC DESIGN

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of thermal and hydraulic design for Light Water Reactors

Secondary - None

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. Design Specific Review Standard (DSRS) Section 4.4 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) 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 the calculations upon which the technical specifications are based are adequately supported by tests and experiments applicable to the mPower™ design and consistent with 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.

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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 hold down requirements.

5. The review evaluates the uncertainty analysis methodology and the uncertainties of variables and correlations such as CHF. 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), 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. <u>Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)</u>. For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this DSRS section in accordance with DSRS Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this DSRS section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with DSRS Section 14.3.

9. <u>COL Action Items and Certification Requirements and Restrictions</u>. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

   For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

<u>Review Interfaces</u>

Other DSRS 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 DSRS

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 thermal and hydraulic performance of the reactor coolant system during postulated loss-of-coolant accidents conditions is performed under DSRS Section 15.6.5.

3. The review of anticipated transients without scram (ATWS) is performed under DSRS Section 15.8.

4. The review of the adequacy of components and structures under accident loads and the preoperational vibration test program is performed under DSRS Sections 3.9.3 and 3.9.6.

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

6. The review of inadequate core cooling (ICC) monitoring system hardware to determine compliance with the requirements applicable to information systems important to safety is performed under DSRS Section 7.5.

7. The review of the applicant's training program is performed under DSRS Sections 13.2.1 and 13.2.2.

8. 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 ICC conditions, is performed under DSRS Section 13.3.

9. The review of the human factors aspects of information displays is performed under DSRS Chapter 18.

10. For new plant applicants, the review of shutdown risk assessment is performed under SRP Chapter 19.

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).

II. <u>ACCEPTANCE CRITERIA</u>

<u>Requirements</u>

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.

3. Section 52.47(b)(1) Title 10 of the Code of Federal Regulations (10 CFR 52.47(b)(1)), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Atomic Energy Act (AEA), and the U.S. Nuclear Regulatory Commission's (NRC) regulations.

4. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the AEA, and the NRC's regulations.

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are set forth below. The DSRS is not a substitute for the NRC's regulations, and compliance with it is not required. Identifying the differences between this DSRS section and the design features, analytical techniques, and procedural measures proposed for the facility, and discussing how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria, is sufficient to meet the intent of 10 CFR 52.47(a)(9), "Contents of applications; technical information." The same approach may be used to meet the requirements of 10 CFR 52.79(a)(41) for COL applications.

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

1. DSRS 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 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.

The following is an example of acceptable approaches to meeting this criterion:

A. For departure from nucleate boiling ratio (DNBR) 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 condition during normal operation or AOOs.

Correlations of CHF 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 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 "VIPRE-01: A Thermal-Hydraulic Code for Reactor Core," NP-2511-CCM-A, Electric Power Research Institute, August 1989, should be used to calculate local fluid conditions within fuel assemblies for use in 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.

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

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 (e.g., "Reactor Vessel Model Flow Tests," BAW-10037 (nonproprietary version of BAW-10012), Revision 2, Babcock and Wilcox Company, September 1968).

   B. Core flow distribution (e.g., BAW-10037).

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 (RG) 1.68, as it relates to measurements and the 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 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.

11. Thermal-hydraulic stability performance of the core during an ATWS event should not exceed acceptable fuel design limits. DSRS Sections 15.8 and 15.9.A describe an acceptable method for performing such an analysis for the mPower™ core.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this DSRS 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. REVIEW PROCEDURES

The COL review also encompasses the proposed technical specifications to assure that they are adequate with regard to safety limits, limiting safety system settings, and conditions of operation.

The review procedures described below are based on the identified DSRS acceptance criteria. For deviations from these specific acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

For DC and COL applications submitted under Part 52, the level of information reviewed should be consistent with that of a final safety analysis report (FSAR) submitted in an operating license (OL) application. However, verification that the as-built facility conforms to the approved design is performed through the ITACC process.

1. Programmatic Requirements - In accordance with the guidance in NUREG-0800 "Introduction," Part 2 as applied to this DSRS Section, the staff will review the programs proposed by the applicant to satisfy the following programmatic requirements. If any of the proposed programs satisfies the acceptance criteria described in Subsection II, it can be used to augment or replace some of the review procedures. It should be noted that the wording of "to augment or replace" applies to nonsafety-related risk-significant SSCs, but "to replace" applies to nonsafety-related nonrisk-significant SSCs according to the "graded approach" discussion in NUREG-0800 "Introduction," Part 2. Commission regulations and policy mandate programs applicable to SSCs that include:

    A. Maintenance Rule Standard Review Plan (SRP) Section 17.6 (DSRS Section 13.4, Table 13.4, Item 17, RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." and RG 1.182; "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants".

    B. Quality Assurance Program SRP Sections 17.3 and 17.5 (DSRS Section 13.4, Table 13.4, Item 16).

    C. Technical Specifications (DSRS Section 16.0 and SRP Section 16.1) – including brackets value for DC and COL. Brackets are used to identify information or characteristics that are plant specific or are based on preliminary design information.

    D. Reliability Assurance Program (SRP Section 17.4).

    E. Initial Plant Test Program (RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants,"DSRS Section 14.2, and DSRS Section 13.4, Table 13.4, Item 19).

    F. ITAAC (DSRS Chapter 14).

2. In accordance with 10 CFR 52.47(a)(8),(21), and (22), for new reactor license applications submitted under Part 52, the applicant is required to (1) address the proposed technical resolution of unresolved safety issues (USIs) and medium- and high-priority generic safety issues (GSIs) that are identified in the version of NUREG-0933

current on the date six months before application and that are technically relevant to the design; (2) demonstrate how the operating experience insights have been incorporated into the plant design; and, (3) provide information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island (TMI) requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v). Reference: 10 CFR 52.47(a)(21), 10 CFR 52.47(a)(22) , and 10 CFR 52.47(a)(8), respectively. These cross-cutting review areas should be addressed by the reviewer for each technical subsection and relevant conclusions documented in the corresponding SER section.

3. 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.

4. 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.

5. The reviewer compares the information in the applicant's safety analysis report (SAR) or design control document (DCD) for new plants to the documents referenced by the applicant or included in this DSRS 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.

6. 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.

7. 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. On

occasion (e.g., if a new design or new design method is proposed), the reviewer performs 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.

8. 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 DSRS Section 7 should detail the instrumentation design and logic.

9. 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.

10. The reviewer establishes that the thermal-hydraulic design and its characterization by minimum 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 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.

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

12. For mPower™ applicants proposing operation with one of the reactor coolant pumps out of operation (i.e., (N-1) pump 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) pump 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 minimum critical heat flux ratio. When performing review of thermal-hydraulics instabilities resulting from (N-1) pump and other operational circumstances, the reviewer should use the guidance found in the requirements of DSRS Section 15.9.A. The reviewer should also verify that the applicant has addressed the possibility for pump

vibration during (N-1) pump operation. For mPower™ applicants proposing to operate with less than the maximum number of eight reactor coolant pumps (RCPs), the reviewer confirms that continued plant operation with fewer than eight RCPs 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.

13. 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.

14. 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 (DSRS 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 DSRS Sections 3.9.3 and 3.9.6. The review to determine that a coolable core geometry is maintained is performed under DSRS Section 4.2.

15. 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 RG 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 DSRS Sections 13.2.2.

16. 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 DSRS Sections 3.9.3 and 3.9.6 and provides technical consultation to the primary organization reviewer on the need for permanent vibration monitoring provisions for the plant under review.

17. 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) 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.

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.

18. For mPower^{TM}, 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, such as Generic Letter No. 88-17 and NUREG-1449, and should encompass thermodynamic and physical states 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.

19. The reviewer determines whether the applicant's proposed preoperational and initial startup test programs are consistent with the intent of RG 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.

20. 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.

21. For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the FSAR meets the acceptance criteria. DCs have referred to the FSAR as the DCD. The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

    For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

22. For review of both DC and COL applications, DSRS Section 14.3 should be followed for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the staff's technical review and analysis, as augmented by the application of programmatic requirements in accordance with the staff's technical review approach in the DSRS Introduction, 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.

1. 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.

2. 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 GDC 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 RG 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 RG 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.

3. For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this DSRS section.

4. In addition, to the extent that the review is not discussed in other SER sections, the findings will summarize the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

V. IMPLEMENTATION

The staff will use this DSRS section in performing safety evaluations of mPower™-specific DC, or COL, applications submitted by applicants pursuant to 10 CFR Part 52. The staff will use the method described herein to evaluate conformance with Commission regulations.

Because of the numerous design differences between the mPower™ and large light-water nuclear reactor power plants, and in accordance with the direction given by the Commission in SRM- COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010 (ML102510405), to develop risk-informed licensing review plans for each of the small modular reactor (SMR) reviews including the associated pre-application activities, the staff has developed the content of this DSRS section as an alternative method for mPower™ -specific DC, or COL submitted pursuant to 10 CFR Part 52 to comply with 10 CFR 52.47(a)(9), "Contents of applications; technical information."

This regulation states, in part, that the application must contain "an evaluation of the standard plant design against the SRP revision in effect six months before the docket date of the application." The content of this DSRS section has been accepted as an alternative method for complying with 10 CFR 52.47(a)(9) as long as the mPower™ DCD FSAR does not deviate significantly from the design assumptions made by the NRC staff while preparing this DSRS section. The application must identify and describe all differences between the standard plant design and this DSRS section, and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria. If the design assumptions in the DC application deviate significantly from the DSRS, the staff will use the SRP as specified in 10 CFR 52.47(a)(9). Alternatively, the staff may supplement the DSRS section by adding appropriate criteria in order to address new design assumptions. The same approach may be used to meet the requirements of 10 CFR 52.79(a)(41), and COL applications.

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.”

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