U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

14.3.4 REACTOR SYSTEMS - INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of iPWR reactor systems

Secondary - None

I. AREAS OF REVIEW

This section provides technical review guidance for both evolutionary and passive safety-system designs to ensure that (1) the top-level design information regarding the reactor systems in the design control document (DCD) Tier 2 is appropriately included in Tier 1 and (2) appropriate inspections, tests, analyses, and acceptance criteria (ITAAC) are developed for each top-level system, structure, and component (SSC) within the scope of reactor systems to ensure acceptability of the as-built facility to meet the requirements of Title 10, Code of Federal Regulations (10 CFR), Part 52. The scope of ”reactor systems” encompasses the reactor core, fuel, control rods, reactor vessel, reactor coolant system, loose parts monitoring system, and emergency core cooling systems (active and passive) that are significantly related to normal operation, transients, and accidents.

The Tier 1 design certification material as submitted by the applicant in its DCD includes the top-level design features and performance standards that pertain to the safety of the plant and include descriptive text and supporting figures. The top-level design features and performance standards are those that are most important to safety, including safety-related and defense-in-depth features and functions, and non-safety-related systems that potentially impact safety. The Tier 1 information is derived from Tier 2. (See Appendix A of Design Specific Review Standard (DSRS) 14.3 for definitions of Tier 1 and Tier 2). In general, many of the reactor and core cooling systems are classified as safety-related, and therefore, many of the characteristics and features of these systems are judged to have safety significance. This is reflected in a relatively higher level of detail in Tier 1 for these systems than other systems of the standard design. Thus, the Tier 1 portion of the DCD as derived from Tier 2 information is the focus for this review for the aforementioned SSCs for Reactor Systems identified as Tier 1.

ITAAC include (1) design commitments; (2) identification of those inspections, tests, and analyses (observations, tests, or examinations) to determine if the commitment was met; and (3) acceptance criteria that demonstrate that the design commitment was, in fact, met. Successful completion of all ITAAC will demonstrate that the plant was constructed in accordance with a certified design, regulations, and the license.

The specific fuel, control rod and core designs presented in Tier 2 will constitute an approved design that may be used for the combined operating license (COL) first cycle core loading,

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without further U.S. Nuclear Regulatory Commission (NRC) review. If any other core design is requested for the first cycle, the COL applicant or licensee will be required to submit for staff review that specific fuel, control rod and core design analyses. No ITAAC are required for Tier 1 information in the fuel, control rod, and core design areas because of the requirement for prior NRC approval of any proposed changes to the approved design. Post-fuel-load testing programs (e.g., startup and power-ascension testing) verify that the actual core performs in accordance with the analyzed core design.

The specific areas of review are as follows:

1. Tier 1 information identified as such in the DCD and the process by which the applicant identified this information from its Tier 2 SSC descriptions. Tier 1 should include those SSCs that could affect the operation of the reactor and core cooling systems [e.g., the following chapters of the DSRS: Chapter 4–Reactor, Chapter 5–Reactor Coolant Systems and Connected Systems, Chapter 6–Section 6.3 on Emergency Core Cooling Systems, Chapter 9–Auxiliary Systems, Chapter 15–Transients and Accidents Analyses].

2. The design features and functions of those SSCs for the reactor and core cooling systems determined to be safety-significant from probabilistic risk assessment (PRA) insights and other sources.

3. Those systems that might be classified as non-safety-related by the designer or applicant but are important to safety or otherwise provide defense-in-depth functions.

4. Policy, technical, and licensing issues for evolutionary and passive designs as identified by NRC generically and for a given design, including, as an example, the use of design acceptance criteria (DAC), for a limited set of technical issues, as acceptance criteria for ITAAC.

5. ITAAC format and content.

6. For a DC application:

   A. The staff reviews 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 plant that incorporates the design certification is built and will operate in accordance with the design certification, the Atomic Energy Act (AEA), and the U.S. Nuclear Regulatory Commission (NRC) regulations.

   B. The staff reviews the justification that compliance with the interface requirements is verifiable through ITAAC. The staff also reviews the method that is to be used for verification of the interface requirements.

7. For a COL application:

   A. The staff reviews 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, the facility has been constructed and will operate in conformity with the combined license, the AEA, and the NRC regulations.

B. If the application references a standard design certification, the staff verifies that the ITAAC contained in the certified design apply to those portions of the facility design that are approved in the design certification.

8. COL Action Items and Certification Requirements and Restrictions. For a Design Certification (DC) application, the review will also address combined license (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.

Review Interfaces

Other DSRS sections interface with this section as follows:

1. The identification of those design features and functions of the SSCs that should be addressed in Tier 1 based on severe accident, PRA, and shutdown safety evaluations, respectively, is evaluated and determined under DSRS Section 19.

2. DSRS Section 14.3 provides general guidance on ITAAC information.

3. Acceptability of ITAAC information regarding the ability of SSCs to withstand various natural phenomena is reviewed under DSRS Section 14.3.2.

4. Acceptability of ITAAC information for piping design is reviewed under DSRS Section 14.3.3.

5. Acceptability of ITAAC information for Instrumentation and Controls is reviewed under DSRS Section 14.3.5.

6. Acceptability of ITAAC information for electrical systems and components is reviewed under DSRS Section 14.3.6.

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. 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 plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the AEA, and the NRC's regulations.

2. 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.

<u>DSRS Acceptance Criteria</u>

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.

1. Appendix A of DSRS 14.3 describes and provides guidance relative to the content of the DCD for a design certification application and defines Tier 1 and Tier 2 design-related information that is to be ultimately incorporated by reference into the design certification rules. The basis for identifying Tier 1 information as derived from Tier 2 information, which is essentially the same information as is required for a design certification application, is that the top-level design features and performance standards (Tier 1) are those that are most important to safety, including safety-related and defense-in-depth features and functions, and non-safety-related systems that potentially impact safety.

   Tier 1 should be reviewed to verify that plant safety analyses, such as for core cooling, transients, overpressure protection, steam generator tube rupture, and anticipated transients without scram (ATWS), are adequately addressed. Applicants should provide tables in DCD Tier 2 Section 14.3 to show how the important input parameters used in the transient and accident analyses for the design are verified by the ITAAC. For intersystem loss-of-coolant accidents, the design pressure of the piping of the systems that interface with the reactor coolant pressure boundary should be specified in the design descriptions or figures.

   The specific fuel, control rod, and core designs presented in Tier 2 constitute an approved design that may be used for the COL first-cycle core loading without further NRC staff review. If any other core design is requested for the first cycle, the COL applicant or licensee will be required to submit for staff review those specific fuel, control rod, and core design analyses as described in DCD Tier 2 Chapters 4, 6, and 15. Much of the detailed supporting information in Tier 2 for the nuclear fuel, fuel channel, and control rods, if considered for a change by a COL applicant or licensee that references the certified standard design, would require prior NRC approval. Therefore, for the evolutionary and passive designs, the staff concluded that this information should be designated as Tier 2* information (see Appendix A of DSRS Section 14.3 for a definition). However, staff will allow some of the Tier 2* designations to expire after the first full-power operation of the facility when the detailed design has been completed and the core performance characteristics are known from the startup and power-ascension test programs. The NRC bears the final responsibility for designating which material in Tier 2 is Tier 2*.

The following issues are identified to ensure comprehensive and consistent treatment of Tier 1 based on the safety significance of the system being reviewed:

A. System purpose and functions

B. Location/functional arrangement of system

C. Key design features of the system

D. System operation in various modes

E. Seismic and American Society of Mechanical Engineers (ASME) code classifications

F. Materials—weld quality and pressure-boundary integrity

G. Controls, alarms, and displays

H. Logic

I. Interlocks

J. Class 1E electrical power sources and divisions

K. Equipment to be qualified for harsh environments

L. Valve qualification and operation

M. Interface requirements with other systems

N. Numeric performance values (flow rates, capacities, etc.)

O. Accuracy and quality of figures

P. Active systems that provide defense-in-depth functions designated as non-safety systems

Appendix C to DSRS 14.3 provides ”checklists” for the fluid systems as an aid for establishing consistency and comprehensiveness in the review of the system.

2. The source of information used to determine safety significance of SSCs for the design of reactor and core cooling systems include applicable rules and regulations, general design criteria, unresolved safety issues, and generic safety issues, NRC generic correspondence, PRA, insights from the standard design’s safety and severe accident analyses, and operating experience.

Inputs from the PRA review, including shutdown safety evaluations, and severe accident analyses ensure important insights and design features from these analyses are incorporated into Tier 1. For both PRA and severe accident analyses, although large uncertainties and unknowns may be associated with the event phenomena, design features important for severe accident prevention and mitigation resulting from these analyses should be selected for treatment in Tier 1.

3. The passive-designed reactors use safety systems that employ passive means (natural forces), such as gravity, natural circulation, condensation and evaporation, and stored energy, for accident mitigation. These designs also include active systems that provide defense-in-depth capabilities for reactor-coolant makeup and decay heat removal. These active systems are the first line of defense to reduce challenges to the passive systems in the event of transients or plant upsets. SECY-95-132, ”Policy and Technical Issues Associated with the regulatory treatment of non-safety systems (RTNSS) in passive plant designs (SECY-94-084)” provides certain guidance and positions for ensuring consistent and complete treatment of those systems that might be classified as non-safety-related by the designer or applicant but are important to safety or otherwise provide defense-in-depth functions.

4. Applicable regulatory guidance from the Commission for selected policy and technical issues related to particular design should be followed. For the severe accident analyses, the basis for the staff's review for the evolutionary and passive standard designs was the Commission guidance related to SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements.” SECY-93-087, ”Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs” generically presents guidance and NRC positions on evolutionary and passive LWR design certification issues. For guidance, positions, and issues related to specific designs, guidance is available in such documents as SECY-97-044, ”Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design” or SECY-92-327, ”Reviews of Inspections, Test, Analyses, and Acceptance Criteria (ITAAC) Requirements for the General Electric (GE) Advanced Boiling Water Reactor (ABWR).” Regarding DAC, SECY-02-0059, ”Use of Design Acceptance Criteria for the AP1000 Standard Plant Design,” presents staff conclusions on acceptable use of DAC for instrumentation and control (I&C), control room, and piping design areas, contingent upon Westinghouse’s and the staff’s agreeing on adequate DAC during the design certification review. In SECY-92-053, ”Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews ,” the staff noted that DAC is defined as ”a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies, in a limited number of technical areas, in making a final safety determination to support a design certification.”

   In some instances, an applicant may employ DAC to provide the staff with information to support its safety determination process. In SECY-92-053, the staff noted ”the concept of DAC would enable the staff to make a final safety determination, subject only to satisfactory design implementation and verification by the COL licensee through appropriate use of ITAAC.” The staff defined DAC as ”a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies, in a limited number of technical areas, in making a final safety determination to support a design certification. The DAC are to be objective (measurable, testable, or subject to analysis using pre-approved methods), and must be verified as part of the ITAAC performed to demonstrate that the as-built facility conforms to the certified design. That is, the acceptance criteria for DAC become the acceptance criteria for ITAAC, which are part of the design certification.” The use of DAC by applicants use for I&C is considered acceptable given the rapidly changing technology for digital I&C systems.

For many of the design features, it might be impractical to test their functionality because of the absence of simulated severe accident conditions. An example might be the ability of the reactor cavity to absorb the heat and radiation effects of a molten core. Consequently, the existence of the feature on a figure, subject to a basic configuration walkdown and confirmatory test reports or analysis, may be considered sufficient Tier 1 treatment. Another example in which passive designs would be difficult to verify prior to fuel loading as related to normal operations involves natural circulation. Passive designs, compared to previous designs, can include elongated-reactor-core designs to create the pressure differential for establishing natural circulation. Evidence of prior testing and analysis providing conclusive results may have to suffice for suitable acceptance criteria for ITAAC purposes.

5. Appendix D of DSRS 14.3 lists acceptable “Standard ITAAC Entries” in the standard three-column format for ITAAC entries for configuration of systems, hydrostatic tests, net positive suction head for pumps, divisional power supply, etc., that should be contained in the overall set of ITAAC entries, as appropriate.

   Regulatory Guide (RG) RG 1.206, ”Combined License Applications for Nuclear Power Plants (LWR Edition),” contains guidance for developing ITAAC assuming that a COL applicant does not reference a certified design and/or an early site permit (ESP). Guidance in Section III for COLs referencing a certified design notes that the ITAAC contained in the certified design must apply to those portions of the facility design that have been approved. Appendix C.II.2-A provides ”general ITAAC development guidance” on fluid, I&C, and electrical systems.

6. In accordance with 10 CFR 20.1406 applications must describe how contamination and generation of radioactive waste are minimized. RG 4.21 provides guidance for meeting these requirements. RG 4.21 describes an acceptable method for demonstrating compliance with 10 CFR 20.1406. In association with RG 4.21, DC/COL-ISG-06 provides further clarification of the evaluation and acceptance criteria used to meet the requirements of 10 CFR 20.1406 and the guidelines of RG 4.21.

7. 10 CFR 52.47(b)(1) specifies that the application of a DC should contain proposed ITAAC for SSCs necessary and sufficient to assure the plant is built and will operate in accordance with the DC. 10 CFR 52.97(b) specifies that the COL identifies the ITAAC for SSCs necessary and sufficient to assure that the facility has been constructed and will be operated in conformity with the license. DSRS 14.3 provides guidance for reviewing the ITAAC. The requirements of 10 CFR 52.47(b)(1) and 10 CFR 52.97(b) will be met, in part, by identifying inspections, tests, analyses, and acceptance criteria of the top-level design features of the reactor systems and components in the DC application and the COL, respectively.

8. Programmatic requirements. Commission regulations and policy mandate a number of specific “programs” applicable to SSCs that include:

- Maintenance Rule (RG 1.160 and 1.182; DSRS Sections 17.6 and 13.4Table 13.4, Item 17])

- Technical Specifications(TS) (DSRS Sections 16.0 and 16.1)

- Reliability Assurance Program (DSRS Section 17.4).

- Availability controls (Regulatory Treatment for Nonsafety Systems [RTNSS] and RG 1.206, Section C.IV.9).

- Initial Plant Test Program (RG 1.68, DSRS Sections 14.1 and 13.4 [Table 13.4, Item 19]).

- ITAAC (RG 1.215 and DSRS Section 14.3).

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. Application of 10 CFR 52.47(b)(1), as it relates to ITAAC (for design certification) provides reasonable assurance that the SSCs in this area of review will operate in accordance with the design certification, the provisions of the AEA, and NRC's regulations.

2. Application of 10 CFR 52.80(a), as it relates to ITAAC (for combined licenses) provides reasonable assurance that the SSCs in this area of review have been constructed and will be operated in conformity with the combined license, the provisions of the AEA, and the NRC's regulations.

3. Tier 1 should be reviewed for treatment of design information proportional to the safety significance of the SSC for that system. SSCs involving the reactor and core-cooling systems, such as the overpressure protection system, may be classified or judged to be important to safety and thus should be included in Tier 1.

4. NRC rules and regulations, generic correspondence, PRA insights, and operating experiences provide important sources for identifying significant design and features for inclusion in Tier 1.

5. Those active systems classified as non-safety systems are potentially the first line of defense to reduce challenges to the passive systems in the event of transients or plant upsets. While the passive systems are designed to perform their safety functions independently of operator action or off-site support for 72 hours after an event, these non-safety or active systems are capable of supplying water to the passive systems or directly performing core and containment heat removal functions and, therefore, should be considered as Tier 1. RTNSS evaluations provide a systematic determination of non-safety systems' impact that should be included in Tier 1.

6. The Commission provides applicable guidance for selected policy and technical issues related to a particular design that should be used by the reviewer. Examples of such guidance are contained in SECY-93-087.

7 Where a COL applicant references a certified standard design, the ITAAC, as contained in the standard certified design, must apply to those portions of the design that are covered by the design certification rule, as contained in the appendices to 10 CFR 52.

8. 10 CFR 20.1406 requires the design of a nuclear power plant to address the minimization of contamination of the facility and the environment. This is accomplished by considering the design features and operation of SSCs that contain or handle radioactive material as described in the COL technical submittal. Regulatory positions C.1 through C.4 of RG 4.21 describe concepts to be implemented to provide reasonable assurance that inadvertent spills, leaks, and discharges of liquid, gaseous, and solid radioactive effluents are prevented, detected, and corrected.

III. REVIEW PROCEDURES

These review procedures are based on the identified DSRS acceptance criteria. For deviations from these 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.

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. TS (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 6 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 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 safety evaluation report (SER) section.

3. Follow the general procedures for review of Tier 1 contained in Section III, ”Review Procedures” of DSRS Section 14.3, including those for ”Preparation for the Review” as well as the ”General Review Procedures.” Ensure that the DCD is consistent with the guidance and definitions as presented in Appendix A to DSRS Section 14.3. Review the functional review responsibilities for Tier 1 as presented in Appendix B to DSRS Section 14.3 to provide additional guidance on primary and secondary review assignments.

4. Ensure that an applicant for a COL referencing a certified design appropriately adopts the ITAAC for the certified portion of the design in the application.

5. Ensure that all Tier 1 information is consistent with Tier 2 information since all Tier 1 information is derived from Tier 2. Figures and diagrams should be reviewed to ensure that they accurately depict the functional arrangement, location, and requirements of the systems. Reviewers should use the review checklists in Appendix C to DSRS Section 14.3 as an aid in establishing consistent and comprehensive treatment of systems. Additionally, Tier 1 should be reviewed for consistency with the initial test program as described in DCD Tier 2 Chapter 14.

6. Ensure that the reactor systems are clearly described in Tier 1, including the key performance characteristics and safety-related functions of SSCs based on their safety significance.

7. Ensure that appropriate ITAAC are specified for those SSCs performing safety-related functions for Tier 1 Reactor Systems in the prescribed format as presented in Appendix A to DSRS Section 14.3.

8. Ensure that appropriate ITAAC are specified for verifying elevation differences between the reactor core and storage pools and tanks that provide core cooling for passive plants.

9. Ensure that appropriate ITAAC are specified for verifying design pressures of piping systems that interface with the reactor coolant boundary used to validate interfacing-systems loss-of-coolant accident analyses.

10. Ensure that appropriate guidance is provided to other branches such that reactor and core-cooling-systems issues in Tier 1 are treated in a consistent manner among branches.

11. Ensure that inputs from other branches regarding (a) PRA, including shutdown safety evaluations, and (b) severe accident analyses are appropriately treated in Tier 1.

12. Ensure that appropriate ITAAC are specified for verifying those important input parameters used in transient and accident analyses.

13. Ensure that standard ITAAC entries in Appendix D to DSRS Section 14.3 related to reactor systems are included, where appropriate, in the systems of the standard design. The reviewer should ensure consistent application and treatment of the standard ITAAC, and in particular for the basic-configuration ITAAC and the net-positive-suction-head ITAAC (for safety-related pumps).

14. Ensure that design features from the resolutions of selected policy and technical issues are adequately addressed in Tier 1 based on the safety significance of the design features. Ensure that the appropriate Commission guidance, requirements, bases, and resolutions for these items are clearly documented in the SER.

15. Ensure that any Tier 2* information is clearly designated in Tier 2, and consider expiration of these items at first full power, if appropriate. The staff's basis for designating the information as Tier 2* and the rationale for its decision, which requires prior NRC approval to change, should be specified in the SER. (See also the discussion in Appendix A to DSRS Section 14.3.)

16. Review Tier 1 definitions, legends, interface requirements, and site parameters to ensure that reactor-systems issues are treated consistently and appropriately.

17. Review Appendix C.II.2-A of RG 1.206 to understand the guidance and related rationale provided to applicants in developing ITAAC for fluid, I&C, and electrical systems as might be applicable to Reactor Systems.

18. 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 technical submittal meet the acceptance criteria. 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 technical submittal .

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

19. For review of both DC and COL applications, DSRS Section 14.3 should be followed for the review of ITAAC.

20. Implementation of ITAAC will be inspected in accordance with NRC Inspection Manual Chapter IMC-2503, "Construction Inspection Program: Inspections of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Related Work".

21. The reviewer should ensure that the guidance contained in the issued final Interim Staff Guidance (ISG) documents associated with applications for new reactors is followed:

   - Final Interim Staff Guidance – Evaluation and Acceptance Criteria for 10 CFR 20.1406 to support Design Certification and Combined License Applications (DC/COL-ISG-06).

   - Interim Staff Guidance on Post-Combined License Commitments (DC/COL-ISG-015).

   - Final Interim Staff Guidance DC/COL-ISG-019, "Review of Evaluation to Address Gas Accumulation Issues in Safety Related Systems

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 reviewer verifies that sufficient information has been provided to satisfy the requirements of DSRS Section 14.3 and this DSRS section, and concludes that the ITAAC is acceptable. A finding similar to that in the Evaluation Findings section of DSRS Section 14.3 should be provided in a separate section of the SER.

2. 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.

3. The reviewer verifies that sufficient information has been provided to satisfy the requirements of 10 CFR 20.1406 and the guidance of RG 4.21.

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 6 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 final safety analysis report 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 Part 20, Section 1406, "Minimization of Contamination."

2. 10 CFR Part 52, Section 47, "Contents of Applications; Technical Information."

3 10 CFR 52, Section 80, "Contents of Applications."

4. 10 CFR Part 52, Section 97, "Issuance of Combined Licenses."

5. SECY-02-0059, "Use of Design Acceptance Criteria for the AP1000 Standard Plant Design," April 1, 2002.

6. SECY-97-044, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," February 19, 1997.

7. SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084)," May 22, 1995 .

8. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," April 2,1993.

9. SECY-92- 327, "Reviews of Inspections, Test, Analyses, and Acceptance Criteria (ITACC) Requirements for the General Electric (GE) Advanced Boiling Water Reactor (ABWR)," September 22, 1992.

10. SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews," February 19, 1992.

11. SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," January 12,1990.

12. RG 1.68, "Initial Test Program for Water-Cooled Nuclear Power Plants."

13. RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.”

14. RG 1.182, “Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.”

15. RG 1.206, ”Combined License Applications for Nuclear Power Plants (LWR Edition).”

16. RG 1.215, “Guidance for ITAAC Closure Under 10 CFR Part 52.”

17. RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning.”

18. NRC Inspection Manual Chapter IMC-2503, ”Construction Inspection Program: Inspections of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Related Work,” October 3, 2007.

19. Interim Staff Guidance DC/COL-ISG-06, “Final Interim Staff Guidance – Evaluation and Acceptance Criteria for 10 CFR 20.1406 to Support Design Certification and Combined License Applications.”

20. Interim Staff Guidance DC/COL-ISG-015, “Post-Combined License Commitments.
21. Interim Staff Guidance DC/COL-ISG-019, “Review of Evaluation to Address Gas Accumulation Issues in Safety Related Systems.“