U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

3.2.1 SEISMIC CLASSIFICATION

REVIEW RESPONSIBILITIES

Primary - Organization responsible for mechanical engineering reviews

Secondary - Organizations responsible for the review of safety systems and risk assessment and, component performance and testing.

I. AREAS OF REVIEW

General Design Criterion (GDC) 2 of 10 CFR Part 50, Appendix A, in part, requires that nuclear power plant structures, systems, and components (SSCs) important to safety should be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. Important to safety SSCs are those SSCs that provide reasonable assurance that the facility can be operated with adequate protection to the health and safety of the public. The earthquake against which these plant features are designed is defined as the safe shutdown earthquake (SSE) in 10 CFR Part 100, Appendix A, and 10 CFR Part 50, Appendix S. The SSE is based upon an evaluation of the maximum earthquake potential and is that earthquake which produces the maximum vibratory ground motion for which SSCs important to safety are designed to remain functional. Appendix S also, requires consideration of surface deformation. Those plant features that are designed to remain functional if an SSE occurs are designated Seismic Category I in Regulatory Guide (RG) 1.29.

Requirements for SSCs whose function is necessary for continued operation during and following an operating basis earthquake (OBE) are discussed in 10 CFR Part 50, Appendix S.
An alternative approach to classify SSCs important to safety is identified in 10 CFR 50.69 as a risk-informed categorization process that applies industry guidelines for categorizing SSCs according to a risk-informed safety class. The risk-informed approach described in Regulatory Guide (RG) RG 1.201 is optional and subject to the limitations of 10 CFR 50.69. Successful application of an acceptable risk-informed categorization approach depends on a high quality PRA. Considering that RG 1.201 currently is to be used only as interim guidance for trial use and that an acceptable risk-informed method to assign risk-informed safety class does not exist, this Design Specific Review Standard (DSRS) section does not include criteria for reviewing a risk-informed categorization approach. Guidance in other referenced standard review plans can support a risk-informed classification approach. For iPWR designs, SECY11-0024 describes a risk-informed approach to enhance the safety focus for small modular reactor reviews.
Risk-informed classification review guidance in RG 1.201 may assist in iPWR reviews when combined with pilot studies. As discussed in NUREG-1242, other industry standards managed for the U.S. electric utility industry by EPRI such as EPRI ALWR URD can be used to address the seismic design requirements for the design of Advanced Light Water Reactor (ALWR).

ADAMS Accession No. ML12272A013

The specific areas of review are as follows:

1. This DSRS section reviews the seismic classification design criteria and application of that criterion on a sampling basis to those SSCs (including their foundations and supports) that are important to safety and are designed to withstand, without loss of function, the effects of a SSE and specified as Seismic Category I by the applicant's safety analysis report (SAR). The review covers identification of SSCs that are not required to remain functional following a seismic event, but whose failure could reduce the functioning of any Category I SSCs to an unacceptable safety level, or could result in incapacitating injury to control room occupants, and therefore must be analyzed and designed to maintain their integrity under seismic loading from the SSE, referred to as Seismic Category II in NUREG-1242. In addition, the staff reviews the identification of radioactive waste management SSCs that require seismic design considerations as specified in RG 1.143.

2. This review, which may be coordinated with each branch that has primary review responsibility for these plant features when exceptions to RG 1.29 are identified, is performed for combined license (COL) applications. The staff review of Seismic Category I and Seismic Category II items includes the following plant features: structures, dams, ponds, cooling towers, reactor internals, fluid systems important to safety that are identified in RG 1.29, NUREG-1242, safety-related instrument sensing lines that are identified in RG 1.151, ventilation systems, standby diesel generator auxiliary systems, fuel handling systems, and cranes.

3. The staff reviews Seismic Category I SSCs that are identified in RG 1.189 to establish the design requirements of fire protection to withstand seismic loading from the SSE. This RG identifies portions of fire protection SSCs requiring seismic design consideration.

4. The applicant's proposed seismic classification may in part be presented in the form of a table[1] that identifies those SSCs that are designated Seismic Category I. The table should identify all activities affecting the safety-related functions of these Seismic Category I plant features that should also meet GDC 1 and the pertinent quality assurance (QA) requirements of 10 CFR Part 50, Appendix B. The acceptability of QA applied to SSCs that have seismic design considerations is determined in accordance with RG 1.29. Details of the seismic classification of these plant features may be shown on plot plans, general arrangement drawings, and piping and instrumentation diagrams.

   If the applicant has set OBE Ground Motion to the value one third of the SSE Ground Motion, per Staff Requirements Memorandum to SECY 93-087 approved by the Commission, OBE is eliminated from the design of SSCs. OBE will serve as an “inspection level earthquake” below which the effect on health and safety of the public would be insignificant and above which the licensee would be required to shut down the plant and inspect the damage. A list of the SSCs necessary for continued safe operation during and following an OBE is not required at the design stage. However, a list of necessary SSCs is needed to support plant inspections for damage after an earthquake; timing for providing the list should be determined by NUREG-0800 Standard Review Plan (SRP) or DSRS Chapter 13 reviewers or those responsible for pre-earthquake planning and post-earthquake inspection procedures in Section 3.7.4

[1]See DSRS Section 3.2.2 - "System Quality Group Classification," for guidance.

Certain equipment that is only designed to withstand OBE may be addressed by specific regulatory guidance, such as RG 1.143.

5. Where portions of structures and fluid systems are Seismic Category I, they also must be clearly identified. For fluid systems important to safety, the classification tables in the SAR should identify system components such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves, have suitable footnotes defining interfaces, and be in sufficient detail so that there is a clear understanding of the extent of those portions of the system that are classified as Seismic Category I.

6. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For DC and COL reviews, the staff reviews the applicant's proposed ITAAC associated with the SSCs related to this DSRS section in accordance with SRP 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 SRP Section 14.3. The application should include ITAAC or an equivalent alternative process to verify seismic classification.

7. COL Action Items and Certification Requirements and Restrictions. 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. For more specific guidance see RG 1.206.

Review Interfaces

Other SRP and DSRS sections that interface with this section include:

1. The acceptability of the quality group classification of system components is determined in accordance with DSRS Section 3.2.2. The quality group classification information may be combined and/or cross-referenced with the seismic classification information reviewed in this DSRS section to minimize repetition of similar information (e.g., tables or lists of components, system drawings, etc.).

2. Verification is performed on the systems and components important to safety that are designated as Seismic Category I items that are designed in accordance with the regulatory guides, industry codes and standards that are referenced in SRP and DSRS Sections 3.2.2, 3.9.1 through 3.9.3, 3.10, and 3.11.

3. The adequacy of the qualification and inservice testing program for pumps and valves is determined in accordance with DSRS Section 3.9.6.

4. Consistency with seismic requirements for electrical equipment is evaluated in the DSRS sections for Chapter 8. The seismic qualification of equipment is assessed in accordance with SRP Section 3.10.

5. The radioactive waste management SSCs is reviewed in accordance with DSRS Sections 11.2 through 11.4.

6. The seismic design of fire protection systems installed in safety-related areas is reviewed in accordance with SRP Section 9.5.1

7. The quality assurance program for design, construction and operation is reviewed in accordance with SRP Sections 17.5.

8. The classification and design of safety-related structures are reviewed in accordance with DSRS Sections 3.8.1 through 3.8.5.

9. Consistency with seismic requirements for reactor pressure vessel internals is reviewed in accordance with DSRS Section 3.9.5.

10. The list of SSCs necessary for continued safe operation that must remain functional during and following an OBE to support plant inspections for damage after an earthquake is reviewed in SRP or DSRS Chapter 13 and those responsible for pre-earthquake planning and post-earthquake inspection procedures in Section 3.7.4. RG 1.206 clarifies guidance for the list of SSCs.

11 Identification of risk-significant non safety-related SSCs that are important to safety, including regulatory treatment of non-safety related system (RTNSS) SSCs, is primarily reviewed in SRP Section 17.4 and SRP Section 19.0.

To assist in the review of seismic classification, the staff in other branches that review information presented in other SRP and DSRS sections referenced in this DSRS section will coordinate evaluations that interface with the overall review of system seismic classification addressed in those sections as follows:

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

II. ACCEPTANCE CRITERIA

Requirements

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

1. GDC 1, and the pertinent QA requirements of 10 CFR Part 50, Appendix B, as they relate to applying QA requirements to activities affecting the safety-related functions of SSCs designated as Seismic Category I commensurate with their importance to safety.

2. GDC 2, as it relates to the requirements that SSCs important to safety shall be designed to withstand the effects of earthquakes without loss of capability to perform necessary safety functions.

3. GDC 61, as it relates to the design of radioactive waste systems, and other systems that may contain radioactivity, to assure adequate safety under normal and postulated accident conditions.

4. 10 CFR Part 100, Appendix A and 10 CFR Part 50, Appendix S, as it relates to certain SSCs being designed to withstand the SSE and remain functional.

5. 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's) regulations.

6. 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.

7. 10 CFR 52.47 which requires that the information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant. The Commission will require, before design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if the information is necessary for the Commission to make its safety determination.

8. 10 CFR 50.34 and 10 CFR 52.47 which require that the final safety analysis report (FSAR) include the design bases and the technical justification upon which the design requirements have been established. Design bases as defined in 10 CFR Part 50.2 means that information which defines the specific functions to be performed by SSCs and the specific values or ranges of values chosen for controlling parameters as reference bounds for design.

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for review described in this DSRS section. 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."

1. To meet the requirements of GDC 2, 10 CFR Part 100, Appendix A, and 10 CFR Part 50, Appendix S regarding seismic design classifications are met by using guidance provided in RG 1.29 "Seismic Design Classification." This guide describes an acceptable method

of identifying and classifying those plant features that should be designed to withstand the effects of the SSE. An alternative to RG 1.29 is the EPRI URD classification approach evaluated in NUREG-1242. RG 1.151 provides guidance with regard to seismic design requirements and classification of safety-related instrumentation sensing lines.

RG 1.143 provides guidance used to establish the seismic design requirements of radioactive waste management SSCs to meet the requirements of GDC 2 and 61 as they relate to designing these SSCs to withstand earthquakes. The guide identifies several radioactive waste SSCs requiring some level of seismic design consideration.

RG 1.189 provides guidance used to establish the design requirements of fire protection to meet the requirements of GDC 2 as it relates to designing these SSCs to withstand earthquakes. This guide identifies portions of fire protection SSCs requiring seismic design consideration.

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. Compliance with GDC 1 and 10 CFR Part 50, Appendix B, requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. GDC 1 requires, in part, that a QA Program be established and implemented in order to provide adequate assurance that SSCs important to safety will satisfactorily perform their safety functions. 10 CFR Part 50, Appendix B, establishes QA program requirements for the design, construction, and operation of SSCs important to safety. The requirements of 10 CFR Part 50, Appendix B apply to activities affecting the safety-related functions of those SSCs, including those SSCs defined by the guidance of RG 1.29 as Seismic Category I SSCs. Specifying and using proven quality standards and requirements for the design of SSCs important to safety minimizes the potential for failures of those SSCs, including Seismic Category I SSCs that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

2. Compliance with GDC 2 requires that nuclear power plant SSCs important to safety be designed to withstand the effects of natural phenomena, including earthquakes, without loss of capability to perform their safety functions. Also, compliance with 10 CFR Part 100, Appendix A and 10 CFR Part 50, Appendix S, requires that certain SSCs be designed to withstand the SSE and remain functional. The SSCs are those necessary to ensure: (1) the integrity of the reactor coolant pressure boundary; (2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100. RG 1.29 describes an acceptable method of identification and classification of those SSCs that should be designed to withstand the SSE. RG 1.29 states that systems and components required for safe shutdown, including their foundations and supports, are designated as Seismic Category I and should be designed to withstand the effects of the SSE and remain functional. In addition, this guide recommends that systems, other than radioactive waste management systems, that contain, or may contain, radioactive material and whose postulated failure would result in potential offsite whole body (or equivalent) doses that are more than 0.005 Sv (0.5 rem), should also be classified as

Seismic Category I. Compliance with RG 1.29 assures that, by designing the SSCs identified in the guide to withstand the effects of an SSE, a designed-in safety margin is provided for bringing the reactor to a safe, shutdown condition, while also reducing potential offsite doses from seismic events. RG 1.151 positions C.2 and C.3 provide guidance for the proper seismic classification of safety-related instrumentation sensing lines. Application of this guidance ensures that the instrument sensing lines used to actuate or monitor safety-related systems will be appropriately classified and will be capable of withstanding the effects of the SSE. RG 1.189 positions 3.2.1, 6.1.1.2, and 7.1 provide guidance for the proper seismic classification of fire protection systems. Application of this guidance ensures that (1) the fire protection systems for manual firefighting in areas containing safety related equipment, (2) containment penetrations and (3) RCP lube oil will be properly classified and analyzed for safe-shutdown earthquake loads. Compliance with the above requirements and guidance assures that the SSCs important to safety that are required to function during an SSE are properly classified as Seismic Category I and will function during such events enabling accomplishment of the safety functions described above.

3. Compliance with GDC 61 requires that radioactive waste management systems, and other systems that may contain radioactivity, be designed to assure adequate safety under normal and postulated accident conditions. Postulated conditions considered with respect to seismic design and classification of SSCs include losses of SSC integrity and potential radioactive releases as a result of seismic events. RG 1.143 provides acceptable methods and guidance relative to seismic design and classification for radioactive waste management SSCs. This RG provides classification information and design criteria to assure that components and structures used in radioactive waste management systems are designed, constructed, installed and tested in a manner that protects the health and safety of the public and the plant operating personnel. Designing and constructing the radioactive waste management SSCs to meet the requirements of GDC 61 and the guidance on seismic design and classification contained in RG 1.143 provides assurance that SSCs containing radioactivity will be properly classified and radiation exposures as a result of seismic events will be as low as reasonably achievable.

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 and Guidance - 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.

Examples of those programs and associated guidance follows:

- Maintenance Rule SRP Section 17.6 (DSRS Section 13.4, Table 13.4, Item 17, Regulatory Guides 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".

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

- 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.

- Reliability Assurance Program (SRP Section 17.4).

- Initial Plant Test Program (Regulatory Guide 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).

- ITAAC (DSRS Chapter 14).

2. In accordance with 10 CFR 52.47(a)(8),(21), and (22), and 10 CFR 52.79(a)(17) and (20), for new reactor license applications submitted under Part 52, the applicant is required to (1) address the proposed technical resolution of unresolved safety issues and medium- and high-priority generic safety issues which are identified in the version of NUREG-0933 current on the date up to 6 months before the docket date of the application and which 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). 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. RG 1.29, which identifies SSCs of light-water-cooled reactors on a functional basis, is the principal document used for identifying those plant features important to safety which, as a minimum, should be designed to Seismic Category I requirements. RG 1.151 provides guidance for the seismic classification of safety-related instrument sensing lines. RG 1.29 also recommends that systems, other than radioactive waste management systems, that contain, or may contain, radioactive material and whose postulated failure would result in conservatively calculated potential offsite whole body (or equivalent to any part of the body) doses that are more than 0.005 Sv (0.5 rem), should also be classified as Seismic Category I. RG 1.143 provides seismic design requirements for radioactive waste management system SSCs. Those radioactive waste management systems requiring seismic design considerations should be clearly identified. RG 1.189 provides guidance for seismic classification and analysis of fire protection systems SSCs. Those portions of fire protection systems requiring seismic design considerations should be clearly identified.

The staff review should establish whether the applicant has indicated compliance with Regulatory Guides 1.29, 1.143, 1.151, and 1.189 in the SAR. Where there are differences with respect to the Guides, these differences should be identified. An alternative seismic classification approach described in the EPRI URD and evaluated in NUREG-1242 that applies Seismic Category I, Seismic Category II and Non-Seismic is acceptable for advanced light-water-cooled reactors.

4. The information in the SAR identifying Seismic Category I and Seismic Category II SSCs is reviewed for completeness and to assure there is sufficient detail to permit identification of specific items. This may include a review of the SAR text, tables, plot plans, general arrangement drawings, structural drawings, and piping and instrumentation diagrams, as appropriate. Where portions of a system are classified Seismic Category I, the boundary limits of that portion of the system designed to Category I requirements are reviewed on the piping and instrumentation diagrams. For fluid systems that are partially Seismic Category I, the Category I portion of the system should extend to the first seismic restraint beyond the isolation valves that isolate the part that is Seismic Category I from the Non-Seismic portion of the system. At the interface between Seismic and Non-Seismic Category I piping systems, the Seismic Category I dynamic analysis will be extended to either the first anchor point in the Non-Seismic system or to a sufficient distance in the Non-Seismic system so as not to degrade the validity of the Seismic Category I analysis. In addition, where portions of a structure are classified Seismic Category I, those portions of the building foundations and supports designed to Category I requirements are identified on the plant arrangement drawings. The interfaces between components and associated support structures designed to Seismic Category I requirements are then checked to assure compatibility.

   The reviewer verifies that the seismic classification of safety-related instrumentation sensing lines is in accordance with the guidance in RG 1.151 positions C.2 and C.3.

5. SSCs that are classified Seismic Category I and Seismic Category II are also reviewed to assure that these plant features are within the scope of an applicant's QA Program. This QA Program should be in compliance with the pertinent QA requirements of 10 CFR Part 50, Appendix B. In accordance with RG 1.29, the pertinent QA requirements of Appendix B to 10 CFR Part 50 should be applied to all activities affecting the safety-related functions of Seismic Category I SSCs. If there are items designated Seismic Category I that are not identified as within the scope of the 10 CFR Part 50, Appendix B, QA Program, then this information is transmitted to the staff for resolution of the issue. The seismic classification review of SSCs important to safety and the review verifying that these plant features are constructed in accordance with a 10 CFR Part 50, Appendix B, QA Program is normally performed concurrently with the quality group classification review of DSRS Section 3.2.2.

   Other SSCs that may be required for operation of the facility (excluding electrical features) need not be designed to Seismic Category I requirements. Those SSCs not required to be designed to Seismic Category I requirements include those portions of Seismic Category I systems such as vent lines, drain lines, fill lines and test lines on the downstream side of isolation valves and those portions of the system not required to perform a safety function.

6. The information in the SAR is reviewed to identify SSCs whose continued function is not required following a seismic event, but whose failure could reduce the functioning of any

Seismic Category I feature to an unacceptable safety level, or could result in incapacitating injury to control room personnel, to assure that such items will be analyzed and designed to maintain their integrity under seismic loading from the SSE.

The information in the SAR is also reviewed to identify radioactive waste management system and fire protection SSCs to assure that those SSCs requiring seismic design considerations have been identified consistent with those systems specified in RG 1.143 and RG 1.189.

7. In the event an applicant intends to take exception to RG 1.29, 1.143, 1.151, and/or 1.189 but has not provided an adequate justification for resultant proposed seismic classifications, the staff prepares questions whose answers may require additional documentation or analysis to establish an acceptable basis for the proposed seismic classification. The staff may also prepare comments requesting clarification in order to assure a clear understanding of the seismic classification assigned to a system by the applicant.

   If the staff's questions are not resolved in a satisfactory manner, a staff position is taken requiring conformance to RG 1.29, 1.143, 1.151, 1.189 and with the positions discussed in the above Review Procedures.

8. 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 design control document (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 or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

   For review of both DC and COL applications, SRP 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.

9. 10 CFR 52.47 also states that the Commission will require, before design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if the information is necessary for the Commission to make its safety determination. The staff may elect to credit an ITAAC or audit available design documents such as design specifications, system description and schematics or piping & instrumentation diagrams, if applicable; QA lists; and procurement documents associated with the seismic classification of risk-significant systems and mechanical components. An audit should be scheduled based on the availability of design documents and prior to the design certification and/or COL application. The staff review may include an assessment of the degree of completeness of design information supporting classifications and how seismic classification is translated into design documents. The audit may also be used to support resolution of seismic classification open items identified during the review of the application. Depending on the audit plan, the scope may be limited to a review of the

design classification process and a sampling of risk-significant systems and mechanical components to validate that the applicant has an appropriate classification process in place.

10. GDC 2 requires SSCs that are important to safety be designed to withstand earthquakes. To support compliance with GDC 2, such SSCs should be appropriately classified to ensure that they are designed to withstand earthquakes. In addition to safety-related SSCs, non safety-related SSCs are to be designed to withstand earthquakes if they perform an important to safety function. The extent that these non safety-related SSCs are to be classified and designed for earthquakes depends on the specific need to be functional or to preclude their failure, consistent with risk insights. The risk informed approach that utilizes the probabilistic risk assessment (PRA) can be utilized to supplement and enhance the deterministic approach in order to identify these non safety-related SSCs that are credited in the PRA. Where industry consensus standards for seismic classification are consistent with NRC regulations, they may be acceptable for licensing purposes. Various approaches, such as the RTNSS process, may be considered to designate appropriate seismic requirements, including industry consensus standards and regulatory guidance. The RTNSS process described in the consolidation of SECY-94-084 and SECY-95-132 dated 7/24/95, is generally applied to passive designs and is considered for advanced reactors on a case by case basis (RG 1.206, C.IV.9). The designer will impose design requirements commensurate with risk significance. For example, in SECY 96-128 and its associated SRM, it was decided that certain RTNSS SSCs for passive ALWRs (RTNSS B) need not be safety-related or subject to dynamic qualification, but their anchorage did need to be designed for seismic events and their equipment enclosed in a seismically designed structure. Depending on the particular safety function, other design classifications and criteria may be considered for non safety-related SSCs, such as the application of experience data (NUREG-1242). As part of the PRA, the seismic margins analysis may also be considered in determining which SSCs are credited for seismic events and the degree to which they are to be designed for seismic events.

IV. EVALUATION FINDINGS

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

SSCs (excluding electrical features) that are important to safety and that are required to withstand the effects of an SSE and remain functional have been classified as Seismic Category I items and have been identified in an acceptable manner in Tables 3.X.X and 3.X.X, and on system piping and instrumentation diagrams in the SAR. Other SSCs not identified as Seismic Category I, but whose failure could reduce the functioning of any Seismic Category I feature to an unacceptable safety level or injure control room personnel, are identified for analysis to assure the SSE will not cause such failures.

The staff concludes that the SSCs important to safety that are within the scope of this review have been properly classified, are within the scope of the applicant's QA Program, and thus meet the relevant requirements of GDC 1, 2, and 61, 10 CFR Part 50, Appendix B, 10 CFR 50.34(a)(1), and 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

This conclusion is based on:

1. The applicant’s having met the requirements of GDC 1 by providing a commitment in the SAR that Seismic Category I SSCs will be designed, constructed and operated under a QA Program, in compliance with the requirements of 10 CFR Part 50, Appendix B.

2. The applicant’s having met the requirements of GDC 2, 10 CFR Part 100, Appendix A and 10 CFR 50, Appendix S, by having properly classified SSCs important to safety as Seismic Category I items in accordance with the positions of RG 1.29, "Seismic Design Classification," RG 1.151, "Instrument Sensing Lines" and RG 1.189 “Fire Protection for Nuclear Power Plants.” The identified SSCs are those plant features necessary to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent and mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.

3. Those SSCs not identified as Seismic Category I, but whose failure could reduce the functioning of any Seismic Category I feature to an unacceptable safety level or result in incapacitating injury to control room personnel, having been identified for analysis to assure they will not fail during a SSE (e.g. Seismic Category II).

4. Radioactive waste system and fire protection SSCs requiring seismic design considerations having been identified consistent with the positions of RG 1.143. and RG 1.189.

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.

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 reviews including the associated pre-application activities, the staff has developed the content of this DSRS section as an alternative method for mPower^{TM} -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 Standard Review Plan (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^{TM} 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) for COL applications.
.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records."

2. 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection Against Natural Phenomena."

3. 10 CFR Part 50, Appendix A, GDC 61, "Fuel Storage and Handling and Radioactivity Control."

4. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

5. 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants."

6. RG 1.29, "Seismic Design Classification."

7. RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."

8. RG 1.151, "Instrument Sensing lines."

9. RG 1.189, “Fire Protection for Nuclear Power Plants.”.

10. 10 CFR 50, Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants.”

11. 10 CFR 50.34, Contents of application, technical information.

12. 10 CFR 52.47, Contents of application, technical information

13. NUREG-1242, Vol.2, Pt.1, “NRC review of Electric Power Research Institute’s Advanced Light water Reactor Utility Requirements Document, Evolutionary Plant Designs Chapter 1”

14. Electric Power Research Institute (EPR),Advanced Light water Reactor (ALWR) Utility Requirement Document (URD) Volume I, II and III.

15. SECY-11-0024, “Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews”

16. SECY-93-087, “Policy, Technical, and Licensing issues pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs.

17. SECY-96-128, Policy and Key Technical issues pertaining to the Westinghouse AP600 Standardized Passive Reactor Design.

18. SECY-94-084, Policy and Ttechnical Issues associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs.

19. SECY-95-132, Policy and Technical Issues associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084).

20. RG 1.201, “Guidelines for Categorizing Structures, Systems and Components in Nuclear Power Plants, According to their Safety Significance.”

21. RG 1.206, “Combined License Applications for Nuclear power Plants”

22. 10 CFR 50.69, “Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors.”

23. Branch Technical Position 3-2, “Classification of BWR/6 Main Steam and Feedwater Components Other than the Reactor Coolant Pressure Boundary.”

24. NUREG-800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.