U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER^{TM} iPWR DESIGN

3.8.3 CONCRETE AND STEEL INTERNAL STRUCTURES OF STEEL CONTAINMENTS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for structural analysis reviews

Secondary - None

I. AREAS OF REVIEW

The containment internal structures are designed to provide structural support to the following systems and components, or portions thereof, which are inside the containment:

- Reactor vessel
- Reactor Coolant System
- Emergency Core Cooling System
- Reactor Coolant Inventory and Purification System
- Fuel Transfer Canal and Fuel Handling Systems
- Overhead Crane.

The specific areas of review are as follows:

1. Description of the Internal Structures. The descriptive information, including plans and sections of the various internal structures, is reviewed to establish that sufficient information is provided to define the primary structural aspects and elements relied upon to perform the safety-related functions of these structures. To perform safety-related functions, these structures must be capable of resisting loads and load combinations to which they may be subjected and should not become the initiator of a loss-of-coolant accident (LOCA). If such an accident were to occur, the structures should be able to mitigate its consequences by protecting the containment and other engineered safety features from the accident's effects such as jet forces and whipping pipes.

   The major containment internal structures that are reviewed, together with the primary structural function of each structure and the extent of descriptive information required for each structure, are indicated below. This Design Specific Review Standard (DSRS) section reviews the intervening structural elements between distribution systems including their supports (e.g., cable trays; conduit; heating, ventilation, and air conditioning; and piping and equipment supports) and building structural steel/concrete (e.g., steel platforms, building frame members, embedment plates, and building steel members beyond the jurisdictional boundary of supports to mechanical components).

   i. Concrete/Steel Supports for Reactor

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The mPower^{TM} reactor vessel should be supported and restrained to resist normal operating loads, seismic loads, and loads induced by postulated pipe rupture, including LOCAs. The support and restraint system should restrain the movement of the vessel to within allowable limits under the applicable loading combinations. However, the support system should minimize resistance to the thermal movements expected during operation.

With these functional requirements in mind, the review evaluates the general arrangement and principal features of the reactor vessel supports, with an emphasis on the methods of transferring loads from the vessel to the support and ultimately to the structure and its foundations.

ii. Primary Shield Wall and Reactor Cavity

The primary shield wall forms the reactor cavity and usually supports and restrains the reactor vessel. It is often a thick wall that surrounds the reactor vessel and may be anchored through the liner plate to the containment base slab.

The review evaluates the general arrangement and principal features of the wall and cavity, including the reinforcement and anchorage system.

iii. Other Interior Structures

The review also evaluates other major interior structures of the mPower^{TM} containment, including concrete refueling pool walls, refueling water storage tank, operating floor, other intermediate floors and platforms, and the overhead crane supporting elements.

Plant designs may also use modular construction methods for the major containment internal structures. Wall modules are typically constructed from large, prefabricated sections of steel plates spaced apart with intermittent steel members, joined with other modules at the site, and then filled with concrete. The concrete fill used in wall modules could be structural concrete with reinforcement (composite construction) or fill concrete of low strength without reinforcement, or heavy concrete for radiation shielding. Floor modules consist of prefabricated steel members and plates and are combined with poured concrete to create a composite section. In view of the new application of modules to nuclear power plants, the structural module design, fabrication, configuration, layout, and connections will be reviewed on a case-by-case basis.

2. <u>Applicable Codes, Standards, and Specifications</u>. The review evaluates the information pertaining to design codes, standards, specifications, and regulatory guides, as well as industry standards that is applied in the design, fabrication, construction, testing, and surveillance of the containment structures. The specific editions, dates, or addenda identified for each document are also reviewed.

3. <u>Loads and Loading Combinations</u>. The review evaluates the information pertaining to the applicable design loads and associated load combinations. The loads normally applicable to containment internal structures include the following:

A. Loads encountered during construction of containment internal structures, including dead loads, live loads, prestress loads (if applicable), temperature, wind, earth pressure, snow, rain, and ice, and construction loads that may be applicable, such as material loads, personnel and equipment loads, horizontal and vertical construction loads, loads that are induced by the proposed construction sequence and by the differential settlements of the soil under and to the sides of the reactor service building, erection and fitting forces, equipment reactions, and form pressure.

B. Loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads from operating temperature, and hydrostatic loads such as those in the refueling pool and reactor water storage tank. In addition, hydrodynamic loads resulting from actuation of the automatic depressurization valves (ADVs) into the refueling water storage tank (RWST) and manifested as drag load, jet impingement, wall pressure loads, and building dynamic response loads should be considered.

C. Loads to be sustained during severe environmental conditions, including those induced by the operating-basis earthquake (OBE) specified for the plant site. Subsection II.3.A of this DSRS defines the condition for which the OBE load is required for design of containment internal structures.

D. Loads to be sustained during extreme environmental conditions, including those induced by the safe-shutdown earthquake (SSE) specified for the plant site.

E. Loads to be sustained during abnormal plant conditions. The design-basis LOCA is a significant abnormal plant condition during which most of the containment internal structures have to perform their primary function. Ruptures of other pipes should also be considered. Time-dependent and dynamic loads induced by such accidents include elevated temperatures and differential pressures across compartments, jet impingement, impact forces associated with the postulated ruptures of piping, and may include building dynamic response loads.

The various combinations of the above loads that are normally postulated and reviewed include construction loads, normal operating loads, normal operating loads with severe environmental loads, normal operating loads with extreme environmental loads, normal operating loads with abnormal loads, normal operating loads with severe environmental and abnormal loads, and normal operating loads with extreme environmental and abnormal loads.

4. Design and Analysis Procedures. The review evaluates the design and analysis procedures used for the containment internal structures, with an emphasis on the extent of compliance with the applicable codes as indicated in Subsection II.2 of this DSRS. The review includes the design and analysis procedures applicable to the following areas:

A. mPower^{TM} Containment Internal Structures

i. Concrete/Steel Supports for Reactor Vessel

The support system for the reactor vessel, as described in Subsection I of this DSRS section, should be designed to resist various combinations of loadings, including normal operating loads, seismic loads, and loss-of-coolant and other pipe rupture accident loads.

DSRS Section 3.7.2 describes the analytical procedures for determining seismic loads.

After the procedures for determining individual loads and combinations thereof are reviewed, the design and analysis methods used for the supports are considered, including the type of analysis, the methods of load transfer, and the assumptions of boundary conditions.

ii. Primary Shield Wall and Reactor Cavity

The primary shield wall should withstand all applicable loads, including those transmitted through the reactor supports. The wall is subjected to most of the loads described in Subsection I.3 of this DSRS section and should be designed and analyzed for all applicable load combinations. During normal plant operation, the attenuation heat of gamma and neutron radiation originating from the reactor core generates a thermal gradient across the wall. Insulation and cooling systems may be provided to reduce the severity of this gradient by limiting the rise in temperature to an acceptable level.

Procedures for determining seismic loads on the primary shield wall are reviewed in accordance with DSRS Section 3.7.2.

LOCA loads applicable to the primary shield wall include a differential pressure created across the reactor cavity by a pipe break in the vicinity of the reactor nozzles. Such a transient pressure may act on the entire cavity or on portions of the cavity. Procedures for determining such pressures are reviewed in accordance with guidance provided in DSRS Section 6.2.1.

Other LOCA loads that apply are those transmitted to the wall through the reactor supports, including pipe rupture reaction forces which may induce simultaneous shear forces, torsional moments, and bending moments at the base of the wall. The elevated temperature within and around the primary shield created by the accident may also produce transient thermal gradients across the thick wall. The review evaluates these design and analysis procedures accordingly.

iii. Other Interior Structures

Many of the other interior structures reviewed are combinations of slabs, walls, beam, and columns classified as seismic Category I structures. These structures are subjected to most of the loads and load combinations described in Subsection I.3 of this DSRS section. The review evaluates the analytical techniques for these structures on the same basis as the review of the structures described above.

B. Design Reports

The applicant's design report, as described in Appendix C to DSRS Section 3.8.4, is reviewed.

C. Structural Audit

A structural audit, as described in DSRS Section 3.8.4, Appendix B is conducted.

5. Structural Acceptance Criteria. The review evaluates the design limits imposed on the various parameters that quantify the structural behavior of the various interior structures of the containment, particularly with respect to stresses, strains, deformations, and factors of safety against structural failure, with emphasis on the extent of compliance with the applicable codes indicated in Subsection II.5 of this DSRS.

6. Materials, Quality Control, and Special Construction Techniques. The review evaluates the information provided on the materials that are used in the construction of the containment internal structures. Concrete ingredients, reinforcing bars and splices, structural steel, and various supports and anchors are among the major materials of construction reviewed.

The review evaluates the quality control program proposed for the fabrication and construction of the containment internal structures, including nondestructive examination of the materials to determine physical properties, placement of concrete, and erection tolerances.

Special, new, or unique construction techniques, such as the use of modular construction methods, if proposed, are reviewed on a case-by-case basis to determine their effects on the structural integrity of the completed containment internal structures.

In addition, the following information should be provided:

A. The extent to which the materials and quality control programs comply with American Concrete Institute (ACI) 349, as supplemented by additional guidance provided by Regulatory Guides (RG) 1.142 and 1.199 for concrete and anchors (steel embedments), respectively, and American National Standards Institute/American Institute of Steel Construction (ANSI/AISC) N690-1994 including Supplement 2 (2004), for steel, as applicable.

B. If welding of reinforcing bars is proposed, describe the extent to which the applicant complies with the applicable sections of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code (hereafter referred to as Code) Section III, Division 2, Subsection CC, as supplemented with additional guidance provided by RG 1.136. Provide justification for any exceptions.

7. Testing and Inservice Surveillance Programs. For seismic Category I structures inside containment, the review evaluates information on structures monitoring and maintenance requirements.

   For containment internal structures, it is important to accommodate inservice inspection of critical areas. The review includes any special design provisions (e.g., sufficient physical access, alternative means for identification of conditions in inaccessible areas that can lead to degradation, remote visual monitoring of high radiation areas) to accommodate inservice inspection of containment internal structures.

   Postconstruction testing and inservice surveillance programs for containment internal structures and periodic examination of inaccessible areas are reviewed on a case-by-case basis.

8. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). 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 NUREG-0800 Standard Review Plan (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.

9. 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.

Review Interfaces

Other SRP and DSRS sections interface with this section as follows:

1. Determination of structures that are subject to quality assurance programs in accordance with the requirements of 10 CFR 50, Appendix B, is performed in accordance with DSRS Sections 3.2.1 and 3.2.2. The review of safety-related structures is performed on that basis.

2. Determination of pressure loads from higher-energy lines located in safety-related structures is performed in accordance with SRP Section 3.6.1. The loads thus generated are included in the load combination equations of this DSRS section.

3. Determination of loads generated from pressure under accident conditions is performed in accordance with DSRS Section 6.2.1. The loads thus generated are included in the load combinations in this DSRS section.

4. Distribution systems including their supports (e.g., cable trays; conduit; heating, ventilation, and air conditioning) and equipment supports are reviewed in accordance with SRP Sections 3.9.2 and 3.9.3.

5. The organization responsible for quality assurance performs the reviews of design, construction, and operation phase quality assurance programs under SRP Chapter 17. In addition, while conducting regulatory audits in accordance with Office Instruction NRR-LIC-111 or NRO-REG-108, "Regulatory Audits," the technical staff may identify quality-related issues. If this occurs, then the technical staff should contact the organization responsible for quality assurance to determine if an inspection should be conducted.

6. Review of the Probabilistic Risk Assessment is performed under SRP Section 19.

II. ACCEPTANCE CRITERIA

Requirements

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

1. 10 CFR 50.55a and 10 CFR 50, Appendix A, General Design Criterion (GDC) 1, as they relate to the design, fabrication, erection, and testing of containment internal structures in accordance with quality standards commensurate with the importance of the safety function to be performed.

2. GDC 2, as it relates to the ability of the containment internal structures without loss of capability to perform their safety function, to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and the appropriate combination of all loads.

3. GDC 4, as it relates to the protection of containment internal structures against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

4. GDC 5, as it relates to safety-related structures not being shared among nuclear power units, unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions.

5. GDC 50, as it relates to the design of containment internal structures with sufficient margin of safety to accommodate appropriate design loads.

6. 10 CFR 50, Appendix B, as it relates to the quality assurance criteria for nuclear power plants.

7. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (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, and the NRC's regulations.

8. 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 Atomic Energy Act, 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 as follows for the 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. Description of the Internal Structures. The descriptive information in the safety analysis report (SAR) is considered acceptable if it meets the criteria set forth in Section 3.8.3.1 of RG 1.70 or 1.206.

   During the application acceptance review, the reviewer identifies deficient areas of descriptive information and initiates a request for additional information. New or unique design features that are not specifically covered in RG 1.70 or RG 1.206 may require a more detailed review. The reviewer determines whether additional information is required to accomplish a meaningful review of the structural aspects of such new or unique features.

   RG 1.206 provides the basis for evaluating the description of structures to be included in a DC or a COL application.

2. Applicable Codes, Standards, and Specifications. The design, materials, fabrication, erection, inspection, testing, and inservice surveillance, if any, of containment internal structures are covered by codes, standards, and guides that are applicable either in their entirety or in part. The following codes and guides are acceptable:

Code, Standard, or Specification	Title
ACI 349	Code Requirements for Nuclear Safety-Related Concrete Structures (supplemented with additional guidance by RG 1.142 and 1.199)
ASME Code	Section III, Division 2, Subsection CC, "Code for Concrete Reactor Vessels and Containments"

ANSI/AISC N690-1994 including Supplement 2 (2004)	Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities

Regulatory Guides

1.69	Concrete Radiation Shields for Nuclear Power Plants
1.136	Materials, Construction, and Testing of Concrete Containments
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)
1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in LWR Plants
1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants
1.199	Anchoring Components and Structural Supports in Concrete

3. Loads and Load Combinations. The loads and load combinations for containment internal structures described in Subsection I.1 of this DSRS are acceptable if they are consistent with the guidance given below.

A. Concrete Structures

All loads and load combinations are to be in accordance with ACI 349 and RG 1.142. Supplemental criteria on the use of loads and load combinations are presented below.

Dead loads include hydrostatic loads, and, for equipment supports, they include static and dynamic head and fluid flow effects.

Live loads include any movable equipment loads and other loads that vary with intensity and occurrence. For equipment supports, they also include loads caused by vibration and any support movement effects. Alternate load cases in which the magnitudes and locations of the live loads are arranged so that worst-case conditions are included in the design should be investigated, as appropriate.

As per 10 CFR 50, Appendix S, the OBE is only associated with plant shutdown and inspection unless the applicant specifically selects it as a design input. If the OBE is set at one-third or less of the SSE ground motion, an explicit response or design analysis is not required. If the OBE is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the containment internal structures remain functional and are within applicable stress, strain, and deformation limits. DSRS Section 3.7.1 and 3.7.2 provides further guidance on the use of OBE.

Hydrodynamic loads resulting from LOCA and/or actuation of ADVs into RWST including wall pressure loads; direct loads such as drag loads, jet impingements, impact loads, and reaction loads; and building dynamic response loads, need to be considered. Fluid structure interaction associated with these hydrodynamic loads and those from earthquakes should be taken into account. The definition of the loads, phasing of the loads, and method for combination of the loads (e.g., absolute sum, square root of the sum of the squares) are reviewed on a case-by-case basis.

The design of concrete structures needs to consider the loads and load combinations that may occur during their construction. These loads consist of dead loads, live loads, temperature, wind, snow, rain, and ice. Applicable construction loads include material loads, personnel and equipment loads, horizontal and vertical construction loads, erection and fitting forces, equipment reactions, and form pressure. Structural Engineering Institute (SEI)/ASCE Standard 37 provides additional guidance on construction loads. This standard may be used for supplemental guidance. When the standard and the Code/DSRS provide conflicting criteria, the criteria provided in the Code/DSRS governs.

B. Steel Structures

All loads and load combinations are to be in accordance with ANSI/AISC N690-1994 including Supplement 2 (2004). This specification uses the allowable stress design (ASD) method. Use of the load and resistance factor design (LRFD) version of the specification (N690L) is reviewed on a case-by-case basis. The supplemental criteria on the use of loads and load combinations presented above for concrete structures also apply to steel structures.

4. <u>Design and Analysis Procedures</u>. The design and analysis procedures used for the containment internal structures are acceptable if found to be in accordance with the following:

A. mPower™ Containment Internal Structures

i. Primary Shield Wall and Reactor Cavity

The design and analysis procedures used for the shield wall are acceptable if found to be in accordance with ACI 349 with additional guidance provided by RG 1.142. This code is based on the strength design method. The design and analysis of anchors (steel embedments) used for component and structural supports on concrete structures are acceptable if found to be in accordance with ACI 349, Appendix B, with additional guidance provided by RG 1.199.

Analyses for LOCA loads applicable to the primary shield wall, such as the cavity differential pressure combined with pipe rupture reaction forces, are acceptable if these loads are treated as dynamic time-dependent loads. This requires that either a detailed time-history analysis be performed or a static analysis using the peak of the forcing function amplified by an appropriate chosen dynamic factor be employed. Elastic behavior of the wall should be maintained under the differential pressure. However, for the concentrated accident loads, such as Y_r, Y_j, or Y_m, elasto-plastic behavior may be assumed if the deflections are limited to maintain functional requirements. Simplified methods for determining effective dynamic load factors for elastic behavior are acceptable if found to be in accordance with recognized dynamic analysis methods.

ii. Other Interior Structures

Most of the other interior structures that are reviewed are combinations of reinforced concrete slabs, walls, beams, and columns, and steel beams and columns, which are classified as seismic Category I structures subject to the loads and load combinations described in Subsection II.3 of this DSRS section.

Analytical techniques for these structures are acceptable if found to be in accordance with those described in ACI 349, and with additional guidance provided by RGs 1.142 and 1.199 for concrete and anchors (steel embedments), respectively, and with ANSI/AISC N690-1994 including Supplement 2 (2004) for steel.

B. For all containment internal structures, the design and analysis methods described in Subsections II.4 of DSRS Sections 3.8.2 and 3.8.4, which are applicable to the containment internal concrete and steel structures, also need to be considered. These items include assumptions on boundary conditions, axisymmetric and nonaxisymmetric loads, transient and localized loads, shrinkage and cracking of concrete, computer programs, and evaluation of liner plates and anchors.

C. Design of structures that use modular construction methods are reviewed on a case-by-case basis. NUREG/CR-6486 provides guidance related to the use of modular construction methods. Appendix B to NUREG/CR-6486 includes proposed modular construction review criteria.

D. A structural design audit is conducted as described in Appendix B to DSRS Section 3.8.4.

E. The applicant's design report is considered acceptable if it satisfies the guidelines of Appendix C to DSRS Section 3.8.4.

5. Structural Acceptance Criteria. The structural acceptance criteria for containment internal structures described in Subsection I.1 of this DSRS section are acceptable if found to be in accordance with the guidance given below. The structural acceptance criteria for structures that use modular construction methods are reviewed on a case-by-case basis. See Section II.4.C of this DSRS section for criteria relating to modular construction.

A. Concrete Structures

ACI 349 and RG 1.142 define the structural acceptance criteria for concrete structures. The structural acceptance criteria for anchors (steel embedments) used for support of systems and components to concrete structures are acceptable if found to be in accordance with Appendix B to ACI 349, with additional guidance provided by RG 1.199.

B. Steel Structures

ANSI/AISC N690-1994 including Supplement 2 (2004) defines the structural acceptance criteria for steel structures. This specification uses the ASD method. Use of the LRFD version of the specification (N690L) is reviewed on a case-by-case basis.

6. Materials, Quality Control, and Special Construction Techniques. The specified materials of construction and quality control programs are acceptable if found to be in accordance with the public code or standard as indicated in Subsection I.6 of this DSRS section.

   Special construction techniques, if any, are treated on a case-by-case basis. For modular construction, the materials, quality control, and special construction techniques are also reviewed on a case-by-case basis. See Section II.4.C of this DSRS section for further information.

7. Testing and Inservice Surveillance Requirements.

   For seismic Category I structures inside containment, structures monitoring and maintenance requirements are acceptable if found to be in accordance with 10 CFR 50.65 and RG 1.160.

   It is important that seismic Category I structures inside containment accommodate inservice inspection of critical areas. The staff considers that monitoring and maintaining the condition of the containment internal structures is essential for plant safety. Any special design provisions (e.g., providing sufficient physical access, providing alternative means for identifying conditions in inaccessible areas that can lead to degradation, remote visual monitoring of high radiation areas) to accommodate inservice inspection of containment internal structures are reviewed on a case-by-case basis.

Technical Rationale

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

1. Compliance with 10 CFR 50.55a requires that SSCs be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

   DSRS Section 3.8.3 cites RGs 1.69, 1.136, 1.142, 1.143, 1.160, 1.199, and 1.221 for guidance regarding design, construction, quality control, tests, and inspections that are acceptable. ACI 349, with additional guidance provided in RGs 1.142 and 1.199; ASME Code, Section III, Division 2, Subsection CC, with additional guidance provided by

RG 1.136; and ANSI/AISC N690-1994 including Supplement 2 (2004), contain criteria for concrete and steel structures.

Meeting these requirements and criteria provide assurance that the SSCs described herein will perform their safety function and limit the release of radioactive materials.

2. Compliance with GDC 1 requires that (1) SSCs important to safety be designed, fabricated, erected, and tested in accordance with quality standards commensurate with the importance of their safety function, (2) a quality assurance program be established and implemented, and (3) sufficient and appropriate records be maintained. When generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function.

   DSRS Section 3.8.3 provides guidance related to static and dynamic loadings and evaluation criteria for containment internal structures. It also describes acceptable materials, design methodology, quality control procedures, construction methods, and inservice inspections, as well as documentation criteria for design and construction controls.

   DSRS Section 3.8.3 cites ACI 349, with additional guidance given in RG 1.142; ASME Code, Section III, Division 2, Subsection CC, with additional guidance provided by RG 1.136; ANSI/AISC N690-1994 including Supplement 2 (2004); and regulatory guidance describing design methodology, materials testing, and construction techniques that are commensurate with the importance of the safety function to be performed. Conformance with these requirements imposes specific restrictions to ensure that containment internal structures will perform acceptably, commensurate with their intended safety function, when designed in accordance with the above standards.

   Meeting these requirements and criteria provide assurance that the SSCs described herein will perform their intended safety function.

3. Compliance with GDC 2 requires that SSCs important to safety be designed to withstand the effects of expected natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches, without a loss of capability to perform their safety functions. The design bases for these SSCs shall reflect appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.

   To ensure that structures of a nuclear power plant are designed to withstand natural phenomena, it is necessary to consider the most severe natural phenomena that have been historically reported with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated. These data shall be used to specify the design requirements of nuclear power plant components to be evaluated as part of construction permit, operating license (OL), COL, and ESP reviews, or for site parameter envelopes in the case of DCs, thereby ensuring that components important to safety will function in a manner that will maintain the plant in a safe condition.

   DSRS Section 3.8.3 provides detailed acceptance criteria and cites appropriate regulatory guidance for design methodology, materials testing, and construction techniques that are acceptable to the staff. GDC 2 requires that containment internal structures be designed to withstand the effects of natural phenomena, combined with those of normal and accident conditions, without a loss of capability to perform their safety function. Load combinations and specifications cited in this DSRS section provide acceptable engineering criteria to accomplish that function.

Meeting these requirements and criteria provide assurance that safety-related structures inside containment will be designed to withstand the effects of natural phenomena and will perform their intended safety function.

4. Compliance with GDC 4 requires that nuclear power plant SSCs important to safety be designed to accommodate the effects of, and be compatible with, environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. In addition, these SSCs must be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

   DSRS Section 3.8.3 provides methods acceptable to the staff, including load combinations, acceptance criteria, standards, and codes to assure compliance with GDC 4. Meeting these requirements and criteria provide assurance that the containment internal structures will withstand loads from internal events, such as those described above, and from external sources such as earthquakes, thus decreasing the probability that these events will damage containment internal structures.

   Meeting these requirements and criteria provide assurance that the internal structures of the containment will function as designed, be capable of maintaining their structural integrity, and perform their intended safety function.

5. Compliance with GDC 5 prohibits the sharing of structures important to safety among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

   The requirements of GDC 5 are imposed to ensure that the use of common structures in multiunit plants will not significantly affect the orderly and safe shutdown and cooldown in one plant in the event of an accident in another. The load combination equations combine loads from normal operation and design-basis accidents so that the resulting structural designs provide for mutual independence of shared structures.

   Meeting this requirement provides assurance that containment internal structures and their associated components are capable of performing their required safety functions, even if they are shared by multiple nuclear power units.

6. Compliance with GDC 50 requires that the reactor containment structure, including access openings, penetrations, and the containment heat removal system, be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

DSRS Section 3.8.3 provides detailed acceptance criteria and cites appropriate regulatory guidance for design methodology, material testing, and construction techniques that are acceptable to the staff. GDC 50 requires that the internal structures of the containment be designed to withstand the effects of natural phenomena, combined with those of normal and accident conditions, including LOCA loads, without a loss of capability to perform their safety function. The load combinations and specifications cited in this DSRS section provide acceptable engineering criteria to accomplish that function.

Meeting these requirements and criteria provide assurance that the internal structures of the containment will perform their intended safety function with sufficient margin when subjected to LOCA loads in combination with other applicable loads.

7. Compliance with 10 CFR 50, appendix B, requires that applicants establish and maintain a quality assurance program for the design, construction, and operation of SSCs.

   DSRS Section 3.8.3 provides guidance specifically related to design, construction, testing and inservice surveillance of structural concrete and steel of containment internal structures. Subsection II.2 of this DSRS section cites ACI 349, with additional guidance provided by RG 1.142 and 1.199; ASME Code, Section III, Division 2, Subsection CC, with additional guidance provided by RG 1.136; ANSI/AISC N690-1994 including Supplement 2 (2004); and other regulatory guides to satisfy the requirements of 10 CFR 50, Appendix B.

   Meeting these requirements and criteria provide assurance that structures covered in this DSRS section will meet the requirements of 10 CFR 50, Appendix B and thus perform their intended safety function.

III. <u>REVIEW PROCEDURES</u>

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. 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.

2. <u>Description of the Internal Structures</u>. After the type of structure and its functional characteristics are identified, the reviewer will obtain information on similar containment internal structures previously licensed for reference. Such information, which is

available in SARs and amendments of previous license applications, can identify differences in the case under review. These differences require additional justification and evaluation for meaningful review. New and unique features that have not been used in the past are examined in greater detail.

The reviewer evaluates the information furnished in the SAR for completeness in accordance with RG 1.70 for a CP or an OL (for applications submitted in accordance with 10 CFR Part 50) or RG 1.206 for a DC or a COL (for application submitted in accordance with 10 CFR Part 52).

3. Applicable Codes, Standards, and Specifications. The list of codes, standards, guides, and specifications is checked against the list in Subsection II.2 of this DSRS section. The reviewer verifies the use of the appropriate code or guide and the acceptability of the applicable edition and stated effective addenda.

4. Loads and Loading Combinations. The reviewer verifies that the loads and load combinations are consistent with those specified in Subsection II.3 of this DSRS section. Any deviations from the acceptance criteria for loads and load combinations are reviewed for adequate justification.

5. Design and Analysis Procedures. The reviewer becomes familiar with the design and analysis procedures that are generally used for the type of structures being reviewed. Because the assumptions regarding the expected behavior of the structure and its various elements under loads may be significant, the reviewer evaluates their acceptability based on the acceptance criteria provided in Section II. The design and analysis procedures, including the behavior of the structures under various loads and the manner in which these loads are treated in conjunction with other coexistent loads are reviewed to establish compliance with procedures delineated in Subsection II.4 of this DSRS section. These include the criteria for computer programs, consideration of concrete cracking, design reports, and the structural audit.

   As discussed in Subsection II.4.C of this DSRS section, the use of modular construction methods is reviewed on a case-by-case basis using guidance provided in NUREG/CR-6486.

6. Structural Acceptance Criteria. The limits on allowable stresses and strains in the structural elements, including concrete, reinforcement, structural steel, and anchors, are compared with those specified in Subsection II.5 of this DSRS section. The reviewer evaluates the justification provided to demonstrate that the functional integrity of the structure will not be affected, when the applicant proposes to exceed some of these limits for certain load combinations and at certain localized points on the structure.

7. Materials, Quality Control, and Special Construction Techniques. The information provided on materials, quality control programs, and special construction techniques, if any, is reviewed and compared with that specified in Subsection II.6 of this DSRS section. For a new material that has not been used in prior license applications, the applicant is requested to provide sufficient test and user data to establish the acceptability of such a material. Similarly, the reviewer evaluates any new quality control programs or construction techniques to ensure that no degradation of structural quality that might affect the structural integrity of the structure will occur.

8. Testing and Inservice Surveillance Requirements. Proposed testing and inservice surveillance programs are reviewed on a case by case basis.

   For containment internal structures, the reviewer verifies that structure monitoring and maintenance requirements are in accordance with 10 CFR 50.65 and RG 1.160.

   Any special design provisions (e.g., providing sufficient physical access, providing alternative means for identification of conditions in inaccessible areas that can lead to degradation, remote visual monitoring of high radiation areas) to accommodate inservice inspection of containment internal structures are reviewed on a case-by-case basis.

   Any other proposed testing and inservice surveillance programs are reviewed on a case by case basis.

9. Design Certification/Combined License Application Reviews. 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 final safety analysis report (FSAR) meets 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 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).

   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.

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.

The staff concludes that the design of the containment internal structures is acceptable and meets the relevant requirements of 10 CFR 50.55a, 10 CFR 50, Appendix B, and GDC 1, 2, 4, 5, and 50. This conclusion is based on the following:

1. The applicant has met the requirements of Section 50.55a and GDC 1 with respect to ensuring that the containment internal structures are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the safety function to be performed.

2. The applicant has met the requirements of GDC 2 by designing the containment internal structures to withstand the most severe earthquake that has been established for the site with sufficient margin, as well as the combinations of the effects of normal and accident conditions with the effects of environmental loadings, such as earthquakes and other natural phenomena.

3. The applicant has met the requirements of GDC 4 by ensuring that the design of the containment internal structures is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

4. The applicant has met the requirements of GDC 5 by demonstrating that SSCs are not shared between units or that sharing will not impair their ability to perform their intended safety functions.

5. The applicant has met the requirements of GDC 50 by designing the containment internal structures to accommodate, with sufficient margin, the design leakage rate, calculated pressure, and temperature conditions resulting from accident conditions and by ensuring that the design conditions are not exceeded during the full course of the accident condition. In meeting these design requirements, the applicant has used the recommendations of the regulatory guides and industry standards indicated below. The applicant has also performed appropriate analysis to demonstrate that the ultimate capacity of the structures will not be exceeded and to establish the minimum margin of safety for the design.

6. The applicant has met the requirements of 10 CFR 50, Appendix B by providing a quality assurance program that includes adequate measures for implementing guidelines relating to structural design.

The criteria used in the design, analysis, and construction of the containment internal structures to account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime conform to established criteria, codes, standards, and specifications acceptable to the NRC staff. This includes meeting the positions of RGs 1.69, 1.136, 1.142, 1.143, 1.160, 1.199, and 1.221, as well as industry codes and standards ACI-349, with additional guidance in RG 1.142; ANSI/AISC N690-1994 including Supplement 2 (2004); and ASME Code, Section III, Division 2, Subsection CC, with additional guidance provided by RG 1.136.

The use of these criteria, as defined by the applicable codes, standards, and specifications; loads and loading combinations; design and analysis procedures; structural acceptance criteria; materials, quality control programs, and special construction techniques; and testing and inservice surveillance requirements, provides reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within the containment, the containment internal structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions.

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^{TM}-specific design certification (DC), combined license (COL), or early site permit (ESP) 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, COL, or ESP applications 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™ 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 52.17 (a)(1)(xii) and 10 CFR 52.79 (a)(41), for ESP and COL applications, respectively.

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

VI. REFERENCES

1. 10 CFR 50.55a, "Codes and Standards."

2. 10 CFR 50.65, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

3. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

4. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

5. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Dynamic Effects Design Basis."

6. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems, and Components."

7. 10 CFR Part 50, Appendix A, General Design Criterion 50, "Containment Design Bases."

8. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

9. 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants."

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

11. ACI 349, "Code Requirements for Nuclear Safety Related Concrete Structures," American Concrete Institute.

12. ANSI/AISC N690-1994 including Supplement 2 (2004), "Specification for Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities."

13. ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsection CC, "Code for Concrete Reactor Vessels and Containments," American Society of Mechanical Engineers.

14. NUREG/CR-6486, "Assessment of Modular Construction for Safety-Related Structures at Advanced Nuclear Power Plants," March 1997.

15. Regulatory Guide 1.69, "Concrete Radiation Shields for Nuclear Power Plants."

16. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

17. Regulatory Guide 1.136, “Materials, Construction, and Testing of Concrete Containments.”

18. Regulatory Guide 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (other than Reactor Vessels and Containments)."

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

20. Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

21. Regulatory Guide 1.199, "Anchoring Components and Structural Supports in Concrete."

22. Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

23. SEI/ASCE 37, "Design Loads on Structures During Construction," American Society of Civil Engineers, 2002.