U.S. NUCLEAR REGULATORY COMMISSION

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

3.8.2 STEEL CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Organization responsible for structural analysis reviews

Secondary - None

I. AREAS OF REVIEW

This section describes the review of areas relating to the mPower^{TM} steel containment while Design Specific Review Section (DSRS) 3.8.5 addresses the foundation of the mPower^{TM} containment.

The specific areas of review are as follows:

1. Description of the Containment

A. The descriptive information is reviewed, including plans and sections of the structure, to establish that sufficient information is provided to define the primary structural aspects and elements relied upon to perform the containment function. In particular, the type of steel containment is identified and its structural and functional characteristics are examined.

The mPower^{TM} containment geometry is an upright cylinder topped with a dome and supported on a concrete foundation built around the bottom portion of the steel shell, which is an inverted dome. The steel containment shell is continuous into the concrete foundation that supports the entire reactor service building. As such, the steel containment shell is expected to provide the pressure resisting structural integrity and leak tight barrier for the containment system. The concrete foundation that supports the containment, containment internals, and the reactor service building is addressed under DSRS Section 3.8.5.

The geometry of the containment, including sketches showing plan views at various elevations and sections in at least two orthogonal directions are reviewed. The arrangement of the containment and the relationship and interaction of the steel containment shell with its surrounding buildings and with its interior compartments, walls, and floors to determine the effect that these structures could have upon the design boundary conditions and the expected behavior of the steel containment shell when subjected to the design loads are reviewed. The ultimate heat sink (UHS), which is connected to but not part of the steel containment pressure boundary, is also reviewed with respect to its structural interaction with the steel containment.

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B. The general information related to the containment shell including the following is reviewed:

i. Since the bottom of the steel containment is continuous through an inverted dome, the staff will review the method by which the inverted dome and its supports are anchored to the concrete foundation. DSRS Section 3.8.5 covers the concrete foundation.

ii. The cylindrical portion of the shell, including major structural attachments to the inside and outside surface of the shell, such as beam seats, pipe restraints, crane brackets, shell stiffeners (if any) in the hoop and vertical directions, and external cooling flow channels, if applicable.

iii. The dome of the steel containment, including any steel reinforcement at the dome/cylinder junction, penetrations, and attachments to the inside and outside surface of the dome, such as supports for attached piping and any stiffening of the dome. Also, the UHS tank is reviewed with respect to the structural interaction with the steel containment, in order to determine its effects on the containment. In addition, the connection of the UHS to the containment shell is reviewed to determine the jurisdictional boundary between the tank and the containment.

iv. The major penetrations, or portions thereof, of the steel containment, to the limits defined by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereafter referred to as the ASME Code), Section III, Division 1, Subsection NE. This includes portions of the penetrations that are intended to resist pressure but are not backed by structural concrete, sleeved and unsleeved piping penetrations, mechanical system penetrations such as fuel transfer tubes, electrical penetrations, and access openings such as the equipment hatch and personnel locks.

2. <u>Applicable Codes, Standards, and Specifications</u>. The reviewer evaluates the information pertaining to design codes, standards, specifications, and regulatory guides, and other industry standards that are used in the design, fabrication, construction, testing, and inservice surveillance of the steel containment. The specific editions, dates, or addenda identified for each document are reviewed.

3. <u>Loads and Loading Combinations</u>. The information pertaining to the applicable design loads and various load combinations, with emphasis on the extent of compliance with ASME Code, Section III, Division 1, Subsection NE; Regulatory Guide (RG) 1.57; and Subsection II.3 of this DSRS section are reviewed. The loads normally applicable to steel containments include the following:

A. Those loads encountered during preoperational testing.

B. Those loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads resulting from operating temperatures, and hydrostatic loads such as those present in the UHS tank.

C. Those loads to be sustained during severe environmental conditions, including those induced by design wind (if not protected by a concrete building) and the operating-basis earthquake (OBE).

D. Those loads to be sustained during extreme environmental conditions, including those induced by the design-basis tornado and hurricane (if not protected by a concrete building) and the safe-shutdown earthquake (SSE) specified for the plant site.

E. Those loads to be sustained during abnormal plant conditions, which include loss-of-coolant accidents (LOCAs). The main abnormal plant condition for containment design is the design-basis LOCA. The staff will also consider other accidents involving various higher-energy pipe ruptures. Loads induced on the containment by such accidents include elevated temperatures and pressures and possibly localized loads such as jet impingement and associated missile impact, and may include building dynamic response loads. Also included are external pressure loads generated by events inside or outside the containment.

F. Those loads to be sustained, if applicable, after abnormal plant conditions, including flooding of the containment subsequent to a LOCA for fuel recovery.

G. Those hydrodynamic loads resulting from actuation of the automatic depressurization valves (ADVs) into the refueling water storage tank (RWST), which may include building dynamic response loads.

H. Those thermal loads associated with passive cooling of the external surface of the steel containment shell, including consideration of nonuniform distribution in the circumferential and meridional directions, if applicable.

I. Those loads associated with combustible gas generation from a metal-water reaction of the fuel cladding, in accordance with 10 CFR 50.44.

J. Those loads encountered during construction, including dead loads; live loads; temperature, wind, snow, rain, and ice; and construction loads that may be applicable such as material loads, personnel and equipment loads, erection and fitting forces, and equipment reactions.

Various combinations of the above loads are normally postulated and reviewed, including testing loads; normal operating loads; normal operating loads with severe environmental loads; normal operating loads with severe environmental loads and abnormal loads; normal operating loads with extreme environmental loads and abnormal loads; post-LOCA flooding loads with severe environmental loads, if applicable; and construction loads. Subsection II.3 of this DSRS section delineates specific and more detailed information on these combinations.

Unless a concrete building protects the steel containment, other site-related design loads might also be applicable, including those described in Subsection I.3 of DSRS Section 3.8.4.

4. Design and Analysis Procedures. The design and analysis procedures used for the steel containment, with emphasis on the extent of compliance with ASME Code, Section III, Division 1, Subsection NE, and RG 1.57.

   The review will place particular emphasis on the following subjects:

   A. Treatment of nonaxisymmetric and localized loads
   B. Treatment of local buckling effects
   C. Computer programs used in the design and analysis
   D. Ultimate capacity of steel containment
   E. Structural audit
   F. Design report

5. Structural Acceptance Criteria. The design limits imposed on the various parameters that serve to quantify the structural behavior of the containment, specifically with respect to allowable stresses, strains, and gross deformations, and emphasizing the extent of compliance with ASME Code, Section III, Division 1, Subsection NE, and RG 1.57 are reviewed. For each specified load combination, the reviewer will compare the proposed allowable limits with the acceptable limits delineated in Subsection II.5 of this DSRS section. These allowable limits include the following major parameters:

   A. Primary stresses, including limits on general membrane stress (P_m); local membrane stress (P_L); and primary bending stress (P_b) plus local membrane stress (P_L)

   B. Primary stress (P_L+P_b) plus secondary stress (Q)

   C. Primary stress (P_L+P_b) plus secondary stress (Q) plus peak stress (F)

   D. Buckling criteria

6. Materials, Quality Control, and Special Construction Techniques

   A. The information provided on the materials to be used in the construction of the steel containment, with emphasis on the extent of compliance with Article NE-2000 of ASME Code, Section III, Division 1, Subsection NE are reviewed. The major materials reviewed include the following:

      i. Steel plates used as shell components

      ii. Structural steel shapes used for stiffeners, beam seats, and crane brackets

   B. The quality control program proposed for the fabrication and construction of the containment with emphasis on the extent of compliance with Articles NE-2000, NE-4000, and NE-5000 of ASME Code, Section III, Division 1, Subsection NE, are reviewed including the following:

i. Nondestructive examination of the materials, including tests to determine their physical properties

ii. Welding procedures

iii. Erection tolerances

Any special construction techniques, if proposed, are reviewed on a case-by-case basis to determine their effects on the structural integrity of the completed containment.

7. Testing and Inservice Surveillance Program. The staff will review the preoperational structural test programs for the completed containment and for individual class MC components, including the objectives of the test and the acceptance criteria, with emphasis on the extent of compliance with Article NE-6000 of ASME Code, Section III, Division 1, Subsection NE. Structural tests for components such as personnel airlocks and equipment hatch locks are also reviewed.

   The inservice surveillance programs for components relied upon for containment structural integrity are reviewed with emphasis on whether it complies with ASME Code, Section XI, Subsection IWE, and additional requirements delineated in 10 CFR 50.55a.

   The staff should review any special testing and inservice surveillance requirements proposed for new or previously untried design approaches. It is important that the design accommodates inservice inspection of critical areas. The review will include 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 the steel containment.

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. The determination of structures that are subject to quality assurance programs in accordance with the requirements of Appendix B to 10 CFR Part 50 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. The determination of pressure loads from higher-energy lines located in safety-related structures other than containment is performed in accordance with SRP Section 3.6.1. The loads thus generated are accepted for inclusion in the load combination equations of this DSRS section.

3. The determination of loads generated by pressure under accident conditions is performed in accordance with DSRS Section 6.2.1. The loads thus generated are accepted for inclusion in the load combinations in this DSRS section.

4. 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.

5. The reviews of containment performance and the satisfaction of severe accident criteria are performed in accordance with SRP Section 19.0.

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 Part 50, Appendix A, General Design Criterion (GDC) 1 as they relate to designing, fabricating, erecting, testing, and inspecting steel containments to quality standards commensurate with the importance of the safety function to be performed

2. GDC 2, as it relates to designing steel containments to be capable of withstanding the most severe natural phenomena such as winds, tornados, hurricanes, floods, and earthquakes and the appropriate combination of all loads

3. GDC 4, as it relates to the capability of steel containments to withstand the dynamic effects of equipment failures, including missiles, pipe whipping, and blowdown loads associated with LOCAs

4. GDC 16, as it relates to the capability of the steel containment to act as a leak-tight membrane to prevent the uncontrolled release of radioactive effluents to the environment

5. GDC 50, as it relates to designing steel containments with sufficient margin of safety to accommodate appropriate design loads

6. 10 CFR 50.44, as it relates to the capability of the steel containment to resist those loads associated with combustible gas generation from a metal-water reaction of the fuel cladding

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.

<u>DSRS Acceptance Criteria</u>

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. <u>Description of the Containment</u>. The descriptive information in the safety analysis report (SAR) is acceptable if it meets the criteria set forth in Section 3.8.2.1 of RG 1.206.

   If the steel containment has new or unique features that RG 1.206 does not specifically cover, adequate information necessary to accomplish a meaningful review of the structural aspects of these new or unique features need to be presented such that an evaluation can be made that it is equivalent in function and complies with the applicable requirements.

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

2. <u>Applicable Codes, Standards, and Specifications</u>. Codes, standards, and specifications, acceptable either in their entirety or in part, cover the design, materials, fabrication, erection, inspection, testing, and inservice surveillance of steel containments. The following codes and guides are acceptable:

Code/Guide	Title
ASME Code	Section III, Division 1, Subsection NE, "Class MC Components"
ASME Code	Section XI, Subsection IWE, “Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants”
RG 1.7	Control of Combustible Gas Concentrations in Containment
RG 1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components
RG 1.216	Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure
RG 1.221	Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants

3. Loads and Loading Combinations. Currently, ASME Code, Section III, Division 1, Subsection NE does not explicitly state the loads and load combinations that should be considered in the design of steel containments. RG 1.57, “Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components” provides additional guidance for design requirements, including loads and load combinations, which should be considered in the design of steel containments.

The specified loads and load combinations are acceptable if found to be in accordance with the following:

A. Loads

D - Dead loads

L - Live loads, including all loads resulting from platform flexibility and deformation and from crane loading, if applicable

P_t - Test pressure

T_t - Test temperature

T_o - Thermal effects and loads during startup, normal operating, or shutdown conditions, based on the most critical transient or steady-state condition

R_o - Pipe reactions during startup, normal operating, or shutdown conditions, based on the most critical transient or steady-state condition

P_o - External pressure loads resulting from pressure variation either inside or outside containment

E - Loads generated by the OBE, including sloshing effects, if applicable

E' - Loads generated by the SSE, including sloshing effects, if applicable

P_a - Pressure load generated by the postulated pipe break accident (including pressure generated by postulated small-break or intermediate-break pipe ruptures) and hydrodynamic loads

Note: For loading combinations B, Service Conditions (iii), for (1)(d), (3)(c), and (4)(b), a small or intermediate pipe break accident is postulated; for all other load combinations, the design-basis LOCA is postulated.

T_a - Thermal loads under thermal conditions generated by the postulated pipe break accident and hydrodynamic reaction loads

Note: For loading combinations B, Service Conditions(iii), for (1)(d), (3)(c), and (4)(b), a small or intermediate pipe break accident is postulated; for all other load combinations, the design-basis LOCA is postulated.

R_a - Pipe reactions under thermal conditions generated by the postulated pipe break accident and hydrodynamic reaction loads

Note: For loading combinations B, Service Conditions(iii), for (1)(d), (3)(c), and (4)(b), a small or intermediate pipe break accident is postulated; for all other load combinations, the design-basis LOCA is postulated.

P_s - All pressure loads that are caused by the actuation of ADV discharge, including hydrodynamic loads, if applicable

T_s - All thermal loads that are generated by the actuation of ADV discharge, including hydrodynamic thermal loads, if applicable

R_s - All pipe reaction loads that are generated by the actuation of ADV discharge, including hydrodynamic reaction loads, if applicable

Y_r - Equivalent static load on the structure generated by the reaction on the broken pipe during the design-basis accident

Y_j - Jet impingement equivalent static load on the structure generated by the broken pipe during the design-basis accident

Y_m - Missile impact equivalent static load on the structure generated by or during the design-basis accident, such as pipe whipping

F_L - Load generated by the post-LOCA flooding of the containment, if applicable

P_{g1} - Pressure load generated from 100-percent fuel clad metal-water reaction

P_{g2} - Pressure loads generated by hydrogen burning, if applicable

B. Loading Combinations

The loading combinations for which the containment might be designed or subjected to during the expected life of the plant include the following:

i. Testing Condition

This includes the testing condition of the containment to verify its leak integrity. The loading combination in this case includes -

$D + L + T_t + P_t$

ii. Design Conditions

These include all design loadings for which the containment vessel or portions thereof might be designed during the expected life of the plant. Such loads include design pressure, design temperature, and the design mechanical loads generated by the design-basis LOCA. The loading combination in this case includes -

$D + L + P_a + T_a + R_a$

iii. Service Conditions

The load combinations in these cases correspond to and include Level A service limits, Level B service limits, Level C service limits, Level D service limits, and the post-flooding condition. The loads may be combined by their actual time history of occurrence taking into consideration their dynamic effect upon the structure.

(1) Level A Service Limits

These service limits are applicable to the service loadings to which the containment is subjected, including the plant or system design-basis accident conditions for which the containment function is required, except only those categorized as Level B, Level C, Level D, or testing loadings. The loading combinations corresponding to these limits include the following:

(a) Normal operating plant condition

$D + L + T_o + R_o + P_o$

(b) Operating plant condition in conjunction with the actuation of ADVs

$D + L + T_s + R_s + P_s$

(c) Design-basis LOCA

$$D + L + T_a + R_a + P_a$$

(d) ADV actuations in combination with small- or intermediate-break accident

$$D + L + T_a + R_a + P_a + T_s + R_s + P_s$$

(2) Level B Service Limits

These service limits include the loads subject to Level A service limits plus the additional loads resulting from natural phenomena during which the plant must remain operational. The loading combinations corresponding to these limits include the following:

(a) Design-basis LOCA in combination with OBE (if $E \leq$ one-third E', only its contribution to cyclic loading needs to be considered)

$$D + L + T_a + R_a + P_a + E$$

(b) Operating plant condition in combination with OBE (if $E \leq$ one-third E', only its contribution to cyclic loading needs to be considered)

$$D + L + T_o + R_o + P_o + E$$

(c) Operating plant condition in combination with OBE and ADV actuations (if $E \leq$ one-third E', only its contribution to cyclic loading needs to be considered)

$$D + L + T_s + R_s + P_s + E$$

(d) Design-basis LOCA in combination with a single active component failure causing ADV discharge

$$D + L + T_a + P_a + R_a + T_s + R_s + P_s$$

(3) Level C Service Limits

These service limits include the loads subject to Level A service limits plus the additional loads resulting from natural phenomena for which safe shutdown of the plant is required. The loading combinations corresponding to these limits include the following:

(a) Design-basis LOCA in combination with SSE

$$D + L + T_a + R_a + P_a + E'$$

(b) Operating plant condition in combination with SSE

$$D + L + T_o + R_o + P_o + E'$$

(c) ADV actuations in combination with small- or intermediate-break accident and SSE

$$D + L + T_a + R_a + P_a + T_s + R_s + P_s + E'$$

(d) Dead load plus pressure resulting from an accident that releases hydrogen generated from 100-percent fuel clad metal-water reaction accompanied by hydrogen burning (10 CFR 50.44)

$$D + P_{g1} + P_{g2}$$

Note: In this load combination, $P_{g1} + P_{g2}$ should not be less than 310 kilo Pascals (kPa) or 45 pounds per square in gauge (psig).

(4) Level D Service Limits

These service limits include other applicable service limits and loadings of a local dynamic nature for which the containment function is required. The load combinations corresponding to these limits include the following:

(a) Design-basis LOCA in combination with SSE and local dynamic loadings

$$D + L + T_a + R_a + P_a + Y_r + Y_j + Y_m + E'$$

(b) ADV actuations in combination with small- or intermediate-break accident, SSE, and local dynamic loadings

$$D + L + T_a + R_a + P_a + Y_r + Y_j + Y_m + P_s + T_s + R_s + E'$$

(5) Post-flooding Condition

This includes the post-LOCA flooding of the containment in combination with OBE-basis earthquake

$$D + L + F_L + E$$

C. Construction Loads

Temporary construction loads and the effects of environmental loads during the construction stage need to be considered. ASME Code, Section III, Subsection NE, does not address this. The sections of Structural Engineering

Institute/American Society of Civil Engineers (SEI/ASCE) Standard 37-02 pertaining to steel structures may be used for guidance.

D. External Environmental Loads

A concrete building typically protects steel containments from the environment. If environmental loads external to the steel containment (e.g., wind, tornado, hurricane, external flooding) either directly or indirectly impose loads on the steel containment, the design of the steel containment also needs to consider these loads. Load combinations and acceptance criteria that are consistent with those specified in RG 1.136 for concrete containments should be used.

As noted in 10 CFR 50, Appendix S, the OBE is only associated with plant shutdown and inspection, unless specifically selected by the applicant as a design input. If the OBE is set at one-third or less of the SSE ground motion, explicit analysis is not required. The only exceptions are the post-flooding condition and cyclic loading considerations. The staff requirements memorandum for SECY-93-087 provides guidance on the treatment of cyclic loading for the OBE. If the OBE is set at a value greater than one-third of the SSE, explicit analysis must be performed to demonstrate that the applicable load combinations meet the Service Level B stress, strain, deformation, and fatigue limits.

4. <u>Design and Analysis Procedures</u>. Article NE-3000 of ASME Code, Section III, Division 1, Subsection NE, covers design and analysis procedures for steel containments. The procedures given in the ASME Code, with additional guidance provided in the applicable provisions of RGs 1.7, 1.57, and 1.216, constitute an acceptable basis for design and analysis. Moreover, for the specific areas of review described in Subsection I.4 of this DSRS section, the following criteria are acceptable:

A. Treatment of Nonaxisymmetric and Localized Loads

For most containments, the nonaxisymmetric loads encountered are the horizontal seismic loads and associated sloshing loads, as well as hydrodynamic loads caused either by LOCA or by ADV actuation. Other possible nonaxisymmetric and localized loads are those induced by pipe rupture, such as reactions, jet impingement forces, and missiles. For such localized loads, the analyses should include a determination of the local effects of the loads. These effects should then be superimposed on the overall effects. For the overall effects of nonaxisymmetric loads on shells of revolution, an acceptable general procedure is to expand the load by a Fourier series. Any other applicable methods proposed for a large thin shell, will be reviewed on a case-by-case basis.

B. Treatment of Buckling Effects

Earthquake loads and localized pressure loads require consideration of shell buckling. An acceptable approach to the problem is to perform a nonlinear dynamic analysis. If a static analysis is performed, an appropriate dynamic load factor should be used to obtain the effective static load.

Subarticle NE-3133 of ASME Code, Section III, Division 1, Subsection NE, is acceptable to address buckling of shell geometries and loadings covered therein. Buckling of shells with more complex geometries or loading conditions than those covered by Subarticle NE-3133 may be considered in accordance with the criteria described in ASME Code Case N-284, Revision 1, with additional guidance provided in RG 1.193. Each application of ASME Code Case N-284, Revision 1, is subject to review on a case-by-case basis.

Buckling of shells under internal pressure (e.g., torispherical heads) may also be considered in accordance with the criteria described in ASME Code Case N-284, Revision 1, with guidance provided in RG 1.193. Each application of ASME Code Case N-284, Revision 1, is subject to review on a case by case basis.

The staff will review the use of alternate methodologies to address the buckling of steel containments on a case-by-case basis.

RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III” and RG 1.193, "Code Cases Not Approved for Use,” provide additional guidance for code case acceptability which should be considered in the design of steel containments. Any Code cases not currently approved by NRC requires review on a case by case basis.

C. Computer Programs

The computer programs used in the design and analysis should be described and validated by procedures or criteria described in Subsection II.4.D of DSRS Section 3.8.4.

D. Ultimate Capacity of Steel Containment

In accordance with GDC 50 and 10 CFR 50.44, a determination of the internal pressure capacity for containment structures, as a measure of the safety margin above the design-basis accident pressure, is needed. The design and analysis procedures are acceptable if performed in accordance with RG 1.216.

E. Structural Audit

Structural audits are conducted as described in DSRS Section 3.8.4, Appendix B.

F. Design Report

The design report is considered acceptable when it satisfies the guidelines provided in DSRS Section 3.8.4, Appendix C.

5. <u>Structural Acceptance Criteria</u>. Stresses at various locations of the shell of the containment for various design loads are determined by analysis. Total stresses for the combination of loads delineated in Subsection II.3 of this DSRS section are acceptable if found to be within the limits defined by ASME Code, Section III, Division 1, Subsection NE, Subarticles NE-3221.1, NE-3221.2, NE-3221.3, and NE-3221.4 for Service Levels A, B, C, and D, respectively.

For the post-flooding load combination (Subsection II.3.B(iii)(5)), Service Level C limits apply to primary stress, and Service Level B limits apply to primary plus secondary stress. Evaluation of primary plus secondary plus peak stress is not required.

If external environmental loads need to be considered in the steel containment design, the staff will review the adequacy of the approach and acceptance criteria on a case-by-case basis.

6. Materials, Quality Control, and Special Construction Techniques

   A. The materials of construction are acceptable if they are in accordance with Article NE-2000 of ASME Code, Section III, Division 1, Subsection NE. The organization responsible to review material properties will review corrosion protection.

   B. Quality control programs are acceptable if they are in accordance with Articles NE-2000, NE-4000, and NE-5000 of ASME Code, Section III, Division 1, Subsection NE.

   C. The acceptability of special construction techniques, if any, are evaluated on a case-by-case basis.

   D. The staff will review the consideration of temporary construction loads and the effects of environmental loads during the construction stage on a case-by-case basis.

7. Testing and Inservice Surveillance Requirements

   A. Procedures for the preoperational structural proof test are acceptable if the procedures are in accordance with Article NE-6000 of ASME Code, Section III, Division 1, Subsection NE.

   B. For steel containments, 10 CFR 50.55a requires examination be conducted as outlined in ASME Code Section XI, Subsection IWE. Subsection IWE provides preservice examination, inservice inspection, and repair/replacement requirements and corresponding acceptance criteria. The scope of Subsection IWE includes the steel containment shell; integral attachments; containment hatches and airlocks; seals, gaskets, and moisture barriers; and pressure-retaining bolting. 10 CFR 50.55a(b)(2) specifies the acceptable edition of the ASME Code and additional requirements beyond those contained in Subsection IWE. 10 CFR 50.55a (b)(2)(viii)(E) requires that licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas.

   C. The staff will review 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 the steel containment on a case-by-case basis.

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 requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of their safety function, that a quality assurance program be established and implemented, and that sufficient and appropriate records be maintained. Where generally recognized codes and standards are used, they should be identified and evaluated to determine their applicability, adequacy, and sufficiency and should be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

   This DSRS Section provides guidance related to static and dynamic loadings and evaluation programs for steel containments. 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.

   This DSRS Section cites RG 1.57 for guidance regarding load combination equations, and ASME Code, Section III, Division 1, Subsection NE, provides acceptable design guidance and acceptance criteria.

   Meeting these criteria provides reasonable assurance that engineering analysis and design of steel containments for nuclear power plants will comply with GDC 1, and that steel containments will perform their intended safety function to prevent or mitigate the spread of radioactive material.

2. Compliance with GDC 2 requires that SSCs important to safety be designed to withstand the effects of expected natural phenomena such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without 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 the containment of a nuclear power plant is designed to withstand natural phenomena, it is necessary to consider the most severe natural phenomena that have been reported historically with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated. These data should be used to specify the design requirements of nuclear power plant components to be evaluated as part of construction permit, operating license, COL, and Early Site Permit 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.

   This DSRS Section and RG 1.57 provide guidance related to load combination equations, and ASME Code, Section III, Division 1, Subsection NE, provides acceptable stress and deformation limits for evaluating the effects of natural phenomena, in combination with normal and accident conditions.

   Meeting these criteria provides reasonable assurance that steel containment structures will be designed to withstand the effects of natural phenomena without loss of capability to perform their intended function as required by GDC 2.

3. 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. It also requires that they 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.

   This DSRS Section provides methods, including load combinations, acceptance criteria, standards, and codes, to ensure compliance with GDC 4.

4. Compliance with GDC 16 requires that reactor containment and associated systems be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to ensure that design conditions important to safety are not exceeded for as long as required for postulated accident conditions.

   Steel containments should be designed, constructed, and tested in accordance with ASME Code, Section III, Division 1, Subsection NE, to provide a leak-tight barrier. A typical steel containment consists of a thin-walled steel cylinder and top closure head. The wall thickness should be increased around penetrations to compensate for the openings. Penetrations (e.g., personnel locks, equipment hatches, and mechanical and electrical penetrations) should also be designed in accordance with ASME Code, Section III, Division 1, Subsection NE. Seals provided at the penetrations must be designed to maintain containment integrity for design-basis accident conditions, including pressure, temperature, and radiation. Leak-tightness of the containment structure must be tested at regular intervals during the life of the plant in accordance with the requirements of 10 CFR Part 50, Appendix J as described in DSRS Section 6.2.6.

   This DSRS Section provides methods, including load combinations, acceptance criteria, standards, and codes, acceptable to the staff to ensure compliance with GDC 16. Meeting these criteria provides reasonable assurance that an uncontrolled release of radioactivity to the environment will be prevented and that the design conditions of the reactor coolant pressure boundary will be maintained for as long as required.

5. Compliance with GDC 50 requires that the reactor containment structure, including access openings, penetrations, and containment heat removal systems, be designed so that the structure and its internal compartments will have the capability to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA. The calculated margin should reflect consideration of (1) the effects of potential energy sources that have not been included in the determination of the peak conditions and, as required by 10 CFR 50.44, the energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

   This DSRS Section provides acceptable methods, including load combinations, acceptance criteria, standards, and codes to ensure that the design of the containment can withstand the pressure loads and temperature conditions resulting from any LOCA.

This DSRS Section provides a deterministic methodology for estimating the ultimate pressure capacity of steel containments.

Meeting these criteria provides reasonable assurance that the containment structure, including the penetrations and the internal compartments, will be able to withstand the loads resulting from pressure and temperature conditions resulting from any LOCA, and will perform its designed safety function.

6. Compliance with 10 CFR 50.44 requires that containments accommodate loadings associated with combustible gas generated from a metal-water reaction of the fuel cladding.

   This DSRS Section provides load combinations and acceptance criteria that demonstrate that the steel containment structural integrity is maintained under these loads. RG 1.7 and RG 1.216 provide further guidance on the analytical technique, loading combination, and acceptance criteria.

   Meeting these criteria provides reasonable assurance that the containment will be able to withstand loads from the sources specified above and will perform its intended safety function as required by 10 CFR 50.44.

7. Compliance with 10 CFR 50.55a requires that (1) SSCs be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed, (2) containments, systems, and components of nuclear power reactors meet the requirements of the ASME Code, and (3) RG 1.84 and 1.147 provides guidance related to NRC approved ASME Code cases that may be applied to the design, fabrication, erection, construction, testing, and inspection of containments, systems, and components.

   Compliance with 10 CFR 50.55a also requires that examination of steel containments be performed in accordance with the requirements of ASME Code, Section XI, Subsection IWE, and supplemental requirements specified in 10 CFR 50.55a(b)(2)(ix). Subsection IWE provides requirements for preservice examination and inservice inspection, acceptance criteria, and repair/replacement requirements.

   This DSRS Section provides review guidance to ensure that the requirements of 10 CFR 50.55a have been appropriately addressed for steel containments.

Meeting the criteria of this subsection provides assurance that the containment structure will perform its safety function to limit the release of radioactive material throughout its licensing period.

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. 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. Description of the Containment. After the type of containment and its functional characteristics are identified, the reviewer will obtain information on similar steel containments previously licensed for reference. Such information, which is available in SARs and amendments of previous license applications, enables the identification of differences for the case under review that require additional scrutiny and evaluation. New and unique features that have not been used in the past are of particular interest and are thus 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 reviewer will check the list of codes, standards, guides, and specifications against the list in Subsection II.2 of this DSRS section. The reviewer will verify that the applicable edition and effective addenda are used.

4. Loads and Loading Combinations. The reviewer will verify that the loads and load combinations are consistent with those specified in Subsection II.3 of this DSRS section. Loading conditions that are unique, and not specifically covered in Subsection II.3, are treated on a case-by-case basis. The reviewer will identify any deviations from the acceptance criteria for loads and load combinations that have been adequately justified.

5. Design and Analysis Procedures. The reviewer will verify that the applicant is committed to the design and analysis procedures delineated in Article NE-3000 of ASME Code, Section III, Division 1, Subsection NE. Any exceptions to these procedures will be reviewed and evaluated on a case-by-case basis. The areas of review contained in Subsection I.4 of this DSRS section will be evaluated for conformance with the acceptance criteria.

6. Structural Acceptance Criteria. The reviewer will consider the limits on allowable stresses in the steel shell and its components and compare them with the acceptable limits specified in Subsection II.5 of this DSRS section. If, the applicant proposes to exceed some of these limits for some of the load combinations and at some localized points of the structure, the reviewer will evaluate the justification provided to show that the structural integrity of the containment will not be affected.

7. Materials, Quality Control, and Special Construction Techniques. The reviewer will compare the information provided on materials, quality control programs, and special construction techniques, if any, with that referenced in Subsection II.6 of this DSRS section. If a material used is not covered by the ASME Code, the applicant needs to provide sufficient test and user data to establish the acceptability of the material. Similarly, the reviewer will evaluate any new quality control programs or construction techniques to ensure that no degradation of structural quality will occur that may affect the structural integrity of the containment and its various components.

8. Testing and Inservice Surveillance Requirements. The reviewer will evaluate the initial structural overpressure test program and compare it with that indicated as acceptable in Subsection II.7 of this DSRS section. Any proposed deviations will be considered on a case-by-case basis. The staff will review inservice inspection programs in accordance with the requirements of 10 CFR 50.55a.

   The staff will review any special design provisions (e.g., providing sufficient physical access, providing alternative means for the identification of conditions in inaccessible areas that can lead to degradation, remote visual monitoring of high radiation areas) to accommodate inservice inspection 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 steel containment is acceptable and meets the relevant requirements of 10 CFR 50.44, 10 CFR 50.55a, and GDC 1, 2, 4, 16, and 50. This conclusion is based on the following:
>
> 1. The applicant has met the applicable requirements of 10 CFR 50.44 by designing the containment to withstand the pressure loads generated by fuel clad metal-water reaction and the subsequent burning of hydrogen, using the appropriate ASME Code service limits.

2. The applicant has met the requirements of 10 CFR 50.55a and GDC 1 with respect to ensuring that the steel containment is designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with its safety function to be performed by meeting the guidelines of the regulatory guides and industry standards indicated below.

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

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

5. The applicant has met the requirements of GDC 16 by designing the steel containment so that it is an essentially leaktight barrier to prevent the uncontrolled release of radioactive effluents to the environment.

6. The applicant has met the requirements of GDC 50 by designing the steel containment 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 regulatory guides and industry standards indicated below. The applicant has also performed appropriate analysis, which demonstrates that the ultimate capacity of the containment will not be exceeded and establishes the minimum margin of safety for the design.

The criteria used in the analysis, design, and construction of the steel containment structure to account for anticipated loadings and postulated conditions that may be imposed upon the structure during its service lifetime are in conformance with established criteria, codes, standards, and guides acceptable to the staff. These include compliance with the criteria of ASME Code, Section III, Division 1, Subsection NE and guidance provided in RG 1.57.

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

A seismic Category I concrete building protects the steel containment from the effects of wind, tornados, hurricanes, and various postulated accidents occurring outside the concrete building, and/or the steel containment has been evaluated for and will withstand the effects of wind, tornados, hurricanes and various postulated accidents occurring outside the containment that induce loading, either directly or indirectly, on the steel containment.

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^{TM} 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^{TM}-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^{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 52.17 (a)(1)(xii) and 10 CFR 52.79 (a)(41), for ESP and COL applications, respectively.

The referenced regulatory guides contain the implementation schedules for conformance to parts of the method discussed herein.

VI. REFERENCES

1. American Society of Mechanical Engineers. Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components," ASME.

2. American Society of Mechanical Engineers. Boiler and Pressure Vessel Code, Section XI, Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants," ASME.

3. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident."

4. Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components."

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

6. Regulatory Guide 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III."

7. Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1."

8. Regulatory Guide 1.193, "ASME Code Cases Not Approved for Use."

9. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standard and Records."

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

11. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Dynamic Effects Design Bases."

12. 10 CFR Part 50, Appendix A, General Design Criterion 16, "Containment Design."

13. 10 CFR Part 50, Appendix A, General Design Criterion 50, "Containment Design Basis."

14. 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

15. 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants."

16. 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors."

17. 10 CFR 50.55a, "Codes and Standards."

18. NUREG/CR-6906, "Containment Integrity Research at Sandia National Laboratories," July 2006.

19. SEI/ASCE Standard 37-02, "Design Loads on Structures During Construction," American Society of Civil Engineers, 2002.

20. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

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

22. Regulatory Guide 1.216, “Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure.”

23. Regulatory Guide 1.221, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants.”