U.S. NUCLEAR REGULATORY COMMISSION

# STANDARD REVIEW PLAN

## 3.8.2 STEEL CONTAINMENT

**REVIEW RESPONSIBILITIES**

**Primary -** Organization responsible for structural analysis reviews

**Secondary -** None

I. AREAS OF REVIEW

The staff will review areas relating to steel containments or to other Class MC steel portions of steel/concrete containments.

The specific areas of review are as follows:

1. Description of the Containment

   A. The descriptive information, 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 staff will review various types of steel containments, including the following:

      i. Steel boiling-water reactor (BWR) containments using the pressure-suppression concept, including the Mark I (lightbulb/torus), the

Mark II (over/under), and the Mark III (with horizontal venting between a centrally located cylindrical drywell and a surrounding suppression pool)

ii. Steel pressurized-water reactor (PWR) containments using the pressure-suppression concept with ice-condenser elements

iii. Steel PWR dry containments

These containments have various geometries. The geometry most commonly encountered is an upright cylinder topped with a dome and supported on either a flat concrete base mat covered with a liner plate or on a concrete foundation built around the bottom portion of the steel shell, which is an inverted dome. Although applicable to any geometry, the specific provisions of this Standard Review Plan (SRP) section are best suited to the cylindrical-type steel containment, surrounded by a Category I concrete shield building. If containments with other types of geometry are reviewed, the necessary modifications to this SRP section are made on a case-by-case basis.

iv. Steel components of concrete containments that resist pressure and are not backed by structural concrete (e.g., the drywell head in a BWR).

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 shield building 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.

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

i. The foundation of the steel containment, including the following:

(1) If 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. SRP Section 3.8.5 covers the concrete foundation.

(2) If the bottom of the steel containment is not continuous, and a concrete base slab, topped with a liner plate, is used for a foundation, the staff will review the method of anchorage of the steel cylindrical shell walls into the concrete base slab and the connection between the floor liner plate and the steel shell. SRP Section 3.8.1 provides review guidance.

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 (e.g., AP600, AP1000).

iii. The dome of the steel containment, including any reinforcement at the dome/cylinder junction, penetrations, and attachments to the inside and outside surface of the dome, such as supports for containment spray piping, any stiffening of the dome, and external cooling flow channels (e.g., AP600, AP1000).

iv. The major penetrations, or portions thereof, of steel or concrete containments, 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.

v. The ice-condenser containments, with special emphasis on those areas that are unique to this type of design, such as the connection between the ice-condenser and the containment.

vi. The BWR pressure suppression systems, paying special emphasis to piping that channels steam and air and is necessary for the containment function. This includes, but is not limited to, the torus, vent lines, vent header, equalizing ring header, and downcomers. In addition, the staff reviews the drywell/ventline junction, vent header/downcomers junctions, and penetrations to determine the expected behavior of the structure when subjected to the design loads.

2. <u>Applicable Codes, Standards, and Specifications</u>. The reviewer evaluates the information pertaining to design codes, standards, specifications, and regulatory guides (RGs), 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; RG 1.57; and Subsection II.3 of this SRP 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 pressure-suppression containments using water.

C. Those loads to be sustained during severe environmental conditions, including those induced by design wind (if not protected by a shield 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 (if not protected by a shield 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 high-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. 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 that are associated with BWR suppression pool swell phenomena and are produced as a result of the purging of air and steam in the drywell and vent system into the submersion pool during a postulated LOCA and/or the actuation of safety/relief valve (SRV) discharge. Such loads include bubble pressure, bulk swell, and froth swell loads, drag pressure, pool boundary chugging loads, and other pool well loads associated with these phenomena. The staff will also consider those loads resulting from fluid-structure interaction caused by seismic and/or pool swell.

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 (e.g., AP600, AP1000).

I. Those loads that are generated as a result of the LOCA in the ice-condenser. These loads are categorized as nonsymmetric dynamic transient pressure loads that in the first few seconds might produce compressive stresses in the containment because of the differential pressure across the containment.

J. For an existing plant subject to the requirements of Title 10 of the *Code of Federal Regulations*, Paragraph 50.34(f) (10 CFR 50.34(f)), those loads that are generated by pressure and dead loads during an accident that releases hydrogen generated from 100-percent fuel clad metal-water reaction and accompanied by either hydrogen burning or added pressure from postaccident inerting (10 CFR 50.34(f)(3)(v)(A)(1)); and those loads that are generated as a result of an inadvertent full actuation of a postaccident inerting hydrogen control system, excluding seismic or design-basis accident loadings (10 CFR 50.34(f)(3)(v)(B)(1)).

K. Those loads associated with combustible gas generation from a metal-water reaction of the fuel cladding, in accordance with 10 CFR 50.44, which defines the percent fuel cladding to be considered for existing and new plants.

L. 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 SRP section delineates specific and more detailed information on these combinations.

Unless a shield building protects the steel containment, other site-related design loads might also be applicable, including those described in Subsection I.3 of SRP Section 3.8.1.

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 SRP 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 hatches locks are also reviewed.

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

Any special testing and inservice surveillance requirements proposed for new or previously untried design approaches. For new reactors, it is important to accommodate 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 SRP section in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this SRP 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 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 SRP 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 high-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 SRP section.

3. The determination of loads generated by pressure under accident conditions is performed in accordance with SRP Section 6.2.1. The loads thus generated are accepted for inclusion in the load combinations in this SRP 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 exclusion of postulated pipe ruptures from the design basis is generally referred to as the leak before break. The review of those applications that propose to eliminate consideration of design loads associated with the dynamic effects of pipe rupture is performed in accordance with SRP Section 3.6.3.

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

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

## 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, 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 leaktight 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 50.34(f), as it relates to the capability of the steel containment of specific identified plants to resist (1) those loads that are generated by pressure and dead loads during an accident that releases hydrogen generated from 100-percent fuel clad metal-water reaction and accompanied by either hydrogen burning or added pressure from postaccident inerting, and (2) those loads that are generated as a result of an inadvertent full actuation of a postaccident inerting hydrogen control system, excluding seismic or design-basis accident loadings[1]

7. 10 CFR 50.44, as it relates to the capability of the steel containment of existing plants and new plants to resist those loads associated with combustible gas generation from a metal-water reaction of the fuel cladding

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[1]For Part 50 applicants not listed in 10 CFR 50.32(f), the provisions of 50.34(f) will be made a requirement during the licensing process.

8. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAACs that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act (AEA), and the U.S. Nuclear Regulatory Commission's (NRC's) regulations;

9. 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 COL, the provisions of the AEA, and the NRC's regulations.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for the review described in this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

1. Description of the Containment. 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.

   RG 1.70 provides guidance for information to be submitted with an application for construction permit (CP) or operating license (OL).

2. Applicable Codes, Standards, and Specifications. 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 |

3. Loads and Loading Combinations. Currently, ASME Code, Section III, Division 1, Subsection NE, and RG 1.57 do not explicitly state the loads and load combinations that should be considered in the design of steel containments. The staff has issued as a proposed revision to RG 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components." This draft guide or subsequent revision to RG 1.57 provides additional guidance for design requirements, including load 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), pool swell, and subsequent 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, pool swell, and subsequent 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, pool swell, and subsequent 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 SRV discharge, including pool swell and subsequent hydrodynamic loads, if applicable

$T_s$ — All thermal loads that are generated by the actuation of SRV discharge, including pool swell and subsequent hydrodynamic thermal loads, if applicable

$R_s$ — All pipe reaction loads that are generated by the actuation of SRV discharge, including pool swell and subsequent 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

$P_{g3}$ — Pressure load from postaccident inerting, assuming carbon dioxide is the inerting agent, 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 postflooding 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 multiple SRVs

$D + L + T_s + R_s + P_s$

(c) Design-basis LOCA

$D + L + T_a + R_a + P_a$

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

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

(e) Normal operating plant conditions in combination with inadvertent full actuation of a postaccident inerting hydrogen control system (10 CFR 50.34(f)(3)(v)(B)(1))

$D + L + T_o + R_o + P_o + P_{g3}$

(f) Pressure test load to ensure that the containment will safely withstand the pressure calculated to result from carbon-dioxide inerting (10 CFR 50.34(f)(3)(v)(B)(2))

$D + 1.10 \times P_{g3}$

(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 multiple SRV 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 one SRV 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) Multiple SRV 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 50.34(f)(3)(v)(A)(1), 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).

(e) Dead load plus pressure resulting from an accident that releases hydrogen generated from 100-percent fuel clad metal-water reaction accompanied by the added pressure from postaccident inerting, assuming carbon dioxide as the inerting agent (10 CFR 50.34(f)(3)(v)(A)(1))

$D + P_{g1} + P_{g3}$

Note: In this load combination, $P_{g1} + P_{g3}$ should not be less than 310 kPa or (45 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) Multiple SRV 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) Postflooding 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 shield building typically protects steel containments from the environment. If environmental loads external to the steel containment (e.g., wind, tornado, 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 SRP Section 3.8.1 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 postflooding 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. Design and Analysis Procedures. 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 RG 1.57, constitute an acceptable basis for design and analysis. Moreover, for the specific areas of review described in Subsection I.4 of this SRP section, the following criteria are acceptable:

   A. Treatment of Nonaxisymmetric and Localized Loads

      For most containments, the nonaxisymmetric loads that apply are the horizontal seismic and associated sloshing loads, pool swell, and its related hydrodynamic loads caused either by LOCA or by SRV actuation. Other possible nonaxisymmetric and localized loads are those induced by pipe rupture, such as reactions, jet impingement forces, and missiles. For the PWR ice-condenser containment, the design-basis accident may result in a nonaxisymmetric pressure load caused by compartmentation of the containment interior. 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 (such as those encountered in PWR ice-condenser containments) 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 require 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.e of SRP Section 3.8.1.

D. Ultimate Capacity of Steel Containment

For new reactors, regulatory criteria require a determination of the internal pressure capacity for containment structures, as a measure of the safety margin above the design-basis accident pressure.

One methodology acceptable to the staff for cylindrical steel containments is to estimate the capacity based on attaining a maximum global membrane strain away from discontinuities (i.e., the hoop membrane strain in a cylinder) of 1.5 percent.

To conduct the necessary analysis, both nonlinear material behavior and nonlinear geometric behavior must be considered. The stress-strain curve for the steel containment material should be based on the code-specified minimum yield strength and a stress-strain relationship above yield that is representative of that specific grade of steel. The stress-strain curve must be developed for the design-basis accident temperature.

Analyses of noncylindrical containments and analyses of cylindrical containments that use alternate failure criteria will be subject to detailed staff review, on a case-by-case basis.

The NRC has published guidance on computer modeling of steel containments for internal pressure capacity calculations in NUREG/CR-6906.

Note: In applying the analysis methodology to existing containment structures, it is permissible to use as-built material properties for the steel containment material. Sufficient material certification data must be available to establish with reasonable confidence a lower bound, a median, and an upper bound value for the important material parameters. These values must be adjusted for the design-basis accident temperature. For deterministic assessments, the lower bound values should be used. For probabilistic risk assessment, calculations of failure probability vs. pressure should consider the statistical distribution of the material properties.

<u>Containment Penetrations</u>: The methodology described above applies to the containment structure. A complete evaluation of the internal pressure capacity must also address major containment penetrations, such as the removable

drywell head and ventlines for BWR designs, equipment hatches, personnel airlocks, and major piping penetrations. Other potential containment leak paths through mechanical and electrical penetrations should also be addressed.

Special Considerations for Steel Ellipsoidal and Torispherical Heads: Under internal pressure, a potential failure mode of steel ellipsoidal and torispherical heads is buckling, resulting from a hoop compression zone in the knuckle region. This potential mode of failure needs to be evaluated, to determine if it is the limiting condition for the pressure capacity of the containment. The analysis should consider nonlinear material and geometric behavior and address the effect of initial geometric imperfections either explicitly (direct modeling) or implicitly (through the use of appropriate imperfection sensitivity knockdown factors). If appropriately demonstrated, residual postbuckling strength can be considered in determining the pressure capacity.

The details of the analysis and the results should be submitted in a report form with the following identifiable information:

i. Original design pressure, P, as defined in ASME Code, Section III, Division 1, Subsection NE, Subarticle NE-3112.1

ii. Calculated static pressure capacity

iii. Equivalent static pressure response calculated from dynamic pressure

iv. Associated failure mode

v. Criteria governing the original design and the criteria used to establish failure

vi. Analysis details and general results

vii. Appropriate engineering drawings adequate to allow verification of modeling and evaluation of analyses employed for the containment structure

E. Structural Audit

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

F. Design Report

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

5. Structural Acceptance Criteria. 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 SRP 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 postflooding 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 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 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 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 SRP 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.

   SRP Section 3.8.2 describes 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.

   SRP Section 3.8.2 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 assurance that engineering analysis and design of steel containments for nuclear power plants will comply with 10 CFR Part 50, 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 CP, OL, COL, and early site permit (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.

   SRP Section 3.8.2 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 this requirement provides assurance that steel containment structures will be designed to withstand the effects of natural phenomena without loss of capability to perform their intended function.

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.

   SRP Section 3.8.2 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 leaktight 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 are designed, constructed, and tested in accordance with ASME Code, Section III, Division 1, Subsection NE, to provide a leaktight barrier. A typical steel containment consists of a thin-walled steel cylinder and top closure head. The wall thickness is increased around penetrations to compensate for the openings. Penetrations (e.g., personnel locks, equipment hatches, and mechanical and electrical penetrations) are also 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. Leaktightness 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 the SRP Section 6.2.6.

   SRP Section 3.8.2 provides methods, including load combinations, acceptance criteria, standards, and codes, acceptable to the staff to ensure compliance with GDC 16. Meeting these criteria provides 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 (such as energy in steam generators) 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.

   SRP Section 3.8.2 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.

SRP Section 3.8.2 provides a deterministic methodology for estimating the ultimate pressure capacity of steel containments, and also provides review guidance when alternate methodologies are implemented.

Meeting these requirements provides 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 design safety function.

6. Compliance with 10 CFR 50.34(f)(3)(v)(A) and (B) requires that the containments for specific plants be capable of resisting loads associated with hydrogen generation equivalent to 100-percent metal-water reaction of the fuel cladding, accompanied by hydrogen burning or the added pressure of inerting system actuation. At a minimum, 10 CFR 50.34(f)(3)(v)(A)(1) requires that containment structures be designed to withstand a combined dead load and internal pressure of 310 kPa (45 psig).

   SRP Section 3.8.2 provides load combinations and acceptance criteria based on the specific provisions of 10 CFR 50.34(f)(3)(v)(A) and (B).

   Meeting the requirements of 10 CFR 50.34, specifically 10 CFR 50.34(f)(3)(v)(A) and (B), provides assurance that the containment will be able to withstand loads from the sources specified above and will perform its intended safety function.

7. 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.

   SRP Section 3.8.2 provides load combinations and acceptance criteria that demonstrate that the steel containment structural integrity is maintained under these loads. RG 1.7, Revision 2, is referenced for further guidance on the analytical technique, loading combination, and acceptance criteria.

   Meeting these requirements provides assurance that the containment will be able to withstand loads from the sources specified above and will perform its intended safety function.

8. 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 boiling and pressurized water-cooled nuclear power reactors meet the requirements of the ASME Code, and (3) RGs 1.84 and 1.147 provide 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.

SRP Section 3.8.2 provides review guidance to ensure that the requirements of 10 CFR 50.55a have been appropriately addressed for steel containments.

Meeting the requirements 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

The reviewer will select material from the procedures described below, as may be appropriate for a particular case.

These review procedures are based on the identified SRP 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. Description of the Containment. After the type of containment and its functional characteristics are identified, information on similar and previously licensed applications is obtained 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).

2. 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 SRP section. The reviewer will verify that the applicable edition and effective addenda are used.

3. 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 SRP 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.

4. 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 SRP section will be evaluated for conformance with the acceptance criteria.

5. 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 SRP 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.

6. 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 SRP 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.

7. 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 SRP 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 50.55.

   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.

8. 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. DCs have referred to the FSAR as the design control document. 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 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 50.34(f), 10 CFR 50.44, 10 CFR 50.55a, and GDCs 1, 2, 4, 16, and 50. This conclusion is based on the following:

1. The applicant has met the applicable requirements of 10 CFR 50.34(f) and 10 CFR 50.44 by designing the containment to withstand the pressure loads generated by fuel clad metal-water reaction, and either the subsequent burning of hydrogen or the added pressure from postaccident inerting, 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 RGs 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 RGs 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 Category I concrete shield building protects the steel containment from the effects of wind and tornados and various postulated accidents occurring outside the shield building, and/or the steel containment has been evaluated for and will withstand the effects of wind and tornados and various postulated accidents occurring outside the containment that induce loading, either directly or indirectly, on the steel containment (e.g., as for AP600, AP1000).

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 SRP 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 SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications submitted 6 months or more after the date of issuance of this SRP section, unless superseded by a later revision.

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

## VI. REFERENCES


1. ASME, Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components."

2. ASME, 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."

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

4. RG 1.57, “Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components.”

5. RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants.”

6. RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III.”

7. RG 1.147, “Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1.”

8. RG 1.193, “ASME Code Cases Not Approved for Use.”

9. 10 CFR Part 50, Appendix A, GDC 1, “Quality Standard and Records.”

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

11. 10 CFR Part 50, Appendix A, GDC 4, “Environmental and Dynamic Effects Design Bases.”

12. 10 CFR Part 50, Appendix A, GDC 16, “Containment Design.”

13. 10 CFR Part 50, Appendix A, GDC 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 50.34, “Contents of Application; Technical Requirements.”

17. 10 CFR 50.44, “Combustible Gas Control for Nuclear Power Reactors.”

18. 10 CFR 50.55a, “Codes and Standards.”

19. NUREG/CR-6906, “Containment Integrity Research at Sandia National Laboratories,” July 2006.

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

21. 10 CFR Part 52, ”Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants.”

22. RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition).“

**PAPERWORK REDUCTION ACT STATEMENT**

The information collections contained in the Standard Review Plan are covered by the requirements of 10 CFR Part 50 and 10 CFR Part 52, and were approved by the Office of Management and Budget, approval number 3150-0011 and 3150-0151.

**PUBLIC PROTECTION NOTIFICATION**

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

**SRP Section 3.8.2**
**“Steel Containment”**
**Description of Changes**

Revision 3 to SRP Section 3.8.2 updates Revision 2 of this section, dated March 2007, to reflect the following changes:

1. This SRP section is administratively updated by the Office of New Reactors, per request from Juan D. Peralta, Branch Chief, Quality and Vendor Branch 1, Division of Construction, Inspection, and Operational Programs, memorandum dated February 17, 2010 (ADAMS Accession No. ML10090148).