



U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD for NuScale DESIGN

3.8.2 STEEL CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Organization responsible for mechanical engineering reviews

Secondary - Organizations responsible for structural engineering and materials engineering reviews

I. AREAS OF REVIEW

This section describes the review of areas relating to the NuScale steel 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 NuScale containment vessel is an upright cylinder with closure heads at the top and bottom. A support skirt at the bottom of the containment vessel provides both vertical and horizontal support. In addition, lateral support lugs near the top of the vessel provide lateral support against the reactor compartment walls. The containment vessel along with the support skirt is immersed in the below-grade reactor pool which provides a passive heat sink. The containment vessel and the reactor pool are housed in the reactor building. The reactor building is a Seismic Category I reinforced concrete structure and also provides radiation protection to plant operation and maintenance personnel. The reactor building along with the below-grade, water-filled, stainless steel-lined, reinforced concrete reactor pool is reviewed in Design-Specific Review Standard (DSRS) Sections 3.8.4 and 3.8.5.

The geometry of the containment vessel, including sketches showing plan views at various elevations and sections in at least two orthogonal directions, is reviewed. The arrangement of the containment vessel shell wall and the reactor building pool is reviewed to determine the effect of interaction between the containment vessel and the reactor pool when subjected to the design loads.

The potential effects of any containment internals attached to the containment vessel shell are also reviewed.

- B. The general information related to the containment shell, including the following, is reviewed:
 - i. the cylindrical portion of the shell, including all penetrations, access openings, and attachments to the inside and outside surfaces of the shell
 - ii. the top and bottom closure heads of the steel containment, including closure head bolting and/or welding details and any steel reinforcement at the head/cylinder junction, all penetrations, access openings, and attachments to the inside and outside surfaces of the heads
- 2. Applicable Codes, Standards, and Specifications. The reviewer evaluates the information pertaining to design codes, standards, specifications, 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.

It is noted that the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code), Section III, Subsection NCA-2134(c) states that “Containment vessels classified as Class MC in their Design Specification may be constructed and stamped in accordance with the rules of Subsection NB, provided the rules of NE-7000 are applied in lieu of the rules of NB-7000 for protection against overpressure.” Because of similarities between the NuScale steel containment vessel and Class 1 pressure vessels covered by Subsection NB, the design certification (DC) applicant may follow the rules of ASME BPV Code Section III, Subsection NB, as incorporated by reference in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a, “Codes and Standards,” for construction and stamping of the containment vessel, and follow the rules of NE-7000 for protection against overpressure in lieu of the rules of NB-7000.

- 3. Loads and Loading Combinations. The information pertaining to the applicable design loads and various load combinations, with emphasis on the extent of compliance with ASME BPV Code, Section III, Division 1, Subsection NB, as incorporated by reference in 10 CFR 50.55a; conformance with RG 1.57; and consistency with Subsection II.3 of this DSRS section are reviewed. Loads should be compared for consistency with those established for other components, such as the reactor vessel, that are designed in accordance with ASME BPV Code, Section III, Division 1, Subsection NB, as reviewed under Section 3.9.3 of the Standard Review Plan (SRP). The loads applicable to the NuScale steel containment include the following:
 - A. Those loads encountered during pre-operational testing.
 - B. Those loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads resulting from operating temperatures, and external pressure loads due to submergence in the reactor pool.

- C. Those loads to be sustained during severe environmental conditions, including those induced by the operating-basis earthquake (OBE), if applicable. Dynamic fluid-structure interaction must be considered.
- D. Those loads to be sustained during extreme environmental conditions, including those induced by the design-basis safe-shutdown earthquake (SSE) specified for the plant site. Dynamic fluid-structure interaction should be considered.
- E. Those loads to be sustained during abnormal plant conditions, which include loss-of-coolant accidents (LOCAs). The staff will also consider other accidents involving various high-energy pipe ruptures, as applicable. 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.
- F. Those loads associated with combustible gas generation from a metal-water reaction of the fuel cladding, in accordance with 10 CFR 50.44.
- G. Those loads encountered during shop fabrication, transportation, and onsite erection.
- H. Because of the unique design of the NuScale reactor pressure vessel (RPV)/steel containment module, the steel containment is potentially subject to more severe cyclic temperature and pressure loads, compared to traditional containment designs. The fatigue evaluation of the NuScale containment vessel should be reviewed using the ASME BPV Code Section III, Subsection NB fatigue evaluation rules.

Various combinations of the above loads are normally postulated and reviewed. Subsection II.3 of this DSRS section delineates specific and more detailed information on these combinations.

- 4. Design and Analysis Procedures. The design and analysis procedures used for the steel containment are reviewed, with emphasis on the extent of compliance with ASME BPV Code, Section III, Division 1, Subsection NB, as incorporated by reference in 10 CFR 50.55a, and conformance with RG 1.57. The unique design features of the NuScale steel containment require consideration of the effects of radiation exposure on the material properties and fracture toughness. This is reviewed in SRP Section 6.2.7.

The review will place particular emphasis on the following subjects:

- A. treatment of nonaxisymmetric and localized loads
- B. treatment of local buckling effects
- C. consideration of cyclic fatigue

- D. effects of submergence in the reactor pool (i.e., normal operating external pressure loading, dynamic fluid-structure interaction under seismic loading)
- E. computer programs used in the design and analysis (other than those addressed in SRP Section 3.9.1)
- F. ultimate capacity of steel containment

Because of the unique design of the NuScale RPV/steel containment module, the steel containment vessel and the RPV should be analyzed as coupled structures for seismic loading, and potentially also for other loads (e.g., hydrodynamic loads, thermal loads) applied to either the containment vessel or to the RPV. Interaction forces at the RPV-to-containment vessel supports must be included in the structural assessment of the containment vessel.

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 BPV Code, Section III, Division 1, Subsection NB, and conformance with RG 1.57 are reviewed. Regarding references to Subsection NE in RG 1.57 for design limits, the corresponding criteria in Subsection NB shall be used. 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 major materials used in the construction of the containment vessel include the following:
 - i. steel plates used as shell components, if applicable
 - ii. structural steel shapes used for stiffeners, beam seats, and crane brackets, if applicable
 - iii. steel forgings used as shell components
 - iv. stainless steel cladding on wetted surfaces

Although many materials are acceptable for steel containments, in accordance with ASME BPV Code, Section III, Division 1, Subsection NB, the basic materials to be used for the containment vessel should be selected considering radiation effects and fracture toughness. The radiation effects and fracture toughness are reviewed in SRP Section 6.2.7.

- B. The quality control program proposed for the fabrication of the containment with emphasis on the extent of compliance with Articles NB-2000, NB-4000, and NB-5000 of ASME BPV Code, Section III, Division 1, Subsection NB, as incorporated by reference in 10 CFR 50.55a, 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, such as shop fabrication, transport to site, and hoisting into place, 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 pre-operational 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 NB-6000 of ASME BPV Code, Section III, Division 1, Subsection NB, as incorporated by reference in 10 CFR 50.55a.

Components relied upon for containment structural integrity that are constructed in accordance with ASME BPV Code, Section III, Division 1, Subsection NB, should be subject to periodic examination in accordance with ASME BPV Code, Section XI, Subsection IWB, as incorporated by reference 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 (ISI) 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 ISI of the steel containment.

- 8. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For 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 SRP Section 14.3.2, "Structural and Systems Engineering— Inspections, Tests, Analyses, and Acceptance Criteria," and 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 and SRP Section 14.3.2.

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 (QA) 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 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. Overall evaluation of the adequacy of the structural integrity of components constructed in accordance with ASME BPV Code Section III, Subsection NB, is performed in accordance with SRP Section 3.9.3. This review includes the fatigue analysis and the combination of design, service, and postulated event loadings. To the extent that the design of the containment uses this information, reference to Section 3.9.3 of the application or safety evaluation report (SER) may be appropriate.
4. 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.
5. The organization responsible for QA performs the reviews of design, construction, and operation phase QA 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 the audit identifies quality related issues, then the technical staff should contact the organization responsible for QA to determine if an inspection should be conducted.
6. 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, "Quality Standards and Records"
2. GDC 2, "Design Bases for Protection Against Natural Phenomena"
3. GDC 4, "Environmental and Dynamic Effects Design Bases"
4. GDC 16, "Containment Design"
5. GDC 50, "Containment Design Basis"
6. GDC 53, "Provisions for Containment Testing and Inspection"
7. 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.
8. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the DC has been constructed and will be operated in conformity with the DC, the provisions of the Atomic Energy Act, 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 Atomic Energy Act, and the NRC's regulations

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are set forth below. The DSRS is not a substitute for the NRC's regulations, and compliance with it is not required. As an alternative, and as described in more detail below, an applicant may identify the differences between a DSRS section and the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and discuss how the proposed alternative provides an acceptable method of complying with the NRC regulations that underlie the DSRS acceptance criteria.

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 should 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. 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 the NuScale steel containment. The following codes and guides are acceptable:

<u>Code/Guide</u>	<u>Title</u>
ASME BPV Code	Section III, "Rules for Construction of Nuclear Facility Components," Division 1, Subsection NE, "Class MC Components" ^a
ASME BPV Code	Section III, "Rules for Construction of Nuclear Facility Components," Division 1, Subsection NB, "Class 1 Components"
ASME BPV Code	Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Division 1, Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants"
ASME BPV Code	Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Division 1, Subsection IWB, "Requirements for Class 1 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

^a NE-7000 should be used for protection against overpressure in lieu of the rules of NB-7000.

3. Loads and Loading Combinations. Currently, ASME BPV Code, Section III, Division 1, Subsection NB, 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; external hydrostatic pressure, and buoyancy effects due to submergence in the reactor pool
- E Loads generated by the OBE, including sloshing effects, if applicable, and fluid-structure interaction effects due to submergence in the reactor pool
- E' Loads generated by the SSE, including sloshing effects, if applicable, and fluid-structure interaction effects due to submergence in the reactor pool
- P_a Pressure loads 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 atmospheric dump valve (ADV) and/or safety/relief valve (SRV) discharge, including hydrodynamic loads, if applicable

T_s All thermal loads that are generated by the actuation of ADV and/or SRV discharge, including hydrodynamic thermal loads, if applicable

R_s All pipe reaction loads that are generated by the actuation of ADV and/or SRV 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 loads from an accident that releases hydrogen generated from 100-percent fuel clad metal-water reaction

P_{g2} Pressure loads resulting from uncontrolled 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 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 ADVs and/or SRVs

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

(c) Design-basis LOCA

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

- (d) ADV and/or SRV 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 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) ADV and/or 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 50.44)

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

Note: In this load combination, $P_{g1} + P_{g2}$ should not be less than 310 kilopascals (kPa) or 45 pounds per square inch, 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 and/or 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

The NuScale containment vessel will be constructed in an enclosed fabrication facility protected from environmental effects and shipped to the plant site. Temporary loads encountered during shop fabrication, transportation, and onsite erection need to be considered.

D. External Environmental Loads

A concrete building protects the NuScale steel containment vessels from the external environmental loads.

In addition, according to 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 (SRM) 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 should 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 NB-3000 of ASME BPV Code, Section III, Division 1, Subsection NB, covers design and analysis procedures for Class I components. The procedures given in the ASME BPV 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 of steel containments. 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 and/or SRV actuation. For NuScale, nonaxisymmetric fluid-structure interaction loads due to horizontal seismic excitation must be evaluated. 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 NB-3133 of ASME BPV Code, Section III, Division 1, Subsection NB, 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 BPV Code Case N-284, Revision 2, "Metal Containment Shell Buckling Design Methods, Class MC Section III, Division 1," which is accepted for use by the NRC staff in RG 1.184, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III."

Buckling of shells under internal pressure (e.g., torispherical heads) may also be considered in accordance with the criteria described in ASME BPV Code Case N-284, Revision 2.

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, "ASME Code Cases Not Approved for Use," provide additional guidance for Code Case acceptability, which should be considered in the design of steel containments. Revision 1 of Code Case N-284 included errata, misprints, recommendations, and errors and is unacceptable to the NRC, as discussed in RG 1.193. Any Code Case not currently approved by NRC requires review on a case-by-case basis.

C. Effects of Long-Term Exposure to High Radiation Levels and Potential Loss of Fracture Toughness

Because of the close proximity of the NuScale containment shell wall to the RPV outer surface, there is the potential for degradation of the containment material properties over a limited region, due to long-term radiation exposure. Review of the methods used to evaluate this possible condition and its potential effects on the specified Code allowables will be performed under SRP 6.2.7.

D. Consideration of Cyclic Fatigue

Because of the close proximity of the NuScale containment shell wall to the RPV outer surface, and consideration that the NuScale containment is submerged in a pool of water, the potential for significant cyclic thermal loading needs to be evaluated. The staff will perform a detailed review of the methods used to evaluate this possible condition and the resulting cyclic fatigue calculations. The staff will also review the consideration of reduced fatigue strength due to long-term radiation exposure. The OBE contribution to cyclic loading, including fluid-structure interaction effects, is also reviewed.

E. 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 or SRP Section 3.9.1, as appropriate.

F. 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 the guidance in RG 1.57 or applicable guidance in RG 1.216 or both. The potential effect of long-term radiation exposure should be addressed.

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 DSRS section are acceptable if found to be within the limits defined by ASME BPV Code, Section III, Division 1, Subsection NB, as incorporated by reference in 10 CFR 50.55a. The potential effect of long-term radiation exposure on the specified Code limits should be addressed.

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.

6. Materials, Quality Control, and Special Construction Techniques

- A. The basic materials to be used for containment vessel must be selected considering radiation effects and fracture toughness. This is reviewed under SRP Section 6.2.7.
- B. Quality control programs are acceptable if they are in accordance with Articles NB-2000, NB-4000, and NB-5000 of ASME BPV Code, Section III, Division 1, Subsection NB.
- C. The acceptability of special construction techniques, if any, is evaluated on a case-by-case basis.

7. Testing and Inservice Surveillance Requirements

- A. Procedures for the pre-operational structural proof test are acceptable if the procedures are in accordance with Article NB-6000 of ASME BPV Code, Section III, Division 1, Subsection NB.
- B. Consistent with construction in accordance with ASME BPV Code, Section III, Division 1, Subsection NB, procedures for periodic examination of the containment vessel and its components relied upon for containment structural integrity are acceptable if the procedures are in accordance with ASME BPV Code, Section XI, Subsection IWB, as incorporated by reference in 10 CFR 50.55a.
- 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 ISI 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 QA program be established and implemented; and that 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 ISIs, 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 BPV Code, Section III, Division 1, Subsection NE, provides acceptable design guidance and acceptance criteria. Provisions of NCA-2134(c) are used for construction of containment vessel using the rules of Subsection NB, provided the rules of NE-7000 are applied in lieu of the rules of NB-7000 for protection against overpressure.

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, in part, that SSCs important to safety be designed to withstand the effects of expected natural phenomena such as earthquakes, tornados, hurricanes, floods, tsunamis, 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 are used in developing the design requirements associated with a particular site for a CP, operating license (OL) or COL application, or for site parameter envelopes in the case of DCs. In doing so, the applicant will ensure that the containment 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 BPV Code, Section III, Division 1, Subsection NB, 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 or 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 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 including penetrations and penetration seals should be designed, constructed, and tested to provide a leaktight barrier and 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 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 containment 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, will be able to withstand the loads resulting from pressure and temperature conditions caused by any LOCA and will perform its designed safety function.

6. Compliance with GDC 53 requires that the design of steel containment shall have the provisions for containment testing and inspection. As applicable to the containment design, the containment should include features designed to permit:
 - A. appropriate periodic inspection of all important areas, such as penetrations
 - B. an appropriate surveillance program
 - C. periodic testing at containment design pressure of the leaktightness of penetrations that have resilient seals and expansion bellows

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.

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, RG 1.216, and RG 1.57 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.

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 nuclear power reactors meet the requirements of the ASME BPV Code, and (3) RGs 1.84 and 1.147 provide guidance related to NRC-approved ASME BPV 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 BPV 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 ISI, 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. Selected Programs and Guidance—In accordance with the guidance in NUREG-0800, "Introduction – Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light-Water Small Modular Reactor Edition" (NUREG-0800, Intro Part 2), as applied to this DSRS Section, the staff will review the information proposed by the applicant to evaluate whether it meets the acceptance criteria described in Subsection II of this DSRS. As noted in NUREG-0800, Intro Part 2, the NRC requirements that must be met by an SSC do not change under the small modular reactor (SMR) framework. Using the graded approach described in NUREG-0800, Intro Part 2, the NRC staff may determine that, for certain SSCs, the applicant's basis for compliance with other selected NRC requirements may help demonstrate satisfaction of the applicable acceptance criteria for that SSC in lieu of detailed independent analyses. The design-basis capabilities of specific SSCs would be verified, where applicable, as part of completing the applicable ITAAC. The use of the selected programs to augment or replace traditional review procedures is shown in Figure 1 of NUREG-0800, Intro Part 2. Examples of such programs that may be relevant to the graded approach for these SSCs include:
 - 10 CFR Part 50, Appendix A, GDC, Overall Requirements, Criteria 1–5
 - 10 CFR Part 50, Appendix B, Quality Assurance (QA) Program
 - 10 CFR 50.49, Environmental Qualification of Electrical Equipment (EQ) Program
 - 10 CFR 50.55a, Code Design, Inservice Inspection, and Inservice Testing (ISI/IST) Programs
 - 10 CFR 50.65, Maintenance Rule requirements
 - Reliability Assurance Program (RAP)
 - 10 CFR 50.36, "Technical Specifications"

- Availability Controls for SSCs Subject to Regulatory Treatment of Nonsafety Systems (RTNSS)
- Initial Test Program (ITP)
- Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

This list of examples is not intended to be all inclusive. It is the responsibility of the technical reviewers to determine whether the information in the application, including the degree to which the applicant seeks to rely on such selected programs and guidance, demonstrates that all acceptance criteria have been met to support the safety finding for a particular SSC.

2. In accordance with 10 CFR 52.47(a)(8), (21), and (22), and 10 CFR 52.79(a)(17), (20), and (37), for DC or COL applications submitted under 10 CFR 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, "Resolution of Generic Safety Issues," 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), for a DC application, and except paragraphs (f)(1)(xii), (f)(2)(ix), (f)(2)(xxv), and (f)(3)(v), for a COL application. These cross-cutting review areas should be addressed by the reviewer for each technical subsection and relevant conclusions documented in the corresponding safety evaluation report (SER) section.
3. 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 construction permit (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 applications submitted in accordance with 10 CFR Part 52).

4. 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.
5. 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.

6. Design and Analysis Procedures. The reviewer will verify that the applicant is committed to the design and analysis procedures delineated in Article NB-3000 of ASME BPV Code, Section III, Division 1, Subsection NB. 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.
7. 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.
8. 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 BPV 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.
9. 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.6 of this DSRS section. Any proposed deviations will be considered on a case-by-case basis. The staff will review ISI 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 ISI on a case-by-case basis.

10. 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 Sections 14.3 and 14.3.2 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 SER. 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 GDCs 1, 2, 4, 16, 50, and 53. 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 BPV 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.

7. The applicant has met the requirements of GDC 53 by ensuring that the steel containment has appropriate provisions for containment testing and inspection.

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 BPV Code, Section III, Division 1, Subsection NB, 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 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 regulations in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), and 10 CFR 52.79(a)(41) establish requirements for applications for ESPs, DCs, and COLs, respectively. These regulations require the application to include an evaluation of the site (ESP), standard plant design (DC), or facility (COL) against the SRP revision in effect 6 months before the docket date of the application. While the SRP provides generic guidance, the staff developed the SRP guidance based on the staff's experience in reviewing applications for construction permits and operating licenses for large light-water nuclear power reactors. The proposed SMR designs, however, differ significantly from large light-water nuclear power plant designs.

In view of the differences between the designs of SMRs and the designs of large light-water power reactors, the Commission issued Staff Requirements Memorandum

(SRM)-COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights To Enhance Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010. In the SRM, the Commission directed the staff to develop risk-informed licensing review plans for each of the SMR design reviews, including plans for the associated preapplication activities. Accordingly, the staff has developed the content of the DSRS as an alternative method for evaluating a NuScale-specific application submitted pursuant to 10 CFR Part 52, and the staff has determined that each application may address the DSRS in lieu of addressing the SRP, with specified exceptions. These exceptions include particular review areas in which the DSRS directs reviewers to consult the SRP and others in which the SRP is used for the review. If an applicant chooses to address the DSRS, the application should identify and describe all differences between the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and the guidance of the applicable DSRS section (or SRP section, as specified in the DSRS), and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria.

The staff has accepted the content of the DSRS as an alternative method for evaluating whether an application complies with NRC regulations for NuScale SMR applications, provided that the application does not deviate significantly from the design and siting assumptions made by the NRC staff while preparing the DSRS. If the design or siting assumptions in a NuScale application deviate significantly from the design and siting assumptions the staff used in preparing the DSRS, the staff will use the more general guidance in the SRP, as specified in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), or 10 CFR 52.79(a)(41), depending on the type of application. Alternatively, the staff may supplement the DSRS section by adding appropriate criteria to address new design or siting assumptions.

VI. REFERENCES

1. *U.S. Code of Federal Regulations*, "Standards for Protection against Radiation," Part 20, Chapter I, Title 10, "Energy."
2. *U.S. Code of Federal Regulations*, "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter I, Title 10, "Energy."
3. *U.S. Code of Federal Regulations*, "Early Site Permits, Standard Design Certifications, and Combined Licenses for Nuclear Power Plants," Subparts A, B, and C, Part 52, Chapter I, Title 10, "Energy."
4. GDC 61, "Fuel Storage and Handling and Radioactivity Control."
5. GDC 19, "Control Room."
6. GDC 4, "Environmental and Dynamic Effects Design Bases."
7. U.S. Nuclear Regulatory Commission, "Control of Combustible Gas Concentrations in Containment," RG 1.7, Agencywide Documents Access and Management System (ADAMS) Accession No. ML070290080.

8. U.S. Nuclear Regulatory Commission, "Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," RG 1.112, ADAMS Accession No. ML070320241.
9. U.S. Nuclear Regulatory Commission, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," RG 1.183, ADAMS Accession No. ML003716792.
10. American National Standards Institute/American Nuclear Society, "Radioactive Source Term for Normal Operation of Light Water Reactors," ANSI/ANS 18.1-1999, La Grange Park, IL.
11. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980, ADAMS Accession No. ML051400209.
12. *U.S. Code of Federal Regulations*, "Environmental Radiation Protection Standards for Nuclear Power Operations," Part 190, Chapter I, Title 40, "Protection of Environment."
13. U.S. Nuclear Regulatory Commission, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," RG 1.89, ADAMS Accession No. ML003740271.
14. U.S. Nuclear Regulatory Commission, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," RG 1.143.
15. U.S. Nuclear Regulatory Commission, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," RG 1.26.
16. U.S. Nuclear Regulatory Commission, "Seismic Design Classification," RG 1.29.
17. U.S. Nuclear Regulatory Commission, "Tornado Design Classification," RG 1.117, ADAMS Accession No. ML003739346.
18. U.S. Nuclear Regulatory Commission, "Combined License Applications for Nuclear Power Plants (LWR Edition)," RG 1.206.
19. Electric Power Research Institute, "Pressurized Water Reactor Primary Water Chemistry Guidelines."
20. Electric Power Research Institute, "Pressurized Water Reactor Primary Water Zinc Application Guidelines."
21. Electric Power Research Institute, "Advanced Light Water Reactor Utility Requirements Document, Volume III, ALWR Passive Plant."
22. U.S. Nuclear Regulatory Commission, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document, Passive

Plant Designs,” NUREG-1242, Volume 3, Parts 1 and 2 (ADAMS Accession Nos. ML070600372 and ML070600373).

23. Electric Power Research Institute, “Cobalt Reduction Guidelines.”
24. U.S. Nuclear Regulatory Commission, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable,” RG 8.8, ADAMS Accession No. ML003739549.
25. Staff Requirements Memorandum COMGBJ-10-0004/COMGEA-10-0001, “Use of Risk Insights To Enhance Safety Focus of Small Modular Reactor Reviews,” dated August 31, 2010. ADAMS Accession No. ML102510405.

APPENDIX A: STRUCTURAL DESIGN AUDIT OF STEEL CONTAINMENT

1. Introduction

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, Section 47, “Contents of Applications; Technical Information,” requires:

The application must contain a level of design information sufficient to enable the Commission to judge the applicant’s proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. The information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC [U.S. Nuclear Regulatory Commission], and procurement specifications and construction and installation specifications by an applicant. The Commission will require, before design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if the information is necessary for the Commission to make its safety determination.

Paragraph (11) in 10 CFR 52.79(a) requires that a combined license application provide a description of the programs and their implementation necessary to ensure that the systems and components meet the requirements of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code) and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) in accordance with 10 CFR 50.55a.

This appendix provides guidelines for the NRC staff’s implementation of structural design audits.

2. Objectives

The audit has the following objectives:

- A. to verify that detailed design documents implement the structural design criteria described in the application
- B. to verify that the key structural design calculations implement an acceptable methodology, consistent with the application and the guidance in this design-specific review standard (DSRS) section and related Standard Review Plan (SRP) and DSRS sections
- C. to identify and assess the safety significance of those areas where the plant structures were designed and analyzed using methods other than those recommended by this DSRS section

3. Audit Procedures

The general procedures for conducting the audit should follow established NRC internal guidance.

4. Conduct of the Audit

A. Overview of the Plant Design

The applicant should present an overview of each of the key structures including a brief description, assumptions, modeling techniques, and features of the design, as well as any deviations from those committed to in the application.

B. Audit of Design Report

The NRC staff should audit the design report for the initial design report prepared by the applicant to understand detailed design and construction information beyond the information contained in the application. This audit also confirms that the containment vessel meets the design criteria and that the analytical methods and criteria satisfy the guidance provided in relevant SRP and DSRS sections and the rules of ASME BPV Code Section III. The design report should include information from the actual design computations and the design results, in accordance with ASME BPV Code Section III, NCA-3260. The design report should also provide criteria for reconciliation between design and as-built conditions.

The following outline presents the information typically expected in a design report.

I. CONTAINMENT DESCRIPTION AND GEOMETRY

1. Structural Geometry and Dimensions
2. Key Structural Elements and Description
3. Floor Layout and Elevations
4. Conditions of Vicinity and Supports
5. Special Structural Features

II. STEEL STRUCTURAL MATERIAL REQUIREMENTS

1. Grade
2. Ultimate Tensile Strength
3. Yield Stress

III. STRUCTURAL LOADS

1. Live and Dead Load Floor Plans
2. Determination of Transient and Dynamic Loads
3. Manufacturer's Data on Equipment Loads
4. Environmental Loads

5. Torsional Effects

IV. STRUCTURAL ANALYSIS AND DESIGN

1. Design Computations of Critical Elements
2. Stability Calculations
3. Engineering Drawings including Details of Connections and Joints
4. Discussion of Unique Features and Problem Resolution

V. SUMMARY OF RESULTS

1. Individual Load Contributions
2. Capacities versus Capacities Required for Different Failure Modes (Bending, Shear, Axial Load)
3. Summary of the Maximum Total Stress, Deformation and Cumulative Usage Factor Values
4. Margins of Safety Provided

VI. CONCLUSIONS

C. Audit of Design Calculations

The auditing personnel should audit the design calculations for the structures identified during the review of the applicant's design report. The participants in the audit should discuss and resolve any questions such as those regarding the structural modeling, analysis, proportioning of the members, and computer runs. If resolution of the questions requires additional engineering data from and further analysis by the applicant, the specific followup action items should be noted in the audit summary and documented in a request for additional information, as appropriate.

5. Input to the Safety Evaluation Report

The audit is an integral part of the review process and supports the staff's conclusions on the acceptability of the application. The audit should be described in the safety evaluation report (SER) with reference to publicly available audit reports. The basis for the staff's decision, however, must be in docketed correspondence, such as the design certification (DC) application, from the applicant.