U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

3.8.1 CONCRETE CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Organization responsible for structural analysis reviews

Secondary - None

I. AREAS OF REVIEW

This section describes the review of areas relating to concrete containments or to concrete portions of steel/concrete containments, as applicable.

The specific areas of review are as follows:

1. Description of the Containment. The staff reviews 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 on to perform the containment function. In particular, the review identifies the type of concrete containment and examines its structural and functional characteristics. The following are among the various types of concrete containments reviewed:

   A. Reinforced and prestressed concrete 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).

USNRC STANDARD REVIEW PLAN

This Standard Review Plan, NUREG-0800, has been prepared to establish criteria that the U.S. Nuclear Regulatory Commission staff responsible for the review of applications to construct and operate nuclear power plants intends to use in evaluating whether an applicant/licensee meets the NRC's regulations. The Standard Review Plan 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 an acceptable method of complying with the NRC regulations.

The standard review plan sections are numbered in accordance with corresponding sections in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)." Not all sections of Regulatory Guide 1.70 have a corresponding review plan section. The SRP sections applicable to a combined license application for a new light-water reactor (LWR) are based on Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

These documents are made available to the public as part of the NRC's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Individual sections of NUREG-0800 will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience. Comments may be submitted electronically by email to NRR_SRP@nrc.gov.

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B. Reinforced concrete pressurized-water reactor (PWR) containments using the pressure-suppression concept with ice condenser elements.

C. Reinforced concrete PWR containments designed to function under subatmospheric conditions.

D. Reinforced and prestressed concrete PWR dry containments designed to function at atmospheric conditions.

E. Reinforced and prestressed concrete PWR or BWR containments using special features or modifications of the above-listed types. As an example, for any application submitted for design certification (DC) that incorporates unique features such as modular construction, passive systems with pools, single foundation for the nuclear island structures, and concrete walls and floors integrally connected to the concrete containment, the applicant needs to provide information such that an adequate review and evaluation can be done.

Various geometries have been used for these containments. The geometry most commonly encountered is an upright cylinder topped with a dome and supported on a flat concrete base mat. Although applicable to any geometry, the specific provisions of this SRP section are best suited to the cylindrical-type containment topped by a dome. Reviews of containments with other types of geometry will make the necessary deviations from this SRP section on a case-by-case basis.

The geometry of the containment is reviewed, including sketches showing plan views at various elevations and sections in at least two orthogonal directions. The reviewer examines the arrangement of the containment and the relationship and interaction of the shell with its surrounding structures and with its interior compartment walls and floors to determine the e ffect these structures could have on the design boundary conditions and expected structural behavior of the containment when subjected to design loads.

The review encompasses general information related to the containment shell, including the following:

A. The base foundation slab, including the main reinforcement; the floor liner plate and its anchorage and stiffening system; and the methods by which the interior structures are anchored through the liner plate and into the slab, if applicable

B. The cylindrical wall, including the main reinforcement and prestressing tendons, if any; the wall liner plate and its anchorage and stiffening system; the major penetrations and the reinforcement surrounding them, including the personnel and equipment hatches and major pipe penetrations; major structural attachments to the wall that penetrate the liner plate such as beam seats, pipe restraints, and crane brackets; and external supports, if any, attached to the wall to support external structures such as enclosure buildings

C. The dome and the ring girder, if any, including the main reinforcement and prestressing tendons; the liner plate and its anchorage and stiffening systems; and any major attachments to the liner plate made from the inside

D. Other special features such as the containment refueling seals and drain, seismic gaps between the containment and adjacent structural elements inside and outside containment, rock anchors, subfoundation drainage system, use of waterproofing membrane, and containment settlement monitoring systems.

SRP Section 3.8.2 covers steel components of concrete containments that resist pressure and are not backed by structural concrete.

2. Applicable Codes, Standards, and Specifications. The review evaluates information pertaining to design codes, standards, specifications, regulations, and regulatory guides and other industry standards that are applied in the design fabrication, construction, testing, and inservice surveillance of the containment. The specific editions, dates, or addenda identified for each document are also reviewed.

3. Loads and Loading Combinations. The staff reviews information pertaining to the applicable design loads and various combinations thereof, with emphasis on the extent of compliance with Article CC-3000 of Section III, Division 2, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereafter referred to as the ASME Code) and Regulatory Guide (RG) 1.136 (see Subsection II.3 of this SRP section). The loads normally applicable to concrete containments include the following:

A. Those loads encountered during construction of the containment, including dead loads, live loads, prestress loads, temperature, wind, earth pressure, snow, rain, and ice, and construction loads that may be applicable such as material loads, personnel and equipment loads, horizontal construction loads, erection and fitting forces, equipment reactions, and form pressure.

B. Those loads encountered during preoperational testing.

C. Those loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads resulting from operating temperatures, hydrostatic loads, and hydrodynamic loads resulting from safety/relief valve (SRV) actuation such as those present in pressure-suppression containments using water. Appendix A to this SRP section further describes loads associated with SRVs.

D. Those loads to be sustained during severe environmental conditions, including those induced by the design wind and the operating-basis earthquake (OBE) specified for the plant site. Subsection II.3.C of this SRP section defines the condition for which the OBE load is required for design of concrete containment.

E. Those loads to be sustained during extreme environmental conditions, including those induced by the design-basis tornado and the safe-shutdown earthquake (SSE) specified for the plant site.

F. 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. Other accidents involving various high-energy pipe ruptures are also considered. 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. For BWR containments, the review should consider the LOCA- and SRV-related hydrodynamic loads in suppression pools manifested as jet loads and/or pressure loads. Appendix A to this SRP section describes loads associated with LOCAs.

G. Those loads to be sustained, if applicable, after abnormal plant conditions, including flooding of the containment subsequent to a LOCA to maintain core cooling and/or fuel recovery.

H. For those plants to which 10 CFR 50.34(f)(3)(v) applies, pressure and dead loads during an accident that releases hydrogen generated from 100-percent metal-water reaction of the fuel cladding and accompanied by either hydrogen burning or added pressure from postaccident inerting, and the loadings produced by the inadvertent full actuation of a postaccident inerting hydrogen control system, excluding seismic or design-basis accident loadings.

I. 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 applicable plants that must consider this loading and the percent fuel cladding to consider.

The various combinations of the above loads that are normally postulated and reviewed include construction loads; testing loads; normal operating loads; normal operating loads with severe environmental loads; normal operating loads with extreme environmental loads; normal operating loads with abnormal loads; normal operating loads with severe environmental and abnormal loads; normal operating loads with extreme environmental and abnormal loads; and post-LOCA flooding loads with severe environmental loads, if applicable.

The loads and load combinations described above are generally applicable to all containments. However, other site-related design loads may also apply. Any loads, which are not normally combined with abnormal loads are reviewed on a case-by-case basis. They include those loads induced by floods, potential aircraft crashes (nonterrorism-related incidents), explosive hazards in proximity to the site, and projectiles and missiles generated from activities of nearby military installations.

4. Design and Analysis Procedures. The staff reviews the design and analysis procedures used for the containment with emphasis on the extent of compliance with Article CC-3000 of the ASME Code and RG 1.136 (see SRP subsection II.3 of this section), particularly with respect to the following:

A. Assumptions on boundary conditions

B. Treatment of axisymmetric and nonaxisymmetric loads

C. Treatment of transient and localized loads

D. Treatment of the effects of creep, shrinkage, and cracking of the concrete

E. A description of the computer programs used in the design and analyses

F. The treatment of the effects of seismically induced tangential (membrane) shears

G. The evaluation of the effects of variations in specified physical properties of materials on analytical results

H. The treatment of the large, thickened penetration regions

I. The treatment of the steel liner plate and its anchors (SRP Section 3.8.2 covers steel penetration closures)

J. Ultimate capacity of the concrete containment

K. Structural audit

L. Design report submitted for review

5. Structural Acceptance Criteria. The staff reviews the design limits imposed on the various parameters that quantify the structural behavior of the containment, with emphasis on the extent of compliance with Article CC-3000 of the ASME Code and RG 1.136 (see Subsection II.3 of this SRP section), specifically with respect to allowable stresses, strains, deformations, and other parameters that quantify the margins of safety. For each load combination specified, the reviewer compares 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. Compressive stresses in concrete, including membrane, membrane plus bending, and localized stresses

   B. Shear stresses in concrete, particularly those tangential (membrane) stresses induced by lateral loads

   C. Tensile stresses in reinforcement

   D. Tensile stresses in prestressing tendons

   E. Tensile or compressive strain limits in the liner plate, including membrane and membrane plus bending

   F. Force/displacement limits in the liner plate anchors, including those induced by strains in the adjacent concrete

6. Materials, Quality Control, and Special Construction Techniques. The staff reviews the information provided on materials that are used in construction of the containment with emphasis on the extent of compliance with Article CC-2000 of the ASME Code and RG 1.136 (see Subsection II.3 of this SRP section). The following are among the major materials of construction reviewed:

   A. The concrete ingredients

B. The reinforcing bars and splices

C. The prestressing system

D. The liner plate

E. The liner plate anchors and associated hardware

F. The structural steel used for embedments such as beam seats and crane brackets

G. The corrosion-retarding compounds used for the prestressing tendons

The staff reviews the quality control program that is proposed for the fabrication and construction of the containment with emphasis on the extent of compliance with Articles CC-4000 and CC-5000 of the ASME Code and RG 1.136 (see Subsection II.3 of this SRP section). This includes examination of the materials, as well as tests to determine the physical properties of concrete, reinforcing steel, mechanical splices, the liner plate and its anchors, and the prestressing system, if any; placement of concrete; and erection tolerances of the liner plate, reinforcement, and prestressing systems.

The review covers any proposed special, new, or unique construction techniques, such as slip forming, on a case-by-case basis to determine their effects on the structural integrity of the completed containment.

7. Testing and Inservice Surveillance Requirements. For concrete containments, it is important to accommodate inservice inspection of critical areas. The review includes 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, or providing remote visual monitoring of high-radiation areas) to accommodate inservice inspection of concrete containments.

   The review covers the preoperational structural testing program for the completed containment and for individual components, such as personnel and equipment locks and hatches, and includes the objectives of the test program and acceptance criteria, with emphasis on the extent of compliance with Article CC-3000 of the ASME Code. The review also includes inservice surveillance programs, such as the periodic surveillance and inspection of the containment and prestressing tendons, if any, and examines the applicable technical specifications, at the operating license stage. The staff reviews the inservice surveillance programs, for components relied upon for containment structural integrity, with emphasis on the extent of compliance with ASME Code, Section XI, Subsection IWL. The review of programs for the examination of inaccessible areas, monitoring of ground water chemistry, and monitoring of settlements and differential displacements proceeds on a case-by-case basis. The staff also reviews special testing and inservice surveillance requirements proposed for new or previously untried design approaches on a case-by-case basis.

8. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this 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. Determination of structures which are subject to quality assurance programs in accordance with the requirements of 10 CFR Part 50, Appendix B 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. Determination of pressure loads from high-energy lines located in safety-related structures 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. 10 CFR 50, Appendix A, General Design Criterion (GDC) 4, allows the exclusion of dynamic effects of pipe ruptures, if analyses (i.e., leak-before-break analyses) demonstrate that the probability of rupture is extremely low. For containment design, the applicability of these analyses is limited to localized effects only. 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 guidance provided in SRP Section 3.6.3.

4. Determination of loads generated because of pressure under accident conditions is performed in accordance with guidance provided in SRP Section 6.2.1. The loads thus generated are accepted for inclusion in the load combinations in this SRP section.

5. ~~The review for quality assurance is coordinated and performed in accordance with SRP Chapter 17.~~ The organization responsible for quality assurance performs reviews of design, construction, and operations 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, the technical staff should contact the organization responsible for quality assurance to determine if an inspection should be conducted.

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6. Determination that the containment performance meets severe accident criteria is performed in accordance with guidance provided in Chapter 19.

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 GDC 1, as they relate to concrete containment being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

2. GDC 2, as it relates to the design of the concrete containment being able to withstand the most severe natural phenomena such as winds, tornadoes, floods, and earthquakes and the appropriate combination of all loads.

3. GDC 4, as it relates to the concrete containment being 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.

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

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

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

7. 10 CFR 50.34(f), as it relates to demonstrating containment integrity of applicable plants for loads associated with an accidental release of hydrogen generated from metal-water reaction of the fuel cladding, accompanied by hydrogen burning or added pressure from postaccident inerting[1]

8. 10 CFR 50.44, as it relates to demonstrating the structural integrity of BWRs with Mark III type containments, all PWRs with ice condenser containments, and all containments used in future water-cooled reactors for loads associated with combustible gas generation

9. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;

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

10. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.

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 considered acceptable if it meets the criteria set forth in Section 3.8.1.1 of RG 1.206. If the concrete containment has new or unique features that are not specifically covered in RG 1.206, the reviewer determines whether the information necessary to accomplish a meaningful review of the structural aspects of these new or unique features is presented.

   RG 1.206 provides the basis for evaluating the description of Seismic Category I 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. The design, materials, fabrication, erection, inspection, testing, and inservice surveillance of concrete containments are covered by codes, standards, specifications, and guides that are applicable either in their entirety or in part. The following codes and guides are acceptable:

Codes	Title
ASME Code	Section III, Division 2, Subsection CC, "Code for Concrete Reactor Vessels and Containments"
ASME Code	Section XI, Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Plants"
ASME Code	Section XI, Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Concrete Components of Light-Water Cooled Power Plants"

RG	Title
1.7	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident

1.35	Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments
1.35.1	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments
1.90	Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons
1.91	Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants
1.107	Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures
1.115	Protection Against Low Trajectory Turbine Missiles
1.136	Materials, Construction, and Testing of Concrete Containments

3. Loads and Loading Combinations. The specified loads and load combinations are acceptable if found to be in accordance with Article CC-3000 of the ASME Code with the exceptions listed below applied to the requirements specified in Table CC-3230-1. RG 1.136, "Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments," provides additional guidance for design requirements, including load and load combinations, which should be considered in the design of concrete containments.

   A. The maximum values of P_a, T_a, R_a, R_{rr}, R_{rj}, and R_{rm} should be applied simultaneously, where appropriate, unless a time-history analysis is performed to justify doing otherwise.

   B. Hydrodynamic loads resulting from LOCA and/or SRV actuation should be combined as indicated in the appendix to this SRP section. Fluid structure interaction associated with these hydrodynamic loads and those from earthquakes should be considered.

   C. As noted in Appendix S to 10 CFR Part 50, 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, an explicit response or design analysis is not required. If the OBE is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the containment remains functional and is within applicable stress, strain, and deformation limits. SRP Section 3.7 provides further guidance on the use of OBE.

      When the OBE is defined as less than one-third of the SSE (and therefore the OBE does not need to be considered in design), certain structural elements of the containment (e.g., penetrations or bellows) still need to be evaluated for fatigue resulting from the OBE-induced stress cycles. In these instances, the guidance for determining the number of earthquake cycles for use in fatigue calculations should be the same as the guidance provided in the staff requirements memorandum (SRM) for SECY-93-087 for piping systems. The number of

earthquake cycles to consider is two SSE events with 10 maximum stress cycles per event. Alternatively, the number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE Standard 344-1987.

D. Where post-LOCA flooding is a design consideration for the plant, the load combination in the ASME Code containing LOCA flooding along with OBE should be considered. Where post-LOCA flooding is combined with the OBE set at one-third or less of the SSE for the plant, this load combination may be eliminated provided the load combination is shown to be less severe than one of the other load combinations.

E. For those plants to which 10 CFR 50.34(f)(3)(v) applies, the requirements regarding loads and loading combinations include the following:

Containment integrity should be maintained by meeting the requirements of Subarticle CC-3720 of the ASME Code (considering pressure and dead load alone) during an accident that releases hydrogen generated from 100-percent metal-water reaction of the fuel cladding and accompanied by either hydrogen burning or added pressure from postaccident inerting (assuming carbon dioxide is the inerting agent). At a minimum, the ASME Code requirements will be met for a combination of dead load and an internal pressure of 310 Kilo Pascals (KPa) or 45 pounds per square in gauge (psig)

The containment structure should be designed against the loadings produced by the inadvertent full actuation of a postaccident inerting hydrogen control system (assuming carbon dioxide), excluding seismic or design-basis accident loadings. Under these conditions, the loadings should not produce strains in the containment liner in excess of the limits established in Subarticle CC-3720 of the ASME Code.

The requirements of Subarticle CC-3720 of the ASME Code should be met when the containment structure is exposed to the following loading conditions:

i. For the factored load category:

$$D + P_{g1} + [P_{g2} \text{ or } P_{g3}]$$

ii. For the service load category, the strains in the containment liner should not exceed the limits set forth in Subarticle CC-3720 when exposed to pressure P_{g3}.

iii. As a minimum design condition for either condition i or ii above, the following load combination must be satisfied:

D + 310 kPa (45 psig) where

D = Dead load

P_{g1} = Pressure resulting from an accident that releases hydrogen generated from 100-percent metal-water reaction of the fuel cladding

P_{g2} = Pressure resulting from uncontrolled hydrogen burning

P_{g3} = Pressure resulting from postaccident inerting, assuming carbon dioxide is the inerting agent

F. 10 CFR 50.44 requires that an analysis be performed that demonstrates that the containment structural integrity is maintained under loads resulting from combustible gases generated from metal-water reaction of the fuel cladding. An analytical technique accepted by the NRC staff should demonstrate the containment structural integrity. This analysis should include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. RG 1.7 presents further guidance on the analytical technique, loads, loading combination, and acceptance criteria.

G. Other site-related or plant-related loads applicable to containment such as floods, explosive hazards in proximity to the site, potential aircraft crashes (nonterrorism-related incidents), and missiles generated from activities of nearby military installations or turbine failures need to be considered. The staff reviews the inclusion of these loads in the factored load combinations on a case-by-case basis.

H. The review considers those loads encountered during construction of the containment, which include dead loads, live loads, prestress loads, temperature, wind, earth pressure, snow, rain, and ice, and construction loads that may be applicable such as material loads, personnel and equipment loads, horizontal construction loads, erection and fitting forces, equipment reactions, and form pressure. Structural Engineering Institute (SEI)/American Society of Civil Engineers (ASCE) Standard 37 gives additional guidance on construction loads for use in the load combination for construction given in Table CC-3230-1 of the ASME Code. When SEI/ASCE Standard 37 and the ASME Code/SRP provide conflicting criteria, then the ASME Code/SRP should govern.

4. Design and Analysis Procedures. The procedures for design and analysis used for the concrete containment, including the steel liner, are acceptable if found in accordance with those stipulated in Article CC-3300 of the ASME Code and RG 1.136 (see Subsection II.3 of this SRP section). In particular, for the areas of review outlined in Subsection I.4 above, the following procedures are, in general, acceptable:

A. Assumptions on Boundary Conditions. The boundary conditions depend on the methods of analysis to be used and the portions of the containment shell to be separately analyzed. If the analysis is to involve the use of the finite element technique and is to include the foundation media, the boundary would be the demarcation lines separating the foundation mass taken into consideration in the analysis from the surrounding media. The boundaries of the foundation mass considered should be selected to provide comparable or conservative results to those corresponding to a further extension of the boundaries. This is reviewed on a case-by-case basis.

If the analysis considers only the containment shell and its foundation mat, then the bottom of the foundation slab is the boundary of the analytical model. The foundation media should be represented by appropriate soil springs.

If separate analyses of the containment shell and the base mat are to be used, it is considered acceptable if strain compatibility of the bottom portion of the shell with the base mat is maintained.

B. Axisymmetric and Nonaxisymmetric Loads. Even with the large penetrations and buttresses that may be used in the shell, the overall behavior of the shell has been shown to be axisymmetric under pressure. Therefore, it is acceptable to make such an assumption with respect to the containment geometry. However, for loads such as those induced by wind, tornadoes, earthquakes, and pipe rupture, the analysis should consider the nonaxisymmetric effect of these loads.

C. Transient and Localized Loads. During normal operation, a linear temperature gradient across the containment wall thickness may develop. After a LOCA, however, the sudden increase in temperature in the steel liner and the adjacent concrete may produce a nonlinear transient temperature gradient across the containment wall thickness. The analysis should consider the effects of such transient loads.

In a PWR ice condenser containment, nonaxisymmetric and transient pressure loads resulting from compartmentalization inside the containment will develop after a LOCA. For a BWR pressure-suppression containment, the analysis should consider nonaxisymmetric and transient pressure loads resulting from earthquakes, LOCA, and/or SRV actuation (including fluid-structure interaction).

For the effects of such localized and transient loads, the overall behavior of the containment structure should first be determined. A portion of the containment shell, within which the localized or transient load is located, should then be analyzed, using the results obtained from the analysis of the overall vessel behavior as boundary conditions.

D. Creep, Shrinkage, and Cracking of Concrete. Creep and shrinkage values for concrete should be established by tests performed on the concrete to be used in the containment structure or from data obtained on completed containments constructed of the same kind of concrete. In establishing these values, the analysis should consider the differences in the environment between the test samples and the actual concrete in the structure.

For some containments, cracking of concrete is expected to occur based on the structural integrity test performed in accordance with Article CC-6000 of the ASME Code. Also, based on load combinations that include the design pressure load with earthquake loads, additional concrete cracking would be expected to occur. Concrete cracking can cause redistribution of member forces because of the various loadings applied to the structure. Concrete cracking can also affect the stiffness of the containment and cause shifting of the natural frequency, thereby affecting the response/loads used to design the containment. Accordingly, the analysis used to calculate the dynamic response of the containment resulting from dynamic loads such as earthquake and hydrodynamic loads (if applicable) needs to consider the potential effects of concrete cracking, if significant. The approach used should include the effect of redistribution of the various loads caused by concrete cracking. With improvements in the development of computer programs for analysis of concrete structures, the evaluation of concrete cracking can be analyzed directly within the finite element model. Alternatively, additional analyses can treat the effect of concrete cracking by determining the response of the containment to variation in the stiffness characteristics of the containment shell (e.g., shear stiffness and tensile membrane stiffness reduction). As stated in CC-3320 of the ASME Code, the effects of reduction in shear stiffness and tensile membrane stiffness resulting from cracking of the concrete should be considered in methods for predicting the maximum strains and deformations of the containment. Thus, concrete cracking needs to be considered depending on the stress levels caused by the most severe seismic load combination. Provide technical justification, if cracking is not considered or is determined to be insignificant. Sections 3.1.3 and C 3.1.3 of ASCE 4-98 provide additional guidance for modeling the stiffness of concrete elements.

The staff reviews the methods used for considering creep, shrinkage, and concrete cracking, or the justification for not considering these effects, on a case-by-case basis.

E. Dynamic Soil Pressure. Consideration of dynamic lateral soil pressures on embedded walls of a concrete containment (if applicable) is acceptable if the lateral earth pressure loads are evaluated for two cases. These are (1) lateral earth pressure equal to the sum of the static earth pressure plus the dynamic earth pressure calculated in accordance with ASCE 4-98 Section 3.5.3.2 and (2) lateral earth pressure equal to the passive earth pressure. If the above methods are shown to be overly conservative for the cases considered, then any alternative methods proposed will be reviewed on a case-by-case basis.

F. Computer Programs. The computer programs used in the design and analysis should be described and validated by any of the following procedures or criteria:

i. The computer program is recognized in the public domain and has had sufficient history of use to justify its applicability and validity without further demonstration.

ii. The computer program's solutions to a series of test problems have been demonstrated to be substantially identical to those obtained by a similar and independently written and recognized program in the public domain. The test problems should be demonstrated to be similar to or within the

range of applicability of the problems analyzed by the public domain computer program.

iii. The computer program's solutions to a series of test problems have been demonstrated to be substantially identical to those obtained from classical solutions or from accepted experimental tests or to analytical results published in technical literature. The test problems should be demonstrated to be similar to or within the range of applicability of the classical problems analyzed to justify acceptance of the program.

A summary comparison should be provided for the results obtained in the validation of each computer program.

G. Tangential Shear. Design and analysis procedures for tangential shear are acceptable if in accordance with those contained in Article CC-3000 of the ASME Code. The regulatory staff should note the exceptions taken to the provisions of this article, as contained in Subsection II.5 of this SRP section.

H. Variation in Physical Material Properties. For the analysis of the effects of possible variations in the physical properties of materials on the analytical results, the upper and lower bounds of these properties should be used, wherever critical. The physical properties that may be critical include the soil modulus, modulus of elasticity, and Poisson's ratio of concrete.

I. Thickened Penetrations. The effect of the large, thickened penetration regions on the overall behavior of the containment may be treated by the same method used for localized loads as discussed in Subsection II.4.C.

J. Steel Liner Plate and Anchors. For the design and analysis of the liner plate and its anchorage system, the procedures furnished are found adequate and acceptable if in accordance with the provisions of Subarticle CC-3600 of the ASME Code. In general, the liner plate analysis should consider deviations in geometry resulting from fabrication and erection tolerances and variations of the assumed physical properties of the liner and anchor material. Since the liner plate is usually anchored at relatively closely spaced intervals, the analysis procedures are acceptable if based on either the classical plate or beam theory. Since the concrete shell is much stiffer than the liner plate, the strains in the liner will essentially follow those in the concrete. The strains in the concrete under the various load combinations as obtainable from the analysis of the shell are thus imposed on the liner plate, and the resulting strains and stresses in the liner and its anchors should be lower than the allowable limits defined in Tables CC-3720-1 and CC-3730-1 of the ASME Code.

K. Ultimate Capacity of Concrete Containment. 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.

i. Reinforced Concrete Containments

One acceptable methodology for cylindrical reinforced concrete 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 percent. The specific location of interest is the steel reinforcement in the hoop direction, closest to the inside surface of the concrete. The inside radius of the concrete wall should be used in calculating the strain in the hoop reinforcing steel.

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

The use of an alternate failure criteria for the analyses of noncylindrical containments and cylindrical containments are reviewed on a case by case basis.

Guidance on computer modeling of reinforced concrete containments for internal pressure capacity calculations appears 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 reinforcing steel and concrete. Sufficient 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 versus pressure should consider the statistical distribution of the material properties.

ii. Prestressed Concrete Containments

One acceptable methodology for cylindrical prestressed concrete 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 0.8 percent. This strain limit is applicable to all materials which contribute to resisting the internal pressure (i.e., tendons, rebars, and liner (if considered)). When calculating the pressure capacity contribution from the tendons, the above-specified strain limit is applicable to the full range of strain (from 0.0 psi at 0.0-percent strain up to the tendon contribution to pressure capacity at 0.8-percent strain).

The other items described previously for reinforced concrete containment, after the first paragraph identifying global strain limits, are also applicable to the approach used for prestressed concrete containments. The criteria presented for consideration of nonlinear material behavior of the reinforcing steel also apply to the tendons.

iii. Containment Penetrations

The methodologies described above apply 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. The analysis should also address other potential containment leak paths through mechanical and electrical penetrations.

iv. Special Considerations for Steel Elliptical 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. The analysis needs to evaluate this potential mode of failure 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 report form with the following identifiable information:

(1) The original design pressure, P_a, as defined in the ASME Code

(2) Calculated static pressure capacity

(3) Equivalent static pressure response calculated from dynamic pressure

(4) The associated failure mode

(5) The stress-strain relation of the liner steel and reinforcing and/or prestressing steel and the behavior of the liner under the postulated loading conditions in relation to that of the reinforcing and/or prestressing steel

(6) The criteria governing the original design and the criteria used to establish failure

(7) Analysis details and general results

(8) Appropriate engineering drawings adequate to allow verification of modeling and evaluation of analyses employed for the containment structure

L. Structural Audit. Appendix B to SRP Section 3.8.4 describes the conduct of a structural audit.

M. Design Report. The design report is considered acceptable when it satisfies the guidelines of Appendix C to SRP Section 3.8.4.

5. Structural Acceptance Criteria

A. For the structural portions of the containment, the specified allowable limits for stresses and strains are acceptable if they are in accordance with Subsection CC-3400 of the ASME Code and RG 1.136 (see Subsection II.3 of this SRP section), with the following exceptions:

CC-3421.5

For existing (older vintage) plants where a portion of the tangential shear stress, v_c, was permitted to be carried by the concrete, v_c is limited to 276 kPa or 40 pounds per square inch (psi) and 414 kPa (60 psi) for the load combinations of Table CC-3230-1, representing abnormal/severe environmental and abnormal/extreme environmental conditions, respectively. The criteria for design of steel reinforcement to resist the excess shear load above v_c should meet the provisions of the code of record for the containment design.
For other plants, the concrete should carry no tangential shear stress as indicated in Subsection CC-3421.5 of the ASME Code. The tangential shear strength provided by orthogonal reinforcement should be limited to the following:

$$0.833\sqrt{f'_c}(MPa);\left[10\sqrt{f'_c}(psi)\right]$$

where the value of f'_c is in units of MPa and psi in the first and second expression, respectively, in accordance with the ASME Code.

For prestressed concrete containments, the principal tensile stress should not exceed the following:

$$\tfrac{1}{3}\sqrt{f'_c}(MPa)\left[4\sqrt{f'_c}(psi)\right]$$

where the value of f'_c is in units of MPa and psi in the first and second expression, respectively, in accordance with the ASME Code.

B. For the liner plate and its anchorage system, the specified limits for stresses and strains are acceptable if in accordance with Tables CC-3720-1 and CC-3730-1 of the ASME Code, respectively.

6. Materials, Quality Control, and Special Construction Techniques

A. The specified materials of construction are acceptable if found to be in accordance with Article CC-2000 of the ASME Code with additional guidance provided by RG 1.107 and 1.136.

B. Quality control programs are acceptable if found to be in accordance with applicable portions of Articles CC-4000 and CC-5000 of the ASME Code with

additional guidance provided by RG 1.136 for quality assurance requirements.

C. Special construction techniques, if any, are reviewed on a case-by-case basis.

7. Testing and Inservice Surveillance Requirements

A. Procedures for the postconstruction, preoperational structural proof test proposed for the containment are acceptable if found in accordance with those delineated in Article CC-6000 of the ASME Code.

B. For reinforced and prestressed concrete containments, 10 CFR 50.55a imposes the examination requirements of Section XI, Subsections IWL and IWE, of the ASME Code. These subsections provide preservice examination, inservice inspection, and repair/replacement requirements, and acceptance criteria. The scope of Subsection IWL includes the concrete and unbonded posttensioning systems. Subsection IWE covers examination requirements for steel liners of concrete containments and their integral attachments; metallic shell portions of containment (e.g., steel head); containment hatches and airlocks; seals, gaskets and moisture barriers; and pressure-retaining bolting. The regulations in 10 CFR 50.55a(b)(2) specify the acceptable edition of the ASME Code and additional requirements beyond those contained in these subsections of the ASME Code. 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. For concrete containments, it is important to accommodate inservice inspection of critical areas. The staff considers that monitoring and maintaining the condition of containments is essential for plant safety. The staff reviews on a case-by-case basis 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 containments.

For plants with nonaggressive ground water/soil (i.e., pH > 5.5, chlorides < 500 ppm, sulfates <1500 ppm), an acceptable program for normally inaccessible, below-grade concrete walls and foundations is to (1) examine the exposed portions of below-grade concrete for signs of degradation, when excavated for any reason; and (2) conduct periodic site monitoring of ground water chemistry, to confirm that the ground water remains nonaggressive.

For plants with aggressive ground water/soil (i.e., exceeding any of the limits noted above), an acceptable approach is to implement a periodic surveillance program to monitor the condition of normally inaccessible, below-grade concrete for signs of degradation.

D. For prestressed concrete containments, inservice surveillance requirements for the tendons, as presented in the technical specifications of the operating license, are acceptable if in accordance with Section XI, Subsection IWL of the ASME Code; 10 CFR 50.55a; and RG 1.35 and 1.35.1 for ungrouted tendons and 1.90

for grouted tendons, respectively.

E. SRP Section 6.2.6 presents the preoperational and inservice integrated leak-rate testing criteria.

F. For new and unique containment designs (e.g., incorporating integrally connected passive systems with pools), the preoperational tests and inspections of containment discussed above need to consider items included in these unique features.

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 assure a quality product in keeping with the required safety function.

   SRP Section 3.8.1 provides guidance related to static and dynamic loadings and evaluation programs for concrete 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 3.8.1 cites RG 1.136 for guidance regarding load combination equations and ASME Code, Section III, Division 2, Section CC, for design guidance and acceptance criteria.

   Meeting these requirements and criteria provide assurance that engineering analysis and design of concrete containments for nuclear power plants will comply with 10 CFR Part 50 and that concrete 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, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these SSCs should 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 historically reported with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated. These data should be used to specify the design requirements of nuclear power plant components to be evaluated as part of construction permit, operating license, COL, or 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 3.8.1 and RG 1.136 provides guidance related to acceptable load combination equations, and ASME Code Section III, Division 2, Section CC, provides stress and deformation limits for evaluating the effects of natural phenomena, in combination with normal and accident conditions.

Meeting these requirements and criteria provide assurance that concrete 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, and 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 3.8.1 provides guidance related to load combinations, acceptance criteria, standards, and codes to assure compliance with GDC 4. Meeting these requirements and criteria provide assurance that concrete containments will withstand loads from internal events, such as those described above, and from external sources such as explosive hazards in proximity to the site, potential aircraft crashes (nonterrorism-related incidents), and missiles generated from activities of nearby military installations or turbine failures, thus decreasing the probability that these events would damage the containment and cause release of radioactive material.

4. Compliance with GDC 16 requires that reactor containment and associated systems be provided to establish an essentially leak-tight barrier against 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.

   Reinforced and prestressed concrete containments are designed, constructed, and tested in accordance with ASME Code, Section III, Division 2, Subsection CC, to provide a leak-tight barrier. A typical concrete containment consists of a thick-walled reinforced concrete cylinder and roof dome, with or without prestressing tendons. A steel liner is attached to the inside surface to provide the leak-tight barrier. The liner plate is thickened around penetrations to compensate for the openings. Steel closure heads and penetrations (e.g., personnel locks, equipment hatches, and mechanical and electrical penetrations) are designed in accordance with ASME Code, Section III, Division 1, Subsection NE. Seals provided at the penetrations must be designed to maintain containment integrity for design-basis accident conditions, including pressure, temperature, and radiation. Leak-tightness of the containment structure must be tested at regular intervals during the life of the plant in accordance with the provisions of Appendix J to 10 CFR Part 50, as described in SRP Section 6.2.6.

   SRP Section 3.8.1 provides guidance related to methods, including load combinations, acceptance criteria, standards, and codes, to ensure compliance with GDC 16. Meeting these requirements and criteria provide 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 which 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 3.8.1 provides guidance for 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 3.8.1 also describes a deterministic methodology for estimating the ultimate pressure capacity of reinforced concrete and prestressed concrete containments and also provides guidance for reviewing alternate methodologies.

   Meeting these requirements and criteria provide 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 caused by any LOCA and will perform its design safety function.

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

   SRP Section 3.8.1 provides guidance specifically related to design, construction, testing and inservice surveillance of concrete containments. Subsection II.2 of this SRP section cites the ASME Code, Section III, Division 2, Subsection CC, and Section XI, Subsections IWL and IWE, with additional guidance provided by RG 1.35, 1.35.1, 1.107, and 1.136 to satisfy the requirements of 10 CFR Part 50, Appendix B.

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

7. 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 should be designed to withstand a combined dead load and internal pressure of 310 kPa (45 psig).

   SRP Section 3.8.1 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.

8. 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 3.8.1 provides load combinations and acceptance criteria acceptable to the staff to demonstrate that the 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.

9. 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 BWRs and PWRs meet the requirements of the ASME Code; and (3) RG 1.84 and 1.147, lists NRC approved ASME Code cases that may be applied to design, fabrication, erection, construction, testing and inspection of containments, systems, and components.

   Compliance with 10 CFR 50.55a also requires that reinforced and prestressed concrete containments, and the metal components of concrete containments be examined in accordance with the requirements of Section XI, Subsections IWL and IWE, of the ASME Code, and supplemental requirements specified in 10 CFR 50.55a(b)(2)(viii) and 10 CFR 50.55a(b)(2)(ix). These subsections of the ASME Code provide requirements for preservice examination and inservice inspection, acceptance criteria, and repair/replacement requirements.

   SRP Section 3.8.1 provides review guidance to ensure that the requirements of 10 CFR 50.55a are appropriately addressed for concrete containments.

   Meeting these requirements 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 identifying the type of containment and its functional characteristics, the reviewer obtains information on similar and previously licensed applications for reference. Such information, which is available in SARs and amendments of previous license applications, enables identification of differences for the case under review. These differences require additional scrutiny and evaluation. New and unique features are evaluated and examined in greater detail for acceptability. 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 checks the list of codes, standards, guides, and specifications against the list in Subsection II.2 of this SRP section. The reviewer verifies that the applicant has used the appropriate code, standard, or guide and the applicable edition and stated effective addenda.

3. Loads and Loading Combinations. The reviewer verifies that the loads and load combinations, as described by the applicant, are consistent with those referenced in Subsection II.3 of this SRP section. The reviewer treats loading conditions that are unique to the site, such as potential aircraft crashes, and that are not specifically covered in Subsection II.3, on a case-by-case basis. Any deviations from the acceptance criteria for loads and load combinations should be adequately justified are identified.

4. Design and Analysis Procedures. The reviewer verifies that the applicant has committed to use the design and analysis procedures delineated in Article CC-3000 of the ASME Code and RG 1.136 (see Subsection II.3 of this SRP section). The reviewer evaluates any exceptions to these procedures on a case-by-case basis. The areas of review are evaluated for conformance with acceptance criteria contained in Subsection II.4 of this SRP section.

5. Structural Acceptance Criteria. The reviewer examines the limits on allowable stresses and strains in the concrete, reinforcement, the liner plate, and its anchors and in components of the prestressing system, if any, and compares them with the acceptable limits referenced 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 on the structure, the reviewer evaluates 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 evaluates the information provided on materials, quality control programs, and special construction techniques, if any, and compares to ensure that it meets the acceptance criteria referenced in Subsection II.6 of this SRP section. If the applicant is utilizing a material not used in previously licensed applications, the applicant needs to provide sufficient test and user data to establish the acceptability of the material. Similarly, the reviewer evaluates any new quality control programs or construction techniques to ensure that there will be no degradation of structural quality that might affect the structural integrity of the containment, the liner plate, and its anchorage system.

7. Testing and Inservice Surveillance Requirements. The reviewer examines the initial structural overpressure test program and compares it with that indicated as acceptable in Subsection II.7 of this SRP section. Proposed deviations are considered on a case-by-case basis. The review of inservice surveillance programs, including those for the prestressing tendons, if any, as presented in the technical specifications of the operating license, proceeds in a similar manner. These surveillance programs include preservice and inservice inspection in accordance with Subsection XI of the ASME Code, supplemental requirements of 10 CFR 50.55a, and regulatory guides identified in Subsection II.7 of this SRP section for any prestressing tendons. The reviewer evaluates the information related to preoperational and inservice integrated leak-rate tests of

containment in accordance with SRP Section 6.2.6.

The reviewer evaluates 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 containments on a case-by-case basis.

8. For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

   For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit (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 concrete containment is acceptable and meets the relevant requirements of 10 CFR 50.34(f); 10 CFR 50.44; 10 CFR 50.55a and GDC 1, 2, 4, 16, and 50. This conclusion is based on the following:

1. The applicant has met the requirements of 10 CFR 50.55a and GDC 1 with respect to ensuring that the concrete containment is designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with its safety function.

2. The applicant has met the requirements of GDC 2 by designing the concrete 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.

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

4. The applicant has met the requirements of GDC 16 by designing the concrete containment so that it is an essentially leak-tight barrier to prevent the uncontrolled

release of radioactive effluents to the environment.

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

6. The applicant has met the requirements of Appendix B to 10 CFR Part 50 in that the quality assurance program provides adequate measures for implementing guidelines relating to structural design.

7. If applicable, the applicant has met the requirements of 10 CFR 50.34(f) by designing the plant to accommodate hydrogen generation equivalent to 100-percent metal-water reaction of the fuel cladding and loads associated with hydrogen burning or inerting system actuation.

8. The applicant has met the requirements of 10 CFR 50.44 by designing the plant to accommodate the loads associated with hydrogen gas generated from a metal-water reaction of the fuel cladding so that there is no loss of containment structural integrity.

The criteria used in the analysis, design, construction, testing and inservice surveillance of the concrete containment structure to account for anticipated loadings and postulated conditions that the structure may experience during its service lifetime conform with established criteria and with codes, standards, guides, and specifications acceptable to the staff. These include meeting the positions of RG 1.7, 1.35, 1.35.1, 1.90, 1.107, 1.136, and industry standard ASME Code, Section III, Division 2, and Section XI, Subsections IWL and IWE.

The use of these criteria, as defined by applicable codes, standards, guides, and specifications; 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, provides reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated accidents occurring within and outside the containment, the structure will withstand the specified design conditions without impairment of structural integrity or safety function of limiting the release of radioactive material.

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 six months or more after the date of issuance of this SRP section, unless superseded by a later revision.

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

VI. REFERENCES

1. 10 CFR 50.34, "Contents of Applications; Technical Information."

2. 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors."
3. 10 CFR 50.55a, "Codes and Standards."

4. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

5. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

6. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Dynamic Effects Design Bases."

7. 10 CFR Part 50, Appendix A, General Design Criterion 16, "Containment Design."

8. 10 CFR Part 50, Appendix A, General Design Criterion 50, "Containment Design Basis."

9. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

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

11 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

12. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

13. ASCE 4-98, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," American Society of Civil Engineers (Sections 3.1.3 and C3.1.3 for concrete cracking, and Section 3.5.3.2 for embedded walls).

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

Concrete Reactor Vessels and Containments." New York, NY.

15. American Society of Mechanical Engineers. Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Concrete Components of Light-Water Cooled Power Plants." New York, NY.

16. American Society of Mechanical Engineers. Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Plants." New York, NY.

17. Institute of Electrical and Electronics Engineers., "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," Appendix D, "Test Duration and Number of Cycles," IEEE 344-1987. June 1987.

18. U.S. Nuclear Regulatory Commission. "Containment Integrity Research at Sandia National Laboratories, An Overview." NUREG/CR-6906. Washington, DC. July 2006.

19. Regulatory Guide 1.7, Revision 2, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident."

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21. Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments."

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

23. Regulatory Guide 1.90, "Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons."

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25. Regulatory Guide 1.107, "Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures."

26. Regulatory Guide 1.115, "Protection Against Low Trajectory Turbine Missiles."

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

29. SEI/ASCE 37-02, "Design Loads on Structures During Construction," Structural Engineering Institute/American Society of Civil Engineers, 2002.

30. SRM SECY 93-087, Staff Requirements Memorandum, “SECY-93-087—Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs.”

31. Regulatory Guide 1.84, “Design, Fabrication, and Material Code Case Acceptability, ASME Section III.”

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APPENDIX A TO SRP SECTION 3.8.1

STAFF POSITION ON BOILING-WATER REACTOR CONTAINMENT POOL DYNAMICS

1. LOSS-OF-COOLANT ACCIDENTS

   A. Loads associated with loss-of-coolant accidents (LOCAs) (such as drywell pressure; annulus pressurization; suppression pool pressures including chugging, condensation oscillation, vent clearing, and pool swell; and jet loads and drag loads acting on submerged structures/components and those above the water surface) should be treated as abnormal pressure loads, P_a. Appropriate load combinations and load factors should be applied accordingly.

   B. All of the various LOCA-induced loads may be combined in accordance with their actual time-dependent mutual occurrence.

2. SAFETY/RELIEF VALVE DISCHARGE

   A. Safety/relief valve (SRV) loads (such as single valve, single valve plus adjacent valve, automatic depressurization system valves, and all valves) should be included in all load combinations as defined in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 2, Subsection CC. However for the load combination that contains 1.5 P_a, a load factor of 1.25 should be applied to the appropriate SRV loads. Also, for the severe environmental load combination, the load factor should be 1.3, which is consistent with the practice of treating SRV as a live load.

   B. A single active failure causing one SRV discharge must be considered in combination with the design-basis accident.

   C. Appropriate multiple SRV discharges should be considered in combination with the small-break accident and intermediate-break accident.

   D. Thermal loads resulting from SRV discharge should be treated as T_o for normal operation and T_a for accident conditions.

3. COMBINATION OF DYNAMIC LOADS

   Revision 1 of NUREG-0484 presents guidance on the acceptable methods for combining LOCA, SRV, and the safe-shutdown earthquake (SSE). This report states that the square root of sum of the squares (SRSS method is appropriate for the following:

   A. Combination of SSE and LOCA dynamic responses for all ASME Class 1, 2, or 3 systems, components, or supports. For dynamic responses resulting from the same initiating event, when the time-phase relationship between the responses cannot be established, the absolute summation of these dynamic responses should be used.

   B. Combining responses of dynamic loads other than LOCA and SSE, if a

nonexceedance probability of 84 percent or higher is achieved for the combined SRSS response.

Although the above criteria were prepared for ASME systems, components, and supports, they are applicable to seismic Category I structures as well.

REFERENCES

1. NUREG-0484, Revision 1, “Methodology for Combining Dynamic Responses,” May 1980.

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.

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