U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

3.8.1 CONCRETE CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - ~~Structural Engineering Branch (SEB)~~Civil Engineering and Geosciences Branch (ECGB)[1]

Secondary - None

I. AREAS OF REVIEW

The following areas relating to concrete containments or to concrete portions of steel/concrete containments, as applicable, are reviewed.

1. Description of the Containment

   The descriptive information, including plans and sections of the structure, is reviewed 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 concrete containment is identified and its structural and functional characteristics are examined. Among the various types of concrete containments reviewed are:

   a. Reinforced- and prestressed-concrete, boiling-water-reactor (BWR)[2] containments utilizing the pressure-suppression concept, including the Mark I (modified 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) and the ABWR (with steel liner, upper and lower drywell chambers, and suppression chamber)[3].

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

b. Reinforced-concrete, pressurized-water-reactor (PWR)[4] containments utilizing 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 utilizing special features or modifications of the above-listed types.

Various geometries have been utilized 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. If containments with other types of geometry are reviewed, the necessary ~~modifications to~~ deviations from[5] this SRP section are made 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 arrangement of the containment and the relationship and interaction of the shell with its surrounding structures and with its interior compartment walls and floors are reviewed to determine the effect which these structures could have upon the design boundary conditions and expected structural behavior of the containment when subjected to design loads.

General information related to the containment shell is reviewed, including the following:

a. The base foundation slab, including the main reinforcement; the floor liner plate and its anchorage and stiffening system; and[6] 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 which 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. Steel components of concrete containments that resist pressure and are not backed by structural concrete are covered by SRP~~Standard Review Plan~~[7] Section 3.8.2.

2. Applicable Codes, Standards, and Specifications

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 is reviewed. The specific editions, dates, or addenda identified for each document are reviewed.

3. Loads and Loading Combinations

Information pertaining to the applicable design loads and various combinations thereof is reviewed with emphasis on the extent of compliance with Article CC-3000 of the ASME Section III, Division 2, "Code for Concrete Reactor Vessels and Containments," (Reference~~.~~ 1)[8] (hereafter "the Code"). The loads normally applicable to concrete containments include the following:

a. Those loads encountered during preoperational testing.

b. Those loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads due to operating temperature, and hydrostatic loads and hydrodynamic loads resulting from safety relief valve (SRV) actuation such as those present in pressure-suppression containments utilizing water.

c. Those loads to be sustained during severe environmental conditions, including those induced by the design wind and the operating basis earthquake specified for the plant site.

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

e. Those loads to be sustained during abnormal plant conditions, which include loss-of-coolant accidents (LOCAs). The main abnormal plant condition for containment design is the design basis LOCA. Also considered are other accidents involving various high-energy pipe ruptures. Loads induced on the containment by such accidents include elevated temperatures and pressures and possible localized loads such as jet impingement and associated missile impact. For BWR containments the LOCA-related or LOCA/SRV-related hydrodynamic loads in suppression pools manifested as jet loads and/or pressure loads should be considered.

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

g. Pressure and dead loads during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction 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.[9]

The various combinations of the above loads that are normally postulated and reviewed include the following:

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 might be applicable also. Such 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, explosive hazards in proximity of the site, and projectiles and missiles generated from activities of nearby military installations.

4. Design and Analysis Procedures

The design and analysis procedures utilized for the containment are reviewed with emphasis on the extent of compliance with Article CC-3000 of the Code, 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 utilized 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. Steel penetration closures are covered by ~~Standard Review Plan~~SRP[10] Section 3.8.2.

j. Ultimate capacity of the concrete containment.

k. Structural audit.

l. Design report submitted for review.

5. Structural Acceptance Criteria

The design limits imposed on the various parameters that serve to quantify the structural behavior of the containment are reviewed, with emphasis on the extent of compliance with Article CC-3000 of the Code, specifically with respect to allowable stresses, strains, gross deformations, and other parameters that identify quantitatively the margins of safety. For each load combination specified, the proposed allowable limits are compared with the acceptable limits delineated in subsection II.5 of this SRP section. Included in these allowable limits are 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

Information provided on materials that are used in construction of the containment is reviewed with emphasis on the extent of compliance with Article CC-2000 of the Code. Among the major materials of construction that are reviewed are the following:

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 quality control program that is proposed for the fabrication and construction of the containment is reviewed with emphasis on the extent of compliance with Articles CC-4000 and CC-5000 of the Code, including the following:

Examination of the materials, including 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.

Special, new, or unique construction techniques, if proposed, such as slip forming, 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 Requirements

The preoperational structural testing program for the completed containment and for individual components, such as personnel and equipment locks and hatches, is reviewed, including the objectives of the test program and acceptance criteria, with emphasis on the extent of compliance with Article CC-3000 of the Code. Inservice surveillance programs, such as the periodic surveillance and inspection of the prestressing tendons, if any, are also reviewed, including the applicable technical specifications, at the operating license stage. Special testing and inservice surveillance requirements proposed for new or previously untried design approaches are also reviewed on a case-by-case basis.

Review Interfaces[11]

The ~~SEB~~ECGB[12] coordinates other branches' evaluations that interface with structural engineering aspects of the review, as follows:

1. Determination of structures which are subject to a quality assurance program in accordance with the requirements of Appendix B to 10 CFR Part 50 is performed by the Mechanical Engineering Branch ~~(MEB)~~(EMEB)[13] as part of its primary review responsibility for SRP Sections 3.2.1 and 3.2.2. The ~~SEB~~ECGB[14] will perform its review of safety-related structures on that basis.

2. Determination of pressure loads from high-energy lines located in safety-related structures other than containment is performed by the ~~Auxiliary Systems Branch (ASB)~~Plant Systems Branch (SPLB)[15] as part of its primary review responsibility as described for SRP Section 3.6.1. The ~~SEB~~ECGB[16] accepts the loads thus generated as approved by the ~~ASB~~SPLB,[17] to be included in the load combination equations of this SRP section.

3. General Design Criterion 4 allows the exclusion of dynamic effects of pipe ruptures if analyses (i.e., leak-before-break analyses) demonstrate the probability of rupture is extremely low. For containment design, the applicability of these analyses is limited to

localized effects only. The Materials and Chemical Engineering Branch (EMCB) performs a review of those applications that propose to eliminate consideration of design loads associated with the dynamic effects of pipe rupture, as part of its primary review responsibility for SRP Section 3.6.3 (to be developed).[18]

4. Determination of loads generated due to pressure under accident conditions is performed by the ~~Containment Systems Branch (CSB)~~ Containment Systems and Severe Accident Branch (SCSB)[19] as part of its primary review responsibility for SRP Section 6.2.1. ~~SEB~~ECGB[20] accepts the loads thus generated, as approved by the ~~(CSB)~~(SCSB),[21] to be included in the load combinations in this SRP section.

5. The review for quality assurance is coordinated and performed by the ~~Quality Assurance Branch~~Quality Assurance and Maintenance Branch (HQMB)[22] as part of its primary review responsibility for SRP ~~Section 17.0~~Chapter 17.[23]

6. The SCSB verifies that containment performance meets severe accident criteria as part of its primary review responsibility for SRP Section 19.2 (proposed).[24]

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding primary branch.

II. ACCEPTANCE CRITERIA

~~SEB~~ECGB[25] acceptance criteria for the design of the concrete containment ~~is~~are[26] based on meeting the relevant requirements of the following regulations:

A. 10 CFR Part 50.55a and General Design Criterion 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.

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

C. General Design Criterion 4 as it relates to the concrete containment being capable of withstanding the dynamic effects of equipment failures including missiles and blowdown loads associated with the loss-of-coolant accident.

D. General Design Criterion 16 as it relates to the capability of the concrete containment to act as a leaktight membrane to prevent the uncontrolled release of radioactive effluents to the environment.

E. 10 CFR 50.34 and[27] General Design Criterion 50 as ~~it~~they[28] relate to containment internal structures being designed with sufficient margin of safety to accommodate appropriate design loads.

The regulatory guides and industry standards identified in ~~item 2 of this subsection~~subsection II.2[29] provide~~s~~[30] information, recommendations, and guidance and, in general, define~~s~~[31] a basis acceptable to the staff that may be used to implement the requirements of 10 CFR ~~Part 50, §~~ 50.34, 10 CFR[32] 50.55a, and ~~GDC~~General Design Criteria[33] 1, 2, 4, 16, and 50. Also, specific acceptance criteria necessary to meet the relevant requirements of these regulations for the areas of review described in subsection I of this SRP section are as follows:

1. Description of the Containment

   The descriptive information in the safety analysis report (SAR) is considered acceptable if it meets the minimum requirements set forth in Subsection 3.8.1.1 of ~~the~~Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." If the concrete containment has new or unique features that are not specifically covered in ~~the Standard Format~~Regulatory Guide 1.70,[34] the reviewer determines that the information necessary to accomplish a meaningful review of the structural aspects of these new or unique features is presented, as appropriate.

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 either applicable in their entirety or in part. The following codes and guides are acceptable.

Code	Title
ASME, Section III, Division 2	"Code for Concrete Reactor Vessels and Containments"

Regulatory Guides	Title
~~1.10~~	~~Mechanical (Cadweld) Splices for Reinforcing Bars of Category I Concrete Structures~~
~~1.15~~	~~Testing of Reinforcing Bars for Category I Concrete Structures~~
~~1.18~~	~~Structural Acceptance Testing for Concrete Primary Reactor Containments~~
~~1.19~~	~~Nondestructive Examination of Primary Containment Liner Welds~~[35]
1.35	Inservice Surveillance of Ungrouted Tendons in Pre-stressed Concrete Containment Structures

1.35.1 Determining Prestressing Forces for Inspection of Prestressed Concrete Containments[36]

~~1.55 Concrete Placement in Category I Structures[37]~~

1.90 Inservice Surveillance in Prestressed Concrete Containments with Grouted Tendons

1.94 Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel during the Construction Phase of Nuclear Power Plants

~~1.103 Post-Tensioned Prestressing System for Concrete Reactor Vessels and Containments[38]~~

1.107 Qualification for Cement Grouting for Prestressing Tendons in Containment Structures

1.136 Material for 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 Code with the exceptions listed below taken to the requirements specified in Table CC-3230-1.

~~a. In the third combination under "abnormal/severe environmental condition "0.5" under Ess should be replaced by the word "or."[39]~~

~~b~~a. The maximum values of P_a, T_a, R_a, ~~Y(r), Y(j) and Y(m)~~ $_{rr}$, R_{rj}, and R_{rm}[40] should be applied simultaneously, where appropriate, unless a time-history analysis is performed to justify doing otherwise.

~~c~~b. Hydrodynamic loads resulting from LOCA and/or SRV actuation should be combined as indicated in the appendix to this SRP section.

~~d~~c. Where post-LOCA flooding is a design consideration, the following combination should also be considered in the factored load category:

1.0 D + 1.0 L + 1.0 F + 1.0 Feqo, where D, L, Feqo are as defined in the Code and F is the load generated by the post-LOCA flooding of the containment.

10 CFR 50.34(f)(3)(v) requirements regarding loads and loading combinations include the following:

d. Containment integrity should be maintained by meeting the requirements of Subarticle CC-3720 of the ASME Code, "Factored Load Category," (considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction and accompanied by either hydrogen burning or added pressure from postaccident inerting (assuming carbon dioxide is the inerting agent). At a minimum, the code requirements will be met for a combination of dead load and an internal pressure of 310 kPa (45 psig)[41].[42]

e. 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 Code.[43]

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

1. For the Factored Load Category:

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

2. For the Service Load Category, the strains in the containment liner shall not exceed the limits set forth in Subarticle CC-3720 when exposed to pressure P_{g3}.

3. As a minimum design condition for either condition 1 or 2 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% fuel clad metal-water reaction

   P_{g2}= Pressure resulting uncontrolled hydrogen burning

   P_{g3}= Pressure resulting from postaccident inerting assuming carbon dioxide is the inerting agent[44]

4. <u>Design and Analysis Procedures</u>

The procedures of design and analysis utilized for the concrete containment, including the steel liner, are acceptable if found in accordance with those stipulated in

Article CC-3300 of the Code. In particular, for the areas of review outlined in subsection I.4 of this SRP section, 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 be accomplished through 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 have to be so selected that any further extension of the boundaries will not affect the results by more than 15~~ percent~~%.[45]

If only the containment shell and its foundation mat are taken into consideration in the analysis, 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 comparability 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 utilized in the shell, the overall behavior of the shell has been shown to be axisymmetric under pressure. Therefore, it is acceptable if such an assumption is made with respect to the containment geometry. However, for loads such as those induced by wind, tornadoes, earthquakes, and pipe rupture, the nonaxisymmetric effect of these loads should be considered.

c. Transient and localized loads

During normal operation, a linear temperature gradient across the containment wall thickness may develop. After a ~~the loss-of-coolant accident (~~LOCA~~)~~,[46] 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. Effects of such transient loads should be considered.

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

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 which is 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, consideration should be given to the differences in the environment between the test samples and the actual concrete in the structure. Cracking of the concrete may be considered in either of the following two ways: (i) the moments, forces, and shears under load may be obtained on the basis of an uncracked section for all loading combinations. In sizing the reinforcing steel required, however, the concrete shall not be relied upon for resisting tension. Thermal moments may be modified to take creep and cracking into consideration. (ii) For axisymmetrical loadings, cracking of the concrete may be considered through the use of computer programs which are capable of treating such cracking by an iterative process. However, for nonaxisymmetric loadings, most of the computer programs available do not have the capability of considering cracking, since the structure itself becomes nonaxisymmetric when concrete cracking is to be considered iteratively. Accordingly, if the concrete is cracked under any load combination involving axisymmetric and nonaxisymmetric loadings, a method should be described for considering cracking. Such methods are reviewed on a case-by-case basis.

e. 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 a recognized program in the public domain and has had sufficient history of use to justify its applicability and validity without further demonstration.

(ii) The computer program solution to a series of test problems has 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 solution to a series of test problems has 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.

f. Tangential shear

Design and analysis procedures for tangential shear are acceptable if in accordance with those contained in Article CC-3000 of the Code. The exceptions taken by the regulatory staff to the provisions of this article, as contained in subsection II.5 of this SRP section, are to be noted.

g. Variation in physical material properties

For considering 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 in the analysis, wherever critical. Among the physical properties that may be critical include the soil modulus, ~~and~~[47] modulus of elasticity, and Poisson's ratio of concrete.

h. Thickened penetrations

The effect of the large, thickened penetration regions on the overall behavior of the containment may be treated in the same manner as for localized loads discussed in ~~item (c)~~ subsection II.4.c.[48]

i. 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 Code. In general, the liner plate analysis should consider deviations in geometry due to 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 Code.

j. Ultimate capacity of concrete containment

An analysis should be performed to determine the ultimate capacity of the containment.

The pressure-retaining capacity of localized areas as well as of the overall containment structure should be determined.

The analysis should be made on the basis of the allowable material strength specified in the Code. However, if the actual material properties (such as concrete cylinder compressive strength, mill test results of reinforcing steel and liner plate, strength variations indicated by mill test certificates) and other uncertainties are available, the lower and upper bounds of the containment capacity may be established statistically.

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

(1) The original design pressure, P_a, as defined in the 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; and

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

k. Structural Audit

Structural audit is conducted as described in Appendix B to SRP Section 3.8.4.

l. Design Report

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 Code, ~~but~~ [49]with the following exceptions:

CC-3421.5

Under no conditions shall the tangential shear stress carried by the concrete, v_c, exceed 276 kPa (40 psi)[50] and 414 kPa (60 psi~~,~~[51] for the load combinations of Table CC-3230-1, representing abnormal/severe environmental and abnormal/extreme environmental conditions, respectively.

For prestressed concrete, the principal tensile stress shall not exceed:

$$27.6\sqrt{f'c}\ (\text{kPa})\ [4\sqrt{f'c}\ (\text{psi})]$$

where the value of f'c is always in units of psi, in accordance with the Code.[52]

CC-3431.1

The 33⅓% increase in allowable stresses is permitted only for temperature loads and not for OBE or wind loads.

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 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 Code and[53] augmented by Regulatory Guides ~~1.103,~~[54] 1.107 and 1.136.

b. Quality control programs are acceptable if found to be[55] in accordance with applicable portions of Articles CC-4000 and CC-5000 of the Code as augmented by Regulatory Guides ~~1.10 for Cadweld reinforcement splicing (Ref. 3), 1.15 for testing of reinforcing bars (Ref. 4), 1.19 for the nondestructive examination of the liner plate welds (Ref. 5), 1.55 for concrete placement (Ref. 6), and~~[56] 1.94 and 1.136[57] 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 Code ~~as augmented by the provisions delineated in Regulatory Guide 1.18 (Ref.7).~~[58]

b. 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 Regulatory Guides 1.35 ~~(Ref. 8)~~[59] and 1.35.1[60] for ungrouted tendons and 1.90 for grouted tendons ~~(Ref. 9)~~,[61] respectively.

Technical Rationale[62]

The technical rationale for application of these acceptance criteria to reviewing concrete containments is discussed in the following paragraphs:[63]

1. Compliance with GDC 1 requires that structures, systems, and components 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.

   SRP Section 3.8.1 describes staff positions related to static and dynamic loadings and evaluation programs for concrete containments. It describes acceptable materials, design methodology, quality control procedures, construction methods, and inservice inspections as well as documentation criteria for design and construction controls at nuclear power plants.

   Meeting these criteria provides assurance that the engineering analysis and design of concrete containments for nuclear power plants will comply with 10 CFR Part 50 and that containments will perform their intended safety function.

2. Compliance with GDC 2 requires that systems, structures, and components important to safety be designed to withstand the effects of expected natural phenomena when combined with the effects of normal and accident conditions without loss of capability to perform their safety function.

   To ensure that the containment of a nuclear plant is designed to withstand natural phenomena, it is necessary to specify the most severe natural phenomena event that may occur as a function of the frequency of occurrence. To meet the requirements of GDC 2 for all natural phenomena related to meteorological events (e.g., earthquakes, snow and ice load, meteorological conditions affecting the ultimate heat sink, tornado parameters, and wind speed), it is necessary to review historical data and obtain the expected frequency of the most severe occurrences. These data are then used to specify the design requirements for nuclear plant components to be evaluated as part of construction permit (CP), OL, combined license (COL), or early site permit reviews or for site parameter envelopes in the case of design certifications, thereby ensuring that components important to safety will function in a manner that will maintain the plant in a safe condition.

Meeting this requirement provides assurance that the containment will be designed to withstand the effects of natural phenomena without loss of capability to perform its intended function.

3. Compliance with GDC 4 requires that nuclear power plant structures, systems, and components 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.

   SRP Section 3.8.1 cites acceptance criteria, standards, and codes so that concrete containments will resist dynamic effects, including missiles and pipe whipping, discharging fluids, and other events, including LOCA effects.

   Meeting this requirement provides assurance that the containment will withstand dynamic effects such as missiles impacts associated with tornadoes or other external sources, including aircraft, thus decreasing the probability that these events would cause damage to the containment that could result in release of radioactive material.

4. Compliance with GDC 16 requires that a reactor containment and its associated systems be provided to establish an essentially leaktight 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.

   GDC 16 applies to this SRP section because a typical concrete containment has a liner plate on the inside face of the containment structure that forms a leaktight barrier. To serve as a leaktight barrier, the liner plate is designed as a Category I structure in accordance with the provisions of the ASME Code, Article CC-3000.

   Meeting these requirements provides assurance that the containment will perform its intended safety function and that uncontrolled releases of radioactivity to the environment will be prevented.

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 itself 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 caused by a LOCA.

   These requirements apply to this SRP section because the containment structure design is based on the elastic behavior of the material used. That is, when a strength design approach is used, the structure is dimensioned so that (a) the combination of loads multiplied by appropriate load factors will not cause a permanent deformation within the structure and (b) the stress at all points is less than the yield or buckling stress. Design criteria for containment structures are provided in the ASME Code, as supplemented by the regulatory guides listed in subsection II.2 of this SRP section. Penetrations are generally analyzed using the finite element method, taking into consideration temperature gradient, cracking of concrete, anchorage of sleeves, and shrinkage and creep.

Meeting these requirements provides assurance that the containment structure, the penetrations, and the internal compartments will be able to withstand loads resulting from pressure and temperature conditions and will perform their design safety function.

6. Compliance with 10 CFR 50.34 requires that plant designs (a) accommodate loadings associated with hydrogen generation equivalent to 100% 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)(A)(1) requires that containment structures should be designed to withstand a combined dead load and internal pressure of 310 kPa (45 psig).

   These requirements have been incorporated into SRP Section 3.8.1 by means of the load combination equations provided in subsection II.5.b.

   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.

7. Compliance with 10 CFR 50.55(a) requires that structures, systems, and components be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of their safety function and that suitable optional Code Cases be applied to such structures, systems, and components.

   The SRP Section 3.8.1 cites Regulatory Guides 1.35, 1.35.1, 1.90, 1.94, 1.107, and 1.136, which provide guidance regarding construction, quality control, tests, and inspections that is acceptable to the staff. ASME Code Section III, Division 2, addresses concrete containments. The Code requirements impose specific restrictions to ensure that structures, systems, and components will perform their intended safety function.

   Meeting these requirements provides assurance that the containment structure will perform its safety function to limit the release of radioactive material.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes material from the review procedures described below as may be appropriate for a particular case.

1. Description of the Containment

   After the type of containment and its functional characteristics are identified, information on similar and previously licensed applications is obtained for reference. Such information, which is available in safety analysis reports 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 that have not been used in the past are of particular interest and are examined in greater detail. The information furnished in the SAR is reviewed for completeness in accordance with the Regulatory Guide 1.70~~"Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition" (Ref. 2)~~.[64] A decision is then made with regard to the sufficiency of the descriptive information provided in the SAR. Any additional required information not provided is requested from the applicant at an early stage of the review process.

2. Applicable Codes, Standards, and Specifications

The list of codes, standards, guides, and specifications ~~are~~is[65] checked against the list in subsection II.2 of this SRP section. The reviewer ~~assures~~ensures[66] that the applicable edition and stated effective addenda are utilized.

3. Loads and Loading Combinations

The reviewer verifies that the loads and load combinations, as described by the applicant, are as conservative as those referenced in subsection II.3 of this SRP section. Loading conditions that are unique to the site, such as potential aircraft crashes, and that are not specifically covered in subsection II.3 are treated on a case-by-case basis. Any deviations from the acceptance criteria for loads and load combinations that have not been adequately justified are identified as unacceptable, and this information is transmitted to the applicant for further consideration.

4. Design and Analysis Procedures

The reviewer ~~assures himself~~ensures[67] that the applicant has committed to utilize design and analysis procedures delineated in Article CC-3000 of the Code. Any exceptions to these procedures are reviewed and evaluated on a case-by-case basis. In particular, the areas of review contained in subsection II.4 of this SRP section are evaluated for conformance with the acceptance criteria.

5. Structural Acceptance Criteria

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, are reviewed and compared with the acceptable limits referenced in subsection II.5 of this SRP section. Where the applicant proposes to exceed some of these limits for some of the load combinations and at some localized points on the structure, the justification, provided to show that the structural integrity of the containment will not be affected, is reviewed. If such justification is unacceptable, the applicant is required to submit additional justification or otherwise comply with the acceptance criteria delineated in subsection II.5 of this SRP section.

6. Materials, Quality Control, and Special Construction Techniques

The information provided on materials, quality control programs, and special construction techniques, if any, is reviewed and compared with that referenced in subsection II.6 of this SRP section. If a material not used in previously licensed applications is utilized, the applicant is requested to provide sufficient test and user data to establish the acceptability of the material. Similarly, any new quality control programs or construction techniques are reviewed and evaluated to ~~assure~~ensure[68] 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 initial structural overpressure test program is reviewed and compared with that indicated as acceptable in subsection II.7 of this SRP section. Proposed deviations are considered on a case-by-case basis. Inservice surveillance programs, particularly for the prestressing tendons, if any, as presented in the technical specifications of the operating license, are similarly reviewed.

In the ABWR and System 80+ design certification FSERs the Staff accepted an exemption to the 10 CFR 100, Appendix A requirement that all safety-related SSCs be designed to remain functional and within applicable stress and deformation limits when subjected to an OBE. The Staff reviewed the controlling load combinations and concluded that, in most cases, load combinations incorporating an OBE load do not control the design of concrete structures. As a result, the Staff concluded that there would be no reduction in the safety margin of concrete structures due to the elimination of the OBE as a design requirement.[69]

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[70]

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided to satisfy the requirements of this SRP section and concludes that ~~his~~the[71] evaluation is sufficiently complete and adequate to support the following type of conclusive statement to be included in the staff's safety evaluation report (SER):[72]

The staff concludes that the design of the concrete containment is acceptable and meets the relevant requirements of 10 CFR Part 50, §§ 50.34 and[73] 50.55a, and General Design Criteria 1, 2, 4, 16, and 50. This conclusion is based on the following:

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

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 condition 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 leaktight 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 ~~assuring~~ ensuring[74] 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 10 CFR 50.34 by designing the plant (a) to accommodate hydrogen generation equivalent to 100% metal-water reaction of the fuel cladding and loads associated with hydrogen burning and/or inerting system actuation.[75]

The criteria used in the analysis, design, and construction of the concrete 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 and with codes, standards, guides, and specifications acceptable to the Regulatory staff. These include meeting the positions of Regulatory Guides ~~1.10, 1.15, 1.18, 1.19,~~[76] 1.35, 1.35.1,[77] ~~1.55,~~[78] 1.90, 1.94, ~~1.103~~,[79] 1.107, 1.136, and industry standard ASME Boiler and Pressure Vessel Code, Section III, Division 2.

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, provide 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 design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.[80]

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52.[81] Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section.[82]

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

VI. REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, Division 2, "Code for Concrete Reactor Vessels and Containments."[83]

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

~~3. Regulatory Guide 1.10, "Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures."~~

~~4. Regulatory Guide 1.15, "Testing of Reinforcing Bars for Category I Concrete Structures."~~

~~5. Regulatory Guide 1.19, "Nondestructive Examination of Primary Containment Liner Welds."~~

~~6. Regulatory Guide 1.55, "Concrete Placement in Category I Structures."~~

~~7. Regulatory Guide 1.18, "Structural Acceptance Test for Concrete Primary Reactor Containments."~~[84]

~~8~~3.[85] Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containments."

4. Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments."[86]

~~9~~5.[87] Regulatory Guide 1.90, "Inservice Surveillance in Prestressed Concrete Containments with Grouted Tendons."

~~10~~6.[88] Regulatory Guide 1.94, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants."

~~11. Regulatory Guide 1.103, "Post-Tensional Prestressing Systems for Concrete Reactor Vessels and Containments."~~[89]

~~12~~7.[90] Regulatory Guide 1.107, "Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures."

~~13~~8.[91] Regulatory Guide 1.136, "Materials, Construction, and Testing of Concrete Containments."

~~14~~9.[92] 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standard and Records."

1~~5~~0.[93] 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

1~~6~~1. 10 CFR Part 50, Appendix A, General Design Criterion 4, ~~"Environmental and Missile Design Bases."~~"Environmental and Dynamic Effects Design Bases."[94]

1~~7~~2.[95] 10 CFR Part 50, Appendix A, General Design Criterion 16, "Containment Design."

1~~8~~3.[96] 10 CFR Part 50, Appendix A, General Design Criterion 50, "Containment Design Basis."

14. 10 CFR Part 50.34, "Contents of Applications; Technical Information."[97]

1~~9~~5.[98] 10 CFR Part 50, 50.55a, "Codes and Standards."

Appendix to SRP Section 3.8.1

~~STRUCTURAL ENGINEERING BRANCH~~
CIVIL ENGINEERING AND GEOSCIENCES BRANCH[99] POSITION
U.S. NUCLEAR REGULATORY COMMISSION

BWR MARK III CONTAINMENT POOL DYNAMICS

1. POOL SWELL

   a. Bubble pressure, bulk swell and froth swell loads, drag pressure, and other pool swell loads should be treated as abnormal pressure loads, P_a. Appropriate load combinations and load factors should be applied accordingly.

   b. The pool swell loads and accident pressure may be combined in accordance with their actual time-dependent mutual occurrence.

2. SAFETY RELIEF VALVE (SRV) DISCHARGE

   a. The SRV loads should be treated as live loads in all load combinations with the exception of the combination that contains 1.5 P_a where a load factor of 1.25 should be applied to the appropriate SRV loads.

   b. A single active failure causing one SRV discharge must be considered in combination with the design basis accident (DBA).

   c. Appropriate multiple SRV discharge should be considered in combination with the small-break accident (SBA) and intermediate break accident (IBA).

   d. Thermal loads due to SRV discharge should be treated as T_0 for normal operation and T_a for accident conditions.

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Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	Current PRB name and abbreviation	Changed PRB to Civil Engineering and Geosciences Branch (ECGB).
2.	Editorial	Defined "BWR" as "boiling water reactor."
3.	Potential Impact 25965	Editorial change to revised the description of BWR containment types to include the ABWR containment. This change is editorial in that it only involves the addition of a brief description of the containment type and does not reflect any staff positions or requirements.
4.	Editorial	Defined "PWR" as "pressurized water reactor."
5.	Editorial	Substituted "deviations from" for "modifications to" for clarification.
6.	Editorial	Added "and" before last item in a series.
7.	Editorial	Used "SRP" as defined in item 4 above (global change for this section).
8.	SRP-UDP Format Item, Reformat References	Spelled out "Reference".
9.	Integrated Impact No. 540	Added design criteria to reflect the provisions of 10 CFR 50.34(f)(3).
10.	Editorial	Abbreviated "Standard Review Plan" as "SRP".
11.	SRP-UDP format item	Added "Review Interfaces" to AREAS OF REVIEW and put in numbered paragraph form to describe how other branches support the review of the concrete containments.
12.	Current PRB abbreviation	Changed PRB to ECGB.
13.	Current review branch abbreviation	Changed review interface branch to EMEB.
14.	SRP-UDP format item	Changed PRB to ECGB.
15.	Current review branch name and abbreviation	Changed review interface branch to Plant Systems Branch (SPLB).
16.	Current review branch abbreviation	Changed review interface branch to ECGB.
17.	Current review branch abbreviation	Changed review interface branch to SPLB.

Item	Source	Description
18.	Potential Impact 21731.	Added a review interface with SRP Section 3.6.3 regarding application of leak-before-break to eliminate dynamic loads associated with pipe ruptures from the structural design basis. The EMCB was selected as the primary review branch based on comments received on ROC # 88 (SRP Section 3.6.2).
19.	Current review branch name and abbreviation	Changed review interface branch to Containment Systems and Severe Accident Branch (SCSB).
20.	Current PRB abbreviation	Changed PRB to ECGB.
21.	Current review branch abbreviation	Changed review interface branch to SCSB.
22.	Current review branch name and abbreviation	Changed review interface branch to Quality Assurance and Maintenance Branch (HQMB).
23.	Editorial	Corrected SRP citation from "Section 17.0 to "Chapter 17.
24.	SRP-UDP format item, Review Interfaces	Added a Review Interface to proposed SRP Section 19.2, "Severe Accident Containment Performance." (See PI 24154).
25.	Current PRB abbreviation	Changed PRB to ECGB.
26.	Editorial	Corrected to provide noun-verb agreement.
27.	Integrated Impact 540.	Added acceptance criterion to reflect the requirements of 10 CFR 50.34.
28.	Editorial	Changed "it" to "they" to provide number agreement.
29.	SRP-UDP format item	Replaced "item 2 of this subsection" with "Subsection II.2."
30.	Editorial	Changed verb form to agree with plural subject.
31.	Editorial	Changed verb form to agree with plural subject.
32.	Integrated Impact No. 540	Added PRB review responsibility to reflect the requirement of 10 CFR 50.34.
33.	Editorial	Changed "GDC" to "General Design Criteria" to accommodate plural usage.
34.	SRP-UDP format item	Added RG number to aid the reviewer.
35.	Integrated Impact No. 542	RGs 1.10, 1,15, 1.18, and 1.19 were withdrawn with the issuance of RG 1.136.
36.	Integrated Impact No. 539	Added new RG 1.35.1.
37.	Integrated Impact No. 542	RG 1.55 was withdrawn with the issuance of RG 1.136.

Item	Source	Description
38.	Integrated Impact No. 542	RG 1.103 was withdrawn with the issuance of RG 1.136.
39.	SRP-UDP format item	Deleted obsolete item and renumbered subsequent items.
40.	SRP-UDP format item	Changed to reflect current abbreviations used in the Code.
41.	Conversion to SI units	Converted 45 psig to 310 kPa.
42.	Integrated Impact No. 540	Added design criteria to reflect the provisions of 10 CFR 50.34(f)(3).
43.	Integrated Impact No. 540	Added paragraph to describe provisions of 10 CFR 50.34(f)(3) regarding design of containments against loads from postaccident inerting hydrogen control system.
44.	Integrated Impact No. 540	Added load combination equations to reflect the provisions of 10 CFR 50.34(f)(3).
45.	Editorial	Changed "percent" to "%" for consistency.
46.	Editorial	Modified because LOCA was previously defined in text.
47.	Editorial	Deleted extra "and."
48.	SRP-UDP format item	Changed to reflect SRP subsection designation.
49.	Editorial	Deleted "but" as unnecessary.
50.	Conversion to SI units	Converted 40 psi to 276 kPa.
51.	Conversion to SI units	Converted 60 psi to 414 kPa.
52.	Conversion to SI units	Expressed the principal tensile stress in SI units.
53.	Editorial	Added words to improve clarity.
54.	Integrated Impact No. 542	Deleted RG 1.103, which has been withdrawn.
55.	Editorial	Added words to improve clarity.
56.	Integrated Impact No. 542	Changed to reflect withdrawal of RGs 1.10, 1.15, 1.19, and 1.55.
57.	Integrated Impact No. 542	Changed to reflect applicability of RG 1.136.
58.	Integrated Impact No. 542	Changed to reflect withdrawal of RG 1.18.
59.	SRP-UDP format item	Deleted unnecessary reference callout.
60.	Integrated Impact No. 539	Changed to reflect applicability of RG 1.35.1.
61.	SRP-UDP format item	Deleted unnecessary reference callout.

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Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
62.	SRP-UDP format item: develop technical rationale	Added "Technical Rationale" to ACCEPTANCE CRITERIA and put in numbered paragraphs to describe the bases for referencing the GDCs and 10 CFR paragraphs.
63.	SRP-UDP format item: develop technical rationale	Added lead-in sentence for "Technical Rationale."
64.	SRP-UDP format item	Added the specific Regulatory Guide in place to the Regulatory Guide title.
65.	Editorial	Changed to provide noun-verb agreement.
66.	Editorial	Changed "assures" to "ensures."
67.	SRP-UDP format item	Changed "assures himself" to "ensures."
68.	Editorial	Changed "assure" to "ensure."
69.	Integrated Impact 543	Added information relating to the Staff's acceptance in the evolutionary FSERs an exemption to eliminate the OBE from seismic design requirements.
70.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
71.	Editorial	Changed "his" to "the" to eliminate gender reference.
72.	Editorial	Provided "SER" as initialism for "safety evaluation report."
73.	Integrated Impact 540	Modified EVALUATION FINDINGS to include the requirements of 10 CFR 50.34.
74.	SRP-UDP format item	Changed "assuring" to "ensuring."
75.	Integrated Impact No. and 540	Added evaluation finding pertaining to pressure resulting from hydrogen burning to reflect the requirements of 50.34.
76.	Integrated Impact No. 542	Changed to reflect withdrawal of RGs 1.10, 1.15, 1.18, and 1.19.
77.	Integrated Impact No. 539	Changed to reflect issuance of RG 1.35.1.
78.	Integrated Impact No. 542	Changed to reflect withdrawal of RG 1.55.
79.	Integrated Impact No. 542	Changed to reflect withdrawal of RG 1.103.
80.	SRP-UDP Format Item, Implement 10 CFR 52 Related Changes	To address design certification reviews a new paragraph was added to the end of the Evaluation Findings. This paragraph addresses design certification specific items including ITAAC, DAC, site interface requirements, and combined license action items.

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Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
81.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard sentence to address application of the SRP section to reviews of applications filed under 10 CFR Part 52, as well as Part 50.
82.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
83.	Integrated Impact No. 544	The citation of ASME Boiler and Pressure Vessel Code, Section III, Division 2, should be updated to cite the latest version of the standard.
84.	Integrated Impact No. 542	Deleted RGs 1.10, 1.15, 1.19, 1.55, and 1.18 from REFERENCES.
85.	Integrated Impact No. 542	Changed reference number.
86.	Integrated Impact 539	Added reference to reflect issuance of RG 1.35.1.
87.	SRP-UDP format item	Changed reference number.
88.	SRP-UDP format item	Changed reference number.
89.	Integrated Impact No. 542	Deleted RG 1.103 from REFERENCES.
90.	SRP-UDP format item	Changed reference number.
91.	SRP-UDP format item	Changed reference number.
92.	SRP-UDP format item	Changed reference number.
93.	SRP-UDP format item	Changed reference number.
94.	SRP-UDP format item	Changed reference number and title.
95.	SRP-UDP format item	Changed reference number.
96.	SRP-UDP format item	Changed reference number.
97.	Integrated Impact No. 540	Added reference to 10 CFR 50.34.
98.	SRP-UDP format item	Changed reference number.
99.	Current PRB name	Changed PRB to Civil Engineering and Geosciences Branch.

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SRP Draft Section 3.8.1

Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
539	Add RG 1.35.1 to Acceptance Criteria.	ACCEPTANCE CRITERIA, II.2 and II.7 EVALUATION FINDINGS, IV.7 REFERENCES, VI.4
540	Revise SRP Section 3.8.1 to comply with the 10 CFR 50.34(f)(3)(v) related to coping with hydrogen generated from 100% fuel clad metal-water reaction.	AREAS OF REVIEW, II.3 ACCEPTANCE CRITERIA, II.E and II.3 EVALUATION FINDINGS, V.6 REFERENCES, VI.14
541	Title of GDC 4 was changed	REFERENCES, VI.11
542	RGs 1.10, 1.15, 1.18, 1.19, 1.55, and 1.103 have been withdrawn with the issuance of RG 1.136.	ACCEPTANCE CRITERIA, II.2 and II.6 EVALUATION FINDINGS, IV.7 REFERENCES, VI
543	Reflects position in SECY 93-087 regarding decoupling of the OBE from the SSE for evolutionary plants.	REVIEW PROCEDURES
544	The ASME Section III, Division 2, "Code for Concrete Reactor Vessels and Containments," referenced in the SRP Section 3.8.1 is to be reviewed for acceptability by the staff.	This is a placeholder integrated impact and will not be processed further.
545	Revise SRP Section 3.8.1 to include a design check for the loads resulting from rapid releases of energy (high-pressure core melt ejection with direct containment heating, or hydrogen combustion), as well as gradually evolving releases, such as those from decay heat, and noncombustible gas generation.	No changes to SRP.
546	RG 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," a proposed addition to SRP Section 3.8.1, endorses the use of ANSI/ASTM C512-76, "Standard Test Method for Creep of Concrete in Compression," with exceptions. The current version of ANSI/ASTM C512 was issued in 1987.	No changes to SRP Section 3.8.1.

Integrated Impact No.	Issue	SRP Subsections Affected
47	RG 1.94 cites ACI 309-72 as the basis for determining the adequacy of the equipment for concrete consolidation and of the operation technique. The current version of ACI 309 was issued in 1987. In addition, RG 1.94 endorses ANSI N45.2.5-74. This document has been incorporated into ANSI/ASME NQA-2, and the current version was issued in 1992.	No changes to SRP Section 3.8.1.
719	RG 1.90 endorses ACI 201 (no date of issue is specified). The current version of ACI 201R was issued in 1992. In addition, RG 1.90 refers to RG 1.18 and ASTM C597-71. RG 1.18 has been withdrawn and the current version of ASTM C597 is cited as ASTMR-87.	No changes to SRP Section 3.8.1.
738	RG 1.35 endorses ACI 201 1968, ASTM D512, ASTM D 3867 (no date of issue specified for ASTM D512 and 3867). The current versions, are ACI 201R-92, ASTM D512-89 and ASTM D3867-90.	No changes to SRP Section 3.8.1.
739	RG 1.107 endorses ASTM C 150-74, ASTM C191-74, and ASTM D512-67. The current versions of these documents were issued in 1994, 1992, and 1989 respectively. In addition, RG 1.107 endorses ACI 212 (no date of issue) and references the American Public Health Association's "Standard Method for Examination of Water and Waste Water," dated 1971. Current versions of these documents were issued in 1986 and 1992, respectively. Further, Appendix B to RG 1.107 references the following ASTM standards: <u>Cited</u> — <u>Date of Current Version</u> C109-73 — 1992 C260-74 — 1989 C494-71 — 1992 D992-71 — 1990 D516-74 — 1990 D596-74 — 1983 D1129-74 — 1990 D1293-65 — 1984 R90	No changes to SRP Section 3.8.1.
740	RG 1.136 endorses ACI 18-77 and ACI 349-76. The current versions were issued in 1992 and 1990, respectively. In addition, RG 1.136 cites AISI 4140 steel (no further information regarding steel designation given), ASME/ACI 359-80, and ACI 308-71. The current versions of these documents were issued in 1992.	No changes to SRP Section 3.8.1.

SRP Draft Section 3.8.1

Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
1198	Revise the Acceptance Criteria and Review Procedures to incorporate the requirements from proposed rulemaking 59 FR 979.	This is a placeholder II and will not be processed further.
1241	Revise the SRP to incorporate the new and revised requirements from proposed rulemaking 59 FR 52255.	This is a placeholder II and will not be processed further.