U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

3.8.3 CONCRETE AND STEEL INTERNAL STRUCTURES OF STEEL OR CONCRETE CONTAINMENTS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for structural analysis reviews

Secondary - None

I. AREAS OF REVIEW

The specific areas of review are as follows:

1. Description of the Internal Structures

The descriptive information, including plans and sections of the various internal structures, is reviewed to establish that sufficient information is provided to define the primary structural aspects and elements relied upon to perform the safety-related functions of these structures. To perform safety-related functions, these structures must be capable of resisting loads and load combinations to which they may be subjected and

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should not become the initiator of a loss-of-coolant accident (LOCA). If such an accident were to occur, the structures should be able to mitigate its consequences by protecting the containment and other engineered safety features from the accident's effects such as jet forces and whipping pipes.

The major containment internal structures that are reviewed, together with the primary structural function of each structure and the extent of descriptive information required for each structure, are indicated below. This Standard Review Plan (SRP) section reviews the intervening structural elements between (distribution systems including their supports, e.g., cable trays; conduit; heating, ventilation, and air conditioning; and piping and equipment supports) building structural steel/concrete (e.g., steel platforms, building frame members, embedment plates, and building steel members beyond the jurisdictional boundary of supports to mechanical components).

A. Pressurized-Water Reactor (PWR) Dry Containment Internal Structures

i. Concrete Supports for Reactor

The PWR vessel should be supported and restrained to resist normal operating loads, seismic loads, and loads induced by postulated pipe rupture, including LOCAs. The support and restraint system should restrain the movement of the vessel to within allowable limits under the applicable loading combinations. However, the support system should minimize resistance to the thermal movements expected during operation.

With these functional requirements in mind, the review evaluates the general arrangement and principal features of the reactor vessel supports, with an emphasis on the methods of transferring loads from the vessel to the support and ultimately to the structure and its foundations.

ii. Concrete Supports for Steam Generator

Steam generators should be supported and restrained to resist normal operating loads, seismic loads, and loads induced by pipe rupture. The support system should prevent the rupture of the primary coolant pipes from a postulated rupture in steam or feedwater pipes and vice versa. However, the system should minimize resistance to the thermal movements expected during operation.

With these functional requirements in mind, the review evaluates the general arrangement and principal features of the steam generator supports, with an emphasis on the methods of transferring loads from the vessel to the support and eventually to the structure and its foundations.

iii. Primary Shield Wall and Reactor Cavity

The primary shield wall forms the reactor cavity and usually supports and restrains the reactor vessel. It is often a thick wall that surrounds the reactor vessel and may be anchored through the liner plate to the containment base slab.

The review evaluates the general arrangement and principal features of the wall and cavity, including the main reinforcement and anchorage system.

iv. Secondary Shield Walls

The secondary shield walls surrounding the primary loops form the steam generator compartments and protect the containment from the effects of pipe rupture accidents inside the compartment. These walls may also support intermediate floors and the operating floor. The review evaluates the general arrangement and principal features of these walls, with an emphasis on the method of structural framing and expected behavior under compartment pressure loads and jet forces, particularly those associated with a LOCA.

v. Other Interior Structures

The review also evaluates other major interior structures of PWR dry containments in a similar manner, including the concrete refueling pool walls, refueling water storage tank (if applicable), the operating floor, other intermediate floors and platforms, and the polar crane supporting elements.

B. PWR Ice-Condenser Containment Internal Structures

The following elements, in addition to the applicable structures reviewed in dry PWR containments, are reviewed for PWR plants using an ice-condenser containment system:

i. The Divider Barrier

In the PWR ice-condenser containment system, which uses the pressure-suppression concept, the divider barrier surrounds the reactor coolant system. The upper portion of the divider barrier is nearly surrounded by the ice-condenser, which is bounded by the containment shell on the outside and by the divider barrier wall on the inside. Several venting doors connect the space inside the divider barrier to the ice-condenser.

In the event of a LOCA, the divider barrier will contain the steam released from the reactor coolant system and, while temporarily acting as a pressure-retaining envelope, will channel the steam through the venting doors and into the ice-condenser. The ice will condense the steam, thus minimizing the energy released to the containment.

Following such a LOCA and before blowdown is complete, the divider barrier will be subjected to differential pressure and possibly jet forces. Any structural failure in its boundary may result in steam bypassing the ice-condenser and flowing directly into the containment, possibly generating a containment pressure higher than that for which it has been designed.

With this functional requirement in mind, the review evaluates the general arrangement and principal features of the divider barrier with an emphasis on structural framing and expected behavior when subjected to the design loads .

ii. Ice-Condenser

A major feature of the ice-condenser containment are the baskets of ice forming the heat sink essential for pressure suppression. The structurally significant components of the ice-condenser reviewed are the vent doors, ice baskets, brackets, couplings and lattice framings, lower and upper supports, and insulating and cooling panels.

The review evaluates the general arrangement and principal features of these major components, with an emphasis on the structural framing, supports, and expected behavior when subjected to design loads.

C. Boling-Water Reactor (BWR) Containment Internal Structures

This SRP section is oriented toward the BWR Mark III containment concept. Other BWR containment types are reviewed in a similar manner.

The following major BWR containment internal structures are reviewed, together with the primary structural function of each structure and the extent of descriptive information required for each structure:

i. Drywell

In the BWR containment system, which uses the pressure-suppression concept, the drywell surrounds the reactor coolant system. The lower portion of the drywell is surrounded by the suppression pool which is bounded by the containment shell on the outside and by a weir wall located just inside the drywell wall. A series of vent holes connects the drywell to the suppression pool. In the event of a LOCA, the drywell will contain the steam released from the reactor coolant system and, while temporarily acting as a pressure-retaining envelope, will channel the steam through the vent holes and into the suppression pool. The pool water will condense the steam, thus minimizing the energy released to the containment.

Following such a LOCA and before blowdown is complete, the drywell will be subjected to a differential pressure and possibly to jet forces. Any structural failure in its boundary would result in steam bypassing the suppression pool and flowing directly into the containment, possibly generating a containment pressure higher than that for which it was designed.

With this functional requirement in mind, the review evaluates the general arrangement and principal features of the drywell, with an emphasis on structural framing and expected behavior under loads. Because the drywell geometrically resembles, to a certain degree, a containment, the descriptive information reviewed is similar to that reviewed for containments in Subsection I.1 of SRP Section 3.8.1. The major components of the drywell reviewed, other than the main body of the drywell, include the bottom vent region, the roof and drywell head, and major penetrations.

ii. Weir Wall

The weir wall forms the inner boundary of the suppression pool and is located inside the drywell. It completely surrounds the lower portion of the reactor coolant system. The review evaluates the general arrangement and principal features of the weir wall, with an emphasis on structural framing and behavior under loads.

iii. Refueling Pool and Operating Floor

The refueling pool walls are located on top of the drywell. The outer walls form a rectangular pool that is usually subdivided by two interior crosswalls. The base slab of the pool is common to the drywell roof slab. The pool may be filled continuously with water for shielding purposes during operation.

The review evaluates the general arrangement and principal features of the refueling pool, with an emphasis on structural framing and behavior under loads.

The operating floor is intended to provide laydown space for refueling operations and is usually a combination of reinforced concrete and structural steel framing. The containment walls and the refueling pool walls may support the floor.

The review evaluates the general arrangement and principal features of the operating floor.

iv. Concrete Supports for Reactor and Recirculation Pump

The support systems of the BWR vessel and recirculation pumps have the same functions as the support systems for PWR vessels and pumps and thus are similarly reviewed.

v. Reactor Pedestal

The reactor pedestal is usually a cylindrical structure located below and supporting the reactor vessel, which is anchored to the top of the pedestal. The review evaluates the general arrangement and principal features of the reactor pedestal, with an emphasis on structural framing, main reinforcement, and the manner in which the pedestal is anchored to the containment base slab.

vi. Reactor Shield Wall

The reactor shield wall is usually a cylindrical wall surrounding the reactor vessel for radiation shielding purposes. It is supported on the reactor pedestal. The wall may be lined on both surfaces with steel plates which also may act as the main structural components of the wall. In addition, the wall may be used as an anchor for pipe restraints.

The review evaluates the general arrangement and principal features of the reactor shield wall, with particular emphasis on structure framing and behavior under loads.

vii. Other Interior Structures

The review evaluates the other major interior structures constructed of reinforced concrete or structural steel, or combinations thereof, including the floors located inside the drywell and in the annulus between the drywell and the containment and the polar crane supporting elements. The general arrangement and principal features of these structures are reviewed.

Some recent applications (e.g., AP600 and AP1000) also use modular construction methods for the major containment internal structures. Wall modules are typically constructed from large, prefabricated sections of steel plates spaced apart with intermittent steel members, joined with other modules at the site, and then filled with concrete. The concrete fill used in wall modules could be structural concrete with reinforcement (composite construction) or fill concrete of low strength without reinforcement, or heavy concrete for radiation shielding. Floor modules consist of prefabricated steel members and plates and are combined with poured concrete to create a composite section. In view of the new application of modules to nuclear power plants, the structural module design, fabrication, configuration, layout, and connections will be reviewed on a case-by-case basis.

2. Applicable Codes, Standards, and Specifications

The review evaluates the information pertaining to design codes, standards, specifications, and Regulatory Guides (RGs), as well as industry standards, that is applied in the design, fabrication, construction, testing, and surveillance of the containment structures. The specific editions, dates, or addenda identified for each document are also reviewed.

3. Loads and Loading Combinations

The review evaluates the information pertaining to the applicable design loads and associated load combinations. The loads normally applicable to containment internal structures include the following:

A. Loads encountered during construction of containment internal structures, 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 and vertical construction loads, loads that are induced by the proposed construction sequence and by the differential settlements of the soil under and to the sides of the containment building, erection and fitting forces, equipment reactions, and form pressure.

B. Loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads from operating temperature, and hydrostatic loads such as those in refueling and pressure suppression pools. In addition, hydrodynamic loads resulting from actuation of safety relief valves (SRVs) and manifested as drag load, jet impingement, and/or pressure loads should be considered. The appendix to SRP Section 3.8.1 includes a further description of the loads associated with SRVs.

C. Loads to be sustained during severe environmental conditions, including those induced by the operating-basis earthquake (OBE) specified for the plant site. Subsection II.3.A of this SRP defines the condition for which the OBE load is required for design of containment internal structures.

D. Loads to be sustained during extreme environmental conditions, including those induced by the safe-shutdown earthquake (SSE) specified for the plant site.

E. Loads to be sustained during abnormal plant conditions. The design-basis LOCA is the most critical abnormal plant condition during which most of the containment internal structures have to perform their primary function. Ruptures of other high-energy pipes should also be considered. Time-dependent and dynamic loads induced by such accidents include elevated temperatures and differential pressures across compartments, jet impingement, impact forces associated with the postulated ruptures of piping, and loads applicable to some structures, such as drag forces in the PWR ice-condenser containment. In addition, for structures or structural components located in or above the suppression pools of BWR containments, the review should consider the applicable LOCA-related and SRV-related hydrodynamic loads manifested as jet loads and/or pressure loads. The appendix to SRP Section 3.8.1 further describes the loads associated with LOCAs.

The various combinations of the above loads that are normally postulated and reviewed include construction 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, and normal operating loads with extreme environmental and abnormal loads.

4. Design and Analysis Procedures

The review evaluates the design and analysis procedures used for the containment internal structures, with an emphasis on the extent of compliance with the applicable codes as indicated in Subsection II.2 of this SRP. The review includes the design and analysis procedures applicable to the following areas:

A. PWR Dry Containment Internal Structures

i. Concrete Supports for Reactor Coolant System

The support system for the reactor vessel and steam generators, as described in Subsection I of this SRP section, should be designed to resist various combinations of loadings, including normal operating loads, seismic loads, and LOCA and other pipe rupture accident loads.

SRP Section 3.7.2 describes the analytical procedures for determining seismic loads.

After the procedures for determining individual loads and combinations thereof are reviewed, the design and analysis methods used for the supports are considered, including the type of analysis, the methods of load transfer, and the assumptions of boundary conditions.

ii. Primary Shield Wall and Reactor Cavity

The primary shield wall should withstand all applicable loads, including those transmitted through the reactor supports. The wall is subjected to most of the loads described in Subsection I.3 of this SRP section and should be designed and analyzed for all applicable load combinations. During normal plant operation, the attenuation heat of gamma and neutron radiation originating from the reactor core generates a thermal gradient across the wall. Insulation and cooling systems may be provided to reduce the severity of this gradient by limiting the rise in temperature to an acceptable level.

Procedures for determining seismic loads on the primary shield wall are reviewed in accordance with SRP Section 3.7.2.

LOCA loads applicable to the primary shield wall include a different pressure created across the reactor cavity by a pipe break in the vicinity of the reactor nozzles. Such a transient pressure may act on the entire cavity or on portions of the cavity. Procedures for determining such pressures are reviewed in accordance with guidance provided in SRP Section 6.2.1.

Other LOCA loads that apply are those transmitted to the wall through the reactor supports, including pipe rupture reaction forces which may induce simultaneous shear forces, torsional moments, and bending moments at the base of the wall. The elevated temperature within and around the primary shield created by the accident may also produce transient thermal gradients across the thick wall. The review evaluates these design and analysis procedures accordingly.

iii. Secondary Shield Walls

The secondary shield walls surrounding the primary loops and supporting the operating floor should be designed for loads similar to those applicable to the primary shield wall, including loads of fluid jets from a postulated break of a primary pipe which can impinge on them. The analytical techniques used for these walls are reviewed, including their structural framing and behavior under loads. When elasto-plastic behavior is assumed and the ductility of the walls is relied upon to absorb the energy associated with jet loads, the review evaluates the procedures and assumptions, with particular emphasis on such areas as modeling techniques, boundary conditions, force-time functions, and assumed ductility. For the time-dependent differential pressure, however, elastic behavior is required, and the review considers the methods of determining an equivalent static load accordingly.

iv. Other Interior Structures

Many of the other interior structures reviewed are combinations of slabs, walls, beam, and columns classified as Category I structures. These structures are subjected to most of the loads and load combinations described in Subsection I.3 of this SRP section. The review evaluates the analytical techniques for these structures on the same basis as the review of the structures described above.

v.

B. PWR Ice-Condenser Containment Internal Structures

i. Divider Barrier

Because the divider barrier has to maintain a certain degree of leaktightness during a LOCA and is thus a critical structure with respect to the proper functioning of the containment, it is treated on the same basis as the containment.

The loads that usually govern the design of the divider barrier are those induced by the LOCA, including the time-dependent differential pressure across the barrier and any concurrent concentrated jet impingement loads. Because the divider barrier is typically a combination of walls and slabs framed together, the design and analysis procedures are conventional. The review evaluates them accordingly, with an emphasis on the assumed boundary conditions and behavior under loads. Since the differential pressure and jet impingement loadings are dynamic impulsive loads that vary with time, the review considers the techniques used to determine their equivalent static loads.

C. Ice-Condenser

The design of the ice-condenser and its various components may be based on a combination of analysis and testing. The review includes the analytical and testing procedures for the ice baskets and brackets (couplings), the lattice frames and columns, including attachments; the supporting structures comprising the lower supports; the wall panels; and cooling duct and supports of various auxiliary components.

The ice-condenser and its components should be analyzed or tested for various loads and combinations thereof, including dead and live loads, thermal loads induced by differential thermal expansion within the various elements, seismic loads, and loads induced by a LOCA. Accident loads include pressure differential drag loads and loads induced by the change of momentum of the flowing steam.

Elastic analysis is usually used for the ice-condenser and its components. However, plastic analysis may also be used as an alternate approach. Accordingly, the review evaluates the load factors that are applied to each of the applicable loads and their basis and justification.

When experimental verification of the design using simulated load conditions is employed, the review evaluates the procedures used to account for similitude relationships which exist between the actual component and the test model to ensure that the results obtained from the test are a conservative representation of the load-carrying capability of the actual component under the postulated loading.BWR Containment Internal Structures

This SRP section is oriented toward the BWR Mark III containment concept. Other BWR containment types are reviewed in a similar manner.

i. Drywell

The drywell, which has to maintain a certain degree of leak tightness during a LOCA, is critical to the proper functioning of the containment. Because it geometrically resembles a containment, the design and analysis procedures used for the drywell are reviewed on a basis similar to that used to review containments as described in Subsection I.4 of SRP Sections 3.8.1 and 3.8.2 for concrete and steel portions, respectively.

ii. Weir Wall

One of the major loads to which the weir wall may be subjected is a jet impingement load induced by a pipe rupture in a nearby recirculation loop. Under such a concentrated load, the weir wall should not deform to an extent that it might impair or degrade the pressure-suppression performance. Accordingly, the review evaluates the procedures used to analyze the wall for such dynamic time-dependent loads with particular emphasis on modeling techniques, assumptions on boundary conditions, and behavior under loads.

iii. Refueling Pool and Operating Floor

The refueling pool is assumed to be continuously filled with water to provide biological shielding above the reactor. The operating floor, which may be supported on the walls of the refueling pool on one side and on the containment shell on the other side, is a combination of reinforced concrete and structural steel. The design and analysis procedures for the refueling pool and the operating floor are conventional and are reviewed accordingly, with particular emphasis on the structural framing and behavior under loads. In cases in which the floor beams are supported vertically on the containment shell, they should be laterally isolated to minimize interaction between the containment and its interior.

iv. Concrete Supports for Reactor and Recirculation Pump

The design and analysis procedures used for the reactor and recirculation pump supports are reviewed in a manner similar to that used for the PWR reactor and reactor coolant system supports, as already described in this SRP section.

v. Reactor Pedestal

The reactor pedestal supports the reactor and must withstand the loads transmitted through the reactor supports. It is thus subjected to most of the loads described in Subsection I.3 of this SRP section and is designed and analyzed for all applicable load combinations. Because of the similarity in geometry and function of the BWR reactor pedestal to the PWR primary shield wall, their design and analysis procedures are similar. Hence, the review evaluates the reactor pedestal using the method previously discussed in this SRP section.

vi. Reactor Shield Wall

This cylindrical wall, which surrounds the reactor and provides biological shielding, is also subjected to most of the loads described in Subsection I.3 of this SRP section. In most cases, the wall is used to anchor pipe restraints in the vicinity of the reactor nozzles. A pipe rupture in this area may pressurize the space within the wall. The wall is usually lined on both faces with steel plates which may constitute the major structural elements relied upon to resist the design loads.

The review evaluates the analytical and design techniques used to determine the effect of the design loads on the wall with particular emphasis on the assumed boundary conditions and the behavior of the wall under loads.

vii. Other Interior Structures

Several platforms exist within the BWR containment, some of which are inside the drywell while others are outside the annulus between the drywell and the containment. Platforms inside the drywell are usually constructed of structural steel, and their main structural function is to provide foundations for the pipe restraints inside the drywell. Platforms outside the drywell are usually constructed of a combination of steel and concrete and must be designed to resist the various applicable loads, particularly the effects of pool swell during a LOCA and/or SRV actuation. The analytical procedures for determining pool swell loads are reviewed in accordance with guidance provided in SRP Section 6.2.1. The review evaluates the design and analysis procedures for these platforms, with particular emphasis on the framing and structural behavior under loads.

D. Design Reports

The applicant's design report, as described in Appendix C to SRP Section 3.8.4, is reviewed.

E. Structural Audit

A structural audit, as described in SRP Section 3.8.4, Appendix B, is conducted.

5. Structural Acceptance Criteria

The review evaluates the design limits imposed on the various parameters that quantify the structural behavior of the various interior structures of the containment, particularly with respect to stresses, strains, deformations, and factors of safety against structural failure, with emphasis on the extent of compliance with the applicable codes indicated in Subsection II.5 of this SRP.

6. Materials, Quality Control, and Special Construction Techniques

The review evaluates the information provided on the materials that are used in the construction of the containment internal structures. Concrete ingredients, reinforcing bars and splices, structural steel, and various supports and anchors are among the major materials of construction reviewed.

The review evaluates the quality control program proposed for the fabrication and construction of the containment internal structures, including nondestructive examination of the materials to determine physical properties, placement of concrete, and erection tolerances.

Special, new, or unique construction techniques, such as the use of modular construction methods, if proposed, are reviewed on a case-by-case basis to determine their effects on the structural integrity of the completed containment internal structures. In addition, the following information should be provided:

A. The extent to which the materials and quality control programs comply with American Concrete Institute (ACI) 349, as supplemented by additional guidance provided by RGs 1.142 and 1.199 for concrete and anchors (steel embedments) respectively, and American National Standards Institute/American Institute of Steel Construction (ANSI/AISC) N690-1994 including Supplement 2 (2004,) for steel, as applicable.

B. If welding of reinforcing bars is proposed, describe the extent to which the applicant complies with the applicable sections of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereafter, referred to as Code) Section III, Division 2, Subsection CC, as supplemented with additional guidance provided by RG 1.136. Provide justification for any exceptions.

7. Testing and Inservice Surveillance Programs

For Category I structures inside containment, the review evaluates information on structures monitoring and maintenance requirements.

For containment internal structures, it is important to accommodate inservice inspection of critical areas. The review includes any special design provisions (e.g., sufficient physical access, 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 containment internal structures.

Postconstruction testing and inservice surveillance programs for containment internal structures, such as pressure testing of the drywell/wetwell in a BWR containment and periodic examination of inaccessible areas, are reviewed on a case-by-case basis.

The structural integrity test for the drywell of the BWR containment is reviewed in a similar manner to that used to review the containment.

8. Inspections, Tests, Analyses, and Acceptance Criteria

For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed Inspections, Tests, Analyses, and Acceptance Criteria (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 that are subject to quality assurance programs in accordance with the requirements of Title 10 of the *Code of Federal Regulations* (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 included in the load combination equations of this SRP section.

3. The exclusion of postulated pipe ruptures from the design basis is generally referred to as the “leak before break.” The review of those applications that propose to eliminate

consideration of design loads associated with the dynamic effects of pipe rupture is performed in accordance with SRP Section 3.6.3.

4. Determination of loads generated from pressure under accident conditions is performed in accordance with SRP Section 6.2.1. The loads thus generated are included in the load combinations in this SRP section.

5. Distribution systems including their supports (e.g., cable trays; conduit; heating, ventilation, and air conditioning) and equipment supports are reviewed in accordance with SRP Sections 3.9.2 and 3.9.3.The organization responsible for quality assurance performs the reviews of design, construction, and operation phase quality assurance programs under SRP Chapter 17. In addition, while conducting regulatory audits in accordance with Office Instruction NRR-LIC-111 or NRO-REG-108, “Regulatory Audits,” the technical staff may identify quality-related issues. If this occurs, then the technical staff should contact the organization responsible for quality assurance to determine if an inspection should be conducted.

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

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. 10 CFR 50.55a and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1, as they relate to the design, fabrication, erection, and testing of containment internal structures in accordance with quality standards commensurate with the importance of the safety function to be performed.

2. GDC 2, as it relates to the ability of the containment internal structures without loss of capability to perform their safety function, to withstand the effects of natural phenomena, such as earthquakes, tornadoes, floods, and the appropriate combination of all loads.

3. GDC 4, as it relates to the protection of containment internal structures 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 5, as it relates to safety-related structures not being shared among nuclear power units, unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions.

5. GDC 50, as it relates to the design of containment internal structures 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 52.47(b)(1), which requires that a DC application contain the proposed ITAACs that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act (AEA), and the U.S. Nuclear Regulatory Commission's (NRC) regulations. 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 AEA, and the NRC regulations.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC regulations identified above are as follows for review described in Subsection I of 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 Internal Structures. The descriptive information in the Safety Analysis Report (SAR) is considered acceptable if it meets the criteria set forth in Section 3.8.3.1 of RG 1.70 or 1.206. During the application acceptance review, the reviewer identifies deficient areas of descriptive information and initiates a request for additional information. New or unique design features that are not specifically covered in RG 1.70 or 1.206 may require a more detailed review. The reviewer determines whether additional information is required to accomplish a meaningful review of the structural aspects of such new or unique features.
RG 1.206 provides the basis for evaluating the description of structures to be included in a DC or a COL application.

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

2. Applicable Codes, Standards, and Specifications. The design, materials, fabrication, erection, inspection, testing, and inservice surveillance, if any, of containment internal structures are covered by codes, standards, and guides that are applicable either in their entirety or in part. The following codes and guides are acceptable:

Code, Standard, or Specification	Title
ACI 349	Code Requirements for Nuclear Safety-Related Concrete Structures (supplemented with additional guidance by RGs 1.142 and 1.199)
ASME Code	Section III, Division 2, Subsection CC, "Code for Concrete Reactor Vessels and Containments"
ASME Code	Section III, Division 1, Subsection NE, "Class MC Components"
ANSI/AISC N690-1994 including Supplement 2 (2004)	Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities

Regulatory Guides

1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment
1.69	Concrete Radiation Shields for Nuclear Power Plants
1.136	Materials, Construction, and Testing of Concrete Containments
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)
1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in LWR Plants
1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants
1.199	Anchoring Components and Structural Supports in Concrete

3. Loads and Load Combinations The loads and load combinations for containment internal structures described in Subsection I.1 of this SRP are acceptable if they are consistent with the guidance given below. The loads and load combinations for the divider-barrier and ice-condenser elements of the ice-condenser PWR containment and

the drywell of the BWR containment are presented following the general criteria given for concrete and steel structures.

A. Concrete Structures

All loads and load combinations are to be in accordance with ACI 349 and RG 1.142. Supplemental criteria on the use of loads and load combinations are presented below.

Dead loads include hydrostatic loads, and, for equipment supports, they include static and dynamic head and fluid flow effects.

Live loads include any movable equipment loads and other loads that vary with intensity and occurrence. For equipment supports, they also include loads caused by vibration and any support movement effects. Alternate load cases in which the magnitudes and locations of the live loads are arranged so that worst-case conditions are included in the design should be investigated, as appropriate.

As per 10 CFR Part 50, Appendix S, the OBE is only associated with plant shutdown and inspection unless the applicant specifically selects it 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 internal structures remain functional and are within applicable stress, strain, and deformation limits. SRP Sections 3.7.1 and 3.7.2 provide further guidance on the use of OBE.

For structures or structural components subjected to hydrodynamic loads resulting from LOCA and/or SRV actuation, such loads should be considered as indicated in the appendix to SRP Section 3.8.1. Fluid structure interaction associated with these hydrodynamic loads and those from earthquakes should be taken into account.

The design of concrete structures must consider the loads and load combinations that may occur during their construction. These loads consist of dead loads, live loads, temperature, wind, snow, rain, and ice. Applicable construction loads include material loads, personnel and equipment loads, horizontal construction loads, erection and fitting forces, equipment reactions, and form pressure. Structural Engineering Institute (SEI)/ASCE Standard 37 provides additional guidance on construction loads. This standard may be used for supplemental guidance. When the standard and the Code/SRP provide conflicting criteria, the criteria provided in Code/SRP governs.

B. Steel Structures

All loads and load combinations are to be in accordance with ANSI/AISC N690-1994 including Supplement 2 (2004). This specification uses the allowable stress design (ASD) method. Use of the load and resistance factor design (LRFD) version of the specification (N690L) is reviewed on a case-by-case basis. The supplemental criteria on the use of loads and load combinations presented above for concrete structures also apply to steel structures.

C. Divider Barrier and Ice-Condenser of the PWR Ice-Condenser Containment

Specific load and load combination criteria applicable to the divider barrier and ice-condenser elements are given below. Supplemental criteria presented in Subsection II.3.A of this SRP section are also applicable.

i. Divider Barrier

Because the structural integrity of the divider barrier and, to a certain extent, its leak tight integrity are important to the proper functioning of the ice-condenser containment system, it is treated, for design purposes, in a manner similar to the containment itself. Accordingly, for concrete pressure-resisting portions of the divider barrier, the loads and load combinations of Article CC-3000 of the ASME Code, Section III, Division 2, with additional guidance provided by applicable portions of SRP Section 3.8.1 and RG 1.136.

For other concrete portions of the divider barrier, the loads and load combinations as defined in Subsection II.3.A apply.

Steel portions of the divider barrier that resist the design differential pressure and are not backed by concrete, such as penetrations, hatches, locks, and guard pipes, should be designed in accordance with the appropriate sections of Subsection NE of the ASME Code, Section III, Division 1, with additional criteria provided by applicable portions of SRP Section 3.8.2 and RG 1.57 apply.

For other steel portions of the divider barrier, the loads and load combinations as defined in Subsection II.3.B apply.

ii. Ice-Condenser Elements.

The structural integrity of the ice baskets, ice-bed framing, and their supports is important to the functional integrity of the ice-condenser containment system. Loads and load combinations for the ice-condenser elements are acceptable if found to be in accordance with ANSI/AISC N690-1994 including Supplement 2 (2004). For the ice-condenser, the load P_a is the LOCA pressure load induced by drag and change in the momentum of flowing air and steam.

D. BWR Containment Drywell

This SRP section is oriented toward the BWR Mark III containment concept.

Other BWR containment types are reviewed in a similar manner.

Because the structural integrity of the drywell and, to a certain extent, its leak tight integrity are critically important to the proper functioning of the pressure-suppression system, the drywell is treated, for design and testing purposes only, in a manner similar to the containment itself. Accordingly, for the concrete pressure-resisting portions of the drywell, the loads and loading combinations of Article CC-3000 of ASME Code, Section III, Division 2, will apply, with additional criteria provided by applicable portions of SRP Section 3.8.1 and RG 1.136. For steel components of the drywell that resist pressure and are not backed by concrete, the appropriate sections of Subsection NE of ASME Code, Section III, Division 1, should be used with additional guidance provided by applicable portions of SRP Section 3.8.2 and RG 1.57. Specifically, the loads and load combinations of Subsection II.3 of SRP Section 3.8.2 apply.

Additional criteria presented in Subsection II.3.A of this SRP section are also applicable to the BWR containment drywell.

For the lower vent portion of the drywell, the following conditions apply:

i. If the main reinforcement of the drywell is carried down between the vent holes, and the reinforced concrete section is relied upon for structural purposes, the criteria that apply to concrete portions of the drywell as described above will apply.

ii. If the main reinforcement of the drywell is terminated above the vent holes, and two steel plates lining both faces of the drywell are used for structural purposes, the criteria that apply to steel portions of the drywell as described above will apply.

iii. If other structural systems are used in the vent region, the loads and load combinations are reviewed and judged on a case-by-case basis.

4. Design and Analysis Procedures

The design and analysis procedures used for the containment internal structures are acceptable if found to be in accordance with the following:

A. PWR Dry Containment Internal Structures

i. Primary Shield Wall and Reactor Cavity

The design and analysis procedures used for the shield wall are acceptable if found to be in accordance with ACI 349 with additional guidance provided by RG 1.142. This code is based on the strength design method. The design and analysis of anchors (steel embedments) used for component and structural supports on concrete structures are acceptable if found to be in accordance with ACI 349, Appendix B, with additional guidance provided by RG 1.199.

Analyses for LOCA loads applicable to the primary shield wall, such as the cavity differential pressure combined with pipe rupture reaction forces, are acceptable if these loads are treated as dynamic time-dependent loads. This requires that either a detailed time-history analysis be performed or a static analysis using the peak of the forcing function amplified by an appropriate chosen dynamic factor be employed. Elastic behavior of the wall should be maintained under the differential pressure. However, for the concentrated accident loads, such as Y_r, Y_j, or Y_m, elasto-plastic behavior may be assumed if the deflections are limited to maintain functional requirements. Simplified methods for determining effective dynamic load factors for elastic behavior are acceptable if found to be in accordance with recognized dynamic analysis methods.

ii. Secondary Shield Walls

Design and analysis procedures used for the secondary shield walls are acceptable if found to be in accordance with conventional beam/slab design and analysis procedures described in ACI 349, with additional guidance provided RG 1.142. The design and analysis of anchors (steel embedments) used for component and structural supports on concrete structures are acceptable if found to be in accordance with ACI 349, Appendix B, with additional guidance provided by RG 1.199.

Similar to the primary shield wall, the secondary shield walls are also subject to dynamic LOCA loads and the methods described in Subsection II.4.A.i are, therefore, applicable and acceptable.

iii. Other Interior Structures

Most of the other interior structures that are reviewed are combinations of reinforced concrete slabs, walls, beams, and columns, and steel beams and columns, which are classified as Category I structures subject to the loads and load combinations described in Subsection II.3 of this SRP section.

Analytical techniques for these structures are acceptable if found to be in accordance with those described in ACI 349, and with additional guidance provided by RG 1.142 and 1.199 for concrete and anchors (steel embedments,) respectively, and with ANSI/AISC N690-1994 including Supplement 2 (2004) for steel.

B. PWR Ice-Condenser Containment Internal Structures

i. Divider Barrier

The most important loads that usually govern the design of the divider barrier are those induced by a LOCA, including the differential pressure across the barrier and any concentrated jet impingement loads. Because the structural integrity of the divider barrier and, to a certain extent, its leak tight integrity are important to the proper functioning of the ice-condenser containment system, it is treated, for design purposes, in a manner similar to the containment itself. Accordingly, for concrete pressure-resisting portions of the divider barrier, the design and analysis procedures of Article CC-3000 of the ASME Code, Section III, Division 2, apply with additional guidance provided by applicable portions of SRP Section 3.8.1 and RG 1.136. For the other concrete portions of the divider barrier, the design and analysis procedures are acceptable if found to be in accordance with ACI 349, with additional guidance provided by RGs 1.142 and 1.199.

These methods are based on linear elastic design methods unless the structure is subjected to concentrated accident loads, as discussed in Subsection II.4.A.i, in which elasto-plastic behavior may be assumed.

For steel portions of the divider barrier that resist pressure but are not backed by structural concrete, the design and analysis procedures are acceptable if found to be in accordance with the applicable provisions of Subsection NE of the ASME Code, Section III, Division 1 apply, with additional guidance provided by applicable portions of SRP Section 3.8.2 and RG 1.57.

ii. Ice-Condenser Elements

The design and analysis procedures for the ice-condenser and its various components are acceptable if found to be in accordance with either the elastic/linear design method of Part 1 of ANSI/AISC N690-1994 including Supplement 2 (2004), or the plastic design method of Part 2 of the same specifications. For components using experimental testing to verify the design, the testing procedures are acceptable if found to be in accordance with recognized prototype or model testing procedures that consider the effect of scaling and similitude.

D. BWR Containment Internal Structures

This SRP section is oriented toward the BWR Mark III containment concept. Other BWR containment types are reviewed in a similar manner.

i. Drywell

The design and analysis procedures used for concrete portions of the drywell are acceptable if found to be in accordance with Subsection II.4 of SRP Section 3.8.1. For steel portions of the drywell that resist pressure but are not backed by structural concrete, the design and analysis procedures are acceptable if found to be in accordance with the applicable provisions of SRP Section 3.8.2, Subsection II.4.

ii. Weir Wall

One of the major loads to which the weir wall may be subjected is a jet impingement load induced by a pipe rupture in a nearby recirculation loop. The deflection of the wall under such a load must be limited so as not to impair the pressure-suppression performance. The procedures used to analyze the wall for such a dynamic time-dependent load are acceptable if a detailed time-history dynamic analysis is performed or if an equivalent static analysis is performed using the peak of the jet load amplified by an appropriately chosen dynamic load factor. The design and analysis procedures for concrete weir walls are acceptable if found to be in accordance with conventional methods described in ACI 349, with additional guidance provided by RGs 1.142 and 1.199, for concrete and anchors (steel embedments,) respectively.

iii. Refueling Pool and Operating Floor

The refueling pool and the operating floor, which may be supported on the walls of the refueling pool on one side and on the containment shell on the other side, are constructed of a combination of reinforced concrete and structural steel. The design and analysis procedures are acceptable if found to be in accordance with conventional methods described in ACI 349, with additional guidance provided by RGs 1.142 and 1.199, for concrete and anchors (steel embedments,) respectively, and in ANSI/AISC N690-1994 including Supplement 2 (2004) for structural steel.

iv. Supports for Reactor

The support system for the reactor vessel, described in Subsection I of this SRP section, should be designed to resist various combinations of loadings as indicated in Subsection II.3 of this SRP section. Among the major loads that should be considered are normal operating loads, seismic loads, and LOCA loads.

The design and analysis procedures used for the reactor supports (beyond the jurisdictional boundary of the ASME-designed supports) are acceptable if found to be in accordance with the same criteria for concrete and steel that apply to the refueling pool and operating floor.

v. Reactor Pedestal

The reactor pedestal, which supports the reactor and must withstand the loads transmitted through the reactor supports, should be subjected to most of the loads described in Subsection II.3 of this SRP section and should be designed for all applicable load combinations.

The design and analysis procedures used for the reactor pedestal are acceptable if found to be in accordance with the same criteria for concrete applicable to the refueling pool and operating floor.

vi. Reactor Shield Wall

This cylindrical wall, which surrounds the reactor and provides biological shielding, should be subjected to most of the loads described in Subsection II.3 of this SRP section. In many cases, the wall is used to anchor most of the pipe restraints placed around the reactor coolant system piping. A pipe rupture in the vicinity of the reactor nozzles may pressurize the space within the wall. The wall may be lined on both faces with steel plates which may constitute the major structural elements relied upon to resist the design loads. Like the reactor pedestal, the biological shield wall is also subjected to dynamic LOCA loads and the same methods are, therefore, applicable and acceptable.

The design and analysis procedures used for the reactor shield wall are acceptable if found to be in accordance with the same criteria for concrete that apply to the refueling pool and operating floor. If the shield wall is constructed from steel plates filled with unreinforced concrete, then the design and analysis procedures are reviewed on a case-by-case basis.

vii. Miscellaneous Platforms

Platforms inside the drywell are usually constructed of structural steel and their main structural function is to provide foundations for the pipe restraints inside the drywell. Platforms outside the drywell are usually combinations of steel and concrete. The design and analysis procedures used for miscellaneous platforms are acceptable if found to be in accordance with the same criteria for concrete and steel that apply to the refueling pool and operating floor. Of particular interest are the dynamic loads induced on these floors by pool swell during a LOCA.

E. For all containment internal structures, the design and analysis methods described in Subsections II.4 of SRP Sections 3.8.1 and 3.8.2, which are applicable to the containment internal concrete and steel structures, respectively, also need to be considered. These items include assumptions on boundary conditions, axisymmetric and nonaxisymmetric loads, transient and localized loads, shrinkage and cracking of concrete, computer programs, and evaluation of liner plates and anchors.

F. Design of structures that use modular construction methods are reviewed on a case-by-case basis. NUREG/CR-6486 provides guidance related to the use of modular construction methods. Appendix B to NUREG/CR-6486 includes proposed modular construction review criteria.

G. A structural design audit is conducted as described in Appendix B to SRP Section 3.8.4.

H. The applicant's design report is considered acceptable if it satisfies the guidelines of Appendix C to SRP Section 3.8.4.

5. Structural Acceptance Criteria

The structural acceptance criteria for containment internal structures described in Subsection I.1 of this SRP section are acceptable if found to be in accordance with the guidance given below. The acceptance criteria for the divider-barrier and ice-condenser elements of the ice-condenser PWR containment and the drywell of the BWR containment are presented following the criteria given for concrete and steel structures. The structural acceptance criteria for structures that use modular construction methods are reviewed on a case-by-case basis. See Section II.4.E of this SRP section for criteria relating to modular construction.

A. Concrete Structures

ACI 349 and RG 1.142 define the structural acceptance criteria for concrete structures. The structural acceptance criteria for anchors (steel embedments) used for support of systems and components to concrete structures are acceptable if found to be in accordance with Appendix B to ACI 349, with additional guidance provided by RG 1.199.

B. Steel Structures

ANSI/AISC N690-1994 including Supplement 2 (2004) defines the structural acceptance criteria for steel structures. This specification uses the ASD method. Use of the LRFD version of the specification (N690L) is reviewed on a case-by-case basis. Divider Barrier and Ice-Condenser of PWR Ice-Condenser Containment

i. Divider Barrier

For concrete pressure-resisting portions of the divider barrier, the specified limits for stresses and strains are acceptable if found to be in accordance with Subsection CC-3400 of ASME Code Section III, Division 2, with additional guidance provided by applicable portions of SRP Section 3.8.1 and RG 1.136. For steel portions of the divider barrier that resist pressure but are not backed by structural concrete, the design should be similar to that of steel containments. Accordingly the stress limits are acceptable if found to be in accordance with Subsection NE of the ASME Code, Section III, Division 1, with additional guidance provided by applicable portions of SRP Section 3.8.2 and RG 1.57.

For the other concrete and steel portions of the divider barrier, the specified limits for stresses and strains are acceptable if found to be in accordance with those provided in Subsections II.5.A and B for concrete and steel, respectively.

ii. Ice-Condenser Elements

For load combination delineated in Subsection II.3 of this SRP section, the specified limits for stresses and strain are acceptable if found to be in accordance with those given in ANSI/AISC N690-1994 including Supplement 2 (2004).

D. BWR Containment Drywell

This SRP section is oriented toward the BWR Mark III containment concept. Other BWR containment types are reviewed in a similar manner.

For concrete and steel portions of the drywell, the specified limits for stresses and strain are acceptable if found to be in accordance with the acceptance criteria of item II.5.C.i as described for the divider barrier.

For the lower vent portion of the drywell, the following conditions apply:

i. If the main reinforcement of the drywell is carried down between the vent holes, and the reinforced concrete section is relied upon for structural purposes, the structural acceptance criteria are the same as for item II.5.C.i above for concrete.

ii. If the main reinforcement of the drywell is terminated above the vent holes, and two steel plates lining both faces of the wall are used for structural purposes, the acceptance criteria are reviewed on a case-by-case basis.

iii. If other structural systems are used in the vent region, the acceptance criteria are also reviewed on a case-by-case basis.

6. <u>Materials, Quality Control, and Special Construction Techniques</u>

The specified materials of construction and quality control programs are acceptable if found to be in accordance with the public code or standard as indicated in Subsection I.6 of this SRP section.

Special construction techniques, if any, are treated on a case-by-case basis. For modular construction, the materials, quality control, and special construction techniques are also reviewed on a case-by-case basis. See Section II.4.E of this SRP section for further information.

7. <u>Testing and Inservice Surveillance Requirements</u>

BWR containment drywells, such as those used for the Mark III containment, should be subjected to a structural proof test. Such a test is acceptable if found to be in accordance with the following:

A. The drywell should be subjected to an acceptance test that increases the drywell internal pressure in three or more approximately equal pressure increments ranging from atmospheric pressure to at least the design pressure. The drywell should be depressurized in the same number of increments. Measurements should be recorded at atmospheric pressure and at each pressure level of the pressurization and depressurization cycles. At each level, the pressure should be held constant for at least 1 hour before the deflections and strains are recorded.

B. So that the overall deflection pattern can be determined in prototype drywells, radial deflections should be measured at a minimum of three points along each of at least three meridians equally spaced around the drywell, including locations with varying stiffness characteristics. Radial deflections should be measured at the lower vent region, about mid-height, and near the top of the cylindrical design. Measurement points may be relocated, depending on the distribution of stresses and deformations anticipated in each particular design.

C. In prototype drywells only, strain measurements sufficient to permit an evaluation of strain distribution should be recorded for at least two opposing meridians at the following locations on the wall:

   i. At the bottom of the wall
   ii. At mid-height of the wall

   These strain measurements should be made at a minimum of three positions within the wall section - one at the center and one each near the inner and outer surfaces.

D. In nonprototype drywells, deflection and strain measurements need not be made if strain levels have been correlated with deflection measurements during the acceptance test of a prototype drywell when measured strains and deflections are within the predefined tolerance of their predicted responses. Any reliable system of displacement meters, optical devices, strain gauges, or other suitable apparatus may be used for the measurements.

E. If the test pressure drops as a result of unexpected conditions to or below the next lower pressure level, the entire test sequence should be repeated. Significant deviations from the previous test should be recorded and evaluated.

F. If any significant modifications or repairs are made to the drywell following, and because of, the initial test, the test should be repeated.

G. A description of the proposed acceptance test and instrumentation requirements should be included in the preliminary SAR.

H. The following information should be submitted before the performance of the test:

   i. The numerical values of the predicted responses of the structure which will be measured

   ii. The tolerances to be permitted on the predicted responses

   iii. The bases on which the predicted responses and the tolerances were established

J. The following information should be included in the final test report:

   i. A description of the actual test and instrumentation

   ii. A comparison of the test measurements with the allowable limits (predicted response plus tolerance) for deflections and strains

   iii. An evaluation of the accuracy of the measurements

   iv. An evaluation of any deviations (i.e., test results that exceed the allowable limits), the disposition of the deviations, and the need for corrective measures

   v. A discussion of the calculated safety margin provided by the structure as deduced from the test results

For Category I structures inside containment, structures monitoring and maintenance requirements are acceptable if found to be in accordance with 10 CFR 50.65 and RG 1.160.

It is important that Category I structures inside containment accommodate inservice inspection of critical areas. The staff considers that monitoring and maintaining the condition of the containment internal structures is essential for plant safety. Any special design provisions (e.g., providing sufficient physical access, providing alternative means for identifying conditions in inaccessible areas that can lead to degradation, remote visual monitoring of high radiation areas) to accommodate inservice inspection of containment internal structures are reviewed on a case-by-case basis.

Technical Rationale

The technical rationale for application of these requirements and/or acceptance criteria to the areas of review addressed by this SRP section is discussed in the following paragraphs

1. Compliance with 10 CFR 50.55a requires that SSCs be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

   SRP Section 3.8.3 cites RGs 1.57, 1.69, 1.136, 1.142, 1.143, 1.160, and 1.199 for guidance regarding construction, quality control, tests, and inspections that are acceptable. ACI 349, with additional guidance provided RGs 1.142 and 1.199; ASME Code, Section III, Division 1, Subsection NE, and ASME Code, Section III, Division 2, Subsection CC, with additional guidance provided by RGs 1.57 and 1.136, respectively; and ANSI/AISC N690-1994 including Supplement 2 (2004) contain criteria for concrete and steel structures.

   Meeting these requirements and criteria provide assurance that the SSCs described herein will perform their safety function and limit the release of radioactive materials.

2. Compliance with GDC 1 requires that (1) SSCs important to safety be designed, fabricated, erected, and tested in accordance with quality standards commensurate with the importance of their safety function, (2) a quality assurance program be established and implemented, and (3) sufficient and appropriate records be maintained. When generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function.

   SRP Section 3.8.3 provides guidance related to static and dynamic loadings and evaluation criteria for containment internal structures. 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 3.8.3 cites ACI 349; ASME Code Section III, Division 1, Subsection NE, and ASME Code, Section III, Division 2, Subsection CC, with additional guidance provided by RGs 1.57 and 1.136, respectively; ANSI/AISC N690-1994 including Supplement 2 (2004); and regulatory guidance describing design methodology, materials testing, and construction techniques that are commensurate with the importance of the safety function to be performed. Conformance with these requirements imposes specific restrictions to ensure that containment internal structures will perform acceptably, commensurate with their intended safety function, when designed in accordance with the above standards. Meeting these requirements and criteria provide assurance that the SSCs described herein will perform their intended safety function.

3. 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 a loss of capability to perform their safety functions. The design bases for these SSCs shall reflect appropriate combinations of

the effects of normal and accident conditions with the effects of the natural phenomena.

To ensure that structures of a nuclear power plant are 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 shall be used to specify the design requirements of nuclear power plant components to be evaluated as part of CP, OL, COL, and early site permit (ESP) reviews, or for site parameter envelopes in the case of DCs, thereby ensuring that components important to safety will function in a manner that will maintain the plant in a safe condition.

SRP Section 3.8.3 provides detailed acceptance criteria and cites appropriate regulatory guidance for design methodology, materials testing, and construction techniques that are acceptable to the staff. GDC 2 requires that containment internal structures be designed to withstand the effects of natural phenomena, combined with those of normal and accident conditions, without a loss of capability to perform their safety function. Load combinations and specifications cited in this SRP section provide acceptable engineering criteria to accomplish that function.

Meeting these requirements and criteria provide assurance that safety-related structures inside containment will be designed to withstand the effects of natural phenomena and will perform their intended safety function.

5. 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. In addition, these SSCs must 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 3.8.3 provides methods acceptable to the staff, including load combinations, acceptance criteria, standards, and codes to assure compliance with GDC 4. Meeting these requirements and criteria provide assurance that the containment internal structures will withstand loads from internal events, such as those described above, and from external sources such as earthquakes, thus decreasing the probability that these events will damage containment internal structures.

   Meeting these requirements and criteria provide assurance that the internal structures of the containment will function as designed, be capable of maintaining their structural integrity, and perform their intended safety function. Compliance with GDC 5 prohibits the sharing of structures important to safety among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

   The requirements of GDC 5 are imposed to ensure that the use of common structures in multiunit plants will not significantly affect the orderly and safe shutdown and cooldown in one plant in the event of an accident in another. The load combination equations combine loads from normal operation and design-basis accidents so that the resulting structural designs provide for mutual independence of shared structures.

Meeting this requirement provides assurance that containment internal structures and their associated components are capable of performing their required safety functions, even if they are shared by multiple nuclear power units.

6. Compliance with GDC 50 requires that the reactor containment structure, including access openings, penetrations, and the containment heat removal system, be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

   SRP Section 3.8.3 provides detailed acceptance criteria and cites appropriate regulatory guidance for design methodology, material testing, and construction techniques that are acceptable to the staff. GDC 50 requires that the internal structures of the containment be designed to withstand the effects of natural phenomena, combined with those of normal and accident conditions, including LOCA loads, without a loss of capability to perform their safety function. The load combinations and specifications cited in this SRP section provide acceptable engineering criteria to accomplish that function.

   Meeting these requirements and criteria provide assurance that the internal structures of the containment will perform their intended safety function with sufficient margin when subjected to LOCA loads in combination with other applicable loads.

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

   SRP Section 3.8.3 provides guidance specifically related to design, construction, testing and inservice surveillance of structural concrete and steel of containment internal structures. Subsection II.2 of this SRP section cites ACI 349, with additional guidance provided by RGs 1.142 and 1.199; ASME Code, Section III, Division 1, Subsection NE, and ASME Code, Section III, Division 2, Subsection CC, with additional guidance provided by RGs 1.57 and 1.136, respectively; ANSI/AISC N690-1994 including Supplement 2 (2004); and other RGs to satisfy the requirements of 10 CFR Part 50, Appendix B.

   Meeting these requirements and criteria provide 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.

III. <u>REVIEW PROCEDURES</u>

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 Internal Structures. After the type of structure and its functional characteristics are identified, information on similar and previously licensed plants is obtained for reference. Such information, which is available in SARs and amendments of previous license applications, can identify differences in the case under review. These differences require additional justification and evaluation for meaningful review. New and unique features that have not been used in the past are examined in greater detail.

   The reviewer evaluates the information furnished in the SAR for completeness in accordance with RG 1.70 for a CP or an OL (for applications submitted in accordance with 10 CFR Part 50) or RG 1.206 for a DC or a COL (for application submitted in accordance with 10 CFR Part 52).

2. Applicable Codes, Standards, and Specifications. The list of codes, standards, guides, and specifications is checked against the list in Subsection II.2 of this SRP section. The reviewer verifies the use of the appropriate code or guide and the acceptability of the applicable edition and stated effective addenda.

3. Loads and Loading Combinations. The reviewer verifies that the loads and load combinations are consistent with those specified in Subsection II.3 of this SRP section. Any deviations from the acceptance criteria for loads and load combinations are reviewed for adequate justification.

4. Design and Analysis Procedures. The reviewer becomes familiar with the design and analysis procedures that are generally used for the type of structures being reviewed. Because the assumptions regarding the expected behavior of the structure and its various elements under loads may be significant, the reviewer evaluates their acceptability based on the acceptance criteria provided in Section II. The design and analysis procedures, including the behavior of the structures under various loads and the manner in which these loads are treated in conjunction with other coexistent loads are reviewed to establish compliance with procedures delineated in Subsection II.4 of this SRP section. These include the criteria for computer programs, consideration of concrete cracking, design reports, and the structural audit.

   As discussed in Subsection II.4.E of this SRP section, the use of modular construction methods is reviewed on a case-by-case basis using guidance provided in NUREG/CR-6486.

5. Structural Acceptance Criteria. The limits on allowable stresses and strains in the structural elements, including concrete, reinforcement, structural steel, and anchors, are compared with those specified in Subsection II.5 of this SRP section. The reviewer evaluates the justification provided to demonstrate that the functional

6. integrity of the structure will not be affected, when the applicant proposes to exceed some of these limits for certain load combinations and at certain localized points on the structure.

7. 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 specified in Subsection II.6 of this SRP section. For a new material that has not been used in prior license applications, the applicant is requested to provide sufficient test and user data to establish the acceptability of such a material. Similarly, the reviewer evaluates any new quality control programs or construction techniques to ensure that no degradation of structural quality that might affect the structural integrity of the structure will occur.

8. Testing and Inservice Surveillance Requirements. Procedures for the structural test of the BWR containment drywell are reviewed and compared with the procedures described in Subsection II.7 of this SRP section. Any other proposed testing and inservice surveillance programs are reviewed on a case-by-case basis.

   For containment internal structures, the reviewer verifies that structure monitoring and maintenance requirements are in accordance with 10 CFR 50.65 and RG 1.160.

   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 containment internal structures are reviewed on a case-by-case basis.

   Any other proposed testing and inservice surveillance programs are reviewed on a case-by-case basis.

9. Design Certification/Combined License Application Reviews. For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the Final Safety Analysis Report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document. The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

   For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an ESP or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

   For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

The staff concludes that the design of the containment internal structures is acceptable and meets the relevant requirements of 10 CFR 50.55a, 10 CFR Part 50, Appendix B, and GDCs 1, 2, 4, 5, and 50. This conclusion is based on the following:

1. The applicant has met the requirements of Section 50.55a and GDC 1 with respect to ensuring that the containment internal structures are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the safety function to be performed.

2. The applicant has met the requirements of GDC 2 by designing the containment internal structures to withstand the most severe earthquake that has been established for the site with sufficient margin, as well as 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 containment internal structures is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

4. The applicant has met the requirements of GDC 5 by demonstrating that SSCs are not shared between units or that sharing will not impair their ability to perform their intended safety functions.

5. The applicant has met the requirements of GDC 50 by designing the containment internal structures 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 the RGs and industry standards indicated below. The applicant has also performed appropriate analysis to demonstrate that the ultimate capacity of the structures will not be exceeded and to establish the minimum margin of safety for the design.

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

The criteria used in the design, analysis, and construction of the containment internal structures to account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime conform to established criteria, codes, standards, and specifications acceptable to the NRC staff. This includes meeting the positions of RGs 1.57, 1.69, 1.136, 1.142, 1.143, 1.160, and 1.199, as well as industry codes and standards ACI-349; ANSI/AISC N690-1994 including Supplement 2 (2004,) and, depending on the structure, ASME Code, Section III, Division 1, Subsection NE, or ASME Code, Section III, Division 2, Subsection CC, with additional guidance provided by RGs 1.57 and 1.136, respectively.

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

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this SRP section.

In addition, to the extent that the review is not discussed in other SER sections, the findings will summarize the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

V. IMPLEMENTATION

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

The provisions of this SRP section apply to reviews of applications submitted 6 months or more after the date of issuance of this SRP section, unless superseded by a later revision Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced RGs.

VI. REFERENCES

1. 10 CFR 50.55a, "Codes and Standards."

2. 10 CFR 50.65, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

3. 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records."

4. 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection Against Natural Phenomena."

5. 10 CFR Part 50, Appendix A, GDC 4, “Environmental and Dynamic Effects Design Basis.”

6. 10 CFR Part 50, Appendix A, GDC 5, “Sharing of Structures, Systems, and Components.”

7. 10 CFR Part 50, Appendix A, GDC 50, “Containment Design Bases.”

8. 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.”

9. 10 CFR Part 50, Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants.”

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

11. ACI 349, “Code Requirements for Nuclear Safety Related Concrete Structures,” American Concrete Institute.

12. ANSI/AISC N690-1994 including Supplement 2 (2004), “Specification for Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities.”

13. ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsection CC, “Code for Concrete Reactor Vessels and Containments,” ASME.

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

15. NUREG/CR-6486, “Assessment of Modular Construction for Safety-Related Structures at Advanced Nuclear Power Plants,” March 1997.

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

17. RG 1.69, “Concrete Radiation Shields for Nuclear Power Plants.”

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

19. RG 1.136, “Materials, Construction, and Testing of Concrete Containments.”

20. RG 1.142, “Safety-Related Concrete Structures for Nuclear Power Plants (other than Reactor Vessels and Containments).”

21. RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants.”

22. RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.”

23. RG 1.199, “Anchoring Components and Structural Supports in Concrete.”RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition).”

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

PAPERWORK REDUCTION ACT STATEMENT

SRP Section 3.8.3
“Concrete and Steel Internal Structures of Steel or Concrete Containments”
Description of Changes

Revision 3 to SRP Section 3.8.3 updates Revision 2 of this section, dated March 2007, to reflect the following changes:

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

SRP Section 3.8.3
"Concrete and Steel Internal Structures of Steel or Concrete Containments"

Description of Changes

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in Revision 3, dated May 2010of this SRP. See ADAMS Accession No. ML12353A377

The technical changes incorporated in Revision XXXXX, dated YYYY:

I. AREAS OF REVIEW

1. Enhanced SRP Section 3.8.3 I.3 "Loads and Load Combinations" item A, to include loads induced by the construction sequence and differential settlements. See item 2 in SRP Section 3.8.5, "Description of Changes, II Acceptance Criteria," for the technical rationale for this change.