

CODE FOR CONCRETE REACTOR VESSELS AND CONTAINMENTS

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14.1 INTRODUCTION

This Chapter describes the bases and provisions of the Code for Concrete Reactor Vessels and Containments. After a short description of the provisions for Concrete Reactor Vessels, the Chapter describes the concrete containment general environment, types of existing containments, future containment configurations, and background development including the regulatory bases of concrete containment construction code requirements. The description covers sequentially the following topics: Introduction, Concrete Reactor Vessels, Concrete Reactor Containments, Types of Containments, Future Containments, Regulatory Bases for the Code Development, Background Development of the Code, Reinforced Concrete Containment Behavior, Containment Design Analysis and Related Testing, Code Design Requirements, Fabrication and Construction, Construction Testing and Examination, Containment Structural Integrity Testing, Containment Overpressure Protection, Stamping and Reports, Containment Structure and Aircraft Impact, Containment and Severe Accident Considerations, Other Relevant Information, Summary and Conclusion.

The previous editions of this Chapter were developed by John D. Stevenson. The basic format of this chapter is kept the same as in the previous editions. The updates and additional information relating to the regulatory bases for the code requirements, future containments and considerations for future revisions of the Code included in this update are based on contributions from Hansraj Ashar, Barry Scott, and Joseph Artuso.

14.1.1 Concrete Reactor Vessels

Concrete reactor vessel construction requirements are contained in Subsection CB of Section III, Division 2 of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code. Fort St. Vrain is the only concrete reactor vessel built in the United States. The Subsection CB requirements were developed in parallel with the design and construction of Fort St. Vrain. The reactor is no longer in operation. Because there are no plans to construct concrete nuclear reactor

vessels in the foreseeable future, the current text of Subsection CB, though still being published in the periodic Code editions, is no longer being actively maintained by the ASME Boiler and Pressure Vessel Code committees.

14.1.2 Concrete Reactor Containments

Concrete containment construction requirements are contained in Subsection CC of Section III, Division 2 of the ASME Boiler and Pressure Vessel Code.

In general there are two types of barriers intended to resist release of radioactive material at nuclear facilities: confinements and containments. A confinement is a barrier intended to keep unpressured, usually solid radiological material from being released from its intended location and is not generally required to be leak-tight. In many instances it is operated at a slight negative pressure (i.e., $\frac{1}{2}$ in. water gauge: 0.018 psi, 122 Pa) to ensure that any confinement leakage during normal operation would occur only into the confinement space. Confinement construction is not addressed by the ASME Boiler and Pressure Vessel Code.

Containment as considered herein is often referred to as secondary containment and acts as a final barrier to radiological releases to the environment. The primary containment is typically considered fuel element cladding in a water-cooled and moderated nuclear reactor. Secondary containment acts as a barrier to the release of radioactive fluids under pressure and is designed to be leak tight under pressure. In general, large nuclear power, water-cooled reactors require containment vessels or structures surrounding a high pressure and temperature reactor coolant system. Confinements are often used as barriers to release of unpressurized or solid spent nuclear fuel and high-level waste storage and processing facilities, and also have been used as barriers for release of reactor coolant in gas-cooled nuclear power plants.

Containment construction¹ criteria, which include design loads, load combinations and acceptable behavior applicable to concrete nuclear containments, developed originally as a unique combination of mechanical and civil structural engineering procedures. They are composed of procedures considered in design and analysis of boiler

¹Typical design and layout characteristics for both BWR and PWR systems are shown in Table 14.1. Construction as defined by the ASME Code includes the prescription of administrative, documentation, material selection and material qualification, design analysis, fabrication, erection, examination, start-up testing, structural integrity testing, overpressure protection, and Code-stamping requirements.

TABLE 14.1 RANGE OF WATER REACTOR CONCRETE CONTAINMENT DESIGN AND LAYOUT PARAMETERS

Type	Design Pressure, psi	Percentage above Maximum Calculated Pressure	Enclosure Volume $\times 100 \text{ ft.}^3$	Height to Springline $\times 100 \text{ ft}$	Diameter $\times 100 \text{ ft}$	Base Material Thickness, ft.	Cylinder Wall Thickness, ft.	Dome Thickness, ft.
PWR I-S [Note (1)]	45–60	10 to 20	1.2–2.4	1.2–1.5	1.05–1.20	8–10	3.0	2.0
PWR I-L [Note (2)]	47–75	10 to 20	2.25–3.3	1.2–2.0	1.24–1.5	8–12	3.0–4.5	2.0–2.5
PWR II-S	45	10	3.0	1.5	1.3	—	—	—
PWR III-L [Note (3)]	12–15	10	1.2	1.2	1.15–1.2	8.0	4.5	2.0
BWR Mark I [Note (4)]	—	10 to 20	—	N/A	—	—	N/A	N/A
BWR Mark II	—	10 to 25	—	N/A	—	—	N/A	N/A
BWR Mark III	23	10 to 20	1.5	1.2	1.2	3.0	3.0	2.5

General Notes:

- PWR I-SI: small prestressed concrete containment; see Fig. 14.1.
- PWR I and II, S and I: small and large deformed-bar concrete containment; see Fig. 14.2.
- PWR I-L: large prestressed concrete containment; see Fig. 14.3.
- PWR III-L: large deformed bar concrete containment; see Fig. 14.4.
- PWR III-L: large hybrid steel and concrete containment; see Fig. 14.5.
- BWR Mark I: steel containment; see Fig. 14.6.
- BWR Mark I: deformed-bar concrete containment; see Fig. 14.7.
- BWR Mark II: deformed-bar concrete containment; see Fig. 14.8.
- BWR Mark III: deformed-bar concrete containment; see Fig. 14.9.
- BWR Mark III: hybrid concrete and steel containment; see Fig. 14.10.

Notes:

- The small PWR deformed-bar reinforced-concrete containment is similar in appearance to a large deformed-bar reinforced-concrete containment, except smaller in size.
- The large deformed-bar reinforced-concrete containment, except for the elimination of the tendon access gallery, is similar in appearance to the Fig. 14.3 PWR I-L reinforced, prestressed concrete containment; small = 600 MWe $\pm$ 25%; large = 1000 MWe $\pm$ 25%.
- Only two all-concrete ice condensers were constructed, D.C. Cook Units 1 and 2.
- Only one Mark I concrete containment was constructed, Brunswick Station.

and pressure vessel components developed by the ASME Boiler and Pressure Vessel Code Committee, and those procedures used in design of conventional concrete building structures as developed by the American Concrete Institute (ACI) 318 Code Committee. This situation naturally follows from an understanding that such containments perform a dual function: (1) to be a building structure used to house, and to protect from design-basis hazards, nuclear safety-related structures, mechanical and electrical components, and distribution systems associated with a reactor coolant system; and (2) to serve a primary function as an engineered safeguard to contain the postulated radiological consequences of a loss-of-coolant accident in the nuclear steam supply system.

As a result of such design basis accidents, the containment structure in the existing greater than 1,000 MWe light-water reactor nuclear power plants may use internal steam vapor design pressures as high as 4 atmospheres (60 psi, 420 KPa) and a design temperature of 350°F (177°C) for a short duration design basis, loss-of-coolant accident, and 650°F (377°C) for local hot spots with the concurrent occurrence of a 10^{-4} year or 10,000 year return period earthquake with design-basis mean peak ground surface accelerations that range from 0.1 to 0.75 g.

Normal long-term operation temperatures in the containment concrete are limited generally to 150°F (66°C) and in local areas (such as near penetrations) to 200°F (93°C).

Concrete containments in the United States are also required to resist the effects of tornadoes with maximum wind speeds ranging

from 240 to 360 mph concurrent with differential pressure drops and design basis tornado missiles.

Normally, the design requirements to contain pressure and temperature effects are more severe (except for earthquake-induced membrane shear and possibly membrane tension in a concrete wall segment) than those required to protect the safety-related components from extreme natural hazard effects, such as earthquake, tornado, and flooding, and human-induced design-basis loads such as blast, small airplane crash, and external missiles. Hence, concrete containment structures tend to follow current pressure vessel design more closely than they do building design practice.

This section is a brief description of the construction requirements, techniques, and procedures developed by the joint ACI 359 and ASME B&PV Code Section III, Division 2, Subsection CC committees for concrete containments. They bridge the gap between steel pressure component construction developed by the ASME and nuclear safety-related concrete building structures developed by the ACI. Inherent in this discussion is the noting of differences between working stress design (WSD) normally used by ASME and ultimate strength design (USD) used by ACI codes, as well as the use of load factors instead of allowable stresses to provide necessary design margins. Understanding these differences in philosophy between the two technical societies and the codes and standards they have developed is important to the understanding of the content and application of the Joint ACI-ASME Concrete Containment Code for construction of nuclear containments.

TABLE 14.1.1 CONCRETE CONTAINMENT DESIGN PARAMETERS

Type	Design Press. kPa	% Margin over Calc. P	Volume $10^3 \times \text{m}^3$	Diam m	Basemat Thickness m	Wall Thickness m	Dome Thickness m	Remarks
Pressurized Water Reactors								
PWR-Dry	310–433	10–20	51–93	35–46	2–3.5	1.0–1.4	0.6–0.8	Reinf & Prestr Fig. 14.1 Reinforced Fig. 14.2 PC-Containm RC-Outer Shield Fig. 14.6
PWR-Ice	80–84	10	34	36	2.2	1.4	0.6	
EPR-Planned	427	—	80	47	—	1.3	1	
Boiling Water Reactors								
BWR- Mk I	420	10–20	4.5 DW 7 Torus	Vary1	2–3 Below DW Conc. Base	1–2	Steel Head	Reinforced Fig. 14.3
BWR-MkII	380	10–25	7.1 DW 7.5–8 Wetwell	Vary2	2–3	1.2–2	Steel Head	RC & PC Fig. 14.4
BWR- Mk III	207-DW 103-WW	10–20	39.6 DW 11 WW	37	1	1	.8	Reinforced Fig. 14.5
ABWR-Planned	310	15	7.35 DW 5.96 WW	29	1.6	2	Steel Head	Reinforced Fig. 14.7
ESBWR-Planned	310	15	7.2 DW 9.9 WW	18 UDW 5.6 LDW	5.1	2	Steel Head	Reinforced Fig. 14.8
The parameters shown are approximate and meant to be for general information								

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Explanation: DW – Drywell, WW – Wetwell, Vary1 – Inverted Light Bulb, Vary2 – Truncated Conical Drywell & Cylindrical Wetwell, UDW – Upper Drywell includes Suppression Pools, LDW – Lower Drywell around Reactor Pedestal

14.1.2.1 Subsection CC requirements are too stringent for confinement structure design and construction. Reactor Buildings of Mark I, Mark II containments may be defined as low leakage confinement buildings. They are designed using the requirements of ACI 349. In the definition of “Defense in Depth,” fuel element cladding and RC loop (including RPV) are the primary barrier to the release of radioactive materials. However, they are not defined as containments. Hence the containment, as envisioned in Subsection CC, implies primary containments.

Table 14.1.1 has been added in this update to show parameters for the containments of standardized designs (proposed to be used in the future reactors), e.g. ABWR, ESBWR, and EPR.

14.1.3 Types of Containment Systems

In the United States there are two basic types of commercial nuclear power plants, both of which consist of water-cooled and moderated reactors. The Boiling Water Reactor (BWR) in the United States is a product of the General Electric Co. It typically operates at a design pressure of 1,100 psi and 600°F with the direct cycle of coolant from the reactor coolant system through the electricity-generating steam turbine. Pressurized Water Reactor (PWR) systems in the United States have been provided by the Combustion Engineering Co., the Babcock Wilcox Co., and the Westinghouse Electric Co. These types of reactor coolant systems have a subcooled design pressure of 2,500 psi and 650°F with an intermediate steam generator heat exchanger between the reactor coolant system (primary) and the steam system used to drive the electricity-generating turbine system (secondary).

Commercial nuclear power plant containments are designed to effectively contain without leakage the total primary coolant system inventory released into the containment volume. Because of the larger reactor coolant inventory at rated power in a BWR-type commercial power reactor system, the containment for BWRs employs a water-based pressure suppression system, which includes a dry well that surrounds the reactor coolant system. In the event of a loss-of-coolant accident, the resultant steam and air in the dry well blow down into a wet-well water pool where the steam contained in the reactor coolant system condenses, thereby significantly reducing the design pressure of the dry- and wet-well containment system.

Essentially two different sizes, 600 MWe and 1,000 MWe $\pm$ 25%, are used in both BWR and PWR plant design. Three distinct types of containment structures, Mark I, Mark II, and Mark III, are used for BWR pressure-suppression-type nuclear power plant containments. Three different types of containment structures are used for PWRs in the United States: dry, reduced pressure, and pressure-suppression ice containments.

The Mark I steel containment system was in general use between 1964 and 1972. The Mark I type of containment for a BWR-type reactor consists mostly of a steel dry-well (light bulb) and wet-well (torus) design as shown in Fig. 14.6 with a 4 in. gap between the steel light bulb and concrete shield structure. The single exception to the Mark I steel containment design was the deformed bar-reinforced concrete design used in the Brunswick Nuclear Power Station shown in Fig. 14.7. This containment type for a BWR reactor system was followed by the Mark II, which

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consists mostly of a concrete base mat, a concrete cylindrical section housing the wet well, and a concrete conical section housing the dry well, as shown in Fig. 14.8. This containment type design for BWRs was popular in the United States from 1972 to 1974. The most recent BWR containment structural design in the United States after 1974 is the Mark III containment, which consists of concrete deformed bar type as shown in Fig. 14.9 or a hybrid deformed bar-reinforced concrete-slab base mat and a deformed bar-reinforced concrete cylinder reinforcing a steel liner vapor barrier in the wet-well region and a steel cylinder and dome above the wet well region of the containment, as shown in Fig. 14.10.

It should be noted that there have been no new containment designs employed in the United States since 1976. Other countries, however, have constructed BWR designs since that time; in Japan, its advanced BWR designs have used a modified version of the Mark II type of containment.

Because of its relatively smaller reactor coolant system inventory, most PWR-type reactors have generally employed the so-called dry containment designed to simply contain the inventory of the reactor coolant system. However, pressure suppression systems of two types have been employed in some cases. The Stone Webster Corp. designed several containments with reduced internal air pressure typically in the range of 5–10 psig, which had the effect of significantly reducing the design pressure in the containment subjected to a design basis loss-of-coolant accident.

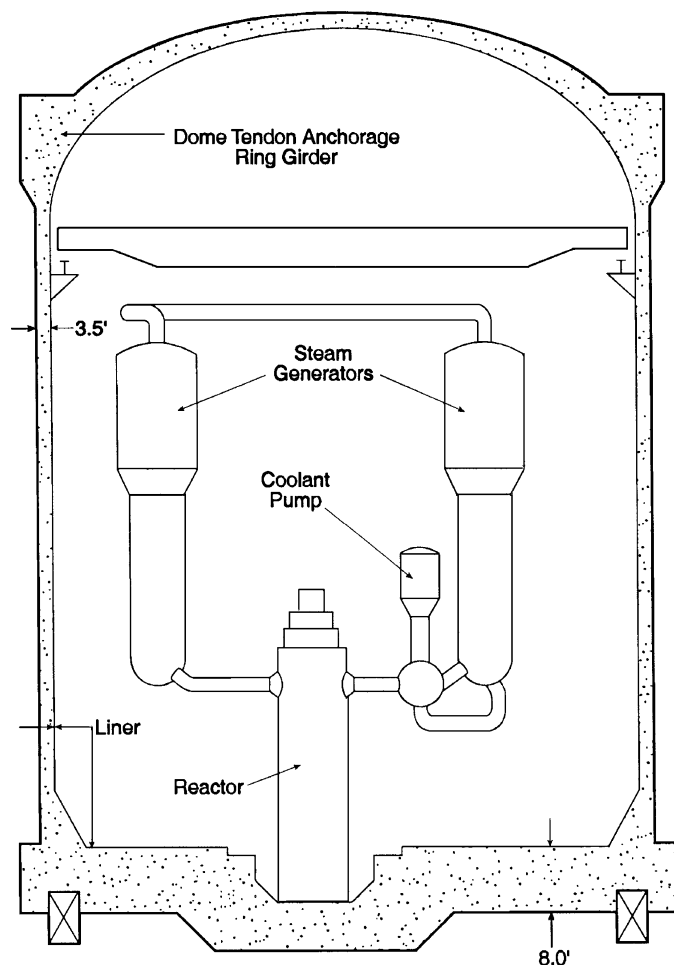


FIG. 14.1 SMALL PWR PRESTRESSED CONCRETE CONTAINMENT

Most dry and reduced-pressure PWR concrete containment volumes vary from 1.5 to 3.2 million ft³ depending on the volume of the reactor coolant system size of the unit and the containment design pressure. They consist of a reinforced and deformed-bar concrete base mat typically 10 ft thick; either a reinforced-concrete 4½ ft thick deformed bar or a 3½ ft thick prestressed cylinder; and a 2–2½ ft thick hemispherically and elliptically shaped dome as shown in Figs. 14.1, 14.2, and 14.3. Two different sizes of containments were used: one enclosing a 600 MWe ± 25% and 1,000 MWe ± 25%. Table 14.1 gives typical design parameters for containments built in the United States.

In 1968 Westinghouse Electric Co. introduced a pressure-suppression-type containment called the ice condenser. For the most part the ice condenser containments are a hybrid concrete and steel mixture having a reinforced-concrete, deformed-bar base mat with steel cylinders and hemispherical domes, as shown in Fig. 14.5. The single exception was the deformed-bar, reinforced-concrete ice condenser containment used for the D.C. Cook power plant, as shown in Fig. 14.4.

The containment design pressures applied to most concrete containments are usually statically applied as a function of the several seconds it takes pressure to build up from the postulated initial discharge of the reactor coolant system into containment. Exceptions to this static application of load for containment design is the area of the BWR wet wall that is exposed to pool swell, oscillation-condensation, jet impingement, and chugging phenomena associated with the dry-well blow down into the wet well, which are dynamic events. These localized dynamic loads in BWR Mark III-type containments have generally required the use of reinforced concrete in the region of the wet well rather than steel shells. The localized differential pressure loads across interior barriers of the ice condenser and ice component doors impact, as well as steam and air flow through the ice beds, are also dynamic in

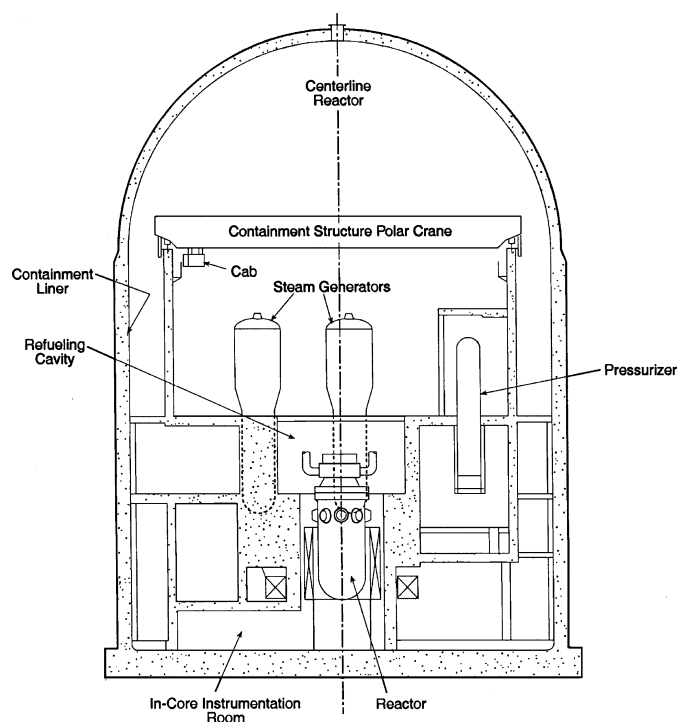


FIG. 14.2 LARGE OR SMALL PWR DEFORMED-BAR CONCRETE CONTAINMENT

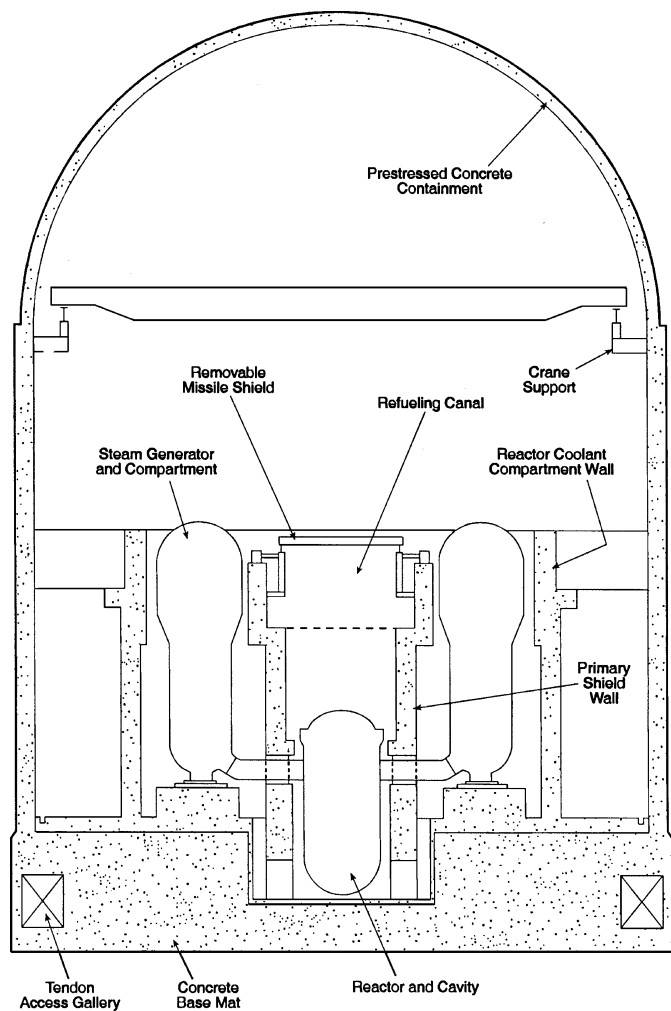


FIG. 14.3 LARGE PWR PRESTRESSED REINFORCED-CONCRETE CONTAINMENT

nature. In addition, most BWR and PWR containments, whether steel or reinforced concrete, are subject to dynamic jet impingement loads from postulated rupture of adjacent steam and feedwater lines. Concrete containment structures are subject to tornado missiles and, in very few cases, postulated design basis ruptured steam turbine and aircraft impact missiles. While not considered as a “design basis” loading required to meet ASME Code design criteria, containments have also been evaluated for the effects of postulated hydrogen deflagrations and detonations.

Internal concrete structures of the containment, which support the reactor system, are generally subject to differential pressure loads during the blowdown of the reactor coolant system. However, they are generally not required to remain leak-tight, so the ACI-349 concrete structures Code is used as their design basis. In the early containment designs, internal structures supporting or protecting the reactor coolant slab were supported on 2 or 3 ft thick deformed-bar, reinforced-concrete slabs locate above the containment base mat liner. In later containment designs, penetrations through the containment base liner were provided in order to anchor the internal structure and major reactor coolant system components directly to the containment base mat.

The dynamic analysis of containment structures for earthquake loads have progressed from a few two-dimensional lumped three or four mass stick models employing response spectrum modal

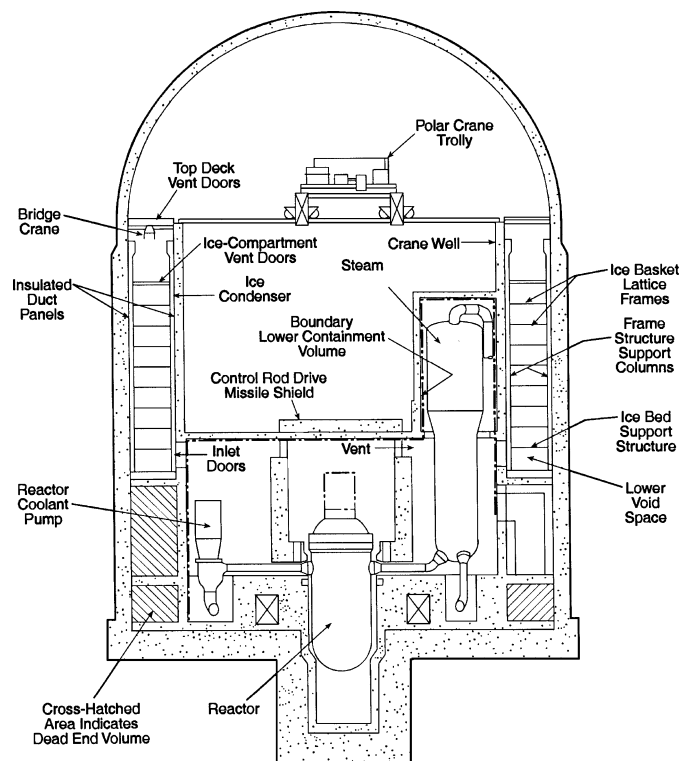


FIG. 14.4 PWR CONCRETE ICE CONTAINMENT

analysis (in the late 1960s) to complex three-dimensional hundreds to thousands of degrees of freedom finite element models (in the 1970s and 1980s). The dynamic modeling of containments has generally included Soil-Structure Interaction (SSI) effects

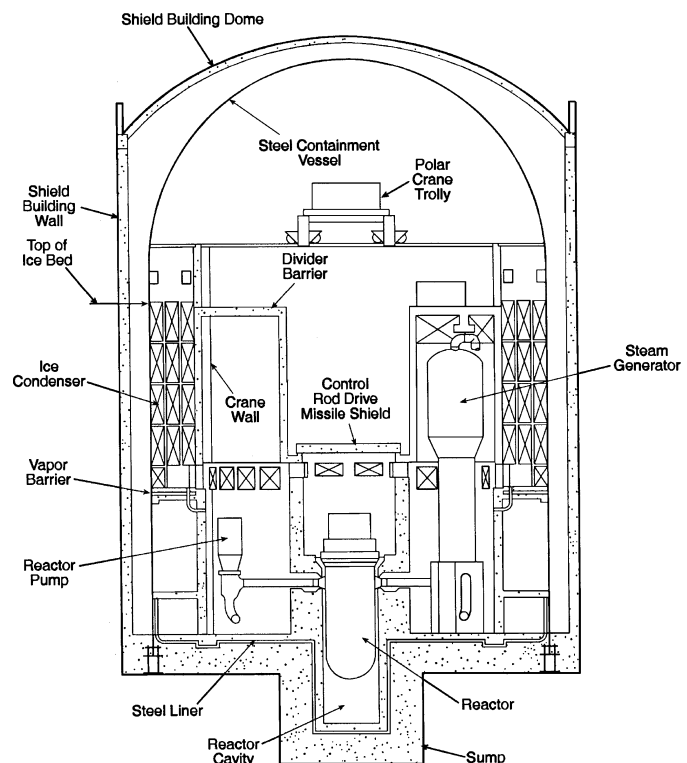


FIG. 14.5 PWR HYBRID ICE CONTAINMENT

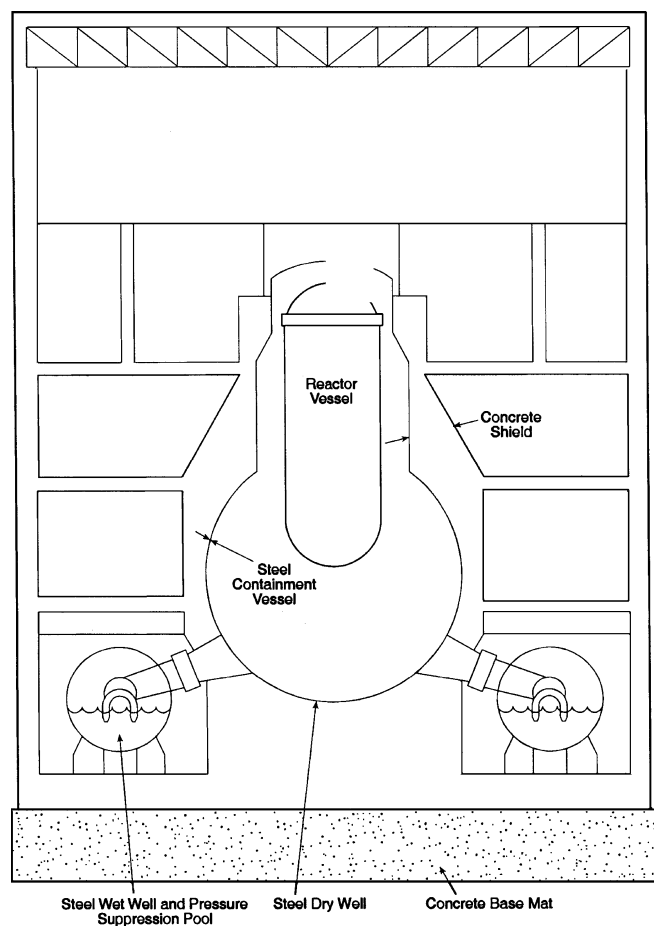


FIG. 14.6 BWR MARK I STEEL CONTAINMENT

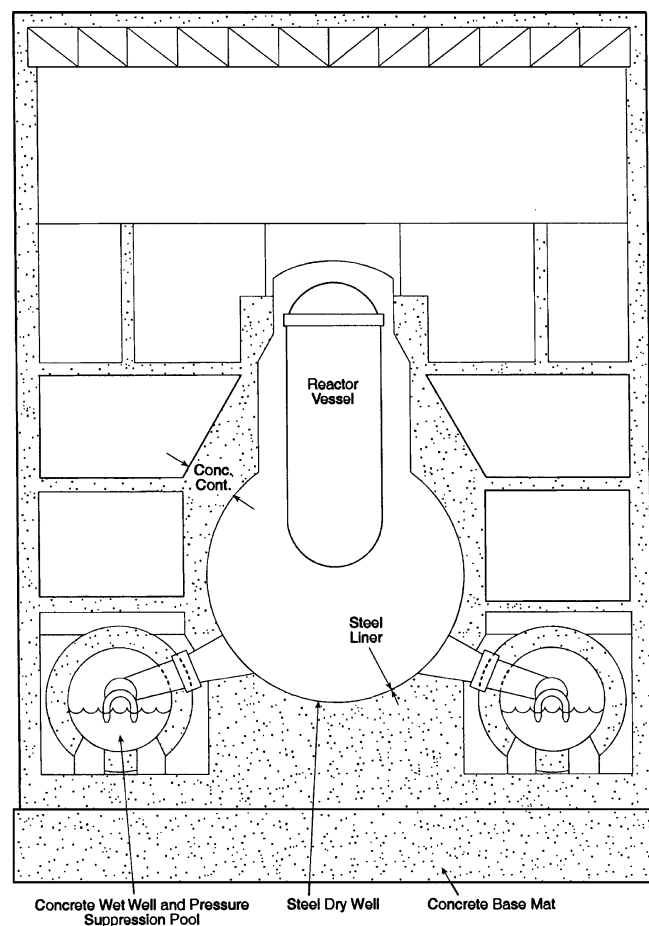


FIG. 14.7 BWR MARK I CONCRETE WET-WALL CONTAINMENT

when the shear wave velocity of the supporting foundation material is less than 1,100 m/s. This SSI modeling has progressed from lumped mass and stiffness elements and simple springs and dash pots to three-dimensional finite element representations of the soil capable of representing the nonlinear response of the foundation material as a function of strain levels.

14.1.3.1 Based on the analysis of Three Mile Island-2 accident in 1979, NRC promulgated two regulatory requirements in (1) 10 CFR 50.34, and (2) 10 CFR 50.44. As a result of these requirements, Regulatory Guide 1.136 (which endorses the Containment Code requirements) has been revised (March 2007) to include, among other provisions, guidance for incorporating the requirements of 10 CFR 50.34 and 10 CFR 50.44.

14.2 FUTURE CONTAINMENT DEVELOPMENT

Recent events have suggested that nuclear reactor containments in the future may be quite different from current designs and configurations. The first event of significance is associated with the potential use of large aircraft as a weapon against a nuclear reactor installation. The second consideration is the increased pressure to locate large nuclear power reactors near or in metropolitan areas.

Currently the ability of existing nuclear reactor containment structures to resist the effects of a large commercial aircraft crashing into it is somewhat in doubt. Currently containments were not designed to resist the effects of such a load, hence the margin to resist such loads are not known. In general it is expected that such margins are considerably less than those associated with extreme natural phenomena such as a 10,000-year return period earthquake or a Fujita Class 3 or 4 (200 mph) tornado. This suggests future containments will be of a shorter design as shown in Figure 4.11A.

It should also be noted that current nuclear containments are not designed to resist the worse possible accident—that of a reactor vessel rupture and core melt. Such a design would be required for any metropolitan siting. A conceptual design to meet both requirements has been defined since the mid 1970s, and parts of the concept have been incorporated into the Westinghouse current AP600 and AP1000 advanced reactor designs. The key elements of such a design would be to increase the structural load capacity of the containment structure. What is envisioned is a 8–10 ft. thick reinforced-concrete shell designed to resist 10 atmospheres of pressure rather than a 3.5 to 4.5 ft. thick cylinder wall designed to resist 4 atmospheres of pressure. Added to this would be a secondary steel containment shell and radiator 1.5 inches thick and an additional 2.5–3.5 ft. thick steel lined reinforced-concrete shield structure as shown in Figure 14.11. Unique features of the containment would be large melting chamfers below the normal

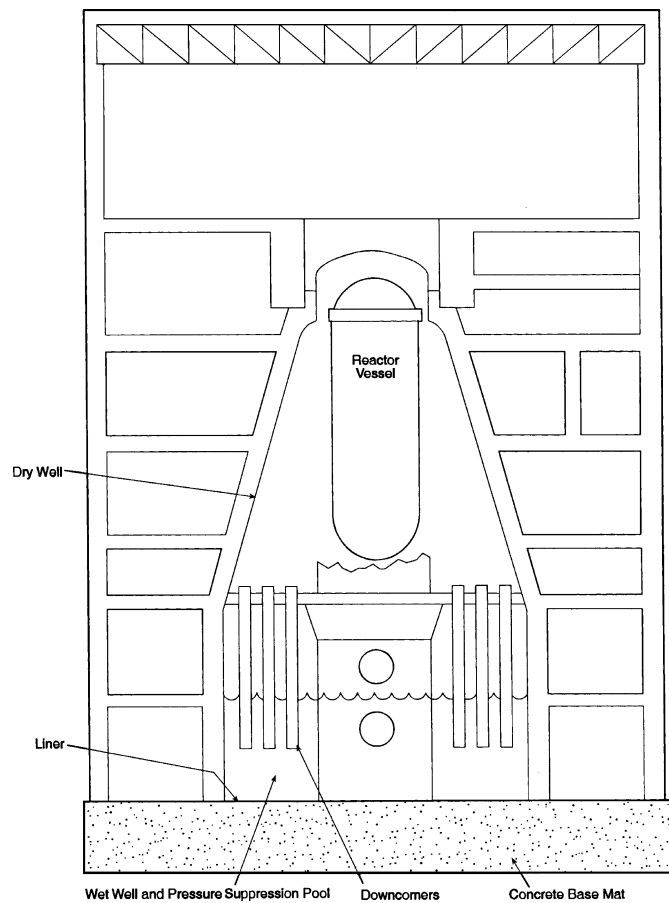


FIG. 14.8 BWR MARK II CONCRETE CONTAINMENT

floor of the reactor compartment to catch, absorb, and contain the residue from a reactor rupture and core melt. Another key feature of this containment system would be the passive nature of the cooling system. Steam could be released to the space between the inner concrete barrier and the steel shell and condensed by external cooling sprays on the steel shell, thereby controlling the condensable pressure buildup. In addition, the overall system would be at least three times stronger to resist airplane crash effects than current containment designs. Finally, the steel shell would provide a high temperature resistant barrier to any burning fuel. The steel shell can retain its structural integrity up to temperatures approaching 1000°F, when concrete begins to seriously degrade at temperatures below 500°F. In addition, the smooth steel shell would provide an excellent surface that promotes runoff of jet fuel and the ability to apply flame extinguishing spray.

14.2.1

For more information on acceptable new standardized reactor designs, refer to 10 CFR 52, Appendices A to D. New appendices will be added as new designs, such as ESBWR and EPR, when these get approved.

14.2.2 Concrete Containments of Future Reactors

Based on the preliminary new reactor licensing application received by the Nuclear Regulatory Commission in 2007, AP 1000 by Westinghouse LLC, ABWR and ESBWR by GE Energy, and EPR by Framatome ANP, Inc. are the likely reactors of the future. Among them, ABWR, ESBWR, and EPR designs will

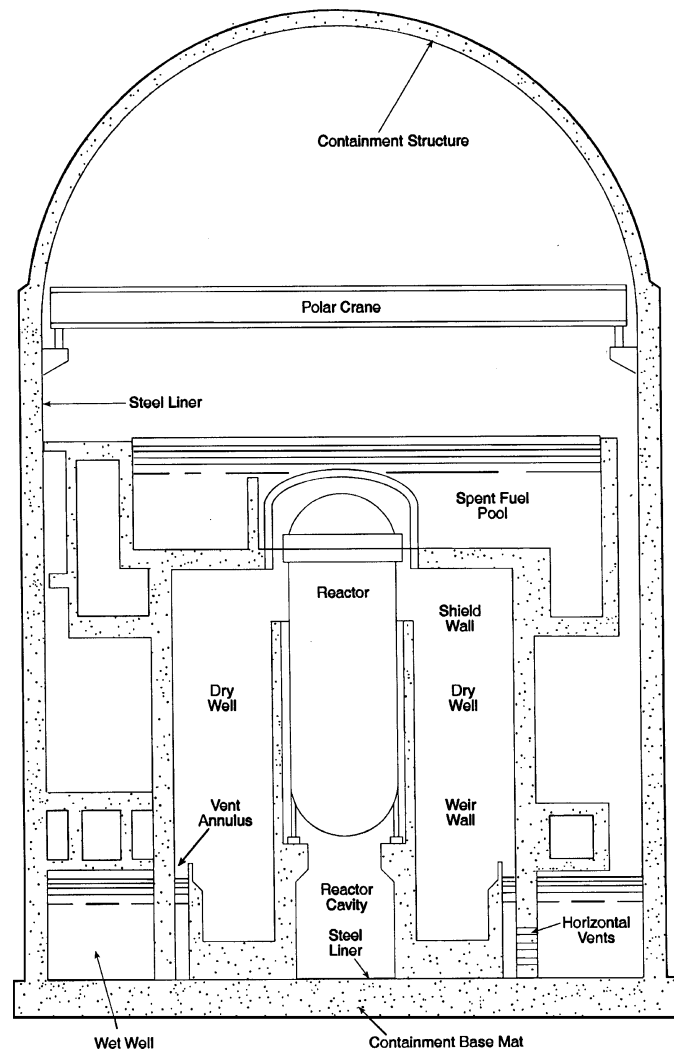


FIG. 14.9 BWR MARK III CONCRETE CONTAINMENT

have concrete containments. The following paragraphs summarize the containments of these reactors. The information and the Figures provided are for general information and should not be construed as design information.

14.2.2.1 Containment of Advanced Boiling Water Reactors (ABWR) The 29 meter (inside diameter) reinforced concrete steel lined containment, surrounds the NSSS components, consists of an Upper Drywell, a Lower Drywell, a Suppression Pool chamber (also called wetwell) and the vents connecting the drywells and the wetwell. From the top of the basemat to the top of the upper drywell, its height is 31.8 m. A steel drywell head caps the containment. The entire containment is surrounded by a reinforced concrete secondary containment.

14.2.2.2 Containment of Economic Simplified Boiling Water Reactor (ESBWR) The reinforced concrete steel lined containment surrounds the NSSS components, and consists of an Upper Drywell, Suppression Pool chamber (also called wetwell), and connecting vents. The inside diameter of the upper drywell is 36 m and it is 19.95 m high. The Lower Drywell has an inside diameter of 11.2 m and it is 15.05 m high. The entire containment is surrounded by a reinforced concrete reactor building envelop.

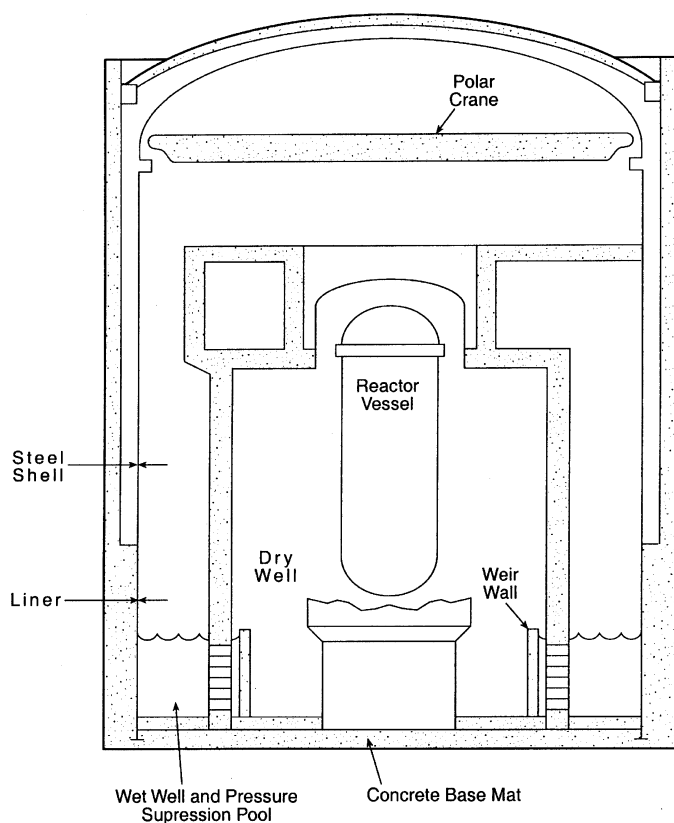


FIG. 14.10 BWR MARK III HYBRID CONTAINMENT

14.2.2.3 Containment of U. S. EPR The containment of the proposed pressurized water reactor consists of an inside prestressed concrete steel lined containment having an inside diameter of 46.8 m, and a surrounding reinforced concrete shield building, having an inside diameter of 53 m. Both the structures are supported on a common basemat.

14.3 BACKGROUND DEVELOPMENT OF CONCRETE CONTAINMENT CONSTRUCTION CODE REQUIREMENTS

14.3.1 Development of Original Concrete Containment Design Requirements

The first concrete containment used to house a large nuclear power reactor (>1,000 MW_e) was the Connecticut Yankee, Northeast Utilities deformed-bar, reinforced-concrete containment for a Westinghouse PWR reactor with a design started in 1962. The plant construction was completed in 1967 by the Stone and Webster Engineering Co. The design essentially used the working strength design provisions of the then-current ACI Standard 31863 Building Code augmented by agreements made between the Atomic Energy Commission (AEC) and the Utility Owner of the plant as documented in the Safety Analysis Report for the plant.

In the early winter of 1965, a meeting was held in the offices of the consulting engineering firm of Praeger, Kavanaugh and Waterbury in New York City. This meeting was attended by representatives of Rochester Gas & Electric Co., owner of the Ginna Nuclear Power Station; the Westinghouse Electric Co. (Turnkey

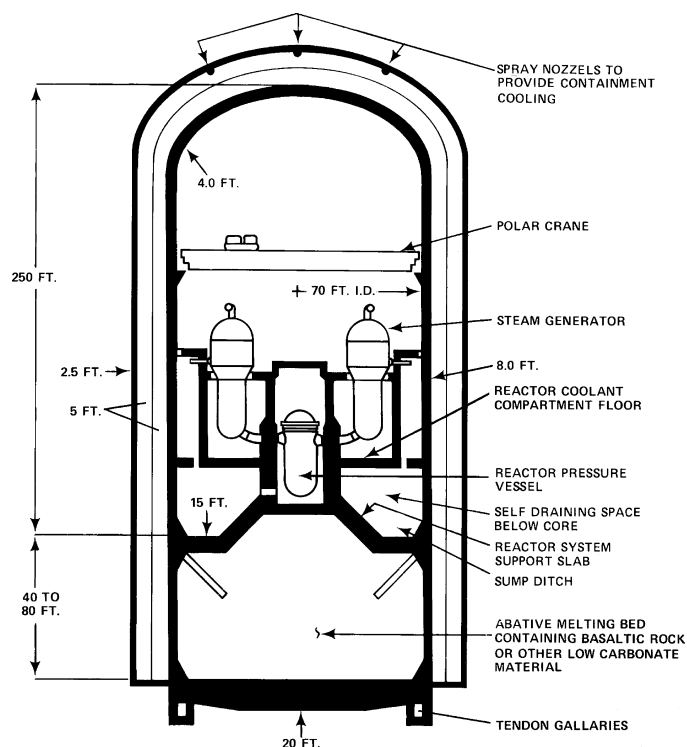


FIG. 14.11 PASSIVE DRY CONTAINMENT STRUCTURE USING RADIATOR CONCEPT FOR CONTAINMENT COOLING SYSTEMS FOR PWR

Project Manager); and Gilbert Associates, the architect-engineer for the Ginna Plant. The purpose of this meeting was to establish the loads and load combinations to be used in design of the R.E. Ginna vertically prestressed, horizontally deformed-bar, reinforced-concrete containment for a Westinghouse PWR reactor. As a result of that meeting, three load combinations were defined effectively for concrete containment design purposes.

$$L_c = D \pm 0.05D + 1.5P + T'' \quad (14.1)$$

$$L_c = D \pm 0.05D + 1.25P + T' + 1.25(E \text{ or } W) \quad (14.2)$$

$$L_c = D \pm 0.05D + P + T + E' \quad (14.3)$$

where

L_c = critical load

D = dead load of structure and equipment, including allowance for future additions plus any other permanent loading such as hydrostatic or soil; also live loads associated with operating condition floor and snow loads

E' = loading resulting from the assumed hypothetical earthquake, which is the maximum earthquake that could reasonably be expected to occur at the site at the present geological time

E = loading resulting from the assumed design earthquake, which is the maximum earthquake that could reasonably be expected to occur at the site during the plant life

T' = thermal load effect associated with design accident pressure and temperature

T' = thermal load effect associated with 1.25 factored accident pressure and temperature

T'' = thermal load effect associated with the 1.5 factored accident pressure and temperature

P'' = design basis accident pressure: the maximum calculated peak pressure increased by a factor of 10% determined as a result of a postulated single, double-ended rupture of the largest reactor coolant pipe²

Starting in 1967, design for tornado loads in the United States for sites situated east of the Rocky Mountains began to be required as follows:

$$L_c = (D \pm 0.05D + W_t) \quad (14.4)$$

where

W_t = load resulting from tornado wind assumed at a maximum 300 mph rotation, 60 mph translation coupled with reduced atmospheric pressure equal to a maximum 3.0 psig at the center of the tornado, and 1.5 psi at the point of maximum wind speed plus the impact of postulated tornado missiles at the point of maximum wind pressure

A few of the PWR containments have used rock anchors with resulting base slabs less than 2 ft thick; otherwise, base mats have varied from 9.0 to 30.0 ft thick concrete. While most of the connection between the containment base mat and cylindrical wall have been designed to be moment resisting, a few have used *hinged*-based cylindrical connection designs. Several containments were pile supported. Both bearing as well as friction piles were used.

The Ginna Nuclear Power Plant containment became the first power reactor containment in the world to be designed using factored loads and ultimate strength behavior design criteria rather than working strength design procedures.

Concrete containments were not applied to BWR-type reactor systems until the New Brunswick Nuclear Power Plant in 1970, which essentially was a Mark I containment consisting of a reinforced-concrete, deformed-bar dry and wet well. Most BWR Mark II containments are reinforced concrete and BWR Mark III containments are hybrid steel and reinforced-concrete containments.

In 1968, the U.S. AEC began to require the same safety margins when considering the earthquake loading, which later became known as the Operational Basis Earthquake and the ANSI A58.1 Standard Wind (no one-third increase in allowable stress for these as would be permitted by National Building Codes, nor the use of a 0.75 load factor as would be permitted for normal loads). This was ultimately reflected in the ACI 359-74 Containment Code by incorporating the Severe Environmental Load Category with allowable stresses typically equal to one-half those given in the Factored Load Category, which results in the equivalent load factor of 2.0. The ACI 359 and ASME B&PV Code Section III, Division 2, Subsection CC Code is unique among concrete design codes in that it uses load factors in conjunction with working stress analysis procedures.

The load combinations (but not the ultimate strength behavior criteria) with the factors as originally developed in 1965 and as supplemented in 1967 and 1968 agreements with the Nuclear Regulatory Commission (NRC) were incorporated into the 1975 ACI 359 and ASME B&PV Code Section III, Division 2, Joint Committee Subsection CC Code, and are as currently defined in the ACI 359-98 Code in Table CC-3230-1 (shown in Table 14.2).

TABLE 14.2 ASME B&PV CODE SECTION III, DIVISION 2, SUBSECTION CC LOAD COMBINATIONS AND LOAD FACTORS

Category	D	L [Note (1)]	F	P_t	G	P_a	T_t	T_o	T_a	E_o	E_s	W	W_t	R_o	R_a	R_r	P_v	H_a
Service																		
Test	1.0	1.0	1.0	1.0	—	—	1.0	—	—	—	—	—	—	—	—	—	—	—
Construction	1.0	1.0	1.0	—	—	—	—	1.0	—	—	—	1.0	—	—	—	—	—	—
Normal	1.0	1.0	1.0	—	1.0	—	—	1.0	—	—	—	—	—	1.0	—	—	1.0	—
Factored																		
Severe environmental	1.0	1.3	1.0	—	1.0	—	—	1.0	—	1.5	—	—	—	1.0	—	—	1.0	—
	1.0	1.3	1.0	—	1.0	—	—	1.0	—	—	—	1.5	—	1.0	—	—	1.0	—
Extreme environmental	1.0	1.0	1.0	—	1.0	—	—	1.0	—	—	1.0	—	—	1.0	—	—	1.0	—
	1.0	1.0	1.0	—	1.0	—	—	1.0	—	—	—	—	1.0	1.0	—	—	1.0	—
Abnormal	1.0	1.0	1.0	—	1.0	1.5	—	—	1.0	—	—	—	—	—	—	1.0	—	—
	1.0	1.0	1.0	—	1.0	1.0	—	—	1.0	—	—	—	—	—	1.25	—	—	—
	1.0	1.0	1.0	—	1.25	1.25	—	—	1.0	—	—	—	—	—	1.0	—	—	—
Abnormal/severe environmental	1.0	1.0	1.0	—	1.0	1.25	—	—	1.0	1.25	—	—	—	—	1.0	—	—	—
	1.0	1.0	1.0	—	1.0	1.25	—	—	1.0	—	—	1.25	—	—	1.0	—	—	—
	1.0	1.0	1.0	—	1.0	—	—	1.0	—	1.0	—	—	—	—	—	—	—	1.0
	1.0	1.0	1.0	—	1.0	—	—	1.0	—	—	—	1.0	—	—	—	—	—	1.0
Abnormal/extreme environmental	1.0	1.0	1.0	—	1.0	1.0	—	—	1.0	—	1.0	—	—	—	1.0	1.0	—	—

Note:

(1) Includes all temporary construction loading during and after construction of containment. See Section 14.5 for load definitions.

²Containment design pressures typically have been defined as 10%–25% above maximum calculated pressure. Initial design of containment internal structures typically used a 40% margin for containment barriers and 20% for interior structures that did not serve a containment barrier function. These margins were typically reduced to half these values when all the geometry of the interior compartments are known as part of the final design. An even lower margin could be used if rigorous modeling and dynamic load effects are considered.

14.3.2 Historical Code Development

The ACI-ASME Code requirements for concrete containments had their beginnings in 1966 as the result of meetings between the ACI and ASME representatives. It was decided that the ACI would concentrate on development of a Concrete Containment Code by means of ACI Committee 349 and the ASME would concentrate on preparing a prestressed Concrete Reactor Vessel Code. At the time, it was expected that prestressed concrete reactor vessels would be required in the design of gas-cooled reactors to be built in the United States. This parallel effort continued until 1970, when the ACI Committee had completed its preparation of a draft Code. At that time the ASME attempted to adopt the ACI-developed Code to the format and jurisdictional and inspection requirements of the ASME Code. It was at this time that basic differences in philosophy between the ACI, which considered the containment to be a structure, and the ASME, which considered it to be a pressure vessel, became apparent.

In 1971 it was decided to combine the efforts of both the ACI and ASME into a joint committee effort. The ACI effort under Committee ACI-349 was limited to concrete structures in nuclear service other than containment, and a new Committee ACI-359 and the ASME B&PV Code Section III, Division 2, Subsection CC were formed as the Joint Committee. This new joint committee had its first technical meeting in September of 1971.

The Section for Concrete Reactor Vessels and Containments was published jointly as Section III, Division 2 of the ASME Boiler and Pressure Vessel Code and ACI Standard 359 in January of 1975. It originally consisted of the following three subsections:

- (1) CA: General Requirements
- (2) CB: Concrete Reactor Vessel
- (3) CC: Concrete Containment

In the 1977 B&PV Code edition, the CA Subsection of Division 2 and the NA Subsection of Section III, Division 1 were combined into one subsection, applicable to both divisions, entitled NCA.

The technical content of the ACI-ASME Concrete Containment Code, Subsection CC, is similar to that developed by ACI Committee 349 and published in the ACI Journal in January of 1972. It also contains much of the design philosophy of the ACI Building Code 318-71. The organization and format of the ACI-359 Code and much of its procedures, inspections, and reporting requirements follow more closely the ASME Boiler and Pressure Vessel Code Section III requirements. The differences in technical content, which appear in the joint standard as compared to earlier ACI versions, can be traced in large part to the different design approaches taken by the two Societies. The two major differences are as follows:

- (1) allowable stress versus load factor design; and
- (2) working stress versus strength design.

The ACI-ASME Concrete Containment Code uses the load factor approach of the ACI but retains the working stress material design stress allowables of ASME.

It should also be noted that in ASME B&PV Code Section III, Division 1 design it is the Owner who prescribes the loads and load combination and their acceptance criteria (Design, Service Levels A, B, C, D, and Test) to be used in the Design Specification. The ACI followed its normal practice of specifying the load combinations and acceptance criteria in the body of the Code. The joint Concrete Containment Code follows the ACI practice in this area.

The ASME Code for steel pressure-retaining components typically uses the allowable or working stress concept where stresses are limited to some fraction of limiting behavior stresses (typically the lesser of 0.67 of yield or 0.25 or 0.33 of ultimate strength). These stress levels are increased as a function of the Service Level selected when behavior consequence or low probabilities of occurrence would permit a reduced margin of safety.

The load factor concept typically used by the ACI develops various safety margin levels by increasing the applicable load by some percentage as a function of individual importance to safety, loading, and response uncertainty.

With respect to working stress versus strength design in the Concrete Code design, the nonlinear behavior of reinforced concrete has been recognized since the early 1950s. For this reason, current ACI Concrete Design Codes permit the development of design-limiting internal forces and moments to be determined based on nonlinear analysis. Load factors are applied to input loads to determine resultant member forces and moments. These forces and moments are compared to forces and moments that could be developed using limiting yield or ultimate strength properties of the reinforcing steel and concrete materials. Working stress design uses elastically computed stresses and compares them with allowable stresses, which are limited by a built-in safety margin. Alternatively, the required safety margin is developed by increasing the expected load by a load factor. With strength design procedure, it is not normally possible to define the level of material stress or strain under the actual load combinations.

The ASME Code working stress approach has always defined design adequacy in terms of allowable stresses, even in those instances where fictitious or pseudoelastic stress limits are used. This permits stress levels into the inelastic range. Even in the case where inelastic or plastic analysis is permitted, design adequacy is defined in terms of allowable stress rather than strain.

The existence of these dichotomies between the ACI concrete design and the ASME steel component design required a great deal of compromise in developing a Concrete Containment Design Code. As a result, the ACI-ASME Code has features of both the ACI and ASME Codes. The ACI-ASME Code employs the ACI concept of defining explicitly the loading conditions that will be used in design, as shown in Table 14.2. However, it follows the ASME approach of defining behavior limits in terms of allowable stresses rather than internal forces and moments to be compared to limiting forces and moments based on ultimate strength design procedures.

Serviceability or functionality requirements of the Code relative to excessive concrete cracking or deformation are to ensure containment leak-tight integrity currently provided by the Factored Load Conditions also defined in Table 14.2. Service Category allowable stresses are limited to approximately one-half of those used for factored loads. The definition of loads contained in Table 14.2 are given in Section 14.5.

14.3.3 Regulatory Bases for Containment Design, Construction, Inspection and Testing

General Design Criteria (GDC) 16 and 50 of 10 CFR 50, Appendix A form the basis for designing the containment structure and associated systems, which essentially require that the containment structure must be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated loss-of-coolant, steam line, or feedwater line break accidents. The containment structure must also maintain functional integrity in the long term following a postulated accident; i.e., it must remain a low leakage barrier against the release of fission

products for as long as postulated accident conditions require. GDC 50, moreover, requires margin in the design of containment structure and associated systems which requires consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators, and as required by 10 CFR 50.44, energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculation model and input parameters.

In addition to the basic requirement, GDC 2 requires that structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed. GDC 4 requires that the structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall 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.

GDCs 51, 52, 53, moreover, require the containment structure components to be designed and constructed, so that they can (1) withstand low temperature effects, (2) accommodate periodic integrated leak rate testing, and (3) permit (a) appropriate periodic inspection of all important areas, such as penetrations, (b) an appropriate surveillance program, and (c) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

The NRC Standard Review Plan (NUREG-0800) Sections 3.8.1 related to concrete containment is developed to satisfy the regulatory requirements.

The GDC regulatory requirements, and the SRP guidance collectively provide a framework for the design, material selection and testing, fabrication of containment components, and installation and construction of concrete containments, which are codified in the Containment Code. The GDC requirements, also, provide bases for containment pressure testing, construction inspection, and periodic leak rate and structural testing of concrete containments.

Associated with these requirements are the quality assurance (QA) requirements of 10 CFR 50, Appendix B, which to an extent, are covered by Subsection NCA of Section III of the ASME Code. Other QA procedural requirements are codified in the ASME's NQA documents.

14.4 REINFORCED-CONCRETE CONTAINMENT BEHAVIOR

Containment behavior under internal pressure and temperatures up to about 100°C is mainly governed by the stress-strain laws of reinforced concrete, before and after cracking. In general, in concrete cylindrical shell containments that have a vessel diameter to wall thickness ratio greater than 10, the concrete may be assumed to be in a biaxial stress state and not require consideration of triaxial stresses.

14.4.1 Prestressed versus Deformed-Bar Concrete Reinforcements

Both the deformed-bar and prestressed posttensioned reinforced-concrete containment shells have been widely used. Prestressed-concrete containments have the advantage of thinner concrete wall and dome cross sections because the extra thickness is not required for placement of the very large quantity of deformed-bar reinforcements. In addition, prestressing permits the elimination of the need to provide tangential shear reinforcement.

The primary disadvantage of prestressed containments is the greater construction precision and attention to detail needed to erect the containment. Prestressed containments in time have also undergone the greatest changes in design. Initially, prestressed containments used six vertical buttresses to anchor horizontal prestressing, which were placed in one-third circumference segments. Also, a ring girder located at the springline cylinder to dome intersection was used to anchor vertical tendons in the cylinder and dome tendons. Almost all tendons were posttensioned and initially of the 90 wire BBRV type. Posttensioning is still used, but with 180 wire BBRV tendons. The six vertical buttress designs have given way to a three-buttress design in the United States, with each horizontal extending two-thirds of the way around the circumference. The springline ring girder anchor design has given way to the so-called hairpin design, where vertical tendons are anchored in a gallery in or below the base mat and extended over the hemispherical dome and down the other side of the cylinder.

While metal frame inserts in the concrete wall section were used in some of the earlier concrete containments to reinforce openings, current designs use secondary reinforcement to support the primary membrane reinforcement, which is now curved around the opening. In deformed-bar, reinforced-concrete containments, the cylindrical shell region surrounding the equipment batch opening is thickened by about 60% to accommodate the supplemental reinforcement around the opening.

14.4.2 Temperature Effects

Elevated temperature, which occurs under pressurized accident conditions, can reduce the strength of concrete significantly as a function of the magnitude of the temperature above about 200°F to 93°C.³

The increase in containment internal temperature under accident conditions tends to expose the liner, which tends to load the restraining concrete in tension. However, the expansion of the concrete shell under accident pressure more than compensates for the heat effect, which is restraint-of-deformation limited.

³When concrete surface temperatures rise, the water in the concrete tends to migrate away from the heated surface. Rate of temperature increase is important to the structural integrity of the concrete when temperatures exceed 212°F (100°C), where the heat rise is capable of converting the water in the concrete to steam.

For reinforced, prestressed concrete containments, the general design intent is to provide sufficient prestressing to a point such that the concrete does not crack in either hoop or longitudinal tension caused by the pressure. In such cases the uncracked concrete without steel reinforcement is assumed to carry longitudinal shear. The effect of thermal loads on the seismic resistance of containment structures has been evaluated and indicates a decrease in rigidity from thermal cracking. However, thermal cracking does not affect the ultimate seismic capacity of the structures.

14.4.3 Metal Vapor Barrier Linear Plate Systems

The containment of a nuclear power plant must maintain its leak-tight and its structural integrity in case of any design-basis accident to prevent the release of radioactive materials into the environment.

The containment structures for Light-Water Reactor (LWR) plants are designed to withstand a variety of service and factored loading conditions, including severe and extreme environmental events and abnormal (design-basis accidents) and combined severe environmental and abnormal and extreme environment and remain leak-tight in the process. The ACI-ASME Code accomplishes this by requiring a leak-tight steel liner attached to the inside face of the concrete containment boundary shells and slabs. It is important to note that in the design of concrete containments the liner is not relied on to provide any strength to the concrete section to which it is attached.

Liner thicknesses typically vary from 0.25 in. in prestressed concrete containments and 0.375–0.5 in. in thickness in deformed-bar concrete containments. Liner anchorages to the concrete in prestressed containment usually consist of embedded stick-welded angles, while headed studs are used for deformed-bar concrete containment liner anchors.

14.4.4 Radial Shear

It is essential in the design of the concrete containment structure to understand the mechanism of shear resistance of the reinforced deformed bar and prestressed concrete. Because of the complex interaction of the concrete and reinforced bars in a reinforced deformed-bar biaxially cracked containment shell, experimental work has been widely used to explain the shear action and develop design expressions. Both perpendicular and inclined reinforcing-steel patterns have been tested. The parameters included pre-cracked and uncracked concrete and changes in the reinforcement ratio. The work resulted in experimental equations for the prediction of the ultimate shear strength. For radial shear that typically occurs in a shell when shell membrane displacement is restrained, such as at the base mat cylinder junction and dome-to-cylinder juncture and near-large penetrations under pressure loading, good performance has been observed in full-scale structural integrity tests.

14.4.5 Tangential Shear

Tangential shear in containment shells is due to the lateral load on the containment structure from earthquake and wind. The effect of biaxial tension on the capacity of the deformed-bar, reinforced-concrete shells to resist tangential shear must be considered. It is in this area of tangential shear design that lateral loads from earthquakes or wind influence the design of the containment.

A direct method of resisting tangential shear is through the use of diagonal reinforcement in the plane of the shell.⁴ Since little previous test data existed relative to reinforced deformed-bar concrete orthogonal reinforcement to tangential shear load, full-scale wall elements with orthogonal reinforcement were subjected to lateral shear and biaxial tension loads. These tests concluded that significant shear stress can be resisted by the orthogonally reinforced concrete elements designed to resist pressure stresses; however, displacements will be larger than with the diagonal-reinforcing steel. These tests form the basis of the current tangential shear equations that appear in the Code.

14.5 CONCRETE REACTOR CONTAINMENT DESIGN ANALYSIS AND RELATED TESTING

14.5.1 Analysis and Design of Concrete Containment Structures

Only one BWR Mark I concrete containment was constructed in the United States. Concrete containments for BWR Mark II and parts of Mark III are mainly reinforced deformed-bar concrete, whereas for the PWRs both deformed-bar and prestressed concrete containments are common. The design loads consist mainly of internal pressure, temperature, tornado, and earthquake, all of which result in different states of stresses in the concrete elements such as tension, compression, and radial and tangential shears. These loads are also categorized as axisymmetric loads, such as the internal pressure and temperature, and non-axisymmetric loads, such as tornadoes and earthquakes. Prestressing forces are also considered as an additional load in prestressed containments. Although containment structures are mainly symmetrical in shape, consisting of base slab, cylindrical shell wall, and a dome, there are also a number of discontinuity areas near large penetrations and hatches. Early containment designs and Code requirements inherently included considerable conservatism in these discontinuity areas because of the lack of analytical tools and actual containment structural and leak-tight integrity testing experience.

In recent years, detailed two- and three-dimensional finite element modeling of these discontinuity regions to include the effects of cracking and nonlinear concrete behavior have significantly reduced the uncertainty of structural response in these regions.

It has been said that in the late 1960s it took two days to design a containment and two weeks to analyze it before the release of the design for construction. Today it would still take about two days to design a containment but about two years to analyze it. This is particularly interesting, for the actual amounts of concrete and deformed-bar or prestressing reinforcing steel have changed little in location and quantity over the past 35 years.

14.5.2 Concrete Containment Design Examples

The American Society of Civil Engineers (ASCE) Manual 58 contains example problems that illustrate the basic design and analysis procedures to be applied to reinforced-concrete, deformed-bar, and prestressed containments [2].

⁴Earlier deformed-bar, reinforced-concrete containments placed the diagonal reinforcement at the mid-plane of the cylindrical shell. More recent designs have placed the diagonal reinforcement near the outer surface of the containment to facilitate placement.

14.5.2.1 Computation of Design Loads In deformed-bar, reinforced-concrete containments:

- (1) All hoop reinforcement in the dome and cylinder is designed for membrane stresses and is usually controlled by the $1.5P_a$ load case.
- (2) Meridian or longitudinal reinforcement in the dome and middle portion of the cylinder is designed for membrane stresses and is usually controlled by the $1.5P_a$ load case.
- (3) The meridian reinforcement at the bottom of the cylinder is designed to resist the interaction of the discontinuity moment at the base and tension in the cylinder resulting from accident pressure and seismic uplift. The governing load case is usually either $1.5P_a + D$ or $1.25P_a + 1.25E_o$.
- (4) As a first step in design it is necessary to define the various loads to be considered; seismic or wind tangential reinforcement is provided as diagonal bars or horizontal or vertical and is designed to act with the hoop and meridian reinforcement to resist the combined action of accident pressure, seismic or wind uplift, and seismic or wind tangential shear. The controlling load case is usually either $1.25P_a + 1.25E_o + D$, or $P_a + E_s + D$.
- (5) Radial shear reinforcement in deformed-bar containments is provided in the lower portion of the cylinder in the form of bent bars to resist the discontinuity shear at the base. The controlling load case is usually $1.5P_a$. There is also a discontinuity shear at the springline and at penetration.

14.5.2.2 Concrete Section Reinforcement In the ASCE Manual 58, the following deformed-bar containment section design examples are presented:⁵

- (1) hoop deformed-bar wall thickness and reinforcement (dome and cylinder);
- (2) longitudinal reinforcement deformed bar (cylinder);
- (3) base mat reinforcement deformed bar;
- (4) moment and radial shear reinforcement at discontinuities at the dome-cylinder and cylinder-base intersections;
- (5) tangent shear reinforcement (cylinder);
- (6) base mat thicknesses and reinforcing; and
- (7) large penetration thickness and reinforcing.

14.5.2.3 Concrete Section and Reinforcement Prestressed In the ASCE Manual 58, the following prestressed concrete section design examples are presented:⁶

- (1) containment cylindrical wall (hoop) prestressing;
- (2) containment elliptical, hemispherical segment, or hemispherical dome and cylindrical wall longitudinal (vertical) prestressing;
- (3) containment elliptical, hemispherical segment, or hemispherical dome prestressing;
- (4) deformed-bar reinforcing for moments and radial shear at discontinuities;
- (5) base mat thickness and deformed-bar reinforcing; and
- (6) large penetration thickness and deformed-bar reinforcement.

14.5.2.4 Containment Testing Concrete containments for large nuclear power plants are unique for concrete structures in that they

routinely undergo, on each containment, a pneumatic structural integrity test to 1.15 times their design pressure and a leak-tight integrity test before going into service. They are also periodically tested for leak-tightness while in service or after modification or repair. These requirements do not exist routinely for any other concrete structure.

14.6 CODE DESIGN LOADS

Table CC-3230-1 of the ACI-ASME Joint Code (see Table 14.2) has two categories of loads: service and factored.

14.6.1 Service Loads and Load Combination

14.6.1.1 Normal Loads

- D = dead loads, including hydrostatic and permanent equipment loads
- L = live loads, including any moveable equipment loads and other loads that vary with intensity and occurrence, such as soil pressures
- F = loads resulting from the application of prestress
- G = loads resulting from relief valve or other high-energy device actuation
- T_o = thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition
- R_o = pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady-state condition
- P_v = external pressure loads resulting from pressure variation either inside or outside the containment

14.6.1.2 Construction Loads Construction loads are loads that are applied to the containment from the start to the completion of construction. The definitions for D , L , F , and T_o given in CC-3221.1 are applicable but shall be based on construction conditions.

14.6.1.3 Test Loads Test loads are loads that are applied during structural integrity or leak-rate testing. The definitions for D , L , and F given in CC-3221.1 are applicable but shall be based on test conditions. In addition, the following shall also be considered:

- P_t = pressure during the structural integrity and leak rate tests, normally taken equal to $1.15 P_a$ pressure.
- T_t = thermal effects and loads during the test

14.6.2 Factored Loads and Load Combination

The factored loads identified in Section CC-3220 of the Code are defined in the following paragraphs.

14.6.2.1 Severe Environmental Loads Severe environmental loads are loads that could be encountered infrequently during the plant life.

- W = loads generated by the design wind specified for the plant site
- E_o = loads generated by the operating basis earthquake (only the actual dead load and existing live load weights need be considered in evaluating seismic response forces)

⁵In future containing design, it is anticipated that E_o will be eliminated as a design basis.

⁶Normally controlled by required radiation protection shielding.

14.6.2.2 Extreme Environmental Loads Extreme environmental loads are loads generated by natural or human-induced phenomena, which are credible but highly improbable, that are considered as part of the design basis for the containment.

- E_s = loads generated by the safe shutdown earthquake; weights considered shall be the same as for E_o
- W_t = tornado loading including the effects of missile impact to include
 - W_{tq} = the loads from tornado wind pressure
 - W_{tp} = the differential pressure loads from rapid atmospheric pressure change
 - W_{tm} = the tornado-generated missile impact effects; the type of impact, such as plastic or elastic, together with the ability of the structure to deform beyond yield shall be considered in establishing the structural capacity necessary to resist the impact

Included in the extreme environment loads would be loading phenomena associated with blast, airplane crash, high-energy turbine failure, volcano, etc., if defined as a design basis in the plant's Safety Analysis Report.

14.6.2.3 Abnormal Loads Abnormal loads are loads generated by the design-basis accident (DBA).

- P_a = design pressure load within the containment generated by the DBA
- T_a = thermal effects and loads generated by the DBA including T_o
- R_a = pipe reaction including thermal conditions generated by the DBA including R_o
- R_r = the local effects on the containment from the DBA to include
 - Y_{rr} = load on the containment generated by the reaction of a ruptured high-energy pipe during the postulated event of the DBA; the time-dependent nature of the load and the ability of the containment to deform beyond yield shall be considered in establishing the structural capacity necessary to resist the effects of R_{rr}
 - Y_{rj} = load on the containment generated by jet impingement from a ruptured high-energy pipe during the postulated event of the DBA; the time-dependent nature of the load and the ability of the containment to deform beyond yield shall be considered in establishing the structural capacity necessary to resist the effects of R_{rj}
 - Y_{rm} = the load on the containment resulting from the impact of a ruptured high-energy pipe during the DBA; the type of impact—for example, plastic or elastic—together with the ability of the containment to deform beyond yield shall be considered in establishing the structural capacity necessary to resist the impact
- H_a = loads resulting from flooding of containment following a DBA

Combinations of severe and extreme environmental loads with abnormal loads are shown in Table 14.2.

14.7 ALLOWABLE BEHAVIOR CRITERIA

As there are two distinct load categories, service and factored, there are also two corresponding distinct behavior criteria as shown in Tables CC-3421-1 and CC-3431-1 of the Code. Since

concrete containments must act as highly reliable pressure vessels, there are serviceability restrictions on deformations for the factored load category to ensure leak-tightness not typically found in building structures.

Structural response for all load conditions, including factored loads, are limited to essentially elastic behavior. As such, behavior limits are placed on allowable stresses in the concrete and steel components for both loading categories instead of developing section capacities, as is done in the strength design of conventional concrete structures. For the factored load category, the allowable stresses are essentially those that define limiting material behavior such as yield in steel reduced by a suitable capacity reduction factor, as defined in the ACI-349 Code. For the service load category, allowable stresses are generally taken as half of those permitted for factored loads. To many ACI designers this allowable stress criteria would appear to be a step backward to the pre-1960 days of working stress design of concrete. However, the criteria are intended to meet the serviceability requirement as well as provide a rational compromise between the load factor strength design criteria of ACI and the working load allowable stress criteria of ASME. It should also be noted, as shown in Tables CC-3421-1 and CC-3431-1 of the Code, that ASME nomenclature associated with primary and secondary and membrane and bending stress allowables has also been introduced into the Code. Such distinctions between primary and secondary stresses are not usually made in building design structural Codes.

14.8 ANALYTICAL MODELS AND DESIGN PROCEDURES

It is certainly possible to design the basic concrete containment cylindrical shells and domes using simple classic shell theory. The recent tendency has been toward more complex finite-element analysis, which considers partial cracking and other nonlinear behavior of the concrete shell under internal pressure, seismic and thermal loads, and compatibility of deformations of the liner and the various layers of reinforcing steel. In regions of large discontinuity at the shell-to-base mat and dome juncture, large penetrations, high concentrated loads, and at buttresses, the finite-element analysis procedures are routinely employed to develop detailed stress resultants, which in general would not be possible using continuum shell theory. Also, finite-element representations of the foundation media as well as the containment base mat have become routine.

It is interesting to note that the computerized analysis and structural design verification typically employed on concrete containments today are perhaps ten times more complex and time consuming than the analysis performed in 1965. However, the actual locations and quantities of reinforcement and concrete that would be used today have changed little since the late 1960s.

14.9 SPECIAL DESIGN FEATURES

From a structural design standpoint, concrete containments are similar to conventional shear wall-type structures with three important differences: (1) design pressure-induced tensile stresses, (2) temperature-induced stresses, and (3) potential use of diagonal reinforcement to carry biaxial membrane seismic loads in non-prestressed containments.

The design to resist membrane or axial tensile stresses in the typically $3\frac{1}{2}$ – $4\frac{1}{2}$ ft thick concrete cylinder shell is by far the dominant design requirement. This typically results in the equivalent of

12.0 in.² or more of 60 ksi yield strength horizontal, deformed-bar reinforcement, or equivalent prestressing per foot of cylindrical wall in the hoop direction. Approximately 60% of this quantity in the vertical direction is used to carry both pressure and vertical seismic-induced stresses.

Design temperature differentials of 70°F, and temperature gradients through the containment wall of 120°F or more, typically develop restraint moments and forces that are considered in design by recognition that such temperature-induced moments and forces are deformation limited or secondary in nature. This results in maximum limited differential strain limits of twice yield being imposed on reinforcement to carry temperature-induced secondary stresses, rather than an elastic stress limitation, which is applicable to primary load-caused stresses.

Seismic-induced membrane shears in combination with pressure-induced tensile stress in non-prestressed concrete containments requiring use of diagonal reinforcement for all membrane shear stresses is controlled by the percentage of reinforcing steel, the ultimate shear and moment, and the concrete compressive strengths. The relatively low value set on concrete to carry shear is the result of the concrete being in a cracked or biaxial tension state where the mechanism for shear transfer includes aggregate interlock and dowel action.

14.10 CURRENT ORGANIZATION OF THE CODE

14.10.1 Introduction

The overall organization of the Concrete Containment Design Code is contained in the ASME B&PV Code Section III, Division 2, Subsection CC. As in other subsections of Section III, the administrative requirements, including documentation requirements, are contained in Subsection NCA, which is applicable to both Divisions 1 and 2 of Section III.

Material qualification and selection requirements for concrete containments are contained in Article CC-2000 of the Code. Article CC-3000 presents design requirements, and Article CC-4000 contains fabrication and construction requirements. Examination requirements are given in Article CC-5000. Article CC-6000 provides structural integrity test requirements after the containment is completed. Article CC-7000 presents overprotection requirements if desired, and Article CC-8000 gives ASME Code stamping requirements.

There are several Mandatory Appendices to Section III, Division 2, which are identified as follows:

- (1) Appendix I: Tables of (Allowable) Prestressing and Liner Material.
- (2) Appendix II: Glossary of Terms and Nomenclature.
- (3) Appendix III: Approval of New Material.
- (4) Appendix IV: Rounded Indications.
- (5) Appendix V: Qualifications of Concrete Inspection Personnel.
- (6) Appendix VI: Nondestructive Examination Methods.
- (7) Appendix VII: Submittal of Technical Inquiries to the Boiler and Pressure Vessel Committee.
- (8) Appendix VIII: Qualifications for Arc Welding Reinforcing Bars.

Non-Mandatory Appendices associated with Subsection CC are identified as follows:

- (1) Appendix A: Typical CRV Load Combinations (optional).
- (2) Appendix B: Non-Mandatory Preheat Procedures.
- (3) Appendix C: Certification of Level I and II Concrete Inspection Personnel.
- (4) Appendix D: Liner Dimensional Tolerances.
- (5) Appendix E: Certified Material Test Reports for Liner Materials.
- (6) Appendix F: Reinforcement Fabrication and Placing Tolerances.

14.10.2 Section III Design Specifications

Design Specifications are required for all Section III components including concrete containments. The Owner of the containment is responsible for the preparation of the Design Specifications.

The Design Specifications shall contain sufficient detail to provide a complete basis for Division 2 design. All Design Specifications shall include the following:

- (1) the functions and boundaries of the items covered;
- (2) the design requirements including all required overpressure protection requirements, if any;
- (3) the environmental conditions, including radiation;
- (4) the Code classification of the items covered;
- (5) material requirements including impact test requirements;
- (6) reference to other appropriate documents that specify the operating requirements when operability of a component is a requirement; and
- (7) the effective Code edition, addenda, and Code Cases to be used for construction.

Design Specifications shall be provided for each concrete containment serving in a single power-generating unit for multiple concrete containments at the same site. In addition to the requirements in the preceding list, the Design Specifications shall also include the following:

- (1) Design life.
- (2) Corrosion effects.
- (3) Structural acceptance testing requirements (CC-6000).
- (4) Shielding requirements.
- (5) Construction surveillance required by the designer.
- (6) Foundation type and allowable loading.
- (7) Loads from internal structures (8) Identification of those components and / or parts that require a preservice examination including the following:
 - (a) edition and addenda of ASME B&PV Code Section XI to be used;
 - (b) category and method;
 - (c) qualifications of personnel, procedures, and equipment; and
 - (d) Weld information, including surface conditioning requirements and identification marking system to be used.

Responsibilities and distribution of the Design Specifications are given in Table 14.3.

14.10.3 Section III Design Report

The designer shall prepare a Design Report in sufficient detail to show that the applicable stress limitations are satisfied when the concrete containment component is subject to the loading conditions specified in the Design Specifications and Subsection CC. The Design Report prepared by the designer shall contain calcula-

TABLE 14.3 DOCUMENT DISTRIBUTION FOR DIVISION 2 CONSTRUCTION

Document	Prepared by	Reviewed by	Certified by	Approved by	Provided to [Note (1)]	Available on Request
Design Specification (NCA-3250)	O	O	O	—	D, C, I, J	—
Construction Specification (NCA-3340)	D	O	D	O	O, C, F, M	I, J
Design Drawings (NCA-3340)	D	O	D	O	O, C, F, M	I, J
Design Report (NCA-3350)	D	O	D	O	O	I, J
Construction Procedures [Note (1)] (NCA-3451)	C, F	D	—	D	D, O	I, J
Certified Material Test Reports or Certificates of Compliance [Note (1)] (CC-2130)	M	C, F	M	—	C, F, O	I, J, L, D
Shop and Field Drawings [Note (1)] (NCA-3452)	C, F	D	—	D	C, F	I
Construction Report (NCA-3454)	C	D	D, I	O	D, O, J	I, J
Data Report C-1 (NCA-8410)	C	—	D, C, I	—	O	I, J
Data Report N-2 (NCA-8410)	F	—	F, I	—	C	I, J
Data Report N-3 (NCA-8420)	—	O, I	—	J	—	—

General Notes:

- (a) Symbols used: C Constructor; D Designer; F Fabricator; I Inspector; J Enforcement Authority; M Material Manufacturer; and O = Owner or Designee.

Note:

- (1) Information provided to the indicated participants when required to satisfy their designated responsibilities under this Section. Other information provided only by specific arrangement with the Owner. Participants are required to furnish only such information as is necessary to permit the recipient to perform his or her duties in conformance with this Section; other information may be furnished at the discretion of the responsible parties.

tions and sketches substantiating the design's accordance with the Design Specifications and Subsection CC. Distribution of the Design Report is shown in Table 14.3.

14.10.4 Subarticle NCA-3200

This paragraph contains the responsibilities for the various documents required by Section III, Division 2, Subsection CC associated with concrete construction. In particular it requires the preparation of a Construction Specification by the designer, the contents of which are as follows:

- (1) material specifications (text or by reference);
- (2) material shipping, handling, and storage requirements;
- (3) inspection requirements;
- (4) appropriate Code references;
- (5) requirements for personnel or equipment qualification;
- (6) material or part examination and testing requirements;
- (7) acceptance testing requirements;
- (8) leak testing requirements;
- (9) requirements for shop drawings;
- (10) requirements for batching, mixing, placing, and curing concrete;
- (11) requirements for the fabrication and installation of the prestressing system, reinforcing steel, embedments, and all other parts;
- (12) identification of parts requiring a Code stamp;
- (13) design life for parts and materials where necessary to establish compliance with the Design Specifications;

- (14) construction surveillance to be performed by the designer as required by the Design Specifications;
- (15) construction documents that require review by the designer, as well as those that require both review and approval by the designer (as a minimum, these will include the requirements of Table NCA-3200-1); and
- (16) tolerances and limits not otherwise defined in Subsection CC.

The Construction Specification is the primary document by which the designer makes known to the constructor the construction requirements for the concrete containment. As such the Construction Specifications contain many of the requirements placed in the Design Specifications by the Owner passed through to the constructor by the designer.

In addition, the Designer prepares Design Drawings that include the following:

- (1) concrete and steel liner thickness
- (2) size and location of reinforcing steel
- (3) size and location of prestressing tendons
- (4) size and location of penetrations
- (5) all other details necessary to construct the item in accordance with the requirements of the Design Specification, the Construction Specification and Subsection CC.

The Construction Specification, Design Drawings and Design Report shall be reviewed and certified to be correct and in accordance with the Design Specification and Subsection CC by one or more Registered Professional Engineers competent in the

field of design of concrete components and qualified in accordance with the requirements of Appendix XXIII of Section III Appendices.

14.10.5 Article CC-1000

This article contains the scope and general requirements applicable to concrete containments. A containment is defined as having a design pressure greater than 5 psi (35 KPa). This pressure load design requirement is what distinguishes containments from confinements in nuclear service. Concrete confinements are normally designed to the requirements of the ACI-349 Code. Article CC-1000 also provides some guidance regarding boundaries of jurisdiction that are required to be defined in the Owner's Design Specifications.

14.10.6 Article CC-2000: Material

This article presents the general and specific material requirements for concrete containments. It includes material requirements for concrete and constituents (cement, water, aggregates, and admixtures), requirements for concrete mix design and proportioning and requirements for cement grout. Also included are material requirements for deformed bar, including mechanical connectors, prestressing wire and strand, liner metal and embedded anchors, along with welding requirements and requirements for a Material Manufacturer's Quality System Program.

14.10.6.1 Subarticle CC-2221: Cement The cement used shall be that used in establishing the required strength.

14.10.6.2 Subarticle CC-2222: Aggregates The aggregates shall conform to the ASTM C33

Specification with the exception that the Engineer may approve variations in gradation provided that concrete design properties referred to in CC-2231.1 are met.

The term "cubical" implies a crushed aggregate particle that is not thin or elongated.

The size limitations on aggregates are provided to ensure proper encasement of reinforcement and to minimize honeycomb conditions. These limitations may be waived by the Engineer if the workability and consolidation methods employed are deemed suitable to enable concrete placement without honeycomb or voids.

14.10.6.3 Subarticle CC-2223: Mixing Water Mixing water shall be tested as specified. Generally any natural water that is drinkable (potable) and has no pronounced taste or odor will be satisfactory as mixing water for concrete. Salts or other deleterious impurities contributed from the mixing water need to be added to those contributed from other ingredients (aggregates, admixtures, etc.) when evaluating the acceptability of total impurities in the concrete.

14.10.6.4 Subarticle CC-2224: Admixtures Liquid admixtures should be certified as "non-chloride". Any chloride levels present should only be trace amounts caused by small levels by impurities.

14.10.6.5 Subarticle CC-2232: Selection of Concrete Proportions Requirements for durability will often dictate that a lower water-cement ratio be employed than that qualified by field experience and/or trial mix data.

14.10.6.6 Subarticle CC-2233: Proportioning on the Basis of Field Experience and/or Trial Mixtures It is recommended that

field test records and trial mix data used to verify concrete proportions for use be produced by a testing agency that is conforming to ASTM C1077 and accredited by a nationally recognized evaluation agency.

Once use of proportions established using trial mixtures have produced a suitable field record of tests, a standard deviation can be calculated from the field data and then used to select proportions from the strength curve having lower cement factors and/or higher water-cement ratios. However, the maximum water-cement ratio required for durability must not be exceeded.

14.10.7 Article CC-3000

14.10.7.1 Subarticle CC-3100: General Requirements CC-3100 presents the general design requirements for deformed-bar and prestressed concrete, metallic liner, including anchorage and penetration assemblies used in nuclear power plant containments. Also included are definitions of terms applicable to concrete not otherwise included in the glossary of Section III (NCA-9000) and categorization of the forces and stresses required to define allowable stresses. Finally, this paragraph discusses how tolerances are to be considered. Table 14.4 is a reprint of Table CC-3136.6-1 that gives a classification of forces and stresses as a function of the containment loads and locations.

14.10.7.2 Subarticle CC-3200: Load Criteria This subarticle presents the load categories, normal, construction, test, severe environmental, extreme environmental, and abnormal, to be considered in design. It also defines load combinations to be considered as well as load definitions in the static, cyclic, impulse, and impact categories.

14.10.7.3 Subarticle CC-3300: Containment Design Analysis Procedures This subarticle presents general analysis procedures for concrete shells, base mat, frames, box-type structures, and assemblies of slabs as well as penetrations and openings.

14.10.7.4 Subarticle CC-3400: Concrete Containment Structural Design (Behavior) Allowables This subarticle presents the allowable stresses or behavior criteria as a function of the applied loading. This paragraph is unique in its current application to concrete structures in that it still gives the acceptance criteria as allowable computed stresses for both service and factor loads in the concrete and steel reinforcement rather than using limits of strength design procedures. This allowable stress behavior criteria, rather than the strength design, is used to limit the deformations that the containment can undergo to essentially elastic behavior to better ensure its leak-tight integrity. However, it should be noted that the stresses computed also include load factors different from 1.0 in the loading combination equations of CC-3200 for factor loads.

14.10.7.5 Subarticle CC-3500: Containment Design Details This subarticle provides detailed design requirements for reinforced-concrete sections, both deformed-bar and prestressed for flexure and axial, tangential shear, radial shear, peripheral shear, and torsion shear stresses. Also included are deformed-bar splice and anchorage requirements and deformed-bar cover and spacing requirements. For prestressing, system requirements are given associated with loss of prestress, tendon anchor reinforcement, placement of curved tendons, and radial tension reinforcement. Containment separation from other structures and foundation requirements are also discussed in general.

TABLE 14.4 CLASSIFICATION IN CONCRETE CONTAINMENTS OF STEEL REINFORCING AND CONCRETE ALLOWABLE STRESSES

Location	Origin of Loads	Type of Force	Classification
Regions away from discontinuities	External [Note (2)]	Membrane	Primary
		Bending	Primary
		Shear [Note (3)]	Primary
	Volume change effects such as creep shrinkage and thermal strains	Membrane	Secondary
		Bending	Secondary
		Shear	Primary
Regions at and near gross changes in shell geometry	External [Note (2)]	Membrane	Primary
		Bending	Primary
			[Note (4)]
	Volume change effects such as creep shrinkage and thermal strains	Shear	Primary
		Membrane	Secondary
		Bending	Secondary
Regions near large openings	External [Note (2)]	Shear	Primary
		Membrane	Primary
		Bending	Primary
	Volume change effects such as creep shrinkage and thermal strains	Shear	Primary
		Membrane	Secondary
		Bending	Secondary
		Shear	Primary

Notes:

(1) Allowable stresses for concrete may be considered secondary in the region defined in CC-3422.1(c)(3).

(2) Includes prestressing.

(3) Includes both radial and tangential shear force.

(4) For allowable stresses, bending at discontinuities due to external loads is considered primary.

14.10.7.6 Subarticle CC-3600: Liner Design and Analysis Procedures In this subarticle, procedures used in liner, liner anchor, penetration assemblies, brackets, and attachments are discussed in general.

14.10.7.7 Subarticle CC-3700: Liner Design In this subarticle, specific behavior limits are given or referenced for the liner, liner anchors, penetrations, brackets, and attachments.

14.10.7.8 Subarticle CC-3800: Liner Design Details This sub-article covers detailed requirements for liner anchors and penetrations as well as transition zone from concrete to steel components designed to contain pressure. These design procedures apply to what are typically called hybrid containments where part of the containment structure resisting pressure loads is reinforced concrete and part is steel plate.

14.10.7.9 Subarticle CC-3900: Design Criteria for Impulse Loadings and Missile Impact This subarticle defines the ductility limits allowed under abnormal, extreme environment, and combined extreme environment and abnormal impact- and impulse-type loading. Also presented is a discussion of missile penetration formula limitations, local missile impact areas, and the effective mass of the containment responding during impact.

14.11 ARTICLE CC-4000: FABRICATION AND CONSTRUCTION

The general and specific construction or placement requirements for concrete are given in this article. It includes requirements for storing, batching & mixing of materials, placing and curing of concrete including preplaced aggregates concrete, curing, formwork and joints, hot and cold weather conditions, repairs

and cement grout for tendons. Also, included are the fabrication and placing requirements for steel reinforcement systems and the fabrication and welding requirements for liners.

14.11.1 Subarticle 4210: General

The construction specification should be descriptive and specific in identifying requirements for all phases of concrete construction. Wholesale reference to other documents for instructions should be avoided.

14.11.2 Subarticle 4220: Storing, Batching, Mixing, and Placing

Batching facilities should be inspected prior to the start of production to insure that requirements of ASTM C94 can be achieved. It is recommended that batching facilities and delivery units be certified by the National Ready-Mix Concrete Association.

Excessively long mixing times should be avoided to guard against grinding of the aggregates.

For truck mixing operations, uniformity for each different mixer drum design needs to be verified. Field additions of water or admixtures must be accompanied by proper mixing to insure uniform distribution of any field added ingredient. Field water additions may only be added to full loads in order to adjust slump within the allowable range and only if the addition does not exceed the maximum allowable water of the mix. Water added after discharge has effectively begun is considered re-tempering and is not permitted.

14.11.3 Subarticle CC-4224: Conveying

Some conveying equipment can alter the properties of concrete. In these instances, concrete needs to be tested before and after conveyance in order to establish concrete control properties at the

point of delivery which are needed to assure that concrete at point of placement has the concrete qualities that are specified.

14.11.4 Subarticle CC-4225: Depositing

The possibility of segregation during handling and placement increases with higher slumps. It is good practice to place concrete at the lowest slump that enables complete filling of the forms and full encasement of reinforcing and embedments.

The height at which concrete can be dropped without segregation will vary from mix to mix and is highly dependent on the maximum size of aggregate used and the slump of concrete as placed. The engineer should consider these concrete properties as well as expected placement conditions prior to establishing limits in the Construction Specification.

14.11.5 Subarticle CC-4226: Consolidation

Mortar used at congested rebar conditions should be placed immediately before depositing concrete and should neither be too stiff or too fluid.

The Engineer must determine what types of placement conditions will require mockups to be constructed but typically these would include special types of congested reinforcement conditions and unique placement techniques such as pre-placed aggregate concreting.

14.11.6 Subarticle CC-4240: Curing

Curing and protection of mass concrete to prevent thermal induced cracking requires special practices and precautions such as maintaining temperature differentials between surface concrete and interior concrete at a minimum (usually 36°F max).

Reference ACI-207.IR for further information and references, [13]

14.12 ARTICLE CC-5000: CONSTRUCTION TESTING AND EXAMINATION

This article lists the requirements for nondestructive examinations, procedure qualifications and evaluations, and requirements for qualification and certification of non destructive personnel. Also listed are requirements for qualification of concrete inspection and testing personnel and requirements for concrete laboratory qualifications. A complete listing of tests and testing frequencies for concrete materials, reinforcing systems and welds is also provided. This Code article is unique with respect to the scope of material tests and testing frequencies listed which are required to be included in the Construction Specification.

14.12.1 Subarticle CC-5232: Concrete Properties

Procedures for sampling should be implemented that ensure random selection of samples for strength testing.

Concrete samples for acceptance testing should represent the quality of the concrete at the point of placement into the forms. To achieve this, samples in some instances will need to be taken at the discharge end of conveyance systems.

Due to variations in production and material qualities, strength test specimens will occasionally fail to meet the specified strength (about 1 test in 100) at the designated age. Some concrete mixes, especially those with pozzolanic mineral admixtures, will continue to gain strength well beyond 28 days and the casting of additional test specimens for later age testing of these types of concrete mixtures is recommended.

Cement factors should be increased whenever unusual frequency of failing tests is encountered. Concrete mixes should also be monitored and evaluated as to whether or not they are achieving the strength levels indicated in the mix submittal documentation. Steps should be initiated to identify and correct quality deficiencies before test results fall below specified strength levels.

14.13 ARTICLE CC-6000: STRUCTURAL INTEGRITY TEST OF CONCRETE CONTAINMENTS

This article presents requirements for the following subjects associated with Containment Structural Integrity Test:

- (1) general requirements;
- (2) test and instrument plan;
- (3) testing of ASME B&PV Code Section III, Division 1 parts, appurtenances, and components;
- (4) classification of containments as prototype or nonprototype;
- (5) test procedure;
- (6) structural test requirements;
- (7) pressurization;
- (8) data required;
- (9) evaluation of results; and
- (10) analysis of data and preparation of containment structural integrity test report.

Nuclear power reactor concrete containments are unique among large concrete structures in that they routinely require a design proof test before being placed into service.

14.14 ARTICLE CC-7000: OVERPRESSURE PROTECTION

Concrete nuclear power plant containments are also unique in that they do not normally contain any over- or underpressure protection devices. Steel containments usually contain an internal negative pressure-relief device in the event of an inadvertent operation of containment sprays or fan coolers, causing a cooling down of the containment atmosphere and, hence, a negative pressure transient that can be as high as 2.0 psig. Concrete containments are not susceptible to any failures resulting from containment negative pressure transients. However, there is no wording in CC-7000 that would preclude the installation of a pressure-relief device if required by the Design Specifications. In some countries such devices have been installed in concrete containments to provide a filtered, vented containment capability.

14.15 ARTICLE CC-8000: NAMEPLATES, STAMPING, AND REPORTS

This article references Article NCA-8000 for nameplates, stamping, and report requirements. NCA-8000 covers the following attributes, which are required to obtain ASME Code Certification:

- (1) authorization to perform Code activities;
- (2) scope of Certificate;
- (3) required inspection agreement;
- (4) Quality Assurance Program requirements;
- (5) application for accreditation;

- (6) evaluation (by ASME);
- (7) nameplates; and
- (8) data reports.

14.16 PRACTICAL NUCLEAR POWER PLANT CONTAINMENT DESIGNED TO RESIST LARGE COMMERCIAL AIRCRAFT CRASH AND POSTULATED REACTOR CORE MELT

14.16.1 Historical Development of Containment Structure Design Requirements

The initial design of large nuclear power plant light water reactor system containments in the U.S. was developed in the early to middle 1960s and was directed toward protecting the environment and in particular the public from a set of prescribed accident conditions associated with reactor operation. The design basis for the containment structure design was to contain essentially without leakage the result of the instantaneous double ended guillotine rupture of the largest reactor coolant pipe. Other accident conditions were also postulated, including missiles in the form of valve and instrument parts ejected from the high pressure reactor coolant system up to and including the rupture of a control rod drive mechanism. Since there was no practical way to develop containment redundancy, robustness of design and high quality construction practices were developed that essentially treated the Design Basis Accident Loads as normal design loads, which included the increased design margins associated with such loads.

Starting in the mid 1960s a second mission was given to containments and that was to protect the reactor coolant and protection systems from external extreme events. The first of these was extreme earthquake resistant design, followed in the late 1960s by tornado and external extreme flood resistance as design requirements with loads several times those normally considered in conventional industrial facility design.

Initially, the accident design load was not considered mechanistically, but simply as an energy release into the containment volume with a resultant static pressure load on the containment vessel. However, as time progressed the local effects of postulated consequential main coolant pipe rupture, fluid jets, and reaction loads at the break location and pipe whip began to be considered as part of the design basis, which greatly complicated containment internal design. In the late 1980s the containment design basis was modified to consider only the reactor coolant system total energy release, and localized effects were limited to branch piping connected to the main reactor coolant system and other high energy piping and then only when leak before break could not be demonstrated.

It should be noted that there remained a number of potential accidents that were not considered as design basis accidents, as summarized in Table 1, that could challenge containment integrity that were considered of such low probability as not to be considered a credible design basis.

14.16.2 Recent Developments

Recent developments and experience suggest that some of the extreme load phenomena that have not been considered as a design basis in the past may no longer be considered incredible. In addition, the conventional practice of locating nuclear power plants (NPP), away from population centers may have become so expensive as to justify use of the concept of metropolitan siting. These two new conditions have prompted the consideration of the concept

TABLE 14.5 SEVERE ACCIDENT AND EXTERNAL EVENT SCENARIOS NOT CONSIDERED IN EXISTING NPP-PWR CONTAINMENT DESIGN

- Simultaneous Primary Reactor Coolant and Secondary Heat Transport Steam Systems Blowdown
- Reactor Core Melt
- Reactor Vessel Rupture
- Hydrogen Deflagration
- Hydrogen Detonation
- Large Commercial Aircraft Crash at Controllable Flight Speed (400 mph)

of NPP containment structures and systems, which can accommodate the worst possible accident and extreme environmental load phenomena of reactor vessel rupture, core melt, hydrogen detonation, deflagration, and a large commercial aircraft crash.

14.16.3 The Concept

In Figure 14.11 is shown the conceptual design of a PWR containment structure designed to resist those postulated failures listed in Table 14.5 that are not currently considered in containment design. In developing this conceptual design, the materials contained in References 4 through 9 were reviewed for applicability to develop input design parameters. Once these input design parameters were defined as shown in Table 14.6, the containment structure was sized and preliminarily designed in accordance with the requirements of existing standards.⁽⁶⁾ It should be understood that the design basis defined in Table 14.6 was based on a review of the current literature as applicable. No attempt was made within the limited scope of this effort to develop a detailed design of the containment structure based on basic detailed NSSS and site dependent input data and detailed calculation. It is hoped that interest in the apparent advantages of a passive containment system capable of withstanding all conceivable accidents and external threats as described herein may promote the funding of such a detailed effort in the near future. In Figure 14.12 is the application of this proposed design concept to a BWR containment. It should be noted that this concept would eliminate the need for an active pressure suppression type of containment for a BWR reactor system.

This design concept at least would permit the elimination of conventional containment safety-related accident heat removal systems in favor of an externally cooled radiator. In this concept the containment atmosphere within the primary prestressed concrete containment could be circulated to an external steel shell through pressure relief valves where external water spray would cool and condense the containment vapors. Surrounding the intermediate steel containment radiator would be a more or less conventional biological shield and environmental protection concrete shield structure. A postulated large airplane crash would first have to penetrate the environmental shield building and the radiator steel shell before it would impact and be stopped by the 8.0 feet thick primary prestressed concrete containment shell.

14.16.4 Design Features

14.16.4.1 The Containment Structure The passive containment internal structural design above the reactor coolant compartment floor as shown in Figure 14.11 and 14.12 would be very similar to conventional prestressed concrete single barrier containment designs for PWR dry and BWR Mark III type light water reactors.

TABLE 14.6 PWR PASSIVE CONTAINMENT SYSTEM DESIGN PARAMETERS

Design Parameter	Cause	Calculated Value
1. Prompt Pressurization	Simultaneous blowdown of primary reactor coolant and secondary heat transport system in a PWR	105 psi
2. Delayed Pressurization	Vaporization of water by the molten core after vessel rupture, including:	
	(a) water in reactor cavity	95 psi
	(b) water of crystallization from concrete	5 psi
	(c) hydrogen from zirconium water reaction	5 psi
	(d) CO ₂ generated from concrete	45 psi
	TOTAL	150 psi
	+20% Design Margin	30 psi
	Design Pressure	180 psi
	Design Temperature	375°F
OR		
3. Hydrogen Deflagration		25 psi
OR		
4. Hydrogen Detonation		150 psi
OR		
5. Large (Boeing 747) Aircraft Crash at 400 mph Not Combined with RCS or Heat Transport System Failure		

However, the water suppression system for the BWR type reactors would no longer be necessary. The containment cylinder walls and dome would be approximately 8.0 and 4.0 feet thick rather than the current 3.5 and 2.5 feet, respectively. This increased thickness would be required to resist the increased containment design pressure as shown in Table 14.6. In addition, the 8.0 feet (2.5 m)

selected is approximately equal to the 7.75 feet (2.38 m) used for the BASF reactor burst protection structure for the reactor vessel that had a function of containing potential missiles from a burst vessel as developed in Reference 4 and would effectively stop a large commercial aircraft at controllable flight speed impact at 400 mph.

It should be noted that the much simpler, so called “dry,” rather than pressure suppression containment system, could be used for a BWR light water reactor because the basic containment design as described herein is capable of containing both the primary coolant as well as the secondary heat transport system energy releases.

The design pressure selected is 180 psi, which is three times the design pressure typically used in current prestressed concrete 1300 MWe PWR-NPP containments and 2.25 times that developed in a typical pressure suppression BWR dry well. This design pressure was selected based on the analysis presented in References 4 through 8. Also, it should be noted that the postulated simultaneous blowdown of the secondary as well as primary systems in a PWR and BWR could give rise to a containment pressure approximately twice that currently considered in design of PWR containments and 1.6 times current BWR dry well design.

The containment section below the reactor coolant compartment floor and above the reactor system support slab is space that does not exist in current PWR containment designs and could be used to house safety-related components and systems currently housed in the reactor auxiliary building. For this proposed containment system to be effective, it would be imperative that the amount of water and normal concrete in potential contact with the molten core be minimized to limit the potential for water molten core reaction and CO₂ gas generation. In the design, the reactor system support slab would also function as water barrier from the reactor coolant system compartment of the containment from the ablative melting bed compartment.

The containment configuration below the reactor coolant compartment floor is essentially the same as proposed by Van Erp⁽²⁾. It is anticipated that ablative melting cavity would be steel lined as

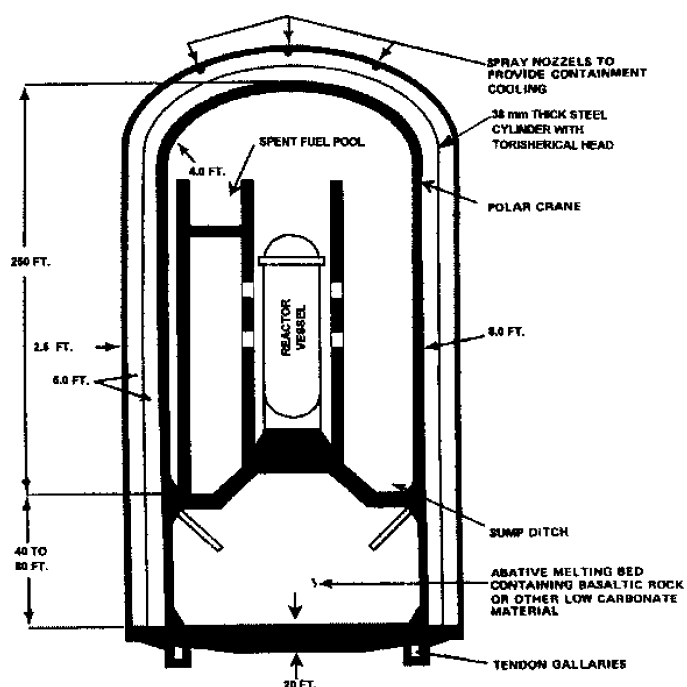


FIG. 14.12 PASSIVE DRY CONTAINMENT STRUCTURE USING RADIATOR CONCEPT FOR CONTAINMENT COOLING SYSTEMS FOR A BWR

is the rest of the containment to provide a vapor barrier and refractory baffles used or alternatively, the steel liner would be faced with refractory to insure no melt-through of the liner.

In Figure 14.11 and 14.12 a flat bottom, free-standing steel shell would be used to act as a giant radiator to provide containment cooling and thereby reduce or eliminate the need to provide containment cooling as an internal engineered safeguard. Practical limits of anchorage of the steel shell to a flat concrete base slab would limit internal design pressure on the steel shell to approximately 15 psi. Essentially, this approach has been used on the bell jar hybrid steel and concrete containment concepts used on PWR ice condenser NPP and Mark III type BWR containments.

14.16.4.2 Emergency Core Cooling Systems In the passive containment design suggested herein, emergency core cooling system (ECCS) would no longer be required to ensure public health and safety since the containment system can accommodate a reactor core melt. As such, it would be downgraded to a normal plant operating system whose design would be dictated by economic (potential loss of plant investment) rather than public safety considerations. The ECCS would logically be reduced to a single train instead of three trains, with perhaps additional active components so that the systems would still be operative during maintenance outage. Besides the obvious reduction in direct capital costs, the congestion in the reactor building for system support and distribution systems, requirements for in-service inspection, testing, maintenance, control, instrumentation, and emergency power requirements would all be greatly reduced. Finally, the design of the reactor coolant systems (RCS) would be simplified with the reduction of the number of ECCS injection nozzles in the RCS with their attendant effect as a discontinuity and potential for localized stress concentration, which could lead to failure.

14.16.4.3 Containment Cooling System In the concept shown in Figures 14.11 and 14.12, an internal containment cooling system (CCS), including containment spray, would no longer be a safety requirement since the external steel radiator and sprays would provide this function as needed.

14.16.4.4 Reactor Coolant System Component Supports Currently, a prime function of, and the one that largely dictates their design in PWRs, is the requirement that the RCS support structures isolates the effects of a postulated rupture of the primary reactor Coolant system from the secondary heat transport steam system and vice versa. This criterion in existing designs requires that the RCS support structures as a safety function be designed for equivalent static reaction loads, which vary between 2.5 and 8.0×10^6 pounds. These loads may be as much as an order of magnitude greater than the supports see in normal operation and under earthquake loads.

In the U.S., these extremely large RCS or steam rupture loads also have been combined with the loads associated with the Safe Shutdown Earthquake (SSE), adding even more to the support design load. Such additional seismic loads typically range between 10 to 25 percent of the LOCA load effect taken alone. This new passive containment concept of Figures 14.11 and 14.12 would no longer require RCS supports to be designed for postulated RCS pipe rupture loads as a safety issue. Elimination of the

LOCA or steamline break isolation requirement combined with SSE would have a significant effect on the reduction in RCS component support capital costs and the congestion caused by the design of massive load resisting supports. Postulated pipe break loads from all high energy systems could all also be eliminated as a design basis.

14.16.4.5 Elimination of Pressure Suppression While the effects on costs associated with elimination of pressure suppression have not been evaluated in this study, it would no longer be necessary to provide a water pressure suppression system for large commercial BWR type reactors since the containment would be able to accommodate both the primary reactor coolant as well as the secondary heat transport system's simultaneous total energy release.

14.16.5 Cost Comparison

In Table 14.7 is presented a quantity estimate comparison between conventional prestressed concrete cylindrical containments for a typical 1300 MWe PWR-NPP and the concept suggested in this paper as shown in Figures 14.11. In Table 14.8 is presented a relative cost estimate and comparison between current conventional and the passive containment types described in Table 14.7 with the cost of current containments normalized to 1.0. Credits are also considered associated with reduction in the complexity and redundancy associated with a PWR's emergency core cooling system in downgrading such a system from a safety-related engineered safeguard to a system whose design is dictated by economic consideration associated with protection against the loss of the power plant investment.

An added significant effect on overall plant costs presented in Table 14.8 is the reduced requirement for in-service inspection and maintenance during plant life, which would be achieved by reducing the complexity of the ECCS and associated elimination of redundant safety trains and downgrading it to a plant operating system, since it is no longer necessary for public health and safety.

Also presented in Table 14.8 is the estimated significant reduction in the cost of major PWR component supports, which are designed to accommodate the extremely large loads induced in the reactor coolant system when isolation of the primary RCS and secondary heat transport system is no longer a safety issue.

14.16.6 Summary and Conclusions

Based on the cost estimate summarized in Section 14.15.5, it would appear the passive containment concept and revised design bases presented in Section 14.15.3 would add about 1.1 times the containment structure cost to an overall 1300 MWe PWR NPP costs. Given that the containment structure adds about 6.0 percent to overall plant costs, the total increase in PWR plant cost for a passive containment system design for all severe accident scenarios, plus large airplane crash suitable for metropolitan siting, would be about 6.7 percent. This should merit further study from an economic as well as improved safety standpoint.

This is particularly true when considering a large aircraft crash as a design basis event. In addition, based on this preliminary study, there does appear to be a containment and nuclear power plant design option, at least from a postulated external event hazard or accident standpoint, which would permit consideration of metropolitan siting.

14.17 ITEMS WHICH SHOULD BE CONSIDERED IN FUTURE REVISIONS OF THE CODE

The ACI-318 code has determined that cementitious mineral admixtures having pozzolanic qualities (notably natural pozzolans, flyash, ground granulated blast-furnace-slag and silica fume) contribute many of the same durability properties associated with portland cement. Therefore, weights of these materials are added

**TABLE 14.7 QUANTITY ESTIMATES CONVENTIONAL PRESTRESSED CONCRETE
COMPARED TO CONCRETE PASSIVE CONTAINMENT DESIGN**

I. Conventional Containment—140' I.D. Height to Springline 140' Diameter Prestressed Concrete Wall 3.5 ft. thick and Hemispherical Dome 2.5 ft. thick, Design Pressure = 60 psi

Element	Concrete Cu. Yards	Rebar Tons	Prestressing Ft. of 170 Wire Tendons	Liner Sq. Ft.	Misc.
Reactor Sump	600	125		2,500-1/40" PL	
Base Mat	6,950	2,450		14,000-1/4" PL	
Cylinder	8,380	350	17,000 Horizontal 13,500 Vertical	62,000-1/4" PL	
Dome	2,950	150	10,250	31,000-1/4" PL	
Interior Concrete	8,500	1,600			
TOTALS	27,380	4,675	41,000	109,500	

II. Passive Containment—140' I.D., Height from Reactor Coolant Compartment Floor to Springline 110', Height from Containment Base Mat to Top of Reactor Coolant Compartment Floor 120', Primary Containment Wall 8.0 ft. thick and Hemispherical Dome 4.0 ft. thick, Design Pressure = 180 psi. Primary Containment surrounded by Steel Shell Radiator with an average 1.0 inch thickness and Reinforced-Concrete-Shield Building 2.5 ft. thick walls and 2 ft. thick Dome.

Element	Concrete Cu. Yards	Rebar Tons	Prestressing Ft. of 170 Wire Tendons	Liner Sq. Ft.	Misc.
Base Mat	18,850	8,500		25,500-1/4" PL	
Cylinder	31,700	1,000	85,000 Horizontal 65,000 Vertical	101,200-1/4" PL	
Dome	6,500	500	32,000	34,000-1/4" PL	
Interior Concrete	17,500	3,800			
Basalt Rock					45,600 yd ³
Radiator Shell				163,000-1" PL	
Shield Bldg.	14,300	500			
TOTALS	88,850	14,300	182,000	160,000-1/4" PL 163,000-1" PL	45,600 yd ³

to the weight of portland cement and a water-cementitious ratio is then calculated and compared to the established water-cement ratio limits for durability. The present code only recognizes water-cement ratio not water-cementitious ratio.

The ACI-318 Code requires concrete exposed to freezing and thawing in a moist condition to have a maximum w/c ratio of 0.45. The present code permits a maximum w/c ratio of 0.50 for these conditions.

The ACI-318 Code requires concrete to be maintained above 50°F and in a moist condition for the first 7 days after placement. Curing requirements given in CC-4240 are either left to the Construction Specification, are at variance with ACI guidelines or represent incorrect interpretations of ACI as in (d) where it is stated that "the protection period can be reduced if concrete achieves 500 psi". ACI simply states, that concrete that has achieved strength of at least 500 psi will not be damaged by exposure to a single freezing cycle. The 306R procedures for cold weather curing and protection should be followed. Article CC-4260 references as-placed temperatures of concrete be based upon ACI 305R and ACI 306R but neglects to reference the curing and protection temperatures and duration periods given in these references.

14.18 SUMMARY

It should be emphasized that, although containment analysis and design verification have grown more detailed and complex and the specific loads considered in design are more numerous,

the concrete containments in terms of concrete and reinforcement placement constructed in the late 1970s and early 1980s in the United States are still very similar to the containments that were constructed 10–15 years earlier, before the direct application of ASME Code requirements. It is anticipated that future nuclear power plant concrete containments, if designed to the same construction and loading requirements as defined in the ASME B&PV Code Section III, Division 2, Subsection CC, will be similar to those currently in operation.

Most of the requirements discussed in this chapter are based on the knowledge and understanding of how the containment design and analysis have been performed in the existing operating reactors. Only a few containments have been actually stamped using Article CC-8000 of the code. The applicants for design certifications issued by the NRC for future reactor containments have all committed to use the code requirements, except as discussed in Standard Review Plan 3.8.1 and Regulatory Guide 1.136 (issued in 2007). Though requirement for analyzing the containments to withstand the impact of various types of aircrafts is under review by the NRC, some applicants have voluntarily taken actions to make the future reactor containments robust enough to withstand the impact from the most commercial aircrafts without loss of intended function of the containments.

The future reactor applicants are required to perform probabilistic risk assessments (PRAs) of the designed plants so that the consequences of severe accidents and mitigating characteristics of the containments could be quantified and compared against the safety goals established by the NRC [11, 12].

TABLE 14.8 RELATIVE COST COMPARISON BETWEEN CONVENTIONAL AND PASSIVE CONTAINMENT, CONTAINMENT COOLING, AND EMERGENCY CORE COOLING SYSTEMS AND EQUIPMENT SUPPORTS WITH CONVENTIONAL CONTAINMENT NORMALIZED TO 1.0

		Total Cost
I.	Conventional Containment	1.0 ⁽¹⁾
II.	Passive Containment with Radiator 88,850 yd ³ concrete 14,300 tons rebar 182,000 ft. 170 wire tendons 160,000 ft ² liner PL 1/4" 163,000 ft ² radiator PL 1" 45,600 yd ³ basalt rock 49,000 ft ² refractory brick	3.68
III.	Emergency Core Cooling System Conventional Containment (ECCS) Maintenance & ISI for 40 yrs.	1.20 <u>0.33</u>
TOTAL		1.53
	Passive Containment ECCS Maintenance & ISI 10/yr × 40 yrs.	0.49 <u>0.13</u>
TOTAL		0.62
DIFFERENCE		(0.91)
IV.	Containment Cooling System Conventional CCS Maintenance & ISI for 40 yrs.	0.50 <u>0.17</u>
TOTAL		0.67
	Passive Containment CCS Maintenance & ISI for 40 yrs.	0.15 <u>0.03</u>
TOTAL		0.17
DIFFERENCE		(0.5)
V.	NSSS Component Supports Conventional Containment Maintenance & ISI for 40 yrs.	0.21 <u>0.07</u>
TOTAL		0.28
	Passive Containment ECCS Maintenance & ISI 10/yr × 40 yrs.	0.11 <u>0.03</u>
TOTAL		0.15
DIFFERENCE		(0.15)
		Relative Cost
VI.	Cost Summary Conventional Containment Passive Contain with Radiator Conventional ECCS Passive ECCS Conventional CCS Passive (Radiator) CCS Conventional NSSS Supports Passive NSSS Supports	1.00 3.68 1.53 0.62 0.67 0.17 0.28 0.11
VII.	Cost Differentials Containment (3.68–1.00) ECCS (1.53–0.62) CCS (0.67–0.17) NSSS Supp. (0.28–0.11)	+2.68 –0.91 –0.50 –0.15
	(Net increased in plant cost over conventional containment design. 1.10 times conventional containment design.)	<u>1.13</u>

(1) Cost of conventional single barrier prestressed concrete 1,300 MWe PWR Containment. Assuming total plant costs in 2004 dollars is 2 billion, the conventional containment cost is $0.06 \times 2,000,000 = \$120,000,000.00$.

14.19 REFERENCES

1. Timoshenko, S. P., and Woinowsky, Krieger S. Theory of Plates and Shells, 2nd edition. New York: McGraw-Hill Book Co.; 1959.
2. ASCE Manual 58, *Structural Analysis and Design of Nuclear Plant Facilities*. New York: American Society of Civil Engineers; 1980.
3. NUREG/CR-5640, "Overview and Comparison of United States Commercial Nuclear Power Plants." U.S. Nuclear Regulatory Commission; 1990.
4. "Appendix VIII Physical Processes in Reactor Meltdown Accidents to Reactor Safety Study," WASH-1400 (NUREG - 75/014), U.S. Nuclear Regulatory Commission, October 1975.
5. Van Erp, J. B., "Containment Systems Aimed at Protecting Against the Consequences of Hypothetical Core-Melt Accidents," Trans. American Nuclear Society, 22, 468, (1975).
6. HSK-AN-4626, "Position of the Swiss Federal Safety Inspectorate Regarding the Safety of the Swiss Nuclear Power Plants in the Event of an Intentional Aircraft Crash," Swiss Federal Nuclear Safety Inspectorate, March 2003.
7. Design Criteria, Federal Republic of Germany Reactor Safety Commission (Proposed) BASF Power Station in Ludwigshaven on Rhine.
8. Simos, N., "Assessment of a Nuclear Power Plant Containment Structure Safety Related to an Aircraft Crash (DRAFT)," NATO Grant SST.CLG.978967, Brookhaven National Laboratory, May 2004.
9. ACI-ASME Joint Committee, ASME Boiler and Pressure Vessel Code, Section III-Division 2 and ACI Standard 359, 2001.
10. Stevenson, J. D., "Preliminary Design of a Containment to Withstand Core Melt for a 1300 MWe LWR System," Nuclear Engineering and Design, Vol. 48 (1978) North Holland Publishing Co.
11. NRC Policy Statement, "The Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," 60 FR 42622, August 16, 1995.
12. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water-Cooled Reactor Designs," ADAMS Accession No. ML003708021, April 2, 1993, and the related Staff Requirement Memorandum, ADAMS Accession No. ML003708056, July 21, 1993.
13. ACI 207.1R, Guide to Mass Concrete.

