CHAPTER

14

Code for Concrete Reactor Vessels and Containments

HANSRAJ ASHAR, BARRY SCOTT, JOSEPH F. ARTUSO and John D. Stevenson

14.1 INTRODUCTION

Insert A

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. Only one concrete reactor vessel has been built in the United States that followed the requirements contained in the current Subsection CB, and that 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.~~

Insert B to replace above 14.1.1

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 construction[1] 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 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 $M\ W_t$ light-water reactor nuclear power plants may use internal stream 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

[1] Typical 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 × 100 ft.^{3}	Height to Springline × 100 ft.	Diameter × 100 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:
(a) PWR I-SI: small prestressed concrete containment; see Fig. 14.1.
(b) PWR I and II, S and I: small and large deformed-bar concrete containment; see Fig. 14.2.
(c) PWR I-L: large prestressed concrete containment; see Fig. 14.3.
(d) PWR III-L: large deformed bar concrete containment; see Fig. 14.4.
(e) PWR III-L: large hybrid steel and concrete containment; see Fig. 14.5.
(f) BWR Mark I: steel containment; see Fig. 14.6.
(g) BWR Mark I: deformed-bar concrete containment; see Fig. 14.7.
(h) BWR Mark II: deformed-bar concrete containment; see Fig. 14.8.
(i) BWR Mark III: deformed-bar concrete containment; see Fig. 14.9.
(j) BWR Mark III: hybrid concrete and steel containment; see Fig. 14.10.

Notes:
(1) The small PWR deformed-bar reinforced-concrete containment is similar in appearance to a large deformed-bar reinforced-concrete containment, except smaller in size.
(2) 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 MW_e ± 25%; large = 1000 MW_e ± 25%.
(3) Only two all-concrete ice condensers were constructed, D.C. Cook Units 1 and 2.
(4) Only one Mark I concrete containment was constructed, Brunswick Station.

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.

Inserts C, D and E

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* ± 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.

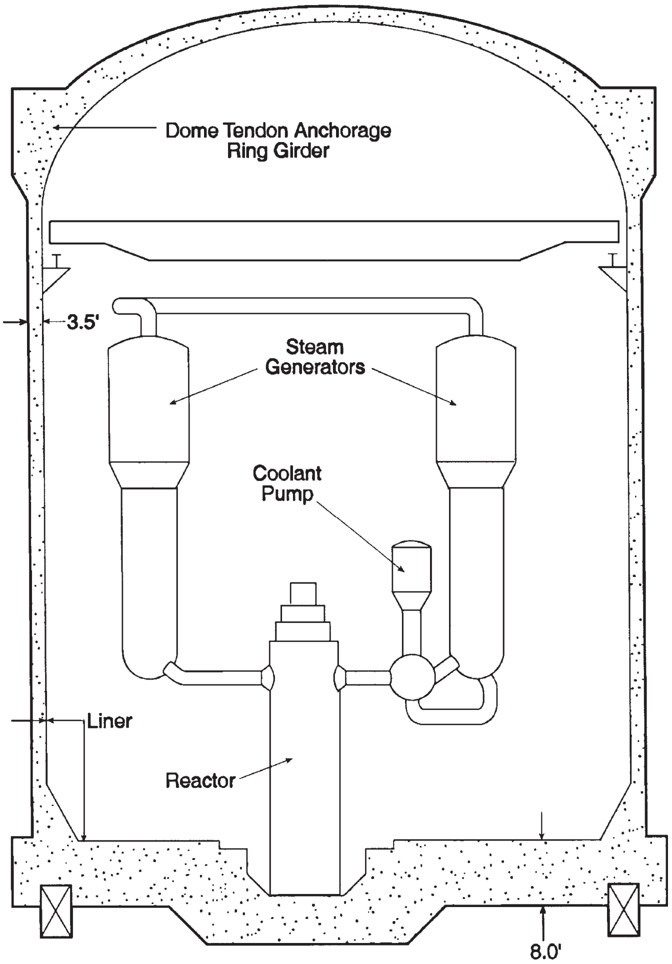

FIG. 14.1 SMALL PWR PRESTRESSED CONCRETE CONTAINMENT

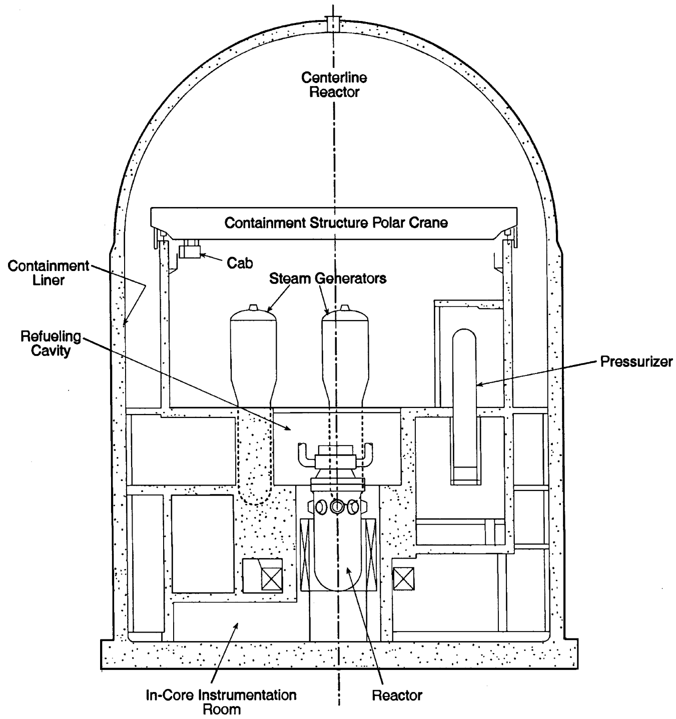

FIG. 14.2 LARGE OR SMALL PWR DEFORMED-BAR CONCRETE CONTAINMENT

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 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.

Most dry and reduced-pressure PWR concrete containment volumes vary from 1.5 to 3.2 million ft^3 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 $4\frac{1}{2}$ ft thick deformed bar or a $3\frac{1}{2}$ ft thick prestressed cylinder; and a $2-2\frac{1}{2}$ 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 $\pm$ 25% and 1,000 MWe $\pm$ 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

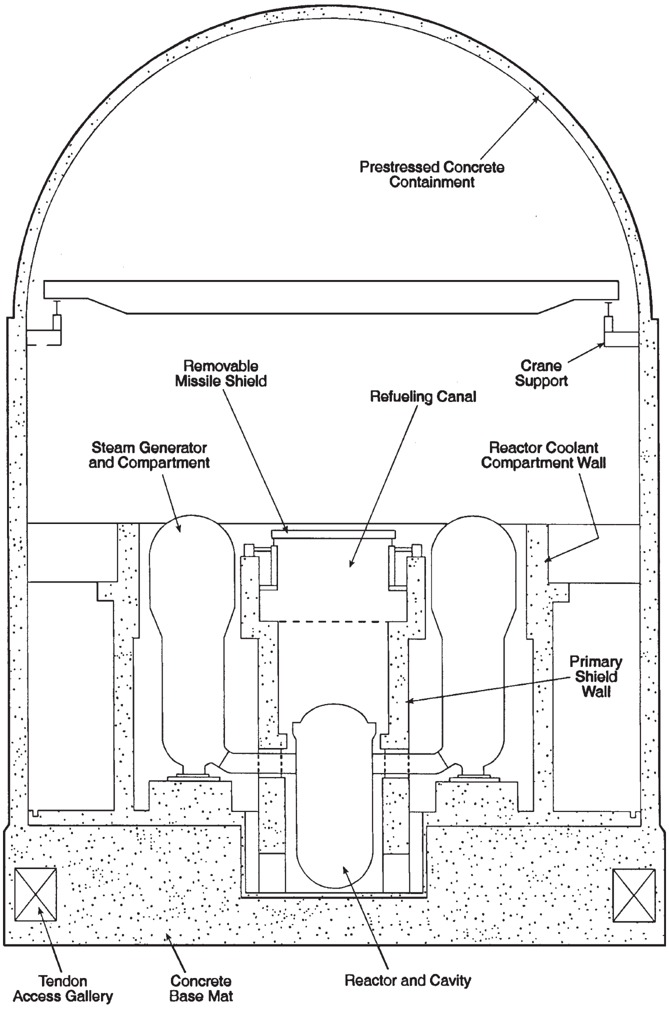

FIG. 14.3 LARGE PWR PRESTRESSED REINFORCED-CONCRETE CONTAINMENT

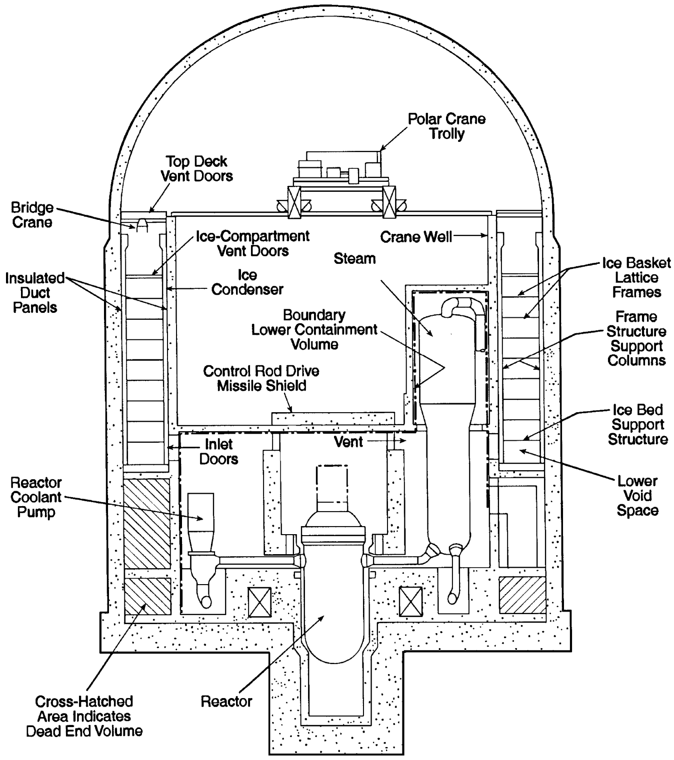

FIG. 14.4 PWR CONCRETE ICE CONTAINMENT

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 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.

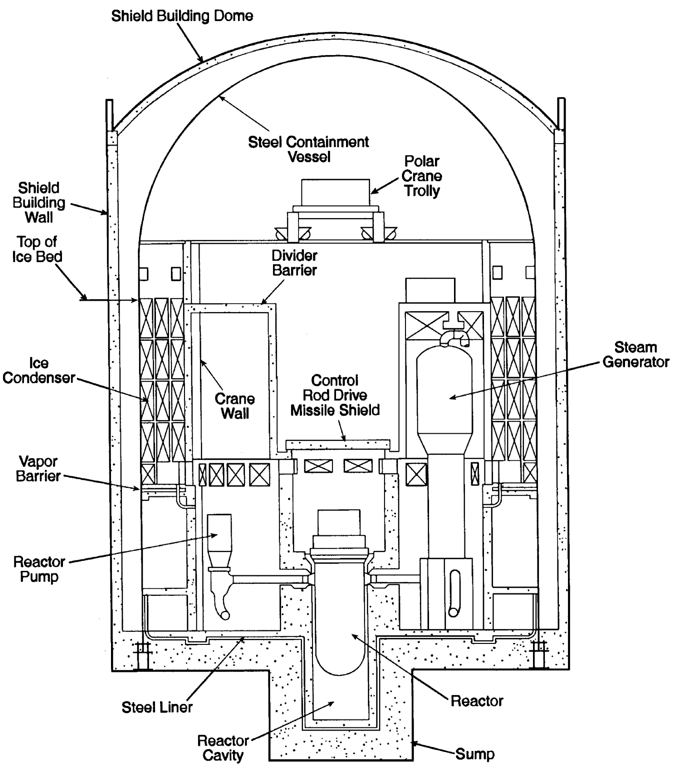

FIG. 14.5 PWR HYBRID ICE CONTAINMENT

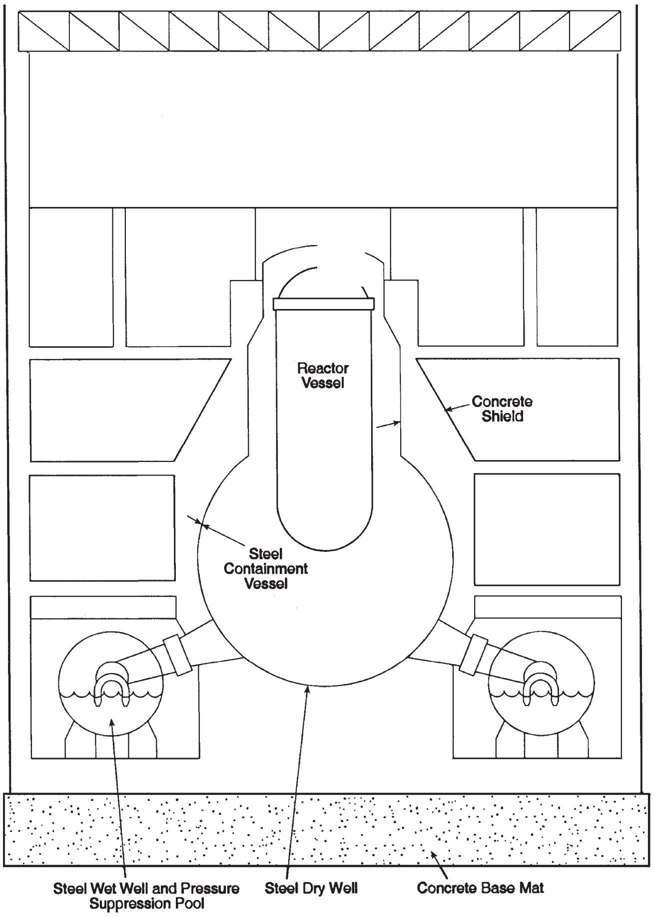

FIG. 14.6 BWR MARK I STEEL CONTAINMENT

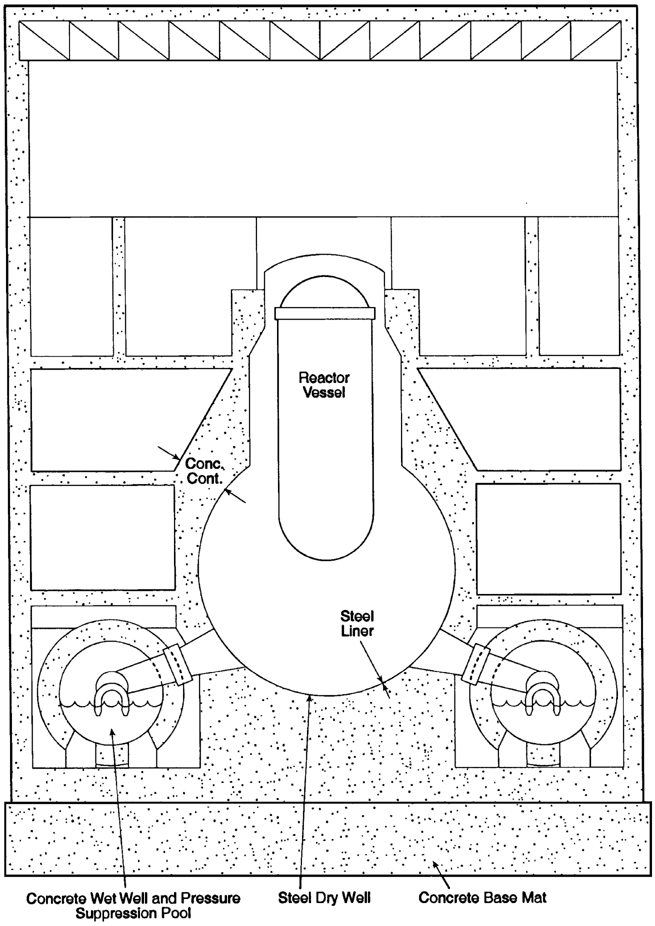

FIG.14.7 BWR MARK I CONCRETE WET-WALL CONTAINMENT

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 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 when the shear wave velocity of the supporting foundation material in 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.

Insert F

14.2 FUTURE CONTAINMENT DEVELOPMENT

Recent events have suggested that nuclear reactor containments in the future may be quite difrent 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~~A~~. Unique features of the containment would be large melting champers below the normal 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

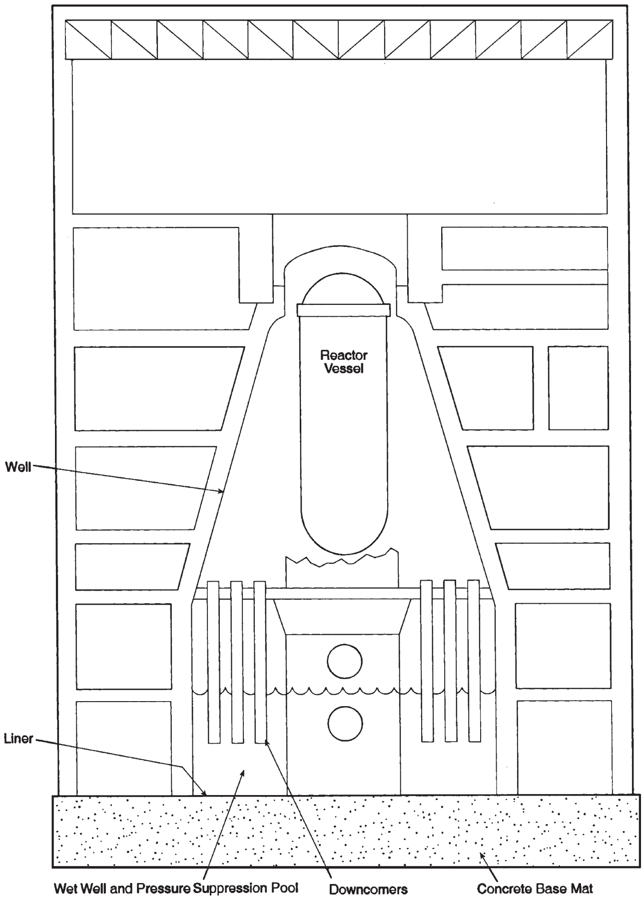

FIG. 14.8 BWR MARK II CONCRETE CONTAINMENT

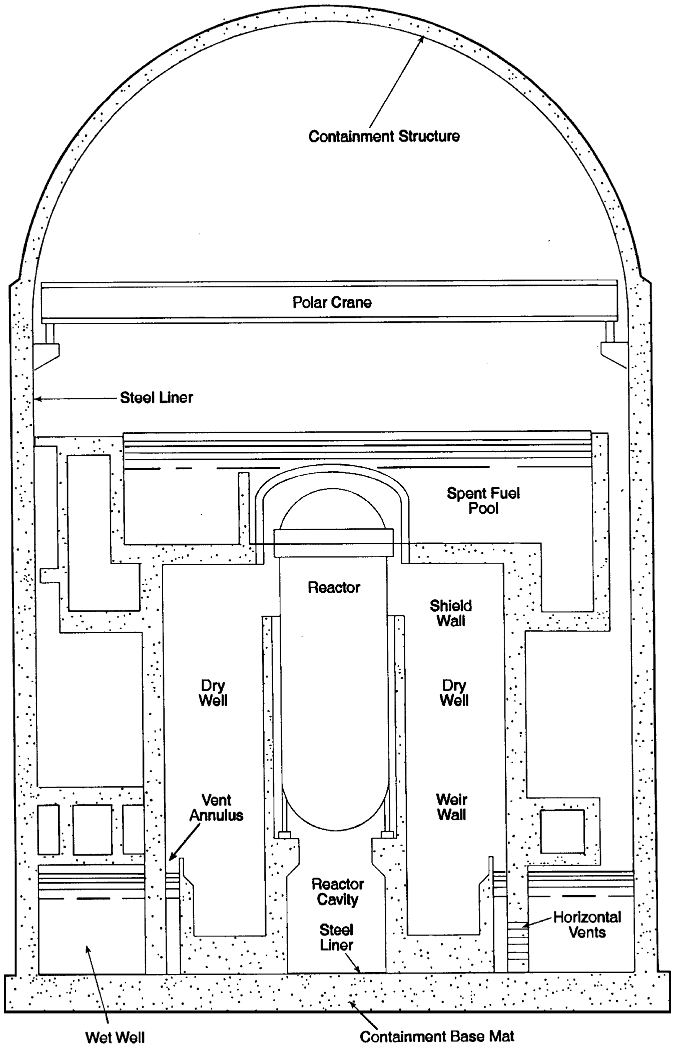

FIG. 14.9 BWR MARK III CONCRETE CONTAINMENT

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.

Inserts G and H

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_t) 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 318-63 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 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

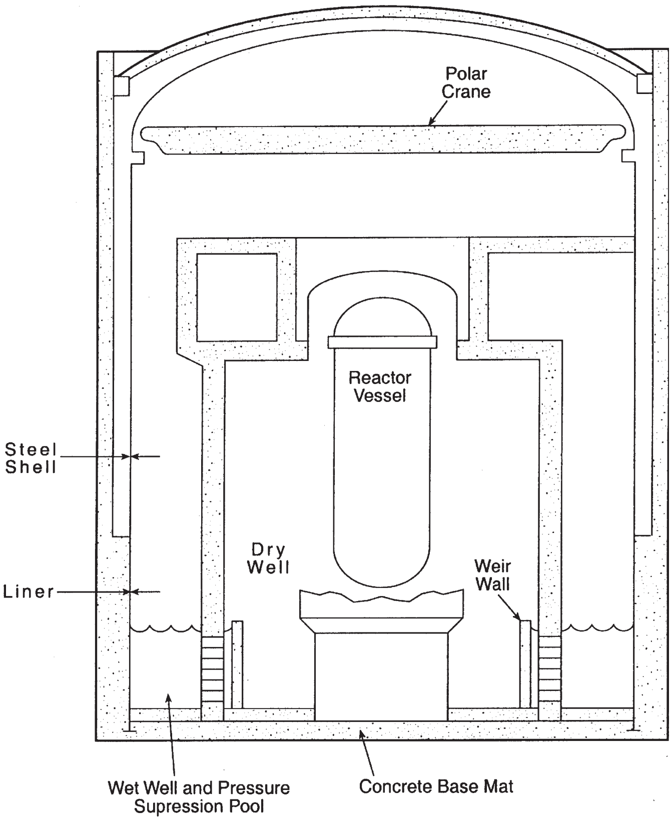

FIG. 14.10 BWR MARK III HYBRID CONTAINMENT

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 pipe[2]

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:

$$L_c = (D \pm 0.05D + W_t) \tag{14.4}$$

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

[2]Containment 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.

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).

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

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.

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.

Insert I

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.

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

[3] When 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.

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-axisymmet-ric 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].

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 $1.25P_a + 1.25E_o + D$.
(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);

[4]Earlier 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.

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

(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)

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 bythe 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_{jrj} = 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

[6]Normally controlled by required radiation protection shielding.

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 ~~installation~~ 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: Concrete Multiaxial Compressive Strength Modifications (primarily used for prestressed concrete reactor vessels, Subsection CB).[7]~~
(~~3~~ 2) Appendix ~~III~~ II: Glossary of Terms and Nomenclature.
(~~4~~ 3) Appendix ~~IV~~ III: Approval of New Material.
(~~5~~ 4) Appendix ~~VI~~ IV: Rounded Indications.
(~~6~~ 5) Appendix ~~VII~~ V: Qualifications of Concrete Inspection Personnel.
(~~7~~ 6) Appendix ~~IX~~ VI: Nondestructive Examination Methods.
(~~8~~ 7) Appendix ~~X~~ VII: ~~Preparation~~ Submittal of Technical Inquiries to the Boiler and Pressure Vessel Committee.
(~~9~~ 8) Appendix ~~XI~~ VIII: Qualifications for Arc Welding Reinforcing Bars.

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

~~(1) Appendix B: Design Specification (to Section III).~~
~~(2) Appendix C: Design Report (to Section III).~~
(~~3~~ 1) Appendix ~~C~~ A: Typical CRV Load Combinations (~~applicable to concrete reactor vessel~~ optional).
(~~4~~ 2) Appendix ~~D~~ B: Non-Mandatory Preheat Procedures.
(~~5~~ 3) Appendix ~~E~~ C: Certification of Level I and II Concrete Inspection Personnel.
(~~6~~ 4) Appendix ~~F~~ D: ~~Liner Dimensional Tolerances.~~
(~~7~~ 5) Appendix ~~G~~ E: Certified Material Test Reports for Liner Materials.
(~~8~~ 6) Appendix ~~H~~ F: Reinforcement Fabrication and Placing Tolerances.

~~Appendix B, Design Specification to Section III, is also applicable to Subsection CC. The content of Appendix B that is applicable to concrete containment is summarized in the following paragraphs.~~

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 1 construction or~~ 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 ~~CC for Concrete Containment~~ 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 reactor vessels or~~ 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.
(8) Identification of
(7) Loads from internal structures~~, identifying~~ those components and/or parts that require a preservice examination ~~and~~ including the following:
 (a) edition and addenda of ASME B&PV Code Section XI to be used;
 (b) category and method; ~~and~~
 (c) qualifications of personnel, procedures, and equipment~~.~~; and
(~~8~~ 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. ~~Appendix B of Section III contains guidance regarding the content and organization of the Design Specifications~~.

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 calculations 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. ~~Appendix C of Section III contains guidance regarding the content and organization of the Design Report.~~

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;

~~[7]Subsection CB for concrete reactor vessels continues to be published with new Code edition; however, it is no longer being maintained by the joint ACI-ASME Code Committee.~~

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)] (~~CB-2130,~~ CC-2130)	M	C, F	M	—	C, F, O	I, J, L, D
Ship 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~~ -	~~—~~ O, I	~~O, I~~ -	~~—~~ J	~~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.

(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.

Insert J

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.

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~~14.10.6 Article CC-2000~~

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

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.

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)]
		Shear	Primary
	Volume change effects such as creep shrinkage and thermal strains	Membrane	Secondary
		Bending	Secondary
		Shear	Primary
Regions near large openings	External [Note (2)]	Membrane	Primary
		Bending	Primary
		Shear	Primary
	Volume change effects such as creep shrinkage and thermal strains	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.

14.10.7.3 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.

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 subarticle 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.

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14.11 ARTICLE CC-4000: FABRICATION AND CONSTRUCTION

~~The general and specific construction or placement requirements are given in this article. The fabrication requirements for steel reinforcement systems are also covered. This article is unique in ASME B&PV Code Section III, Division 1, in that Construction Specifications are required. In the Construction Specifications, the designer has the ability to define construction and fabrication parameters and procedures that supplement the requirements contained in Article CC-4000 but that are not intended to change the requirements of CC-4000.~~

14.12 ARTICLE CC-5000: CONSTRUCTION TESTING AND EXAMINATION

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~~This article lists all the tests, test frequencies, examination procedures, and qualification and certification of nondestructive examination personnel either directly or by references, which are requirements that are applicable to all parts of the containment during construction.~~

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 stamping
(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 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

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)

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

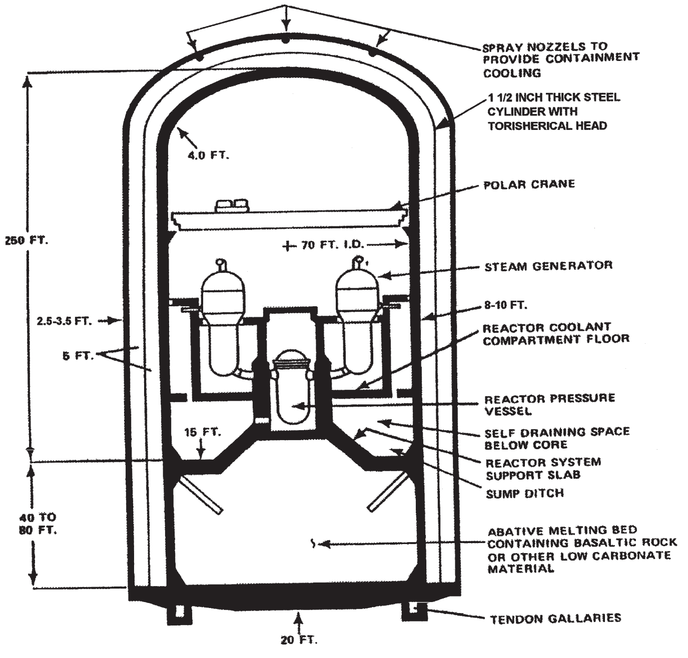

FIG. 14.11 PASSIVE DRY CONTAINMENT STRUCTURE USING RADIATOR CONCEPT FOR CONTAINMENT COOLING SYSTEMS FOR PWR

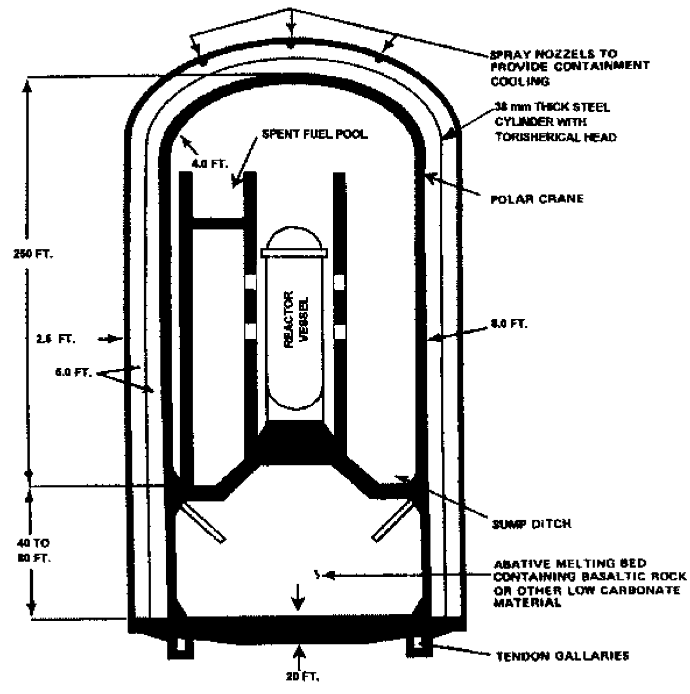

FIG. 14.12 PASSIVE DRY CONTAINMENT STRUCTURE USING RADIATOR CONCEPT FOR CONTAINMENT COOLING SYSTEMS FOR A BWR

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 moltant 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		

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. 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 CO_2 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 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

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

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 $\times 10^6$ 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

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	3.68
	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	
III.	Emergency Core Cooling System	
	Conventional Containment (ECCS)	1.20
	Maintenance & ISI for 40 yrs.	0.33
TOTAL		1.53
	Passive Containment ECCS	0.49
	Maintenance & ISI 10/yr × 40 yrs.	0.13
TOTAL		0.62
DIFFERENCE		(0.91)
IV.	Containment Cooling System	
	Conventional CCS	0.50
	Maintenance & ISI for 40 yrs.	0.17
TOTAL		0.67
	Passive Containment CCS	0.15
	Maintenance & ISI for 40 yrs.	0.03
TOTAL		0.17
DIFFERENCE		(0.5)
V.	NSSS Component Supports	
	Conventional Containment	0.21
	Maintenance & ISI for 40 yrs.	0.07
TOTAL		0.28
	Passive Containment ECCS	0.11
	Maintenance & ISI 10/yr × 40 yrs.	0.03
TOTAL		0.15
DIFFERENCE		(0.15)
		Relative Cost
VI.	Cost Summary	
	Conventional Containment	1.00
	Passive Contain with Radiator	3.68
	Conventional ECCS	1.53
	Passive ECCS	0.62
	Conventional CCS	0.67
	Passive (Radiator) CCS	0.17
	Conventional NSSS Supports	0.28
	Passive NSSS Supports	0.11
VII.	Cost Differentials	
	Containment (3.68 − 1.00)	+2.68
	ECCS (1.53 − 0.62)	−0.91
	CCS (0.67 − 0.17)	−0.50
	NSSS Supp. (0.28 − 0.11)	−0.15
(Net increased in plant cost over conventional containment design. 1.10 times conventional containment design.)		1.13

(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 × 2,000,000 = $120,000,000.00.

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.

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~~14.17~~ 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.

Insert O:

14.18 REFERENCES

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