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Recent advances in concrete containment vessels in Japan

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An outline of the development of concrete containment vessels in Japan for use in nuclear power plants is described, where the emphasis is laid on the reinforced concrete containment vessel (RCCV) recently developed for the advanced boiling water reactor (ABWR). Also explained are the salient features of concrete containment vessel design which are unique in Japan; namely stringent seismic requirements, thorough verification and novel containment concept. Finally the design principles applied to RCCV are presented along with the design standard.

1. Introduction

A majority of nuclear power plants in Japan presently in operation, under construction and in preparation belong to light water reactors (LWR). Namely 52 units out of 53 units are LWR's, which consist of 23 units of pressurized water reactors (PWR) and 29 units of boiling water reactors (BWR) [1].

As can be seen from table 1, the prestressed concrete containment vessel (PCCV) was first adopted in PWR's for the 1160 MWe Tsuruga Power Station Unit 2 which started operation in 1987, and since then all the 1100 MWe class PWR's built in Japan have employed PCCV. In BWR's the RCCV was first introduced into the 1356 MWe ABWR Kashiwazaki-Kariwa Power Station Unit 6 and 7 (K6/7) which are expected to begin their operation in 1996 and 1997 respectively. The use of RCCV for ABWR is one of the unique features of ABWR by which a significant improvement is expected to be made to the operability, safety and economy of BWR. The development work for ABWR was completed in 1985 and it is anticipated that the future BWR type power reactors will be dominated by ABWR.

Now it must be mentioned that there are some unique features to be noted in the concrete containment vessel development in Japan. The salient features among them are as follows:

- *Stringent seismic requirements*
- *Thorough verification* by a series of extensive experimental studies
- *Novel containment concept* building/containment (RCCV) integrated structure for ABWR.

In this paper the presentation is centered on these three important features. Finally a brief explanation is given of the design principles and standard relevant to concrete containment vessel.

2. Seismic requirements

2.1. Basic requirements

2.1.1. Basic principles

An overview of the seismic design procedure for nuclear power plants in Japan is shown in fig. 1. In general, seismic design is conducted in accordance with the following sequence in such a way as to conform to the regulatory guides and standards [2–6].

(1) Preparation of basic information, including identification of the earthquakes to be considered in design,
(2) Estimation of earthquake ground motions and input motions induced by the earthquakes thus identified,
(3) Estimation of seismic forces acting on the plant by seismic response analysis and static seismic requirement,

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Table 1
Concrete containment vessels in Japan

Type of CCV	Reactor Type	Power Company	Plant	Output (MWe)	Techical Standard	Starting of Operation (Scheduled)
PCCV	PWR	Japan Atomic Power Company	Tsuruga Unit 2	1160	Technical Standard of Prestressed Concrete. Containment Vessels for Nuclear Power Plant	Feb.,1987
		Kansai Electric Power Company	Ohi Unit 3	1180	(Permission of Particular Design by MITI Ordinance No.62, Article 3)	(Dec.,1991)
			Ohi Unit 4	1180		(Feb.,1993)
		Kyusyu Electric Power Company	Genkai Unit 3	1180		(Mar.,1994)
			Genkai Unit 4	1180	Technical Standard for Concrete Containment Vessel for Nuclear Power Plant (MITI Notification No.452)	(Jul.,1997)
RCCV	BWR (ABWR)	Tokyo Electric Power Company	Kashiwazaki -Kariwa Unit 6	1356		(Jul.,1996)
			Kashiwazaki -Kariwa Unit 7	1356		(Jul.,1997)

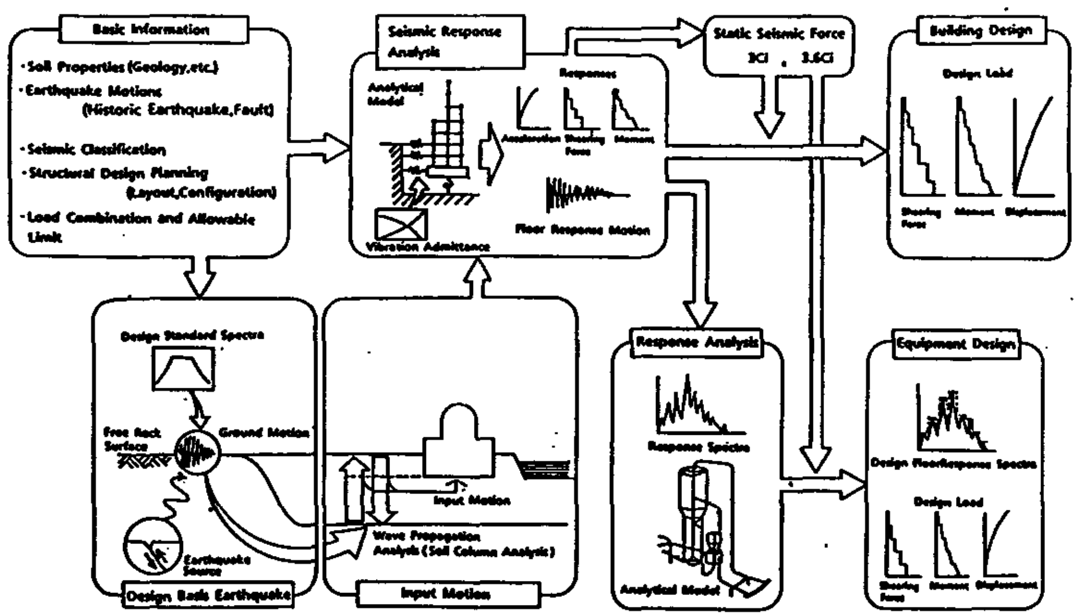


Fig. 1. Flow of seismic design procedures for nuclear power plants in Japan.

(4) Estimation of stresses, strains, deformations, etc. resulting from seismic forces,
(5) Finally review of structural integrity and safety function of the plant in light of acceptance criteria.

In practice, all the plant items are first classified into three categories, A, B and C according to their importance. There is one more class, that is As, on which the more stringent requirements are imposed in addition to the requirements as A. Containment vessels belong to this As class. In principle, using the design basis earthquake ground motions, a dynamic analysis is performed for class A items to obtain the seismic forces other than the static analysis. A time dependent seismic response analysis technique is usually employed for the analysis of buildings and structures, but this technique is used for some of the major equipment and piping as well. A static analysis is required for all classes, and the intention is to determine the minimum seismic forces (so-called "seismic floor") to be taken into account in the design, on the basis of the requirements set forth in the Building Standard Law, Building Standard Law Enforcement Order, Notifications of Ministry of Construction and relevant regulations (hereafter referred to as "Building Standard Law").

It can be said that there are three main features in the seismic requirements and practices prevailing in Japan. They are S_1 and S_2 design basis earthquake ground motions, seismic classification and static analysis requirement to arrive at design seismic forces.

2.1.2. Design basis earthquake ground motions

Presented herein is an outline of how the design basis earthquake ground motion is defined in Japan. Figure 2 shows the locations of nuclear power plants in Japan and the two levels of the maximum accelerations of design basis earthquake ground motions S_1 and S_2 employed for these plants. Although the maximum acceleration is not a good measure of the damage potential of earthquake ground motions, they are shown here as an indicator of the seismic intensity at each site. It is seen that they range from 180 to 450 gal for S_1 and 270 to 630 gal for S_2. The Japanese Guide [3] (hereafter referred to as "Guide") requires that the design basis earthquake ground motions be classified into S_1 and S_2 as described below. Ground motion S_1 is induced by the S_1 design earthquake that is the maximum design earthquake thought probable to occur, and ground motion S_2 is induced by the S_2 design earthquake that is the extreme design earthquake thought possible to occur.

According to the Guide, the design basis earthquake ground motions are defined as the ground motions at the free surface of the base stratum of a site. The Guide also says that "the free surface of the base stratum" is a nearly flat surface of the base stratum extending over a considerable area, and above which neither surface layers nor structures are assumed to be present. The base stratum is firm bedrock which was formed in general in the Tertiary or earlier era and

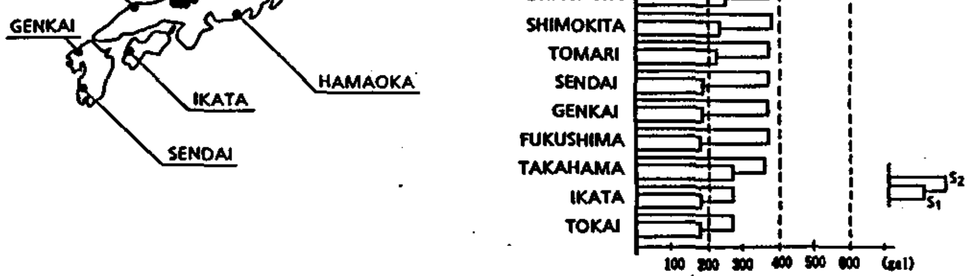


Fig. 2. Design earthquake acceleration levels in Japan (gal: unit of acceleration, cm/s^2).

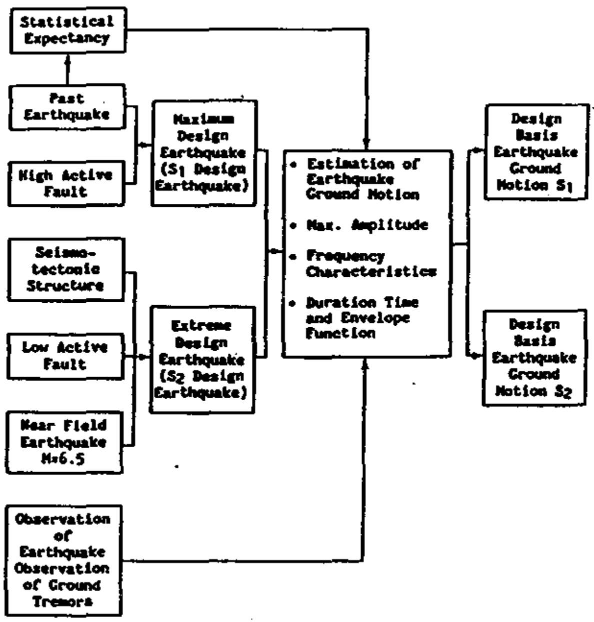


Fig. 3. Flow chart for determining design basis earthquake ground motions S_1 and S_2.

which is not significantly weathered nor fissured. In Japan bedrock is, in general, considered to exhibit a shear wave velocity greater than 700 m/s.

Figure 3 shows how the design basis earthquake ground motions S_1 and S_2 be established in Japan for use in nuclear power plant design. First of all, it is required to determine the S_1 and S_2 design earthquakes which give the design motions S_1 and S_2. An S_1 earthquake is determined primarily on the basis of the records of historic earthquakes and highly active faults. Statistical expectancy based on the records of historic earthquakes is also taken into account in estimating the intensity of S_1 earthquake motions. An S_2 earthquake is determined on the basis of seismo-tectonic structure at a site region and the active fault with relatively low activity. In addition an earthquake of magnitude 6.5 occurring directly underneath the site must be assumed to occur as the S_2 design earthquake.

The earthquake ground motions are characterized by the maximum amplitude, frequency characteristics, duration time and time-dependent variation of amplitude envelope curve. Based on the S_1 and S_2 design earthquakes, these parameters can be determined. Using these information, thus the design basis earthquake ground motions S_1 and S_2 are established in the form of design spectra and synthesized ground motions.

2.1.3. *Seismic classification and design seismic force*

Table 2 gives the definition and examples of seismic classification. It is a mandatory requirement in Japan that all the plant items be classified into the three categories A, B and C in accordance with their importance in terms of the public safety. Classified as class A are the items containing or related to highly radioactive material, and whose loss of function might lead to the release of radioactive material to the atmosphere, and such items required to protect the public from nuclear hazards in the event of a nuclear accident. Essential items among class A items such as reactor containment, shutdown devices, and primary coolant system are classified as As. Such items related to radioactive material but having relatively minor effectiveness except those classified as class A are classified as class B. Class C items are those not classified as

Table 2
Seismic classification

Seismic Classification	Definition	Example
Class As	Facilities extremely essential to plant safety among Class A items	Reactor containment, Reactor coolant pressure boundaries, Core shutdown system, etc.
Class A	Facilities important to plant safety or related to radioactive material	Reactor auxiliary building, Emergency core cooling system, Emergency off-gas system, etc.
Class B	Same as Class A but whose rupture might lead to less serious consequences	Turbine Bldg. (BWR), Rad-waste treatment system, etc.
Class C	Facilities not classified as A or B, and the same degree of safety as ordinary industrial facilities	Turbine Bldg. (PWR), Turbine Generator, etc.

Table 3
Design seismic forces

Seismic Classification	Static		Dynamic	
	Horizontal	Vertical	Horizontal	Vertical
Class As	—	—	S_2	V_2
Class A	3Ci	1.0Cv	S_1	V_1
Class B	1.5Ci	—	—	—
Class C	1.0Ci	—	—	—

1) S_2, S_1 : Dynamic forces derived from design basis earthquake ground motions S_2 and S_1
2) V_2, V_1 : Uniform Vertical forces based on 1/2 maximum acceleration amplitude (gal) of S_2 and S_1 motions divided by acceleration of gravity (980)
3) C_i : Shearing force coefficient to be determined from the standard shearing coefficient of 0.2 and other considerations such as response characteristics of building and soil
4) C_V : Vertical seismic coefficient of 0.3 and is uniform value irrespective of height
5) For equipment and piping, the above static value must be multiplied by a factor of 1.2

Table 4
Load combinations and allowable limits – Basic principles

Seismic Classification	Load Combination									Analysis			Allowable Limit		
	Operational	Accidental	Seismic												
			Horizontal				Vertical								
			Static		Dynamic										
	D-O	L	Ci	Q	S_1	S_2	Cv	$S_1/2$	$S_2/2$	Linear	Non-Linear	Limit Analysis	Elastic Limit	Adequate Margin to Ultimate State	Ultimate Strength
As	1					1			1		○			○	
As	1	1			1			1			○			○	
A	1				1			1		○			○		
A	1		3*				1			○			○		
A	1			(1)			1					○		○	
A	1	1									○			○	
B	1		1.5*							○			○		
B	1				[1/2]					○			○		
B	1			(1)								○		○	
C	1		1*							○			○		
C	1			(1)								○			○

Note,
* Multiplied by 1.2 for equipment and piping.
( ) Applicable to building and structure only.
[ ] Required in case of resonant vibration only.
Ci Story shear coefficient
Cv Vertical seismic coefficient
Q Required horizontal ultimate strength

class A and B, which are only required to maintain the same degree of safety as ordinary industrial facilities.

Table 3 shows the relation between design seismic forces and seismic classification. It is required according to the Guide that all the class A items including class As items be designed to the design basis earthquake ground motions S_1, while only the class As items are required to be designed to the design basis earthquake ground motion S_2. The basic concept behind the use of two levels of design earthquake S_1 and S_2 is that a nuclear power plant must remain intact and can continue its operation during and after the maximum design earthquake S_1 which is thought probable to occur, in addition to maintaining its safety function during and after the extreme design earthquake S_2 which is thought possible to occur.

It will be noticed that the static forces are larger in the order of importance, namely in the horizontal direction 3, 1.5 and 1.0 for A, B and C respectively. It will also be seen that the seismic forces in the vertical direction are only required for class A, and no dynamic analysis is presently required in the vertical direction. It should also be pointed out that the static forces are 20% larger for equipment and pipings than for buildings and structures. This is because of the consideration for the minimum response amplification relative to buildings.

2.1.4. Acceptance criteria for seismic qualification

The acceptance criteria for seismic qualification of nuclear power plants in Japan are outlined with emphasis placed on load combinations and allowable limits. Table 4 presents the basic principle of load combinations and allowable limits prevailing in Japan. Basically it is required for class A items to take into account an occurrence of the maximum design earthquakes S_1 under normal or upset condition. For class As items such as containment vessels, It is further required that a simultaneous occurrence of the design accident and maximum design earthquake S_1 be considered although it is a very remote probability, in addition to a combination of normal or upset condition plus an occurrence of the extreme design earthquake S_2.

In principle the class A items are required to remain elastic under S_1 loading condition, and the details are stipulated in the Japan Electric Association's Technical Guide [6] to meet this intent in accordance with the stress category in the case of equipment and piping for example, such as primary stress, secondary stress and local stress, while the newly established MITI Notification [7] is applied to concrete containment vessels.

In the case of class A and B buildings and structures, it is required to follow the allowable limits for short term loading stipulated in the Building Standard Law, if they need to remain elastic. Although it is allowed for the most essential As items to exceed elastic limits, they are required to possess a sufficient margin for deformation capability and a certain appropriate margin against ultimate state or strength for the sake of retention of safety function of a plant.

2.2. Requirements for concrete containment vessels

Seismic forces acting on concrete containment vessels are estimated in accordance with the above-mentioned seismic design sequence, and in the case of PCCV for a typical recent PWR plant, its response at the top of the dome is obtained at approximately 2700 gal and over 3000 gal under the S_1 and S_2 earthquake intensity level of 365 gal and 532 gal respectively (gal: unit of acceleration, cm/s^2). The maximum story shear coefficient at the bottom of containment vessel under the above condition is estimated at 1.31 and 1.57 [8].

In the case of RCCV's, since the center of gravity of deeply embedded containment/building structure is fairly lowered as compared to other LWR's, the design is dictated by static forces rather than dynamic forces.

It can be noted from the above examples that the stringent seismic requirements were one of the incentive to employ PCCV's in the case of PWR's, and that the use of RCCV/building combined structure has an advantage in terms of seismic resistance capability. Upon the introduction of PCCV's and development of RCCV's, for the purpose of proving the seismic qualification and appropriateness of design approaches an extensive research and development works were carried out as mentioned hereafter in section 3.

3. Verification by research and development

3.1 Prestressed Concrete Containment Vessel (PCCV)

Upon construction of the Tsuruga Power Station Unit 2 which is a 1160 MWe class PWR, the Japan Atomic Power Company (JAPC) decided to employ PCCV based on the comparison of four different type of containment vessel; ordinary steel spherical vessel, high tensile steel cylindrical vessel with ice-condenser, RCCV and PCCV. The reason for selecting PCCV was that PCCV has been widely used in the USA and

Table 5
Development of PCCV in Japan

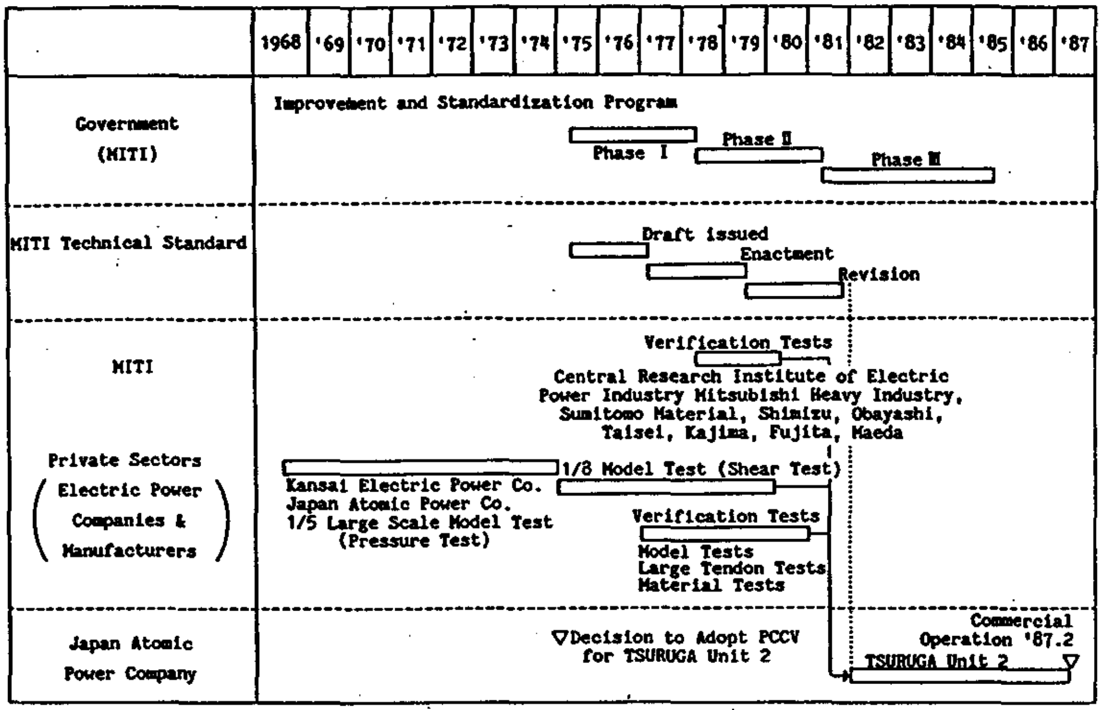


Table 6
Experimental studies for the development of PCCV

	Item	Outline
MITI Verification Tests	In-plane Shear	• In-plane shear tests of RC plate, RC & PC cylindrical model and PC cylindrical model with dome, thus evaluating ultimate in-plane shear strength.
	Out-of-plane Shear	• Push-off tests of RC blocks, thus investigating shear transfer mechanism and evaluating shear strength. • Internal pressure tests of RC cylindrical wall, thus evaluating ultimate out-of-plane shear strength of base of cylindrical wall based on resistance of circumferential rebar reinforcement.
	Thermal Stress	• Thermal stress tests of RC beams, thus studing reduction ratio of thermal stress of RC beam. • Internal pressure + thermal load tests of base of cylindrical wall, thus confirming that ultimate shear strength of the base is not affected by thermal stress.
Large Capacity Tendon Tests	Friction Loss, etc.	• Stressing tests of 1000 ton class tendons by full scale partial model, thus investigating friction loss coefficients and constructability.
Material Tests	Concrete Properties	• Concrete properties test on actual concrete for Tsuruga 2, thus investigating suitable concrete mixture, creep properties and thermal properties of concrete.

Europe and it has some advantages over the others in terms of seismic resistance capability [9].

Although over 100 PCCV's have been completed or are under construction in the world, a number of verification studies have been conducted in Japan in addition to the establishment of standard because PCCV is the first large structure of its kind in Japan. These studies include optimization of structure (shape of dome, buttress, tendon capacity, bonding, etc.), conceptual design and numerous verification tests. Indicated in table 5 is the schedule of PCCV development activities.

The Ministry of international Trade and Industry (MITI) organized a committee in 1975 for the establishment of technical standard of concrete containment vessel for nuclear power plants, which looked into the relevant standards in Japan as well as the ASME code and others. When the first draft of the standard was prepared in 1977, the following two comments were made by the committee.

(1) Lack of studies both in Japan and overseas regarding the in-plane and out-of-plane shear stress when the containment vessel is subjected to a combined stress of membrane force resulting from internal pressure and shear force.
(2) Necessity for studying the method of evaluating stiffness in estimating thermal stress.

On the basis of this review, the decision was made to perform verification tests as part of the MITI LWR standardization activities to confirm the requirements for evaluation of out-of-plane and in-plane shear stress and thermal stress [10]. In addition electric utilities with assistance from industries carried out various verification tests using models subjected to loads such as

Table 7
Major parameters of PCCV

	Plant Name	Tsuruga Unit 2	Ohi Unit 3&4	Genkai Unit 3&4
Configuration	Shape of CCV	Cylindrical Shell with Spherical Dome		
	Cylinder, Thickness of wall Height(Internal) Diameter Dome, Thickness Height (Internal) Radius(Internal)	1.3 m 43.0 m 43.0 m 1.1 m 22.6 m 21.5 m		
Design Criteria	Design Pressure	4.0 kg/cm^2		
	Test Pressure	4.6 kg/cm^2		
	Design Temperature	144 °C		
Materials	Specified Design Strength of Concrete	Fc=420 kg/cm^2	Fc=450 kg/cm^2	Fc=420 kg/cm^2
	Reinforcing Bar	SD40, D51		
	Liner	6.4 mm		
PS System	Anchorage	Unbonded type (Grouting with grease)		
	Tendon	Unbonded type		
	Buttress	3 - Buttress	2 - Buttress	Unit 3 3-Buttress Unit 4 2-Buttress
	Tendon Layout	Cylinder : Hoop, Dome : 3 Ways reversing U, Hoop (partially)		
	Tendon Composition	163wires of 7mm	55 strands of 12.5mm	163 wires of 7mm
	Tendon Capacity	1,000 tons class		

internal pressure, temperature and horizontal force, and experimental studies on large capacity tendons (see table 6) [11,12].

In light of the outcome of the above-mentioned studies, "MITI Technical Standard (Tentative)" was established in 1979 which was then revised in 1981 [13]. Construction permit has been issued to the Tsuruga Unit 2, Ohi Unit 3&4 and Genkai Unit 3 after submitting the PCCV Technical Guideline which was prepared for each plant-by-plant basis in accordance with the Technical Standard.

Major parameters of the PCCV's are given in table 7. It is obvious from this table that they are identical in terms of dimension and design requirements except for minor differences in construction material and prestressing system.

In-Service Inspection (ISI) has been conducted for the Tsuruga Unit 2 one and three year after the start of operation, and its structural integrity has been confirmed by checking prestressing force of tendon, anchorage, grease, concrete, etc.

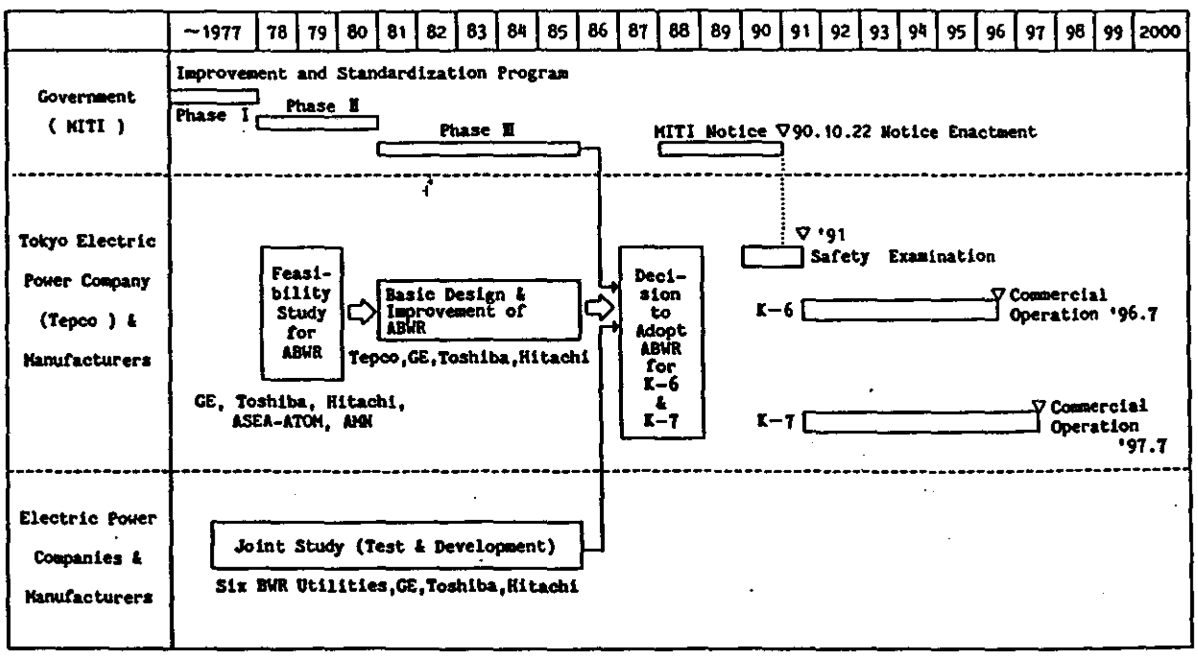
3.2. Reinforced Concrete Containment Vessel (RCCV)

A feasibility study has been undertaken in 1978 by the world's five BWR manufactures (GE, Toshiba, Hitachi, ASEATOM, ANSALDO) aiming at developing an advanced version of the BWR (ABWR) on the basis of technologies of conventional BWRs. Since that time numerous studies have been conducted, i.e. 3rd Phase LWR Improvement and Standardization by the MITI, Conceptual Design by the Tokyo Electric Power Company (TEPCO) and Industries, Structural Evaluation and Verification Tests for Establishment of Standard by the Electric Utilities/Industries Joint Study. Consequently ABWR has been successfully developed and adopted for TEPCO's Kashiwazaki-Kariwa Unit 6 and 7, which are deemed to be equivalent to the MITI's 3rd phase standard LWR. Listed below are the main features of ABWR [14].

(1) Adoption of an internal pump for reactor coolant recirculation, thereby eliminating the outer pump.

(2) Adoption of a reinforced concrete containment

Table 8
Development of ABWR



K-6 : Kashiwazaki-Kariwa No.6
K-7 : Kashiwazaki-Kariwa No.7

vessel which is structurally combined with the reactor building.
(3) Adoption of improved control rod drive mechanism.
(4) Adoption of improved reactor core.
(5) Scaling up of plant output by the use of high-efficiency turbine, etc.

As part of the end-product of this development, a cylindrical RCCV was developed as listed in the above (2) which is integrated with the building to make a single combined structure having the following advantages:
(1) Freedom in shape, which leads to a reasonable shape meeting the equipment and piping layout requirements.
(2) Ideal hybrid structure with steel liner functioning as leak-proof membrane and concrete as pressure sustaining structure, shielding and seismic wall of reactor building.
(3) Smaller size of RCCV and lower gravity center, thus enhancing seismic resistance capability.
(4) Shortening of construction period.

Table 8 shows the schedule of RCCV development activities [14]. Since RCCV is the first structure of its kind newly adopted in Japan, a trial design was performed in accordance with the above-mentioned Technical Standard in addition to a series of verification tests as the joint effort of BWR utilities aiming at the structural evaluation and establishment of the reinforced concrete containment vessel. From the structural viewpoint, there were three outstanding aspects to be investigated as follows:
- The RCCV structure is under a complex combined stress condition resulting from a tensile membrane stress by internal pressure and stresses by other loads.
- The RCCV top slab is stiffened by pool girders.
- The RCCV is combined with the building through building slabs.

Accordingly the following experimental studies were conducted.
(1) Basic design data has been accumulated on the shear strength under an axial tensile state by experimental means using beam and disc models. The results thus obtained were reflected upon the MITI Notice 452 along with the knowledge derived from the previous shear tests.
(2) A partial model of top slab was tested to failure by internal pressure, thereby confirming the ample margin for ultimate strength (maximum pressure at failure was four times the design pressure.)
(3) An entire scale model of the RCCV/building slab

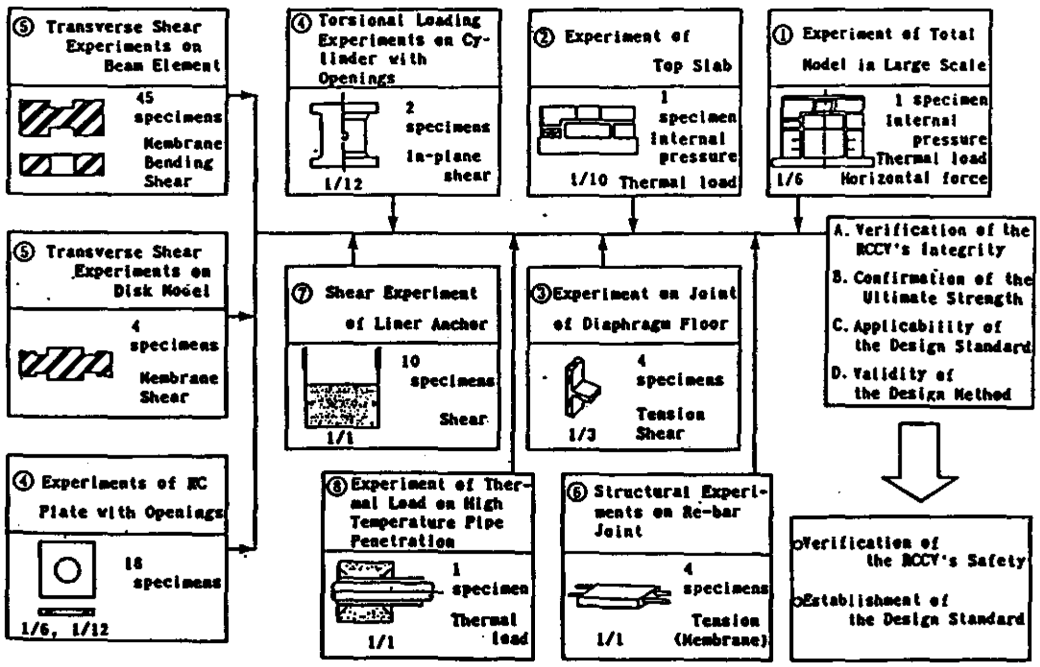


Fig. 4. Flow chart of experimental study for RCCV structure.

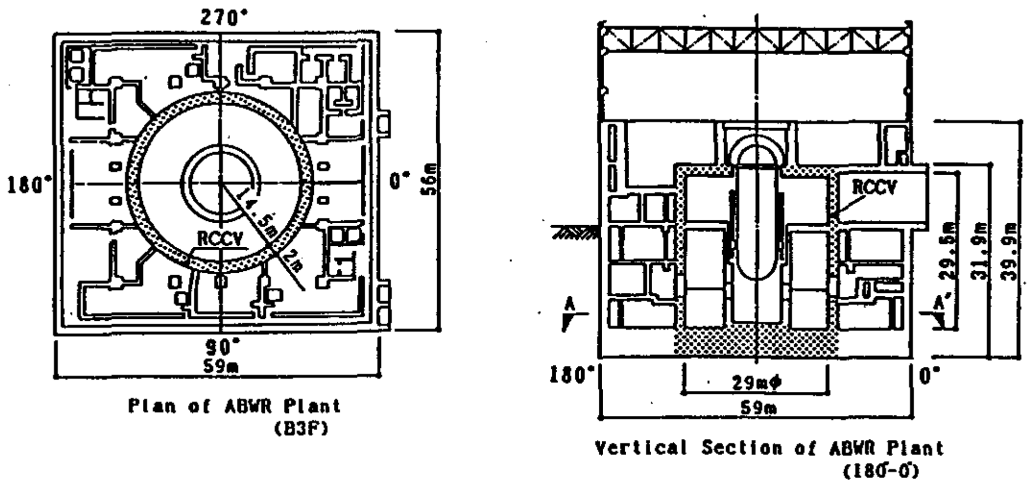


Fig. 5. Outline of ABWR/RCCV structure.

combined structure was tested by loading internal pressure, temperature and horizontal seismic force, and compared with the analysis results to confirm the validity of design method, and then the model was tested to failure by horizontal force, thereby assuring the ample margin for ultimate strength.

Figure 4 is a flow chart showing the entire experimental research work, which consists of basic study, verification studies by partial structural element models and an entire scaled model.

The structural integrity under design loading conditions as well as the ultimate strength and its margin were confirmed. The outcome of these studies was already presented in the 10th SMiRT Conference [15].

Figure 5 and table 9 show an outline of the structure and major design parameters of RCCV's for K6/7. K6/7 is presently in the preparatory stage for construction, and an effort is under way to study construction method and sequence, In-Service Inspection of RCCV structure, etc.

Table 9
Major parameters of RCCV

Plant Name		Kashiwazaki-Kariwa Unit No.6 & 7
Shape of CCV		Hybrid Cylinder and Top Slab with Steel Liner and Reactor Building
Configuration	Thickness of Cylinder Height of Cylinder (Internal) Inner Diameter Thickness of Top Slab Total Height	2.0 m 29.5 m 29.0 m 2.2 m ~ 2.4 m about 36 m (To Drywell Head)
Design Criteria	Start of Operation (Scheduled)	Unit 6 June, 1996 , Unit 7 July, 1997
	Design Pressure	3.16 kgf/cm²
	Test Pressure	3.56 kgf/cm²
	Design Temperature	171 °C
Materials	Concrete	Specified Design Strength Fc=330kgf/cm²
	Reinforcing Bar	SD40-D41 and larger , SD35-D38 and smaller
	Liner Plate	6.4 mm

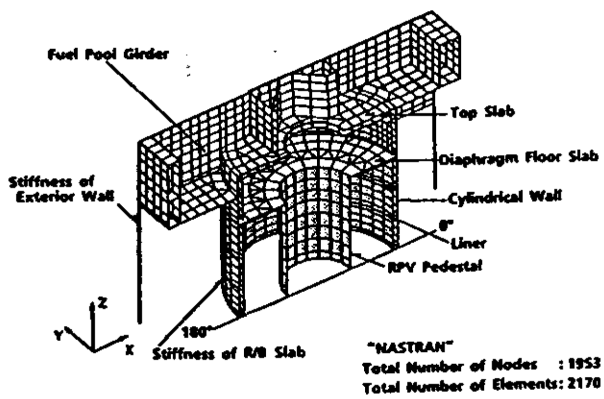


Fig. 6. 3-Dimensional FEM model of RCCV for structural design.

Since RCCV is a complicated structure surrounded and connected with structural elements of the building, and in addition it must be designed to various loading conditions, its structural analysis is carried out for final confirmative purpose by using a 3-dimensional FEM model as shown in fig. 6 [15].

4. Design

4.1. General

As was previously mentioned, each PCCV was designed and approved by the technical guideline prepared on a plant-by-plant basis in accordance with the MITI Technical Standard for Concrete Containment Vessel (revised in 1981) [13]. However, in view of the anticipation for the continued adoption of concrete containment vessels in the future, MITI established a committee and issued the 'Technical Standard for Concrete Containment Vessels for Nuclear Power Plants' as MITI Notice 452 (hereafter referred to as "Standard") in 1990 [7].

The basic principles of the Standard are as follows:
(1) Reference was made mainly to the relevant standards in Japan as well as the intent of existing equivalent foreign standards such as the ASME Sec. III Div. 2. Furthermore experiences gained from the design, construction and operation of previous concrete containment vessels were well reflected upon the Standard.
(2) Load combinations are classified into four Load Categories I–IV, which in principle correspond to those of the ASME Sec. III Div. 2.
(3) Requirements for seismic design are stipulated.
(4) Consideration is given that regulations for steel portion are consistent with those of the MITI Notice 501 (Technical Standard for Structural Design of Mechanical Components of Nuclear Power Facilities).

The Standard applies to concrete portion, steel liner plate, liner anchor, penetration sleeve, penetration anchor, attachment to liner plate, and so on. Those portions consisting of steel only are subject to regulations stipulated in the MITI Notice 501. The Standard is composed of four chapters which are outlined in table 10.

4.2. Load categories, loads and load combinations

Load combinations are classified into four categories "Load Categories I–IV" depending upon the

Table 10
Outline of MITI Notification No.452

Chapter	Contents
1 Introduction	• Scope and Definition of Terms
2 Concrete	• Materials (Concrete, Reinforcing Bars, Prestressing Tendons, etc.) and Design Criteria (Loads, Specified Design Strength and Design Allowables) • Design for Concrete Structures (Cylinder, Top Slab, Base Mat and Penetrations) • Other Details of Design (Minimum Reinforcement, Layout of Re-bars and Prestressing Tendons, Anchorage, Splicing, Covering, Spacing, etc.
3 Liner Plate, Liner Anchor etc.	• Steel Material and Design Criteria (Loads, Load Combinations, Design Allowables of Liner Plates, Liner Anchors etc.
4 Knuckle Parts and Shell Anchor Parts	• Material and Design Criteria.

frequency and simultaneous occurrence of loading, and each structural element is designed in accordance to the requirements to each Load Category.

Load Category I is defined as normal operating condition, while Load Category II as safety relief valve operating condition, test condition and snow load condition, and basically under these I and II conditions the plant is required to maintain its function for the long-term operation.

Load Category III is defined as abnormal condition other than I and II such as accidental and seismic loading conditions. Under these shortterm loading conditions, the plant is basically required to remain below elastic limits. Load Category IV is defined as extreme condition postulated in safety design where the plant is required to maintain its safety function.

Loads and load combinations for the four Load Categories are indicated in table 11. It must be mentioned that the test pressure is defined as 1.125 times the maximum design pressure as in the case of the steel containment vessel. As can be seen from table 11, altogether 14 loads and 15 load combinations are considered in concrete containment design.

4.3. Allowable limits

4.3.1. Concrete

Allowable limits for concrete are grouped into two categories; limits for membrane force and bending moment, and shear force:

– *Allowable limits for membrane force and bending moment* – Allowable stress limits for concrete are defined to Load Category I, II and III as shown in tables 12 and 13. Regarding the allowable compressive stress limits for concrete, two stress conditions are defined, i.e. stress condition 1 which does not include thermal stress and stress condition 2 which includes all stresses. Allowable limits for stress condition 2 are increased compared with those for stress condition 1 by some factors. The ultimate strength is set as an

Table 11
Load Categories and Load Combinations

Load Category	Load / load Condition	Dead Load	Live Load	Prestress Loads	Normal Operating Pressure	Normal Operating Piping Loads	Normal Operating Thermal Loads	L-accident Pressure	L-accident Piping Loads	L-accident Thermal Loads	Jet Force	S_1 seismic Load	S_2 seismic Load	Snow Loads	Wind Loads	Test Pressure	Equivalent of ASME Sec.III Div.2
I	Normal operating	1.0	1.0	1.0	1.0	1.0	1.0										Normal
II	Safety relief value operating	1.0	1.0	1.0	1.0	1.0	1.0										Normal
II	Testing	1.0	1.0	1.0												1.0	Test
II	Snow	1.0	1.0	1.0	1.0	1.0	1.0							1.0			Normal
III	Storm	1.0	1.0	1.0	1.0	1.0	1.0								1.0		Severe
III	S_1 seismic	1.0	1.0	1.0	1.0	1.0	1.0					1.0					Severe
III	L(1)-accident	1.0	1.0	1.0				1.0	1.0	1.0							Abnormal
III	L(2)-accident + S_1	1.0	1.0	1.0				1.0	1.0	1.0		1.0					Abnormal/Severe
IV	S_2 seismic	1.0	1.0	1.0	1.0	1.0							1.0				Extreme
IV	L(3)-accident	1.0	1.0	1.0				1.5*	1.0								Abnormal
IV	J-accident	1.0	1.0	1.0							1.0						Abnormal/Extreme
IV	L(4)-accident + S_1	1.0	1.0	1.0				1.0	1.0			1.0					Abnormal/Severe
IV	L(5)-accident + Snow	1.0	1.0	1.0				1.25*	1.0					1.25*			Abnormal/Severe
IV	L(5)-accident + Storm	1.0	1.0	1.0				1.25*	1.0						1.25*		Abnormal/Severe

Note : (1) When used for liner plate and liner anchor, all the values with "*" may be taken equal to 1.0 because they are not required to possess pressure-resisting function
(2) Safety relief valve operating condition in Load Category II is applicable to BWR only
(3) L(1)-accident condition in Load Category III includes peak loads immediately after LOCA
(4) L(2)-accident condition in Load Category III is long-sustaining loading condition 10^{-1} year after LOCA which is combined with S_1
(5) L(3)-accident condition in Load Category IV is LOCA loading condition where 1.5 times the design pressure is taken into account
(6) L(4)-accident condition in Load Category IV is LOCA loading condition combined with S_1 where the maximum pressure and piping loads are taken into account
(7) L(5)-accident condition in Load Category IV is LOCA loading condition combined with snow and storm where 1.25 times the maximum pressure and piping loads are taken into account

Table 12
Allowable stress (1. Compressive stress of concrete)

Load Category	Allowable Compressive Stress	
	Stress Condition 1	Stress Condition 2
I & II	Fc / 3	9Fc / 20
III	2Fc / 3	3Fc / 4

Note : Fc is Specified design strength of concrete (kgf/cm^2)

Table 13
Allowable stress (2. Shear stress of concrete)

Load Category	Allowable Shear Stress
I & II	Lesser of $\frac{Fc}{30}$ and $5 + \frac{Fc}{100}$
III	The above × 1.5

Note : Fc is Specified design strength of concrete(kgf/cm^2)

allowable limit for the combination of membrane force and bending moment for Load Category IV, and the strain limits of concrete and rebar in estimating the above ultimate strength are determined to be the values indicated in table 14. Additionally it is stipulated that the allowable limit for the compressive stress of concrete under Load Category IV be below 2/3 of the specified design strength.

– Allowable limit for shear force for shell portion – With regard to the allowable limits for in-plane shear force and out-of-plane shear force, and the limit for out-of plane shear force acting on the bottom of the cylinder which is induced by axi-symmetric loading, the ultimate shear strength derived from the test results is used as the basis for an allowable limit for Load Category IV. The allowable limit for Load Category I and II is defined as 1/2 of the ultimate shear strength used for Load Category IV, while the limit for Load Category III is defined as 3/4 of the same.

The following is the equation used for estimating the ultimate strength.

(i) In-plane shear strength

On the basis of horizontal loading test results using reinforced and prestressed concrete cylindrical models, the in-plane shear strength is determined from the strength obtained from the assumption of only steel being effective as restraining force and the upper limit of concrete strength as follows;

$$\tau_U = \tfrac{1}{2}\{(p_{t\phi} f_y + \sigma_{p\phi} - \sigma_{0\phi}) + (p_{t\theta} f_y + \sigma_{p\theta} - \sigma_{0\theta})\},$$

and $\tau_U \le 3.5\sqrt{F_c}$,

where

τ_u : ultimate tangential shear stress,
$p_{t\phi}$, $p_{t\theta}$: reinforcement ratios in meridional (ϕ) and circumferential (θ) directions respectively,
$\sigma_{0\phi}$, $\sigma_{0\theta}$: Membrane stresses in ϕ and θ directions respectively which are induced by an external force except for a prestressed force (these values become positive for tensile stress and 0 for compressive stress),
$\sigma_{p\phi}$, $\sigma_{p\theta}$: effective prestressed stresses in ϕ and θ directions, respectively,
f_y : specified yield strength of bars,
F_c : specified design strength of concrete.

(ii) Out-of-plane shear strength

For the out-of-plane shear strength, a new equation is used which is proposed based on the test results such as push-off tests, shear tests for columns and beams and reinforced concrete containment vessel, where consideration is given to the coefficient of reduction due to shear span ratio.

$$\tau_R = \Phi\{0.1(p_t f_y - \sigma_0) + 0.5 p_w f_y + 0.75\sqrt{F_c}\}, \text{ and}$$

$$\tau_R \le 3.5\sqrt{F_c},$$

where

τ_R : ultimate out-of-plane shear stress,
p_t : ratio of reinforcement to total cross section,
σ_0 : membrane stress caused by external forces (this value becomes positive for tensile stress),
p_w : ratio of out-of-plane shear reinforcement,
Φ : Coefficient of reduction by M/Qd, $\Phi = 1/\sqrt{M/(Qd)}$, $0.58 \le \Phi \le 1$, where
M : maximum bending moment of cross section,
Q : maximum shear force of cross section,
d : effective cross section.

Table 14
Strain limits for concrete and reinforcing bar in Load Category IV

Material	Value of Limit Strain	
Concrete	compressive	: 0.003
Reinforcing bar	tensile and compressive	: 0.005

The maximum compressive stress of concrete must be less than 0.85 Fc.
Tensile and compressive stress of reinforcing bars must be less than those of allowables in Load Category III.

Table 15
Strain allowables for liner plate

Load Category	Classification of Strains	Design Allowable	
		Membrane	Membrane + bending
I & II	Compressive strain	0.002	0.004
	Tensile strain	0.002	0.004
III & IV	Compressive strain	0.005	0.014
	Tensile strain	0.003	0.010

Table 17
Displacement allowables for linear anchor

Load Category	Displacement Allowables
I & II	$\delta a = 0.25\ \delta u$
III & IV	$\delta a = 0.5\ \delta u$

Note : δu is displacement of liner anchor at fracture, which may be determined on the basis of theoretical or experimental results.

(iii) Out-of-plane shear strength at bottom of shell

The following equation based on the yielding of re-bars in circumferential direction is used for estimating the ultimate out-of-plane shear strength at the bottom portion of the shell connected with the foundation when it is subjected to axi-symmetric loading.

$$\tau_H = 10 p_{t\theta} f_y / \left(13.2\sqrt{\beta} - \beta\right),$$

where

τ_H : ultimate out-of-plane shear stress,
$p_{t\theta}$: reinforcement ratio in θ direction,
f_y : specified yield strength of reinforcing bar,
$\beta = r/t$ r : radius to center of wall,
t : thickness of containment.

4.3.2. Liner plate and liner anchor

Liner plate – Allowable limits for the strain induced in the liner plate by constraint or forced deformation are indicated in table 15. The provisions for class 2 support structure in the MITI Notice 501 are to be followed when liner plate is subjected to external load (mechanical load) and is thought to function as support structure transmitting this load to concrete, and when liner plate is not supported by concrete (in case the liner is subjected to negative pressure) and is thought to function as structural element, the provisions for class 2 vessel in the MITI Notice 501 are to be followed.

Liner anchor – Table 16 shows the allowable load limits of the liner anchor when subjected to mechanical loads. Indicated in table 17 are the allowable deformation limits for liner anchor which are induced in liner anchor by forced strain of liner plate. When the liner anchor thought to act as support structure, it is stipulated that the provisions in the MITI Notice 501 are to be followed.

Table 16
Allowables for liner anchor

Load Category	To Mechanical Loads
I & II	Lesser of Fa=0.67 Fy and Fa=0.33 Fu
III & IV	Lesser of Fa=0.9 Fy and Fa=0.5 Fu

Note : Fy is yield strength of liner anchor.
Fu is ultimate strength of liner anchor.
Fy and Fu may be determined on the basis of theoretical or experimental results.

5. Conclusions

Owing to the above-mentioned extensive verification studies and actual design and construction experiences, CCV's are proved to function satisfactorily as a sound safety barrier in earthquake-prone countries like Japan.

The Kashiwazaki-Kariwa Power Station Unit 6 and 7, the first ABWR's employing RCCV, are currently under licensing review and is expected to start its construction in 1991.

In light of further advancement of technology, an effort is presently being made by on-going studies to develop a more advanced method for use in concrete containment vessels in the near future, for example, to upgrade the quality of concrete during construction and to develop the optimum non-destructive method to inspect the structural integrity of CCV's during operation by automatic devices.

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Containments design for the Advanced and Simplified Boiling Water Reactor standard plants

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The Advanced Boiling Water Reactor (ABWR) design is based on construction and operating experience of nuclear power plants in Japan, United States, and Europe. To optimize the plant arrangement of the Advanced Boiling Water Reactor (ABWR) and to verify the structural feasibility to carry design loads a study was conducted. To arrive at an optimized plant arrangement with a minimum size reactor building (RB), a circular cylindrical reinforced concrete containment vessel (RCCV) with a flat top slab and a monolithically connected diaphragm slab has been selected.

The Simplified Boiling Water Reactor (SBWR) is being developed as a standardized 600 MWe Advanced Light Water Reactor. The design concept of the SBWR is based on simplicity and passive features to enhance safety and reliability, improve performance and increase economic viability. Due to the use of passive containment cooling, SBWR has features that are different from those of existing designs.

The objectives of the study for the ABWR containment and RB are to perform a structural analysis of the containment and RB and to evaluate the structure for conformance to the U.S. NRC requirements. The main objective of the studies for the SBWR is to demonstrate the structural design feasibility of the containment for the pressure and the temperature response associated with the passive systems adopted for the SBWR.

1. Introduction

The Advanced Boiling Water Reactor (ABWR) design is based on construction and operating experience of nuclear power plants in Japan, United States and Europe. General Electric and Bechtel performed studies in 1984 to optimize the plant arrangement of the ABWR and to verify the structural feasibility to carry design loads [1,2]. A comparison of major plant specifications for the ABWR with those of the current generation of Japanese BWR can be found in ref. [3].

The Simplified Boiling Water Reactor (SBWR) is based on simplicity and passive features to enhance safety and reliability, improve performance and increase economic viability. Use of the pressure suppression system, gravity-driven cooling system (GDCS) and passive containment cooling system (PCCS) allows the elimination of safety grade emergency diesel generators, core cooling pumps and heat removal pumps thus simplifying plant design and reducing plant costs. Reference [4] gives a comparison of features for the SBWR with those for the current conventional BWR and the ABWR.

The objectives of the study for the ABWR containment and RB are to perform a structural analysis of the containment and RB and to evaluate the structure for conformance to the U.S. NRC requirements.

The main objective of the studies for the SBWR is to demonstrate the structural design feasibility of the containment for the pressure and the temperature response associated with the passive systems adopted for the SBWR and to demonstrate that a 30-month construction schedule can be achieved. More detailed information can be found in ref. [5].

2. Description of the containments and the reactor buildings

To arrive at an optimized plant arrangement with a minimum size reactor building, a circular cylindrical reinforced concrete containment vessel (RCCV) with a flat top slab and a monolithically connected diaphragm slab has been selected. The flat top slab is integrated with the fuel pool girders which are framed into the RB structural walls and floors.

The ABWR containment has 29.0 m inside diameter (ID) and is integrated with the reactor building. The containment and the reactor building are supported by a common foundation mat. The bottom of

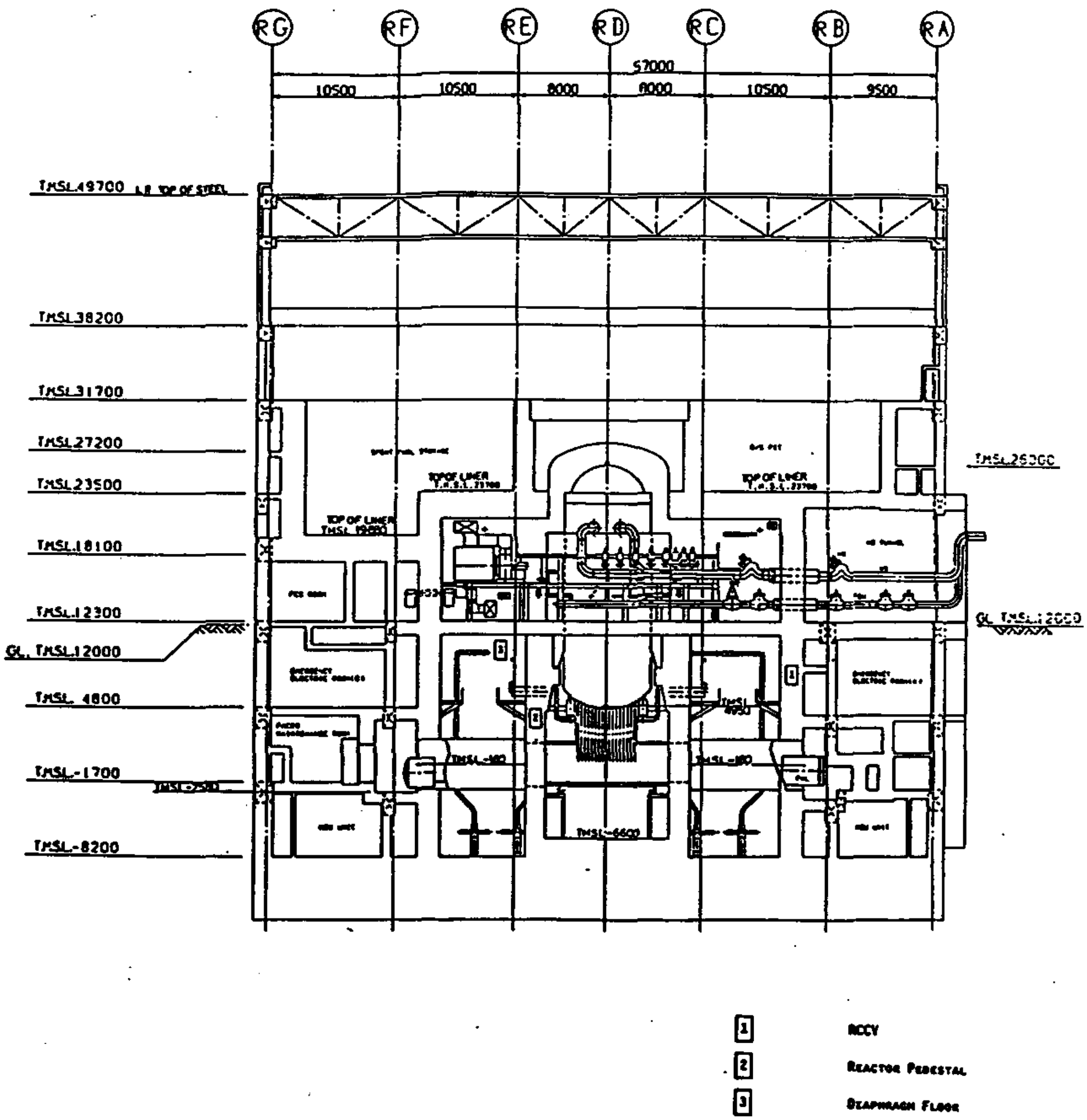
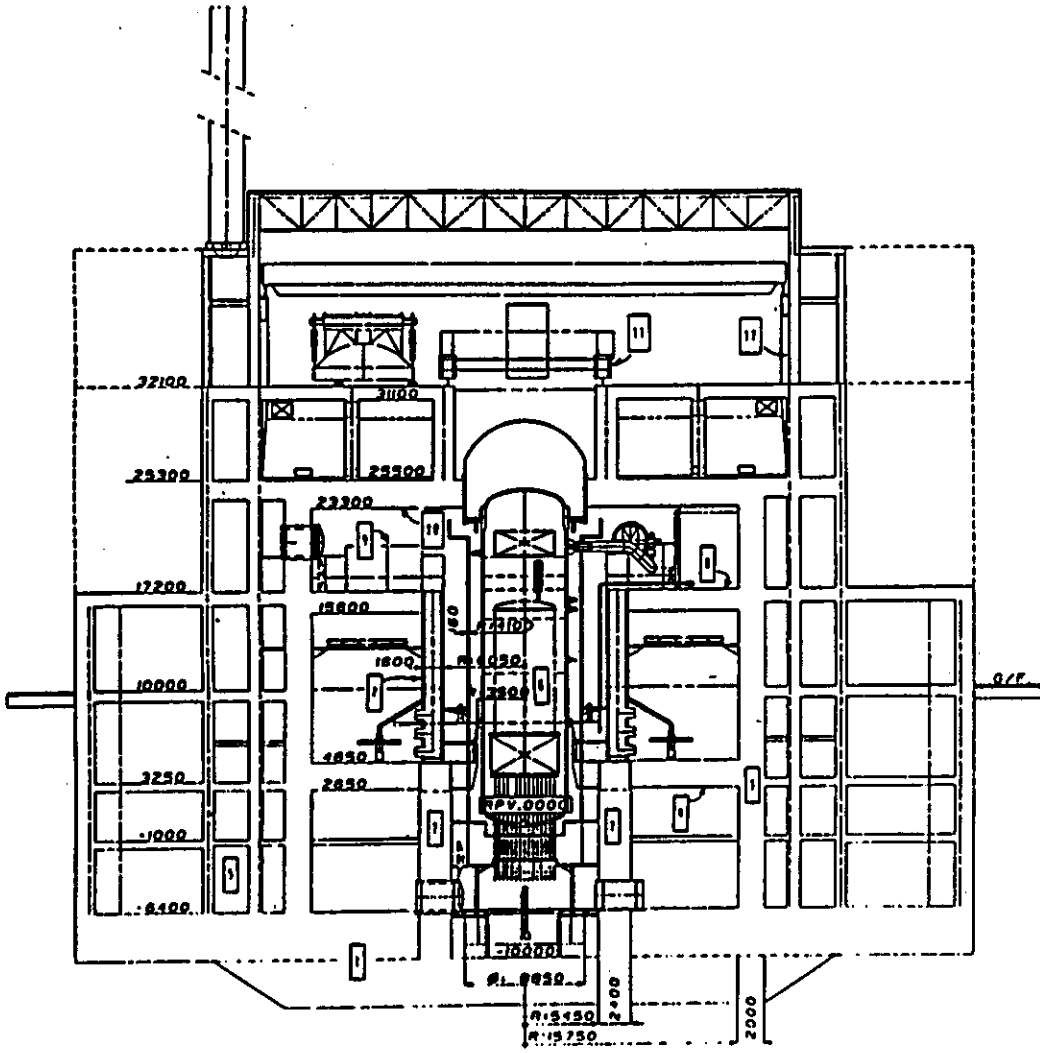


Fig. 1. ABWR containment and reactor building.



DESIGNATION	STRUCTURE/ EQUIPMENT	MODULE
1	Base-mat	Rebar Module Upper and Lower Mat and Pedestal Dowels
2	Pedestal	Rebar Module with Liner Plate
3	Containment Wall	Rebar Module with Liner Plate
4	Suppression Pool Slab	Steel Lower Form Module Radial Rebar In-Site Concentric Hoop Rebar Module
5	R/B Equip. Area	Precast Wall Module Rebar Module Prefabricated Piping Composite Module
6	Reactor	RPV Module
7	Vent Wall Structure	Vent-Structure & RPV Support Steel Module with In-situ Filled Concrete
8	D/F Slab	Drywell Floor Steel Module
9	Drywell Equip. Piping Support Structures	Structure/Equipment/Piping/Valves Module(s)
10	Drywell Top Slab	Top Slab Steel Liner-Rebar Module with In-situ Filled Concrete
11	Bridge Crane & Fuel Handling Machine	Bridge Crane & Fuel Hnd. Mach. Module
12	Super-Structure	Rebar Module

Fig. 2. SBWR containment and reactor building.

the foundation mat is embedded in the ground 25.9 m (85 ft) below grade. The major containment internal structures consist of the reactor pedestal, the reactor shield wall, and the diaphragm floor. The reactor pedestal is a composite steel and concrete structure, the pedestal consists of two concentric steel shells tied together by vertical steel diaphragms. A reinforced concrete circular diaphragm floor slab serves as a barrier between the drywell and the wetwell. The diaphragm floor is supported by the containment wall and the reactor pedestal. The top of the RPV is supported by the reactor shield wall by means of RPV stabilizer truss.

The RB of the ABWR is a 59 m (193.5 ft) by 56 m (183.75 ft) reinforced concrete structure. The building has six reinforced concrete floors which are monolithically connected to the containment. The operating floor at elevation 26.7 m (87.6 ft) is not directly connected to the containment, but is connected to the fuel pool girders which are supported by the containment and the RB. The interior walls and the floor beams are not connected to the containment structure. The arrangement of the RB and the containment is shown in fig. 1.

The SBWR plant due to the use of passive containment cooling, has features that are different from those of other existing designs. The Isolation Condensers (I.C.'s) and the passive containment cooling system (PCCS) that removes decay heat by natural convection and evaporation are in pools which are located on top of the drywell. To maintain long term cooling and water coverage of the reactor core, the suppression pool (SP) is elevated to such a level that water can flow by gravity from the SP to the reactor after LOCA.

The SBWR containment has 31.5 m ID and is partially integrated with the RB. It consists of the reactor pressure vessel (RPV) pedestal, SP floor slab, the cylindrical containment wall and the drywell top slab. The drywell top slab supports the IC pools and service pool. The IC pool girders on the drywell top slab provide strength to resist containment pressure loads. The top slab has a large opening ($D = 9.4$ m) in the middle for the drywell head and four openings ($d = 3.2$ m) for the IC's and PCCS. The vent wall structure and the diaphragm floor slab are steel structures filled with concrete. The RPV pedestal supports the reactor vessel, reactor shield wall, vent wall structure and the suppression pool.

The RB structures for the SBWR consists of the RCCV and three rectangular "boxes" supported on a common basemat of 66.3 m x 66.3 m with interconnected slabs at various elevations as shown in fig. 2. The structures are primarily of reinforced concrete construction. The bottom of the foundation mat is embedded in the ground 23.0 m (76 ft) below grade. In the present design for SBWR, the RPV pedestal forms part of the containment pressure boundary. It was decided to adopt a reinforced concrete pedestal with liner plate on the inner face acting as a leak tight boundary. This was judged to be more desirable than a steel-concrete composite pedestal, based on consideration of applicable design codes, severe accident conditions and construction requirements including modularization.

3. Design criteria

The containment structures are designed in accordance with the ASME boiler and Pressure Vessel Code, Section III Div. 2 [6].

The containment temperature and pressure conditions for normal, testing and LOCA conditions considered in the study are shown in table 1. Pool hydrodynamic loads and the corresponding containment pressure are also considered. Temperatures greater than 150°F are postulated to last a long period of time after a LOCA or postulated severe accident conditions in the SBWR. Degradation of material properties is expected and, therefore, temperature dependent material properties are considered in the analysis and design.

4. Seismic analysis

The seismic analyses for the feasibility study of the SBWR standard plant were performed with lumped mass model as shown in fig. 3. A range of soils was considered in terms of shear wave velocity (v). Evalu-

Table 1
Pressure and temperature loads

Condition	Pressure (psig)		Temperature (°F)	
	Drywell	Wetwell	Drywell	Wetwell
Test (1)	63.3	63.3	60	60
Test (2)	63.3	34.6	60	60
Normal	2.0	2.0	135	95
LOCA	55.0	55.0	340 [a]	220 [a]

[a] For SBWR these temperatures may exist for up to 30 days after LOCA and have been considered in the design.

ated soils include soft soil with $v = 300$ m/s, 500 m/s, intermediate soil with $v = 1000$ m/s and hard soil with $v = 1500$ m/s and 3000 m/s. Two sets of input motion were used. One based on a peak horizontal ground acceleration of 0.3 g (SSE) with a response spectrum per US NRC Reg. Guide 1.60. The other based on Japanese MITI Standardization program for LWR in Japan. In addition, a parametric seismic analysis was performed for various idealized site conditions as presented in ref. [9].

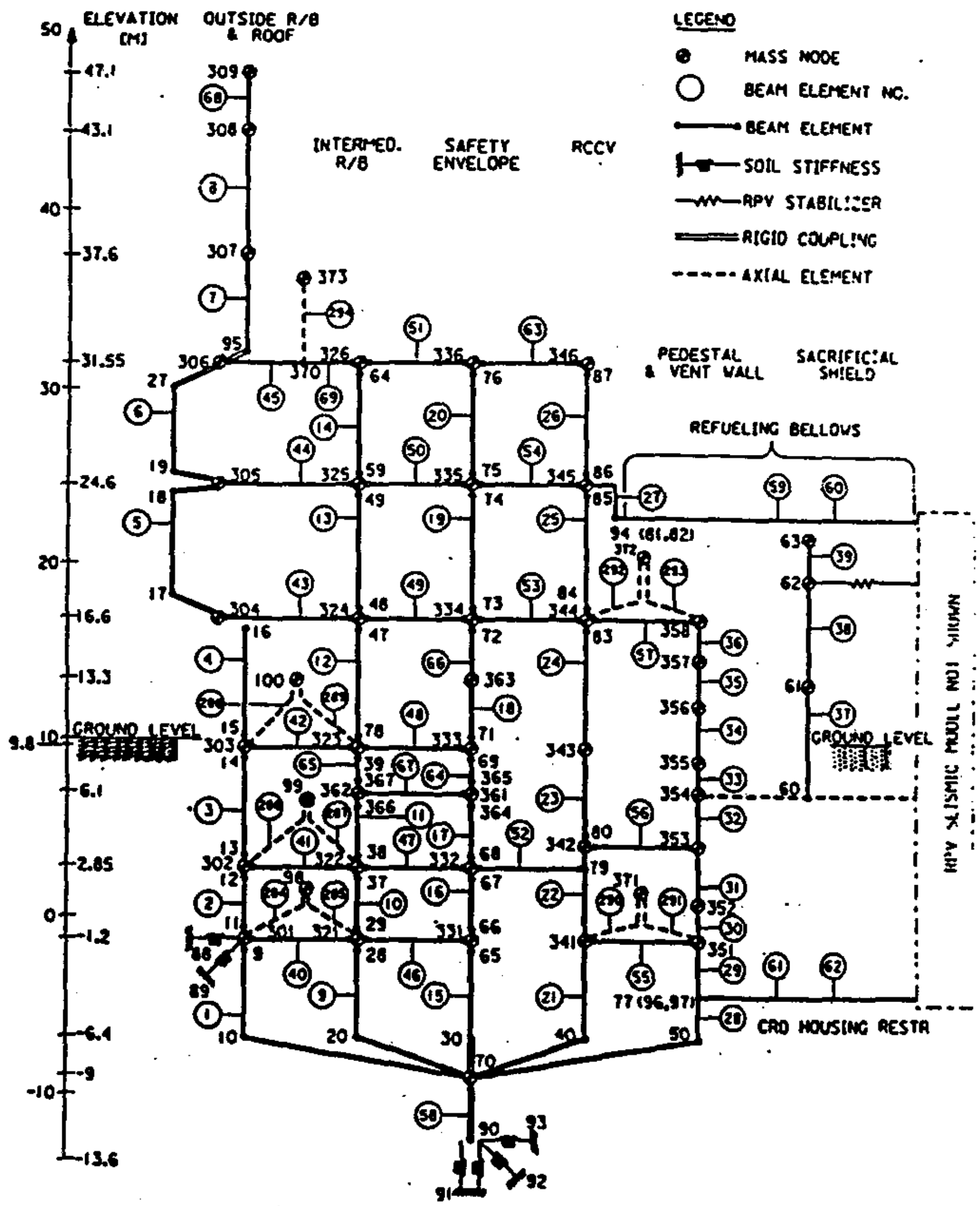


Fig. 3. SBWR lumped mass seismic model.

5. Analytical models and structural analysis

The containment and the RB are analyzed as one integrated structure utilizing the finite element computer program STARDYNE. The structures are idealized as a three-dimensional assemblage of linear elastic beam and plate elements.

The models include the geometry and the material properties of major structural components consisting of the containment wall, reactor pedestal, reactor shield wall, reactor vessel, foundation mat, diaphragm floor, containment top slab, fuel pool girders, and the RB floors, walls, columns, and roof. The underlying foundation soil was represented by spring elements. The side soil was not included in the model. The foundation soil was adjusted to include embedment effects. The lateral soil pressure was considered during the evaluation of the RB outer walls. The finite element representation of the structure for the ABWR is shown in fig. 4 and for the SBWR in fig. 5. For the ABWR, because of symmetry, 180° model was used where as for the SBWR, 360° F.E. model was used.

The structural analysis consisted of four steps:
- the formulation and decomposition of the stiffness matrix,
- the static analysis for the load cases,
- the combination of loads, and
- the stress analysis of rebar and concrete.

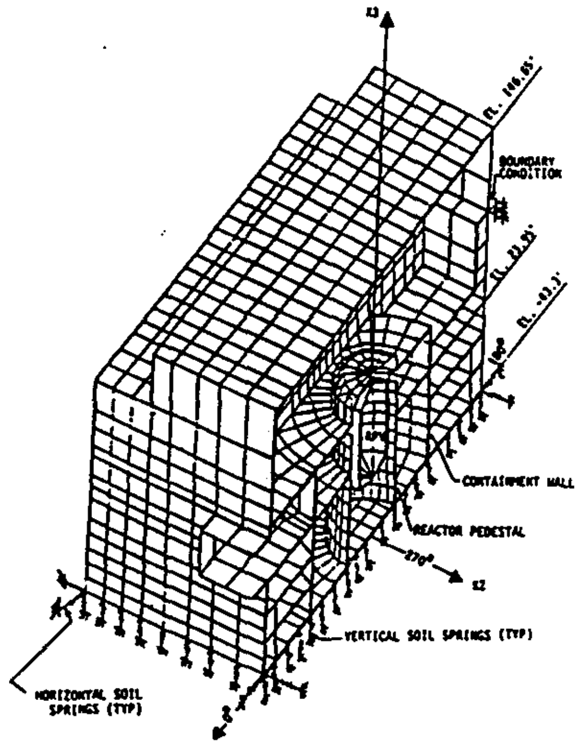


Fig. 4. Finite element model for ABWR containment and reactor building.

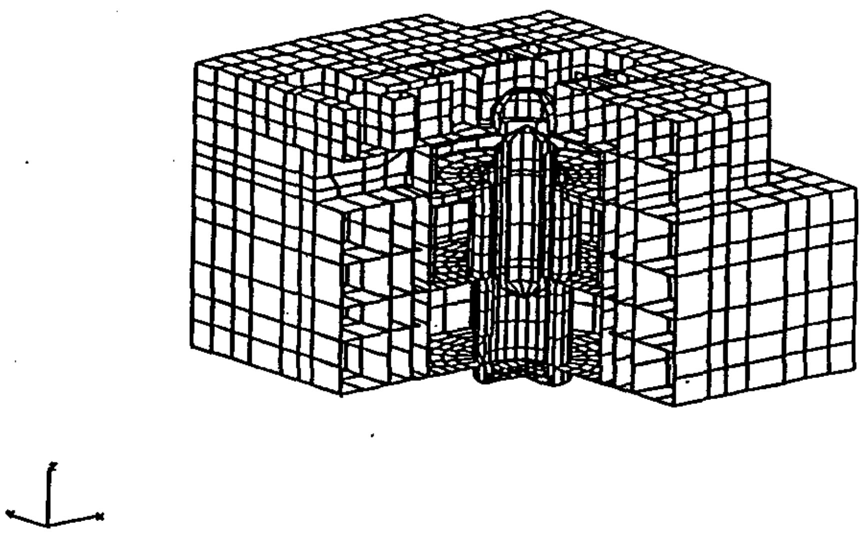


Fig. 5. Finite element model for SBWR containment and reactor building.

The rebar and concrete stress analysis was performed using Concrete Element Cracking Analysis Program (CECAP). The element represents a section of a concrete shell or plate, layers of reinforcing steel, and a liner plate. External forces, as input, consist of moments in two directions, axial forces in two perpendicular directions, in plane and transverse shear forces. The program outputs stresses and strains along the element in the concrete, reinforcement, and the liner plate. CECAP assumes linear strain relationships for steel and concrete in compression. Concrete is assumed to have no tensile strength. The solution is an iterative process, whereby tensile stresses found initially in concrete are relieved due to concrete cracking and redistributed in the element. The equilibrium of non-thermal loads is preserved. For thermal effects, the element is assumed free to expand inplane, but is fixed against rotation. The capacity for expansion and cracking generally results in a reduction in thermal forces and moments from the initial condition.

6. Structural assessment and conclusions

Although the design criteria and seismic analysis discussed in sections 3 and 4 respectively, are for the SBWR, similar design criteria and seismic analysis based on a range of soil conditions were used for the ABWR study. The results of the ABWR study show that with the present configuration and with RCCV wall thickness of 2 m, the containment and RB can be designed as standardized plant or generic site to meet the ASME Section III, Division 2, and the NRC requirements.

The reactor building and the containment for both the ABWR and the SBWR were analyzed by finite element methods using plate and shell elements. The concrete cracking effects were evaluated by successive iterations. The evaluation results demonstrate that the RCCV wall of 2.0 meter thickness can be adequately reinforced to resist the loads discussed. The RCCV, as designed for LOCA conditions, can withstand a severe accident pressure of at least two times the design pressure value together with associated temperatures. Similar conclusions were reached in ref. [7] for a slightly larger SBWR containment but with somewhat different configuration.

The details of analysis and results for the SBWR top slab are presented in ref. [8]. The results show that the design is feasible with the specified number of large openings in the slab.

Similar to the ABWR design, the RCCV of the SBWR plant is integrated with the RB by RB floor slabs at various elevations and by the pool girders on the top. For the evaluation, a finite element model was prepared, and analysis was performed using the STAR-DYNE computer code. In a series of iterations the stiffness of highly stressed elements was reduced to allow for redistribution of forces due to concrete cracking. The study results showed that for the present configuration with integrated RCCV, RB, and pool girders, the structural design is feasible. Even though large shear stresses are induced in the pool girders due to the thermal growth of the RCCV due to LOCA thermal loads, based on the initial elastic analysis, the thermal stresses are significantly reduced after accounting for relaxation due to cracking of concrete.

The reactor building for the SBWR has floor slabs at various heights. Integration of the RCCV with the RB is advantageous from seismic design considerations. However, when the design for LOCA pressure and temperature and construction are considered, integration of all floor slabs with the RCCV is not desirable. To optimize the structural responses it was decided that the two structures be integrated only at:

- Suppression pool bottom floor slab level
- Suppression pool top slab level
- RCCV top slab level
- Operating floor.

It was decided not to integrate the remaining floors of the RB with the RCCV. A separation gap has been provided between these slabs and the RCCV wall, with appropriate detail for the required leak tightness against flooding, fire, etc. This will prevent containment pressure and thermal loads from being transmitted to these floor slabs, thus making their design more economical. Also, in absence of floor integration with the RCCV, the construction of RCCV can be expedited.

The vent wall structure is made up of two concentric steel cylinders with vertical stiffeners in-between. There are 8 vent pipes that are equally spaced between stiffeners. The remaining spaces are filled with concrete. This type of construction lends itself well to modularization and off-site prefabrication. The design in feasible with 30 mm thick steel plates.

Diaphragm floor slab is a steel structure consisting of continuous top and bottom plates with circumferential stiffeners and radial vertical web plates in between. This lends itself to modular construction and provides for easy anchorage of GDCS pool steel framing and the pipe support structure in the drywell.

The evaluation results of SBWR show that the structural design is feasible for the pressure and temperature responses associated with the passive systems adopted for the SBWR in conjunction with seismic loads derived from various soil conditions to represent generic site.

Study was also performed about constructability of the RB. The results showed that by using large scale prefabrication and modularization and based on use of 1000 ton crane with rolling 4×10's, work week (70 working hours per week), a 30-month construction schedule from start of structural concrete to the fuel load is achievable for "*n*-th of a kind" plant. Figure 2 shows some of the large structural modules considered in this study.

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Containment concepts for High Temperature Reactors

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Different containment concepts have been proposed for High Temperature Reactors. In the paper the confinement, the gastight pressurized containment and the vented confinement are discussed. For a small HTR such as the Modul it seems to be possible to provide a vented confinement instead of a gastight containment. The German Reactor Safety Commission has given a positive statement. Due to the specific safety characteristics of the HTR the safety concepts can differ in part quite considerably from current LWR standard solutions.

1. Introduction

During the past three different containment concepts have been proposed for High Temperature Reactors. The prototype reactor THTR-300 which was under construction from 1972 to 1985 and is now out of operation has a confinement. For the HTR 1160 project of the mid-1970s a gastight pressurized containment was considered necessary, while for the recent plant concepts of medium and small power, the HTR-500 and the Modul, vented confinements are proposed. In the paper these different confinement and containment types are discussed.

2. Containment requirements

The requirements for containments are specified in the 'Safety Criteria for Energy-Producing Plants with Gas-Cooled High-Temperature Reactors' [1]. According to these criteria, the plant must have a containment to fulfil its safety-related functions in normal operation and during accidents.

In conjunction with the coolant confinement and other retention barriers for radioactive substances, the containment must ensure that the requirements stipulated by the Atomic Energy Act and the Radiation Protection Ordinance are met for the assumed discharge or release of radioactive substances into the environment in normal operation and during accidents.

The confinement of the reactor coolant must be accommodated within the containment. Any other plant parts containing radioactive substances must also be accommodated within the containment unless the requirements of the Radiation Protection Ordinance are met by other suitable measures.

The containment including all penetrations, airlocks and auxiliary equipment, as functionally required for accident control, must be designed in such a way that it can withstand static, dynamic and thermal loads in normal operation and during accidents to the extent required in order to fulfil its safety-related function. The containment must maintain its integrity in the case of external impacts.

The requirement that the containment must maintain its integrity in the case of external impacts means:
- The leak tightness and load-bearing capacity of the containment must be ensured if proof of compliance with the provisions of the Radiation Protection Ordinance for accidents can only be furnished under the tightness condition.
- Only the load-bearing capacity of the containment need be ensured if proof of compliance with the provisions of the Radiation Protection Ordinance can also be furnished without the tightness criterion.

Containments of reinforced and prestressed concrete must comply with DIN 25 459 [9].

3. Containment concepts

3.1. Confinement of the THTR-300

The confinement system of the THTR-300 consists of the following parts [2,3] (Fig. 1):

- the burst-proof reactor pressure vessel of prestressed concrete in which the primary circuit is located,
- the so-called vent wall around the pressure vessel,
- the so-called safety shell composed of the walls between rooms R_1, respectively R_2 and R_3,
- the auxiliary systems for retention and filtering of possible leakages.

The R_1 rooms contain systems which are in contact with the primary circuit during operation or accidents. They are of pressure-resistant design. The maximum accidental gas leakage, which is limited to 5.5 kg/s, can be discharged directly into the stack through a separate depressurization system. The R_2 rooms also contain primary gas conducting systems. Possible gas leakages (max. ≤ 1.83 kg/s) are controlled and discharged into the stack by the normal exhaust air system. The R_3 rooms do not contain any systems which are in direct contact with the primary system. All systems connected with systems in R_1 and R_2 rooms can be isolated.

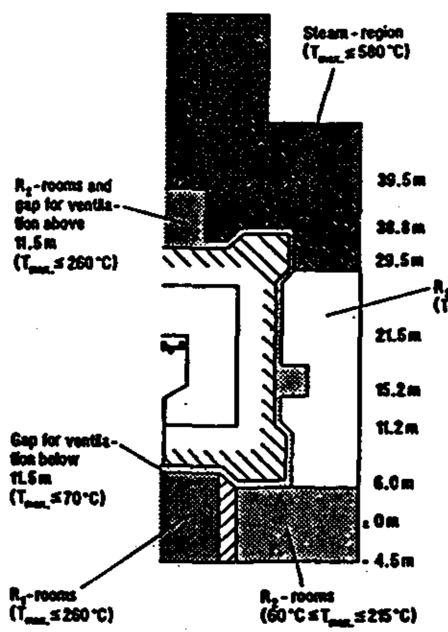

Fig. 1. Confinement of the THTR-300.

Activity release in R_3 rooms is suppressed by structural design measures. In addition, a partition is provided in two afterheat removal systems.

3.2. Reactor protection building as a gastight pressurized containment of the HTR-1160

The basic design of the planned large HTR-1160 plant includes a containment similar to those usually provided in LWRs, i.e. a gastight pressurized containment [4]. The reactor protection building differs from the reactor hall of the 300 MWe THTR nuclear power station with respect to the building requirements and type of construction. It serves both as the primary system containment and the surrounding building. This purpose defines the functions and the safety-related significance of the reactor protection building, i.e. to withstand all internal loads arising from the plant and all external loads including, in particular, also aircraft crash, chemical explosions and earthquake.

The THTR containment function was basically assigned to the reactor pressure vessel of prestressed concrete without depressurization system and with a gastight liner. For this reason, great significance was attached to the proof of liner integrity in the design philosophy for the THTR prestressed concrete reactor pressure vessel.

The HTR reactor protection building differs from previous light-water reactor containments (with the exception of Grundremmingen 2 featuring a prestressed concrete containment) in that there is no annular gap between the containment and the surrounding reactor building. The concrete structure of the building must therefore accommodate both the external loads and the internal pressure in the event of a loss-of-coolant accident, whereas the liner must ensure the tightness of the building. The tightness of the reactor protection building was specified with a leak rate of 0.3%/d under accident conditions at 385°C and 4.05 bar.

The reactor protection building essentially consists of a cylindrical shell which is placed on a circular foundation slab and covered by a hemi-ellipsoidal dome (fig. 2). The inner surface of the concrete structure is sealed from the atmosphere in the reactor protection building by a liner – a steel lining fitted directly to the concrete and anchored therein. The penetrations through the concrete structure are designed as gastight pipe penetrations.

The cylindrical part of the reactor protection building is prestressed in the circumferential and vertical directions. The horizontal tendons are anchored on 3 butresses (120° pitch) on the cylinder outer wall. The vertical tendons are anchored in an annular support at the upper end of the cylinder and, at the bottom, in the outer prestressed gallery of the foundation slab. The tendons of the dome are anchored in the ring girder at the upper end of the cylinder. Due to the double function of the reactor building as both the safety containment and the surrounding building, it has to meet the requirement according to [1] criteria 2.6 and 8.1, that any release of radioactive substances must be prevented even after external impacts, in particular aircraft crash.

There are two possibilities of complying with this requirement which is decisive for the concept:
- furnishing proof that the reactor proctection building is still sufficiently tight after an external impact, especially after an aircraft crash, or
- furnishing proof that no loss-of-coolant accident occurs due to external impacts if the leak-tightness cannot maintained after external events.

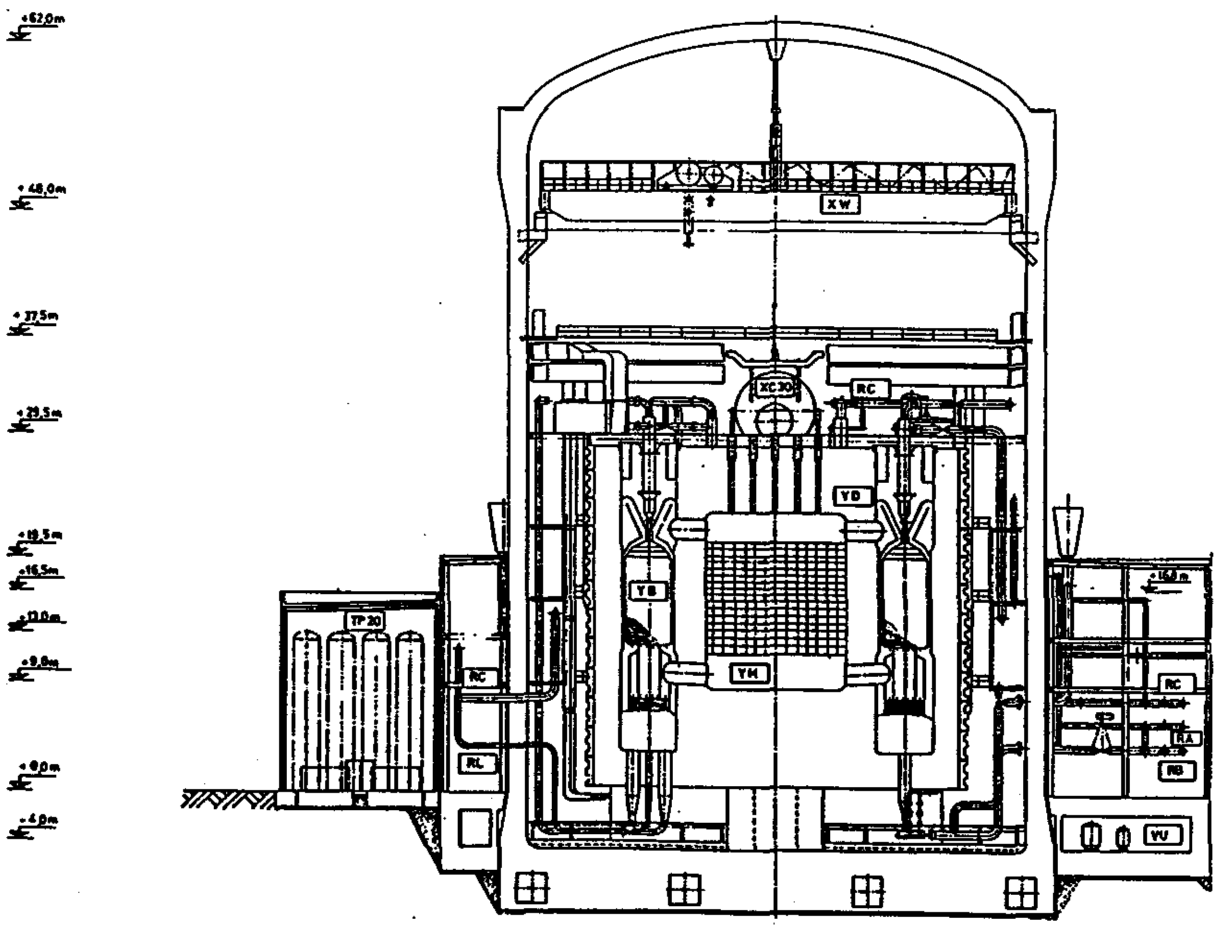

Fig. 2. Reactor protection building of the HTR-1160.

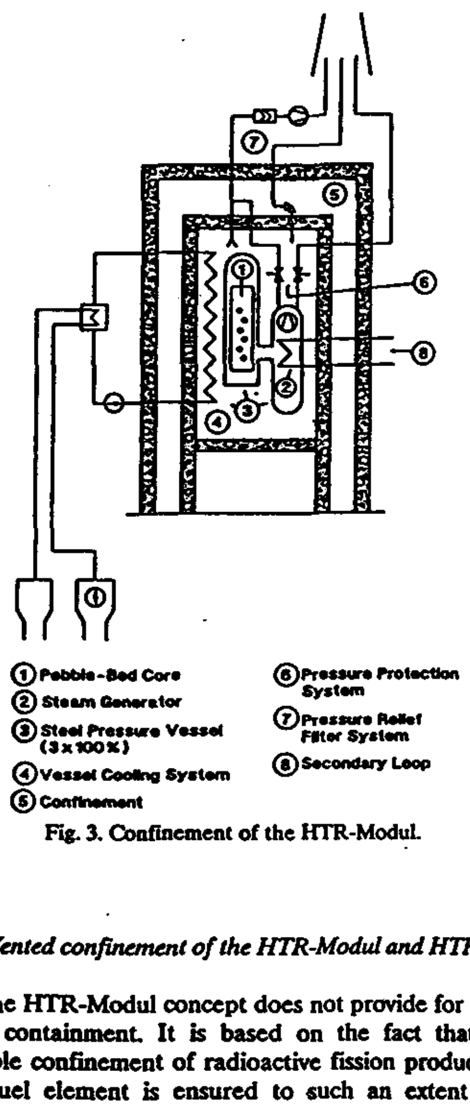

Fig. 3. Confinement of the HTR-Modul.

3.3. Vented confinement of the HTR-Modul and HTR-500

The HTR-Modul concept does not provide for a gas tight containment. It is based on the fact that the reliable confinement of radioactive fission products in the fuel element is ensured to such an extent that environmental exposure remains below the permissible limits in all accidents [5,6].

The reactor protection building without liner has a double function in that it protects the reactor against external impacts and ensures controlled activity release of primary circuit leakages into the environment (fig. 3). This means that the design does not aim at completely confining the activity over long periods of time, as generally practiced, but major leakages (ranging from 0.5 to 11.5 kg of helium/s) are discharged through the depressurization system and minor leakages through the exhaust air system with filters into the stack and the environment.

The primary rooms are connected by openings in order to achieve pressure equalization as rapidly as possible. After having reached the equalization pressure of 1 bar, the pressure relief flaps close automatically and directional ventilation is established again in the building. The relief ducts are additionally provided with a remotely closing flap each.

Ventilation of the rooms in the reactor building is designed for the selective release of radioactive fission products. Primary circuit leakages up to a leak size of 2 m^2 can be accommodated and filtered by the ventilation system.

The German Reactor Safety Commission has evaluated the proposal and does not have any safety-related objections to the concept of activity confinement. 'The concept is suited to ensure that the regulations of the Radiation Protection Ordinance for normal operation and design basis accidents are complied with.'

In the United States also the concept of a Modular High-Temperature Gas-Cooled Reactor (MHTGR) is pursued. The reactor building of this design does not provide a leaktight, pressurized containment, but controlled venting instead. In NUREG-1338 [8] the NRC explains:

The staff recognizes that a design without a conventional containment building presents a significant departure from past practice on LWRs and that under certain situations LWR containment buildings have been effective components of the defense-in-depth approach. Therefore, designs that deviate from such practice need to be reviewed to ensure that an equivalent level of safety as that of current-generation LWRs is maintained and that uncertainties in design and performance are properly accounted for. The staff believes that such designs are possible, although the ultimate acceptance of such designs will require extensive review, testing, and demonstration. Accordingly, the staff proposes criteria to be met in order to certify a reactor design without a containment building with the understanding that in reviewing a design against these criteria, a large burden will rest with the applicant to demonstrate compliance, particularly in view of the uncertainties associated with a new design.

The following are proposed criteria that advanced-reactor designers must meet for NRC certification of a design without a containment building:

(1) The design should contain multiple barriers to radiation release that limit radiation release at least equivalent to that of current-generation LWRs.
(2) The fission-product-retention capability of the design must be demonstrated via a testing program utilizing a full-size prototype plant consisting of at

least one reactor module and the associated systems, structures, and components necessary to demonstrate safety.

(3) Different emphasis and types of QA, surveillance, in-service inspection, and inservice testing over and above that traditionally employed on LWRs should be provided.

(4) Protection of safety-related systems, structures, and components from sabotage and external events should be provided that is at least equivalent to that for current-generation LWRs.

(5) The design should have specific measures to ensure that core heat up accidents, accidents with significant positive reactivity feedback, or other accidents with the potential of a large radiation release, such as graphite fires, have lower frequencies than 10^{-7} per plant-year.

(6) An assessment of the potential improvement in safety if a containment building were added would have to be made. Judgment would then be used to determine the need for a containment building based on the cost and change in risk.

These criteria are intended to maintain at least the same level of protection of the public and environment, by specifying equivalent dose guidelines and protection, as is provided by current-generation LWRs.

4. Conclusion

For High Temperature Reactors of small power such as the Modul it seems to be possible to provide vented confinements instead of the gastight pressurized containments of the current generation LWRs. The German Reactor Safety Commission has given a positive statement. The final determination of the acceptability by the NRC is contingent on evaluation of additional information. Due to the specific safety characteristics of the High Temperature Reactor the safety concepts can differ in part quite considerably from current LWR standard solutions.

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Containment for the Low Temperature District Nuclear-Heating Reactor

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An integral arrangement is adopted for the Low Temperature District Nuclear-Heating Reactor. The primary heat exchangers, control rod drives and spent fuel elements are put in the reactor pressure vessel together with the reactor core. The primary coolant flows in natural circulation through the reactor core and the primary heat exchangers. The primary coolant pipes penetrating the wall of the reactor pressure vessel are all of small diameters. The reactor vessel constitutes the of the reactor vessel can be used for the reactor. Design principles and functions of the containment are the same as for the containment of a PWR. But the adoption of a small sized containment brings about some benefits such as a short period of manufacturing, relatively low cost, and ease for sealing. A loss of primary coolant accident would not be happened during a rupture accident of the primary coolant pressure boundary inside the containment owing to its intrinsic safety.

1. Introduction

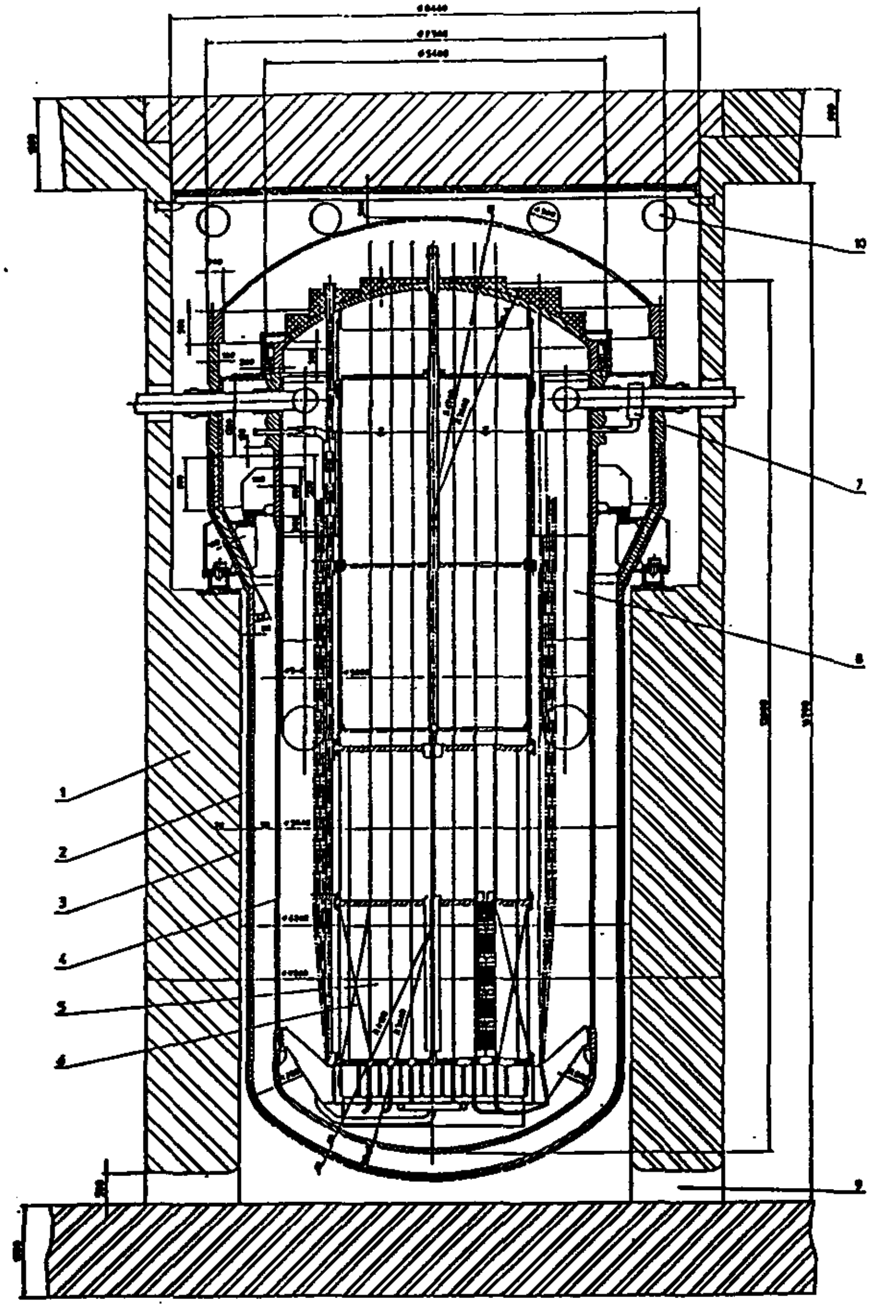
The 200 MW Low Temperature District Nuclear-Heating Reactor (LTHR-200) is used for district heating in cities. An integral arrangement is adopted for the components of its primary circuit system. Reactor core, primary heat exchangers, control rod drives and spent fuel elements are all put in the reactor pressure vessel. The control rods are driven by a hydraulic cylinder. The primary coolant goes through the reactor core, the primary heat exchangers and then returns to the reactor core in natural circulation. It is not necessary to have primary pumps, pipings and pressurizer among the primary system components. The reactor pressure vessel forms the main part of the primary coolant pressure boundary of LTHR-200. Figure 1 shows the arrangement of LTHR-200.

Primary coolant pipes penetrating the wall of the containment, for example, water pipes for control rod driving, water pipes of the primary coolant purification system, pipes of pressure relief, etc. are all of small diameters. The maximum diameter of these containment penetrations is 100 mm. Two isolation volves on each penentration for primary coolant are separately put inside and outside the containment wall. In addition to the reactor vessel, only composite electromagnetic valves of control rod driving systems, small pipes, isolation valves and some other small things like cable plugs etc. are inside the containment. In this case the normal huge containments used for PWR and BWR are not suitable and a small sized metallic containment with a full sized closure head can be used for LTHR-200.

The containment is supported inside the biologic shielding. The reactor vessel is supported on the containment wall. The composite electromagnetic valves and all other internals of the containment are arranged in the upper part inside the containment.

Thermal insulations are attached to the lower wall of the containment and to the upper wall of the reactor vessel. The arrangement of thermal insulations is showed in fig. 1. In normal operating condition the temperature is not higher than 130°C in the upper space inside the containment, and the temperature is about 100°C in the upper part of the containment wall. Under these temperatures the electrical installations and sealing parts are working normally.

There is a narrow gap between the concrete wall of biologic shielding and the thermal insulation of the containment. Air comes through the lower ventilating holes into the reactor cavity inside the biologic shielding concrete wall, then goes up around the containment and out of the biologic shielding from the upper ventilating holes by natural convection. Most of the



1	Biologic Shielding	6	Storage of Spent Fuel Elements
2	Thermal Insulation	7	Composite Valve of Control Rod Drive
3	Containment	8	Primary Heat Exchanger
4	Reactor Pressure vessel	9	Lower Ventilating Hole
5	Reactor Core	10	Upper Ventilating Hole

Fig. 1. Arrangement of LTHR-200, containment and primary shielding.

heat from the containment is removed by the air. The temperatures in the biologic shielding concrete are kept below 70°C.

2. Considerations about a small sized metallic containment

On account of the integral arrangement and excellent safety property of LTHR-200, the adoption of a small sized metallic containment is suitable. That brings about some benefits. The construction period will be shorter, overall cost will be lower and the safety properties will be better than by using a big containment. Compared with a big containment, at least some points as follows could be considered.

2.1. Low cost and short manufacturing period

A big containment used for a PWR or a BWR is manufactured and fabricated on the construction site. But a small metallic containment can be made in a factory. That would directly lead to the results including easy manufacturing, good quality, low cost and short manufacturing period.

2.2. Expected low leakage rate

It is different from the containment for a PWR that the air locks are not necessary for the LTHR-200 containment. Joints between penetrations and containment wall are reliably sealed. It is expected that the overall leakage rate of the containment would be quite low.

2.3. Ease for pressure test and leakage rate measurement

The volume of LTHR-200 containment is very small. It is only about 1% of the containment for a PWR. This small volume makes the containment easy to be pressurized and the inner pressure variation will be more sensitive to leakage rate than a big containment during an air pressure test and leakage rate inspection. The test and measurement can be performed in a short time, for instance within 10 hours.

2.4. Without loss of primary coolant accident (LOCA) and core melt down accident during break accidents of primary circuit pressure boundary

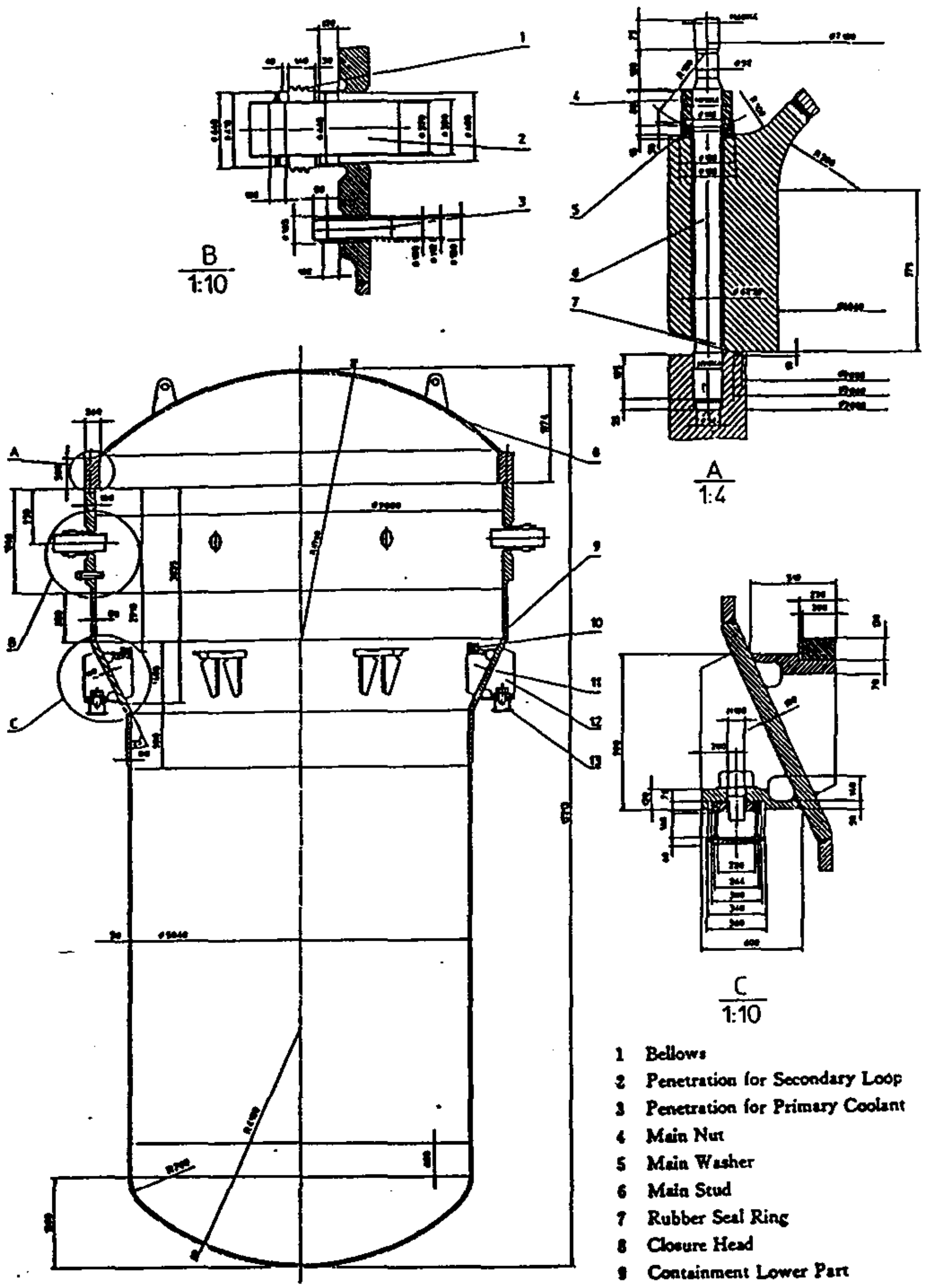
The air pressure inside the containment is 1 bar in normal condition. Because of the small volume, if the most serious accidnet should happens, a break occurs on the bottom of the reactor vessel, primary coolant will gush from the reactor vessel into the containment. In this case, accident analysis shows that the amount of primary coolant out of the reactor vessel will get to a maximum. But the level of primary coolant inside reactor vessel will be at least 1500 mm over the reactor core during the accident. In fact, if the air space in the containment were fully filled with primary coolant, the level will still be 500 mm higher than the reactor core.

2.5. Advantageous to leakage monitoring of primary coolant and to break monitoring of primary pressure boundary

Primary coolant leakage monitoring and primary pressure boundary break monitoring of a PWR are difficult or not sensitive by the methods of water level measurement in the reactor vessel, and pressure measurement and temperature measurement of the containment air. But the monitoring of LTHR-200 is much easier and more sensitive. The air space of LTHR-200 containment is so small that if one kg primary coolant leaks into the containment, the air pressure will rise in 0.01 bar only. Pressure monitoring of the containment air will give an obvious indication for the pressure variation. By means of the pressure monitoring of the containment air, the break accident of primary pressure boundary can be judged and the leakage rate of the primary coolant into the containment can be estimated.

2.6. Simplifying in-service inspection of the reactor vessel

The break of the primary pressure boundary of a PWR and a BWR will probably cause a LOCA and a core melt down accident. Especially if a break of the reactor vessel happens, the accident of core melt down would be unavoidable. That is not allowed. To avoid a break of the reactor vessel, an ultrasonic test is required during in-service inspection. LTHR-200 is distinguished from a PWR and a BWR in view of a break accident of the reactor vessel and the consequences. No LOCA and no core melt down accidents would be caused by the break of the reactor vessel. In addition, the integral flux of fast neutrons in the core belt of the reactor vessel is only about 1×10^{16} n/cm^2, much lower than in case of a PWR and a BWR vessel. Operating conditions are advantageous. The break probability of the reactor vessel is very low. Based on the facts mentioned above, simplifying in-service inspection for the vessel is reasonable. An ultrasonic test would not be necessary any more.



1 Bellows
2 Penetration for Secondary Loop
3 Penetration for Primary Coolant
4 Main Nut
5 Main Washer
6 Main Stud
7 Rubber Seal Ring
8 Closure Head
9 Containment Lower Part
10 Bearing Block
11 Support Stand for Reactor Vessel
12 Support Bracket
13 Bearing Stand

Fig. 2. LTHR-200 containment.

3. Description of the design of the containment structure

Figure 2 shows the LTHR-200 containment structure. The bottom of the containment is a big closure head. The closure head is bolted onto the cylindrical containment shell. The upper part of the shell is expanded with a diameter of 7000 mm. All containment penetrations are arranged in the upper part for ease of installation, inspection and maintenance. The lower part of the shell is reduced to a diameter of 5840 mm. A cone-type transition part is in the middle of the shell. Six containment support brackets and six support stands for the reactor vessel are welded to the transition part.

The main parameters of the containment are as follows:

Material	16MnHR (or SA516),
Height	15213 mm,
Outer diameter	7360 mm,
Weight	210 ton,
Design pressure	15 bar,
Design temperature	200°C.

Weld joints between penetrations and containment wall are adopted as far as possible. Rubber seal rings are used for sealing between flanges, including the seal between closure head flange and upper end of the containment shell.

The stress analysis method is used in the design of main parts of the containment. Design, material selection, fabrication and examination have met the requirements for metallic containment in accordance with ASME BPV-III-I-MC. The structural integrity of the containment is ensured in full life and in all credible operating conditions.

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