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STRUCTURAL EVOLUTION OF CONTAINMENT FOR INDIAN PHWRS

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Pressurised heavy water reactors form the mainstay of the first stage of the Indian Nuclear Power Programme. Each of these units is provided with containment as an ultimate barrier for escape of radioactivity between the inside of the reactor building and the atmosphere. These containments have thus the function of limiting leakage of gases from inside the reactor building to the external atmosphere under the most adverse accidental conditions. To achieve this, beside proper design and construction of containments, suitable energy management features are also incorporated to minimise the pressure and temperature buildup during an accidental condition.

Present paper deals with the step by step evolution of containments for Indian PHWRs. This starts from the use of single containment with energy management system based on active components such as dousing system, to a complete double containment with passive energy management system in the form of vapour suppression pool. This evolution is based on design criteria and functional requirements as well as various research and developmental work done in different areas related to containment design.

The text initiates the discussion with the brief description of containment systems for different power reactors in India. This is followed by the design criteria and the functional requirements adopted for the containments. Identification of different areas chosen for developmental work and their influence on the design/construction of our containments are described next. It concludes with the indication of future thrust in this area.

1. Introduction

Structural evolution of containment for Indian PHWRs is a story of realisation of improvement in design/construction/performance/reliability of a system through systematic developmental work done in various related areas. These areas include the stress analysis, seismic analysis, thermal hydraulic and blowdown analysis, hydrodynamic analysis etc. Beside these, proper choice of construction material, improvement in construction procedure and proper layout of components so to have less interference with containment system, have also contributed to this evolution.

To obtain a safe and reliable system, it is first necessary to identify various loads which can come onto the containment system. This requires postulation of various accidental conditions. These conditions may be due to accident in reactor system or also due to abnormality in environment, such as earthquake. In case of PHWRs, the accidental conditions in a reactor system may be due to coolant channel break or coolant header break, which will cause pressure and temperature build up in the containment. With the choice of vapour suppression pool as energy absorber above accidents may also lead to hydrodynamic loads on the containment system. Other accidental conditions in a reactor system may be due to hydrogen blast in some pockets of the enclosed atmosphere. In order to check blast resistance capability in the adjoining structure, the pressure time history is required to be constructed and the structure should be analysed for this time history. Other category of loads may be due to jet load arising out of a rupture of a pipe lying nearer the containment wall. This may also lead to the loading due to impact of moving pipe and any missiles generated consequent to a pipe rupture. To assess the effects of loads due to earth quake, necessary response spectra is to be generated. This makes it dependent on the site and hence changes for every reactor location. This also inhibits the process of standardisation of the design.

After identifications of loadings, it is necessary to choose proper configurations, construction materials and design criteria depending upon the functional requirements. The choice of containment configuration and construction material have been done keeping safety as the top most concern, which some times over-shadowed

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the economic considerations. The latest design calls for a complete prestressed double containment, with minimum openings for less leakages. For design criteria, it is necessary to decide the load combinations arising out of many events and the design code to be followed. In India ASME code section 3 along with IS code are adopted for this purpose. A detailed stress analysis by finite element technique is done to locate zone of high stresses and configuration is modified suitably to bring down the stresses. Experimental stress analysis is also done to generate necessary inputs for the analytical studies.

Testing of the containment system is done after the construction for a test pressure, which is 15% higher than the maximum pressure it has to see during most severe accidents. Integrity of each part of the containment is checked during testing and percentage leak rate is assessed to compare with the design value.

2. Brief descriptions of containment systems for Indian PHWRs

The pressurised heavy water type power reactor was first constructed at Kota in Rajasthan (RAPP). This is a 200 MWe twin unit station, which is basically a heavy water cooled and heavy water moderated reactor. This reactor has reinforced concrete containment wall which is 1.2 m thick (fig. 1), the thickness of wall has been mainly from shielding considerations. This containment has been designed for 0.51 kg/cm^2 pressure when max-

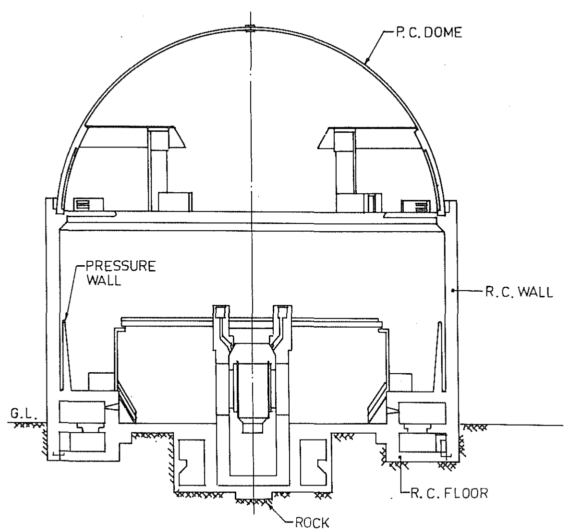

Fig. 1. RAPP containment system.

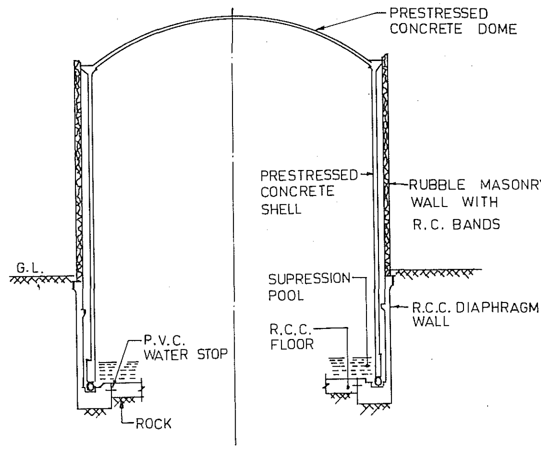

Fig. 2. MAPP containment system.

imum accidental pressure inside is expected to be limited to 0.42 kg/cm^2. This containment also has a prestressed concrete dome. To limit pressure and temperature during a severe accident, a "dousing water system" is used. For this purpose, building houses 400 000 gallon capacity dousing water tank near its top.

Subsequent to this the Madras Atomic Power Project (MAPP) came under construction, which is a 235 MWe PHWR twin unit station. In this case, the improved design philosophy used a suppression pool at the base level, instead of dousing water system. This containment is designed for 1.16 kg/cm^2 pressure. An additional feature of MAPP is the use of two containment walls replacing one wall used at RAPP (fig. 2). The outer wall has been constructed in stone rubble masonary and does not cover the dome area. This rests on a R.C. diaphragm wall. The inner wall along with the dome is a prestressed concrete wall. The inner wall and diaphragm wall are founded on a ring raft. The advantage of double wall containment is that the leakage is only into the annulus gap between R.C. shell and outer shell in masonry as this gap is connected to the ventilation system and always kept under a slight negative pressure. The leakage would be into the gap either from primary containment or from outside atmosphere. From this consideration, this containment is expected to function more effectively.

Subsequent to this, we undertook the design and construction of a nuclear power station at Narora. This station has altogether different foundation conditions. The station is founded in alluvium zone of Ganga basin and located in zone-IV as per seismic zoning map of Indian standard. The experience gained on earlier power stations was utilised in modifying the station. The main modification in containment system came in the provision of internal containment structure in fully prestressed concrete with a roof of flat cellular concrete slab also in prestressed concrete. The other containment building, which is in reinforced concrete, covers the roof area also (fig. 3). This change was necessitated from seismic considerations. Both inner and outer containment walls are founded on a common 4 m thick concrete raft. In this reactor the steam generators are mounted such a way, so that a part of the steam generator is inside the primary containment and the part outside the primary containment. A number of similar units are also under construction.

While the above projects were in progress, the department took the decision to go ahead with the reactors of 500 MWe capacity. The containment system underwent a detailed review and following major changes are proposed to be made in the basic concept of the containment (fig. 4).

(a) Fully double containment building.
(b) The entire reactor system and the steam generators are taken inside the primary containment.
(c) The inner prestressed building is provided with prestressed concrete dome.
(d) Four openings are provided in the dome for steam generator erection to offer greater flexibility during erection/removal.

3. Functional requirements and design criteria

Before we move into a discussion on the developmental work done on various related topics, a look at the functional requirements and design criteria for Indian PHWR containment systems would be in order.

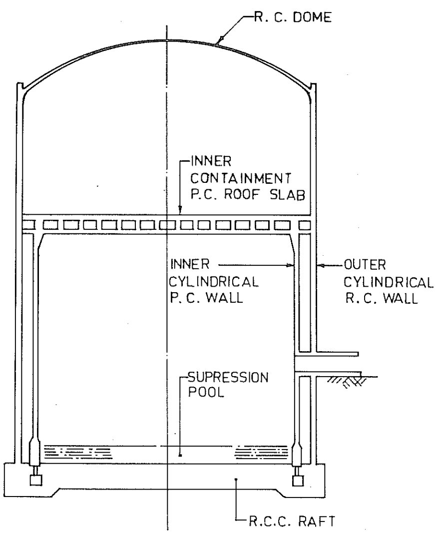

Fig. 3. NAPP containment system.

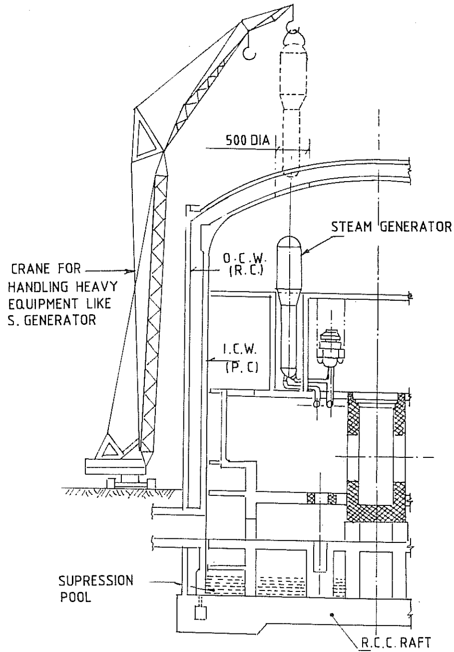

Fig. 4. 500 MW containment system.

The functional requirements are

(a) To completely isolate the plant from the external atmosphere in case of an accident.
(b) Minimise ground level leakage to near zero value during an adverse accidental conditions.
(c) Provide radiation shielding.
(d) Eliminate active components in so far as working of energy management feature is concerned.
(e) Enable better visibility conditions in accessible parts of the reactor building to permit safe escape of personnel following an accident.

Structural integrity of the containment has to be assured during the following postulated initiating events.

(a) Loss of coolant accident (LOCA)
(b) Safe shutdown earthquake (SSE).

The other initiating events such as aircraft impact have so far not formed part of design basis as they get eliminated on grounds of low probability of occurrence through the use of siting criteria currently in vogue. Extreme flood and wind loads may, however, become important in some cases.

Different types of loads (some of them correspond to above mentioned two initiating events) and their load combinations that have to be considered for an acceptable design are given in table 1.

The design is based on the working stress method for all combinations. The amount of prestressing force applied is such that under even worst combinations of loading the net compressive stresses are maintained in the concrete at critical sections.

4. Developmental work

As mentioned above, a series of developmental programmes have been implemented to evolve the containment configuration and its associated energy management features towards a better, safer and more reliable containment. Above mentioned structural evolution of different containments from reactor to reactor is the result of these developmental activities. Here we give a brief summary of these work for important areas.

4.1. Work related to static stress analysis

Stress analysis of containment for different load combinations constitute a major design procedure for every containment system. Finite element method is used for this purpose. Well known computer codes such as SAP4, PAFEC, SESAM along with other indigenously developed codes are used for this purpose. The analysis is done mainly in two phases. The first phase is essentially for fixation of profile and dimensions of geometry of different parts of the containment. This phase consists of three analyses, such as axisymmetric model analysis with all axisymmetric loadings, local analysis to pick up stress multipliers for non-symmetric openings such as air locks, holes in dome etc. and finally an analysis on a simplified model to pick up stresses due to nonsymmetric loadings. Superimposition of stresses by these three analyses gives the actual picture of stresses in the containment. Modifications of profile and dimensions are done at different levels of working whenever stress is found to be higher than its limiting value. Figs. 5A and 5B show deformed shape of the 500 MWe containment for test pressure and also due to accident pressure and temperature respectively. Phase-2 analysis correspond to a complete 3-D model for the containment considering all the openings at a time. All the individual loads and their combinations are imposed on this model and stresses are analysed critically. The effect of interaction between near by openings can be picked up by this analysis.

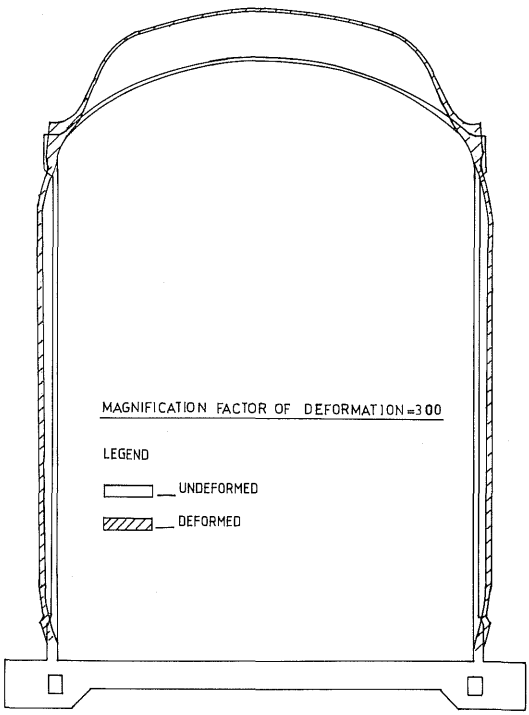

Fig. 5A. Deformed shape of 500 MW containment.

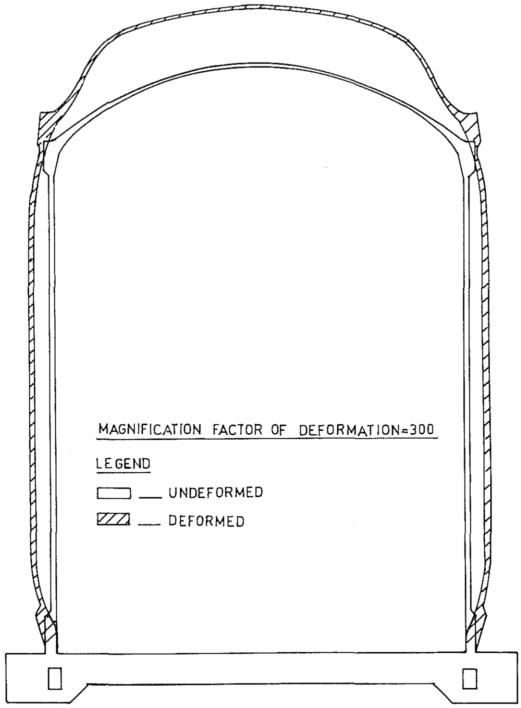

Fig. 5B. Deformed shape of 500 MW containment under accidental pressure and temperature.

Table 1

Loads	
1. Dead load	D
2. Live load	L
3. Temperature during normal operation	To
4. Temperature during leak test	Tt
5. Accident pressure	Pa
6. Test pressure (15% more than accident pressure)	Pt
7. Peak temperature under accident condition	Ta
8. Loads due to operating basis earthquake	Eo
9. Loads due to safe shutdown earthquake	Es
10. Wind load	W
11. Prestressing	F

Load combinations	
1. Normal	D + F + L + To + Eo(or W)
2. Test	D + F + Tt + Pt
3. Extreme environmental	D + F + L/2 + Es + To
4. Abnormal	D + F + Pa + To(or Ta)
5. Extreme environmental	D + F + L/2 + Pa + To(or Ta) + Es

4.2. Work related to seismic analysis

A large part of our country lies between moderate to active seismic zones. A seismic design of containment is therefore of vital importance to safety of our power plants. Two levels of design earthquakes are considered in the design. Safety related structures are designed to withstand the safe shutdown earthquake (SSE). In addition to these, structures are checked for the operating basis earthquake (OBE). Such dual earthquake design basis is considered important in view of different allowable stress limits as well as different damping values that are relevant for the two levels of earthquake loading. It is common practice to keep operating basis earthquake to 50% of the safe shutdown earthquake. Depending on the availability of historical data the level for the OBE can be fixed at a value which can reasonably be expected to occur once during operating life of the plant.

It is generally found necessary to analyse internals along with containments seismically. This is due to connections of internals with containment at some of the floor levels. Various analyses are done to calculate moments and shear forces on different models. These models are 1-D beam, equivalent axisymmetric 2-D and also 3-D models. Major task in case of 1-D and 2-D analyses is the calculation of moment of inertia and shear area at different levels of reactor building. This

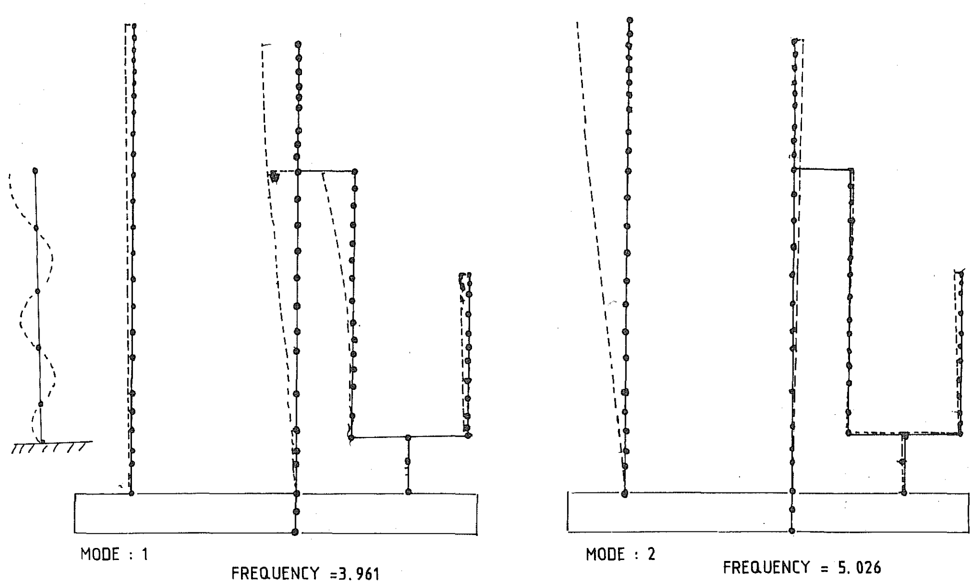

Fig. 6A. Mode shapes of 500 MW containment with internals from 1-D analysis.

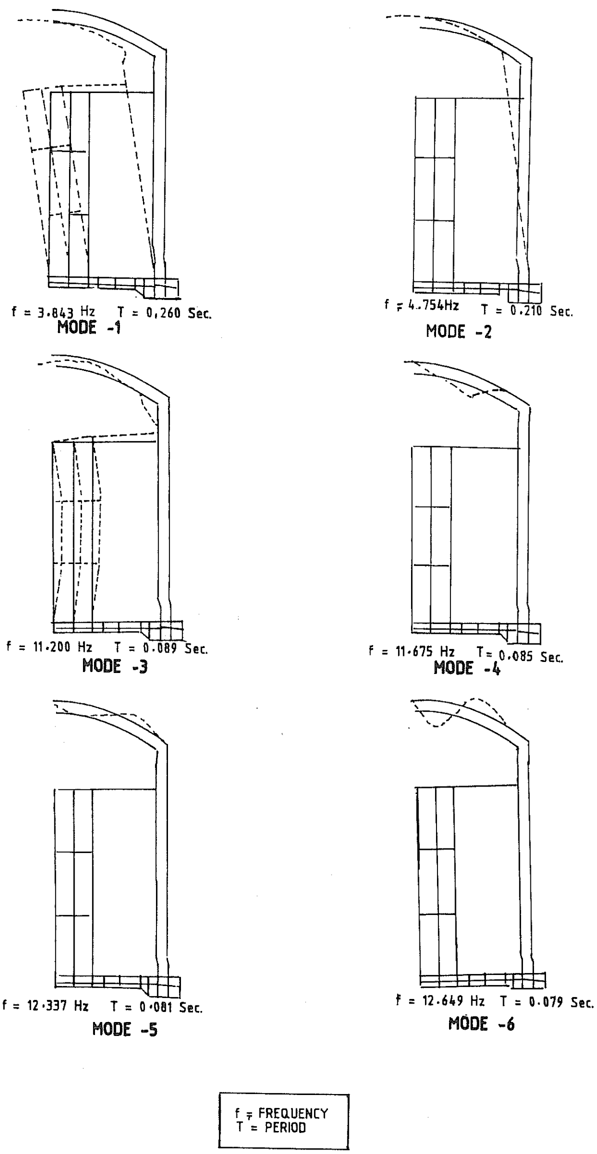

Fig. 6B. Mode shapes of 500 MW containment along internals from equivalent axisymmetric analysis.

sometimes necessitates local analyses and experiments. Fig. 6A shows the mode shapes as obtained from 1-D model of 500 MWe reactor containments and internals. Same thing is shown for 2-D equivalent axisymmetric model in fig. 6B. Fig. 6C shows the comparison of displacements, accelerations, shear forces and bending moments for inner and outer containments as obtained from 1-D and 2-D models.

In contrast to the loading that occurs during the loss of coolant accident the seismic loads are site dependent and unless a suitable standardisation procedure is adopted, a fresh design exercise is inevitable for each nuclear power plant site. A way out of this is to work out the envelope spectra for various sites that are likely and to establish a standardised design.

4.3. Work related to energy management features

One of the postulated events for the containment design is loss of coolant accident (LOCA). In PHWRs the rupture of primary heat transfer system (PHT) inlet header is felt to be the most severe loss of coolant accident. Hence, it is necessary to assess different types and magnitudes of loadings on containment during a PHT header rupture.

To limit pressure and temperature buildup in the containment during LOCA, it is found necessary to have a suitable energy management feature. This may be in terms of releasing a large quantity of spray water as in case of dousing tank system or by forcing steam air mixture to go through a pool of water and in turn condensing the steam. Though the containment at RAPP reactors is of dousing tank type, the same for MAPP onwards is of vapour suppression pool type. Hence, here we concentrate on the developmental work done related to vapour suppression pool type energy management features.

4.3.1. Pressure–temperature buildup

The pressure and temperature histories within the containment volume are primarily dictated by the containment volumes on either side of suppression pool and resistance of interconnecting passages vis-a-vis the total mass and energy released into the containment consequent to a header rupture. To assess the pressure–temperature buildup, the developmental work is done in theoretical as well as experimental fronts. Theoretical work consists of the development of computer codes for calculating discharge and energy release rates following a PHT header rupture and also pressure and temperature transients in containment building because of corresponding discharge and energy release rate.

The experimental setup consists of PHT and containment models. The PHT model consists of a cylindrical shell having three compartments filled with water to simulate different masses of heavy water in reactor at different average temperatures. These three compartments are connected with each other by pipes simulating actual resistances in reactor. A rupture disc is

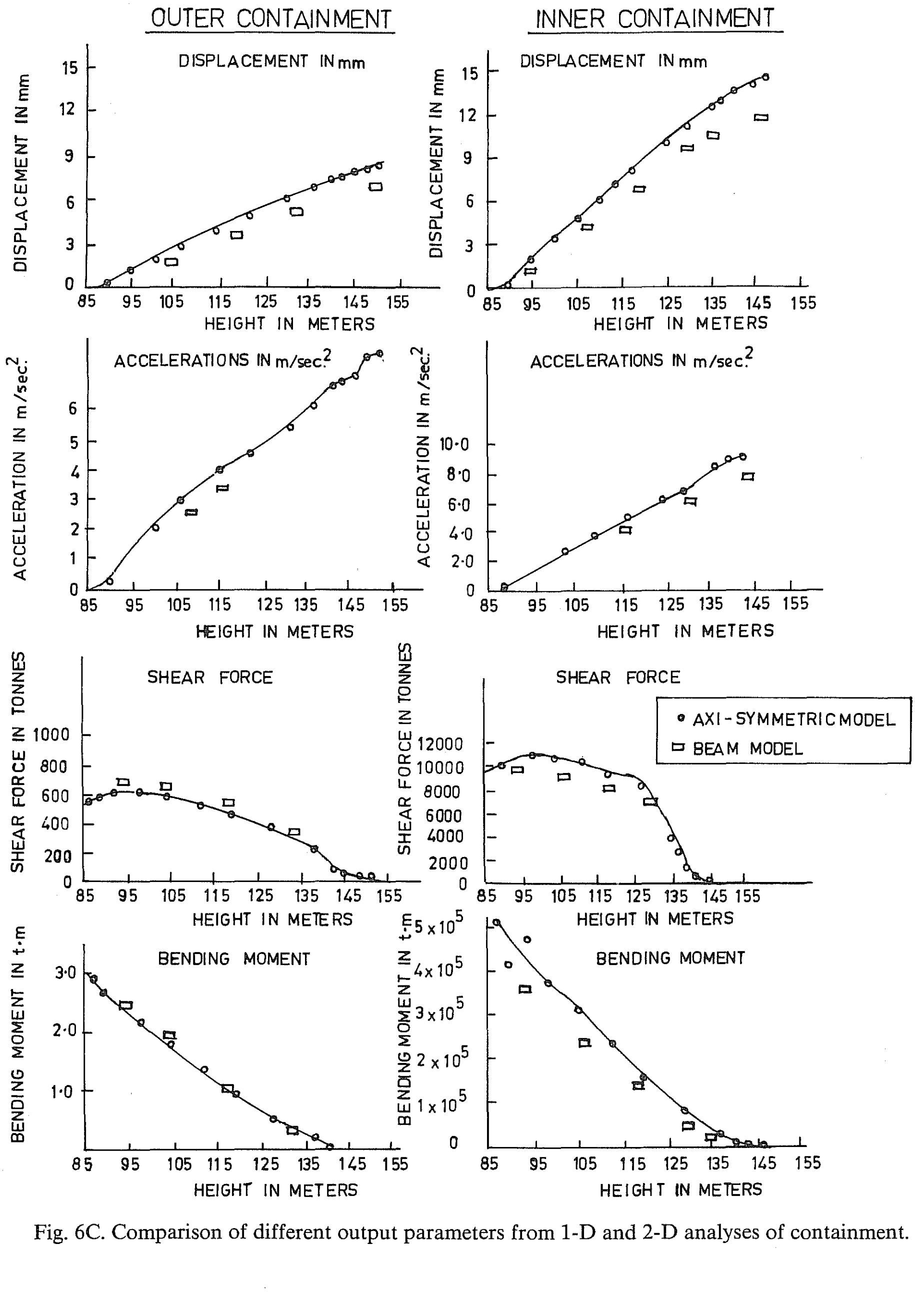

Fig. 6C. Comparison of different output parameters from 1-D and 2-D analyses of containment.

mounted in one of the compartments to simulate PHT header rupture. The discharge from this model is sent to containment model at a proper location. The pressure–temperature transients are then recorded by fast recorder at different points of the containment model. Fig. 7A shows the PHT model, whereas 7B shows the pressure transients as obtained experimentally for different temperatures of reactor inventory. Figs. 8A and 8B show typical pressure-temperature transients in containment following LOCA.

4.3.2. Vapour suppression pool hydrodynamics

In the event of LOCA the discharge of high pressure temperature water flashes into steam in the drywell space and forces the air–steam mixture in the suppression pool via vent system. The containment and the

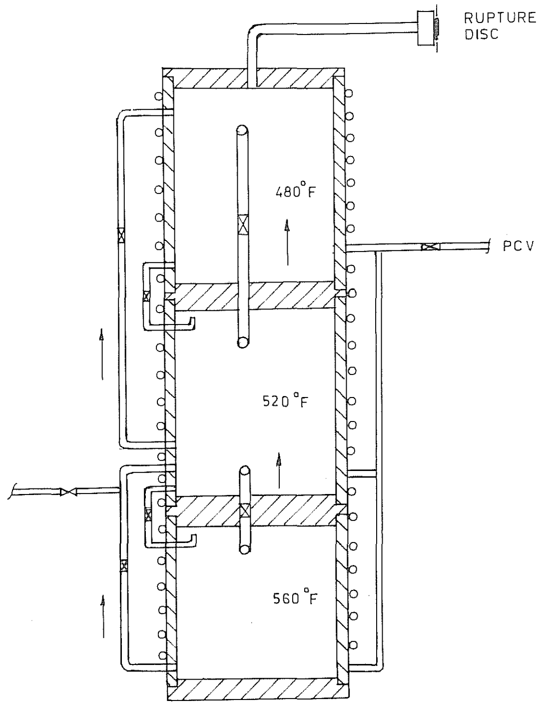

Fig. 7A. PHT simulated model for containment pressure temperature transient experiment.

pool internals need to withstand the hydrodynamic loading due to jet impingement and general motion of pool water during water clearing phase from the vents. Fig. 9 gives a schematic representation for the poolswell and vent clearing along with the typical poolswell and vent clearing transients that have been analysed. The results are compared with published experimental results. Chugging phenomena is expected to occur when steam flow rate through vent pipe becomes lower than around 30–35 kg/m^2 s. In PHWRs, in view of large drywell volume one rarely reaches conditions of pure steam condensation and as such severity of chugging phenomena is expected to be lower. A quantitative assessment of this can however be made only through detailed experimentation.

4.3.3. Hydrogen explosion and blast loading

In PHWR design, the volumes available within the containment vis-a-vis the total quantity of hydrogen likely to be liberated are so large that uniform hydrogen concentration to the levels of explosive limits are unlikely to be reached and as such global explosion of hydrogen within the containment is not postulated. A local pocket involving hydrogen concentration beyond the threshold and possibility of a local blast loading, however needs to be evaluated and carefully considered

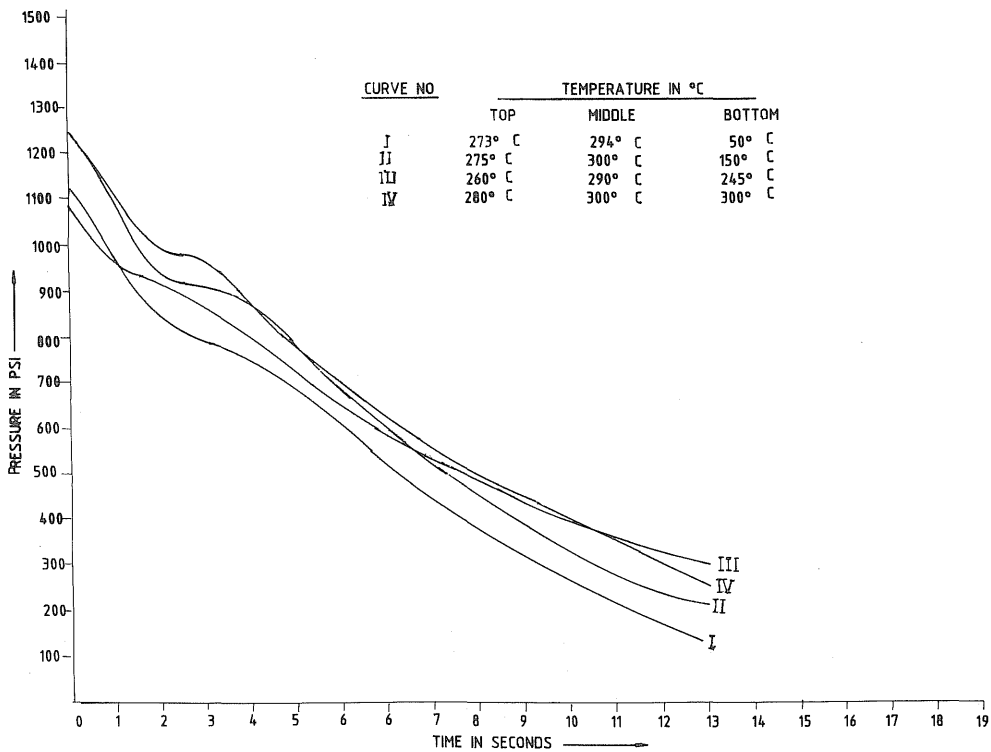

Fig. 7B. Pressure transient history in PHT simulated model for different compartment temperatures.

in design. In order to check blast resistance capability in the adjoining structure the pressure time history is required to be constructed. In the mean time guidelines for assumption of capacity of nuclear power plant structure to resist blast loading as given in US-NRC document NUREG/CR2462 can be used.

4.3.4. Rupture loads

An additional load during the loss of coolant accident that needs to be accounted for is the load arising out of fluid jet reaction or fluid jet force that would occur consequent to pipe rupture. This load will affect the ruptured pipe as well as nearby structures through jet impingement as well as the loading that would be caused by impact of moving pipe and any missiles generated consequent to a pipe rupture. Various design philosophies are in vogue for assuming safety against pipe rupture. One way is to provide adequate pipe supports suitably located to take care of all possible rupture locations. This approach tends to cause severe hindrance to the inservice inspection programme on such piping. However, the problems of inservice inspection may outweigh the protection provided by such supports and hence the new philosophy which gives larger emphasis on superior quality design, manufacture and inservice inspection is fast becoming popular. Fig. 10 shows a typical time history of the blowdown force.

As mentioned earlier missile loads arising out of aircraft impact are not considered in design on account of very low probability for such loading. However, internally generated missiles due to rupture of high energy piping or due to failure of rotating component would still arise. A force-time history is required to be constructed for analysis of the structure for missile loads. This time history is dependent on the velocity and stiffness characteristic of missile. Work to establish quantitative analytical procedures for such analysis is in

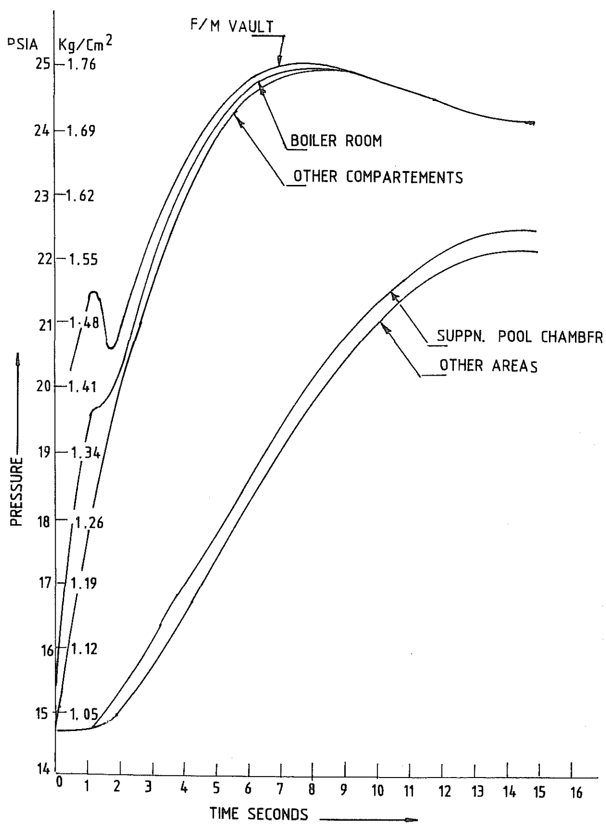

Fig. 8A. Reactor building pressure transient following LOCA.

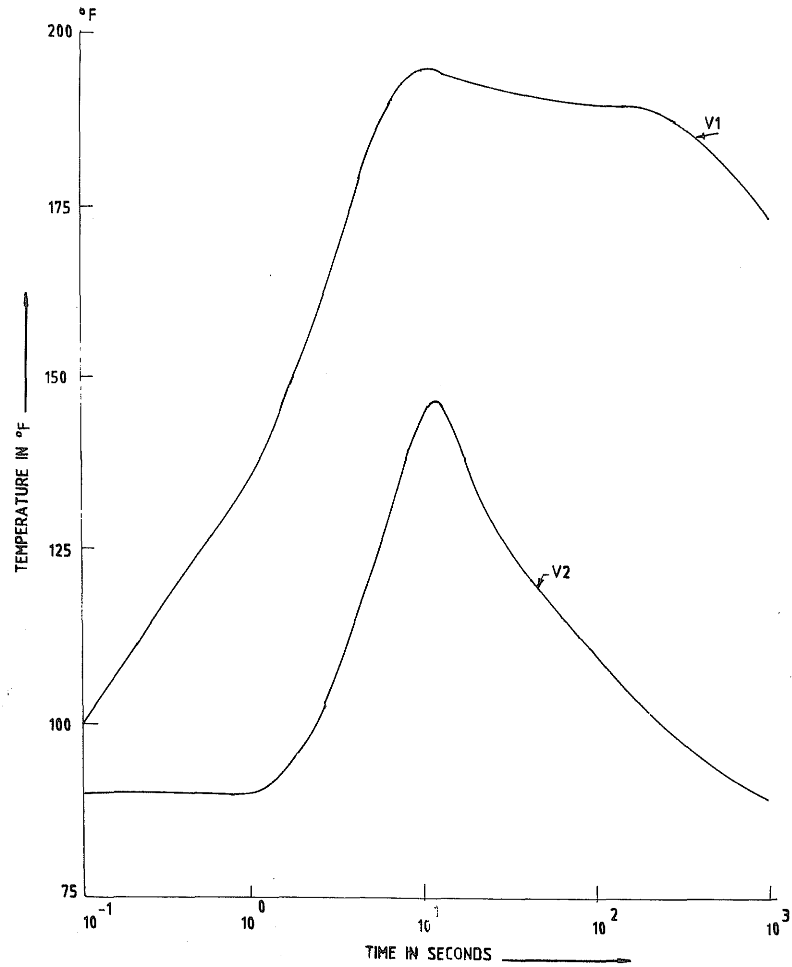

Fig. 8B. Reactor building temperature transient following LOCA.

progress. In the mean time a greater emphasis has to be given on protection by physical barriers.

5. Conclusions and future work

The development work done in various related areas has yielded a safe and reliable containment system for Indian PHWRs. It has been decided to keep the concept of prestressed double containment and vapour suppression pool for our all future reactors. The number of openings may vary from site to site depending upon the convenience, but their number will be kept small for better leak tightness.

In India the progress in nuclear power reactors is in two fronts. On one side the country is in a position to produce a number of 235 MWe PHWRs and in parallel the necessary developmental work has been started for the construction of a series of 500 MWe reactors. The nature of developmental work now required for our containments are also of two types. For 235 MWe reactors, though the procedure for design, construction and testing are well known, but for every containment this cycle is to be repeated and should be completed in the shortest period of time. This requires development of necessary package for standardisation of CAD/CAM design procedure, inspection procedure during construction, quality control and acceptance procedure from the test data.

The developmental work related to 500 MWe PHWR containment is still in progress. Necessary data are being collected from thermal-hydraulic analysis, hydro-

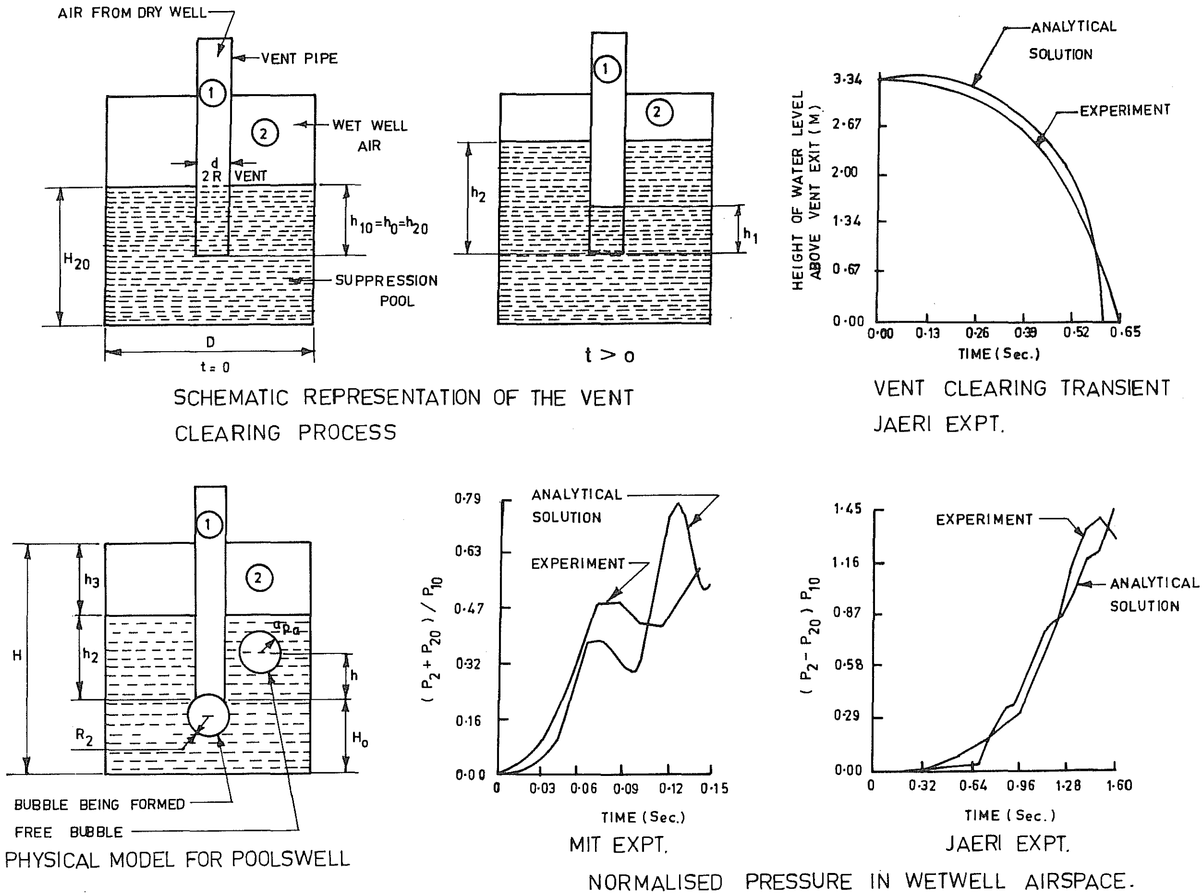

Fig. 9. Hydrodynamic load calculations in vapour suppression pool following LOCA.

dynamic and chugging analysis of vapour suppression pool, seismic analysis and also stress analysis of different parts of the containment to check their integrity under different combination of loadings. Number of openings are being decided to facilitate movements of the large size components. An effort is being put to

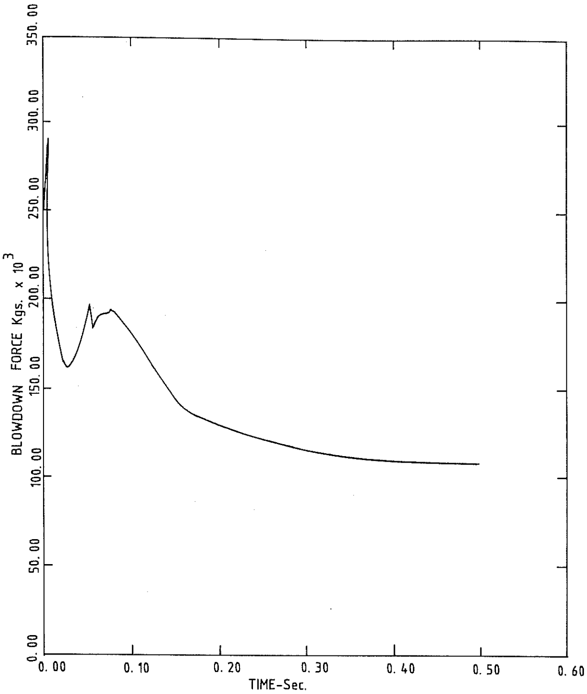

Fig. 10. Blowdown force for circumferential break.

develop a seismic analysis procedure to make the design of the reactor site independent. This is done by working out the envelop spectra for various sites.

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