"Evolution of Nuclear Reactor Containments in India: Addressing the Present Day Challenges"

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Key words

Protection of public, chronological sequence of containment structural design development, energy management and thermal hydraulics assessment with dousing, vapor suppression and spray systems, engineered safety features, beyond design basis and extreme events, ultimate load capacity and BARCOM round robin, aircraft impact and external threats, hydrogen management, catalytic recombiner development, designs with and without liner, radiological consequences and present day challenges, public acceptance

Abstract

Indigenously developed Pressurized Heavy Water Reactors (PHWRs) that form the backbone of current stage of nuclear power development in India have seen continuous evolution of their containment systems. This evolution that has taken place over implementation of 18 PHWRs (200/220/540 MWe) has encompassed all aspects of containment design, viz. the structural system, energy management system, radio-activity management and hydrogen management system. As a part of ongoing efforts towards strengthening of safety performance, India is also ready with the design of Advance Heavy Water Reactor (AHWR), which represents a technology demonstrator for advanced reactor systems and for thorium utilization. This reactor has a number of improved passive safety features and it is capable of meeting the demanding safety challenges that future reactor system would be expected to meet as a result of emerging expectations in the background of accidents over the past three decades viz. those at Three Mile Island (1979), Chernobyl (1986) and most recently at Fukushima (2011). In this lecture I shall focus on the evolution of nuclear reactor containments in India and highlight the design, associated structural and thermal hydraulics safety assessment made over the years for the improvement of containment performance.

1. Introduction

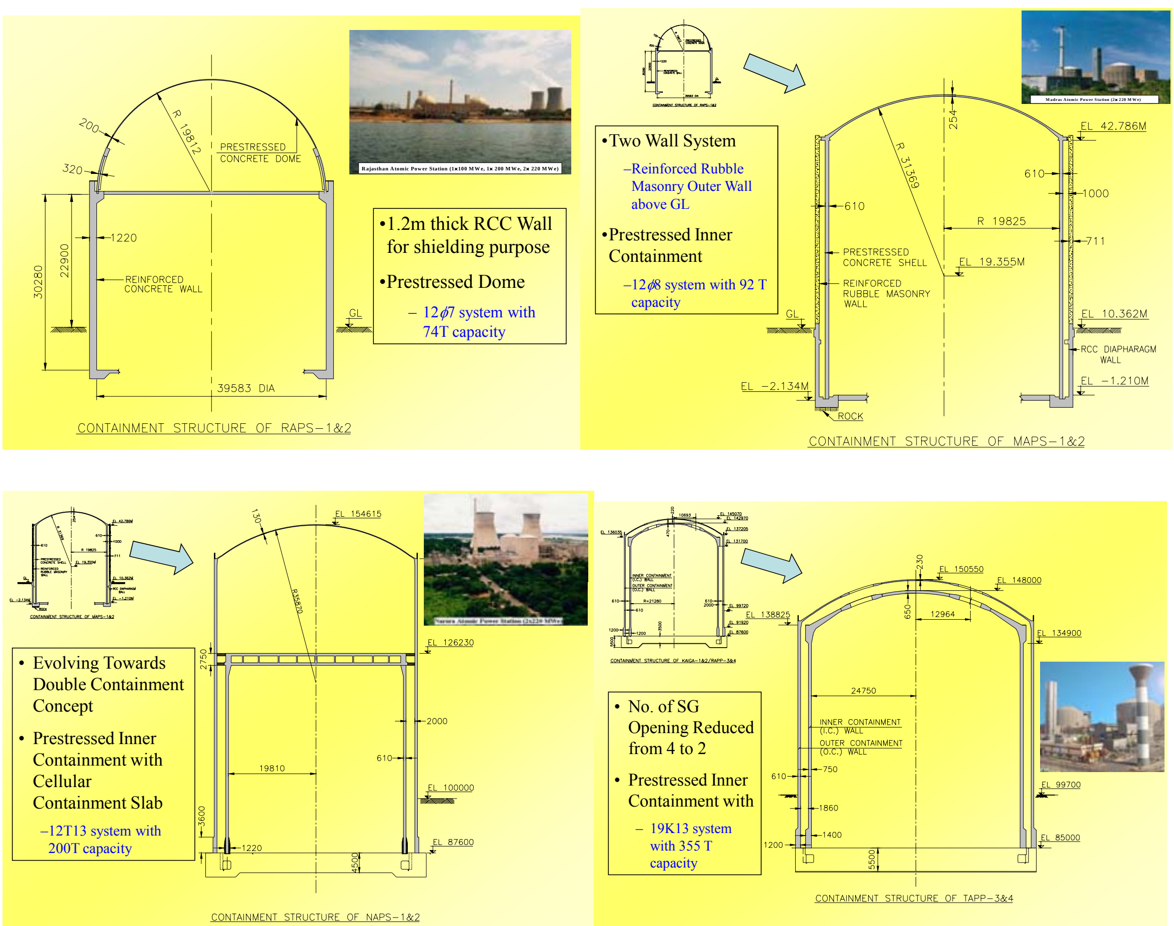
It is an honour for me to present the Thomas Jaeger Lecture at this SMiRT-21 conference here in India. Being a reactor engineer with structural engineering background, I have been following SMiRT conferences almost since their beginning. They have been an important platform to look at structural mechanics comprehensively in the broader context of developments on nuclear reactors around the world. Clearly SMiRT has made an important contribution to development of this field in India and indeed all over the world. Bringing SMiRT to India has been an idea that we have pursued for a long time. I am happy that the conference is taking place here in India and I would like to thank IASMiRT for their agreement to do so. I would also like to compliment Dr. B.K. Dutta and all other colleagues of mine for their effort in organizing this conference.

Nuclear reactor containment constitutes the most crucial barrier to protect surroundings of a nuclear reactor from a reactor accident as also to protect the reactor from external events. Progressively the containment designs in conjunction with the reactor designs are being improved to keep the exposure of people to within acceptable limits even for severe accidents. With Linear No Threshold (LNT) philosophy as the basis for radiation protection, defining acceptable intervention levels for protection of people in public domain has become a challenge in view of the large integrated consequences that one would calculate. This leads to evacuation of a very large population to safe locations in the event of a severe accident causing public trauma. Accidents at Chernobyl and recently at

Fukushima have shown that the trauma caused in public mind is disproportionately large in relation to actual risks to which the population is exposed. Chernobyl has provided evidence of larger health consequences as a result of such traumatic experience.[1] While the scientific debate on this issue will hopefully lead to a satisfactory resolution some day, further improvements in the design of the reactor and containment systems to further minimize and if feasible eliminate the impact in public domain would be a step in right direction. In the evolution of the Indian nuclear power programme, containment design has continuously evolved in this direction. The latest Advanced Heavy Water Reactor [2] and its containment in fact realize the objective of near zero impact in public domain.

2. Early evolution in containment system design

The containment system of Indian PHWRs has seen several modifications in successive projects. The first pair of Indian PHWRs at Rajasthan Atomic Power Station (RAPS) had a single containment envelope of reinforced concrete cylindrical section with a pre-stressed concrete dome. In RAPS, pressure suppression in the containment is achieved by dousing of the containment volume with water stored in a large tank, at the topmost elevation.



Evolution of PHWR containments

Actuation of fast-acting dousing valves on sensing release of high enthalpy water in the containment is necessary. In Madras Atomic Power Station (MAPS) and all subsequent 220 MWe and 540 MWe plants, active dousing for pressure suppression is replaced by passive vapour suppression pool system located at the bottom of the containment. In MAPS, a partial double containment was used with primary containment of pre-stressed concrete, and secondary containment of rubble masonry around the cylindrical portion. From Narora Atomic Power Station (NAPS) onwards, all reactors have full double containment. In Narora and Kakrapar Atomic Power Stations, for effective separation of light water and heavy water areas, the light water portion of steam generators were kept protruding out of the primary containment through a flat roof with provision of a sealing bellow. To limit pressure rise in secondary containment due to steam line break in these reactors, blow-out panels that open to outside

Details of Containment Structures of Indian PHWRs

Design Parameter	RAPS-1&2	MAPS-1&2	NAPS-1&2 / KAPS-1&2	KGS-1to4 / RAPS-3to8	TAPS-3&4
Containment Volume (m^3)	40286	47784	32200	54000	82267
Test Pressure [Kg/cm^2 (g)]	0.55	1.44	1.44	1.73	1.44
Design Pressure [Kg/cm^2 (g)]	0.42	1.16	1.25	1.73	1.44
Peak Ground Accl^n (PGA)	0.05g	0.1g	0.3g / 0.2g	0.2g / 0.1g	0.2g
Temperature due to Design Basis Accident	71^oC	96^oC	120^oC	153^oC	125^oC
Prestressing System & Capacity	12 ϕ 7 74T	12 ϕ 8 92T	12 T 13 200T / 220T	19K13 355T	19K13 355T

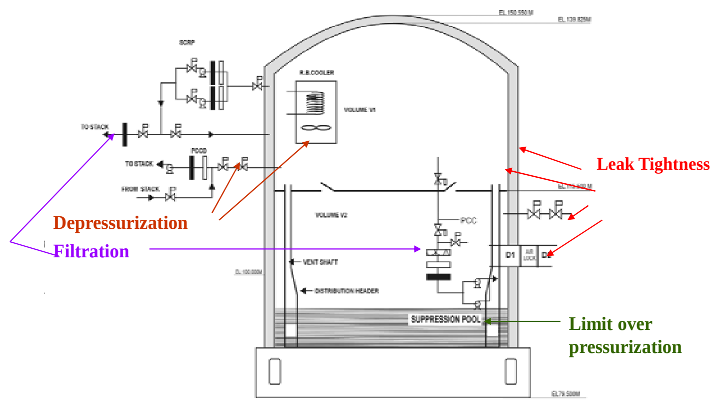
Design parameters of Indian PHWR Containments

atmosphere are provided. However, subsequently for all further PHWRs, based on experience with leakage integrity of heavy water systems, erection of Steam Generator bellows and construction of flat roof with pressure withstanding capability; it was decided to adopt double containment that envelopes all reactor systems. The annular space between the primary and secondary containment envelopes is provided with a purging arrangement to maintain a negative pressure in the space. This arrangement significantly reduces the escape of radioactivity at the ground level. Figure and Table above show the evolution of various PHWR containment structures and their design parameters respectively. The primary containment uses epoxy coating as a liner for added leak-tightness and ease of decontamination. Automatic isolation is initiated in the event of (i) pressure rise or (ii) activity build-up in the containment. The primary containment building is divided into two accident-based volumes, volume V1 ('drywell') which houses all high enthalpy systems and volume V2 ('wet well'), which is separated from dry well by leak-tight walls and floors and connected to it only through a water filled vapour suppression pool. In the event of a rupture in any of high pressure systems, the pressure build up in drywell gets vented into the wet well through suppression pool which traps steam and significant part of radioactivity and also helps limiting the peak pressure in the containment. The suppression pool water also forms part of the long-term recirculation mode of emergency core cooling. To cool down and thereby depressurize the containment following an accident in as short a time as possible, a system of building air-coolers, distributed at various locations in the dry well are used. These coolers are used during normal operation as well. The coolers are supplied from an assured process water supply and their fan motors are driven by power supplies backed by diesel generators (on-site electric power supply). Like all other containment-related engineered safety features, these coolers are designed to work in post-loss of coolant accident environment conditions within the containment. In order to further depressurize at low pressures, there is a provision for controlled gas discharge to the stack via filters. This system can also be used up to full containment overpressure for delayed containment venting if warranted due to accumulation of non-condensable during the post-accident phase. Usually operation of this system is not envisaged before 48 h following an accident.

For post-accident clean-up of the atmosphere in the containment, two systems are used:

(a) *Primary containment filtration and pump-back system*: In this system, air flow is recirculated within the primary containment through charcoal filters, to perform containment atmosphere clean-up operation on a long-term basis after an accident. Significant reduction in the concentration of iodine in the primary containment would be effected over a period of time, so that by the time the controlled gas discharge system is operated, say after 48 h, the associated stack releases will be low.

(b) *Secondary containment filtration, recirculation and purge system*: This system provides multi-pass filtration and mixing by recirculation within the secondary containment space, and also maintains negative pressure within it. The negative pressure maintained in the secondary containment space brings the net ground level release down to very low values.



	V1 /V2 Ratio	Suppression Pool Energy Efficiency
MAPS	2.6	14%
NAPS / KAPS	1.4	30%
KAIGA	3.2	4%
TAPP 3 & 4	2.0	9.6%

Pressure Suppression Parameters

3. Containment Structure

Being an important barrier to protect the population and the environment, the containment structure must have the capability to perform under extreme conditions and be free from cliff edge effects. The loads for which the containment structure is designed include normal operational and construction loads, abnormal loads that can arise during an accident in the reactor and extreme environmental loads (wind and earthquake). Combinations such as loss of coolant accident occurring simultaneously with an extreme earthquake are also considered.

The structural analyses at several levels including scenarios during construction, normal operation, relatively more frequent occurrences as well as rare occurrences is necessary to assure a balanced design with appropriate safety margins under all conditions. A set of enveloping loads to be considered for the design need to be arrived at through detailed analyses of potential accident sequences as well as of extreme environmental conditions of sufficiently low probability. Since the design of containment system itself has a role in determining the magnitude of these loads, an iterative process is invariably involved. The prestress loads which are imparted on the structure to counter the tensile stresses generated during design basis accident condition, vary both spatially as well as in time on account of short term and long term losses. The short-term losses are due to friction and wobble effect, anchorage draw-in and elastic shortening of concrete structure due to stage wise stressing of cables during the process of prestressing. Shrinkage and creep of concrete and relaxation (creep) of prestressing steel contribute towards the long-term losses of prestress loads.

Finite element analysis is carried out in order to evaluate the membrane forces due to various types of loads, for the inner containment (IC) structure. Normally, the IC structure can be modeled using thick shell elements, which can take care of the shear deformation. In order to obtain the stresses in the discontinuity areas such as ring beam more accurately, such regions in the containment structure are modeled using 3D solid brick elements. The effect of the duct openings and the reinforcing/prestressing steel is accounted for appropriately at the design stage. The design of the inner containment structure is checked both for limit state of serviceability as well as for limit state of strength (also called as ultimate limit state) as per the provisions of standards (RCC-G in our case). It is ensured that a membrane compressive stress of at least 1 MPa is available even under accident conditions. Presence of voids in the concrete section due to cable duct openings led to development of special tools for carrying out the design check under limit state of serviceability and limit state of strength. The entire inner containment is checked against the requirement of Limit State of Strength. In general the load factor adopted for the permanent loads is 1.35 and that for the variable loads is 1.5.

The normal stress in the thickness direction, also called as radial stress, is generated in the inner containment structure due to (i) C*urvature effect,* as a result of the prestressed cables embedded inside the concrete with a curvature, exert radial pressure towards the center of curvature along the inner edge of the cable sheath leading to development of radial stress, (ii) *transition effect*, arising at locations where the shell thickness changes rapidly within a short distance and (iii) *Stress concentration effect,* in which the in-plane membrane compressive stresses also generate radial tensile stress around the cable duct openings. Normally, the radial stresses in the prestressed concrete inner containment structure are larger during the construction stage. We had experienced de-lamination of inner containment dome during construction of Kaiga reactor. Several improvements in design and construction practices have been put in place since then. The radial reinforcements in the IC dome are designed to carry the radial tensile force resulting from the curvature effect and transition effect where the cable spacing is large. Where cable spacing is close, reinforcements are provided to take care of the local tensile force due to stress concentration effect also.

A reduced value of permissible stress is considered while designing the reinforcements to limit cracking in concrete. The strain measurements in order to capture both membrane as well as radial stresses at different locations covering the entire IC are necessary to monitor the development of stresses in the IC structure at different stages of prestressing. The deflection and strains in the IC are also measured during proof testing mainly to confirm agreement with results of analytical studies. Based on the recorded data, the stresses developed in the IC are compared with the specified allowable stress limits of the design code RCC-G.

Equivalence Study with Respect to ASME, Sec-III, Div-2

The design of containment structures of Indian PHWRs has been based on the methodology evolved considering the Indian construction practices. French code RCC-G is being primarily used for design as the double containment without metallic liner adopted for Indian PHWRs is similar to the one adopted in French PWRs. In order to assess the level of safety margins under design basis accident scenario, an attempt has been made to study the equivalence of the various provisions for design of primary (Inner) containment structure with respect to those adopted in various international standards The study has been carried out to establish the equivalent partial safety factor for accident pressure (P_a) in factored load combination under limit state of strength. For this purpose, a detailed design check has been carried out for some critical areas located in different regions of Inner Containment structure, such as, (i) thickened area near large openings in IC dome, (ii) IC dome general area, (iii) IC wall-raft junction, (iv) IC wall general area and (v) IC dome springing area etc. under accidental load combinations as per the provisions of various international standards viz. Canadian Standard, ASME, Sec-III, Div-2 and RCC-G.

Based on the equivalence study, the following conclusions are drawn:

(i) Though the design philosophy and partial safety factors for material and loads, pre-stress loss computations are different in various international standards, the design provisions and the associated design parameters suggested by various International codes pertaining to design of containment structure are laid down in such a balanced way that the final design lies in the same range.

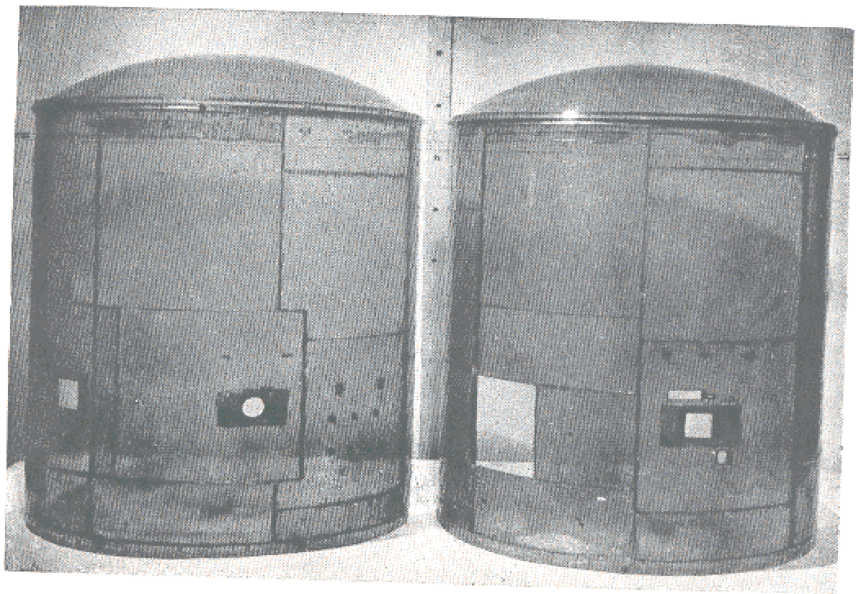
(ii) In general, it is noticed that the load combination involving accident pressure along with the design philosophy suggested in RCC-G is equivalent to the design provisions adopted in the Canadian standard as well as the ASME Code.

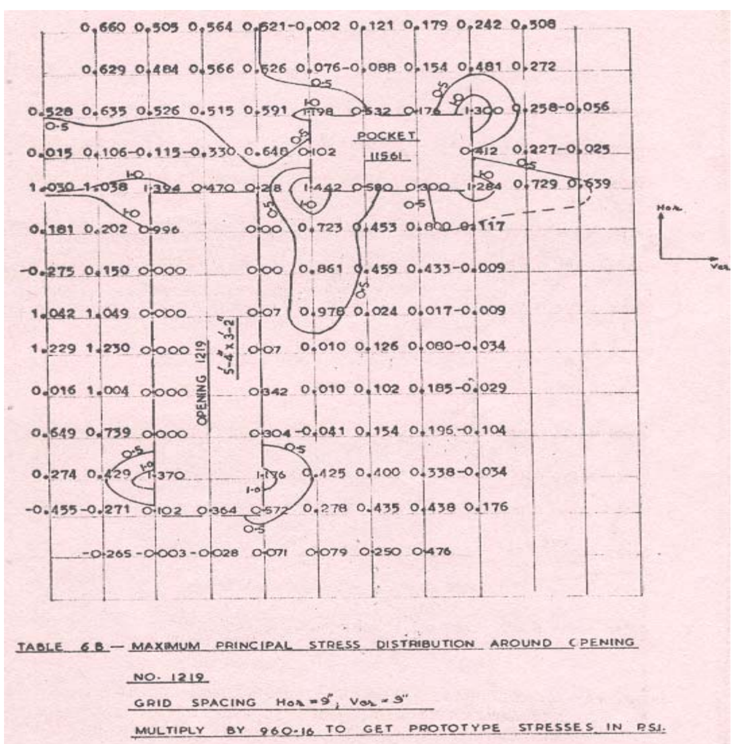
4. Containment Research and Development

Importance of Research and Development on nuclear reactor containment system was recognised right since the early stages of the nuclear power programme in India and was pursued in tune with the evolution of containment systems.

MAPS Containment

Indigenous design and development efforts for the containments began with Madras Atomic Power Station (MAPS) PHWR units. To achieve reduction in ground level release of radioactivity and to facilitate safe egress of personnel who might be in the reactor building during an accident, partial double containment and vapour suppression pool respectively were introduced. The design had provision of structural hinge for the IC wall-raft junction to prevent transfer of moment to the wall and Neoprene pads were provided at the base of IC wall to facilitate radial movement of wall for effective transfer of hoop forces during the pre-stressing of horizontal cables. The break out panels in this design are structurally disconnected with the wall but through provision of loading jacks local pre-stressing is ensured. The stresses around various openings and break out panel were experimentally evaluated through 3D photoelastic model(3) and concerns with regard to localized concrete cracking and its possible influence on the leakage rate were suitably addressed.





Stress Contours from MAPP Reactor Containment Photo-Elastic Model

Another study was conducted on a 1:12 scale micro-concrete pre-stressed model at Structural Engineering Research Laboratory[(4)]. The model was instrumented for measurement of displacements, strains and forces in pre-stressing wires besides pressures and temperatures. The experiment confirmed the theoretical predictions of pressure at which first crack would occur. Also that the bellows arrangement employed for avoiding transfer of transverse pressure load on the peripheries of various openings though causes disturbance in the axisymmetric behavior and results in minor tilt, still the permissible limits of deflections as envisaged in the design were met. Subsequently during the ultimate load capacity assessment of this containment[5], the identified failure modes were studied rigorously in detail with inelastic finite element codes and a thorough review was made on the observed leakage rate data obtained during the initial commissioning and subsequent in-service tests and the containment performance has been found to be satisfactory till today.

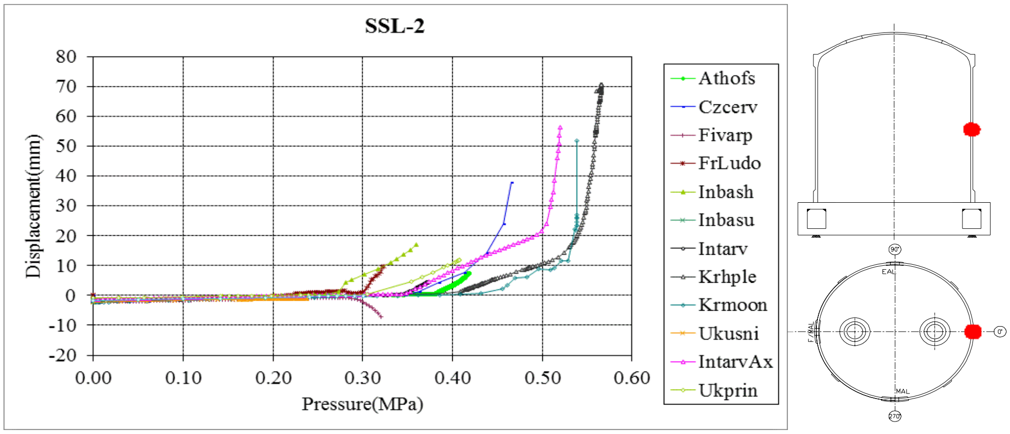
BARC Containment (BARCOM) International Round Robin Program

The severe accidents at Three Mile Island-1979 (USA), Chernobyl-1986 (former USSR) and more recently at Fukushima-2011 (Japan) nuclear plants have led to wide spread attention to the ultimate load capacity assessment of nuclear containments structures. This has become important not only for safety authorities but also for public acceptance of nuclear power plants. This needs verification through the functional and structural failure mode tests that enable ruling out containment structural failure following a severe accident with adequate margin over the design pressure. Further it would be important to study post failure behavior of the containment. Bhabha Atomic Research Centre (BARC), Trombay has organized an International Round Robin Analysis program to carry out the ultimate load capacity assessment of BARC Containment (BARCOM) test model [(6)-(8)]. The test model located in Tarapur; is a 1:4 scale representation of pre-stressed concrete inner containment structure of Tarapur Atomic Power Station (TAPS) unit 3&4. Around 1200 sensors installed in BARCOM include vibratory wire strain gauges (VWSG), surface mounted electrical resistance strain gauges, dial gauges, earth pressure cells, tilt meters and high resolution digital camera systems for structural response, crack monitoring and fracture parameter measurement to evaluate the local and global behavior of the containment test model. The objective of the present test program is to obtain the pressure, displacement and strain data related to various functional and structural failure modes of BARCOM in terms of the loss of pre-stress in the membrane and discontinuity regions of major openings, first appearance of concrete surface cracks followed by first through thickness cracks, first yielding of reinforcement / tendons and significant loss of leak tightness, the maximum pressure sustained by the model before significant leakage to identify functional failure pressure of the test-model and finally the maximum static pressure sustained by the model for the structural failure assessment. All these assessments would finally lead to predictive capability for assessment of ultimate pressure that the containment can withstand and estimation of functional and structural behavior of the containment model up to the ultimate pressure.

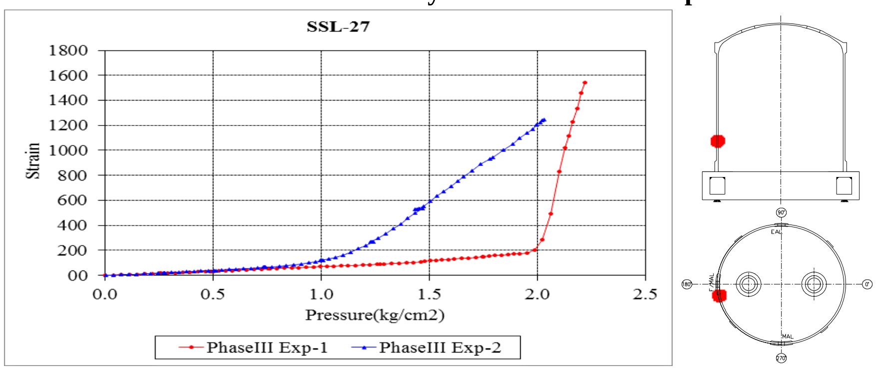
Pre-Stressing, Phase-II Design Pressure Test and Phase-III Over-Pressure Test Results

The strain data obtained during the construction period including pre-stressing (Oct 2007-Dec 2008) and beyond up to a period of one more year were obtained from embedded VWSG sensors. Pre-stressing load induced strains at these locations were identified after separating the creep and shrinkage strain components from the overall strain data with help of pre-stressing schedule. VWSGs responses in hoop (Specified Sensor Locations -10 & 12) and longitudinal directions (Specified Sensor Locations -11 & 13) on the inner and outer surfaces of BARCOM wall are 272-353 $\mu\varepsilon$ and 97-109 $\mu\varepsilon$ respectively as compared to average computed value of 292 $\mu\varepsilon$ and 99 $\mu\varepsilon$ respectively based on the jack load, which shows consistent response of embedded VWSGs. The Phase-I pressurization of BARCOM Test Model up to 0.049 MPa (0. 35Pd) was carried out for the functionality tests of various sensors and data logger systems. Subsequently, BARCOM Test Model was pressurized up to the design pressure (Pd) of 1.44 kg/cm^2 (0.1413 MPa) in August, 2010 and since then three more design pressure tests have been completed during Oct-Dec, 2010 as a part of the Phase-II experiments programme. The commissioning and proof test of BARCOM has been completed successfully and data from all the sensors were recorded during the pressurization and depressurization cycles. In addition, consistent and repeatable leakage-rate was obtained during all Phase-II experiments. Further under the Phase-III over-pressure test (OPT) program, the BARCOM Test Model was pressurized up to a pressure of 0.2207 MPa (1.56 Pd) in December 2010. The milestone with regard to “first appearance of crack” was recognized with online monitoring of the inelastic strain developed in the discontinuity regions of Main Air Lock (MAL) and Emergency Air Lock (EAL) with embedded VWSGs, which was also confirmed with soap bubble test during depressurization at 0.0981 MPa. A second OPT was also conducted during Jan 12-15, 2011 to check the repeatability of the test data and obtain the localized strain field in the fracture process zone (FPZ) at few selected locations. The strain pattern obtained showed development of parallel cracks with localized strain field in the fracture process zone (FPZ) near MAL, EAL locations and the first through thickness cracks in BARCOM test-model were identified successfully. With the availability of all these inputs that were presented in the post-test meeting and international workshop[6-8] during Feb 04-07, 2011, the failure mode of BARCOM has been identified. Round Robin Participants have been asked to submit their post-test analysis results in the prescribed format, which will be compared with the recently concluded Phase-III test up to 1.68-1.78 Pd of July and October 2011. I am glad that this conference would provide an opportunity to take these discussions forward[9]. This experiment has demonstrated that even after functional failure of the containment and leakage from primary containment where the cracks are tight, the observed leakage rates will be in controllable and manageable limits and shielding cover will be retained. Moreover the Engineered Safety Features will in addition assist in controlling the ground leakage and releases to the environment. Our endeavors to demonstrate the available safety margin against over-pressurization of BARCOM will address important issues with regard to containment safety under extreme events. It is important to note that under such extreme events the responses of the lined / unlined containments need to be established with confidence and after the functional failure such as liner tearing or major through thickness cracking in double containment system, the overall stability of the containment structure needs to be ensured for public acceptance for nuclear power plants.

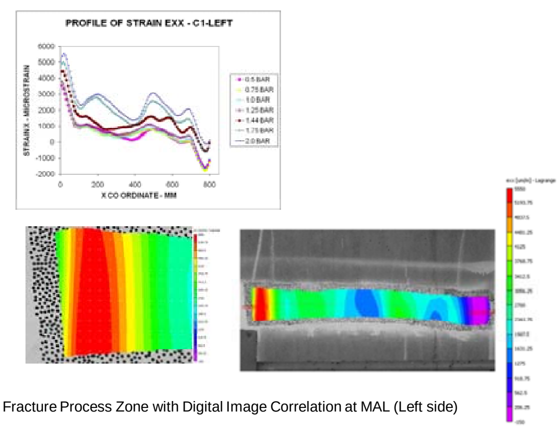
BARC Containment (BARCOM) Test-Model (Design Pressure Pd 0.1413 MPa) at BARC-Tarapur Test Facility with details of Embedded Sensors and Cable Panels



One PreTest Prediction by Round Robin Participants



Response during Over-Pressure Test in BARCOM near MAL at SSL-27

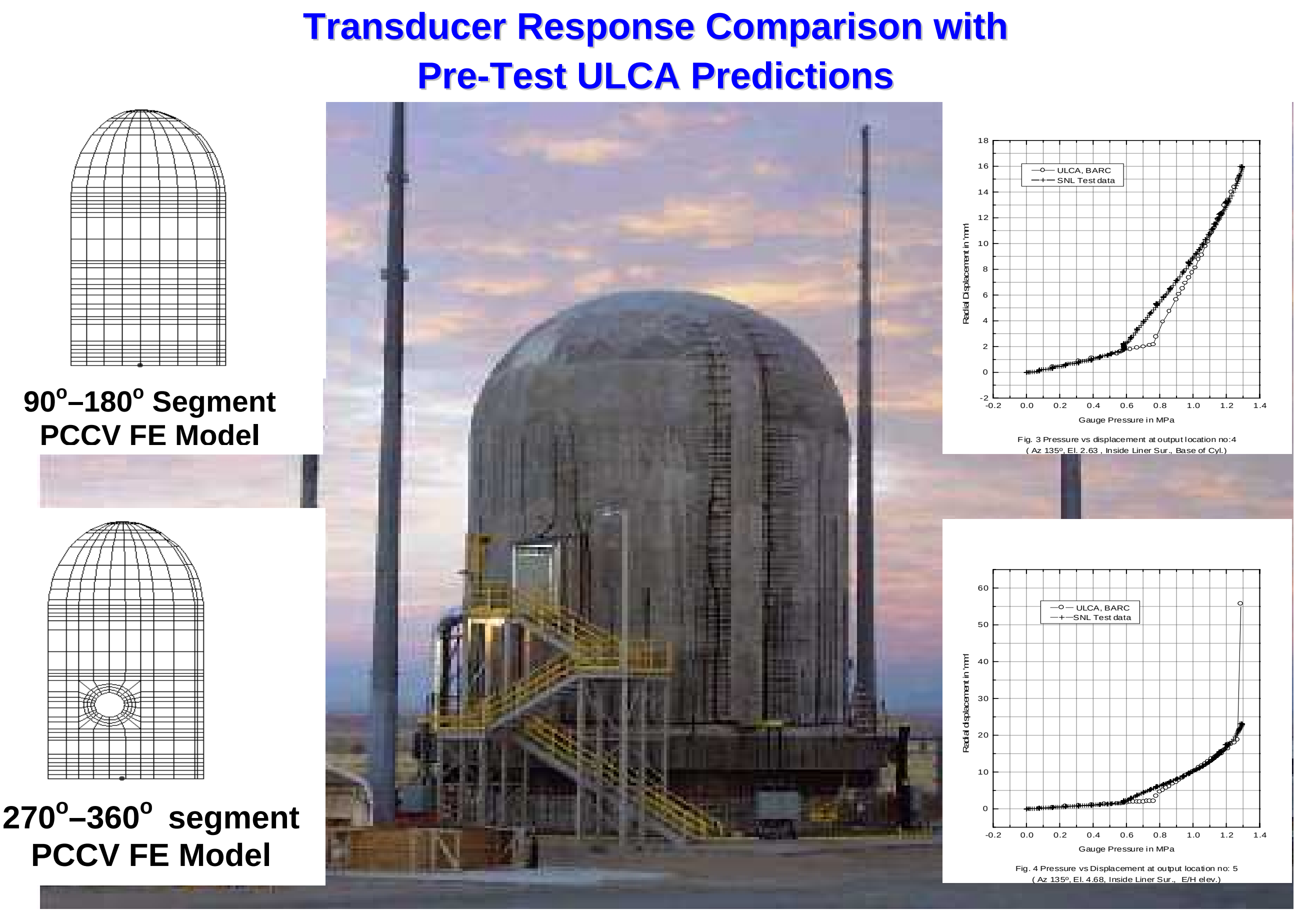


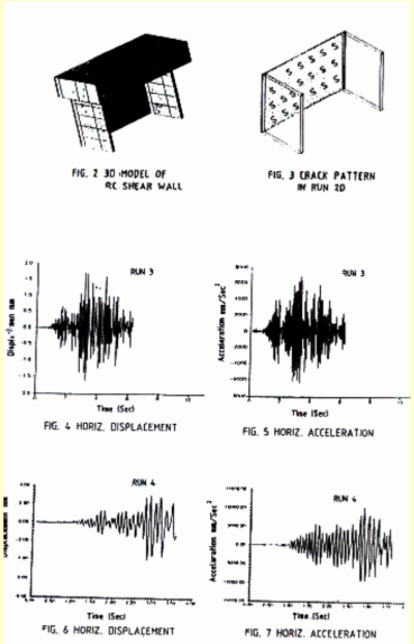
Fracture Process Zone with Digital Image Correlation at MAL (Left side)

First Appearance of Crack in BARCOM Test Model during Over-Pressure Test
Phase-III Experiment and verification with Optical Crack Profiling (Dec 17 2010 and Jan 2011)

Assessment of Ultimate Load Capacity of Nuclear Containments - Concrete Fracture Mechanics

With a view to ensure adequacy of containment as the ultimate barrier, assessment of structural safety margins under severe accident conditions assume importance. Ultimate load capacity assessment of Indian PHWRs was therefore taken up. In-house finite element codes ULCA(10-12) and ARCOS3D(13-15) were developed. These codes have been extensively benchmarked with experimental test results of Sandia Laboratory, USNRC sponsored containment model test and NUPEC, Japan seismic shear wall test. Limit State Test (LST) Pressure & Structural Failure Mode Test (SFMT) Pressure estimated using in-house code ULCA were in excellent agreement with PCCV Sandia Test results. Fracture mechanics studies with regard to quasi-brittle, heterogeneous and composite concrete and RC/PC materials were initiated for evolving sound engineering practices for nuclear containment structures[16-17]. Using validated codes, Indian containments have been shown to have a factor of safety of more than 2 over the design pressure.





In-house FE Code ARCOS-3D for static and transient dynamic loads qualified with NUPEC Japan sponsored seismic shear wall test.

The code has a visco-plastic model with yield and strength limit surfaces and strain rate dependant constitutive formulation suitable for cyclic and transient dynamic loads.

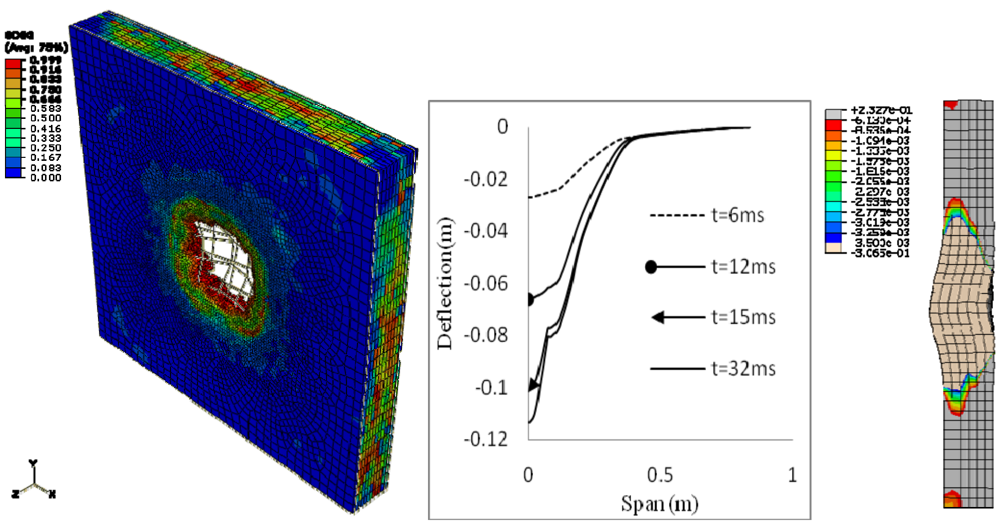
Numerical predictions in excellent agreement with the test results for all the test runs on shake table.

Maximum Top Slab Displacement (mm)

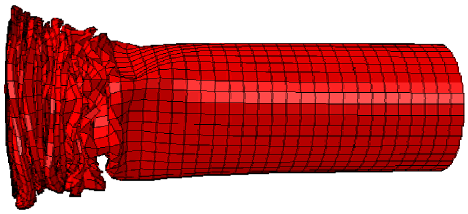
U1	U2	ARCOS-3D	RUN
0.29	0.35	0.25	RUN-1
0.58	0.89	0.74	RUN-2
1.04	----	1.39	RUN-2A
1.63	2.68	1.70	RUN-3
3.71	3.91	4.00	RUN-4
11.1	11.5	7.04	RUN-5

Maximum Top Slab Displacement with Static Run after accounting previous cracking history ~ 11 mm 30

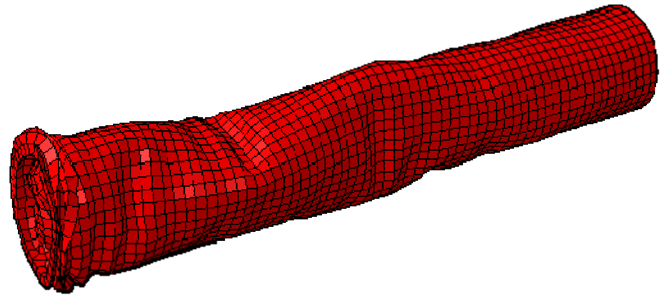
Aircraft Impact Studies for Containment Integrity



Punching and Flexure failure modes in RC slab



Meppen Test: Missile after rebound at 55ms

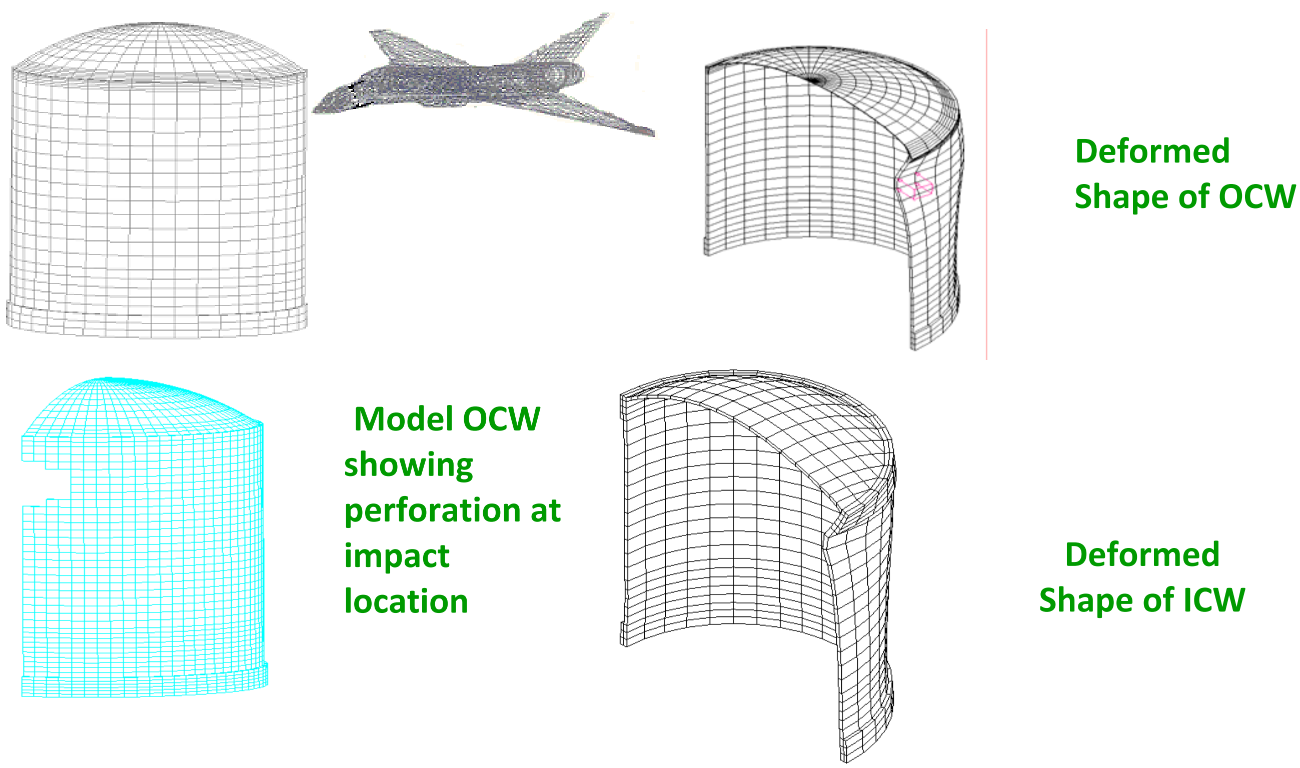


VTT Bending Test: Missile after rebound 50ms

Assessment of capability of nuclear containments to withstand aircraft impact has been carried out for 540 MWe PHWR plant. To validate in-house and commercial inelastic finite element transient dynamic codes, Nuclear Energy Agency (NEA) sponsored benchmark studies on flexural and punching shear assessment of reinforced concrete slabs were carried out[18] and the data obtained from simulation were verified with the benchmark experiments with regard to the missile and slab deformations and strains. A coupled multi-body transient dynamic analysis procedure for missile and slab system was evolved.

It has been assessed that outer containment wall (OCW) of 540 MWe PHWR plant could suffer local perforation with a peak local deformation of 117mm at 0.19 sec. The overall integrity of OCW structure is maintained with negligibly small displacement of ~ 5 -10 mm at other locations. There will be local cracking and rebar yielding in inner containment wall (ICW) with maximum displacement of 115mm, but there would be no perforation in ICW.

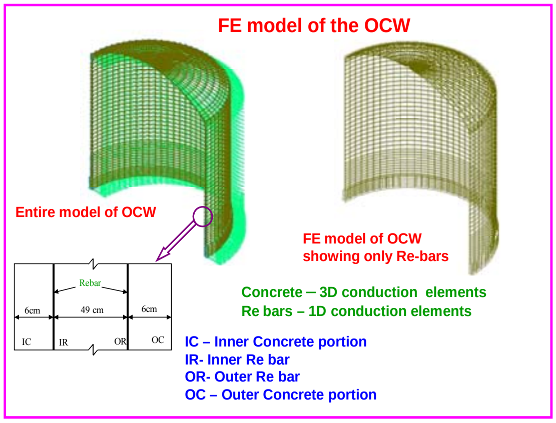
The double wall (OCW and ICW) would be capable of sustaining the full impulsive load of Boeing 707-320 & Air Bus A300B4-200[19-20].



540 MWe PHWR Accidental Aircraft Impact Simulation for ICW and OCW – Multiple Barrier Assessment

Fire Load Analysis of 540 MWe PHWR Double Containment Following Aircraft Impact

The effect of the fire induced thermal load on containment structure has assumed importance especially after witnessing the World Trade Centre (WTC) collapse due to aircraft impact where reported studies have revealed that the main cause for WTC collapse was the loss of structural strength of the steel columns, due to prolonged heating to high temperatures (~ 800-900 ºC) caused by burning aviation turbine fuel (ATF). This necessitates the study on the behavior of the nuclear containments under fire due to the spillage of aircraft fuel. Effect of fire ball induced heat load and radiation from fuel spilled on the ground surrounding the containment structure of 540 MWe double containment system has been studied.[21]



Effect of fire ball duration on containment response

Containment response has been evaluated by varying the fire ball duration (t_f) and keeping the heat input (100% of fire ball energy) as constant

Temperature (K)at various locations

Location	t_f = 1s	t_f = 5s	t_f = 10s
OC	1238.8	1232	1231
OR	332.12	332.12	332.09
IR	300.88	300.8	300.8
IC	300.58	300.5	300.5

Fire ball duration has a negligible effect on containment response.

Heat flux due to radiation exchange between ground fire and the OCW can be considered as negligible as the fuel spread on the ground burns off quickly compared to that of fire ball. This is true with an assumption of fuel spread over uniformly and ignited simultaneously. However, in reality, the aircraft disintegrates and its wreckage viz. wings, tailfin, fuselage portion, engines etc will be scattered around and spillage of fuel takes place in an irregular pattern. Pockets of fire surround the containment site for much longer durations and hence the radiation effect on to the OCW needs to be analyzed. For the present study, it is assessed that the radiation effect on the OCW due to ground fire is approximately 1% of the heat flux generated due to the fire ball.

Following broad conclusions have been drawn as a result of these studies;

- The outer concrete cover portion of the OCW will be affected under thermal load due to fuel spillage
- .However, the central concrete portion will remain intact due to fire thermal load.
- Temperature rise in the inner concrete cover portion rebars is comparatively negligible.
- The fire ball duration has a negligible effect on containment response.

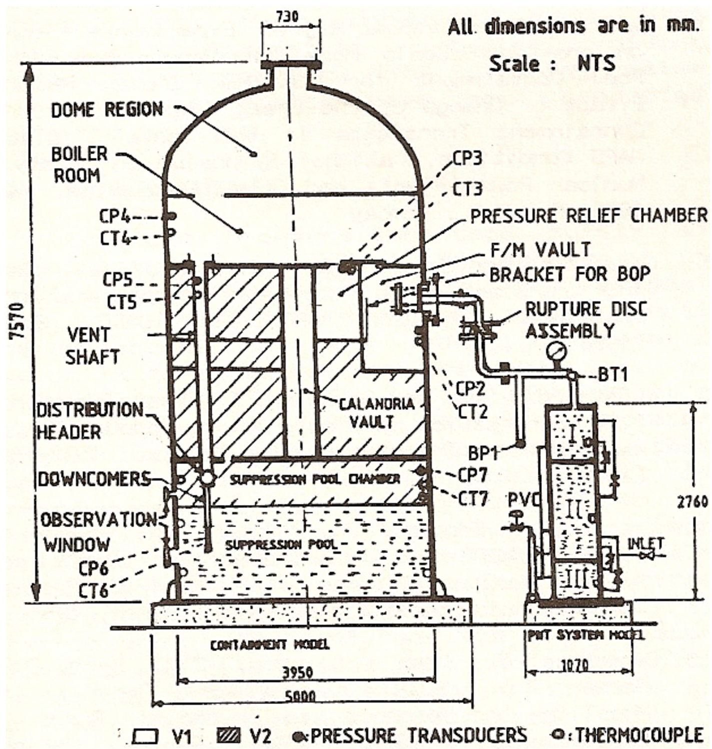
The present evaluations are conservative as per the reported concrete strength degradation [22] under fire conditions and a more detail study with coupled hygro-thermal models for assessing the concrete strength under intense fire due to moisture loss are underway.

Thermal Hydraulics Experiments for MAPS & NAPS Containments

During the early evolution of containment system, thermal hydraulics experiments were carried out on a one tenth length scale model of PHWR at Kalpakkam[23-24]. These experiments had following features:

- The containment model simulating major compartments of the prototype with the volumetric scaling ratio of 1:1000.

- 1:1 scale for downcomer submergence as well as for the suppression pool water depth located at the lowest elevation in the model.
- The Primary Heat Transport system model simulating the inventory of water at three temperatures with volumetric scaling factor of 1000, located outside the containment model for convenience.
- The containment response for LOCA conditions and effectiveness of vapor suppression pool were demonstrated.
- Experiments were carried out for both MAPS and NAPS containments by adjusting the ratio of drywell and wet well volumes.



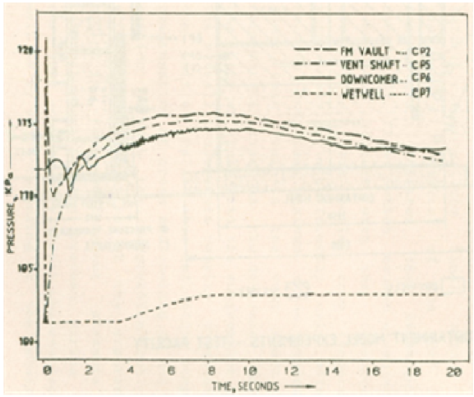
Containment Thermal Hydraulic Studies

The major findings from these model studies were as follows:

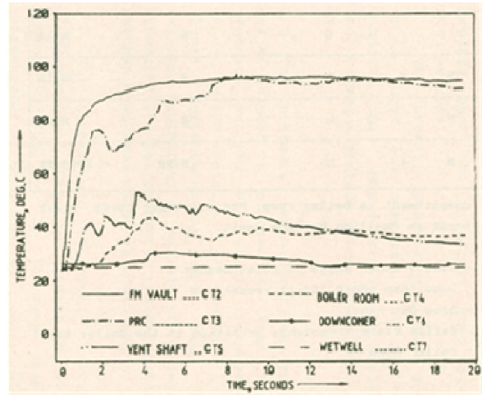
- The experimental data confirmed that the vapor suppression pool is highly effective in condensing the steam as well as in cooling hot air passing through the pool, thus limiting the pressure and temperature rise in containment.

- In all the experiments, no temperature rise was observed in the wet well.

- Heat transfer to structures played prominent role in later part of transients and after around 5 minutes the dry well pressure was found to be less than the wet well pressure.



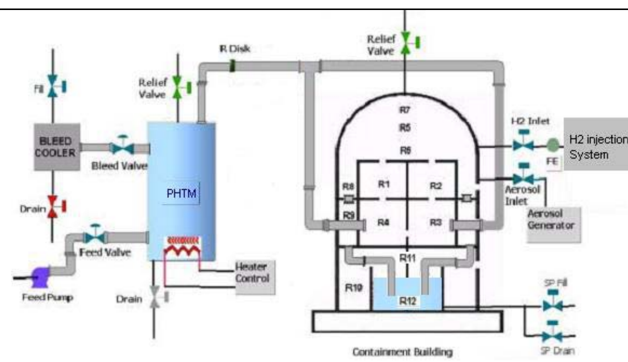
Pressure Transients



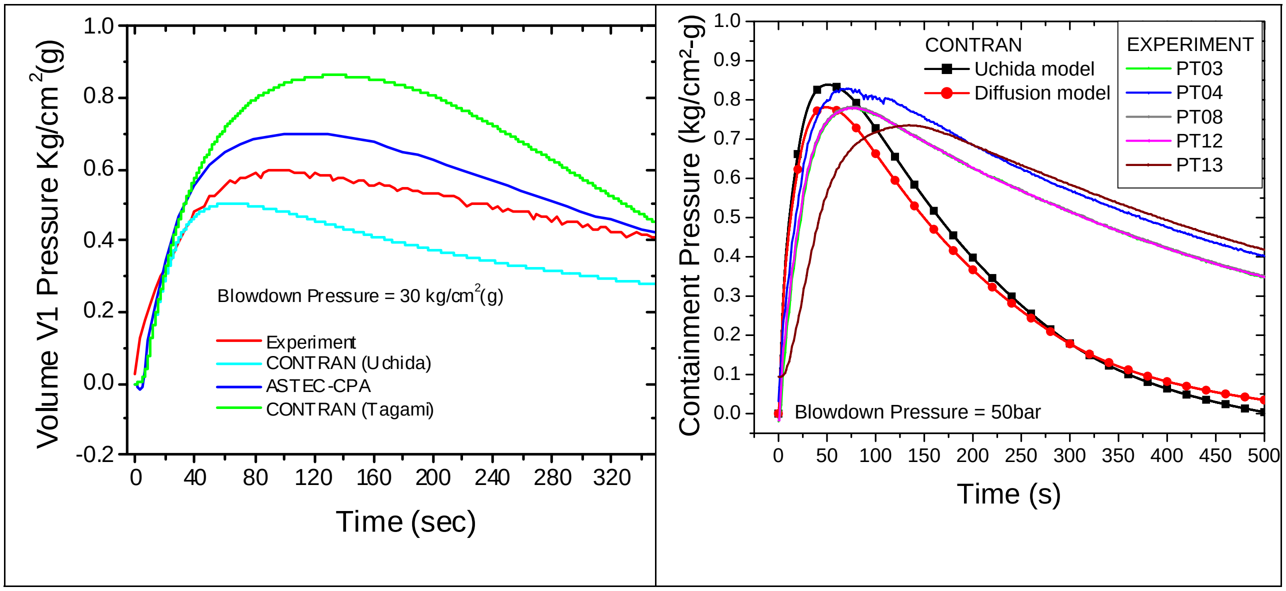
Temperature Transients

Containment Studies Facility (CSF):

Containment Studies Facility (CSF) constructed at BARC, Trombay represents a larger scale (1:250 volumetrically scaled) model of a 220 MWe Indian PHWR for more refined thermal hydraulics experiments. It consists of reinforced concrete containment and a Primary Heat Transport Model (PHTM) vessel along with associated instrumentation for simulation of the Loss of Coolant Accident induced blowdown and hydrogen/helium/aerosol transport studies. A detail instrumentation system consisting of thermocouples, pressure transmitters, pressure and level gauges along with a data logger system is employed for recording the pressure and temperature transients during the controlled LOCA/MSLB (Main Steam Line Break) experiments in this facility. The containment model of CSF is divided into high enthalpy V1 volume (dry well) and low enthalpy V2 volume (wet well). The V1 volume is further divided into many compartments to simulate pump room, fuelling machine vault. The V2 volume is connected to V1 volume through vent pipes and suppression pool. PHTM vessel was pressurized to a predetermined pressure and then it was allowed to blowdown into V1 volume of containment using a double rupture disk assembly.



Schematic of the Containment Studies Facility

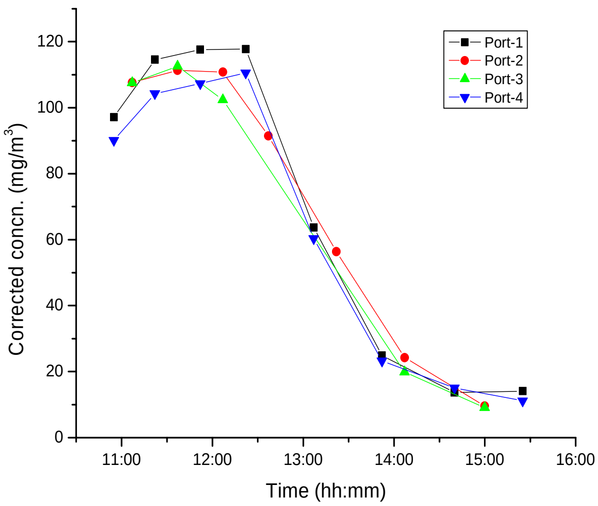


Experiments and analysis for Blow Down

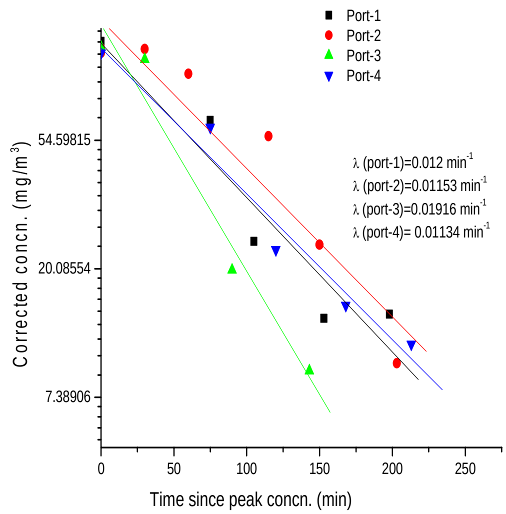
Pretest analysis has been carried out for different PHTM pressure ranging from 10 bar to 100 bar at saturated conditions. Parametric studies were conducted to assess the influence of a large number of thermodynamic and geometrical parameters which are known to affect the transients and alter the peak pressure and temperature values. Experiments were carried out in CSF at blowdown pressures of 30 and 50 bar. Results were compared with numerical in-house code CONTRAN and French code ASTEC and reasonable agreements were obtained between the three results(25).

Containment Aerosol Behavior Studies(26)

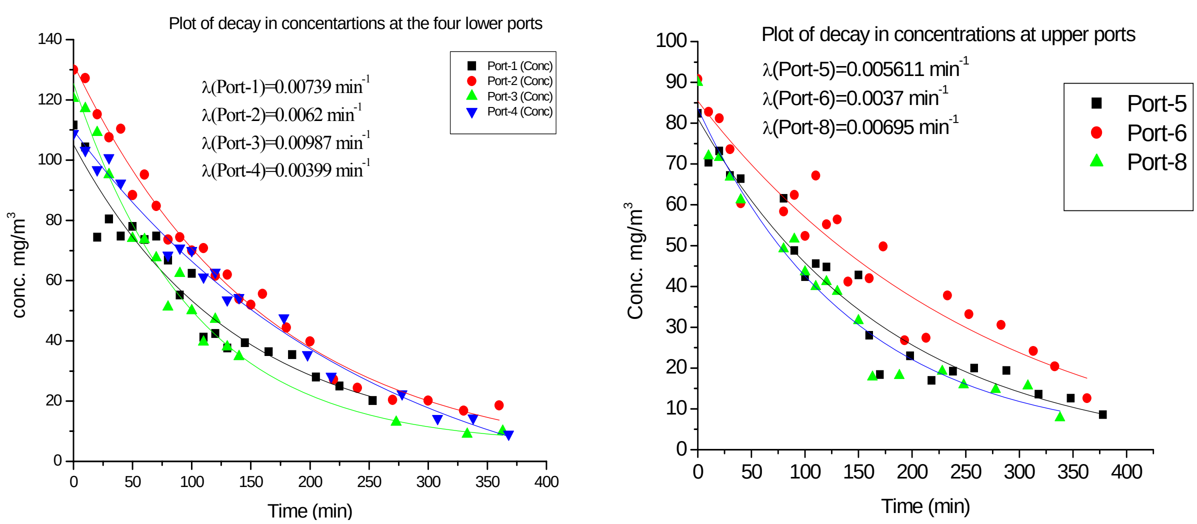
Aerosols get generated within the containment during a severe accident. Understanding of their behavior is crucial to assessment of radioactivity transport and management. CSF was utilized to study the temporal behaviour and physical characteristics of aerosols of combustion. Tests were conducted in a single compartment volume and two inter-connected compartments of the containment model. During the tests, aerosol concentration was allowed to build up for about one hour and samples for estimation of gross concentration were collected at various time intervals. The decay profile of the suspended aerosol mass was obtained and the aerosol mass depletion constants were calculated. The Mass Size distribution parameters viz., MMAD (Mass Median Aerodynamic Diameter) and GSD (Geometric Standard Deviation) were estimated using the Cascade Impactor. Online measurements of number concentrations were recorded using Grimm Optical particle counter, which gives the count of particles in 14 size channels ranging from 0.3 µm to 20.0 µm. The time variations in the concentrations with the average corrected flow rates were obtained. The aerosol mass depletion constants were calculated for the concentrations at each of the ports from the linear fit plot of the concentration with time. The decay in aerosol concentration at the four ports corresponding to the peak concentrations was studied. The Andersen eight stage cascade impactor was used to estimate the size segregated mass concentrations. The variation in no. concentration with time for particles of different size ranges shows a second peak observed in all the sizes at about 14:30 hrs in the upper room. This rise in concentration could be attributed to some ingression of aerosols from the lower room to the upper one. As part of the studies planned further, the Plasma Torch Aerosol Generator is being commissioned. The facility will be further effectively utilized to generate the requisite information with regard to aerosol transport to address the severe accident issues.



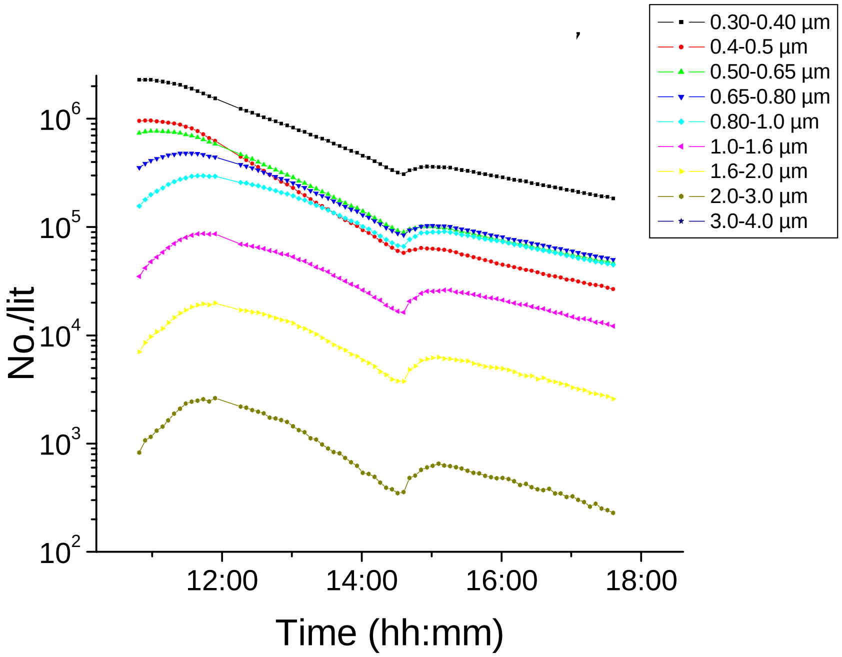
Temporal variation of the corrected mass concentrations at the four sampling ports of CSF



Estimation of aerosol depletion constants from the linear fit to the concentration



Estimation of aerosol mass depletion constants from the exponential fit to the concentration vs. time plot at the lower and upper sampling ports



Variation in no. concentration for different size ranges

Hydrogen Mitigation Technology Development at BARC

Behaviour of hydrogen in the containment of nuclear power plants under accident conditions has been a subject of worldwide research for quite some time. Experimental as well as analytical research programmes have been underway in many countries to address these issues with a specific aim of improving the safety of nuclear power plants under accident conditions. Such studies have also helped to improve regulatory compliance for safe operation of nuclear power plants. As a part of programme for development of technologies to address hydrogen related safety issues during a severe accident, several analytical as well as experimental studies are underway at BARC. Development of passive catalytic recombiners is an important part of this programme. Effective design and deployment of such recombiners in the containment can prevent overpressure in the containment as a result of hydrogen release as well as protect the containment atmosphere from becoming flammable under severe accident conditions.

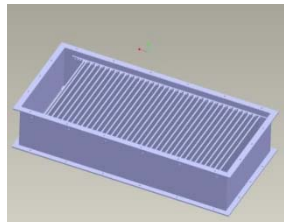
Hydrogen Mitigation Test Facility (HYMIS)-at BARC

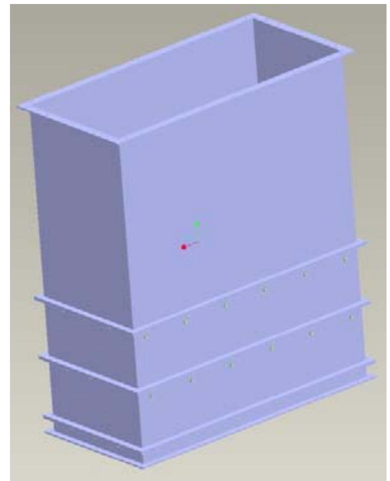
Development of catalytic recombiner elements for this purpose was undertaken in BARC along with the design of a recombiner module. **HY**drogen **MI**tigation **S**tudies (HYMIS) test Facility has been set up at BARC. The 7 m. tall and 2 m. in dia. HYMIS test set up has a total volume of about 22.0 m^3. Hydrogen Distribution experiments and subsequently hydrogen recombiner performance evaluation tests were performed in this test set-up. A preliminary assessment of the efficacy of the recombiner device with the catalyst samples was performed in HYMIS under dry conditions (without steam) and hydrogen concentrations below flammability limits (4%).

Tests in Containment Studies Facility (CSF)

As a next step, the performance evaluation of the recombiner device, is required to be carried out under wet environments (air + hydrogen + steam) and higher hydrogen concentrations before possible deployment in the NPPs.

Subsequently, more elaborate studies are being planned in a multi-compartment Containment Studies Facility (CSF) which has been set up at BARC for containment related safety research. These studies will assess the hydrogen related concerns in presence of steam. Experimental studies are planned to be carried out in a phased manner in CSF to analyse the thermal hydraulics, hydrogen and aerosol behaviour under accidental conditions.





Hydrogen Recombiner Test Facility (HRTF)

Schamitic Passive Hydrogen Recombiner Device

Hydrogen Recombiner Test Facility (HRTF) [27]

A single compartment Hydrogen Recombiner Test Facility (HRTF) that can simulate different reactor containment environments under controlled conditions for testing of the recombiner modules at higher hydrogen concentrations (upto 8 % by volume) is under advanced commissioning stage at Tarapur. HRTF consists of a cylindrical carbon steel vessel of around 3 m diameter and 9 m height. The vessel has a volume of 60 m^3 and would be equipped with various types of instruments for measurement of pressure, temperature and hydrogen & steam concentration etc. Steam condensation due to loss of heat to the walls would be minimized by insulating the vessel.

5. Advanced Heavy Water Reactor : futuristic steps towards public protection[2]

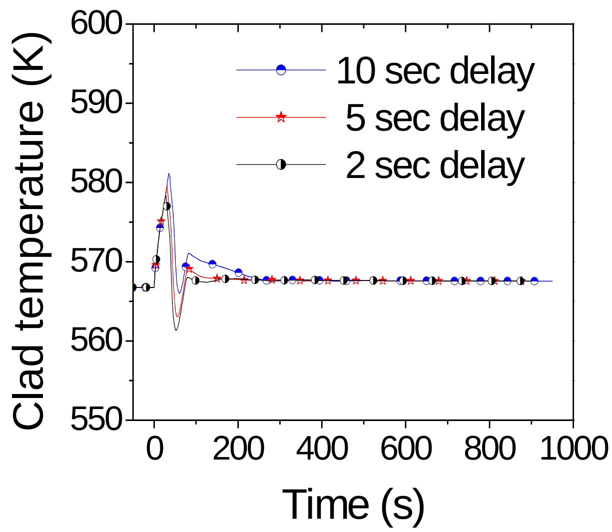
In the design of Advanced Heavy Water Reactor (AHWR) a novel configuration for the reactor and the containment systems enables important attributes such as;

i) long (3 days) grace period for operator to react to a situation,
ii) negligible impact in public domain even under extremely low probability situations, and
iii) high level of immunity against insider threats.

These features are achieved in AHWR essentially through passive means leading to a high level of confidence in terms of protection of public. Some of the important passive features incorporated in the reactor and containment design are;

i) large inventories of water stored under appropriate pressure/elevation for emergency core cooling, decay heat removal, containment cooling and isolation etc.
ii) core cooling by natural recirculation including for primary heat transport
iii) passive shut down system

The reactor characteristics are such that reactor cooling remains assured even under extremely low probability events thus virtually eliminating excessive demands on the containment in terms of release of radioactivity and hydrogen that takes place under core damage situations. In AHWR, the peak clad temperature hardly rises even in the extreme condition of complete station blackout and failure of both primary and secondary shut down systems.

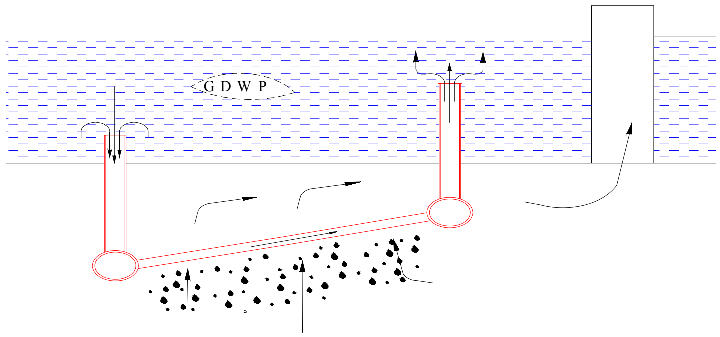


Peak clad temperature profile during station black out and failure of both primary and secondary shutdown systems

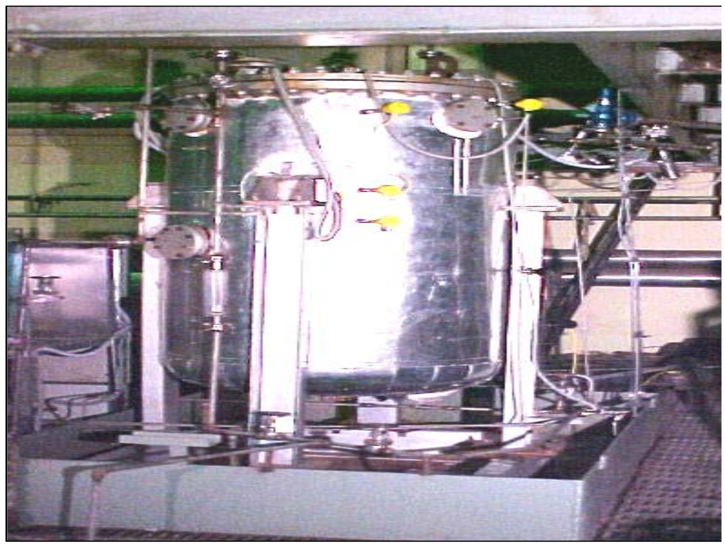
There are other containment specific features in AHWR such as 100 years design life and better accessibility inside the containment during operation.

Containment energy management using passive containment cooling systems (PCCS)

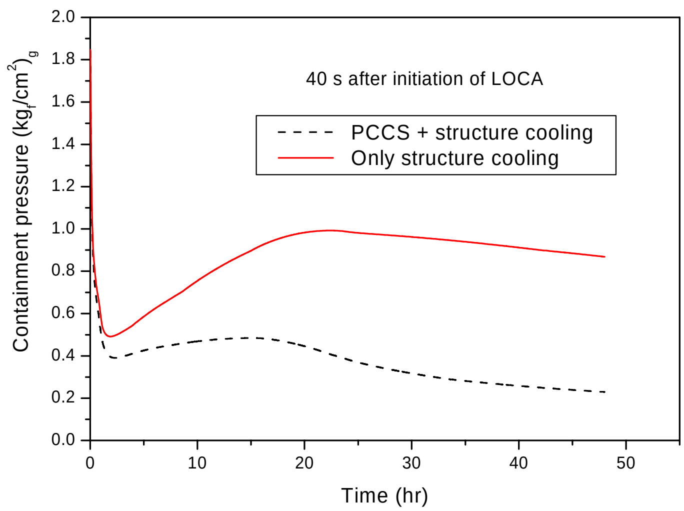
In Indian AHWR, the PCCS consists of Passive External Condensers (PECs) which are connected to a gravity driven water pool above it. The steam condenses on the outer surface and water flows by natural circulation inside the condenser tubes. The condensation of steam in the presence of non-condensable gas becomes an important phenomenon during LOCA when steam released from the coolant system mixes with the containment air. Experiments have been performed to determine the heat transfer coefficient in presence of non-condensable gas. Thermal hydraulic analysis for determining pressure and temperature transients in primary containment of AHWR have been carried out following a loss of coolant accident in the containment following 200% inlet header break.



Schematic of Passive external condenser



Test Facility to study condensation in presence of non-condensable gas



Containment pressure transients considering PCCS and concrete cooling

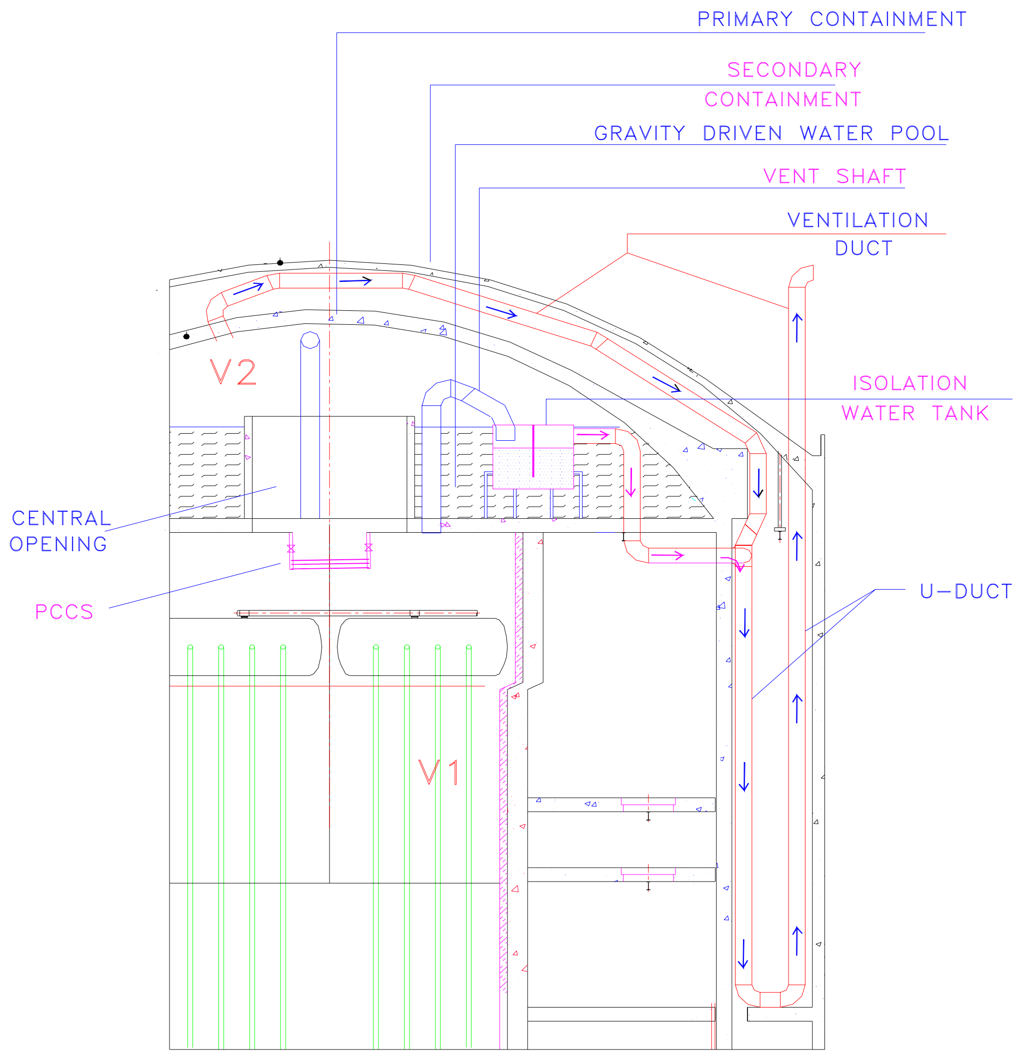
Passive containment Isolation system

Under accident conditions the containment need to be isolated to prevent the direct ground level release of radioactivity (if any) to the atmosphere. This requires isolation of ventilation ducts (both supply and exhaust). Active isolation dampers are provided in both the ducts to isolate the containment from the atmosphere. In addition to this, a passive system to isolate the containment is also provided to minimize the consequences due to failure of active type isolation damper based on sensors and actuators. Under LOCA conditions, V1 and V2 regions witness a pressure transient. Until the opening of break out panels (BOP), the V1 pressure rises more rapidly than V2 pressure. This transient differential pressure is used to fill U shaped ventilation duct that isolates the containment from outside by establishing a liquid U- seal in the ventilation duct.

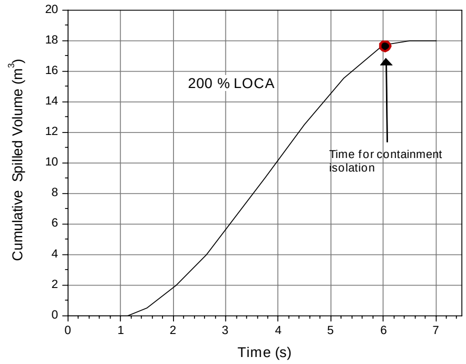
The Passive Containment Isolation System (PCIS) consists of a water tank (PCIT – Passive Containment Isolation Tank) and an exit pipe with initial water level maintained at a preset value. The space above the water level in the tank is kept in communication with volume V1 through a vent shaft. An exit pipe from the bottom of the tank is connected to ventilation duct. It may be noted that the volume V2 is ventilated to atmosphere through a ‘U’ duct. Under any operating conditions, the air space above the water level in the PCIT would experience the pressure of V1 volume while the exit pipe connected to the bottom of the tank would experience the pressure of V2 volume. Following an event of LOCA, the V1 and V2 zone undergo a pressure transient. The difference in the V1 and V2 pressure drives the water from containment isolation tank to U-shaped ventilation duct through exit pipe. The spilled water in the ventilation duct forms the U-shape liquid seal against the V2 pressure that prevents the release of gas/activity from the containment to atmosphere.

It may be noted that, although the V1 zone is also in communication with ventilation duct through a purge line, the pressure in the exit pipe of passive containment isolation system would be that of V2 pressure even with failure of purge line active isolation dampers as the volume of V2 zone is much more than that of V1 zone and the purge line is a small duct which cannot quickly pressurize the V2 region.

An experimental facility PCIF has been commissioned and a few set of experiments have been conducted to assess the performance of the system and is found to be satisfactory. It was observed that following the pressure transient, adequate water spills into U-duct of PCIS within 6s-7s.



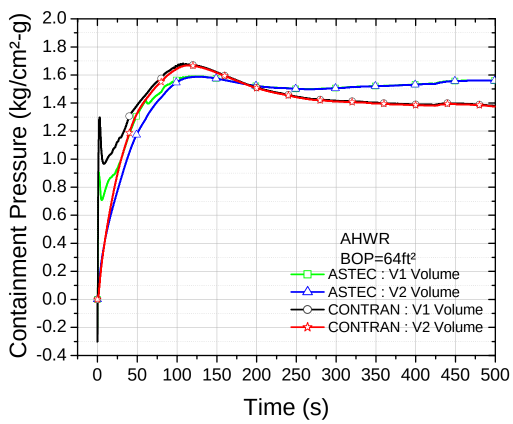
AHWR Passive Containment Isolation System

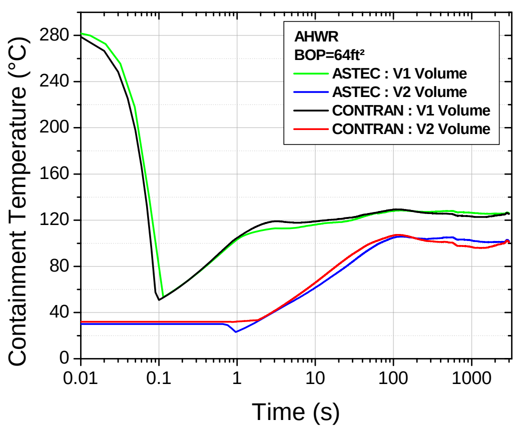



1/6 scale test facility of PCIS **Cumulative Water Spillage in U duct**

AHWR Containment System Evaluation

AHWR uses passive safety features for efficient and reliable management of potential extreme events. The pressure temperature transients in the containment were estimated and used in design of the containment structure along with other loads and the requirement for 100 years design life. Evaluation of all the passive safety features such as passive containment isolation system (PCIS) and passive containment coolers was also accomplished for the entire range of reactor inlet header break sizes(28).

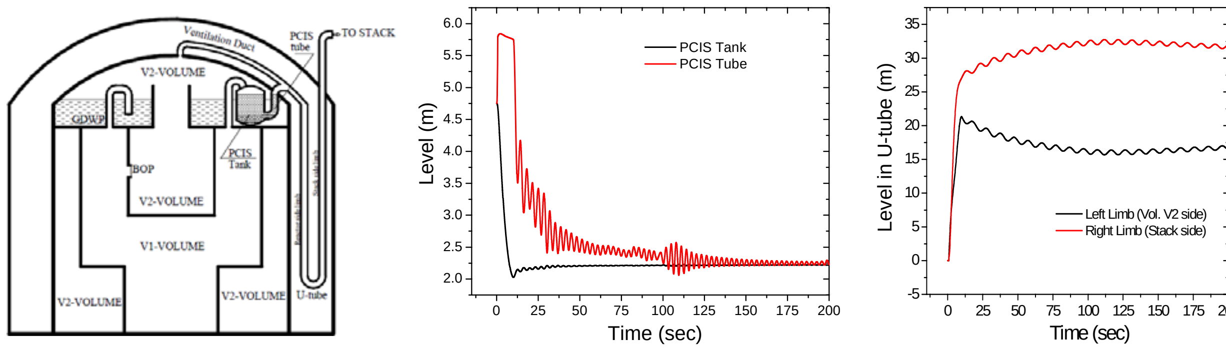




Pressure Temperature Transients for AHWR

Evaluationof Passive Containment Isolation System (PCIS)

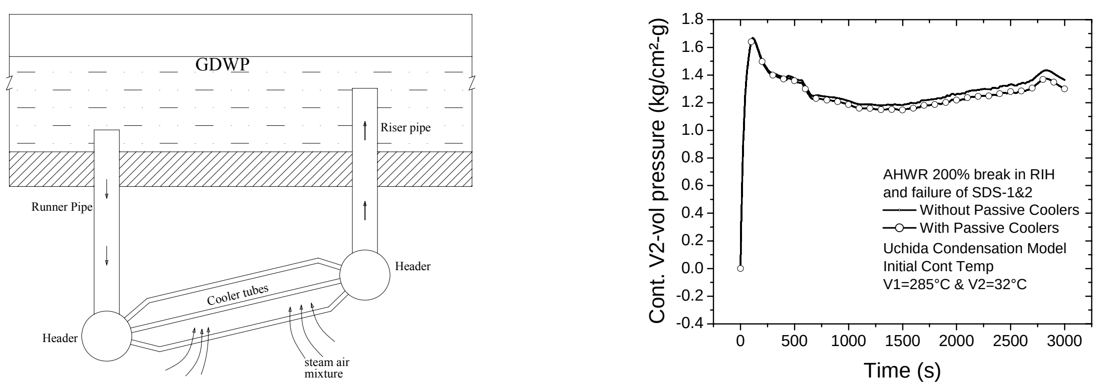
The containment will be isolated by PCIS in a passive manner, to prevent the escape of radioactivity to environment. Performance of PCIS has been evaluated under LOCA covering the entire range of RIH break sizes from 2 % to 200 %.(29) It is observed from the analysis that the PCIS will perform its intended function to isolate the containment within 15 seconds for all break sizes.



Passive Isolation System for AHWR

Passive Containment Coolers

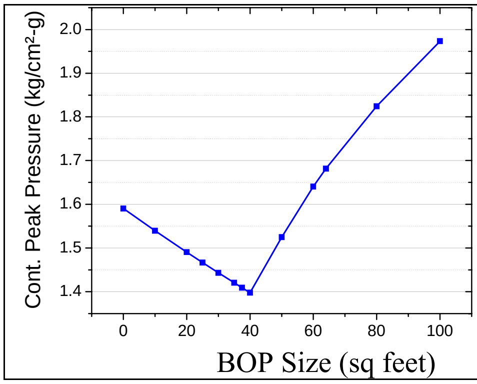
Passive containment coolers located at the top of AHWR containment (in volume V2 area) beneath Gravity Driven Water Pool (GDWP) are primarily meant for long term containment cooling by way of natural circulation. Their performance was evaluated under a loss of coolant accident as well as main steam line break[28].

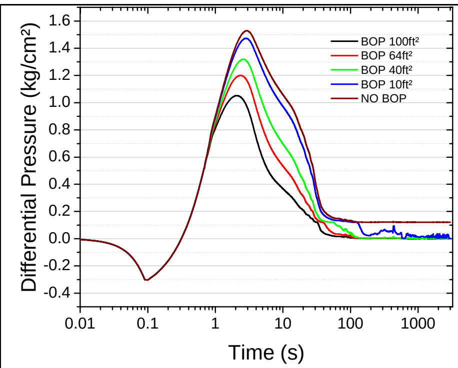


Passive Containment Coolers for AHWR

Optimization Studies for Break-Out Panel for AHWR[28]

Effect of BOP (Blow out Panel) size on containment peak pressure and differential pressure across V1 and V2 volumes have been studied for AHWR. BOP is provided in tail pipe tower structural wall separating V1 and V2 volume. Function of BOP is to limit differential pressure coming into wall in case of LOCA. The BOP ruptures if the differential pressure exceeds the set pressure and then pressure equilibrium is established between V1 and V2 volumes. Parametric study on the influence of the size of BOP on containment peak pressure has been carried out[28]. It was observed that the containment peak pressure passes through a minimum. Initially as BOP size increases, while steam flow bypasses to the V2 volume through BOP the transfer of air from V1 to V2 is lower and hence reduces containment peak pressure, with further increase in BOP area, very little or no flow passes through vent pipe leading to very low condensation of steam in the pool. Thus peak containment pressure rises again. This study was carried out to optimize the peak containment pressure.

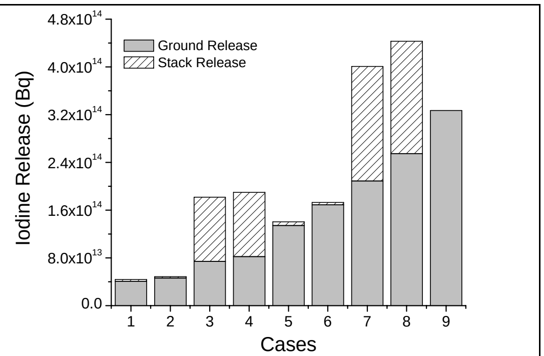




PSA for AHWR[30]

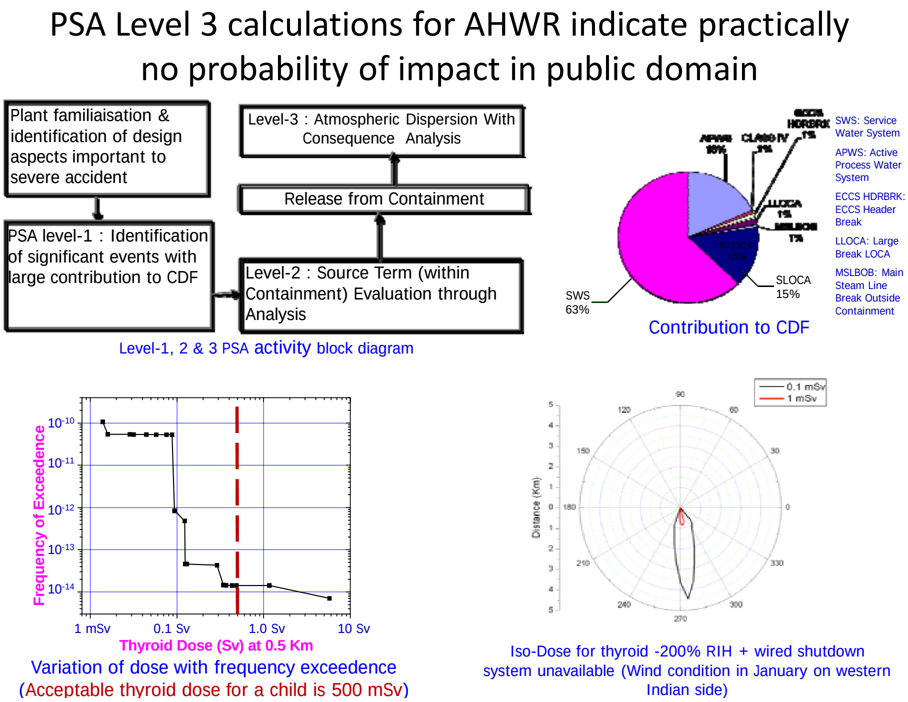
A comprehensive PSA study was carried out for AHWR. The study has indicated practically no impact in the public domain. Specifically analysis of most severe accident sequence (large LOCA with failure of SDS 1&2) with and without availability of various containment ESFs was carried out. Source term which consists of ground level release, stack level release and total activity release from containment was evaluated for 72 hours from the initiation of the accident. Effect of various Containment Engineered Safety Systems on iodine release from the containment has been evaluated.

	Containment Coolers	PCFPB	SCFRP	PCCD
CASE-1	ON	ON	ON	ON
CASE-2	ON	ON	OFF	ON
CASE-3	ON	OFF	ON	ON
CASE-4	ON	OFF	OFF	ON
CASE-5	OFF	ON	ON	ON
CASE-6	OFF	ON	OFF	ON
CASE-7	OFF	OFF	ON	ON
CASE-8	OFF	OFF	OFF	ON
CASE-9	OFF	OFF	OFF	OFF



Iodine release (in Bq) for various cases

Level 3 probabilistic safety analysis of AHWR has shown the benefits of this unique reactor configuration using on passive systems. As can be seen the core damage frequencies are very low as also the release of radioactivity in the environment is very low indicating practically no probability of any serious impact in the public domain.



6. Closing Remarks

Experience with accidents that have taken place at TMI, Chernobyl and Fukushima; has revealed significant gap between the risk perceived by people and the actual risk to which they were exposed. One would have imagined that with progressively greater familiarity, this gap would decrease as has happened with many new technologies that have appeared on the horizon. That somehow has not happened in case of nuclear energy. It seems to me that the cause for this situation lies in excessive over conservatism in estimating public consequence following the LNT principle. Studies with population living in high natural background areas have shown no increase in consequences attributable to radiation up to the levels of radiation that exist there. The Indian experience with regard to the dose to the public at the fence of the nuclear power plant for old generation of Indian PHWRs is 0.04 mSv per year, which has further reduced to 0.02 mSv per year for the new reactors. Thus a margin of 25 to 50 over the permissible dose rate of 1 mSv per year has been ensured under normal reactor operation. Further it may be noted that the studies conducted by Health Physics Association of United States[32] has concluded that the life time dose of 100 mSv over natural radiation results into no risk to the human life. Hence there is a threshold dose of 100 mSv. Indian data on High Background Radiation (HBR) areas is in agreement with this finding on this threshold value of radiation which will not cause any harm to the public.

It is important that we bring in greater realism in risk assessments estimates to bring them closer to actual observations that we now have in case of some of the severe accidents that have been witnessed. [32] This is necessary for a balanced view of different risks that the society is exposed to. With the threats of climate change looming large, we could in fact be increasing the risk to which humanity is exposed, unless carbon free energy supply shares a larger proportion of energy demand. With growing economy in countries with large population like India, China and others, which constitute nearly half of the world population, the demand for energy is bound to go up significantly. Nuclear energy seems to be the only viable option to meet this growing demand in an environmentally sustainable manner at least in the short run. Thus, while bringing in greater realism in understanding of risks in public domain remains an important objective, reconfiguring nuclear reactor systems including their containment systems in a manner that nearly eliminates adverse impact in the public domain is an urgent necessity. Evolution of nuclear reactors and their containment systems in India has been a journey towards continuous enhancement of safety not only in terms of reduction of risk but also in the context of virtual elimination of any impact in public domain that can be caused by the reactor. In this lecture, I have tried to give an overview of this evolution of containment systems in India with the hope that this would present a good case study. I am aware of nuclear power reactor and containment system development with similar objectives that is taking place around the world. These developments are urgently required to sustain nuclear renaissance that we expect would address the threats of climate change effectively.

Acknowledgement

I would like to acknowledge contribution of a large number of my colleagues in BARC and NPCIL who have worked on different aspects containment development and whose work is reflected in this lecture. In particular, I would like to thank Dr. R.K.Singh and Shri K.K.Vaze for putting together this manuscript.

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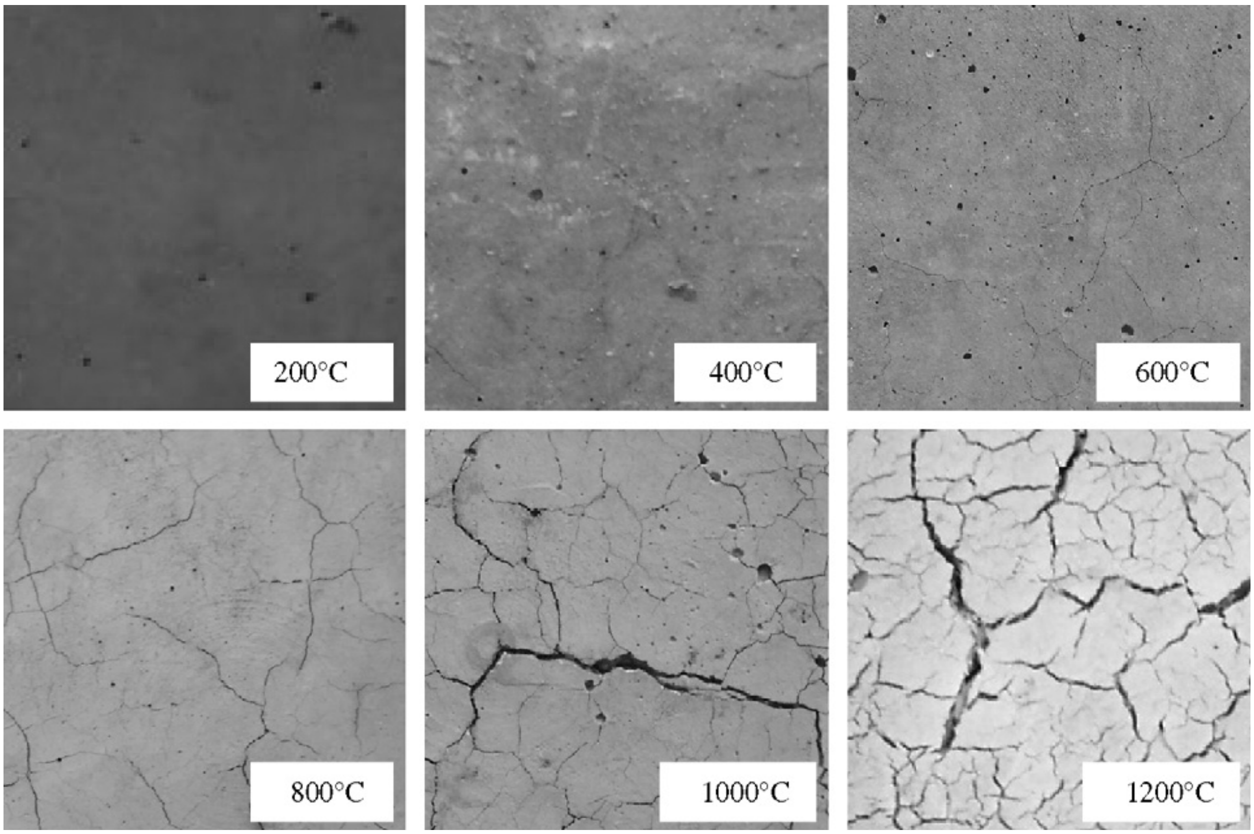
Backup Information on the Comments and editorial corrections

1. The editorial corrections are made in blue font.

2. The concrete strength degradation due to fire is as per the following from reported literature which has been added in the reference list and comments with regards to hygroscopic model which addresses the problem in a coupled manner has been indicated in the text.



3. A para has been added in the "7.0 Closing Remarks" on Indian Experience so far with the operation of 18 PHWRs and reference of Health Physics Association of United State findings of life time threshold dose of 100 mSv over natural radiation.