NUCLEAR SAFETY STUDIES RELATED TO CONTAINMENT THERMAL HYDRAULICS, SOURCE TERM ANALYSIS AND NUCLEAR SYSTEMS/COMPONENTS

Vishnu Verma, I. Thangamani, P. Goyal and R. K. Singh
Reactor Safety Division

Shri Vishnu Verma is the recipient of the DAE Scientific & Technical Excellence Award for the year 2013

Abstract

This paper briefly describes studies related to containment thermal hydraulics in nuclear reactor containments, source term evaluation, flow accelerated corrosion study, thermo-mechanical analysis of metallic fuel, containment safety related study in experimental facility, heat load analysis of ADSS components and fire test analysis in transportation cask.

Keywords

Containment Thermal Hydraulic, FAC, Metallic Fuel, Joule Heated Ceramic Melter, ADSS components

Comprehensive Nuclear Safety Studies related to containment thermal hydraulics, Source Term Analysis and Nuclear Systems/Components have been carried out for Indian nuclear program and are briefly described in the subsequent sections.

Containment Thermal Hydraulics

AHWR Containment Analysis for DBA

Containment thermal hydraulics analysis for AHWR containment has been carried out for evaluating containment peak pressure and temperature under Design Basis Accident (DBA) condition i.e. 200% break LOCA in Reactor Inlet Header (RIH) [1]. Results were compared with the containment having no liner (i.e. only concrete structure) and containment peak pressures found to be almost the same as obtained for the case of lined containment. An optimisation study for peak pressure variation with respect to Blow Out Panel area (see figure-1) has been carried out. BOP panels are required to limit the load generating on the walls due to differential pressure arising between two compartments in case of accident. Performance of Passive Containment Isolation System (PCIS) through Inclined Fuel Transfer Machine has been assessed for LOCA conditions [2]. Level variations in Transfer Fuel Storage Bay (TFSB) pool (see figure-2) and in primary ramp were evaluated during the transient for 200% break LOCA in RIH. It was found
that remaining water left in TFSB pool is sufficient to maintain the shielding requirements for pool and containment isolation.

![AHWR CONTAINMENT DBA ANALYSIS](image)

**Fig.1:** Variation of pressure in V1 and V2 volume for various BOP areas

PCIS analysis was also carried out for containment isolation from the environment in case of accident condition LOCA by forming U tube water seal due to pressure differential between V1 and V2 volume. Performance of PCIS has been evaluated for LOCA covering the entire range of RIH break sizes from 2 % to 200 %. It is observed from the analysis that the PCIS will perform its intended function within 15 seconds for all the break sizes.

![Level transient in TFSB for 200% break LOCA condition](image)

**Fig.2:** Level transient in TFSB for 200% break LOCA condition
Experiments Studies related to Containment Thermal Hydraulics

Containment Studies Facility (CSF) is designed to study containment performance under DBA and beyond DBA conditions for Indian PHWRs. The CSF (see figure-3) consists of a PHTM, a Containment System (CM) model and a Control and Instrumentation room.

![Containment Studies Facility](image1)

**Fig.3: Containment Studies Facility**

The containment model is approximately 1:250 volumetrically scaled down model of the prototype 220 MWe PHWR containment of Kaiga NPP. Like Indian PHWR containment system, the CM is also divided into V1 volume (dry well) and V2 volume (wet well). The V1 volume is further divided into many compartments to simulate the various rooms such as pump room, fuelling machine vault etc. present in actual reactor containment. The V2 volume is connected to V1 volume through vent pipes and suppression pool. The containment model is a cylindrical structure with an ellipsoidal dome, made out of RCC with epoxy painting on the inner surface. The outer diameter of the model is 6.9 m and its height is 10.95 m. Containment thermal hydraulics experiments have been performed for Blowdown pressure of 30, 50 and 75 kg/cm\(^2\) (g) [3]. The experimental data [3] were compared with the results of in-house containment code CONTRAN and ASTEC codes, for simulated blow down conditions and results were found to be in good agreement (see figure 4).

![Comparison: Experiment Vs. Analytical](image2)

**Fig.4: Comparison: Experiment Vs. Analytical (100 kg/cm\(^2\)-g)**
Suitable condensation model has been identified for the thermal hydraulic codes through this study. The present experiment and its validation have given requisite confidence on the PHWR / AHWR containment pressure and temperature transient evaluation with in-house CONTRAN and French code ASTEC. Besides with a systematic evaluation of various condensation models, the adequacy of Uchida model for containment transient calculations has been demonstrated.

Source Term Analysis

Fukushima Daichi Unit -1

Severe accident analysis of Fukushima Daichi Unit -1 has been carried under Station Black Out (SBO) condition initiated by Seismic and Tsunami events. In the analysis, containment thermal hydraulics, Molten Corium Concrete Interaction and Fission product (FP) transport analysis has been considered using ASTEC code. Analysis was carried out up to hydrogen detonation time in unit-1. For fission product transport in the containment, 18 species including Xe, Kr, Cs and Iodine and corresponding cases of decay heat have been modelled. Except Xe and Kr all other FPs including Iodine have been taken as aerosol form in the numerical model. Containment pressure & temperature transients and deposited and suspended mass of the FPs with time have been evaluated in all the zones. Hard vent opening and enhanced leakage at containment high pressure has been considered. Source term (ground and stack level release from containment to Environment) is evaluated up to H2 detonation. Inert atmosphere as initial condition has been considered in Primary Containment Vessel and in suppression chamber. Continuous release of corium from Reactor Pressure Vessel in case of vessel failure and external water injection were modelled. A large relocation of the molten corium on the floor of reactor cavity and consequent Molten Corium Concrete Interaction (MCCI) was observed releasing large amount of hydrogen and carbon monoxide, making primary containment to further pressurize and the reactor building susceptible to explosion. H2 and CO concentration has been plotted on Shapiro diagram for deflagration/detonation limits. It was found that containment pressure (see figure-5) reaches up to twice the containment design pressure and top region of reactor building enters into deflagration zone. Fig. 6 shows cavity erosion due to MCCI at the end of one day transient in Unit-1.

![Fig. 5: Containment Pressure Transient](image1)

![Fig. 6: Cavity Erosion due to MCCI](image2)
By and large, the present analysis [4] predictions agree well with the event chronology, recorded plant parameters and measured activity and with the analysis prediction of Nuclear and Industrial Safety Agency, Japan.

**IAEA- Benchmarking of severe accident computer codes for PHWR**

Containment analysis for severe accident conditions was carried out for 700 MWe CANDU reactor under IAEA Coordinated Research Project (CRP) [5]. Objective of this CRP was benchmarking of severe accident computer codes for Pressurized Heavy Water Reactor applications under prolonged Station Black Out conditions (SBO). Containment Pressure, Temperature transient, concrete ablation rate due to MCCI, hydrogen and CO distribution, fission product distribution in the containment have been evaluated using ASTEC code. Containment was divided into 14 zones. Flammability limits of H\(_2\) and CO has been checked with time on Shapiro diagram. Results were compared among 7 International Participants from different countries and found to be in good agreement. Fig. 7 shows comparison of time predicted by participant for different chronological events during progression of the severe accident. Figure 8 shows total amount of hydrogen generated during accident as predicted by all the participant.

![Fig. 7: Comparison of time predicted by participant for different chronological events](image1)

![Fig. 8: Total amount of hydrogen generated during accident](image2)
Thermal Analysis of VWSB-IP1 at Tarapur

Thermal Design and analysis of Vitrified Waste Storage Block-IP1 at Tarapur has been carried out [6]. The vitrified waste has high heat generation rate due to decay heat and needs interim storage under surveillance. The waste needs to be cooled continuously until major portion of the decay heat is dissipated. VWSB-IP1 is designed for interim storage of waste generated of 30yrs of Integrated Plant-1 under plant operation. VWSB-IP1 will be consisting of about 2192 canisters housed in 548 locations. Natural circulation air cooling was considered to cool the canisters. Canisters are placed in a storage vault and cooled by induced flow of air with the help of stack (Fig. 9). Each location houses four canisters stacked axially based on the parametric study.

![Fig. 9: Schematic view of the Facility](image)

Air from the inlet duct is introduced at the lower plenum of all the storage locations and flows axially through the annular gap between thimble and sleeve. The flow emerging from the top of each storage location is combined at the top plenum and goes to the stack. The air flow in the annular space between sleeve and thimble gets heated and rises due to buoyancy. The hot air rises through the stack due to chimney effect and maintains the flow through the vault. Through parametric study, stack dimensions, duct dimensions at inlet and outlet, plenum height, flow gap between sleeve and thimble etc. have been arrived based on concrete and canister temperature limitation. Analysis is divided in two steps; in first step converged flow rate has been obtained based on the 1-D flow model. Thereafter detail temperature field within the canister has been obtained using CFD code CFD ACE+ (Fig. 10). To validate results one miniature experimental facility was planned and the results were validated at IIT-Bombay.

![Fig. 10: Axial temperature variation along canister centre line, canister surface, Thimble and Ventilation pipe](image)

Metallic fuel for future FBRs

2-D thermo-mechanical analysis of Metallic fuel (U+15 % Pu) for future fast Breeder Reactors have been carried out [7]. These metallic fuel have merit with high breeding ratio beside power production. The fuel consists of U-Pu binary alloy (containing 15 % Pu) cladded in T91 alloy with Zr layer between the fuel slug and cladding material and helium gas in semicircular groove...
region. Two geometries (two and four groove, see figure 11) of fuel are considered for the analysis.

![Schematic diagram of the cross-section of (U-Pu) metallic fuel](image)

Fig. 11: Schematic diagram of the cross-section of (U-Pu) metallic fuel

Smeared density of fuel with these grooves is about 85%. The length of the fuel slug is 1000 mm. Thermal performance of the fuel has been evaluated based on maximum allowable Linear Heat Rate depending on the maximum permissible fuel centerline and clad temperature. For closed gap condition various models were studied for gaseous conductance and solid to solid conductance. Parametric study has been done and effect of gas thermal conductivity and contact pressure on gap conduction (gaseous and solid to solid) has been studied. Detailed temperature distribution was estimated using Finite Element Method. Contact pressure between clad and fuel slug due to differential thermal expansion was also evaluated with the help of finite element method and gap conductance was modified accordingly. Contact elements were used in-between clad and fuel slug. It was found that LHR limitation is first reached based on maximum clad temperature rather than fuel centerline temperature (see figure-12). Input data required for gap conductance was obtained from various models available for gaseous and solid to solid conductance.

![Variation of fuel centerline and clad inner temperature with LHR](image)

Fig. 12: Variation of fuel centerline and clad inner temperature with LHR (FEM analysis for geometry with four semicircular groove)

**Flow Accelerated Corrosion Study**

Flow Accelerated Corrosion (FAC) studies in Pipes and Elbows have been carried out for number of geometries. Analysis was done using CFD software. In each case wall shear stress has
been evaluated. Wear rate can be obtained by knowing shear stress with the help of correlation. FAC rate has been evaluated for RAPS-2 outlet feeders and Surry unit-2 feed water pipe failure (Incident occurred in Virginia, USA) using Colburn analogy and compared with the Lister model based on shear stress. Exact location of the pipe failure has also been predicted by CFD analysis [8]. Fig. 13 shows location of the pipe failure which occurred on the intrados of the second elbow of the feeders.

![Velocity contour plot](image)

**Fig.13: Velocity contour plot**

**Thermal Test Analysis of Radioactive Transportation Cask**

The transportation cask/packages are used to transport of radioactive material from one place to other. The design of packages should satisfy the regulatory requirements for safe transport of radioactive material. Different types of packages are used for transport of radioactive material. The package should demonstrate its compliance to various tests under normal and accident conditions, and in addition to the general design qualifications as required by regulatory authority. The overall objectives of all these tests are to demonstrate that the loss of radioactive contents or dose to the public do not exceed the respective limits specified by the regulatory guides. For thermal test, package should be tested on fully engulfed fire for at least of 30 minutes. A fire should be controlled to an extent that it sufficiently engulfs a test package and develops at least the required minimum heat flux (based on the temperature of 800 °C to ambient temperature) to the package.

A package may be thermally tested in a furnace if acceptable conditions in the furnace can be achieved. Alternately, the same can be achieved by carrying out requisite thermal analysis. The thermal analyses of a number of casks such as cask for transportation of Cesium source for irradiation, calibration lab for calibrating wide range of radiation monitoring instruments, marine product irradiator, and transportation of a coolant tube of PHWR etc. have been carried out in RSD in the recent past as per IAEA/AERB guidelines. Temperature distribution and lead melting if any have been evaluated for normal and accidental fire conditions. Analysis has been done using CFD ACE+ software. The analysis covers normal transport conditions, half an hour external conditions representing fire and also the post fire cool down period. Fig.14 shows temperature contour plot for Portable Research Irradiator (PRI) cask at the end of fire test.
Fig. 14: Temperature Contours in the PRI Cask at the end of fire test (30 minutes)

**Joule heated Ceramic Melter**

Thermal analysis of Joule Heated Ceramic Melter (JHCM) has been carried out for AVS-1 (Advance Vitrification System) and AVS-2 at Tarapur. In AVS, concentrated high level liquid waste and glass former (in the form of granules) is fed into JHCM, where electrodes immersed in the glass generates heat by the Joule heating due to electric current passing between electrodes through glass. The vitrified product glass melted in the furnace is withdrawn periodically by using a bottom freeze valve (central or side drain) consisting of Joule heated section followed by induction heated section into the stainless steel canister. Temperature at selected location were compared for both types the melter for normal condition (without draining), and different melt draining conditions (main drain and side draining). A temperature contour plot for side draining case is shown in figure-15. This analysis had given insight for further improvement in the JHCM design.

![Temperature Contours in the PRI Cask at the end of fire test](image)

Fig. 15: Temperature contour plot of AVS2 for side draining of glass

**ADSS components**

A linear accelerator comprising of Radio frequency quadruple (RFQ) and drift tube linac (DTL) is being developed by BARC under LEHIPA (Low Energy High Intensity Proton Accelerator) as
front end injector of the 1 GeV accelerator for the ADS programme. This DTL accelerates protons from energy of 3 MeV to 20 MeV. In DTL particles are accelerated by longitudinal electric fields at the gap crossings between drift tubes. The permanent magnet quad-poles are placed inside the hermetically sealed drift tubes and provide a constant magnetic field gradient in the beam aperture. The drift tubes are mounted concentrically inside the resonating DTL tank and are attached to the tank body with stems. Heat is generated in each drift tube due to Joule losses because of RF heating. Hence drift tubes should be continuously cooled to limit the temperature and resulting thermal deflection. Thermal deflection is required to evaluate frequency shift in the cavity. Analysis was done using CFD code CFD-ACE+ as thermo-mechanical coupled analysis simulating conjugate heat transfer and thermal deflection. Each drift tube have different geometry (higher length in axial direction) and heat generation rate. A number of drift tubes for different tanks have been analyzed for thermal deflection, pressure drop and temperature distribution [9]. Effect of flow velocity and change in the outer channel flow gap has also been studied on drift tube temperatures. Based on feedback from thermal analysis, outer channel size was suggested and the design was modified. Fig. 16 shows velocity vector plot in inner and outer channel of drift tube.

![Fig. 16: Velocity vector plot in a Drift Tube](image)

![Fig. 17: Temperature contour plot for DTL tank](image)

DTL tank is a very complex structure having opening for waveguide, vacuum port, cooling channels etc. About 115 kW of power is dissipated due to RF heating which is to be removed to limit the temperature and frequency shift due to cavity expansion. The incident heat flux on the wall is about 24 kW/m². Heat transfer analysis was carried out for 3-D geometry and minimum flow rate required to limit the temperature below the acceptable limit has been evaluated. Before detail 3-D analysis coolant channels size optimization was done with the help of 2-D analysis and modification were suggested. Fig. 17 shows temperature contour plot for the case of DTL tank.

**Closing Remarks**

With thermal analysis, CFD and suitably designed experiments, the relevant design and safety issues related to nuclear reactor containment thermal hydraulics, flow accelerated corrosion, thermo-mechanical analysis of metallic fuel, thermal analysis of ADSS components and fire test analysis in transportation cask have been addressed as presented in this paper.
References