CHAPTER 1
INTRODUCTION

1.1 FOREWORD

Nuclear power is one of the promising and environmentally sustainable options to cater to the ever increasing energy demand. Among the various types of nuclear reactors, the fast reactors have a strong potential to use a large fraction (about 75%) of uranium resource and burn the waste generated in thermal reactors. Considering these factors many developed as well as developing countries pursue active research in the area of science and technology of Fast Breeder Reactors (FBRs). In the wake of untoward sequence of events at Fukushima Daiichi nuclear accident, more emphasis has been placed on safety perspective of nuclear reactors all over the globe. With the advancement of numerical methods and computing technology in the recent past, Computational Fluid Dynamics (CFD) has been harnessed to address and solve many of the thermal hydraulic challenges in the design and safety of FBRs. This is evident from the published work of Tenchine for French FBRs and Velusamy et al. for Indian FBRs (Tenchine, 2010 and Velusamy et al., 2010).

In the design and safety analysis of FBR, post accident core material relocation and post accident heat removal assume primary importance, demanding systematic heat transfer analysis towards devising strategies to mitigate the consequences of extremely low probability beyond design basis events (Velusamy, 2008). The present research work aims at quantifying molten fuel relocation time to the main vessel lower plenum for a few postulated accident scenes and proposes a new conceptual core catcher - marking a few baby steps in this mammoth field full of interesting challenges and complexities.

All the studies performed as part of this research work belong to the less explored grey areas that lie interwoven within the core melt down accident sequences. Full fledged
experimental simulation of a severe nuclear accident is very difficult due to the small time scales and large quantum of heat generation involved, combined with significant spatial and temporal variations. An ideal approach would be to conduct small scale model experiments, complemented by computational modelling. However, in the present work, recourse to numerical simulation has been taken and validation has been achieved with benchmark data and empirical correlations.

Physical processes accompanying nuclear reactor accidents are rather complicated and computational capacities for the comprehensive analysis of the whole accident are limited. It is usual practice to evaluate the progression of these accidents in several stages, using separate computational modules which are robust for each of the particular processes and then integrating the findings of the separate effect studies. In the present research work, natural convection in cold sodium plenum, time for core-melt relocation to core catcher and heat transfer studies of a multi layer core catcher are investigated in succession, making use of the results obtained from one study as the input for the subsequent study as shown below.

Attempt has been made to state the envisaged sequence of events in unambiguous terms and carry out the mathematical modelling with appropriate physical justification for the adopted model. It is a tough task to define any nuclear accident progression precisely, in clear cut terms. Considering the example of core debris settling on the core catcher, this challenge is best described by Kayser who has illustrated different possible configurations of the fuel and stainless steel debris on the core catcher (Muller and
Gunther, 1982). His comical portrayal of the phenomenon is given in Appendix – 1. Though he has depicted it as a cartoon, it is serious enough to stir the thought of researchers, drawing their attention to the level of uncertainty that can exist in the definition of accident scenario. Therefore it is worth highlighting here that the usefulness and credibility of any accident analysis hinges on the initial axioms that are postulated at the beginning of the study and this particular study is no exception.

All the analyses have been performed for a whole core accident, in line with the targeted objective of conceptualizing a core catcher to handle debris generated from a whole core-melt. Heat generating mass consists of molten fuel component only, devoid of stainless steel and sodium. Relocation of the whole core material in the downward direction is improbable because a fraction of the molten core may be dispersed in the upward direction with reference to active core region and can settle on available flat surfaces on the upper part of the hot sodium pool. Another possibility of a part of the core to be left behind in its original location cannot be ruled out either. Nonetheless, whole core is considered to arrive at conservative estimates for the analysed output parameters such as core-melt relocation time to the core catcher and the thickness of the delay bed in the multi layer core catcher.

1.2 OVERVIEW OF CDA

The vertical section of a typical medium sized pool type fast reactor is depicted in Fig 1.1. The main components immersed in the large inventory of liquid sodium (~1100 t) inside the main vessel of a pool type FBR are the primary sodium pumps, intermediate heat exchangers, the entire lot of core subassemblies (SAs) which includes fuel, blanket, control rod, reflector and shielding SAs, grid plate supporting the SAs, inner vessel separating the cold and hot sodium pools, core support structure and the core catcher.
assembly. The core catcher is provided to protect the reactor main vessel from thermal and mechanical loads produced by the relocated core-melt / debris following an accident.

Argon acts as cover gas above the free surface of sodium. The roof slab which forms the thermal and biological shield accommodates the control plug, small rotatable plug and large rotatable plug facilitating fuel handling. Grid Plate (GP) consists of two stainless steel plates each of thickness 5 cm, separated by a sodium volume of 1 m axial gap in which the two primary sodium pumps, feed cold sodium through four sodium pipes to all the subassemblies.

The overall description of a core disruptive accident is given in the next few sections. An Unprotected Loss Of Flow Accident (ULOFA) or a Loss of Heat Sink Accident (LOHS) in a fast reactor can lead to whole core melt down depending upon the
severity of the accident. The accident is termed unprotected if the shutdown systems (control rods) of the reactor fail to trip/shut down the reactor by SCRAM (Safety Control Rod Accelerated Movement) on demand. Unprotected Transient Overpower Accident (UTOPA) can be arrested and terminated early due to the fuel squirting behaviour (Sathiyasheela, 2008). Fast reactors are provided with two diverse and fast acting shutdown systems, viz., Shut Down System – 1 (SDS-1) and Shut down System – 2 (SDS – 2), which are capable of bringing the reactor to a safe shutdown state independently, in case of an initiator of an accident.

Traditionally, SDS-1, the first group of control rods is used for power and reactivity control as well as for shutting down the reactor by SCRAM whereas SDS-2 which comprises of another group of control rods is used only for tripping the reactor by way of SCRAM. Each system is capable of shutting down the reactor independently with adequate safety margin. Also, single failure criterion is applied such that effective reactor shut down is ensured even when one control rod in a group does not act. Diverse concepts are employed in the design of the two systems. The reactor parameters that are continuously monitored to trigger SCRAM are divided into two different groups each activating one shutdown system on demand. An optical link is provided between the two systems to ensure that both systems act even when one is actuated by a SCRAM parameter crossing the threshold. The probability of simultaneous failure of these two devices is kept below $10^{-6}$/reactor year because of the extreme care taken in their design. Therefore, Core Disruptive Accident (CDA) has been categorized as Beyond Design Basis Events (BDBE) in fast reactor design.

Nevertheless, analysis of the severity of these events becomes necessary in order to assess the damage they can cause. Such assessment helps in the design of certain engineered safeguard features which can be incorporated in the reactor to mitigate the
consequences of the BDBE. This philosophy is called the defense in depth philosophy which aims at minimising the hazard and risk associated with nuclear reactors. The chain of events describing a CDA in a fast breeder reactor is pictorially represented in Fig. 1.2.

![Diagram of CDA Scenario in FBR](image)

**Fig. 1.2 CDA Scenario in FBR**

The progression of CDA is generally analysed step by step using the cause and effect relationship under different phases namely, predisassembly phase, transition phase, disassembly phase, system response phase and post accident heat removal phase (Walter and Reynolds, 1981). A brief description of these phases is given below.

1.2.1 **Predisassembly Phase**

This phase of LOFA / TOPA analysis involves the calculations of core neutronics, reactivity feedbacks, thermal hydraulics, sodium boiling, fuel pin mechanics leading to failure, cladding and fuel slumping and their relocation and fuel coolant interaction. Predisassembly phase lasts till the fuel reaches the boiling point and starts dispersing the
core material. Beyond this point, fuel displacement reactivity feedback dominates with short time scales and this phase is known as disassembly phase. Predisassembly phase provides the initial conditions for the disassembly phase.

1.2.2 Transition Phase

The predisassembly phase can lead to early termination of the accident if there are sufficient negative feedbacks. On the other hand if reactivity addition rates are large, it leads to energetic dispersal of the core which is called the disassembly phase. A third possibility comes into picture when there are insufficient negative feedbacks to terminate the accident and insufficient reactivity addition for core explosion. In such a case, large segments of core could slowly melt and proceed to involve the whole core. This slowly developing boiling process has been termed the Transition Phase, since it is sandwiched between predisassembly and disassembly phases. In this phase, the multichannel model of predisassembly phase breaks down and one cannot use the disassembly idealization of treating the entire core as fluid. The end of the neutronic event may come either by energetic disassembly or due to gradual boiling or melting of the core.

1.2.3 Disassembly Phase

In this phase, one calculates mainly the fuel displacement feedback coupled with the core neutronics and core hydrodynamics. This phase lasts till the reactor attains subcriticality due to core dispersal. The timescale of disassembly phase is very small, of the order of milliseconds.

1.2.4 System Response Phase

At the end of the disassembly phase, though neutronically the reactor is in its shutdown state, core is still in its expanding state and thermal energy release is capable of performing significant mechanical work on the system. This mechanical work is augmented by the Fuel Coolant Interaction (FCI) and consequent vapourisation and
expansion of coolant. Once the source pressure term is established from FCI or normal reactor material expansion, the response of the reactor system is analysed.

1.2.5 Post Accident Heat Removal (PAHR)

The final phase in the analysis is the evaluation of PAHR. That is one wants to know how the fuel comes to rest in various parts of the system ultimately, where it can be permanently cooled. This present research work addresses a few issues related to this phase of CDA.

1.3 ANALYSIS OF LOFA

Since UTOPA gets terminated early due to the fuel squirting behaviour, only ULOFA exemplifies the generic behavior over the whole range of CDA spectrum of circumstances. Hence it can be used to adequately characterize the spectra of energetic consequences. One of the initiators for ULOFA is loss of electrical power supply to the primary pumps. In this type of accident, coolant boiling in coolant channels takes place first. Due to positive void coefficient, power shoots up which then leads to fuel pin failure and fuel slumping.

Four characteristics of LMFBR cores – positive sodium void worth, core geometry not in its most reactive configuration, high fuel reactivity worth and an extensive background of experience in accident analysis, led to choice of ULOFA as the principal scenario for study, in the international community. In the safety analysis of both SUPER PHENIX (Rigoleur, 1982) and EFR (Dufour, 2007), only ULOFA was considered as whole core accident for safety analysis. A possible scenario that leads to ULOFA in a FBR is shown schematically in Fig. 1.3.

The transient is initiated due to loss of primary coolant flow resulting from power supply failure to the primary pumps. The pump speed reduces gradually due to the large inertia of the flywheel. Flow reduction leads to coolant temperature rise in core that gives
positive reactivity. The heating of the spacer pads results in negative reactivity. These negative reactivity components dominate over the positive reactivity components from clad and coolant heating and hence the net reactivity is negative. This results in decrease
in power. However, the power to flow ratio is high which leads to coolant temperature rise and ultimately coolant boiling (voiding) in the upper part of the core. Subsequently, sodium voiding spreads radially outward and axially downward. Due to the positive reactivity introduced when the sodium voiding propagates into the central part of the core, the net reactivity begins to increase and becomes positive. It leads to power excursion and finally to rapid increase in clad and fuel temperatures that result in clad and fuel melting. At this stage, molten fuel may be swept out of the core by shearing force of fission gas pressure in irradiated fuel. This will lead to large negative reactivity addition and reactor shutdown. A part of the fuel propagates downwards, melting the underlying materials. To model this transition phase, detailed evaluation of the fuel and clad movement and the consequent reactivity effects are required (Srinivasan, 2008).

1.4 ANALYSIS OF PROTECTED LOSS OF HEAT SINK (PLOHS) ACCIDENT

A fast breeder reactor can enter into an extremely low probability event of core melt down called PLOHS when there is no means of decay heat removal generated within the nuclear fuel even after the reactor is shut down. Normally, FBRs are provided with two decay heat removal systems. The probability of combined failure of both these systems is below $10^{-7}$/reactor year. Nevertheless, their integrity could be jeopardized due to an earthquake of very large magnitude or a combination of natural calamities for which they are not designed. This can culminate in a PLOHS accident.

1.5 MOTIVATION FOR THE CURRENT RESEARCH

As stated in the foreword, this research aims at addressing a few thermal hydraulic issues that fall under the scope of severe accident analysis following a core disruptive / melt down accident in a FBR. The molten nuclear fuel which is flowing out of the disrupted core is called ‘core-melt’. Eventually it becomes ‘core debris’ after being quenched by liquid sodium. Core-melt is a heat source not only by virtue of its initial
high temperature but mainly because of the decay heat of contained fission products. The fission products decay into stable isotopes with course of time. Therefore decay heat is a rapidly decreasing function of time.

In the first part of the research work, natural convection of liquid sodium in lower plenum of reactor main vessel is investigated. Empirical correlations for convective heat transfer in liquid metals are rather limited because of inherent difficulties in conducting experiments with them. Internal natural convection in an enclosure heated from above and cooled at the cylindrical side wall is of paramount importance in the post accident scenario of a FBR. In the present study, the vertical cylindrical enclosure containing liquid sodium is heated from above and cooled along the side wall. Heat transfer correlations for such a geometry and low Prandtl number (0.004) are not reported in open literature. A few correlations are available for a hot plate facing upward / downward in an infinite liquid metal pool or a differentially heated cavity where one of the side walls is maintained at a higher temperature than the other side wall which is cold. Yadav and Kant (2008) have experimentally studied conjugate heat transfer from a vertical flat plate of finite thickness under natural convective cooling and concluded that conductivity ratio and aspect ratio greatly influence the heat transfer behaviour. Gazit (1998) in his study with mercury bath has quoted a very early work by Stewart and Weinberg stating that natural convection in enclosures depends strongly on geometry and quantitative results for one geometry cannot be derived from another. Jabbar (2001) has reviewed natural convective heat transfer coefficients for high Prandtl number liquids in two and three dimensional enclosures and summarized the discrepancies between various correlations. The discrepancies are found to be upto a factor of 5 for vertical surfaces, a factor of 4 for horizontal surfaces facing upward and upto a factor of 8 for horizontal surfaces facing downward for high prandtl number liquids. This large uncertainty associated with heat
transfer from downward facing hot surfaces in enclosures has further motivated the present study of developing natural convective heat transfer correlations. These correlations are essential for prescribing realistic boundary condition for the numerical investigation related to grid plate melting which is the next part of the study.

In the second part of the study, a mathematical model is formulated using the explicit enthalpy method and a finite difference code HEATRAN-1 is developed for solving heat conduction heat transfer in the grid plate with phase change. Heat transfer problems involving phase change are nonlinear and are not amenable to analytical solutions except in very simple cases. In the problem at hand, nonlinear heat source and boundary conditions add further to the complexity of the problem which has necessitated the development of a computer code for a reasonably good prediction of grid plate melting. The melting models available in many commercial codes do not offer scope for prescribing separate properties for solid and liquid phases. Moreover tracking of the melt-front with respect to time and displacement of the melting substance which is in contact with a denser hot source are very difficult to be implemented in these codes. To overcome these difficulties, HEATRAN-1 code has been developed to analyse grid plate melting problem.

As we know, core-melt relocation time is one of the crucial input parameters for the design of core catcher. Therefore, having obtained the time estimate for grid plate melt-through, a conceptual core catcher is hypothesized to withstand a whole core meltdown and this forms the final part of the research. For generation IV sodium fast reactors, in-vessel accommodation of a whole core accident is desired for enhancing the safety of the reactor. The need for such an improved core catcher for future FBRS has given the impetus for this part of the study. Sharma et al. (2009), in their study have proposed a triple tray core catcher for Indian FBRs. But distribution of core debris
equally on all the trays is a challenge. In the upcoming Japanese reactor JSFR, multiple tray approach is adopted with special guide tubes to effect tray to tray debris transfer (Sato, 2009). After considering various design options for FBR core catchers, particular emphasis has been placed on multi layer core catcher, in the present work. A multi layer core catcher comprising of a top sacrificial layer, a middle delay bed and a base layer of high mechanical strength is proposed and its adequacy is substantiated by detailed heat transfer analysis.

1.6 SPECIFIC PROBLEMS ADDRESSED

1.6.1 Natural Convection in Cold Sodium Plenum

During natural convection in enclosures, the fluid is driven by density variations in a body-force field, and natural convection heat transfer is highly sensitive to the heating or cooling conditions at the boundaries (Akins, 1986). Flow and temperature fields in these systems are governed by the continuity equation, Navier-Stokes equations and energy equation. Turbulence is modelled by solving two additional transport equations, one for turbulent kinetic energy and the other for its dissipation rate. Due to the non linearity of the equations and a strong coupling between momentum and energy equations, general analytical solutions are still not possible except for a few simple cases. Most research efforts have been based on experimental work and, more recently, on numerical approaches particularly for liquid metals (Wenxian, 2004).

In this research work, CFD analysis is carried out to estimate heat transfer coefficient for prescribing proper convective boundary condition at the bottom of the grid plate for its melting studies. Based on detailed parametric studies, Nusselt number correlations are obtained as a function of Boussinesq number for both isothermal and isoflux boundary conditions for the top heated and side wall cooled sodium filled cylindrical cavity. In the next case, the lower sodium plenum is approximated as a
cylindrical enclosure. Here, both transient and steady state heat transfer correlations are developed for natural convection setting in liquid sodium in the enclosure heated from above and cooled at the side wall. The effect of the finite thickness of top plate on heat transfer is also investigated by imposing constant temperature boundary on the top surface of the plate. A correlation is developed which shows the dependence of Nusselt number on Boussinesq number, conductivity ratio and height ratio for a thick plate.

1.6.2 Core-melt Relocation Studies

The second part of the research deals with core-melt relocation time to the core catcher. Heat transfer studies have been carried out to obtain the time taken for core-melt relocation by penetration through the underlying steel structures before settling on the core catcher. This time estimate is essential in defining initial thermal load on the core catcher (Roychowdhury, 2003). ULOFA and PLOHS are taken to be bounding events which would give an estimate of minimum and maximum time for core-melt relocation to the core catcher. In the case of ULOFA it is envisaged that the core-melt is swept out of the core due to the combined effect of core bubble and fission gas pressures and it gets deposited on the grid plate. The disrupted core and the core-melt lying above the bottom plate of grid plate are depicted in Fig. 1.4. The starting point of the present calculation is that the core-melt has reached the lower plate of the grid plate and rests on it, thereby melting it. Hence, grid plate is the only structure considered to undergo melting to determine core-melt relocation time. The case of PLOHS event is perceived to lead to the slowest of core-melt progression, where it is assumed that the active core is molten and subassemblies remain intact with melting starting from the bottom axial blanket region and progressing through successive underlying regions. Here, the molten material movement is idealistically taken to resemble that of a candle burning slowly and melting away. In the case of ULOFA, core-melt reaches the grid plate within a few seconds after
the accident and the lower plate of the grid plate is a solid obstacle for the molten corium to melt through, before settling on the in-vessel core catcher.

![Fig. 1.4 Vertical section of a FBR with degraded core after a CDA](image)

1- Inner vessel 2- Main Vessel 3- Safety Vessel
4- Core catcher 5- Grid plate 6- Core-melt

**Fig. 1.4 Vertical section of a FBR with degraded core after a CDA**

This problem of ‘grid plate melt-through’ for some possible core-melt configurations and boundary conditions is addressed as part of the investigation of ULOFA scenario using HEATTRAN-1 code. Heat transfer analysis has been carried out using the code for a medium sized mixed oxide fuelled reactor. Decay heat as a function of time is the volumetric heat source in the core-melt. The code is validated with both
semi-analytical solution and available benchmark data for BN 800 reactor problem. A salient feature of the code is that it employs dynamic adaptation of the computational mesh, giving way for the molten grid plate material to be displaced as the heavier molten fuel sinks into it. The thickness of the thin molten film underneath the core-melt is calculated by the analytical solution obtained from the work of Moallemi and Viskanta who have extensively worked in the field of migrating/moving heat sources on melting substrates. The code can capture the evolution of phase change front with time and also can handle separate properties for the solid and liquid phases.

In the next case, PLOHS event is analysed to determine melt relocation time. The vertical section of a fuel subassembly and the computational domain for this PLOHS event are illustrated in Fig. 1.5.

Core-melt progression takes place slowly starting from the bottom axial blanket region, moving through the fission gas plenum, tail piece, discriminator and grid plate top plate. Porous body formulation is adopted and effective thermophysical properties are defined for each region which are made up of different components such as uranium oxide, sodium, stainless steel cladding etc. For this analysis, it is assumed that boiling fuel is in constant touch with the underlying composite regions.

1.6.3 Multi Layer Core Catcher

Finally, the heat bearing ability of the new proposed core catcher is analysed using the core-melt relocation time obtained from the previous study to define the initial decay heat load on it. In general, for nuclear reactors, there are two approaches for permanent retention of core debris. The retention may be achieved either, in-vessel, within the primary system or, ex-vessel, external to the primary system but within the containment building.
Fig. 1.5 Schematic of (a) FBR fuel subassembly and (b) mathematical model for PLOHS

The invessel core catchers are generally single or multiple plates with inclusion of sacrificial layers. Ex-vessel core catchers adopt large spreading compartment type catchers as in EPR or multiple crucible catchers as in VVER. But in the case of sodium
cooled fast breeders, because of hazardous reaction of sodium with air and water, main vessel has to be intact and invessel core catcher is the best option. Therefore, the core catcher is provided at the bottom of the main vessel to protect it from intense thermal and mechanical loads following an accident. Nevertheless, there are many possible design options which might enhance core debris retention capacity. The choice of core catcher material and its configuration leading to its optimization is an important scientific task which needs to be addressed for future fast breeders. Several design options of core catchers for fast breeders are discussed in the present work with particular emphasis on the multi layer core catcher. A multi layer core catcher comprising of a sacrificial layer of Molybdenum, delay bed of thoria or magnesia and the bottom most base layer made of stainless steel is proposed and its adequacy is substantiated by heat transfer analysis.

1.7 ORGANIZATION OF THE THESIS

Chapter 1 gives introduction to the research problem and outlines the scope of the work. Chapter 2 provides the review of literature in the relevant research areas. Mathematical formulation of the problems and solution procedures are elucidated in Chapter 3. Chapter 4 deals with natural convection studies in top heated and side wall cooled sodium filled cylindrical enclosures in general followed by analysis of the lower cold sodium plenum of the reactor vessel in particular. The development of HEATRAN-1, the transient heat conduction code which includes melting phenomenon is detailed and its application to the grid melting problem following ULOFA is presented in Chapter 5. Core-melt relocation following PLOHS accident is also presented in the same chapter. The concept of multi layer core catcher is put forth along with relevant heat transfer analysis and the results are highlighted in Chapter 6. The major conclusions drawn from the studies are summarized in Chapter 7.