# Thermal Hydraulics Studies for Future Indian Fast Breeder Reactors

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**Abstract**

Prototype Fast Breeder Reactor is a 500 MWe pool type liquid sodium cooled nuclear reactor presently under commissioning at Kalpakkam, India. The design for next generation higher capacity Fast Breeder Reactors 1&2 (FBR1&2) has been commenced. The experience gained from the design and construction of PFBR has been utilized in the optimized design of FBR1&2 with enhanced safety and improved economy as the main targets. FBR1&2 is a pool type fast breeder reactor adopting twin-unit concept. The design changes envisaged for FBR1&2 includes, (i) two primary pipe per sodium pump, (ii) inner vessel with single torus lower shell, and (iii) reduced main vessel diameter with narrow gap cooling baffles. This paper discussed about the 3D computational fluid dynamic studies carried out for the FBR1&2 in the following topics: (a) Studies on optimization of the location & number of anti-gas entrainment baffles towards reducing the gas entrainment from the sodium free surface, (b) Prediction of velocity and temperature distribution in the inlet & outlet window of intermediate heat exchanger (IHX) for providing the input to FIV studies of tube bundle and also to estimate the temperature variation in the outlet plenum of the IHX, (c) Assessing the adequacy of the decay heat exchanger (DHX) toward removing the decay heat, (d) Transient response of hot pool to reactor thermal hydraulic transients for estimating the transient thermal load on the hot pool components, (e) Steady & transient study of the integrated hot pool – cold pool for estimating the heat transfer through inner vessel and also the transient thermal load on the cold pool and hot pool components,

## D:\AITHAL\ACAD Drgs\CFBR\600 MWe\Homogeneous Core\CBR_600MWe_RA_GA_Colour_16-08-2016_wo Shade.jpgINTRODUCTION

Prototype Fast Breeder Reactor (PFBR) [1] is a 1263 MWt, 500 MWe sodium cooled pool type reactor presently under construction in Kalpakkam. The design for next generation higher capacity Fast Breeder Reactors (FBR1&2) [2] has been commenced with enhanced safety and improved economy as the main targets (Fig. 1). The experience gained from the design and construction of PFBR has been utilized in the optimized design of FBR1&2 with enhanced safety and improved economy as the main targets. FBR1&2 is pool type reactor with innovative modifications to achieve specific targets like higher reactor power, core optimization for higher breeding ratio, specific material inventory reduction, simplified systems and components, integrated manufacture and erection, twin units sharing non-safety systems, reliability enhanced decay heat removal systems and enhanced in-service inspection and repair features. Due to the modified design, there are many dimensional and flow changes in reactor components. The design changes envisaged for FBR1&2 includes, (i) two primary pipe per sodium pump, (ii) inner vessel with single torus lower shell, and (iii) reduced main vessel diameter with narrow gap cooling baffles. This paper discussed about the 3D Computational Fluid Dynamics (CFD) thermal hydraulics studies carried out for the FBR1&2 in the following topics (a) Studies on optimization of the location & number of anti-gas entrainment baffles towards reducing the gas entrainment from the sodium free surface, (b) Prediction of velocity and temperature distribution in the inlet & outlet window of intermediate heat exchanger (IHX) for providing the input to FIV studies of tube bundle and also to estimate the temperature variation in the outlet plenum of the IHX, (c) Assessing the adequacy of the decay heat exchanger (DHX) toward removing the decay heat, (d) Transient response of hot pool to reactor thermal hydraulic transients for estimating the transient thermal load on the hot pool components, (e) Steady & transient study of the integrated hot pool – cold pool for estimating the heat transfer through inner vessel and also the transient thermal load on the cold pool and hot pool components.

*Fig. 1 Vertical section of FBR1&2*

2. OPTIMIZATION OF ANTI-GAS ENTRAINMENT BAFFLES

The sodium inventory inside the reactor is divided by inner vessel (IV) into two portions, viz., hot pool and cold pool. The sodium volume contained within the IV is called hot pool. The sodium volume contained outside the IV (and within main vessel) is called cold pool. There is an argon cover gas region above the sodium pools. The argon cover gas is maintained above hot and cold pool sodium inside reactor to avoid air ingress and to accommodate volumetric changes of sodium. The hot pool sodium after coming out of core takes a 90° turn, flow along the IV and reaches the free surface. The free surface is turbulent in nature and the maximum free surface velocity is to be limited to 0.5 m/s [3] to avoid the entrainment of argon into sodium flow. The maximum free surface velocity can be limited by adding a baffle attached to the IV. The location of baffle and width of baffle influences the maximum free surface velocity. So it is necessary to investigate the effect of gas entrainment baffle on the free surface velocity. Towards this, three dimensional CFD study of flow and temperature distribution of sodium in hot pool of FBR1&2 is carried out using ANSYS FLUENT [4]. The computational mesh size is 1.05 million and the time taken for 1 simulation in a 16 core CPU is 3 days. For all the subsequent works, high Re k-ε model is used for modelling turbulence [5] and standard wall function with y+ > 30 is used for modelling velocity profile near wall. A maximum free surface velocity of 0.89 m/s is observed in the hot pool which is more than the allowable value of 0.5 m/s from gas entrainment considerations. The maximum free surface velocity decreases rapidly when the baffle width is increased from 100 mm to 500 mm and after 500 mm, the velocity remains nearly constant. So, a baffle of 500 mm width provided at 1.3 m below the free level reduced the maximum free surface velocity to 0.57 m/s. Though the free surface velocity is above 0.5 m/s, the percentage of area where velocity is above 0.5 m/s is 2.9 % only. The addition of vertical baffle also has less say in reducing the free surface velocity. Figures 2a and 2b shows the velocity vectors of sodium at the free surface with/without baffle. Because of location of fuel handling port and dissimilar no of IHX/Pump (4 IHX, 3 pumps), geometrical symmetricity among major components is not achievable. Because of this, velocity vectors in between IHX-1/Pump-1 and IHX-2/Pump-2 are not similar as in Figs. 2a and 2b.

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| Vec-Top-Baf_0 | Vec-Top-Baf_500 |
| *Fig. 2a Velocity vectors (m/s) of sodium at the free surface without baffle* | *Fig. 2b Velocity vectors (m/s) of sodium at the free surface with baffle* |

3. PREDICTION OF FLOW & TEMPERATURE DISTRIBUTION IN THE INLET WINDOW OF IHX

The Intermediate Heat Exchanger (IHX) connects hot pool and cold pool. The hot primary sodium which comes out of the Subassembly (SA) reaches hot pool and from there it flows through the IHX to the cold pool. Sodium from hot pool flowing in the shell side of IHX transfers heat to secondary sodium flowing inside the tubes of IHX. The temperature distribution of secondary sodium in the IHX depends on the temperature and velocity distribution of primary sodium at the inlet window. Moreover, the cross flow velocity of primary sodium faced by the tube bundle is important for the Flow Induced Vibration (FIV) studies of the IHX tubes. Prediction of velocity and temperature distribution in the inlet & outlet window of IHX is important for providing input to FIV studies of tube bundle and also to estimate the temperature variation in the outlet plenum of the IHX. Towards this, a three dimensional CFD study of flow and temperature distribution of sodium in hot pool is carried out to estimate the velocity distribution at the IHX inlet window. The computational mesh size is 1.05 million and the time taken for 1 simulation in a 16 core CPU is 3 days. The maximum resultant velocity predicted at the inlet window of IHX is 0.8 m/s. From the inlet temperature profiles, a maximum ΔT of 4 K is observed. Figures 3 and 4 show the velocity and temperature distribution over the inlet window of IHX for various circumferential positions.

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| *Fig. 3Variation of resultant velocity of sodium over the inlet window of IHX at different circumferential positions.* | *Fig. 4 Variation of sodium temperature over the inlet window of IHX at different circumferential positions.* |

4. ASSESSING THE ADEQUACY OF THE DECAY HEAT EXCHANGER

Decay Heat Removal (DHR) systems are envisaged to remove decay heat generated in the core after reactor shutdown. The Operation Grade DHR system transfers the decay heat to steam-water circuit through secondary sodium circuit, when at least one secondary sodium loop, DHR related steam-water circuit and offsite power supply system are available. Safety Grade Decay Heat Removal System (SGDHRS) is the other path which is available even under station blackout condition. In the conceptual design of the reactor, it is proposed to have SGDHRS with four independent circuits; each circuit having 10 MW capacity when the hot pool temperature is 547 °C and ambient air temperature is 40 °C. Each circuit consists of a Na-Na Decay Heat exchanger (DHX) dipped in the hot pool and a Na-Air heat exchanger (AHX) placed on the Steam Generator (SG) building, an expansion tank, a storage tank, associated piping and valves, air dampers and a stack (Fig. 5). Thermal hydraulics design of DHX-A for FBR 1&2 has been carried out. The computational mesh size is 1.2 million and the time taken for 1 simulation in a 16 core CPU is 4 days. From the study, it is found that the DHX is capable of removing the required capacity of 10 MW. In order to confirm the capacity of DHX, three dimensional thermal hydraulics analysis of DHX is carried out. Based on the analysis, it is seen that the DHX-A is capable of removing 11.3 MW of heat from the hot pool.

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| *Fig. 5 Schematic of passive circuit of SGDHRS.* | *Fig. 6 Schematic of DHX - A* |

5 PREDICTION OF TRANSIENT THERMAL LOAD ON HOT POOL COMPONENTS

Hot sodium from various SA in the core reaches hot pool and from there it flows through the IHX to cold pool. Sodium from the hot pool flowing in the shell side of IHX transfers heat to secondary sodium flowing inside the tubes. In case reactor SCRAM is initiated due to some event happening in the plant, power produced in the core reduces suddenly to decay heat level. Moreover, after SCRAM, automatic coast down of primary and secondary sodium pump speeds also take place. Since no fission power is generated, the temperature of sodium exiting the core reduces much below that present in the hot pool. Hence, there is a possibility of thermal stratification conditions to be developed in the hot pool where the cold sodium from the core tries to settle at the bottom of the pool. Subsequently, the stratification interface moves upwards because of continues inflow of cold sodium from the core. Thermal hydraulic effects of these phenomenon in hot pool needs to be investigated to establish transient thermal loading on various components (IHX, pump, control plug (CP), Inner Vessel (IV) etc). The cold sodium front from the core moves towards IHX inlet window and IHX start receiving a mixture of sodium cold stream from the core and hot stream prevailing in the hot pool. For a significant duration of time, IHX receive hot and cold streams. The tube bundle of IHX would then be subjected to hot sodium and cold sodium streams resulting in differential temperature profile among tubes. Knowledge of transient velocity and temperature distribution of sodium flow in IHX windows is essential for the estimation of thermal loading on IHX tube bundle and for FIV studies. Towards this, a transient three dimensional CFD analysis has been carried out to evaluate transient temperature and velocity evolution of sodium in hot pool. Variable time step in the range of 0.001 s to 0.1 s is considered for the simulation carried out for total duration of 600 s. The computational mesh size is 1.05 million and the time taken for 1 simulation in a 30 core CPU is 15 days. Though, the two IHX of a loop are not symmetrically placed in the hot pool, total flow rate through each is nearly same due to the large resistance offered by the tube bundle and anti-vibration belts compared to the resistance of the flow paths in hot pool. At the CP surface, the temperature contours shows a stratification interface above holes in the shell. Stratification interfaces are also observed on the IV surface at time t=15 s which moves upwards with time. The velocity of sodium entering IHX is maximum at the bottom and a maximum velocity of 0.82 m/s is observed in the inlet window of IHX-1. At the IHX inlet window, a ΔT of 5 K is observed in the sodium stream entering it initially and the same increases to as high as 39 K during the transient. Figure 7 shows the temperature contours of sodium in the plane passing through IHX at various instances.

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| Temp_Time_0_s | Temp_Time_10_s | Temp_Time_40_s | Temp_Time_80_s |
| Temp_Time_150_s | Temp_Time_200_s | Temp_Time_300_s | Temp_Time_600_s |
| *Fig. 7 Temperature contours (K) of sodium in the plane passing thorough IHX-1 at various instances.* | | | |

6. INTEGRATED TRANSIENT HOT POOL – COLD POOL STUDY

Hot and cold pools along with immersed components represent the primary heat transport system of a pool type fast reactor. Cold pool along with inner vessel is enveloped by Main Vessel (MV). MV and IV represent most important structural components of a Fast Reactor. The primary heat transfer loop including coolant (liquid sodium), core, heat exchangers, pumps etc., is enclosed within main vessel. Cold pool of reactor assembly houses several critical components viz., grid plate, core support structure, primary pipes etc. During normal operation, major load on main vessel and internals in cold pool is primarily mechanical in nature due to the relatively lower prevalent temperatures (about 670 K). Components in hot pool along with IV on the other hand face significant thermal stresses due to the prevailing higher temperatures (bulk mean temperature of hot pool is >800 K). Steady state flow and temperature fields in reactor pools have been determined and this would allow estimation of structural loads due to thermal hydraulic factors for reactor conditions prevalent during normal steady state operation. However, during reactor transients, both IV and MV along with immersed components are subjected to rapid temperature changes, making thermal loads important for both. Primary coolant, viz., liquid sodium exacerbates the problem of thermal loads on IV and MV and their internals due to its high thermal conductivity. In view of this, transient temperature evolution of reactor pool components during reactor SCRAM is determined using an integrated fully coupled model of reactor pool. Due to the complex flow physics to be resolved, both hot and cold pools along with immersed components are modelled and analysed in a coupled form. Development of a three dimensional model of the complete primary heat transport system of FBR 1&2 is a challenging task due to the widely different scales that need to be modelled. Further inclusion of internal structures like spherical headers, primary piping etc. complicates the task of mesh generation. During a transient like SCRAM, control rods are inserted into reactor core to stop fission heat generation. At the same time, primary sodium pumps coast down to 20% speed. Secondary sodium pumps also coast down to 20% speed. The above sequence of events leads to significant temperature changes at core outlet and subsequently at IHX outlet. Sodium entering hot pool, after exit from core top, influences components in hot pool that includes IV. IHX outlet temperatures affects both MV and IV along with other cold pool components. An additional influence arises out of the temperature changes in MV cooling system, affecting upper parts of IV and MV. Towards this, an integrated CFD model of hot and cold pools of FBR 1&2 with conjugate heat transfer model has been used to study the transient pool thermal hydraulic behaviour during reactor SCRAM. The computational mesh size is 2.6 million and the time taken for 1 simulation in a 30 core CPU is 28 days. Time step 0.01 s is considered for the simulation carried out for total duration of 600 s. Due to the capabilities of the present model, simultaneous evolution of flow and temperature in hot and cold pools has been analysed (Fig. 8). Temperatures of important structural components like inner vessel and main vessel have been determined. The influence of heat transfer though inner vessel during SCRAM on flow and temperatures of hot and cold pool has been characterised. Temperatures of all important immersed components viz. pump standpipe, IHX standpipe, Spherical header, primary piping etc. have been resolved. These results are important for future thermo-mechanical analysis of reactor assembly components.

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| 0 s | 20 s |
|  |  |
| 100 s | 600 s |
| *Fig. 8 Transient evolution of temperatures on symmetry plane* | | |

7. CONCLUSION

Detailed 3D high fidelity CFD simulations are carried out to investigate 3D thermal hydraulics of FBR1&2 systems and components towards validating the various design modification proposed for FBR1&2. Conclusion from each study. Towards reducing gas entrainment in the hot pool, a horizontal baffle of 500 mm width provided at 1.3 m below the free surface level, reduces the free surface velocity to 0.57 m/s and the percentage of free surface area where the velocity is above 0.5 m/s is only 2.9 %. The maximum resultant velocity predicted at the inlet window of IHX is 0.8 m/s. The predicted velocity profile is useful for FIV design IHX tubes. From the inlet temperature profiles, a maximum ΔT of 4 K is observed at the IHX inlet. The heat removal capacity of DHX-A of SGDHRS of FBR1&2 is estimated using 3D CFD simulation and is predicted to be 11.3 MW compared to design value of 10 MW. By a 3D transient CFD study of hot pool, the temperature contours predicted at the CP surface shows a stratification interface above holes in the shell. Stratification interfaces are also observed on the IV surface at time t=15 s which moves upwards with time. At the IHX inlet window, a ΔT of 5 K is observed in the sodium stream entering it initially and the same increases to as high as 39 K during the transient. By using an integrated CFD model of hot and cold pools of FBR 1&2, transient temperatures of important structural components like inner vessel and main vessel have been determined. The influence of heat transfer though inner vessel during SCRAM on flow and temperatures of hot and cold pool has been characterised. Temperatures of all important immersed components viz. PSP standpipe, IHX standpipe, Spherical header, primary piping etc. have been resolved.

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