Computational modelling of inter-wrapper flow and primary system temperature evolution in FBTR under extended Station Blackout

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Abstract. To handle a station black-out (SBO) event, sodium cooled fast reactors are equipped with passive systems to remove decay heat. Decay heat removal in FBTR, depends on natural convection driving flow through the core subassemblies and inter-wrapper gap(s). The relatively low decay heat during an extended SBO presents a risk of coolant freezing in the primary circuit inlet pipes. This impacts coolability of core, as flow through subassemblies would be impeded and major heat removal path would become ineffective. A two dimensional axi-symmetric computational fluid dynamics model of primary plenum is developed. Flow and temperature in reactor plenum during the transient is predicted and compared against available safety limits. These predicted temperatures are extended using a separate two dimensional model, to predict fuel and clad temperatures inside fuel subassemblies. Results reveal that fuel clad temperatures reach design safety limit(s) after two days after initiation of SBO. It is concluded that adequate time is available for deploying Emergency Diesel Generator(s) and initiation of double envelope cooling.
Key Words: fast reactor, extended station black out, computational fluid dynamics, decay heat.

1. Introduction

The concept of extended Station Black-Out came into picture after the Fukushima Daiichi nuclear disaster in Japan [1], wherein restoration of power required more than 7 days.

During a station black out event of limited duration, FBR's are designed with a Safety Grade Decay Heat Removal System (SGDHR) to ensure core coolability. This system is a completely passive system capable of removing decay heat without any external power supply. But the reactors around the world are now considering a situation in which the availability of power would be interrupted for more than a week and coolability of reactor core must be ensured. Fast Breeder Reactors using liquid sodium as coolant could face a risk of coolant freezing in the pipelines.Fast Breeder Test Reactor (FBTR) is a 40 MW thermal, loop type sodium cooled fast reactor operating at Kalpakkam, India. A schematic of FBTR is shown in FIG.1 [2]. FBTR has a relatively small core and as a result whenever the reactor is shutdown the resulting decay heat is small as well. The probability of sodium freezing in the primary inlet pipe in such a system is high.

Freezing of sodium in primary pipe would have an immediate adverse effect on ability of the decay heat removal system. Decay heat removal in FBTR, depends on natural convection driving flow through sodium circuit with the steam generator casing providing the heat sink. Any sodium freezing would cause an obstruction to this flow and steam generator would be cut off from the flow path. The buoyancy head available to sodium due to decay heat emanating from the reactor core would drive flow predominantly in the interwrapper space (FIG.2). Sodium would circulate within the hot plenum. Possible cases for sodium freezing in primary piping of FBTR are listed as follows:

- Line heaters that maintain sodium in liquid state are not available under SBO.
- Local under cooling of sodium piping in cold spots is possible.
- As the temperature of sodium goes down, impurities start building up at the cold spots.

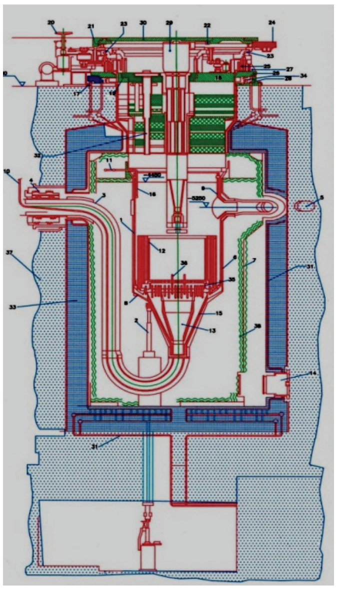

FIG.1.Schematic of FBTR

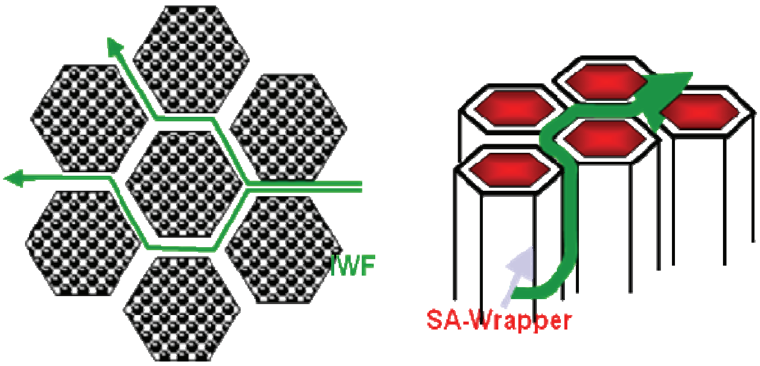

FIG. 2 Schematic showing Inter wrapper flow: top view on left and side view on the right

The freezing would take place at the coldest point on the primary loop that also would coincide with the point of lowest elevation on the loop. This would make remelting difficult as well. The aim of this study is to understand the consequences of sodium freezing in primary piping, on its core. The study deals with a scenario in which FBTR faces a station black out condition and the reactor is shutdown. After shutdown the decay heat will be dissipated to the atmosphere through natural convection in the primary coolant circuit. After 24 hrs of shutdown, due to sodium freezing the primary pipe loses its functionality leading to loss of continuity for sodium flow and suppression of natural convection driven flow through the primary loop. Under these conditions decay heat would raise the temperature of primary sodium present within the core. The inter-wrapper sodium would carry part of this heat to the hot pool (outside the core) through natural convection. This heat in turn would be lost to atmosphere due to heat losses through structural components. Temporal core temperature evolution under these conditions, are quantified, which would help in estimating time available for taking corrective action.

2. Methodology

The whole study is divided into three major parts. The first part involves prediction of temperatures of all major components of FBTR reactor pile. A one dimensional heat transfer model based on lumped body analysis is developed. Next, a two dimensional axi-symmetric CFD model of FBTR hot plenum along with reactor vessel is developed. This model takes boundary conditions from the one dimensional model. It is used to predict thermal hydraulic behaviour of sodium inside reactor vessel. Maximum pool temperatures are extracted and used to supply boundary conditions for the third study. In the third part a two dimensional symmetric finite volume based conduction model of fuel subassembly (SA) cross section is developed to estimate the peak fuel pin temperature.

2.1. Thermal analysis of FBTR block pile

First, one dimensional lumped heat transfer model of reactor pile is developed. A schematic of the model developed, is shown in FIG. 3. The components included in the model from left to right are reactor vessel (RV), double envelope (DE), safety vessel (SV), thermal insulation (TI), borated concrete (BC) and structural concrete (SC). The core is to the left side of reactor vessel. Heat flow through all these components is by conduction. Inter-component spaces are filled with nitrogen (N2). A temperature node is placed at the centroid of each component and heat balance equations are described with respect to this arrangement. All the three modes of heat transfer viz. conduction through nitrogen, convection at component surface(s) and radiation between component surfaces are modelled.

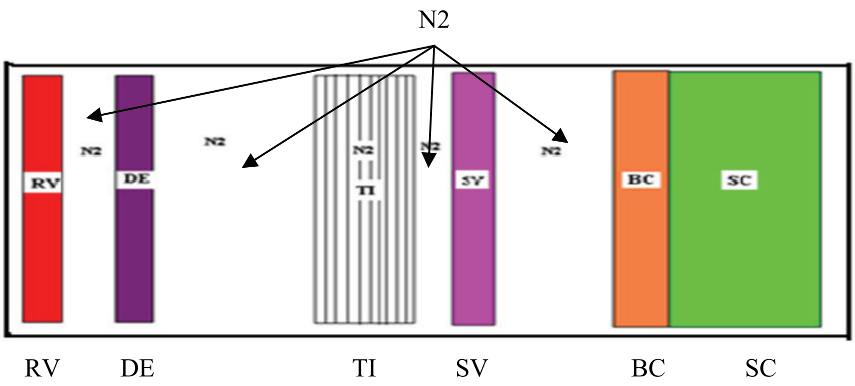

FIG. 3 Schematic showing one dimensional model of the reactor pile

The system gets heated due to decay heat generation in core. This decay heat is a function of time and is calculated from the following equation:

$$P = P_0\left(6.48\times10^{-3}\right)\left(t^{-0.2}-\left(t+T\right)^{-0.2}\right) \quad (1)$$

Where,
P is the power at time **t**, $\mathbf{P_0}$ is the operating power of reactor and **T** is the reactor operating time before shutdown. All time quantities are expressed in days.

There is heat generation in borated concrete due to irradiation by neutron and gamma radiation. This volumetric heat generation is a function of reactor power and is obtained from reactor physics calculations. The general form of heat balance equation implemented for each component participating in heat transfer can be represented as:

$$m\,C_p\,\frac{T_i^{\,n+1}-T_i^{\,n}}{\Delta t} = \frac{T_{i-1}^{\,n}-T_i^{\,n}}{\left(R_{i-1}^{\,i}\right)^n} - \frac{T_i^{\,n}-T_{i+1}^{\,n}}{\left(R_i^{\,i+1}\right)^n} \quad (2)$$

Where, **m** is mass of a component, $\mathbf{C_p}$ is the specific heat of its material, **T** is temperature and t is time, **R** represents thermal resistance between nodes represented by sub and superscripts, subscript/superscript **i-1, i, i+1** represent the downstream, native and upstream nodes, superscript **n, n+1** represent new and old time steps respectively.

The development of this model is done to fulfil two major requirements. The first is to estimate and track the temperatures of the several important components that are part of this model. The second is to generate boundary conditions for the more detailed CFD model(s) as representing all the components that are a part of the one dimensional model into the CFD model would be computationally too costly. The sequence of events during a typical run is described in section 3.

2.2. Axi-symmetric two dimensional CFD model of primary plenum

Next, a two dimensional axi-symmetric model of the primary plenum is developed to study the transient evolution of core and primary plenum temperatures during extended SBO. Since the transient runs into a few days and computational economy is given due importance. A three dimensional analysis proved to be infeasible. All major components in primary plenum are modelled appropriately. Components like reactor vessel, grid plate, control plug, neutron shields etc. are directly axi-symmetric and are represented in the model without any approximation. The core having hexagonal subassemblies is lumped row wise. Radial and axial distribution of decay heat generation in core is modelled accurately. Total decay heat is taken from equation 1. A cross section of the model developed is shown in FIG. 4.

The model shown in FIG 4 is meshed using quadratic elements. The aim of the study is to analyze the flow and temperature due to natural convection of sodium in the reactor primary plenum. Standard conservation equations of mass momentum and energy [3] are solved in a finite volume based framework. The pressure velocity coupling is effected using the PISO algorithm [4]. The temperature dependent buoyancy force is modelled using the Boussinesq approximation. Turbulence is modelled using the RNG - k-ε high Reynolds model [5]. This model does not assume a constant turbulent Prandtl number as the standard model does and recalculates it during every time step as a function of flow field. Apart from the fluid volumes filled with sodium there are several solid structures primarily made of stainless steel. To estimate temperature distribution in these structures the conjugate heat transfer model is invoked as well. The outer surface of the reactor vessel serves as the only heat sink of the model, where heat flux boundary condition is imposed with heat flux being calculated using the one dimensional model of section 2.1. The core region holding fuel and blanket subassemblies including the surrounding shielding subassemblies is modelled as porous zones.

The core (including fuel region) that holds distinct hexagonal subassemblies is modelled as a continuous porous medium with pressure drop in the radial and axial directions being supplied using a momentum sink. The correlations taken from [6] are presented below:

For Radial (x) direction:

$Eu_x = 1034\ Re_x^{-0.88}$ $(Re_x < 2000)$ (3)

$Eu_x = 12.82\ Re_x^{-0.2232}$ $(Re_x > 2000)$ (4)

Pressure drop per unit length in x direction (dP/dL) is given by:

$$S_x = \frac{Eu_x \rho |u|}{2\delta\varphi^2} \quad (5)$$

Where,

Eu is Eulers number, **Re** is Reynolds number, **ρ** is fluid density, **δ** is subassembly pitch, **φ** is area porosity in radial direction and **u** is superficial velocity along radial direction.

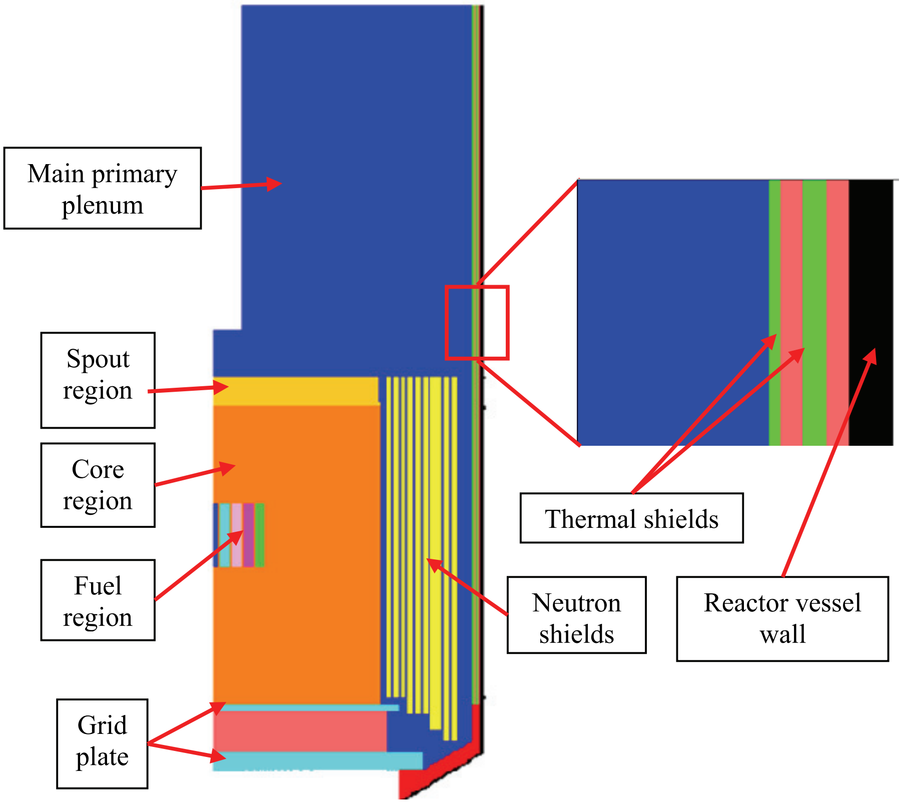

FIG.4. Schematic of Two Dimensional Axi-Symmetric model of reactor pool

For Axial (y) direction:

Friction factor, $f=96/Re_y$ $(Re_y<5000)$ (6)

$f=0.0766\ Re_y^{-0.0985}$ $(Re_y>5000)$ (7)

Pressure drop per unit length in y direction (dP/dL) is given by:

$$S_y = \frac{f\rho|v|}{2\varphi^2 d} \qquad (8)$$

Where,

d is hydraulic diameter for flow in axial direction, **v** is superficial velocity along axial direction

The momentum sink term(s) in the conservation equations for x and y momentum would appear as **$-S_x$ u** and **$-S_y$ v** respectively. The top exit of core is modelled as a series of blocked and unblocked flow areas. Sodium can flow only through the unblocked areas which have been modelled to be equal (row wise) to actual flow area available for sodium flow. This is done in order to account for entry and exit losses. Apart from effect on flow, fuel and structural material would affect the effective heat capacity of the core region. An effective heat capacity for fuel region and the rest of core is calculated and is imposed into the model using a heat sink term in the energy equation. Similarly heat generation taking place in the fuel region due to decay heat is also modelled using a heat source term in the energy equation.

2.3. Thermal analysis of hexagonal subassembly

A two dimensional conduction model of a single fuel subassembly was generated. The model is a two dimensional periodic model. Periodicity is in the axial direction. The aim of this study is to use the pool temperature predictions obtained in the previous study and estimate the maximum clad and fuel temperatures inside the subassembly. This is required because many of the safety limits are based on the peak fuel and clad temperatures. A 60° sector is modelled since the subassembly is hexagonal. FBTR uses 61 pin subassemblies. The model developed has fuel pins, stainless steel clad, inter clad sodium volumes and subassembly wrapper. The sodium is assumed to be static. During the transient, the primary coolant flow path is blocked. The grid plate would not receive any sodium from the primary pipe. However, there is a possibility of sodium entering the grid plate through the peripheral subassemblies and exiting through the central subassemblies. This phenomenon is beyond the scope of this present work as this would require a full three dimensional study. A static sodium assumption would ensure conservatism in the results. The contact between fuel pin and clad is taken to be perfect and gap resistance is not considered in the model. The decay heat generation, 24 hrs after shutdown would become about 0.3% of full power. The linear heat rating of each pin would reduce with the same ratio. Based on this data, the temperature difference across the fuel-clad gap is estimated to be < 3°C. Hence, in the detailed subassembly model, gap resistance is not modelled to minimize the computational time. The maximum temperature obtained in the fuel region of primary plenum model is imposed as constant temperature boundary condition on the outer surface of the hexagonal wrapper. The fuel pin with the maximum linear heat rating is chosen for the study. Heat generation in the form of source term is imposed in the energy equation to match with decay heat based on this linear heat rating. It may be pointed out that this treatment makes the study further conservative and the calculated values would envelope actual temperatures inside subassembly. The model developed for this study along with the mesh employed in shown in FIG 5.

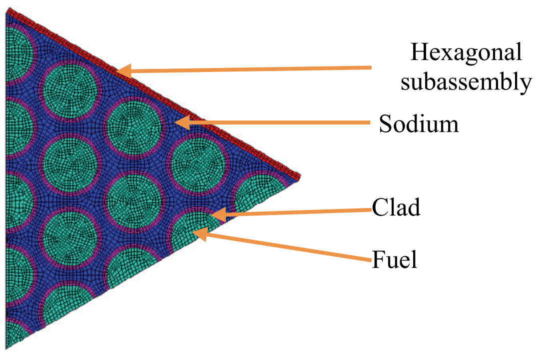

FIG.5. Meshed model of 60^0 section of cross sectional view of a hexagonal subassembly

3.0. Results and Discussion

3.1. Thermal analysis of block pile

As described in section 2.1, the lumped body model of reactor block pile is to predict the transient temperatures of various reactor structures during the extended SBO. However, before starting the transient a steady state solution is to be obtained which would serve as the initial condition. The primary flow is assumed to get choked 24 hrs after reactor SCRAM. This assumption is based on results from [7]. After 24 hrs, the pool is expected to be at an average temperature of 200°C maintained by the SG casing heat removal. The steady state temperature distribution in various components is determined for this pool temperature imposed on the inner surface of reactor vessel. Starting from this initial state the evolution of temperatures of various components is presented in FIGS. 6 and 7. The transient is simulated for 170 hrs. The average temperature of reactor pool reaches 1105 K after 170hrs. Safety vessel attained a maximum temperature of 537 K after 170hrs. The temperature of different sections of borated concrete and structural concrete are plotted in FIG.7. Structural concrete attained a maximum of 329 K and borated concrete attained a maximum of 445 K. It is also seen that the temperature change for the outer components takes a considerable time due to the high thermal inertia of the system.

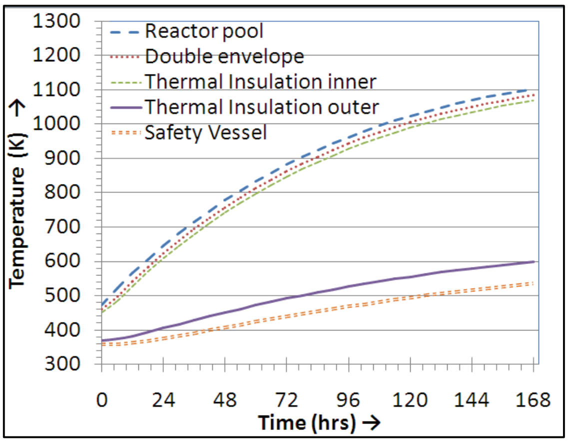

FIG.6. Temperature evolution in reactor vessel, double envelope, thermal insulation and safety vessel

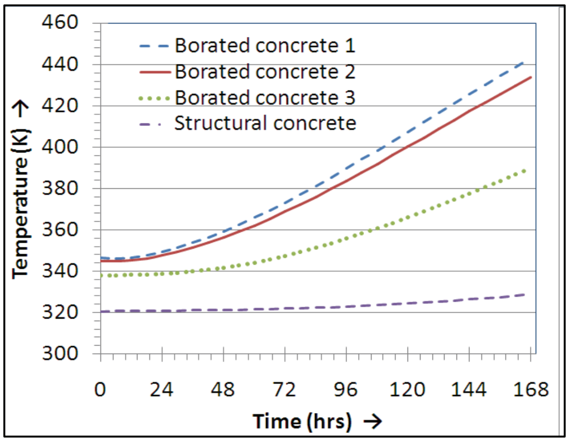

FIG.7. Temperature plot for Borated concrete sections and Structural concrete

3.2. Axi-symmetric two dimensional CFD model of primary plenum

The results obtained from pool thermal hydraulic analysis of reactor pool are presented using temperature contour plots and velocity vector plots. FIG. 8 shows the evolution of temperature within the reactor pool at different times ranging from 600 seconds to 3 days.

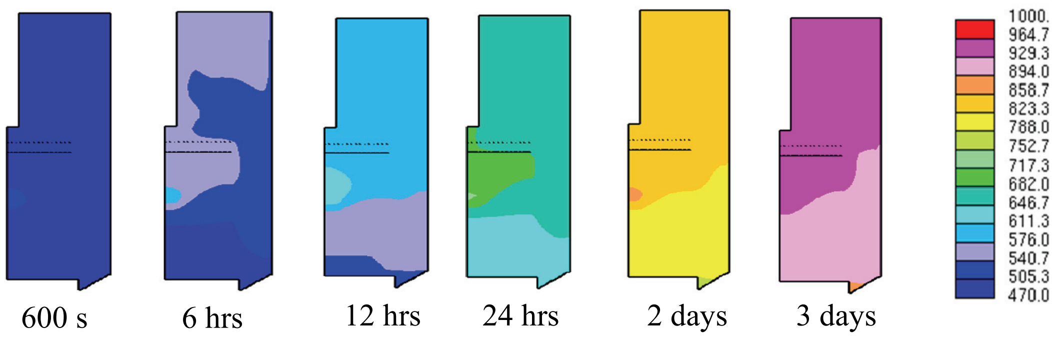

FIG.8. Evolution of temperature (K) of primary plenum during after sodium freezing in primary

The initial condition of this study was an isothermal primary plenum temperature of 200°C. This initial state is chosen based on the premise that, after shutdown hot pool temperature is brought down to and maintained at 200°C. Twenty four hours after shutdown most of the pool would be at this temperature except local spots of higher temperature in the fuel region due to decay heat release. To resolve these temperatures, we would require a more explicit model including IHX and primary piping which would increase computational cost enormously. However, owing to the small heat capacity of this zone and the focus of the study being at times far-off from the initial state, this assumption would be appropriate. After the primary inlet gets blocked, the decay heat starts heating up the sodium present in primary plenum itself. Sodium gets heated inside the fuel region of reactor core and would gain a positive buoyancy force. The sodium present inside the subassemblies would be unable to rise and exit the subassembly due to the bottom region being blocked. However, the inter-wrapper sodium is free to flow and would rise up. As it rises up, cooler sodium from the rest of the plenum would replace this sodium. This would set up a convection driven plume of sodium which would carry heat from the core to the rest of primary plenum and lose some heat to reactor structures through the reactor vessel. A major part of the heat would lead to an increase in the average temperature of primary plenum sodium. Natural convection controlled systems are quite unstable. The amount of heating in reactor core is low and it further decreases with passage of time. The buoyancy head available is quite low as a result. The plume formation described above is not a stable construct and the same gains and loses strength contiguously as the transient progresses in time. This trend can be seen clearly in FIG. 9.

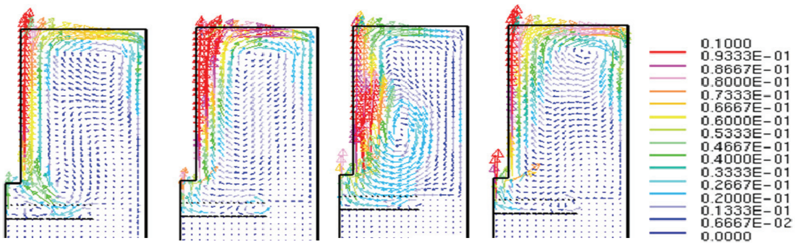

FIG.9.Evolution of velocity vectors (m/s) in primary plenum

The average and maximum temperatures of various components is plotted in FIGS. 10 and 11.

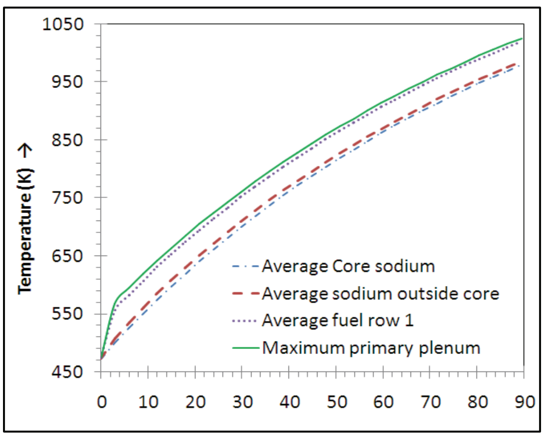

FIG.10 Evolution of core temperatures

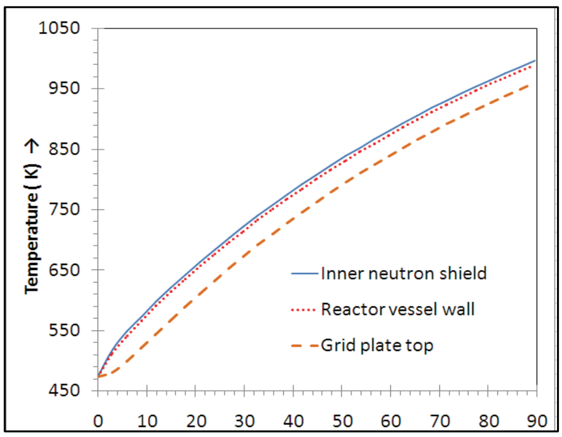

FIG.11. Evolution of structural temperatures

From FIG. 10 it can be seen that the maximum temperature in the pool is 1025 K after 89.5 hrs. The maximum plenum temperature is observed in the innermost row fuel region. The average temperature of this row follows the maximum temperature quite closely.

3.3. Thermal analysis of hexagonal subassembly

After estimation of maximum sodium temperature in the pool, the temperatures inside subassembly are estimated. The pool sodium temperature is imposed as a boundary condition on the outer surface of hexagonal wrapper. The evolution of internal temperatures is shown in temperature contours of FIG. 12.

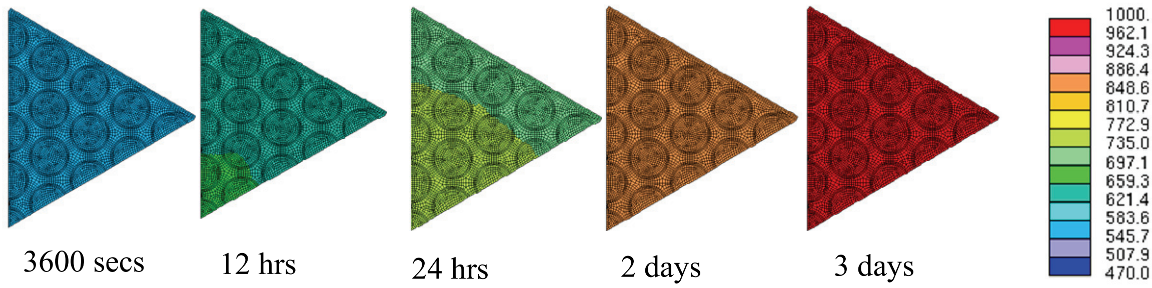

FIG.12. Evolution of fuel and clad temperatures (K) with time

The maximum temperature inside subassembly would stay above the maximum pool temperatures calculated in section 3.2. The maximum pool temperature predicted from the section 3.2 of the study and average pool temperature from the section 3.1 is used to calculate a peak factor 1 (ratio of maximum pool temperature and average pool temperature).

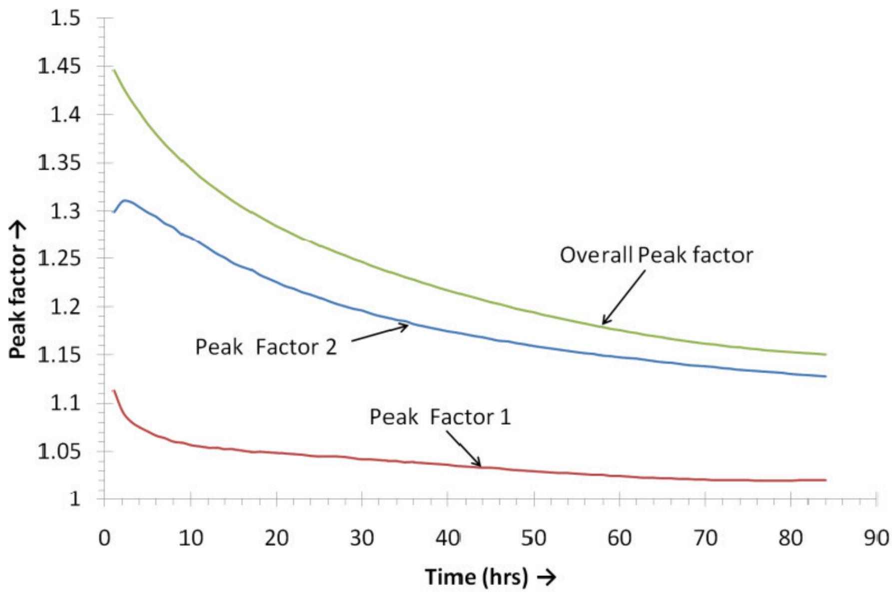

FIG.13. Evolution of peak factors with time

Similarly, the maximum fuel temperature predicted in the section 3.3 of the study and maximum pool temperature predicted from the section 3.2 of the study is used to calculate the peak factor 2 (ratio of maximum pool temperature to the maximum fuel temperature). The multiplication of both the factors gives the overall peak factor. These peak factors can be used to get a reasonable estimation of maximum clad temperatures using average temperatures calculated from the one dimensional model. FIG 13 shows the variation of peak factors over the time period. The average values of peak factor 1, peak factor 2 and overall peak factor are 1.18, 1.03 and 1.233 respectively.

4.0. Conclusion

An integrated analysis of extended SBO in FBTR has been carried out. First a one dimensional model of hot pool and its surrounding structures is developed and pool temperatures under failure of decay heat removal system are estimated as a function of time. The results reveal that hot pool average temperature value after 50hrs is 787K and 873K after 70hrs. This study also provided the heat loss from the outer surface of reactor vessel. This becomes the boundary condition for the subsequent two dimensional axi-symmetric porous body CFD study, which allows estimation of peak temperatures in hot pool. Hot pool temperature crosses 873 K after around 50hrs, 898K after 56hrs and 923K after 61hrs. These temperatures represent the design safety limits for pool temperatures in FBRs. Finally with the help of these estimated hot pool temperatures the peak fuel pin temperature is estimated using a two dimensional conduction model. The maximum temperature of central fuel pin reached a maximum of 873K after 47 hrs, 898K after 52 hrs and 923K after 58 hrs. The design safety limits will be reached after 2 days of sodium freezing which decides the time available to deploy diesel generators.

5.0. Acknowledgements

The neutron and gamma heating in Borated concrete estimated by Dr.D.Sunil Kumar and Dr.G.Pandikumar, Reactor shielding and data division, IGCAR is duly acknowledged.

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