International Atomic Energy Agency's Coordinated Research Project

On

Development of Small Reactors Without On-Site Refuelling

IAEA Research Contract Number: 13090/Regular Budget Fund (RBF)

Title of the Research Project: A study of enabling technologies for an Indian reactor for non-grid based electricity applications

Institute where research was carried out: Bhabha Atomic Research Centre, Mumbai, Pin code: 400 085, India

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Progress Report on Third Year's Work (December 2006 - November 2007)

(Third Year's Progress Report on IAEA CRP on Development of Small Reactors Without On-site Refuelling)

Design of a multipurpose nuclear power pack for satisfying energy related needs for remote Indian villages

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1.0 Introduction

The project aims to explore the potential of small reactors without on-site refuelling to supply electricity to remote villages, islands, and small-scattered communities in India, which are located in regions not connected to electricity grid. The aim of the project is to arrive at a feasible design of small portable nuclear power packs, having long core life, passive safety and reactor core heat removal features, and not needing skilled man-power for operation. To satisfy the energy related needs of the regions mentioned above the unit size of the reactor was estimated to be 5 MWth [1]. A reactor configuration consisting of thorium fuel cycle based metallic fuel, BeO moderator, BeO and graphite reflector, and molten lead alloy based coolant was selected for study [1]. The proposed specification of the reactor [1] is shown in Table-1.

Table-1: Broad specification of the reactor

Attributes	Property
Reactor power	5 MW(th)
Core life	Around 10 years
Fuel	Metallic ^{233}U+ ^{232}Th +Zr
Fuel clad	Zircalloy
Moderator	BeO
Reflector material	BeO and graphite
Coolant	Pb-Bi eutectic alloy
Core height	1000 mm
Core inlet temperature	450 ^{0}C
Core outlet temperature	600 ^{0}C
No. of fuel assemblies	30
No. of control locations	31
No. of fuel pins per assembly	12
Fuel pin ID	8 mm
Fuel pin OD	10 mm
Pitch	140 mm
Top reflector height	150 mm
Bottom reflector height	150 mm
Coolant tube OD	45 mm

2.0 Description of reactor geometry

The core of the reactor is based on tri-angular lattice arrangement [1,2]. It contains 30 fuel assemblies, 12 in the inner ring and 18 in the outer ring. The cross-sectional view of the reactor core is shown in the Figure-1. In each fuel assembly, fuel pins are located in graphite fuel tubes, which also act as coolant tubes. These fuel tubes are located in moderator blocks. These moderator blocks are in turn surrounded by BeO reflector blocks, which also contain reactor control devices. There are 7 graphite blocks inside the fuel assemblies and 24 BeO reflector blocks outside the fuel assemblies. Graphite reflector blocks surround these BeO reflector blocks. Core height is 100 cm with additional 15-cm top reflector and 15-cm bottom reflector.

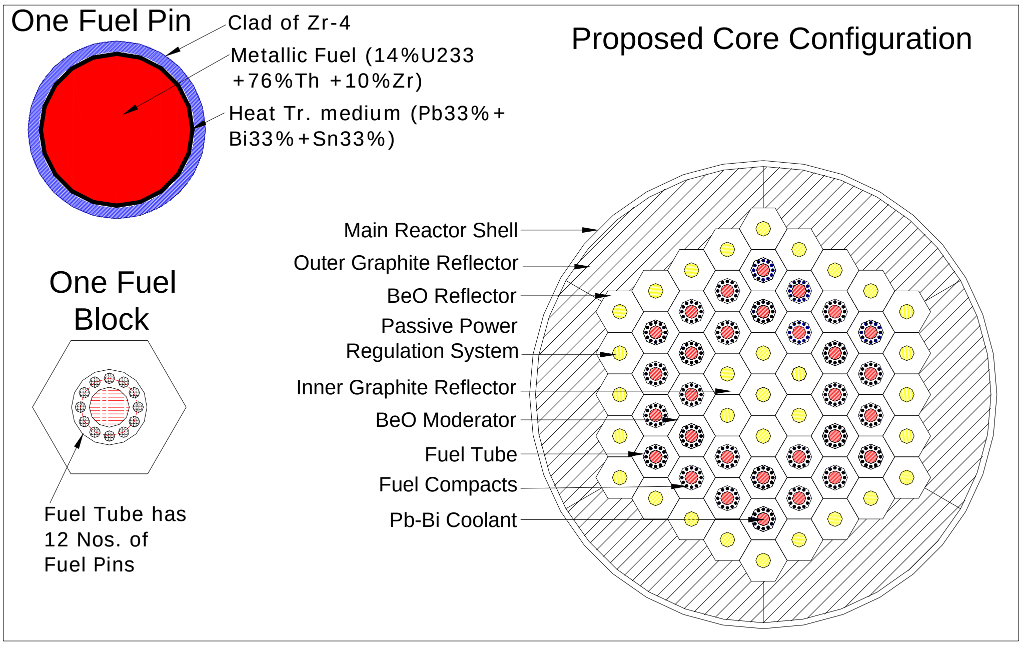


Figure-1: Configuration for fuel pin, fuel block and core

3.0 Reactor physics design

Cross-sectional view of a fuel pin and the core configuration is shown in Figure-1. Earlier [2] the reactor physics design of the proposed configuration was carried out. Following paragraphs briefly features of the design:

a) A lattice pitch of 14 cm has been found to be adequate.
b) Content of 20% U^{233} in the alloy gives a core life of 15 years. However, cold to hot swing in reactivity becomes positive with the introduction of Gd to control the extra reactivity.
c) With 14% U^{233} available burn up is 3000 FPDs.
d) Initial K-eff is very large necessitating the introduction of burnable poison in the core.
e) 300 gm Gd has been mixed in each of 12 inner fuel assemblies.
f) To maintain K-eff=1 at hot operating condition, all control rods should be 39.5 cm inside the core. In this situation maximum worth of a control rod is 2.9 mk.
g) Total U^{233} inventory in the whole core is 28.88 Kg.
h) Fuel Temperature coefficient at 775^0C is negative [-1.695×10^{-5} per ºC], which is satisfactory.

4.0 Analysis of inadvertent control rod withdrawal incident

The preliminary point kinetics calculations were done to analyse the inadvertent withdrawal of maximum worth control rod. The transient was analysed with temperature feedbacks and the variation of power and temperatures were studied. The computations were done for two cases: with addition of 2.9 mk of reactivity in 5.0 sec and in 60.0 sec so as to simulate both subsequent paragraphs.

4.1 The model used in these analyses is the point reactor model with temperature feedback. The transient is analyzed in case of an externally added reactivity due to the inadvertent withdrawal of control rod of maximum worth at criticality and the variation of power and temperatures with time are studied with a computer code MRIF [3]. In the code, the fundamental point kinetics equations with six delayed neutron precursors are solved with temperature feedback.

4.2 Point kinetics setup with temperature feedback

The fundamental equations for reactor kinetics for the point reactor model with delayed neutron precursors, is given as follows:

$$\frac{dn}{dt} = \frac{k\left(\rho(t) - \beta\right)n}{l} + \sum_i \lambda_i C_i$$
$$\frac{dC_i}{dt} = k\frac{\beta_i\, n}{l} - \lambda_i C_i \quad , i = 1...NDG \tag{1}$$

If temperature feedback is considered, the above equations have to be solved with the following heat transfer equations:

$$m_f c_f \frac{dT_f}{dt} = \gamma n - h(T_f - T_c)$$
$$m_c c_c \frac{dT_c}{dt} = h(T_f - T_c) - 2 w_c c_c (T_c - T_i) \tag{2}$$

Here, the subscripts f and c denote the fuel and the coolant, *m* and *c* are mass and the specific heat. T_f, T_c and T_i are the average fuel, average coolant and inlet coolant temperatures. w_c is the coolant mass-flow and h, γ are heat transfer and power normalization coefficients.

4.3 Basic data used for the analysis

Fuel properties assumed	
Mass of U^{233}	28.88 kg
Mass of Th^{232}	156.78 kg
Mass of Graphite	169.62 kg
Mass of Zirconium	20.61 kg
Mass of Gadolinium	3.6 kg in 12 inner FAs
Thermal properties assumed	
At 500 °C, Specific Heat of U^{233}	145 J °K^{-1} kg^{-1}
At 500 °C, Specific Heat of Th^{232}	176 J °K^{-1} kg^{-1}
At 500 °C, Specific Heat of Graphite	1630 J °K^{-1} kg^{-1}
At 500 °C, Specific Heat of Zirconium	340 J °K^{-1} kg^{-1}

Coolant properties assumed						
Mass of coolant in one coolant tube	16.826 kg					
Total coolant Mass (6.35×30)	504.78 kg					
Specific Heat of Pb-Bi eutectic	146.5 J °K^{-1} kg^{-1}					
Temperatures						
Average fuel temperature, T_f	575 °C					
Average coolant temperature, T_c	525 °C					
Inlet coolant temperature, T_i	450 °C					
Outlet coolant temperature, T_o	600 °C					
Temperature Coefficients α						
Fuel temperature coefficient	-1.695×10^{-5} °C^{-1}					
Coolant temperature coefficient	-1.0×10^{-6} °C^{-1}					
External Reactivity						
External reactivity added	2.9 mk					
Power						
Operating power	5 MW_{th}					
Point Kinetics Parameters						
Neutron Mean Life Time (l)	0.46571×10^{-4} sec					
Delayed Neutron Fraction (β)	0.00476					
β_i / β	0.06723	0.18908	0.13866	0.28361	0.11344	0.20588
λ_i	0.01589	0.03992	0 .14111	0 .35310	0.95636	2.86020

4.4. Numerical Results

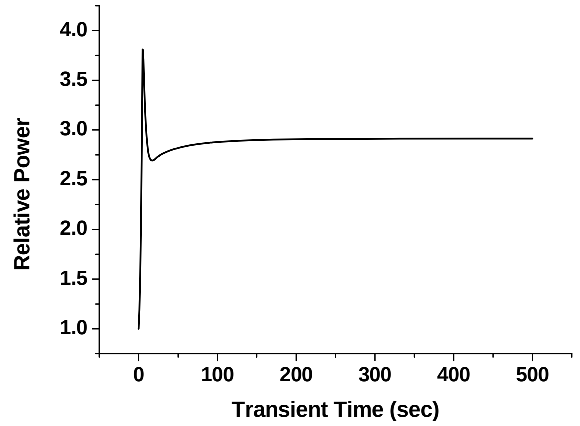
In the earlier analysis [4,5], it was found that when all 30 control rods are grouped together into one bank, the maximum reactivity insertion due to inadvertent withdrawal of a control rod at hot operating condition was of the order of 2.9 mk at criticality. The values of fuel temperature coefficient as calculated by the TRIHTR [6] code is as given in Table-1. The calculations of kinetic parameters have been performed with the computer code CLUB [7-8] using 172-group IAEA library obtained from the ENDF/B-VI point data. In this code, the multigroup integral transport equation is solved by the combination of interface current formalism and collision probability method. The present analysis has been carried out using 3 terms in the expansion of angular flux at region interfaces.

A small program was written to calculate various coefficients required as input to MRIF. The fuel (U^{233}+Th^{232}+Zr) and graphite fuel tubes were considered for the analysis along with the coolant. The transient time was taken as 500 seconds with appropriate time-steps. The results are given in the form of three types of plots: power, reactivity and temperatures variation against transient time (Figures-2 & 3). In all the plots, the temperatures are given in °C. The analysis were done for the following two cases:

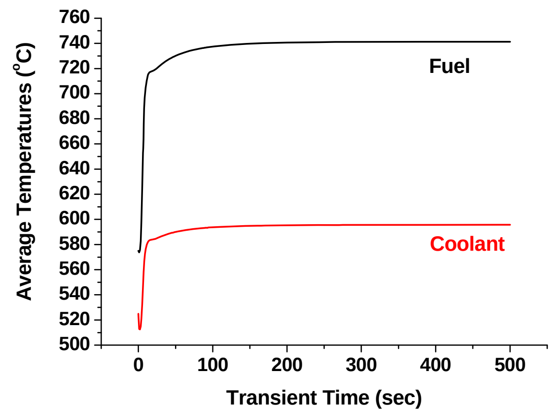
[i] 2.9 mk +ve reactivity is added in 5 seconds,

[ii] 2.9 mk +ve reactivity is added in 60 seconds.

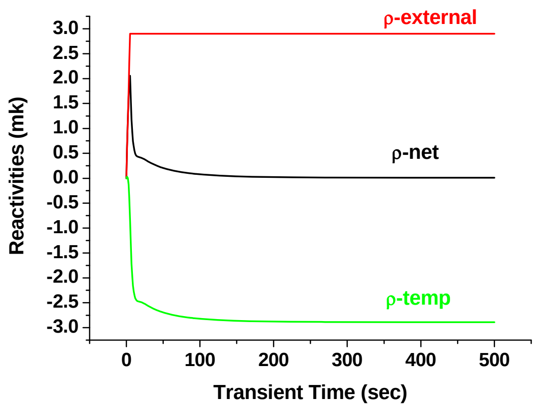
It can be seen from the Figure-2 for first case, when an external reactivity of 2.9 mk is added in 5 seconds, power surges to about 4.5 times the initial power for a short time and quickly subsides due to negative reactivity inserted by temperature feedback (Figure-2A). The average fuel and coolant temperatures rise to about 742 °C and 596 °C adding a negative reactivity (Figure-2B). As a result the power reduces and stabilizes at about 3 times the initial power. The reactivity variation with transient time is given in Figure-2C.



Figure–2A Power variation with transient time-Addition of 2.9 mk reactivity in 5 sec

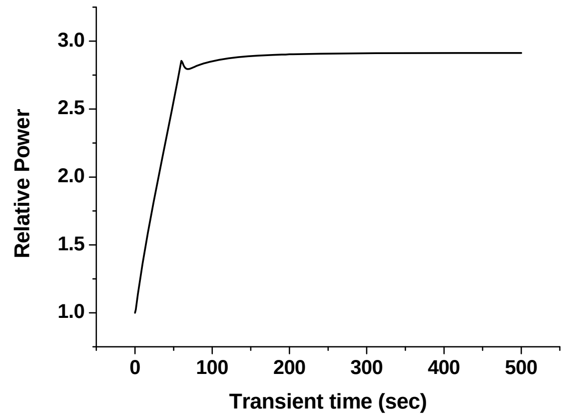


Figure–2B Temperature variation with transient time-Addition of 2.9 mk reactivity in 5 sec

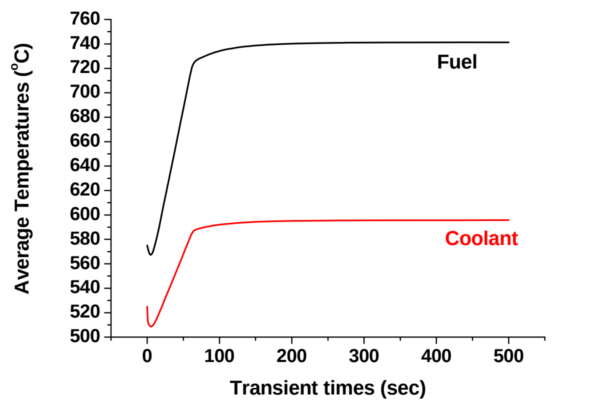


Figure–2C Reactivity variation with transient time-Addition of 2.9 mk reactivity in 5 sec

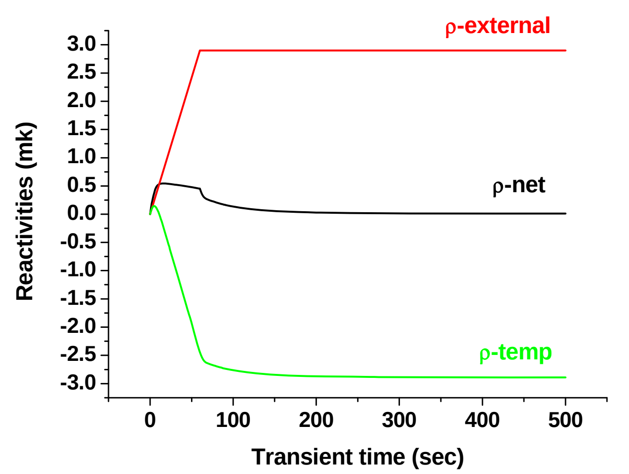
In the second case, where an external reactivity of 2.9 mk is added in 60 seconds, power surges to about 3 times the initial power and stabilizes at the value (Figure-3A). The average fuel and coolant temperatures rise to about 742 °C and 596 °C as in the previous case (Figure-3B). The reactivity variation with transient time is given in Figure-3C.



Figure–3A Power variation with transient time-Addition of 2.9 mk reactivity in 60 sec



Figure–3B Temperature variation with transient time-Addition of 2.9 mk reactivity in 60 sec



Figure–3C Reactivity variation with transient time-Addition of 2.9 mk reactivity in 60 sec

4.5 Summary of the results

The results of the point reactor model analysis for an inadvertent withdrawal of a control rod were carried out. An external reactivity of 2.9 mk, which is equal to the maximum worth a control rod in critical configuration, was added in 5 seconds (fast transient) and in 60 seconds (slow transient). The rise in power due to addition of reactivity increases the temperature of fuel as well as fuel tube. In both the transient cases, the power rises and stabilizes to about 3 times the initial power. More importantly, the averaged fuel temperature rises to about 742 °C. It may be mentioned that these results were obtained with a very basic point reactor model with temperature feedbacks. It would be worthy to analyse the given transient of inadvertent withdrawal of maximum worth control rod with more accurate 3-D models with detailed thermal-hydraulics feedback.

5.0 Modified thermal hydraulic design of the reactor

Thermal-hydraulic design of the reactor was modified so as to have a system of insulated down comers for return of the cold coolant from the upper plenum of the reactor to the lower plenum. This was needed to provide the necessary driving head for natural circulation. Earlier design [2] involving 60 nos. of downcomer tubes of required dimension could not be accommodated in the compact core. Computer code developed earlier for carrying out thermal hydraulic analysis was used for carrying out design modifications.

A system based on 36 downcomer tubes was found to be the most appropriate and the thermal-hydraulic analysis was carried out to find the required chimney height. The required chimney height was found as 850 mm instead of 250 mm as analysed earlier [2]. The average mass flow rate and average velocity of the coolant in the coolant tube remains same as found [2] earlier, i.e. 227 kg/s and 47 cm/s respectively. A simplified geometry of the modified loop and results of the analysis are shown in Figures-4 and 5 respectively.

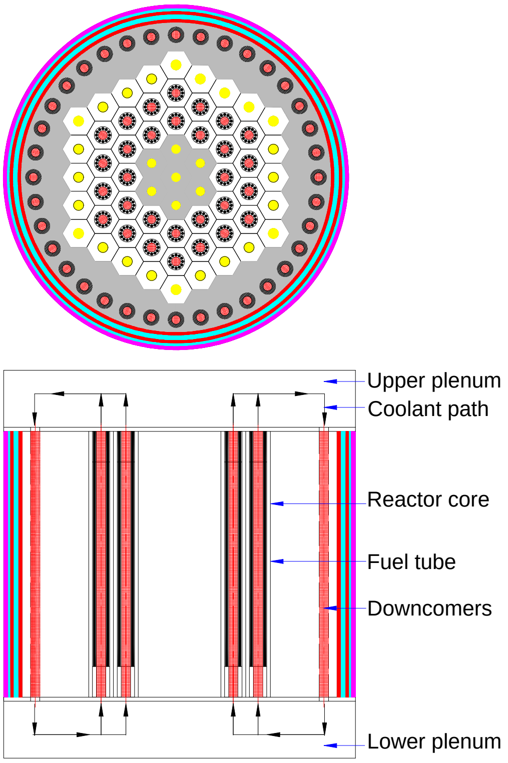


Figure-4: Simplified geometry of the coolant circulation loop

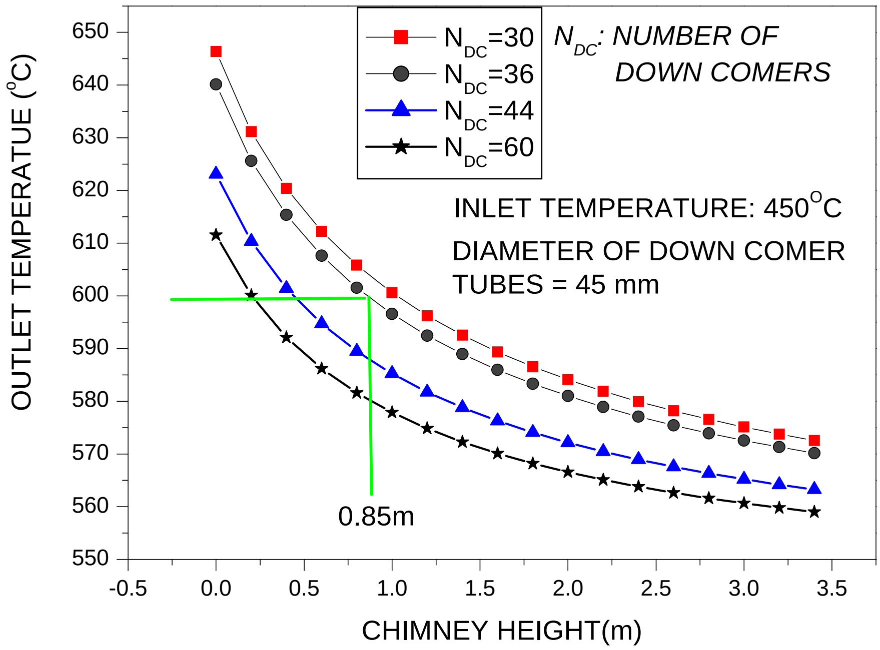


Figure-5: Variation of outlet temperature with chimney height for varying downcomer tube numbers

6.0 Thermal analysis under normal operating conditions

A steady state analysis of the reactor was carried out using finite element method considering conduction heat transfer mode in order to determine the prevalent temperatures in the various components of the reactor. Utilizing the 30° r-θ symmetry of the reactor core, a 1/12th three-dimensional sector of the core and reflector was modelled along with shell for this analysis as shown in Figure-6.

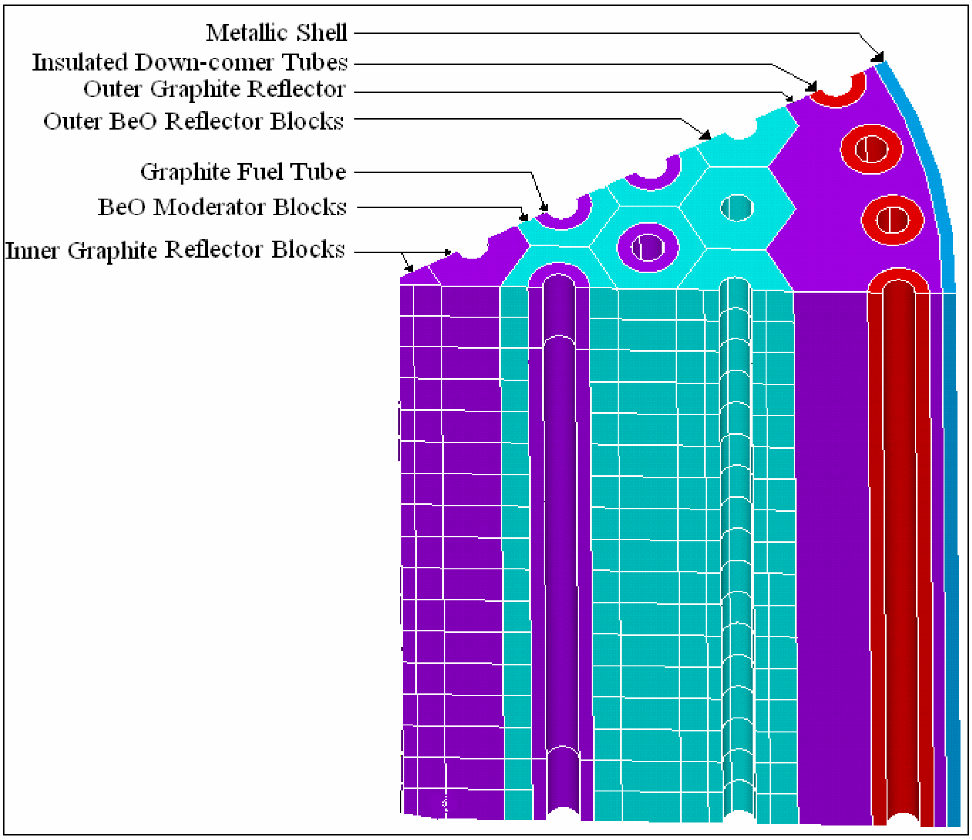


Figure-6: 30° Sector used for the analysis

A 20-noded element was used. The thermal contact resistance, between the graphite fuel tubes and beryllium oxide blocks and between beryllium oxide and graphite blocks, was assumed to be negligible. A uniform volumetric heat generation load was applied in each of the fuel tube to simulate the effect of nuclear heat production. An additional 250 kW of volumetric heat generation was assumed as total moderator heat generation rate. Appropriate convection heat transfer coefficients were applied to the inner diameter of the fuel tubes. The coolant temperature in the fuel tube was assumed to be 450°C at inlet and 600°C at outlet. All other surfaces were assumed to be adiabatic. The temperature contours so obtained are shown in Figure-7. The maximum temperature seen by the fuel during normal operating conditions is about 1010ºC.

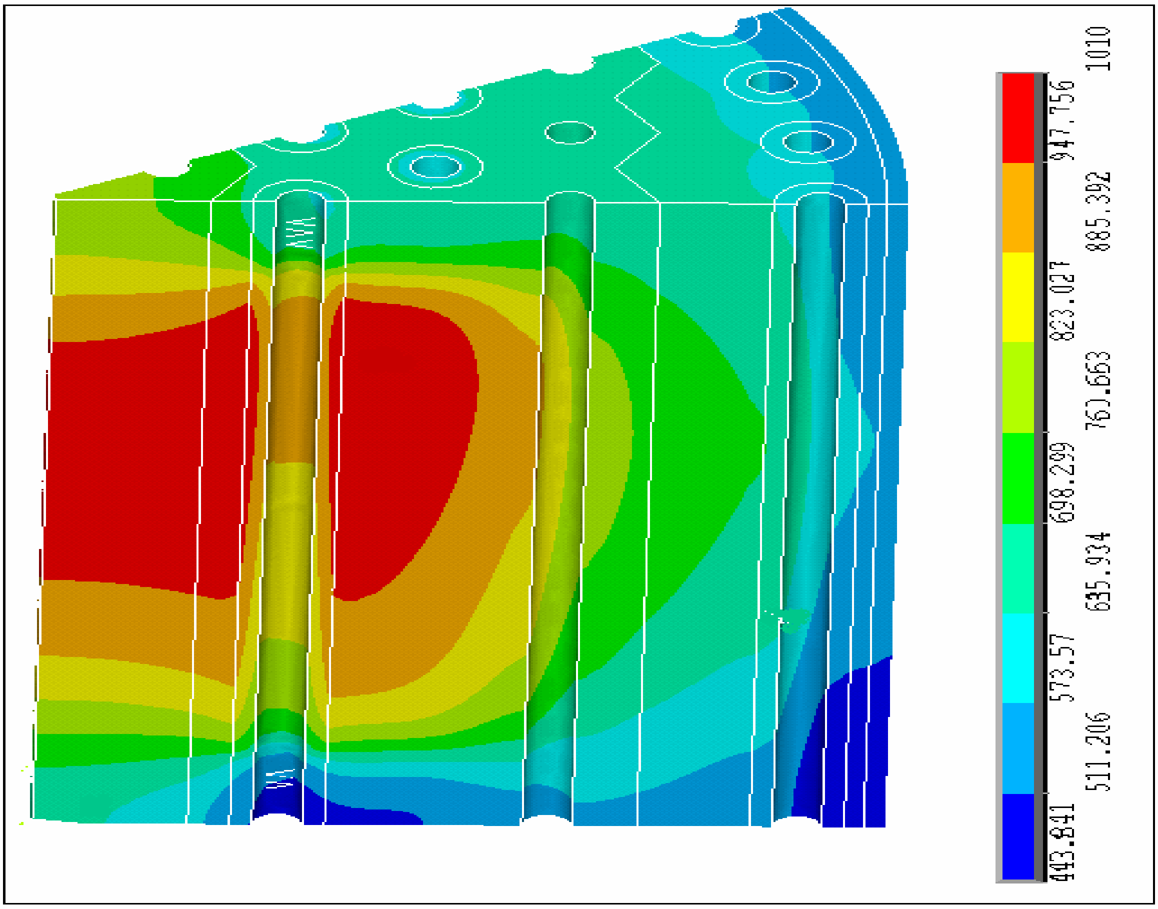


Figure-7: Temperature contours (in ºC) calculated under steady state conditions

7.0 Stress analysis of the core components

7.1 Stress analysis of graphite fuel tube

The temperature distribution obtained in the three dimensional thermal analysis was used to estimate the stresses expected in the fuel tube, under normal operating conditions, primarily arising from thermal gradients. It is seen from the analysis that the maximum principal stress in the fuel tube comes out to be around 4.2 MPa, which is lower than the allowable stress in the case of graphite. Figure-8 shows the stress contours for the fuel tube.

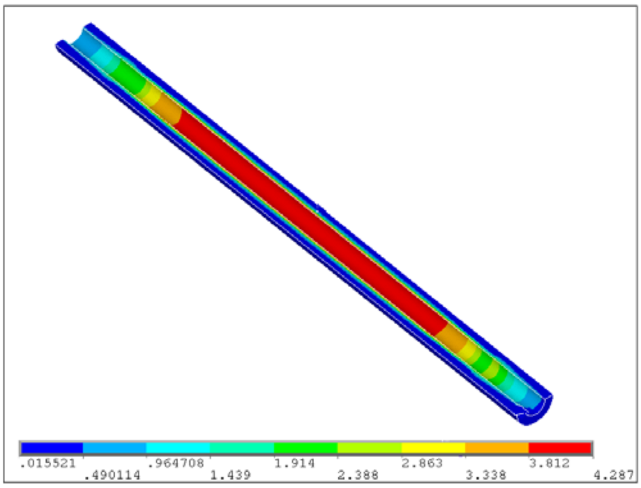


Figure-8: Principal stress contour for the fuel tube

7.2 Stress analysis of BeO moderator blocks

From the thermal analysis it is evident that the outer ring moderator block located just above the centreline of the core is subjected to the maximum thermal gradients. Initially the individual block thickness was kept at 77 mm. The stresses were calculated and found higher. Hence, the moderator block thickness was reduced to 50 mm, whereupon the first principal stress reduced to acceptable values. The stress contours are shown in Figure-9.

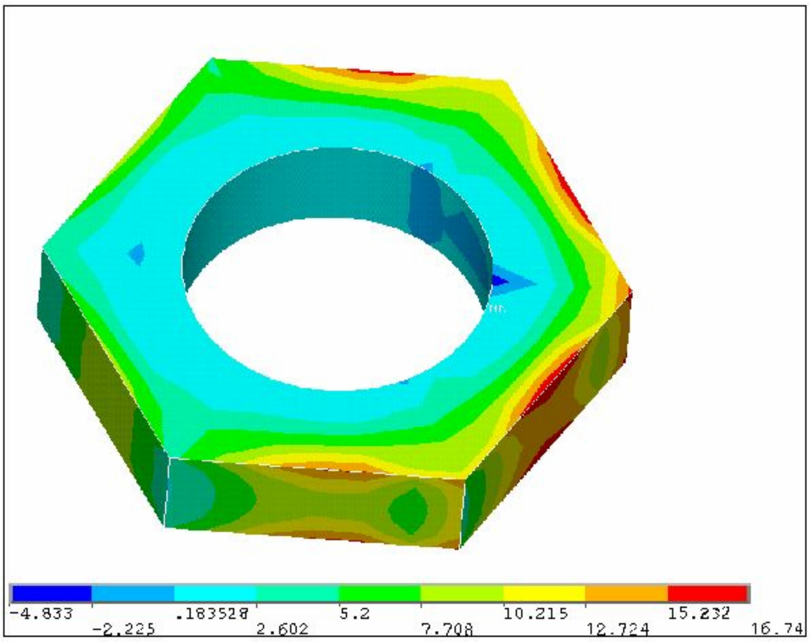


Figure-9: Principal stress contour for BeO moderator block

7.3 Stress analysis of graphite reflector blocks

Initially outer graphite reflector as a single piece was considered for analysis. Full reflector block of about 1300 mm height as a single piece was considered for analysis. The maximum principal stress obtained were higher than allowable. In the current configuration the graphite reflector blocks are segmented into 6 numbers of equal (angle 60 °) circular sectors. The first principal stresses are much below the allowable limit. Stress contours are shown in Figure-10.

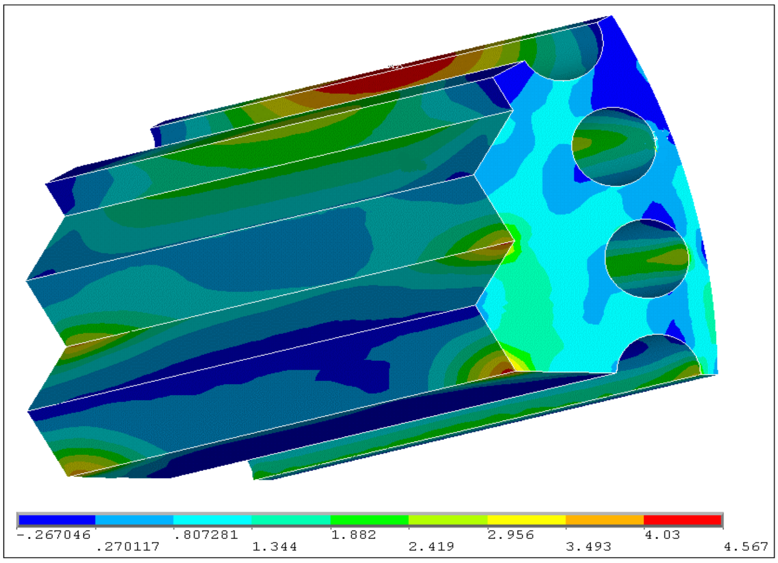


Figure-10: Principal stress contour for graphite reflector block

7.4 Stress analysis of BeO reflector blocks

The block considered for analysis was selected based on the maximum temperature gradient, as obtained from the thermal analysis. The block height obtained from analysis of the moderator block was 50 mm. For the analysis the height of the reflector blocks were kept as 50 mm. Analysis showed much higher stresses. Then the block segmented into two parts were analysed. In such a configuration the stresses were found to be within limits. However this aspect need to be studied in greater details from all considerations, including manufacturability of such blocks. In future alternate designs of these blocks would be worked out for further analysis.

8.0 Stress analysis of passive power regulation system

This reactor incorporates a Passive Power Regulation System (PPRS) [2]. This system includes a gas header filled with helium gas at moderate pressure. The header is attached to a driver tube, which contains lead-bismuth eutectic alloy as driven liquid. The driver tube is housed within a control tube that contains an annular control rod made of boron carbide with a material compatible to Pb-Bi at that temperature. The annular space between the driver and control tube contains lead-bismuth eutectic, on which the control rod floats. The space above the liquid level is filled with helium. The PPRS gas header, located in the top plenum, is submerged in the coolant and senses the coolant temperature immediately downstream of the heat pipes. Under normal operating conditions, the gas header is surrounded by coolant at 450 °C, the temperature resulting after removal of the reactor power by the heat pipes. Any condition (such as non availability of heat pipes), which causes the coolant to return at a temperature higher than the normal, would also cause the gas in the gas header to heat up. This would lead to a rise in gas pressure in the driver tube and would result in a pressure imbalance between the driver and the control tube. This, in turn, would cause the level of liquid in the driver tube to go down and that in the control tube to go up. Since the absorber rod floats on liquid, it would also rise with the liquid level in the reactor core, thus inserting negative reactivity. Depending on the temperature rise sensed, the system would stabilise at a particular value of reactivity insertion. The PPRS operation was analysed using an in-house developed dedicated computer code. The stress analysis of the PPRS components were carried out. The results were found to be within limits.

9.0 Decay heat removal system

The decay heat removal system consists of a passive means of establishing a heat transport path across the gas gaps. The system is shown schematically in Figure-11. Liquid metal is stored in a reservoir above the upper plenum block. The gas gaps are connected to this reservoir by a bank of bent tubes called siphon tubes. The gas above the liquid metal in the reservoir communicates with a bulb, which is dipped into the coolant in the upper plenum. When all heat removal paths are unavailable, the temperature of the coolant will rise. This will cause the gas inside the immersed bulb to increase in temperature and hence raise the pressure above the liquid metal in the reservoir. As a result of this, the liquid metal level in the siphon tube will rise, bend the corner and ultimately a siphon is established. As the liquid metal inside the reservoir drops, the pressure inside the reservoir decreases, which may lead to interruption in the siphon action. To prevent this a connecting tube is provided. This tube is partially dipped into the liquid metal in the reservoir at one end and communicates with the gas tank at the other end. As the liquid metal gets transferred from the reservoir the level in the reservoir drops below the opening of the connecting tube thus equalising the gas pressure in the gas gap and in the reservoir (the gas gaps being also connected to the gas tanks), hence allowing continued transfer of liquid metal.

The trigger temperature, or the temperature at which the transfer of liquid metal from the reservoir to the gas gaps is initiated determines the height of the siphon tube. If the trigger temperature is set too close to the core outlet temperature conditions under normal operating conditions, it may lead to spurious trigger of the system. Hence a 100°C margin is provided between the trigger temperature and the normal operating outlet temperature. With molten tin (at 600°C) as the working liquid, the siphon tubes were designed.

The liquid level was determined by the need to provide adequate inventory of liquid metal in order to fill the gas gaps. As the gas gaps get filled up, the difference in liquid metal levels in the reservoir and gas gaps decrease and hence decreasing the driving head for the siphon. It is therefore necessary to provide some extra inventory in the reservoir so that adequate level difference is maintained as the gas gaps get nearly filled up, thus ensuring their filling in reasonable time. In the current design an excess inventory of 30% has been assumed.

Dimensions of the reservoir were decided from available space point of view and are about 1000 mm ID and 1550 mm OD. From redundancy considerations, the reservoir is subdivided into twelve segments to ensure at least partial filling of the gas gaps if one of them fails. Each of these segments has its own complement of gas bulb, siphon tube, connecting tube and siphon tubes and hence can act independent of the others. Each reservoir is provided with six sets of siphon tubes in order to provide optimum flow area.

The flow of liquid metal once the siphon has been established can be modelled by using the one-dimensional form of momentum conservation equation. In the current design a finite volume methodology has been used to solve the equation while taking care of all minor losses like bends, entries and exits. The variation of level in the gas gaps as obtained for the current design is shown in Figure-12, with molten tin at 600°C as the working fluid. The level of 1300 mm, which corresponds to the combined height of the core and the axial reflectors, is marked. This corresponds to the region from where removal of heat is most critical and is attained in about 7.2 seconds. It may be noted that the equilibrium level is attained at a liquid level much higher than 1300 mm as extra inventory has been provided for in the reservoir.

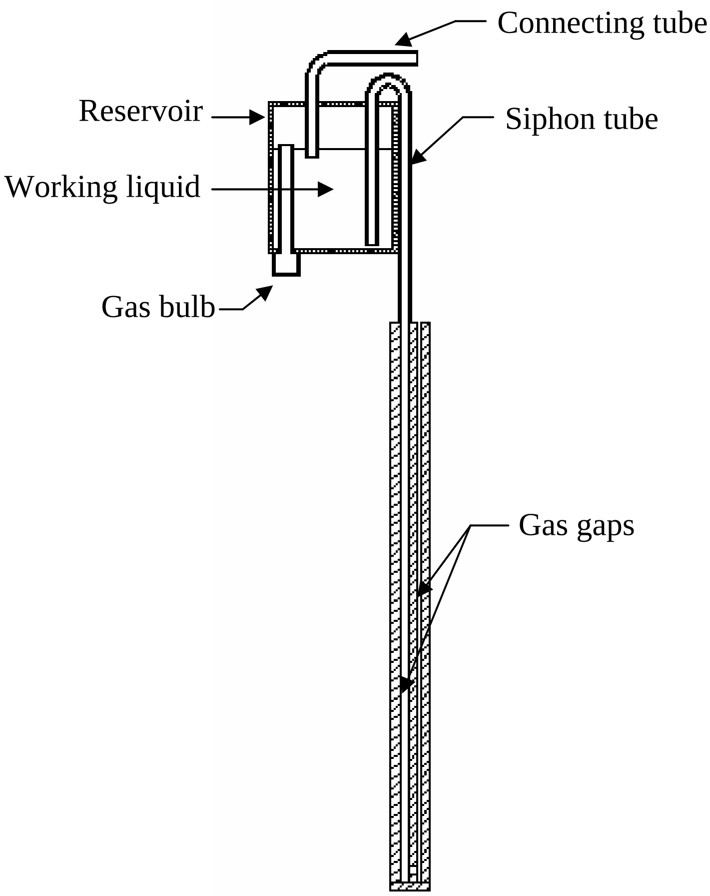


Figure-11: Schematic of decay heat removal system

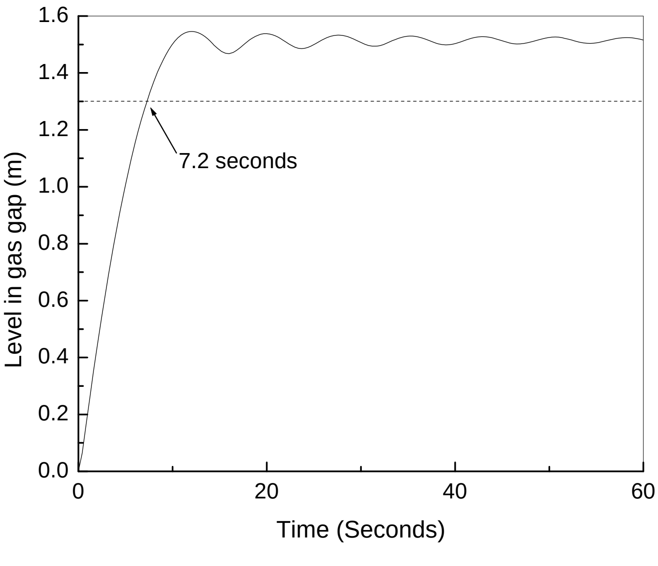


Figure-12: Variation of level of liquid metal in gas gap with time during operation of gas gap filling system. Level of 1.3 m corresponds to height of core and axial reflector is marked.

9.1 Temperature distribution during decay heat removal

During the reactor shutdown period the decay heat is produced in the reactor, which is expected to be around be 5 – 6% of the reactor power. In this reactor a passive system has been incorporated to dissipate the decay heat produced. This system involves filling the gas gaps with molten metal so as to facilitate conduction flow of reactor heat to outside heat sink.

An analysis of the reactor was carried out using finite element method considering conduction heat transfer mode in order to determine the prevalent temperatures in the various components of the reactor during the reactor shut down. Utilizing the 30° r-θ symmetry of the reactor core, a 1/12th two-dimensional sector of the core and reflector was modelled along with shells gas gaps and fins for this analysis. A 4-noded element was used. The thermal contact resistance between the graphite fuel tubes and beryllium oxide blocks and between beryllium oxide and graphite blocks was assumed negligible. A uniform volumetric heat generation load 300 kW was applied in the each fuel tube to simulate the effect of decay heat. No convection heat transfer coefficients were applied to the inner diameter of the fuel tubes. The coolant temperature in the fuel tube is assumed to be 600° C. All other surfaces were assumed to be adiabatic. The analysis is done with three alternate filling materials in the gas gap e.g. Tin, Indium, and Pb-Bi. The temperature contours so obtained are shown in Figure-11. A comparison of the temperature profiles for the three molten metals are shown in Figure-12

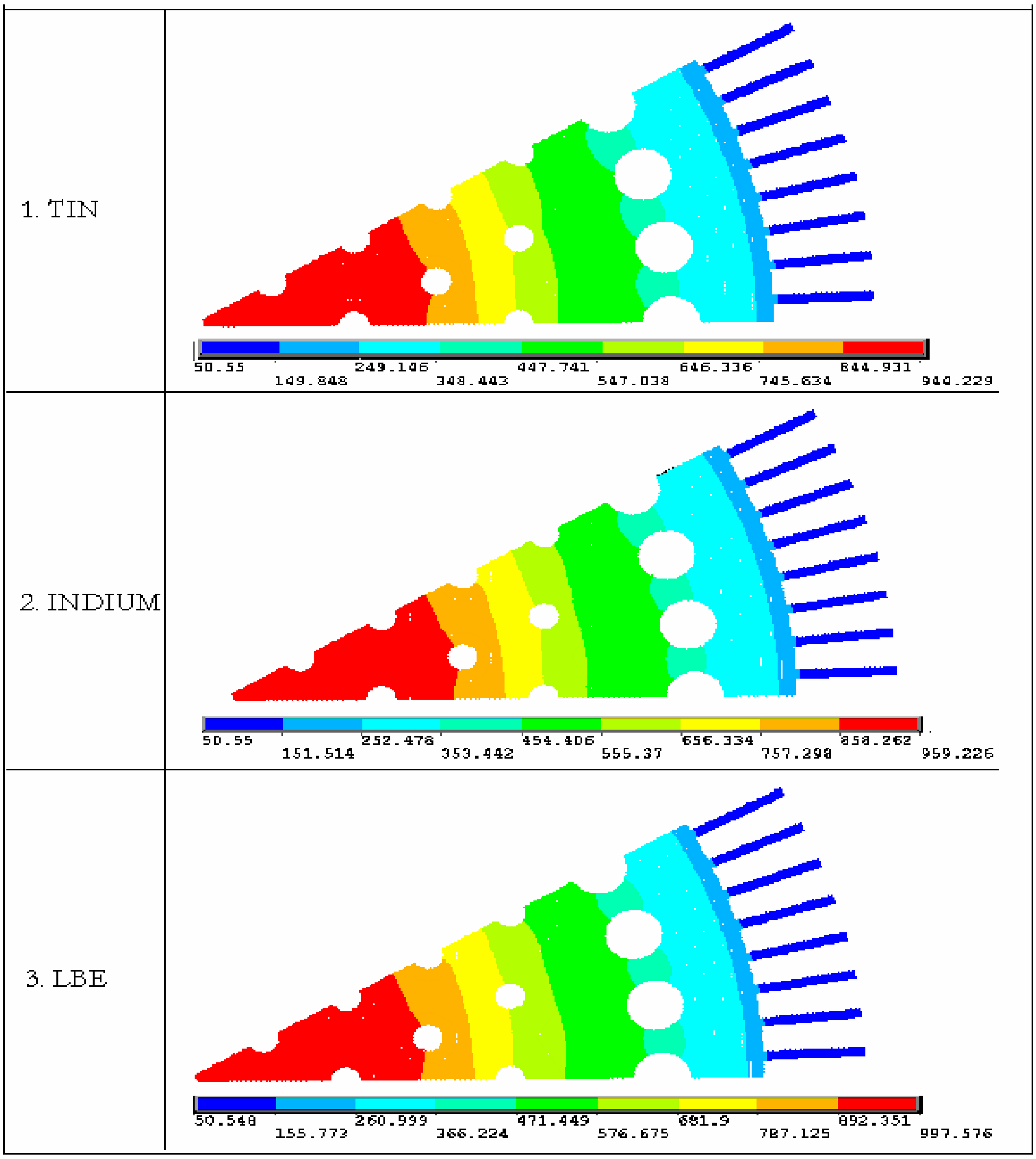


Figure-11: Temperature contours (in ºC) during decay heat removal by gas gap filling system

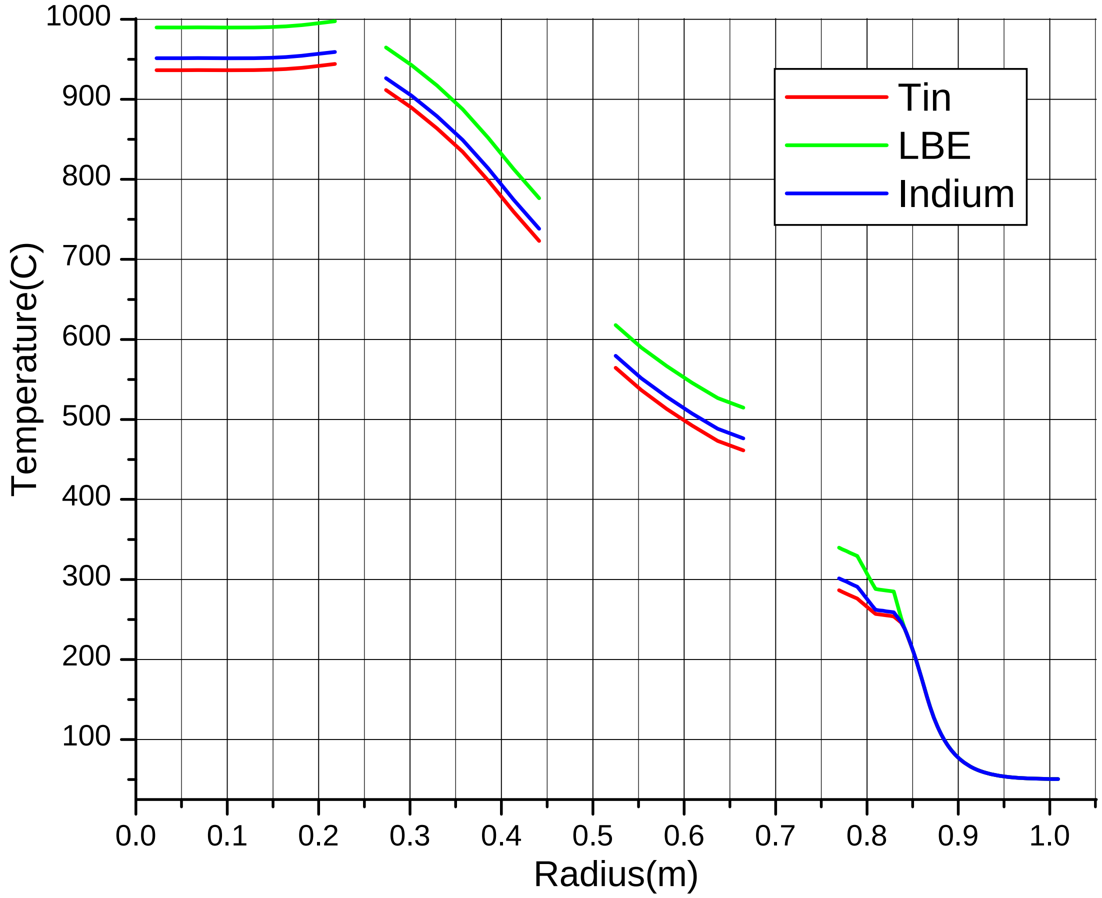


Figure-12: Temperature profiles for gas gaps filled with the three molten metals

10.0 Design of interface for heat removal during normal operations

In order to utilise the high temperature heat, a system of heat utilizing interface system vessels has been incorporated in the design. These vessels would facilitate suitable interface hardware, area, environment etc. for various high temperature heat utilizing systems such as direct thermo-electric conversion systems to produce electricity, hydrogen production system, etc. These vessels would be acting as secondary heat exchangers and thus would prevent contamination of the above-mentioned heat utilizing system components. This vessel basically consists of a pool of Pb-Bi eutectic coolant. This receives heat from the upper plenum of the reactor through a system of potassium heat pipes. The coolant in the pool can either be circulated through an intermediate heat exchanger. Alternately thermo-electric generators could be mounted on the walls of these vessels to produce electricity passively.

11.0 Development of point kinetics based coupled neutronics-thermal analysis code

A point kinetics based coupled neutronics-thermal analysis code has been developed. The thermal model used in this system is the one-dimensional transient thermal diffusion equation with temperature dependent material properties. Due to the dependence of material properties on temperature, the diffusion equation becomes non-linear and does not possess, in general, an analytical solution. The thermal model was translated into a computer code for obtaining a numerical solution. The computer code has been validated for a few problems in which the degree of complexity varied. During these validation studies it was noted that the thermal model required a relatively fine mesh, especially at locations with sharp change in temperature. The thermal code is capable of tackling problems only in one space dimension (along the cylindrical radial direction) and time. Modelling a complex system to fit such a code will result in certain assumptions. The assumptions as related to modelling of the reactor core are:

1) The mass of the entity being modelled is preserved in an equivalent sense. The reactor core is very complex with hexagonally shaped beryllia blocks and irregularly shaped graphite reflector blocks. In a one – dimensional code, it is difficult to capture this geometric complexity completely. However, to simulate the core as nearly as possible, entities (such as group of fuel tubes, beryllia blocks or the graphite reflector blocks) are modelled as successive rings. To ensure the validity of analysis using this assumption, an order of magnitude comparison was made with results of a steady state thermal analysis performed on the core in two – dimensions using the finite element method. The match between the results is fair, if one allows the fact that results of a one – dimensional code are being compared with a two – dimensional code and some basic assumptions are different.

2) The fuel is not modelled independently. It is assumed to be homogenously mixed with the surrounding graphite tube material. This mixture is now identified as ‘fuel’. As a consequence of this, material properties need to be calculated using properties of fuel and graphite matrix material and mass averaging is performed.

3) The core cross-section modelled is across the mid-plane of the reactor core. The coolant boundary conditions at various fuel tubes are specified accordingly.

4) It is assumed that the heat transferred to the coolant from the reactor core, during a transient, heats up the coolant. This leads to a time-dependent coolant temperature boundary condition. The heat transfer coefficient, however, is maintained constant.

5) The natural circulation of coolant is assumed to proceed unchanged, being assumed to be unaffected by any transient power surge.

6) The heat sink is assumed to be capable of removing all the heat transferred to the coolant. This allows for the fixed down comer tube coolant temperature.

In the present stage of development the point reactor kinetics model for neutronics calculations has been used. In the point kinetics model, it is assumed that the power function can be separated into the space and time functions. The spatial flux is assumed to be constant and only amplitude changes with time. This approximation is adequate in transients when flux-shape is not changing much. In the present model, six groups of delayed neutrons have been considered. The required neutronics parameters such as delayed neutron fraction, decay constants, mean neutron lifetime and reactivity coefficients of temperature etc. were calculated using lattice analysis codes. The algorithm for coupling the two parts of the code viz. the thermal and neutronic parts utilises the fact that there is interdependence of power (and hence neutron flux) and temperature. The temperature at any instant depends on the power being produced and the power in turn depends on the fuel temperature via its effect on overall reactivity. The initial temperature data is supplied to the code as input. At this steady state temperature, the net reactivity is zero. Any transient, initiating from this point, will result in a change in reactivity. As soon as the reactivity changes, the overall neutron flux (and hence the power) changes. However, this causes a change in the fuel temperature. The total reactivity calculation utilises a model in which the fuel temperature contribution is included. This cyclic dependence drives the system to a new steady state point, only if the reactivity coefficient is negative. The fuel is a distributed entity in the thermal model and hence a mass average value is calculated and used in the neutronics model. The code is currently undergoing extensive validation. In future, the aim is to use fully implicit approach and ultimately develop full 3-dimensional neutronics and thermal hydraulics models. This task would be performed through step-by-step development in several stages.

12.0 Summary

To satisfy energy related needs of Indian population staying in zones, which is difficult to be connected to national or regional grid system; conceptual design of a 5 MWth reactor having a long core life is being carried out. The reactor physics design of the reactor was studied for inadvertent control rod withdrawal incident. Following points summarises the findings:

a) An external reactivity of 2.9 mk, which is equal to the maximum worth a control rod in critical configuration, was added in 5 seconds (fast transient) and in 60 seconds (slow transient).
b) The rise in power due to addition of reactivity increases the temperature of fuel as well as fuel tube. In both the transient cases, the power rises and stabilizes to about 3 times the initial power.
c) The averaged fuel temperature rises to about 742 °C. It may be mentioned that these results were obtained with a very basic point reactor model with temperature feedbacks. It would be worthy to analyse the given transient of inadvertent withdrawal of maximum worth control rod with more accurate 3-D models with detailed thermal-hydraulics feedback.

The reactor has been designed to have natural circulation of lead-bismuth eutectic alloy coolant to remove reactor heat under normal operation. Thermal hydraulic design of the reactor was modified to have reduced number of downcomer tubes. Three dimensional temperature distribution of the entire core was carried out. The results were found to be in agreement with earlier 3-D analysis carried out. The stress analysis of the core components of the reactor was carried out. For the fuel tube, BeO moderator blocks, and graphite reflector blocks the stresses were found to be within allowable limits. However the BeO reflector block design would have to be reviewed considering higher prevalent thermal stresses. Stress analysis of the passive power regulation system was also carried out and the components dimensions were accordingly modified. A passive decay heat removal system design was worked out. It was found that with this system it was possible to remove decay heat without exceeding the acceptable fuel temperature. A point kinetics based coupled neutronics-thermal analysis code was developed.

13.0 Future work planned

a) Analysis of the reactor for the transient conditions using the developed point kinetics based coupled neutronics-thermal analysis code of the reactor.
b) Preparation of overall layout of different components of the reactor.
c) Modified design of the BeO reflector blocks

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