Comparison of Fuel Assemblies in Lead Cooled Fast Reactors

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Abstract

This paper presents a comparison of the thermal-fluid processes in the core, fuel heat transfer, and thermal power between two fuel assemblies: square and hexagonal, in a lead-cooled fast reactor (LFR). A multi-physics reduced order model for the analysis of LFR single channel is developed in this work. The work focused on a coupling between process of neutron kinetic, fuel heat transfer process and thermal-fluid, in a single channel. The thermal power is obtained from neutron point kinetics model, considering a non-uniform power distribution. The analysis of the processes of thermal-fluid considers thermal expansion effects. The transient heat transfer in fuel is carried out in an annular geometry, and one-dimensional in radial direction for each axial node. The results presented in comparing these assemblies consider the temperature field in the fuel, in the thermal fluid and under steady state, and transient conditions. Transients consider flow of coolant and inlet temperature of coolant. The mathematical model of LFR considers three main modules: the heat transfer in the annular fuel, the power generation with feedback effects on neutronic, and the thermal-fluid in the single channel. The modeling of nuclear reactors in general, the coupling is crucial by the feedback between the neutron processes with fuel heat transfer, and thermo-fluid, where is very common the numerical instabilities, after all it has to refine the model to achieve the design data. In this work is considered as a reference the ELSY reactor for the heat transfer analysis in the fuel and pure lead properties for analyzing the thermal-fluid. The results found shows that the hexagonal array has highest temperature in the fuel, respect to square array.

1. INTRODUCTION

Due to the interest of many countries to use nuclear energy as a source to meet present and future needs of electricity, in 2000 the Generation IV International Forum (GIF) was created to research and develop the next generation of systems advanced nuclear power [1].

This generation is made up of six systems; advanced thermal reactors are the VHTR (Very High Temperature Reactor), the SCWR (Super Critical Water-cooled Reactor) and MSR (Molten Salt Rector). Reactor fast neutron spectrum that includes the SFR (Sodium-cooled Fast Reactor), the GFR (Gas-cooled Fast Reactor) and LRF (Lead-cooled Fast Reactor).

The cooled lead system has been selected by several countries, in Asia can highlight systems: CLEAR (China LEAd-based Reactor) from China, PEACER (Proliferation-resistant, Environmental-friendly, Accident-tolerant, Continuable, and Economical Reactor) by Korea and LSPR (LBE-cooled long-life safe single small portable proliferation-resistant reactor), and 4S (Super Safe, Small and Simple) of Japan. Meanwhile Russia has several programs that include the SVBR (Svintsovo Vismutovyi Bystriy Reaktor) cooled by LBE (lead-bismuth eutectic), and BREST (Bystriy Reaktor Estestrennoy Bezopasnosti). In USA has the reactor SSTAR (Small Secure Transportable Autonomous Reactor) stands out [2]. In Europe, the selected design was the ELSY (European Lead-cooled System), which emerged within the EU-FP6-ELSY project in 2006.

The reactor design ELSY is focused to satisfy the targets set for Generation IV Systems, being economically competitive thanks to the compact configuration pool type reactor, with an imminent security in the primary system that it is at atmospheric pressure, removal system heat with natural circulation, and low temperature difference across the core. This reactor is sustainable because it contributes to the reduction of waste generation, and uranium mining, in addition to the closed fuel cycle that uses prevents the proliferation of nuclear weapons [3][4].In this work is considered as a reference the ELSY reactor for the heat transfer analysis in the fuel and pure lead properties for analyzing the thermal-fluid.

In this work we present a comparison of the thermal-fluid processes in the core, fuel heat transfer, and thermal power between two fuel assemblies: square and hexagonal, in a lead-cooled fast reactor (LFR). The work focus on a coupling between process of neutron kinetic, fuel heat transfer process and thermal-fluid, in a single channel.

When performing the analysis found that the hexagonal array is an option to make more compact the design of the core reactor. In state transient the effects in square array is higher than hexagonal array for different mass flow rate and inlet temperature, respectively. This condition could be because square has more rods per assembly than hexagonal array.

2. DESCRIPTION OF THE REACTOR

The ELSY reactor is a pool type reactor cooled with pure lead, which has a thermal power of 1500 MW. They have been considered two types of fuel; the mixed oxide (MOX) of highly enriched UO2 and PuO2-nitride fuel (as an advanced option) for cases of this study is considered the MOX [5]. Table 1 specifications of the ELSY reactor are presented.

Table I. Features of ELSY reactor

Feature	Value
Thermal power	1500 MW
Efficiency	40 %
Fuel	MOX
Coolant	Lead
Length	1.2 m
Diameter	4.54 m
Coolant inlet temperature	673.15 K
Coolant outlet temperature	750.15 K
Maximum speed of lead	2 m/s

The design of the reactor core ELSY has been evaluated in two basic arrangements; hexagonal and square [6]. Table II the characteristics of the fuel assemblies studied in this work are presented. In this table can be seen that the hexagonal array has more fuel assemblies than the square, however a square assembly has more fuel rods per assembly. Figure 1 the core and fuel assembly for both arrangements are shown.

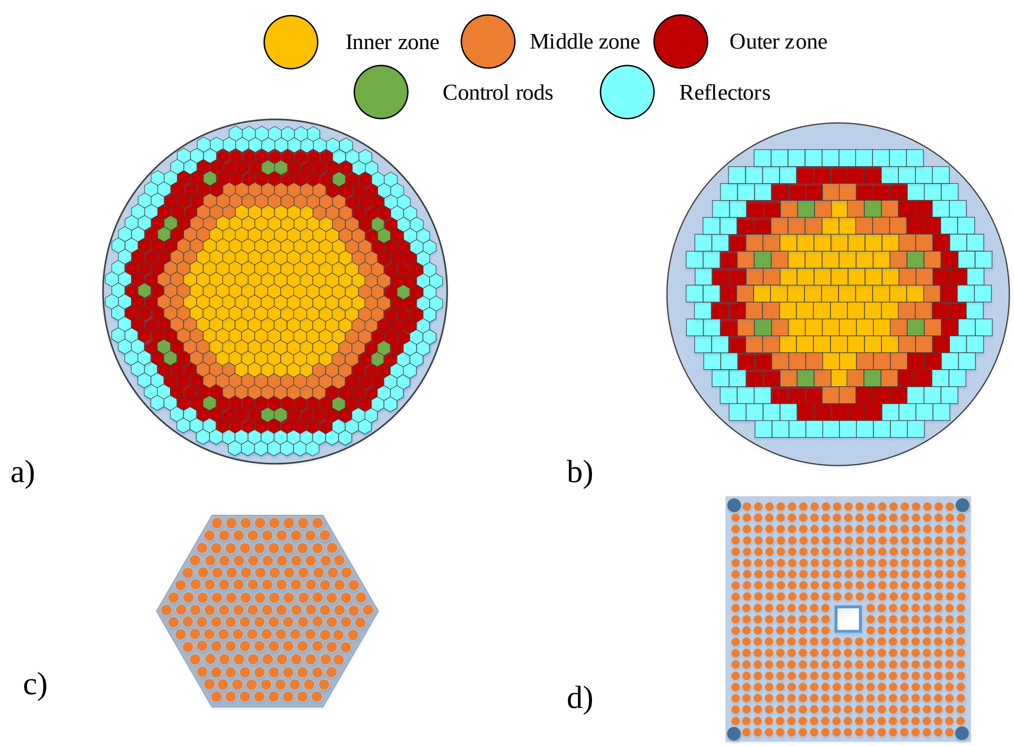

Figure 1. a) Core in hexagonal array, b) Core in square array, c) Hexagonal assembly fuel d) Square assembly fuel (adapted [5])

Table II. Features of core in hexagonal and square array

Feature	Hexagonal array	Square array
Total assemblies	325	170
Rods/assemblies	165	428
Thermal power/rod (kW/rod)	27.31	20.62

Each of these arrangements, the core distribution assemblies depend enrichment having the oxide mixture. For the case of the square array enrichment is 15.5%, 16.5% and 19.5% for the inner zone, middle and outer, respectively. While that for hexagonal array are 14.45%, 17.53% and 20.50%, respectively [7].

3. MATHEMATICAL MODEL

The mathematical model of LFR considers three main modules: (1) the heat transfer in the annular fuel, (2) the power generation with feedback effects on neutronic, and (3) the thermal-fluid in the single channel. The Modeling of nuclear reactors in general, the coupling is crucial by the feedback between the neutron processes with fuel heat transfer, and thermo-fluid, where is very common the numerical instabilities, after all it has to refine the model to achieve the design data.

3.1 Neutronic power

The power analysis will be performed for each rod. The thermal power per rod is given by the following equation

$$P(t) = P_o(t)n(t)\psi(z) \tag{1}$$

where P_o is the nominal thermal power per fuel rod, $n(t)$ is the neutron density, and $\psi(z)$ is the axial power distribution.

To calculate the nominal thermal power in each rod is necessary divide nominal thermal power of reactor between numbers of rods on core. Table II shows the values for hexagonal and square arrays.

For determination of the neutron density, the neutron point kinetic equations are applied [8]:

$$\frac{dn(t)}{dt} = \frac{\rho(t) - \beta_{eff}}{\Lambda} n(t) + \sum_{i=1}^{6} \lambda_i C_i(t) \tag{2}$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda} n(t) - \lambda_i C_i(t) \tag{3}$$

where ρ is the reactivity, β_{eff} is the effective fraction of delayed neutron, Λ is the generation average lifetime of instantaneous neutron, λ is the decay constant of delayed neutron precursor, C_i is the concentration of the i-th neutron delayed precursor. The first and second terms on right side of the Eq. (3) represents the rate of formation of the precursors and radioactive decay of the i-th group, respectively.

Reactivity is influenced by fuel temperature according (ρ_D) to fuel reactivity feedback coefficient α_f, mainly due to Doppler effect (-1 pcm/C), and Reactivity by control rods (ρ_{CR}) are taken into account by means of a second coefficient α_z representing an ideal control rod. Reactivity feedbacks are thus evaluated as follows:

$$\rho = \rho_{CR} + \rho_D \tag{4}$$

where, $\rho_{CR} = \alpha_z \Delta z$, and $\rho_D = \alpha_f \Delta T_f$.

3.2 Fuel heat transfer

The fuel mathematical model considers heat transfer in annular fuel pellets of hexagonal and square arrays, as is illustrated in Figure 2. In this figures can be observed that fuel pellets has differences in dimensions for each assembly design.

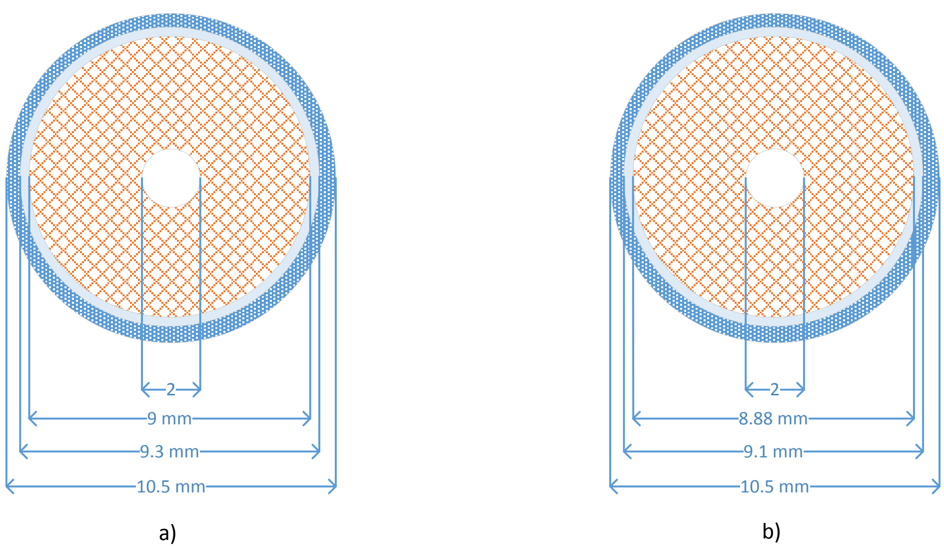

Figure 2. Cross section of fuel rod, a) hexagonal array, b) square array. (Adapted [3])

The fuel heat transfer formulation is based on the following fundamental assumptions:

- Axis-symmetric radial heat transfer, i.e., $\mathbf{e}_r \cdot \nabla T = 0$ at $r = 0$.
- The heat conduction in the axial direction is negligible with respect to the heat conduction in the radial direction.
- The volumetric heat rate generation in the fuel is uniform in radial direction.
- The gap spacing was considered as uniform.

Under these assumptions, the transient temperature distribution in each physical region of the fuel:

$$\rho_f Cp_f \frac{\partial T_f}{\partial t} = \frac{k_f}{r}\frac{\partial}{\partial r}\left(r\frac{\partial T_f}{\partial r}\right) + q'''(t), \qquad Fuel \qquad r_a < r \le r_f \tag{6}$$

$$\rho_g Cp_g \frac{\partial T_g}{\partial t} = \frac{k_g}{r}\frac{\partial}{\partial r}\left(r\frac{\partial T_g}{\partial r}\right), \qquad Gap \qquad r_f < r \le r_g \tag{7}$$

$$\rho_{cl} Cp_{cl} \frac{\partial T_{cl}}{\partial t} = \frac{k_{cl}}{r}\frac{\partial}{\partial r}\left(r\frac{\partial T_{cl}}{\partial r}\right), \qquad Clad \qquad r_f < r \le r_{cl} \tag{8}$$

The subscript f refers to fuel radio, subscript g refers to gap radio and cl refers to clad radio. The initial condition is given by $T(r,0) = f(r)$, boundary condition are:

$$\frac{dT_f}{dr} = 0 \qquad at \qquad r = r_a \tag{9}$$

$$-k_g \frac{dT_g}{dr} = h_g(T_f - T_g) \qquad at \qquad r = r_f \tag{11}$$

$$-k_{cl} \frac{dT_{cl}}{dr} = h_{cl}(T_g - T_{cl}) \qquad at \qquad r = r_g \tag{12}$$

$$k_{cl} \frac{dT_{cl}}{dr} = H(T_{cl} - T_{lead}) \qquad at \qquad r = r_{cl} \tag{13}$$

In these equations ρ is the density, Cp is the specific heat, k is the thermal conductivity, T is the temperature, H is the heat transfer coefficient and q''' refers to heat source, and the subscript $lead$ refers of coolant,. Figure 2 shows the radial dimensions of fuel for the hexagonal array and square array shown. Table III presents the transport properties for the gap (helium) and cladding (T91), which are considered constant in this work. The heat source term is given by

$$q'''(t) = \frac{P(t)}{V_f} \tag{14}$$

where P is the reactor power in the reactor given by Eq. (1), and V_f is the fuel volume.

Table III. Transport properties for gap and cladding

Property	Gap	Cladding
Density (kg/m^3)	2.425	7700.0
Specific heat ($J/kg\,K$)	5191.0	622.0
Thermal conductivity ($W/m\,K$)	$15.8\times10^{-4}T^{0.7}$	26.0

3.3 Thermal-fluid analysis

The thermo-fluid is modeled with mass, energy and momentum balance that considers thermal expansion effects.

Mass balance

$$\alpha\,\rho_{lead}\frac{dT_{lead}}{dt}+\frac{\partial G}{\partial z}=0 \tag{15}$$

where $\rho=\rho(T)$, and thermal expansion coefficient is given by $\alpha=\frac{1}{\rho}\left(\frac{\partial\rho}{\partial T}\right)$, and G is mass flux.

Energy balance

$$\frac{\partial T_{lead}}{\partial t}=\frac{P_m H\left(T_{clad}-T_{lead}\right)}{A_f\left(\rho_{lead}Cp_{lead}-h\alpha\rho_{lead}\right)}-\frac{G}{\rho_{lead}}\frac{\partial T_{lead}}{\partial z} \tag{16}$$

where P_m wetted perimeter, A_f is the flow area (cross-sectional area), and h is the enthalpy.

Momentum balance

$$\frac{\partial G}{\partial t}=-\frac{\xi_{fr}}{2}\left(\frac{G^2}{\rho_{lead}L_{rod}}\right)-\frac{\partial\left(G^2/\rho_{lead}\right)}{\partial z}-\rho_{lead}\,g \tag{17}$$

where L_{rod} is the length of the fuel rod, and the friction coefficient is given by

$$\xi_{fr}=\frac{0.210}{\mathrm{Re}^{0.25}}\frac{L_{rod}}{D_h}\left[1+\left(\frac{l_p}{d_{rod}}-1\right)^{0.32}\right] \tag{18}$$

where l_p is the rod pitch, d_{rod} is the rod diameter, Re is the Reynolds number, and D_h the hydraulic diameter.

The thermo-hydraulic parameters depend on the arrangement of the reactor. Figure 3 shows the flow area for hexagonal array and square array, where the rod pitch for hexagonal array is 15.5 mm, and for square array is 13.9mm. The hydraulic diameter for each array is given by:

$$D_h = \begin{cases} \dfrac{4}{\pi d_{rod}}\left(\dfrac{\sqrt{3}}{2} l_p^2 - \dfrac{\pi d_{rod}^2}{4}\right), & \textit{for hexagonal array} \\ \dfrac{4}{\pi d_{rod}}\left(l_p^2 - \dfrac{\pi d_{rod}^2}{4}\right), & \textit{for square array} \end{cases} \tag{19}$$

3.4 Numerical solution

The numerical solution of the neutron point kinetics equations is applying Runge-Kutta 4th Order Method.

The fuel heat transfer differential equations described in Section 3.2 are transformed into discrete equations using the control volume formulation technique in an implicit form for annular region of the fuel pin, fuel, gap and cladding:

$$a_j T_{j-1}^{t+\Delta t} + b_j T_j^{t+\Delta t} + c_j T_{j+1}^{t+\Delta t} = d_j \tag{20}$$

where $T_{j-1}^{t+\Delta t}$, $T_j^{t+\Delta t}$ and $T_{j+1}^{t+\Delta t}$ are unknowns, a_j, b_j, c_j and d_j are coefficients, which are computed at the time t. When these equations are put into a matrix form, the coefficient matrix is tri-diagonal whose solution procedure is the Thomas algorithm, which is the most efficient for this type of matrices.

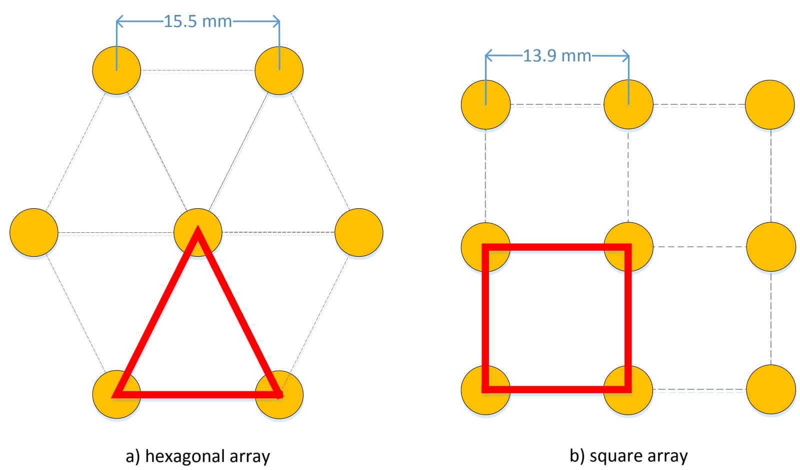

Figure 3. Area flow for calculation of thermo-fluid parameters

The numerical solution for energy balance and momentum balance was applying the Euler method.

4. MULTI-SCALE AND MULTIPHYSICS COUPLING

The coupling of multiphysics phenomena is a very complex subject with several possible combinations. An extensive description of this is given in [9], and some details and key points are found in [10-12]. The analysis is performed on a model coupling the multiscale and multiphysics phenomena of *reducer order*. The core model of LFR is a multiscale problem due to involves the neutron scale (Scale I), pellet scale for heat transfer (Scale II), fuel rod that includes the gap and cladding (Scale III), fuel assembly (Scale III), and core scale (Scale IV), where Scale I << Scale II << Scale III << Scale IV. Also the modeling of LFR represents of multiphysics problem due that involves of neutron movement and related theory, fuel heat transfer by molecular mechanism, and mechanism simultaneous of energy and momentum transport of the thermal-fluid of the lead as cooling. The term *reducer order* is introduced in this work to indicate that the multiscale, and multiphysics problem is carry out in a single channel (average) of the LFR, for steady state and transient analysis.

5. Numerical experiments

Numerical experimentation involves varying fundamental parameters such as the reactor power, temperature inlet in the channel and mass flow rate of the lead.

Figures 4-7 shows temperature distribution in fuel rod for 25%, 50%, 75% and 100% of rated power, respectively.

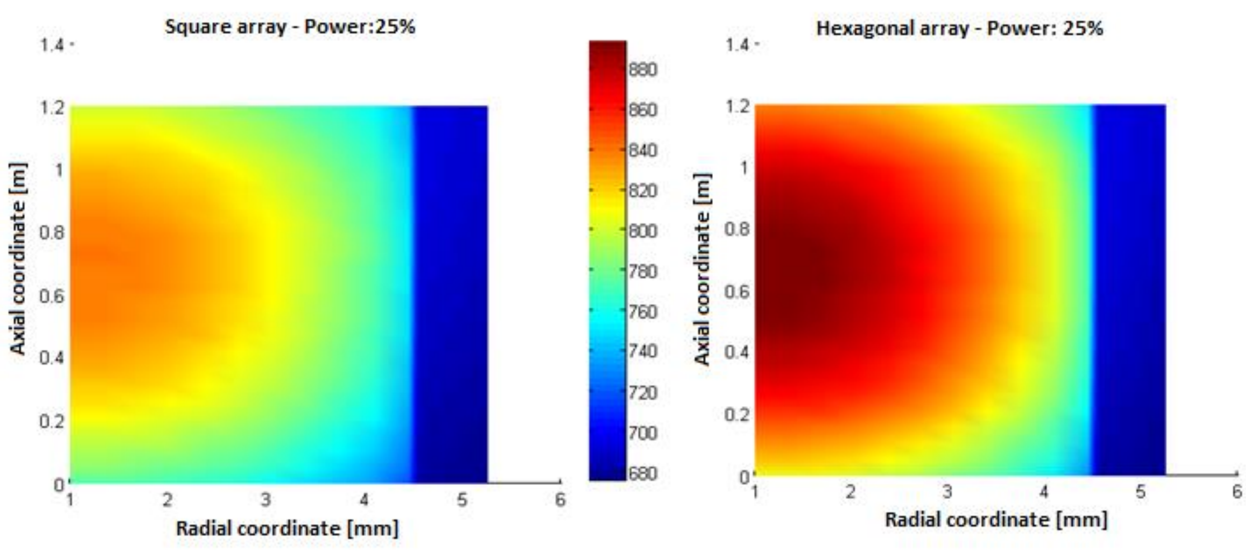

Figure 4. Temperature distribution for 25% of power

In this figures can be observed that the temperature distribution presents values that are lower for the square assembly with respect to hexagonal assembly. For 25% of rated power (Figure 4) the square assembly presents a minimum temperature of the cladding 675.15 K, and the maximum temperature of the fuel is 838.80 K, while for hexagonal assembly are 675.86 K and 893.07 K, respectively.

For 50% (Figure 5) minimum temperature in square assembly is 677.45 K, and the maximum temperature of the fuel 1029.44 K, while for hexagonal assembly are 678.93 K and 1157.43K, respectively.

In case of 75% (Figure 6) for hexagonal array maximum temperature is 1476.45 K and the minimum is 682.25; while for square assembly temperatures are 1249.36 K and 679.88 K respectively

For 100% (Figure 7) minimum temperature in square assembly is 682.45 K, and the maximum temperature of the fuel 1503 K, while for hexagonal assembly are 685.84 K and 1858.16 K, respectively.

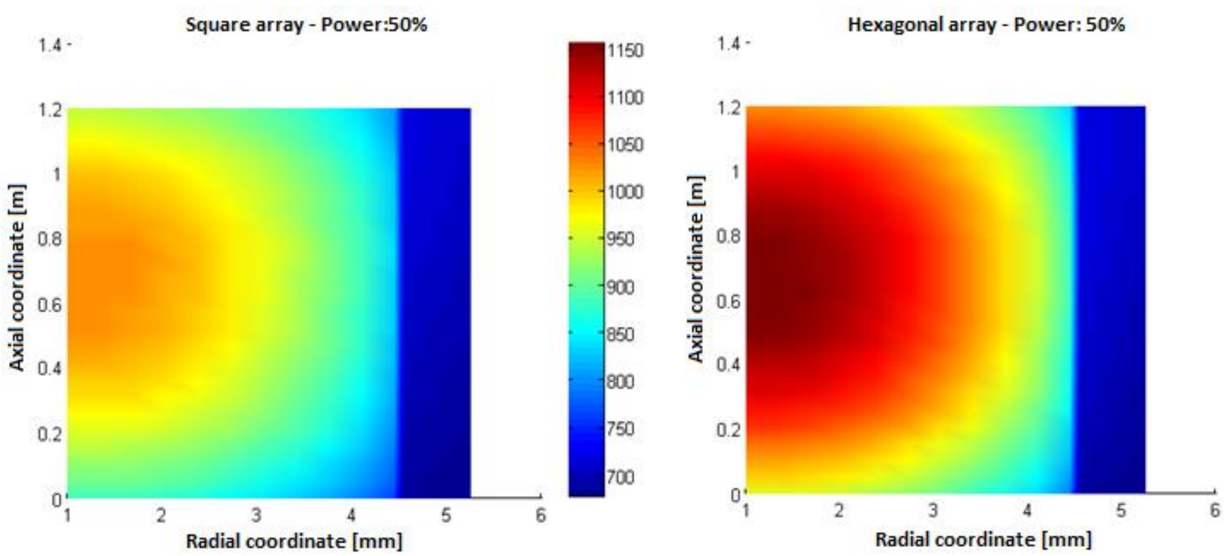

Figure 5. Temperature distribution for 50% of power

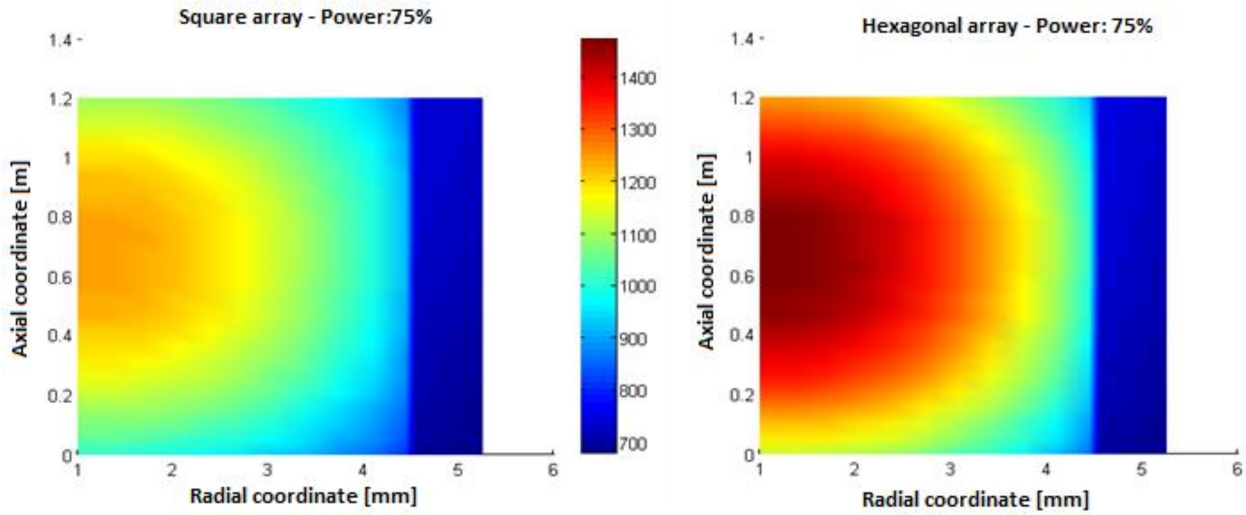

Figure 6. Temperature distribution for 75% of power

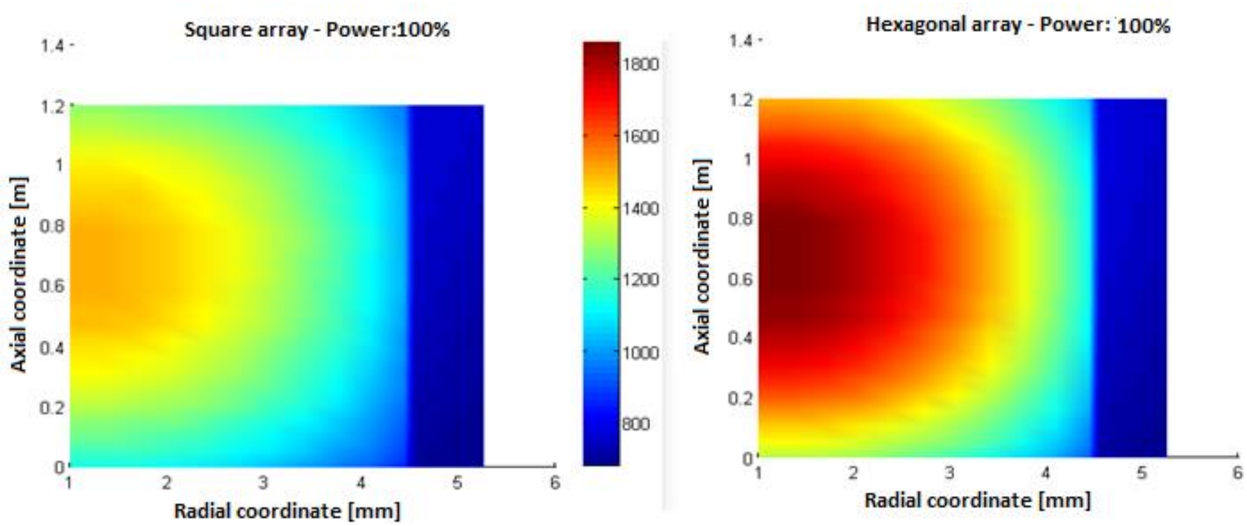

Figure 7. Temperature distribution for 100% of power

The temperature distribution shows a low temperature in square array that hexagonal array, this difference can be for the number of rod per assembly in each array.

The neutronic processes are very important for the operation of the reactor, because these processes could determine if the reactor is in state critical, super-critical or sub-critical. Figure 8 shows the neutron density for both arrays with different flows of coolant during transient condition. The effects in square array are higher than hexagonal array because it has more rods per assemble.

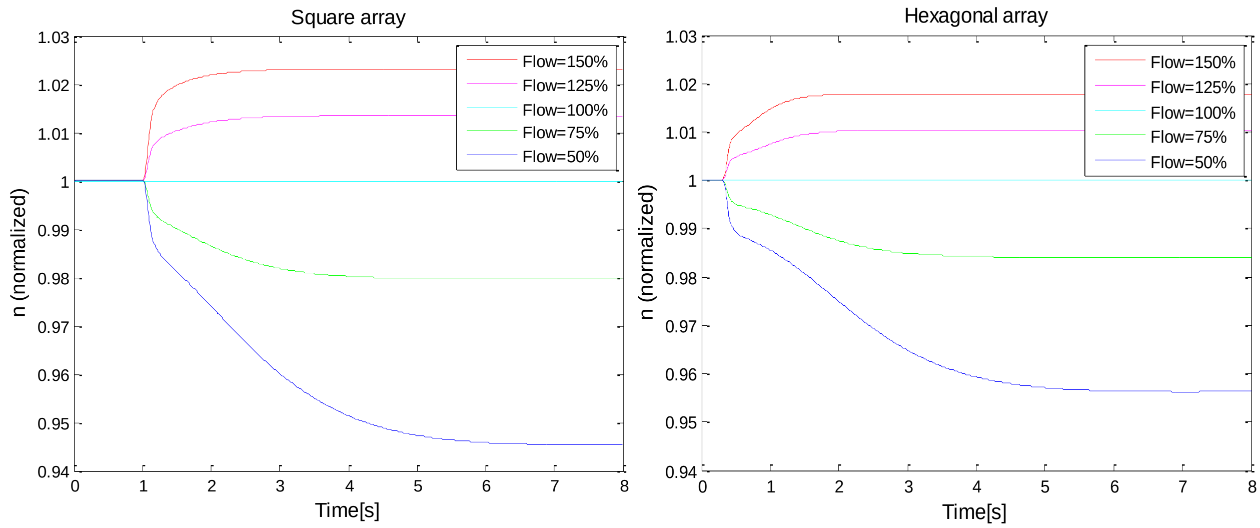

Figure 8. Neutron density normalized in square and hexagonal array with different flows of coolant

Figure 9 shows the neutron density for arrays with different inlet temperature of coolant during transient condition. These results are similar to previous, i.e., the effects to inlet temperate for square array is higher.

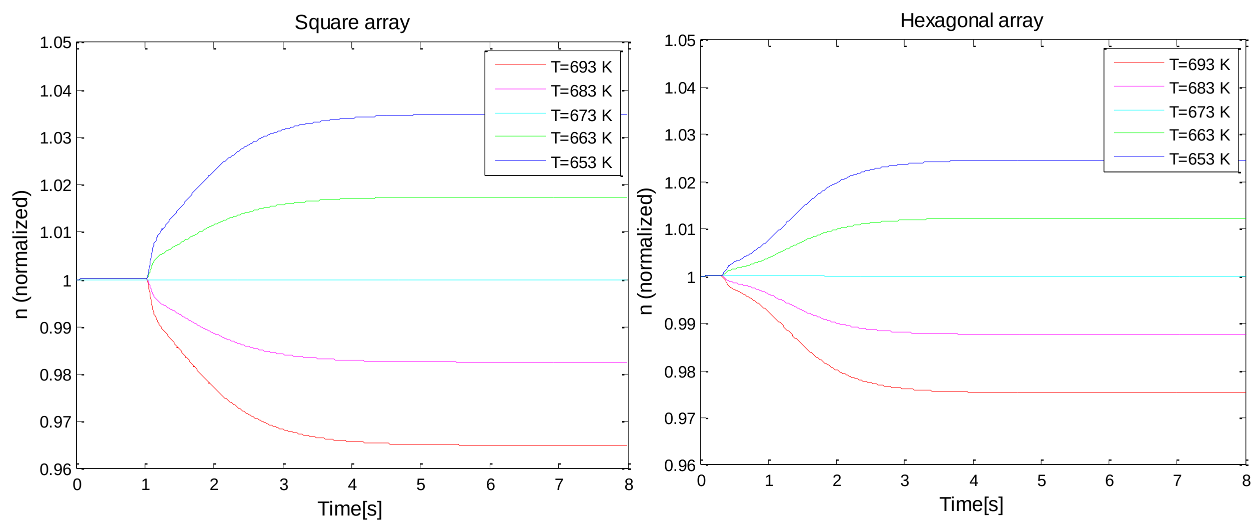

Figure 9. Neutron density normalized in square and hexagonal array with different inlet temperature of coolant

CONCLUSIONS

A Lead-cooled fast reactor can be use two different basic arrays in the core, hexagonal and square array. Each array improves different characteristics to operation. The geometry in the core can modify the temperature of fuel and coolant.

The hexagonal array is an option to make more compact the design of the core reactor, but these option increases the temperature in the fuel, Figures 4 to 7 it can see that in different power the fuel temperature is higher than square array.

In state transient the effects in square array is higher than hexagonal array how as can be seen in Figures 8 and 9, for different mass flow rate and inlet temperature, respectively. This condition could be because square has more rods per assembly than hexagonal array.

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