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Thermal Hydraulic Study of the SMART (System-integrated Modular Advanced Reactor) most Powerful Assembly

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ABSTRACT

SMART is a modular advanced nuclear reactor inspired by PWR, originally designed by the Korean Atomic Energy Research Institute (KAERI). As a modular reactor, SMART is designed to be manufactured and transported to provide electrical energy in strategic regions. The main proposal of the SMART project is to meet the energy demand in isolated regions such as the Earth's poles and some regions in the Middle East. SMART provides a nominal power output of 330 MWth and employs UO_2 as its nuclear fuel, with light water as the reactor coolant and Zircaloy-4 as the cladding material. The present work aims to perform the computational simulation of the most powerful SMART nuclear reactor fuel assembly, operating with a traditional PWR power distribution, using the Open-source Field Operation and Manipulation (OpenFOAM) computational code at steady state. The OpenFOAM is an open-source computational fluid dynamics (CFD) software package is based on solving conservation equations using the Finite Volume Method (FVM). It provides a wide range of capabilities for simulating fluid flow, heat transfer, and other related phenomena. The data for performing the calculation were provided by the KAERI report for the SMART operation configuration, as well as the configuration and position of the simulated fuel assembly. The developed thermal model considers the solid region as boundary condition for calculating the thermal physical parameters of the coolant. Inlet temperature and pressure were taken from the average inlet value of the reactor, as documented in the KAERI report. The main results of the thermal-hydraulic analysis include the distribution of pressure and temperature along the thermal hydraulic channels and the coolant velocity. For future work, efforts will be made to enhance the thermal model to consider the solid region of the fuel through conjugate heat transfer simulations (CHT), where both fluid and solid fields will be resolved. Furthermore, other types of fuel for use in the SMART reactor will be investigated.

1. INTRODUCTION

The continuous increase in electrical demand has prompted extensive research into new technologies capable of providing the necessary supply [1]. In this context, Small Modular Reactors (SMRs) are being researched worldwide to meet the electrical demand in isolated regions. This type of nuclear reactor has the advantage over large nuclear power plants of being able to be fabricated and licensed at the factory, then transported to the region of interest [2]. In 1997, the Korean Atomic Energy Research Institute (KAERI) began developing the conceptual design of the System-Integrated Modular Advanced Reactor (SMART) [3]. The nominal power of the reactor is 330 MW_{Th}, providing an electrical efficiency of around 30%, which corresponds to 10% of the electrical energy of large plants like the Angra 2 Pressurized Water Reactor (PWR), located in Rio de Janeiro, Brazil [4]. The SMART reactor is a small-sized PWR, using traditional UO_2 nuclear fuel, Zircaloy-4 as cladding material, and light water as coolant, moderator, and reflector. Fig. 1. presents a diagram of the reactor pressure vessel and Fig. 2. presents a detailed vision of the reactor core.

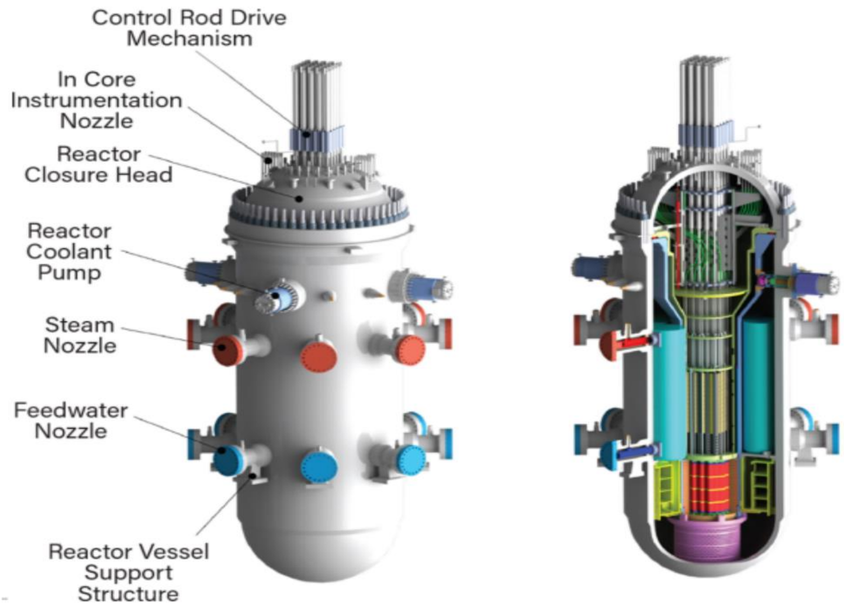
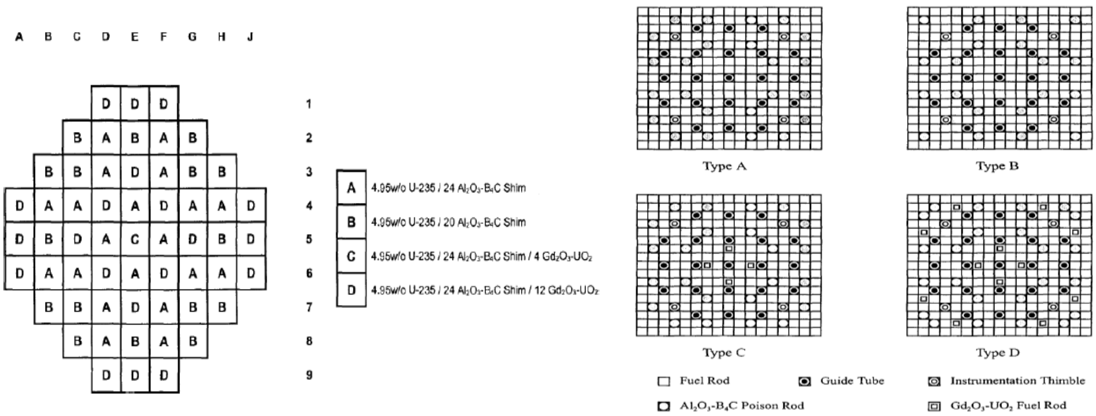


Fig. 1. SMART nuclear reactor diagram [3].

The reactor core contains 57 fuel assemblies, with a 17x17 rod configuration. The assembly is configured in four different assembly types: A, B, C and D. The most powerful assembly in SMART was identified as A type through the analysis of assembly-wise power distribution available in the KAERI report [3]. Fig. 2-a. presents a diagram of SMART core configuration and Fig. 2-b. presents the diagram of SMART four types of assembly.



(a) (b)

Fig. 2. SMART core diagram: (a) SMART core assembly configuration and (b) the four types of assemblies diagram, including the assembly simulated (A type) [3].

The present work aims to conduct a thermal-hydraulic study of the coolant region of the most powerful assembly in SMART nuclear reactor core, operating with a typical PWR power distribution. The choice of the most powerful assembly was made considering that it is the critical region of the reactor, making it crucial for a more detailed analysis. To conduct the study, the Open-source Field Operation and Manipulation (OpenFOAM) code was used. Computational Fluid Dynamics (CFD) codes, such as OpenFOAM, are very useful for performing detailed local analysis of physical problems. All analyses presented in the present work are in steady state, using the OpenFOAM-v2206 and solver buoyantSimpleFOAM.

2. METHODOLOGY AND MATHEMATICAL FORMULATION

2-1 GOVERNING EQUATIONS

The simulation was done using the suite OpenFOAM-v2206 (Open Field Operation and Manipulation), which is a free and open source code. The flow was treated as compressible. Except the density, the other thermodynamic properties of the fluid were assumed as constant. The equations that govern a single-phase Newtonian flow are the continuity equation 1, the momentum 2 and the energy equation 3, in terms of the density ρ and the velocity u_j. The first can be written as:

$$\frac{\partial \rho}{\partial t} + \frac{\partial(\rho u_i)}{\partial x_j} = 0 \tag{1}$$

In equation 1, x_j is the position and t is the time. The moment equation can be written as follows:

$$\frac{\partial(\rho u_j)}{\partial t} + \frac{\partial(\rho u_j u_i)}{\partial x_j} = -\frac{\partial p}{\partial x_i} + \frac{\partial}{\partial x}\left(\tau_{t,_{ij}} + \tau_{ij}\right) + \rho \mathrm{g_i} \tag{2}$$

where g_I is the acceleration of gravity, p is the pressure, $\tau_{t,ij}$ is the turbulent stress tensor, and τ_{ij} is the laminar stress tensor due to molecular viscosity. Finally, the energy equation is written in terms of the enthalpy h:

$$\frac{\partial(\rho h)}{\partial t} + \frac{\partial(\rho h u_j)}{\partial x_j} - \frac{\partial p}{\partial t} = \frac{\partial}{\partial x_k}\left(k_{eff}\frac{\partial T}{\partial x_k}\right) \tag{3}$$

Where k_{eff} is the effective conductivity.

OpenFOAM is a pack of codes that consists in the implementation of the Finite Volume Method (FVM) to solve differential equations. The FVM converts the divergences in the conservation equations into flux over the control volumes (finite volumes) surfaces using Gauss theorem, which later are converted in algebraic equations (computationally solved). A special case of differential equation that OpenFoam is capable to solve are the conservation of mass and momentum expressed by the Navier-Stokes equations using one of it solvers, the BuoyantSimpleFoam. This solver is capable to deal with the turbulence by Reynolds Average Navier Stokes (RANS) method. In the present simulation the turbulence is solved by the RANS method. During the simulations, two turbulent models were evaluated, the $\kappa - \epsilon$ and the $\kappa - \omega$. The results presented are from the $\kappa - \omega$ turbulence model, recommended for more accurate modeling of the fluid near the rods [6].

3 COMPUTATIONAL MODEL

To perform the simulations the 1/8 symmetry of the PWR most powerful assembly were used. The symmetry consideration is justified by the fact that the simulations were performed in a deterministic computational code, i.e. the same numerical equation would be solved by the same numerical model 8 times providing equal results. To perform the thermal-hydraulic study, a computational modeling was implemented for the calculation of the pin power profile for the rods. The thermal nominal power of the nuclear reactor (330 MW_{Th}) was distributed to the 57 fuel assemblies by using the assemblywise power distribution available in KAERI report [3], where the power of the most powerful assembly power was discovered by multiplying the average assembly power (5,7895 MW_{Th}) by the radial power factor (1,455). Next step was taken by distributing the assembly power at the rods, where each rod power was discovered. Fig. 3. presents a typical PWR pin power distribution.

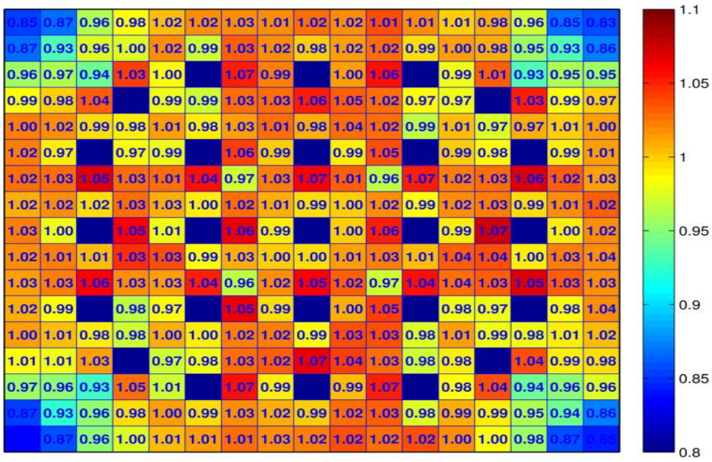


Fig. 3. Pin power distribution used during the simulations [5].

The PWR pin power distribution was obtained in a work available on literature [5]. The same procedure used for the assembly power distribution was implemented for the discovery of the pin's power distribution. The power of each rod was discovery multiplying the average rod power times the rod power factor (Fig. 3.). For each rod, the power distribution by unit of surface was modeled by equation 4, according to the neutron diffusion equation, as function of the amplitude A, the natural frequency w and the phase φ in OpenFOAM as boundary condition for each rod wall. The modeling was made keeping the power value in each surface element of the rods as an exclusive function of the height z.

$$P(z) = Asin(wz + \varphi) \qquad (4)$$

The parameters w and φ were defined to provide power equal to zero at the rods edges ($z = 0\ m$ and $z = 2{,}166\ m$) ($w = 3{,}14159/2.166m^{-1}$ e $\varphi = 0$). The amplitude A was defined independently for each rod by the condition that the integration of equation 4 over each rod surface must be equal to the total rod power. By the integration of the equation 4 over the rod (2,166 meters height and $9{,}5.10^{-3}$m radius) surface, equation 5 was obtained for the amplitude $A[\frac{W}{m^2}]$ of each rod, that depends on the rod radial power factor F, presented in Fig. 3, for the entire assembly.

$$A = 12{,}1494m^{-2} < P > F \qquad (5)$$

Where $< p > = 31907{,}895$ W is the average power per rod.

3.1 OPENFOAM MODEL

To construct a case in OpenFOAM the spatial discretization is required (Mesh construction), provide thermal physics boundary conditions for the algorithm solver as well system condition of operation defined in a thermal hydraulic model context that are presented in the next sections. Fig. 4. Shows the mesh used for the simulations.

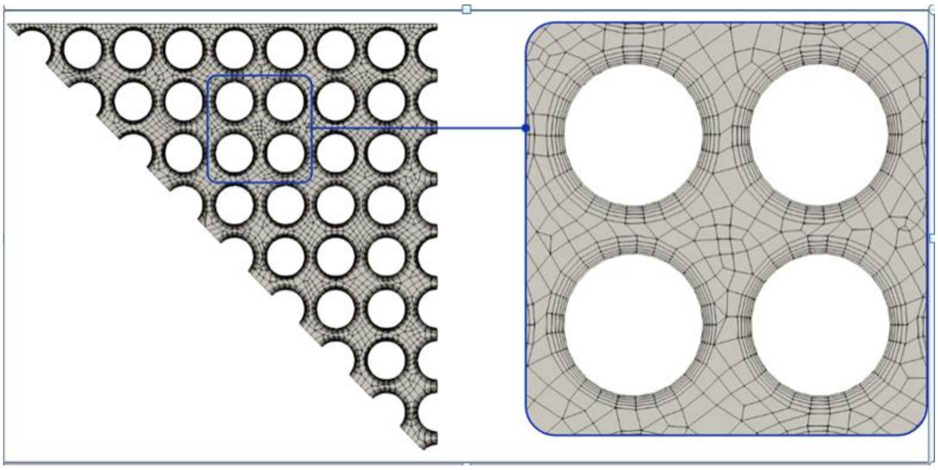


Fig. 4. Mesh dicretization for the simulations.

The mesh was produced based on a high discretization close to the rods (typical procedure in FVM for the higher temperature gradient region). The tab. 1. presents the main parameters of the mesh.

Tab. 1. Main mesh parameters.

Parameter	Value
Number of Axial layers	150
Number of Inflation Layers	5
Total Number of Cells	878400
Faces	2728806
Maximum Skewness	1.00516
Maximum Non-orthogonallity	64.1403
Average Non-orthogonallity	7.08751
Maximum aspect ratio	145.606

3.2 BOUNDARY CONDITIONS

The simulation using OpenFOAM requires initial boundary conditions for the solution of the differential equations solved during the calculations. The initial thermal physical properties field, boundary conditions, used for the steady state simulation are presented at the Tab. 2.

Tab. 2. Boundary conditions of the Most Powerful Assembly of SMART Nuclear Reactor [3].

Parameter	Value
Inlet Pressure (Bar)	150
Inlet Temperature (°C)	270
Inlet Coolant Velocity (m/s)	1.3
Assembly height (m)	2.166
Rods diameter (cm)	0.95
Outlet Pressure (Bar)	150
Outlet Velocity (m/s)	1.3
Outlet Temperature (°C)	270

The values of the boundary conditions are according to the SMART design parameters available on literature [3]. The thermophysical properties of the water were imposed in OpenFOAM for the simulations as fixed values for the present work as shown in Tab.3. The values applied was obtained from the literature for a reference temperature of $T = 290\ °C$ and a reference pressure $P = 150\ Bar$ [7].

Tab. 3. Thermophysical properties of water [7].

Property	Value
Specific Heat (J/kg°C)	5264
Density (kg/m^3)	744,6
Dynamic Viscosity (Pa.s)	$9{,}2.10^{-5}$
Prandtl number	0,87

4. RESULTS

The main results provided by the simulation include the coolant temperature, pressure and flow velocity fields. Fig. 5. presents the heat flux at the rods walls were the Z axis taken as the axial channel direction.

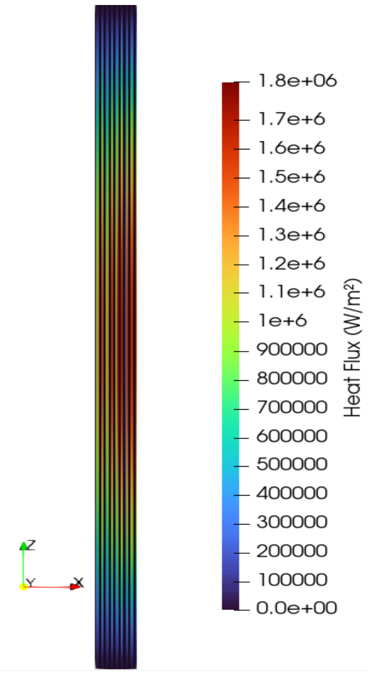
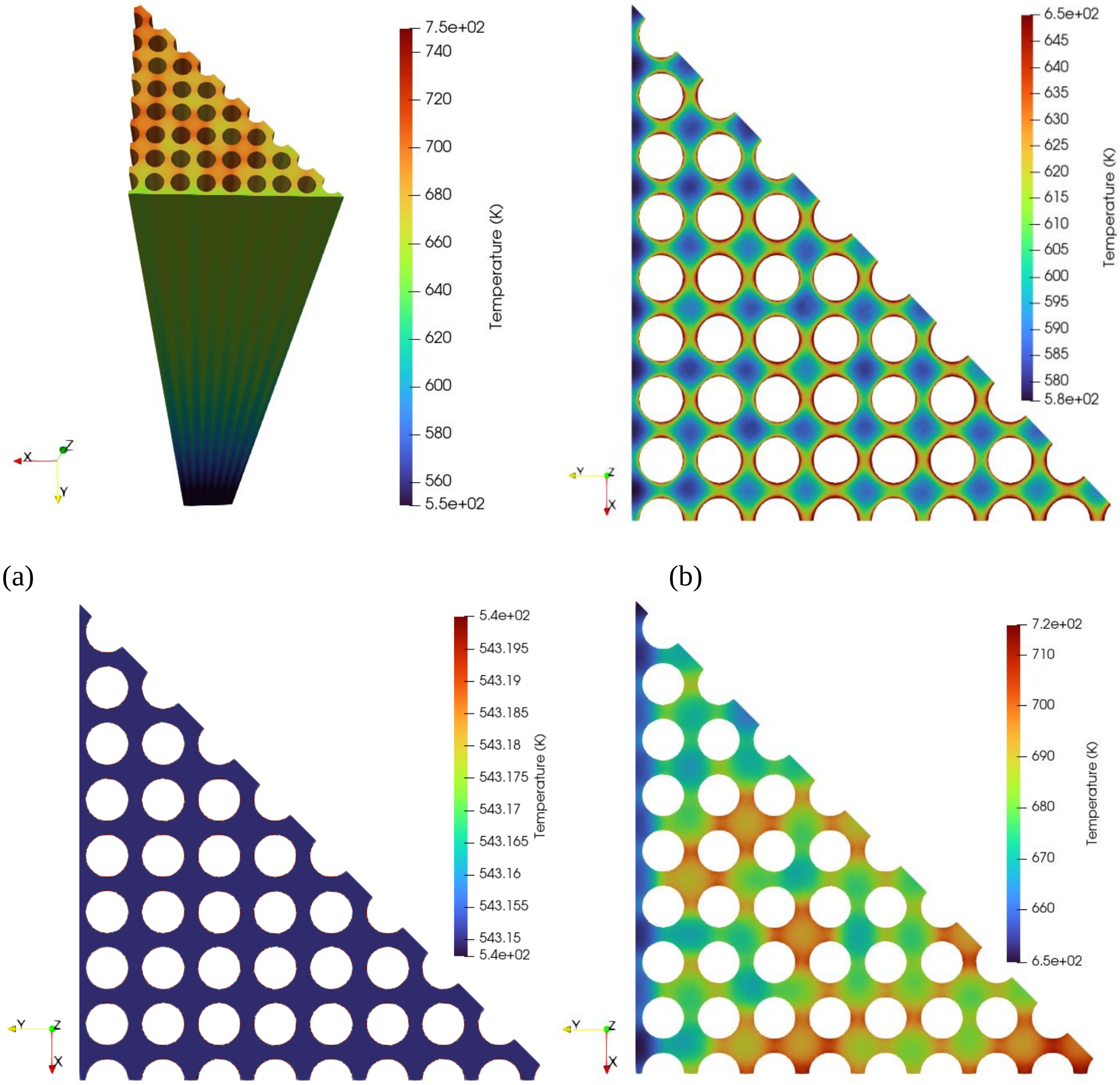


Fig. 5. Heat Flux Distribution on the 1/8 of the Most Powerful Assembly.

The distribution presented on the fig. 5. was produced by the axial sinusoidal boundary condition imposed for the fuel rods, where from the amplitude is modeled by the equation 4, for each fuel rod present in the assembly, according to the power distribution presented at Fig. 3. The temperature distribution for the coolant, provided by the conditions of operation impose for the simulations, are shown at Fig. 6. At Fig. 6-a., the entire assembly temperature is introduced and in the Fig. 6-b. a radial distribution profile approximately in the center of the assembly (1 meter height) is presented. The inlet temperature in Fig. 6-c. and the outlet temperature radial Fig. 6-d. profile are also detailed in Fig. 6.



(a) (b)

(c) (d)

Fig. 6. Temperature distribution: (a) Temperature all over the fuel assembly, (b) Assembly center (1m height) radial temperature distribution, (c) Inlet temperature and (d) outlet temperature.

The temperature distribution profile shown at Fig. 6. is according to the power profile density imposed at the rods (Fig. 5.). The temperature drop obtained, shown in Fig. 6-b, is proportional to the coolant distance of the rod, as expect by heat transference condition. In parallel, the simulation of the steady state of the SMART nuclear reactor provided the pressure profile and the flow velocity field. The Fig. 7. presents the distribution of the entire assembly drop of pressure (Fig. 7-a) and the velocity radial distribution (Fig. 7-b) for the condition simulated.
In the Fig. 7-a. the coolant presents a uniform pressure drop with axial height of the assembly and the flow velocity are according with the expected profile, taken into account the boundary condition that the coolant velocity must be equal to zero at the rods surfaces.

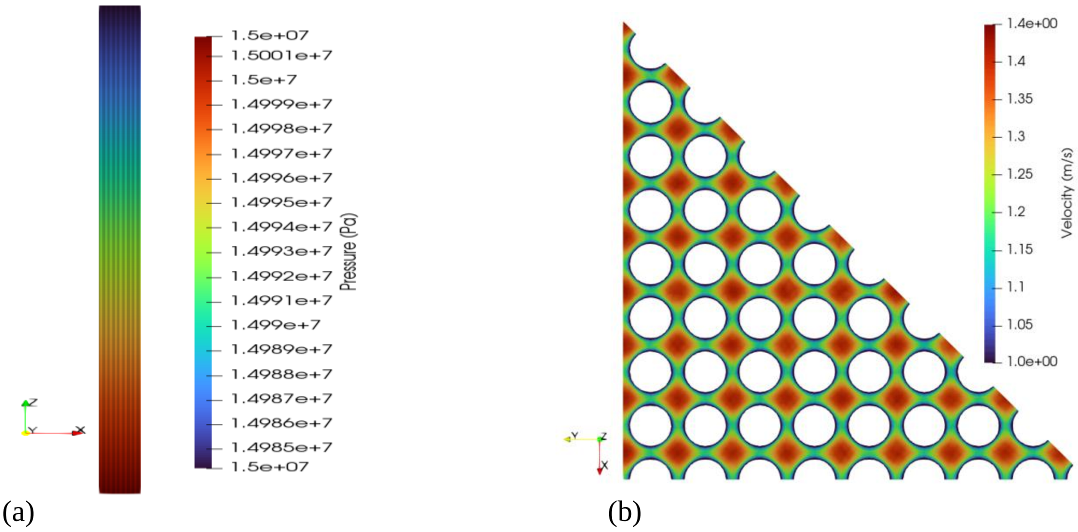


Fig. 7. Thermal physics properties calculated: (a) Pressure distribution in the SMART assembly and (b) Assembly center velocity radial distribution.

5. CONCLUSION

The present work was capable of perform the steady state analysis of the SMART reactor coolant most powerful assembly using OpenFOAM. Thermal physics properties were found for the fields of pressure, temperature and flow velocity with a typical PWR power distribution. Results profiles are according with the expect behavior for an increasing gradient of temperature near to the rods. In the next studies, the solid region (fuel, gap and cladding) will be simulated to calculate the heat transport in the rods region.

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