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An Overview of Molten Salt Reactors and Preliminary Calculations

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1. Introduction

MSR is characterized by using molten salt as a coolant, which could also contain fissile radioisotopes to work as nuclear fuel. The former type is called Solid Fuel MSR and the other is Fluid Fuel MSR. Both types could operate at thermal spectrum using graphite as moderator (mainly in the form of TRISO particles - also called VHTR), or in the fast spectrum (especially the Fluid Fuel MSR type). Although MSRs could use any fissile material, a concept using 233U and a blanket of 232Th salt (the Two-fluid concept), is of great interest nowadays given the 232Th abundance [10]. The initial concept of Molten Salt Reactors (MSR) was first developed in the United States, at Oak Ridge National Laboratory (ORNL) in the last half of the 1940s as part of a program that aimed to develop an aircraft that could overcome the limitations of jet-fueled bombers. Despite the development of intercontinental ballistic missile make this nuclear propulsion aircraft obsolete, its success led to the development of the Molten Salt Reactor Experiment (MSRE), an experimental molten salt reactor constructed by 1964 and operated until 1969 when ORNL proposed the first Molten Salt Breeder Reactor (MSRB), a Single Fluid design that introduces thorium with reprocessing online. Although not constructed, the concept established the bases for the current MSR, which was included as one of six concepts choose by the Generation IV International Forum at the beginning of this century (21st) [5, 6].

The growing interest is not limited to the academic level but, extends also to private companies that have commercial interests [8]. The main concepts under development with highlighted in recent years include is the FUJI- Japan, the Russian concept MOSART, the MSFR proposed by France, the Chinese concepts TMSR-LF (Liquid-Fuel) and the TMSR-SF (Solid-Fuel), the Liquid Fluoride Thorium Reactor (LFTR) developed by a private American company Flibe Energy and the TMSR by ThorCon company [6]. The attention in MSR comes from the several operational and safety advantages over solid fuel designs, how security against damage caused by the intrinsic nature of the fuel; elevated boiling point (much higher than the operational temperature); fissile concentrations are easily adjusted on a continuous basis, resulting in no excess reactivity and no need for burnable poisons; operation in low-pressure levels with higher efficiency; and the gaseous and volatile isotopes (Xe, Kr, He, among others) are continuously removed from fuel salt, contributing to reduce the risks associated with them in case of leakage [2]. Current research for MSR has as main focus the study of new alloys for corrosion resistance, the fuel reprocessing processes, which still lacks analysis, and mainly the development of calculation methodologies [10], since this is a technology that differs radically from what we have today, and which is the object of research of this work.

2. Methodology

For fixed fuel, the neutronics and thermal fluid dynamics methodologies does not differ from the usual methods already in use when considering other types of reactors. However, for fluid fuel MSRs, new methodologies should be investigated. Since the fuel is in motion, the equation for the delayed neutron precursors concentration must include a convective term to take in account its motion inside the reactor, which could affect the dynamics of the reactor (for example, the effective β calculation) [9]. The equations describing the neutronics has the form of Eq. 1, derivative from Transport Theory, and the precursor's concentrations by Eq. 2:

$$\begin{aligned} \frac{1}{V}\frac{\partial}{\partial t}\,\Phi\,(\underline{\mathbf{r}}\,,\,\underline{\Omega}\,,\,\mathbf{E}\,;\,\mathbf{t}) = -\underline{\Omega}\,\boldsymbol{.}\,\nabla\Phi\,(\underline{\mathbf{r}}\,,\,\underline{\Omega}\,,\,\mathbf{E}\,;\,\mathbf{t}) - \Sigma_t\,(\underline{\mathbf{r}}\,,\,\mathbf{E}\,,\,\mathbf{t})\;\Phi(\underline{\mathbf{r}}\,,\,\underline{\Omega}\,,\,\mathbf{E}\,;\,\mathbf{t}) + \\ \int_0^\infty d\mathbf{E}' \int_{4\pi} d\underline{\Omega}'\;\Sigma\,(\underline{\mathbf{r}}\,;\,\mathbf{E}'\to\mathbf{E}\,;\,\underline{\Omega}'\,\boldsymbol{.}\,\underline{\Omega}\,;\,t)\;\Phi\,(\underline{\mathbf{r}}\,,\,\Omega'\,,\,\mathbf{E}'\,;\,\mathbf{t}) + \\ \frac{\mathcal{X}(\mathbf{E})\;(1-\beta)}{4\pi}\int_0^\infty d\mathbf{E}' \int_{4\pi} d\underline{\Omega}'\;\nu\,(\mathbf{E}')\;\Sigma_f\,(\underline{\mathbf{r}}\,,\,\mathbf{E}'\,,\,t)\;\Phi\,(\underline{\mathbf{r}}\,,\,\underline{\Omega}'\,,\,\mathbf{E}'\,,\,\mathbf{t}) + \\ \beta\sum_{i=1}^{6}(\frac{\mathcal{X}_{i,d}}{4\pi})\,\lambda_i\;\mathcal{C}_i\,(\underline{\mathbf{r}}\,,\,\mathbf{t});\;\; i=1,2,...,G \end{aligned} \tag{1}$$

$$\begin{aligned} \frac{\partial}{\partial t}\,(\,\frac{\mathcal{X}_{i,d}\,(\mathbf{E})}{4\pi}\;\mathcal{C}_i\,(\underline{\mathbf{r}}\,,\,\mathbf{t})\,)\;+\;\nabla\,(\,\frac{\mathbf{u}\,\mathcal{X}_{i,d}\,(\mathbf{E})\,\mathcal{C}_i\,(\underline{\mathbf{r}}\,,\,\mathbf{t})}{4\pi}\,)\;=\;-\;\lambda_i\;\frac{\mathcal{X}_{i,d}\,(\mathbf{E})}{4\pi}\;\mathcal{C}_i\,(\underline{\mathbf{r}}\,,\,\mathbf{t}) \\ +\;\beta_i\;\frac{\mathcal{X}_f(\mathbf{E})}{4\pi}\int_\Omega d\underline{\Omega}'\int_0^\infty d\mathbf{E}'\;\nu\,(\mathbf{E}')\;\Sigma_f\,(\underline{\mathbf{r}}\,,\,\mathbf{E}')\;\Phi\,(\underline{\mathbf{r}}\,,\,\underline{\Omega}'\,,\,\mathbf{E}'\,,\mathbf{t})\,;\;\; i\;=\;1\,,\,2\,,\,3\,,\,...\,,\,6. \end{aligned} \tag{2}$$

Where V is the Neutron group velocity; $\Phi\,(\underline{\mathbf{r}}\,,\,\underline{\Omega}\,,\,\mathbf{E}\,;\,\mathbf{t})$ the Angular flux or Angular particle track length in space $\underline{\mathbf{r}}$, direction $\underline{\Omega}$, energy $\mathbf{E}$ and time $\mathbf{t}$; $\Sigma_t\,(\underline{\mathbf{r}}\,,\,\mathbf{E}\,,\,\mathbf{t})$ the Total macroscopic cross section; $\Sigma\,(\underline{\mathbf{r}}\,;\,\mathbf{E}'\to\mathbf{E}\,;\,\underline{\Omega}'\,\boldsymbol{.}\,\underline{\Omega}\,;\,t)$ is the Differential transfer cross section, that describe the probability that a particle with an initial energy $\mathbf{E}'$ and direction $\underline{\Omega}'$ undergoes a collision at $\underline{\mathbf{r}}$ in a time $\mathbf{t}$, resulting in a change of direction and energy; $\mathcal{X}(\mathbf{E})$ is the energy spectrum of the fission neutrons; $\nu(\mathbf{E}')$ is the number of neutrons emitted by fissions induced by neutrons with energy $\mathbf{E}$; $\Sigma_f\,(\underline{\mathbf{r}}\,,\,\mathbf{E}'\,,\,t)$ the Fission macroscopic cross section; $\mathcal{C}_i$ is the expected density of precursors in the i° group, in which i = 1, 2, ..., 6.; λ_i the decay constant; $\mathcal{X}_{i,d}$ is the normalized energy spectrum of the delayed neutrons; β the delayed neutron fraction; and β_i the fraction of the total neutrons emitted by fission expected that come from group i of precursors.

$\mathbf{u}$ is the velocity field of moving salt, in m/s, which can be obtained using the principles of Fluid Mechanics and its equations (Mass, Moment and Energy). Since, even in a steady-state, the physical density, and consequently the atomic density, depends upon the position ($\mathbf{r}$) inside the reactor core, the core and the molten salt (fuel and coolant fluid) could not be considered as homogeneous. In this manner, the solution to find the power distribution is a complex multi-physics task, yet to be completely solved.

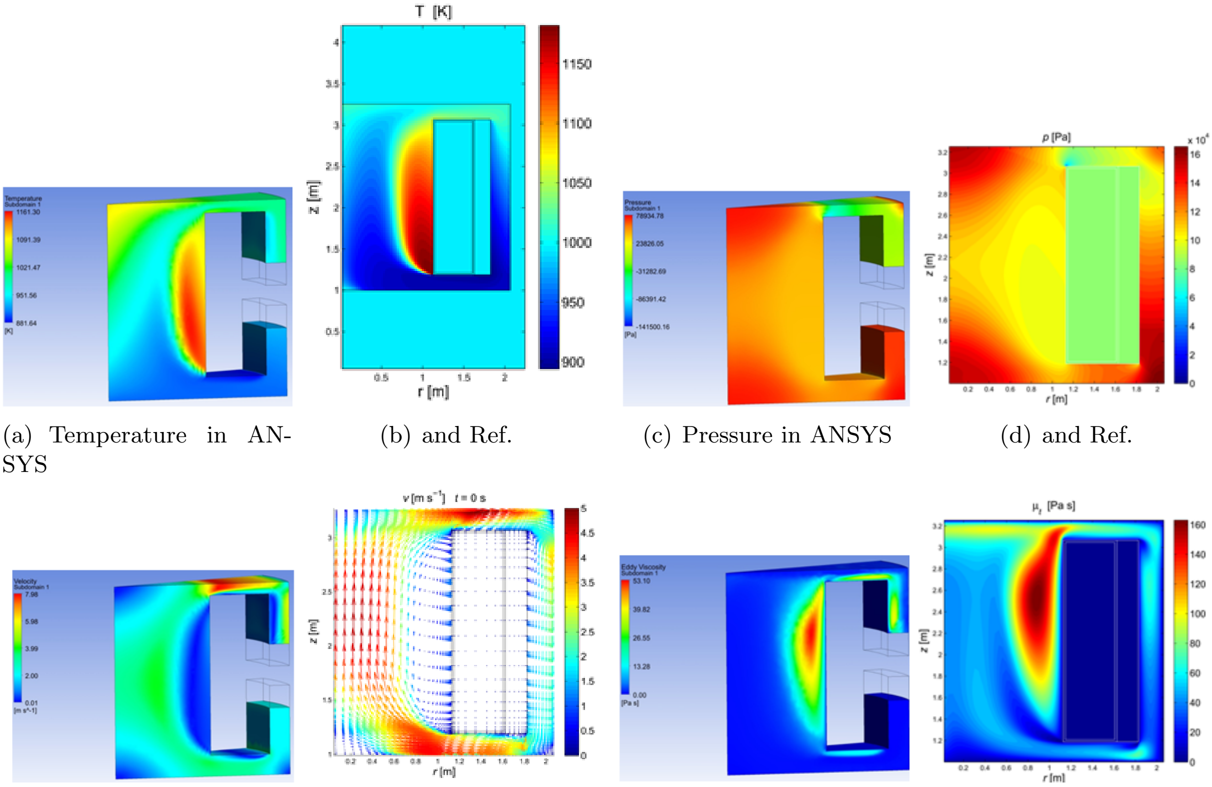
This work intends to solve Eq. 1 in steady-state, using the Código XSDRN [4] in R-z geometry to simulate the EVOL Benchmark [3] and to calculate the power density distribution which is going to be the input for the CFD code, ANSYS-CFX® [1], using k-ϵ and SST turbulence models. However just as a first approach we assume that the core is a homogeneous bare cylinder and that the power distribution is given by Eq. 3:

$$\mathbf{q}'''\,(r\,,z)\;=\;\frac{3.63\,P\,J_0\,(4.405\,\frac{r}{R})\,cos(\frac{z}{H})}{Vol} \tag{3}$$

In which P is the reactor power, J_0 is the Bessel function of the zero-order, r is azimuthal position at the core with reference to the core axis, R is the core reactor radius, z is the height position with respect to the core center, H is the core height and Vol is core volume.

3. Results and Discussion

The objective is to reach as much as possible the values obtained by Linden [7] as well as a similar profile for the term fluid dynamics properties (temperature, pressure, velocity, etc). The results obtained in the validation of models are shown graphically in Fig. 1.



(a) Temperature in ANSYS (b) and Ref. (c) Pressure in ANSYS (d) and Ref.

(e) Velocity in ANSYS (f) and Ref. (g) Viscosity in ANSYS (h) and Ref.

Figure 1: k-ϵ model - Validation.

As could be observed, the results obtained with ANSYS-CFX® are very similar to the reference. Despite it, additional efforts should be spent in the investigation of the discrepancies found in pressure and eddy viscosity profiles as well as in the values of the properties evaluated.

Other simulations were performed, changing turbulence model from k-ϵ to SST and considering the variation of density and thermal conductivity. It was noticed that occurred slight changes in the temperature, velocity, and eddy viscosity profiles while more significant changes occur in the pressure profile. Since the SST model treats more suitable the turbulence within and out of the boundary layers or in the locals where the gradients are elevated, the results shown were considered more accurately. Despite it, additional simulations are required to investigate the sensibility regarding the change in turbulence models.

4. Conclusions

The Molten Salt Reactors attend the requirements of the Generation IV reactors class and draws attention to its several operational and safety advantages over solid fuel designs. Besides, it could be the first reactor to introduce the thorium fuel cycle sustainably, adding even more benefits. There are a lot of open research opportunities, in the technology of the salt, reprocessing technologies, and mainly in the calculation methodologies to simulate this system. As part of the present paper, some preliminary results were presented. They consist of qualitative analyses obtained with numerical CFD simulations, and which were compared with results found in the literature. The preliminary results demonstrate a good agreement within the benchmark work, which means that the models selected, and boundary conditions are suitable to simulate the fluid dynamics of this kind of reactor. Despite it, additional efforts should be spent in the future aiming to improve the accuracy of the results, mainly by producing a power density with results from a transport code.

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