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# NEUTRONIC ANALYSIS OF THE CORE OF A MOLTEN SALT REACTOR (MSR)

**Tororó, Magda M. A.[1]; Cabral, Ronaldo G.[1]; Oliveira, Cláudio L.[1]; Vellozo, Sérgio O.[1]**

[1] Military Institute of Engineering (IME), Rio de Janeiro, Brazil.,
magda.azevedo@ime.eb.br, eng.magdamaia@gmail.com

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## ABSTRACT

This study aims to evaluate, analytically, the physics of a **Molten Salt Reactor – (MSR)**. The set of steady-state equations, including one group delayed neutron, are solved. The fuel velocity (**u**) is considered within the framework of the diffusion theory. To solve these equations, the Finite Sine Fourier Transform (FSFT) is applied under specific boundary conditions. Understanding the neutron distribution, core reactivity and energy production is crucial for the development of the MSR design. However, MSR neutronics presents unique challenges, such as the corrosion of core materials by molten salt and the need for more accurate computational models. This paper describes a detailed analytical approach for neutron and precursor concentration. The results obtained using the FSFT offer a significant advantage over direct methods and show agreement with the reference work.

## 1. INTRODUCTION

Nuclear power plants have long faced safety concerns, primarily stemming from accidents like Three Mile Island, Chernobyl, and Fukushima, as well as the challenge of nuclear waste management. Molten Salt Reactors (MSRs), a type of Generation IV reactor, offer potential solutions to these issues. Countries worldwide, including the USA, China, France, India, the UK, Canada, Indonesia, Denmark, Japan, and others, are investing in MSR development as a means to transition to carbon-free energy sources. This reactor concept, originally proposed in the 1950s and 1960s, is gaining renewed attention. MSRs are considered a promising technology due to their high operating temperatures, energy efficiency, intrinsic safety features, and fuel flexibility. [**3, 5, 8-10, 13, 16-17**]

The neutronics of the MSR core are central to its safe and efficient operation. Key areas of focus include:

- **Moderator:** Graphite is commonly used in MSRs due to its low neutron absorption and compatibility with molten salt;
- **Fuel:** MSR can use various fuels, such as uranium, thorium, or plutonium, dissolved in molten salt;
- **Neutron Spectrum:** The neutron spectrum influences the rate of nuclear reactions, including fission, radioactive capture, and scattering;
- **Kinetic Parameters:** Control rods are used to regulate the MSR core's reactivity by absorbing neutrons and

- **Safety Analysis:** Neutronics are essential for safety analysis, encompassing accident studies and criticality analysis.

### 1.1. Brief description of MSR

In a MSR (for example liquid fluoride), the fuel, in the form of salt, contains the fissile material and is transported through a metal pipe to the reactor vessel, where structures, usually graphite, act as neutron moderators, increasing the likelihood of fissions occurring. These reactions deposit energy in the form of heat within the salt itself, which increases in temperature and flows out of the vessel, where nuclear reactions are very unlikely due to the absence of a moderator (with the exception of Fast Reactors). The fluid then goes to heat exchangers in which it heats a coolant in an independent system and then continues the cycle back to the reactor vessel. The coolant heats the turbine's working fluid, where helium gas is generally used. The heat transfer properties of salt, combined with the high working temperatures, allow gases to be used in the turbine, which is advantageous in terms of thermal efficiency. In addition to this range of advantages, the MSR has inherent safety, meaning that it does not require operator action or electricity to shut down safely in the event of an emergency. [**3-5, 8-10, 13, 16-17**]

## 2. METHODOLOGY

### 2.1. The MSR under diffusion model

The diffusion equation for neutrons is a powerful tool for analyzing the behavior of neutrons in a fissionable medium. By understanding the meaning of each term, it is possible to understand the physical processes involved and apply this equation in various areas of nuclear engineering. [**1-2, 4, 6-7, 12, 15**]. The **figure 1** shows the unit cell used for calculations.

#### 2.1.1. Analysis of the Meaning of the Terms of the Neutron Diffusion Equation: [1-2, 4, 6-8, 11-12, 14]

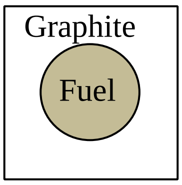


**Fig. 1 Unit cell of MSR core for homogenization. [4]**

For the fuel fluid homogenized assembly, the equations are:

$$D\nabla^2\phi_f(z) + [\nu\Sigma_f(1-\beta) - \Sigma a(z)]\,\phi_f(z) + \lambda C_0(z) = 0 \quad (1)$$

$$u\frac{dC_0(z)}{dz} = \beta\nu\Sigma_f\phi_f(z) - \lambda C_0(z) \quad (2)$$

And for core reflector graphite region:

$$D_G\nabla^2\phi_G - \Sigma_a^G\phi_G = 0 \quad (3)$$

Where:

- $D_f$ and $D_G$ is the fuel and graphite diffusion coefficient, respectively;
- $\phi_f \equiv \phi_f(z)$ and $\phi_G \equiv \phi_G(z)$ is the fuel and graphite scalar neutron flux, respectively;
- $\nu$ is the average number of neutrons emitted per fission;
- $u$ is the fuel axial velocity;
- $\Sigma_f$ is the fission macroscopic cross section;
- $\beta$ is the fraction of delayed neutrons;
- $\Sigma_a^f$ and $\Sigma_a^G$ is the fuel and graphite absorption macroscopic cross section, respectively;
- $\lambda$ is the decay constant of one group delayed neutrons;
- $C \equiv C(z)$ is the delayed neutron precursors concentration;
- $k$ is the effective multiplication factor.

Having the following Axial Boundary Conditions:

$$\begin{cases} \phi_f(z=0)=0 \\ \phi_f(z=L)=0 \\ \frac{dC}{dt}(t=0)=0 \end{cases} \qquad 0 \leq Z \leq L \qquad (4)$$

**2.2. Steps to a pply the FSFT to the Diffusion Equation:**

In the intended evaluation of the neutronics project, this set of static equations [1-3] will be used, from neutronics applied to an energy group and a delayed neutron group for MSR for a reactor with fuel velocity $u$ in the theory of diffusion of a group with a delayed neutron group are read as [**1-2, 4, 6-7, 13-15**]:

We start from equation (**1**) with appropriate Laplacian.

$$D\frac{\partial^2}{\partial z^2}\phi + \left[\frac{\nu\Sigma_f}{k}(1-\beta) - \Sigma a\right]\phi + \lambda C = 0 \qquad (5)$$

The **FSFT** [**14**] is applied to the second derivative term in (**1**)**:**

$$\mathcal{F}_S\left\{\frac{\partial^2\phi}{\partial z^2}\right\} = -\frac{n^2\pi^2}{L^2}\phi_s + \frac{n\pi}{L}\{\phi(0) - \phi(L)\cos(n\pi)\} = -\frac{n^2\pi^2}{L^2}\phi_s \quad n = 1, 2, \ldots \qquad (6)$$

where the boundary conditions for $\phi$ was employed.

Having $K_1 = [\nu\Sigma_f(1-\beta) - \Sigma a]$ the flux ($\phi_S$ ) is isolated as follows: $\phi_s = \frac{\lambda}{\{D\frac{n^2\pi^2}{L^2} + K_1\}}C_s$ **(7)**

Having $K_2 = \beta\nu\Sigma_f$ has been $\frac{dC}{dt} = K_2\phi(z) - \lambda C$ **(8)**

The **FSFT** [14] applied in (**8**) with (ϕ) is substituted we have $C_s = K_2\phi_s - \lambda C_s$ **(9)**

Having $k_3 = \left(\frac{K_2\lambda}{\{D\frac{n^2\pi^2}{L^2} + K_1\}} - \lambda\right)$ has been $\frac{\mathrm{d}}{\mathrm{dt}} C_s = K_3 C_s$ it is solved by 1st Order ODE,

generating the Constant $K_n$ : $C_s = K_n e^{K_3 t}$ for infinite solutions depending on "n" **(10)**

$K_n$ is calculated by applying the boundary conditions then the **Inverse Fourier Transform** [**14**] is applied:

$$\phi(z,t) = \frac{2}{L}\sum_{n=1}^{\infty} \phi_s \sin\left(\frac{n\pi z}{L}\right) \quad \therefore \quad \phi(z, t = 0) = \sin\left(\frac{n\pi z}{L}\right) \text{ (normalized)} \qquad (11)$$

$$\mathcal{F}_S\left\{\sin\frac{\pi z}{L}\right\} = \int_{z=0}^{L} \sin\frac{\pi z}{L} * \sin\frac{n\pi z}{L} dz = \frac{L}{2} \leftrightarrow (\mathbf{n = 1}) \text{ and for } (\mathbf{n \neq 1}) = \mathbf{0} \qquad (12)$$

## 3.RESULTS AND DISCUSSIONS

We take equation (**2**) as start point with the first Fourier term, $n = 1$, for the flux. Equation (**2**) can be integrated resulting:

$$C(z) = \frac{\beta\nu\Sigma_f}{ku} e^{-\frac{\lambda}{u}z} \int_0^z e^{-\frac{\lambda}{u}w} \phi(w)dw \qquad \phi(w) \sim \sin\left(\frac{\pi}{L}w\right) \qquad (13)$$

$$C(L) = \frac{\beta\nu\Sigma_f}{ku} \frac{2\pi}{L\left[\left(\frac{\lambda}{u}\right)^2 + \left(\frac{\pi}{L}\right)^2\right]} \qquad (14)$$

Equation (**14**) can easily be solved. The **Fig.2** shows the plot solution for $C(z)$ for four different fuel salt velocities. Note that, as predicted by equation (**14**), $C(L)$ is no longer zero.

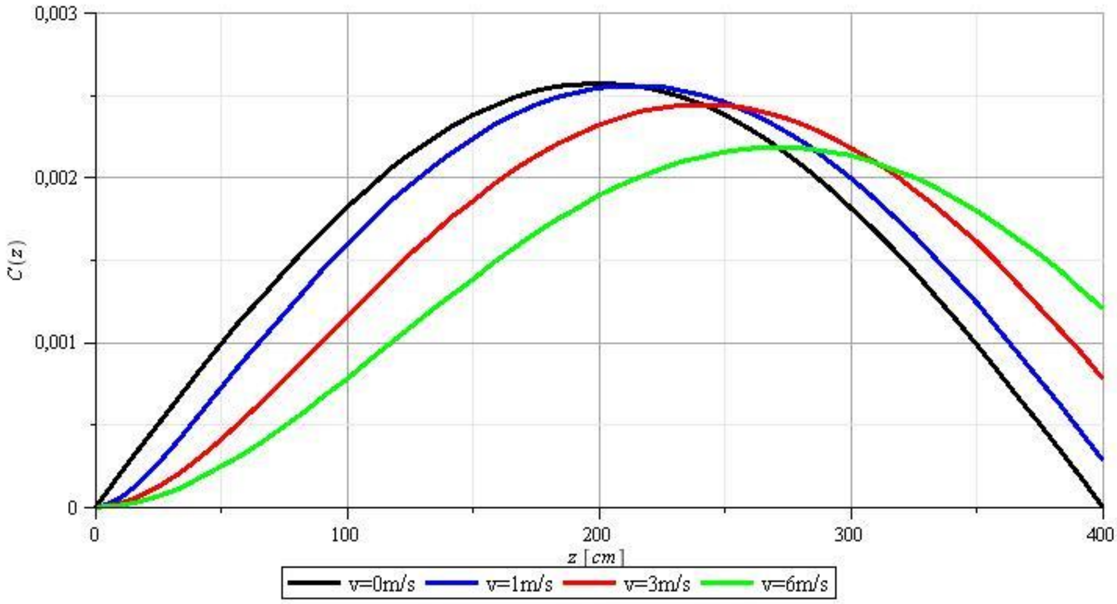


**Fig. 2 – Precursor axial distribution.**

Reference [**1**] was followed for reactor calculation. The composition of the fluid fuel can be seen in Table 1 below.
Segue-se a referência [1] para cálculo do reator. A composição do combustível fluido pode ser vista na Tabela 1 abaixo.

**Tab 1 – Fraction volume**

| Salt composition | % / mol |
|---|---|
| $Li^7F$ | 71.7 |
| $BeF_2$ | 16 |
| $ThF_4$ | 12 |
| $UF_4$ | 0.3 |
| Total | 100 |

The SCALE-6.2 code system was used to generate the macroscopic cross sections at 1 energy group and the multiplication factor $k$.

**Tab 2 – X-sections**

| Macroscopic X-sections | $1/cm$ |
|---|---|
| $\Sigma_{tr}$ | 0.0348579 |
| $\nu\Sigma_f$ | 4.03788e-2 |
| $\Sigma_a$ | 3.87067e-3 |
| $\boldsymbol{k}$ | **1.046957** |

This $\boldsymbol{k}$ value differs from the reference [1] by **0.84%.** The **Keef** generated in reference [**1**] by the SCALE-5.1 code system is **1.03824** and the one found (**Tab 2**) is **1.046957**. Applying:

$$\mathbf{100} * \left|\frac{\boldsymbol{k-Keff}}{\boldsymbol{Keff}}\right| = \left|\frac{\mathbf{1.046957} - 1.03824}{1.03824}\right| \cong \mathbf{0,84\%} \quad \mathbf{(15)}$$

## 4. CONCLUSIONS

This work shows how a simple model can be used preliminarily to MSR design. The X-sections was generated by SCALE code system with ENBF-B VII library. The agreement on $k$ by less than 1% is a strong motivation to test simple models. This procedure establishes safety and reliability in complex design. [**1-2, 4**, **8-12, 14-15**]

## 5. ACKNOWLEDGMENT

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