

Three-dimensional space and time-dependent analysis of molten salt reactors

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Abstract

This paper presents the development of a 3D time dependent calculation scheme for graphite moderated molten salt reactors (MSRs). The neutronics was modelled by diffusion and the delayed neutron precursor equations were extended with a convection term to take into account the drift of precursors. To account for the heat transfer in the fuel, a 1D heat convection equation was applied. All fuel channels were calculated individually. Heat transfer in the moderator was described by the 3D heat conduction equation, thus thermally connecting the fuel channels. The computational scheme was applied for the Molten Salt Reactor Experiment (MSRE). Steady-state and time dependent simulations were carried out, including the investigation of a local fuel channel blocking event where the space dependent effects are especially relevant.

KEYWORDS: *molten salt reactor; delayed neutron precursor; coupled reactor physics - heat transfer calculation*

1. Introduction

In recent years, the scientific interest renewed toward molten salt reactors (MSR) due to their promising capabilities in the field of transmutation. The history of the development and research dates back to the 1960's when the Molten Salt Reactor Experiment (MSRE) was built and operated at the Oak Ridge National Laboratory [1]. Although the research program was cancelled after a sh

ort period, it proved the viability of the molten salt technology and provided experimental data for the development of molten salt reactors.

The unique feature of these reactors is that the fuel is dissolved in molten salt which serves as coolant as well. Since the fuel is circulated in the primary loop of the reactor, delayed neutron precursors move in the core and emit delayed neutrons at a different location than that of the fission. Moreover, they can decay outside the core. Therefore, the kinetic behavior of MSRs is essentially different from the behavior of conventional reactors utilizing solid fuel.

The heat transfer features of the molten salt reactors also substantially differ from that of conventional fixed-fuel reactors. Contrary to fixed fuel reactors, most of the fission heat is deposited directly into the coolant (i.e. fuel). However, in case of graphite moderated thermal reactors, a reasonable fraction (4-7%) of the heat is deposited in the graphite due to the neutrons slowing down and the gamma heating of the core. Hence the molten fuel cools the moderator.

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In the recent past, according to the increasing demand in the field of nuclear waste management, an increasing number of publications have appeared discussing the physics of molten salt reactors. Some of those discuss the theoretical aspects of the physics of fluid fuel reactors [2-4]. Others report the time dependent, one dimensional simulation tools and results for coupled neutronics and heat transfer calculations [5,6]. A benchmark of such calculations was carried out in the frame of the MOST project [7]. However, no 3D, coupled neutronics-heat transfer investigation of MSRs has been published until now. This paper is devoted to present the work performed to develop a 3D and time dependent calculation scheme including feedback effects.

2. Model

2.1. Neutronics

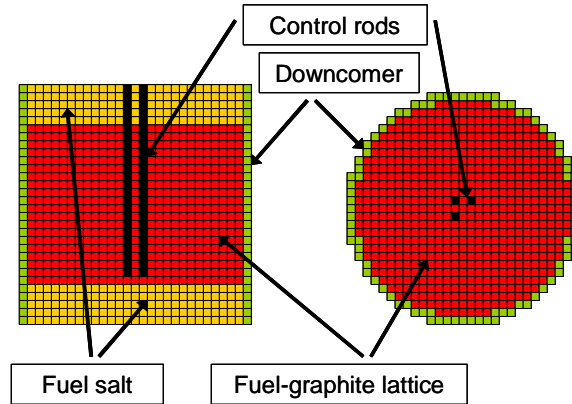
A modified version of the in-house developed DALTON diffusion code is used to solve the 3D diffusion and precursor equations. The precursor equations are extended by convection terms [2]:

$$\frac{1}{v_g} \frac{\partial \Phi_g}{\partial t} = \nabla D_g \nabla \Phi_g - \Sigma_g^r \Phi_g + \sum_{g \neq g'}^G \Sigma_{g \rightarrow g'}^s \Phi_{g'} + \chi_p \sum_{g'}^G (1 - \beta) \nu \Sigma_{g'}^f \Phi_{g'} + \sum_i^I \lambda_i \chi_d C_i, \quad (1)$$

$$\frac{\partial C_i}{\partial t} = \sum_{g'}^G \beta_i \nu \Sigma_{g'}^f \Phi_{g'} - \lambda_i C_i - \nabla u C_i,$$

where standard notation is used and u denotes the velocity of the fuel. The fuel velocity is assumed to be parallel to the axis of the reactor core. The concentrations of the precursors are calculated in the external loop of the reactor too.

Figure 1: Vertical and horizontal cross section of the diffusion model of the MSRE



The equations were discretized using the method of finite volumes:

$$\frac{\partial C_i}{\partial t} = -\nabla u C_i \quad (2)$$

$$\Delta V \frac{\partial C_i^k}{\partial t} = u^{k-1/2} \Delta A^{k-1/2} C_i^{k-1/2} - u^{k+1/2} \Delta A^{k+1/2} C_i^{k+1/2}$$

In order to reduce the numerical dispersion occurring in the precursor concentrations, a total variation diminishing (TVD) scheme was employed. On the other hand, to estimate the surface concentrations, the van Leer limiter was applied [8]:

$$C_i^{k+1/2} = C_i^k + \frac{1}{2} \Psi \left(\frac{C_i^{k+1} - C_i^k}{C_i^k - C_i^{k-1}} \right) (C_i^k - C_i^{k-1}) \quad (3)$$

$$\Psi(r) = \frac{r + |r|}{1 + r}$$

The concentration on the opposite wall of the control volume ($C_i^{k-1/2}$) can be estimated analogously.

The model of the MSRE was built in the modified in-house developed DALTON code. The horizontal and vertical cross section of the model can be seen in Fig. 1. The core consisted of a lattice of fuel salt in channels of graphite slabs. The graphite served as moderator as well as flow baffle. The core was straddled between two layers of fuel salt without graphite. The control rods were modeled by internal albedo boundaries. Although the DALTON code is able to perform calculations in cylindrical coordinates, Cartesian coordinates were applied in order to maintain the original rectangular structure of the MSRE. The circumference of the core was consequently approximated by a series of rectangular slabs. Homogenized 8-group neutron cross sections were produced with the aid of the code package SCALE [9].

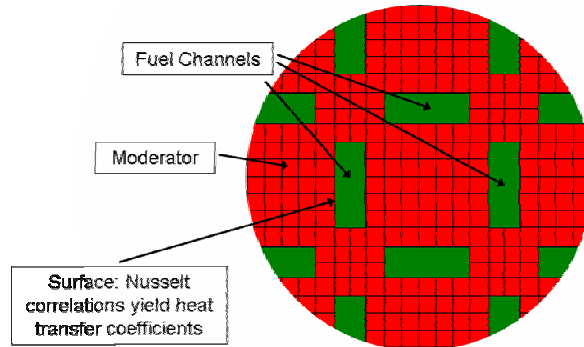
2.2. Heat transfer

The heat transfer model was divided into two parts. Fuel channels were described individually by the 1D (parallel to the axis of the core) heat convection equation (Eq. 4):

$$\rho c_p \frac{\partial T}{\partial t} = c_p \frac{\partial}{\partial z} uT + p_f, \quad (4)$$

where standard notation is used and p_f is the power density deposited in the fuel. Uniform velocity profile was assumed in the core and the thermal expansion was neglected. The primary loop of the reactor was implemented by a single pipe and the heat exchanger was modelled with a heat removal term in one control volume of the primary loop.

Figure 2: A detail of the horizontal cross section of the thermal model of the MSRE



The temperature distribution of the surrounding moderator slab was calculated by applying the 3D heat conduction equation:

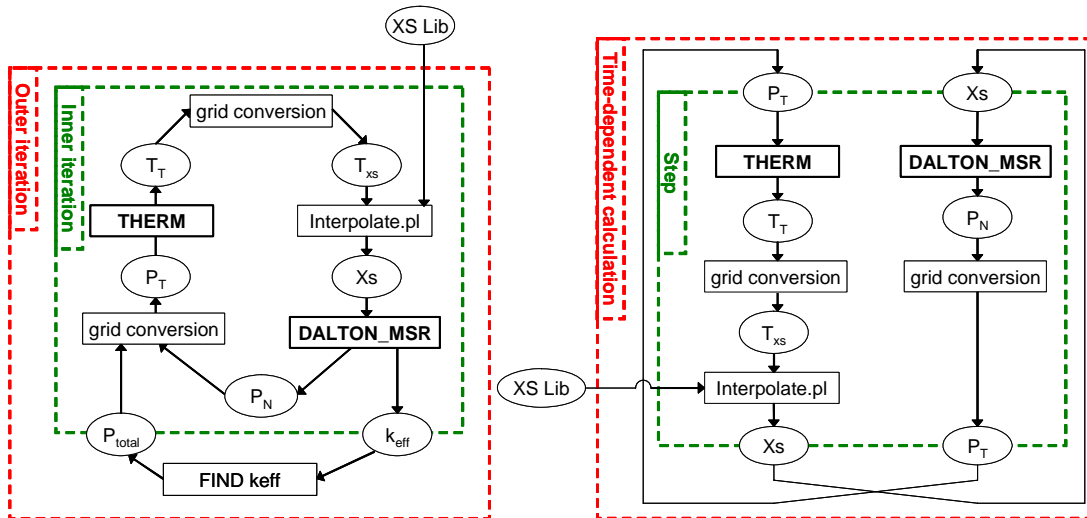
$$\rho c_p \frac{\partial T}{\partial t} = \nabla \lambda \nabla T + p_m, \quad (5)$$

where p_m is the power density deposited in the moderator. The connection between equation (4) and (5) was set up using heat transfer coefficients based on Nusselt correlations. In order to appropriately calculate the surface temperature of the moderator and the heat transfer between the moderator and the fuel, high spatial resolution is required in the model of the moderator. A small detail of the horizontal cross-section of the MSRE's heat transfer model is displayed in Fig. 2. In normal operational conditions the Reynolds number was $Re \approx 900$ and the Nusselt number was $Nu \approx 22$.

2.3. Coupling

The coupling of the neutronics and thermal calculations is based on the iteration of mutual exchange of the power distribution and temperature fields. The scheme for steady-state and time-dependent calculations is shown in Fig. 3.

Figure 3: The coupling scheme of the steady-state (left) and the time-dependent (right) calculations



In the case of steady state calculations, the inner loop (marked with a green rectangle) was responsible for the calculation of neutron flux, temperature field and k_{eff} at a given total power. The outer loop (red rectangle) calculated the total power required for a critical state ($k_{eff} = 1$, **FIND_keff** module in Fig. 3). In the case of time-dependent calculations, an explicit Euler method was applied. Both schemes were based on a neutron cross section library containing a matrix of cross section sets for various fuel and moderator temperatures with a temperature step of 100 K. The interpolation of the cross sections was carried out on a spatial mesh containing 30x9x9 regions.

3. Results

3.1. Feedback coefficients

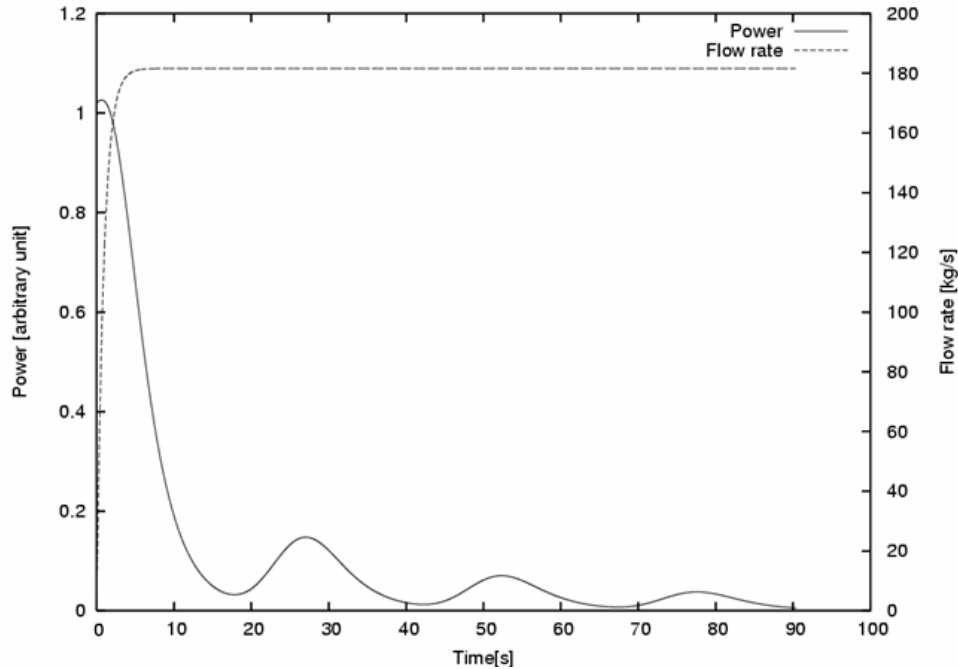
Since the feedback coefficients play a key role in coupled neutronics-thermal calculations, the fuel and moderator feedback coefficients of the MSRE were calculated. The results are displayed in Tab. 1. The coefficients were calculated with uniform fuel and moderator temperature distributions, in accordance with the experiments [10].

Table 1: Fuel and moderator temperature feedback coefficients of the MSRE

	Fuel temp. coefficient	Moderator temp. coefficient
Calculation	-9.77 pcm/K	-6.31 pcm/K
Measurement [10]	-8.46 pcm/K	-4.68 pcm/K
Difference	14 %	26 %

3.2. Pump startup transient

Figure 4: Power as a function of time during pump startup transient



Pump startup is the most interesting zero-power transient of molten salt reactors, since the change of the drift of delayed neutron precursors affects heavily the neutron economy of the reactor. A pump startup transient case of the MSRE was calculated. At the beginning, the reactor power was sustained and the fuel was stationary. At 0 s the fuel pump was started and the fuel velocity was increased to the nominal value by an exponential rise of $u(t) = u_0 \left(1 - e^{-\frac{t}{\tau}} \right)$, where

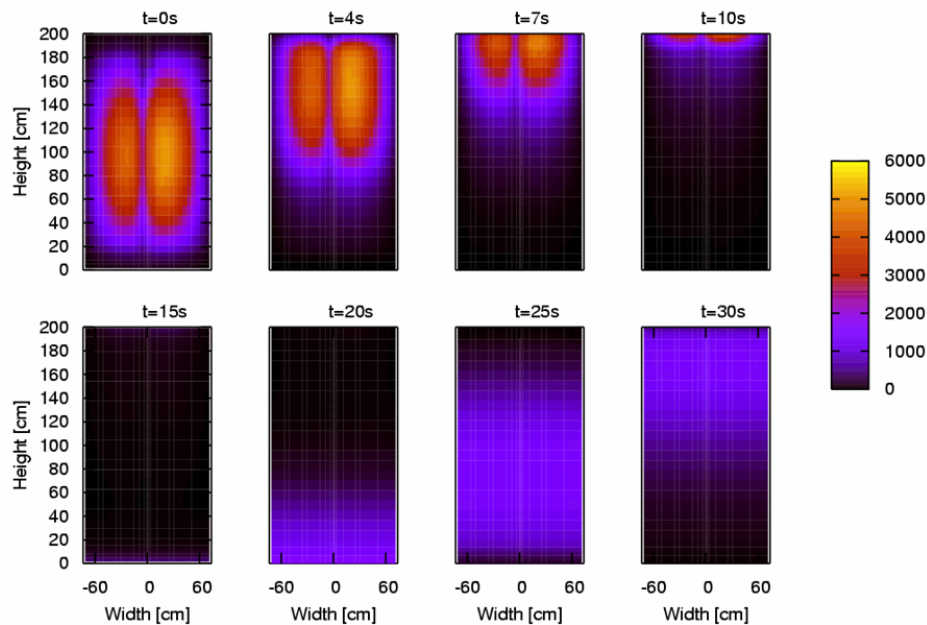
τ is the characteristic time of the hydraulics and pump inertia of the primary circuit. The calculation was performed using a two group test library.

Fig. 4 shows the power of the reactor and the mass flow rate of the fuel as a function of time, while Fig. 5 shows the vertical sections of the concentration of the precursor group with longest half-life, at various times. The section plane is close to the vertical midplane of the core.

It is visible that oscillations are present in the power of the reactor. These oscillations can be attributed to the fact that up to $t = 14$ s practically all precursor nuclei have exited the core, causing the collapse of the neutron flux. Later, the remaining precursors re-enter the core, providing a quasi source of neutrons and increasing the power for a short time (see Fig. 5).

A small power peak can be observed at the very beginning of the simulation. This can be attributed to the slightly positive initial reactivity (~ 15 pcm) of the reactor.

Figure 5: Concentration of the precursor group with longest half-life, at various times



3.3. Steady-state coupled calculation

By adjusting the control rods, the power was set to 8.59 MW which is close to the maximal power of the MSRE (10 MW planned, 7.3 MW realized), thus the normal operating state of the MSRE was calculated. The thermal neutron flux is shown in Fig. 6, where the influence of the control rods can be observed. Fig. 7 displays a small detail of the horizontal section of the temperature field. It is visible that the fuel temperature is substantially lower than that of the nearby graphite moderator, and the temperature of the graphite moderator has local maximum regions between two individual fuel channels while the moderator temperature has a large scale gradient (see Fig. 8). Fig. 8 and Fig. 9 show the horizontal and vertical cross sections of the temperature field.

Figure 6: Thermal neutron flux in normal operating state of the MSRE. Power: 8.59 MW
(Three horizontal and one vertical cross section, distorted plots)

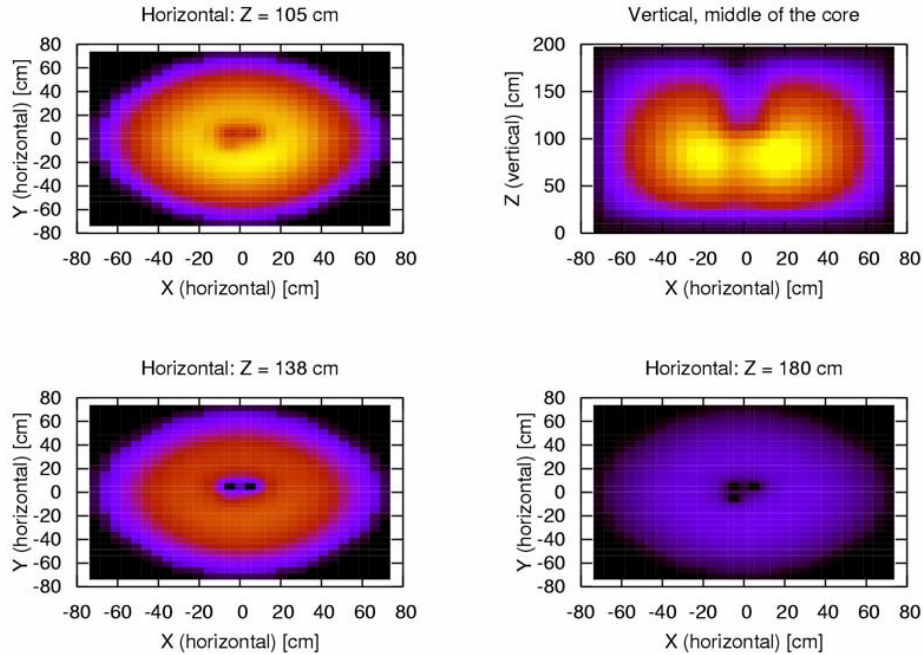


Figure 7: Detail of the horizontal cross section of the temperature distribution in normal operating state of the MSRE

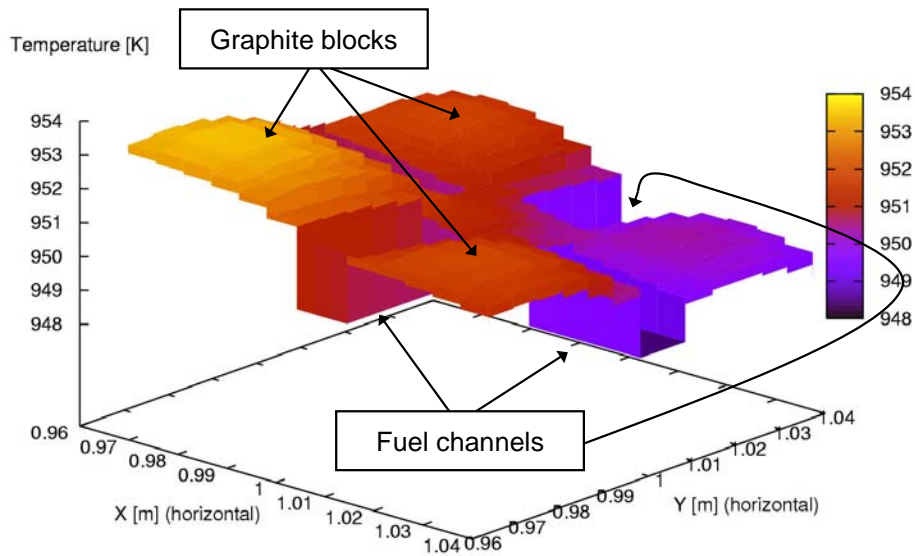


Figure 8: Horizontal cross section of the temperature field in normal operating state of the MSRE (left: moderator temperature, right: fuel temperature)

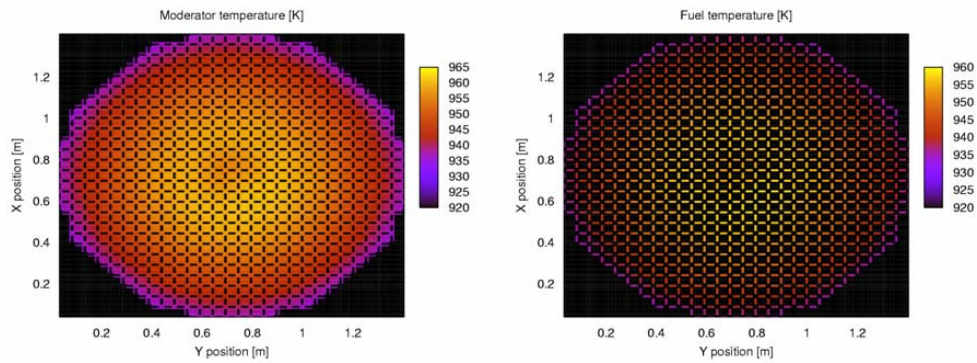
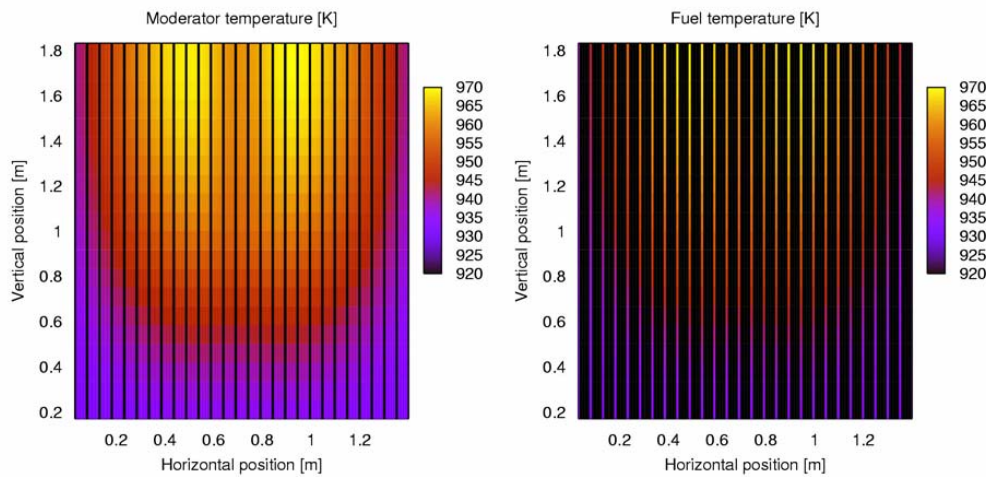


Figure 9: Vertical cross section of the temperature field in normal operating state of the MSRE (left: moderator temperature, right: fuel temperature)



3.4. Transient due to local blocking of fuel channels

In order to demonstrate the 3D capabilities of the code, a transient induced by a local blocking of fuel channels was simulated. A total of 45 fuel channels were blocked, which was simulated by reducing their mass flow to 20%. The total mass flow of the core was maintained. The affected channels can be recognized in Fig. 11. The result of the previous calculation (see section 3.3.) was used as stationary initial state for the simulation, and the blocking occurred at 0 s. The power as a function of time is displayed in Fig. 10. Starting from 8.59 MW, after a valley of 7.6 MW the power finally sustains at 8.32 MW. The newly evolved steady-state temperature field can be seen in Figs. 11 and 12. It should be noted that heat conduction gains significance in the cooling of the blocked volume since the gradient of the temperature field is an order of magnitude higher than in the case of the unblocked core. (By comparing Figs. 9 and 12)

Figure 10: Power as a function of time during the channel blocking transient

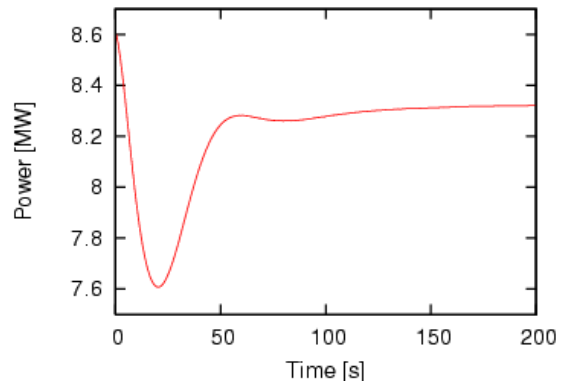


Figure 11: Power as a function of time during the channel blocking transient

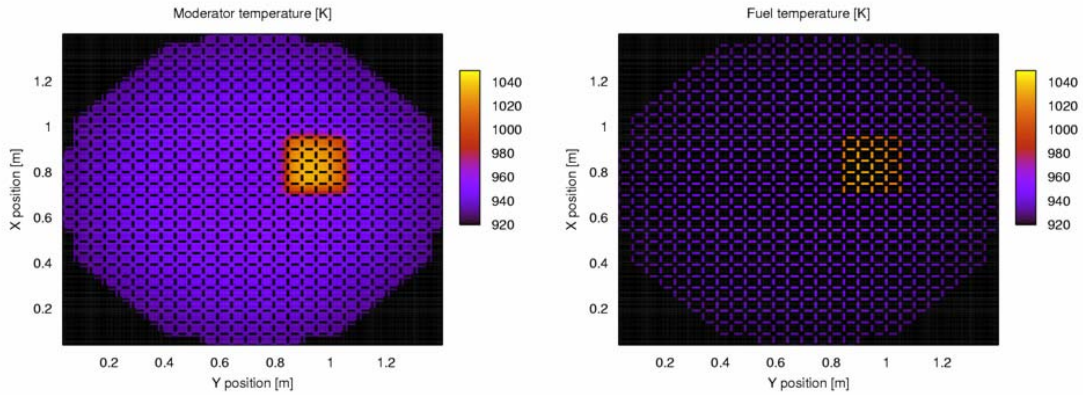
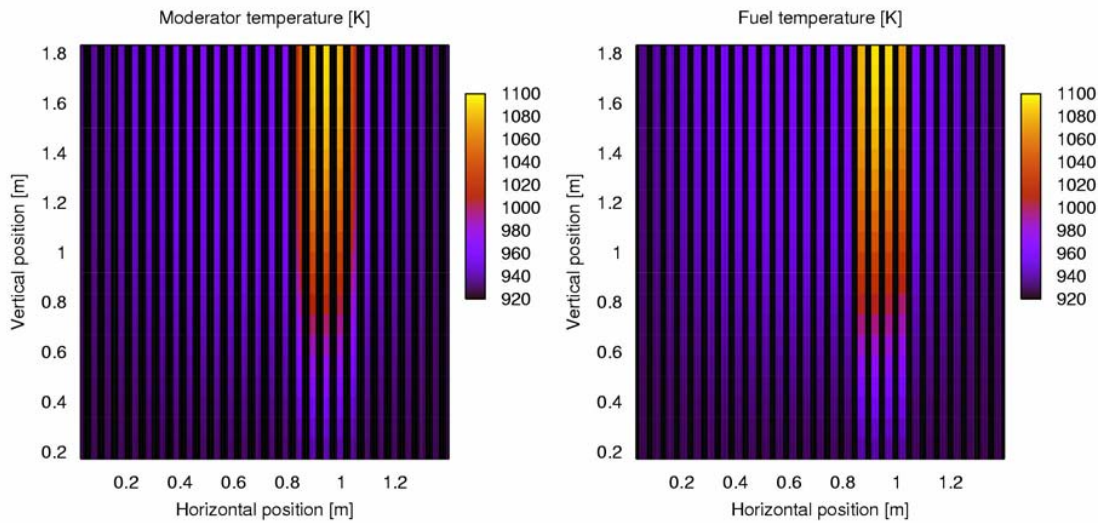


Figure 12: Power as a function of time during the channel blocking transient



4. Conclusion

A time dependent, 3D calculation scheme was presented for coupled reactor physics /heat transfer investigation of molten salt reactors. The scheme is based on the neutron diffusion equation, is able to account for the convection of delayed neutron precursors and supports detailed modelling of the heat transfer of the fuel and moderator. The scheme was applied for the MSRE, and demonstration calculations were presented, including a fuel channel blocking transient, which is a true three dimensional case.

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