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Steady-state and dynamic behavior of a moderated molten salt reactor

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ABSTRACT

The moderated Molten Salt Reactor (MSR) is an attractive breeder reactor. However, the temperature feedback coefficient of such a system can be positive due to the contribution of the moderator, an effect that can only be avoided with special measures. A previous study (Nagy et al., 2010) aimed to find a core design that is a breeder and has negative overall temperature feedback coefficient. In this paper, a coupled calculation scheme, which includes the reactor physics, heat transfer and fluid dynamics calculations is introduced. It is used both for steady-state and for dynamic calculations to evaluate the safety of the core design which was selected from the results of the previous study. The calculated feedback coefficients on the salt and graphite temperatures, power and uranium concentration prove that the core design derived in the previous optimization study is safe because the temperature feedback coefficient of the core and of the power is sufficiently negative. Transient calculations are performed to show the inherent safety of the reactor in case of reactivity insertion. As it is shown, the response of the reactor to these transients is initially dominated by the strong negative feedback of the salt. In all the presented transients, the reactor power stabilizes and the temperature of the salt never approaches its boiling point.

1. Introduction

The Molten Salt Reactor (MSR) is one of the six reactor types chosen in the Generation IV initiative (Generation IV International Forum, 2002), which aims for safer, more efficient and proliferation resistant reactor designs. This reactor uses a liquid fuel consisting of molten fluoride salts which is circulated in the primary loop through the core and the heat exchanger. The actinides are dissolved in the salt mixture as a fluoride salt, while other fluoride salts (such as LiF, BeF_2, NaF, ZrF_4) are used in the mixture to provide a low melting point. The heat produced by fission in the core is mainly deposited in the salt itself. This heat from the radioactive liquid is then transferred to a clean liquid and ultimately either to a steam or a gas cycle.

The MSR is a unique reactor design and it has some characteristics which cannot be found in other reactor types. Firstly, the spatial distribution of the delayed neutron precursors is decoupled from the flux distribution because the delayed neutron precursors travel through the core and the whole primary loop with the fuel salt. As a result, the delayed neutrons are emitted in a different location than that of the fission event which produced the precursor and a part of the precursors will decay outside the reactor core. Thus, the kinetics of the MSR are different from other types of reactors. Secondly, most of the heat is directly deposited into the salt which acts as the coolant of the reactor. In case of a moderated system, part of the fission heat is deposited in the moderator by gamma and neutron heating. This heat is removed from the core by the salt as well. Therefore, the moderator is at a higher temperature than the salt during operation.

Several publications have discussed the physical aspects of liquid fuel reactor systems, from basic reactor physics problems (Lapenta and Ravetto, 2000), to theoretical aspects of these dynamic systems, which include the generalization of the quasi-static method for the MSR (Dulla et al., 2004), and the dynamic space and frequency dependent response of MSRs to stationary perturbations (Pazsit and Jonsson, 2011). One-dimensional coupled neutronics and heat transfer programs were developed for time-dependent analysis (Krepel et al., 2005; Lecarpentier and Carpentier, 2003) and benchmarked against experimental data of the MSRE together with many other codes (Deplech et al., 2003) in the framework of the MOST project (Kophazi et al., 2003).

Two independent code systems were developed for 3D dynamic calculations of a moderated MSR. These were used for simulation of various reactivity- and pump-driven transients as well as accident scenarios, such as fuel channel blockage of the MSRE (Haubenreich and Engel, 1970) and MSBR (Robertson, 1971). The first code was realized by the modification of computational tools designed for pressurized water reactors whereby the original

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thermal–hydraulic model was extended with the model of the graphite moderator (Krepel et al., 2007, 2008). The other coupled code system consisted of a diffusion code, which was modified to calculate the precursor drift, and a full 3-D heat transfer model of the salt and graphite in the reactor core (Kophazi et al., 2009). In the latter coupled system, the flow field is parallel with the channels in the entire primary loop which includes the plena below and above the core and the loop outside the reactor vessel. Thus, the mixing of the precursors in the plena was not incorporated in the calculations, although it was demonstrated to have an impact on the kinetic behavior of the reactor (Kophazi et al., 2007) and it is expected to have an impact on the temperature distribution in the core as well.

The development of coupled codes dedicated to fast MSRs has started as well. In these studies the Molten Salt Actinide Recycler and Transmuter (MOSART) reactor (Ignatiev et al., 2007; Rineiski et al., 2006) was considered. The SIMMER-III two-dimensional code was modified and the conducted research focused on fluid dynamics simulations and optimization of the MOSART core (Wang et al., 2006). The development of a 2-D coupled neutronics – fluid dynamics code was reported (Nicolino et al., 2008) and applied to the full core modeling of the MOSART reactor. Finally, a steady-state (Zhang et al., 2009a) and dynamic (Zhang et al., 2009b) code system was developed to provide a more basic understanding of the coupling of flow calculations to neutron kinetics and heat transfer. These studies considered a simplified 2-D model of the MOSART reactor.

For decades, the graphite-moderated MSR designs were preferred over the non-moderated ones as breeder reactors. However, the focus of the research has moved to the non-moderated designs. One of the reasons for this is the positive temperature feedback of the graphite moderator and its impact on reactor safety. The safety of such a moderated breeder MSR is evaluated in this paper by calculating the feedback coefficients and by simulating the transient behavior of the reactor. The reactor design chosen for this study is a power reactor which proved to be self-breeding and have favorable safety characteristics in a previous study (Nagy et al., 2011). The 3-D coupled neutronics-thermal–hydraulics code system for moderated MSRs, which is used to perform the steady-state and transient calculations, is presented in this paper. The steady-state calculations are used to obtain the feedback coefficients of the reactor.

The thermophysical (melting point, density, heat capacity) and fluid transport properties (viscosity, thermal conductivity) of molten salts are key parameters when it comes to modeling the MSR behavior accurately. However, measuring and predicting some of these properties (typically the heat capacity and the thermal conductivity) is difficult and no accurate data is available for the salt mixtures used in the MSR. Some models exist; however, two reviews (Konings and Van der Meer, 2003; Williams et al., 2006) have clearly shown that differences exist between these model predictions and the measured data for different salt compositions. Therefore, one objective of the study is to show the impact of the substance properties on the nominal operating conditions of the reactor and on the reactivity feedback coefficients.

Finally, several pump-driven and temperature induced full power transients are calculated and discussed. These transients are selected to demonstrate that the power production stabilizes without any intervention and that the temperature of the salt does not reach its boiling point. The changing mass flow of the salt has two separate effects on the reactor, it changes the precursor density in the core and it influences the salt temperature. To demonstrate the effects of the changes in the precursor flow, the zero-power pump-driven transient is also presented.

2. Calculation model

The computational model consisted of several parts specially developed for MSRs in order to investigate the behavior of such a system. Separate computer codes calculated the neutron flux and precursor distribution, the flow field in the reactor, and the fuel and moderator temperature. The in-house developed neutron diffusion code DALTON (Boer et al., 2009) was extended to incorporate the drift of the precursors. The calculations were performed by DALTON using a 2-D or 3-D flow field of the fuel rather than using only a slug flow. This velocity field was provided by another in-house developed code (van Wijk, 2008) which calculated the radial distribution of the axial velocities in the fuel channels and the flow fields in the plena. In the heat transfer calculations, the temperature field in the salt and in the graphite was calculated on the true geometry – all the fuel channels were modeled separately and the thermal coupling between those was realized by a 3-D heat conduction model of the moderator. The cross-sections for the neutronics calculations were provided by 1-D cell calculations of the SCALE code (SCALE, 2009). As the codes have worked on different meshes, mesh conversion codes help the data transfer in the coupling scheme.

2.1. Neutron diffusion

The in-house developed DALTON code solved the multigroup diffusion equations on a structured grid (xyz or $rz\theta$) in 3D. The original DALTON code had been applied to high-temperature reactor neutronics (Boer et al., 2009; Ding et al., 2009) and to time-dependent perturbation theory (Van Rooijen and Lathouwers, 2008). Previously, the code had been modified to incorporate the drift of the precursor and was applied to time-dependent calculations of the MSRE (Kophazi et al., 2009). For this study, the precursor equations were extended with a convection term to take into account the flow of the fuel salt. The new equations read as:

$$\frac{1}{v_g}\frac{\partial \Phi_g}{\partial t} = \nabla \cdot D_g \nabla \Phi_g - \Sigma_g^r \Phi_g + \sum_{g \neq g'}^{G} \Sigma_{g' \to g}^s \Phi_{g'} + \chi_g^p \sum_{g'=1}^{G} (1-\beta) \nu \Sigma_{g'}^f \Phi_{g'} + \sum_i^l \lambda_i \chi_g^{d,i} C_i \quad (1)$$

$$\frac{\partial C_i}{\partial t} = \sum_{g'=1}^{G} \beta_i \nu \Sigma_{g'}^f \Phi_{g'} - \lambda_i C_i - \nabla \bar{u} C_i \quad (2)$$

Here Φ_g, and D_g note the scalar flux, and the diffusion coefficient in group g, and Σ_g^r, Σ_g^s and Σ_g^f represent the removal, scatter and fission cross-section in group g, respectively. Furthermore, ν, χ_g^p and $\chi_g^{d,i}$ note the average number of neutrons produced per fission, and the group-wise energy spectra of the prompt neutrons and the delayed neutrons of delayed neutron group i. In Eq. (2), C_i, β_i and λ_i represents the precursor concentration, delayed neutron fraction and decay constant of precursor group i, and $\bar{v}$ denotes the velocity of the fuel. The diffusion equation was solved only for the reactor core but the precursor concentration equation was solved for the whole primary loop. The primary circuit is modeled as a one-dimensional pipe. This way the precursors were mixed in the external part of the loop. In the volumes of the primary loop the neutron flux was zero.

The equations were discretized using the finite volume method. The neutron leakage term was calculated with the central difference scheme while a total variation diminishing scheme (Harten, 1983) was implemented for the precursor equation. In the latter scheme the van Leer limiter (Van Leer, 1974) was applied. In the DALTON model, the cross-sections of the core were homogenized in every unit cell. As a result, the calculated