Nuclear Engineering and Design 316 (2017) 112–124

Contents lists available at ScienceDirect

Nuclear Engineering and Design

journal homepage: www.elsevier.com/locate/nucengdes

Transient coupled calculations of the Molten Salt Fast Reactor using the Transient Fission Matrix approach

A. Laureau *, D. Heuer, E. Merle-Lucotte, P.R. Rubiolo, M. Allibert, M. Aufiero

LPSC, Université Grenoble-Alpes, CNRS/IN2P3, 53 rue des Martyrs, F-38026 Grenoble Cedex, France

HIGHLIGHTS

- Neutronic 'Transient Fission Matrix' approach coupled to the CFD OpenFOAM code.
- Fission Matrix interpolation model for fast spectrum homogeneous reactors.
- Application for coupled calculations of the Molten Salt Fast Reactor.
- Load following, over-cooling and reactivity insertion transient studies.
- Validation of the reactor intrinsic stability for normal and accidental transients.

ARTICLE INFO

Article history:
Received 12 March 2016
Received in revised form 18 October 2016
Accepted 18 February 2017

Keywords:
MSFR
Transient calculation
Neutronics
Thermal hydraulics
Transient Fission Matrix

ABSTRACT

In this paper we present transient studies of the Molten Salt Fast Reactor (MSFR). This generation IV reactor is characterized by a liquid fuel circulating in the core cavity, requiring specific simulation tools. An innovative neutronic approach called "Transient Fission Matrix" is used to perform spatial kinetic calculations with a reduced computational cost through a pre-calculation of the Monte Carlo spatial and temporal response of the system. Coupled to this neutronic approach, the Computational Fluid Dynamics code OpenFOAM is used to model the complex flow pattern in the core. An accurate interpolation model developed to take into account the thermal hydraulics feedback on the neutronics including reactivity and neutron flux variation is presented. Finally different transient studies of the reactor in normal and accidental operating conditions are detailed such as reactivity insertion and load following capacities. The results of these studies illustrate the excellent behavior of the MSFR during such transients.

1. Introduction

The reference design of the Molten Salt Fast Reactor (MSFR) is a 3 GWth liquid fuel reactor with a fuel salt volume of 18 m^3 and an average fuel salt temperature of 975 K (Brovchenko et al., 2013; Heuer et al., 2014; Allibert et al., 2016). It comprises three distinct circuits: the fuel circuit, the intermediate circuit and the power conversion system. This paper focuses on the modeling of the fuel circuit in transient calculations. The fuel salt considered is a molten binary fluoride salt with 77.5% of lithium fluoride; the other 22.5% are a mix of heavy nuclei fluorides (thorium and fissile matter). The proportion of fissile matter is adjusted to reach criticality. The circulation period of the fuel salt is around 4 s. As shown in Fig. 1, the fuel circuit includes a fertile blanket to improve breeding and a bubbling system to extract non-soluble fission products. The fuel salt is circulated out of the core via the pumps through the heat exchangers where the heat generated is removed from the fuel salt and transferred to the intermediate circuit.

This system's evolution during transient situations such as reactivity insertion and load following depends strongly on the neutronics and thermal hydraulics coupling due to the heat motion and the delayed neutron precursor circulation. The neutronics impacts the thermal hydraulics through the distribution of the power produced and of the precursor creations. The thermal hydraulics has feedback effects through the distributions of the delayed neutron sources and of the temperature in the core.

The Transient Fission Matrix (TFM) approach has initially been developed to model the neutronics of this kind of reactor. As discussed in Section 2, this TFM approach is designed to reproduce Monte Carlo neutronic calculations with a reduced computational cost. The neutron propagation in the reactor is pre-calculated once, prior to the transient calculation, using the Serpent Monte Carlo code (Leppänen, 2015). This information is then stored in matrices

* Corresponding author.
E-mail address: laureau.axel@gmail.com (A. Laureau).

http://dx.doi.org/10.1016/j.nucengdes.2017.02.022
0029-5493/

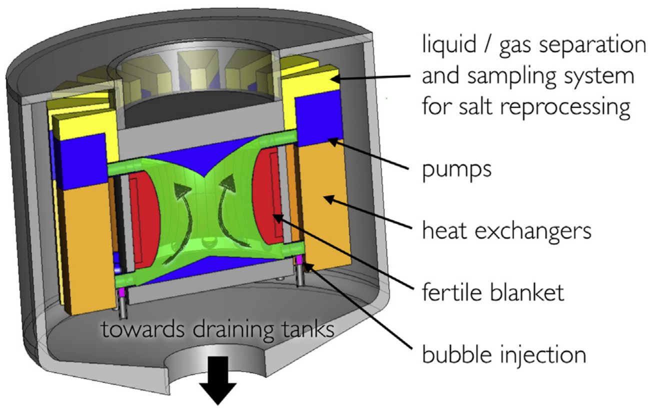

Fig. 1. MSFR fuel circuit global scheme.

for subsequent use in the kinetics calculations. An interpolation model is presented to take into account accurately the evolution of these matrices for different perturbations such as the thermal hydraulics feedback effects studied in this paper. Concerning the thermal hydraulics modeling, the complex flow pattern in the reactor can not be reproduced with sub-channel code. Thanks to the reduced fluid–solid interface in the core, the CFD (Computational Fluid Dynamics) resolution presented in Section 3 can be used for this reactor with a reasonable execution time. These two resolutions are finally coupled (Section 4) to obtain a solution of the reactor at steady state (Section 5) and to study transient scenarios (Section 6).

2. Neutronics modeling: Transient Fission Matrix

This section presents a brief introduction of the elements of TFM approach used in the current work, including the neutron kinetics equations solved during the coupling to the thermal hydraulics. This approach and its validation on different nuclear systems such as the Flattop experiment are detailed in Laureau et al. (2015) and Laureau (2015). Note that Section 2.5 is dedicated to the presentation of a new interpolation model adapted to the MSFR specificities: a fast spectrum reactor with an homogeneous fuel.

2.1. Fission matrix introduction

Fission matrices are usually employed to accelerate the source convergence in Monte Carlo neutronics codes (Carney et al., 2014; Dufek and Gudowski, 2009) or to estimate the different modes of the neutron source distribution (Carney et al., 2012).

The information contained in fission matrices is the transport of neutrons during one generation from each neutron emission-position j in the reactor to all its fission-positions i. Using a spatially discretized reactor, the emission and production positions i and j are associated to cells (or volume elements). This quantity can be directly estimated using a Monte Carlo neutronics calculation; the fission neutron production in cell i produced by a neutron created in cell j is scored in line i and column j of the fission matrix. This neutron propagation is represented in Fig. 2, the generation of the fission matrices pre-calculates and condenses the neutron propagation, and can be used a posteriori to propagate any source neutron distribution in the core.

The objective of this approach is to perform transient calculations, so that an innovative temporal aspect is added to the fission matrix approach.

2.2. TFM additional operators

From the usual fission matrices, additional operators have been added to develop the TFM approach. The first kind of operators are defined to take into account the distinct behavior between prompt and delayed neutrons. Different matrices are then calculated, $\underline{\underline{G_{\chi_x \nu_x}}}$, where χ_x represents the prompt or delayed emission spectrum χ_p or χ_d, and ν_x represents the prompt or delayed production of neutrons ν_p or ν_d. The second kind of operators concerns the kinetic aspect: the $\underline{\underline{T_{\chi_p \nu_p}}}$ matrix represents the average time response from cell j to cell i associated to the prompt neutron production $\underline{\underline{G_{\chi_p \nu_p}}}$.

This approach has been implemented in a modified version of the Serpent code. For each fission neutron source created in the core during a critical calculation, this neutron has an attribute corresponding to the cell number j of its birth, and another attribute indicating if this neutron is a delayed neutron. Then, at each interaction in all the cells i during the neutron transport, the probability of creating a fission neutron, prompt or delayed, is scored. The neutron lifetime weighted by the production of fission neutrons is also scored. Finally, the spatially discretized operators $\underline{\underline{G_{\chi_p \nu_p}}}, \underline{\underline{G_{\chi_p \nu_d}}}, \underline{\underline{G_{\chi_d \nu_p}}}, \underline{\underline{G_{\chi_d \nu_d}}}$ and $\underline{\underline{T_{\chi_p \nu_p}}}$ are estimated in one calculation using these j to i estimators.

2.3. Kinetic parameter calculations

The effective generation time and the effective fraction of delayed neutrons can be deduced from the properties of the generated matrices (Laureau et al., 2015). The kinetic calculations with the TFM approach presented in this paper are using the effective fission to fission time, estimated by combining the data of the fission matrix $\underline{\underline{G_{\chi_p \nu_p}}}$ and of the time propagation matrix $\underline{\underline{T_{\chi_p \nu_p}}}$. The calculation of this effective parameter is detailed in this section.

This quantity requires an estimation of the equilibrium source neutron distribution and of the importance map. The eigenvector of $\underline{\underline{G_{\chi_p \nu_p}}}$ corresponds to the equilibrium prompt neutron sources $\boldsymbol{N_p}$ in the reactor, its propagation through the fission matrix corresponds to a dilatation of the prompt multiplication factor: $\underline{\underline{G_{\chi_p \nu_p}}}\boldsymbol{N_p} = k_p \boldsymbol{N_p}$. The transposed fission matrix corresponds to the backward transport of the neutron, the probability that a neutron created in j comes from a previous history stated in i. The associated eigenvector $\boldsymbol{N}_p^*$ is the importance map of the neutrons, it represents the proportion of neutrons coming from each position.

Finally, using $\underline{\underline{G_{\chi_p \nu_p}}} \cdot \underline{\underline{T_{\chi_p \nu_p}}}$ the element by element multiplication matrix, the effective fission to fission time $l_{f_{eff}}$ can be calculated using Eq. (1). This equation consists in a local time response weighted by the local neutron production to obtain the global time response. Both numerator and denominator are adjoint weighted by the importance map to obtain the effective value. It corresponds to the effective prompt lifetime l_{eff} since non-fission reactions have a zero-importance and the importance of fission events is the importance of the produced neutrons.

$$l_{eff} = \frac{\boldsymbol{N}_p^* \left(\underline{\underline{G_{\chi_p \nu_p}}} \cdot \underline{\underline{T_{\chi_p \nu_p}}} \right) \boldsymbol{N_p}}{\boldsymbol{N}_p^* \underline{\underline{G_{\chi_p \nu_p}}} \boldsymbol{N_p}} \tag{1}$$

2.4. Neutron kinetics equations

The kinetics equations use the effective prompt lifetime l_{eff} calculated using $\underline{\underline{T_{\chi_p \nu_p}}}$ discussed in Section 2.3 dealing with the kinetic