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Calculating the effective delayed neutron fraction in the Molten Salt Fast Reactor: Analytical, deterministic and Monte Carlo approaches

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ABSTRACT

This paper deals with the calculation of the effective delayed neutron fraction (β_{eff}) in circulating-fuel nuclear reactors. The Molten Salt Fast Reactor is adopted as test case for the comparison of the analytical, deterministic and Monte Carlo methods presented. The Monte Carlo code SERPENT-2 has been extended to allow for delayed neutron precursors drift, according to the fuel velocity field. The forward and adjoint eigenvalue multi-group diffusion problems are implemented and solved adopting the multi-physics toolkit OpenFOAM, by taking into account the convective and turbulent diffusive terms in the precursors balance. These two approaches show good agreement in the whole range of the MSFR operating conditions. An analytical formula for the circulating-to-static conditions β_{eff} correction factor is also derived under simple hypotheses, which explicitly takes into account the spatial dependence of the neutron importance. Its accuracy is assessed against Monte Carlo and deterministic results. The effects of in-core recirculation vortex and turbulent diffusion are finally analysed and discussed.

1. Introduction

The effective delayed neutron fraction (β_{eff}) is an important reactor kinetics parameter. The contribution of delayed neutrons is of primary importance for the safe control of any nuclear reactor. β_{eff} is often adopted as unit of experimental reactivity (*dollar*) (Bell and Glasstone, 1979). Many efforts have been devoted to the measurement (e.g., Sakurai et al., 1999; Rudstam et al., 2002) and calculation (e.g., Meulekamp and van der Marck, 2006; Carta et al., 2011) of physical and effective delayed neutron fractions. The traditional definition of β_{eff} involves the calculation of both the forward and adjoint solution of the neutron transport equations, which makes its accurate calculation by means of continuous energy Monte Carlo codes a hard task, even for static-fuel reactors (Nagaya et al., 2010). In circulating-fuel systems, the motion of delayed neutron precursors complicates the calculation of the effective delayed neutron fraction. In molten salt reactors, the adoption of the Thorium cycle (with ^{233}U as fissile) or the envisaged incineration of Minor Actinides (Merle-Lucotte et al., 2011), and the fuel motion itself (Guerrieri et al., 2013), may lead to β_{eff} values much lower than those typical of Light Water Reactors (LWR). Moreover, loss of flow accidental scenarios in fluid-fuel systems involve the introduction of positive reactivity due to delayed neutron source term redistribution (Guerrieri et al., 2012). The quantification of the introduced reactivity requires the calculation of the difference in delayed neutron effectiveness between static and circulating conditions. For these reasons, accurate and reliable β_{eff} calculations for molten salt reactors are desirable.

In circulating-fuel reactors, the effective delayed neutron fraction (β_{eff}) differs from the physical delayed neutron fraction (β_0) for two distinct reasons. The first reason (common to solid-fuelled reactors) is that the emission spectrum of delayed neutrons is softer than that of prompt neutrons: on average, the former are emitted with a lower energy. This may imply a difference in the importance of delayed and prompt neutrons. The second reason is that delayed neutron precursors are transported by the fluid flow in the fuel circuit and might decay in position of low importance and even out of the core. Spatial effects due to fuel motion are more relevant and always reduce the values of β_{eff}. Energy effects are, in general, of lesser relevance and might reduce or increase the effective delayed neutron fraction, according to the neutronic characteristics of the core.

One-dimensional approaches have often been adopted to correct the effective delayed neutron fraction calculation in order to take into account the fuel motion. One of the most common

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Nomenclature

Latin symbols

c	concentration of delayed neutron precursors (m^{-3})
c^*	importance of delayed neutron precursors ($m^{-2}\ s^{-1}$)
cf_{circ}	correction factor for the effective delayed neutron fraction defined in Eq. (1) (–)
D	diffusion coefficient (m)
$D(\mathbf{r}')$	space dependence of the delayed neutron precursors decay probability (–)
D_T	turbulent mass diffusivity ($m^2\ s^{-1}$)
H_a	active height (m)
H_e	extrapolated height (m)
$I(\mathbf{r}')$	space dependence of the neutron importance (–)
$K(\mathbf{r}',\mathbf{r})$	conditional precursor decay probability (–)
k_{eff}	effective multiplication factor (–)
k_p	effective multiplication factor without delayed neutrons (–)
Pr_T	turbulent Prandtl number (–)
r	radial coordinate (m)
R_a	active radius (m)
R_e	extrapolated radius (m)
$S(\mathbf{r})$	space dependence of the precursors source (–)
Sc_T	turbulent schmidt number (–)
T	fuel circulation period (s)
$\mathbf{u}$	fuel velocity field ($m\ s^{-1}$)
z	axial coordinate (m)

Greek symbols

β_0	total physical delayed neutron fraction (–)
β_{eff}	total effective delayed neutron fraction (–)
γ	in-core-to-total fuel volume ratio (–)
λ	decay constant of precursors (s^{-1})
ν	average total number of neutron emitted per fission (–)
ν_T	eddy viscosity ($m^2 s^{-1}$)
Σ_a	absorption cross section (m^{-1})
Σ_f	fission cross section (m^{-1})
Σ_s	scattering cross section (m^{-1})
ϕ	forward neutron flux ($m^{-2}s^{-1}$)
ϕ^*	adjoint neutron flux ($m^{-2}s^{-1}$)
χ_p	delayed neutron yield (–)
χ_p	prompt neutron yield (–)

Superscripts

c	relative to circulating-fuel conditions
s	relative to static-fuel conditions

Subscripts

d	relative to delayed neutrons
g	relative to the gth neutron energy-group
i	relative to the ith delayed neutron precursor group
p	relative to prompt neutrons

Acronyms

CFD	Computational Fluid-Dynamics
EVOL	Evolution and Viability Of Liquid fuel fast reactor systems
IFP	Iterated Fission Probability
LWR	Light Water Reactor
MCNP	Monte Carlo N-Particle transport code
MSFR	Molten Salt Fast Reactor
MSRE	Molten Salt Reactor Experiment
OpenFOAM	Open-source Field Operation And Manipulation
PWR	Pressurized Water Reactor
RANS	Reynolds-Averaged Navier–Stokes
TRU	Transuranic elements

employed method consists in the correction of static-fuel β_{eff} calculations (obtained by means of commonly available Monte Carlo or deterministic codes, and allowing for energy effects) by means of the fraction of precursors decaying inside the reactor core and not in the out-of-core part of the loop (Lecarpentier, 2001; Cammi et al., 2011; Guerrieri et al., 2013). This fraction is calculated under the hypothesis that the production of precursors has a flat or sinusoidal dependence on the axial coordinate in the core. This approach neglects the effects on β_{eff} due to the inhomogeneous spatial importance of neutrons inside the core. More precisely, this method considers the delayed neutrons produced within the core to have a relative effectiveness of one, regardless of their position, while those produced outside the core have zero importance. On average, precursors are created in a position with a higher spatial neutron importance with respect to the place where they will decay. For this reason, this approach may lead to a significant overestimation of the effective delayed neutron fraction in circulating conditions. In Section 3.1, a new analytical formula for the β_{eff} correction factor is derived, which takes into account the in-core spatial importance effect. Moreover, the proposed approach allows for radial redistribution of the delayed neutron precursors that re-enter in the core.

A more accurate approach involves the adoption of deterministic neutronic calculations to solve the adjoint and forward neutron transport problem. Then, the effective (i.e., adjoint weighted) delayed neutron fraction is calculated. For example, Mattioda et al. (2000) adopted the multi-group neutron diffusion approximation along with precursors drift in one-dimensional geometry. Kópházi et al. (2009) proposed a model for the analysis of moderated MSRs based on neutron diffusion and precursor convection and applied it to the MSRE (Molten Salt Reactor Experiment). In Section 3.2, the solution of the equations of neutron diffusion and precursor transport by means of the multi-physics open-source toolkit OpenFOAM (Weller et al., 1998) is discussed. The proposed solver allows the solution of the forward and adjoint eigenvalue problems for arbitrary geometries based on detailed spatially-dependent velocity fields.

Monte Carlo has often been regarded as reference method for verifying the results of deterministic calculations. Commonly available Monte Carlo codes are not suitable for β_{eff} calculation in circulating-fuel system. Kópházi et al. (2004) evaluated the reactivity loss due to fuel circulation in the MSRE with an extended version of Monte Carlo N-Particle Transport Code (MCNP). In that work, the decay position of delayed neutron precursors was recalculated under simple hypotheses, and neutrons emitted outside of the core were considered lost. More recently, Kiedrowski (2012) also considered the effect of precursors diffusion in fissile solution systems. In both works, the velocity profiles, adopted as input in the Monte Carlo simulations, were simple analytical functions of the position. In Section 3.3, an extension of the Monte Carlo code SERPENT-2 (SERPENT, 2011) for β_{eff} calculation in circulating-fuel systems is presented. The extended version features a suitable algorithm for the transport of delayed neutron precursors. The algorithm is intended for an accurate and efficient tracking of the precursors,