



Beam transient analyses of Accelerator Driven Subcritical Reactors based on neutron transport method



Mingtao He^a, Hongchun Wu^a, Youqi Zheng^{a,*}, Kunpeng Wang^b, Xunzhao Li^a, Shengcheng Zhou^a

^a School of Nuclear Science and Technology, Xi'an Jiaotong University, Xi'an 710049, Shaanxi, China

^b Nuclear and Radiation Safety Center, PO Box 8088, Beijing 100082, China

HIGHLIGHTS

- A transport-based kinetics code for Accelerator Driven Subcritical Reactors is developed.
- The performance of different kinetics methods adapted to the ADSR is investigated.
- The impacts of neutronic parameters deteriorating with fuel depletion are investigated.

ARTICLE INFO

Article history:

Received 17 July 2015

Received in revised form

18 September 2015

Accepted 25 October 2015

ABSTRACT

The Accelerator Driven Subcritical Reactor (ADSR) is almost external source dominated since there is no additional reactivity control mechanism in most designs. This paper focuses on beam-induced transients with an in-house developed dynamic analysis code. The performance of different kinetics methods adapted to the ADSR is investigated, including the point kinetics approximation and space–time kinetics methods. Then, the transient responds of beam trip and beam overpower are calculated and analyzed for an ADSR design dedicated for minor actinides transmutation. The impacts of some safety-related neutronics parameters deteriorating with fuel depletion are also investigated. The results show that the power distribution varying with burnup leads to large differences in temperature responds during transients, while the impacts of kinetic parameters and feedback coefficients are not very obvious. Classification: Core physic

© 2015 Elsevier B.V. All rights reserved.

1. Introduction

Currently, partitioning and transmutation of transuranics (TRUs), especially minor actinides (MAs), is an attractive research field of reactor physics (Greene et al., 2008). During the last decade, reactor systems with thermal and fast neutron spectrums have been intensively studied as the dedicated transmuter in various fuel cycles. However, transmuting recycled MAs in thermal reactors like Light Water Reactors (LWRs) would generate very high inventory of ²⁵²Cf in fuel cycle facilities. It results in an unacceptably strong neutron source at fuel fabrication and requires expensive investments in radioactive shielding. In contrast, using fast spectrum systems would reduce the inventory of ²⁵²Cf by

more than two orders of magnitude (Salvatores and Palmiotti, 2011).

The performance of MAs transmutation in fast spectrum systems, including Accelerator Driven Subcritical Reactors (ADSRs) and critical Fast Reactors (FRs), has been investigated and compared extensively (OECD/NED, 2002; Salvatores, 2009). As a matter of fact, introducing MAs into a reactor usually significantly affects safety-related neutronics parameters. For example, the magnitude of the negative Doppler feedback effect would be reduced, the coolant void coefficient would be increased and the effective delayed neutron fraction would be smaller (Wallenius, 2012; Yang, 2012). Consequently, these impacts of neutronics become the major limitation to the amount of MAs loaded in critical FR cores (Tesinsky et al., 2012a,b). It has been found that the ADSR is preferred due to the potential flexibility to utilize various fuel types and to load with large amounts of TRUs with very different ratios of MAs over plutonium.

* Corresponding author. Tel.: +86 29 8266 8692; fax: +86 29 8266 8916.
E-mail address: yqzheng@mail.xjtu.edu.cn (Y. Zheng).

This paper focuses on the transient analyses of an ADSR design (Zhou et al., 2014) dedicated for MAs burning. Studies are based on a new developed code for ADSR transient analyses named as DAISY based on neutron transport method. The anisotropic distribution of the neutron flux which caused by the strong external neutron source in core center could be accommodated by employing more accurate transport method (Aizawa et al., 2013). In this code, the conventional integral parameters based point kinetics approximation method is adopted to give useful information on physical characteristics of the ADSR. Besides, the improved quasi-static method and the direct difference method are also included in the code for space dependent analyses, and these methods are adopted to estimate the performance of the point kinetics approximation.

Among serials of transients, only accelerator beam induced transients are carried out in this study. In order to maintain a sufficient safety margin, the ADSR is usually operated under deep subcritical level with the effective multiplication factor around 0.97 (Kin et al., 2002). However, for the purpose of simplifying system complexity and reducing extra potential accident indicator (Yang and Khalil, 2001), there is no additional reactivity control mechanism in most designs, i.e., the ADSR is almost external source dominated. Consequently, beam-induced transients have become important matters that require further research to accommodate engineering needs. Two typical beam transients, the beam trip (BT) and the beam overpower (BOP) process, are performed and analyzed. Maximal temperature limits for fuel and cladding materials are compared. The impacts of deterioration of some safety-related neutronics parameters caused by fuel depletion are also investigated.

The rest of this paper is organized as following. Section 2 introduces the theory of space-time neutron kinetics and discusses its implications in the ADSR system. The transient code development and the corresponding verification are given in Section 3. Then, in Section 4, the code is employed to analyze the beam transient process of an ADSR design. Conclusions are summarized in Section 5.

2. Neutron kinetics equations

In an source-driven subcritical system, time-dependent neutron transport equations with delayed neutron precursors can be expressed as follows with standard symbols (Prinja and Larsen, 2010):

$$\begin{aligned} \frac{1}{v(\mathbf{r}, E)} \frac{\partial \psi}{\partial t}(\mathbf{r}, E) &= A(\mathbf{r}, E) \psi(\mathbf{r}, E) + F_p(\mathbf{r}, E) \psi(\mathbf{r}, E) \\ &+ \frac{1}{4\pi} \sum_{j=1}^{Nd} \chi_j(\mathbf{r}, E) \lambda_j C_j(\mathbf{r}, t) + Q(\mathbf{r}, \Omega, E, t) \frac{\chi_j(\mathbf{r}, E)}{4\pi} \frac{\partial C_j(\mathbf{r}, t)}{\partial t} \quad (1) \\ &+ \frac{\chi_j(\mathbf{r}, E)}{4\pi} \lambda_j C_j(\mathbf{r}, t) = F_{d,j}(t) \psi(\mathbf{r}, E), \quad j = 1, \dots, Nd \end{aligned}$$

and subject to corresponding initial and boundary conditions:

$$\begin{aligned} \psi(t) &= \psi^{\text{boundary}}(\mathbf{r}, \Omega, E, t), \quad \mathbf{r} \in \partial V, \quad \Omega \cdot \mathbf{n} < 0, \quad 0 < E < \infty, \quad 0 < t \\ \psi(t_0) &= \psi^{\text{initial}}(\mathbf{r}, \Omega, E), \quad \mathbf{r} \in \partial V, \quad \Omega \in 4\pi, \quad 0 < E < \infty \\ C_j(\mathbf{r}, t_0) &= C_j^{\text{initial}}(\mathbf{r}), \quad \mathbf{r} \in V, \quad j = 1, \dots, Nd \\ Q(t_0) &= Q^{\text{initial}}(\mathbf{r}, \Omega, E), \quad \mathbf{r} \in \partial V, \quad \Omega \in 4\pi, \quad 0 < E < \infty \end{aligned} \quad (2)$$

where $\psi(t) \equiv \psi(\mathbf{r}, \Omega, E, t)$ and $C_j(\mathbf{r}, t)$ denote the angular neutron flux and delayed neutron precursors concentrations, respectively, and Nd is the group number of precursors, v is the neutron velocity. The external neutron source is taken into account by Q which is also time-dependent. The A is the sum of neutron scattering, leakage and

interaction, while F_p and $F_{d,j}$ represent the time-dependent prompt and delayed neutron fission, respectively:

$$\begin{aligned} A(t) \psi(t) &= \int_0^\infty \int_{4\pi} \sum_s (\mathbf{r}, \Omega' \cdot \Omega, E' \rightarrow E, t) \psi \\ &\times (\mathbf{r}, \Omega', E', t) d\Omega' dE' - \Omega \cdot \nabla \psi(\mathbf{r}, \Omega, E, t) \\ &- \sum_t (\mathbf{r}, E, t) \psi(\mathbf{r}, \Omega, E, t) \end{aligned} \quad (3)$$

$$\begin{aligned} F_p(t) \psi(t) &= \frac{\chi_p(\mathbf{r}, E) \cdot [1 - \beta(\mathbf{r})]}{4\pi} \int_0^\infty \int_{4\pi} v \sum_f (\mathbf{r}, E', t) \psi \\ &\times (\mathbf{r}, \Omega', E', t) d\Omega' dE' \end{aligned} \quad (4)$$

$$\begin{aligned} F_{d,j}(t) \psi(t) &= \frac{\chi_j(\mathbf{r}, E) \cdot \beta_j(\mathbf{r})}{4\pi} \int_0^\infty \int_{4\pi} v \sum_f (\mathbf{r}, E', t) \psi \\ &\times (\mathbf{r}, \Omega', E', t) d\Omega' dE' \end{aligned} \quad (5)$$

where Σ_s , Σ_t and $v\Sigma_f$ are time-dependent cross sections, χ_p and χ_j are the prompt and delayed fission spectrum, β_j and λ_j are the fraction and decay constant of precursors, respectively. In general case, directly solving these equations by the implicit difference method is mathematical rigorous. However, to obtain comprehensive physical information on system kinetic behavior and also to reduce computational costs, the integral parameters based kinetics methods such as improved quasi-static method and point kinetics approximation are very meaningful. Actually, these methods make assumptions on the space-dependence in transient process by parameter integration.

According to the basic idea of improved quasi-static method (Dulla et al., 2008), the 'spatial shape' of neutron flux varies with time much slower than that of the 'amplitude' does. So, it can be represented as following that the amplitude depends only on time while the shape function depends on all variables:

$$\begin{aligned} \psi(\mathbf{r}, \Omega, E, t) &= n(t) \psi(\mathbf{r}, \Omega, E, t) \\ C_j(\mathbf{r}, t) &= m_j(t) C_j(\mathbf{r}, t) \end{aligned} \quad (6)$$

where $n(t)$ and $m_j(t)$ are amplitude functions while $\tilde{\psi}(t)$ and $C_j(\mathbf{r}, t)$ are shape functions. Here, a weighting function is required to obtain the amplitude function equations. There are many discussions on determining the weighting function in the source-driven subcritical system. In this study, the adjoint neutron flux of initial steady state is adopted (Rabiti and Rineiski, 2004; Rineiski et al., 2005):

$$A^*(t_0) \psi^* + F_p^*(t_0) \psi^* + \frac{1}{4\pi} \sum_{j=1}^{Nd} F_{d,j}^*(t_0) \psi^* = 0 \quad (7)$$

where $\psi^* \equiv \psi^*(\mathbf{r}, \Omega, E, t_0)$ is the neutron importance in the initial steady state t_0 , the star in Eq. (7) represents adjoint operator. Consequently, the estimated reactivity worth of perturbation is accurate to first order in the shape function variation (Stacey, 2007). The factorization above could be arbitrary, so constraint conditions are added to make them unique and should be fulfilled during whole

Download English Version:

<https://daneshyari.com/en/article/6760602>

Download Persian Version:

<https://daneshyari.com/article/6760602>

[Daneshyari.com](https://daneshyari.com)