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Original Article

Sensitivity Analysis of Core Neutronic Parameters in Electron Accelerator-driven Subcritical Advanced Liquid Metal Reactor

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ABSTRACT

Calculation of the core neutronic parameters is one of the key components in all nuclear reactors. In this research, the energy spectrum and spatial distribution of the neutron flux in a uranium target have been calculated. In addition, sensitivity of the core neutronic parameters in accelerator-driven subcritical advanced liquid metal reactors, such as electron beam energy (E_e) and source multiplication coefficient (k_s), has been investigated. A Monte Carlo code (MCNPX_2.6) has been used to calculate neutronic parameters such as effective multiplication coefficient (k_{eff}), net neutron multiplication (M), neutron yield ($Y_{n/e}$), energy constant gain (G_0), energy gain (G), importance of neutron source (φ^*), axial and radial distributions of neutron flux, and power peaking factor (P_{max}/P_{ave}) in two axial and radial directions of the reactor core for four fuel loading patterns. According to the results, safety margin and accelerator current (I_e) have been decreased in the highest case of k_s, but G and φ^* have increased by 88.9% and 21.6%, respectively. In addition, for LP1 loading pattern, with increasing E_e from 100 MeV up to 1 GeV, $Y_{n/e}$ and G improved by 91.09% and 10.21%, and I_e and P_{acc} decreased by 91.05% and 10.57%, respectively. The results indicate that placement of the Np–Pu assemblies on the periphery allows for a consistent k_{eff} because the Np–Pu assemblies experience less burn-up.

1. Introduction

About 55 years ago, to reduce radioactive waste, several nuclear systems were studied. Transmutation of plutonium with minor actinides and long-lived fission products is a promising approach to reduce radioactive waste and its long-term radio toxicity. Furthermore, one of the design goals of subcritical reactors driven by an external neutron source is to produce neutron excess and energy for the purpose of transmutation of waste [1].

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Therefore, the ratio of energy produced by fission in the subcritical reactor core to energy of particle beams twisted into a neutron target, which is defined as the energy gain of the system, is necessary. Importance of neutron parameter indicating the relative contribution of neutron source to neutron fission is another important neutron parameter [2–4].

Moreover, neutronic parameters are dependent on core composition. During the extended reactor operation, core composition is changed due to fuel burn-up. Therefore, these reactor parameters might differ from their conditions at the beginning of the cycle. A good design assures the safe conditions of the reactor in the life-cycle period. Based on specific features of accelerator-driven subcritical reactors (ADSRs), fuel burn-up effects required accelerator current [5,6].

In this study, electron accelerator-driven systems are used to reduce harmful emissions. In most countries, costly technologies are used to transmute long-lived fission products [7,8]. Therefore, in this research, a natural uranium cylindrical target and an accelerated electron beam have been simulated utilizing the MCNPX code (version 2.6; Los Alamos National Laboratory, Los Alamos, NM, USA) [9]; then neutron spectra produced by the photonuclear process in the target have been located in the subcritical advanced liquid metal reactor (ALMR) core [7]. Finally, the sensitivity of core neutronic parameters to the accelerator-related parameters, such as source multiplication coefficient and electron beam energy, has been investigated.

2. Materials and methods

In this study, a natural uranium cylindrical target, with a thickness of 10 cm and a diameter of 4 cm, is used, which is exposed to an electron beam of 1 cm diameter, 1,000 MeV energy, and a parabolic spatial distribution. The parts of modified ALMR core include 120 fuel assemblies with 271 fuel rods, with the central fuel assembly containing a natural uranium target and 234 fuel rods located in the interior. The fuel alloy was composed of 11 w/o TRU (isotopes with higher atomic numbers than uranium are termed transuranic) within 89 w/o Zr, and molten sodium as a coolant and graphite as a reflector [7,10].

The dimensional characteristics, based on the ALMR design, are given in Table 1. A description of the fuel loading patterns is given in Table 2 [7]. Figs. 1–5 show the radial simulated fuel assembly, the target located in the interior, and loading patterns using the Monte Carlo code MCNPX. In addition, the axial configuration of the core is shown in Fig. 6 using this code. Furthermore, the TRU isotopic compositions utilized in fuel assemblies are given in Tables 3–6 [10,11].

Table 1 – Reactor design characteristics.

Characteristics	Value (cm)
Core fuel height	107
Outside core barrel height	111.4
Outside core barrel radius	103.10
Fuel radius	0.372
Outer cladding radius	0.428
Fuel rod pitch	0.890568
Fuel assembly pitch	61.14
Reflector thickness	18

Table 2 – Loading pattern summary.

Loading pattern	TRUs present in regions listed		
	Inner region	Middle region	Outer region
1	Np–Pu–Am–Cm		
2	Am	Cm	Np–Pu
3	Cm	Np–Pu	Am
4	Cm	Am	Np–Pu

TRU, transuranic.

2.1. Calculation of neutronic parameters

In this study, the effects of accelerator parameters, such as source multiplication coefficient (k_s) and electron beam energy (E_e), on the ALMR core neutronic parameters have been investigated. The MCNPX code has been used to calculate neutronic parameters, including effective multiplication coefficient (k_{eff}), net neutron multiplication (M), neutron yield ($Y_{n/e}$), energy constant gain (G_0), energy gain (G), importance of neutron source (φ^*), axial and radial distributions of neutron flux, and power peaking factor (P_{max}/P_{ave}) in two axial and radial directions of the reactor core for four fuel loading patterns.

Accelerator power (P_{acc}) in MW has been calculated according to Eq. (1) [2–4].

$$P_{acc} = i_e \cdot E_e \tag{1}$$

where E_e is the electron energy in GeV and i_e is the accelerator current in mA.

The net neutron multiplication has been calculated according to Eq. (2) [2–4,12]:

$$M = \frac{1}{1 - K_s} \tag{2}$$

The energy gain of the system has been calculated according to Eq. (3) [2–4,12]:

$$G = \frac{Y_{n/e} E_f K_s}{V(1 - k_s) E_e} \tag{3}$$

where $Y_{n/e}$ is the number of neutrons produced per proton in the target, E_f is the average energy per fission, and V is the average neutron yield per fission.

According to energy constant gain, G_0, the equation is modified as follows [2–4,12]:

$$G_0 = \frac{Y_{n/e} E_f}{V E_e} \Rightarrow G = G_0 \frac{K_s}{(1 - k_s)} \tag{4}$$

Importance of neutron source is defined as the ratio of the average importance of the external neutron source to the average importance of the fission neutrons, which has been calculated according to Eq. (5) [2–4,12]: