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Sensitivity analysis of core neutronic parameters in accelerator driven subcritical reactors

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ABSTRACT

In this paper, sensitivity of the ADSRs core neutronic parameters to the accelerator related parameters such as beam profile, source multiplication coefficient (k_s) and proton beam energy (E_p) has been investigated. TRIGA reactor has been considered as the case study of the problem. Monte Carlo code MCNPX has been used to calculate neutronic parameters such as: effective multiplication coefficient (k_{eff}), net neutron multiplication (M), spallation neutron yield ($Y_{n/p}$), 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 three eigen values levels (k_s) including: 0.91, 0.97 and 0.99. According to the results, using a parabolic spatial distribution instead of a uniform spatial distribution increases the relative differences of spallation neutron yield, net neutron multiplication and energy gain by 4.74%, 4.05% and 10.26% respectively. In consequence the required accelerator current (I_p) will be reduced by 7.14% to preserve the reactivity. Although safety margin is decreased in highest case of k_s, but energy gain increases by 93.43% and the required accelerator current and importance of neutrons source decrease by 48.3% and 2.64% respectively. In addition, increasing E_p from 115 MeV up to1 GeV, improves spallation neutron yield and energy gain by 2798.71% and 205.12% and decreases the required accelerator current and power by 96.83% and 72.44%, respectively. Therefore, our results are indicative of the fact that investigating sensitivity of the core neutronic parameters to the accelerator related parameters are necessary in order to optimally design a cost-efficient ADSR.

1. Introduction

Accelerator Driven Subcritical Reactors (ADSRs), called hybrid reactor, are modern nuclear reactors producing energy and transmutation of radioactive wastes in a clean and safe manner. One of the essential issues for an ADSR is the condition that the reactor core is designed in such a way, that it remains sub-critical under normal operation and off-normal conditions.

Number of neutrons produced per proton-nucleus reaction in the spallation target is an effective parameter in a subcritical reactor driven by a proton accelerator. On the other hand, one of the objectives in designing an ADSR is to obtain as high power as possible in the core using as low proton beam power as possible (Kadi and Revol, 2001; Nifenecker et al., 2001, 2003).

Furthermore, the system is driven by an accelerator with a high-energy proton beam that smashes a target atom into many atomic fragments producing a large number of neutrons. Therefore, the optimum proton beam energy for production of neutrons by spallation depends on heavy metal target, in terms of costs, system efficiency and etc. (Eriksson et al., 2005; Nifenecker et al., 2001).

Since the construction of a reliable high-power proton accelerator is a difficult technical task and its operation is very expensive, investigating sensitivity of the core neutronic parameters to the accelerator related parameters has a significant impact on the overall design of a future ADSR and on the economy of its operation.

Therefore, in this paper we study the effects of several variations in the accelerator related parameters such as beam profile, k_s and E_p on the core neutronic parameters of the TRIGA reactor (Borio di Tigliole et al., 2010) as the case study of the problem using MCNPX (version 2.4) code (Hughes et al., 2002).

The TRIGA reactor is a pool-type research reactor moderated and cooled by light water which is utilized in TRADE project (TRIGA Accelerator Driven Experiment). This project is based on coupling of a linear proton accelerator with TRIGA reactor. The conceptual design is carried out. Due to a lack of enough fund, the proton accelerator was replaced with californium sources and a small D–T neutron generator placed in the centre of the reactor fuel in 2004 (Borio di Tigliole et al., 2010; Rubbia et al., 2002a,b, 2004).

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2. Materials and method

2.1. Subcritical TRIGA core model

Monte Carlo analyses were all performed using an identical geometrical model of the subcritical TRIGA core shown in Fig. 1 (Hassanzadeh et al., 2013). The core is divided into seven rings. Some of the regions contain fuel pins, graphite dummy elements and control rods as well. Central channel has been designed for loading a neutron source.

Three different cases of simulations are considered as followings (Rubbia et al., 2002a,b, 2004):

I. Criticality operation mode: The core configuration consists of 108 fuel elements of 20% enrichment, four B_4C control rods, some tubes loaded for various experiments, eleven graphite rods and a tungsten neutron source.
II. Subcritical operation mode: For k_s = 0.97–0.98, the structure is similar to the critical structure, except that 102 fuel pins in addition to three control rods have been loaded into the core.
III. Low level subcritical operation mode: This core configuration has been studied with 79 fuel pins and three control rods. One of control rods is fully extracted; the second one is positioned at 32% fuel −68% absorber and the third one is positioned at 46.5% fuel −53.5% absorber that results in k_s = 0.90–0.91.

Also, spallation target neutronic source parameters for two cases of uniform and parabolic beam spatial distribution have been investigated in accelerator driven subcritical TRIGA reactor. Moreover, these parameters have been calculated at k_s = 0.97 for proton energies of 115, 300, 600, and 1000 MeV using the MCNPX code. Details of the TRIGA core characteristics are presented in Table 1. A schematic view of the fuel rod is shown in Fig. 2.

2.2. Calculation of neutronic parameters

MCNPX code reads the input file, where the geometry, the materials, the neutron source and etc are described. Results of interest can be scored by using tallies. A tally is a specification of what should be included in the problem output, for example the neutrons flux (F2 or F4) through a certain area or the number of neutrons in a particular energy interval. In MCNPX code it is possible to calculate integrals of the following equation:

$$\mathrm{F2,4} = C\int \Phi(E)R(E)\,dE \qquad (1)$$

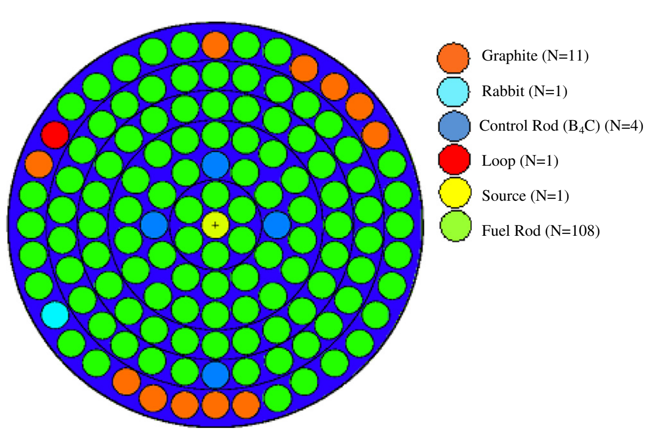

Fig. 1. Schematic view of the horizontal cross section of the TRIGA core.

Table 1
TRIGA core characteristics.

Description	Value
Initial fuel mixture	UZrH
Initial fuel mass (g)	235.2
Initial U concentration (%)	8.5
Initial fissile enrichment (%)	20
Proton beam energy (MeV)	115
Accelerator current (mA)	2
Number of fuel elements	116
Number of control rods (B_4C)	4
Diameter of the core (cm)	56
Height of the core (cm)	72
AISI-304 thickness (cm)	0.508
Density of AISI-304 (g/cm^3)	7.5
Density of fuel mixture (g/cm^3)	5.8
Density of clad (Zr) (g/cm^3)	6.49
Clad thickness (cm)	0.25
Density of graphite rod (g/cm^3)	2.25
Diameter of the graphite rod (cm)	8.7
Height of the graphite rod (cm)	3.63
Pin external diameter (cm)	3.73
Pin active length (cm)	38.1
Pitch (cm)	3.97
External fuel radius (cm)	1.815
Internal fuel radius (cm)	0.25
Fuel volume in a pin (cm^3)	387.407
Pin exchange surface (cm^2)	446.46

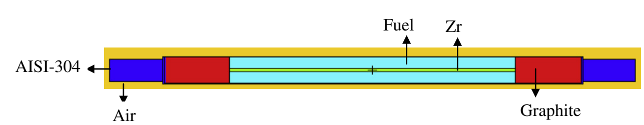

Fig. 2. A schematic view of the fuel rod in TRIGA core.

where C is a multiplication constant, $R(E)$ is any combination of sums and products of energy-dependent quantities known to MCNPX and $\Phi(E)$ is the neutron flux. In this way, reaction rates with different materials can be determined (Hughes et al., 2002).

Heating and energy deposition (MeV/g) can be determined by MCNPX code using F6 and F7 cell tallies according to the Eq. (2) (Hughes et al., 2002):

$$\mathrm{F6,7} = \rho_a/\rho_g \int H(E)\Phi(E)dE \qquad (2)$$

where ρ_a is atom density, ρ_g is gram density, $\Phi(E)$ and $H(E)$ are the neutron flux and heating response respectively. Power peaking factors within the core has been calculated using these tallies.

Another powerful property of MCNPX code is the possibility of criticality calculation using KCODE card to give an estimation of k_{eff}. Continuous energy cross section data from nuclear data evaluation file, version ENDF/B-VII.0, has been used. Additionally, thermal neutron scattering cross sections has been used from the ENDF/B-VI library (Hughes et al., 2002).

Neutronic parameters such as $Y_{n/p}$, k_s, k_{eff}, M, G_0, G, I_p, φ^*, P_{acc}, axial and radial distributions of neutron flux and P_{max}/P_{ave} ratio in two axial and radial directions of TRIGA reactor core model have been calculated for three eigen values levels (k_s) including: 0.91, 0.97 and 0.99.

In the present study, the parameter φ^*, which represents the relative efficiency of external source neutrons is defined as the ratio of the average importance of the external neutrons source to the average importance of the fission neutrons (Gudowski et al., 2001). The φ^* describes the difference between the real external source multiplication and the multiplication inherent to the distribution of neutrons corresponding to the fundamental mode according to the Eq. (3) (Seltborg, 2003, 2005):

$$\varphi^* = \left(\frac{1}{k_{eff}} - 1\right) \Big/ \left(\frac{1}{k_s} - 1\right) \qquad (3)$$