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Calculation of the power and absolute flux of a source driven subcritical assembly using Monte Carlo MCNP code

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ABSTRACT

The reactor power is an essential parameter used for calculating the coefficients of a nuclear reactor such as the absolute flux, reaction rate, power density, fuel burn-up, and the source term. The power is dependent on the driving neutron source strength in the subcritical reactors. In this work, MCNP5 code based on Monte Carlo method was used to model the 3D configuration of the subcritical assembly and its driving source. The aim of this paper is to calculate the amount of energy produced by the subcritical assembly, to estimate the absolute neutron flux along the radial and axial axis, and to determine other coefficients such as K_{eff} and ν. Using MCNP, the reactor power and flux computations were performed using two methods: the fixed source (nps) calculations and the criticality (K_{code}) calculations. This research compared the results obtained from the two methods, and discussed any significant deviation in the computation results.

1. Introduction

A subcritical assembly (SCA) is a nuclear reactor that cannot sustain a chain reaction in the absence of an external source of neutrons. This inherent safety feature of SCA makes its operation unsusceptible to reactivity accidents, thus making it one of the most suitable teaching tools for nuclear engineering students.

Subcritical systems have been proposed as feasible burners for the transmutation of actinides and fission products in spent nuclear fuel (SNF) (Chen et al., 2015; Wang et al., 2015; Velasquez et al., 2015; Bowman et al., 1992). Driven by a strong source of neutron from spallation reactions, the accelerator-driven subcritical system (ADS) has also been proposed as an energy-producing reactor (Clausse et al., 2015; Kapoor, 2002; Rubbia et al., 1995). In the design of a nuclear reactor, the reactor energy or reactor power is a crucial variable used for determining several important parameters such as the absolute flux, reaction rate, power density, fuel burn-up, and source term. These parameters cannot be calculated without determining the reactor power. The results of nuclear calculations are usually normalized per neutron or fission; thus, it must be scaled to the reactor energy to give a meaningful value.

Unlike critical reactors in which the reactor power is controlled and set at different levels, subcritical reactors power is dependent on the strength of driving neutron source. The aim of this work is to calculate the amount of energy produced by the subcritical assembly, to estimate the absolute neutron flux along the radial and axial axis, and to determine other coefficients such as K_{eff} and ν. Using Monte Carlo N-Particle Transport Code (MCNP), the flux computations will be performed using two methods: the fixed source (nps) calculations and the criticality (K_{code}) calculations.

1.1. Subcritical assembly description

The SCA at the Jordan University of Science and Technology (JUST) is a low-enriched uranium (LEU) oxide fueled light water reactor. It also contains 313 fuel rods, which are enriched with 3.4 wt% U^{235}, cladded with zirconium alloy (Zr-4) in a square lattice of 19.1 mm pitch, and loaded into a water-filled vessel to make up the core. Each fuel rod contains 43 pellets, each of which is 10 mm in height; thus, the total length of fuel rod holding the 216.173 g of uranium is 43 cm. The diameter of the core is 40 cm and its height, which is 55 cm, is positioned in the center of a water tank whose diameter and height are 120 cm and 132 cm, respectively, thus surrounding the core with 40 cm side reflector and 38.5 cm top-and-bottom reflectors. The driving neutron source is a plutonium-beryllium (Pu-Be) located under the core at the centerline (Xoubi, 2013). The subcritical reactor facility is mainly used as a practical educational tool for teaching and training nuclear engineering students in advanced universities, including JUST. A layout of the subcritical assembly (Xoubi, 2013) with the position of the Pu-Be source is shown in Fig. 1.

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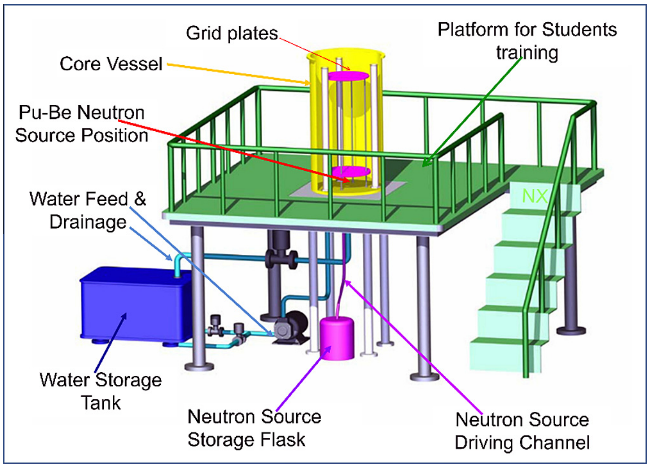

Fig. 1. Layout of the subcritical assembly, showing the location of the driving neutron source under the core grid plate, and other major system.

1.2. Fission energy

The total energy released from the fission of U^{235} nucleus is approximately 202 MeV. As depicted in Table 1, this amount of energy is distributed as kinetic energy of fission products, neutrons and radiation. On the one hand, most of this energy (88%) is deposited in the fuel within a short distance of the fission location; on the other hand, the remaining energy (12%), apart from the energy carried out by the neutrinos, is deposited within the reactor. The total recoverable energy is approximately 193.72 MeV, and this quantity of energy can be recovered from the reactor (Ma et al., 2013; Lamarsh and Baratta, 2013).

2. Methodology

In this research, Monte Carlo calculations were performed using MCNP code, which is capable of simulating complex three-dimensional nuclear systems of several types of radiation such as neutrons, photons, and electrons as well as their interactions (Monte Carlo team, 2003). MCNP methods can stochastically simulate nuclear processes by sampling the probability of individual events successively and by following each particle lifecycle, from its production stage to its destruction stage. This method of simulation accounts for all plausible events such as absorption, fission, scattering and escape (Monte Carlo team, 2003).

In MCNP, the calculations can be executed using two methods: the fixed source (nps) calculations and the criticality (K_{code}) calculations. In both cases, all the output tallies including surface current and flux, track length estimate of cell flux, point detector flux, track length estimate of energy deposition, track length estimate of fission energy deposition and other reactions are normalized per source neutron (Monte Carlo team, 2003), thus absolute values of these tallies require the use of the accurate normalization constant. In this paper, both types of calculations will be used to compute reactor power and the neutron flux absolute value.

2.1. Subcritical assembly modeling

The SCA is modeled based on the actual design parameters, and a three-dimensional, full detail MCNP model is developed for this nuclear reactor. This simulation model is used with the continuous energy neutron ENDF/B-VII cross-section data libraries and is developed as a nuclear analysis computational tool to calculate nuclear parameters and verify the design (Xoubi, 2013; Monte Carlo team, 2003) using the MCNP5 version 1.51.

Criticality and reactor physics calculations were also performed using the MCNP5 Monte Carlo code (Monte Carlo team, 2003), which has been verified and approved by the nuclear community worldwide. The MCNP-5 code is based on pure transport theory and is accepted by the USA NRC for performing such calculations. The fuel pellets, clad, helium, fuel elements loading arrangement, grid plates, water moderator/reflector and the vessel tank are explicitly modeled in this work. An x-z view of the reactor core, showing the reactor core surrounded by the water (purple) reflector and the neutron source located below the core, is shown in Fig. 2.

2.2. Reactor power calculations

The energy generated in the reactor and the reactor power were determined using fixed source calculations (nps) in MCNP. Also, the Pu-Be neutron source driving the assembly was inputted into the model as a volumetric source on the sdef card. The position of the neutron source, the weight of 1.191E+06 n/s and 3.455E+09 gamma/s, energy spectra, and the probability distribution of emitting neutrons and gammas at the corresponding energy are used in MCNP model to sample the distributions and to track the particle history. To calculate the reactor power of the SCA, the four tallies used in estimating the energy generated from the fission process in the core are as follows:

1. The energy is calculated from the total number of fissions in the fuel caused by the neutron source and then multiplied by the energy released by fission of U^{235} (193.72 MeV), as the main fission reaction is from this isotope.
2. The energy is calculated from the track length estimate of the fission energy deposition (F7) tally, which includes the energy of fission products, neutrons, and prompt gammas, thus accounting for 180.89 MeV out of 193.72 MeV. The F7 tally does not include the energy of the delayed gammas, betas, or capture gammas, all of which should be accounted for when determining the total power of the SCA. This tally gives the average energy over fuel cell (MeV/g); thus, the output is normalized by the cell mass.

Table 1
The distribution of energy released from the fission of U^{235}.

	Emitted Energy (MeV)	Range (cm)	Recoverable Energy (MeV)	Deposition site
Kinetic energy of fission fragments	169.12	0.0001	169.12	Fuel
Kinetic energy of prompt neutrons	4.790	1–10	4.790	Coolant
Kinetic energy of prompt gamma	6.970	10–100	6.970	Fuel cladding coolant
Kinetic energy of delayed neutrons	0.0074	1–10	0.0074	Coolant
Kinetic energy of delayed gammas	6.330	10–100	6.330	Fuel cladding coolant
Energy released by delayed betas	6.500	0.1	6.500	Fuel
Energy carried away by neutrinos	8.750	Infinity	0	Outside reactor
Total energy released per fission	202.47		193.72	