Progress in Nuclear Energy 108 (2018) 43–53

Contents lists available at ScienceDirect

Progress in Nuclear Energy

journal homepage: www.elsevier.com/locate/pnucene

Reactivity and flux characterization of the Jordan subcritical assembly

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ARTICLE INFO

Keywords:
Subcritical assembly
Reactivity
Criticality
Flux characterization
JSA

ABSTRACT

Nuclear subcritical assemblies are widely used for research, education, and training due to their inherently safe characteristics and operation at subcritical state. Following the construction of Jordan Subcritical Assembly (JSA) at the Jordan University of Science and Technology campus, a detailed computational neutronic analysis of the assembly was performed to investigate the criticality as well as the neutron spectrum within the facility. A three dimensional model of the JSA, as built, was developed using various computational tools including MCNP5, KENO-V.a (SCALE), and Serpent. The range of the k_{eff} of the JSA using different codes and libraries is 0.96151± 0.00009 – 0.96571± 0.00007 with a maximum difference of 420 pcm. The reference k_{eff} value was obtained at the optimal lattice pitch of 19.1 mm, moderator level of 132 cm, and reflector thickness of 40 cm. The effect of the change in the moderator temperature and fuel temperature on the reactivity of the JSA was investigated and found to be negative, which means that the facility is inherently safe. The value of the total effective delayed neutron fraction (β_{eff}) for the facility is 6.645E-03. The radial and axial flux characterization in the JSA were performed in this study. In addition, the neutron flux was simulated in the seven experimental channels in JSA which can be used for validation purposes. The neutron spectrum inside JSA was calculated at different axial locations using 252 energy groups and the results showed that the fast flux dominates the spectrum at regions close to the external neutron source and diminishes further away from the source.

1. Introduction

Subcritical assemblies are nuclear reactors that produce fission without achieving criticality. They are useful facilities for students, trainees and researchers to gain practical experience and to learn how to calculate the main reactor physics parameters. A reactor is subcritical if the amount of fissionable material present, coupled with its physical arrangement and other supplementary materials such as those that comprise the fuel lattice structure, moderators, reflectors, fuel rod cladding, etc., are not sufficient to achieve a self-sustained fission chain reaction. The reactor state is expressed by the effective multiplication factor (k_{eff}) being less than 1 (Vega-Carrillo et al., 2015). Subcritical reactors have been used as neutron sources, for energy production, education and research as well as spent fuel transmutation (Shahbunder et al., 2010). Such subcritical Assemblies include; Yalina-Booster subcritical assembly (Bécares et al., 2013; Talamo et al., 2011; Persson et al., 2005), Subcritical Assembly in Dubna (SAD) (Polanski et al., 2006; Shvetsov et al., 2006), Delphi subcritical assembly (Szieberth et al., 2015), and others. In 2007, Jordan University of Science and Technology (JUST) established the first and only Nuclear Engineering Department in Jordan, with a mission to provide Jordan's Nuclear Energy program with qualified nuclear engineers and support its human-capacity building program. Jordan Subcritical Assembly (JSA) is the first nuclear facility to be built in Jordan for the purpose of education, training, and experimental research.

Perssson et al. (Persson et al., 2005) performed reactivity analysis on the subcritical experiment Yalina using different reactivity determination methods. The methods considered were: the slope fit, the Sjöstrand, and the source jerk Methods. In addition, the results are compared with Monte Carlo simulations performed with different nuclear data libraries. The Monte Carlo simulations showed a good agreement with slope fit results. Kamalpour et al. (2014) investigated the effect of different fuel, moderator, reflector types on the criticality and flux of a subcritical assembly. The study is computational, in which the MCNPX code is used to perform the analysis. The paper concluded that subcritical assembly with UO_2 fuel have larger reactivity coefficient, lower criticality, and lower average total flux than metal U fuel. However, the assembly using the two fuel types are inherently safe due

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https://doi.org/10.1016/j.pnucene.2018.05.003
Received 7 January 2018; Received in revised form 20 April 2018; Accepted 9 May 2018
0149-1970/ Published by Elsevier Ltd.

to the negative temperature coefficient of reactivity for fuel and moderator. Polanski et al. (2006) presented a Monte Carlo modeling of a subcritical assembly in Dubna (SAD) to investigate the possibility of a power upgrade. SAD is loaded with MOX fuel and the study concluded after investigation of the effect of different parameters on the criticality that SAD could be run for specific experiments at a power level of 100 kW while k_{eff} is approximately 0.972.

Previous work on JSA was limited due to the latest establishment of this facility. Xoubi (2013) firstly introduced the structure of the facility with its objectives to fulfill the training, education, and research needs in JUST. This study reported the k_{eff} of the facility based on a core configuration of 313 fuel rods. Xoubi (2016) calculated the amount of energy produced by the JSA to estimate the absolute neutron flux in both radial and axial directions. The power and flux calculations were performed using two methods in MCNP5 (X-5 Monte Carlo Team, 2003): (1) the fixed source (nps) and (2) the criticality (Kcode) calculations. The two methods revealed that JSA is nearly zero power with an order of magnitude of 10^{-5} W. However, the previous works did not address the sensitivity of criticality of JSA if operating or geometrical conditions change. In addition, experimental evaluation of neutron flux and multiplication factor were not addressed in any of the previous works. Kinetic parameters of the facility were not also addressed in previous literature.

Following the construction of Jordan Subcritical Assembly (JSA); a detailed computational and experimental study of the assembly should be performed. In this paper, a detailed computational analysis of the JSA using Monte Carlo simulation is presented. In particular, criticaltiy, reactivity, neutron flux, and spectrum investigation of the JSA facility is performed using Monte Carlo code MCNP5. The work discusses neutronic and design analysis of JSA. Neutronic parameters such as effective multiplication factor, optimal fuel lattice pitch, optimal moderator level, optimal reflector thickness, and neutron flux distribution in the facility were determined. The effect of moderator temperature and fuel temperature on the reactivity of JSA was also studied along with the determination of critical fuel loading for subcritical assembly with UO2 fuel type and Zircaloy cladding. The radial, axial, and 3D neutron flux in different energy ranges were calculated in this study.

The remaining sections of this paper are organized as follows: Section 2 presents the configuration and structure of the JSA facility, which was modeled in MCNP5. Section 3 describes the MCNP5 model developed for this study to investigate the criticality and flux of the facility. Section 4 presents the results obtained by this study along with the discussion of these results. The conclusions are presented in Section 5.

2. Jordan subcritical assembly description

The JSA was constructed for the main purpose of training, education, and research activities. JSA adopts a vertical platform structure. The core vessel is placed on a supporting structure with an operating platform surrounding the vessel. The operating platform is used for the purpose of fuel loading and other operation and maintenance activities. Fig. 1 shows the general layout of JSA. The layout is designed to allow students and operators to change core configuration, work safely close to the reactor core, visualize the all reactor features including fuel and moderator.

JSA is designed to remain below criticality at all operating conditions. In this study, this subcriticality is verified. JSA sustains the chain reaction with the presence of external neutron source. The structural pattern of the commercial PWR fuel rod is used in JSA. The fuel pellet is ceramic UO2, with a U-235 enrichment of 3.4 wt %, an He-3 gap, and Zr-4 alloy as cladding material. The total number of fuel pellets in each fuel rod is 43 pellets with a pellet diameter of 0.843 cm. The active height of the fuel rod is 43 cm, with a total height of 55 cm. The fuel is moderated and reflected by high deionized distilled light water. The core is filled with water when the facility is in use and the discharge system drains the water when the facility is idle. The amount of water used is larger than what is required to reflect the neutrons back to the core so that the water also acts as a neutron shield.

The fuel rods are arranged in square arrays and supported by upper and lower grid plates. The upper grid plate is made of organic material Plexiglas, whereas the lower grid plate is made of aluminum alloy. The grid plates can hold up to 313 fuel rods with 1.91 cm lattice pitch, as illustrated in Fig. 2. The outside holes are used to hold spare fuel rods when changing the core configuration. In this work, 17 extra fuel rods were distributed uniformly in the outside holes making a total of 330 fuel rods. This was done to calculate the maximum effective multiplication factor achievable with this core configuration. Fuel specifications and design parameters are listed in Table 1 while a sketch of the JSA core that shows the distribution of the fuel rods and the experimental channels is shown in Fig. 3.

A pneumatically controlled ^{239}Pu-Be neutron source is used to drive JSA. The emission rate of the neutron source is approximately $1.10 \times 10^6 n/s$. When considering the purpose of the assembly for education, training, and experimental research, and to minimize the effect of the neutron source on the neutron environment, the source is driven to a position just below the active core. The control of JSA is achieved mainly by the movement of the neutron source from its storage container below the loading platform into the core. The control room of JSA allows for analysis, monitoring, display, and saving of the experiment data. The control room is also equipped with monitors, cameras, and dosimeters to control the radiation levels.

3. Methodology

In this section, the computational tools used in this study are described. The 3D model for the JSA is presented along with the data libraries used in the simulation. The methods used to calculate the neutron spectrum and the kinetic parameters of the JSA are also discussed.

3.1. Computational tools

The importance of modeling and simulation in many fields, especially engineering, arises from the fact that there are many obstacles that may impede the experimental work, like severity of the experiment, high cost of some experiments as well as incompatibility of the time frame of the experiments with human response. In reactor physics, there are a number of parameters that need to be determined both through calculation and then measured experimentally. Such parameters like reactivity need to be estimated with high accuracy through modeling of the core of the reactor. Deterministic as well as stochastic modeling of Material Testing Research Reactors have been successfully applied (Malkawi and Ahmad, 2000, 2001; Malkawi et al., 2013). Monte Carlo methods differ from deterministic transport methods in several ways. Deterministic methods, the most common of which is the discrete ordinates method, solves the transport equation for the average particle behavior. Conversely, Monte Carlo methods are a class of algorithms based on repeated sampling of probability distributions using random numbers to compute their results. These algorithms are used in simulating physical and mathematical systems with a large number of coupled degree of freedom, such as those found in radiation transport. The Monte Carlo approach obtains results by simulating individual particles and recording some aspects (tallies) of their average behavior.

In this work, three different Monte Carlo codes were used to simulate the JSA facility, namely, MCNP5, Serpent, and KENO-V.a. MCNP5 (Monte Carlo N-Particle transport code) is developed at Los Alamos National Laboratory and it is one of the most frequently-used Monte Carlo neutron transport codes (X-5 Monte Carlo Team, 2003). Serpent is a three-dimensional continuous-energy Monte Carlo particle transport code, developed at VTT Technical Research Centre of Finland, Ltd (Leppänen et al., 2015). KENO-V.a is a three-dimensional Monte