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The Method to Predict First Critical Core Loading for Nuclear Reactors

Senem ŞENTÜRK LÜLE[*1]

Abstract

Nuclear power plants have an important role in carbon free electricity production in the world. One of the important steps of commissioning a nuclear power plant is the first core loading. This is also called approaching the criticality. Since the number of fuel elements for the criticality is not known, precautions must be taken to prevent safety incidents. Although the procedure is performed on-line such that the neutron counts are measured at each loading of fuel elements to calculate sub-critical multiplication and the number of fuel element to reach criticality were predicted, computer simulations can also be used. In this study, inverse sub-critical multiplication method was applied to Istanbul Technical University TRIGA Mark II research reactor first criticality in 1979 by using Monte Carlo simulation code MCNP6.2. Full 3-D model of the reactor was generated for calculations. Both results, experimental and simulation, showed that reactor became critical with 62 fuel elements. The core excess reactivity of 23.1 cents was predicted as 21.7 with the code. The simulation results are in good agreement with experimental results. The methodology and simulations can be used for power reactor analysis as well.

Keywords: Criticality approach, Monte Carlo method, sub-critical multiplication, reactor start-up

1. INTRODUCTION

Currently, 450 operable nuclear power plants with 399 GW_e total installed capacity are producing 2563 TWh carbon free electricity [1]. In addition, 55 GW_e net installed capacity is going to be utilized when 53 nuclear power plants under construction start commercial operation [2]. One of those 53 plants is the one that is being constructed in Akkuyu site in Turkey. Akkuyu project involves the construction of four 1200 MWe VVER-1200 type power plants [3]. The first unit is expected to be online in 2023 [4]. A successful commissioning of a nuclear power plant requires successful initial reactor start-up which involves hydraulic tests, pressure tests, plant heat-up, start-up to minimum load, etc… As a matter of fact, it may take several months. It is very important at the initial reactor start-up that the approach to criticality be performed very slowly and carefully since the actual fuel mass or number of fuel elements required for criticality is unknown. In addition, throughout its lifetime, a nuclear reactor can be started up for various reasons such as start-up after normal shutdown or after refueling. All these start-ups include the step of approaching criticality or start-up to minimum load. This step requires the final effective multiplication factor (k_{eff}) of the reactor becoming one at the end of the start-up [5]. Therefore, it is important to estimate critical conditions such as critical rod positions, critical core inlet temperature, and critical boron

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concentration if power plant is a pressurized water reactor (PWR). Approach to criticality can be measured with subcritical multiplication (M) which reflects the effect of reactivity change on neutron flux [6]. It can simply be defined as the inverse of $(1 - k_{eff})$.

The knowledge of the degree of subcriticality is important not only to understand the reactor response but also to satisfy criticality safety control. If obtained in a timely manner during the operation of nuclear facilities, it could lead to the application of advanced control methods such as more positive usage of neutron absorbers [7].

In general, power plant data are not available to researchers. Therefore, the general practice is to utilize experimental facilities for research and development on criticality safety. There are several experimental facilities in the United States, Japan, Russian Federation, and Europe that are used for criticality safety [7]. On the other hand, research reactors are used to acquire data for research and development activities for many years in nuclear field. In fact, research reactors play an important role in nuclear industry not only for research and development but also human resource development with their flexibility at core designs, powers, flux levels, fuel element types, fuel element shapes, and experimental facilities. The criticality approach can also be experimented in research reactors. There are also computational methods for criticality predictions.

The first criticality prediction and experiment for the Jordan Research and Training Reactor owned by Jordan Atomic Energy Commission (JAEC) was performed and reported in [8]. The Monte Carlo code for Advanced Reactor Design and Analysis (McCARD) with ENDF/B-VII.0 cross section libraries was used for the prediction calculations. The simulation results for the critical control rod position showed good agreement with the experimental data. Least square inverse kinetics method was employed to measure reactivity with source term for the HANARO Research Reactor [9]. This method is widely used for power reactors at high power level when there is no neutron source at the core. But, at subcriticality, the effect of neutron source must be taken into account. The method is proved to be successful to predict reactivity worth. The validation of reactor physics and criticality safety code SCALE 5.1 KENO V.a for seven weight percent ^{235}U fuel was performed with benchmark data from the seven percent critical experiment [10]. The experiment involved the criticality approach procedure. The results of the number of fuel elements for criticality from the experiments and simulations showed a large difference. The first fuel loading of HANARO research reactor was performed both experimentally and analytically in [11]. The fuel elements inserted in the core batch wise and subcritical multiplication versus number of fuel elements graph was drawn. The reactor became critical when four 18-element assemblies and thirteen 36-element assemblies were inserted in the core which required the critical control rod position as 600.8 mm with excess reactivity of 0.71$. WIMS-VENTURE and MCNP codes were used for the simulations. The predicted criticality overestimated the experimental value for both codes.

In this study, the first criticality approach experiment of Istanbul Technical University (ITU) TRIGA Mark II Research Reactor in 1979 was described and then used to develop and verify a Monte Carlo model with MCNP6.2 code [12]. The verified model can be used for future start-up procedures of ITU TRIGA reactor in case there is a new core configuration. In addition, this study can be beneficial if calculations are required for Akkuyu nuclear power plant.

2. ITU TRIGA MARK II RESEARCH REACTOR

The construction of ITU TRIGA Mark II research reactor started in 1977 and first criticality was achieved in 1979. The reactor is designed by General Atomics. It has a 250 kW nominal power and 1200 MW pulse capacity. The light water cooled and graphite reflected reactor core is placed aboveground Aluminum tank of approximately 2 m diameter and 6.4 m height which is centered in a hexagonal reactor structure of heavy concrete for radiation protection and structural integrity [13]. Top and side view of ITU TRIGA Mark II research reactor are shown in Figure 1. Currently, there are 69 fuel elements, which are composed of 19.75% low enriched Uranium Zirconium Hydride ($UZrH_{1.6}$) fuel meat surrounded by stainless steel clad, in the reactor core. The reactivity control is achieved with three control rods: Safety, Regulating, and Transient. There are several irradiation facilities in the core for research and development studies. The central thimble and pneumatic transfer system provide in-core whereas radial, piercing, and tangential beam ports provide out-core irradiation opportunities. Since reactor power is relatively low, natural cooling is enough to remove the heat that is generated by fuel elements from the core [14].

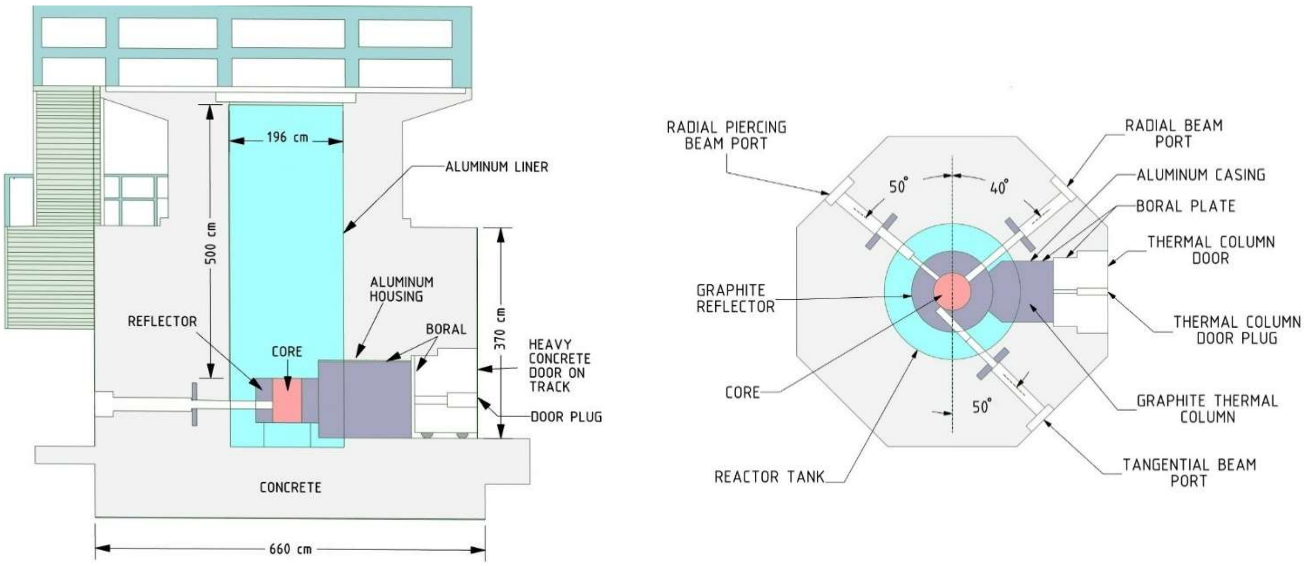

Figure 1 Side and top view of ITU TRIGA Mark II research reactor

3. METHODOLOGY

The multiplication factor of a reactor is defined as in Eq.1 and indicates that the neutrons from one fission reaction induce yet another reaction [15]. If the number of neutrons in one generation is greater than the number of neutrons in preceding generation, k_{eff} becomes greater than one and the reactor is called "*super-critical*". On the contrary, if the number of neutrons are decreasing between two generations, k_{eff} becomes less than one and the reactor is called "*sub-critical*". The ideal condition is where there is balance between the number of neutrons in each generation that results in "*critical*" reactor with k_{eff} equals to one.

$$k_{eff} = \frac{Number\ of\ neutrons\ in\ one\ generation}{Number\ of\ neutrons\ in\ preceding\ generation} \tag{1}$$

Under normal circumstances, a sub-critical reactor will never be self-sustaining since the number of neutrons decreasing in time (with the mean generation time). On the other hand, if there is an external neutron source in the core, neutron population reaches an equilibrium level that is determined by the neutron source strength S such that it can be measured by neutron detectors to provide information to reactor operator. The number of neutrons at any generation for a sub-critical reactor with an external source can be calculated by using Eq.2 [16].

$$\frac{n}{S} = M = \frac{1 - k_{eff}^{m+1}}{1 - k_{eff}} \tag{2}$$

where n is the neutron density level at the m^{th} iteration, S is external source strength, and M is sub-critical multiplication of the reactor. After sufficiently long time, the number of neutrons in the core takes the form defined in Eq.3.

$$n = S \times M = S\frac{1}{1 - k_{eff}} \tag{3}$$

It is not practical to use M to follow criticality approach since as reactor approaches criticality when k_{eff} approaches to one, M becomes infinitely large. Instead, the inverse of sub-criticality multiplication $(1/M)$ can be used since its value will be zero at the point of criticality. Therefore, the approach to criticality is performed by loading fuel elements in the core in batches, measuring the count rates on the detectors after each batch was loaded, and plotting $1/M$ as a function of number of elements loaded. The critical mass then can be predicted by extrapolating the $1/M$ versus the number of fuel elements curve to the horizontal axis. When approach to criticality is performed, long time lapses between the batches are required in order to permit the equilibrium state to be reached. This is particularly important when the reactor gets close to critical. After the last fuel is loaded, the reactor will become slightly super-critical. The core excess reactivity then can be measured by determination of doubling time.

4. RESULTS AND DISCUSSIONS

In this study, criticality approach experiment of the first core loading of ITU TRIGA Mark II research reactor in 1979 was simulated with MCNP6.2 Monte Carlo code. The detailed 3D model of the research reactor was generated for this purpose and can be seen in Figure 2. The Monte Carlo calculations were performed with 45000 initial number of neutrons to complete 2200 active cycles skipping 200 of them to allow Shannon entropy to converge to achieve steady-state value for fission source distribution. ENDF/B-VII.1 cross section library was used. With these arrangements, the standard deviation of the calculated k_{eff} was guaranteed to be below 10^{-3}. The values of active and skipped cycles define not only the magnitude of the error in the results but also the computational time. An increase in active cycles reduces the error but increases the computational time. The value for the skipped cycles must be arranged in a way to allow Shannon entropy to converge. The combination used in this study provided the optimum in terms of accuracy and computational time.

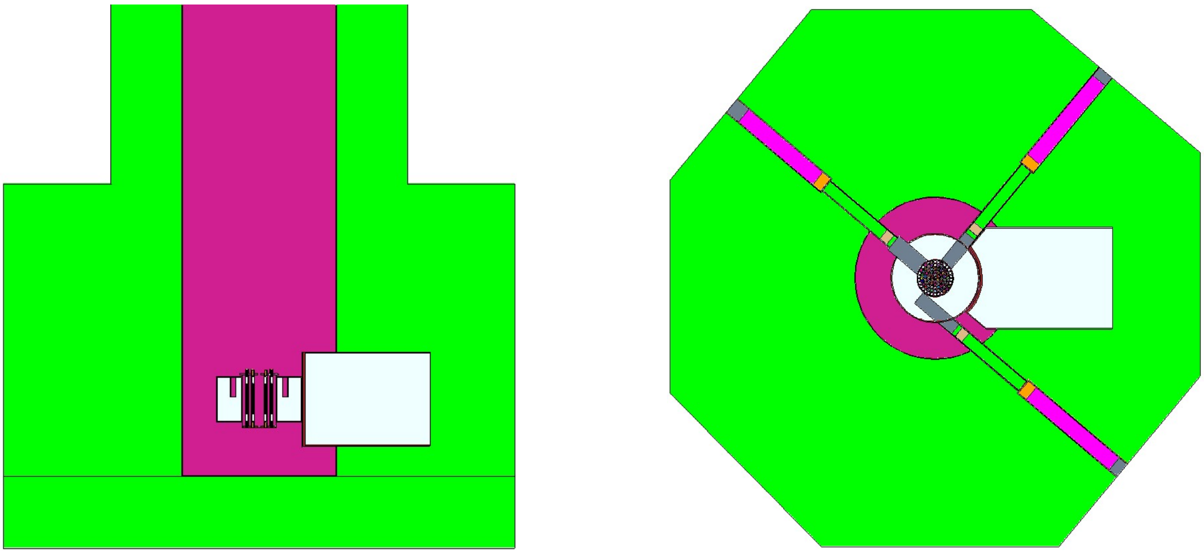

Figure 2 Side and top view from the MCNP model of ITU TRIGA Mark II research reactor

The experiment in 1979 started with withdrawal of all control rods out of the core. Then, 3 Ci Am-Be neutron source providing 3 x 10^6 neutrons/s was placed into the core and count rate was recorded. According to experimental data, 24 fuel assemblies were loaded into the reactor core as a first step. Later, 10, 6, 3, 3, 2, 2, 1, 1, 1, and 1 additional fuel assemblies were loaded into the core. After each step, enough time lapse was given to system to reach steady-state before recording the counted data. The inverse subcritical multiplication was calculated by division of count rates between two steps. Whenever a new data point was obtained, that point and the previous point was used to predict the critical mass by linear extrapolation [17]. When the last fuel was added, the reactor became super-critical with a period of 29.3 s.

The same core loading pattern described above was applied to MCNP simulations. On the contrary to experiments, the sub-critical multiplication was determined by using calculated k_{eff} values. The sample core configurations for 24, 46, 56, and 62 elements that are used for the simulations are shown in Figure 3.

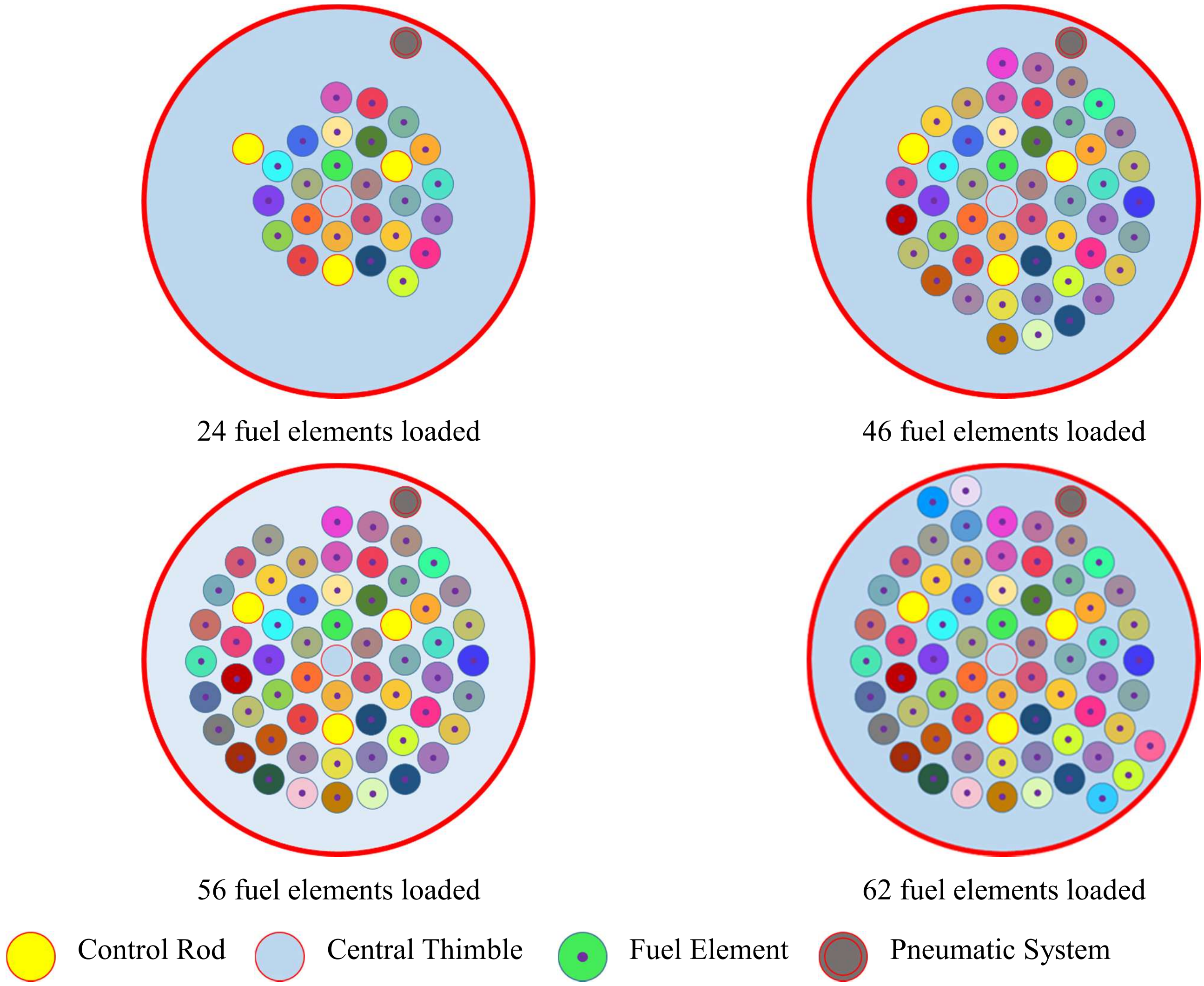

Figure 3 Core loading pattern with various number of fuel elements

The graph of inverse sub-critical multiplication versus the number of fuel elements including both experimental and simulation results is shown in Figure 4. As seen in Figure 4, addition of fuel elements decreases inverse sub-critical multiplication indicating that the core multiplication factor increases therefore it approaches to criticality. It is clear that at the beginning MCNP over predicts core criticality. It is because of the fact that in reality there is not enough fission in the core but MCNP simulates the core as if all materials including fissile isotopes fission while performing criticality calculation. As a result, it over predicts the core multiplication factor. As number of fissions increases with the addition of fuel elements, the results of MCNP and experiment agrees quite well. The error bars on MCNP curve in Figure 4 shows this agreement clearly.

As mentioned above, the prediction of the number of fuel elements that makes the core critical is performed by using the results of two consecutive fuel loading steps. The curves in Figure 4 were used to predict the number of fuel elements required for criticality after each core loading step. Table 1 shows that at first, the prediction is far from the actual value since there is not enough fission in the core. Later, with the addition of new fuel elements the result converges to the actual value. Table 1 indicates that the experimental results are comparable with the simulation results therefore concludes that the simulations and methodology used in this study are appropriate.

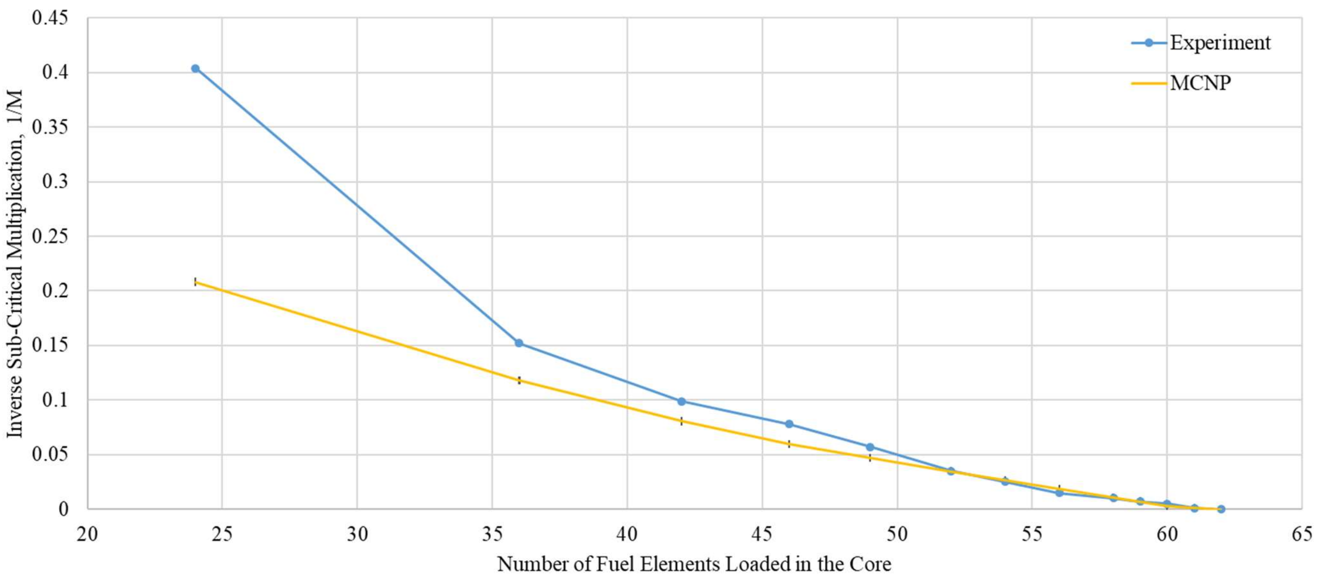

Figure 4 The variation of inverse subcritical multiplication factor with core fuel loading

Table 1
The predictions of experiment and MCNP simulation for the number of fuel elements for criticality

The number of fuel elements loaded	Prediction for the number of fuel elements for criticality	
	Experiment	MCNP
24	-	-
36	43	52
42	53	55
46	61	57
49	57	60
52	57	61
54	59	61
56	59	60
58	62	61
59	61	61
60	62	61
61	61	61

The reactor period at super-criticality state can be used to determine the excess reactivity of the core by using in-hour equation shown in Eq. 4 [16].

$$\rho = \frac{l}{l+T} + \frac{T}{T+l}\sum_{i=1}^{6}\frac{\beta_i}{1+\lambda_i T} \qquad (4)$$

where l is prompt-neutron lifetime, T is reactor period, β_i is the i^{th} group delayed neutron fraction, and $\lambda_i$is the i^{th} delayed neutron group decay constant.

By using the 29.3 s period reported in the experiment, the reactor excess reactivity was calculates as 23.1 cents. The MCNP simulations predicted the excess reactivity as 21.7 cents. It is clear that the result of the simulation agrees well with the experimental result.

5. CONCLUSIONS

The detailed 3D model of the ITU TRIGA Mark II research reactor was generated with MCNP6.2 Monte Carlo code geometry modelling feature to perform calculations to predict the number of fuel elements required to achieve first criticality. The approach to criticality is an important step for the commissioning of nuclear reactors either power or research. The precautions must be taken to prevent safety incidents. Therefore, it is important to have accurate predictions. The inverse sub-criticality method was employed in this study and the experimental results were compared with the simulation results. The simulation results predicted well the number of fuel elements required for the first criticality. The results deviated from the experimental results when there is not enough fission reaction at the reactor core due to small amount of fuel elements loaded in the core. Since MCNP performs criticality simulations assuming all fissile materials fission, which is not the case in reality, it overestimates the core multiplication factor at the beginning of the core loading. On the other hand, this is not a problem when there are great number of fission reactions in the core. The calculated core excess reactivity also agrees well with the experimental result indicating that MCNP6.2 with ENDF/B-VII.1cross section library is capable of accurate simulation. The method of approaching the first criticality can be used for the power reactors as well.

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The author declares that this document does not require an ethics committee approval or any special permission.

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