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A time dependent Monte Carlo approach for nuclear reactor analysis in a 3-D arbitrary geometry

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ABSTRACT

A highly reliable tool for transient simulation is vital in the safety analysis of a nuclear reactor. Despite this fact most tools still use diffusion theory and point-kinetics that involve many approximation such as discretization in space, energy, angle and time. However, Monte Carlo method inherently overcomes these restrictions and provides an appropriate foundation to accurately calculate the parameters of a reactor. In this paper fundamental parameters like multiplication factor (K_{eff}) and mean generation time (t_G) are calculated using Monte Carlo method and then employed in transient analysis for computing the neutron population, proportional to K_{eff}, during a generation time considering precursors decay. Based on this approach, a dynamic Monte Carlo code named MCSP (Monte Carlo dynamic Simulation of Particles tracking) is developed for both the steady state and time-dependent simulation of particle tracking in an arbitrary 3D geometry. MCSP is able to use either continuous or multi-group energy cross section libraries. To speed up the simulation, the MCSP was empowered with parallel processing as well. Several test problems such as C5G7, LMW and TWIGL are examined to assess the performance of the method.

1. Introduction

A rigorous safety analysis is needed to predict the behavior of nuclear reactor in either steady-state or transient modes. The time dependent angular flux spectrum in a time-variable system can be calculated by the time-dependent transport equation which may be written as (Bell and Glasstone, 1970):

$$\frac{1}{v(E)}\frac{\partial\phi(\vec{r},E,\hat{\Omega},t)}{\partial t} + \Omega.\nabla\phi(\vec{r},E,\hat{\Omega},t) + \Sigma_t(\vec{r},E)\phi(\vec{r},E,\hat{\Omega},t)$$
$$= \chi_p(E)(1-\beta)\int_0^\infty\int \bar{\nu}\Sigma_f(\vec{r},E')\,\phi(\vec{r},E',\hat{\Omega}',t)\,dE'd\Omega'$$
$$+ \sum_j \chi_{dj}(E)\,\lambda_j\,C_j(\vec{r},E)$$
$$+ \int_0^\infty\int \Sigma_s(\vec{r},E'\to E,\Omega'\to\Omega)\,\phi(\vec{r},E',\hat{\Omega}',t)\,dE'd\Omega'$$
$$+ S_{ext}(\vec{r},E,\hat{\Omega},t) \quad (1)$$

The corresponding time-dependent precursor equation is:

$$\frac{\partial C_j(\vec{r},t)}{\partial t} = \beta_j\int_{4\pi}\int_0^\infty \bar{\nu}\Sigma_f(\vec{r},E')\,\phi(\vec{r},E',\hat{\Omega}',t)\,dE'd\Omega' - \lambda_j C_j(\vec{r},t) \quad (2)$$

Where $\Sigma_s(\vec{r},E'\to E,\Omega'\to\Omega)\,dE'd\Omega'$is the scattering cross section that of a neutron of direction Ω' and energy E' has a collision, there will scatter a neutron in a direction interval $d\Omega$ about $\hat{\Omega}$with energy in dE about$E$, $\Sigma_t(\vec{r},E)$ and $\Sigma_f(\vec{r},E)$ are total and fission cross section respectively, $\phi(\vec{r},E',\hat{\Omega}',t)$ is the angular flux and $S_{ext}(\vec{r},E,\hat{\Omega},t)$ is the external neutron source, C_j , λ_j and β_j are the population, decay constants and fraction of delayed neutron precursor fission fragment of j^{th} group, β is the total fraction of delayed neutrons, $v(E)$is the neutron speed with energy E and $\bar{\nu}$ is the average number of neutron released per fission.

Since finding an analytic solution for Eq. (Hoffman, 2013) and Eq. (Argonne Code Center, 1977) is difficult except for a few isolated simplified cases, numerous simplifications like spatial homogenization and multi-group energy collapsing are routinely invoked in engineering solutions (Bell and Glasstone, 1970). The complexity of a time varying geometry is usually ignored by replacing them with a fixed system as well. However, the study of time-dependent geometry becomes unavoidable in some scenarios e.g. the investigation of neutron noise caused by the control rod vibration or in the rod ejection transient. Adopting the mentioned simplifications makes the validity of the answer questionable. A reliable alternative for removing the limitations is the Monte Carlo (MC) method which directly simulates the particle

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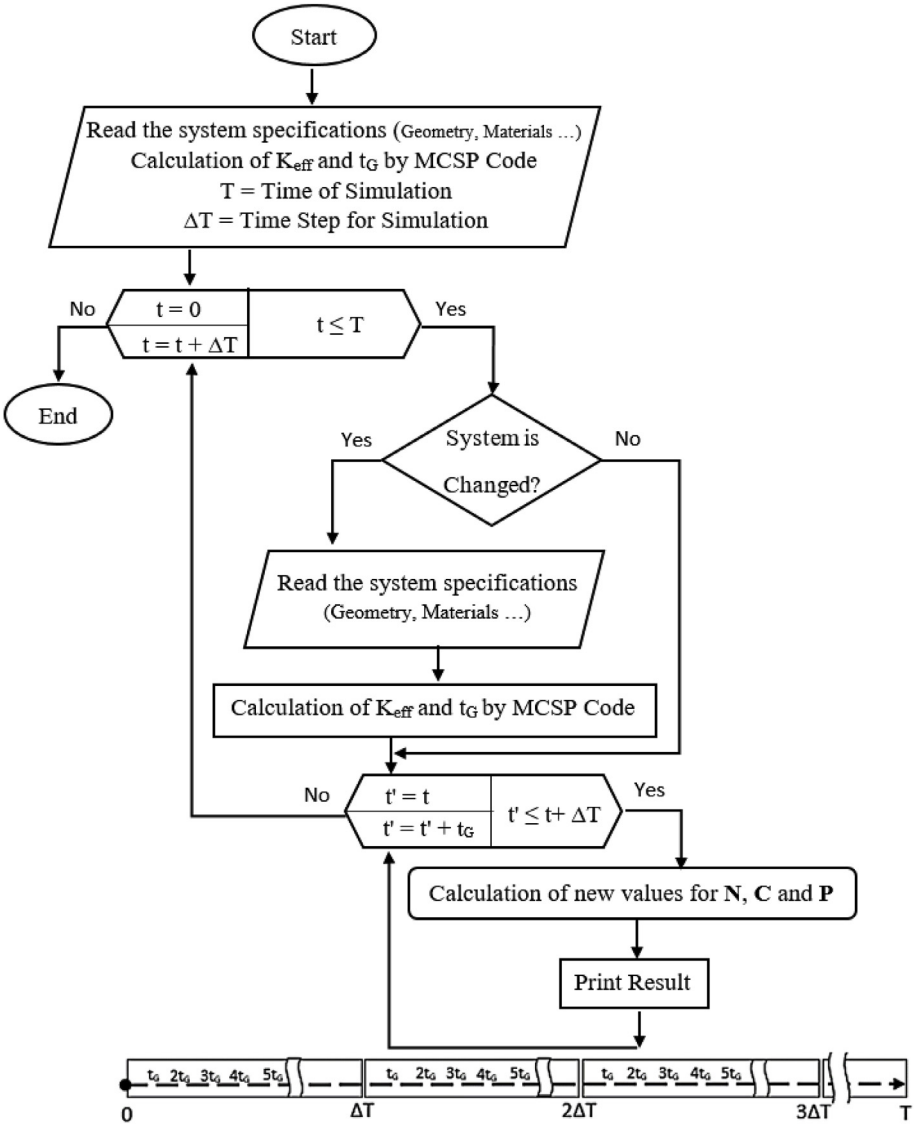

Fig. 1. Flowchart of MCSP for transient analysis.

Table 1
Cross section for benchmark problem 1.

Σ_t (cm^{-1})	Σ_a (cm^{-1})	Σ_f (cm^{-1})	Σ_s (cm^{-1})
1.0000 → 0.9988	0.5882 → 0.5870	0.2500	0.4118

V = 2.2E+4 cm/s, ν = 2.5.

Table 2
Delayed neutron parameters for the benchmark problem 1.

Family	λ (s^{-1})	β
1	0.0127	0.000260
2	0.0317	0.001459
3	0.1156	0.001288
4	0.3110	0.002788
5	1.4000	0.000877
6	3.8700	0.000178

transport phenomena as a stochastic process. The MC does not solve any explicit equation, but estimates the answer through the mean behavior of particles following the record of histories. The MC technique has proved useful in special cases where other methods encounter difficulties. Moreover, it is perfect for detailed variation of cross sections with energy and time, or where the treatment of a moving/fluctuating boundary is necessary. In sum, the MC technique is well suited for the analysis of complicated three-dimensional, time-dependent problems.

There are two main approaches to analysis the time-dependent behavior of a nuclear system. The first one is to turn the problem into an eigenvalue system. Although the system eigenvalues can be classically calculated using deterministic methods, more accuracy is still achievable using the MC based schemes. The time evolution of the power can be approximated, using point-kinetic equations with the largest eigenvalue, (Shayesteh and Shahriari, 2009; S. Yun et al., 2008; Arzhanov, 2002; Abdou, 2005). The second approach is a direct solution of the problem within time steps. The mission might be carried out using several solution techniques including quasi-static methods (J. Kotchoubey, 2015; R. E. Alcouffe and Baker, 2005; S. Goluoglu and H. Dodds, 2001; J. C. Gehin, 1992; K. S. Smith, 1979), Monte Carlo methods (Sjenitzer and Hoogenboom, 2011) and the Method of Characteristics (S. C. Shaner and thesis, 2014; Adam J. Hoffman, 2013).

Based on the MC method, a code named MCSP (Monte Carlo dynamic Simulation of Particles tracking) is developed to simulate the time-dependent transport of neutrons, photons and electrons and positrons in an arbitrary geometry. The concept of fundamental parameters such as multiplication factor (K_{eff}), mean generation time (t_G)

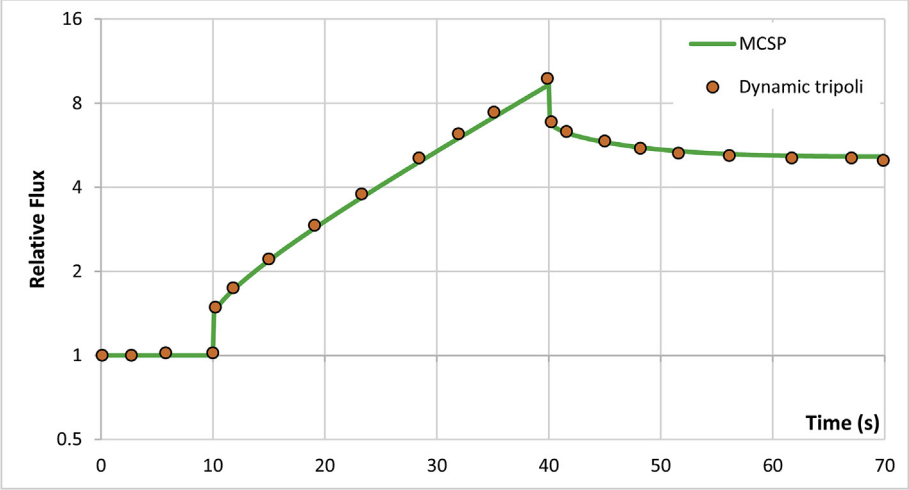

Fig. 2. Relative flux vs. time for benchmark problem 1.

Table 3
The total calculation time for the benchmark problem 1.

1 CPU (Hours)	48 CPUs (Hours)
11.3	0.3

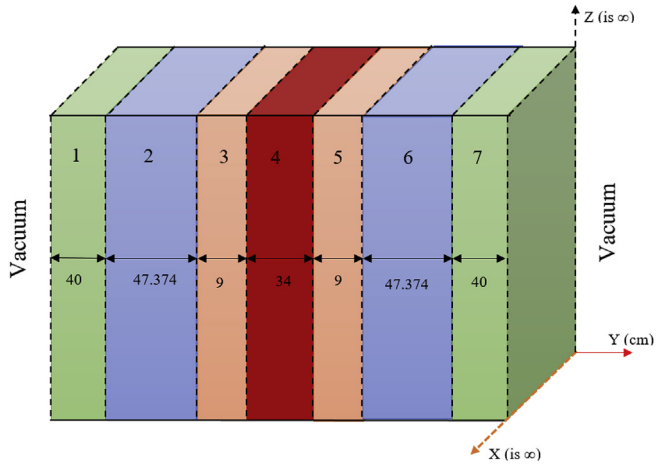

Fig. 3. Schematic of benchmark problem ANL 16-A1.

and delayed neutron precursors are computed by MCSP and applied in the proposed approach. Neutron population grows with K_{eff} in a generation time and precursors decay during this period. The MCSP code is able to calculate time-dependent flux as well as neutronic parameters such as the effective multiplication factor, neutron life time, etc. MCSP might be used either with continuous or multi-group energy cross section library. Rigorous test problems such as C5G7, LMW, TWIGL, etc.

Table 5
Delayed neutron parameters for benchmark problem ANL 16-A1.

Family	$\lambda(s^{-1})$	β
1	0.0129	0.000081
2	0.0311	0.000687
3	0.1340	0.000612
4	0.3310	0.001138
5	1.2600	0.000512
6	3.2100	0.000170

are examined to evaluate the performance of the method. Results are in satisfactory agreement with the reference results.

It is noteworthy that the computation cost depends on many parameters such as the number of particles started (NPS), the multiplication factor, mean generation time of the system, the total time simulated, etc. To speed up the simulations, process is performed in parallel and the elapsed time is given for 1 CPU and 48 CPUs to show advantage of parallel processing. The processors used were 2.2 GHz AMD Opteron (tm).

The rest of the article is organized as follows: In section 2, the methodology employed for the work is described in detail. Numerical simulations and benchmark problems are studied in section 3. The paper is finalized with a discussion on the results as well as a conclusion in section 4.

2. Methodology

The behavior a neutron is considered stochastically within a system based on the MC method. A neutron is absorbed, scattered or leaked out in its life time (t_G), which is too short, while the density of delayed neutron precursors changes slowly and the conditions of a system do not change in this short time. Therefore, the system is treated similar to

Table 4
Cross section for benchmark problem ANL 16-A1.

Region	Group	$\Sigma_t\ (cm^{-1})$	$\Sigma_a\ (cm^{-1})$	$\Sigma_f\ (cm^{-1})$	$\Sigma_s^{g\to g}\ (cm^{-1})$	$\Sigma_s^{g\to g'}\ (cm^{-1})$
1 and 7	1	2.411E-1	3.902E-3	8.3441E-4	2.336E-1	3.598E-3
	2	4.172E-1	1.020E-2	3.2776E-4	4.070E-1	0.0
2, 4 and 6	1	1.849E-1	5.115E-3	7.4518E-3	1.777E-1	2.085E-3
	2	3.668E-1	1.310E-2	1.1061E-2	3.537E-1	0.0
3 and 5	1	9.432E-2	6.893E-3	0.0	8.571E-2	1.717E-3
	2	1.876E-1	1.630E-2	0.0	1.713E-1	0.0

V_1 = 5.4E+8 cm/s, V_2 = 9.19E+7 cm/s, X_1 = 1.0, X_2 = 0.0, ν = 1.0016.

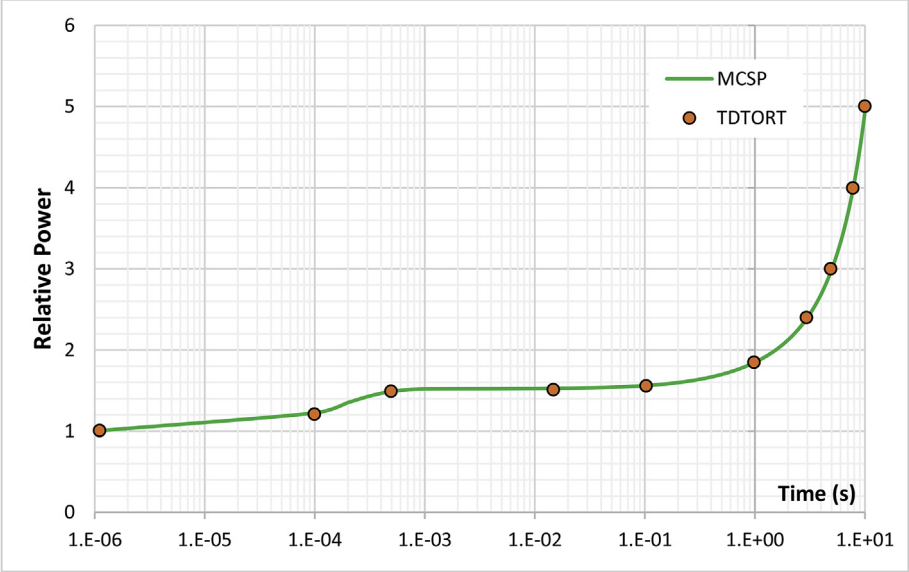

Fig. 4. Relative power vs. time for benchmark problem 16-A1.

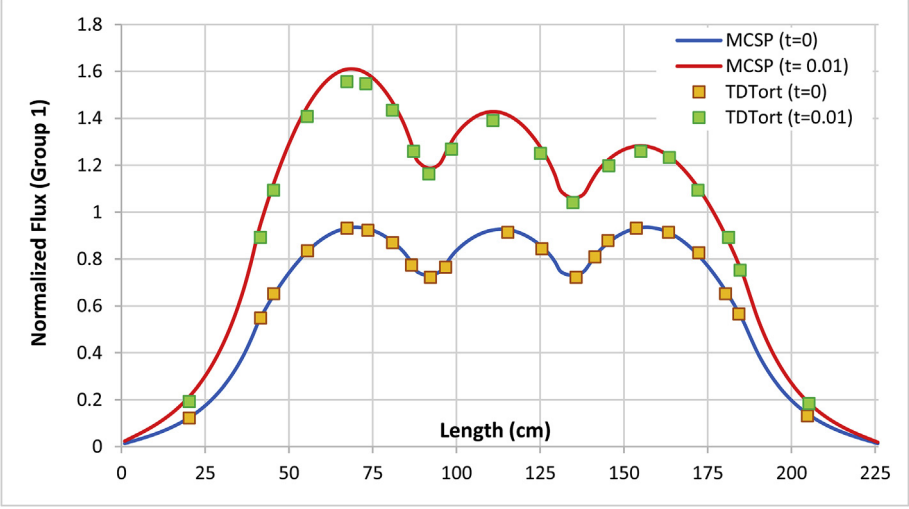

Fig. 5. Normalized flux vs. radial position for problem 16-A1 (group 1).

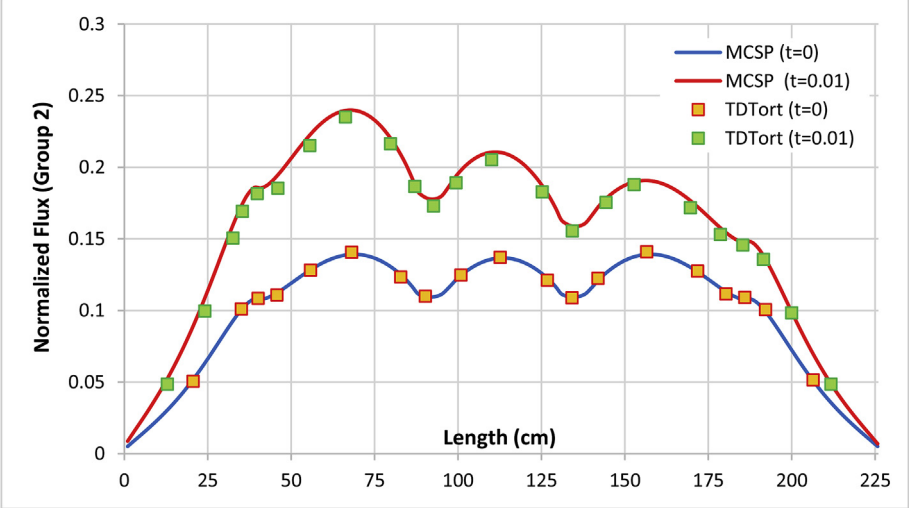

Fig. 6. Normalized flux vs. radial position for problem 16-A1 (group 2).

Table 6
The total calculation time for benchmark problem ANL 16-A1.

1 CPU (Hours)	48 CPUs (Hours)
15.15	0.5

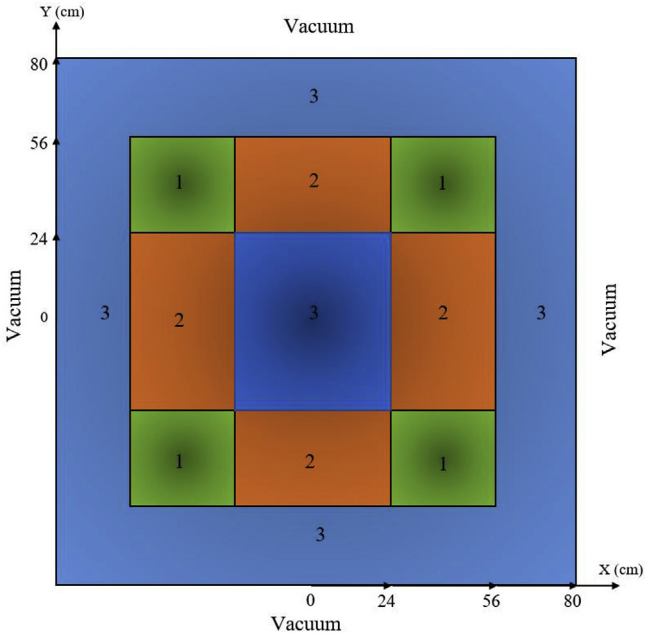

Fig. 7. Schematic of TWIGL transient problem.

a steady state system during a t_G.

The MCSP estimates the flux, the criticality eigenvalue K_{eff} and the removal life time of neutron similar to the MCNPX code (D. B. Pelowitz, 2008). The multiplication factor (K_{eff}) is estimated through averaging three different procedures called absorption, collision and track length estimators. They are obtained from the following equations, respectively.

$$K^A = \sum_{i=1}^{N} W_i \left[\frac{\nu_g \Sigma_{fg}}{\Sigma_{cg} + \Sigma_{fg}} \right] / N \tag{3}$$

$$K^C = \sum_{i=1}^{N} \sum_{c=1}^{C_i} W_{i,c} \frac{\nu_g \Sigma_{fg}}{\Sigma_{tg}} / N \tag{4}$$

$$K^{TL} = \sum_{i=1}^{N} \sum_{c=1}^{C_i} [W_{i,c} S_{i,c}\, \nu_g \Sigma_{fg} + R \nu_g \Sigma_{fg}] / N \tag{5}$$

Where Σ_{fg}, $\Sigma_{cg}$and Σ_{tg} are the macroscopic fission, capture and total cross section in neutron energy group g^{th} respectively and $g$is neutron energy group during the c^{th}collision. $H$is the total number of histories, N is the number of neutron in any history, c is the number of neutron collisions during its lifespan, $W_{i,c}$ is weight of thei^{th} neutron during the c^{th}collision, S is the distance between two collision and R is the distance between the last collision point and boundary of system for neutron leaks.

Then the K_{eff} for h^{th}cycle is estimated as follows:

$$K_{eff}^h = [K^A + K^C + K^{TL}]/3 \tag{6}$$

And the final K_{eff} of the system is estimated by averaging over all active cycles as follows:

$$K_{eff} = \sum_{h=1}^{H} K_{eff}^h / H \tag{7}$$

Furthermore, the MCSP is able to calculate the neutron mean generation time (tG) as follows (Rief and Kschwendt, 1967):

$$tG = \sum_{n=1}^{N} \left(\sum_{c=1}^{C} \frac{t_{n,c} \nu_g \Sigma_{fg}}{\Sigma_{tg}} \right) / \sum_{n=1}^{N} \left(\sum_{c=1}^{C} \frac{\nu_g \Sigma_{fg}}{\Sigma_{tg}} \right) \tag{8}$$

Where $t_{n,c} = \sum_{c=1}^{C} L_{n,c}/vg$ is the time interval of the c^{th}collision of the n^{th} source particle, vg is the speed of neutron in energy groupg^{th} and Σ_{fg} and Σ_{tg} are the macroscopic fission and total cross section in neutron energy group g^{th} respectively.

In MCSP, the total simulation time (T) is divided into time steps each lasts ΔT. The values of the multiplication factor (K_{eff}) and the mean generation time (t_G) are calculated at the beginning of each time step. In the case of time-varying cross sections or changing of boundaries, variations are exerted at beginning of each time step. This method assumes that these values are constant during ΔT and the system will not change during this time step (ΔT). On the other hand, the population of the prompt neutrons and precursors are interdependent and change after each mean generation time (t_G). Therefor each time step (ΔT) divides into sub-steps each lasts t_G and the prompt neutrons and the precursor population are calculated using the time-dependent Eq. (7) in each sub-steps. Production of fission neutrons in next generation (t_G second later) is equal to$K_{eff} \times N$ that the production of prompt neutrons is $(1-\beta) \times K_{eff} \times N$and formation of the precursor of the g'^{th} group is equal to$\beta_{g'} \times K_{eff} \times N$where $\beta = \sum_{g'=1}^{G'} \beta_{g'}$and $\beta_{g'}$ is the fraction of the delayed neutron in groupg'^{th}. The rate of the production of the delayed neutrons in every group is equal to the rate of the radioactive decay of precursors of the same group, i.e. $\lambda_{g'} C_{g'}$. Therefore, the total rate of production of delayed neutrons can be considered as a source with strength $Sd = \sum_{g'=1}^{G'} \lambda_{g'} C_{g'}$where $C_{g'}$and $\lambda_{g'}$ are the population and decay constants of delayed neutron precursor of g'^{th} group respectively. The total number density of prompt neutrons and the power are express as follows:

$$\begin{aligned} N_g^{(t')} &= \chi_g \times (1-\beta) \times K_{eff} \times N^{(t'-t_G)} + \chi_g \times \sum_{g'=1}^{G'} \lambda_{g'} C_{g'}^{(t'-t_G)} \times t_G \\ C_{g'}^{(t')} &= \lambda_{g'} C_{g'}^{(t'-t_G)} \times t_G + \beta_{g'} \times K_{eff} \times N^{(t'-t_G)} \\ P^{(t')} &= \sum_{g=1} E_f \times \Sigma_{f,g}^{(t')} \times v_g \times N_g^{(t')} \times V_F \end{aligned} \tag{9}$$

Where $N^{(t')}$ is the total number of prompt neutrons, $N_g^{(t')}$ is the number of g^{th}group the prompt neutrons, $C_{g'}^{(t')}$is the number of theg'^{th} group the

Table 7
Cross section for the TWIGL benchmark problem.

Region	Group	$\Sigma_t (cm^{-1})$	$\nu\Sigma_f\ (cm^{-1})$	$\Sigma_a\ (cm^{-1})$	$\Sigma_s^{g\to g}\ (cm^{-1})$	$\Sigma_s^{g\to g'}\ (cm^{-1})$
1	1	0.238095	0.007	0.01	0.218095	0.01
	2	0.83333	0.200	0.15	0.68333	0.00
2	1	0.238095	0.007	0.01	0.218095	0.01
	2	0.83333	0.200	0.15	0.68333	0.00
3	1	0.25641	0.003	0.008	0.23841	0.01
	2	0.66667	0.060	0.05	0.616667	0.00

$V_1 = 1.0E+7$ cm/s, $V_2 = 2.0E+5$ cm/s, $X_1 = 1.0$, $X_2 = 0.0$, $\nu = 2.43$, $\beta = 0.0075$, $\lambda = 0.08\ s^{-1}$.

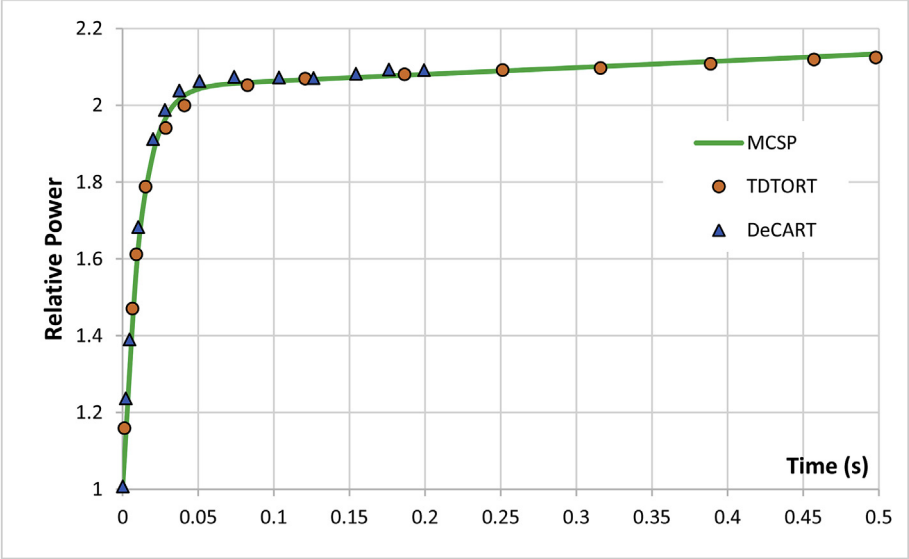

Fig. 8. The relative power vs. time for the step transient in the TWIGL transient problem.

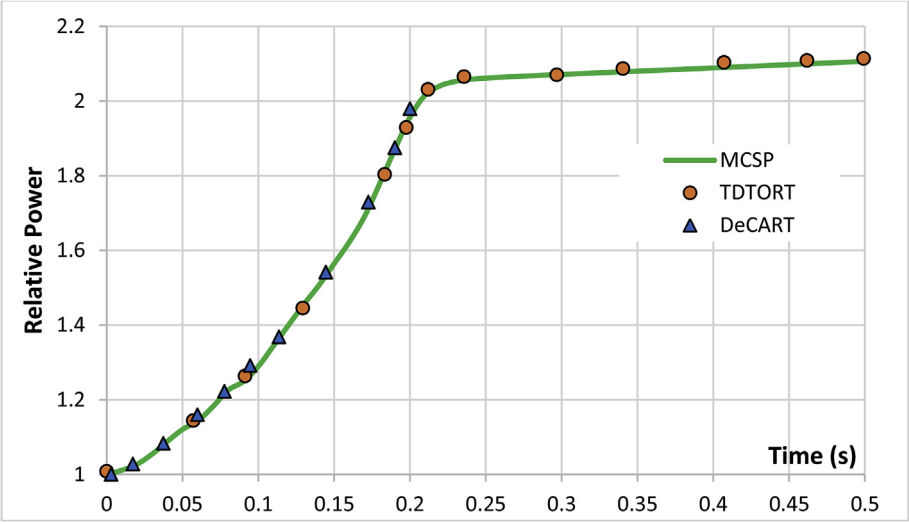

Fig. 9. The relative power vs. time for the ramp transient in the TWIGL transient problem.

Table 8
The total calculation time for the TWIGL transient problem.

Transient mode	1 CPU (Hours)	48 CPUs (Hours)
Step	10.2	0.25
Ramp	101.8	2.4

precursors, $\Sigma_{f,g}^{(t')}$ is the macroscopic fission cross section, $v_g$is the average speed of g^{th}group, E_f is the recoverable energy per fission, V_F is the volume of the fissile elements and $P^{(t')}$ is power generated from all energy groups after $\acute{t}$ second. Fig. 1 presents the flowchart of the MCSP for dynamic transient analysis and how it is divided into time steps and sub-steps. It should be noted that t denotes the coarse time steps and $\acute{t}$ represent fine time sub-steps.

The MCSP uses the implicit capture method. In this method the weight of each neutron (W) is reduced $\left(1-\frac{\sigma_c}{\sigma_t}\right)\times W$ after each collision where σ_c and σ_t are the capture and total cross section of the collision nuclide at the incoming neutron energy. If the new weight is below the problem weight cutoff ($WC_2 = 0.25$), Russian roulette is played and with probability W/WC_1 the particle survives with new weight WC_1 ($WC_1 = 0.5$); otherwise the particle is killed, resulting overall in fewer particles with larger weight. It should be noted, the weight of all sampled neutrons is considered one.

2.1. Numerical results

Each benchmark is executed using 300 cycles leaving first 100 cycles as inactive cycles. The number of source histories per cycle is 10^6. The cross section tables used for each problem are given in the following test problems.

2.2. Problem 1: a cube with time-varying cross sections

As the first example, a system with time-varying cross sections is studied. The system is a small cube of 10 × 20 × 24 cm with vacuum boundaries. It is made of a homogeneous material with one neutron energy group and six delayed neutron precursor groups listed in Table 1 and Table 2. The arrows in the table indicate the cross sections that are step-changed in this transients. Change is applied at t = 10 s and the system is returned to its former state at t = 40 s. The transient is initialized using an eigenvalue problem. We calculated the eigenvalue

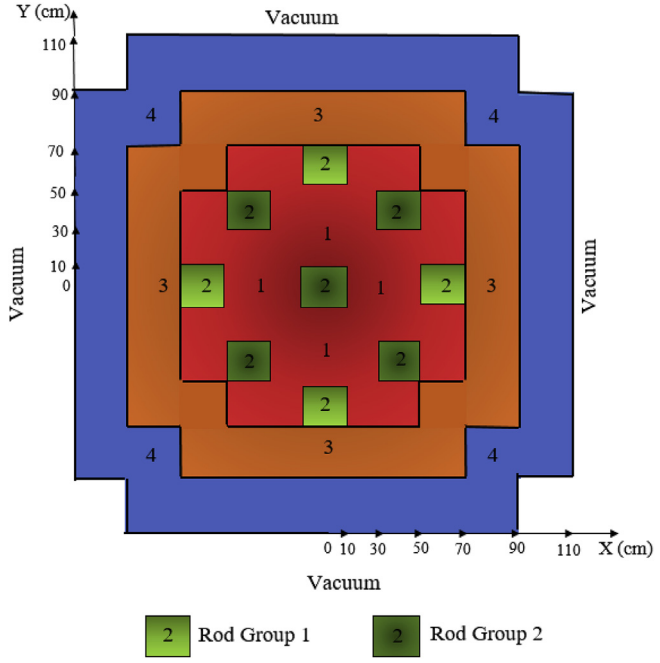

Fig. 10. Schematic of the LMW core layout on the X-Y page.

equal to 1.00000 (± 0.00002) and mean generation time equal to 7.33E-6 (± 3.7E-10) second. The relative flux versus time is shown in Fig. 2 (with time step ΔT = 0.1 s) where an excellent agreement with the results of Dynamic Tripoli which is based on MC method (Sjenitzer and Hoogenboom, 2011) is observed there. MCSP also proves its ability to simulate the long term effects dominated by the delayed neutrons, well. It is noteworthy that the maximum estimated relative standard deviation is 4.68E-03 and is observed at t = 40 s. Moreover, a comparison of the total calculation time is presented in Table 3. As can be seen from Table 3, the time spent with 48 CPUs is much less than computing time with a CPU (It is 37.67 times faster). This indicates an increase in the performance of this method and a large amount of time saving.

It can be clearly seen that there is first a positive reactivity step and then a negative reactivity step. In this case it is the absorption cross section of the entire system that changes, which could represent for example a control rod movement.

2.3. Problem 2: ANL 16- A1 benchmark problem

Problem ANL 16-A1 is a 1-D time-dependent transport problem in a fast reactor. The system consists of a seven-region slab that is surrounded by vacuum. Geometry of the problem is displayed in Fig. 3. The system is composed of homogeneous materials with two neutron energy group and six delayed neutron precursor groups as listed in Table 4 and Table 5. The system is perturbed by increasing the density of material in zone 2 and decreasing the density if material in zone 6 by 5% at time zero. The transient is initialized using an eigenvalue problem. MCSP calculated the eigenvalue equal to 1.00001 (± 0.00004) and mean generation time equal to 3.73E-7 (± 1.2E-10) second. The relative power versus time is shown in Fig. 4 (with time step ΔT = 0.000001 s until t = 0.01s and ΔT = 0.01 s until t = 0.1 s and ΔT = 0.1 s until t = 10 s) and the normalized flux for group 1 and 2 are shown in Figs. 5 and 6 respectively. MCSP shows excellent agreement with the TDTORT results which is a discrete ordinate transport code (Goluoglu and Dodds, 2001). It is noteworthy that the maximum estimated relative standard deviation is 3.69E-03 and is observed at t = 10 s. Moreover, a comparison of the total calculation time is presented in Table 6. As can be seen from Table 6, the time spent with 48 CPUs is much less than computing time with a CPU (It is 30.3 times faster). This indicates an increase in the performance of this method and a large amount of time saving.

2.4. Problem 3: 2D TWIGL benchmark problem

TWIGL is a 2-D reactor, 1.6 m along each side with vacuum boundary conditions. The problem geometry is displayed in Fig. 7. Three different materials with two neutron energy groups and one delayed neutron precursor group are considered for the problem with the data given in Table 7. Two transients are initiated either by decreasing Σ_{a2} in region 1 by 0.0035 as a step perturbation or by decreasing Σ_{a2} in region 1 by 0.0035 as a ramp perturbation for 0.2 s. Each of the TWIGL transients begin from an assumed steady state critical condition (using the change in the value of ν). This state is initialized using an eigenvalue problem. MCSP calculated the eigenvalue equal to 1.00005 (± 0.00005) and mean generation time equal to 4.13E-5 (± 7.2E-08)

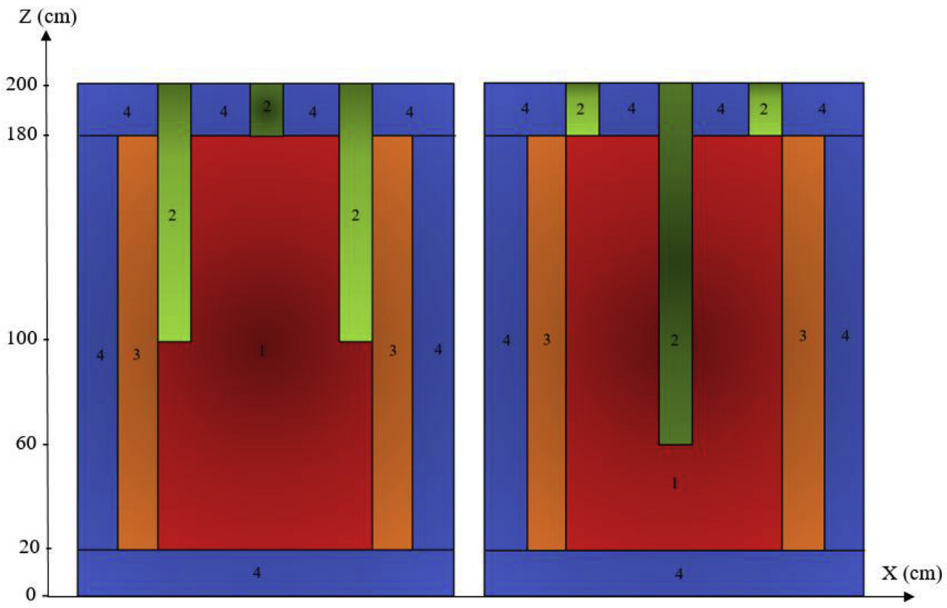

Fig. 11. Schematic of the LMW core layout for the initial state (left) and final state (right) on the X-Z page.

Table 9
Cross section for the LMW transient problem.

Region	Group	$\Sigma_t(cm^{-1})$	$\nu\Sigma_f\ (cm^{-1})$	$\Sigma_a\ (cm^{-1})$	$\Sigma_s^{g\to g}\ (cm^{-1})$	$\Sigma_s^{g\to g'}\ (cm^{-1})$
1	1	0.234097	0.00647769	0.01040206	0.206139	0.0175555
	2	0.935525	0.11273280	0.08766217	0.847863	0.0
2	1	0.234097	0.00647769	0.00109521	0.205589	0.0175555
	2	0.935525	0.11273280	0.09146217	0.844063	0.0
3	1	0.233818	0.07503284	0.01099263	0.205648	0.0171777
	2	0.950822	0.13780040	0.09925634	0.851565	0.0
4	1	0.203970	0.0	0.00266057	0.173713	0.0275969
	2	1.262620	0.0	0.04936351	0.121325	0.0

V_1 = 1.25E+7 cm/s, V_2 = 2.5E+5 cm/s, X_1 = 1.0, X_2 = 0.0, ν = 2.5.

Table 10
Delayed neutron parameters for the LMW transient problem.

Family	$\lambda(s^{-1})$	β
1	0.0127	0.0002470
2	0.0317	0.0013845
3	0.1150	0.0012222
4	0.3110	0.0026455
5	1.4000	0.0008320
6	3.8700	0.0001690

second. As conceived from Fig. 8 and Fig. 9, the relative power for step and ramp perturbation (with time step $\Delta T = 0.01$ s) calculated by MCSP agrees very well with the TDTORT which a discrete ordinate transport code (Goluoglu and Dodds, 2001) and DeCART which is based on MOC method (Adam J. Hoffman, 2013). It is noteworthy that the maximum estimated relative standard deviation is observed at t = 0.5 s equal to 1.14E-03 for step and 1.22E-03 for ramp perturbation. Moreover, a comparison of the total calculation time is presented in Table 8. As can be seen from Table 8, the time spent with 48 CPUs is much less than computing time with a CPU (It is almost 42 times faster). This indicates an increase in the performance of this method and a large amount of time saving.

As seen in Fig. 8, following an initial prompt jump, the power rises relatively slowly and then reaches an asymptotic behavior at the end of the transient.

In contrast, no prompt jump is observed in the ramp transient. The power surges exponentially at first, and then, a relatively slow increment is pursued to becoming almost constant at the end of the transient.

Table 11
The total calculation time for the LMW transient problem.

1 CPU (Hours)	48 CPUs (Hours)
216.9	4.8

Table 12
Delayed neutron parameters for C5G7 transient problem.

Family	$\lambda(s^{-1})$	β
1	0.0127	0.0002470
2	0.0317	0.0013845
3	0.1150	0.0012220
4	0.3110	0.0026455
5	1.4000	0.0008320
6	3.8700	0.0001690

Table 13
Energy group structure for C5G7 transient problem.

Group	E_{top} (eV)	E_{bottom} (eV)	$V_{midpoint}$ (cm/s)
1	2.0E+7	1.0E+6	4.48E+9
2	1.0E+6	5.0E+5	1.20E+9
3	5.0E+5	3.0E+0	6.92E+8
4	3.0E+0	6.25–1	1.86E+6
5	6.25E-1	1.0E-1	8.33E+5
6	1.0E-1	2.0E-2	3.39E+5
7	2.0E-2	1.0E-5	1.85E+5

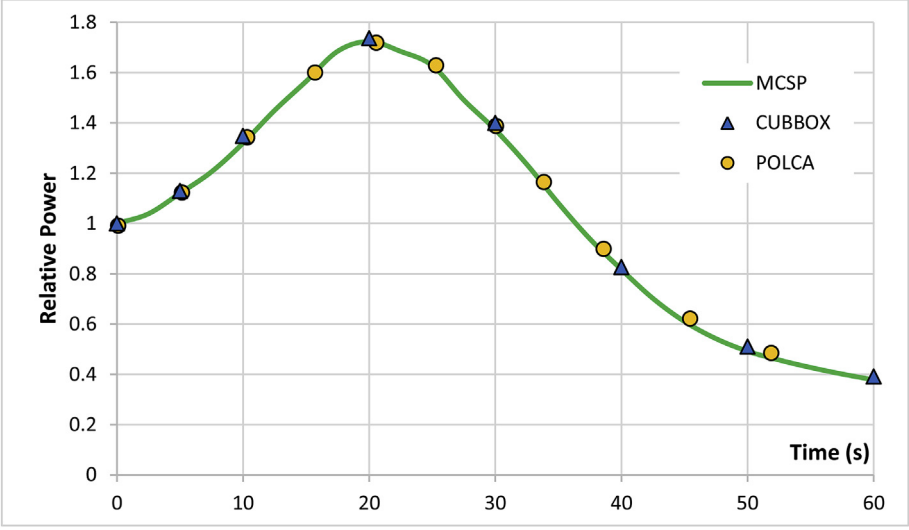

Fig. 12. The relative power vs. time for the LMW transient without Feedback.

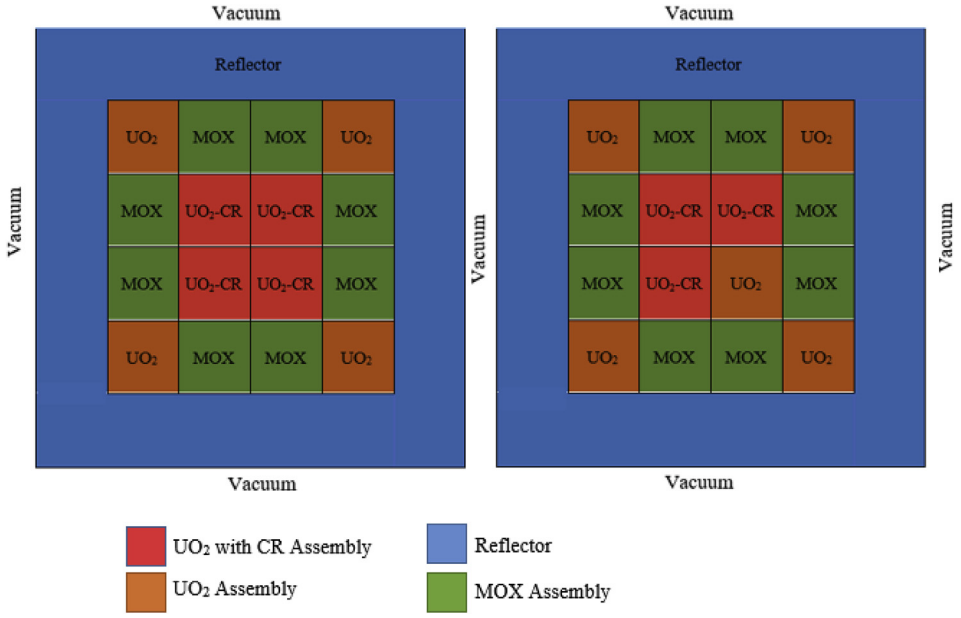

Fig. 13. Schematic of C5G7 core layout for the initial state (left) and final state (right).

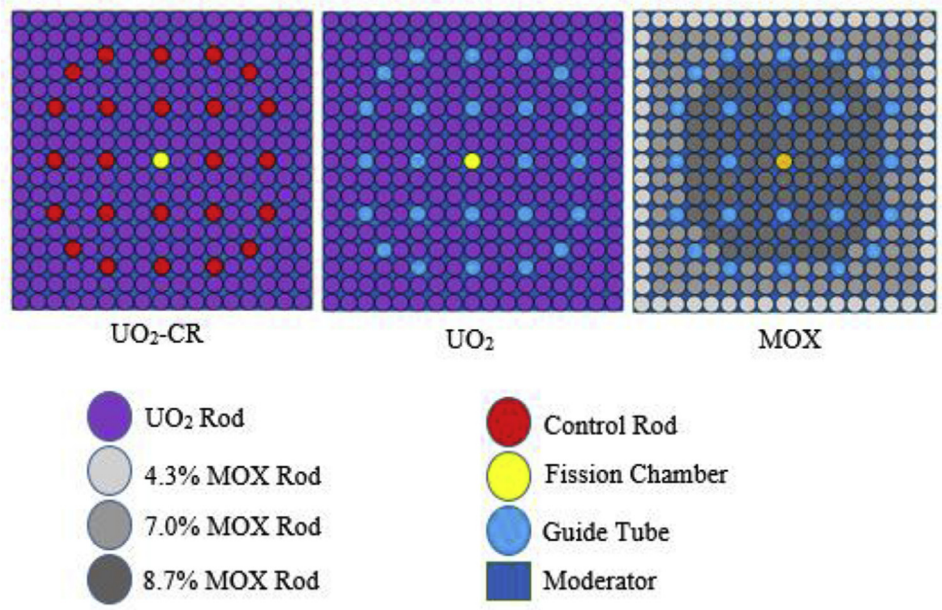

Fig. 14. Schematic of C5G7 fuel assembly layout.

2.5. Problem 4: LMW LWR benchmark problem

The LMW (Langenbuch-Maurer-Werner) represent a simplified LWR transient problem with the geometry sketched in Fig. 10 and Fig. 11. It is made of homogeneous materials with two neutron energy group and six delayed neutron precursor groups as listed in Tables 9 and 10. The transient involves the withdrawal of a bank of four partially-inserted control rod (Rod Group #1 is pulled out at the speed of 3.0 cm/s, $0 \leq t \leq 26.666$ s) and the insertion of a bank of five control rods subsequently (Rod Group #2 is inserted at the speed of 3.0 cm/s, $7.5 \leq t \leq 47.5$ s). The LMW transients begins from an assumed steady-state critical condition (using the change in the value of ν). This state is initialized using an eigenvalue problem. MCSP calculated the eigenvalue equal to 1.00029 (± 0.00005) and mean generation time equal to 3.66E-5 (± 4.5E-08) second. The relative power versus time is shown in Fig. 12 (with time step $\Delta T = 0.1$ s). As observed in this figure, the MCSP poses excellent agreement with the CUBBOX (J. C. Gehin, 1992) and POLCA-T (J. Kotchoubey, 2015) which are diffusion codes. It is noteworthy that the maximum estimated relative standard deviation is 8.09E-03 and is observed at t = 20 s. Moreover, a comparison of the total calculation time is presented in Table 11. As can be seen from Table 11, the time spent with 48 CPUs is much less than computing time with a CPU (It is 45.19 times faster). This indicates an increase in the performance of this method and a large amount of time saving.

As Fig. 12 shows, at first the power increases slowly following the positive reactivity injected by withdrawal of Rods Group #1. The trend is halted back by negative reactivity injected by insertion of Rods Group #2. This process continues until negative reactivity overcome the positive reactivity and after that, the power begins to decrease.

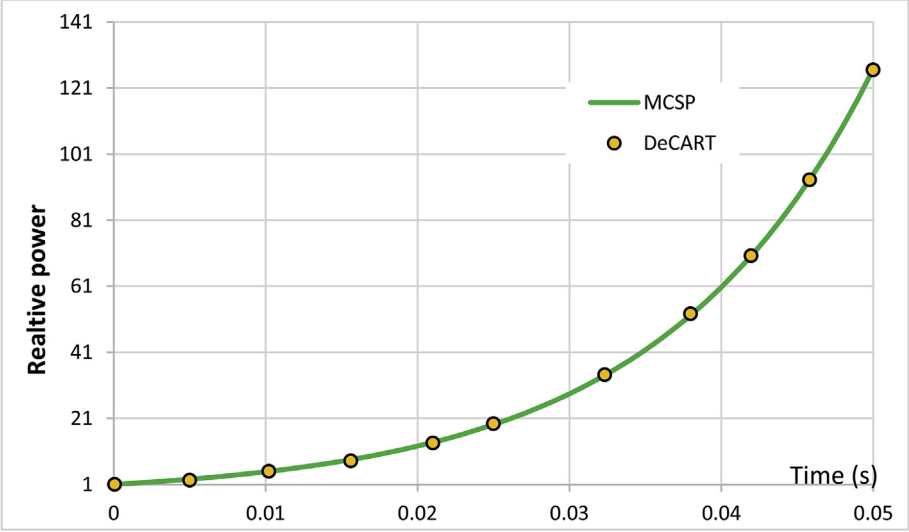

Fig. 15. The relative power vs. time for the step transient in the C5G7.

Table 14
The total calculation time for the C5G7 transient problem.

1 CPU (Hours)	48 CPUs (Hours)
43.95	0.96

Table 15
The total calculation time for the Continuous energy transient problem.

1 CPU (Hours)	48 CPUs (Hours)
21.8	0.6

2.6. Problem 5: the C5G7 benchmark problem

Treating complex geometries without homogenization is one of the main advantages of the Monte Carlo approach. To emphasize on this, a more realistic problem is studied. The C5G7 is a small reactor core with sixteen fuel assemblies and it uses a seven-group cross section library. The data for the problem are given in reference (Adam J. Hoffman, 2013). Parameters used for solving the C5G7 transient problem listed in Table 12 and Table 13. The assemblies are 21.42 cm square and it is surrounded by a water reflector of the same thickness. Each of the assemblies is composed of 17 × 17 fuel pins or guide tubes which are surrounded by moderator. Each pin cell is 1.26 cm square and the fuel pins have a radius of 0.54 cm and are centered in the pin cell (Adam J. Hoffman, 2013). Schematic of C5G7 reactor core layout and its pin cell geometry is displayed in Fig. 13 and Fig. 14.

The transient will be initialized by ejecting control rods in the southeast UO2-CR assembly. Because the C5G7 is a very small reactor, ejecting control rods causes a fast exponential increase in power. The C5G7 transient begins from an assumed steady-state critical condition, (using the change in the value of ν). This state is initialized using an eigenvalue problem. MCSP calculated the eigenvalue equal to 1.14758 (± 0.00005) and mean generation time equal to 3.36E-5 (± 2.82E-08) second. The relative power versus time is shown in Fig. 15 (with time step $\Delta T = 0.001$ s). MCSP shows good agreement with the DeCART results which is based on MOC method (Adam J. Hoffman, 2013). It is noteworthy that the maximum estimated relative standard deviation is 2.01E-03 and is observed at t = 0.05 s. Moreover, a comparison of the total calculation time is presented in Table 14. As can be seen from Table 14, the time spent with 48 CPUs is much less than computing time with a CPU (It is 45.78 times faster). This indicates an increase in

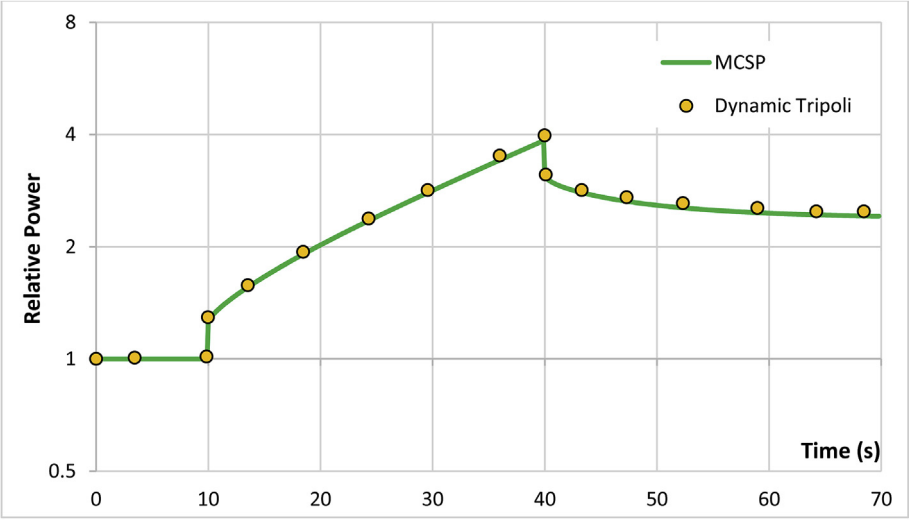

Fig. 16. The relative power vs. time in the problem 6 with Continuous cross section.

the performance of this method and a large amount of time saving.

2.7. Problem 6: a transient system with continuous energy structure

In order to assess the performance of MCSP in the analysis of a transient system with continuous energy, the cuboid of the first benchmark is made of pure ^{235}U. The density is adjusted to make a critical system and the transient is generated by an increase in density. The density increase from 4.4925E-02 atoms/b.cm to 4.5000E-02 atoms/b.cm at t = 10.0 s and the system is returned to its initial state at t = 40.0 s. The transient is initialized using an eigenvalue problem. MCSP calculated the eigenvalue equal to 0.99999 (± 0.00005) and mean generation time equal to 6.00E-9 (± 2.5E-11) second. The relative power versus time is shown in Fig. 16 (with time step ΔT = 0.1 s) and it shows good agreement with Dynamic Tripoli results which is based on MC method (Sjenitzer and Hoogenboom, 2012). It is noteworthy that the maximum estimated relative standard deviation is 5.78E-02 and is observed at t = 40 s. Moreover, a comparison of the total calculation time is also presented in Table 15. As can be seen from Table 15, the time spent with 48 CPUs is much less than computing time with a CPU (It is 36.33 times faster). This indicates an increase in the performance of this method and a large amount of time saving.

3. Conclusions

A new computer code is developed for neutronic analysis of 3-D transient systems with explicit representation of delayed neutrons based on a Monte Carlo method. The code, named MCSP (Monte Carlo dynamic Simulation of Particles tracking), is capable to simulate the systems with time-varying geometry and cross sections with either continuous or multi-group cross section in a 3-D desired geometry without the restrictions of previous methods (numerous simplifications like spatial homogenization, multi-group energy collapsing, etc.). To speed up the calculation The MCSP is programming in parallel processing as well. The MCSP is evaluated with standard benchmark problems such as C5G7, LMW, TWIGL, etc. with satisfactory agreements, and it can be employed to analyze dynamics of complex systems. It should be added that if the MCSP code couples with a thermal-hydraulics code, it capable to investigate of realistic accident scenarios.

Appendix A. Supplementary data

Supplementary data to this article can be found online at https://doi.org/10.1016/j.pnucene.2019.03.024.

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