Calculation of time-dependent neutronic parameters using Monte Carlo method

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1. Introduction

The neutron population in a time-varying system can be modeled by the time-dependent transport equation. The time-dependent neutron transport equation is an integro-differential equation for the neutron angular flux, and may be written as (Bell and Glasstone, 1970):

\[
\frac{1}{\Omega} \frac{\partial \Phi}{\partial t} + \frac{\nabla \cdot \Phi}{\Omega} + \sum_{i} \left[ \Phi(\tilde{r}, \tilde{\Omega}', E, t) s(\tilde{r}, \tilde{\Omega}', \tilde{\Omega}, E, t) - \Phi(\tilde{r}, \tilde{\Omega}, E, t) s(\tilde{r}, \tilde{\Omega}, \tilde{\Omega}', E, t) \right] dv dE
\]

\[
= \int \left[ \sum_{\gamma} \frac{\partial}{\partial t} \Phi(\tilde{r}, \tilde{\Omega}', E, t) + \sum_{\gamma} \int \frac{\partial}{\partial t} \Phi(\tilde{r}, \tilde{\Omega}', E, t) dv dE \right] dv dE
\]

\[
+ \int \left[ \sum_{\gamma} \frac{\partial}{\partial t} \Phi(\tilde{r}, \tilde{\Omega}', E, t) + \sum_{\gamma} \int \frac{\partial}{\partial t} \Phi(\tilde{r}, \tilde{\Omega}', E, t) dv dE \right] dv dE
\]

\[
+ \sum_{\gamma} \int \frac{\partial}{\partial t} \Phi(\tilde{r}, \tilde{\Omega}', E, t) dv dE
\]

\[
+ \sum_{\gamma} \int \frac{\partial}{\partial t} \Phi(\tilde{r}, \tilde{\Omega}', E, t) dv dE
\]

\[
+ \sum_{\gamma} \int \frac{\partial}{\partial t} \Phi(\tilde{r}, \tilde{\Omega}', E, t) dv dE
\]

The corresponding time-dependent precursor equation is:

\[
\frac{\partial}{\partial t} C(\tilde{r}, t) + \lambda C(\tilde{r}, t) = \int \left[ \frac{\partial}{\partial t} \Phi(\tilde{r}, \tilde{\Omega}', E, t) dv dE \right] dv dE
\]

\[
+ \sum_{\gamma} \int \frac{\partial}{\partial t} \Phi(\tilde{r}, \tilde{\Omega}', E, t) dv dE
\]

where \(\Phi(\tilde{r}, \tilde{\Omega}, E, t)\) is the angular flux, \(f(\tilde{r}, \tilde{\Omega}', \tilde{\Omega}, E, t) d\Omega dE\) is the probability that a neutron of direction \(\tilde{\Omega}\) and energy \(E\) has a collision, there will emerge a neutron in a direction interval \(\Omega\) about \(\tilde{\Omega}\) with energy in \(dE\) about \(E\), and \(Q(\tilde{r}, \tilde{\Omega}, E, t)\) is the neutron source.

The complexity of transport equation, even in the absence of delayed neutrons, usually forces one to implement approximate solution methods. In several published works, one-dimensional time-dependent transport equation has been solved in infinite and semi-infinite slabs (Ganapol, 1978; 1980; Fillipone and Ganapol, 1983; El-Wakil et al., 2003, 2004). Several methods were proposed to solve the time-dependent transport equation in finite slabs such as the diamond difference implicit trapezoidal (Barros et al., 1998), the analytical constant nodal (Barros, 1992), the multiple-collision (Windholfer and Pucker, 1985), Pomraning–Eddington approximation (El-Wakil et al., 2005), and the discrete ordinates and semi-analytical numerical (De Oliveira et al., 2002a,b) methods. In some other investigations, the perturbation theory was applied.

Usually, time-dependent problems in reactor physics are treated in systems with fixed boundaries. This is motivated by the fact that real systems have fixed boundaries. Furthermore, treatment of boundary conditions at time-dependent boundaries is complicated and the corresponding problems are difficult to solve. There exists, however, a few distinct classes of practical cases where treatment of a moving/fluctuating boundary is necessary, for instance a tank of liquid fissile material with a free surface at enrichment or reprocessing plants, or the neutron noise induced by the vibration of control rods. Therefore, the treatment of a moving boundary or interface is needed for studying the neutron noise induced by a vibrating absorber rod. Generally, in these problems, small perturbations are considered (Arzhanov, 2002). In some problems, the cross sections may vary with time. In these cases, the time-dependent cross sections may be assumed to be represented by finite power series (Abdou, 2005).

The time-dependent problems are generally very complex and have to be solved numerically. Many computer codes for these types of problems exist in the engineering and science, for instance ANISN, DOT, and TORT for one- two- and three-dimensional geometries. The TDTORT code (by using TORT code), analyzes the time-dependent behavior of neutronic systems using the transport theory with the explicit representation of delayed neutron (Golugu and Dodds, 2001). Neutron transport through matter is essentially a stochastic process. The total cross section is the probability, but not certainly, that a neutron will have a collision while traversing a certain spatial interval. If the neutron does have a collision, the cross sections
for the various processes are the probabilities, but are not certainty, that the collision will be a scattering, radiative capture, fission, and so on, event. Therefore, the neutronic parameters which are obtained from transport equation are actually the mean values.

The Monte Carlo method directly simulates neutron transport as a stochastic process. This method is a numerical procedure based on statistical (or probability) theory. In neutron transport calculations, the applicability of the Monte Carlo techniques arises from the fact that the macroscopic cross section may be interpreted as a probability of interaction per unit distance by a neutron. The locations of actual collisions and results of such collisions are determined from the range of possibilities by set of random numbers. Monte Carlo does not solve an explicit equation, but rather obtains answers by simulating individual particles and recording some aspects of their average behavior. It consists of actually following each of many particles from its source throughout its life to its death, called the history of the particle. Probability distributions are randomly sampled using transport data to determine the outcome at each step of its life. The average behavior of particles in the physical system is then inferred from the average behavior of the simulated particles by using the central limit theorem. The Monte Carlo technique has proved useful in special cases where other methods encounter difficulties. Moreover, when there is considerable detail in the variation of the neutron cross section with energy and time, or where treatment of a moving/fluuctuating boundary is necessary, the Monte Carlo method eliminates the need for making subsidiary calculations. Monte Carlo is well suited to solving complicated three-dimensional, time-dependent problems.

In Monte Carlo methods, there are uncertainties which are not due to the explicit approximations, but to the limitation in the (finite) number of neutrons examined. Such errors are more or less random, and procedures have been developed for reducing the uncertainty associated with a given amount of numerical work.

In this work, a time-dependent neutron transport code (TDMC) has been developed based on Monte Carlo method. In the first step, the results of TDMC have been validated in some static benchmark problems. Then, some time-dependent problems have been chosen and the results of TDMC have been benchmarked on TDTORT time-dependent transport code.

2. Method

The TDMC (time-dependent Monte Carlo) code is developed for neutronic calculations of multiplying systems. This code calculates neutronic parameters such as the effective multiplication factor, neutron life time, and space and time-dependent flux based on Monte Carlo method. The sphere, cylinder, and slab (finite and infinite) geometries are available in TDMC. Both continuous energy-dependent nuclear data and multi-group cross section libraries are available. In the case of moving boundaries and interfaces between two-regions of the system, or in time-varying cross sections, these variations are loaded in the code, and in the start of each cycle, the quantities are initiated based on new boundaries or cross section values.

The definition of particle flux is \( \Phi(\vec{r}, E, t) = v N(\vec{r}, E, t) \). Because \( N(\vec{r}, E, t) \) is the track length density, TDMC estimates the flux by summing \( W T/V \) for all particle tracks where \( W \) is the particle statistical weight, \( T \) is the track length, and \( V \) is the volume of the region. The surface flux is a surface estimator but can be thought of as the limiting case of the track length estimator when the volume becomes infinitely thin. TDMC can calculate \( \int \Phi(E) R(E) dE \) where \( R(E) \) is any energy-dependent quantity such as \( \Sigma_{t}, \Sigma_{c}, \) and \( 1/\rho(E) \). This is useful for calculating quantities such as the criticality eigenvalue \( k_{\text{eff}} \) (or reactivity) and various prompt neutron lifetimes.

Calculating \( k_{\text{eff}} \) consists of estimating the number of fission neutrons produced in one generation or cycle (the life of a neutron from birth in the fission to death by escape or absorption) per fission neutrons started. The number of histories is \( H \), and in each cycle, \( M \) (varying with cycle) source particles are started isotropically and the calculated \( k_{\text{eff}} \) in the previous cycle is used as the initial value (in the first cycle an initial proper guess of \( k_{\text{eff}} \) is selected by the user). Therefore, the weight of each source particle is \( H/M \) and the total source weight is \( H \). For the first cycle, the position of these \( M \) points is selected by the user and for the subsequent cycles, these points are sampled from fission sites of previous cycle. Source particles are transported by the standard random walk process except that fission is treated as capture. The fission neutrons that would have been created are accrued in three different ways to calculate \( k_{\text{eff}} \) for any cycle. These three procedures are absorption, collision, and track length estimators. The absorption estimator is made when a neutron is absorbed by a fissionable nuclide:

\[
    k_{\text{eff}} = \frac{W}{H} \sum_{b=1}^{M} \left[ \nu \Sigma_{t} / (\Sigma_{c} + \Sigma_{t}) \right]
\]

The collision estimate for \( k_{\text{eff}} \) is:

\[
    k_{\text{eff}} = \frac{W}{H} \sum_{b=1}^{M} \sum_{c=1}^{C} \left( \nu \Sigma_{c} / \Sigma_{t} \right)
\]

The track length estimate for \( k_{\text{eff}} \) is accumulated every time the neutron traverses a distance \( L \) in a fissionable material cell:

\[
    k_{\text{eff}} = \frac{W}{H} \sum_{b=1}^{M} \sum_{c=1}^{C} \left( I_{b} \nu \Sigma_{c} + R_{b} \nu \Sigma_{t} \right)
\]

where \( \nu \) is the average number of neutrons produced by fission at the incident energy of this collision, \( \Sigma_{t}, \Sigma_{c}, \) and \( \Sigma_{t} \) are the macroscopic fission, capture, and total cross sections, respectively (these quantities are energy-dependent and they have different values in various collisions), \( H \) is the total number of histories, \( C \) is the number of collision a neutron makes during its histories, \( W \) is the neutron statistical weight (varying with cycle) and \( R_{b} \) is the distance between the last collision point and the outer boundary of the system for the neutron leaks out the system. For each cycle, the mean value of \( k_{\text{eff}} \) is calculated by averaging the three results.

The life times are calculated at the collision points. If \( t_{0}(i) \) (which is equal to \( \sum t_{0}(i) \nu_{0} / \nu_{c} \), where \( \nu_{c} \) is the velocity of neutron in energy group \( g \)), is the time distance at the \( i \) th collision of the \( t \) th neutron history from its origin, the average production time \( \ell_{p} \) (the average time necessary to build up the next neutron generation) can be calculated according to (Rief and Kschwendt, 1967):

\[
    \ell_{p} = \left[ \frac{\sum_{b=1}^{M} \nu \Sigma_{c} / \Sigma_{t}}{\sum_{b=1}^{M} \nu \Sigma_{t} / \Sigma_{t}} \right] \left[ \frac{\sum_{b=1}^{M} \nu \Sigma_{t} / \Sigma_{t}}{\sum_{b=1}^{M} \nu \Sigma_{c} / \Sigma_{t}} \right]
\]

In the same way, the average removal time \( \ell_{r} \) (the average time from the emission of a prompt neutron in fission to the removal of the neutron by escape or absorption processes, or the average time a neutron stays in the system) is given by

<table>
<thead>
<tr>
<th>Thickness (mfp)</th>
<th>TDMC</th>
<th>GMF</th>
<th>DANT</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0.34937 ± 0.00039</td>
<td>0.349466</td>
<td>0.349465</td>
</tr>
<tr>
<td>2</td>
<td>0.66308 ± 0.00058</td>
<td>0.662951</td>
<td>0.662951</td>
</tr>
<tr>
<td>4</td>
<td>1.19898 ± 0.00085</td>
<td>1.19923</td>
<td>1.19923</td>
</tr>
<tr>
<td>5</td>
<td>2.00152 ± 0.00115</td>
<td>2.00193</td>
<td>2.00193</td>
</tr>
<tr>
<td>10</td>
<td>2.22957 ± 0.00121</td>
<td>2.22998</td>
<td>2.22998</td>
</tr>
<tr>
<td>15</td>
<td>2.33555 ± 0.00122</td>
<td>2.3349</td>
<td>2.3349</td>
</tr>
<tr>
<td>20</td>
<td>2.38779 ± 0.00121</td>
<td>2.38783</td>
<td>2.38783</td>
</tr>
<tr>
<td>25</td>
<td>2.38779 ± 0.00121</td>
<td>2.38783</td>
<td>2.38783</td>
</tr>
</tbody>
</table>
The behavior of neutron is considered spontaneous in the system. It means that a neutron is absorbed or leaked out in a short period, while the conditions of region do not change. Therefore they are treated by steady state calculations. On the other hand, the densities of delayed neutron precursors change slowly, and the conditions of a region change after a desired time step. In TDMC the neutrons are treated by using steady state equation at each time point, and the delayed neutron precursors are calculated by using time-dependent equation. For calculating time-dependent power (or fission rate) in a delayed critical system, delayed neutrons can be considered as a source with strength

\[ S_d = \sum_{i=1}^{n} k_i C_i \]

and

\[ \beta = \frac{1}{\sum_{i=1}^{n} \lambda_i} \]

where \( C_i \) and \( \lambda_i \) are the population and decay constants of delayed neutron precursor fission fragments of \( i \)th group. Also, \( n \) is the number of delayed neutron groups and \( \beta \) is the fraction

\[ t_{\text{removal}} = W \frac{\sum_{i=1}^{n} (\beta_i + \lambda_i) + t_{\text{leak}}}{H} \]

where \( t_{\text{leak}} \) (equal to \( \sum_{i=1}^{n} L_{h_i}/v_{h_i} + R_{\text{leak}}/v_{\text{leak}} \)) is the time at which the neutron leaks out of the system.

### Table 2

<table>
<thead>
<tr>
<th>Region 1</th>
<th>Region 2</th>
<th>( k_{\text{eff}} )</th>
</tr>
</thead>
<tbody>
<tr>
<td>( r_{\text{sum}} )</td>
<td>( \sum )</td>
<td>( \sum )</td>
</tr>
<tr>
<td>1.5</td>
<td>0</td>
<td>1</td>
</tr>
<tr>
<td>1.4</td>
<td>0.1</td>
<td>1</td>
</tr>
<tr>
<td>1.3</td>
<td>0.2</td>
<td>1</td>
</tr>
<tr>
<td>1.2</td>
<td>0.3</td>
<td>1</td>
</tr>
<tr>
<td>1.1</td>
<td>0.4</td>
<td>1</td>
</tr>
<tr>
<td>1.0</td>
<td>0.5</td>
<td>1</td>
</tr>
<tr>
<td>0.9</td>
<td>0.6</td>
<td>1</td>
</tr>
<tr>
<td>0.8</td>
<td>0.7</td>
<td>1</td>
</tr>
<tr>
<td>0.7</td>
<td>0.8</td>
<td>1</td>
</tr>
<tr>
<td>0.6</td>
<td>0.9</td>
<td>1</td>
</tr>
</tbody>
</table>

### Table 3

The radius of critical sphere versus the number of neutrons per collision.

<table>
<thead>
<tr>
<th>( \varepsilon )</th>
<th>1.10</th>
<th>1.20</th>
<th>1.40</th>
<th>2.0</th>
</tr>
</thead>
<tbody>
<tr>
<td>TDNC</td>
<td>4.872</td>
<td>3.172</td>
<td>1.986</td>
<td>0.991</td>
</tr>
<tr>
<td>Carlson and Bell</td>
<td>4.87272</td>
<td>3.17227</td>
<td>1.98546</td>
<td>0.99065</td>
</tr>
<tr>
<td>(GFM, TDMC) Atalay</td>
<td>4.87300</td>
<td>3.17207</td>
<td>1.98534</td>
<td>0.99061</td>
</tr>
</tbody>
</table>

### Table 4

The values of \( k_{\text{eff}} \) and mean neutron life time for two-region cylindrical system.

<table>
<thead>
<tr>
<th>( k_{\text{eff}} )</th>
<th>( \varepsilon )</th>
<th>( t_{\text{removal}} )</th>
</tr>
</thead>
<tbody>
<tr>
<td>TDNC</td>
<td>1.02809 ± 0.00378</td>
<td>6.7541E−8 ± 3.7838E−10</td>
</tr>
<tr>
<td>MCNP</td>
<td>1.03489 ± 0.00056</td>
<td>6.7147E−8 ± 7.2578E−10</td>
</tr>
</tbody>
</table>

### Table 5

Thickess of the different regions for problem 5.

<table>
<thead>
<tr>
<th>Region 1</th>
<th>Region 2</th>
<th>Region 3</th>
<th>Region 4</th>
<th>Region 5</th>
<th>Region 6</th>
<th>Region 7</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reflector</td>
<td>Fuel</td>
<td>Absorber</td>
<td>Fuel</td>
<td>Absorber</td>
<td>Fuel</td>
<td>Reflector</td>
</tr>
<tr>
<td>40.00 cm</td>
<td>47.37 cm</td>
<td>9.00 cm</td>
<td>34.00 cm</td>
<td>9.00 cm</td>
<td>47.37 cm</td>
<td>40.00 cm</td>
</tr>
</tbody>
</table>

\[ \text{Axial Distance (cm)} \]

![Fig. 1. Axial neutron flux in a two-region cylinder.](image-url)
belonging to the \(i\)th delayed group. Thus, the number of neutrons and precursors after \(m\)th cycle are

\[
N^{(m+1)} = P^{(m+1)} + D^{(m+1)}
\]

\[
P^{(m)} k^{(m)} + \sum_i \left[ D_i^{(m)} k^{(m)} + C_i^{(m)} (1 - \exp(-\lambda_i^{(m)})) \right]
\]

\[
C_i^{(m)} = C_i^{(m)} \exp(-\lambda_i^{(m)}) + N^{(m)} \beta_i
\]

where \(P^{(m)}\) is the number of prompt neutrons, \(D_i^{(m)}\) is the number of delayed neutrons, \(N^{(m)}\) is the total number of neutrons, and \(C_i^{(m)}\) is the number of \(i\)th group precursors after \(m\)th cycle.

After each cycle, the mean value quantities, \(\bar{x}\), and their standard deviations, \(\Delta x\), are calculated according to (Shayesteh et al., 2007):

\[
\bar{x} = \frac{1}{H} \sum_{h=1}^{H} x_h, \quad \Delta x = \sqrt{\frac{x^2 - \bar{x}^2}{H}}
\]

The Central Limit Theorem states that as the number of histories approaches infinity there is a 68% chance that the true result will be in the range \(\bar{x} \pm \Delta x\) and a 95% chance in the range \(\bar{x} \pm 2\Delta x\). After inactive cycles (the first few cycles where the spatial
source changes from the initial definition to the quasi static distribution, all quantities along with their standard deviations are averaged over all active cycles (Shayesteh et al., 2007).

TDMC allows several variation types that result in reactivity insertions. These variation types may be used to model transients: (I) step or ramp reactivity insertion, (II) time-varying boundaries,

Fig. 3. Normalized distribution of flux versus position for problem 5 (group 1).

Fig. 4. Normalized distribution of flux versus position for problem 5 (group 2).
and (III) time-varying cross sections (one type of the transients or a combination of them).

3. Results and discussion

We have obtained solutions to benchmark criticality problems for the slab, spherical, and cylindrical geometries using TDMC and other methods.

3.1. Slab criticality results

The first test problem is a homogeneous, one-group, one-region slab with outer surface vacuum boundary conditions. The goal is to determine the effective multiplication factor of the slab. In Table 1, we show the effective multiplication factor of the slab with $\Sigma_t = 0.25$, $\Sigma_f = 0.9$, and $\Sigma_r = 1.0$ as the thicknesses of the slab (in terms of mean free path) using TDMC, GFM, and DANT methods. The GFM (by using Green’s function and angular quadratures) calculates integrals to an error of $10^{-5}$. The DANT computations use an angular quadrature order of $S_{96}$, and a convergence criterion of $10^{-8}$ (Kornreich and Parsons, 2004).

The next test considers a two-region slab with isotropic scattering which is solved by the GFM (Kornreich and Parsons, 2004). Nominally, the first region is considered the fuel region and the second region is a reflector. Two sets of thicknesses are chosen. The thicknesses of the first region are 1.0 mfp and 1.5 mfp, respec-

![Graph showing change of reactivity with time for the system with varying boundaries.](image1)

**Fig. 5.** Change of reactivity with time for the system with varying boundaries.

![Graph showing power variation versus time following movement of the interfaces.](image2)

**Fig. 6.** Power variation versus time following movement of the interfaces.
tively, for the two considered cases. In both cases, the thickness of the second region is 1.0 mfp. The first case is sub-critical and the second case is supercritical. The values of $k_{\text{eff}}$ for the two-region slabs using TDMC and GFM methods are shown in Table 2.

3.2. Spherical geometry criticality results

The third test problem is a simple, homogeneous, one-group, spherical geometry with isotropic scattering. The critical dimensions are given upon various values of $c$ eigenvalues (the number of secondary neutrons per collision). Table 3 shows the critical radii versus mean free path using TDMC, Carlson and Bell and Atalay methods (Atalay, 2004). The Atalay method requires the Fourier transformation of angular flux for the position variable, and then an eigenvalue problem is solved in the transform-variable space.

3.3. Cylindrical geometry criticality results

The next test problem is a multiplying system with cylindrical geometry. This system is reflected on the ends and bare on the side (partially reflected). The radius of the core is 40 cm, the core height is 60 cm and the reflector thickness on the ends is 10 cm. The core consists of uranium enriched to $10\% \ ^{235}\text{U}$ and the reflector consists of natural uranium. The six group cross section (Hansen and Roach,
1961) library is prepared for MCNP4C in multi-group cross section format. The multiplication factor and removal lifetime of the system are calculated by using TDMC and MCNP4C, and the results are shown in Table 4 (with 50,000 histories in both cases). The results of the scalar flux along the cylinder axes are shown in Fig. 1. The results are normalized per source particle.

3.4. Time-dependent benchmark problem

The benchmark problem 5 is a time-dependent problem with two neutron energy groups and six delayed neutron precursor groups. This problem simulated neutron transport in a fast reactor. The system comprises a seven-region slab of three combinations of fuel, reflector, and absorber (control rod), that is surrounded by vacuum. The thicknesses of the regions are shown in Table 5. Initial two-group constants and delayed neutron parameters are given in Tables 6 and 7, respectively. The system is critical based on TDTORT calculations, and the value of the neutron generation time is $0.3655 \times 10^{-6}$ s (Goluoglu and Dodds, 2001). The results of TDMC for $k_{ef}$ and neutron generation time are $1.0000 \pm 0.0001$ and $(0.3791 \pm 0.0007) \times 10^{-6}$ s, respectively.

![Fig. 9. Flux variation versus time following decreasing fission cross section in Region 4.](image)

![Fig. 10. Normalized distribution of flux following decreasing fission cross section in Region 4.](image)
3.4.1. Step perturbation

Now, we assume that the system is perturbed by increasing the density of the material in Region 2 by 5% and decreasing the density of the material in Region 6 by 5%, resulting in a step perturbation at time zero. TDMC shows that the multiplicity factor is increased to $k_{eff} = 1.0011$. The relative flux versus time is shown in Fig. 2. The normalized distributions of the scalar flux versus position at time zero and 0.01 s for groups one and two are displayed in Figs. 3 and 4, respectively. In all cases, TDMC shows excellent agreement with the benchmark results (Goluoglu and Dodds, 2001).

3.4.2. Time-varying boundaries

TDMC code can calculate neutronic parameters of a system with moving boundaries or interfaces between the different regions. We assume that at time zero the interface between Regions 3 and 4 is moved 0.001 cm to the left-hand side and the interface between Regions 4 and 5 is moved 0.001 cm to the right-hand side in any cycle. Thus, the thickness of the absorbers decreases with the rate 2.65 cm/ms and the thickness of Region 4 (fuel) increases with the rate 5.30 cm/ms. This transient is the same as withdrawing a control rod from a reactor with solution fuel in a one-dimensional geometry. If we consider short times following the time zero, TDMC results show that the motion of the interfaces corresponds approximately to a ramp reactivity, $\rho(t) = \gamma t$, with $\gamma = 44.7$ s$^{-1}$. The values of reactivity, as a function of time, are given in Fig. 5.

The results of TDMC for flux variation have been compared with the solutions of Point Kinetics equation and are shown in Fig. 6.

With the motion of the interfaces, the distribution of the scalar flux changes with time. This distribution for group 1 is shown in Fig. 7 at several specific times.

3.4.3. Time-varying cross sections

We assume that the system of problem 5 is operating with unit power. Then, at time $t = 0$ the fission cross section of Region 4 is decreasing by 0.1% in any cycle. If we consider short times following the transient, TDMC results show that the reactivity follows $\rho(t) = -0.2422E9 \times t^3 + 0.4592E6 \times t^2 - 0.3401E3 \times t + 0.0002$ where $t$ is time in seconds. After 7.5 ms, the reactivity is equal to $-0.10$, which is about as large a negative reactivity as can be realized in most reactors. The results of TDMC for reactivity insertion following the reduction of fission cross section in Region 4 are shown in Fig. 8.

The results of TDMC and the Point Kinetic equations for flux variation are shown in Fig. 9. The time-dependent flux of group 1 at several specific times is shown in Fig. 10.

4. Conclusions

A new code for neutronic calculations of time-dependent multiplying systems with explicit representation of delayed neutrons has been developed by using a Monte Carlo method. This code, that is named TDMC, can simulate the systems with time-varying boundaries and cross sections. In this code, the number of energy groups and delayed neutron precursors is selected by the user. The distribution of neutron flux for each energy group can be calculated during a transient. TDMC has been evaluated by comparison with computational benchmark problems with satisfactory agreement. TDMC can be used for analyzing the dynamic behavior of complex neutronic systems.

References


