JOINT INSTITUTE FOR NUCLEAR RESEARCH
Frank Laboratory of Neutron Physics

FINAL REPORT ON THE
SUMMER STUDENT PROGRAM

MCNP code for research neutron sources

Supervisor:
Dr. Maksim Bulavin.
Dr. Alexander Verhoglyadov.

Student:
Mohamed Sayed, Egypt
Tomsk Polytechnic University &
Egyptian Russian University

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Contents

Abstract

The Monte Carlo N-Particle code (MCNP) is the internationally recognized code for k_{eff} computation and some neutron computations with specific nuclear material compositions, geometric structures and nucleus size by the Monte Carlo method. For the VVER system, we measured the neutron flux of some geometries, also calculated the keff that arises from the system's nuclear material compositions, geometric structures and fundamental size changes, by two different approaches, the MCNP perturbation technique and the MCNP divergence method. When we simulate the core of a VVER reactor through the MCNP code after reducing the core to the reactor and expressing it with the proportions of the materials inside it, we find that because the physical structures, geometric structures and fundamental size changes in the VVER system are small compared to the whole system, the keff results obtained from the VVER technology MCNP disruption be more efficient and effective. Reliable results from the MCNP divergence method. When the material combinations, engineering structures, and core size changes in the VVER system are large compared to the entire system, both the MCNP perturbation technique and the MCNP difference method can give satisfactory results, and by manipulating the enrichment ratio we clearly see the change in the k_{eff} value. Also calculating the neutron flux and the reduced Keff value in a single fuel bundle.

1. Introduction

MCNP is the Monte Carlo image for common reason N {particles that can be utilized for a neutron, photon, electron, or neutron/photon/electron-associated harbor, counting the capacity to calculate eigenvalues for basic frameworks. The code handles a three-dimensional arbitrary setup of materials in geometric cells bounded by first-order, second-order and fourth-order ellipsoidal surfaces. Pointwise cross-section information is utilized. For neutrons, all responses given in a specific cross-section assessment (such as ENDF/B-VI) are accounted for. Warm neutrons are portrayed by both the free gas and S (α, β) models. For photons, the code takes account of incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, absorption in pair production with local emission of annihilation radiation, and bremsstrahlung. A continuous slowing down model is used for electron transport that includes positrons, k x {rays, and bremsstrahlung but does not include external or self-induced fields. Important standard features that make MCNP very versatile and easy to use include a powerful general source, criticality source, and surface source; both geometry and output tally plotters; a rich collection of variance reduction techniques; a flexible tally structure; and an extensive collection of cross-section data[1].

2. The Monte Carlo Method

Monte Carlo methods, or Monte Carlo experiments, are a broad class of computational algorithms that rely on repeated random sampling to obtain numerical results. The underlying concept is to use randomness to solve problems that might be deterministic in principle. They are often used in physical and mathematical problems and are most useful when it is difficult or impossible to use other approaches. Monte Carlo methods are mainly used in three problem classes: optimization, numerical integration, and generating draws from a probability distribution.

Figure 1. speaks to the irregular history of a neutron occurrence on a chunk of fabric that can experience parting. Numbers between and 1 are chosen arbitrarily to decide what (on the off chance that any) and where interaction takes put, based on the rules (material science) and probabilities (transport information) administering the forms and materials

included. In this specific case, a neutron collision happens at occasion 1. The neutron is scattered within the heading appeared, which is chosen arbitrarily from the physical scattering distribution. A photon is additionally delivered and is briefly put away, or managed an account, for later analysis. At occasion 2, parting happens, coming about within the end of the approaching neutron and the birth of two active neutrons and one photon. One neutron and he photon are kept money for afterward investigation. The primary parting neutron is captured at occasion 3 and ended. The kept money neutron is presently recovered and, by irregular testing, spills out of the piece at occasion 4. The fission-produced photon incorporates a collision at occasion 5 and spills out at occasion 6. The remaining photon produced at occasion 1 is presently taken after with a capture at occasion 7. Note that MCNP recovers banked particles such that the last molecule put away within the bank is the primary molecule taken out. This neutron history is presently total. As increasingly such histories are taken after, the neutron and photon disseminations ended up superior known. The amounts of intrigued (anything the client demands) are counted, beside gauges of the factual accuracy (instability) of the results[1].
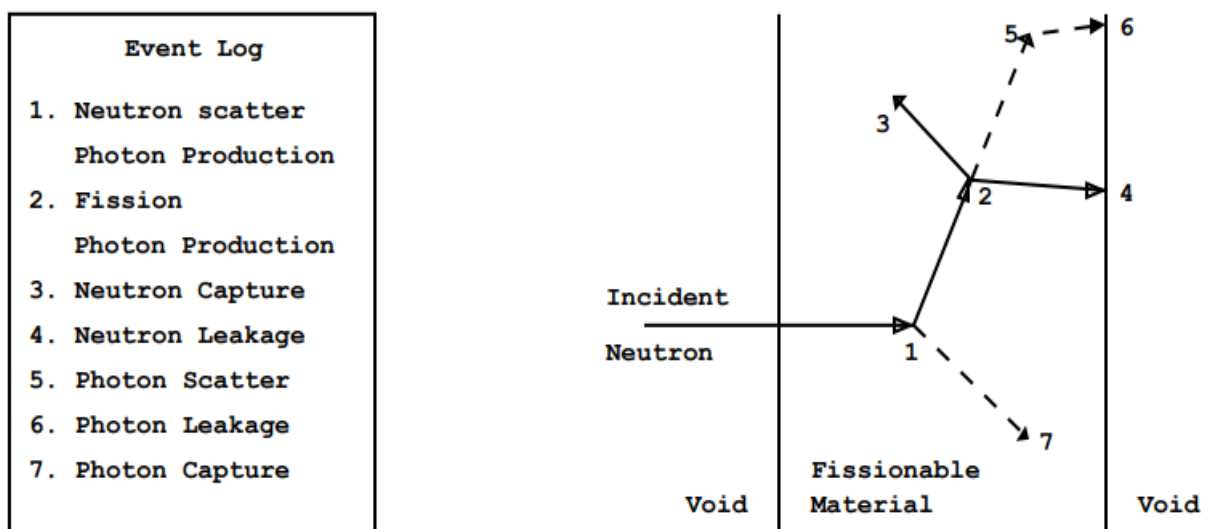

Fig 1. represents the random history of a neutron incident on a slab of material that can undergo fission[1].

3. Examples

In this section, we will apply some examples that explain the writing of Monte Carlo, where you will gradually review some geometric shapes from simple shapes to complex shapes.

3.1 The first Example:

A cylinder inside cube

```
c _____Cell Card_____
    1    1   -10         -1   2   -3
    2    2   -8.9         3  -4  -5   6   7  -8   9
    3    0    #1 #2

c _____Cell surfaces_____
    1         pz          5
    2         pz         -5
    3         cz          2.2
    4         py          5
    5         px          5
    6         py         -5
    7         px         -5
    8         pz          5
    9         pz         -5

c _____Data Card_____
m1     93237.60c          0.38305 $Neptunium
       7014.60c           0.38305 11023.60c          0.06218 26000.50c
0.12535
       24000.50c          0.03091 28000.50c          0.01545
imp:n   1                 1r                0               $ 1, 3
m2     92235.60c 0.0853 8016.60c 0.1709 40000.60c 0.0974 2003.60c
0.0137&
lwtr.04t 0.6327
sdef pos=0 0 0 erg=3 tme=0 wgt=1 par=N
f12:n 1 $Surface flux cell 1
f22:n 2 $Surface flux cell 2
e0 1 12I 14
nps 50000
print
```
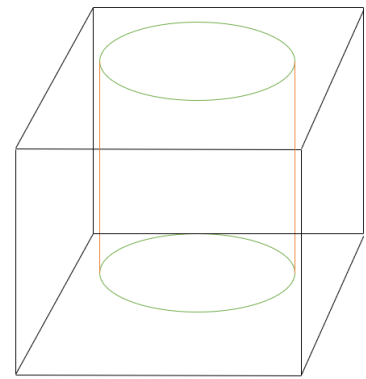
3.2 The second Example (KCODE):

MCNP uses three different estimators for k_{eff}. The advantages of each estimator vary with the problem: no one estimator will be the best for all problems. All estimators and their estimated standard deviations are valid under the assumption that they are unbiased and consistent, therefore representative of the true parameters of the population. This statement has been validated empirically for all MCNP estimators for small dominance

ratios. The batched kef f results table should be used to estimate if the calculated batch-size-of-one kef f standard deviation appears to be adequate.

In this example we have a nuclear reactor core with certain dimensions (diameter 256 cm, height 281 cm), the nuclear fuel used is uranium oxide(U_2O.), zirconium cladding (Zr,1%Nb), and a light water moderator (H_2O.), then we will calculate the inverter required in order to reach the critical temperature, and a reflector made of a light material like graphite or beryllium will also serve as a neutron moderator reducing neutron kinetic energy, while a heavy material like lead or lead-bismuth eutectic will have less effect on neutron velocity, so here the Neutron Reflector Material Be_2C.
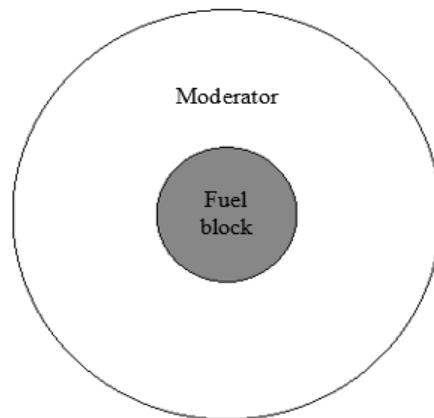

Fig 2. Transformation of the elementary cell into the equivalent one.

Determination of volume shares and nuclear densities of components. These characteristics allows us to calculate macroscopic cross-sections, which allow defining probabilities of different neutron interactions in reactor core. Let's suppose, that a square elementary cell is given, which means that all of the needed sizes are known. And it can be represented as sum of (volume of water); (volume of fuel claddings, which is volume of constriction materials); (fuel volume); (central holes), and (gas gaps). Or turning the elementary cell into equivalent one, we choose fuel as the area with high absorption properties. The materials that we have inside reactor core with a certain present so:

Reactor core:

$$V_{(UO_2)} = 0.2562\ \%, mass\ density\ \gamma_{UO_2} = 10.97\ g/cm^3.$$

$$V_{(H_2O)} = 0.8573\ \%, mass\ density, \gamma_{H_2O} = f(p,t) = 0.70\ g/cm^3.$$

$$V_H = 0.572\ \%, mass\ density, \gamma_H = 0.0899\ g/cm^3\ .$$

$$V_O = 0.2853\ \%, \qquad mass\ density, \gamma_O = 1.429\ g/cm^3\ .$$

$$V_{(gg(helium))} = 0.00477, mass\ density, \gamma_{He} = 1.78 \cdot 10^{-4}\ g/cm^3.$$

$$V_{(Zr,1\%Nb)} = 0.0974.\ mass\ density\ \gamma_{Zr} = 6.52\ g/cm^3\ .$$
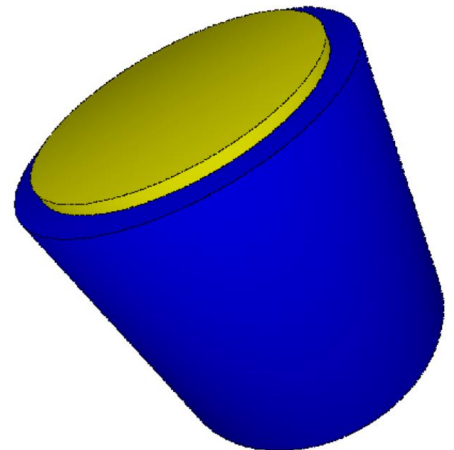
$$V_{(Boron)} = 0.01446.\ mass\ density\ \gamma_{Zr} = 2.46\ g/cm^3\ .$$

Reflector: High Neutron Reflector Materials (Be_2C), with mass density $1.9\ g/cm^3$, and thickness 22 cm.

Let's go write the code. And see what it looks like:

```
c _____Cell Card_____
      1      1   -2.055    -1   2   -3
      2      2   -1.9       3  -1   2   -6
      3      3   -0.7       1  -3  -8
      4      3   -0.7      -2  -3  10
      5      0  -11        #1 #2 #3 #4
      6      0   11

c _____Cell surfaces_____
      1          pz       140.5
      2          pz      -140.5
      3          cz       128
      6          cz       150
      8          pz       150.5
     10          pz      -150.5
     11          so       5000

kcode 20000 1.000000 5 0
ksrc  0.000000 0.000000 0.000000

c _____Data Card_____
m1     92235.60c          0.0549 $UO2(U235+U238(0.2562)),H2O(0.6327),Zr
       92238.50c          0.0304 8016.60c          0.3816 40000.60c
0.0974
       2003.60c           0.0137 1001.60c          0.422
m2     4009.60c           0.667 $Be2C
       6012.50c           0.333
m3     1001.60c           0.667 $H2O
       8016.60c           0.333
imp:n  1                 4r                0                $ 1, 6
f2:n 6 $Surface flux surf 6
e2 0 100I 0.03
ctme 70
print
```
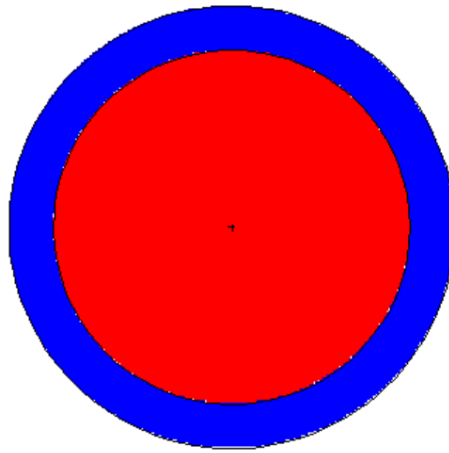
Fig 3. Transformation of the elementary cell into the equivalent one**.**

3.2.1 Graph and result of core

It turns out that in this code we can control and modify the material and thickness to be more suitable for our work. So outside of the Reflector almost numbers of neutrons and its energy closet to zero, it is because there is a reflector and | = final k_{eff} = 0.9965) and the biggest of error almost equal 1 %.
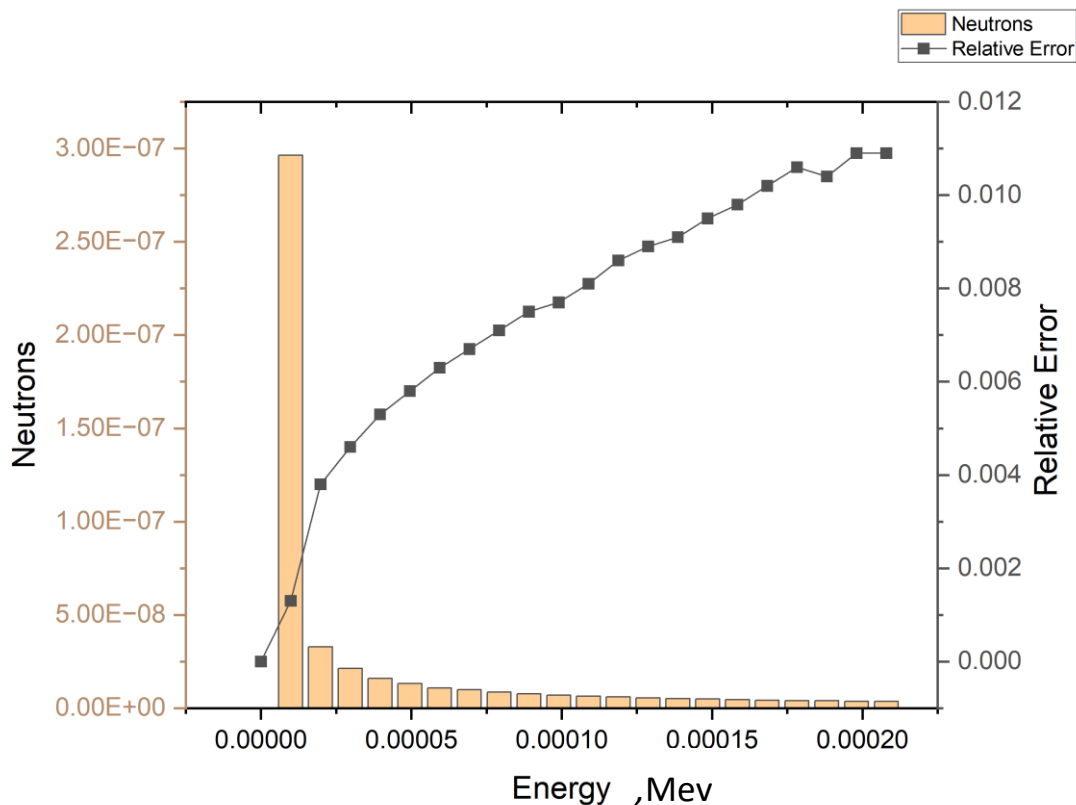

Fig 4. Relation of neutrons and its energy and relative error.

3.3 The third Example (Square BWR fuel assembly array layout)

In this issue we will match the specifications of a parallelogram type fuel assembly for

the Monte Carlo trial, where we have inside the nuclear fuel package a number of different dimensions, fuel uranium 235 rod of an enrichment rate of up to 5%, surrounded by a zirconium cladding and also boron rods to slow down neutrons.
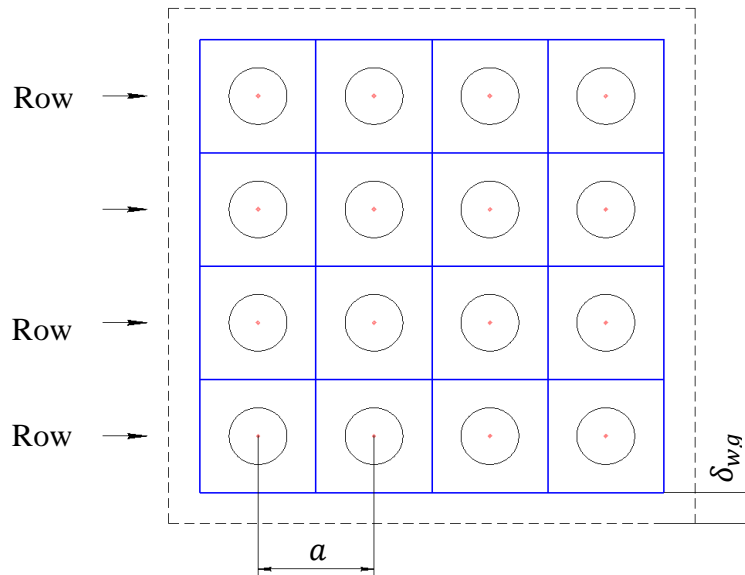

Figure 5. Fuel assembly with square lattice

So, let's see how the code was written:

```
c _____Cell Card_____
    1        0   21   -22   23   -24   5   -6    fill=1
    2        0   11   -12   13   -14   5   -6    u=1 lat=1
            fill=-9:9 -9:9 0:0
       1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 $ROW 1
       1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1 $ROW 2
       1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1 $ROW 3
       1 2 2 2 2 2 4 2 2 4 2 2 4 2 2 2 2 2 1 $ROW 4
       1 2 2 2 4 2 2 2 2 2 2 2 2 2 4 2 2 2 1 $ROW 5
       1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1 $ROW 6
       1 2 2 4 2 2 4 2 2 4 2 2 4 2 2 4 2 2 1 $ROW 7
       1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1 $ROW 8
       1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1 $ROW 9
       1 2 2 4 2 2 4 2 2 4 2 2 4 2 2 4 2 2 1 $ROW 10
       1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1 $ROW 11
       1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1 $ROW 12
       1 2 2 4 2 2 4 2 2 4 2 2 4 2 2 4 2 2 1 $ROW 13
       1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1 $ROW 14
       1 2 2 2 4 2 2 2 2 2 2 2 2 2 4 2 2 2 1 $ROW 15
       1 2 2 2 2 2 4 2 2 4 2 2 4 2 2 2 2 2 1 $ROW 16
       1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1 $ROW 17
       1 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 1 $ROW 18
       1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 $ROW 19
```

```
   3      1   -1.36411   -1          u=2
   4      1   -1.36411    1   -2     u=2
   5      2   -1.9         2          u=2
   6      2   -1.9        -3          u=4
   7      0                3   -4     u=4
   8      0                4          u=4
   9      0               25
  10      0               #9 #1
```

```
c _____Cell surfaces_____
   1      cz       0.38
   2      cz       0.39
   3      cz       0.445
   4      cz       0.565
   5      pz       -50
   6      pz        50
  11      px       -0.6325
  12      px        0.6325
  13      py       -0.6325
  14      py        0.6325
  21      px       -12
  22      px        12
  23      py       -12
  24      py        12
  25      so        60
```

```
c _____Data Card_____
m1     92235.60c          0.033
$UO2(U235+U238(0.2562)),H2O(0.6327),Zr,B
       92238.50c          0.01146 8016.60c          0.34477 40000.60c
0.034
       2003.60c           0.00477 1001.60c          0.572
m2     1001.60c           0.667 $H2O
       8016.60c           0.333
imp:n  1               7r             0            1                 $ 1, 10
sdef pos 0 0 0 par 1
c kcode 1000 0 5 0
c ksrc 0 0 0
f12:n 21 $Surface flux surf 21
f22:n 22 $Surface flux surf 22
e12 0 100I 0.03
e22 0 100I 0.1
ctme 4
print
```
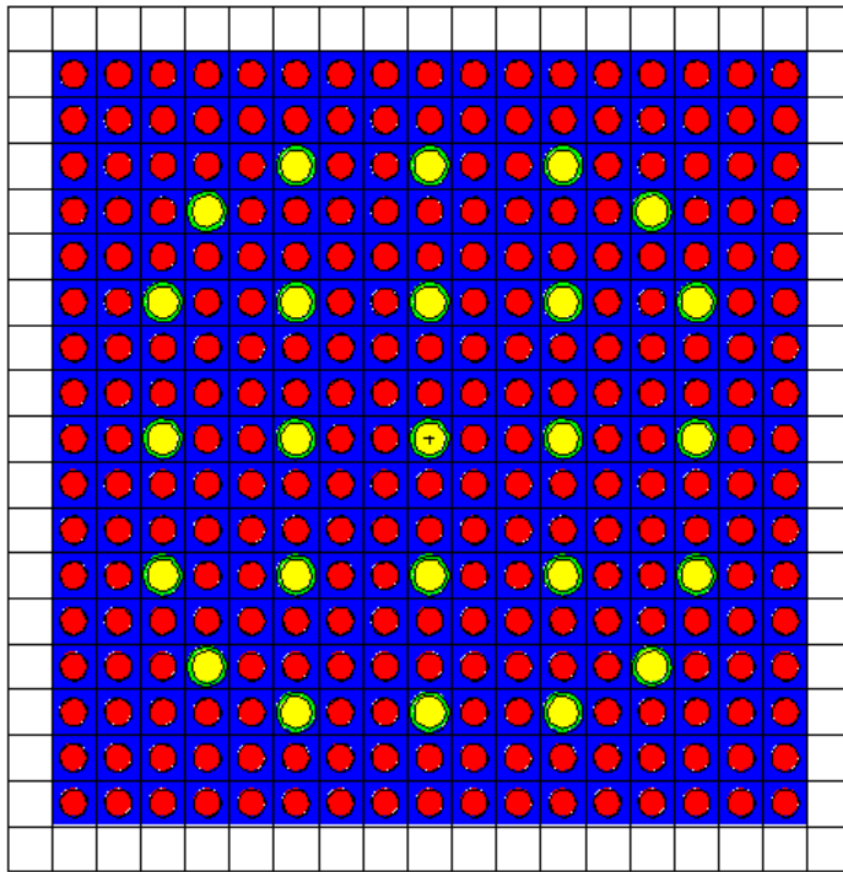
Figure 6. Fuel assembly with square lattice.

Conclusion

By applying or using the Monte Carlo code specifically of the MCNP type, we were able to explain how to write the code and clarify and read after the results, through the exercises provided and graded from the easiest to the most difficult. The data explained the importance of this code in trying what happens inside a nuclear reactor, atomic particles, etc.

The computer software models how electrons, photons, and neutrons interact with the system's substance. Considered include nuclear fission by neutrons as well as scattering and capture processes. It produces a source of secondary particles produced by nuclear processes (neutron, photon, or electron fission) or by the collision of an electron with another electron. It works well to resolve issues in the fields of radiology, radiation protection, and nuclear reactor physics.

Referances

[1] J. F. Briesmeister, "Version 4B," no. March, 1997.

[2] P. Del Moral, A. Doucet, and A. Jasra, "Sequential Monte Carlo samplers," *J. R. Stat. Soc. Ser. B*, vol. 68, no. 3, pp. 411–436, Jun. 2006, doi: 10.1111/j.1467-9868.2006.00553.x.

[3] "Monte Carlo Simulation: History, How it Works, and 4 Key Steps." https://www.investopedia.com/terms/m/montecarlosimulation.asp (accessed Sep. 20, 2022).

[4] "Full article: Supporting Design Analysis of the VTR Using MCNP and TRACE." https://www.tandfonline.com/doi/full/10.1080/00295639.2022.2053488 (accessed Sep. 25, 2022).

[5] "(PDF) Monte Carlo MCNP-4B–Based Absorbed Dose Distribution Estimates for Patient-Specific Dosimetry." https://www.researchgate.net/publication/11998722_Monte_Carlo_MCNP-4B-Based_Absorbed_Dose_Distribution_Estimates_for_Patient-Specific_Dosimetry (accessed Sep. 25, 2022).