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FINAL REPORT ON THE START PROGRAMME

Calculation of the neutron flux around the core of the IBR-2 research reactor using Monte Carlo simulation

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ABSTRACT

In this work, for the IBR-2 research reactor system, we simulate the core of the reactor through the MCNP code to gain the neutron flux of some geometries, also calculated the $\boldsymbol{k_{eff}}$. Reliable results from the MCNP divergence method.

The Monte Carlo N-Particle code (MCNP) is the internationally recognized code for $\boldsymbol{k_{eff}}$ computation and some particles (γ, e^-, e^+, n, , ions and some leptons, baryons, mesons) computations with specific nuclear material compositions, geometric structures and nucleus size by the Monte Carlo method.

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].

1. MONTE CARLO METHOD

The practical application of the Monte Carlo method is due to the fact that the solution of problems related to determining the states of complex systems subject to random perturbations cannot be found analytically. For this purpose, simulation models are used that consider the probabilistic nature of processes (using probability distribution laws, spectral densities, or correlation functions). Due to the random nature of the observed value of the parameter under study, it is necessary to repeatedly reproduce the process with subsequent statistical processing of the data obtained.

The Monte Carlo method is a simulation method for studying complex processes and systems, as well as evaluating the results of a statistical process. The individual probabilistic events that make up the process are modeled sequentially, in accordance with the probability distribution that determines these events. The process is based on the selection of random numbers. The simulation consists of following each particle individually until it dies (absorption, stops, etc.). For this purpose, a "run" of pseudorandom sequences with a given correlation and probability distribution law (Monte Carlo method) is used, simulating random parameter values on a computer with each "run".

The Monte Carlo method models the behavior of each particle, and then the average behavior of the particles is obtained by the central limit theorem – the sum of a sufficiently large number of independent random variables has a distribution close to normal.

The essence of the Monte Carlo method is to obtain perturbations of a real system close to the results based on many "runs" of individual processes. Therefore, the reliability of the results obtained using the Monte Carlo method is crucially determined by the quality of the random number generator [2].

1.1. MATH THEORY OF MONTE CARLO METHOD

A random variable ζ is called continuous if it is defined on a given interval (a, b), which contains all possible values of our random variable, and has a probability density equal to some function p(x):

$$P(a' < \zeta < b') = \int_{a'}^{b'} p(x)dx \tag{1}$$

If p(x) = const, then a random variable is called uniformly distributed over the interval [a, b]. Accordingly, for any random variable, the probability integral over the interval (a, b) – P(a<ζ<b) should be equal to 1.

The Monte Carlo method uses the value γ, uniformly distributed over the interval [0,1] with p(x)=1:

$$\int_{a'}^{b'} p(x)dx = b' - a' \tag{2}$$

The sequence of such numbers γ is given by random number generators.

To obtain the value of a random variable ζ distributed in the interval (a, b) with a given density p(x), Monte Carlo methods show that ζ can be found from the equation

$$\int_{a}^{\zeta} p(x)dx = \gamma \tag{3}$$

i.e., in order to find the value of ζ, it is necessary to solve this equation by choosing the next value γ.

For example, to draw a random variable τ, for which the probability density will correspond to the probability that the particle will travel the distance x, i.e. $p(x) = \Sigma e^{-\Sigma x}$:

$$\int_{0}^{\tau} \Sigma e^{-\Sigma x} dx = \gamma \tag{4}$$

$$\tau = -\frac{1}{\Sigma}\ln(1-\gamma) = -\frac{1}{\Sigma}\ln\gamma \tag{5}$$

The value of 1-γ is distributed exactly the same as γ, so we replaced ln(1− γ) on ln γ. We obtain that τ is the distance to the collision [3].

2. IBR-2M

IBR-2 is a pulsed fast reactor of periodic operation. Its main difference from other reactors consists in mechanical reactivity modulation by a movable reflector.

The core of the IBR-2M reactor has the shape of an irregular hexagonal prism positioned vertically (Fig.1). Two coaxially rotating movable reflectors, the main (MMR) and additional (AMR), which are a reactivity modulator, pass by one of the faces. The remaining five faces are adjacent to stationary reflectors with emergency protection and reactor control units. During rotation, the movable reflectors create periodic reactivity pulses in the IBR-2M, during which the reactor briefly goes into a supercritical state with instantaneous neutrons. As a result, the reactor generates short neutron pulses (200 μs at half height) with a period of 0.2 s and with an average power of 2 MW, the pulse power reaches 1850 MW. Almost all the energy released during the period is generated in a pulse (92%).

The main characteristics of the IBR-2M are given in the table.1. The core, designed to load 69 cassettes (each of seven fuel rods with tablets of plutonium dioxide), is cooled with liquid sodium. The moving parts of a stationary reflector made of tungsten (emergency protection units, compensating organs and an intermediate regulator) and an automatic regulator rod made of beryllium in the matrix of a stationary steel reflector are used as control and protection systems. In the event of a power reset alarm, the emergency protection units move downwards from the active zone at high speed and, during the time between pulses, transfer the reactor from a state of equilibrium pulsed supercriticism to a deeply subcritical state [4].

The movable reflector is a complex mechanical system providing reliable operation of two parts, which determine the reactivity modulation: the main movable reflector and the auxiliary movable reflector.

The rotors of the main and auxiliary movable reflectors rotate in opposite directions with different velocities (300 and 600 revolutions per minute). When both reflectors coincide near the reactor core, a power pulse was generated.

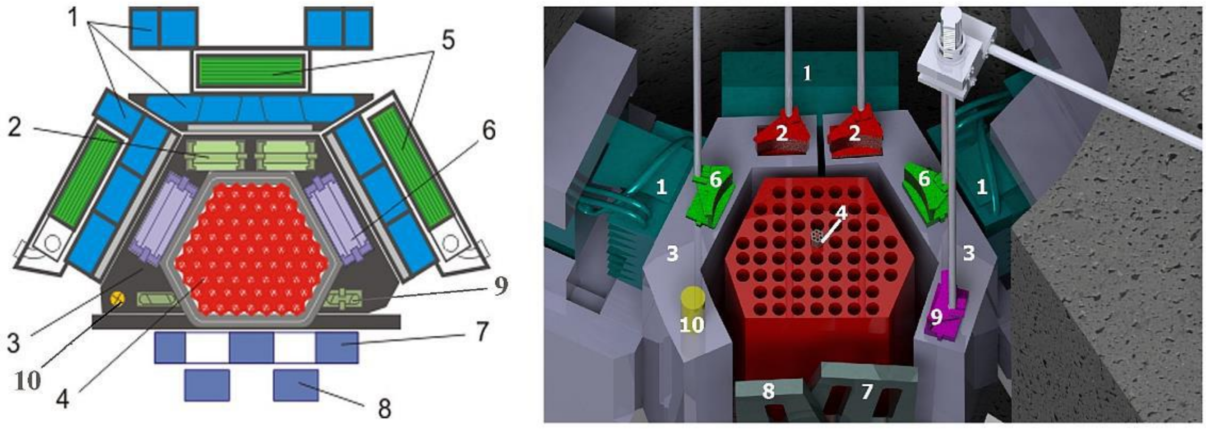

Figure 1. Design scheme of the IBR-2M core with complex of moderators. Horizontal cut-off (1 – water moderators; 2 – emergency protection units; 3 – stationary reflector matrix; 4 – active zone; 5 – cold moderator chambers; 6 – blocks of compensating units; 7 – main movable reflector; 8 – auxiliary movable reflector; 9 – manual regulator; 10 – Automatic regulator.). [5]

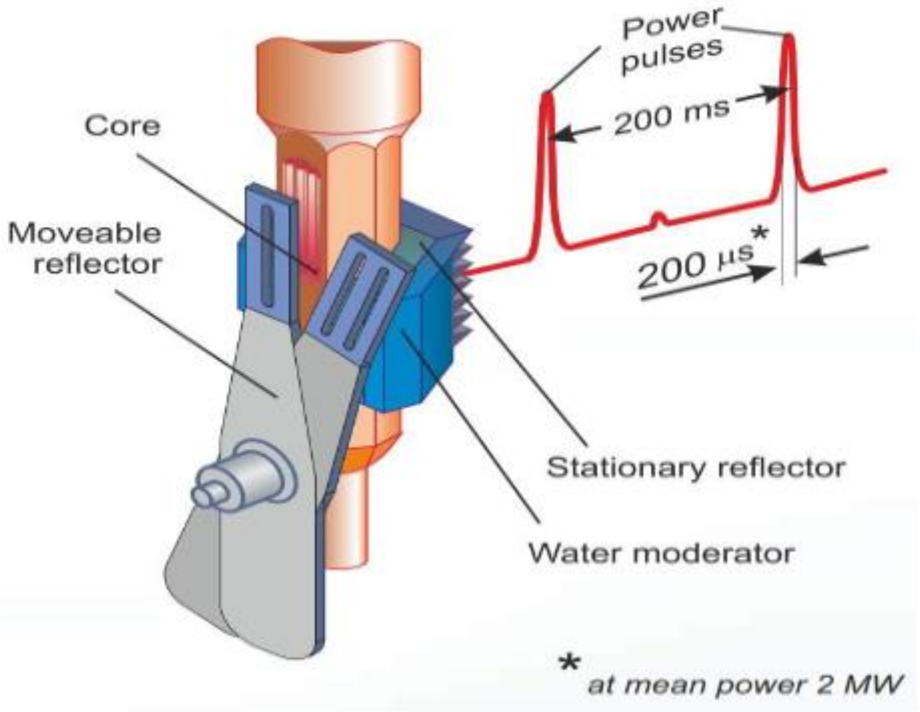

Figure 2. IBR-2M Reactor [5]

Table 1. Parameters of the IBR-2 [6]

Name	Value
Average power, MW	2
Fuel	PuO_2
Number of fuel assemblies	69
Maximum burnup, %	9
Pulse repetition rate, Hz	5; 10
Pulse half-width, μs: fast neutrons thermal neutrons	 200 340
Rotation rate, rev/min: main reflector auxiliary reflector	 600 300
MMR and AMR material	nickel + steel
MR service life, hours	55000
Source of spontaneous fission	Contains 0,43 $\mu g\ of\ {}_{252}^{98}Cf$
Moderator	Ordinary Water (H_2O) and mixture of mesitylene (C_9H_{12}) and meta-xylene ($C_6H_4(CH_3)_2$)

3. PREPARATION FOR SIMULATING

In this work, the geometries of the reactor core provided by the supervisor were used for simulating. The reactor core model did not include both reflectors available in the design. The core was replaced by a hexagonal grid of elements (fuel assemblies).

The core was stripped down on one side; 7 fuel rods (a total of 64 fuel assemblies). In such an assembly, the following number of fuel assemblies is presented line by line: 5-6-7-8-9-10-9-8-7.

The KCODE simulating algorithm was used, which consists in estimating the average number of fission neutrons produced per generation for each fission neutron initiated. The initial point sources of neutrons are located in the middle of the central 7 fuel rods (in the central fuel assembly of the line in front of the line with the largest number of fuel assemblies in the first case).

The flow results are presented as the number of particles passing through the measurement areas divided by the area and normalized by the total number of simulated neutrons. The neutron flux measurement areas are located outside the core and consist of two spheres with a volume of 1 cm^3.

Table 2 describes the parameters used for simulating. Table 3 includes the used materials. Appendix A shows diagrams of the fuel element and the reactor core. Also Appendix A shows diagrams of the reactor core in MCNPX (where we simulated it).

Table 2. Parameters for simulation.

Name	Value
Fuel rod diameter x shell thickness, mm	6,9 × 0,4
Shell for sodium	ЧС-68, ЭИ-847
Diameter of the fuel sleeve, mm	5,9
Diameter of the inner hole of the fuel sleeve, mm	1,5
Height of the fuel part, mm	440
The Sublayer	He
The pitch of the fuel rod, mm	7,4
Tablet density, ρ_{cp}, g/cm3	10,71
Oxygen coefficient in fuel rod	1,955
The material of the fuel assembly covers and housing	08X18H10T
End reflector material	ВНЖ-90

Table 3. Materials for simulation.

Material * Nuclei	Density (g/cm^3) * Fraction (%)
Plutonium Dioxide * Pu-238 * Pu-239 * Pu-240 * Pu-241 * Pu-242 * O	10,71 * 0,2 * 92,61 * 4,01 * 0,17 * 0,022 * 194,024
Helium * He	5,0 * 100

ЧС-68ИД * C * Si * Mn * Ni * S * P * Cr * Mo * V * Ti	7,8 * 0,05-0,08 * 0,3-0,6 * 1,3-2 * 14-15,5 * 0,012 * 0,02 * 15,5-17 * 1,9-2,5 * 0,1-0,3 * 0,2-0,5
08Х18Н10Т * C * Si * Mn * P * S * Cr * Mo * Ni * V * Ti * Cu * W * Fe	7,9 * 0,08 * 0,8 * 2,0 * 0,035 * 0,02 * 17,0-19,0 * 0,3 * 9,0-11,0 * 0,2 * 0,7 * 0,4 * 0,2 * remainder
ВНЖ-90 * Fe * Ni * W	16,8 * 6,8–7,4 * 2,8–3,4 * remainder

RESULTS AND CONCLUSION

For 1000 simulation cycles of 2500 neutrons each for 7-64 configuration (IBR-2M) without both of movable reflector:

- The average energy of neutrons is ~ 2,115 MeV (average energy per neutron);
- The average lifetime of neutrons is ~ 3,708 * 10^{-6} s;
- The average neutron energy deposition in reactor core is ~ 2,094 MeV (average energy deposition per neutron);
- The calculated neutron multiplication factor is $\boldsymbol{k_{eff}}$ ~ 0,409;
- Neutron flux calculation results (2,5 * 10^6 simulated neutrons) is 1,47957 * 10^{-7};
- Delayed neutron fraction calculation results is βeff ~ 0,0018.

We used formula from [6]:

$$\beta_{eff} = 1 - \mathrm{k_p}/\mathrm{k_{eff}} \tag{6}$$

where k_{eff} and k_p is multiplication factor obtained as a result of simulating with and without delayed neutrons (TOTNU CARD with NO parameter), respectively.

In total, about ~35% of neutrons reach the area outside the core shell.

To obtain accurate results, it is necessary to achieve statistics of about 10^{10} neutrons or more (for example, the number of neutrons corresponding to the reactor neutron flux, 10^{14}).

The computer software based on the Monte Carlo method works well to resolve issues in the fields of radiology, radiation protection, and nuclear reactor physics. We able to explain how work detectors, reactors of various shapes and configurations work. We can visualize particle flows, particle self or distribution energy in samples volume. We able to test new types and configurations of products without designing them [8].

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APPENDIX A. CONFIGURATION OF THE FUEL ROD, THE FUEL ASSEMBLY AND THE REACTOR CORE

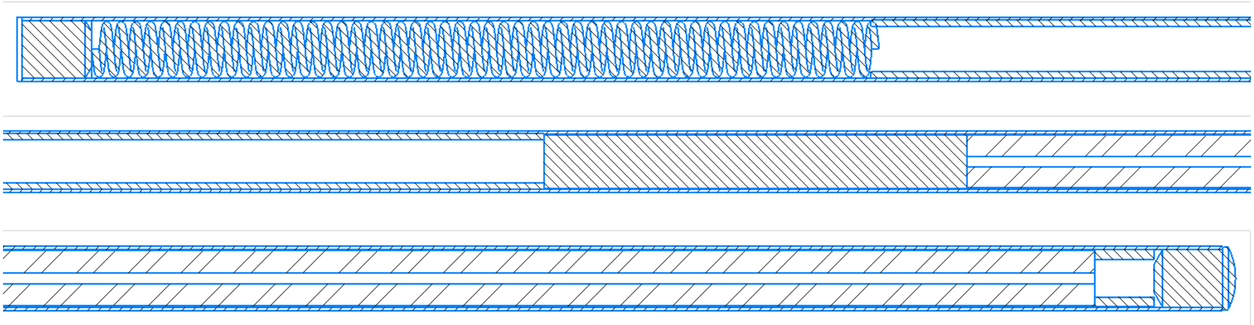

Figure 3. Fuel rod scheme.

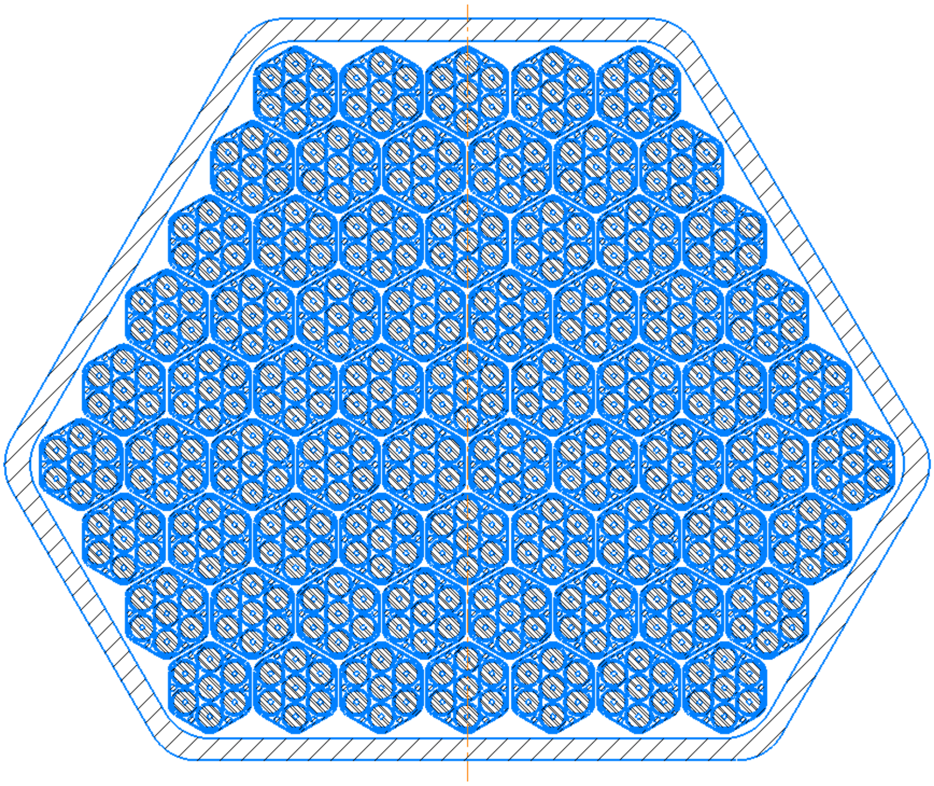

Figure 4. Reactor core. Configuration 7-64.

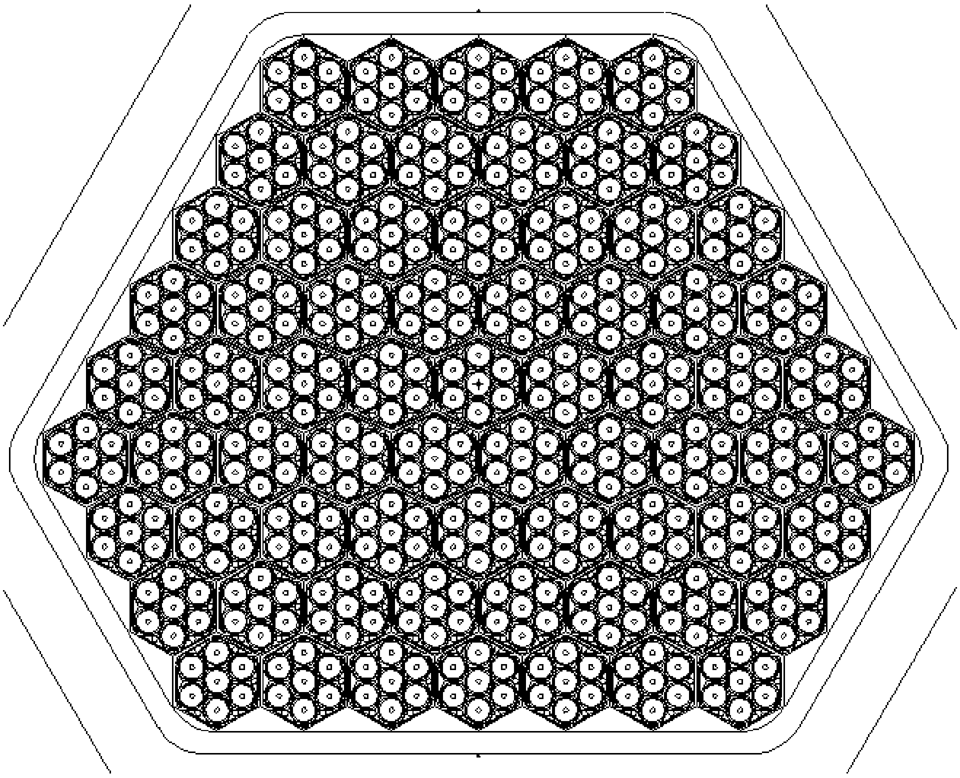

Figure 5. Reactor core in MCNPX. Configuration 7-64.

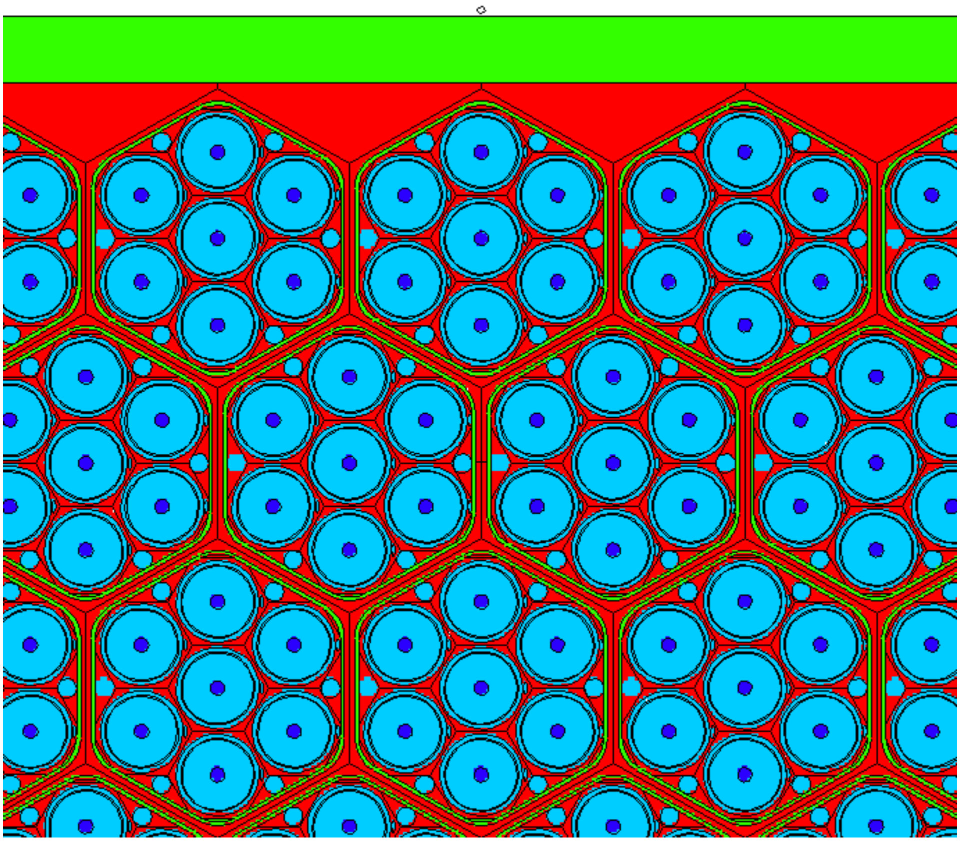

Figure 6. Fragment of reactor core in MCNPX colored by using material. Configuration 7-64.