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# MCNP A GENERAL MONTE CARLO N PARTICLE TRANSPORT CODE

**DARYOUSH MASTI**
***Baku State University***

*One of the strongest calculating codes of the nuclear reactors is MCNP code. This code is designed to base of the Monte Carlo method. The most important restriction on using of the MCNP code is its time consuming and lack of enough memory volume for an extended problem, e.g. neutron calculation of a real reactor core. In this work the calculating codes are used for calculation and determination of the physical parameters of neutron: reactivity, neutron flux,* $K_{eff}$*, etc. of the nuclear reactor with 1000 MW power. The core model has been developed from 1/12 of the core to 1/6 and 1/1 (whole the core) to facilitate calculation of control group's worth. The results are compared with the technical data.*

## 1. Introduction

MCNP is a general-purpose, continuous-energy, generalized-geometry, time-dependent, coupled neutron/photon/electron Monte Carlo transport code. It can be used in several transport modes only: neutron, electron, combined neutron/photon transport where the photons are produced by neutron interactions, neutron/photon/electron, photon/electron, or electron/photon.

The neutron energy regime is from 10-11 MeV to 20 MeV, and the photon and electron energy regimes are from 1 keV to 1000 MeV. The capability to calculate $K_{eff}$ eigenvalues for fissile systems is also a standard feature. One of the benefits of MCNP computer code is to define the detail geometry as well as using continuous cross section library. MCNP has not been successfully vectorized, because the overhead required to set up and break apart vector queues at random decision points is greater than the savings from vectorizing the simple arithmetic between the decision points. MCNP (and any general Monte Carlo code) is little more than a collection of random decision points with some simple arithmetic in between. Because MCNP does not take advantage of vectorization, it is fairly inefficient on vectorized computers. In particular, many workstations and PCs run MCNP as fast or faster than mainframes. MCNP has been made as system independent as possible to enhance its portability, and has been written to comply with the ANSI FORTRAN 77 standard (48,000 lines of FORTRAN and 1000 lines of C).With one source code, MCNP is maintained on many platforms[1].

In this work the neutron parameters of NPP sample core for the first cycle are calculated using MCNP. The core model has been developed from 1/12 of the core to 1/6 and 1/1 (whole the core) to facilitate calculation of control group's worth.The input parameters used in computations numerical are presented in the table 1. The results are compared with the technical data of a sample reactor.

**Tab. (1):Geometric specifications and reactor core materials**

| Parameter | Value | Parameter | Value |
|---|---|---|---|
| Pellet density, $Kg/m3$ | (10.4...10.7) $\times 10^{-3}$ | Absorbing element clad inner diameter, m | $7.2\times 10^{-3}$ |
| Diameter of the fuel pellet central hole, m | $1.5\times 10^{-3}$ | Absorbing material diameter, m | $7.2\times 10^{-3}$ |
| Fuel pellet height, m | (9.0...12.0) $\times 10^{-3}$ | Absorbing element clad outer diameter, m | $8.2\times 10^{-3}$ |
| Helium pressure under fuel rod cladding, MPa | 2.0 ± 0.25 | Nominal height of control rod, m | 4.215 |
| Fuel rod length, m | 3.842 | Clad material | Stainless steel |
| Fuel rod pitch, m | $12.75\times 10^{-3}$ | Equivalent Diameter,m | 3.16 |
| Temperature of fuel rod outside surface, °C | 352 | Active Height, m | 3.53 |
| Max. fuel rod temperature, °C | 1883 | Fuel Assembly number | 163 |
| Clad material | Zr+%1Nb+%0.05 Hf | Fuel weight, Kg | 79840 |
| Inside diameter of the clad, m | $7.73\times 10^{-3}$ | Average inlet coolant temperature, °C | 291 |
| Outside diameter of the clad, m | $9.1\times 10^{-3}$ | Average outlet coolant temperature, °C | 321 |
| Outside diameter of the fuel pellet, m | $7.57 \times 10^{-3}$ | Fuel assembly pitch, cm | 23.6 |
| Fuel type | $UO_2$ pellets | Coolant flow rate, $m^3/h$ | 84000 |
| Number of control rods in fuel assembly, pcs. | 18 | Moderator-fuel area ratio | 1.97 |
| Absorbing material density, $gr/cm3$<br>( $Dy_2 O_3 TiO_2$ )<br>$B_4C$ | 1.7<br>4.9 | Absorbing material height, m<br>( $Dy_2 O_3 TiO_2$ )<br>$B_4C$ | 3.2<br>0.3 |

## 2. Calculated parameters

The calculated parameters are:

Excess reactivity of the core; Negative reactivity of the BAR's (Burnable Absorber Rods); Reactivity worth of working control groups (groups 8, 9 and 10); Variation of core reactivity under sequential movement of CPS AR's (Control and protection system - Absorber rods) working groups (groups 8, 9 and 10); Scram reactivity of all control groups and reactivity curve. control groups position in the core;

Reactivity coefficients at HZP (Hot zero power) condition, including:

i.Boric acid concentration reactivity coefficient ($\partial\rho/\partial CB$);

ii.Coolant density reactivity coefficient ($\partial\rho/\partial\gamma$);

iii.Coolant temperature reactivity coefficient ($\partial\rho/\partial tm$);

iv.Fuel temperature reactivity coefficient ($\partial\rho/\partial t_f$)[2,3].

**1-Excess Reactivity of NPP first core**

It is determined at the beginning of each fuel cycle to compensate fuel burn-up and equilibrium Xe and Sm poisoning. The value of excess reactivity is calculated by assuming there is no boric acid in the coolant (CB=0).

| Excess Reactivity | | |
|---|---|---|
| MCNP | | Sample |
| $K_{eff}$ | ρ(%) | ρ(%) |
| 1.24364 | 19.59 | 18.4 |

**2-Negative Reactivity of BAR's (Burnable Absorber Rods)**

The negative reactivity of BAR's (Burnable Absorber Rods) is calculated by replacing them with fuel assemblies without BAR's. ρΔBAR's=(Reactivity of core without BAR's)–(Reactivity of core with BAR's)

| Negative Reactivity of BAR's | | | |
|---|---|---|---|
| $K_{eff}$ | | MCNP | Sample |
| $K_{eff1}$[1] | $K_{eff2}$[2] | Δ ρ(%) | Δ ρ(%) |
| 1.2713 | 1.2436 | 1.75 | 2.6 |

1] Keff1: without considering BAR's; 2] Keff2: with considering BAR's.

**3- Control and protection system - Absorber rods (CPS-AR) Worth Groups No. 8, 9 and 10**

In the first cycle of NPP, 85 control cluster rods (CPS AR's) will be used.

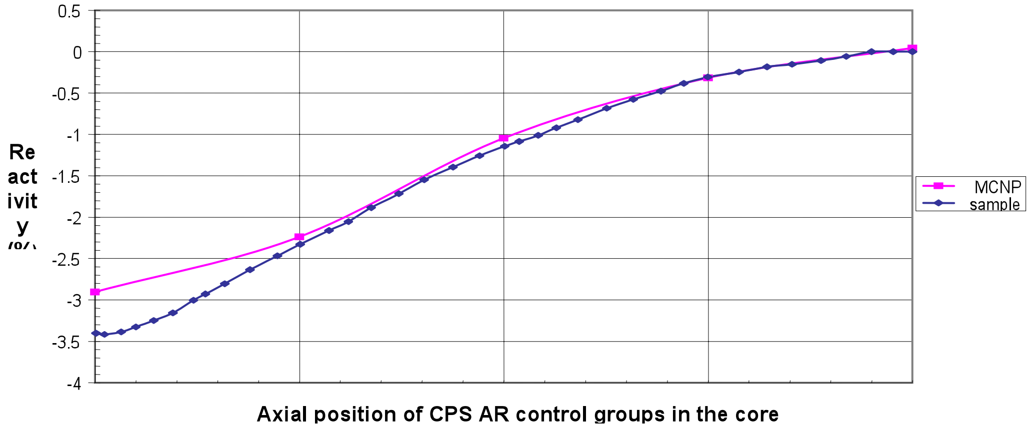
These CPS AR's are divided into 10 groups. Groups No. 8, 9 and 10 are working groups which contain 9, 7 and 6 CPS AR's respectively. The other groups of CPS AR's are exclusively designed for reactor scram.

| CPS AR's | Integral worth (Δ ρ(%)) | |
|---|---|---|
| Control Group No. | MCNP Results | Sample |
| 10 | 0.681 | 0.77- 0.87 |
| 9 | 0.802 | 0.7 – 0.85 |
| 8 | 1.396 | 1.0 – 1.2 |

**4-Variation of core reactivity under sequential movement of CPS AR's working groups (groups 8, 9 and 10).**

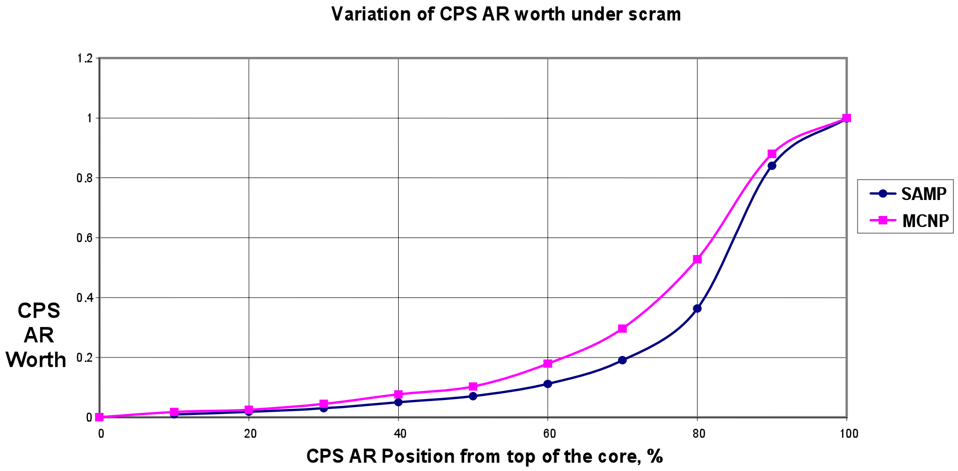
CPS AR's group's No. 10, 9, and 8 are working groups. According to the strategy shown in the following figure for changing the reactor power, these 3 groups are inserted into / removed from the core sequentially. The negative reactivity added to the core versus CPS R's working group's position.

| CPS AR control groups position | MCNP Results | | Sample |
|---|---|---|---|
| Percent of insertion | $K_{eff}$ | Δ ρ(%) | Δ ρ(%) |
| 10 (50%) | 0.99685 | 0.316 | 0.3065 |
| 10 (100%)+9 (50%) | 0.98975 | 1.03562 | 1.143 |
| 10 (100%)+9(100%)+8(50%) | 0.97811 | 2.23799 | 2.327 |
| 10 (100%)+9(100%)+8(100%) | 0.97178 | 2.90395 | 3.4023 |



**5.Scram reactivity of all control groups and reactivity curve the control groups position in the core.**

| CPS AR's position | MCNP Results | | Sample |
|---|---|---|---|
| Percent of insertion | $K_{eff}$ | Δρ(%) | Δρ(%) |
| 10 | 0.99849 | 0.15122 | 0.087 |
| 20 | 0.99788 | 0.21245 | 0.172 |
| 30 | 0.99619 | 0.38245 | 0.278 |
| 40 | 0.99355 | 0.64918 | 0.462 |
| 50 | 0.99137 | 0.87051 | 0.65 |
| 60 | 0.98511 | 1.51150 | 1.026 |
| 70 | 0.97557 | 2.5041 | 1.756 |
| 80 | 0.9573 | 4.46046 | 3.345 |
| 90 | 0.93081 | 7.43331 | 7.733 |
| 100 | 0.92212 | 8.44575 | 9.185 |



## 6-Reactivity coefficients at HZP condition

6-1.Fuel temperature coefficient ($\partial\rho/\partial tf$) at HZP.

Fuel temperature is changed from 280°C to 327°C where other parameters are kept constant.

| Coolant Temperature =280 °C , Coolant Density = 0.76443 g/cm3 Boric Acid Concentration=8.2 g/kg | | | |
|---|---|---|---|
| Fuel Temperature (°C) | $K_{eff}$ | ρ | $\partial\rho/\partial t_f$ (10-5/°C) |
| 280(Criticality) | 0.99947 | -5.30E-04 | -2.452 |
| 327 | 0.99832 | -1.68E-03 | |

6-2.Coolant temperature coefficient ($\partial\rho/\partial t_m$) at HZP. Coolant temperature is changed from 280°C to 290°C considering the effect of variation of its density.

| Fuel Temperature =280 °C , Boric Acid Concentration=8.2 g/kg | | | | |
|---|---|---|---|---|
| Coolant Temp. (°C) | Coolant Density (g/cm3) | $K_{eff}$ | ρ | $\partial\rho/\partial tm$ (10-5/°C) |
| 280(Criticality) | 0.76443 | 0.99947 | -9.21E-04 | -3.905 |
| 290 | 0.74637 | 0.99908 | -1.68E-03 | |

6-3.Coolant density reactivity coefficient ($\partial\rho/\partial\gamma$) at HZP. Coolant density is changed from γ = 0.76443 g/cc (coolant density at critical condition) to γ = 0.8 g/cc .

| Coolant Temperature =280 °C , Fuel Temperature=280 °C Boric Acid Concentration=8.2 g/kg | | | |
|---|---|---|---|
| Coolant Density (g/cm3) | $K_{eff}$ | ρ | $\partial\rho/\partial\gamma$ (1/g/cm-3) |
| 0.76443(Criticality) | 0.99947 | -5.30E-04 | 1.1534E-02 |
| 0.8 | 0.99988 | -1.20E-04 | |

Boric acid concentration coefficient ($\partial\rho/\partial CB$) at HZP 6-4.

The boric acid is changed from $C_B = 8.2$ g/kg (critical value) to $C_B = 8$ g/kg.

| Coolant Temperature =280 °C , Fuel Temperature=280 °C ,<br>Coolant Density = 0.76443 g/cm3 | | | |
|---|---|---|---|
| Boric Acid Conc.(g/kg) | $K_{eff}$ | ρ | $\partial\rho/\partial C_B$ (1/g/kg) |
| 8.2(Criticality) | 0.99947 | -5.30E-04 | -1.7954E-02 |
| 8 | 0.99847 | -3.06E-04 | |

## 3. Comparing the results

We compare our results with exiting sample reactor data:

### Excess and negative reactivity's

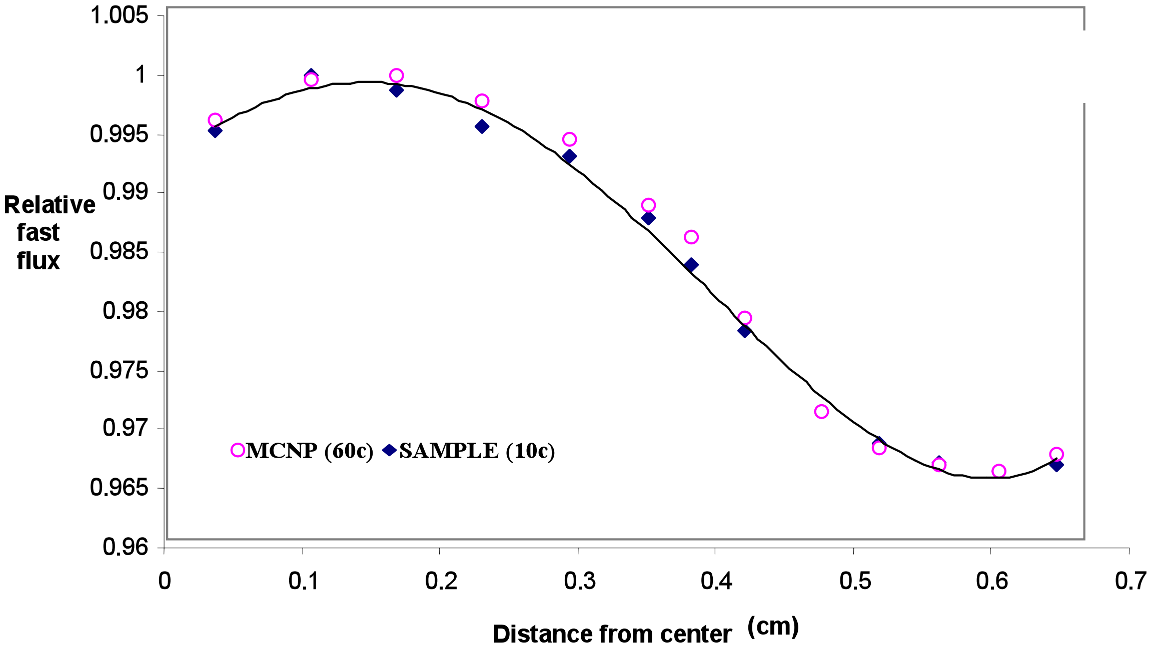
| Reactivity | MCNP | Sample |
|---|---|---|
| Excess Reactivity (ρ (%)) at CZP | 19.59 | 18.4 |
| Negative Reactivity of BAR's (Δρ (%)) at CZP | 1.75 | 2.6 |
| The negative reactivity added to the core by full insertion of all CPS AR's (Δρ (%)) | 8.446 | 9.185 |

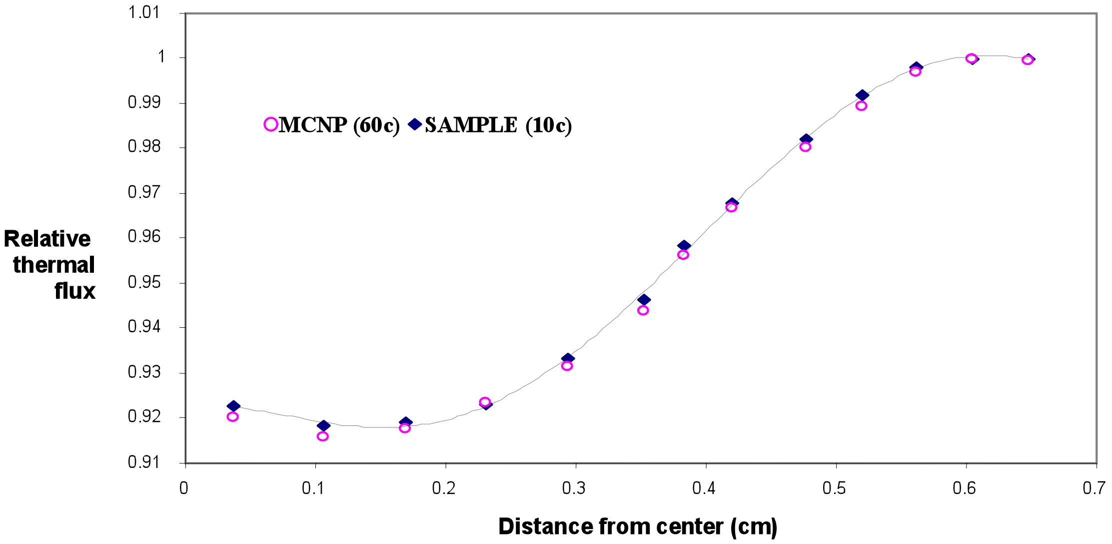
### Integral worth of CPS AR's

| Integral worth of: (Δ ρ(%)) | MCNP | Sample |
|---|---|---|
| Control Group No 10 | 0.681 | 0.77- 0.87 |
| Control Group No 9 | 0.802 | 0.7 – 0.85 |
| Control Group No 8 | 1.396 | 1.0 – 1.2 |

### Reactivity coefficients in different states of the reactor for BOC of the first cycle with CPS AR's withdrawn

| Reactor state | Reactivity coefficients | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Condition | $\partial\rho/\partial\gamma$, 10-2/g/cm-3 | $\partial\rho/\partial t_m$, 10-5/°C | | $\partial\rho/\partial t_f$, 10-5/°C | | | $\partial\rho/\partial_{CB}$, 10-2/g/kg | |
| | MCNP | SAMPLE | MCNP | SAMPLE | MCNP | SAMPLE | MCNP | SAMPLE |
| HZP | 1.153 | 1.14 | -3.905 | -4.05 | 2.452- | -2.70 | 1.795- | -2.07 |





## REFERENCE

[1] Briesmeister, J.F., 2000. MCNP- A General Monte Carlo N-Particle Transport code, Version 4C, Los Alamos National Laboratory Report LA-13709-M.

[2] Members of Cross-Section Evaluating Working Group, Mclane, V., 2001. ENDF-102, Data Formats and Procedures for the Evaluated Nuclear Data File, ENDF-6, National Nuclear Data Center, Brookhaven National Laboratory.

[3] Reactor Physics Constants, Second Edition, ANL-5800, chap 7, section 7.2, 1963

# MCNP: DAŞINMANIN ÜMUMİ N ZƏRRƏCİKLİ MONTE KARLO KODU

## DARYUŞ MASTİ

## XÜLASƏ

Nüvə reaktorlarında ən güclü hesablama metodu MCNP kodu hesab olunur. Bu kod Monte-Karlo metodu əsasında yaranıb inkişaf etdirilmişdir. Böyük hesablama müddətinə, problemin hərtərəfli öyrənilməsinə o qədər də imkan verməyən yaddaşa malik olmasına baxmayaraq, bu metoddan reaktorların aktiv zonalarında neytronların öyrənilməsində geniş istifadə olunur. İşdə Monte-Karlo kodundan 1000MVt gücünə malik olan reaktorda neytronların bir sıra fiziki parametrlərinin: reaktivliyini, neytronlar selinin intensivliyini, $K_{eff}$ və s. hesablanıb və təyin olunmasında istifadə edilir. Aparılan ölçmələrin dəyərini təyin etməkdən ötrü «aktiv zona» modeli reaktorun aktiv zonasının 1/12 hissəsindən 1/6 və 1/1 -ə ( bütöv zona) kimi götürülmüşdür. Alınan nəticələr texniki göstəricilərlə müqayisə edilir.

# MCNP: ОБЩИЙ N-ЧАСТИЧНЫЙ МОНТЕ КАРЛОВСКИЙ КОД ТРАНСПОРТИРОВКИ

## ДАРЮШ МАСТИ

## РЕЗЮМЕ

В ядерных реакторах одним из сильнейших вычислительных методов является Код MCNP. Этот код развит на базе метода Монте-Карло. Несмотря на некоторые недостатки , такие как большое затраченное время, недостаточный объем памяти для всестороннего изучения проблемы, этот код широко используется для вычисления параметров нейтронов в активной зоне реактора. В данной работе Код MCNP используется в реакторе мощностью 1000МВт.для вычисления и определения физических параметров нейтронов: таких как реактивность, интенсивность потока нейтронов, $K_{eff}$, и т.д. Для определения ценности проводимых контрольных измерений модель «активная зона» была развернута от 1/12 части активный зоны реактора до 1/6 и 1/1 (то есть целой зоны). Результаты сравниваются с техническими данными.