

# Research on Primary Shielding Calculation Source Generation Codes

Zheng Zheng<sup>1,\*</sup>, Qiliang Mei<sup>1</sup>, Hui Li<sup>1</sup>, Danhua Shangguan<sup>2</sup>, Guangchun Zhang<sup>2</sup>

<sup>1</sup>Shanghai Nuclear Engineering Research and Design Institute Hongcao Road No. 29, Shanghai, China

<sup>2</sup>Institute of Applied Physics and Computational Mathematics Fenghao East Road No. 2, Beijing, China

**Abstract.** Primary Shielding Calculation (PSC) plays an important role in reactor shielding design and analysis. In order to facilitate PSC, a source generation code is developed to generate cumulative distribution functions (CDF) for the source particle sample code of the J Monte Carlo Transport (JMCT) code, and a source particle sample code is developed to sample source particle directions, types, coordinates, energy and weights from the CDFs. A source generation code is developed to transform three dimensional (3D) power distributions in xyz geometry to source distributions in  $r\theta z$  geometry for the J Discrete Ordinate Transport (JSNT) code. Validation on PSC model of Qinshan No.1 nuclear power plant (NPP), CAP1400 and CAP1700 reactors are performed. Numerical results show that the theoretical model and the codes are both correct.

## 1 Introduction

Primary shielding is composed of in-core internals surrounding the core (including baffle and barrel), water (including water between baffle and barrel, and water between barrel and Reactor Pressure Vessel (RPV)), and RPV and concrete surrounding RPV. PSC plays an important role in evaluating the radiation dose of personnel, insuring the safety of RPV and reactor internals during the whole lifetime, and preventing the concrete shielding as well as the outer components and structures from an excess of activation. PSC mainly consists of the following steps: (1) determine the core neutron and photon sources, (2) perform neutron and photon transport calculations, (3) derive the dose rate, heating rate and displacements per atom (DPA) etc. by multiplying response functions with fluence rates.

Both the Monte Carlo method and the Discrete Ordinate method are usually adopted to perform neutron and photon transport calculations. JMCT<sup>[1]</sup> is a 3D neutron-photon transport Monte Carlo code developed by Institute of Applied Physics and Computational Mathematics (IAPCM). JMCT provides a source particle sample code for user to define space, energy, direction and weight for source particles. JSNT<sup>[2]</sup> is a 3D neutron-photon transport Discrete Ordinate code developed by IAPCM.

In order to facilitate PSC, we develop a source generation code and a source sample code for JMCT, as well as a source generation code for JSNT. Both the CDFs of source particles for JMCT and 3D source distributions of each mesh for JSNT are calculated using 3D power distribution, fission fraction, fission spectrum, number of particles and energy released per fission etc. We apply the source codes on PSC model of Qinshan

No.1 NPP, CAP1400 and CAP1700 reactors, with the aim to validate the source codes, and analyse the effects of the variations of fuel assembly burnup and enrichment on the fluence rates.

## 2 Theoretical Model

### 2.1 JMCT source generation and sample methods

Firstly, the probability density functions (PDF) and CDF of source particles are calculated using 3D power distribution, fission fraction, fission spectrum, number of particles and energy released per fission etc. Secondly, source particle directions, types, coordinates, energy and weights are sampled from the CDFs.

#### 2.1.1 Cumulative distribution functions calculation

The energy PDF of the  $m^{\text{th}}$  assembly is:

$$\chi'(m, g', p) = \frac{\sum_{n=1}^{N'} f'(m, n) \cdot v(n, p) \cdot \chi(n, g', p) \cdot T(g', p)}{\sum_{n=1}^{N'} \left\{ f'(m, n) \cdot v(n, p) \cdot \sum_{g=1}^{G(p)} \chi(n, g, p) \cdot T(g, p) \right\}} \quad (1)$$

Where  $g, g'$  is energy index,  $m$  is assembly index,  $n$  is the fission nuclide index,  $p$  is the particle type index,  $p=1$  is neutron,  $p=2$  is photon,  $N'$  is the number of nuclides,  $\chi'(m, g', p)$  is the PDF of particle  $p$ , group  $g'$  and assembly  $m$ ,  $f'(m, n)$  is the fission fraction of nuclide  $n$  and assembly  $m$ .  $v(n, p)$  is number of

\* Corresponding author: [zhengzheng@snerdi.com.cn](mailto:zhengzheng@snerdi.com.cn)

particles released per fission for particle  $p$  and nuclide  $n$ , particles/s,  $\chi(n, g, p)$  is the probability of particle  $p$ , group  $g$  and nuclide  $n$ ,  $G(p)$  is the number of groups for particle  $p$ ,  $T(g, p)$  is the energy bias factor for particle  $p$  and group  $g$ .

The energy CDF of the  $m^{\text{th}}$  assembly is:

$$E(m, g'', p) = \sum_{g'=1}^{G(p)} \frac{\chi'(m, g', p)}{\sum_{g=1}^{G(p)} \chi'(m, g, p)}, g'' = 1, \dots, G(p) \quad (2)$$

Where  $E(m, g'', p)$  is the CDF of particle  $p$ , group  $g''$  and assembly  $m$ .

The space PDF of the  $q^{\text{th}}$  mesh is:

$$S(i, j, k, p) = P(i, j, k) \cdot \frac{C' \cdot B'(m)}{K'(m)} \cdot \sum_{n=1}^{N'} \left\{ f'(m, n) \cdot v(n, p) \cdot \sum_{g=1}^{G(p)} [\chi(n, g, p) \cdot T(g, p)] \right\} \quad (3)$$

Where  $S(i, j, k, p)$  is the source of particle  $p$  and mesh  $(i, j, k)$ , particles/(cm<sup>3</sup>·s),  $P(i, j, k)$  is the power of mesh  $(i, j, k)$ , W/cm<sup>3</sup>,  $C'$  is unit conversion factor, 6.241508×10<sup>12</sup>MeV/(s·W),  $B'(m)$  is the space bias factor of assembly  $m$ ,  $K'(m)$  is the energy released per fission of assembly  $m$ , MeV.

$f'(m, n)$  and  $K'(m)$  can be calculated by interpolation from library by assembly burnup and enrichment. The library includes fission fraction and energy released per fission of 6 fission nuclides (U-235, U-238, Pu-239, Pu-240, Pu-241 and Pu-242) with 25 burnups and 5 enrichments.

The space CDF of the  $q^{\text{th}}$  mesh is:

$$D(q, p) = \sum_{q'=1}^Q \frac{S'(q', p) \cdot V'(q')}{\sum_{q''=1}^Q S'(q'', p) \cdot V'(q'')}, q = 1, \dots, Q \quad (4)$$

Where  $D(q, p)$  is the CDF of particle  $p$  and mesh  $q$ ,  $S'(q', p)$  is the source of particle  $p$  and mesh  $q'$ , particles/(cm<sup>3</sup>·s),  $V'(q')$  is the volume of mesh  $q'$ , cm<sup>3</sup>,  $Q$  is the number of mesh.

The number of the  $p^{\text{th}}$  particle is:

$$N(p) = \sum_{i,j,k} S(i, j, k, p) \cdot V'(i, j, k) \cdot R(p) \quad (5)$$

Where  $N(p)$  is the number of particle  $p$ ,  $V'(i, j, k)$  is the volume of mesh  $(i, j, k)$ , cm<sup>3</sup>,  $R(p)$  is the particle type bias factor of particle  $p$ .

The particle type CDF of the  $p^{\text{th}}$  particle is:

$$F(p) = \sum_{p'=1}^P \frac{N(p')}{\sum_{p''=1}^P N(p'')}, p = 1, \dots, P \quad (6)$$

Where  $F(p)$  is the particle type CDF of particle  $p$ ,  $P$  is the number of particle types.

Directions of neutrons and photons released in fission are isotropic, which can be determined using isotropic sample subroutine.

Before the biased PDFs and CDFs are calculated, the bias factors should be normalized as follows:

$$T(g, p) = \frac{T'(g, p)}{\sum_{g'=1}^{G(p)} T'(g', p)} \cdot G(p), \sum_{g=1}^{G(p)} T(g, p) = G(p) \quad (7)$$

$$B'(m) = \frac{B''(m)}{\sum_{m=1}^M B''(m)} \cdot M, \sum_{m=1}^M B'(m) = M \quad (8)$$

$$R(p) = \frac{R'(p)}{\sum_{p=1}^P R'(p)} \cdot P, \sum_{p=1}^P R(p) = P \quad (9)$$

Where  $T'(g, p)$  is the un-normalized energy bias factor for particle  $p$  and group  $g$ ,  $B''(m)$  is the un-normalized space bias factor of assembly  $m$ ,  $R'(p)$  is the un-normalized particle type bias factor of particle  $p$ ,  $M$  is the number of assembly.

### 2.1.2 Source particle sample method

Considering the dependent relationship of variables, the source particle sample sequence is the reverse of the CDF calculation sequence, which is: direction, particle type, space and energy.

Directions are determined by isotropic sample.

For particle type, select random number  $\xi_p \in (0, 1]$ , determine index  $p$  which satisfies  $F(p-1) < \xi_p \leq F(p)$ .

For space, select random number  $\xi_q \in (0, 1]$ , determine index  $q$  which satisfies  $D(q-1, p) < \xi_q \leq D(q, p)$ . Select random number  $\xi_{q1}, \xi_{q2}, \xi_{q3} \in (0, 1]$ , calculate particle coordinates as follows:

$$\begin{cases} x = x_q^1 + \xi_{q1} \cdot (x_q^2 - x_q^1) \\ y = y_q^1 + \xi_{q2} \cdot (y_q^2 - y_q^1) \\ z = z_q^1 + \xi_{q3} \cdot (z_q^2 - z_q^1) \end{cases} \quad (10)$$

Where  $x, y, z$  are particle coordinates, cm,  $x_q^1, x_q^2$  are lower and upper boundary of mesh  $q$  along axis  $x$ , cm,  $y_q^1, y_q^2$  are lower and upper boundary of mesh  $q$  along axis  $y$ , cm,  $z_q^1, z_q^2$  are lower and upper boundary of mesh  $q$  along axis  $z$ , cm.

For energy, select random number  $\xi_g \in (0, 1]$ , determine index  $g$  which satisfies  $E(g-1, q, p) < \xi_g \leq E(g, q, p)$ . Select random number  $\xi_{g1} \in (0, 1]$ , calculate particle energy as follows:

$$E' = E_g^1 + \xi_{g1} \cdot (E_g^2 - E_g^1) \quad (11)$$

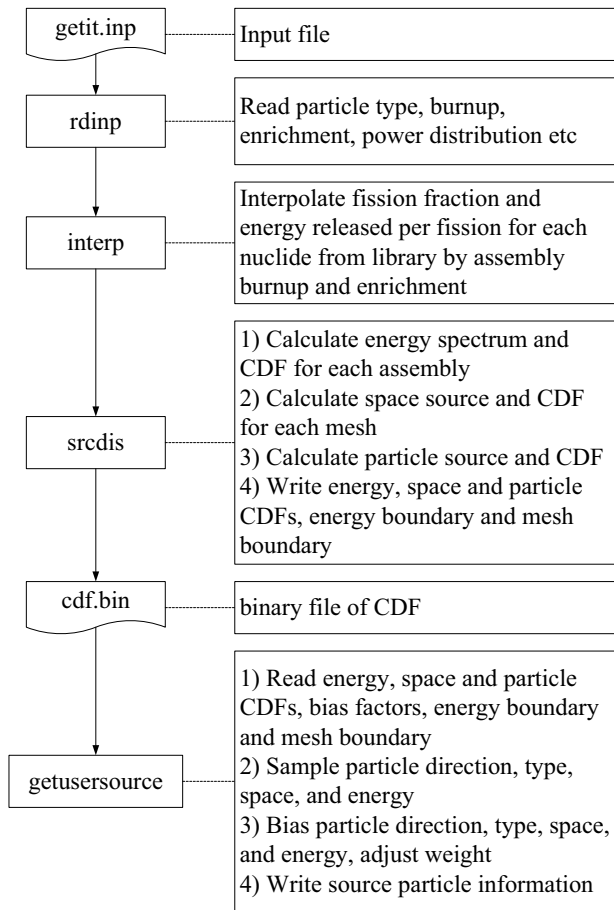
Where  $E'$  is particle energy, MeV,  $E_g^1, E_g^2$  are lower and upper boundary of group  $g$ , MeV.

When the bias factors are used, in order to get unbiased results, source particle weight should be adjusted as follows:

$$w' = \frac{w}{R(p) \cdot B'(m) \cdot T(g, p)} \cdot \frac{N^\dagger}{N^n} \quad (12)$$

Where  $w'$  is biased weight,  $w$  is unbiased weight,  $N^\dagger$  is the number of biased particles,  $N^n$  is the number of unbiased particles.

Based on the CDF calculation method and source particle sample method above, a source generation code and a source sample code for JMCT are developed. Fig. 1 shows the flowchart of the source codes.



**Fig. 1.** Flowchart of CDF calculation and source particle sample for JMCT

## 2.2 JSNT source generation method

Firstly, the pin mesh source is calculated using 3D power distribution, fission fraction, fission spectrum, number of particles and energy released per fission. Secondly, the geometry mesh source is calculated using the pin mesh source by the area or volume weighted method.

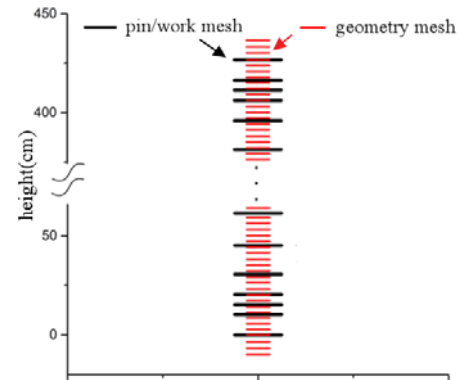
The pin mesh source is calculated as follows:

$$S''(i, j, k, g) = P(i, j, k) \cdot C^\dagger(m, g) \\ = P(i, j, k) \cdot \frac{C' \cdot B'(m)}{K'(m)} \cdot \sum_{n=1}^{N'} [f'(m, n) \cdot v(n) \cdot \chi''(n, g)] \quad (13)$$

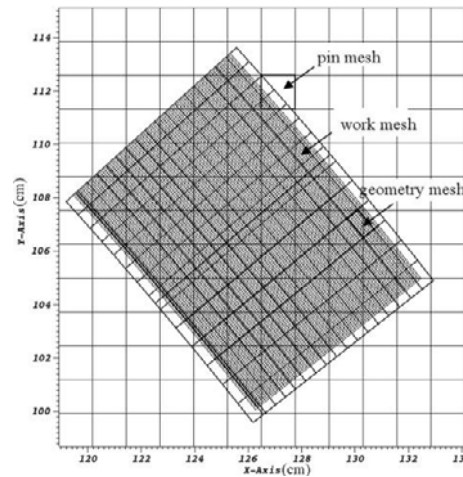
Where  $S''(i, j, k, g)$  is the source of group  $g$  and pin mesh  $(i, j, k)$ , particles/(cm<sup>3</sup>·s),  $C^\dagger(m, g)$  is power to

source transform factor, particles/s/MW,  $\chi''(n, g)$  is the probability of group  $g$  and nuclide  $n$ .

The geometry mesh source is calculated as follows:

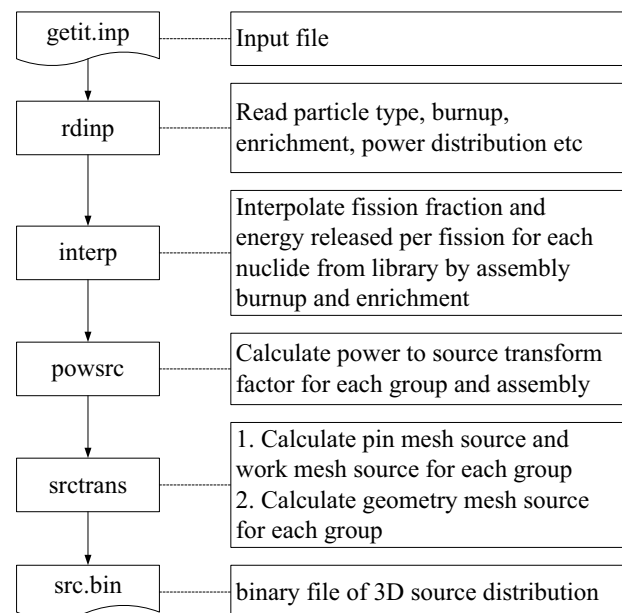


(a) axial mesh



(b) radial mesh

**Fig. 2.** Geometry transformation sketch map



**Fig. 3.** Flowchart of the source generation code for JSNT

1) Define a group of  $r\theta z$  work mesh, for each work mesh, set the source of the work mesh equals to the source of pin mesh if the center of the work mesh lies in the pin mesh.

2) As Fig. 2 and Formula (14) show, calculate the source of each geometry mesh by the area or volume weighted method.

$$S^v(q', g) = \sum_{q \in q'} S^v(q, g) \cdot V'(q) / \sum_{q \in q'} V'(q) \quad (14)$$

Where  $S^v(q', g)$  is the source of group  $g$  and geometry mesh  $q'$ , particles/(cm<sup>3</sup>·s),  $S^v(q, g)$  is the source of group  $g$  and work mesh  $q$ , particles/(cm<sup>3</sup>·s).

Fig. 3 shows the flowchart of the source generation code for JSNT.

### 3 Numerical Results

We perform validation on PSC model of Qinshan No.1 Reactor, and apply these codes on CAP1400 and CAP1700 reactors.

#### 3.1 Qinshan No.1 reactor surveillance capsule Fast Neutron Fluence Rate (FNFR) calculation

Qinshan No.1 Reactor is a 966MW (thermal power) pressurized light-water reactor (PWR) designed by Shanghai Nuclear Engineering Research and Design Institute (SNERDI) of China and placed in operation in December of 1991. It is owned by China National Nuclear Corporation Company (CNNCC) and operated by Qinshan Nuclear Power Limited Company (QNPLC). The core of the Qinshan No.1 Reactor consists of 121 assemblies and is surrounded by the core baffle, barrel, pressure vessel, and biological shield. According to the requirement of RPV material radiation surveillance program, Qinshan NPP shutdowns the reactor on June 6 of 2014 (end of the 15<sup>th</sup> fuel cycle), and removes the 6<sup>th</sup> radiation surveillance capsule (8 in total). The FNFR of this capsule is important for the RPV radiation evaluation. China Institute of Atomic Energy (CIAE) extracts samples from dose box of surveillance capsule, measures the activation, and calculates the neutron fluence rate for energy greater than 0.1MeV, 0.5MeV and 1.0MeV by using a neutron spectrum calculation code. SNERDI calculates the FNFR by using JMCT and JSNT codes with newly developed source codes.

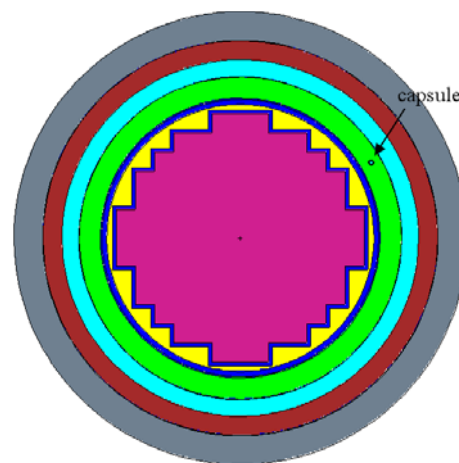
The geometry models of Qinshan are shown schematically in Fig. 4, which also show the locations of the capsule. Both radial and axial core power distributions are processed as follows:

$$\bar{P}(N) = \sum_{n=1}^N [P(n) \cdot T(n)] / \sum_{n=1}^N T(n) \quad (15)$$

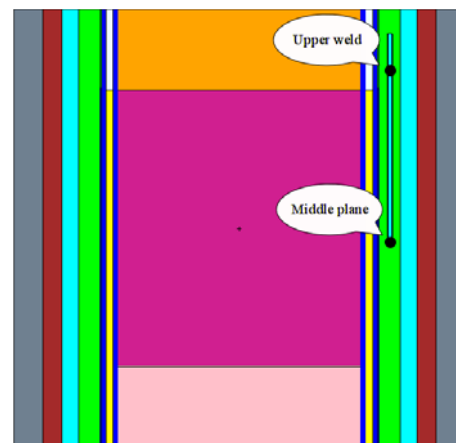
Where  $\bar{P}(N)$  is the average relative power from cycle 1 to cycle  $N$ ,  $P(n)$  is relative power for cycle  $n$ ,  $T(n)$  is the effective full power day (EFPD) for cycle  $n$ , days.

In JMCT calculation, 2E+9 neutrons are simulated. Geometry splitting with Russian Roulette and space bias

factors are used to improve the efficiency. Point detectors are used to tally the FNFRs. In JSNT calculation, the number of meshes is about 200 million for 1/8  $r\theta z$  geometry, multi-group library is BUGLE-96<sup>[3]</sup> library with 47 group neutron and 20 group photon, the order of the Legendre expansion of the scattering cross-section is 3, the order of the fully symmetric angular quadrature set is  $S_8$ , and the point-wise flux convergence criterion is 0.001. The source energy spectrum is taken as U-235 fission spectrum.



(a) xy model



(b) xz model

Fig. 4. Geometry model of Qinshan reactor

Table 1 gives FNFRs of Qinshan reactor for nominal power. As Table 1 shows, the relative errors are within 10% and 30% for the middle plane and upper weld, respectively. The measurement uncertainty is about 15%, which comes from target nuclides weights of detector samples, nuclear cross-sections, and core power etc. The calculation uncertainties are about 10% and 20% for the middle plane and upper weld, respectively, which come from power distribution, geometry and material etc. Considering the uncertainties above, the results of JMCT and JSNT are comparable with measurements.

Table 1. FNFRs of Qinshan reactor

Position	Energy (MeV)	Codes	FNFR (n·cm <sup>-2</sup> ·s <sup>-1</sup> )	Relative error
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				(%)
Middle plane	$\geq 0.1$	Measure	1.404E+11	-
		JMCT	1.449E+11	3.24
		JSNT	1.545E+11	10.05
	$\geq 0.5$	Measure	9.725E+10	-
		JMCT	1.007E+11	3.56
		JSNT	1.068E+11	9.85
	$\geq 1.0$	Measure	6.552E+10	-
		JMCT	6.696E+10	2.21
		JSNT	7.179E+10	9.58
Upper weld	$\geq 0.1$	Measure	1.594E+10	-
		JMCT	1.133E+10	-28.90
		JSNT	1.216E+10	-23.68
	$\geq 0.5$	Measure	1.103E+10	-
		JMCT	8.040E+09	-27.08
		JSNT	8.722E+09	-20.89
	$\geq 1.0$	Measure	7.415E+09	-
		JMCT	5.662E+09	-23.63
		JSNT	6.098E+09	-17.75

### 3.2 CAP1400 reactor RPV neutron and photon fluence rate calculation

CAP1400 is an advanced passive generation III PWR designed by SNERDI of State Power Technology Corporation (SNPTC) of China, which is belonged to State Power Investment Corporation (SPIC). The thermal power of CAP1400 is 4040MW with 193 assemblies and a low-leakage refueling scheme. Different from previous reactors, the core shroud of CAP1400 is a highly heterogeneous structure composed of baffle and 6 half ring segments. Since it is subject to high neutron and photon fluence rates, both thermal stresses and iron DPA become determining parameters for its sizing. The present work aims to calculate neutron and photon fluence rates from barrel to RPV, which can be used to derive heat generation rate and DPA for the core shroud.

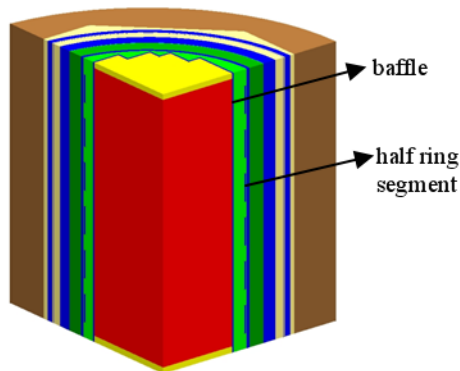


Fig. 5. Geometry model of CAP1400

Fig. 5 shows the geometry model of CAP1400. The cycle-average power distribution of equilibrium cycle (Cycle 6) is used to determine the core source. Fig. 6 and

Fig. 7 are the radial and 3D neutron and photon source distributions of JMCT and JSNT, respectively. In JMCT calculation, number of particles (NPS) is  $5E+9$ , superimposed mesh tallies are used to give fluence rate distributions. In JSNT calculation, almost the same parameters are used as in the previous JSNT calculation for Qinshan No.1 reactor surveillance capsule FNFR calculation. The source energy spectrum of each assembly is a mixed fission spectrum determined by the fission fractions of U and Pu.

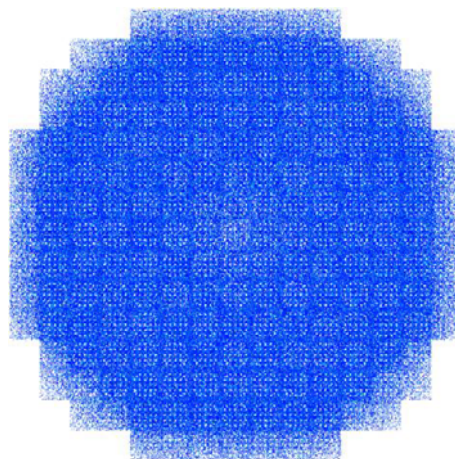


Fig. 6. Radial neutron and photon source of JMCT

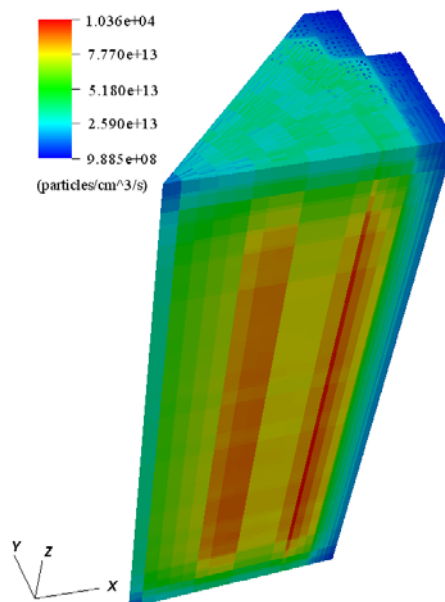
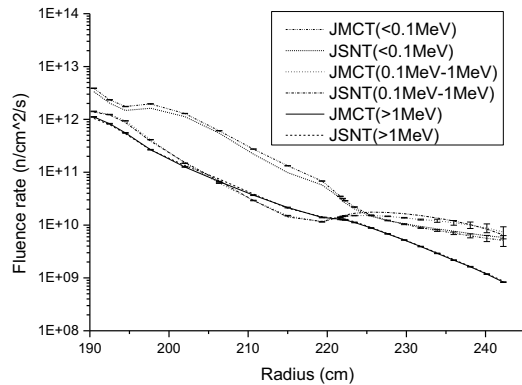


Fig. 7. 3D neutron and photon source of JSNT

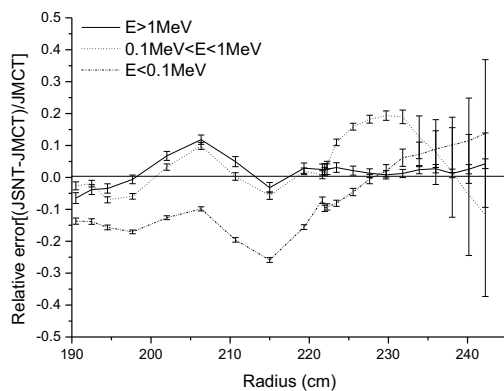
Fig. 8 gives the radial neutron and photon fluence rates and error distributions from barrel to RPV. As Fig. 8 shows, most of the relative errors of neutron fluence rates of JMCT and JSNT are within 20%, which satisfy the requirements for different radiation shielding calculation methods of U.S. Nuclear Regulatory Commission (NRC) RG 1.190<sup>[4]</sup>. While most of the relative errors of photon fluence rates with photon energy greater than 1MeV and between 0.1MeV and 1.0MeV of JMCT and JSNT are within 20%, the relative errors with photon energy less than 0.1MeV are about



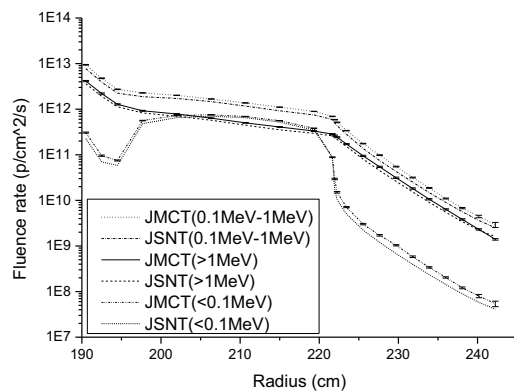
40%. This is probably attributed to the difference from multigroup and continuous libraries. We will investigate this phenomenon in the future.



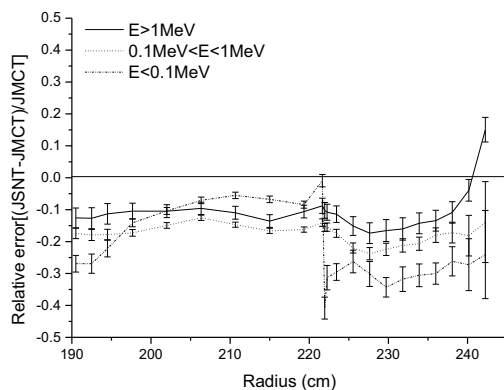
(a) neutron fluence rate



(b) neutron fluence rate error



(c) photon fluence rate



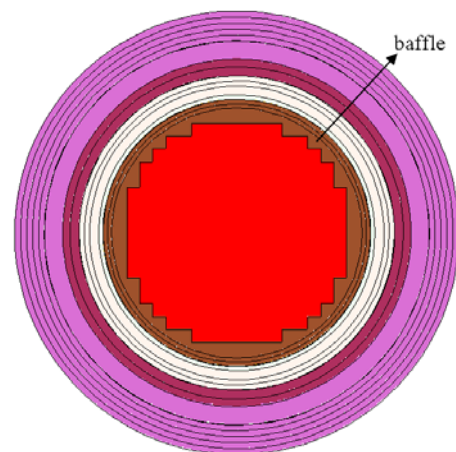
(d) photon fluence rate error

Fig. 8. Neutron and photon fluence rates and error

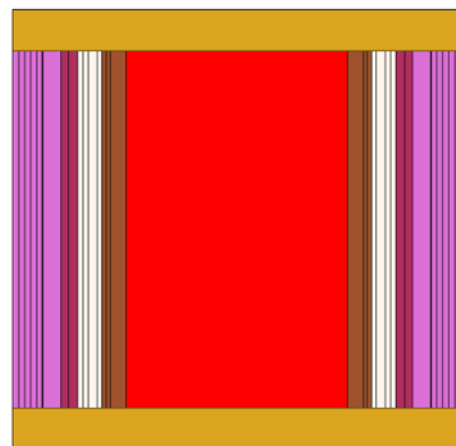
### 3.3 CAP1700 reactor RPV FNFR calculation

CAP1700 is a conceptual innovation advanced passive generation III PWR designed by SNERDI, which is aimed to improve the safety and economics of PWRs. Compared with CAP1400, CAP1700 has a new-style baffle with width increasing from 3cm to greater than 10cm, leading to a lower FNFR on RPV. The objective of this work is to analyse the effects of the fuel assembly burnup and enrichment variations on the FNFR of RPV.

Fig. 9 shows the geometry model of CAP1700. Table 2 gives the maximum FNFR (greater than 1.0MeV) on the inner surface of RPV. As Table 2 shows, the FNFR using U and Pu mixed spectrum is greater than that using U spectrum by 10% to 15%. Compared with U, Pu releases more neutrons and energy for each fission, and the fission spectrum is harder. These factors result in the increase of the FNFR using U and Pu mixed spectrum. Since Pu builds up with the increase of burnup, this effect should be considered in core source determination.



(a) xy model



(b) xz model

Fig. 9. Geometry model of CAP1700

**Table 2.** Maximum FNFR of CAP1700

Codes	Fission spectrum	FNFR ( $\text{n}\cdot\text{cm}^{-2}\cdot\text{s}^{-1}$ )	Relative error (%)
JMCT	U-235	1.07E+08	-
JMCT	U and Pu	1.20E+08	11.68
JSNT	U-235	1.09E+08	1.78
JSNT	U and Pu	1.23E+08	15.33

## 4 Conclusions

As core source is the precondition for reactor shield calculation, the objective of this work is to investigate source generation methods and develop source codes for a Monte Carlo code JMCT and a Discrete Ordinate code JSNT. The source codes consider burnup and enrichment of fuel assembly, and supply particle type, energy and space bias factors for users to improve calculation efficiency. The performed PSCs shows that the source codes developed are correct. The source codes developed can be used to support other Monte Carlo codes and the Discrete Ordinate codes without large modification. We will extend the sources codes for other reactor types in the future.

## References

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2. JSNT User's Manual, Institute of Applied Physics and Computational Mathematics, 2015
3. BUGLE-96, Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications, DLC-185, Oak Ridge National Laboratory, 1999
4. Regulatory Guide 1.190 , Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence, Nuclear Regulatory Commission, 2001