The Monte Carlo Method: Versatility Unbounded In A Dynamic Computing World
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DEVELOPMENT OF RADIOACTIVITY ESTIMATION CODE SYSTEM AFTER FINAL REACTOR SHUT DOWN OF FBR

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ABSTRACT

For the technical discussion of the FBR decommissioning the accurate residual radioactivity estimation around the resonance and fast neutron energy region after the final reactor shut down is very important because the energy peak in the neutron spectrum of the FBR shifts to the higher energy region compared to that in the neutron spectrum of the LWR. Especially the accurate estimation of the radioactivity of the coolant(Na) or the iron of the structural material with the resonance or threshold cross sections is essential. We have performed the comparison of residual radioactivity after the final shut down of FBR under the preparation of the decommissioning between the Sn code (TORT) and Monte Carlo code (MCNP) calculations. We used the weight window cards in MCNP code. We made the weight window parameter by the flux of the TORT. We calculated two spectra cases by MCNP. There were above 10 MeV and under 10 MeV. Because the $^{23}Na(n,2n)^{22}Na$ reaction happens above 10 MeV. It has been clarified that the overall radioactivity of the MCNP is higher than that of the TORT except the radioactivity in the Na. ^{22}Na in coolant(Na) has created same degree both the TORT and the MCNP. It was not remarkable difference by streaming effect between the MCNP and the TORT with biasing angular quadrature cosines.

Key Words: Radioactivity Estimation Code System, Final Reactor Shut Down, FBR, COSMARD, MCNP4C

1 INTRODUCTION

In nuclear facility decommissioning, it is very important to estimate an amount of radioactivity after final reactor shut down. For the FBR decommissioning the accurate residual radioactivity estimation, we are improving a radioactivity estimation code system in the COSMARD system that was development in decommissioning of JPDR in JAERI[1]. We have built in Monte Carlo code (MCNP4C) to the COSMARD system[2]. We discuss how to build in. Furthermore, We discuss a sample calculation of a FBR facility.

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2 CONSTRUCTION OF SYSTEM

2.1 Function of system

This system is worked on Linux system. We show a flow chart of system in Figure 1. This system has nuclear data processing, neutron transport calculation, radioactivity calculation and photon dose rate calculation. Further, It has the GUI interface to an each code and a necessary data of database format. Then it cans efficient calculation of each code.

2.2 Build in MCNP4C

We discuss how to build in the MCNP4C for neutron transport calculation.

The MCNP4C cans calculation of a complicated geometry that is shaped by many surfaces. The MCNP4C is continuously energy calculation. So then it cans accurately estimate at resonance region. Accordingly it is very useful that the MCNP4C is build in and is used a calculation of neutron transport. But an accuracy of Monte Carlo method calculation is proportional to a reciprocal number of a square root of arrived number of particles. Then it is very difficult that does calculation of a nuclear facility where neutron flux is varied over ten figures. Because of it is used long calculation's time to converge of result.

We used next methods for reduced a calculation time.

1. First we calculated a roughly neutron flux by the Sn code (the DOT3.5 or the TORT)[3]. We made a weight window mesh parameter for splitting a particle. Using a weight window mesh parameter, the MCNP4C calculation's time became shorter to can estimate a nuclear facility.
2. It is complicated in the MCNP4C to give a space distribution of neutron source like the method of the TORT. Then we modified the SRCDX routine in the MCN4C for set to the neutron source like the method of the TORT.

Figure 2 shows a flow chart of the above calculation method.

3 SAMPLE OF CALCULATION

3.1 Characteristic of FBR

A neutron spectrum of the FBR is shift to height energy than a neutron spectrum of the LWR. According it is very important for radioactivity estimation that accurately calculate height energy and resonance region. Especially the accurate estimation of the radioactivity of the coolant(Na) or the iron of the structural material with the resonance or threshold cross sections is essential.

3.2 Condition of calculation and process of calculation

The Geometry was 2 dimensional cylindrical model both the MCNP4C and the TORT. The geometry, the composition of material and the distribution of neutron source ware as same as both the MCNP4C and the TORT. According we used a weight window mesh parameter in the

MCNP4C that was made by the flux distribution of the TORT calculation's result. There were above 10MeV and under 10MeV in the MCNP4C. Because the $^{23}Na(n,2n)^{22}Na$ reaction that is very important in the FBR happens above 10 MeV. We used 48 energy groups in the TORT. We used the ENDF/B-VI and the JENDL3.3 for the continuous cross section library in the MCNP4C. Furthermore, We selected 19 regions for the radioactivity calculation by the DCHAIN-SP2001 [4]. There were selected by the material kinds and the neutron flux distribution of 2 dimensional cylindrical model both the MCNP4C and the TORT.

3.3 Result of neutron flux distribution

Figure 3 shows a contour of total neutron flux distribution. The Results of the MCNP4C and the TORT were almost same. The factor of difference between the MCNP4C and the TORT was form 0.5 to 2 times. But it had 5 times at the resonance region of iron. The result of the MCNP4C was larger than the result of the TORT.

Further, It was not remarkable difference by streaming effect between the MCNP4C ant the TORT with biasing angular quadrature cosines.

3.4 Result of radioactivity

The factor of difference between the MCNP4C and the TORT was form 0.3 to 5 times. It has been clarified that the overall radioactivity of the MCNP4C is higher than one of the TORT except the radioactivity in the Na. ^{22}Na in coolant(Na) has created same degree both the TORT and the MCNP4C. The radioactivity of using the JENDL3.3 is about 0.8 times lower than one of the ENDF/B-VI. Table 1 shows the radioactivity in carbon steel both the MCNP4C and the TORT.

4 CONCLUSIONS

We have constructed a method of continuous calculation to the radioactivity in a FBR. So it becomes efficaciously to use the MCNP4C. Further, we used this system to a calculation of a FBR. Then we made clear the fixed quantity effect of the MCNP4C and the TORT. Furthermore we made clear the differential of the ENDF/B-VI and the JENDL3.3.

Further we will extend this method to 3 dimensional of geometry.

5 ACKNOWLEDGMENTS

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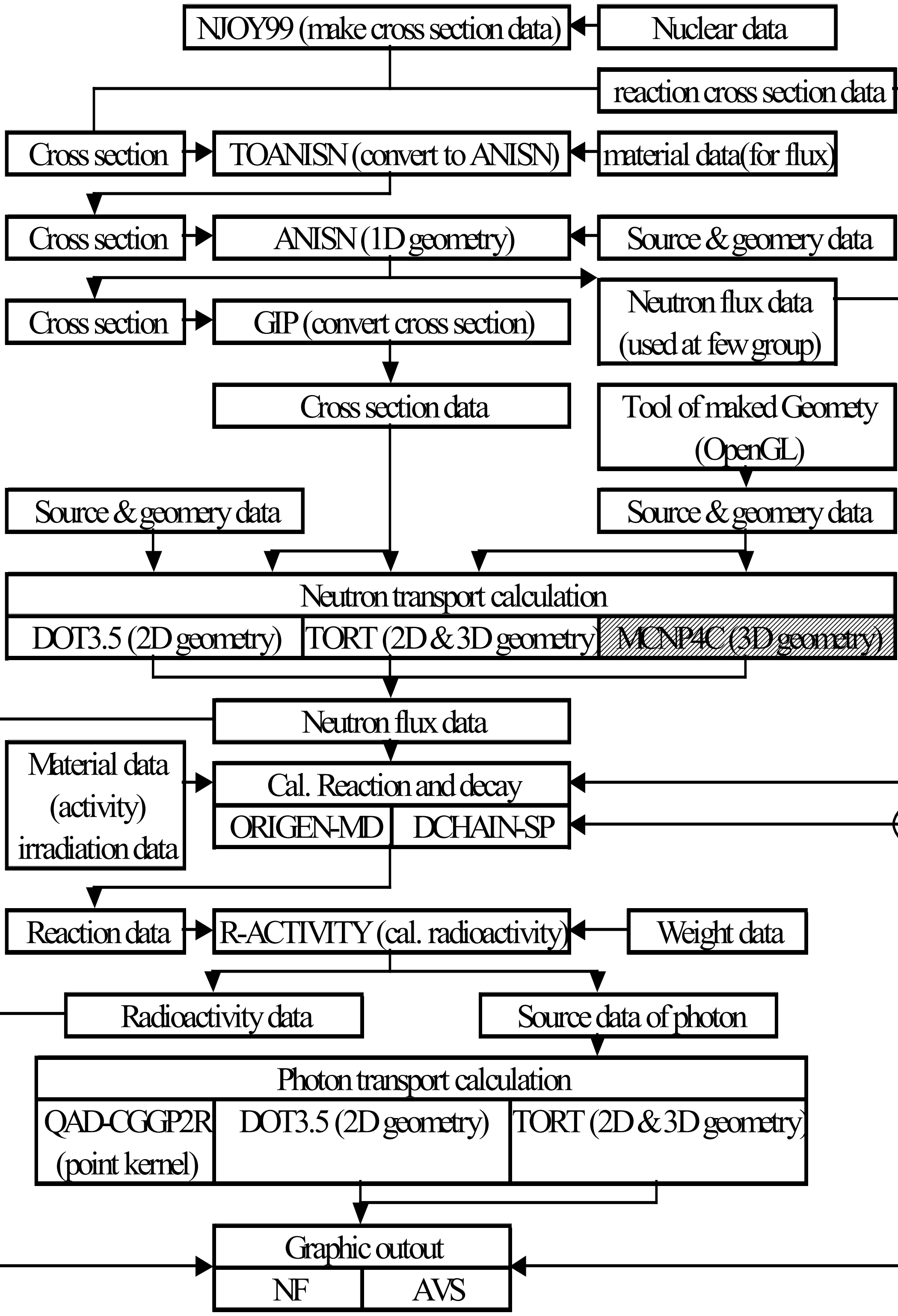

Figure 1. Flowchart of calculation

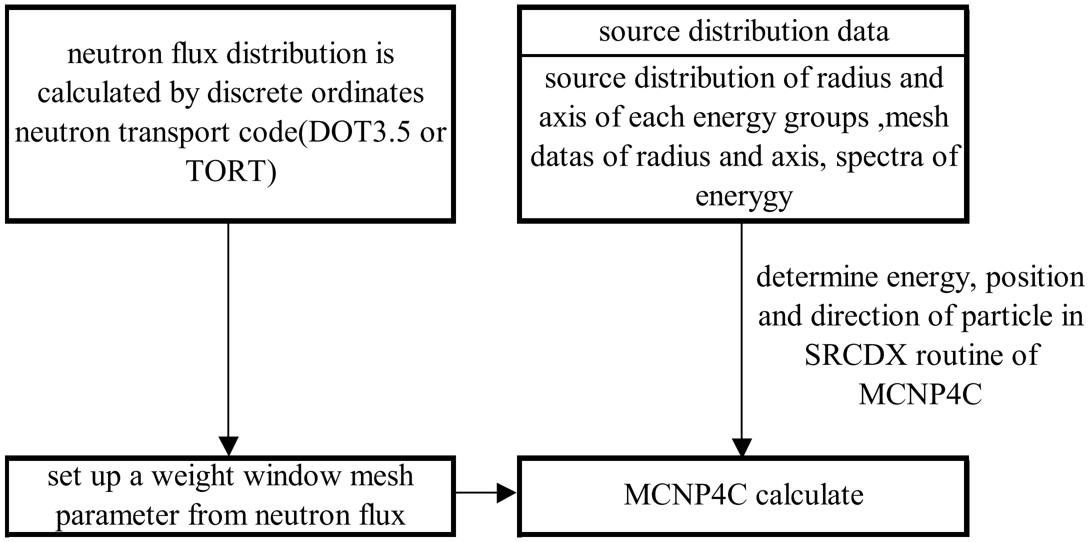

Figure 2. Flow chart of the MCNP4C calculation used weight window mesh parameter.

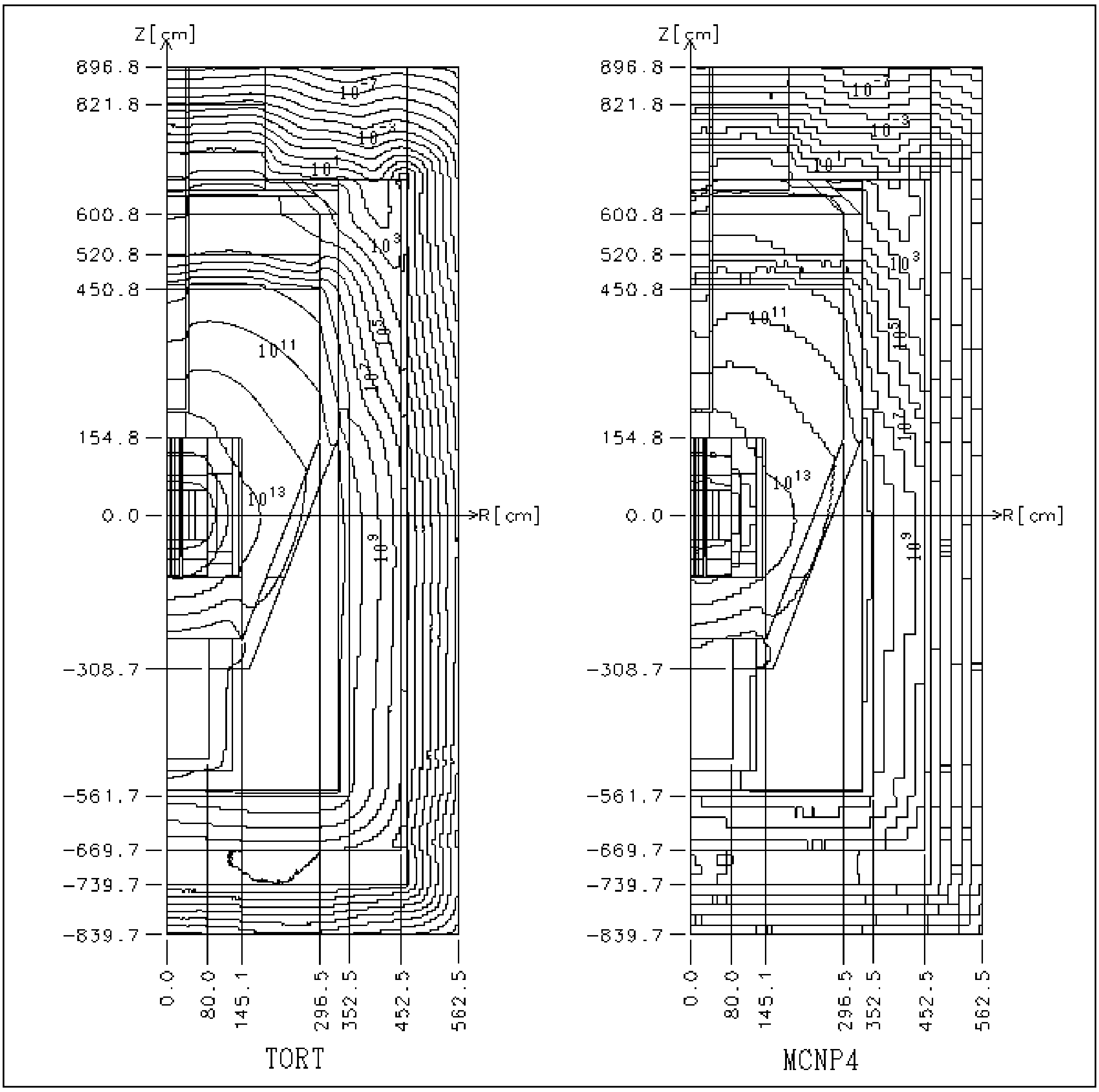

Figure 3. Contour of total neutron flux distribution

unit[n/cm^2/sec]

Table I. Radioactivity of carbon steel

Time after shut down	5 years	10 years	20 years	50 years
Result of MCNP4C [unit:Bq/g]				
^{3}H	3.5 10^{2}	2.6 10^{2}	1.5 10^{2}	2.8 10^{1}
^{56}Fe	1.4 10^{5}	4.0 10^{4}	3.2 10^{3}	1.6 10^{0}
^{60}Co	5.2 10^{4}	2.7 10^{4}	7.2 10^{3}	1.4 10^{2}
^{63}Ni	1.7 10^{3}	1.7 10^{3}	1.5 10^{3}	1.3 10^{3}
^{93}Mo	1.4 10^{1}	1.4 10^{1}	1.4 10^{1}	1.4 10^{1}
^{133}Ba	2.4 10^{1}	1.7 10^{1}	9.0 10^{0}	1.2 10^{0}
^{134}Cs	3.7 10^{1}	6.8 10^{0}	2.4 10^{-1}	1.0 10^{-5}
^{152}Eu	7.5 10^{1}	5.8 10^{1}	3.5 10^{1}	7.5 10^{0}
All of nuclides	2.0 10^{5}	6.9 10^{4}	1.2 10^{4}	1.5 10^{3}
Factor of MCNP4C/TORT				
^{3}H	2.39	2.39	2.39	2.39
^{56}Fe	2.38	2.38	2.38	2.38
^{60}Co	2.29	2.29	2.29	2.29
^{63}Ni	2.39	2.39	2.39	2.39
^{93}Mo	2.30	2.30	2.30	2.30
^{133}Ba	2.25	2.25	2.25	2.25
^{134}Cs	2.38	2.38	2.38	2.38
^{152}Eu	2.43	2.43	2.43	2.43
All of nuclides	2.36	2.34	2.33	2.38