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Analysis of Dogleg Duct Experiments with 14 MeV Neutron Source Using TRIPOLI-4® Monte Carlo Transport Code

Mingzhun LEI, Yannick PENELIAU, Yi-Kang LEE, Yuntao SONG

Abstract—TRIPOLI-4® Monte Carlo transport code, developed by CEA, has been widely used on fission reactor physics and also can be used on fusion device neutronics. In order to verify the calculation features of TRIPOLI-4® code, a simple dogleg duct model was built to simulate the 14 MeV neutron transport based on a SINBAD fusion benchmark, called Dogleg Duct Streaming Experiment. The reaction rates in the bent duct and on the back surface of the experimental assembly for $^{93}\text{Nb}(n,2n)^{92\text{m}}\text{Nb}$, $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ and $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ neutron activation dosimeters were calculated with the TRIPOLI-4® code. To improve the calculation efficiency, variance reduction techniques of TRIPOLI-4® were also performed. The calculation results showed that the variances reduction methods of the TRIPOLI-4® code are helpful and obviously decrease the calculated time and increase the convergence efficiency. The calculation reaction rates results of 11 points inside and outside of the dogleg duct assembly were taken into account. Results from the TRIPOLI-4® simulation were compared with the experimental ones obtained from the measurements of FNS facility in Japan Atomic Energy Agency (JAEA). The benchmark results show that the TRIPOLI-4® code has a good potential to calculate and estimate neutron streaming effects in fusion device design.

Index Terms—TRIPOLI-4®, Monte Carlo, Variance reduction, Benchmark, Neutron transport.

I. INTRODUCTION

NEUTRON shielding is an essential element of fusion nuclear technology. Fusion reactors, such as ITER, CFETR and EU DEMO, usually have auxiliary systems (beam injector system, diagnostics system) ports which penetrate the shield components. It is very important to investigate the effect of neutron streaming through the ports on the radiation dose distributions. Nuclear analyses of fusion reactor are complex activities which include tritium breeding ratio, nuclear heating, radiation dose, gas production and material damage etc. Because the results of the nuclear analyses will affect the design of many fusion reactor systems, detailed and accurate nuclear analyses are required. The advanced nuclear analysis

codes and the verification of the nuclear analyses by experimental benchmarks are very important [1]. Previously, the straight duct and bent duct shielding design for ITER were investigated and had been successfully conducted with FNS (Fusion Neutronics Source) facility in JAEA (Japan Atomic Energy Agency) [2, 3]. The benchmark of nuclear calculation can be done through comparison with the Dogleg Duct experiment results.

TRIPOLI-4®, a Monte Carlo neutron transport code, has been continuously developed at CEA since the 60s. It dedicates to shielding, reactor physics and nuclear instrumentation [4]. With the growing interest in using the TRIPOLI-4® nuclear analyses code for fusion reactor design (such as ITER, CFETR), it is very important to make sure the accuracy and reliability of the released code for the D-T fusion neutronic calculations.

Analyses of the Dogleg Duct model were carried out with TRIPOLI-4® Monte Carlo code. The purpose of analyses was to estimate the reaction rate results for $^{93}\text{Nb}(n,2n)^{92\text{m}}\text{Nb}$, $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ and $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ neutron activation dosimetry. The calculation results were compared with the experiment results from measurements of FNS facility in JAEA. Variance reduction techniques for this shielding calculation are also verified and tested.

II. BENCHMARK MODEL

Firstly the benchmark Dogleg Duct structure was modeled and used for validation of shielding calculations. The Dogleg Duct Streaming Experiments environment of FNS facility in JAEA was referenced for the benchmark model. The layout of the model is shown in Fig.1. The model includes two rooms. One room is used for neutron source and the outline of the room is 5360 mm in length, 5360 mm in width and 4900 mm in height. Another room is bent Dogleg Duct assembly. The room is constructed using concrete material to shield neutron radiation and protect the people around the room. A bend duct with cross section 300mm*300mm goes through the shielding zone, which behaves like the auxiliary system penetrating the fusion reactor shield. The detailed geometrical configuration and sizes of the dogleg duct are shown in Fig.2. The first leg of the duct was horizontal and as high as the neutron source. The second leg is perpendicular to the first leg and third leg. The third leg is parallel to the first leg. The lengths of three legs were 1150 mm, 900 mm and 650 mm[2].

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Fig. 2. Detailed configuration of doubly bent dogleg duct and focus points[2].

III. TRIPOLI-4® CALCULATIONS

A. Calculation procedure

The benchmark structure for TRIPOLI-4® calculation was accurately modeled, including all the structures like experimental environment. We employed the combinatorial way to define the volumes for a simplified model. The shape data is the basic component of a geometrical definition in combinatorial form [4]. For the calculations presented here, a translation of the MCNP data file was done. The boundary representation was translated from MCNP file. For the source definition, two options were used. In the first option, the Monte Carlo transport code TRIPOLI-4® adopts point source with 14 MeV to simulate D-T neutron source in the FNS facility. In the second option, the Fortran patch to MCNP-5 code was compiled with some modifications. The neutron source characteristics were written in external files and translated in TRIPOLI-4® source file format. The option of reading external source particle files was then used. To verify the shielding performance benchmark, detailed simulation results are tracked by setting 11 focused points and compared with the experimental results. Thus, the reliability of the TRIPOLI-4® code simulation for local nuclear responses can be estimated. The nuclear data libraries IRDF-2002 for dosimetry reaction rate calculations and FENDL-2.1 for neutron transport were employed for the present benchmark of shielding calculation. The neutron spectra and reaction rates for $^{93}\text{Nb}(n,2n)^{92\text{m}}\text{Nb}$,

$^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ and $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ were normalized to a source intensity of 1 n/s, which is consistent with the measured experimental value at the dogleg duct.

B. Neutron Spectra

Neutron spectra calculated by TRIPOLI-4® code with FENDL-2.1 nuclear data are shown in Fig.3. Though only 11 positions were selected in the duct and behind streaming assembly to extract the results, it is concluded that the doubly bent duct structure can effectively shield neutron flux, especially for fast neutron. The flux for fast neutron ($>0.1\text{MeV}$) has been reduced by two to three orders of magnitude, but the flux for slow neutron ($<0.1\text{MeV}$) only have a reduction of one to two orders of magnitude.

Fig. 3. Neutron spectra calculated by TRIPOLI-4®.

Figs.4-6 present the calculated and measured neutron spectra above 1MeV at the points 3,5,7. The comparisons are only on higher energy range because the measured neutron spectrum are above 1 MeV[2]. The calculated results are in good agreement with the experimental results with the maximum error ranging from 20% to 30 %. Most of the discrepancy results occur in high-energy regions ($>10\text{MeV}$).

Fig. 4. The comparison of calculated and measured neutron spectra at point 3.

Fig. 5. The comparison of calculated and measured neutron spectra at point 5.

Fig. 6. The comparison of calculated and measured neutron spectra at point 7.

C. Normalized reaction rate

Normalized reaction rates for $^{93}\text{Nb}(n,2n)^{92\text{m}}\text{Nb}$, $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ and $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ were calculated by TRIPOLI-4®. The results are listed in TABLE I, II and III. The results of normalized reaction rate in this doubly bent dogleg duct are obviously different in each leg zone. The reaction rates at the first leg zone are larger than that of other zones like in the duct and behind the streaming assembly due to point 1, point 2 and point 3 do not have any shielding material between these points and neutron source. Compared with the first leg zone, the reaction rate at the second leg decreases by one to two orders of magnitude. The reaction rates for $^{93}\text{Nb}(n,2n)^{92\text{m}}\text{Nb}$, $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ and $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ continue declining.

TABLE I

REACTION RATES IN THE DUCT AND BEHIND THE STREAMING ASSEMBLY -I

Location	Point	Reaction rate for $^{93}\text{Nb}(n,2n)^{92\text{m}}\text{Nb}$		C/E
		Calculation	Measurement	
First leg	1	$1.13 \times 10^{-6} \pm 0.06\%$	$1.73 \times 10^{-6} \pm 2.9\%$	0.66
	2	$6.99 \times 10^{-7} \pm 0.07\%$	$1.06 \times 10^{-6} \pm 3.9\%$	0.66
	3	$4.32 \times 10^{-7} \pm 0.09\%$	$6.91 \times 10^{-7} \pm 3.7\%$	0.62
Second leg	4	$3.40 \times 10^{-9} \pm 1.2\%$	$3.65 \times 10^{-9} \pm 4.6\%$	0.93
	5	$5.99 \times 10^{-10} \pm 3.6\%$	$7.90 \times 10^{-10} \pm 5.9\%$	0.76
Third leg	6	$3.53 \times 10^{-10} \pm 3.68\%$	-	-
	7	$3.56 \times 10^{-11} \pm 11.18\%$	-	-
	8	$5.79 \times 10^{-11} \pm 8.94\%$	-	-
Behind assembly	9	$2.67 \times 10^{-10} \pm 3.79\%$	$3.54 \times 10^{-10} \pm 1.1\%$	0.75
	10	$5.36 \times 10^{-11} \pm 9.72\%$	-	-
	11	$1.98 \times 10^{-11} \pm 17.85\%$	-	-

*C/E, C= Calculated results, E=Experimental results

TABLE II
REACTION RATES IN THE DUCT AND BEHIND THE STREAMING ASSEMBLY -II

Location	Point	Reaction rate for $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$		C/E
		Calculation	Measurement	
First leg	1	$5.76 \times 10^{-7} \pm 0.07\%$	$1.01 \times 10^{-6} \pm 3.0\%$	0.57
	2	$2.56 \times 10^{-7} \pm 0.09\%$	$4.46 \times 10^{-7} \pm 3.5\%$	0.57
	3	$1.49 \times 10^{-7} \pm 0.11\%$	$2.36 \times 10^{-7} \pm 6.9\%$	0.63
Second leg	4	$2.38 \times 10^{-8} \pm 0.38\%$	$2.90 \times 10^{-8} \pm 7.0\%$	0.82
	5	$5.95 \times 10^{-9} \pm 0.86\%$	$8.58 \times 10^{-9} \pm 7.5\%$	0.69
Third leg	6	$2.68 \times 10^{-9} \pm 0.01\%$	$4.60 \times 10^{-9} \pm 7.1\%$	0.58
	7	$6.81 \times 10^{-10} \pm 2.32\%$	$6.53 \times 10^{-10} \pm 20.3\%$	1.04
Behind assembly	8	$6.98 \times 10^{-10} \pm 2.38\%$	-	-
	9	$1.28 \times 10^{-9} \pm 1.57\%$	$1.39 \times 10^{-9} \pm 9.5\%$	0.92
	10	$7.43 \times 10^{-10} \pm 2.37\%$	-	-
	11	$5.58 \times 10^{-10} \pm 3.00\%$	-	-

TABLE III
REACTION RATES IN THE DUCT AND BEHIND THE STREAMING ASSEMBLY -III

Location	Point	Reaction rate for $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$		C/E
		Calculation	Measurement	
First leg	1	$1.66 \times 10^{-4} \pm 1.13\%$	$2.27 \times 10^{-4} \pm 1.1\%$	0.73
	2	$6.96 \times 10^{-5} \pm 1.57\%$	$9.97 \times 10^{-5} \pm 2.1\%$	0.70
	3	$3.38 \times 10^{-5} \pm 2.25\%$	$5.00 \times 10^{-5} \pm 1.7\%$	0.68
Second leg	4	$2.45 \times 10^{-5} \pm 2.77\%$	$2.99 \times 10^{-5} \pm 1.6\%$	0.82
	5	$1.55 \times 10^{-5} \pm 3.44\%$	$2.11 \times 10^{-5} \pm 1.8\%$	0.74
Third leg	6	$1.21 \times 10^{-5} \pm 4.08\%$	$3.06 \times 10^{-5} \pm 1.7\%$	0.40
	7	$3.28 \times 10^{-6} \pm 7.10\%$	$4.08 \times 10^{-6} \pm 2.3\%$	0.73
Behind assembly	8	$1.02 \times 10^{-6} \pm 3.15\%$	$6.65 \times 10^{-7} \pm 5.5\%$	1.54
	9	$9.20 \times 10^{-7} \pm 3.50\%$	$6.79 \times 10^{-7} \pm 4.5\%$	1.36
	10	$1.09 \times 10^{-6} \pm 4.98\%$	$7.09 \times 10^{-7} \pm 4.9\%$	1.54
	11	$7.87 \times 10^{-7} \pm 4.51\%$	$5.57 \times 10^{-7} \pm 3.2\%$	1.41

Fig. 7. Cross section for $^{93}\text{Nb}(n,2n)^{92\text{m}}\text{Nb}$, $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ and $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ reactions from IRDF-2002 nuclear data library.

Fig.7 depicts the cross section for $^{93}\text{Nb}(n,2n)^{92\text{m}}\text{Nb}$, $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ and $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ reactions. The reaction

threshold of Nb, In and Au are 8.93MeV, 0.34 MeV and 1.0×10^{-11} MeV respectively. Based on the figure, we can deduce that the normalized reaction rate for $^{93}\text{Nb}(n,2n)^{92\text{m}}\text{Nb}$ in the benchmark model with 14MeV neutron source will decrease very fast. The measured and calculated results at second leg zone decrease by two to three orders of magnitude. The reaction rates at point 7 is almost at the same level with the points behind the streaming assembly. The threshold of In is smaller than that of Nb. So that the reaction rate for $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ at second leg zone is one to two orders smaller than the values at first leg zone. The reaction threshold of Au is very small, and there is a very particular energy range called Resolved Resonance Range in which cross sections vary a lot. Hence, the reaction rate for $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ decreases slowly in the bent duct. All the results at point 7 are almost at the same level with the points behind the streaming assembly. It is proved that the doubly dogleg duct design can effectively shield neutrons. Auxiliary systems for fusion reactor can adopt the doubly dogleg duct design to penetrate the shield structure.

The C/E (Ratio of Calculated reaction rate and Experimental measured reaction rate) values of $^{93}\text{Nb}(n,2n)^{92\text{m}}\text{Nb}$ shows that the calculated results have the similar variation trend with measured reaction rate. However, the calculated values are generally less than the experimental measured values. It is an underestimation of about 7-34%. The C/E value of $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ is in the range of 0.57 and 1.04. The calculated value is slightly smaller than the measured value except point 7 (error of measured value at this point is 20.3%). Calculated reaction rates of $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ are quite in good agreement with experimental measured values, except for point 6. That's because of the Resolved Resonance Range of the $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ reaction cross section. The cross section is badly evaluated for the benchmark example but the resonance behavior is normal.

D. Results discussions

According to above calculations, neutron flux and reaction rate of $^{93}\text{Nb}(n,2n)^{92\text{m}}\text{Nb}$, $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ and $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ at second leg zone degrade one to two orders of magnitude than that of the points at first leg zone. It is proved that the bent duct can effectively shield the neutron. The reaction rate at the end of third leg shares almost the same level with the points behind the shield. Thus, doubly bent duct structure is a good method to solve the neutron shielding problem. The comparison of the calculated and measured reaction rate proves that these results have the same changing trend except some discrepancy points. For example, the reaction rate of $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ at point 6 have larger errors between calculated and measured results. Fig.3 shows that most neutron flux at point 6 is in the energy range of 10^{-5} MeV to 0.1 MeV. But the cross section for $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ reaction in this energy range fluctuates greatly and it is difficult to obtain very good results. Hence, the discrepancy value is normal. TRIPOLI-4® is a validated code to simulate the neutron shielding problem.

IV. VARIANCE REDUCTION TECHNIQUE OF TRIPOLI-4®

For radiation shielding problems with strong attenuation, biasing the transport of the particles emitted to the trajectories which contribute to the expected results has been employed in order to reduce calculation time and improve the efficiency of the convergence. The biasing scheme of TRIPOLI-4® is based on Exponential Transform, and it can be used for neutron transport or photon transport or both [5]. This biasing method has been used for the dogleg benchmark neutronic simulation.

In order to show the difference between analog and non-analog, neutron collision sites were tracked, as shown in Fig 8 and Fig 9. Comparison of two figures, we can know that there will be very little neutron collision sites at our focus points if we don't use variance reduction techniques. The analog and non-analog calculation time on CEA's parallel cluster are 5155 s and 440 s respectively. The standard deviation of reaction rate for $^{93}\text{Nb}(n,2n)^{92\text{m}}\text{Nb}$, $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$ and $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ are listed in TABLE IV. The non-analog calculation decreases the computation time greatly. Meanwhile, the standard deviation of the non-analog calculation is less than analog calculation. The distribution of the neutron collision sites, the calculation time and the standard deviation illustrate the advantages of the variance reduction technology.

Fig. 8. Two-dimensional view of neutron collision sites(analog)

Fig. 9. Two-dimensional view of neutron collision sites(non-analog)

TABLE IV
THE STANDARD DEVIATION OF REACTION RATE RESULTS OF ANALOG AND
NON-ANALOG CALCULATION

point	$^{93}\text{Nb}(n,2n)^{92m}\text{Nb}$		$^{115}\text{In}(n,n')^{115m}\text{In}$		$^{197}\text{Au}(n,\gamma)^{198}\text{Au}$	
	standard deviation	standard deviation	standard deviation	standard deviation	standard deviation	standard deviation
	analog	non-analog	analog	non-analog	analog	non-analog
1	0.16%	0.01%	0.21%	0.07%	3.19%	1.13%
2	0.20%	0.01%	0.25%	0.09%	4.33%	1.57%
3	0.25%	0.02%	0.32%	0.11%	6.57%	2.25%
4	3.44%	0.44%	1.07%	0.38%	7.84%	2.77%
5	9.76%	0.98%	2.43%	0.86%	8.69%	3.44%
6	9.86%	1.32%	2.86%	1.01%	9.45%	4.08%
7	38.92%	3.39%	6.66%	2.32%	24.14%	7.10%
8	31.33%	5.07%	6.63%	2.38%	39.32%	13.15%
9	10.23%	9.67%	4.30%	1.57%	21.93%	13.50%
10	27.39%	11.82%	6.75%	2.37%	38.51%	14.98%
11	39.71%	9.16%	8.23%	3.00%	39.70%	14.51%

V. CONCLUSIONS

TRIPOLI-4®, the well-known Monte Carlo transport code, was proved to be an accurate and reliable code to carry out the nuclear analyses and neutron streaming effects evaluation. This conclusion is drawn from the good agreement in the comparison between the simulated results by the code and the measured results of JAEA. The advantages of variance reduction technology were also demonstrated in the shielding calculation and the non-analog calculation were proved to decrease the computation time efficiently. Some discrepancies of the comparison results may be caused by the cross section data and still need to be investigated by further study.

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