RICC Usage Report for Fiscal Year 2010

Project Title:

Simulation for radiography of neutron and X-ray as well as the shielding of radioactive rays

Name : 王 盛 (Wang Sheng)

Affiliation: VCAD Modeling Team VCAD System Research Program Innovation Center

Description of the project

1. Background

Investigation of inside structure as well as outside profile is important for many industrial applications. Concerning X-ray imaging systems, they have been applying not only to measure the outside profile but to investigate inside defects such as welding part, cast components and so on. However, X-ray imaging system has limit on penetration depth so that it is hard to measure large industrial components or structures. Neutron beam has advantages on penetration depth over X-ray. Furthermore, material information like strain or temperature that may be obtained by using pulsed neutron source will be quite useful in production engineering. Thus, neutron radiography is expected to be widely utilized in the production processes compared to X-ray CT systems, which are recently becoming popular method for product inspection on-site. Neutron radiography has a number of advantages over X-ray system. However, it requires very large accelerator systems.

For industrial radiography applications, we are planning to construct a compact neutron source using a small proton accelerator combined with a lithium or beryllium target. Primary applications of compact accelerator based neutron source are radiography of industrial components. By utilizing deep penetration depth of neutron beam, investigation of pores inside cast iron parts or other heavy materials are preferable application. Inspection of junction between composite material (carbon fiber structure and steel or aluminium) could be another good application taking advantage of neutron radiography. If transportable neutron source is possible, it can be applied to investigation of large industrial product like aircraft or ship or large scale structures like bridges, buildings.

A preliminary design of accelerator based compact neutron source for industrial and transportable use have been started. We are planning to use low energy nuclear reaction of Be(p,n) or Li(p,n) to produce neutron beam. For laboratory use, size of accelerator system need to be designed to fit 5 to 10 square meter laboratory space. For the purpose of transportable use, accelerator systems are more compactly designed since trailer can carry out approximately 3 to 5 m size and mass of several tons. In this paper, moderator for compact neutron source is simulated and estimated performance of neutron flux is calculated by Monte Carlo code (PHITS, ver. 2.24). Finally, these results are compared with other laboratory scale neuron source.

Monte Carlo calculation is the basic and the most important tool for the compact neutron source design, as well as for the whole radiography system design. As known to all, Monte Carlo calculation is very time-consuming. The calculation cost almost has linear relationship with the particle number. At the same time, Monte Carlo code always has a perfect parallel efficiency. So running the parallel Monte Carlo calculation by using RIKEN Cluster (RICC) is very necessary for the success of this project.

2. Simulation modeling and methods

Neutron flux was simulated by PHITS (Particle and Heavy-Ion Transport code System, ver. 2.24) using RICC. The computational cost for one case is about 24 cores running 12 hours. Neutron source data was based on the results by Gibbons et al. Figure 1 shows schematic diagram of moderator and parameter setting for moderator design. A beryllium plate which has 10 cm diameter was used for target. For reflection of neutron, a beryllium and graphite were adopted. A boric acid resin was used for neutron shielding. To absorb gamma ray, lead was utilized on the outside of boric acid resin as well as output duct. Two detectors, which are located 2m and 5m far from the target, are put to detect the neutron flux. Table 1 shows detailed input parameters for simulation.

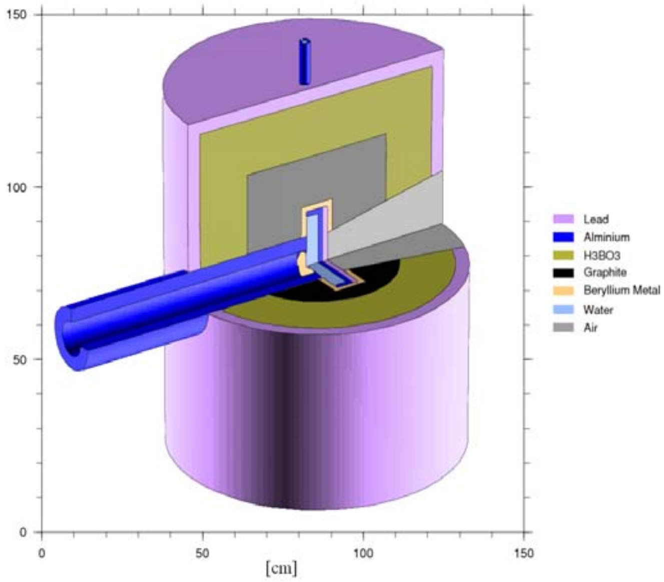


Fig.1. 3D view and parameters setting of compact accelerator system

Table 1. Input parameters for simulation

E_p [MeV]	5.42	Thickness of Be reflector [cm]	2
Radius of beam [cm]	1	Radius of graphite [cm]	30
Thickness of target [cm]	0.2	Height of graphite [cm]	80
Radius of target [cm]	5	Radius of boric acid resin reflector [cm]	50
Housing thickness of moderator [cm]	2	Height of boric acid resin reflector [cm]	120
Radius of duct inside [cm]	6		
Thickness of duct [cm]	5	Radius of gamma shield [cm]	55
Duct length [m]	1	Height of gamma shield [cm]	130

To evaluate moderator performances, flux of thermal neutron at the point of 2 m and 5 m from the target was compared. To investigate optimum condition of neutron flux, 7 types of materials (Light water, heavy water, Be, Graphite, 7Li, Polyethylene, solid methane) were utilized as a moderator, and its performances were compared according the thickness of moderator. In this case, height of moderator was fixed as 12 cm.

After material evaluation as moderator, neutron flux according to the L/D was verified. There are many parameters on the geometry, we assumed that inlet and outlet diameter of output duct (D_{in} and D_{out}) are proportional to the height of moderator ($B=D_{in}$ and $1.2\ D_{in}=D_{out}$). We used two type of distance (2m and 5m), so that L/D is decided by parameter D.

After modification of geometry, TOF (Time of Flight) was calculated. Based on these results, operation frequency and duty ratio was estimated. In this paper, thermal neutron is defined that neutron energy is from 0.01 to 1 eV. Over the 0.1 MeV is defined as fast neutron.

3. Result

3.1 Moderator design

Figure 2 and 3 shows flux distributions of thermal neutron in the distance of 2m and 5m from target according to the moderator materials. In the results, polyethylene shows the highest thermal neutron flux among the seven materials. Except polyethylene, light water also shows high neutron flux around 6 cm thickness. In the case of polyethylene, maximum peak points of thermal neutron flux are located on the 5 cm thickness in the both 2m and 5m distance. Flux of thermal neutron was approximate $2.5x10^6$ n/cm2/mA (2m), and $3.2x10^5$ n/cm2/mA (5m). Figure 4 shows flux distribution of thermal neutron in the case of polyethylene with thickness of 5cm and height of 12 cm. Flux of thermal neutron increases inside of moderator as shown in the figure 4 b. As a first candidate, polyethylene is selected for moderator material, since it is easy to handle and cheap.

Based on these results, thickness of moderator is fixed and height of moderator is swept to verify L/D characteristics. Figure 5 shows flux of thermal neutron according to the L/D at the point of 2 m and 5 m, where moderator material is polyethylene and moderator thickness is 5 cm. For the radiography applications, required minimum L/D is approximately 50. Thus, this graph shows that thermal neutron flux is 2.1 x 10^5 n/cm2/mA (2m), and

2.3 x10⁵ n/cm2/mA (5m) where both L/D is 50.

3.2 TOF calculation

Figure 6 shows the time of flight (TOF) calculation in the case of fast and thermal neutrons. Figure 7 shows TOF calculation of fast and thermal neutron at the point of 2 and 5 m. In both 2 and 5 m cases, fast neutron is almost disappeared in 1 microsecond, and thermal neutron duration is 0.1 to 10 ms in the distance 2 and 5m. Thus, this system can be operated on the 200 Hz, 10 % duty condition.

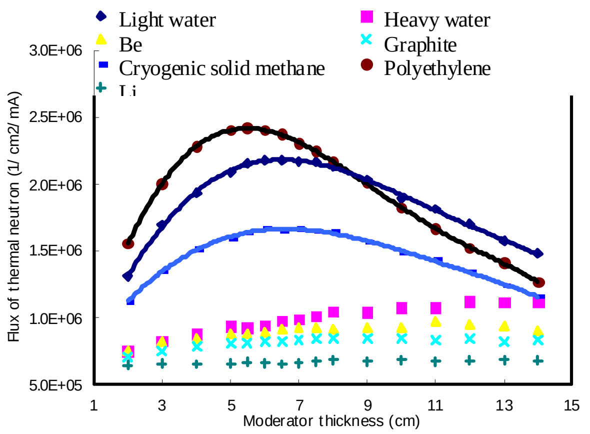


Fig.2. Flux distribution of thermal neutron in the distance of 2m from target according to the moderator materials (height of moderator: 12 cm)

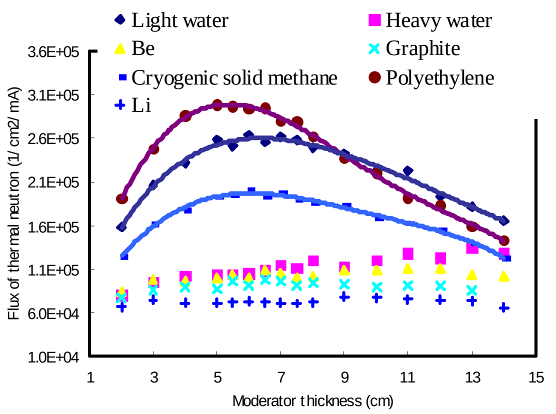


Fig.3. Flux distribution of thermal neutron in the distance of 3m from target according to the moderator materials (height of moderator: 12 cm)

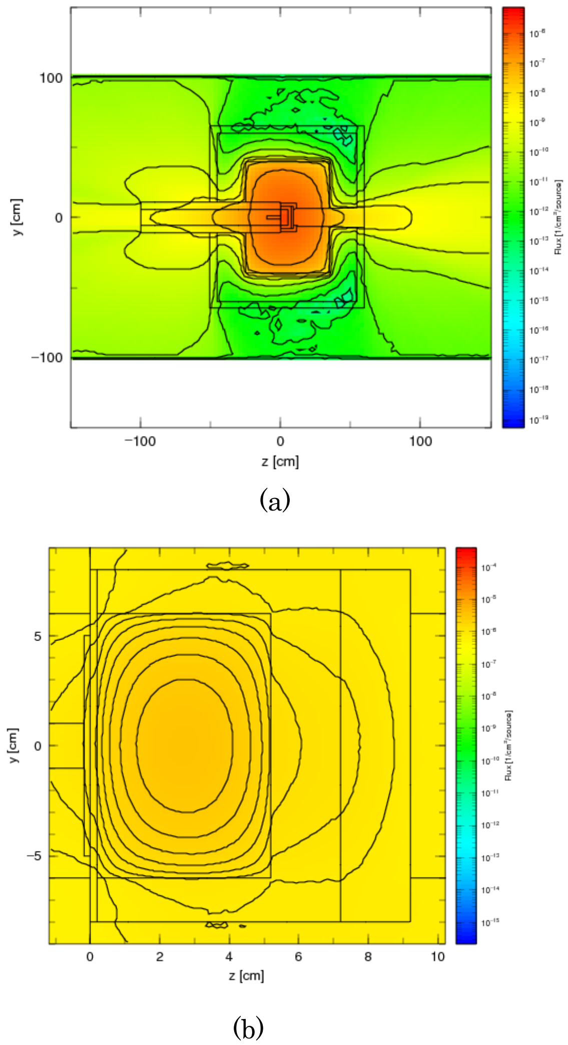


Fig.4. Flux distribution of thermal neutron: (a) entire area of system; (b) around moderator (moderator material: polyethylene, moderator thickness: 5 cm, height of moderator: 12 cm)

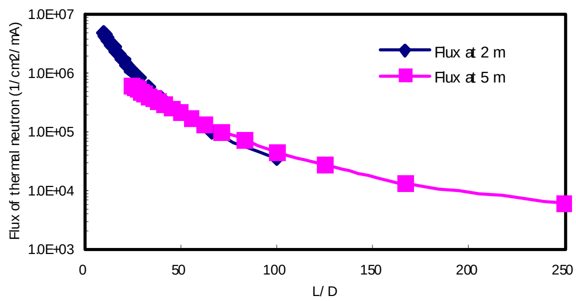
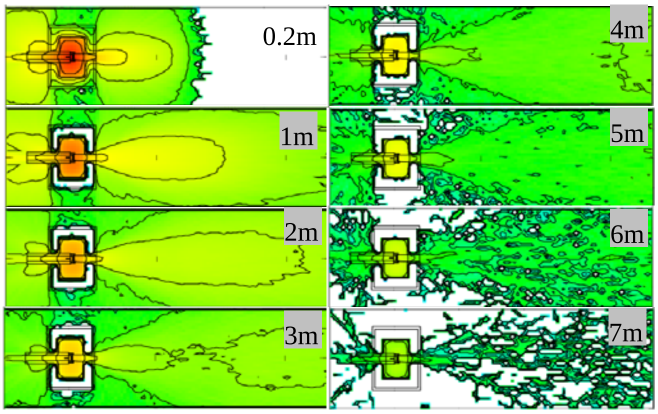
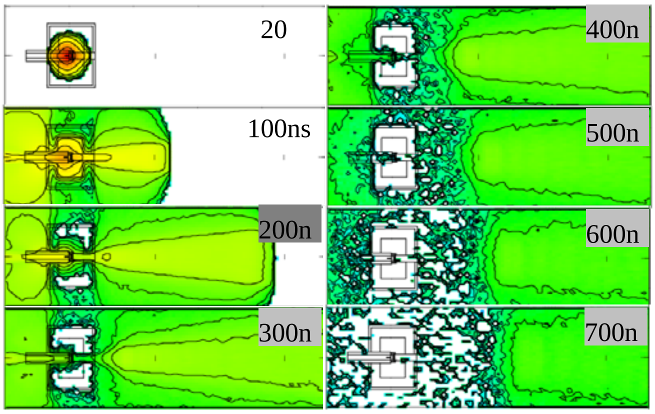


Fig.5. Flux of thermal neutron according to the L/D, (moderator material: polyethylene, moderator thickness: 5 cm)



(a) Thermal neutron



(b) Fast neutron

Fig.6. Time of flight (TOF) calculation in the case of thermal neutron and fast neutron

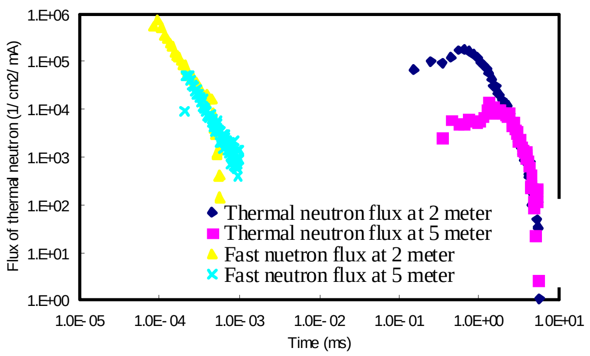


Fig.7. Time of flight (TOF) calculation of fast and thermal neutron at the point of 2m and 5m distance

4. Conclusion

A plan for compact neutron source based on accelerator at RIKEN (The Institute of Physical and Chemical Research) was discussed. First, polyethylene was adopted for material of moderator since thermal neutron flux was higher than others. For the radiography application, high beam quality determined by L/D is necessary. If required L/D is 50, thermal neutron flux is 2.1 x 10^5 n/cm2/mA (2m), and 2.3 x10^5 n/cm2/mA (5m). Then, to use the thermal neutron by pulsed operation, TOF was calculated by simulation. As a result, this system can be operated on the 200 Hz, 10 % duty condition.

This results shows that thermal neutron flux is 2.1 x 10^4 n/cm^2 (2m), and 2.3 x 104 n/cm^2 (5m), where L/D is 50 and average current is 0.1 mA (peak current is 1 mA). For the radiography application, required minimum flux is 10 ^{5} n/cm2. To increase neutron flux, higher proton energy as well as beam current is necessary. Approximately, neutron yield of 7 MeV proton energy on the Be(p.n) reaction is increased 1.9 times than 5.4 MeV. Thus, to get 10 ^{5} n/cm^2 thermal neutron flux, beam current need to be increased up to 0.2 mA. Thus, 7 MeV, 0.2 mA average current system seems to be feasible.

There are still many parameters, which are not well discussed. Further analysis and consideration will be necessary before construction. Therefore, many calculations with PHITS code by using RIKEN Cluster (RICC) will be carried out in the near future.

5. Schedule and prospect for the future

The next step within fiscal year 2011 will focus on the shielding design and the heat removal system design by using PHITS code run on RICC.

6. If you wish to extend your account, provide usage situation (how far you have achieved, what calculation you have completed and what is yet to be done) and what you will do specifically in the next usage term.

Up to now, the moderator design has been finished. To complete the whole compact neutron system design, the shielding system and the heat removal system have to be evaluated and designed. The next usage term will focus on the two systems.

Fiscal Year 2010 List of Publications Resulting from the Use of RICC

[Publication]

Jungmyoung Ju, Sheng Wang, Kenji Mishima, Shin-ya Morita, Katsuya Hirota, Yutaka Yamagata, Yoshihisa Iwashita, Hirohiko M. Shimizu, Yoshiaki Kiyanagi, Hideyuki Sunaga and Akitake Makinouchi, Moderator design and simulation for neutron radiography using a compact accelerator at RIKEN, *Nuclear Instruments and Methods A*, in press.