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Paper No. 2002 **Research on Shielding Performance Measurements of radioactive materials Transportation Package**

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Abstract

Some radioactive sources are often used in irradiation industry and radiotherapy, such as Co-60, Cs-137, I -131, P-32, Sm-153, Mo-99, Sr-90, Sr-89 etc. The amount of radioactive sources transportation is large. Recently, a large number of spent fuels also need to be shipped out. In order to keep all doses exposed lower than the limits during the transport of the radioactive materials, some corresponding radiation protection measures must be taken and the radiation shielding performance of radioactive materials package must meet the requirements of GB11806-2004 (Regulations for the safe transport of radioactive material). The requirements of GB11806-2004 are same with the requirements of IAEA SSR6 (Regulations for the safe transport of radioactive material). Since the state council decree 562(the radioactive goods transportation safety management regulations) issued, the safety supervision of radioactive materials transportation package has been further stressed and implemented in china. The shielding performance measurements of radioactive materials transport package are the important content of supervision. However, some of the problems and difficulties reflected in practice need to be solved. The layout of radioactive sources has a important impact on the monitoring results of the external radiation levels of radiation sources transport container. The neutron dose rate on the surface of spent fuel package is too difficult to be measured precise, the monitoring results are not always reliable while the equipment is not chosen appropriately etc. In this paper, above problems and difficulties were discussed. The monitoring results using different spectrometers were compared and the MCNP runs are done to study the shielding performance measurements of radioactive materials package.

Introduction

With the development of nuclear technology utilization in the field of industry and radiotherapy, irradiation processing technology is extensively applied in materials processing, which make use of physical effects, chemical effects and biological effects during interacting between radiation and materials. Up to 2009 the number of large-scale irradiating equipment is larger than 200, and the total activity exceed 2 thousands of millions Curie. Recently, each Co-60 source used by large-scale irradiating equipment has an radioactive activity of about 8000～14000 Curie, due to the total activity of radioactive source loaded in container is high, the transportation risk is larger. Meantime, with the development of nuclear power, the storage capacity of spent fuel pool is not enough, spent fuels are badly in need of transportation. As a result of the high radioactive activity of spent fuels, the transportation safety is concerned by public and specialist.

The radiation shielding performance of radioactive materials package must meet these requirements of“ Regulations for the safe transport of radioactive material”(GB11806-2004), Since the state council decree 562(the radioactive goods transportation safety management regulations) issued, the safety supervision of radioactive materials transportation package has been further stressed and implemented in china. However, some of the problems and difficulties reflected in practice need to be solved. In this paper above problems and difficulties were discussed and an improvement to the shielding performance measurements technique and management of spent fuel transport is expected.

1 The Deviation of Shielding Performance Measurements of Radioactive Sources container

Sometimes the test sources are different from the practice sources loaded in container during shielding performance test, such as the activity, structure and layout of radioactive sources loaded in container etc. So the test results are influenced by these factors above. CIRP (China Institute for Radiation Protection) launch one research about the shielding performance measurement of one kind of radiation source container [6] and the MCNP [7] simulation was performed. The simulation results show that the outer radioactive level of container is18.98% larger when the only one test source is 4cm far from the center of container. Another simulation results show that the outer radioactive level of container is 44.8% larger when all the twenty test sources are placed in the center rather than scattering all around. Consequently, the layout of test sours should be same with the practice as far as possible and the test results will be more accurate.

2 The Surface Radioactive Level of Container

The surface radioactive level of container must meet the requirement of “Regulations for the safe transport of radioactive material” (GB11806-2004). Normally, the surface radioactive level can be measured using the monitor close to the surface of container. However, the sensitive volume center of the most monitors is 3cm to 5cm even larger. So when the test results of the surface radioactive level is approximate the standard limit, review experts will pay close attention to the test results.

One group of CIRP developed one type of container with steel-lead-steel form, and the MCNP simulation was performed to calculate the gradient of outer radioactive level. The simulation results show that the gradient of outer radioactive level vary from 10%～30% while the monitoring distance vary from 1cm～5cm.So, when the test results of the surface radioactive level is approximate the standard limit, which must be corrected. Furthermore，the maximum capacity of container should be selected according to rules, and sufficient safety margin should be left. The shielding performance of container should be intact under the normal transportation condition (requirements of GB11806-2004), and the increase of surface radioactive level should not be larger than 20%. So, the monitoring distance must be accordant and the influence can be eliminated.

3 The credibility of the monitoring results of neutron radioactive level

CIRP launch one research about the shielding performance measurement of one kind of spent fuel container. The monitoring results of using both the neutron multi-sphere spectrometer [8] and portable neutron survey meter are compared. The monitoring distance is 2m far from the surface of container. The Fig.1 is the Layout of monitoring position. Tab.1 shows the measurement results by use of multi-sphere spectrometer systems, and neutron spectra at each monitoring position is show in Fig. 2.

Tab.1 The measurement results of neutron radiation level of container outer by multi-sphere spectrometer systems

Position	Ambient Dose Equivalent (μSv/h)	Flux（$cm^{-2}.s^{-1}$）			
		TOT 2.58E-8～20Mev	Thermal 2.58E-8～5E-7 Mev	Middle 5E-7～1E-1 Mev	Fast 0.1～20 Mev
L2-1	1.35E+01	4.10E+01	8.24E+00	1.89E+01	1.30E+01
L2-2	2.00E+01	5.35E+01	1.33E+01	2.13E+01	1.78E+01
L2-3	1.36E+01	3.93E+01	7.67E+00	1.83E+01	1.26E+01
R2-1	1.26E+01	3.92E+01	7.33E+00	1.86E+01	1.24E+01
R2-2	1.91E+01	5.28E+01	1.12E+01	2.36E+01	1.70E+01
R2-3	1.33E+01	3.83E+01	8.13E+00	1.72E+01	1.21E+01
U2-1	9.11E-01	5.34E+00	1.02E+00	3.36E+00	8.31E-01
D2-1	1.01E+00	6.60E+00	1.18E+00	4.32E+00	9.43E-01

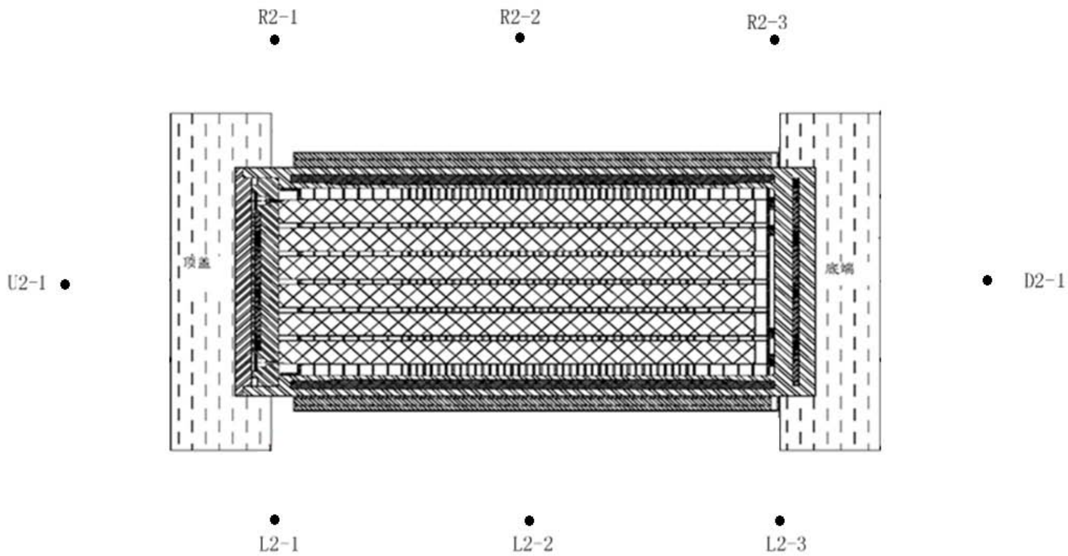

Fig.1 Layout of monitoring position

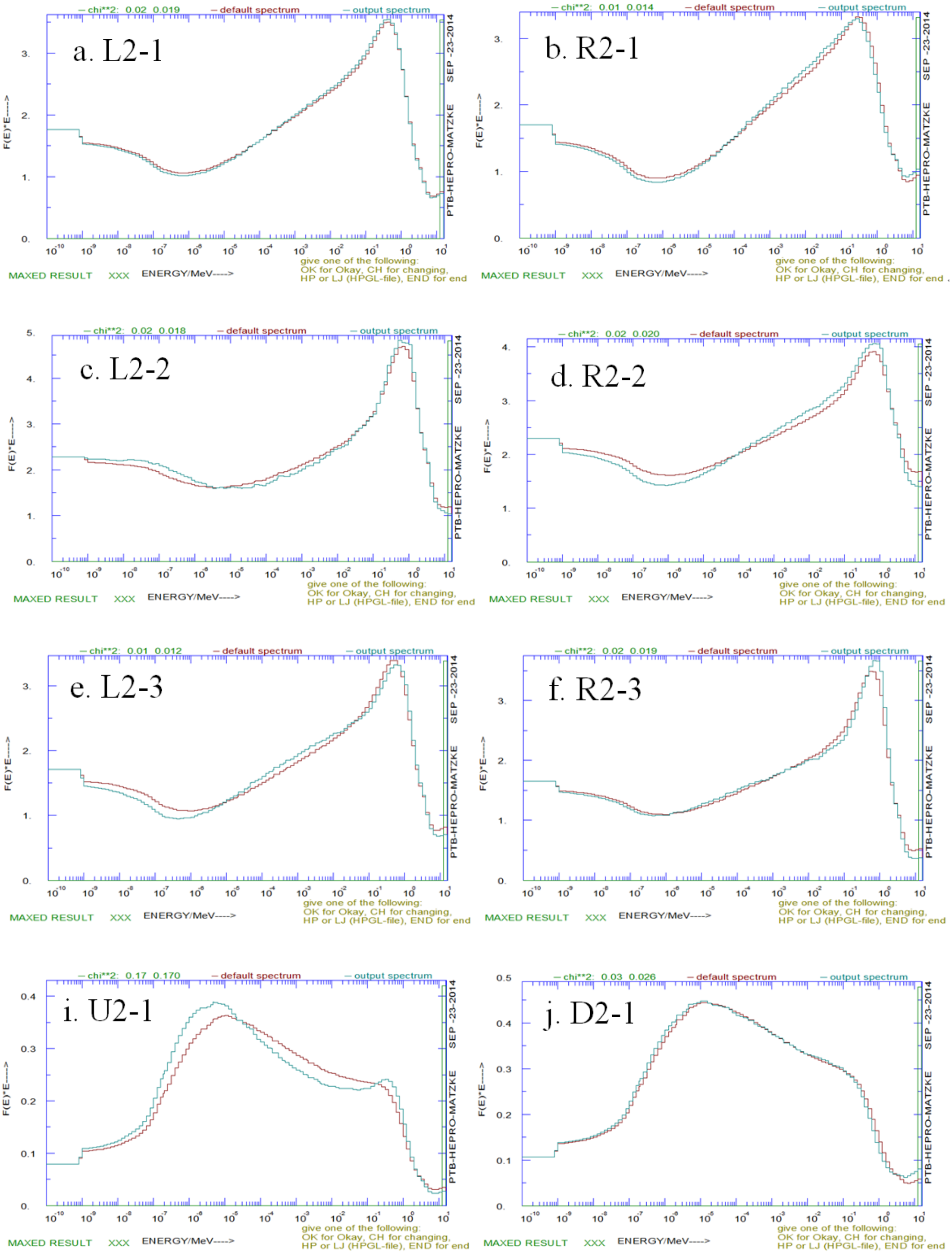

Fig.2 Neutron spectra at each monitoring position

Tab.2 The measurement results of neutron radiation level of container outer by Neutron survey meter and multi-sphere spectrometer systems (μSv/h)

Type of Instrument	L2-1	L2-2	L2-3	R2-1	R2-2	R2-3	U2-1	D2-1
multi-sphere spectrometer	9.05	14.56	9.01	4.82	13.51	7.5	0.71	1.21
Neutron Survey meter	13.50	20.00	13.60	12.60	19.10	13.30	0.91	1.01

The monitoring results using both the neutron multi-sphere spectrometer and portable neutron survey meter are compared in Tab.2. The results show that the varying tendency is the same, and the results using neutron multi-sphere spectrometer is larger. In order to analysis the great difference between the results using these two instruments, the MCNP simulation was performed to checking the test results. The Neutron source is calculated by SCALE [9]. The MCNP mode of spent fuels container is simplified, and some elements do not affect the shielding performance, such as pore, lifting lugs, bolts etc. The Fig.3 is the MCNP model.

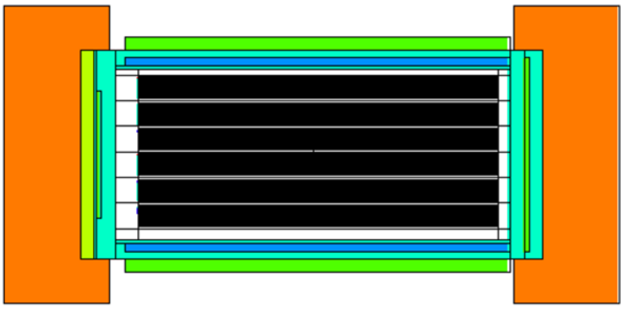

Fig.3 MCNP mode of spent fuels in container

The MCNP simulations take advantage of the continuous energy cross-section, the neutron flux are transformed into dose [10]. The simulation results are show in Tab.3.

Tab.3 Measurement results and the simulation results of neutron radiation level of container outer (μSv/h)

Measurement/Simulation	L2-1	L2-2	L2-3	R2-1	R2-2	R2-3	U2-1	D2-1
multi-sphere spectrometer	9.05	14.56	9.01	4.82	13.51	7.5	0.71	1.21
Neutron Survey meter	13.50	20.00	13.60	12.60	19.10	13.30	0.91	1.01
MCNP simulation	28.10	30.20	22.60	-	-	-	6.28	4.76

The data of Tab.3 show that the varying tendency of outer radioactive level is same by comparing the test results and simulation results. The radioactive level is low at the top and the bottom, and the simulation results are larger than the test results. The transport impact limiters at the top and bottom of container are fabricated from redwood and balsa wood. For conservative consideration, only balsa wood is described in the MCNP model. Furthermore, the test results by use of portable neutron survey meter are lower than the test results by use of the neutron multi-sphere spectrometer. From the Fig.3 neutron spectra one can see that the proportion of middle energy neutron and the fast neutron is large, due to the energy response of portable neutron survey meter is poor in this energy region, the test results is relatively lower than the test results using the neutron multi-sphere spectrometer.

The difference of these results is large. Possible reasons are that the firstly the MCNP simulation is based on the neutron source calculated by SCALE, and the input data are conservative, and the calculation results are uncertainly to some extent. The secondly the instrument is calibrated by use of the ^{241}Am-Be source and ^{252}Cf source, and the neutron spectra is possible different with the unknown neutron field. Furthermore, the calibration condition is different with the site environment.

4 The difficulty of the surface neutron radioactive level measurement

The measurement results of the surface neutron radioactive level is not accuracy because of the absent of proper instrument. Due to effect of neutron elasticity scatter, the deviation of the measurement results is larger. One group of CIRP attempt obtaining the surface neutron radioactive level by use of dosimeter. Because the surface temperature of spent fuel container is high, the counting loss is severe, and the results should be corrected. However, How to correct the results is difficulty.

5 Conclusions

The shielding performance measurements are influent by the test sources feature, such as the activity, structure and layout of radioactive sources in container etc. The test sours should be same with the practice as far as possible and the test results will be more accurate.

The test results of the surface radioactive level must be corrected. Furthermore, the maximum capacity of container should be selected according to rules, and sufficient safety margin should be left.

The difference of these results by use of different instruments is large. The test results by use of portable neutron survey meter are lower. Due to the proportion of the fast neutron is large, the energy response of portable neutron survey meter is poor. The instrument is calibrated by use of the ^{241}Am-Bes source and ^{252}Cf source, and the neutron spectra is possible different with the unknown neutron field. Furthermore, the calibration condition is different with the site environment. The measurement range of instrument should be proper. The MCNP simulation is based on the neutron source calculated by SCALE, and the input data are conservative and the calculation results are uncertainly to some extent. The test results using the neutron multi-sphere spectrometer are relatively accordant with the simulation results, and the energy response range of multi-sphere spectrometer is larger which is proper to the shielding performance measurements of spent fuel container. An appropriate measurement scheme is that the firstly finding out the positions where neutron radioactive level is high by use of the neutron survey meter, the secondly exactly monitoring the neutron radioactive level by use of the multi-sphere spectrometer.

A systemic Research on the surface neutron radioactive level measurement should be developed.

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