Proceedings of the 18th International Symposium on the Packaging and Transportation of Radioactive Materials
PATRAM 2016
September 18-23, 2016, Kobe, Japan

Paper No.2014

Developing a new transportation container for bulk irradiation source

Zhuang Dajie
China Institute for Radiation Protection, Taiyuan, China

Li Guoqiang
China Institute for Radiation Protection, Taiyuan, China

Wang **Renze**
China Institute for Radiation Protection, Taiyuan, China

Sun **hongchao**
China Institute for Radiation Protection, Taiyuan, China

Meng Dongyuan
China Institute for Radiation Protection, Taiyuan, China

Abstract

For irradiation source supply and recycle to the irradiation station both at home and abroad, irradiation source will be transport from production site(Beijing) to irradiation station, as well as from irradiation station to waste repository. China Institute for Radiation Protection(CIRP) was commissioned the work of developing FCTC10 container. FCTC10 container is designed to transport Co-60 special form radioactive sources used in irradiation industry by road, railway or water. With a loading of 180 000 Ci Co-60, the package is a Type B(U), Category III (yellow) package and TI(Transport Index) 9.8. The design life is 20 years. FCTC10 package should comply with the requirements stipulated in IAEA regulations for the transportation of radioactive materials. The container consists of shielding container, hanging basket, protective cover and bracket. The shielding performance mainly rely on the shielding container, which comprises stainless steel shells, tungsten alloy and lead filling. And the structural and thermal integrity under the hypothetical accident conditions has been demonstrated through a series of analyses and test.

Introduction

With the rapid development of nuclear technology in industrial application, irradiation process is applied widely in various fields, such as medical disinfection and sterilization, food preservation, radiation chemical engineering, breeding, environment treatment, etc. Up to now, there are more than 200 large-scale γ irradiation facility with a total activity of more than 2.0×10^8 Ci. In the US, irradiation sterilization facilities with an activity of about 1.0×10^7 Ci were designed and constructed. In China, approximately 200 irradiation facilities were constructed, of which more than 100 facilities were designed to use sources of above 3.0×10^5 Ci, while more than 40 facilities use sources of above 1.0×10^6 Ci. Designed capacity of source loading is more than 1.0×10^8 Ci, and the actual loading is about 3.6×10^7 Ci [1].

^{60}Co is a common used industrial irradiation source. At present, the activity of a ^{60}Co source used in the large-scale irradiation facility is about 8,000 to 14,000 Ci, which is Category Ⅰ source

specified in GB11806-2004, and the transport contender loading such a source is Type B package, accordingly. FCTC10 container with a loading of 1.8×10^5 Ci ^{60}Co source is a Type B(U), Category Ⅲ (yellow) package. Considering the high activity and potential hazards of ^{60}Co source, the container should designed carefully, especially the shielding performance. According to the regulations of authority, tests including water spray, free drop, stacking and penetration for normal conditions of transport have been performed, while drop Ⅰ, drop Ⅱ, thermal test and water immersion have been carried out for accident conditions. In addition, simulations and measurements were conducted to verify design parameters [2-3]. The design was approved by the Chinese authority in 2014.

1 Structure of FCTC10 Container

As shown in Fig. 1, the container consists of shielding container, hanging basket, protective cover and bracket. Shielding container comprises inner cask, lead plug, heat insulation, shock absorber. From inside out, the cask includes 10 mm inner stainless steel (0Cr18Ni9) shell, tungsten alloy bucket, protective cover, lead filling, and 16 mm outer stainless steel (0Cr18Ni9) shell. The tungsten alloy bucket has an outer diameter of 291 mm, a height of 614 mm, and a thickness of 42 mm on the upper side part, 62 mm on the side and 74 mm on the bottom. The protective cover is made of 5 mm stainless steel (0Cr18Ni9). The radial thickness of the lead filling is 147.5 mm. The lead plug consists a lead cylinder and a tungsten alloy cylinder stack. The lead cylinder has a radius of 154.5 mm and a height of 109 mm, while the tungsten alloy cylinder stack has an upper radius of 154.5 mm, upper height of 20 mm, lower radius of 86.5 mm, and lower height of 97 mm. The lead plug is covered with stainless steel (0Cr18Ni9), which is 2.2 mm thick on the side, 12 mm on the bottom, 3 mm in the middle and 25 mm on the top. The heat insulation comprises 6 mm stainless steel shells with 26 mm aluminum silicate fiber blanket in the middle.

The hanging basket, which is 478 mm high and 140 mm in diameter, is placed in the middle of the container. The basket is a rod lattice structure and made of stainless steel (0Cr17Ni12Mo2). The protective cover of 1264 mm×1264 mm×1414 mm consists of an angel steel frame, 4 lifting rugs, 4 side pull tab and a bottom steel net (0Cr18Ni9). The size of the steel frame is 1250 mm×1250 mm×230 mm, while the side of both top and bottom plates is 1250 mm×1250 mm ×10 mm. The inner cask together with the tungsten alloy and lead fillings are mainly the components to provide shielding to the source.

2 Radiation Level Calculation and Measurements

2.1 Calculations

Monte Carlo codes have been used to calculate the radiation level outside the FCTC10 container with the maximum loading [4]. Only steel shells, tungsten alloy and lead fillings were considered in the calculation. The components of each material are listed in Table 1, and the simplified model used in the calculation can be found in Fig. 2 Up to 19 ^{60}Co sources (total activity$\leq1.8\times10^5$ Ci) can be loaded in the container. Calculated dose rates at different positions are shown in Table 2.

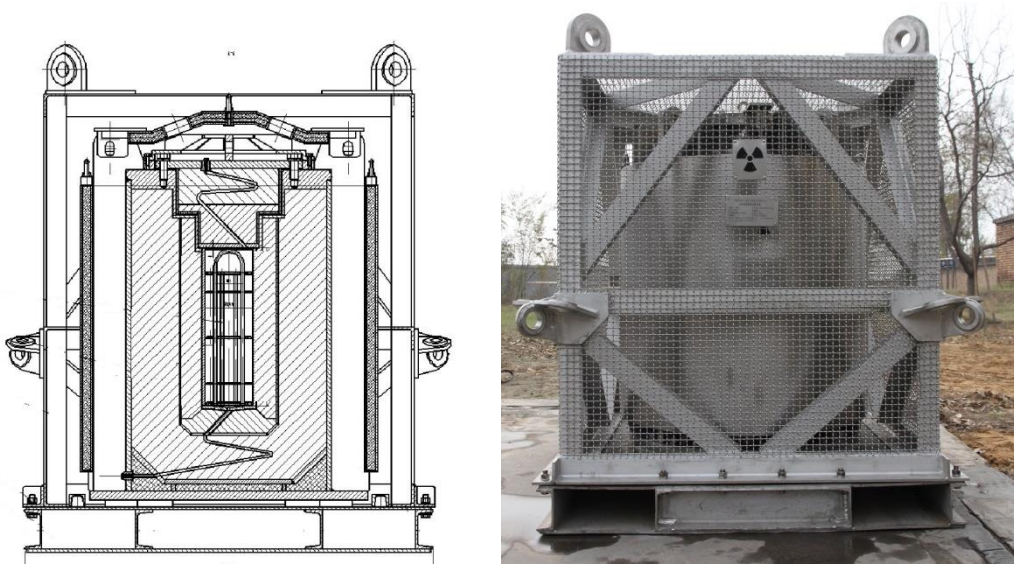

Fig 1 Structure of FCTC10 transport container

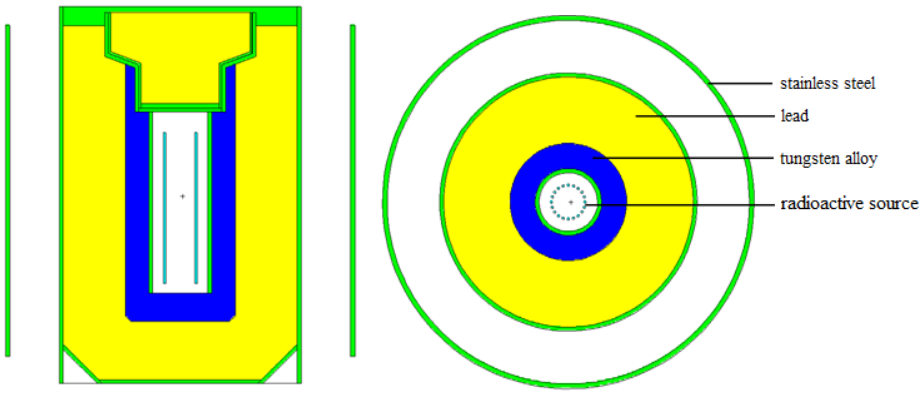

Fig. 2A. Side view **Fig. 2B. Top view**

Fig 2 Shielding calculation model for FCTC10 container

Table 1. Composition of the container components

Material	**Element**	**Atom density (atom· b^{-1}· cm^{-1})**	**Density (g· cm^{-3})**
0Cr18Ni9	Cr	1.65149×10^{-2}	7.92
	Fe	5.95863×10^{-2}	
	Ni	7.3114×10^{-3}	
	C、Si、Mn、S、P	3.8517×10^{-3}	
Lead	Pb	3.30430×10^{-2}	11.34
Aluminum silicate fiber	Al	6.64584×10^{-4}	0.128
	Si	6.67603×10^{-4}	
	O	2.33220×10^{-3}	
	C	2.20007×10^{-4}	

	H	4.40180 $\times 10^{-4}$	
Tungsten alloy	W	6.01918 $\times 10^{-2}$	19.35
	Cu、Ni、Fe	1.88982 $\times 10^{-2}$	

Table 2. Calculated dose rates at different positions

Position	**Shielding container**				**Package**			
	Side		**Bottom**		**Side**		**Bottom**	
	Surface	**1 m**	**Surface**	**1 m**	**Surface**	**1 m**	**Surface**	**1 m**
Dose rate ($\mu Sv \cdot h^{-1}$)	1710.7	90.7	225.7	19.8	825.8	84.6	209.9	6.7
Relative error (%)	0.70	0.34	0.45	0.64	0.53	0.37	1.19	2.66

2.2 Measurements

^{60}Co sources of total 1.77×10^5 Ci was used in the measurement. The activity of each source and the positions in the container can be found in Table 3 and Fig. 3, respectively. The dose rates on the surface of the container, at the distance of 1 m and 2 m from the surface were measured with a FHZ612-10 (Thermo Fisher Scientific Inc., America) dose rate meter, which has a measuring range from 0.1 $\mu Sv\ h^{-1}$ to 10 $Sv\ h^{-1}$, calibration factor of 0.98, and uncertainty of 5% (K=2). Measured results are list in Table 4.

Table 3. Activity list of radioactive sources used in test

Symbol	**Activity (Ci)**	**Symbol**	**Activity (Ci)**	**Symbol**	**Activity (Ci)**
13-441	11896	**13-448**	11860	**13-464**	11916
13-442	11806	**13-450**	11781	**13-468**	11880
13-443	11968	**13-453**	11666	**13-469**	11757
13-444	11982	**13-458**	11767	**13-471**	11689
13-445	11874	**13-460**	11789	**13-472**	11786

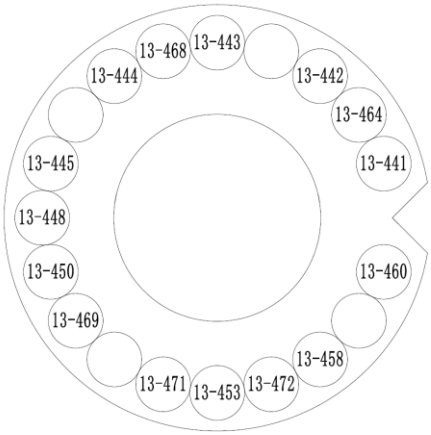

Fig. 3. Positions of radioactive sources in basket (top view)

Table 4. Maximum dose rates measured outside package and shielding container ($uSv \cdot h^{-1}$)

Activity (Ci)	Shielding container Side Surface	1m	Package Side Surface	1m	Top Surface	Bottom Surface
177417	1190	104	595	83.1	270	120
（180000[1])	1207	106	603.7	84.3	273.9	121.7

1) The measured results were normalized to ^{60}Co of 1.8×10^5 Ci.

2.3 Result

The maximum dose rates measured are on the side of the package and the shielding container, at the same height as the source. Calculated dose rates are listed in Table 5 together with measured results.

Table 5. Comparison between MCNP model calculation results and measured results ($uSv \cdot h^{-1}$)

Position / Items	Side of shielding container Surface	1m	Side of package Surface	1m
Calculation value	1710.7	90.7	825.8	84.6
Measured value	1207	106	603.7	84.3
Relative difference	41.7%	-14.4%	36.8%	0.4%

Note: Considering the size of the dose rate meter, the position used in calculations are not exactly the same as in measurements.

FCTC10 container loaded with ^{60}Co of 1.80×10^5 Ci is Type B(U), Category Ⅲ (Yellow) package. According to the regulations, the radiation level should not exceed 10 m Sv h^{-1} at any point on the external surface of the package, while it should be less than 0.1 mSv h^{-1} at 1 m, if it's not for exclusive use. Based on the design, radiation level should be less than 1.5 mSv h^{-1} on the surface,

and it should not exceed 0.098 m Sv h^{-1}. The calculated and measured results show that the radiation level of the container with maximum activity of ^{60}Co meets the design requirements.

3 Conclusion

According to the calculations and measurements, the position of maximum dose rate is on the side of the package. The maximum dose rate on the surface is 825.8 μSv h^{-1}, while it's 84.6 μSv h^{-1} at 1 m from the surface. The radiation level meet the regulations in GB11806-2004, and the design requirements for Type B(U), Category Ⅲ (yellow) package.

References

[1] He Shijun. Protection and safety of ionizing radiation utilization in industry. Beijing. Atomic energy press. 2009:12-13. (in Chinese).

[2] IAEA. Regulations for the Safe Transport of Radioactive Material. IAEA Safety Standards Requirements. 2012. No.SSR-6.

[3] IAE. Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material. IAEA Safety Standards Requirements. 2012. No.SSG-6.

[4] Chen Y F, Chiou Y F, Chang S J. Effects of source and geometry modeling on the shielding calculation for a spent nuclear fuel dry storage cask. Nuclear technology. 2013; 182(2):224-234. (in Chinese).

[5] Tang Ziqiang, Li Guoqiang, Su Buyun, Li Zhiqiang. Numerical Simulation on Impacting Test of Industrial Irradiation Transport Package for Radioactive Material Under the Accident Condition. Journal of TaiYuan University of Technology. 2015; 46(2):247-252. (in Chinese).