3 - 14 Separation of Medical Isotope ^{131}I from Uranium Fission Products through Distillation

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^{131}I is a widely used as diagnosis and treatment isotope for thyroid cancer, hyperthyroidism, and hypothyroidism. At present, Na^{131}I medicines are mainly derived from $^{131}TeO_2$ irradiated by nuclear reactor[1]. Except for coming from nuclear reactor, ^{131}I can also be generated through accelerators[2]. In present work, ^{131}I is prepared by bombarding $^{nat}UO_2$ target with high energy proton and distillation method was adopted to separate the iodine isotopes from the fission products of uranium, considering the good volatility of iodine. The separation process was optimized. The best conditions for efficient ^{131}I separation were determined.

Two steps were involved in the iodine isotope separation process. At the beginning, pulverizing the uranium target at a relatively low temperature by using the oxidation reaction, and then vaporizing iodine from the pulverized target at a high temperature. The correlation between the iodine separation efficiency and vaporizing temperature was investigated and it was found that the separation efficiency increased with the temperature. When the temperature is higher than 900 °C, the separation efficiency of the generated iodine from the target is more than 90%.

The vaporized iodine isotopes were collected by the NaOH solution. The collection solution included ^{132}I, ^{133}I, and ^{131}I. This result can be verified by the γ spectrum of the irradiated $^{nat}UO_2$ target before and after heat treatment (Fig. 1). Since the half-life of ^{131}I is longer than that of the others, the radionuclide purity of Na^{131}I solution can be

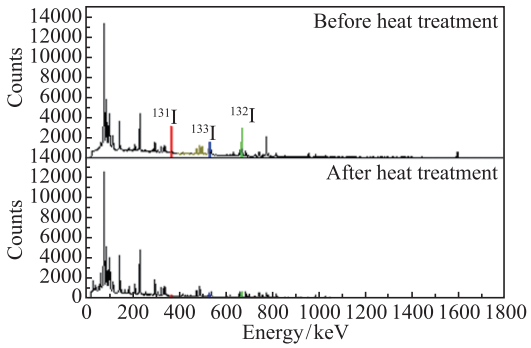

Fig. 1 (color online) The γ spectrum of the irradiated $^{nat}UO_2$ target before and after heat treatment.

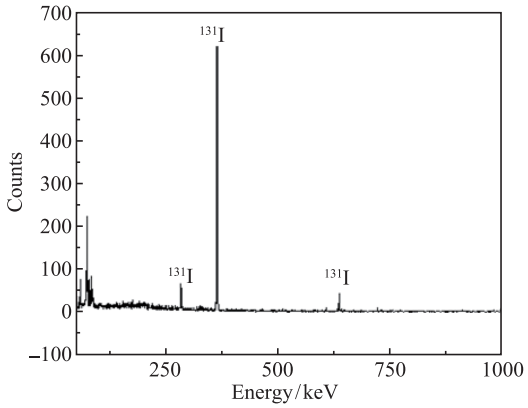

Fig. 2 The γ spectrum of the end product (Na^{131}I).

acquired by leaving the collection solution alone for about 15 days to let the decay of ^{132}I and ^{133}I. Fig. 2 illustrates the γ spectrum of the end product Na^{131}I, which indicate that the present method is practical to get Na^{131}I solution with high radionuclide purity.

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3 - 15 Preliminary Research of Radiopharmaceutical Labeling of Ga-DOTATATE

Gao Ruiqin, Wang Jieru and Qin Zhi

Radio pharmaceuticals are used currently for medical diagnosis and targeted radionuclide therapy. An increase in the availability and accessibility of PET/CT facilities has sparked strong interest in generator-based PET radiopharmaceutical. ^{68}Ga ($t_{1/2}$= 67.71 min) is an excellent positron emitter, with 89% positron branching accompanied

by low photon emission (1.077 keV, 3.22%)[1], it is produced by a ^{68}Ge/^{68}Ga radionuclide generator and is independent on an on-site cyclotron. ^{68}Ga-DOTATATE has been used for presurgical planning in cases of primary bronchial carcinoid and for staging of recurrent pulmonary neuroendocrine tumors[2]. Therefore, it is very crucial to bind ^{68}Ga to ligand stably.

Here, we carry out labeling experiment of Ga-DOTATATE using natural gallium to simulate the radioisotope ^{68}Ga. Firstly, a certain concentration of $GaCl_3$ is prepared, 100 μg DOTATATE was dissolved in water and then added to $GaCl_3$ solution. The pH was adjusted utilizing suitable buffer. The mixture was shaken gently to ensure homogeneity and the reaction was incubated for about 30 min at 95 °C. After cooling, the solution was passed over a C18 cartridge for further purification, then washed with water and ethanol. Finally, the quality of labelled compound was determined by HPLC (high performance liquid chromatography). The retention time of DOTATATE is 22.68 min. Under the same conditions, the retention time of reaction solution changed from 22.68 to 23.62 min, as shown in Fig. 1, which indicates that DOTATATE was already complex with free Ga. It can be deduced that the labeling process is carried out successfully for Ga-DOTATATE.

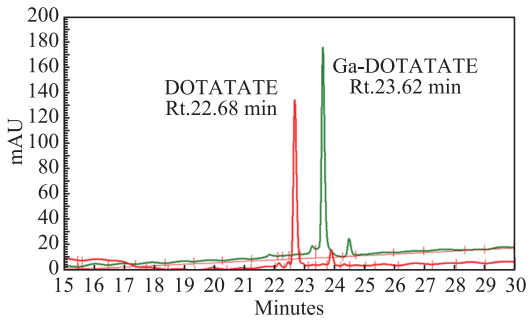

Fig. 1 (color online) HPLC chromatogram of DOTATATE and Ga-DOTA-TATE.

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3 - 16 Characterization of Thorium Targets and the Irradiation Experiments in HIRFL

Tan Cunmin, Wang Jieru, Tan Wei, Huang Qinggang and Qin Zhi

With the rapid development of national economy today, radioisotopes have been widely used in the field of agriculture, industry and national defense. Especially in medical diagnosis and treatment, they play a most important role. Many laboratories have invested substantial funding in the development production routes of radioisotopes, but these efforts have yet to support the use of medical-isotope drugs for so many cancer patients.

As a 100% alpha decay radioisotope, actinium-225 (^{225}Ac) has a very suitable half-life ($T_{1/2}$ = 10.0 d) matching targeted therapy, an average energy of 5.8 MeV and a decay chain that contains four alpha- and two beta-emissions and no long-lived radioactive progeny[1]. Moreover, ^{225}Ac is a parent nuclide for ^{213}Bi ($T_{1/2}$ = 45.7 min) in a ^{225}Ac/^{213}Bi generator. It is noteworthy that ^{213}Bi decays with 440 keV of characteristic gamma rays, so the gamma ray is suitable for SPECT imaging at the same time as treatment[2]. Besides, based on the improvement of the controllable performance of the nuclide chelating group, a new concept of ^{225}Ac/^{213}Bi in vivo generator is be proposed by scientists[3, 4].

However, the international supply of ^{225}Ac comes entirely from the existing ^{229}Th ($T_{1/2}$ = 7 340 y) in a total quantity of about 1.7 Ci y^{-1}. So it cannot meet the increasing demand in clinical research and cancer treatment at all. At present, this isotope has also been called “the rarest drug on the planet”. From the perspective of economy and technology, it is not feasible to increase ^{229}Th production by ^{233}U. The mainstream of production of ^{225}Ac is the method of irradiating ^{232}Th target by high-energy proton beam. However, only a few accelerator of national laboratories in Europe and America can meet the production conditions and carry out some relevant works[5–7].

HIRFL (Institute of Modern Physics, Chinese Academy of Sciences) is a heavy ion research facility. The typical H∼U ions can be accelerated to the energy of 100∼9.5 MeV/u, respectively. It can meet the preliminary research of irradiating ^{232}Th target by high-energy proton beam. This energy value can improve the cross section of ^{225}Ac observably. The equipment will be the first choice platform for the production of ^{225}Ac by high-energy proton irradiation ^{232}Th target in China.