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Solution-Reactor-Produced Mo-99 Using Activated Carbon to Remove I-131

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Introduction

The current production and supply of molybdenum-99, utilized by the United States, is that which comes from Nordion International of Canada. The demand for molybdenum-99, which decays to the medically useful radioisotope technetium-99m, continues to grow. This growth is of concern not only because of the aging reactor that is used to produce the molybdenum-99 but also because of environmental concerns related to the processing of irradiated targets.

This research explores the idea of producing molybdenum-99 in a solution reactor. The Solution High Energy Burst Assembly (SHEBA), located at the Los Alamos Critical Assembly Facility was used to facilitate this study. The goal of this study was to build on work previously completed and to investigate a possible mode of radioactive contaminant removal prior to a molybdenum-99 extraction process. Prior experiments, performed using SHEBA and a single-step sorption process, showed a significant amount of iodine-131 present along with the molybdenum-99 on the alumina that was used to isolate the molybdenum-99. A high concentration of iodine-131 and/or other contaminants present in a sample prohibits the FDA from approving an extraction of that nature for radiopharmaceutical use. However, if it were possible to remove the iodine-131 and other contaminants prior to a molybdenum-99 extraction, a simple column extraction process might be feasible.

Description of Work

Activated charcoal was used to try filtering the iodine-131 contaminant from an irradiated fuel solution. The fuel that was used in this experiment was 20%-enriched

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uranyl nitrate. Approximately 400 mL of the fuel was loaded into a 2-inch diameter stainless steel cylinder which was lowered into the 3-inch experiment port in SHEBA. The experiment port allows the sample to sit down inside the critical assembly vessel for irradiation. The sample was irradiated with 1.73E17 fissions and nine days later the extraction was performed. A Whatman Carbon Cap-75 was used as the activated carbon source. The irradiated uranyl nitrate in the sample cylinder was emptied through the filter and into a flask. The filtration system was pulled through a vacuum and the set-up included a HEPA filter to trap any particles that may have been present (figure 1). A sample of the irradiated fuel was taken prior to filtration. About half of the solution was filtered and a sample of the filtrate was captured from the flask. The filtration was started up again, followed by another sampling of the filtrate. This second filtrate sample was taken directly from the eluant which was coming from the bottom of the carbon filter. From the beginning to end of the experiment, images were taken with a gamma camera to show the movement of the irradiated fuel through the filtration system. The three fuel samples were analyzed via gamma spectroscopy.

Results

Gamma spectroscopy confirmed that the activated carbon trapped a significant amount of the iodine-131, as well as notable amounts of xenon-133, rubidium-105, and barium-140. Most importantly, the carbon traps a diminutive amount of molybdenum-99. The loss of Mo-99 ranged from 10% in the first filtered sample to 38% in the second filtered sample. The increase in Mo-99 absorption is most likely due to either the halt in the experiment half-way through the experiment or possibly to a decreased flow rate in the second half. This experiment should be repeated in order to investigate fully the optimal parameters of a filtration system of this nature.

Tellurium-132 was also present in significant amounts in the filtrate. Te-132 was shown in previous experiments, performed at LANL, to bind with alumina as a

contaminant along with the radioisotope of interest, Mo-99. The chemical separation of Te-132 and Mo-99 should be investigated.

These results are significant because they promote the idea of solution-reactor-produced Mo-99. Solution reactors are much more favorable energetically as well as environmentally. Wastes generated from solution reactor produced Mo-99 are three to four times less than that of current target-irradiation production methods. In order to meet the medical communities 15,000 Ci/wk Mo-99 demands, it would take 1.5 g of U-235 by way of a solution reactor and 21.1 g of U-235 to produce the same amount of Mo-99 by target irradiation methods.[1] A solution reactor could provide enough Mo-99/Tc-99m to support both the current and future radiopharmaceutical needs of the United States.

Reference

1. Glenn, Daniel E., Feasibilty Study for the Utilization of Solution Reactor Production of Molybdenum-99, Masters Thesis, University of New Mexico, December 1995.

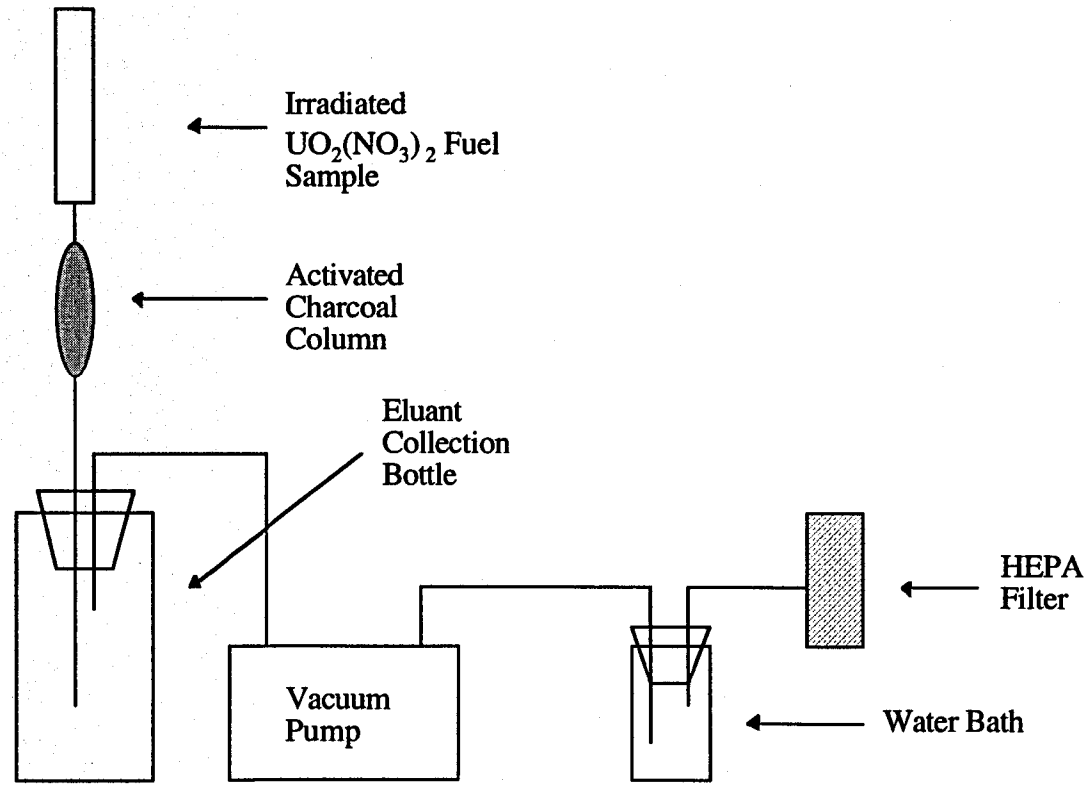

fig.1 Charcoal Filtration Apparatus