Sensitivity Analysis of Molybdenum-99 Production in Oregon State TRIGA Reactor

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INTRODUCTION

Molybdenum-99 (Mo-99) is used throughout the world in technetium-99m generators for medical applications. With the recent unplanned production halt in Canada about two years ago and the planned outage in a Dutch molybdenum-producing reactor, a shortage of the isotope materialized. This shortage has caused many diagnostic imaging procedures to be postponed or cancelled. Alternative domestic ^{99}Mo production options are being considered including irradiation of fissile targets in small research reactors such at the Oregon State University TRIGA Reactor (OSTR).

PRODUCTION OF ^{99}MO IN THE OREGON STATE TRIGA REACTOR

Producing ^{99}Mo in the OSTR involves placing target fuel elements containing low-enriched uranium in the outermost regions of core. ^{99}Mo is produced as a fission product, and is chemically separated from the remaining fuel and other fission products. The traditional Centichem target design is an aluminum cylinder lined with a metallic uranium fuel material (Fig 1a). In this research, we analyzed the impact of new target designs involving concentric cylindrical regions (Fig 1b) on ^{99}Mo production. The effects of changing the target fuel masses, the target thickness, and the radius of the annular regions were also studied. In addition to the changes in geometry, we also investigated the impact of various materials in the annular design on ^{99}Mo production.

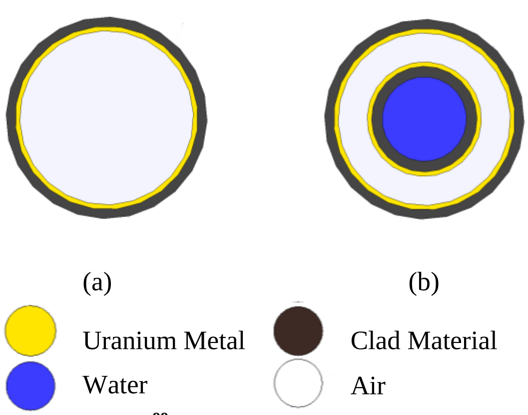

Figure 1: ^{99}Mo Target Designs and Material Compositions

Initially, the traditional Centichem design was altered by changing the clad material to beryllium and adding an inner beryllium slug. Beryllium as a structural material would be beneficial as it undergoes an (n,2n) reaction. The expectation was that that this design would produce the most ^{99}Mo given a specified irradiation time, despite its high material costs.

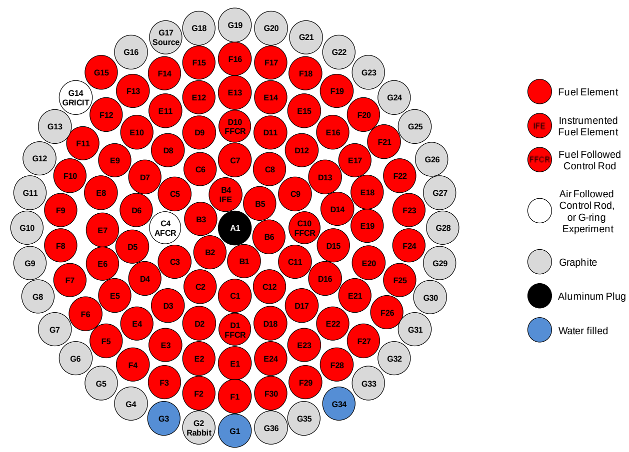

Figure 2: Simulated OSTR Core Configuration. Mo-99 targets placed in graphite slots.

Filling the inner region with water was another method employed to boost the thermal neutron flux throughout a target. This provides a larger amount of moderator material inside the ^{99}Mo target, immediately thermalizing neutrons produced within the target, and significantly increasing the rate of fission reactions in the uranium metal. These design alterations allow more target material to be added to each target element, theoretically producing more ^{99}Mo.

The active height of the target elements is equivalent to that of the active height of the fuel rods in the OSTR, 38.1 cm. The clad thickness remained constant at .0508 cm, standard in both fuel elements and ^{99}Mo targets. The thickness of the uranium metal target in design (a) was varied as a function of mass, increasing from 100g to 1000g of 20% enriched uranium metal in 100g increments. Design (b) was also varied in 100g increments, with the inner and outer targets having an equal thickness. Thus, the outer annulus holds a larger fraction of the target fuel mass.

MCNP5 Simulation of ^{99}Mo Production

MCNP5 was used to perform neutron transport simulations of ^{99}Mo production in the OSTR. The core was arranged in a configuration similar to that of the current OSTR (Fig. 2) operating core. The target elements were placed in the outermost ring of the reactor in all positions that would have graphite elements in the standard OSTR core. Despite the higher fluxes that exist in the center of the core, placing these target elements in the outer ring ensures that the fundamental safety-related characteristics of the TRIGA reactor are maintained.

In each simulation, all available sites in the outer ring are occupied by ^{99}Mo target elements. This allows an assessment of the most optimal locations for ^{99}Mo production. However, operational requirements on core excess and shutdown margin would likely preclude such a configuration.

RESULTS

Both the power per element and the ^{99}Mo produced per element were analyzed for each target design, assuming a nominal reactor power of 1.0 MWth. Figure 3 is a plot of relative ^{99}Mo production in designs (a) and (b), with various target masses and clad materials. The beryllium clad design (b) produces the most ^{99}Mo per element, effectively doubling the production of a comparable mass Centichem design. However, the stainless steel clad element, with an inner annulus of water, still increases the ^{99}Mo production by a factor of 1.5 compared to an equivalent mass Centichem design.

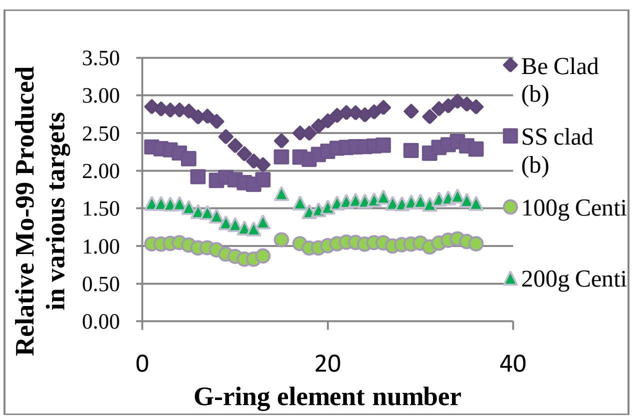

Figure 3: Comparison of Mo-99 Production in Variations of Targets (a) and (b)

Other material factors must also be considered when designing a target element. Figure 4 is a plot of relative ^{99}Mo produced in design (b) as a function of increasing target mass. The ^{99}Mo produced does not increase linearly with mass due to self-shielding in the target uranium. As a result, the finalized target design must balance overall ^{99}Mo produced with an economized fuel use.

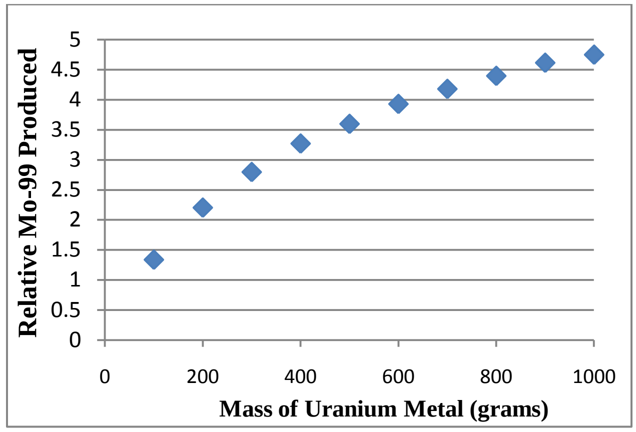

Figure 4: Effects of Variable Mass on ^{99}Mo Produced in Target Design B with Stainless Steel Clad Material

Initial results generally support the change from the classic Centichem target design to a design involving concentric cylinders. Adding the inner water-filled annulus to the original target element also appears to be a design improvement, as an increase in thermal neutron flux increases the rate of ^{99}Mo production, reducing the time needed to obtain the desired amounts of ^{99}Mo. With additional design considerations, such as cost of materials and time, the ^{99}Mo targets can be altered to provide a safe, reliable domestic means of producing ^{99}Mo in a small research reactor.

REFERENCES

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