A FAST RADIOCHEMICAL METHOD FOR THE ^{99m}Tc PRODUCTION BY NEUTRON ACTIVATION AND ITS COST ANALYSIS

A. Salvini, Massimo Oddone, A. Borio di Tigliole, S. Manera, Ezio Previtali, Massimiliano Clemenza, M. Prata, Alessandra Terulla

Laboratorio Energia Nucleare Applicata (L.E.N.A.)
University of Pavia - Italy

3rd - International Nuclear Chemistry Congress -18 - 23 September 2011-Palermo

1.1 Introduction

The World needing of ^{99}Mo is growing and recent publications estimate that about 12.000 6-days Ci for week of this isotope are used.

The actual production is by fission mainly with high enrich Uranium targets and supply chain overly dependent on only five research reactors and four processing facilities.

The aging of the plants and the conversion of the target to low enrichment for facing the proliferation of nuclear weapons makes the situation worse

1.2 Introduction

In recognition of this sustained crisis, many international efforts have been initiated to overcome related challenges and identify options to prevent a recurrence of this problem in the future.

The scientific community is called to work on developing new production methods and alternatives in diagnostic techniques.

Aim of this work is to investigate and test the possibility to use small research reactors as the TRIGA MKII installed in University of Pavia at L.E.N.A. (Laboratory Nuclear Energy Applied) for local ^{99}Mo production

1.1 Facilities

The research was performed using vertical channels of a TRIGA type reactor installed at L.E.N.A. Laboratory.

Neutrons Energy	Energy Range MeV	Neutrons Flux n/cm^2s	Westcott Flux n/cm^2s
Thermal	$< 0{,}55\ 10^{-6}$	$1{,}28\ 10^{12}$	
Epithermal	$0{,}55\ 10^{-6} – 0{,}1$	$1{,}04\ 10^{12}$	
Fast	$> 0{,}1$	$7{,}13\ 10^{11}$	
Total		$3{,}04\ 10^{12}$	$9{,}2\ 10^{12}$

Rotary specimen rack flux (uncertain 10%)

Neutrons Energy	Energy Range MeV	Neutrons Flux n/cm^2s	Westcott Flux n/cm^2s
Thermal	$< 0{,}55\ 10^{-6}$	$7{,}26\ 10^{12}$	
Epithermal	$0{,}55\ 10^{-6} – 0{,}1$	$7{,}52\ 10^{12}$	
Fast	$> 0{,}1$	$6{,}78\ 10^{12}$	
Total		$2{,}16\ 10^{13}$	$5{,}4\ 10^{12}$

Central thimble flux (uncertain 10%)

1.1 Irradiations

Irradiation times were chosen between two to twelve hours (6+6 in two days) at 2,5kW using each time about 500mg of MoO_3 in powder, sintered or compressed.

Irradiations were repeated at 250KW with 100mg of Molybdenum oxide 98% enriched in ^{98}Mo for 30 minutes.

These parameters were decided considering the manipulation of about few kBq of ^{99}Mo for radioprotection issue and in order to perform a good determination by gamma spectroscopy in good geometry.

1.1 Analysis

Analyses were made using HPGe coaxial detector with its electronics calibrated in efficiency with equivalent geometry reference.

The cooling time was enough for reach the equilibrium and the gamma rays of interest were 140,5KeV (^{99m}Tc) and 739,5KeV (^{99}Mo) with Branching Ratio of respectively 0,89 and 0,12.

Results are extrapolated to nominal power and activity is given at the saturation.

1.2 Analysis

Before the experimental test an evaluation of the ^{99}Mo production was made by simulation with MCNP4B.

Analyses were performed also after separation by solvent extraction.

Samples experimental data	Specific activity of ^{99}Mo at saturation and 250kW GBq/gMo
Compressed solid	4,5 ±0,4
Sintered sample	3,7 ± 0,3

Specific activity of ^{99}Mo for compressed and sintered samples extrapolated at saturation and full power

1.3 Analysis

Samples experimental data	Specific activity of ^{99m}Tc at saturation and 250KW GBq/gMo
Compressed solid	4,0 ±0,4
Sintered sample	3,3 ± 0,3

Specific activity of ^{99m}Tc for compressed and sintered samples extrapolated at saturation and full power

MCNP Samples Simulation	Specific activity of ^{99}Mo at saturation and 250KW GBq/gMo
Central thimble	5,3 ±0,5

Specific activity of ^{99}Mo obtained by MCNP simulation at saturation and full power

MCNP Samples Simulation	Specific activity of ^{99}Mo at saturation and 250KW GBq/g^{98}Mo
Central thimble	17,8 ±2

Specific activity of ^{99}Mo obtained by MCNP simulation at saturation and full power with ^{98}Mo enrich target

1.1 Separation

The choice of solvent extraction separation is made in the light of technical, economic and logistic factors; in this case considering the relativity low neutron flux the solvent extraction method can help to reach the specific activity of ^{99}Mo wonted.

The samples irradiated (about 100mg) are dissolved in 50ml of 6M sodium hydroxide on hot plate.
One ml of 3% peroxide is added to oxidize all Mo present.

Then using 100ml of Methyl-ethyl ketone (MEK) in three times the ^{99m}Tc is separated by the ^{99}Mo using a separation funnel.

After this separation the ^{99m}Tc present in organic fraction could be evaporated and recover with sterile isotonic solution at specific activity wonted.

1.2 Separation

Without any automatic system the removed yield of ^{99}Mo was 99,4% and final yield was 80,3%.

The tests were repeat with the ^{98}Mo peroxide enriched in order to verify the possibility of recover the target for new irradiation by water fraction.

No other artificial nuclides are detected by gamma spectroscopy in ^{99m}Tc fractions.

1.1 Cost Analysis

The cost of production is calculated considering the following assumptions:

- Two irradiation of 12 hours for week
- Reactor availability 44 weeks
- Fill the 100cm^3 volume of the central thimble with 45g of Mo oxide enrich target (4,5g/cm^3)
- Daily extractions of 10Ci of ^{99m}Tc activity.
- Daily deliver of the product
- Recover the target for 44 times with 10% of renew
- Each patient needs 20mCi

1.1 Cost Analysis

	Cost evaluated
Annual irradiations	1.050.000€
Target for year	715.000€
Reagents for year	13.200€
Fuel for yaer	15.000€

The evaluation of the final cost for patient is about 16€

Delivery cost are not consider in this table because depends on specific condition; but it is possible in any case consider about 80€ only for case of one deliver at few Kilometers from the laboratory of one batch

1.1 Results and Discussion

The method described is easy and fast.

It is possible to develop one automatic extraction based on this procedures and the final product is in sodium pertechnetate form which is good for preparation in diagnostic work.

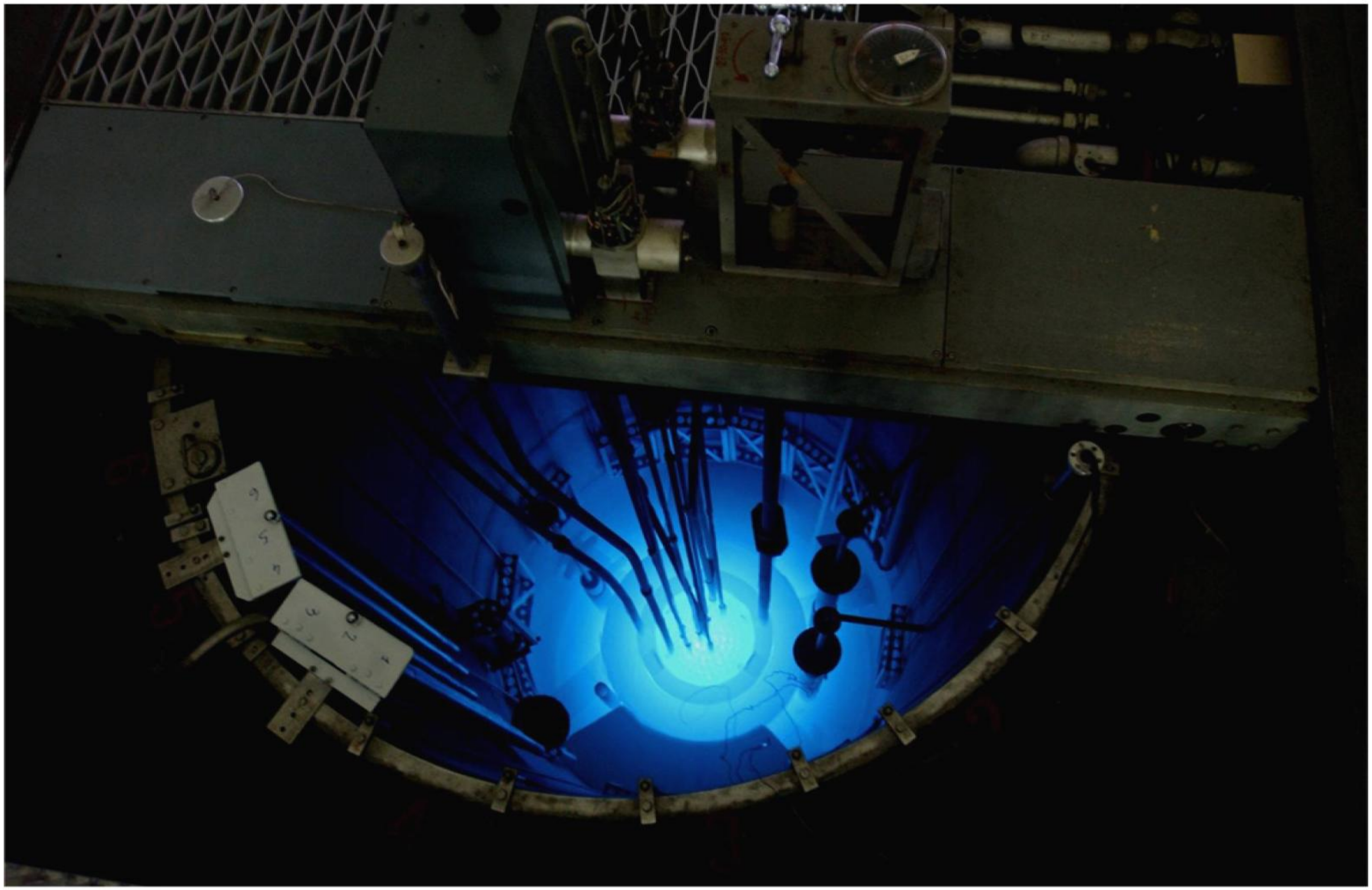
The delivery cost could be shear with ^{18}F distribution network.

1.1 Results and Discussion

The Final product by activation has radionuclide purity better than fission product.

Nowadays the cost of ^{99m}Tc by this activation method is greater than commercial products, but in case of lack of world production a small research reactor can easy effort the needing of near Hospitals.

TRIGA Mark II Reactor



LENA Reactor Core

END