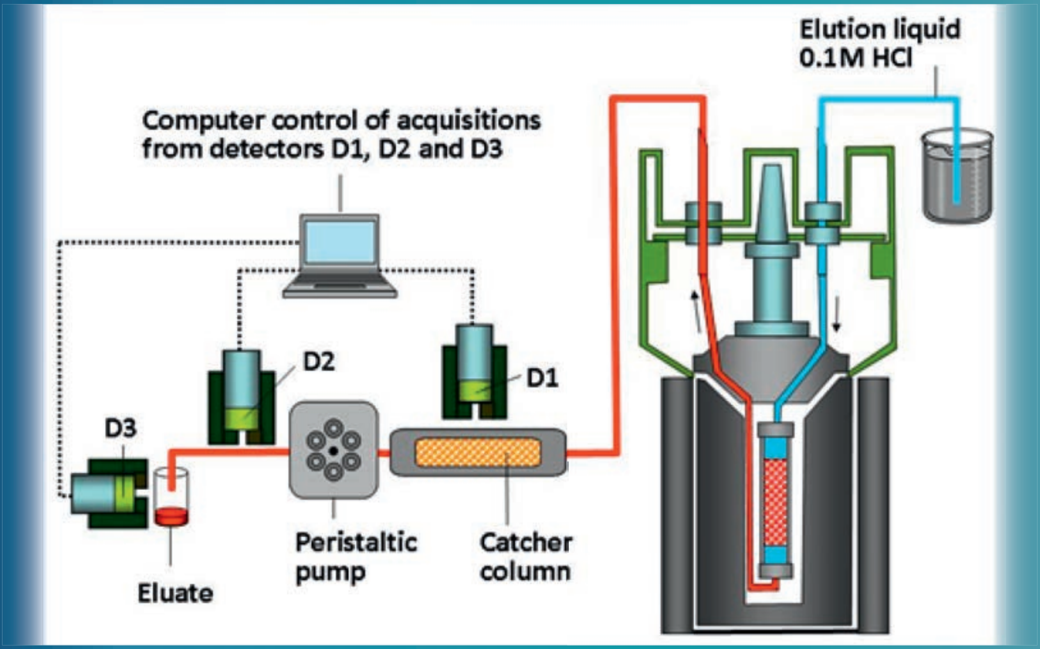
IAEA RADIATION TECHNOLOGY SERIES No. 5



Radiotracer Generators for Industrial Applications

IAEA
International Atomic Energy Agency

IAEA RADIATION TECHNOLOGY SERIES PUBLICATIONS

One of the main objectives of the IAEA Radioisotope Production and Radiation Technology programme is to enhance the expertise and capability of IAEA Member States in utilizing the methodologies for radiation processing, compositional analysis and industrial applications of radioisotope techniques in order to meet national needs as well as to assimilate new developments for improving industrial process efficiency and safety, development and characterization of value-added products, and treatment of pollutants/hazardous materials.

Publications in the IAEA Radiation Technology Series provide information in the areas of: radiation processing and characterization of materials using ionizing radiation, and industrial applications of radiotracers, sealed sources and non-destructive testing. The publications have a broad readership and are aimed at meeting the needs of scientists, engineers, researchers, teachers and students, laboratory professionals, and instructors. International experts assist the IAEA Secretariat in drafting and reviewing these publications. Some of the publications in this series may also be endorsed or co-sponsored by international organizations and professional societies active in the relevant fields.

There are two categories of publications: the **IAEA Radiation Technology Series and the IAEA Radiation Technology Reports**.

IAEA RADIATION TECHNOLOGY SERIES

Publications in this category present guidance information or methodologies and analyses of long term validity, for example protocols, guidelines, codes, standards, quality assurance manuals, best practices and high level technological and educational material.

IAEA RADIATION TECHNOLOGY REPORTS

In this category, publications complement information published in the IAEA Radiation Technology Series in the areas of: radiation processing of materials using ionizing radiation, and industrial applications of radiotracers, sealed sources and NDT. These publications include reports on current issues and activities such as technical meetings, the results of IAEA coordinated research projects, interim reports on IAEA projects, and educational material compiled for IAEA training courses dealing with radioisotope and radiopharmaceutical related subjects. In some cases, these reports may provide supporting material relating to publications issued in the IAEA Radiation Technology Series.

All of these publications can be downloaded cost free from the IAEA web site:

http://www.iaea.org/Publications/index.html

Further information is available from:

Marketing and Sales Unit
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria

Readers are invited to provide feedback to the IAEA on these publications. Information may be provided through the IAEA web site, by mail at the address given above, or by email to:

Official.Mail@iaea.org

RADIOTRACER GENERATORS FOR INDUSTRIAL APPLICATIONS

The following States are Members of the International Atomic Energy Agency:

AFGHANISTAN
ALBANIA
ALGERIA
ANGOLA
ARGENTINA
ARMENIA
AUSTRALIA
AUSTRIA
AZERBAIJAN
BAHRAIN
BANGLADESH
BELARUS
BELGIUM
BELIZE
BENIN
BOLIVIA
BOSNIA AND HERZEGOVINA
BOTSWANA
BRAZIL
BULGARIA
BURKINA FASO
BURUNDI
CAMBODIA
CAMEROON
CANADA
CENTRAL AFRICAN REPUBLIC
CHAD
CHILE
CHINA
COLOMBIA
CONGO
COSTA RICA
CÔTE D'IVOIRE
CROATIA
CUBA
CYPRUS
CZECH REPUBLIC
DEMOCRATIC REPUBLIC OF THE CONGO
DENMARK
DOMINICA
DOMINICAN REPUBLIC
ECUADOR
EGYPT
EL SALVADOR
ERITREA
ESTONIA
ETHIOPIA
FIJI
FINLAND
FRANCE
GABON
GEORGIA
GERMANY
GHANA
GREECE
GUATEMALA
HAITI
HOLY SEE
HONDURAS
HUNGARY
ICELAND
INDIA
INDONESIA
IRAN, ISLAMIC REPUBLIC OF
IRAQ
IRELAND
ISRAEL
ITALY
JAMAICA
JAPAN
JORDAN
KAZAKHSTAN
KENYA
KOREA, REPUBLIC OF
KUWAIT
KYRGYZSTAN
LAO PEOPLE'S DEMOCRATIC REPUBLIC
LATVIA
LEBANON
LESOTHO
LIBERIA
LIBYA
LIECHTENSTEIN
LITHUANIA
LUXEMBOURG
MADAGASCAR
MALAWI
MALAYSIA
MALI
MALTA
MARSHALL ISLANDS
MAURITANIA
MAURITIUS
MEXICO
MONACO
MONGOLIA
MONTENEGRO
MOROCCO
MOZAMBIQUE
MYANMAR
NAMIBIA
NEPAL
NETHERLANDS
NEW ZEALAND
NICARAGUA
NIGER
NIGERIA
NORWAY
OMAN
PAKISTAN
PALAU
PANAMA
PAPUA NEW GUINEA
PARAGUAY
PERU
PHILIPPINES
POLAND
PORTUGAL
QATAR
REPUBLIC OF MOLDOVA
ROMANIA
RUSSIAN FEDERATION
RWANDA
SAUDI ARABIA
SENEGAL
SERBIA
SEYCHELLES
SIERRA LEONE
SINGAPORE
SLOVAKIA
SLOVENIA
SOUTH AFRICA
SPAIN
SRI LANKA
SUDAN
SWEDEN
SWAZILAND
SWITZERLAND
SYRIAN ARAB REPUBLIC
TAJIKISTAN
THAILAND
THE FORMER YUGOSLAV REPUBLIC OF MACEDONIA
TOGO
TRINIDAD AND TOBAGO
TUNISIA
TURKEY
UGANDA
UKRAINE
UNITED ARAB EMIRATES
UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND
UNITED REPUBLIC OF TANZANIA
UNITED STATES OF AMERICA
URUGUAY
UZBEKISTAN
VENEZUELA
VIETNAM
YEMEN
ZAMBIA
ZIMBABWE

The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

IAEA RADIATION TECHNOLOGY SERIES No. 5

RADIOTRACER GENERATORS FOR INDUSTRIAL APPLICATIONS

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2013

COPYRIGHT NOTICE

Marketing and Sales Unit, Publishing Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
fax: +43 1 2600 29302
tel.: +43 1 2600 22417
email: sales.publications@iaea.org
http://www.iaea.org/books

Printed by the IAEA in Austria
April 2013
STI/PUB/1579

IAEA Library Cataloguing in Publication Data

Radiotracer generators for industrial applications. — Vienna : International Atomic Energy Agency, 2013.
p. ; 24 cm. — (IAEA radiation technology series, ISSN 2220–7341 ; no. 5)
STI/PUB/1579
ISBN 978–92–0–135410–5
Includes bibliographical references.

1. Radioisotopes — Industrial applications. 2. Radioactive tracers. I. International Atomic Energy Agency. II. Series.

IAEAL 13–00797

FOREWORD

Radiotracers have been widely used throughout industry to optimize processes, solve problems, improve product quality, save energy and reduce pollution. Their technical, economic and environmental benefits have been recognized by both the industrial and the environmental sectors. The most important radiotracer techniques have been transferred to many developing Member States through IAEA Technical Cooperation projects.

However, in spite of their manifest benefits, radiotracer techniques continue to be underutilized, not only by developing countries but also by more industrialized nations. There are a number of factors that restrict the usage of the radioisotope techniques, but chief among them is the timely availability of suitable radiotracers. Ensuring timely availability of suitable radionuclides is a main hurdle to the use of radiotracer techniques in industry. For developing countries that do not possess radioisotope production facilities, the long time required for import of radionuclides not only completely rules out the use of short half-life nuclides, but also makes it impossible for the radioisotope applications teams to respond to problems of an urgent nature. Many possible radiotracer investigations are not being carried out in developing countries because of this problem.

Even in industrialized countries, radionuclide supply is often a problem, as many of the former suppliers of industrial radionuclides have switched their production facilities to serve the more lucrative radiopharmaceuticals market. Obtaining continuity of supply of radionuclides with which to carry out extended studies in difficult-to-access locations, such as offshore oil platforms, is also a significant challenge.

Making use of tracers from radionuclide generators can alleviate the difficulties associated with radioisotope supply. Two commercially available medical radionuclide generators, ^{99}Mo/^{99m}Tc and ^{113}Sn/^{113m}In, have been used for this purpose, but their use has been restricted because their half-lives are relatively short and only a limited number of radiotracers derived from them have been proven under industrial conditions.

Recognizing the potential usefulness of several radionuclide generators, including ^{68}Ge/^{68}Ga and ^{137}Cs/^{137m}Ba and of radiotracers derived from them, the Agency organized a Coordinated Research Project (CRP) on the Evaluation and Validation of Radionuclide Generator-based Radiotracers for Industrial Applications. The objectives of the CRP, which was implemented over the period 2007–2011, was to coordinate the development of industrial radionuclide generators and to validate generator based radiotracers for use in harsh industrial conditions. It was anticipated that an important outcome would be the production of radionuclide generators designed specifically for industrial purposes. Their cost would be considerably lower than those of medical generators since the high costs of the facilities required to produce generators suitable for in vivo use can be avoided.

The IAEA facilitates the transfer of technology, and an important part of this process is the provision of relevant literature that may be used for reference purposes or as an aid to teaching. This monograph contains guidelines for the selection and preparation of radiotracers derived from radionuclide generators, as well as reports from participants of the CRP. The book has been prepared with contributions from all CRP participants. The IAEA wishes to express thanks to all the contributors and especially to T. Bjørnstad and T.S. Charlton for their contributions to this publication. The IAEA officers responsible for this publication were Joon-Ha Jin and P. Brisset of the Division of Physical and Chemical Sciences.

CONTENTS

1. INTRODUCTION

Radiotracer technology is used in industry to optimize processes, solve problems, improve product quality, save energy and reduce pollution. Creative application of radiotracer methodology may contribute to the economic development and technological advancement of all countries. The evolution of radiotracer technologies has been influenced by enormous developments in radiotracer preparation techniques, production of a wide range of radionuclides of varying half-lives, radiation detection techniques, data processing and visualization practices. Over the years, the IAEA has contributed to the development of radiotracer technology and its transfer to developing Member States. Significant progress has been made, enabling the countries to establish national radiotracer groups, and the most important industrial radiotracer techniques have been transferred to many developing Member States through IAEA Technical Cooperation projects.

The rapid progress in radiotracer investigations realized over the past two decades has stimulated broad interest in the use of short lived gamma emitting radionuclides, as many industrial radiotracer applications typically require tracing the radioactivity for a few minutes to several days. In some cases where gas transport is involved, transit times down to seconds must be measured. However, tracers based on long lived radionuclides have often been used in such applications owing to the difficulty and expense involved in obtaining shorter lived radiotracers. Consequently, at the completion of the tests, residual radioactive nuclides are discharged into commercial products or into environmental waters, or they must be stored safely until the radioactivity dies away. Although the quantities of discharged radionuclides are small (typically less than megabecquerel amounts), government regulations must nevertheless be strictly adhered to. It would be much more satisfactory to use shorter lived radiotracers whenever possible.

In cases where particularly high radioactivity is required, the use of short lived radionuclides is a distinct improvement on the use of long lived radionuclides. Because they decay quickly, they are more acceptable in these days of concern over radioactive contamination of the environment. Additionally, by using short lived tracers, it is possible to repeat measurements at intervals without having to worry about the activity remaining in the system from the previous test.

Although the potential of short half-life gamma emitting radionuclides for industrial tracer investigations is high, there are practical problems associated with their use. Essentially, they can be used only in locations close to their places of manufacture, since their short half-life limits the transportation time and thus the shipping distances. Timely availability of short lived radiotracers at the investigation site is one of the main hurdles that prevent the widespread utilization of short lived radioactive tracers in industry.

Industrial radiotracer generators (IRTGs), based on industrial radionuclide generators (IRNGs), could be an attractive option for obtaining the desired radiotracer, at or in close proximity to the industrial investigation site, as required. Such generators remove the dependence on proximity to radioisotope production facilities. Another advantage is the ease with which the daughter product (short lived radionuclides) can be separated on demand and converted into a radiotracer by some tailor-made labelling procedure on-site. Radionuclide generators also represent a cost effective strategy to ensure a continuous and reliable supply of short lived radionuclides for radiotracer investigations. This facilitates work at remote locations and also makes possible the provision of a rapid response service for problems of an urgent nature.

Radionuclide generators have been in existence for over 50 years, and are widely used for the production of radiopharmaceuticals. In spite of their extensive use in hospitals over the years, commercially available medical radionuclide generators have limited applications in industrial and environmental radiotracer studies because of their relatively short shelf lives and the relatively low energies of the gamma rays from their daughters.

Additionally, concern has also been expressed regarding their use under harsh industrial and environmental conditions where the effects of the temperature, pressure, pH and chemical composition of the process streams may impact adversely on their performance. These potential weaknesses substantially limit the industrial applicability of the radiotracers produced from the generators used in nuclear medicine.

Large scale use of short lived radionuclides in radiotracer application requires that IRNGs should be available. The issues that need to be addressed in fabricating IRNGs are different from those associated with the manufacture of medical generators. In the development of IRNGs, it is essential to utilize mother radionuclides with long physical half-lives of months or years to meet problems related to the logistics of storage and transportation. The production of the radionuclide generator, selection/development of sorbent material, construction of the generator system, devising a chemical method for the separation of the daughter radionuclide from its mother radionuclide and the post-elution chemical manipulation of the daughter radionuclide are factors that should be addressed adequately while developing IRNGs and IRTGs. Furthermore, IRNGs and IRTGs must meet regulatory and quality control requirements. Though several forms of radionuclide generator systems for medical applications have been reported in the literature, the column chromatographic system is most suitable for use in industrial applications as it can readily be adapted for use in harsh environments.

It is anticipated that IRNGs will be inexpensive in comparison to medical radionuclide generators, as there is no need to adhere to the stringent pharmaceutical regulatory requirements. A major deterrent to the use of IRNGs at the present time is their commercial unavailability. The commercial production of IRNGs is perceived to be economically unattractive, as the numbers that are likely to be required are relatively small. As a result, there are very few companies around the world that currently supply IRNGs, and the number is, if anything, decreasing. A practical solution to these problems is to use the expertise and facilities of Member States to develop and validate a few selected IRNGs in order to encourage their utilization in industrial radiotracer studies.

To address the need of IRNGs for radiotracer investigation, the IAEA implemented a Coordinated Research Project (CRP) on the Evaluation and Validation of Radioisotope Generator-based Radiotracers for Industrial Applications from August 2007 to March 2011. The aim of the CRP was to create technical capabilities in developing countries for the fabrication and use of IRNGs for radiotracer application.

The production of the selected IRNGs requires basic infrastructure such as shielded processing facilities, quality control laboratories and waste handling facilities, which are generally available at national or regional facilities. To ensure wide use of the generators in industrial radiotracer applications, the physical properties of the daughter radionuclide and the chemical and physical properties of its labelled compounds should be suitable for a large variety of applications. In many cases, it is necessary to provide specific tracers for specific applications. Once produced, the radiotracer must be validated at the conditions under which it is to be used.

The specific objectives of the CRP were:

- Development of radiotracers from IRNG based radionuclides;
- Development of portable systems for production of radiotracers (IRTGs) using IRNG based radionuclides;
- Provision of guidelines for preparation and validation of IRNG based radiotracers;
- Development of techniques for enhancing the long term stability of IRNGs.

The CRP succeeded in developing a few useful IRNGs and protocols for the preparation and validation of a number of IRNG based radiotracers. Since it is advisable to minimize manual handling in the labelling process, the development of integrated and automatized labelling procedures to produce IRTGs has been a topic of interest. This technology can be utilized by Member States.

Although many mother/daughter pairs have the potential to be used as radionuclide generator systems, only a few were selected for evaluation. The selection was made on the basis of half-lives, radiation type and energy and on ease of production and labelling. This publication describes the results of research undertaken in the utilization of ^{68}Ge/^{68}Ga, ^{137}Cs/^{137m}Ba, ^{99}Mo/^{99m}Tc and ^{113}Sn/^{113m}In generators for radiotracers. This includes development of column materials, development of protocols for the elution of the daughter radionuclide, evaluation of the generators for extended periods of time, preparation of aqueous, organic and solid tracers and quality control of the radiotracers in laboratory experiments conducted under simulated industrial conditions. In addition, some attention has been given to ^{44}Ti/^{44}Sc, ^{144}Ce/^{144}Pr and ^{172}Hf/^{172}Lu generators.

This publication is the output of the work carried out at participating institutions in several Member States under the mandate of the CRP. This publication provides a unique source of information pertaining to the development and use of IRNGs, preparation of radiotracers, IRTGs, and their use in troubleshooting and optimizing industrial processes. The report will generate awareness in the radiotracer community and could be a suitable technology resource for guidance in research, development and applications.

This publication is organized into six sections. Section 1 provides an introduction and Section 2 deals with the theoretical aspects of mother/daughter nuclear relationships, radionuclide generator principles, mother/daughter separation approaches and radiotracer generators. Section 3 provides the nuclear characteristics of several IRNGs, their area of usefulness and limitations. It describes the production and processing of the parent species required for preparation of radionuclide generator systems of current interest. Section 4 provides information relating to the methodology of preparing tracers from the generator derived radionuclides, together with quality assurance guidelines. Section 5 provides general guidelines for chemical preparation of radiotracers from the generators, validation of the generators and safety aspects of their transportation and use. These guidelines should be followed as far as possible. The need for chemical manipulation of the eluted compound is also discussed. Several procedures are described, some of them being sufficiently mature to be adopted for industrial applications. Section 5 also discusses experience in the use of available IRNGs. Section 6 uses case studies to illustrate the validation of IRNGs in the laboratory and their use in the industrial environment. The case studies described in this section are concerned with typical problems in industry and the environment encountered in all countries.

The information contained herein should enable the reader to understand the routes of production, handling and quality evaluation of IRNGs and IRTGs. The references included in individual sections will be very useful to tracer groups for preparing documentation on the production and/or use of generator systems. Commercial availability of IRNGs and IRTGs will ensure timely accessibility of radiotracers based on short lived radionuclides and help Member States to realize more fully the economic benefits to be derived from radiotracer technology.

2. THEORETICAL ASPECTS

2.1. MOTHER/DAUGHTER NUCLEAR RELATIONSHIPS AND THE BATEMAN EQUATION

A nuclear mother/daughter relationship consists of a radionuclide (mother) that, upon radioactive decay, produces another, but shorter lived, daughter radionuclide (i.e. $T_{1/2}$(mother) > $T_{1/2}$(daughter)). The nuclear chart contains more than 110 mother/daughter nuclear relationships from the lightest, ^{28}Mg/^{28}Al, to the heaviest below uranium with some practical use, ^{234}Th/^{234m}Pa. Some of them may be useful for industrial purposes.

The decay process is illustrated in the schematic and simplified decay scheme in Fig. 1. The mother radionuclide is indexed by 1, the daughter radionuclide is indexed by 2 and the granddaughter indexed by 3.

In this decay scheme, $N(t)$ is the number of mother radionuclides at time $t = t$, $N(0)$ is the initial number of mother radionuclides at time $t = 0$, $D(t)$ is the

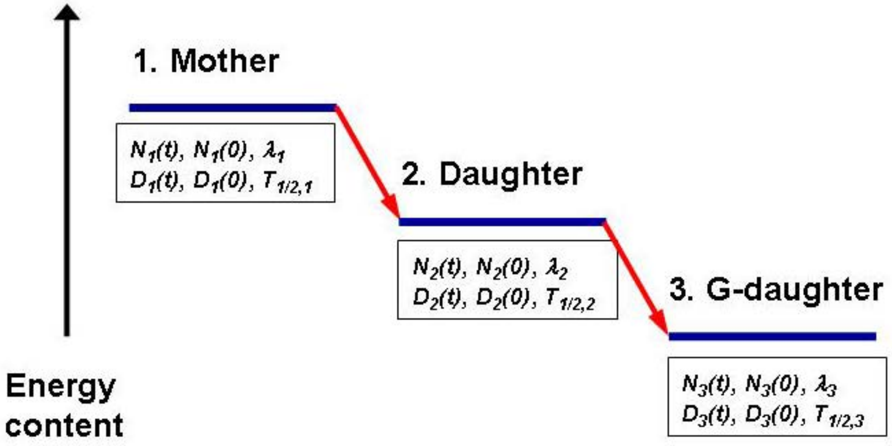


FIG. 1. Simplified decay scheme of a radionuclide generator illustrating the decay of the mother into the daughter and further into the granddaughter.

disintegration rate at time $t = t$, $D(0)$ is the disintegration rate at time $t = 0$, $T_{1/2}$ is the half-life of the nuclide and λ is the decay constant, $\lambda = \ln 2/T_{1/2}$.

The mathematical relationship between the parameters is described by the Bateman equation. The Bateman equation is not a single equation; rather, it is a method of setting up differential equations describing the decay in the chain of interest.

For the first derivation, it is supposed that $N_2(0) = 0$ and $N_3(0) = 0$ and that the granddaughter product is stable, which is the case for most useful radionuclide generators. The stability of the granddaughter is characterized by the fact that $\lambda_3 = 0$.

Then, one can set up the following differential equation for the decrease of the number of parent nuclides as a function of time:

$$\frac{\mathrm{d}N_1(t)}{\mathrm{d}t} = -\lambda_1 N_1(t) \tag{1}$$

The solution of Eq. (1) leads to the well-known decay equation of a single radioactive nuclide, i.e.:

$$N_1(t) = N_1(0)\mathrm{e}^{-\lambda_1 t} \tag{2}$$

Since the disintegration rate is defined by:

$$D_1(t) = \lambda_1 N_1(t) \tag{3}$$

it follows from Eqs (2) and (3) that:

$$D_1(t) = D_1(0)\mathrm{e}^{-\lambda_1 t} \tag{4}$$

Similarly, one may set up a differential equation for the number of daughter nuclides at any time t. Supposing that the only source of the daughter nuclide is from decay of the mother nuclide, i.e. $N_2(0) = 0$, then:

$$\frac{\mathrm{d}N_2(t)}{\mathrm{d}t} = \lambda_1 N_1(t) - \lambda_2 N_2(t) \tag{5}$$

Equation (5) describes the rate of change of the daughter nuclides that is equal to the supply by the decay of the mother and diminished by the rate of loss through its own decay. The solution of this equation results in the following expression:

$$N_2(t) = N_1(0)\frac{\lambda_1}{\lambda_2 - \lambda_1}\left(e^{-\lambda_1 t} - e^{-\lambda_2 t}\right) \tag{6}$$

In analogy with Eq. (4):

$$D_2(t) = D_1(0)\frac{\lambda_2}{\lambda_2 - \lambda_1}\left(e^{-\lambda_1 t} - e^{-\lambda_2 t}\right) \tag{7}$$

For the sake of completeness, the corresponding equations for the granddaughter will also be provided. Equation (7) describes the rate of change of the number of atoms of the granddaughter fed only by the decay of the daughter. Since the granddaughter is stable, the number of atoms is continuously accumulating:

$$\frac{dN_3(t)}{dt} = \lambda_2 N_2(t) \tag{8}$$

The number of stable granddaughter nuclides at any time *t* will then be the initial number of mother nuclides minus the remaining number of mother nuclides at time *t* minus the existing number of daughter nuclides at time *t*, expressed by:

$$N_3(t) = N_1(0) - N_1(t) - N_2(t) \tag{9}$$

Introducing Eqs (2) and (6) gives:

$$N_3(t) = N_1(0) - N_1(0)e^{-\lambda_1 t} - N_1(0)\frac{\lambda_1}{\lambda_2 - \lambda_1}\left(e^{-\lambda_1 t} - e^{-\lambda_2 t}\right) \tag{10}$$

Rearranging Eq. (10) gives:

$$N_3(t) = N_1(0)\left[1 - \frac{1}{\lambda_2 - \lambda_1}\left(\lambda_2 e^{-\lambda_1 t} - \lambda_1 e^{-\lambda_2 t}\right)\right] \tag{11}$$

For more information on these subjects, see Refs [1–3].

2.2. RADIOACTIVE EQUILIBRIUM

Equilibrium is a condition established in a mother/daughter mixture when both mother and daughter are radioactive and when the daughter's half-life is shorter than that of the mother. If the daughter's half-life exceeds that of the parent, equilibrium will never be reached. There are two types of equilibrium: secular and transient.

2.2.1. Secular equilibrium

Strict definitions of secular and transient equilibria hardly exist. However, it may be regarded that if $T_{1/2,1} >> T_{1/2,2}$, a condition of secular equilibrium exists. This means that the activity of the mother radionuclide does not decrease measurably during a reasonable period of time, i.e. subsequent elutions of the daughter with equal spacing in time will yield the same daughter activity (provided the elution efficiency stays constant). This situation for the general case is illustrated in Fig. 2.

A practical example is illustrated in Fig. 3, which shows the activities (disintegration rates) in becquerels for the mother and the daughter nuclides and their activity ratios for the generator system ^{137}Cs/$^{137\text{m}}$Ba, which is an example of secular equilibrium. Here, $T_{1/2}(^{137}\text{Cs}) = 30$ a and $T_{1/2}(^{137\text{m}}\text{Ba}) = 2.55$ min, giving a half-life ratio of $T_{1/2}(^{137}\text{Cs})/T_{1/2}(^{137\text{m}}\text{Ba}) = 6.18 \times 10^6$.

In the equilibrium mixture, the daughter appears to decay with the half-life of the parent. When the daughter is isolated from the mixture, it has its own expected half-life. Perhaps the simplest explanation for their appearing to be equal is that the daughter cannot decay until it is formed, and so the rate of formation of the daughter equals the rate of decay of the parent, which is very slow.

Thus, in secular equilibrium, mixture activity of the mother stays approximately constant while the daughter activity grows to a constant value. Correspondingly, their ratio (mother/daughter) decreases in the beginning and

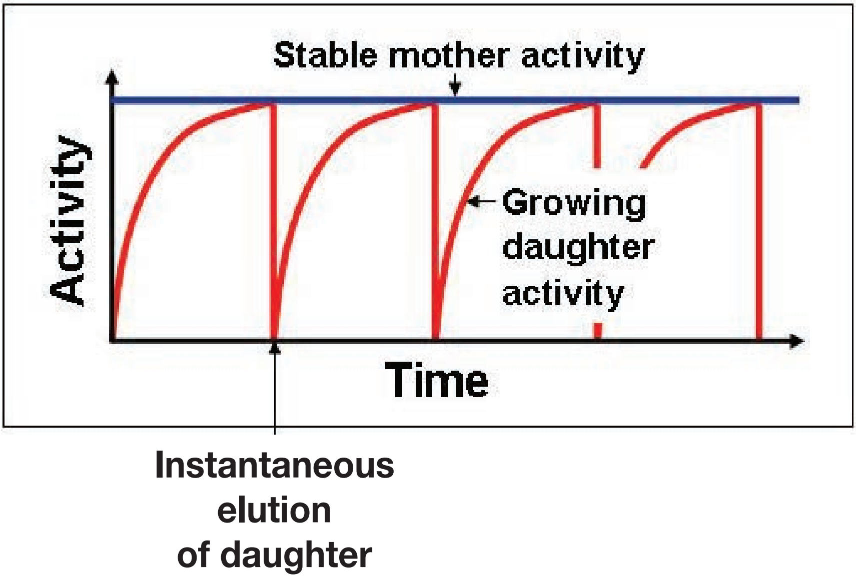


FIG. 2. Illustration of growth of daughter activity with subsequent 'instantaneous' elutions at equal time intervals in a generator system under secular equilibrium.

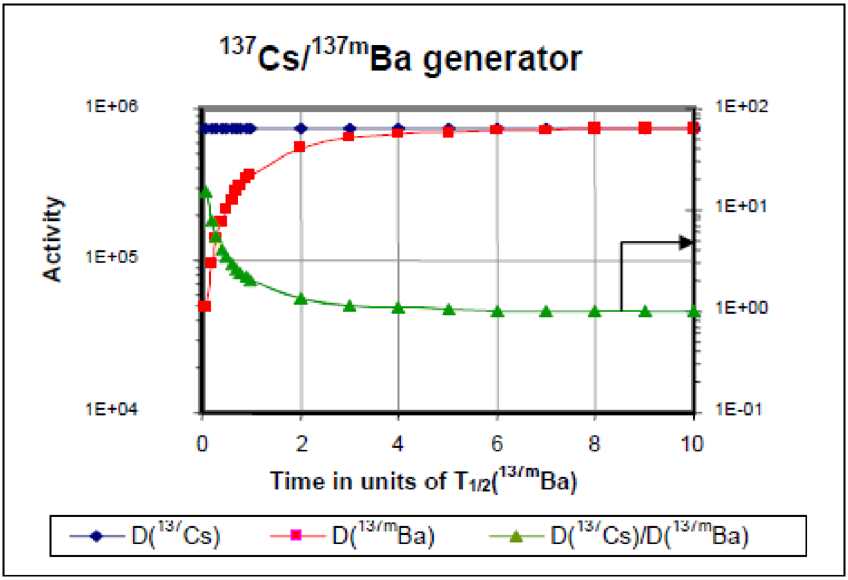


FIG. 3. Activity of mother (^{137}Cs) and daughter (^{137m}Ba) and ratio of their activities (right axis) for a ^{137}Cs/^{137m}Ba generator where the initial number of ^{137}Cs radionuclei has been set to $N_1 = 10^{15}$.

approximates a constant value after a time $t > 10T_{1/2,2}$. Often, one says that for secular equilibrium, Eq. (6) may be simplified to:

$$N_2 \approx \frac{\lambda_1}{\lambda_2 - \lambda_1} \cdot N_1^0 \cdot \mathrm{e}^{-\lambda_1 \cdot t} \tag{12}$$

since $T_{1/2,1} >> T_{1/2,2} \Rightarrow \lambda_1 << \lambda_2 \Rightarrow \exp(-\lambda_1 t) >> \exp(-\lambda_2 t)$ so that the last exponential can be neglected. This leads again to:

$$\frac{N_1}{N_2} \approx \frac{\lambda_2 - \lambda_1}{\lambda_1} \tag{13}$$

and

$$\frac{D_1(t)}{D_2(t)} \approx \frac{\lambda_2 - \lambda_1}{\lambda_2} \tag{14}$$

This expression is only strictly valid for $t > 10T_{1/2,2}$. For $t >> 10T_{1/2,2}$ and $T_{1/2,1} >> T_{1/2,2}$, λ_1 in Eq. (14) can be omitted, and the ratio of the disintegration rates is ~1.

2.2.2. Transient equilibrium

If $T_{1/2,1} \geq T_{1/2,2}$, i.e. the half-life of the mother is not several orders of magnitude larger than the half-life of the daughter and, in addition, the mother decays considerably during the period of application, then transient equilibrium is established. This situation is illustrated generally in Fig. 4.

In order to exemplify transient equilibrium, the mother/daughter relationship ^{72}Se/^{72}As is chosen.

Figure 5 illustrates the activities in becquerel for the mother (D_1) and the daughter (D_2) nuclides and their activity ratios. Here, $T_{1/2}$(^{72}Se) = 8.5 d and $T_{1/2}$(^{72}As) = 26.0 h, giving a half-life ratio of $T_{1/2}$(^{72}Se)/$T_{1/2}$(^{72}As) = 7.85.

There is a rule of thumb regarding how long it takes to reach equilibrium:

- Transient equilibrium is reached in ~$4T_{1/2,2}$;
- Secular equilibrium is reached in ~$6T_{1/2,2}$.

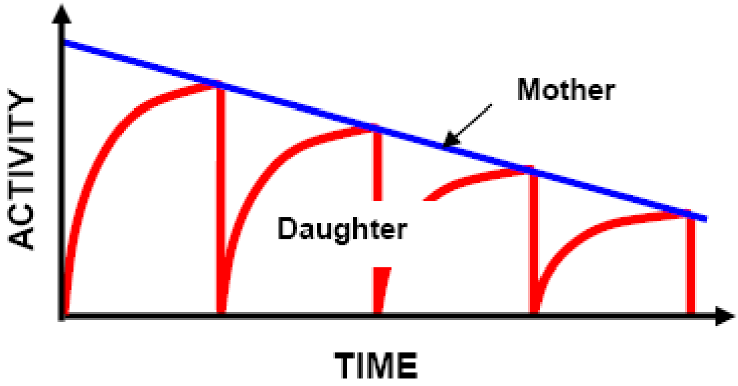


FIG. 4. Illustration of growth of daughter activity with subsequent 'instantaneous' elutions at equal time intervals in a generator system under transient equilibrium.

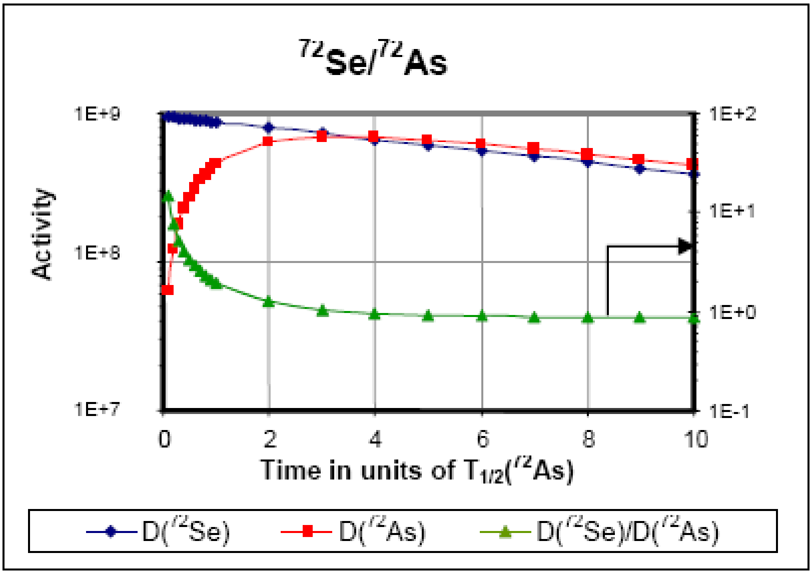


FIG. 5. Activity of mother (^{72}Se) and daughter (^{72}As) and the ratio of their activities for a ^{72}Se/^{72}As generator where the initial number of ^{72}Se radionuclei has been set to $N_1 = 10^{15}$.

The activity of the daughter radionuclide obtained per elution cycle (batch) is the main parameter for the application of a generator system. The activity of the daughter radionuclide generated at a given time can be calculated using the above equations. However, the useful amount of activity for a radiotracer experiment also depends on the elution efficiency of the generator as well as the chemical yield in the labelling process, as will be illustrated later.

It should also be mentioned here that in real applications, interest is focused mainly on the gamma decay rate of a certain gamma energy than on the total disintegration rate of the daughter radionuclide. The gamma intensity (or gamma yield) can be found in decay schemes for the individual radionuclides, for instance, in the ‘Table of Isotopes’ [4] for the different gamma energies.

Such intensities for single gamma lines of special interest following beta decay range between 50 and 100%. For β^+ decay, the intensity can reach 200% for the 511 keV annihilation radiation, here defined by the β^+ branching ratio. It is now possible to define a gamma activity, A_γ, for a special gamma line. The shape of the gamma activity line is identical to the shape of the daughter activity in Figs 2 and 4, and is derived by multiplying these activities by the proper gamma intensity.

2.3. RADIONUCLIDE GENERATOR PRINCIPLE

A radionuclide generator is a self-contained system housing a mother/daughter mixture in equilibrium and designed to produce a daughter radionuclide formed by the decay of a mother radionuclide free from contamination of the mother [5–7]. The mother/daughter nuclear relationships offer the possibility to make radionuclide generators to separate the short lived daughter at suitable time intervals. For practical reasons, most radionuclide generator systems that can be used for industrial radiotracer applications involve secular equilibrium, where the parent radionuclide has a half-life significantly longer than that of the daughter. This scenario permits long shelf life of the generator.

There must be a method of removing the daughter and leaving the mother behind to regenerate more daughter activity. The daughter radionuclide is a different chemical element to the mother, and will therefore exhibit chemical and physical properties different from those of the mother. With this difference in characteristics between mother and daughter radionuclides, the latter can usually be separated by chemical or physical means. It is said that the daughter is *eluted* from the generator (a radionuclide generator is also commonly called a ‘cow’ and the elution process ‘milking of the cow’). Once the activity of the daughter is recovered, there is a growth of the daughter activity on the generator until it again reaches equilibrium with the mother. This separation and growth can be continued as long as there are useful amounts of the parent radionuclide available, and the breakthrough of the mother stays within acceptable limits.

Thus, the useful shelf life of a radionuclide generator depends mainly on the half-life of the mother and on the radiation resistance of the generator constituents. Separation may be performed before equilibrium is reached, and the amount of daughter activity recovered will depend on the time elapsed since the last separation. In most cases, the generator can be eluted after 1–2 half-lives of the daughter; 50–75% of the equilibrium activity has then been generated.

2.3.1. Properties of an ideal IRNG

An ideal IRNG should possess the following properties [8, 9]:

- The chemical and/or physical properties of the daughter must be different from those of the mother to permit efficient separation of daughter from mother.
- The separation of the daughter radionuclide should be easy and efficient using appropriate chemical or physical techniques.
- The daughter radionuclide separation should involve no violent chemical reactions.
- Human intervention in generator operation should be minimal to minimize radiation dose.
- The daughter radionuclide to be used as a tag on tracers for radiotracer investigations should be short lived and gamma emitting.
- In any case, a gamma emitter with a high gamma branching ratio is desirable.
- The elution yield and purity of the daughter radionuclide should be within the acceptable range.
- The physical half-life of the parent should be long enough for extending the shelf life of the generator.
- The generator constituents (i.e. eventual column packing material, liquids involved, tubings and fittings) should be radiation resistant.
- The daughter chemistry should be amenable to the preparation of a wide variety of compounds for radiotracer applications (such as water tracing, oil tracing, particle tracing, etc.).
- Shielding, handling and transportation of the generator, even across national borders, should be straightforward.
- The granddaughter should be stable (or very long lived) to limit concern about site contamination, environmental persistence and waste disposal issues.

2.3.2. Additional specific criteria for an IRNG

Although many parent/daughter pairs have been evaluated as radionuclide generator systems in a clinical context, only a few generators are suitable for routine industrial and research use. While selecting parent/daughter pairs, the following criteria need to be satisfied [8, 9]:

- *Half-life of the mother radionuclide*: It is important to select a generator with an appropriately long half-life of the mother radionuclide. A generator that remains useful for a period of several months to a year or more may be appropriate for industrial applications. In some cases, it should be possible to install the generator ‘permanently’, for instance, to monitor the operations of a multiphase separator over time.
- *Half-life of the daughter radionuclide*: The daughter radionuclide should have a half-life that is compatible with the duration of the radiotracer study. Half-lives in the range of several minutes to several hours or more are useful for most fluid and particle industrial tracing operations. For cases involving tracing of gas transport, transit times of the order of seconds may be expected, and radiotracers with half-lives of the order of minutes are most useful. However, the present document has no special focus on gas monitoring.
- *Type and energy of radiation of daughter radionuclide*: Gamma ray emitting radionuclides are preferred for most industrial applications because of the ability of gamma radiation to penetrate the wall of a vessel or pipeline. For some investigations, a simple gamma ray spectrum (containing one single photo peak) is desirable (for instance, in simple residence time distribution (RTD) experiments, flow rate monitoring, etc.). In other types of investigation, two or more gamma rays with a certain energy difference are desirable (when depth information, for instance, from a chemical reaction vessel, is required).
- *Availability and cost of mother radionuclide*: An ideal situation would be if the selected radionuclide generators (see later) were commercially available. However, in most cases, they are not at the present time, and will probably not be (in industrially suitable versions) for the foreseeable future since the global market probably is too small. The minimum requirement is then that the mother radionuclides should be readily producible in reactors or cyclotrons at reasonable costs.

In line with the criteria cited in paragraphs 2.3.1 and 2.3.2, some nuclear genetic relationships have been found to be suitable for production of IRTGs. Some potential generators are listed and specified in Table 1.

TABLE 1. POTENTIAL RADIONUCLIDE GENERATORS FOR INDUSTRIAL RADIOTRACER APPLICATIONS

Radionuclide generator	Mother nuclides				Daughter nuclides			Applications[a]
	Production	Half-life	Decay mode	Main E_γ (keV)	Half-life	Decay mode	Main E_γ (keV)	
^{44}Ti/^{44}Sc	^{45}Sc(p,2n)^{44}Ti	60 a	EC	67 (94.4%), 78 (96.0%)	3.93 h	β^+	511 (188.7%), 1157 (99.9%)	①–⑨
^{68}Ge/^{68}Ga	^{69}Ga(p,2n)^{68}Ge	270 d	EC	No γ	1.135 h	β^+	511 (178.3%), 1077 (3.2%)	①–⑦, ⑨
^{82}Sr/^{82}Rb	Mo(p,spall.) ^{85}Rb(p4n)^{82}Sr	25.6 d	EC	No γ	1.273 min	β^+	511 (190.9%), 776 (15.1%)	①–⑦, ⑨
^{99}Mo/^{99m}Tc	Fission product	60 h	β^-	740	6.00 h	γ	141 (89.4%)	①–⑤, ⑨
^{113}Sn/^{113m}In	^{112}Sn(n,γ)^{113}Sn	115 d	EC	No γ	1.65 h	γ	392 (64.1%)	①–⑤, ⑨
^{137}Cs/^{137m}Ba	Fission product	30.0 a	β^-	No γ	2.55 min	γ	661 (90.1%)	①–⑤, ⑨
^{144}Ce/^{144}Pr	Fission product	285 d	β^-	133	17.3 min	γ	696 (1.34%)	①–⑤, ⑨
^{172}Hf/^{172}Lu	^{173}Ta(p,2n)^{172}Hf	1.87 a	EC	24, 125	6.7 d	γ	1093 (63.0%)	①–⑤, ⑨

Note: Intensities taken from Nuclear Data Evaluation Research, Japan Atomic Energy Agency Group, http://www.ndc.jaea.go.jp/NuC/index.html, modified on 2011/05/09.

[a] ① Flow rate measurement, ② residence time distribution measurement, ③ leak detection and location, ④ mixing study, ⑤ mass balance, ⑥ positron emission tomography, ⑦ single particle tracing, ⑧ sediment transport study, ⑨ nuclear science education.

After further evaluation of the existing and potential generators and the anticipated demands for industrial investigations, it was decided to put most emphasis on the following three radionuclide generators: ^{68}Ge/^{68}Ga, ^{113}Sn/^{113m}In and ^{137}Cs/^{137m}Ba. In addition, some attention has been paid to ^{44}Ti/^{44}Sc, ^{99}Mo/^{99m}Tc, ^{144}Ce/^{144}Pr and ^{172}Hf/^{172}Lu.

In recent decades, the majority of developments of radionuclide generators has been mainly driven by the need for new short lived radiopharmaceuticals, both for diagnostics and therapy in nuclear medicine. The methods and techniques developed are, of course, tailor-made for the physical, microbiological and chemical system of the human body. Thus, the technology developed on generator design, chemical form of the eluted radionuclides and the tracer molecules derived from these radionuclides cannot necessarily be applied for other purposes.

In the paragraphs below, a survey of different radionuclide generator types are given.

2.4. RADIONUCLIDE GENERATOR TYPES

A radionuclide generator makes available a daughter radionuclide produced from the decay of a parent radionuclide. The goal is to acquire the daughter radionuclide from the parent with a sufficiently large decontamination factor DF = N_M/N_D, where N_M and N_D are the number of atoms of the mother and the daughter, respectively, in the purified daughter radionuclide fraction. In general, the DF should be in the range 10^{-5} to 10^{-7}. There are technical and environmental arguments for these numbers: with regard to the quality of most industrial investigations, a DF of $\approx 10^{-5}$ can be readily accepted.

Environmental considerations may require lower limits of mother breakthrough. The required decontamination factor may depend on the nature of the mother radionuclide. If the mother is not very long lived and decays exclusively by electron capture, a decontamination factor in the lower end may be acceptable.

In principle, a radionuclide generator system designed for use in industrial radiotracer applications should be simple, give reproducible mother/daughter separation results and provide a high yield of the daughter radionuclide.

There are various radionuclide generator principles. Some of them are listed below:

- Mother fixed on solid support in column (ion exchanger/reversed phase), elution of daughter by simple aqueous solution (salinity, pH).

- Mother on solid support in column, elution by aqueous phase with complexing agent (plus salinity and pH), to ease removal of daughter radionuclide, possibly extractable into organic solution.
- Mother on solid support in column, elution by organic solution, possibly with complexing agent to convert daughter directly into a complex extractable into organic solution.
- Mother on solid support in column, elution by cold, hot or wet air (gas) to produce the daughter as a gas tracer.
- Mother on solid support in column, elution by hot air (gas) to volatilize the daughter, which may be converted by chemistry to aqueous, organic or solid tracer.
- Mother dissolved in liquid (organic), separation of daughter by liquid–liquid extraction, whereupon the daughter may be converted to various types of tracers.
- Mother as a gas and the daughter, which may have a much lower vapour pressure, precipitating on container walls. Mother and daughter are then separated by cryopumping followed by conversion of daughter to various tracer types.
- Mother dissolved in aqueous solution. Separation of the daughter is achieved by selective precipitation with carrier.
- Mother dissolved in aqueous solution. Separation of the daughter is carried out by electrodeposition.
- Mother dissolved in aqueous solution. Separation of the daughter performed by electrolysis.
- The supported liquid membrane (SLM) generator where the mother is dissolved in an aqueous solution and separation of mother and daughter takes place over a passive extracting membrane between the mother radionuclide reservoir and an extraction liquid reservoir.
- Generator involving a gaseous component in the decay chain where separation is achieved by electrostatic attraction on a cathode in the gas phase.
- Generators not based on a mother/daughter relationship, but on use of a small isotropic neutron source for production of short lived radionuclides by neutron irradiation, separation of reaction products by chemical or physical methods.

Some of these principles are briefly described below.

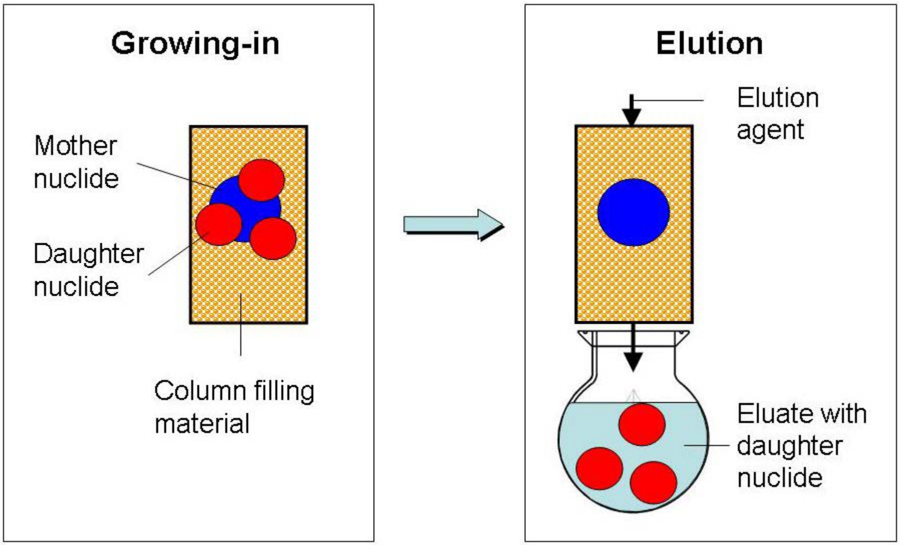


FIG. 6. Principle of a column based radionuclide generator.

2.4.1. Generators based on mother fixed on to solid support in columns

The most common radionuclide generators are based on sorption of the mother radionuclide on a column support material with subsequent and selective elution of the daughter radionuclide from the column with some elution agent (Fig. 6) while the mother remains on the column support. This category includes those mentioned above. Both organic and inorganic supports are used. The most common organic supports are cation and anion exchanger materials based on styrene–divinylbenzene co-polymer with ionogenic groups such as $–SO_3H$ and $–N(CH_3)_3Cl$, respectively.

Here, the H^+ and $Cl^–$ are exchangeable with other cations and anions, respectively. Inorganic materials are most often stable (or even refractory) metal oxides like Al_2O_3, TiO_2, MnO_2, ZnO, SnO, SnO_2, etc., but other types of material are used as well.

For high activity generators, which are intended for a long shelf life (say >100 d), it is important that radiation damage on the support material by radiolysis is kept at a minimum. In these cases, inorganic material is preferred whenever possible.

When performing parent/daughter separation, there are many aspects that must be considered: separation scheme, column type, column dimension, chemical nature of the mobile phase, species of interest, sample throughput and reagent usage for elution of daughter radionuclide [9]. This technique offers the following advantages:

- *Selectivity*: Separation selectivity is ensured by the selection of a suitable separation scheme.
- *Sensitivity*: High sensitivities down to the picomol range could be achieved, which is extremely important in using carrier free mother radionuclide.
- *Speed of separation*: The time necessary to perform the separation becomes an increasingly important aspect because of the short half-life of the daughter radionuclide.
- *Operational simplicity*: A major advantage of this technique in contrast to other techniques is its technical simplicity that not only reduces the radiation exposure to the operating personal, but also precludes the site contamination in an operational set-up.
- *Stability of the separator columns*: The stability of separator columns very much depends on the chemical nature of the packing material being used and the radiation characteristics of the mother and daughter radionuclides (alpha and high energy beta decay is more detrimental than EC decay). The shelf life also depends on the storage conditions (if possible, long time storage of the generator should be carried out with the column in a dried-out form in order to reduce degradation due to radiolysis of the liquid with formation of very reactive radicals).
- *Frequent elution repeatability*: The system can be operated repeatedly during the shelf life of the generator, and the operations can readily be made automatic if required.
- *Product reproducibility*: Product reproducibility can be excellent, provided that measures are taken to ensure standardization in the elution procedure and high radiation stability of the column material.
- *Elution performance*: This system provides unmatched performance in terms of yield and ensures high chemical and radiochemical purity of the daughter radionuclide.

Normally, elution liquids contain no carrier of the daughter radionuclide. Thus, the daughter radionuclide product will be of high specific activity and is referred to as ‘carrier free’. In some cases, the elution and the following chemistry may be eased by addition of a certain amount of stable elemental carrier. Generators of this type are small in size and simple to build, and the daughter radionuclide is availed by a simple elution process. A schematic diagram of a column chromatography generator system appropriate for industrial radiotracer applications is shown in Fig. 7.

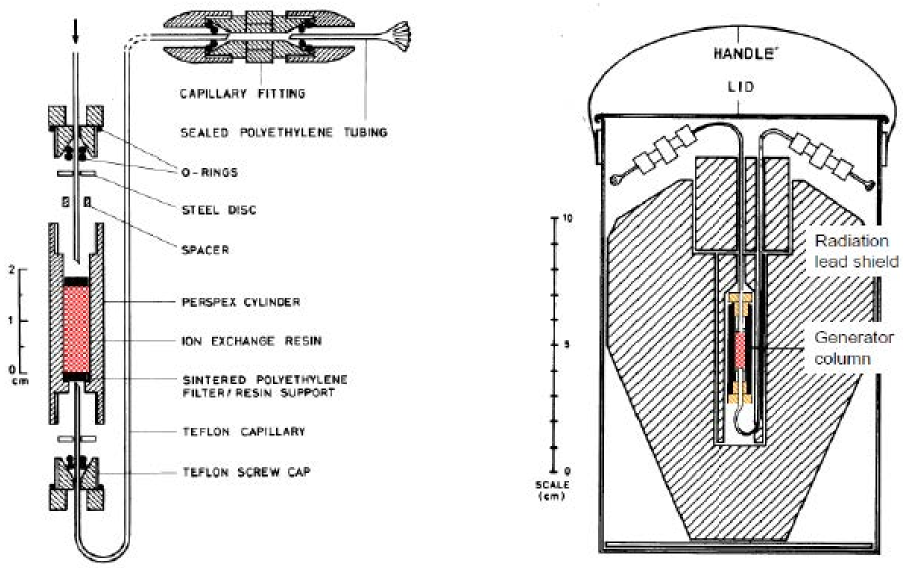


FIG. 7. Schematic diagram of a column based radionuclide generator with a small column system to the left and the column placed in a lead shield to the right [10].

Other column based methods, where the column itself constitutes the generator or where the column separation plays a major role in separating mother and daughter, may, in principle, also include extraction chromatography, partition chromatography, adsorption chromatography and gel permeation chromatography in various forms.

Novel extraction chromatographic resins exhibiting enhanced parent metal ion retention capacity, excellent selectivity, improved radiation and chemical stability offer promise in the development of radionuclide generators [11]. Although extraction chromatography has been shown to provide a means of performing a number of separations of potential use in radionuclide generator systems, the application of the technique to generator development has thus far been limited.

For partition and adsorption chromatography, many of the same arguments may be applied.

Low capacity, limited radiation stability, long term instability of the column under harsh industrial conditions and bleeding organic impurities under repeated elution are among the factors that inhibit its wide scale applicability for the preparation of IRNGs.

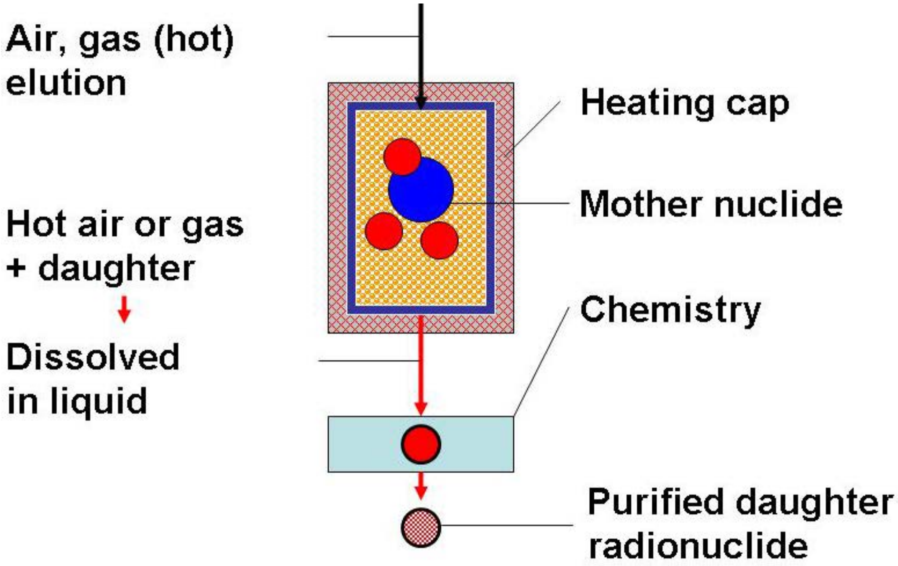


FIG. 8. Principle of a column based radionuclide generator with a high temperature heating cap [8].

Gel permeation chromatography has already, to some extent, been developed for generator production of radionuclides [12, 13], but is not considered further in the present publication.

2.4.2. Column based generator with a heating cap

A variation of the generator type described above is the case where the daughter product is not a gas, but may still be evaporated from the column at elevated temperatures (up to >1000°C). The requirement is that the daughter element has a considerably higher vapour pressure than the mother element in the form in which it is bound to the column material. This kind of separation principle has been developed for laboratory studies but, to our knowledge, has not yet been implemented in radionuclide generators. One example may be the generator ^{128}Ba/^{128}Cs where the mother nuclide is fixed in a ceramic column packing material as BaO, and Cs is removed by evaporation at $T > 1000$°C. The principle is sketched in Fig. 8.

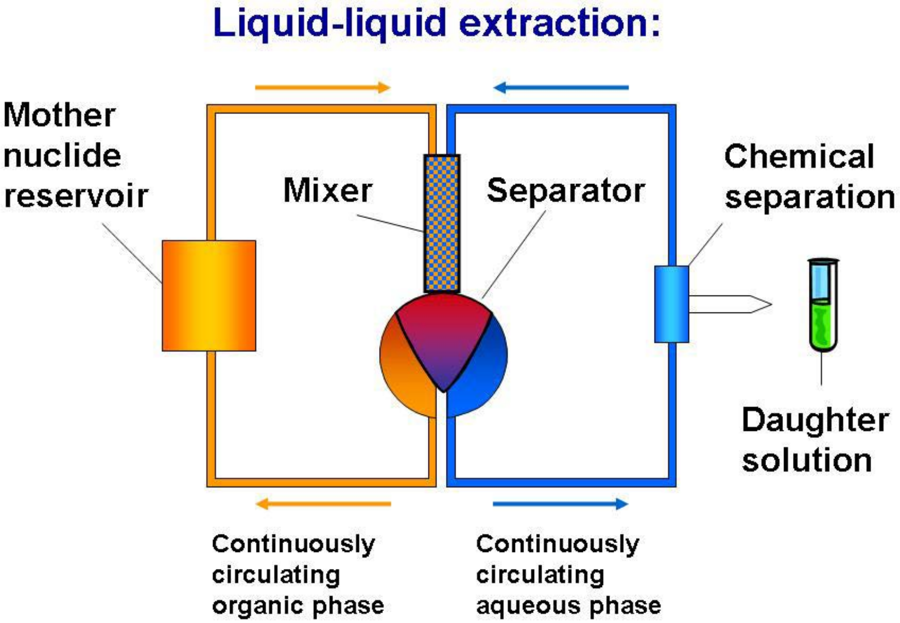


FIG. 9. Illustration of a solvent extraction generator with continuously flowing organic and aqueous phases [10].

2.4.3. Radionuclide generators based on separation by solvent extraction

Solvent extraction is a technique in which an organic solvent, with or without an extractant, is used to separate the daughter radionuclide from its mother [14].

The method is based on the relative solubility of the mother and the daughter radioelements in two immiscible liquids, usually water and an organic solvent. After agitation for a sufficient length of time, the layers are allowed to separate. If the aqueous phase originally contains the mother, the organic liquid, now containing the daughter radionuclide, is removed, and the aqueous layer containing the mother radionuclide is kept for subsequent daughter recovery.

It is also possible for the mother radionuclide to be dissolved in the organic phase. In this case, the daughter would be stripped into the aqueous phase for further processing. The principle of the process is illustrated in Fig. 9 for a generator that is based on continuously flowing liquids and on-line phase separation (either by mixer settlers or by continuously operating centrifuges).

In order to optimize separation, complexing agents may be added to the aqueous phase and extraction agents to the organic phase, and, most often, pH, ion strength and temperature are important parameters to consider. The separation of mother and daughter is governed by their respective distribution ratios in the selected organic–aqueous system. The distribution ratio D for a chemical species is defined as:

$$D_{\mathrm{M}} = \frac{\left(C_{\mathrm{M}}\right)_{\mathrm{org}}}{\left(C_{\mathrm{M}}\right)_{\mathrm{aq}}} \quad \text{and} \quad D_{\mathrm{D}} = \frac{\left(C_{\mathrm{D}}\right)_{\mathrm{org}}}{\left(C_{\mathrm{D}}\right)_{\mathrm{aq}}} \tag{15}$$

where C_{M} and C_{D} are, respectively, the concentrations of the mother and the daughter radionuclides (expressed by their activities corrected for decay and gamma branching ratios), in the organic (org) and aqueous (aq) phases.

In the case where the mother radionuclide is dissolved in the aqueous phase, a good separation is achieved in one separation step if, for instance, $D_{\mathrm{D}} > 100$ and $D_{\mathrm{M}} < 0.01$. If the mother nuclide reservoir consists of the organic phase, a good separation is obtained in one separation step if $D_{\mathrm{M}} > 100$ and $D_{\mathrm{D}} < 0.01$.

An example of the first situation is the separation of ^{144}Pr from ^{144}Ce (the ^{144}Ce/^{144}Pr generator) where ^{144}Ce is dissolved in the aqueous phase. This can be achieved by extraction of $^{144}\mathrm{Ce}^{4+}$ with, for instance, diethylhexyl orthophosphoric acid (HDEHP) dissolved in an organic (kerosene) solvent from an aqueous and weak acidic ($\mathrm{HNO_3}$) solution. The $^{144}\mathrm{Pr}^{3+}$ will remain in the aqueous phase.

A variation of the extraction method is illustrated in Fig. 10. In this example, the mother radionuclide is dissolved in an aqueous solution that is slightly acidified in order to avoid hydroxide formation of metal cations. After growing-in of the daughter to an acceptable level, the aqueous solution is percolated through a column where the column support has been impregnated with an organic phase containing an appropriate extracting agent for the daughter. In this process, the daughter nuclides will be sorbed in a narrow band near the top of the column, together with traces of the mother nuclide. The exit solution from the column containing the mother is pumped back to the mother nuclide reservoir.

In order to purify the daughter fraction, there are two alternative routes to follow:

(i) An elution liquid is applied that effectively removes (strips) any remaining mother activity from the column, leaving pure daughter activity in a somewhat broadened distribution on the column. The daughter is then removed from the column and dissolved into an aqueous phase by stripping with (often) more acidic solution.

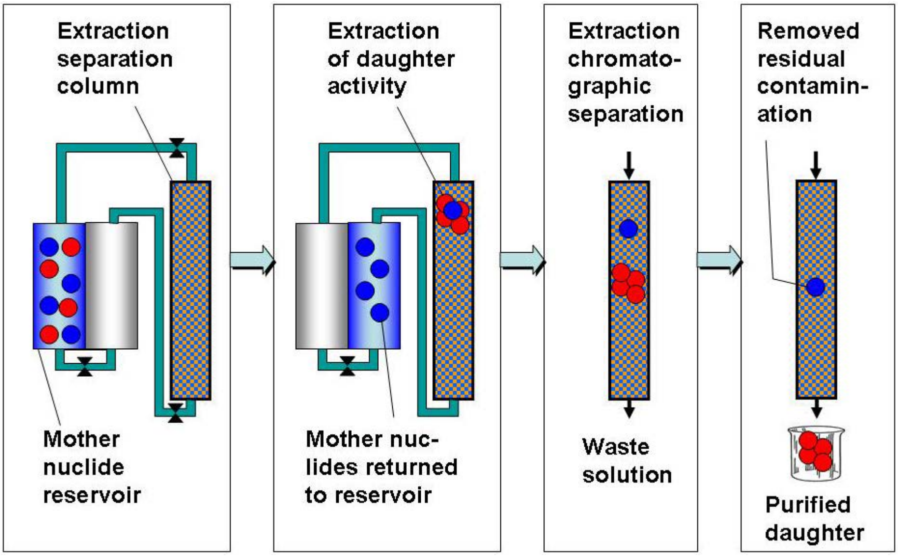


FIG. 10. Illustration of a generator where the mother nuclide is in an aqueous phase and the mother/daughter separation is achieved by percolating the aqueous fluid through an extraction chromatography column [8].

(ii) A specially engineered elution liquid is applied for the chromatographic separation of the daughter and its mother contaminant as the fluid moves through the column at moderate speed. Under these conditions, the mother and daughter have different affinities to the column material. Owing to this difference, the mother and daughter will move with different speeds down the column, and exit at the end in different chromatographic bands. This procedure is illustrated in Fig. 10.

Novel extraction chromatographic resins exhibiting enhanced parent metal ion retention capacity, excellent selectivity, improved radiation and chemical stability offer promise in the development of radionuclide generators [15].

Another version of the solvent extraction method is represented by Ref. [16]. Here, a small static extractor is constructed for extraction of $^{99m}TcO_4^-$ with methyl-ethyl-ketone (MEK) from an aqueous solution of $^{99}MoO_4^{2-}$ with an efficiency of >80% after three extraction cycles.

An example of a generator where the daughter is extracted into the organic phase while the mother remains in the aqueous phase is the $^{99}Mo/^{99m}Tc$ generator based on extraction by MEK. An alkaline solution of $^{99}MoO_4^{2-}$ is extracted with highly purified MEK. The daughter nuclide in the form of $^{99m}TcO_4^-$ is extracted into the MEK solution, while the molybdate remains in the basic aqueous solution. The MEK phase will then be isolated, washed and finally evaporated to dryness. The ^{99m}Tc compound is redissolved in an isotonic saline solution for preparation of radiopharmaceuticals by off-line labelling procedures. This procedure makes use of low specific activity ^{99}Mo and produces ^{99m}Tc at high purity and high radioactivity concentration [12, 17–19].

Advantages of the solvent extraction process are as follows:

- Offers selective separation due to the use of a specific extracting reagent mixture designed for the purpose.
- A high separation (or decontamination) factor is achievable.
- Quick separation process.
- High capacity: low specific activity mother can be used, for instance, by addition of carrier, both for the mother and the daughter to ease the separation process.
- Low running costs: the costs of the chemicals involved are low.
- Regeneration and recharging are possible.
- The process is scalable (volumes and activity) to a certain degree as per the need (see below).

Disadvantages of the solvent extraction process are as follows:

- There may be a limit to the activity concentration due to radiolytic reactions in the solvent chemicals that may, over time, degrade the solvent itself and change the chemical state of both mother and daughter nuclides and thus interfere with the separation process.
- Operation of this system requires highly skilled and trained personnel.
- Considerable volumes of organic extractants may be required to achieve satisfactory performance, dependent on the design of the generator.
- Thus, the operation of the solvent extraction generator may be complicated and cumbersome.
- There is also a potential for cross-contamination of the aqueous stream with the organic solution mainly due to formation of (relatively stable) micro-emulsions.
- The handling of considerable volumes of organic reagents may pose possible fire or explosion hazards. This can be counteracted by selecting solvents with low vapour pressure.

The technical difficulties make this procedure less desirable than the standard chromatographic generator system, and this system is not widely used.

Generators based on the solvent extraction technique are generally not the first choice for the preparation of an IRNG.

2.4.4. Generators based on separation by distillation or sublimation

In this method, the mother/daughter equilibrium mixture is chemically treated to form a compound in which the daughter radionuclide will have lower boiling points, whereas the parent is non-volatile.

There are two versions:

(i) The distillation method, which operates with the mother solution in a liquid solvent under moderate temperatures (<150°C);
(ii) The sublimation method, which operates under reduced pressure (vacuum), either with solid mother nuclide matrices or melts at elevated temperatures (>150°C).

For both versions, on heating the mixture, the vapour will be richer in the daughter radionuclide that is distilled out of the mother solution. The process is illustrated in Fig. 11.

In order to facilitate the separation, a chemical compound may be mixed into the mother solution for version (i) and led over the solid matrix or melt in the form of a gas for version (ii). These components react with the daughter radionuclide and make it more volatile. The evaporate then condenses as a purified compound on a cooled surface, for instance, a cold finger, leaving the non-volatile parent radionuclide behind. Once heating ceases (and the vacuum is removed for version (ii), the purified compound may be collected from the cooling zone, either by removing the cold finger or, as illustrated in Fig. 11, by directing a dissolving liquid into the apparatus and collecting the solution containing the daughter nuclide in a separate vessel. The distillation or sublimation processes are relatively fast, permitting the use of a low specific activity mother, and generating a daughter radionuclide of high purity. However, the success of these methods depends upon the possibility of forming a volatile daughter product compound with sufficient vapour pressure difference with respect to the mother nuclide. The necessary equipment is relatively bulky, partly fragile and requires careful operation. Automated versions suitable for operation under harsh industrial conditions have yet to be developed.

An example of this methodology is one version of the ^{99m}Tc generator that uses the difference in volatilities of the oxides of molybdenum and technetium to achieve a separation. Reactor produced ^{99}Mo is placed in a tube furnace and oxygen is passed over the heated, finely powdered metal. Technetium forms the

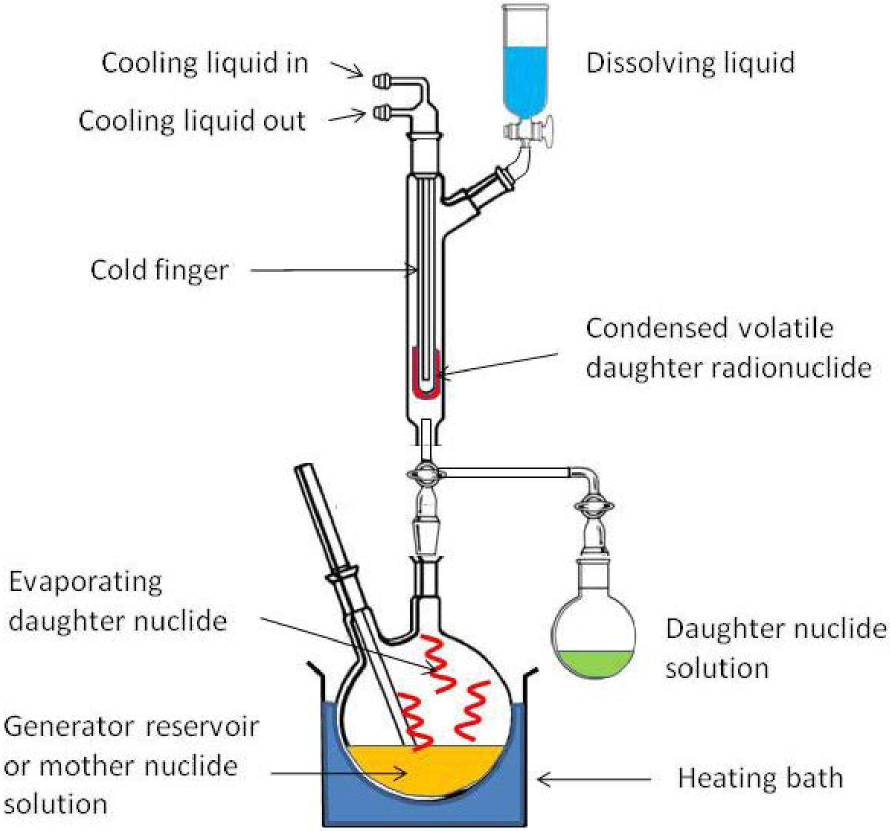


FIG. 11. Radionuclide generator based on sublimation (or distillation) of the volatile daughter radionuclide or a compound of the radionuclide from a mother radionuclide matrix.

oxide species Tc_2O_7, which becomes volatile at temperatures of 550°C, while molybdenum oxides have a much higher sublimation temperature of above 1000°C. The sublimated $^{99m}Tc_2O_7$ can be trapped by cooling followed by dissolution in an aqueous solution [20, 21]. Figure 12 sketches the principle of such a generator, adapted from Ref. [21].

2.4.5. Generators based on cryogenic distillation

Cryogenic separation of a gaseous daughter radionuclide from the mother radionuclide represents an effective strategy to providing an efficient separation with a high purity of the daughter product. It represents the selective removal either of the daughter radionuclide in the gaseous form from the non-volatile

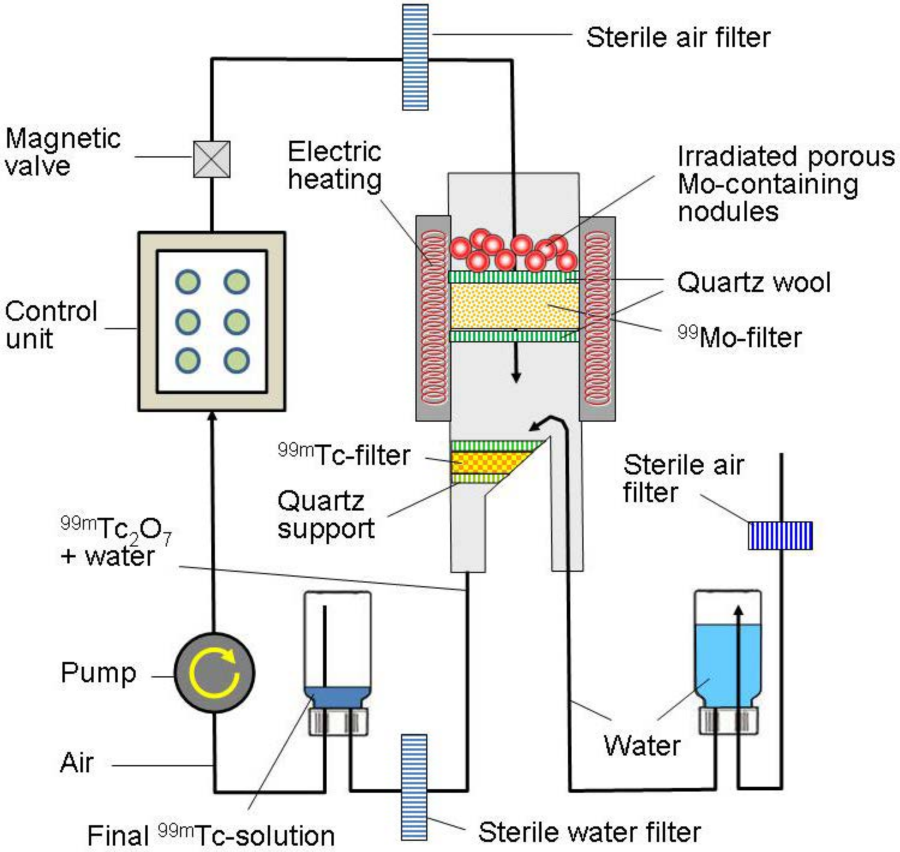


FIG. 12. Concept of the sublimation generator SICCATEC, where ^{99m}Tc *is separated as* $^{99m}Tc_2O_7$ *from the irradiated porous* MoO_3*–*V_2O_5*–SiC target material in a flow of air, caught in the alumina based Tc filter and subsequently redissolved in distilled water or a slightly saline solution [21].*

mother or the removal of the gaseous mother radionuclide from the non-volatile daughter by connecting a closed chamber of the mother/daughter mixture to a cold chamber where either the volatile daughter or the volatile mother is solidified, leaving the other component in the original mixture chamber.

An effective cryogenic separation depends on a low vapour pressure for one of the components and a high vapour pressure for the other component at room (or moderately elevated) temperature. Figure 13 illustrates the principle of a cryogenic distillation.

An example of this type of generator is the ^{122}Xe/^{122}I mother/daughter relationship where the volatile mother ^{122}Xe is cryopumped away from the original mixture, leaving the less volatile ^{122}I behind. Cryogenic pumping is a low

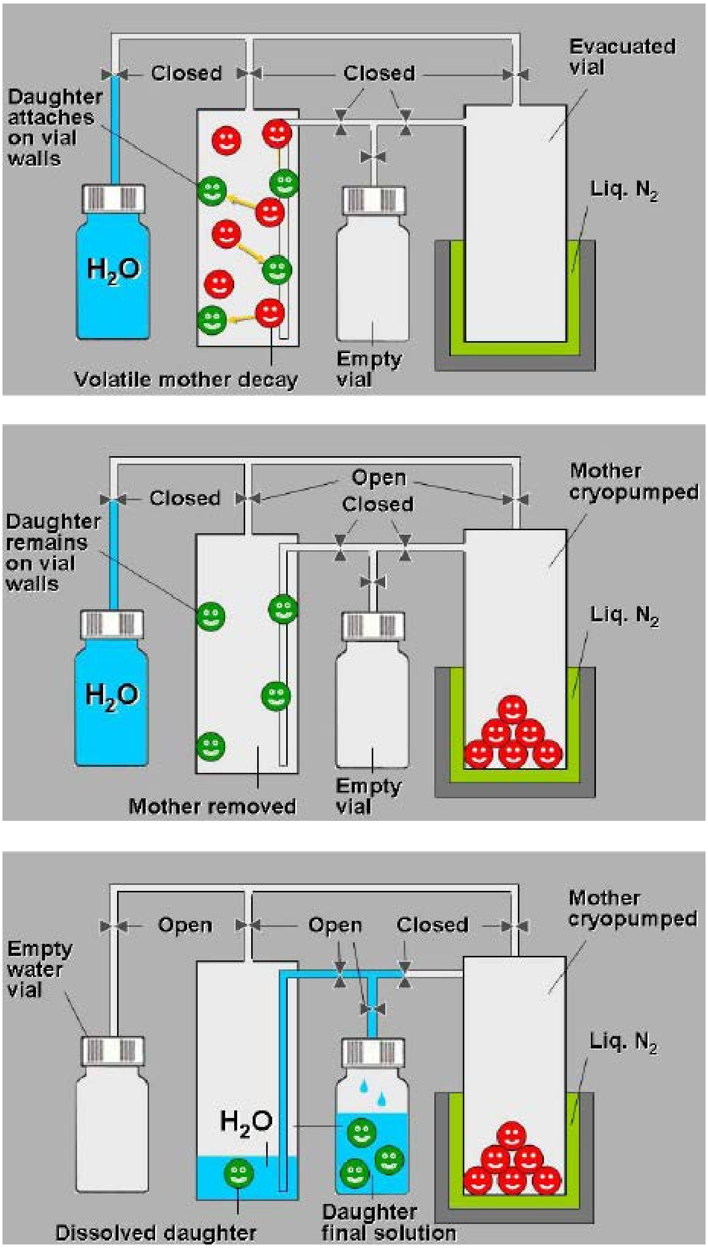


FIG. 13. Schematic illustration of various steps in operation of a cryogenic distillation generator, from top to bottom.

temperature physical approach to separation, in which the mother or daughter radionuclide is isolated directly by phase change.

The main advantage of cryogenic separation is that both a large quantity and a high purity of the daughter radionuclide are achievable. The yield of this separation is generally >90%.

A major disadvantage of cryogenic separation is the amount of energy required to provide the refrigeration necessary for the process, particularly for dilute gas streams. The cooling may be performed with liquid N_2, but at remote locations, electrically cooling equipment may be required. Cryogenic separation requires the use of significant amounts of process equipment at considerable capital cost. It is technically difficult to make a lightweight portable version.

2.4.6. Generators based on chemical precipitation

The relative solubility of the mother and daughter metal ion in a chemical environment can be exploited for the selective separation of the daughter radionuclide from the mother. When solubility of the mother and the daughter metal salts are very different, they can be separated by precipitation. The component that exhibits the lowest solubility will be precipitated first. One converts the component into an insoluble form or by changing the ionic composition of the solution or by adjusting pH and temperature, for instance. Under certain defined conditions, the solubility of every salt is governed by the solubility product K_{sp}. For a chemical reaction of the form:

$$mCn + nAm- \leftrightarrows CmAn$$

the stoichiometric solubility product of the compound *CmAn* is expressed by:

$$K_{\mathrm{sp}} = [C]^m \cdot [A]^n \tag{16}$$

If the actual ionic concentrations are so large that their product according to Eq. (16) exceeds K_{sp} for the particular compound, one has reached the required conditions for precipitation. In a stringent treatment of this subject area, one should, in fact, use the thermodynamic solubility product instead, where the concentrations C and A are exchanged with the expressions $a_C \cdot \gamma_C$ and $a_A \cdot \gamma_A$, respectively, where a is the ionic activity and γ is the activity coefficient. Hence, the condition for precipitation is defined by Eq. (17), where SR is the saturation index:

$$\mathrm{SR} = \frac{[C]^m \cdot [A]^n}{K_{\mathrm{sp}}} \geq 1 \tag{17}$$

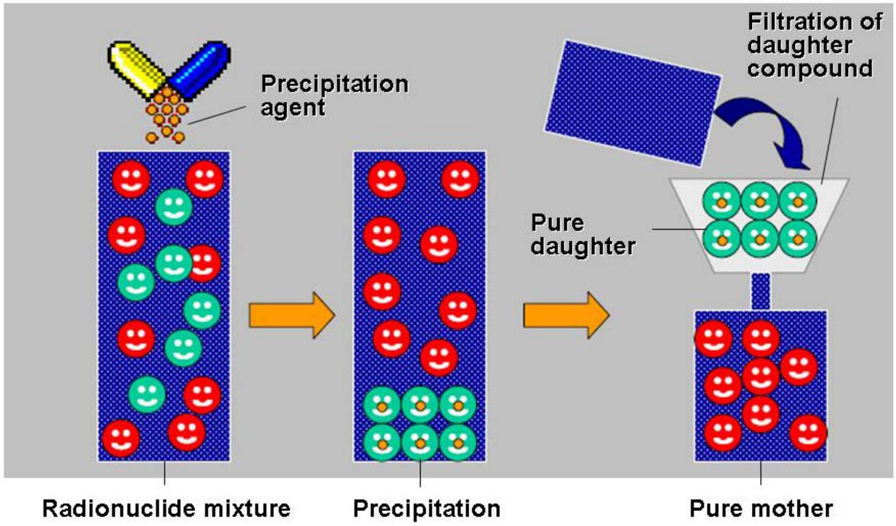


FIG. 14. Schematic illustration of a chemical precipitation system with precipitation of the daughter nuclide.

The practical implementation of the process involves adding a specific precipitating reagent containing the counter ion of the ionic compound of interest to the mother/daughter equilibrium mixture to precipitate either the mother or the daughter ions or ionic compounds.

In order to achieve a high chemical yield and an acceptable decontamination factor for the purified daughter product, carrier substances may be added for both the mother and daughter nuclides. Most often, a large excess of the added precipitation agent is used. This reduces the need for a high carrier concentration. Thus, a higher specific activity can be achieved.

Such precipitation methods can be systematically developed by sequencing of pH, adjusting the added portions of precipitating agents, and selecting the optimum temperature time period for quantitative precipitation. A schematic illustration of a chemical precipitation system is depicted in Fig. 14.

Advantages of the precipitation process are as follows:

- Convenient, simple process that provides the only practical method of separation in some cases;
- Can be highly selective and virtually quantitative (depending on a low K_{sp} value);
- High degree of decontamination is possible (depending on the relative values of $K_{sp,daughter}$ and $K_{sp,mother}$);

- Provides a large range of scale, both with respect to amounts treated and to the number of chemical elements for which the method is applicable;
- Not energy or resource intensive compared with other techniques (e.g. solvent extraction).

Disadvantages of the precipitation process are as follows:

- This system is applicable to those cases where there is a significant difference in solubility product of the actual compound of the mother and daughter radionuclide.
- It is sensitive to many chemical parameters such as concentration of the reactants, order of mixing, reaction temperature, etc., and any deviation from the standard operating procedure may result in failure.
- The multistep separation process to digest, filter or wash the precipitate may be time consuming and labour intensive.
- The precipitation often has to be complemented by some other separation method to purify the daughter product.
- It is hardly possible to achieve a carrier free (non-carrier added) product, so the specific activity (in the case of an isotopic carrier) or the activity concentration (in the case of a non-isotopic carrier) will not be very high.

One example of a generator where precipitation plays a main part is given in Ref. [22], where ^{90}Y is separated from its mother ^{90}Sr by precipitation of the ^{90}Sr followed by subsequent purification of the ^{90}Y by column chromatography.

2.4.7. Generators based on electrochemical separation

The principle of the electrochemical separation of a daughter radionuclide (or its compound) from its mother radionuclide (or its compound) is based on a difference in their electrochemical potential. The practical implementation of the separation process comprises the steps of dissolving the mother/daughter equilibrium mixture in an electrolyte followed by selective deposition of the daughter radionuclide on an electrode in accordance with their electrochemical potential.

This deposition can be achieved either by a spontaneous process or by applying an external voltage. Both variations are treated briefly below.

Spontaneous deposition, also called autodeposition (and sometimes incorrectly even electroplating or electrodeposition), is based on a difference in electronegativity between the mother and the daughter. For preferential deposition of the daughter, it has to exhibit a higher reduction potential than the mother.

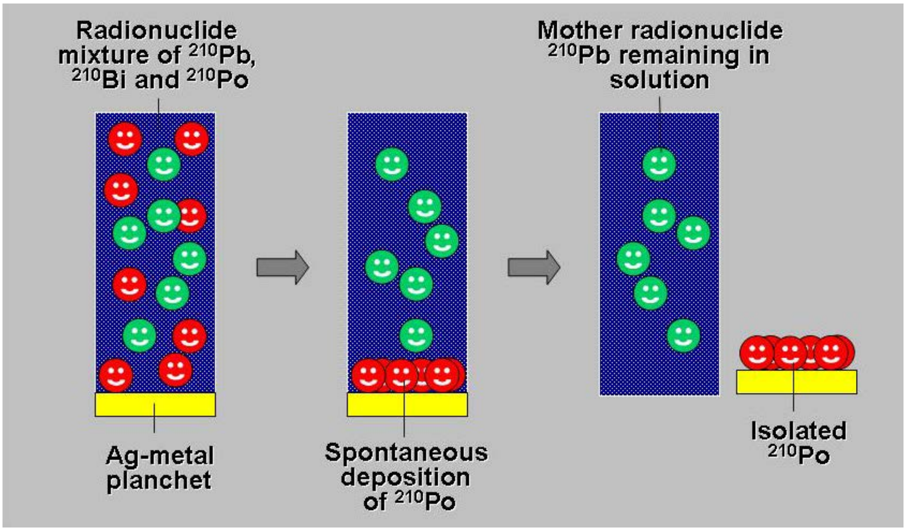


FIG. 15. Schematic illustration of a spontaneous deposition (autodeposition) process illustrated by the separation of ^{210}*Po* ($T_{1/2}$ *= 138.4 d) from its grandmother* ^{210}*Pb* ($T_{1/2}$ *= 22.3 a).*

Spontaneous deposition can take place on metallic surfaces with a lower reduction potential. The process is often demonstrated in primary courses in chemistry by dipping a rod of metallic zinc into an electrolyte of copper ions (Cu^{2+}). Metallic copper is spontaneously deposited (reduced to Cu^0) on the zinc rod, while the zinc rod is correspondingly dissolved (oxidized to Zn^{2+}). For the ^{210}Pb/^{210}Bi/^{210}Po nuclear genetic relationship, the ^{210}Po may be separated by autodeposition of a metallic nickel surface [23]. It is reported to be a very robust method. This creates potential for development of a ^{210}Pb/^{210}Po radionuclide generator for special purpose use. In this case, deposition on to a silver surface would be required since the nickel surface may also partly deposit the progeny ^{210}Bi and ^{210}Pb [24]. A schematic outline of the method is given in Fig. 15. The spontaneous deposition method has not yet been extensively exploited for more general mother/daughter separation in radionuclide generator technology.

The electrolysis method, also called electroplating (or electrodeposition), comprises an electrolyte containing the mother/daughter radionuclide mixture, two electrodes of an inert material like platinum and an external power source with variable high voltage connected to the electrodes, forming a cathode and an anode. The negative potential applied to the cathode causes the daughter radionuclide (as a positively charged ion) to be reduced and deposited on to the surface of the cathode. Reversing the potential causes the deposited daughter radionuclide to return back to the aqueous phase, thus allowing its recovery [11, 19, 25, 26].

Advantages of electrochemical separation are:

- Electrolysis requires few steps and product yield is high.
- Capacity is not limited by the amount of adsorbent or extractant.
- Simple to operate.
- Requires low cost equipment.
- Relatively fast process.
- Permits carrier free and clean separation from the parent.

Disadvantages of electrochemical separation are:

- Not a universal method; still in developmental stage with respect to radionuclide generators.
- Applicable to those systems where there is a significant E^0 difference of parent and daughter.
- A generator version suitable for field application has yet to be developed.
- Production of gaseous components during electrolysis must be taken care of to reduce any explosion danger (H_2 production).

A typical example of this method is the ^{90}Sr/^{90}Y generator developed at Bhabha Atomic Research Centre (BARC), India [25]. The experimental set-up is depicted in Fig. 16. In this generator, the separation of ^{90}Y from ^{90}Sr is achieved due to the difference between the electrochemical potentials of Y^{3+} and Sr^{2+}. On applying a suitable electric potential, ^{90}Y is selectively deposited at the cathode from a mixture of ^{90}Sr and ^{90}Y. The generator involves two electrolysis steps of purification. The first electrolysis is performed for 90 min in $^{90}Sr(NO_3)_2$ feed solution at pH2–3 at a potential of –2.5 V with a 100–200 mA current using platinum electrodes. The second electrolysis is performed for 45 min in 3M HNO_3 at a potential of −2.5 V with 100 mA current. In this step, the cathode from the first electrolysis containing ^{90}Y is used as the anode, whereby the yttrium again redissolves and is deposited on a fresh circular platinum cathode. The ^{90}Y deposited on the circular cathode after the second electrolysis is dissolved in acetate buffer for further production of useful tracer compounds.

2.4.8. Generators based on SLM separation

This separation technique is basically a continuously operated extraction stripping process. It includes the operation of an extracting liquid membrane

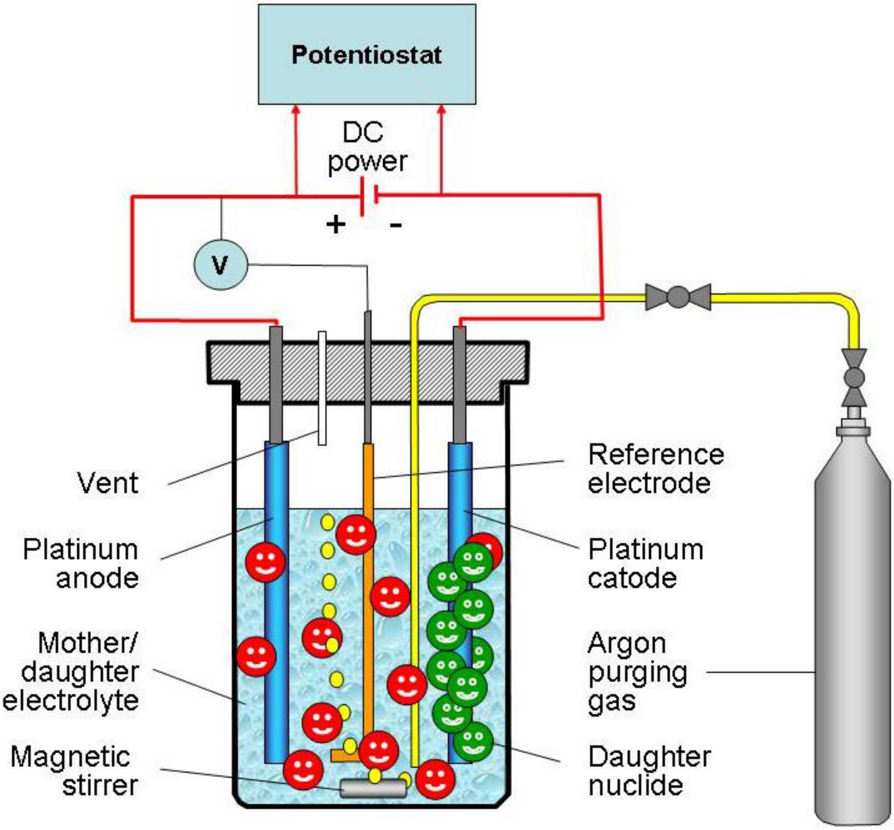


FIG. 16. Schematic diagram of electrolytic separation of ^{90}Y *from a* $^{90}Sr/^{90}Y$ *electrolyte mixture, adapted from Ref. [22].*

where the solute (daughter radionuclide) partitions between three liquid phases: the feed, the membrane and the receiving phases. This may be described as the ‘supported liquid membrane’ (SLM) technique.

The organic liquid (possibly including an extractant) is immobilized within the pores of a porous membrane. The porous membrane itself serves only as a framework or supporting layer for the organic liquid film. Such a membrane can be easily prepared by impregnating a hydrophobic porous membrane with a suitable organic solvent possibly modified with an extractant. The membrane can be defined essentially as a barrier, which separates two aqueous phases and restricts the transport of the mother radionuclide in a selective manner. The transport through the membrane is affected by the concentration gradient across it. The feed aqueous solution initially contains the mother/daughter equilibrium mixture.

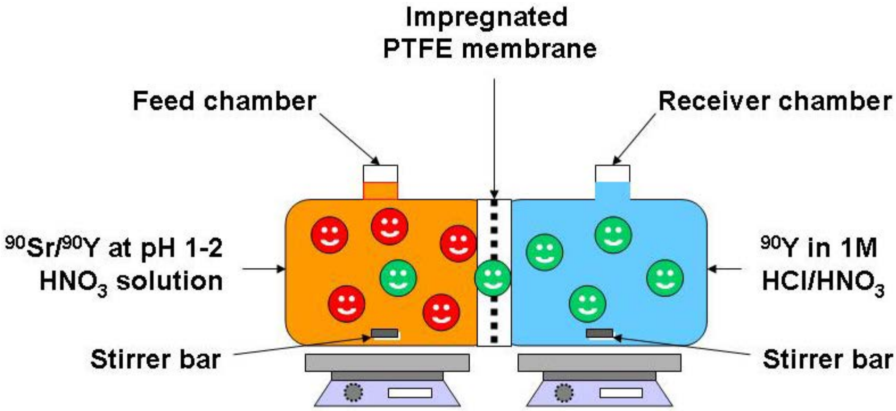


FIG. 17. Two chamber SLM $^{90}Sr/^{90}Y$ generator, adapted from Ref. [22].

Separation of the daughter from the mother by transport of the daughter through the membrane requires that the distribution ratio D_D (Eq. (15)) for the daughter to the organic phase in the membrane is high and the distribution ratio D_M for the mother correspondingly low. Then, the separation factor will be high. The aqueous solution present on the opposite side of the membrane, which is initially free from the daughter radionuclide, is referred to as the strip solution. For an efficient stripping (or back extraction) of the daughter from the organic phase in the membrane, the strip solution has to be designed so as to make the D_D value as low as possible on the stripping side of the membrane.

The performance of the process is strongly affected by the properties of the organic solvent and the selectivity of the possible complexing agent or extraction agent used in the membrane, the carrier concentration for the daughter or holdback carrier for the mother, the fluid dynamic of the phases (viscosity) and the membrane characteristics such as pore size and pore size distribution. In contrast to ordinary solvent extraction, this overall process is governed by kinetics and concentration differences rather than equilibrium parameters [27–29]. The principle of the process is sketched in Fig. 17.

Advantages of the SLM separation technique are:

- It offers selective separation depending on the selectivity of specific extracting reagents.
- Simple operation, both extraction and stripping are feasible in one stage, and the process does not include manual interaction.
- It does not use any external force such as pressure or high tension, but is based on difference of the chemical potential as a driving factor of the process.
- Larger interfacial area permits high efficiency.

Disadvantages of the SLM separation technique are:

- It is not yet a mature technology but still under development. A version suitable for industrial application has yet to be developed.
- Transport processes of the daughter are slow (slow kinetics), so it cannot be used for very short lived daughter activity.
- The equipment is still somewhat mechanically unstable, especially due to short lifetime and instability of the membrane.
- The release of the organic phase from the pores of an SLM into the receiver chamber represents a problem as it leads to chemical impurity in the separated product.
- It is essential to recharge the membrane periodically.

A generator system using the above concept has been developed for the preparation of carrier free ^{90}Y from ^{90}Sr in BARC, India [27] by employing an SLM using 2-ethylhexyl-2-ethylhexyl phosphonic acid (KSM-17 equivalent to PC 88A) supported on a poly-tetrafluoro-ethylene (PTFE) membrane (Fig. 17). After adjusting to pH2 with NaOH, the ^{90}Sr/^{90}Ysolution is used as a feed in a single stage membrane cell partitioned with a PTFE membrane impregnated with KSM-17 and having a feed and receiver compartment with 5.0 mL capacity each. The receiver compartment is filled with a 0.5M HNO_3 or 0.5M HCl stripping solution. Yttrium-90 alone is preferentially transported across the membrane, leaving behind ^{90}Sr in the feed compartment. This technique can yield ^{90}Y in millicurie levels in a pure and carrier free form. The feed can be reused repeatedly after allowing for ^{90}Y buildup.

Out of all the generator principles briefly described above, it was decided that the chromatographic type generator was to be the method of choice for further development in the CRP, which is the basis for this publication. The chromatographic type radionuclide generator has, at this stage, the best overall potential for being further developed into a suitable radiotracer generator for industrial application by coupling a selective chemical treatment for the daughter radionuclide of interest to a suitable radionuclide generator.

2.4.9. Special case: Generator based on electrostatic radionuclide separation

When one of the decay products in the decay chain from the mother radionuclide is gaseous, it may emanate from the mother radionuclide compound. The following decay will produce positively charged daughter products that may be collected on an electrode at negative charge. One example is generation of carrier free ^{212}Pb from decay of ^{228}Th through the following decay chain:

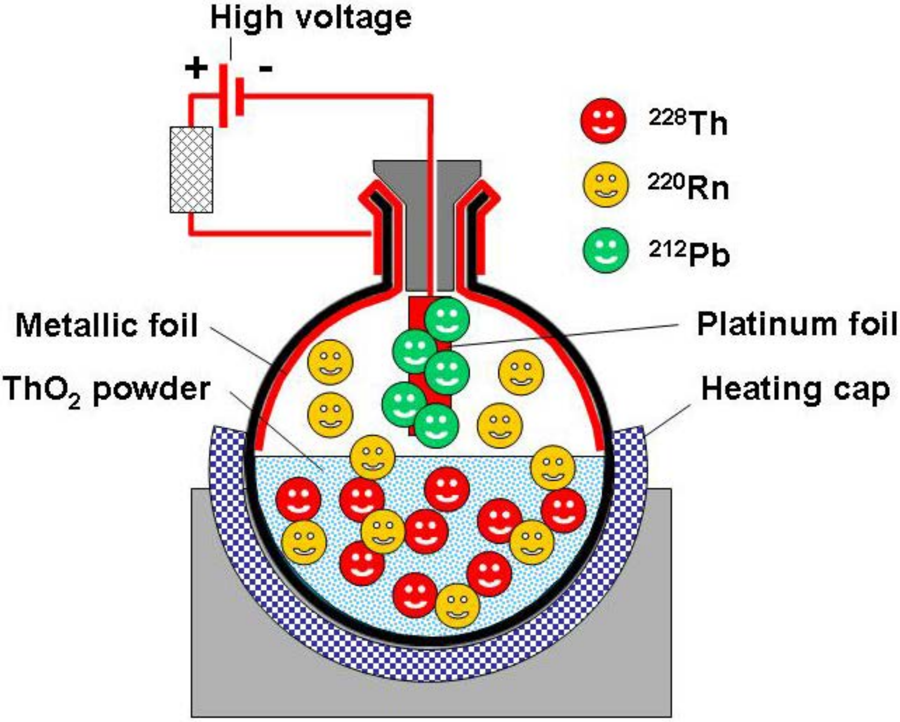


FIG. 18. Schematic diagram of the electrostatic $^{228}Th \rightarrow\cdots\rightarrow {}^{212}Pb$ *generator.*

$$^{228}\mathrm{Th}\ (1.913\ \mathrm{a}) \xrightarrow{\alpha} {}^{224}\mathrm{Ra}\ (3.66\ \mathrm{d}) \xrightarrow{\alpha} {}^{220}\mathrm{Rn}\ (55.6\ \mathrm{s})$$
$$\xrightarrow{\alpha} {}^{216}\mathrm{Po}\ (0.15\ \mathrm{s}) \xrightarrow{\alpha} {}^{212}\mathrm{Pb}\ (10.68\ \mathrm{h})$$

Here, both the very short lived ^{216}Po, as well as ^{212}Pb itself, will be collected at the cathode during alpha decay of their progeny. This generator was first published by Rutherford in 1900 and is often referred to as 'Rutherford's thorium cow' [30].

Thorium-228 is readily found in compounds of the naturally occurring mother ^{232}Th. If the thorium compound is 'old', i.e. at least 60 years since its elemental isolation in order to allow for in-growth of its decay product ^{228}Ra (5.75 a), the ^{228}Th is in nuclear equilibrium with ^{232}Th. Therefore, the emanating compound normally loaded into the generator is ordinary ThO_2. A sketch illustrating the principle of operation of the generator is given in Fig. 18.

The high tension is normally in the range 100–1000 V, the amount of ThO_2 is 10–100 g and the collection time 10–24 h. After the collection period, the platinum foil is removed and the deposited ^{212}Pb rinsed off (dissolved) with dilute nitric acid. If there is a need for several experiments during 1 d (or one collection period), several generators may be connected in parallel (battery of generators) on the same high temperature source. In order to achieve a sufficiently high emanation rate from the ThO_2 powder, it has to be finely grained and preferentially heated to some 50–60°C. A ^{212}Pb activity of 0.5–5 μCi may be obtained per collection. The operation and typical generator yields are given in Ref. [31].

Out of all the generator principles briefly described above, it was decided that the chromatographic type generator was to be the method of choice for further development in the CRP, which is the basis for this publication. The chromatographic type radionuclide generator has, at this stage, the best overall potential for being further developed into a suitable radiotracer generator for industrial application by coupling a selective chemical treatment for the daughter radionuclide of interest to a suitable radionuclide generator.

2.5. RADIOTRACER GENERATORS BASED ON RADIONUCLIDE GENERATORS

Radionuclides from radionuclide generators are the first step on the way to producing radiotracers for use in industrial process studies [32]. Radionuclide generators make radionuclides readily available at locations remote from irradiation facilities (reactors or particle accelerators). In addition, the short half-life of the daughter radionuclide ensures that any radioactive waste remaining after an industrial investigation is minimal and that any contamination of plant, product streams or environment discharges is negligible.

Although it is sometimes possible to use the daughter radionuclide directly to trace process streams, more often, it is necessary to chemically modify it so as to ensure full compatibility with the stream to be traced.

2.5.1. Design of an IRNG

For industrial applications, the most common type of generator is the column type, described in Section 2.4.1. The designs of the components of a column based radionuclide generator will vary from one manufacture to another, but the objective is always to facilitate the separation and elution of the daughter radionuclide from the parent radionuclide. The components in a typical column based generator are given below.

Column: The column, usually glass, containing a bed of sorbent as a support to which the mother radionuclide will bind strongly and not be washed off during the subsequent elution of the daughter radionuclide. Removal of the daughter radionuclide from the sorbent column is attributed to the lack of affinity of the daughter radionuclide for the sorbent, whereas the mother radionuclide is essentially completely bound to the sorbent and cannot be removed by the elution liquid, regardless of the volume of eluent passing through the column. Removal of the mother radionuclide requires an eluent of different chemical composition.

Tubing: Tubing is connected to the column to allow it to be washed with an eluent or rinsing solution. In a traditional generator design, the inlet tube will generally be accessible usually via a needle on to which a vial of the eluent can be attached as needed. Other generator designs provide an internal reservoir of eluent for this purpose. The outlet tube is also accessible and generally terminates as a needle to which an empty collecting vial can be attached. Most generators use an evacuated vial for collection so that eluent from the inlet side is drawn through the column and into the outlet vial. No pumps are needed. The driving force is the evacuated vial that creates a very significant differential pressure over the column since the eluent side of the generator is always at atmospheric pressure.

Filters: Filters serve to maintain the integrity of the sorbent bead within the column so as to prevent fluid channelling that would reduce the elution efficiency. The filters are usually glass frits with pore sizes suitable to remove any small particles of the sorbent from the eluted sample. The filters can also be made of porous plastic materials (for instance, polyethylene or Teflon).

Catcher column: Some generators come with a catcher column. This column contains a bed made of the same absorber material as the main column and is mounted at the outlet side of the main column. The purpose is to catch (bind chemically) any mother radionuclides that might have been eluted from the main column. The catcher column should be readily exchangeable without needing to dismantle the lead shield around the main column (see below).

Shielding: Heavy shielding is required for operator safety, as both the parent and the daughter radionuclides emit radiation. The glass column acts as a partial shield, but cannot stop the penetrating gamma radiation. Most generators will therefore have a lead shield around the column and outlet tubing. Additional shielding is used during the collection process and, of course, the eluted daughter radionuclide must be shielded once it is collected from the generator.

Handles: Generators are contained in a housing and have handles to allow manual or mechanical lifting and positioning. There is a cover to protect the inlet and outlet needles and appropriate labelling on the generator housing.

One typical generator design is shown in Fig. 7. Another design is given in Fig. 19.

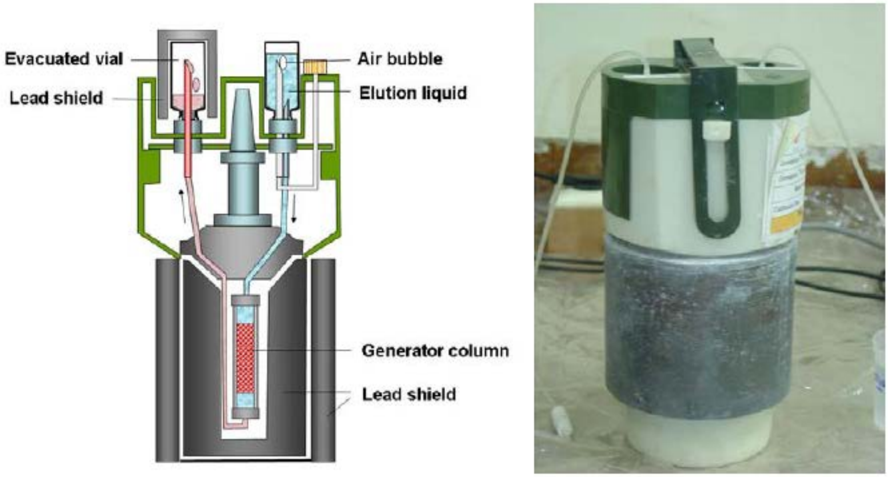


FIG. 19. Left: A column generator in its lead shield. Elution is performed by a vacuum vial, also in a lead shield. Right: A typical lead-shielded and closed radionuclide generator.

A set-up where the generator is eluted with a vacuum pump from an external elution liquid reservoir is illustrated in Fig. 20.

Principal features of the generator to be determined before use are the elution yield of the daughter as a function of the eluant flow rate and the degree of mother breakthrough. The activity requirements must be considered in order to optimize the elution flow rate in relation to the total injected volume, the duration of the examination and the residence time of the system to be studied.

Once the activity of the daughter is eluted, there is a growth of the daughter on the column until it again reaches a maximum and finally is in close equilibrium with the mother. This elution and regrowth can be continued as long as there are useful amounts of the mother radionuclide available. Elution may be performed before equilibrium is reached, and the amount of daughter activity eluted will depend on the time elapsed since the last elution.

Radionuclide generators intended for radiotracer applications must meet regulatory and quality control requirements. The production of the mother radionuclide in nuclear reactions by irradiation in a reactor or a particle accelerator, its separation from the target material, and the chemical and technical construction of the radionuclide generator are factors that must be considered carefully, so that the end result is a generator that is efficient and easy to operate. Processes for the production of radionuclide generator systems for radiotracer applications should follow good manufacturing practice standards.

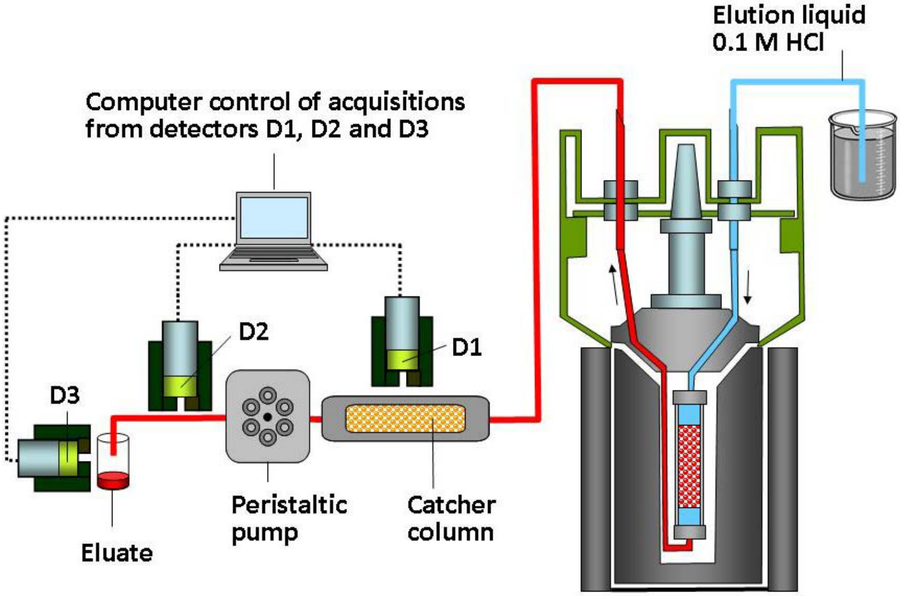


FIG. 20. A generator set-up that is eluted by means of a peristaltic vacuum pump and with a catcher column outside the main generator housing for easy detection of mother radionuclide breakthrough.

For some applications, it is possible to incorporate the generator into the tracer injection system. The system software initiates the elution of the generator at preset intervals and also controls the duration of the elution and the flow rate of the eluent. Such a system can be mounted permanently on a process stream for purposes of routine monitoring.

The choice of a specific daughter radionuclide for any given application depends on the following factors:

- Energy of photon emitted (in some cases, more than one photon energy);
- Half-life;
- Elution yield;
- Physical and chemical form, which leads into the next point;
- Possibility of converting the radionuclide into a useful radiotracer for various tracing scenarios.

2.5.2. Activity requirement

Estimating the required activity is an important step in any industrial experiment with radioactive sources. For radiotracers, the lower limit of the activity is determined by the measurement sensitivity of the detector, desired accuracy, dilution between injection and detection points, and background radiation level. The upper limit is set by radiological safety considerations. The amount of the required activity depends on the following factors:

- Desired accuracy;
- Efficiency of measurement or calibration factor of the detection system;
- Expected level of dilution/dispersion;
- Half-life of the daughter radionuclide;
- Background radiation level and number of repeated measurements;
- Mode of injection and detection (delta function versus continuous or extended pulse width);
- Expected losses of activity (should not really happen).

2.5.3. Industrial tracing situations

Industrial conditions for radiotracer applications vary considerably from case to case. Basically, the following conditions may be encountered in a radiotracer experiment:

- High temperature (up to hundreds or thousands of degrees celsius);
- High pressure (up to hundreds of megapascals);
- Vast volume (up to millions of cubic metres);
- Multiple phases (solids, liquids (aqueous, organic, supercritical), gases);
- Complex processes;
- Quick transit time (down to milliseconds);
- Complex media;
- Unexpected tracer degradation (microbiological attack, sorption, thermal or chemical degradation);
- Adsorption potential;
- Harsh working environment.

The increasing application of generator produced radiotracers in industry has stimulated the extensive development of a multitude of new radiotracers and novel radiolabelling procedures. Radiolabelling with these short lived radionuclides represents a remarkable challenge. Special attention has to be paid to synthesis time and specific labelling techniques due to the short physical half-life of the respective radionuclides. Development of procedures for producing radiotracers for industrial application must take this into account.

2.5.3.1. Aqueous liquid tracing

The radionuclide eluted from a radionuclide generator can be of various chemical forms. In order for the eluted radionuclide to be used directly as a passive radiotracer, the chemical form of the radioactive daughter compound should possess the following properties.

The radioactive material should be soluble in the process stream under the prevailing conditions of pH, temperature and pressure. It should not degrade in any way and not behave differently from the traced material itself.

Ideally, the radionuclide eluted from the generator should be in a chemical form that is physically and chemically compatible with the material to be traced so that it can be injected into the process stream of interest without delay and without complicated chemical manipulation. However, in most cases, the generator eluate does not satisfy all these characteristics. In order to utilize the eluted radionuclides effectively for radiotracer investigation, further chemical manipulation is most often necessary.

Chemical conversion makes the production of a wide spectrum of appropriate radiotracers possible, and opens up the window for utilization of these compounds in different situations. The eluted radionuclide is generally treated with an appropriate ligand to form an inert complex having sufficient thermal stability. For water tracing, the best performing tracers are anionic. In simple one component water tracing, neutral and positively charged compounds may be used depending on the process to be traced. In multiphase flow, where one component is water and the other an oleic phase, a neutral component can only be used if it does not contain a lipophilic fraction, which makes it soluble in the oil phase as well as in the water. In addition, for very short lived radionuclides, the synthesis process must be rapid in order to avoid detrimental decay during the process.

In order to develop synthesis and/or labelling techniques that can be adopted for industrial tracing, it is a good starting point to look at the work conducted for labelling of radiotracers for nuclear medicine purposes. Of course, this is a much simpler system than encountered in most industrial situations, especially when considering the temperature, acidity and chemical compositions of the systems to be investigated.

Radiopharmaceutical scientists have designed, developed and tested many potentially useful and ‘on the spot’ labelling kit formulations using ‘cold’ kits. A cold kit contains the ligand to be complexed, an adequate quantity of reducing or oxidizing agent (if needed), a buffer to adjust the pH to suit the labelling conditions and other required chemical components or modifiers. Radiolabelling with kits is fairly easy, can be easily accomplished with high complexation yield and provides a novel route to make radiotracers. Kit formulations are not only effective in carrying out the tasks, but also save time during field tracer investigations.

Here, two such examples are illustrated:

(i) The eluted chemical form of ^{99m}Tc from the $^{99}Mo/^{99m}Tc$ radionuclide generator is pertechnetate, $^{99m}TcO_4^-$. This anion is a good water tracer provided the temperature is below about 90°C and the chemical conditions are not reducing. At reducing conditions, technetium is reduced from oxidation state +7 to +4 and forms $^{99m}TcO_2$. This is a non-charged molecule that will attach to any solid surface and not behave as a water tracer any more. Hence, ^{99m}Tc has to be stabilized preferably as an anion by complexation with suitable ligands. Such a ligand is EDTA, a readily obtained chelate. It has been shown to be effective in the process evaluation of wastewater treatment plants as it precludes absorption of the radiolabelled molecule on the vessel or pipe walls or on suspended solids during the course of tracing [33].

(ii) A radiotracer experiments was carried out on a model of a rectangular settler with a volume of 3 m^3. Several different aqueous phase tracers were injected consecutively. The tracers were (i) non-radioactive fluorescine (a strongly coloured dye that, in these experiments, is taken as a reference water tracer), (ii) the eluate from a $^{68}Ge/^{68}Ga$ generator eluted with 0.1N HCl solution and no further treatment and (iii) ^{68}Ga complexes with 1,4,7-triazacyclononane-1,4,7-triacetic acid (NOTA) and 1,4,7,10-tetrazacyclododecane-1,4,7,10-tetraacetic acid (DOTA).

The obtained RTD functions indicated that the directly eluted ^{68}Ga (in the form of solvated Ga^{3+}) did not work as a water tracer in the settler. However, the complex compounds of gallium were stable in the water phase and were not adsorbed on the vessel walls. These complexes can be recommended as tracers of the water phase in similar experiments and under similar experimental conditions [34].

The preparation procedure is as follows:

- Elute Ga^{3+} from the generator with about 5 mL of 0.1N HCl solution;
- Add about 1 mL of a 0.01M solution of the macrocyclic compounds DOTA or NOTA;
- Adjust pH to 3.5;
- Heat the mixture to 90–95°C for 20–30 min.

The above procedure for $^{68}Ga^{3+}$ ion complexation with DOTA and NOTA can be recommended as a useful synthetic route for production of ^{68}Ga labelled passive water phase tracers of ^{68}Ga eluted on-site from the $^{68}Ge/^{68}Ga$ generator.

Table 2 summarizes some of the common applications of radiotracers.

TABLE 2. SOME EXAMPLES OF CHEMICAL MODIFICATIONS OF RADIONUCLIDES ELUTED FROM THREE SELECTED GENERATORS AND APPLICATIONS OF THESE ‘MODIFIED’ RADIOLABELLED COMPOUNDS

Labelled phase	Generator		
	$^{137}Cs/^{137m}Ba$	$^{99}Mo/^{99m}Tc$	$^{113}Sn/^{113m}In$
Surface water	^{137m}Ba–EDTA[a] ^{137m}Ba–DTPA[b]	$^{99m}TcO_4^-$ (as eluted)	^{113m}In–DTPA
Groundwater	Too short lived	Unstable?	^{113m}In–DTPA
Water WWTP[c]	^{137m}Ba–EDTA ^{137m}Ba–DTPA	^{99m}Tc–EDTA	^{113m}In–DTPA
Sludge WWTP	Too short lived	$^{99m}TcO_4^- + SnCl_2 \rightarrow {}^{99m}TcO_2$, sorption on to particles	$^{113m}In^{3+}$ will sorb on to particles
Mud in surface water	Too short lived	$^{99m}TcO_4^- + SnCl_2$, reaction as above	$^{113m}In^{3+}$ will sorb on to particles
Organic phase tested <120°C	^{137m}Ba–DC18C6[d]–HDNNS[e]	$^{99m}TcO_4^-$ + TBP[f], i.e. generator eluted directly with TBP	
Catalyst in organic phase tested <120°C			$^{113m}In^{3+}$ will sorb on to particles
HT[g]/HP[h] water (<350°C, 40 bar)		$^{99m}TcO_4^-$ + 8 hydroxy quinoline $^{99m}TcO_4^-$ + TcMAG3[a]	

[a] EDTA: ethylene diamine tetraacetic acid.
[b] DTPA: diethylene triamine pentaacetic acid.
[c] WWTP: wastewater treatment plant.
[d] DC18C6: crown ether: dicyclo-hexano-18-crown-6.
[e] HDNNS: dinonylnaphthalenesulphonic acid.
[f] TBP: tributyl phosphate.
[g] HT: high temperature.
[h] HP: high pressure.

2.5.3.2. Organic phase tracing

By using optimal reaction conditions, the daughter radionuclide can be extracted from an aqueous phase into a non-miscible organic liquid. Several solvent extraction reaction mechanisms are possible, most involving extra complexing agents (for instance, chelate formers) and some also involving adduct formers. Some examples are given below.

(a) Technetium-99^{m} tracer in MEK

The ^{99}Mo/^{99m}Tc generator based on an MEK solvent extraction system (Section 2.4.3) can be used to obtain ^{99m}Tc in MEK. An alkaline solution of 18.5 GBq (500 mCi) sodium molybdate [^{99}Mo] solution (7.4–14.8 GBq (200–400 mCi) per g Mo) is transferred into the extraction system. After extraction, the MEK phase is sampled by extracting an aliquot that can be used directly for organic phase tracing.

(b) Gallium-68 tracer in isopropyl ether

The solvent extraction method was used to prepare a ^{68}Ga tracer suitable for use in a single organic phase. Gallium-68 is eluted in 4N HCl (20 mL) from a ^{68}Ge/^{68}Ga generator based on a TiO_2 column. This acid (and chloride) concentration is sufficient to ensure that the chemical form of gallium is $^{68}GaCl_3$, which is non-charged. 10N HCl (20 mL) was added to the eluted 4N HCl solution in order to make it 6M, which is the optimum molarity to ensure maximum extraction of ^{68}Ga. Isopropyl ether (20 mL) was used to extract ^{68}Ga from the aqueous phase (two sequential extractions) without addition of extra extraction agents. Efficient liquid mixing was achieved by bubbling nitrogen through the combined fluids for 10 min.

(c) Gallium-68 tracer in MEK

The ^{68}Ga is eluted with 1M HCl. A few mL of the aqueous phase containing ^{68}Ga is equilibrated with an equal volume of MEK to extract ^{68}Ga.

(d) Gallium-68 labelled acetylacetonate, $Ga(acac)_3$

The $^{68}Ga^{3+}$ in the form of $GaCl_3$ is eluted from an ionic ^{68}Ge/$^{68}Ga/SnO_2$ generator with 1M HCl. The solution was evaporated by heating to dryness in a test tube. The residue was dissolved in an ethanol solution containing 0.002% acetylacetone. The solution was heated for 20 min in a 65°C water bath to facilitate formation of ^{68}Ga labelled acetylacetonate. The complex is suitable for oil tracing in single phase flow, but it has not been tested sufficiently in multiphase processes.

(e) Barium-137^{m} complex with DC18C6 and HDNNS

A radionuclide column generator consisting of ^{137}Cs absorbed on $K_2[CoFe(CN)_6]$ coated silica gel is eluted with a mixture of 0.1M HCl and 0.145M NaCl. The eluate is vigorously stirred with a solution of DC18C6 and HDNNS (1:2) and 1-dodecanol in isooctane. The resulting complex shows a high distribution ratio, also at elevated temperatures (60°C). It has also been tested with formation waters from oil reservoirs and shows acceptably low back extraction [35]. This complex (patent pending) may be used for multiphase oil tracing under typical oil–water separator conditions in the petroleum industry.

2.5.3.3. Solid tracing

Radiotracing using radiolabelled solid particles is an important tool to obtain invaluable information about movement of solid particulate material in industrial fluid flow, deposition in mines, geological tracing, surface water and river transport and movements of shore sediments. A few methods are described briefly below.

The degree of labelling may be proportional to the volume of the non-porous part of the solid material or proportional to the surface (internal and/or external) of the particle.

(a) Electrostatic sorption/ion exchange techniques

The generator eluate can be directly used for labelling solid particles utilizing electrostatic sorption/ion exchange techniques. This kind of labelling is, in principle, proportional to the surface of the particle. A larger percentage of the daughter radionuclide can be adsorbed at low tracer concentrations because a larger surface area is available compared to the amount of tracer present. Dilution with carrier decreases the amount of tracer adsorbed because the carrier is competing for adsorption, and the relative amount of the radioactive label interacting with the adsorbate is much less. Such a technique depends on the following factors:

- *Chemical state of the radionuclide*: Adsorption increases with increasing charge on the ion.

- *Nature of the surface material*: Surfaces that have a negative charge or that contain hydroxyl groups can interact with cations through electrostatic attraction and hydrogen bonding, respectively. For tracing of geological matter (sands), the charge should typically be positive.
- *Surface area*: Porous particles have larger surface areas than typical crystalline particles and have, therefore, a higher capacity for sorption. In addition, internal surfaces are shielded against wear and erosion of the radiolabel. If the pore size distribution throughout the particle is homogeneous and constant, the labelling in this case can also be taken to be proportional to the *volume* of the particle.
- *pH*: Generally, adsorption decreases with a lower pH (higher hydrogen ion concentration) because the ions interact with negatively charged surfaces, and the hydronium ions compete with other cations for the available binding sites.

Examples include:

- *Labelling of organic ion exchange material*: The daughter radionuclide eluted from the generator can be equilibrated with a specific amount of ion exchange material after adjusting the pH of the external solution. In this way, cation exchangers (with negatively charged sorption sites) can be labelled with a suite of cationic radionuclides. Likewise, anion exchangers (with positively charged sorption sites) can be labelled with radionuclides of anionic form. Organic ion exchangers, for instance, based on polymerization of styrene with divinylbenzene as a cross linker, represent a flexible structure that may expand or contract, depending on the ion strength in the surrounding solution. As such, one can hardly define a difference between surface and volume labelling; in this case, it is both.
- *Labelling of geological and other inorganic material*: Natural inorganic sorbents such as various clays (kaolinite, illite, montmorillonite, vermiculite, etc.), perlite, glass wool, sands (silicate rich or carbonaceous), volcanic ash, etc. can also be used for preparing solid radiotracers. The procedure is similar to the previous one. The process is based on the adsorption of the radiolabel on to, or into (for porous material), the available surface of the solid particles.

While choosing a sorbent, the following points must be considered:

- Rate and capacity of absorption/adsorption;
- Retention capacity (i.e. is the absorption thermodynamically irreversible in the intended tracing process?);

- Physical shielding of the label (i.e. can it be removed by erosion or wear?);
- Ease of application.

(b) Internal labelling

When the starting point is a radionuclide generator, there is only one form of internal labelling (disregarding the above comment on organic ion exchangers): the radiolabelled solid particles are produced by chemical precipitation of the material that contains the radiolabel. It can be direct precipitation with a carrier concentration with stable isotopes of the same chemical element or it can be by co-precipitation with other chemical elements.

Further treatment of the precipitate involves crushing and, possibly, grinding before sieving into special particle size fractions. For precipitation with a carrier of the same element, one always ensures that all the particles are homogeneously labelled (i.e. activity in Bq/g is constant), regardless of the particle size. For the case of co-precipitation, two situations have to be considered:

(i) Precipitation is performed in a thoroughly stirred solution, in which case, one achieves a homogeneous precipitation, i.e. the precipitated material is homogeneously labelled throughout the precipitate volume and the particles fabricated from the precipitate are also all homogeneously labelled (and the activity concentration is constant for all particles).

(ii) If vigorous stirring is not carried out, there is the risk that the precipitation will be non-homogeneous. The first formed core particles have a higher activity concentration than the precipitated layers that grow on the initial particles. Thus, there will be a concentration gradient from high activity in the core towards lower activity at the precipitate surface at the end of the precipitation process. The various particles produced subsequently will then not have a homogeneous volumetric label: those from the surface area will have lower activity and those from the core will have higher activity.

(c) Preparation of radiocolloids

It is also possible to generate radiocolloids by chemical reactions such as hydrolysis. A method of forming an aerosol is to produce a liquid spray with a gas jet. The process can be facilitated by separating the liquid into droplets with electrostatic repulsions by applying a charge to the liquid. Emulsions are usually prepared by vigorously shaking the two immiscible phases together, often with the addition of an emulsifying agent, e.g. a surfactant such as soap, in order to stabilize the product formed. Semi-solid colloids, known as gels, may possibly be formed from the cooling of lyophilic sols that contain large linear molecules and have a much greater viscosity than the solvent.

2.6. EXAMPLES OF RADIOTRACER GENERATORS NOT BASED ON A MOTHER/DAUGHTER NUCLEAR RELATIONSHIP

2.6.1. The methyl bromide generator

Methyl bromide ($CH_3{}^{82}Br$) is a radiolabelled gamma emitting gas used for gas tracing. Since it is oil soluble, it is also used as a tracer for organic liquids in single phase tracing, especially for localization of leaks in underground pipelines. When dealing with strongly radioactive organic liquids, autoradiolysis is always an issue. Ideally, such tracer molecules should be prepared just before use. Below is a description of how to generate $CH_3{}^{82}Br$ on-site from concentrated solutions of reactor produced $^{82}Br^-$.

Neutron irradiation of potassium bromide (KBr) is regularly carried out at reactors to produce $^{82}Br^-$ ($T_{1/2}$ = 36 h) for tracing of aqueous systems, and the availability of this radionuclide is relatively good. The transformation of solid radioactive potassium bromide to gaseous methyl bromide is carried out in a mobile chemical reactor, called a methyl bromide generator. It consists of a lead-shielded chamber containing the irradiated KBr. The KBr is dissolved in sulphuric acid and dimethylsulphate is added. Upon heating, the gaseous reaction product $CH_3{}^{82}Br$ is formed together with non-radioactive $CH_3{}^{79}Br$ and $CH_3{}^{81}Br$ according to the reaction:

$$2KBr + (CH_3)_2SO_4 \rightarrow 2CH_3Br + K_2SO_4$$

The gaseous products are condensed in a second lead shielded and actively cooled chamber from where it can be injected in aliquots. The generator is pictured in Fig. 21.

One important feature in the development of such a system is to enable storage of the CH_3Br gas in a ‘cryocanister’ that prevents its release to the atmosphere. The canister has sufficient capability to store large quantities of $CH_3{}^{82}Br$ for the useful lifetime of the radiolabel (about 1 week).

Depending on the requirements in the various industrial tracing situations, generator activities up to 379 GBq (10 Ci) can be prepared and transported.

Different versions of mobile generators have been constructed with total weights ranging from 100 kg up to 400 kg to enable production of the required

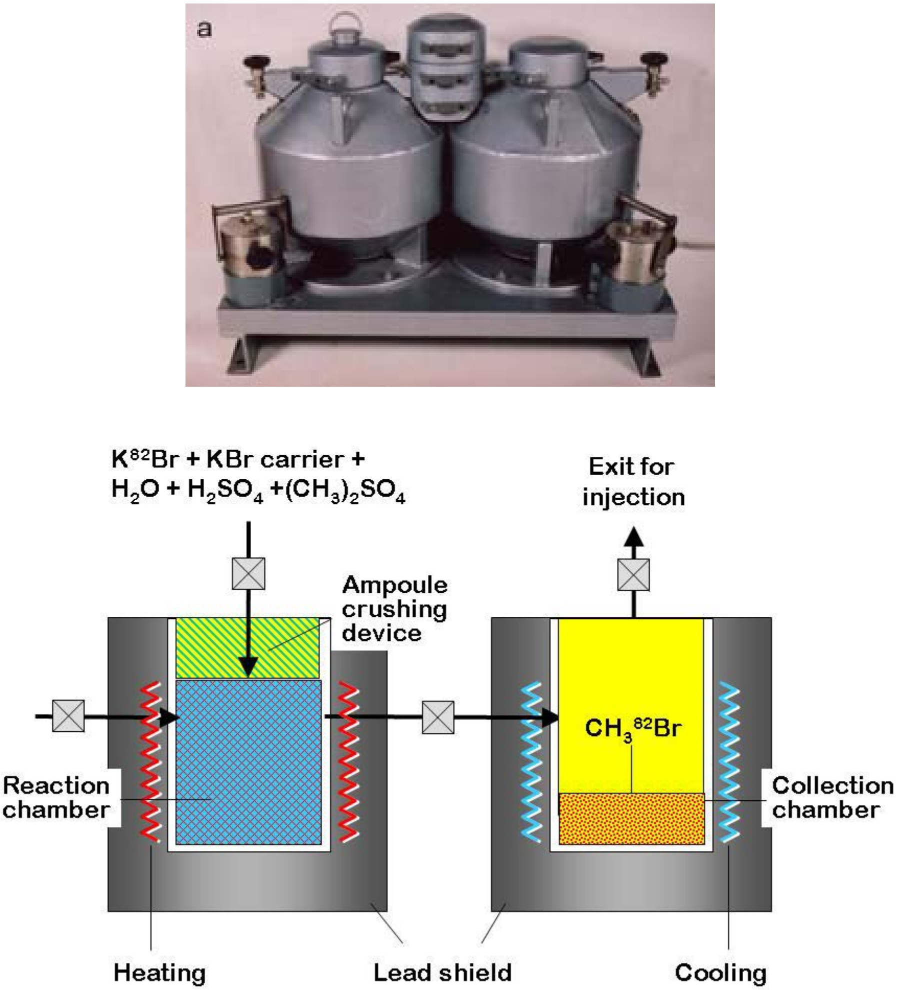


FIG. 21. Photograph and schematic diagram of the $CH_3{}^{82}Br$ radiotracer generator.

activity for various experiments. The fact that this particular generator design allows radioactive KBr to be transported separately in well shielded lead containers, allows for simpler logistics: it is simpler to transport a very small volume of radiolabelled solid salt than the corresponding activity as a radioactive gas.

The production process of $CH_3{}^{82}Br$ at the work site is as follows:

- Crush the quartz ampoule containing the solid salt $K^{82}Br$ using a special integrated crusher.

- Add sulphuric acid and methyl sulphate agents to the reactor for the dissolution of $K^{82}Br$.
- Clean the crusher and make it ready for the next operation.
- Close the valves connected to the reaction chamber to seal it off to prevent any accidental leakages.
- Heat the reactor to 70°C for about 30 min. The chemical reaction starts.
- Close the top valve on the collection chamber and open carefully the valve between the chambers.
- The temperature of the cooling chamber is brought to 5°C (liquefaction temperature of methyl bromide) by submerging the chamber in dry ice. The $CH_3{}^{82}Br$ gas is transported from the reaction chamber to the collection chamber by cryopumping and condensed there.
- Allow natural heating of the collection chamber by removing the dry ice. Once the storage tank reaches ambient temperature, release of $CH_3{}^{82}Br$ gas is possible.
- Connect special containers to the exit valve on the collection chamber and open the valve to transfer $CH_3{}^{82}Br$ to the containers.
- $CH_3{}^{82}Br$ can be dispensed from the container into special ampoules for injection in subsequent tracer experiments.
- Clean and prepare the containers to make the system ready for another synthesis operation.

2.6.2. Radionuclide generators from on-site activation with isotropic neutron sources

Table 3 shows a number of short lived radionuclides that can be produced by irradiation with isotopic neutron sources because of very high activation cross-sections of target nuclides.

The calculated reaction yields are based on the following assumptions:

- Irradiation time: $10T_{1/2}$.
- Thermal neutron flux: 10^9 n·cm^{-2}·s^{-1}.
- Amount of target material: 1 g molecule (mol = 6.023×10^{23} target atoms).

Typical commercial neutron sources are ^{252}Cf (spontaneous fission with emission of neutrons) and ^{241}Am/Be, in which neutrons are generated by the following α particle reaction:

$$^{9}Be + {}^{4}He \rightarrow {}^{12}C + n$$

TABLE 3. NUCLEAR DATA AND CALCULATED REACTION YIELDS (IN NUMBER OF γ EVENTS EXPRESSED AS BECQUERELS OF THE MAIN γ RAY ENERGY) FOR SEVERAL SHORT LIVED RADIONUCLIDES THAT MAY BE PRODUCED BY SMALL ISOTOPIC NEUTRON SOURCES

Radionuclide	Half-life	Nuclear reaction	*n*th cross-section	γ energy (intensity)	Irradiation time	Reaction yield of γ
^{52}V	3.75 min	^{51}V(n, γ)	4.9 b	1434 keV (99.75%)	37.5 min	2.9 GBq
^{56}Mn	2.58 h	^{55}Mn(n, γ)	13.3 b	847 keV (100%)	25.8 h	8.0 GBq
^{60m}Co	10.5 min	^{59}Co(n, γ)	20.7 b	59 keV (100%)	105 min	12.4 GBq
^{80g}Br	17.6 min	^{79}Br(n, γ)	8.3 b	616 keV (50.7%)	176 min	2.5 GBq
^{104}Rh	42 s	^{103}Rh(n, γ)	134 b	556 keV (100%)	7 min	80.4 GBq
^{108g}Ag	2.41 min	^{107}Ag(n, γ)	36 b	663 keV (51.8%)	24.1 min	11.2 GBq
^{110g}Ag	24 s	^{109}Ag(n, γ)	87 b	658 keV (48.2%)	4 min	25.2 GBq
^{116m}In	54 min	^{115}In(n, γ)	73 b	1294 keV (95.7%)	540 min	41.9 GBq

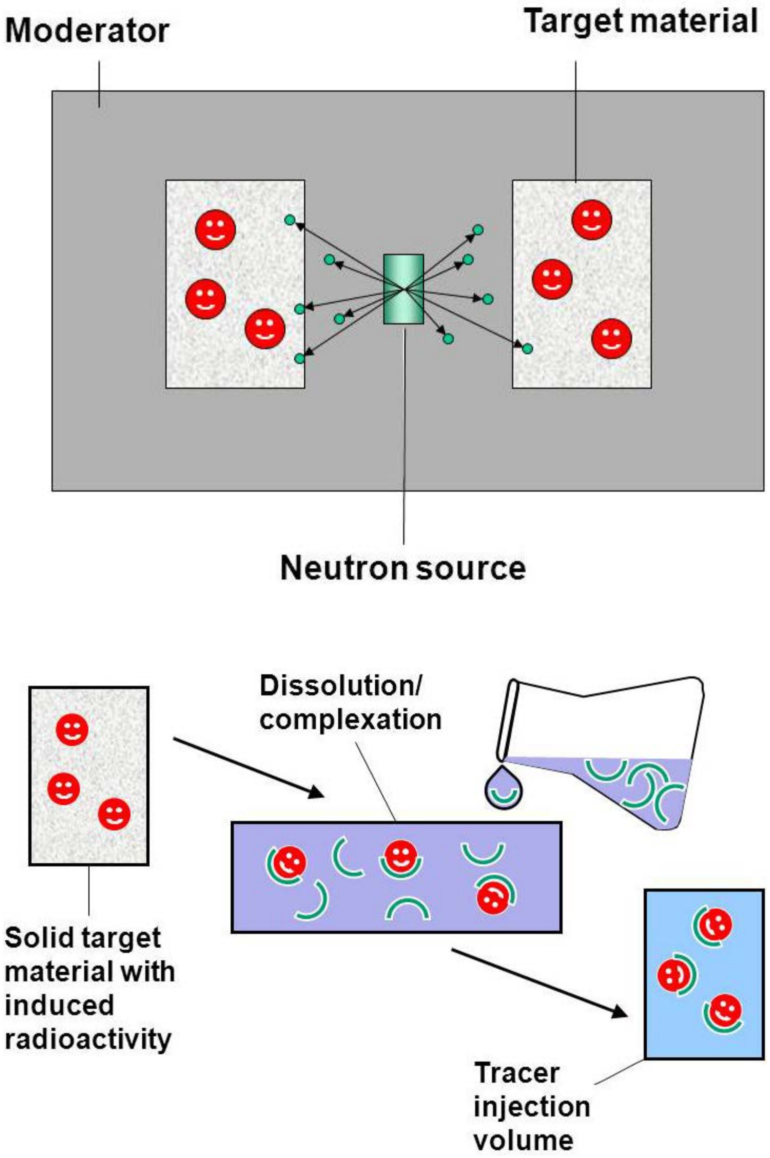


FIG. 22. Activation of solid target with thermal neutrons from a small isotopic neutron source (upper part) followed by dissolution of the target and complexation of the radionuclide of interest (lower part) to produce a suitable tracer compound.

The process is schematically illustrated in Fig. 22.

Decay of ^{252}Cf by alpha emission (96.91% probability) and spontaneous fission (3.09% probability) results in an overall half-life of 2.645 a and neutron emission of 2.314 × 10^6 $s^{-1}\cdot\mu g^{-1}$, with a specific activity of 19.832 GBq per µg (0.536 mCi/µg). Thus, it is an intense neutron emitter that is routinely encapsulated in compact, cylindrical source capsules. A source the size of a person's little finger can emit up to 10^{11} neutrons/s. However, the use of such a source requires bulky and heavy shielding. For instance, a 3 mg source requires a cylindrical transport container of height 121 cm, diameter 112 cm and weight 1723 kg.

The ^{252}Cf sources of several different neutron outputs are, in principle, available commercially, but are expensive in the sizes needed to produce a slow neutron flux of 10^9 $n\cdot cm^{-2}\cdot s^{-1}$, which is used in the calculation of the reaction yields in Table 3. However, a lower flux may still be of interest. The yields given are then simply scaled down linearly with the neutron flux.

The half-life of ^{241}Am is 432 a, and neutron sources based on this radionuclide have therefore a much longer useful lifetime than the ^{252}Cf sources. However, the neutron output is lower. In general, for commercially available sources, the neutron output is 2.2 × 10^6 $n\cdot s^{-1}\cdot Ci^{-1}$. Typical commercial source strengths are in the range 1 µCi (smoke detectors) to >20 Ci (irradiators, oil well logging tools, etc.). The strongest sources will produce fluxes in the range 10^5–10^6 $n\cdot cm^{-2}\cdot s^{-1}$, which is three to four orders of magnitude lower than the flux used for the calculations in Table 3. The results are yields of the order of megabecquerels, which may nevertheless still be useful for some tracing purposes.

Transportable neutron generators utilizing the T(d,n) or the D(d,n) reaction can also be used for radionuclide production, as described briefly in Section 2.6.3 below.

2.6.3. Radionuclide (radiotracer) generators based on on-line use of neutron generators

Tracers to follow a mass flow can be generated in situ if the flowing material contains an activable component. For illustration purposes, the most obvious and simple example is activation of water itself with fast (14 MeV) neutrons from a neutron generator based on the reaction T(d,n)^{4}He. This activation leads to the nuclear reaction:

$$^{16}O + n_{14\ MeV} \rightarrow {}^{16}N + p$$

Nitrogen-16 is radioactive with a half-life of 7.13 s and emits high energy γ rays of 6.1 and 7.1 MeV. This process is possible due to recent development of small sealed tube and transportable neutron generators.

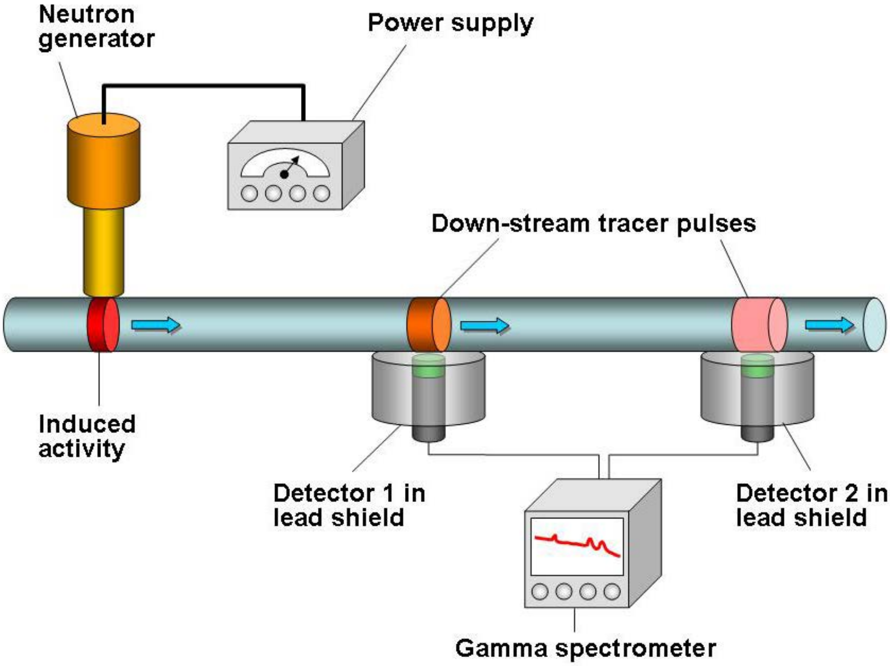


FIG. 23. Schematic diagram of a pulse of a tracer induced in a mass flow by a short (or pulse) irradiation with a neutron generator, and the subsequent downstream detection of gamma radiation.

The principle of activation, or ‘tracer injection’, with subsequent radiation measurement is illustrated in Fig. 23.

Although activation of ^{16}O has been used as an example here, there is a suite of other radionuclides that may be generated in situ by this method, but whether or not the above technique is feasible depends to a considerable degree on the nature and the elemental composition of the flow. If the flow does not have an appropriate concentration of activable elements that produce useful tracers, it is possible to inject easily activable material into it.

Examples of useful activation reactions are $^{12}C(n,2n)^{11}C$ ($T_{1/2}$ = 20.3 min), $^{14}N(n,2n)^{13}N$ ($T_{1/2}$ = 9.97 min), $^{19}F(n,p)^{19}O$ ($T_{1/2}$ = 26.9 s), $^{20}Ne(n,p)^{20}F$ ($T_{1/2}$ = 11.0 s), $^{24}Mg(n,2n)^{23}Mg$ ($T_{1/2}$ = 11.32 s), $^{28}Si(n,p)^{28}Al$ ($T_{1/2}$ = 2.25 min), $^{31}P(n,2n)^{30}P$ ($T_{1/2}$ = 2.50 min), $^{37}Cl(n,\alpha)^{34}P$ ($T_{1/2}$ = 12.4 s), $^{55}Mn(n,p)^{55}Cr$ ($T_{1/2}$ = 3.50 min), etc.

3. SELECTION OF POTENTIALLY USEFUL RADIONUCLIDE GENERATORS AND PRODUCTION OF THE MOTHER RADIONUCLIDES

3.1. REQUIREMENTS FOR IRNGs

In order to conduct proper radiotracer investigations, various requirements need to be met before starting the actual test. The most important of all the requirements is the ‘RADIOTRACER’ itself.

Some medical radionuclide generators like ^{99}Mo/^{99m}Tc and ^{113}Sn/^{113m}In are being used for a few applications in industry and the environment. The use of these generators is limited because of low photon energy and short half-life of the radionuclide and limitations of compatibility and stability of the radiolabelled tracer molecule (direct generator eluate) under harsh industrial conditions. There are several potentially useful nuclear genetic relationships that may form the basis for development of radionuclide generators for industrial applications. For a radionuclide generator to be suitable for industrial use, the following criteria need to be satisfied.

(a) *Availability*: Ideally, the radionuclide generator should be commercially available at a reasonable cost.
(b) *Half-life of the mother radionuclide*: For use at remote locations, it is important to select a generator with an appropriately long half-life. A generator that remains useful for a period of 1 year may be good for industrial applications.
(c) *Half-life of the daughter radionuclide*: The daughter radionuclide, used as a radioactive tracer or tag on the radioactive tracer for industrial applications should have a half-life varying typically from several minutes to several hours.
(d) *Type and energy of radiation from the daughter radionuclide*: Gamma ray emitting radionuclides are preferred for most industrial applications because of the ability of gamma radiation to penetrate the often substantial thickness of material from which the process plant is constructed. Gamma rays of energy 140 keV and upwards have been found to be useful, in some cases, gamma energies >500 keV are required.
(e) *Physicochemical properties of the eluted daughter*: Ideally, the material eluted from the generator should be in a form that is physically and chemically compatible with the material to be traced so that it can be injected into the process stream of interest without delay and without complicated chemical treatment. A radionuclide generator that includes a

suitable chemical processing unit enabling the eluted daughter to be transformed directly into a suitable industrial radiotracer will be preferable.

(f) *Easy logistics*: Packing, transport and subsequent handling at the industrial site must be technically easy and readily implemented within the governing regulations.

3.2. POTENTIALLY USEFUL RADIONUCLIDE GENERATORS FOR INDUSTRIAL AND ENVIRONMENTAL TRACER APPLICATIONS

3.2.1. General

Based on the requirements that need to be met for radiotracer applications under harsh industrial and/or field conditions, the radionuclide generators that have been selected and proposed for use as the basis for generating appropriate short lived radiotracers in developing and industrialized countries are listed in Table 4.

Demands from developing and industrialized countries are somewhat different, mainly due to differing availabilities of the various radionuclide generators. It was supposed that due to less fluent transport and procurement conditions in developing countries, parent nuclides with longer half-lives (at least 100 d) and daughter nuclides with longer half-lives (at least of the order of minutes) should be considered necessary for these countries.

Table 4 lists some important nuclear characteristics such as half-lives and photon energies and the proposed application environment for the various radionuclide generators. The table also indicates whether the given radionuclide generator is commercially available or not.

The following paragraphs give details of the most relevant generator systems, selected from Table 4. For these generator systems, the following, relevant parameters are discussed in detail:

- Daughter gamma energy;
- Daughter half-life;
- Parent half-life;
- Production and expected yield;
- Separation technique and chemical form of daughter;
- Availability and current status;
- Possible applications.

TABLE 4. OVERVIEW OF SELECTED GENERATOR SYSTEMS

Generator	Mother				Daughter		
	Half-life	Decay mode	γ energy (keV)	Production	Half-life	Main emission	Main photon keV (%)
^{44}Ti/^{44}Sc	47.3 a	EC	67, 78	^{45}Sc(p,2n)^{44}Ti	3.92 h	β^+	511 (188.6%) 1157 (99.9%)
^{68}Ge/^{68}Ga	270 d	EC	no γ	^{69}Ga(p,2n)^{68}Ge	1.135 h	β^+	511 (178.3%) 1077 (3.2%)
^{82}Sr/^{82}Rb	25.6 d	EC	no γ	^{85}Rb(p,4n)^{82}Sr	1.273 min	β^+	511 (190.9%) 776 (15.1%)
^{99}Mo/^{99m}Tc	60 h	β^-	740	^{235}U(n,f)	6.00 h	γ	141 (89.4%)
^{113}Sn/^{113m}In	115 d	EC	no γ	^{112}Sn(n,γ)^{113}Sn	1.65 h	γ	392 (64.1%)
^{137}Cs/^{137m}Ba	30.0 a	β^-	no γ	^{235}U(n,f)	2.55 min	γ	661 (90.1%)
^{144}Ce/^{144}Pr	285 d	β^-	133	^{235}U(n,f)	17.3 min	γ	696.5 (1.34%)
^{172}Hf/^{172}Lu	1.87 a	EC	24, 125	^{175}Lu(p,4n)	6.7 d	γ	1093 (63.2%)

3.2.2. Mass attenuation of actual gamma energies

The penetrating power of the various actual gamma energies from the selected radionuclide generators is of interest in order to judge and evaluate the usefulness of a certain radionuclide for the various industrial monitoring operations that may be of interest. The selected absorbers, iron (Fe) and water (H_2O), simulate steels and most actual liquid solutions, respectively.

In order to estimate this penetrating power, let us first repeat some necessary theory. A narrow beam of monoenergetic photons with initial intensity I_0, penetrating a layer of material with mass thickness x, density ρ and linear absorption coefficient μ, exits from the absorber with intensity I_x, given by the Lambert–Beer law:

$$I_x = I_0 \cdot \exp\left[-\left(\frac{\mu}{\rho}\right)\cdot x\right] \tag{18}$$

Let us calculate now the *half-value thickness* $d_{1/2}$ for the various relevant energies for the absorbers iron and water. This $d_{1/2}$ parameter is defined as the linear thickness of the absorber that reduces the exit intensity (I_x) to half the value of the initial intensity (I_0). Solving x from Eq. (18) gives:

$$x = \ln\left(\frac{I_0}{I_x}\right) \Big/ \left(\frac{\mu}{\rho}\right) \tag{19}$$

Setting $I_x = 0.5I_0$ gives:

$$x = 0.693 \Big/ \left(\frac{\mu}{\rho}\right) \tag{20}$$

and since $x = \rho d$:

$$d_{1/2} = 0.693 \Big/ \left\{\left(\frac{\mu}{\rho}\right)\cdot \rho\right\} \tag{21}$$

which is the equation used for calculation of the linear half-value thicknesses.

The primary data on mass absorption coefficients are taken from Ref. [36] and recalculated for the actual gamma energies. The primary data for the energy region of interest are plotted in Fig. 24.

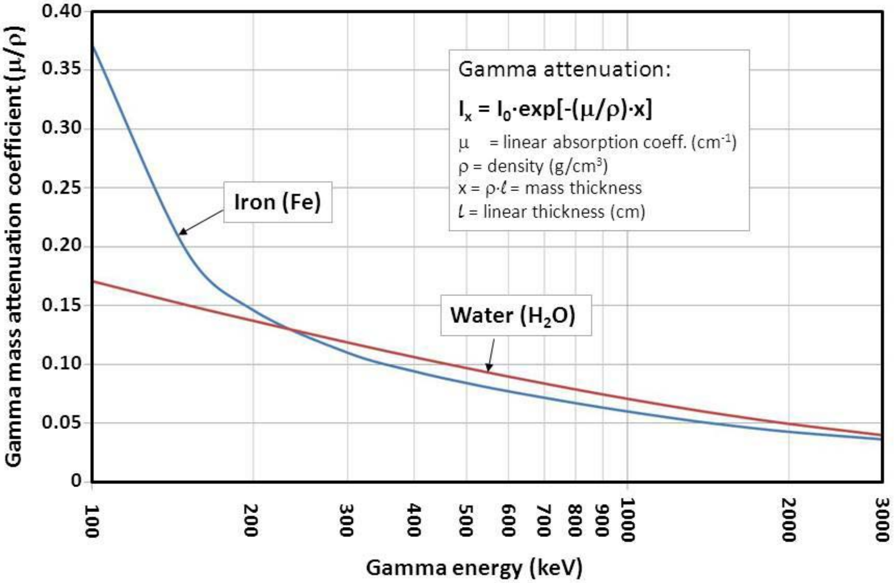


FIG. 24. Mass attenuation coefficients, μ/ρ, as a function of gamma energy in the energy range 100–3000 keV for the two absorbers iron (Fe) and water (H_2O) [36].

Calculated mass attenuation coefficients and the corresponding half-thickness values are given in Table 5.

The $d_{1/2}$ data are plotted in Fig. 25.

3.2.3. The ^{44}Ti/^{44}Sc radionuclide generator

Relevant nuclear data for this radionuclide generator system are given in Table 4 and the decay scheme of ^{44}Sc is given in Fig. 26.

3.2.3.1. Relevant nuclear properties

(a) Daughter photon emission

The high branching positron emission (94.27%) with maximum/mean positron energy of 1.5/0.6 MeV is followed by emission of intense 511 keV annihilation photons. This photon energy is adequate for penetrating metal walls of vessels and pipes for on-line measurements in industry. The annihilation character with two 511 keV photons moving in opposite directions allows for coincidence detection and positron emission tomography (PET). This offers

TABLE 5. MASS ATTENUATION COEFFICIENTS AND THE CORRESPONDING HALF-THICKNESS VALUES

Radionuclide	Gamma energy (keV)	Mass attenuation coefficient (μ/ρ, cm^2/g)	Mass attenuation coefficient (μ/ρ, cm^2/g)	Half-value thickness ($d_{1/2}$, cm)	Half-value thickness ($d_{1/2}$, cm)
		Iron	Water	Iron	Water
^{99m}Tc	141	0.2280	0.1541	0.39	4.50
^{172}Lu	181	0.1652	0.1421	0.53	4.88
^{113m}In	392	0.0953	0.1071	0.93	6.47
^{44}Sc, ^{68}Ga, ^{82}Rb	511	0.0834	0.0961	1.06	7.21
^{137m}Ba	661	0.0740	0.0862	1.19	8.04
^{144}Pr	697	0.0722	0.0843	1.22	8.22
^{82}Rb	776	0.0682	0.0800	1.29	8.67
^{172}Lu	912	0.0630	0.0742	1.40	9.34
^{68}Ga	1077	0.0580	0.0684	1.52	10.13
^{172}Lu	1093	0.0576	0.0679	1.53	10.20
^{44}Sc	1157	0.0559	0.0660	1.58	10.50
^{144}Pr	1489	0.0490	0.0578	1.80	11.99
^{144}Pr	2186	0.0415	0.0476	2.13	14.55

Note: Mass attenuation (or mass absorption) coefficients for the relevant gamma energies in iron (simulating steel) and water (simulating most aqueous and many organic solutions) and the half-thicknesses for the actual combinations of gamma energies and absorbers. The numbers are calculated on the basis of data in Ref. [36].

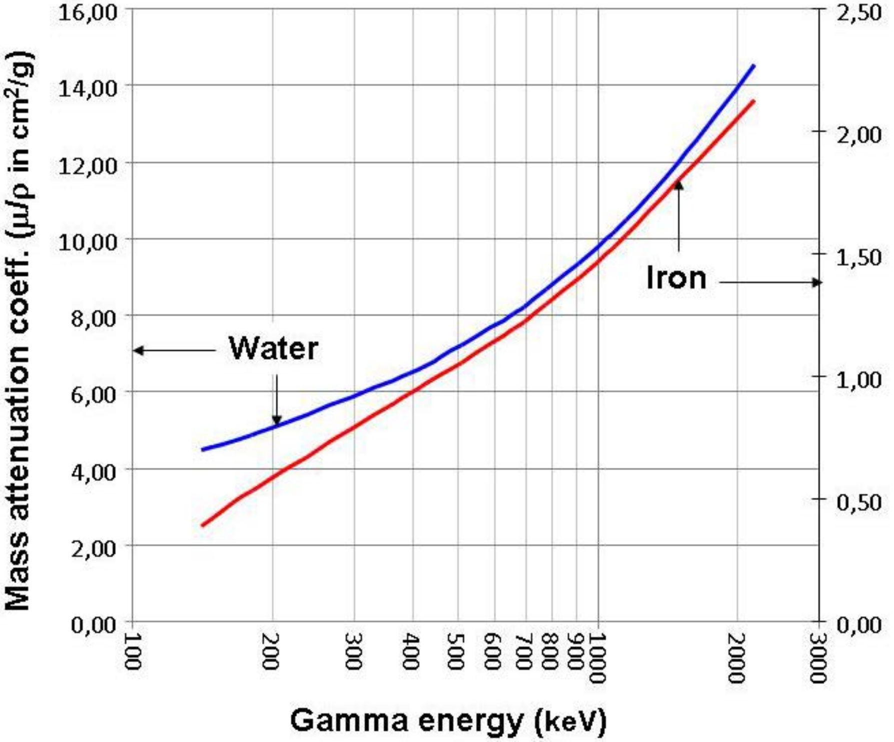


FIG. 25. Half-value thicknesses, d$_{1/2}$, *for water (which simulates with sufficient accuracy most aqueous solutions and a number of organic liquids) and iron (which simulates steels) as a function of gamma energy.*

potential for determination of the radionuclide distribution with high local resolution. In addition, the highly abundant 1157 keV photons (99.9%) allow for on-line measurements of morphological distributions. A gamma spectrum is shown in Fig. 27.

(b) Daughter half-life

The daughter half-life of 3.92 h is suitable for industrial measurements of medium duration.

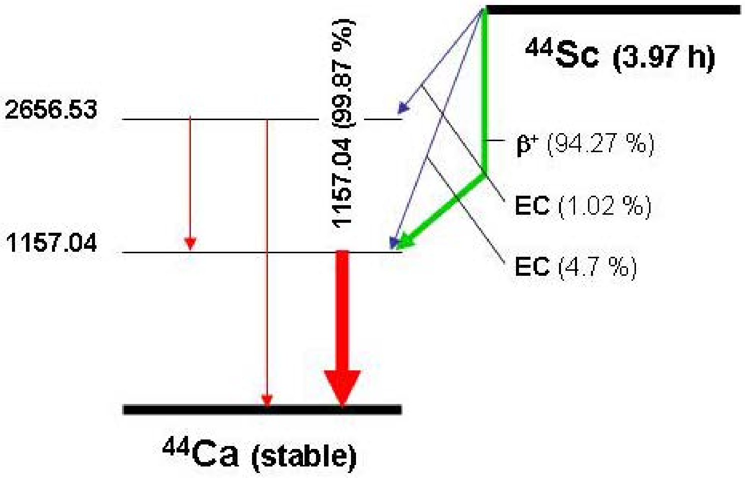


FIG. 26. Decay scheme of the ground state of ^{44}Sc to stable ^{44}Ca [37].

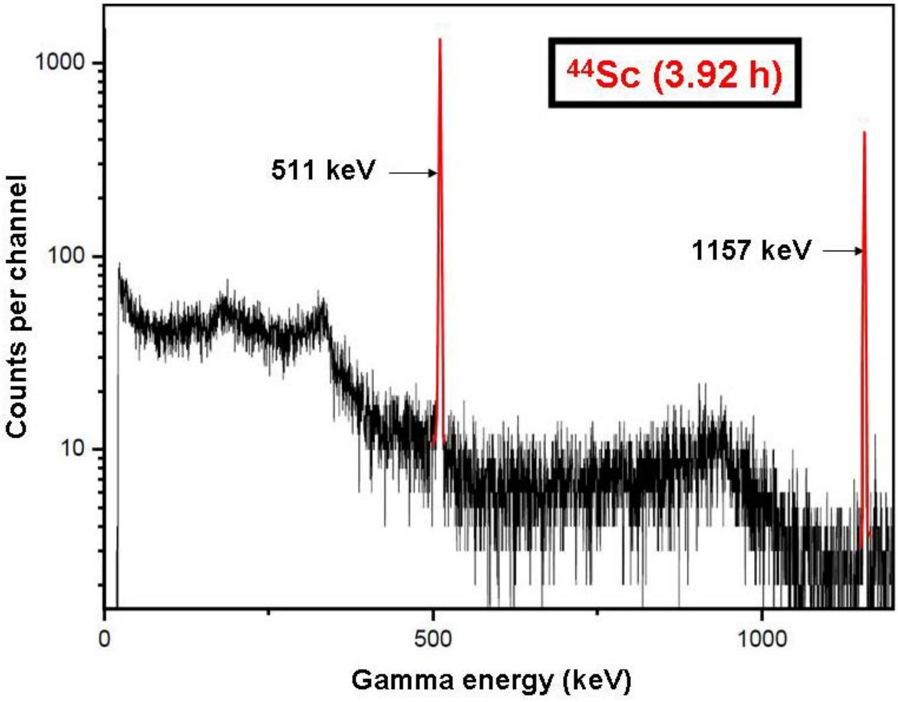


FIG. 27. Gamma spectrum of ^{44}Sc accumulated with an HPGe detector.

(c) Mother half-life

Mother half-life of 47.3 a offers long term application without replacing the generator. The lifetime of the generator is then dominated by the stability of the support (column) material, which again may be affected by the radiation field.

3.2.3.2. *Production and expected yield*

There are several possible production routes for ^{44}Ti. Examples are the ^{45}Sc(p,2n)^{44}Ti reaction and proton induced spallation on V or Cr targets. High production yields of several gigabecquerels per batch are achievable at accelerators, providing that sufficient proton flux and long term irradiations are available.

3.2.3.3. *Generator designs and elution strategy*

The ^{44}Ti is, for instance, fixed on an anion exchanger bed as titanate (TiO_3^{2-}). An early report [38] states that ^{44}Sc in oxidized form (Sc^{3+}) is eluted with 0.1M oxalic acid in 0.2M HCl. The radiochemical yield is 60–70%, the eluent volume 30–50 mL and a breakthrough of ^{44}Ti ~10^{-3}%.

Other reports on the ^{44}Ti/^{44}Sc generator system are:

- Zirconium oxide column bed, 0.01M HCl eluent, 42–46% radiochemical yield and parent breakthrough of 5×10^{-4}% [39];
- Solvent extraction with organic phase of 1% 1-phenyl-3-methyl-4-capryl-pyrazolone-5 in methyl-isobutyl-ketone, with 90% recovery in 10 mL eluent, ^{44}Ti breakthrough is measured at 10^{-6} [40];
- Polyantimonic acid inorganic ion exchanger and HCl/$(NH_4)C_2O_4$ eluent systems give 80% yield and breakthrough of 10^{-7} [41];
- Other metal oxides/HCl eluent systems are under investigation.

Recent developments with similar chemistry [42] have led to improved performance and the only currently working generator system. About 185 MBq (5 mCi) of ^{44}Ti were obtained by the reaction ^{45}Sc(p,2n)^{44}Ti in a cyclotron with a beam energy of 25 MeV and a current of 200 μA. Subsequently, ^{44}Ti was separated from the 1.5 g massive scandium target using cation exchange chromatography on a large column (^{44}Ti/^{Nat}Sc purification, BioRad AG-50Wx8, 200–400 mesh, H^+ form). Ion exchange studies have been performed systematically using AG-1x8 (200–400 mesh Cl^- form) and AG-50x8 (200–400 mesh H^+ form) resin in $H_2C_2O_4$/HCl solution in order to evaluate the potential of ion exchange based generators. The K_d values for both Sc and Ti were determined in batch experiments using different concentrations of HCl/oxalic acid mixtures. Both ^{44}Ti(^{44}Sc) and ^{46}Sc were used as isotopic tracers for Ti(IV) and Sc(III) distributions. The best conditions for efficient separation and for the design of generators are achieved by eluting the AG-1x8 resin with a mixture of 0.2M HCl and 0.1M $H_2C_2O_4$.

For subsequent ^{44}Sc labelling reactions, post-elution processing of the ^{44}Sc fraction may be required. The ^{44}Sc may, for example, be absorbed on-line directly and selectively from the eluate on a cation exchange cartridge. In this case, the mixture of reduced acid concentrations consisting of 0.06–0.008M HCl and 0.005M $H_2C_2O_4$ in the elution liquid offer the optimum K_d value for Sc(III) adsorption.

Further studies with this generator system concerned the effect of ‘direct’ and ‘reverse’ elution procedures. The ‘reverse’ washing procedure after each ‘forward’ elution using the same liquid mixture of 0.2M HCl and 0.1M oxalic acid promotes a long term stability of the ^{44}Ti/^{44}Sc generator. In addition, periodical washing with the same mixture allows for focusing the ^{44}Ti distribution on the ion exchange column. The effect is a smaller breakthrough of ^{44}Ti. The standard forward elution strategy resulted in an increasing breakthrough of ^{44}Ti, which amounts to 50% desorption of ^{44}Ti after about 30 elutions and an almost complete release of ^{44}Ti after 50 elutions. In contrast, the breakthrough of ^{44}Ti in the case of the reverse elution scheme is negligible for the first ten elutions, and increases only slightly in the following 40 elutions. The maximum breakthrough for the reverse elution/washing procedure for ^{44}Ti is about 0.2%. The elution procedure for the final 185 MBq ^{44}Ti/^{44}Sc generator defines an elution volume per batch of 20 mL of a mixture of 0.005M $H_2C_2O_4$ and 0.07M HCl.

3.2.3.4. From eluted radionuclide to ^{44}Sc labelled tracer

Post-elution processing studies were performed to reduce the volume and acidity of ^{44}Sc eluate from the ^{44}Ti/^{44}Sc generator and to reduce the amount of ^{44}Ti in the final product for the syntheses of ^{44}Sc labelled radiotracers.

To reduce the volume of the ^{44}Sc eluate, an additional column filled with the cationic resin BioRad AG-W50x8 (H^+ form) was connected with the primary generator in such a way that the eluate passed directly through the cation exchanger. In this process, the ^{44}Sc(III) is adsorbed on the cationic resin, from which it can be eluted by 3 mL of 0.25M ammonium acetate at pH4.0. This solution is now ready for labelling and complexation. Subsequently, the cation exchanger column is reconditioned by washing with 1 mL of 4M HCl and 1 mL of water in order to prepare the generator system for the next elution sequence.

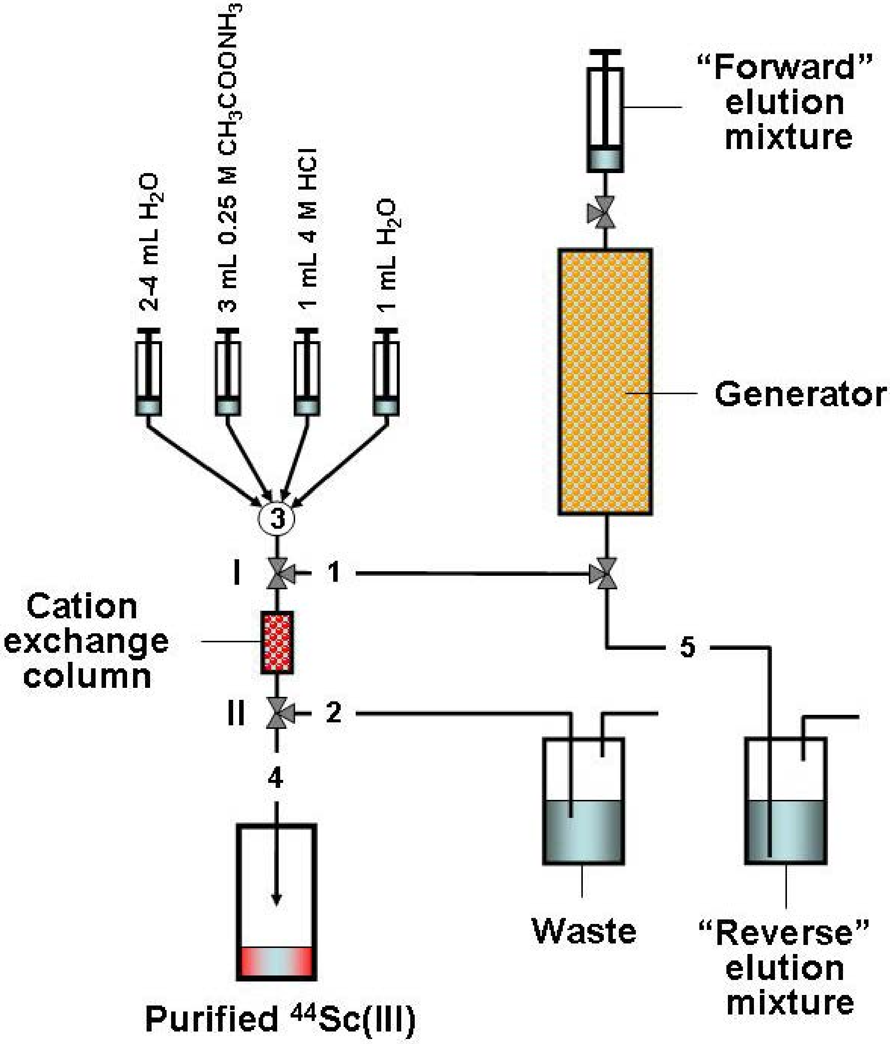


FIG. 28. Schematic diagram of ^{44}Ti/^{44}Sc generator together with post-elution processing of ^{44}Sc eluates and generator associated syntheses of ^{44}Sc labelled compounds [42].

Finally, around 160 MBq of ^{44}Sc is obtained daily in 3 mL 0.25M ammonium acetate buffer (pH4.0). The generator system is sketched in Fig. 28.

The operational procedure is as follows:

- The 20 mL generator elution liquid, consisting of a mixture of 0.005M $H_2C_2O_4$ and 0.07M HCl, is percolated through the ^{44}Ti/^{44}Sc generator (forward elution) with a flow rate of 1 mL/min by use of a syringe.
- The eluted ^{44}Sc(III) adsorbs on-line on the small cationic cartridge between the three-way valves I and II. The ^{44}Sc(III) is retained on the column, whereas the generator eluate continues to the waste vial (line 2).
- The cation exchange resin is then washed with 2–4 mL of H_2O in order to remove the remaining traces of the initial eluate solution. The wastewater is directed to the waste vial. Subsequently, 5 mL of air is blown through the cation exchanger column, and the ^{44}Sc is eluted by pressing slowly 3 mL of 0.25M ammonium acetate buffer (pH4.0) slowly (0.7 mL/min) through the column with a 2 min break after each 1 mL. The ^{44}Sc is collected in an 11 mL glass reaction vial through flow line 4.
- Again, air is passed through the column to remove traces of the ammonium acetate buffer solution remaining in the interstitial volume of the column and in the dead volume of the connecting flow lines.
- Reconditioning of the cation exchange column is achieved by washing with 1 mL of 4M HCl and finally by 1 mL H_2O through flow line 4.
- Finally, the generator column is conditioned by sucking 20 mL of a liquid mixture consisting of 0.2M HCl and 0.1M oxalic acid through the column from the ‘reverse’ elution mixture reservoir by means of a syringe. Thus, the column is prepared for the next elution.

Using this post-processed fraction, radiolabelling yields of >96% were obtained when 30 μL (21.1 nmol) of DOTATOC was added to 3 mL of the ^{44}Sc eluate (pH4.0) and heated in the oil bath for 25 min at 95ºC [43]. Microwave assisted labelling speeded up ^{44}Sc complexation with DOTATOC. After 1 min of microwave heating reaction, the yield was >95% and increased up to 98% during the next 2 min.

3.2.3.5. Availability and current status

Low production rates require production of ^{44}Ti at dedicated accelerators. Nevertheless, ^{44}Ti in >370 MBq (10 mCi) batches are commercially available. To date, no commercial ^{44}Ti/^{44}Sc radionuclide generator systems are available. Research and development are currently in progress.

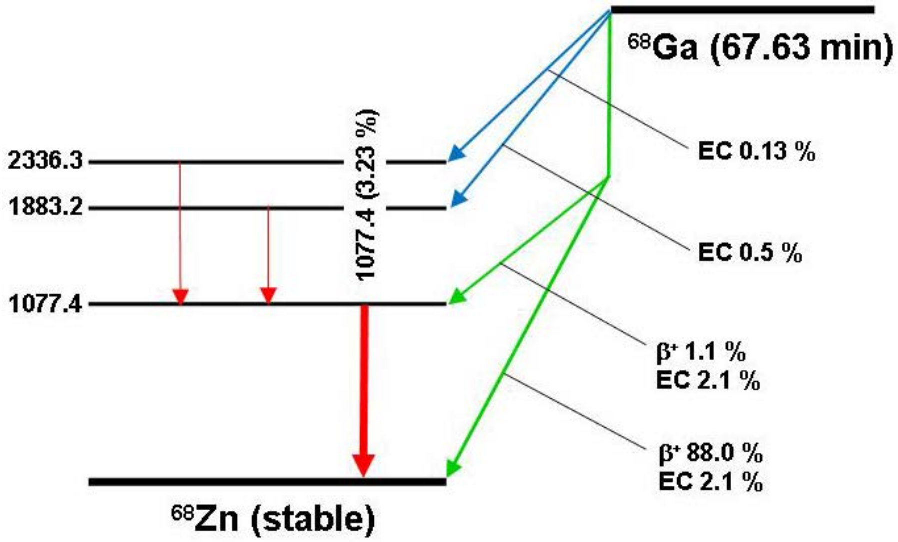


FIG. 29. Decay scheme of ^{68}Ga to stable ^{68}Zn [44].

3.2.3.6. Possible applications

The ^{44}Ti/^{44}Sc radionuclide generator provides an optimum choice of long shelf life, a medium half-life and an optimum decay mode of the daughter. Modules might be designed for synthesis of ^{44}Sc labelled environmental and industrial tracers. The 3.92 h half-life of ^{44}Sc and the high positron branching ratio of 94% may stimulate the application of ^{44}Sc labelled tracers for the study of flow structures in various types of mass transport by the PET technique. In addition, the high energy gamma emission of 1157 keV appears to be ideal for industrial purposes due to the penetrating power of these gamma rays. By measuring the counting rate ratio of the 511 keV and 1157 keV photons using detection equipment with spectroscopic properties, some information about the depth distribution of the traced mass flow components can be achieved.

3.2.4. The ^{68}Ge/^{68}Ga radionuclide generator

Relevant nuclear data for this radionuclide generator system are given in Table 4 and the decay scheme of ^{68}Ga is given in Fig. 29.

3.2.4.1. Relevant nuclear properties

(a) Daughter photon emission

The parent radionuclide ^{68}Ge decays via electron capture to ^{68}Ga ($T_{½}$ = 270.95 d). The ^{68}Ga formed subsequently decays ($T_{½}$ = 67.63 min) to stable ^{68}Zn (see the decay scheme of ^{68}Ga in Fig. 29). Gallium-68 is a positron emitter with 89% positron branching accompanied by low intensity photon emission (1077.4 keV, 3.23%). The ^{68}Ge itself does not emit significant photon radiation.

The high branching positron emission (89%) with maximum and mean positron energies of 1.9 and 0.79 MeV, respectively, provides intense 511 keV annihilation photon emission. This photon energy is adequate for transmission through thick metal walls and detection outside of industrial vessels or tubings, and is therefore suitable for on-line measurements in the industry. As for the case of ^{44}Sc above, the annihilation character with two 511 keV photons moving in opposite directions allows for coincidence detection and PET. This offers the potential for determination of the radionuclide distribution with high local resolution. In addition, the counting rate ratio of the 511 keV photon and the 1077.4 keV gamma ray allows for on-line measurements of depth distributions of the traced material in an industrial mass flow. A gamma spectrum is shown in Fig. 30.

(b) The daughter half-life

The daughter half-life of 67.63 min offers industrial measurements of medium to short duration.

(c) The mother half-life

The mother half-life of 270.95 d offers a shelf life of the generator of about 1.5–2 a.

(d) Main production routes and expected yield of the mother radionuclide

There are several possible production routes for ^{68}Ge. Two of the most convenient are the ^{66}Zn(α,2n) reaction (^{66}Zn abundance: 27.8%) and the ^{69}Ga(p,2n) reaction (^{69}Ga abundance: 60%). The first reaction has a yield of about 37–74 kBq (1–2 μCi) μA/h, while the second reaction has a higher yield [about

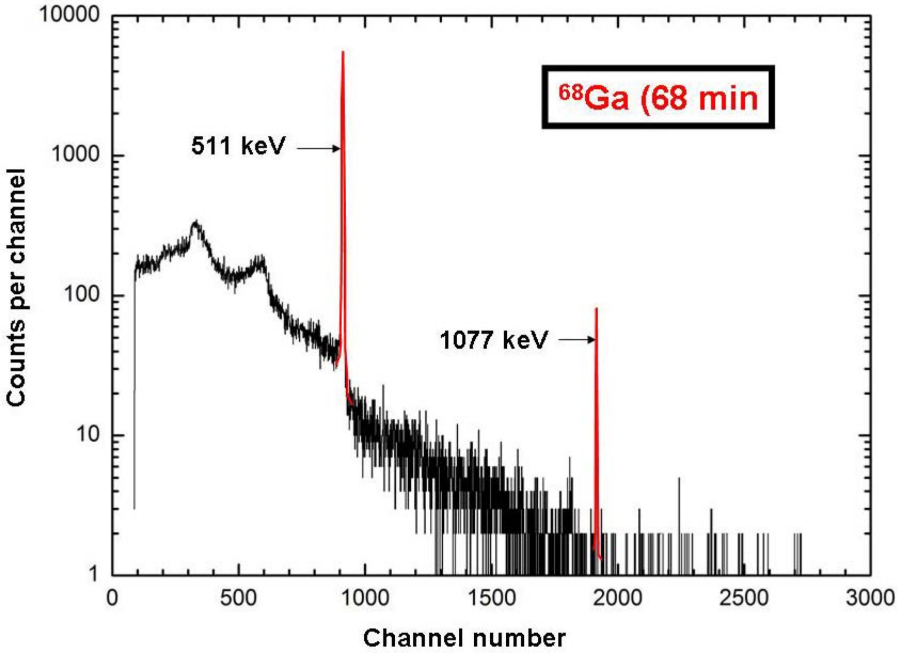


FIG. 30. Gamma spectrum of ^{68}Ga accumulated with an HPGe detector.

740 kBq (20 μCi)], and would be the primary choice for routine production for beam energies around 25 MeV [45]. The useful energy range for production is 35–13 MeV (see the excitation function in Fig. 31). High production yields of several gigabecquerels per batch are achieved at accelerators providing sufficient proton flux (>100 μA).

3.2.4.2. Generator systems, separation techniques and chemical form of the daughter

The chemical details of the individual ^{68}Ge/^{68}Ga generator systems developed during the last 50 years (as mainly stimulated by radiopharmaceutical chemistry and nuclear medical applications of ^{68}Ga labelled tracers for biological systems) are given in Table 6.

Early attempts to adopt liquid–liquid extraction chemistry for routine generator use were not sustainable and concentrated mainly on column based generators.

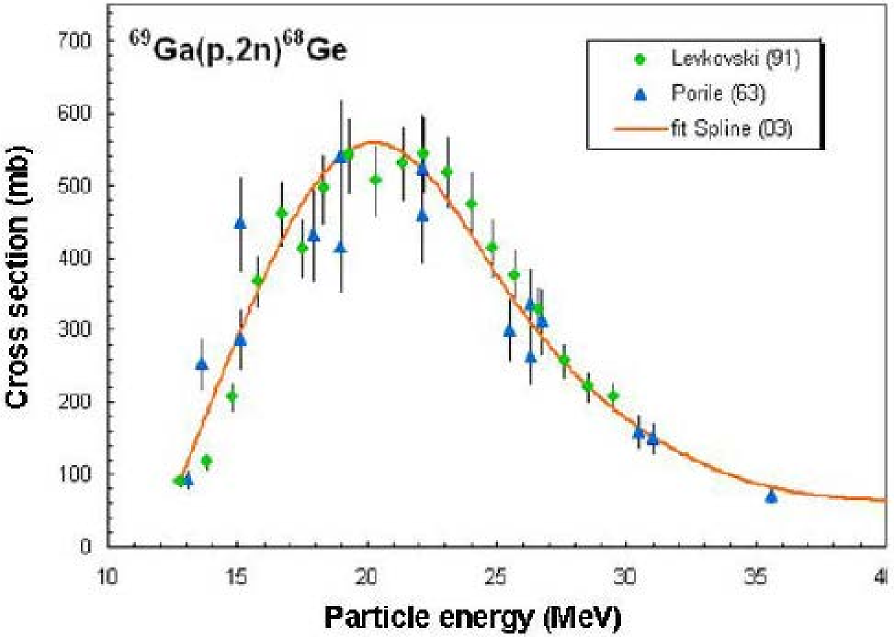


FIG. 31. Excitation function for the reaction ^{69}Ga(p,2n)^{68}Ge [46].

The early column based generator systems eluted ^{68}Ga with 0.005N EDTA solutions from ^{68}Ge, absorbed on alumina or zirconium oxides [46]. It provided non-charged ^{68}Ga–EDTA complexes. Elution efficiencies have been almost quantitative, and the thermodynamically very stable Ga–EDTA complex appears to be a suitable water tracer at moderate temperatures (<80–90°C).

However, the high complex stability prevented the direct use of this eluate for further ^{68}Ga labelling reactions. Analogously, ^{68}Ge was retained on antimony oxide Sb_2O_5 and ^{68}Ga was eluted with oxalate solutions [51]. Anion exchange resins using dilute hydrofluoric acid solutions as eluent allowed high purity separations due to the significant differences of distribution coefficients of the elements [53]. The breakthrough of ^{68}Ge was $<10^{-4}$ for up to 600 elutions, and the ^{68}Ga yield was >90%. In all these cases, further application of the generator eluate for ^{68}Ga labelling reactions was not possible or difficult due to the strong gallium fluoride complex.

As Ge(IV) is known to form a very stable complex with phenolic groups, its adsorption on a 1,2,3-trihydroxybenzene (pyrogallol)–formaldehyde resin was utilized to elute ^{68}Ga^{3+} (or a mixture of chloride complexes) in strong (1M) hydrochloric acid [55]. Average yields of ^{68}Ga of 75% during a period of 250 d were reported. The Ge breakthrough was <0.5 ppm, with no detectable radiolytic

TABLE 6. THE $^{68}Ge/^{68}Ga$ GENERATORS FROM PUBLISHED LITERATURE

Column material	Elution agent	Elution yield	Reference
Al_2O_3, ZrO_2	Elution agent: 0.005M EDTA Ga eluted as EDTA complex	Ga recovery 70%	[47]
Hydrated MnO_2	Elution agent: 1M HNO_3 Probably eluted as Ga^{3+}	Ge ret.>99%, Ga elut.>97%	[48]
Hydrated MnO_2	Elution agent: 1M $HClO_4$ Probably eluted as Ga^{3+}	Ge ret.>99%, Ga elut.>97%	[48]
Hydrated antimony pentoxide	Elution agent: 14M HNO_3 Probably eluted as Ga^{3+}	Ge ret.>99%, Ga elut.>97%	[48]
Acidic aluminium oxide, Al_2O_3	Elution agent: 1M $HClO_4$ Probably eluted as Ga^{3+}	Ge ret.>99%, Ga elut.>97%	[48]
Tin dioxide, SnO_2	Elution agent: 1M HCl Probably eluted as Cl complex	Ge ret.>99%, Ga elut.>97%	[48]
Tin dioxide, SnO_2	Elution agent: 1M HNO_3 Probably eluted as Ga^{3+}	Ge ret.>99%, Ga elut.>97%	[48]
$Al(OH)_3$, $Fe(OH)_3$	Elution agent: weak HCl Probably eluted as Cl complex	—	[49, 50]
SnO_2, ZrO_2, TiO_2	Elution agent: 1M HCl Probably eluted as Cl complex	Ga recovery 70–80% Ge leakage 10^{-6}–10^{-5}% per bolus	[51]

TABLE 6. THE ^{68}Ge/^{68}Ga GENERATORS FROM PUBLISHED LITERATURE (cont.)

Column material	Elution agent	Elution yield	Reference
No column: solvent extraction process	Extraction: 8-hydroxy-quinoline in chloroform at pH5 with Ge^{4+}holdback carrier followed by evaporation to dryness	Ga recovery as Ga–oxine complex: 70–80% Ge breakthrough <0.003%	[51]
Antimony pentoxide Sb_2O_5	Elution agent: oxalate solution Probably eluted as oxalate complex $Ga(Ox)_x^{3-x}$ [a]	—	[52]
Tin dioxide , SnO_2	Elution agent: 1M HCl Probably eluted as Cl complex	Ga recovery 75–90% Ge leakage <0.0002% per elution	[53]
Anion exchanger, BioRad AG1-8x, probably sorbed as GeF_6^{2-}	Elution agent: 0.01M HF Ga eluted as GaF_x^{3-x} [a]	Ga recovery >90% Ge leakage $<10^{-4}$ for 600 elutions	[54]
Anion exchanger, BioRad AG1-8x, probably sorbed as GeF_6^{2-}	Elution agent: 0.5M HCl Form of Ga in eluate as Cl complex	Ga recovery 75% Ge leakage <0.5 ppm	[55]
Hydrous zirconium oxide	Elution agent: not given	Ga recovery 30–40% Ge leakage <10 ppm	[56]
Anion exchanger, probably sorbed as GeF_6^{2-}	Elution agent: weak acid, no further details known	Recovery and leakage not given	[56]
Pyrogallol/formaldehyde condensation product	1M HCl Probably eluted as Cl complex	Ga recovery 60% Ge leakage $<10^{-3}$%	[55]
Pyrogallol/formaldehyde condensation product	Elution agent: 0.3M HCl Probably eluted as Cl complex	Ga recovery 60% Ge leakage $<10^{-3}$% per elution	[57]

TABLE 6. THE ^{68}Ge/^{68}Ga GENERATORS FROM PUBLISHED LITERATURE (cont.)

Column material	Elution agent	Elution yield	Reference
Tin(II) oxide, SnO	Elution agent: HCl (conc. not given) Probably eluted as Cl complex	Recovery and leakage not given	[58]
Tin(IV) dioxide, SnO_2	Elution agent: 1M HCl Probably eluted as Cl complex	Recovery and leakage not given	[59]
Tin(IV) oxide, SnO_2	Elution agent: 1N HCl Ga eluted as Cl complex Ether extraction from 6N HCl Probably extracted as $GaCl_3$	Recovery and leakage not given	[60, 61]
Iron(III) oxide, α-Fe_2O_3 (alpha-ferric oxide)	Elution agent: HCl (pH2) Probably eluted as Cl complex	Ga recovery 50–70% Ge leakage $<2 \times 10^{-4}$% elution	[62]
Cerium(IV) oxide, CeO_2	Elution agent: 0.02M HCl Probably eluted as Cl complex	Ga recovery 45% Ge leakage <0.007	[63]
Cerium(IV) oxide, CeO_2	Elution agent: 0.02M HCl Probably eluted as Cl complex	Ga recovery 56% in 0.5 mL Ge leakage not detected	[64]
Tin(IV) oxide, SnO_2	Elution agent: 1M HCl Probably eluted as Cl complex	Ga recovery 55–75% Ge leakage $\sim 10^{-7}$	[65]
Iron(III) oxide, α-Fe_2O_3 (alpha-ferric oxide)	Elution agent: 0.05M HCl Probably eluted as Cl complex	Ga recovery 90% Ge leakage $<1.8 \times 10^{-4}$% elution	[66]

TABLE 6. THE ^{68}Ge/^{68}Ga GENERATORS FROM PUBLISHED LITERATURE (cont.)

Column material	Elution agent	Elution yield	Reference
Hydrated SnO_2	Elution agent: 0.1M HCl Probably eluted as Cl complex	Ga recovery ~88% in 2 mL eluate Ge leakage not given	[67]
Tin(IV) oxide, SnO_2	Elution agent: 0.1M HCl Probably eluted as Ga^{3+}	Ga recovery 60–70% in 4 mL eluate Ge leakage not given	[68]
Al_2O_3 treated with 0.1M NaOH	Elution agent: 0.1M HCl Probably eluted as Cl complex	Ga recovery 50–70% Ge leakage 2×10^{-4}%	[69]
Styrene–divinylbenzene co-polymer with N-methylglucamine groups	Elution agent: citric or phosphoric acid Ga eluted as a low affinity complex	Elution yield not given Ge leakage <4×10^{-4}%	[70]
Hydrous ZrO_2	Elution agent: 0.1N HCl Probably eluted as Cl complex	Elution yield 30–40% Ge leakage 10^{-3}% at 0.5 mL/min	[56]
Hydrous or crystalline SnO_2	Elution agent: 1M HCl Probably eluted as Cl complex	Elution yield ~65% Ge leakage ~6×10^{-4}%	[71]

[a] Where x = 1 or 2.

by-products for a 370 MBq (10 mCi) generator. The pyrogallol–formaldehyde resin was found to be resistant to radiation degradation. An organic macroporous styrene–divinylbenzene co-polymer containing N-methylglucamine groups was developed to provide ^{68}Ga with a solution of a low affinity gallium chelating ligand such as citric or phosphoric acid [69]. The ^{68}Ge breakthrough was <0.0004% of the ^{68}Ge adsorbed on the resin. These generator strategies are currently taken up by a commercial system distributed by ITG, Germany.

Ionic ^{68}Ge/^{68}Ga generators were developed, avoiding the formation of a stable ^{68}Ga–ligand complex in the eluate system. This strategy is achieved for both $^{68}Ga(OH)_4^-$ and $^{68}Ga^{3+}$, providing eluates. Indeed, ^{68}Ga has been eluted in 0.1M NaOH solution as gallate anion from alumina columns [72]. A comparison of performances of alumina/EDTA, alumina/NaOH and tin oxide/HCl generators [60], however, indicated superior characteristics for the latter system in terms of ^{68}Ge breakthrough and $^{68}Ga^{3+}$ elution efficiency in 1M hydrochloric acid. The ^{68}Ge/^{68}Ga generators, which provide $^{68}Ga^{3+}$ ions (or $^{68}GaCl_x^{3-x}$ complexes where x = 1 or 2) in the eluate, are based on inorganic matrices like Al_2O_3, $Al(OH)_3$, Fe_2O_3, $Fe(OH)_3$, SnO_2, ZrO_2, TiO_2 and CeO_2. The SnO_2 showed the best performance in terms of low ^{68}Ge breakthrough (10^{-6}–10^{-5}% per elution) and high ^{68}Ga elution efficiency (70–80% per elution) with 1M hydrochloric acid. Figure 32 provides a sketch of the various development attempts towards the present-day design of ^{68}Ge/^{68}Ga radionuclide generators for routine application in the medical area, and which, with some adjustment, can also be applied for industrial purposes.

3.2.4.3. Commercial availability of the ^{68}Ge/^{68}Ga radionuclide generator

For commercial generator productions, a modified TiO_2 phase has been used by Cyclotron Ltd, Obninsk, Russian Federation, since about 2000. These generators are eluted with 0.1N HCl and show initial ^{68}Ga elution yields of about 80%, with ^{68}Ge breakthrough of about 1×10^{-3}%. These data deteriorate over time (e.g. after about 1 year) or with increasing number of elutions (e.g. 200), approaching values of about 50% of ^{68}Ga elution and about 10^{-2}% of ^{68}Ge breakthrough.

A similar generator is available from Eckert & Ziegler known as ‘IGG 100’ (Fig. 33(a)), which provides improved elution characteristics. The elution efficiency of ^{68}Ga from the fresh generator is ~70%. This value can drop somewhat if the generator is not eluted for a few days. The ^{68}Ge breakthrough typically ranges from 3×10^{-5}%, when new, to 5.0×10^{-3}% after 200 elutions. Breakthrough can increase up to 1.0×10^{-3}% if the generator is not eluted for a few days. After a break in use, the generator should be pre-eluted with 10 mL of 0.1M HCl 1 d prior to the intended use.

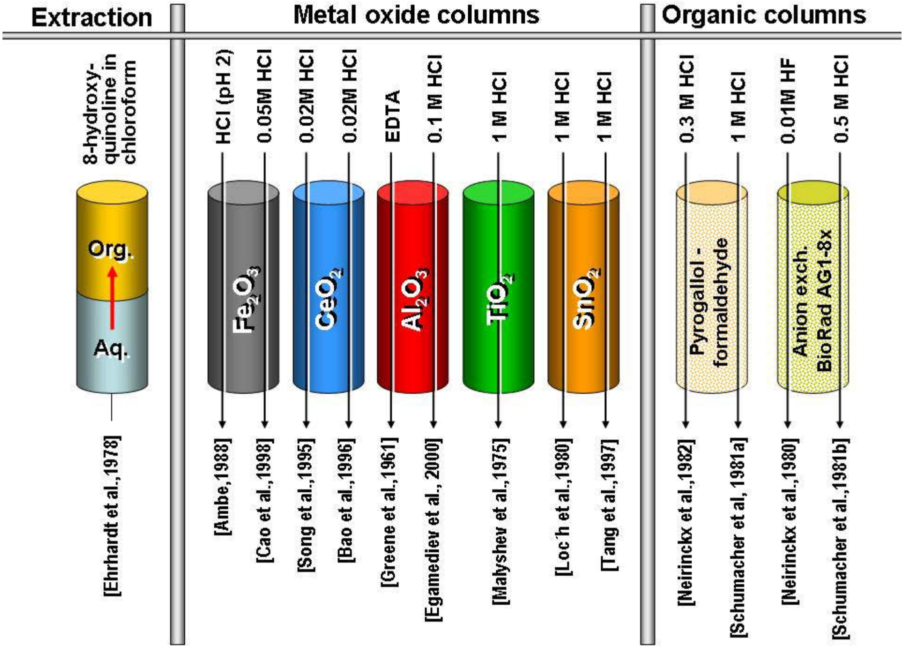


FIG. 32. Summary illustration of some published column concepts: liquid–liquid extraction (left), column based with inorganic column fill material (centre) and column based with organic (polymer) column filling material (right).

Other systems using a SnO_2 based solid phase are produced at Themba Labs, Republic of South Africa, and delivered by IDB Holland B.V., see Fig. 33(b). The generator should be eluated with 0.6M HCl. The yield in 5 mL of eluant is not <80%. The ^{68}Ge breakthrough is <0.002% at the reference date.

Recently, the China Institute of Atomic Energy (CIAE) started delivery of a few test series of generators also based on SnO_2 support. The generators have been made available to participants in a recent IAEA CRP entitled Evaluation and Validation of Radionuclide Generator-based Radiotracers for Industrial Applications, which ended in 2011.

Typical performance data for these generators are displayed in Figs 34 and 35. The generators are produced without the need for sterile conditions because, for most industrial applications, this is not a requirement. Hence, the production process can be made somewhat simpler than for generators intended for medical use, and this should be reflected in a somewhat cheaper price. The elution yield, which is initially around 60%, seems to deteriorate somewhat, but not dramatically, in the long term, as displayed in Fig. 35.

FIG. 33. (a) The IGG 100 ^{68}Ge/^{68}Ga radionuclide generator, Eckert & Ziegler AG, Berlin. (b) The ^{68}Ge/^{68}Ga generator produced by iThemba Labs and delivered by IDB Holland B.V.

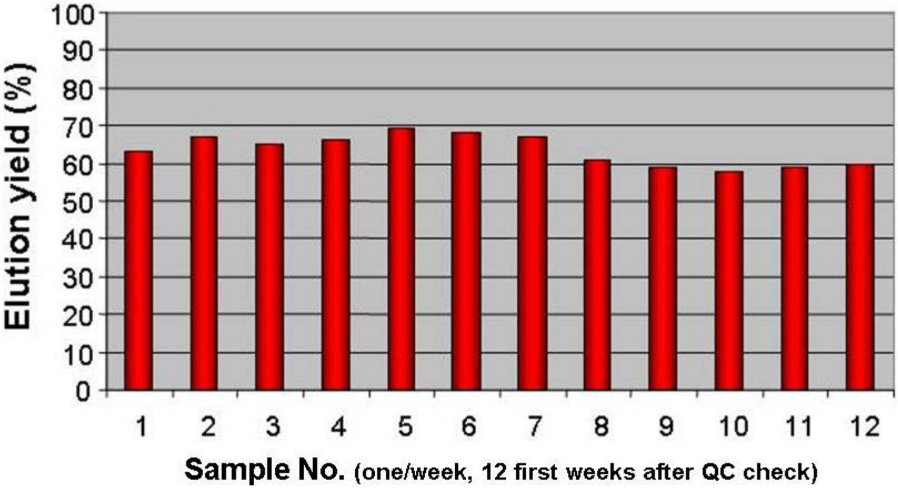


FIG. 34. Typical elution yield of the CIAE ^{68}Ge/^{68}Ga radionuclide generator over the first 12 weeks after the generator was received in the IFE testing laboratory [73].

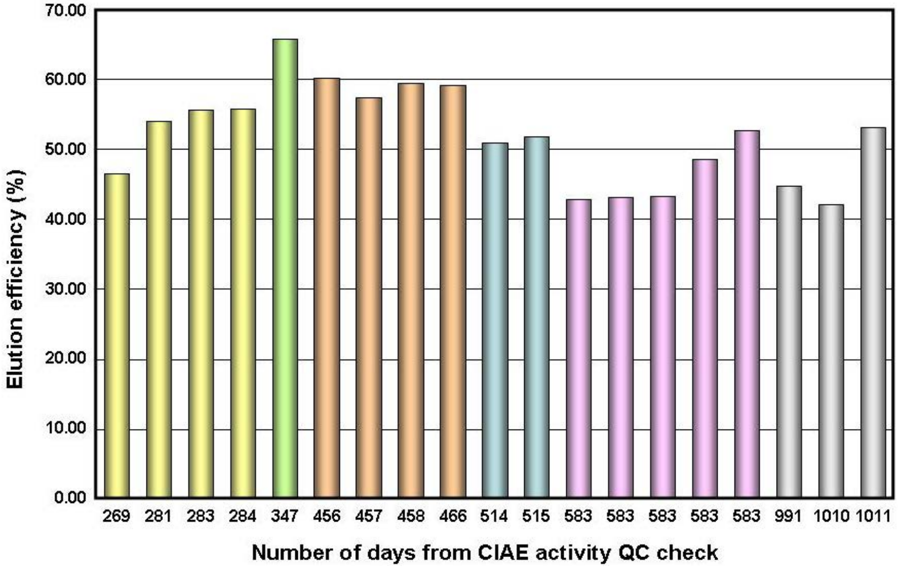


FIG. 35. Example of measured elution yield of the CIAE ^{68}Ge/^{68}Ga radionuclide generator over the first 3 years after the generator quality check at the production laboratory. A 3 year period is about the useful lifetime of a ^{68}Ge/^{68}Ga generator. The generator has been eluted several times within short time periods that are spaced by several months [74].

Variations in ^{68}Ge breakthrough over the first 11 elutions (one per week) of the CIAE produced generator are given in Fig. 36.

Part of the breakthrough is due to tiny particles from the primary column bed containing ^{68}Ge. An acceptable breakthrough fraction should be $<10^{-5}$. Hence, an improved bed filter support and a secondary (catcher) column should be implemented for this generator.

An example of an elution curve is given in Fig. 37. Around 80% of the activity is collected in a volume of 3 mL (fraction 6–8). This is an acceptably small volume for further complexation reactions.

The ^{68}Ge/^{68}Ga radionuclide generator has been investigated for almost 50 years, since the cyclotron independent availability of positron emitting ^{68}Ga via the ^{68}Ge/^{68}Ga system had always attracted research in basic nuclear chemistry, as well as in radiopharmaceutical chemistry. However, it took decades and generations of research to finally reach a level of ^{68}Ge/^{68}Ga radionuclide generator designs adequate to the modern requirements of radiometal labelling chemistry. This landmark can directly be associated to the radiochemical concept of providing cationic ^{68}Ga^{3+} rather than ^{68}Ga fluoride, oxalate or EDTA complexes as eluates. In about 2001, Cyclotron Ltd, Obninsk, Russian

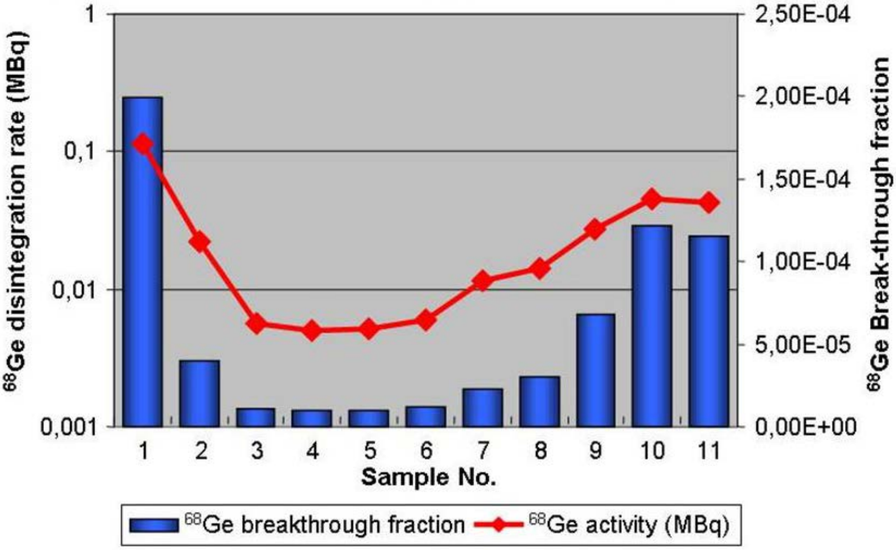


FIG. 36. Example of ^{68}Ge breakthrough during elution of the CIAE ^{68}Ge/^{68}Ga radionuclide generator during the first 11 weeks after the generator was received in the IFE testing laboratory [73].

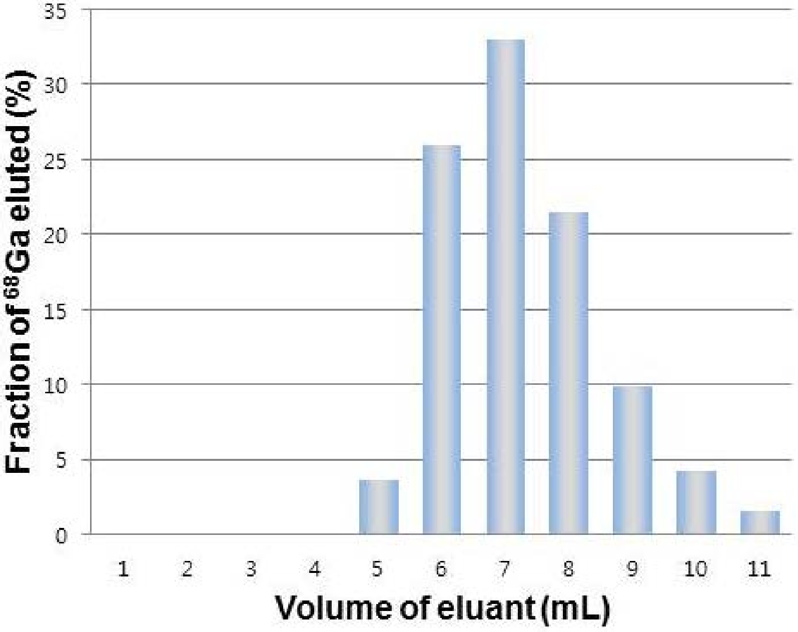


FIG. 37. Example of a ^{68}Ga elution curve of the CIAE ^{68}Ge/^{68}Ga radionuclide generator [75].

Federation, offered such systems with ^{68}Ge absorbed on modified TiO_2 columns and ^{68}Ga eluted using 0.1N HCl.

The exciting perspective for the ^{68}Ge/^{68}Ga radionuclide generator system, however, asks for systematic chemical, radiochemical, technological and radiopharmaceutical efforts, to guarantee reliable, highly efficient and industrially approved ^{68}Ge/^{68}Ga generator systems. The production of the radionuclide generator parent, its separation from the target material, and the chemical and technical concept of the separation of the daughter radionuclide are factors that result in an efficient and easy application of the generator. The IAEA has recently initiated a comprehensive review on the production of several generator mother nuclides which includes discussion of ^{68}Ge [76]. The radiochemistry of ^{68}Ge was covered comprehensively in 1996 [77], and a more recent compilation on radionuclide generators systems including ^{68}Ge/^{68}Ga can be found in Ref. [78]. Recent reviews on the ^{68}Ga radiopharmaceutical and coordination compounds appeared in 2008 [79, 80].

3.2.4.4. Possible applications

Post-processing systems have been developed for fast generator processing, including syntheses and purification of ^{68}Ga tracers. In parallel with the increasing interest for PET imaging related to medicine, there is a potential for application of the positron annihilation radiation and the PET method also in various industrial examinations.

Gallium-68 tracers suitable for aqueous, organic and solid phase tracers for industrial and environmental examinations have partly been developed already, and this development continues. One main feature of the generator is that it can be used three to four times per day at close to maximum activity, allowing repeated studies. Another feature of this radionuclide is the photon energy of 1077 keV, which is coincident with the annihilation radiation, allowing for additional in-depth information from the study of volume flow with ‘ordinary’ RTD measurement techniques.

3.2.5. The ^{82}Sr/^{82}Rb radionuclide generator

Relevant nuclear data for this radionuclide generator system are given in Table 6 and the partial decay scheme of ^{82}Sr is given in Fig. 38.

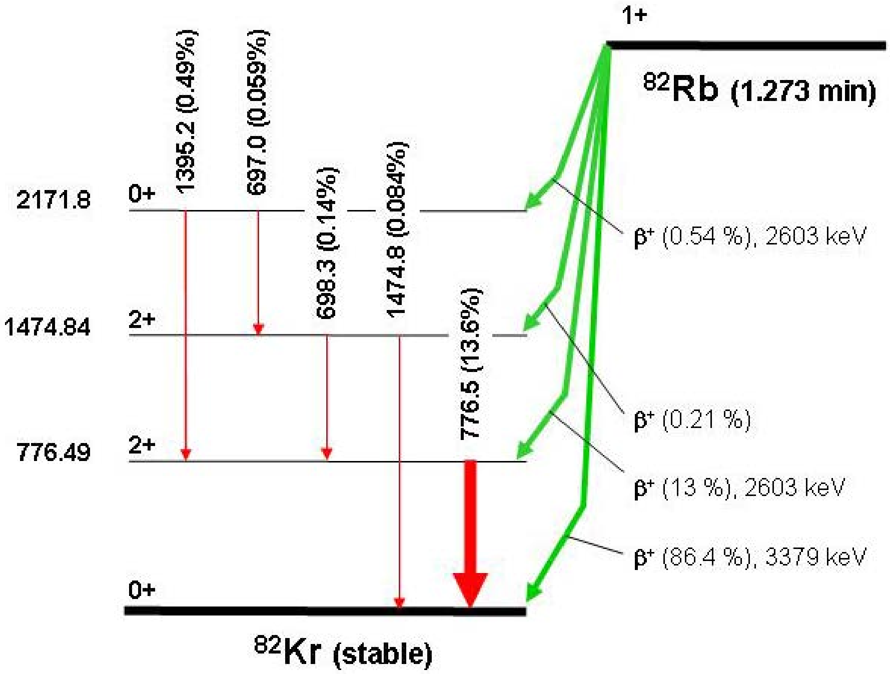


FIG. 38. Decay scheme of ^{82}Rb to stable ^{82}Kr [81]. Newer data suggest that the absolute intensity of the 776.5 keV gamma ray is 15.1%.

3.2.5.1. Relevant nuclear properties

(a) Photon emission of the daughter radionuclide

The ^{82}Rb has a high branching positron emission (96%) with maximum/mean kinetic energy of 3.38 MeV, providing intense 511 keV annihilation photon emission in addition to a relatively strong gamma ray at 776.5 keV with an intensity of about 15%. The photon energies are adequate for transmission through thick metal walls and detection outside of the vessel or tubing, and therefore suitable for on-line measurements in the industry. Annihilation character might allow for coincidence detection with potential of determination with high local resolution of the radionuclide distribution. The single photon, together with the annihilation radiation, provide for some depth resolution close to the vessel walls. A gamma spectrum is given in Fig. 39.

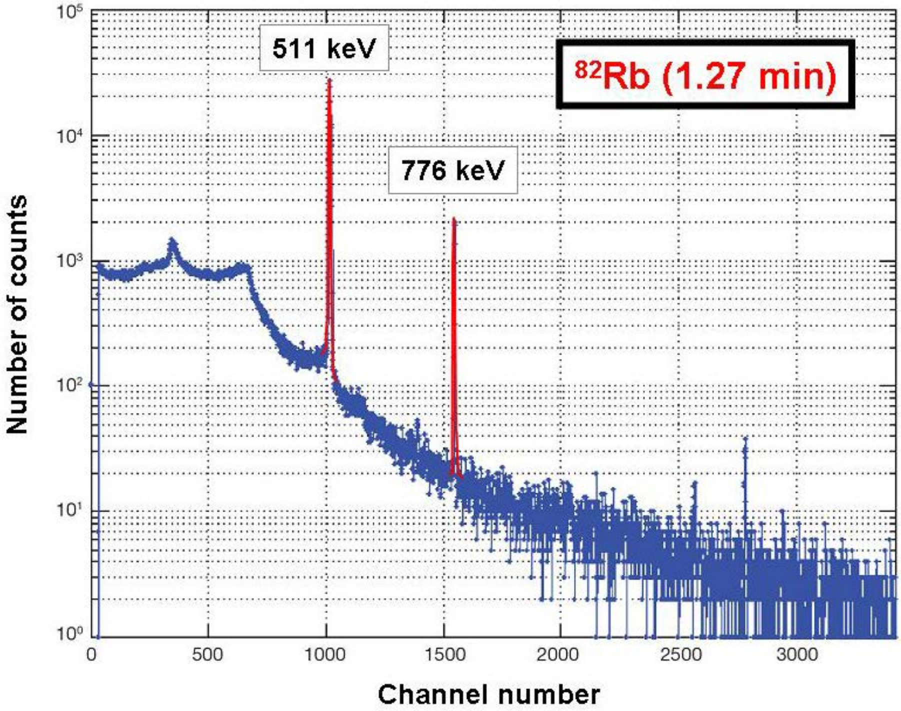


FIG. 39. Gamma spectrum of ^{68}Ga accumulated with an HPGe detector.

(b) Half-life of the daughter radionuclide

The daughter half-life of 1.273 min offers industrial measurements of very short duration and provides for frequent measurement repetitions.

(c) Half-life of the mother radionuclide

The mother half-life of 25.6 d offers the generator a shelf life of about 2 months.

(d) Main production route and expected yield

The most common production route of ^{82}Sr is the ^{85}Rb(p,4n)^{82}Sr reaction, but it is also produced by the ^{82}Kr(^{3}He,3n)^{82}Sr reaction and the ^{Nat}Mo(p,spall.) process. The optimal proton energy range for the (p,4n) reaction is 65–40 MeV, the maximum on the excitation function being at about 50 MeV (Fig. 40). One of

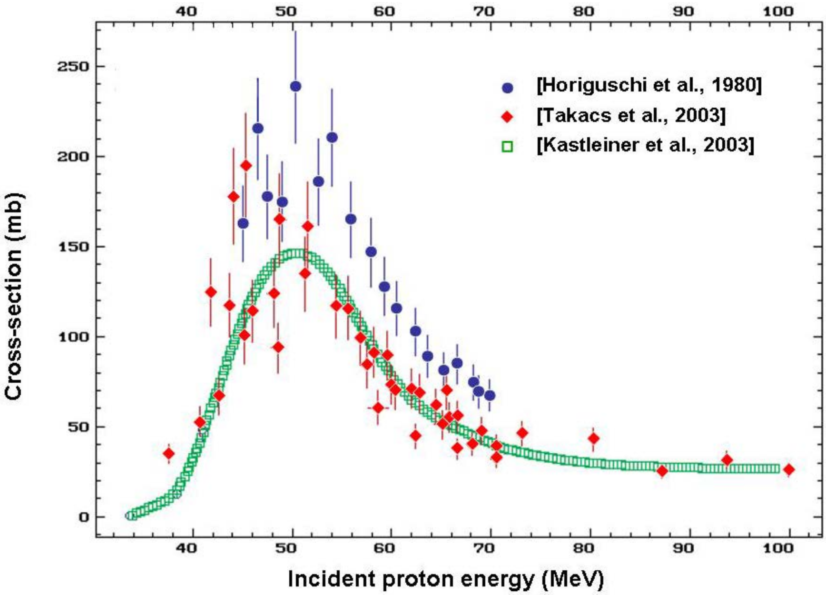


FIG. 40. Excitation function for the ^{85}Rb(p,4n)^{82}Sr reaction [44].

the important contaminants is ^{85}Sr, which is produced simultaneously when natural rubidium is used as the target material. Recently, Qaim et al. [82] determined that an incident proton energy of 60 MeV or above is recommended. The ^{85}Sr impurity then corresponds to <20%.

Some data on target materials and irradiation yields are summarized in Table 7.

According to Fig. 41, the excitation curve maximum for the ^{82}Kr(^{3}He,3n) reaction is around 32–33 MeV, and there is useful production down to about 22 MeV. However, the shape of the curve above 34 MeV is not considered. When natural krypton is used as the target material, the ^{82}Sr production begins at approximately 18 MeV and increases to a maximum of 80 mb at 62 MeV [83]. The ^{83,85}Sr are contaminants simultaneously produced in relative amounts that depend on the utilized effective energy range, and there is no energy range that allows pure ^{82}Sr production alone.

The calculated thick target yield at end of bombardment from 90–20 MeV was 1290 kBq·μA^{-1}·h^{-1}. At low beam intensities (up to 3 μA), the experimental yields were close to the values calculated from the excitation function. However, at 15 μA, the end of bombardment production yield decreased by 50% to 630 kBq·μA^{-1}·h^{-1} owing to gas density reduction in the beam strike area. For the spallation reaction on the Mo target, there is a requirement for 600–800 MeV protons. Commercial generators with up to 3.7 GBq (100 mCi) are available.

TABLE 7. IRRADIATION CONDITIONS AND YIELDS FROM THE CURRENT SUPPLIERS OF ^{82}Sr [83]

Laboratory	Target	Irradiation conditions	Yields (EOB MBq·µA^{-1}·h^{-1})[a]
Brookhaven National Laboratory	RbCl	2 targets (96–74 and 66–44 MeV) 14 d irradiation	10.5
Institute for Nuclear Research, Moscow	Rb metal	100–40 MeV	18
Los Alamos National Laboratory	RbCl	2 targets:	
		97–71 MeV	13.3
		65–45 MeV	8.2
iThemba Labs	RbCl	61–44 MeV	8.6

[a] Batch sizes typically range from approximately 65 to 450 GBq; EOB: end of bombardment.

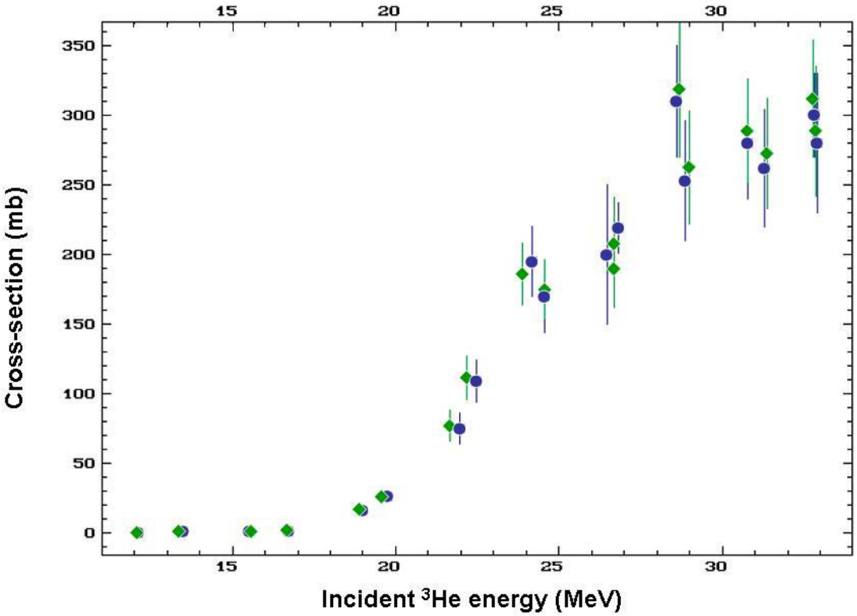


FIG. 41. Excitation function for the ^{82}Kr(^{3}He,3n)^{82}Sr reaction [46, 84, 85].

3.2.5.2. Generator systems, separation techniques and chemical form of the daughter

The ^{82}Sr/^{82}Rb generator has, by now, had more than 30 years of development. Table 8 intends to give a flavour of the various generator types that have been developed and tested during this period.

Synthetic organic resins such as BioRex-70 and Chelex-100 and inorganic ion exchangers such as ZrO_2 and Al_2O_3 were developed and ^{82}Rb eluted with 2% NaCl solutions. Other systems utilize hydrated SnO_2, with ^{82}Rb eluted with physiological NaCl solution (Fig. 42). Elution yields as high as close to 100% and breakthrough as low as 10–9/mL have been reported (see Table 8).

Today, the generator is routinely used, especially in clinical practice in hospitals. As ^{82}Rb is injected intravenously directly from the generator, ^{82}Rb must be obtained from the generator in a sterile pyrogen free physiological medium. This requirement is not necessary in general in industrial applications unless, possibly, when examinations are carried out in the food processing industry.

3.2.5.3. Availability and current status

- Commercial generators with up to 3.7 GBq (100 mCi) are available. Up to July 2011, the company Bracco Diagnostics Inc. provided a generator with the trade name CardioGen-82®. It has recently been temporarily withdrawn from the market. This recall was initiated following discussions with the US Food and Drug Administration regarding reports of unexpected radiation exposure in two individuals who underwent cardiac PET scans with CardioGen-82. This kind of question is, of course, not of importance for industrial applications, but the withdrawal affects commercial generator availability.
- The generator is routinely used in clinical practice, but has not yet been used in industrial studies.
- There is an established radionuclide generator separation chemistry which is easily modified for use in field applications.

3.2.5.4. Possible applications

This generator is useful for the study of mass flow in systems where the flow structure changes frequently, for example, during parameter tuning for process optimization. The short half-life of the daughter radionuclide permits frequent elutions and repeated examinations of a frequency of, perhaps, one elution/injection every 5 min.

TABLE 8. SOME REFERENCES SHOWING THE DIVERSITY OF COLUMN MATERIAL AND GENERATOR PROPERTIES DURING 30 YEARS OF GENERATOR DEVELOPMENT

Column material	Elution agent	Elution yield	Reference
Chelating cation exch.: Chelex-100	0.2M NH_4Cl	Sep. factor Sr/Rb: $>10^7$ Elution yield 80% in 15 mL	[86]
Cation exch.: BioRex 70 Chelating cation exch.: Chelex-100	2% NaCl	Sep. factor Sr/Rb: 10^7–10^8 Sep. factor Sr/Rb: 5×10^4	[87]
SnO_2 (hydrous) $Ti(OH)_4$ polyantimonic acid	Physiological saline = 0.9% NaCl	Best results: ^{82}Sr breakthrough: $<10^{-9}$/mL Elution yield: 90% at a flow rate of 30 mL/min	[55]
Al_2O_3	2% NaCl	Sep. factor Sr/Rb: 10^6–10^7 Elution yield 70–90% in 25 mL	[88]
Dual column: basic Al_20_3 + Chelex-100, $^{82}Sr^{2+}$ loading on the Al_2O_3 at pH9–10	1.8% NaCl, pH9–10	^{82}Sr breakthrough: $<5 \times 10^{-5}$/mL Elution yield 66% at elution rate 18 mL/min	[89]
$SnO_2 \cdot xH_2O$	Physiological saline	Sep. factor Sr/Rb: 10^5 Elution yield 100% in 47 mL at elution rate 25 mL/min	[90]
Hydroxyapatite, general formula $M_{10}(PO_4)_6(OH)_2$ where M is various metals			[91]

TABLE 8. SOME REFERENCES SHOWING THE DIVERSITY OF COLUMN MATERIAL AND GENERATOR PROPERTIES DURING 30 YEARS OF GENERATOR DEVELOPMENT (cont.)

Column material	Elution agent	Elution yield	Reference
SnO_2	Physiological saline	Sep. factor Sr/Rb: 5 × 10^6 Elution yield 10–40% in 15 mL	[92]
SnO_2	Saline, eluted as Rb^+	No significant breakthrough after 20 L saline	[93]
Sodium non-atitanate: $Na_4Ti_9O_{20}\cdot xH_2O$	NaCl (0.001–1M), eluted as Rb^+	Separation factor Sr/Rb >5.5 × 10^4 using 0.1M NaCl	[94]
A special non-regular column approach			
$^{82}Sr^{2+} + {}^{82}Rb^+$ in nuclear secular equilibrium in solution	Selective sorption of $^{82}Rb^+$ by inorganic ion exch.: cryptomelane $MnO_2 = KMn^{(IV)}{}_6Mn^{(II)}{}_2O_{16}$ where K is exchangeable	Separation factor of $^{82}Sr/^{82}Rb$: >10^4 at 0.5M HNO_3 leaving $^{82}Sr^{2+}$ in solution. $^{82}Rb^+$ eluted quantitatively with strong (>6M HNO_3)	Method possible on the basis of Ref. [95]

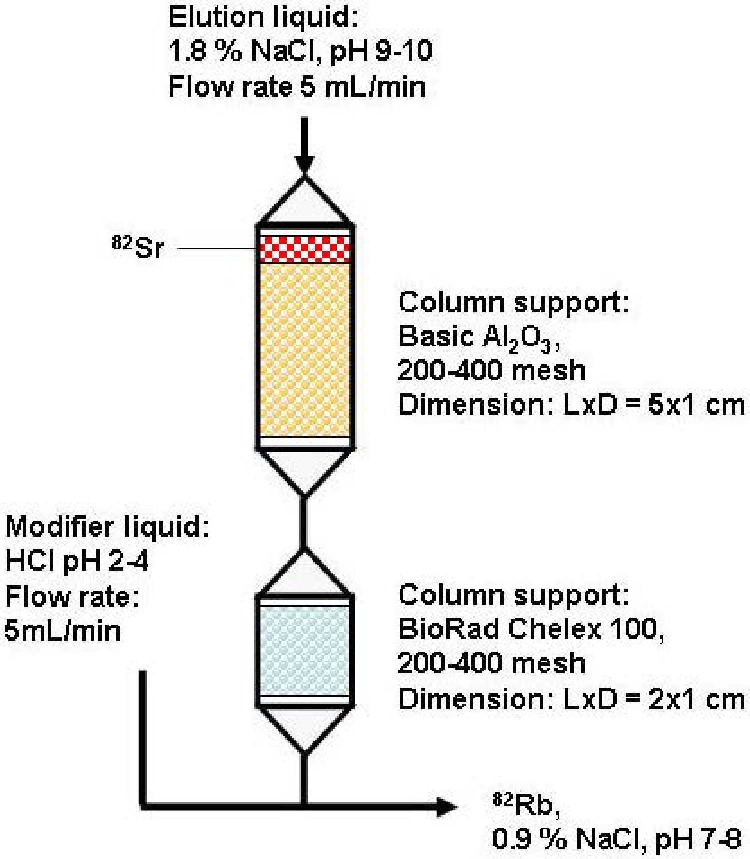


FIG. 42. Example of a ^{82}*Sr/*^{82}*Rb generator with a catcher column to reduce breakthrough of* ^{82}*Sr and a modifier liquid to prepare the* ^{82}*Rb solution, either for direct infusion or for further synthesis into an applicable tracer for various fluid phases [89].*

3.2.6. The ^{99}Mo/^{99m}Tc radionuclide generator

Relevant nuclear data for this radionuclide generator system are given in Table 4, and the simplified and partial decay scheme of ^{99m}Tc is given in Fig. 43.

3.2.6.1. Relevant nuclear properties

(a) Photon energy of the daughter

Decay of the mother radionuclide ^{99}Mo populates several excited states in the daughter ^{99}Tc, but only the two states at 920.5 keV and 142.6 keV are of

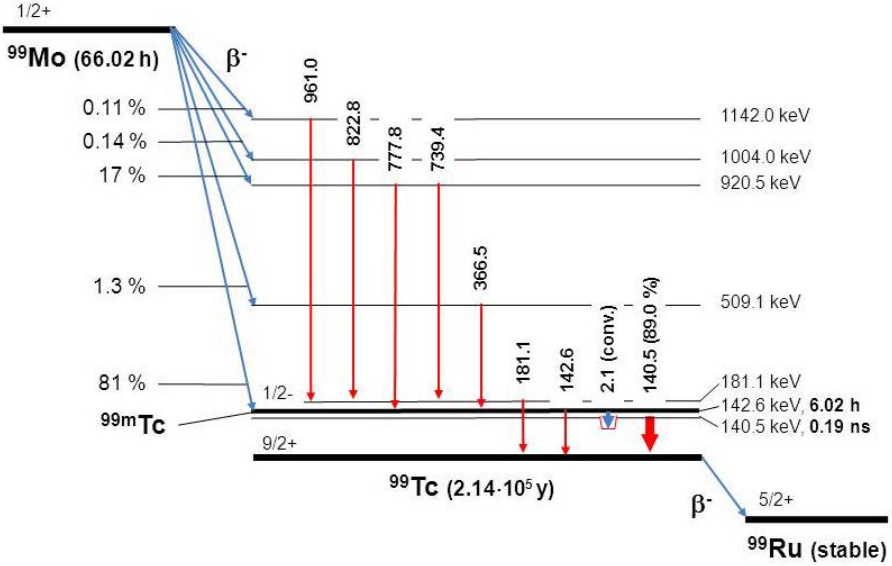


FIG. 43. Decay scheme of ^{99}Mo to excited states in ^{99}Tc including the long lived state at 142.6 keV, which may then be regarded as a separate radionuclide ^{99m}Tc [83].

importance due to spin/parity difference restrictions for the other states. The long lived excited state at 142.6 keV in ^{99}Tc constitutes the isomer ^{99m}Tc. It decays primarily by conversion of electron emission to the first excited state in ^{99}Tc at 140.5 keV, which again decays by internal transition (IT) to the ground state primarily by emission of an intense gamma ray with energy of 140.5 keV. The ground state is very long lived and decays by negatron emission to stable ^{99}Ru.

The ^{99}Mo/^{99m}Tc radionuclide generator has mainly been developed for nuclear medical purposes. Today, ^{99m}Tc, in the form of $^{99m}TcO_4^-$ (and many other compounds) is the main ‘workhorse’ for a variety of examinations in nuclear medicine. Therefore, detection equipment such as position sensitive gamma cameras and other instruments for single photon emission computed tomography (SPECT) (collimators and scintillation crystal thicknesses) have been optimized for 140.5 keV.

Although the gamma energy of 140.5 keV is not optimal for many industrial examinations (too low an energy), the choice of this radionuclide in the present text is due to its wide availability around the world. In several cases, i.e. in laboratory flow equipment and in smaller industrial systems, the energy is sufficient for transmission through the flowing medium and through the enclosing walls of the equipment. The photon energy is appropriate for many common detector materials such as plastic scintillators and smaller detectors such

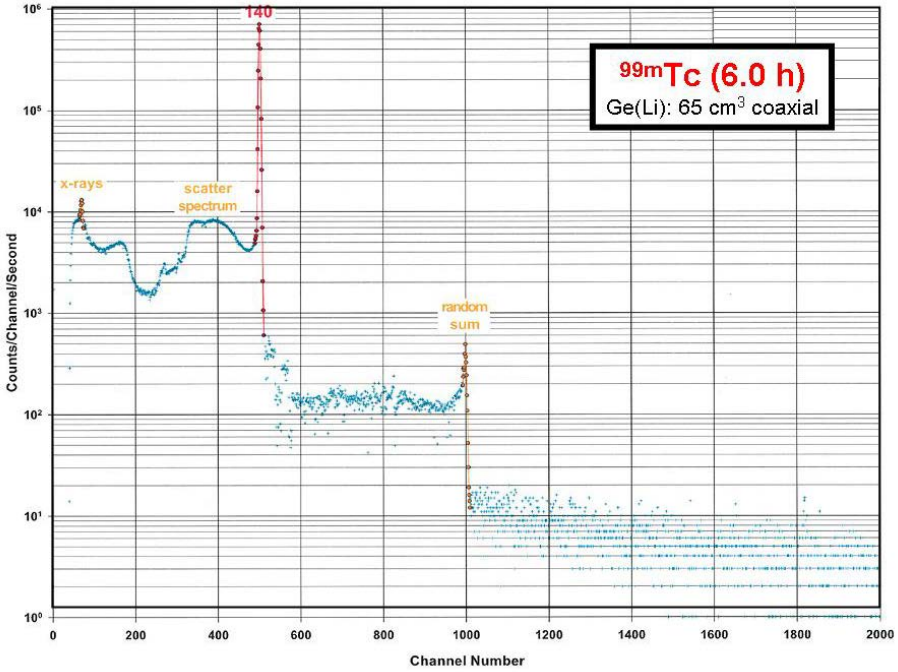


FIG. 44. Gamma spectrum of ^{99m}*Tc accumulated with a Ge(Li) detector.*

as CdTe, but is, of course, also readily detected with other common solid scintillation detectors such as NaI(Tl), CsI(Tl), BGO and BaF_2, by HPGe semiconductor detectors, and even with liquid scintillation adopted to a flowing liquid stream. A gamma spectrum is provided in Fig. 44.

(b) Half-life of the daughter

The daughter half-life of 6.02 h covers industrial experiments with a duration of the order of 1 d, and is the one daughter radionuclide considered in the present text with the longest half-life. A generator can be eluted at a frequency of a few hours and still give adequate activity because generators can be provided with a high initial activity. Thus, several experiments (3–5) can be performed during 1 d following individual elutions of the generator. Alternatively, and especially in the start when the generator is ‘fresh’, one single elution will provide sufficient activity for several experiments/injections.

(c) Half-life of the mother

The mother half-life of 66 h (2.7 d) provides for a shelf life of about 2 weeks. It also allows for transport of the generators to remote locations (for instance, offshore installations), even with surface transportation. A special point of interest is that when such a generator has become useless for hospitals, it may still be useful for technical and industrial tracer examinations. Thus, the generator can be inherited from hospitals after 2.5 half-lives to a cheaper price and utilized the following week before returned to the production company for decommissioning.

(d) Main production routes

(i) Production by fission of ^{235}U

Over 80% of diagnostic nuclear medical imaging today uses radiopharmaceuticals containing ^{99m}Tc, entailing over 30 million investigations per year. Over 95% of the ^{99}Mo required for ^{99m}Tc generators is produced by the fission of ^{235}U targets (^{99}Mo cumulative fission yield 6.1%) in nuclear research reactors. These reactors are multipurpose installations and not dedicated only to medical radionuclide production. In addition, the duty cycle (up-time) of the various reactors differs considerably from about 100 d/year for some to about 300 d/year for others.

During the last two decades a number of research reactors, which previously were available for radionuclide production, have been shut down, and the expression ‘neutron drought’ has been invented to describe the current situation. Hence, there are at present only a few main ^{99}Mo production sites around the world that utilize the fission production method, see Fig. 41.

In most reactors, the target material is highly enriched ^{235}U (HEU) to ensure maximum production for the neutron flux available, while others use uranium of lower ^{235}U content (LEU) with correspondingly lower production rates. This kind of production is followed by chemical processing of the target to isolate ^{99}Mo in a close to carrier free condition. This is a very complicated process, and only a few laboratories around the world are equipped to take on this task. The main processing facilities are shown in Fig. 45. An example of irradiation yield with this method is given in Table 9.

After isolation of ^{99}Mo, the radionuclide is shipped to other laboratories for production of the ^{99}Mo/^{99m}Tc generator. Various laboratories use different methods, but most of the methods are based on a chromatographic or ion exchange column where ^{99m}Tc is extracted by column elution, although the column support may vary somewhat. The most important laboratories are shown in Fig. 45.

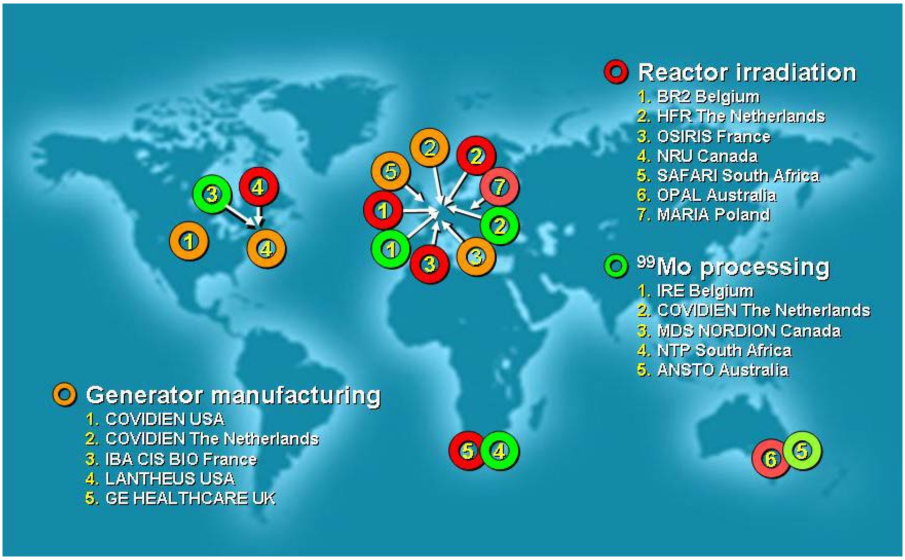


FIG. 45. The most important reactor facilities for irradiation of ^{235}U (HEU or LEU), processing of the irradiated target for isolation of ^{99}Mo and manufacturing facilities for the ^{99}Mo/^{99m}Tc generator.

TABLE 9. EXAMPLE OF IRRADIATION YIELD OF ^{99}Mo FROM FISSION OF A ^{235}U TARGET [96]

Parameter	Value
Neutron flux	1.5×10^{14} n·cm^{-2}·s^{-1}
^{235}U/Al target alloy	4.2 g ^{235}U, 45% enriched
Irradiation time	20–200 h
Irradiation yield	7.4–18.5 TBq (200–500 Ci) per batch

The mother nuclide ^{99}Mo is extracted from a fission mixture in alkaline solution, in the form of molybdate anion, $^{99}\text{MoO}_4^{2-}$. This solution is acidified in order to form oligomers of ^{99}Mo compounds for better adsorption characteristics on the chromatographic column. The ^{99}Mo is fixed on a pre-acidified alumina (Al_2O_3) column. The ^{99m}Tc is eluted from the column easily with 0.9% NaCl solution (physiological solution) in the form of $^{99m}\text{TcO}_4^-$. The ^{99}Mo breakthrough is generally of the order of 10^{-4}.

(ii) Production by neutron activation of ^{98}Mo

Molybdenum-99 may also be produced by thermal neutron capture in the stable molybdenum isotope ^{98}Mo. The isotopic abundance in natural molybdenum is 24.13%, but the reaction cross-section is, however, only 0.14 b. The irradiation yield depends on the following four parameters: (i) isotopic abundance (which may be increased by a factor of four by using isotopically enriched material), (ii) amount of target material irradiated, (iii) neutron flux in the irradiation channel and (iv) irradiation time. An example of an attainable yield is given in Table 10.

However, since one has to irradiate macro-amounts of target material, the specific activity in the irradiated target is rather low. This means that the small size columns, which are sufficient for fission produced ^{99}Mo, will not be applicable here. One needs column material and column sizes large enough to absorb higher quantities of Mo. For this purpose, molybdate gel columns, for instance, have been developed [97].

(iii) Production by charged particle reactions

Much has been written in the literature about the ^{99}Mo/^{99m}Tc generator. Therefore, some useful references on this subject are presented in Table 11 and some brief descriptions on the preparation of a few column based generators and a generator based on solvent extraction are given.

TABLE 10. EXAMPLE OF IRRADIATION YIELD OF ^{99}Mo FROM NEUTRON IRRADIATION OF A Mo TARGET [96]

Parameter	Value
Neutron flux	1.6×10^{14} n·cm^{-2}·s^{-1}
MoO_3 target material, natural abundance	99–120 g
Irradiation time	1 week
Irradiation yield	1.48 TBq (40 Ci) per batch

TABLE 11. SOME USEFUL REFERENCES FOR THE PRODUCTION AND USE OF A ^{99}Mo/^{99M}Tc RADIONUCLIDE GENERATOR

Subject/title	Reference
Fission molybdenum for medical use	[96]
Alternative technologies for ^{99m}Tc generators	[96]
Management of waste from ^{99}Mo production	[96]
Production technologies for ^{99}Mo and ^{99m}Tc	[96]
Manual for reactor produced radioisotopes	[96]
^{99m}Tc generators, chemistry and preparation of radiopharmaceuticals	[98]
^{99}Mo production using LEU targets	[99]
Radionuclide generator technology	[100]
Sol–gel processed ceramics for ^{99}Mo/^{99m}Tc generator materials	[101]
Conversion of ^{99}Mo production from HEU targets	[102]
^{99m}Tc gel generator usage with the aid of post-elution concentration of pertechnetate	[103]
^{99m}Tc generators: technology considerations and the IAEA contributions	[104]
^{99m}Tc pharmaceuticals	[105]
Polymer embedded nanocrystalline titania sorbent for ^{99}Mo/^{99m}Tc generator	[106]
Medical isotope production without highly enriched uranium	[107]
Recovery of ^{99m}Tc from low specific activity $(n,\gamma)^{99}$Mo based on solvent extraction and column chromatography	[108]
Separation of ^{99}Mo from nitric acid medium by alumina	[109]
A novel electrochemical technique for the production of clinical grade ^{99m}Tc using $(n,\gamma)^{99}$Mo	[110]
Magnesium molybdate compounds as a matrix for ^{99}Mo/^{99m}Tc generators	[111]

FIG. 46. Production of ^{99}Mo/^{99m}Tc generators based on chromatographic quality alumina at the Centre of Isotopes in Cuba [112].

3.2.6.2. Generator systems, separation techniques and chemical form of the daughter

(a) Preparation of ^{99}Mo/^{99m}Tc generators with alumina

The adsorption of the molybdate ion on the alumina is strongly affected by the pH. The sorption capacity of the alumina that is used in the preparation of the generators (chromatographic quality) at pH4.0 is ~20 mg of Mo per g of alumina, but it falls quickly to 2 mg/g at somewhat higher pH values. Figure 46 shows a picture from a typical generator production line.

For the preparation of a 3.7 GBq (100 mCi) generator, the following procedure is followed:

- A quantity of chromatographic alumina (80–100 mesh) is washed with distilled water, and the fine particles are eliminated by decantation.

- About 10 g of alumina is transferred to a glass column with appropriate support (Teflon and polyethylene filters) in the bottom and on the top of the bed.
- The column is washed with 0.1M sterile and apyrogen HCl or HNO_3 until a pH of 1–2 is reached in the effluent.
- To prepare the loading solution, transfer the ^{99}Mo to a beaker with 4 mL of distilled water. Add 6M HNO_3 until a pH of 3.0 ± 0.5. At smaller pH values, the molybdenum can precipitate. In such a case, the precipitate may be redissolved by adding NH_4OH. Dilute the solution with distilled water until 25 mL.
- Percolate slowly the ^{99}Mo solution through the column.
- Subsequently, wash the column with 100 mL of 0.1M HCl.
- Wash again the column with 75 mL of isotonic saline solution.
- Capture some drops of the last 10–20 mL of the washing liquid at the exit of the column and transfer them to a test tube with some milligrams of potassium ethyl xantate. Add two drops of 2M HCl. A red colour would indicate that the ^{99}Mo is eluting from the column, which should not happen. In order to qualify the column and measure any trace of ^{99}Mo breakthrough, it is necessary to perform γ spectrometry on a sample of the eluate and follow the decay for a few half-lives of the ^{99m}Tc.
- Finally, the column should be end capped, placed in the shield and autoclaved.

For industrial use, the requirements mentioned above (pyrogen free, sterile, autoclaved) are not necessary for most industrial applications if the generator is not going to be used in the drug or food industry.

Another method for preparation of a generator based on alumina is making use of the reaction of the molybdate with phosphoric acid to produce phosphomolybdate according to:

$$12MoO_4^{2-} + H_3PO_4 + 21HNO_3 \rightarrow PMo_{12}O_{40}^{3-} + 21NO_3^- + 12H_2O$$

The phosphomolybdate is subsequently loaded on to the column. This system is not used for the production of medical generators due to traces of phosphorous in the eluate of $^{99m}TcO_4^-$, but may be a good alternative when preparing generators for industrial radiotracer applications.

(b) Preparation of generators of ^{99}Mo/^{99m}Tc on other solid supports

(i) Hydrated zirconium oxide

Another ^{99m}Tc generator has been developed by using zirconium oxide as the column material (e.g. see Ref. [113]). The ^{99}Mo is loaded on to the zirconium oxide with 0.01M HCl. The ^{99m}Tc is eluted with an acidic solution of MEK solution: 5% v/v in 0.01M HCl. The technetium is complexed as a $H^{99m}TcO_4$–MEK complex that is dissolvable in organic liquids. As such, the complex may be directly used as a tracer of the organic phase.

In order to prepare a tracer for the aqueous phase, for instance, as $^{99m}TcO_4^-$, MEK may be removed by evaporation, and the resulting residue leached with, for instance, physiological saline solution.

(ii) Manganese dioxide

In the same way, one may adsorb the molybdate on to hydrated manganese dioxide. The sorption capacity for ^{99}Mo of the MnO_2 is rather low. Therefore, this procedure is impracticable for ^{99}Mo generated by the $(n,\gamma)^{99}$Mo reaction due to the presence of macro-amounts of natural molybdenum. However, for ^{99}Mo produced by fission, where the carrier concentration is lower, this column support is suitable for a ^{99}Mo/^{99m}Tc generator. Molybdate retention during application on the column consisting of 1 g MnO_2 material of 99.7% has been reported [114]. The pertechnetate can be eluted with acidified water (pH ≈ 5) or 0.9% NaCl. An elution yield >80% in 5 mL was achieved.

(iii) Hydrated titanium dioxide

Hydrated titanium dioxide has a higher absorption capacity for molybdate than alumina. The exchange capacity was found to be 58 mg MoO_3^- per g hydrated TiO_2 and 50 mg MoO_3^- per g $A1_2O_3$ at 0.1M HC1 [115]. The properties of a column containing 2 g anhydrous TiO_2 showed an elution efficiency of ~85% in 10 mL 0.9% NaCl solution, and the breakthrough of ^{99}Mo was reported to be $<10^{-3}$%.

(iv) Zirconium molybdophosphate–^{99}Mo gel

Low and medium specific activity of $(n,\gamma)^{99}$Mo produced by neutron irradiation of MoO_3 can be incorporated in an insoluble gel structure, which after granulation, efficiently releases ^{99m}Tc on elution.

Preparation of the ZrMP–^{99}Mo gel was as follows [116]:

- 2 g of neutron irradiated MoO_3 was dissolved in 5M NaOH by warming with addition of a few drops of H_2O_2 according to the reaction:

  $MoO_3 + 2NaOH \rightarrow Na_2MoO_4 + H_2O$

- The molybdate–^{99}Mo solution formed was converted into molybdophosphate–^{99}Mo by reacting with a calculated amount of $NaH_2PO_4 \cdot 3H_2O$ (160 mg) according to the reaction:

  $12Na_2MoO_4 + NaH_2PO_4 + 25HNO_3 \rightarrow H_3PMo_{12}O_{40} + 25NaNO_3 + 12H_2O$

- HNO_3 was added to adjust the pH to 4.5. The molybdophosphate–^{99}Mo solution formed was added slowly to a zirconium oxynitrate (4.19 g) solution during stirring.

- The temperature was raised to 55–66°C. A zirconium molybdophosphate–^{99}Mo gel was formed according to the reaction:

  $H_3PMo_{12}O_{40} + ZrO(NO_3)_2 \cdot xH_2O \rightarrow ZrMP \cdot xH_2O$ (gel)

The properties of the ZrMP column (2 g) were: elution yield varied in the range 66–90% and the ^{99}Mo breakthrough was $<4 \times 10^{-4}$%.

(v) Silica gel

Other researchers adsorbed the ^{99}Mo on silica gel from a solution of ammonium molybdate. A yield of 95% was achieved after the elution of the ^{99m}Tc with ketone.

(vi) Other generator types

Of the other generator types not treated further here are the ones based on separation by solvent extraction [108], the generator based on electrochemical separation [110] and the generator based on separation by sublimation of volatile Tc_2O_7 [117].

3.2.6.3. Availability and current status

The ^{99}Mo/^{99m}Tc generator is readily available at relatively affordable prices. Activities in the range 2.5–25 GBq are available with 7 d precalibration. There are presently about five main producers of ^{99}Mo, with two more in the pipeline (Fig. 41), and all of them base the production on fission of ^{235}U. In addition, there are four main laboratories that carry out the separation of ^{99}Mo from the target material. An additional seven laboratories are the main manufacturers of the generators, most of them based on small size alumina columns. These are convenient for further refinement in a radiotracer generator system because they are a small size and require moderate shielding.

In addition to the main producers (90–95% of the total production), there are a number of other research reactors that produce locally medium to low specific activity ^{99}Mo by irradiation of a Mo based target. The target separation and the manufacture of the generator are also carried out locally. However, the column sizes are more voluminous and the elution volumes considerably larger, and their use in an IRTG system becomes somewhat more cumbersome.

Long lived ^{99g}Tc, which is the decay product of the ^{99m}Tc, remains below exemption level, even for the highest activity of ^{99}Mo/^{99m}Tc radionuclide generators.

3.2.6.4. Possible applications

The ^{99}Mo/^{99m}Tc radionuclide generators can be used in several types of industrial tracer examinations where:

- Wall thickness does not exceed a few millimetres (for steel);
- The enclosing material container is made of glass, plastic or any other lighter materials;
- A bypass stream is arranged for easier detection;
- Detectors are immersed into the medium to be measured;
- Sampling techniques are used.

The ^{99}Mo/^{99m}Tc generator is often regarded as inappropriate for use in industrial monitoring operations because of the relatively short half-life of the mother and the low energy of the ^{99m}Tc gamma ray. However, the half-thickness of the 140 keV gamma ray (4 mm of steel), while too low for studies on high pressure plants, is still sufficient for use in pipes and vessels with wall thicknesses up to about 10–15 mm. It is also rightly argued that the normal chemical form in the generator eluate, $^{99m}TcO_4^-$, while a useful water tracer at oxidizing (aerobic) conditions, is a strong oxidation agent (valence state 7+) and will easily be reduced to lower valence states under reducing conditions.

For instance, TcO_2 is a common reduced form (valence state 4+). Therefore, labelling techniques are needed to produce useful $^{99m}Tc^-$ labelled tracers for water, organic fluids and solids under varying experimental conditions.

Undisputed positive arguments for industrial use of this generator are that this is the most widely used generator in nuclear medicine. As such, it is available worldwide from a number of suppliers, usually at a few days' notice and at a cost that is not prohibitive. Additionally, from a public relations point of view, prospective clients often perceive a radionuclide that is used for medical applications as more user friendly.

Although the chemistry of technetium is very well known, its multiple states of oxidation can be an obstacle for labelling certain compounds and reagents. In addition, most of the labelling chemistry has the development of methods to produce labelled compounds for in vivo use in the human body as a primary goal. Thus, thermal and chemical stability at elevated temperatures of these compounds have, in general, not been thoroughly examined. It is still largely a remaining task to develop a suite of useful industrially applicable tracers for harsh experimental conditions.

3.2.7. The ^{113}Sn/^{113m}In radionuclide generator

Relevant nuclear data for this radionuclide generator system are given in Table 4 and the simplified and partial decay scheme of ^{113m}In is given in Fig. 47.

3.2.7.1. Relevant nuclear properties

(a) Photon energy of the daughter

The EC decay of ^{113}Sn feeds mainly (98.2%) the first excited state at 391.69 keV in ^{113}In. The transition from this excited state to the ground state is M4 (spin difference $\Delta I = 4$ and parity change), which means that the transition is hindered and therefore also converted to a considerable degree (conversion coefficient measured to $\alpha K = 0.437 \pm 0.007$ [118]). Therefore, the excited state is 'long lived' with a half-life of 99.5 min. Thus, ^{113}In in this excited state may be regarded as a separate radionuclide, ^{113m}In.

The energy of the IT of 391 keV is sufficient for transmission through thin metal or plastic walls and detection outside of the vessel or tubing and therefore also suitable for on-line measurements in the industry. A gamma spectrum is provided in Fig. 48.

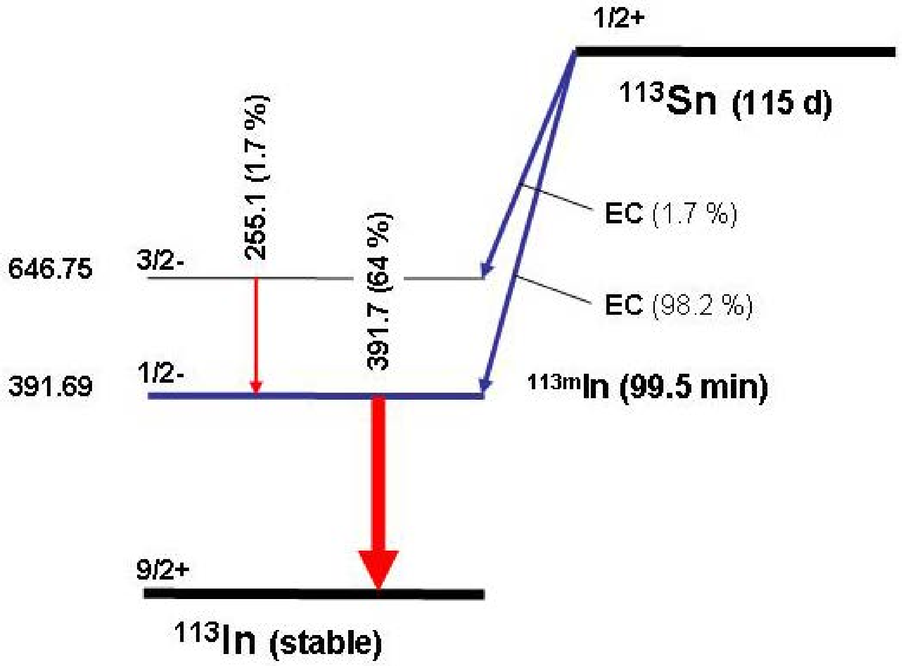


FIG. 47. Decay scheme of ^{113}Sn to excited states in ^{113}In, including the long lived state at 391.69 keV, which may then be regarded as a separate radionuclide ^{113m}In [83].

(b) Half-life of the daughter

The half-life of the daughter is suitable for at least two examinations per day in the same experimental system provided they are spaced by 5 h or more, i.e. the activity from the first injection is largely dead at the time of the second injection, and the buildup of ^{113m}In is sufficient for a new elution.

(c) Half-life of the mother

The half-life of 115 d offers about 1 year of application without replacing the generator.

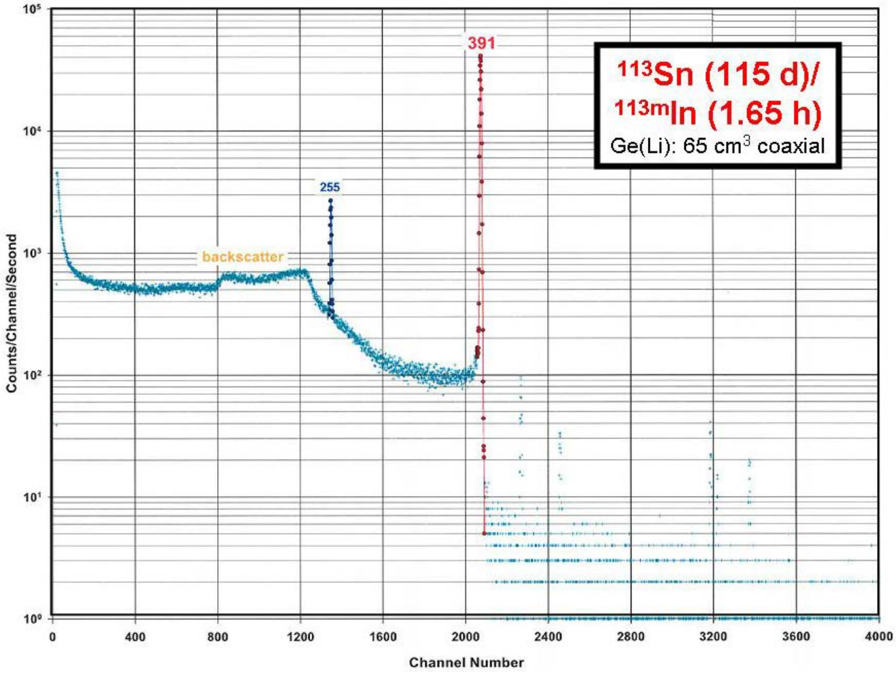


FIG. 48. Gamma spectrum of ^{113m}In accumulated with a Ge(Li) detector.

(d) Main production routes

(i) Neutron reactions

The main production route is through the ^{112}Sn(n$_{th}$,γ)^{113}Sn reaction in a nuclear reactor. Since the isotopic abundance of ^{112}Sn in natural tin is only 0.97% and the thermal cross-section only of the order of 1 b (see excitation function in Fig. 49), the target material should be isotopically enriched with respect to ^{112}Sn. Enrichments >95% are commercially available (Trace Sciences International). By using the experimental parameters given, the average activity obtained (calculated at the end of activation) is about 30 mCi/mg Sn (with 81.7% enrichment in ^{112}Sn).

It is also possible to produce ^{113}Sn by the reaction ^{114}Sn(n,2n)^{113}Sn using 14 MeV neutrons from the D(T,n)^{4}He reaction in a neutron generator. However, this is a marginal production method because the flux in the generators is limited (maximum around 10^{11} n·cm^{-2}·s^{-1}), the reaction cross-section is 1270 ± 115 mb [119] and the isotope fraction for ^{114}Sn only 0.65%. However, isotopically

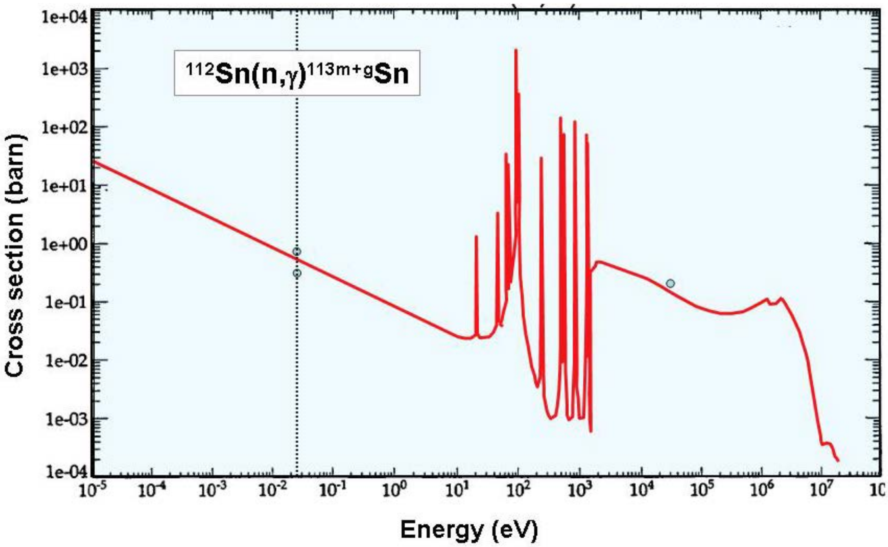


FIG. 49. Excitation function for the neutron capture reaction $^{112}Sn(n,\gamma)^{113m+g}Sn$ [119].

enriched material is commercially available from Trace Sciences International (60–90+%) and from NIDO/Oak Ridge National Laboratory (>61%).

(ii) Charged particle reactions

Charged particle reactions with moderate energies (15–70 MeV) may also be used for production of ^{113}Sn. For these reactions, the target material is, for instance, either indium or cadmium. Some of the most applicable reactions are listed in Table 12.

3.2.7.2. Generator systems, separation techniques and chemical form of the daughter

Presently available ^{113}Sn/^{113m}In generators use high specific activity ^{113}Sn, which cannot be produced in low and medium flux research reactors, even with enriched ^{112}Sn.

However, relatively low specific activity of ^{113}Sn, 75–115 GBq (2–3 Ci) per g, may still be produced in sufficient amounts in low and medium flux research reactors by irradiation of enriched metallic ^{112}Sn. After dissolving the irradiated target in HCl, the resulting $^{113}SnCl_4$ is firmly adsorbed on hydrous zirconium oxide from 3–4M NaCl solutions of pH1.3–1.5 and the elution of ^{113m}In is effected by dilute HCl. The yield of such a generator is typically 70% of

TABLE 12. EXAMPLES OF CHARGED PARTICLE REACTIONS FOR PRODUCTION OF ^{113}Sn

Nuclear reaction	Energy $E_{\sigma\max}$ at cross-section maximum	Cross-section $\sigma_{\max}$ at $E_{\max}$
^{113}In(p,n)^{113}Sn		
^{115}In(p,3n)^{113}Sn	30 MeV	1850 mb ($^{113m+g}$Sn)[a]
^{Nat}In(p,xn)^{113}Sn, x = 1, 3	28 MeV	780 mb[b]
^{113}In(d,2n)^{113}Sn	11–19 MeV	~1000 mb[b]
^{Nat}In(^{3}He,xn)^{113}Sb → ^{113}Sn, x = 3, 5		
^{Nat}In(^{3}He,pxn)^{113}Sn, x = 2, 4	55 MeV	700 mb[c]
^{Nat}In(α,pxn)^{113}Sn, x = 3, 5	85 MeV	850 mb[c]
^{Nat}Cd(^{3}He,xn)^{113}Sn, x = 2–4	38 MeV	400 mb[c]
^{Nat}Cd(α,xn)^{113}Sn, x = 1–5	55 MeV	270 mb[c]

[a] Numerical value from Ref. [120].
[b] Numerical values from Ref. [121].
[c] Numerical values from Ref. [122].

the theoretically available ^{113m}In activity [123]. Breakthrough of ^{113}Sn has been consistently <0.01% of the ^{113m}In activity.

Hydrous zirconium oxides may be produced with varying degrees of hydration, and the different forms may show somewhat different performance characteristics with respect to tin loading, ^{113m}In elution and ^{113}Sn breakthrough. Generators containing up to 25 GBq (700 mCi) of ^{113}Sn have been prepared, yielding pure ^{113m}In suitable for medical use [124].

Other column types reported include the following:

- Glass beads [125] that have been eluted with 0.16M NaCl and gave 73% yield with 0.042% breakthrough.
- Activated carbon [126] with an exchange capacity of Sn(IV) ~7.6 meq/g, elution efficiency of ^{113m}In 80% by using 10 mL of 0.2M HCl–80% acetone with flow rate 1 mL/min. A secondary catcher column of the anion exchanger Dowex 1x8 is used to capture any breakthrough of ^{113}Sn.

- Experiments with various tungstocerate(IV) gel matrices [127, 128], for instance, 12-tungstocerate(IV) column with the formula $H_4[CeO_4 \cdot W_{12}O_{36}] \cdot 18H_2O$, for sorption of ^{113}Sn(IV) from 0.1M HCl solution and elution of ^{113m}In(III) using the same solution.
- Tests on the 12-molybdenocerate(IV) column matrix [129] with focus on the chemical conditions to optimize the loading capacity and efficiency and on the elution efficiency as a function of time.
- Hydrated antimony pentoxide column (2 g/column, 50–100 mesh) [130], elution at 0.5M HCl, elution yield >60% in 18 mL, but decreases with number of elutions.
- Silica gel based on absorption of ^{113}Sn in the presence of bromine water and 0.4M HCl. The absorption capacity was 6 mg Sn per g silica gel, the absorption efficiency >99% and the Sn/In separation factor $>10^4$ for the elution liquid pH > 1.2. In general, the silica gel system had superior operating characteristics when compared to the hydrous zinc oxide system: higher recovery of ^{113m}In over a greater pH range was observed [131].
- Organic Chelex-100 (iminodiacetic functional group) or BioRex-63 (a phosphonic resin) ion exchangers for chromatographic separation [132]. 0.1N hydrochloric acid is used as eluent with a separation yield of 60–70% and a separation factor of 10^5.

The chemical form of ^{113m}In in the eluate from the various generators mentioned here is mainly as the cations In^{3+} at low chloride concentration. At higher Cl^- concentrations, chloride complexes like $InCl^{2+}$, and even $InCl^{2+}$, will occur. At strong Cl^- concentrations, the chemical form may be converted to the anionic form $InCl^{4-}$.

3.2.7.3. Availability and current status

Since this generator has lost some interest in nuclear medicine, the number of commercial producers seems to have been reduced to only a handful on the global scale. Two companies that still deliver the generator are Polatom in Poland and SSC RF-IPPE in the Russian Federation. Alternatively, ^{113}Sn may be purchased, for instance, from Polatom, and the generator produced in the purchaser’s own laboratories according to the procedure outlined in Fig. 50.

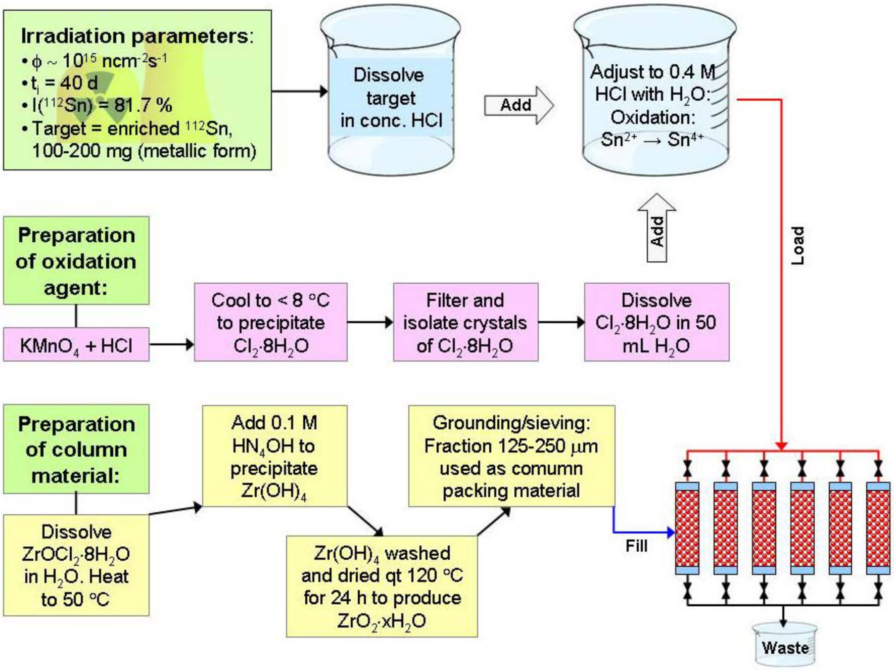


FIG. 50. Example of irradiation–target dissolution–column material preparation–column loading sequence [133].

3.2.7.4. Potential and existing applications

- Potential for application in a variety of industrial situations (chemical, petroleum, metallurgical industries, etc.), with tailor-made ^{113m}In tracers as probes for aqueous, organic or solid material;
- Applications for logging in oil wells [134];
- Used in studies of wastewater treatment plants;
- Used for surface labelling of coal and accompanying minerals;
- Can be used for monitoring movements of fines in surface water flows in environmental situations;
- Syntheses and purification of ^{113m}In tracers for industry is possible due to the experiences from analogous syntheses of ^{111}In labelled tracers used in nuclear medicine and SPECT imaging.

3.2.8. The ^{137}Cs/^{137m}Ba radionuclide generator

Relevant nuclear data for this radionuclide generator system are given in Table 4 and the simplified and partial decay scheme of ^{137m}Ba is given in Fig. 51.

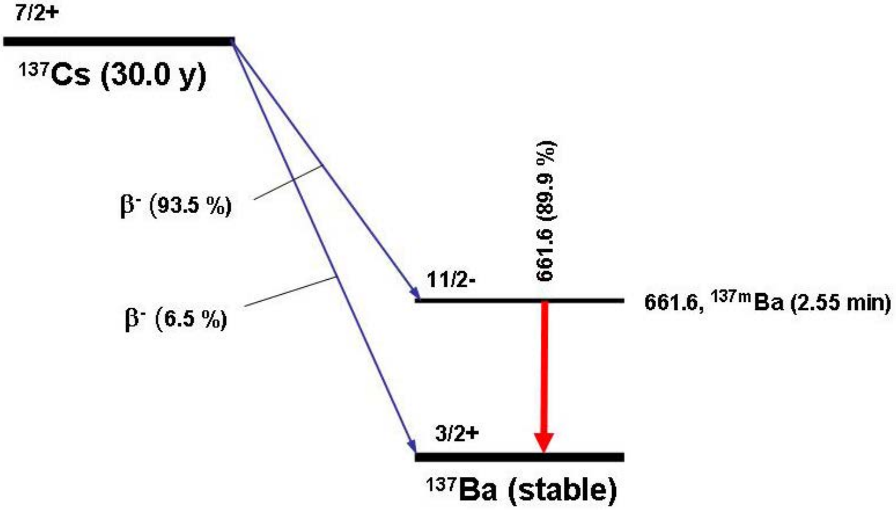


FIG. 51. Decay scheme of ^{137}*Cs to a ‘long lived’ excited state in* ^{137}*Ba at 661.6 keV, which may then be regarded as a separate radionuclide,* ^{137m}*Ba [83].*

3.2.8.1. Relevant nuclear properties

(a) Photon energy of the daughter

The gamma energy (IT) of the daughter of 661 keV, with high intensity (89.9%), is sufficient for transmission through medium thick (<20 mm) metal walls and detection outside of the vessel or tubing. Therefore, it is suitable for on-line measurements in the industry using common sized NaI(Tl) scintillation detectors (for instance, $d \times l$ = 25 mm × 38 mm for narrow detector positions and 50 mm × 50 mm for general applications). A gamma spectrum is given in Fig. 52.

(b) Half-life of the daughter

The daughter half-life of 2.55 min restricts the use to a maximum 10 min per injected batch, but offers possibilities of cyclic industrial measurements of short duration: after about 20 min, the system has no memory of the preceding tracer injection. This is valuable for systems with fast mass transport and which require frequent measurements, especially in situations and periods where running parameters are adjusted in order to optimize the process.

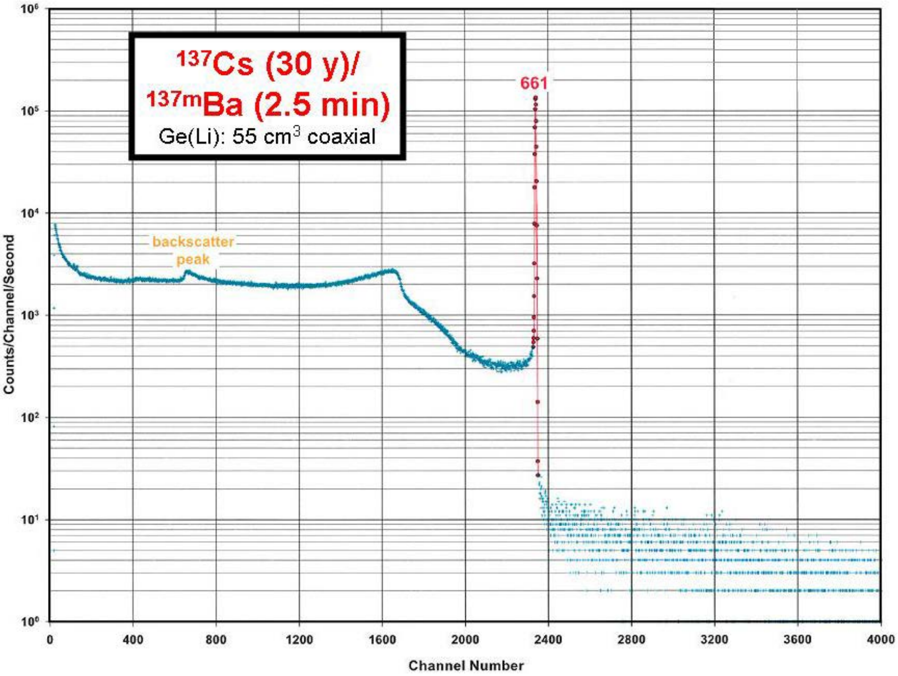


FIG. 52. Gamma spectrum of ^{137m}Ba accumulated with a Ge(Li) detector.

(c) Half-life of the mother

The physical half-life of ^{137}Cs is 30 a. However, the useful life of the generator is significantly less. The shelf life is dependent upon the detailed construction and physicochemical properties of the system. Radiolytic reactions due to beta and gamma interactions in the generator matrix may slowly lead to breakdown of the matrix and thereby ruin the elution properties of the generator.

Still, if column material is selected for the best radiation resistance (mainly inorganic materials), and the generator is properly stored in periods where it is not in use (it should be stored under dry conditions), the shelf life of the generator can be many years, possibly as much as >10 a. This long life of the generator makes it suitable for the provision of tracers at remote locations over an extended period of time, and also for possible use in permanent installations at inaccessible places, such as in subsea oil installation equipment. It is possible to incorporate the generator into installed equipment to automatically provide an ongoing supply of tracer for semi-continuous flow rate measurement.

(d) Main production routes

Neutron reactions: the only viable production route of the mother nuclide ^{137}Cs for large scale industrial use is by fission of uranium through the reaction ^{235}U(n,f)^{137}Cs and recovery of the ^{137}Cs from the mixture of fission products. The accumulated fission yield is 6.236%. Very high activities can be achieved by this method, and the inventory of ^{137}Cs on a global basis is large. The main use of this radionuclide is as a solid source in gamma sterilization plants, mass measurement equipment for the industry (gamma transmission tomography) and in industrial radiography for checks of quality of weldings, for instance.

It is, in principle, also possible to produce carrier free ^{137}Cs in reactors by thermal neutron irradiation of ^{136}Xe to produce short lived (3.83 min) ^{137}Xe followed by beta decay. The total reaction is ^{136}Xe(n,γ)^{137}Xe → ^{137}Cs. However, this production route is not practical due to the long half-life of ^{137}Cs, an isotopic abundance of ^{136}Xe of 8.9% and a low production cross-section of 0.26 b.

Charged particle reactions: it is, of course, also possible to produce ^{137}Cs by charged particle induced fission, for instance, of ^{232}Th or ^{Nat}U in medium to high energy particle accelerators.

Another production method for carrier free ^{137}Cs is by irradiation of ^{136}Xe by deuterons using a small particle accelerator capable of producing 12–15 MeV deuterons. The reaction is ^{136}Xe(d,n)^{137}Cs, but it is not a practical production route due to the same arguments as for the (n,γ) reaction above.

3.2.8.2. Generator systems, separation techniques and chemical form of the daughter

Most of the generator systems developed and studied are based on columns with sorption media of different types: zirconium phosphate, zirconium vanadate, ammonium phosphomolybdate, titanium hexacyanoferrate or cupric cobalt ferrocyanide and several other inorganic adsorbents using EDTA and other eluent solutions. Ion exchange columns eluted with 0.9% NaCl provided highest yield and breakthrough of $<10^{-3}$. A selection of various column systems is summarized in Table 13.

A typical microscopy picture of the granular column support material silica gel coated with the active ion exchange material potassium–copper–hexacyanoferrate is given in Fig. 53. The particle sizes are in the range 0.25–0.85 mm.

TABLE 13. SELECTION OF PUBLISHED DATA ON ^{137}Cs/^{137M}Ba GENERATORS AND COLUMN SUPPORT MATERIAL USED FOR SELECTIVE SEPARATION OF ^{137}Cs FROM COMPLEX REACTION MIXTURES, BUT WHICH HAVE NOT YET BEEN UTILIZED IN GENERATORS

Column material	Column loading fluid and elution agent	Yield and breakthrough	Reference
Cobaltferrocyanides of the form $M[CoFe(CN)_6]$, where M may be Ni, Zn, Fe(II) or Cu(II) The latter is preferred	Loading with 0.1M HCl Elution with isotonic saline (IS) or 0.1M HCl	Yield (IS): 83 ± 7% in 10 mL Yield (HCl): 85 ± 10% in 10 mL Breakthrough: $<10^{-4\%}$	[135] [136]
Zirconium phosphate	Elution with HNO_3	Elution yield: 30% Breakthrough: $<10^{-3}$%	[137]
(Anionic resin + MoO_3^-)$^+$ ($H_2NH_3PO_4$ + 7.5M HNO_3) $\Rightarrow$ R-AMP = $(NH_4)3PO_4 \cdot 12MoO_3$	Loading capacity: 0.48 meq/g dry R-AMP Elution with 0.1M or 0.3M HNO_3	Elution yield: not given Breakthrough: 'completely free from the parent'	[138]
Support bed (various org. polymers) loaded with the extractant dicarbollide H^+ in nitrobenzene, the formula is $((\pi\text{-}(3)\text{-}1,2\text{-}B_9C_2H_{11})_2\ Co^{III})^-$	Various elution agents, optimum conditions for 10^{-4}M HNO_3 + 2.5 × 10^{-3}M $Ba(NO_3)_2$	Elution yield: 65–79% in 10 mL up to 50 elutions Contamination of ^{137}Cs <0.1%	[139]
1. Copper–cobaltferrocyanide, $Cu[CoFe(CN)_6]$ 2. Zirconium phosphate (ZrP)	CCS loading: wide pH range CCS elution: IS or 0.1M HCl ZrP loading: pH2 ZrP elution: 0.5M HNO_3	Yield CCS: high Breakthrough CCS: $<10^{-4}$% Yield ZrP: 58% in 3 mL Breakthrough ZrP: 2%	[140] [141]
Zirconium vanadate (ZrV)–formula $Zr(HVO_4)_2 \cdot nH_2O$	Exch. capacity: 4.02 meq/g Elution with 0.1M NH_4Cl	Elution yield: high Breakthrough: ^{137}Cs not detected	[142]

TABLE 13. SELECTION OF PUBLISHED DATA ON ^{137}Cs/^{137M}Ba GENERATORS AND COLUMN SUPPORT MATERIAL USED FOR SELECTIVE SEPARATION OF ^{137}Cs FROM COMPLEX REACTION MIXTURES, BUT WHICH HAVE NOT YET BEEN UTILIZED IN GENERATORS (cont.)

Column material	Column loading fluid and elution agent	Yield and breakthrough	Reference
6-Tungstocerate(IV) gel matrix	Elution with 0.9% NaCl + 0.1M HCl	Elution yield: high and reproducible over ~1 year Radionuclide purity: >99.99%	[143]
Polyoxometalate of the form $Na_2(H_2O)_4(H_3O)[Al(OH)_6Mo_6O_{18}]$	Exch. capacity for Cs^+: 2.29 mmol/g Loading with 0.01M HCl Elution with 0.5% ascorbic acid	Elution yield: none with 0.01M HCl High but no numbers given with ascorbic acid Breakthrough: no ^{137}Cs detected in eluate	[144]
Potassium–cobalt–hexacyanoferrate on silica gel: $K_2[CoFe(CN)_6]$–SiO_2	Exch. capacity for Cs^+: 0.311 mmol/g Loading and elution with 0.1M HCl + 0.9% NaCl	Elution yield: 65–75% in 10 mL Breakthrough fraction: ~4 × 10^{-6}	[145]
Potassium–copper–hexacyanoferrate on silica gel: $K_2[CuFe(CN)_6]$–SiO_2	Exch. capacity for Cs^+: 0.060 mmol/g Elution with 0.9% NaCl	Elution yield: >80% in 10 mL Breakthrough: <10^{-4}%	[112]
Ammonium molybdophosphate (AMP) impregnated alumina microspheres (AMP–Al_2O_3)	Elution with 0.1M NH_4NO_3 + 0.5M HNO_3 solution	Elution yield: >80% breakthrough: ‘Acceptable radionuclidic purity suitable for industrial radiotracer investigations’	[146]
Interesting column supports not yet further described or utilized in generators			
Titanium phosphates (TiP) and combined titanium phosphate–ammonium molybdophosphate (TiP–AMP)	Not described	Not described	[147]

TABLE 13. SELECTION OF PUBLISHED DATA ON $^{137}Cs/^{137M}Ba$ GENERATORS AND COLUMN SUPPORT MATERIAL USED FOR SELECTIVE SEPARATION OF ^{137}Cs FROM COMPLEX REACTION MIXTURES, BUT WHICH HAVE NOT YET BEEN UTILIZED IN GENERATORS (cont.)

Column material	Column loading fluid and elution agent	Yield and breakthrough	Reference
Selective sorption of radioactive caesium and strontium on stannic molybdophosphate ion exchanger	High sorption efficiency of both Cs^+ and Sr^{2+} (Ba^{2+} has not been tested)	Not described	[148]
Ammonium molybdophosphate–alumina (AMP–Al_2O_3)	Studied kinetics Other properties not described	Not described	[149]
SiO_2–potassium–cobalt–hexacyanoferrate, SiO_2–$K_2CoFe(CN)_6$	Exch. capacity 0.38 meq Cs/g Distribution coeff. for Cs: for HCl 0.001–3M = 7.5×10^4 and 5.8×10^4	Not described	[150]
Organic ion exchanger SuperLig®644 with proprietary structure	Not described	Not described	[151]
Potassium–copper–nickel–hexacyanoferrate(II) with composition $K_{2.3}Cu_{0.65}Ni_{1.7}[Fe(CN)_6]_2 \cdot 13H_2O$ also called K–Cu–Ni–CNF	Exch. capacity: 2.25 mmol/g of Cs Loading efficiency: >99%	Not described	[152] [153] [154]
Ni–CNF, K–Ni–CNF and K–Cu–Ni–CNF absorbed into a porous matrix of polyacrylonitrile, PAN	Not described	Not described	[155]
Ammonium–phosphor–molybdate $(NH_4)_3PO_4(MoO_3)_{12} \cdot 4H_2O$ on activated carbon	Not described	Not described	[156]

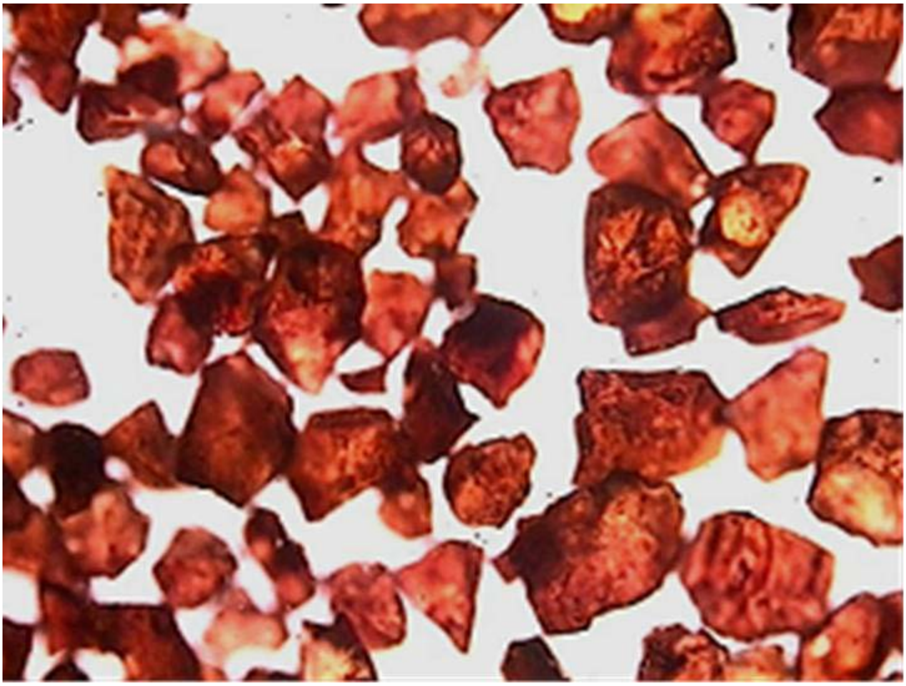


FIG. 53. Micrograph of the granular column support material silica gel ($SiO_2{\cdot}xH_2O$) coated with the active ion exchange material $K_2[CuFe(CN)_6]$ [112].

Generators recently developed and studied by the contributors to the present document are in particular based on the following column support materials:

- Potassium–copper–hexacyanoferrate on silica gel, $K_2[CuFe(CN)_6]$–SiO_2, in short denoted KCuFC–SiO_2, developed at INSTEC in Cuba [112];
- Potassium–cobalt–hexacyanoferrate on silica gel, $K_2[CoFe(CN)_6]$–SiO_2 , in short denoted KCoFC–SiO_2, developed at CIAE in China [145];
- Ammonium molybdophosphate impregnated alumina microspheres, possibly of the form $\{(NH_4)_3PO_4(MoO_3)_{12}–Al_2O_3\}{\cdot}xH_2O$, in short, denoted AMP–$Al_2O_3$, developed at BARC in India [146].

A sketch of the crystal structure of the active ion exchanging part in the column support material is, for illustration purposes only, given in Fig. 54 for the case of the CIAE generator. By changing the cobalt atoms for copper atoms, the figure also illustrates the INSTEC generator. Here, it is the blue spheres (potassium ions) that are exchangeable with $^{137}Cs^+$.

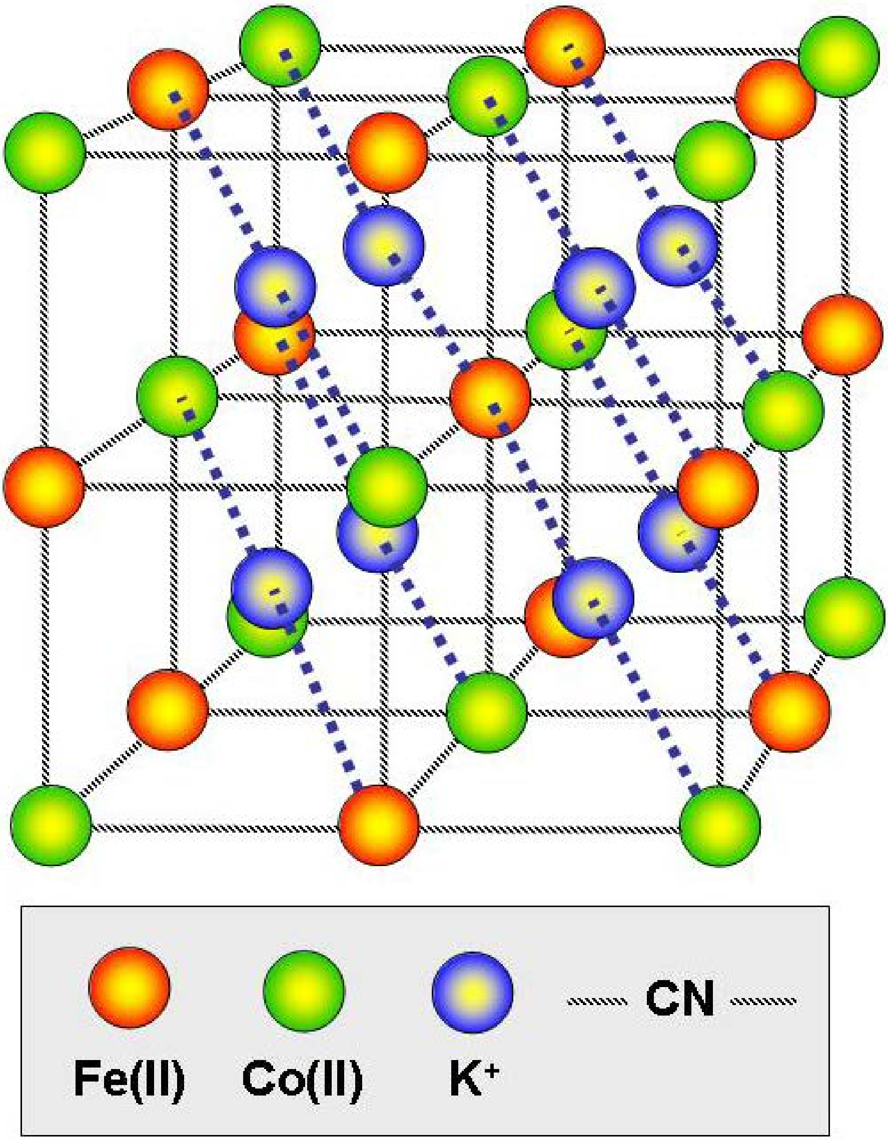


FIG. 54. Crystal structure unit cell of the active ion exchange material $K_2[CoFe(CN)_6]$ [145]. Each connecting line between red and green spheres represents a CN ligand.

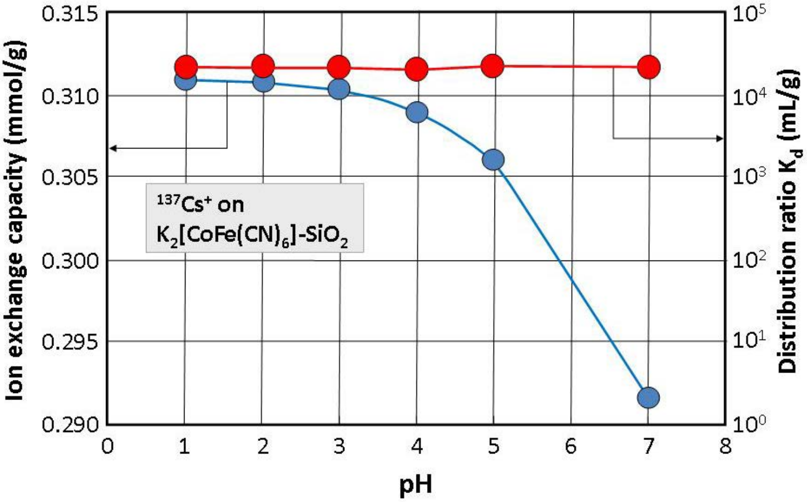


FIG. 55. Ion exchange capacity (in mmol/g) and distribution ratio K_d *(defined by Eq. (*18*)) of* $^{137}Cs^+$*on the column support material* $K_2[CoFe(CN)_6]$–SiO_2 *as a function of pH [145].*

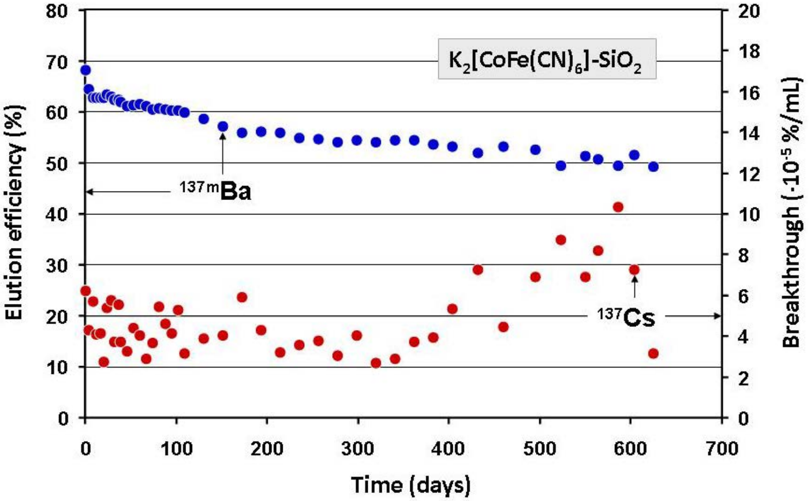


FIG. 56. Elution efficiency of $^{137m}Ba^{2+}$ *and corresponding breakthrough of* ^{137}Cs *in 10 mL of the elution mixture 0.1M HCl + 0.9% NaCl for the column support material* $K_2[CoFe(CN)_6]$–SiO_2 *as a function of time [145].*

The properties of all these three generators seem to be acceptable for sufficiently stable operations over at least a few years. Examples of ion exchange capacities, elution efficiencies and breakthrough properties are given in Figs 55 and 56.

Here, A_{i} is the initial activity concentration (Bq/mL), A_{eq} is the activity concentration (Bq/mL) in the liquid at distribution equilibrium, V is the liquid volume used (mL) and m is the mass of the ion exchanger material or column support material (g):

$$K_{\mathrm{d}} = \frac{(A_{\mathrm{i}} - A_{\mathrm{eq}}) \cdot V}{A_{\mathrm{eq}} \cdot m} \mathrm{mL/g} \tag{22}$$

Figure 57 shows the elution efficiency of a test production of an industrial version of the CIAE generator of the same activity, ~1.7 GBq. The generator was eluted for the first time 330 d after quality check and calibration in the CIAE laboratory, and the measured efficiency amounts to ~60% as compared with the 65% claimed by the manufacturer. The general trend was a reduction to ~40% at the end of the nearly 2 years of application, as compared with ~50% for a corresponding period measured in the manufacturer's laboratory. The abrupt drop

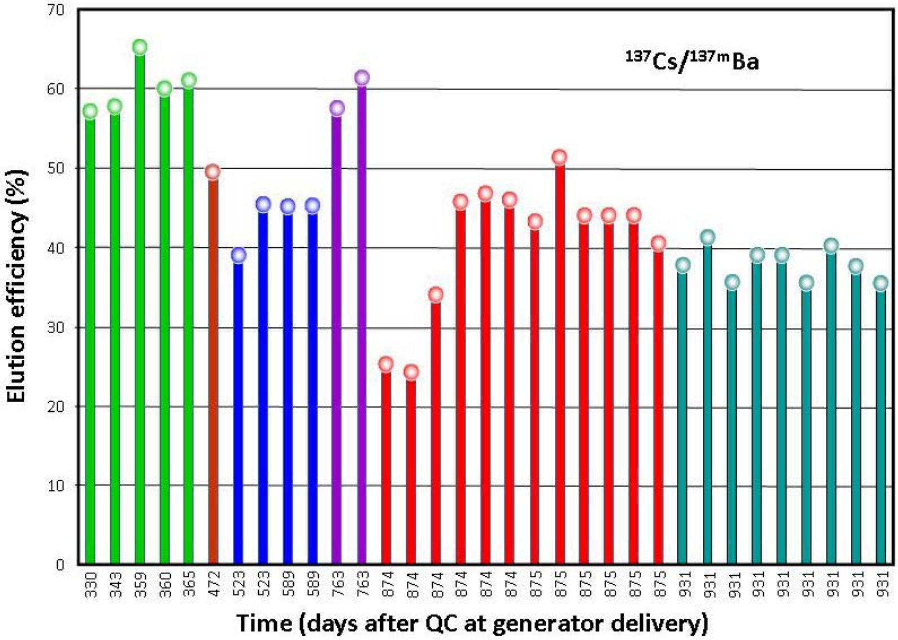


FIG. 57. Elution yield of ^{137m}Ba from $K_2[CoFe(CN)_6]$–SiO_2 (an industrial version of the CIAE generator) as a function of time after it has been used for various industrial tracing operations. The colour code relates to different and narrow spaced elutions for the same colour with different storage periods in between [74].

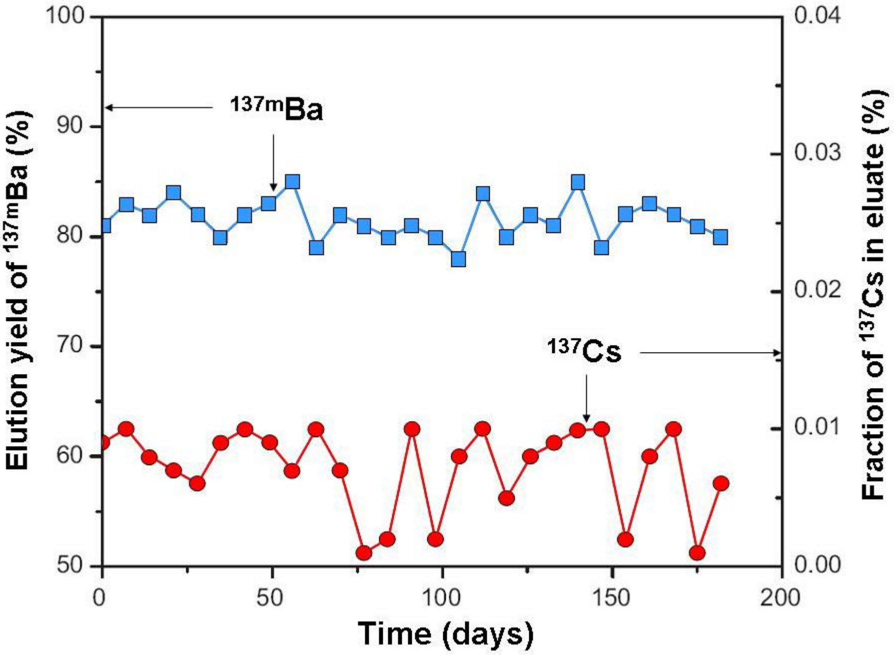


FIG. 58. Elution efficiency of $^{137m}Ba^{2+}$ and corresponding breakthrough of ^{137}Cs in 5 mL of the elution mixture 0.1M NH_4NO_3 + 0.5M HNO_3 for the column support material $[(NH_4)_3PO_4(MoO_3)^-_{12}Al_2O_3]\cdot xH_2O$ as a function of time [146].

in the beginning of the 'red' elution period is explained by improper storage: the generator probably went dry without the operator noticing, hence there is a recovery of the efficiency after a few elution cycles [74].

Figure 58 shows that the ion exchange capacity of the $K_2[CoFe(CN)_6]$–SiO_2 column decreases somewhat, but not dramatically, with increasing pH, while the distribution ratio stays fairly constant over the pH range shown at $K_d = 2.2 \times 10^4$ mL/g, which is somewhat less than that reported by Mardan et al. [150]. In order to secure optimum capacity and minimum breakthrough of caesium, the column is both loaded and eluted with a mixture of 0.1M HCL and 0.9% NaCl.

Figure 58 also shows that the initial elution efficiency of ^{137m}Ba from the same column is around 65%, but decreases to about 50% in the course of the 600 d testing period. The corresponding breakthrough of ^{137}Cs is around 4×10^{-5}% per mL in the beginning, but there is an increasing tendency towards the end of the testing period. The activity of ^{137}Cs on this test generator was ~1.7 GBq (~45 mCi).

The elution efficiency of $^{137\mathrm{m}}$Ba for the APM/Al_2O_3 generator manufactured at BARC (5 mL of the elution mixture 0.1M NH_4NO_3 + 0.5M HNO_3) is shown in Fig. 58. It is, on average, ~80% and stays stable over the testing period of 180 d. This is significantly higher than for the previously described KCoFeC/SiO_2 based generator. The corresponding breakthrough of ^{137}Cs is also fairly stable over the testing period and amounts to some 0.005–0.01% per 5 mL (10^{-3}–2×10^{-3}% per mL), which is acceptable but not as good as for the previously described KCoFeC/SiO_2 based generator [146].

The real shelf lives of the generators are yet to be determined. The main problem is probably the disintegration of the column support material into fines that are not retained by the ordinary filters but transported by the elution liquid. These particles will then also contain the mother nuclide ^{137}Cs. Breakthrough might arise to an unacceptable level. The material breakdown may be due to a combination of radiolysis, chemical interactions from the elution liquid and the physical erosion effect of the eluting liquid.

Measures to increase the shelf life would probably be:

- To mount extra exchangeable filters for ^{137}Cs carrying fines of the column material;
- To install an extra exchangeable catcher column for ionic $^{137}Cs^+$;
- To store the generator between application periods under controlled wet conditions and in a tempered area.

The set-up of column based $^{137\mathrm{m}}$Ba generators may be somewhat different from laboratory to laboratory. One such set-up developed and used at BARC in India [146] is sketched in Fig. 59. It shows the elution operation process where a vial filled with the elution liquid is mounted at the generator top on a needle that penetrates a septum in the lid of the elution flask, thus connecting this liquid to the column support material. Another needle, also penetrating this septum, is leading air into the bottle during the elution process. A second bottle, also with a septum in the lid and originally evacuated, is sucking the elution liquid through the column and is receiving the eluate containing the $^{137\mathrm{m}}$Ba activity. The generator system is a small size and transportable. During transport, a combined lead shield and lid seals off the generator top. The external dimensions of the BARC generator are Ø = 90 mm and h = 163 mm.

Another $^{137\mathrm{m}}$Ba generator recently developed at CIAE in China [145] uses a peristaltic pump to create suction during the generator elution process. The external dimensions are larger, the shielding thicker and the weight correspondingly heavier. The 44 kg still makes it readily transportable.

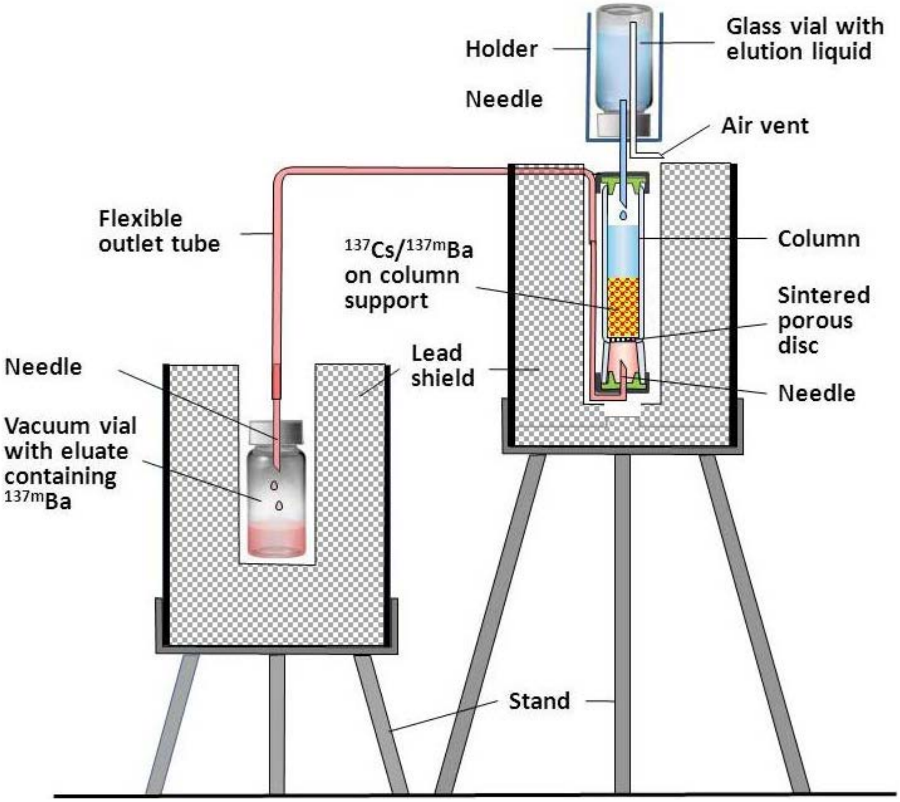


FIG. 59. Schematic diagram of the ^{137}Cs/^{137m}Ba generator set-up developed and used at BARC, India [146].

Extensive reviews of column materials (ion exchangers) and partly also liquid extraction agents for selective separation of ^{137}Cs from aqueous systems are given in Refs [157–159], but elution properties of ^{137m}Ba have not necessarily been considered. However, the experiences described here may be important for further development of radiotracer generators.

An example of an alternative generator is provided by the equilibrium mixture of $^{137}Cs^{+}$ and $^{137m}Ba^{2+}$ which is loaded on to a column of Amberlite IRC-50 modified by incorporating an amide group. Barium is fixed, while caesium runs unhindered through. Subsequently, ^{137m}Ba is recovered from the column by eluting with 0.01M EDTA at a pH of 10 [160]. There are, however, three drawbacks with this method: firstly, the need for handling radioactive solutions of the long lived ^{137}Cs for each separation operation; secondly, these operations steal valuable time which results in a lower separated activity of

^{137m}Ba; thirdly, production of a strong Ba–EDTA complex makes it more difficult and time consuming to use the activity in other labelling operations.

3.2.8.3. Availability and current status

Caesium-137 is primarily used in sealed sources for mass measurements (density), industrial radiography and in large irradiation facilities for treatment of sludges and for sterilization purposes. Therefore, the radionuclide itself is readily available worldwide at a reasonable price because it is separated in large amounts from fission waste.

Commercial generators with a relatively low activity of 0.37 MBq (10 µCi) are available (e.g. from TEL-Atomic Inc.), but they are primarily intended for use as educational instruments.

A 0.74 GBq (20 mCi) ^{137}Cs/^{137}Ba generator has been produced by Atom Hightech Co. Ltd in China, but it is, for the moment, unclear whether this generator is still available. However, both the generator from CIAE in China and the one from BARC in India described in the text above show promise. These organizations are large and have the necessary competence and facilities to be future manufacturers for the world market of industrial versions of the ^{137}Cs/^{137m}Ba generators.

3.2.8.4. Potential and existing applications

- The generator can be used many times per day at maximum activity, allowing repeated studies.
- Post-processing methods to synthesize a variety of industrial ^{137m}Ba tracers for aqueous/environmental, organic, solids and multiphase purposes are under development. Promising results have been obtained recently to synthesize a ^{137m}Ba labelled tracer for oil in multiphase flow experiments, see Ref. [161]. The kinetics of the labelling process is fast and may easily be automatized.
- Simpler labelling, for instance, to generate a stable water tracer, can be done directly in the elution process by application of a suitable complexing agent in the elution liquid.

There are several advantages with this generator:

- *For injection into product streams*: The short half-life ensures that by the time the material leaves the plant, the radioactivity is at a negligible level.
- *For investigations on systems with recirculation*: The short half-life ensures that buildup of activity in the stream does not become a problem.

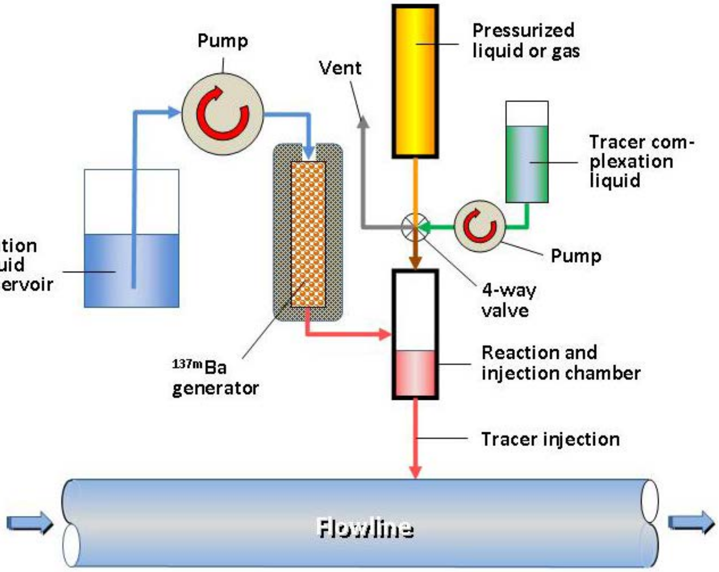


FIG. 60. Schematic diagram of a possible set-up for a ^{137m}*Ba generator integrated with a chemistry unit for complexation of the* Ba^{2+} *ion if needed, and also integrated with a tracer injection system. A number of electronic or pneumatic valves are not shown. The equipment may be operated automatically by a PC with auxiliary electronic equipment.*

- *Where repeat measurements are required*: The system regenerates rapidly so that the tracer is available over an extended period.
- *For solids tracing*: The main application of ^{137m}Ba labelled compounds will be in fluid tracing. However, the primary chemical form eluted for most column support media described above is Ba^{2+}. This can be used directly to label solids that are cation exchangers, such as various clays.

Owing to the short half-life of ^{137m}Ba (2.55 min), it will be desirable to engineer the generator into a system that is automatically operated and which is directly associated with a proper injection system. A possible arrangement is sketched in Fig. 60.

3.2.9. The ^{144}Ce/^{144}Pr radionuclide generator

Relevant nuclear data for this radionuclide generator system are given in Table 4 and the simplified and partial decay scheme of ^{137m}Ba is given in Fig. 61.

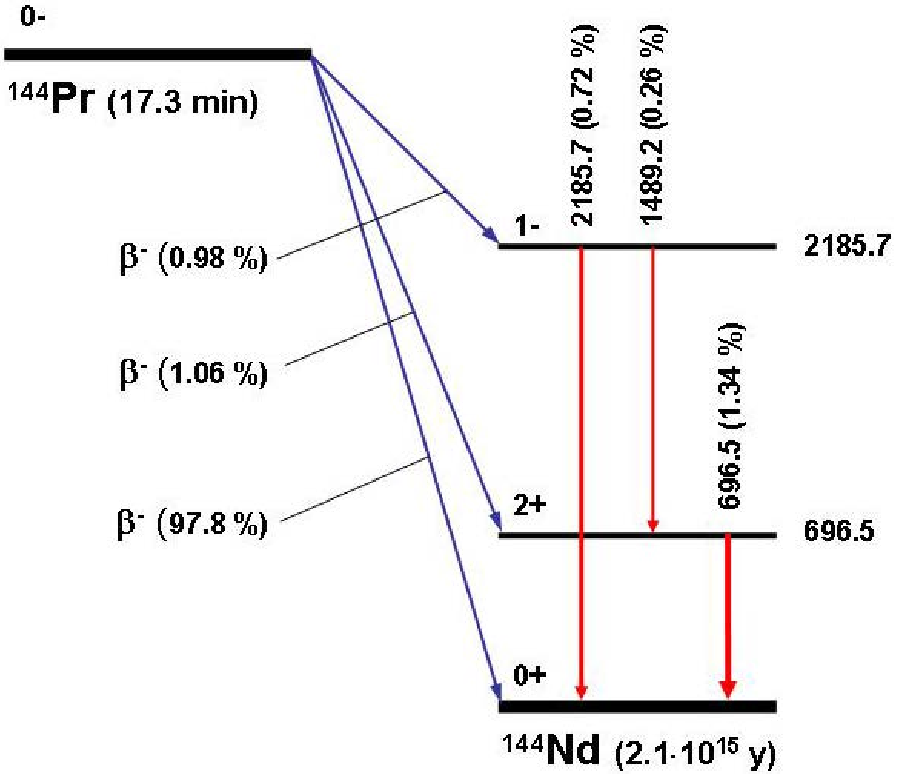


FIG. 61. Partial decay scheme of ^{144}Pr to excited states in ^{144}Nd whose ground state is a very long lived nuclide with α decay to stable ^{140}Ce. In practice, this ground state may be regarded as 'stable' [83].

3.2.9.1. Relevant nuclear properties

(a) Photon energy of the daughter

A total of 97.8% of the beta decay of ^{144}Pr goes directly to the ground state of 'stable' ^{144}Nd, with no associated gamma rays. The most abundant gamma rays at 699.5 keV have an absolute intensity of 1.34%. However, the ^{144}Ce/^{144}Pr nuclear genetic relationship is of interest as an industrial nuclide generator for several other reasons:

- The combination of the mother half-life (relatively long) and the daughter half-life (medium short) covers a time combination not offered by any of the other generators considered in the present document.

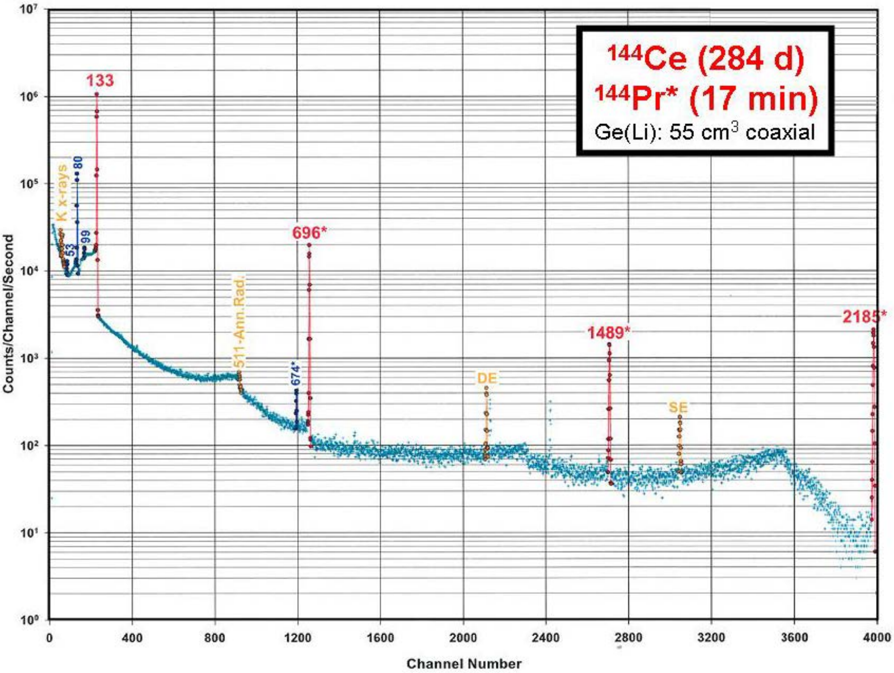


FIG. 62. Gamma spectrum of ^{144}Pr accumulated with a Ge(Li) detector.

- The daughter energy of 699 keV is sufficient for transmission through medium thick (some mm) metal walls and detection outside of the vessel or tubing and offers therefore the possibility for on-line measurements in the industry.
- The two main gamma energies from decay of ^{144}Ce of 133.5 keV (11.1%) and 80.1 keV (1.36%) are easy to shield, so the source strength (activity of ^{144}Ce) can be rather high (several GBq, several hundred mCi) in order to compensate for the low gamma intensity of ^{144}Pr.
- If the source strength is sufficiently high, all three gamma energies may be utilized in industrial measurements. The high energies penetrate several cm of steel, and the combination of the three makes it possible to obtain depth information in tanks or large pipelines. A gamma spectrum is given in Fig. 62.

(b) Half-life of the daughter

A daughter half-life of 17.3 min offers industrial measurements of short-to-medium duration.

(c) Half-life of the mother

A half-life of 285 d offers some 2–3 a of application without replacing the generator.

(d) Main production routes

Fission (for instance, thermal neutron induced fission of ^{235}U or ^{239}Pu or high energy particle induced fission of ^{238}U or ^{232}Th) is the only viable and effective production method for ^{144}Ce. The cumulative chain yield of mass number $A = 144$ for thermal neutron fission of ^{235}U is 5.459%, which is relatively high. The ^{144}Ce is then isolated from the fission mixture. Cerium belongs to the lanthanide series. The oxidation state with highest stability of most lanthanides is 3+. This implies that the chemistry of all lanthanides is very similar and their individual separations are correspondingly challenging. However, cerium is also stable in the 4+ state, and this simplifies the separation, both from other fission products, and from other lanthanides: a lanthanide group separation from other fission products is performed first. This is followed by a specific isolation (mainly by solvent extraction) of four valent cerium from the lanthanide group.

A key element in the production of a $^{144}Ce/^{144}Pr$ radionuclide generator is, of course, the high efficiency separation of ^{144}Ce from the mixture of fission products. Three methods will briefly be mentioned: the first one is based on continuous production comprising electromagnetic separation, the second one on continuous production comprising chemical separations and the third one on batch wise irradiation followed by chemical separations. The first two have not yet developed to their full potential for production of sources of sufficient strength.

Method 1, continuous on-line production:

- Nuclear reaction: High energy particle induced fission of uranium;
- Separation: Element selective surface ionization of the precursor ^{144}Cs ($T_{1/2} = 1.0$ s) followed by electromagnetic separation and collection of the mass separated $A = 144$ beam, which rapidly accumulates in long lived ^{144}Ce.

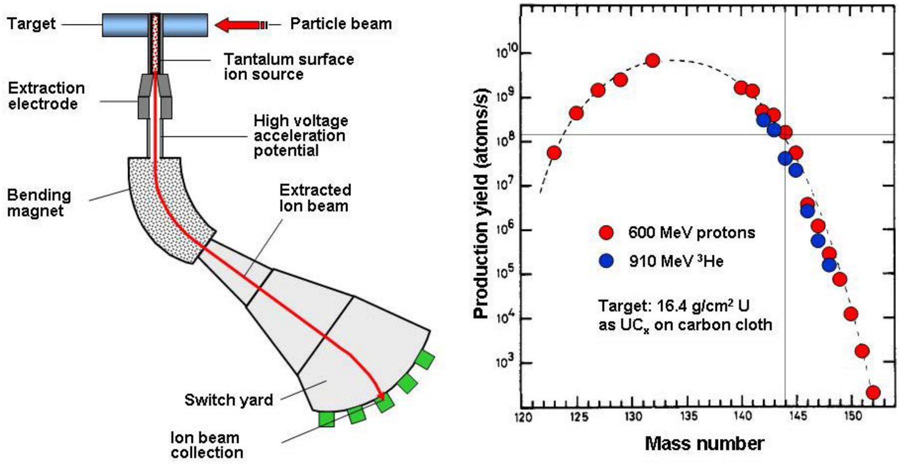


FIG. 63. Method 1 for production of ^{144}Ce.

This process has been implemented, for instance, at the ISOLDE laboratory at CERN. Figure 63 gives somewhat more detailed information of the method. At ISOLDE/CERN, ^{144}Cs is selectively produced at a target temperature of 2000°C and an ionization temperature of 1000°C with an ionization efficiency of 90%.

In Fig. 63, the left picture gives a rough principle sketch of particle induced nuclear reactions in a high temperature target, surface ionization of diffusing nuclear reaction products followed by extraction and acceleration in an electric field of the generated ions before mass analysis in a magnetic field and collection of the mass separated particle beam. In the same figure, the right picture shows experimental production yields at ISOLDE for Cs isotopes normalized to a 1 μA 600 MeV proton and/or a 1 particle μA 910 MeV ^{3}He irradiated target consisting of 16.4 g/cm^2 of uranium impregnated carbon cloth [162].

So far, this method has not been used for large scale production of strong sources of ^{144}Ce. For that to happen, the actual production yield has to increase considerably relative to the numbers in Fig. 63. This can be achieved by:

- Applying a thicker target, say a factor of ten higher than the one mentioned in Fig. 63;
- Applying a higher particle (proton) flux, say 100 μA;
- Using a tungsten surface ionization source and increasing the ionization temperature to 2000°C or higher, resulting in partial ionization also of ^{144}La and ^{144}Ce;
- Keeping the target temperature at >2000°C to maintain a high diffusion rate of relatively volatile fission products out of the target matrix.

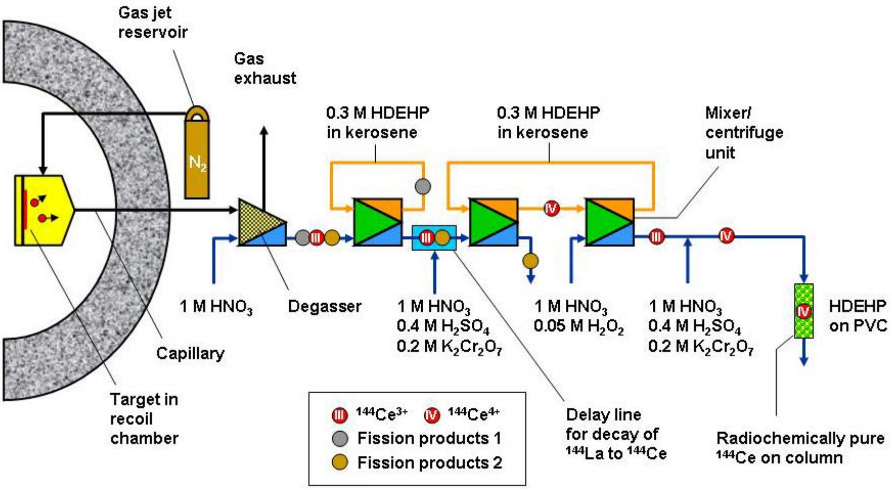


FIG. 64. Schematic diagram of a continuous on-line production and separation set-up for Ce isotopes based on fission of ^{235}U in a thermal reactor.

Assuming the same overall efficiency for production of ^{144}Cs with these parameters might result in a production time of ~1000 s in order to obtain a ^{144}Ce source of 10 MBq.

Method 2, continuous on-line production:

- Nuclear reaction: Thermal fission of ^{235}U in a thin target where fission fragments recoil out of the thin target into a recoil chamber volume;
- Separation: The recoiling fission fragments are trapped in a sweeping gas (most often containing salt clusters of NaCl or KCl) and transported with a gas jet through a thin capillary to a continuously running chemical separation apparatus based on solvent extraction.

This process has been used to isolate and study short lived isotopes of light lanthanide elements in the SISAK equipment. Figure 64 sketches the principles of the process.

The gas jet transport of fission products is guided to the SISAK radiochemical separation system, which is based on solvent extraction and final catchment of ^{144}Ce on a column with a solid support composed of HDEHP impregnated PVC powder. A delay line inserted in the aqueous phase between mixer/centrifuge units 1 and 2 ensures sufficient decay to ^{144}Ce of the precursor ^{144}La ($T_{1/2} = 40$ s).

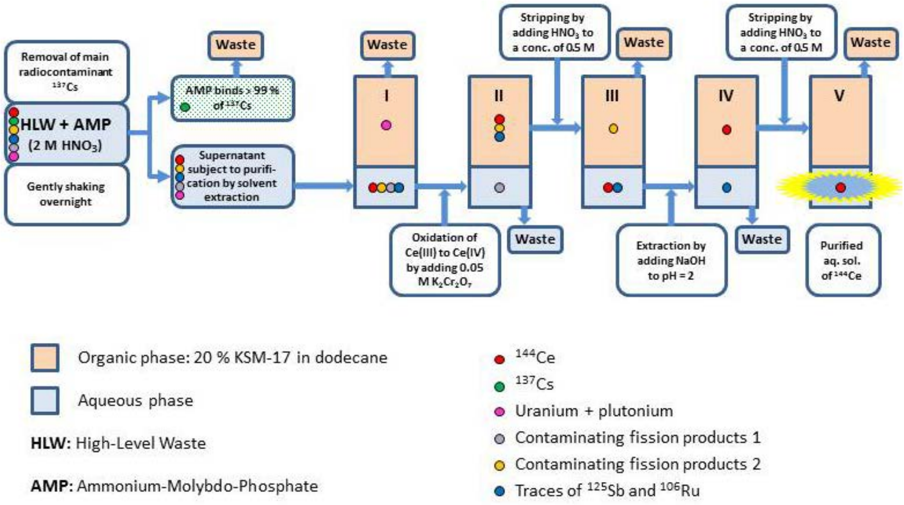


FIG. 65. Schematic diagram off-line and batch wise radiochemical separation procedure for isolation of ^{144}Ce from PUREX HLW.

An alternative route is to apply a system for separation of lanthanum (^{144}La), sorption of the ^{144}La on an ion exchanger and let ^{144}Ce grow in.

This system has not yet been used for production of strong sources of ^{144}Ce, and no attempt has been made here to calculate production yields.

Method 3, batch wise production:

- Nuclear reaction: Thermal neutron induced fission of ^{235}U or high energy particle induced fission of uranium or thorium performed in high flux reactors or at high flux accelerators;
- Separation: Chemical dissolution of the irradiated target followed by a multistep chemical separation scheme, mainly manually operated, in a hot cell.

This procedure and the reprocessing of high level waste (HLW) mainly from the nuclear energy industry, are those primarily used at present for production of radiochemically pure sources and solutions of ^{144}Ce. Some of the relevant methods for off-line batch separation of ^{144}Ce from a fission mixture include chromatographic methods, solvent extraction where electro-oxidation is a key element and more ordinary solvent extraction methods. The separation of ^{144}Ce from HLW is illustrated in Fig. 65.

The acronym KSM-17 (Fig. 65) represents the organic complexant 2-ethylhexyl hydrogen 2-ethylhexyl phosphonate. The overall chemical yield is ~70%, and the decontamination factor >105. Figure 65 is based on a procedure described in Ref. [144].

3.2.9.2. Generator systems, separation techniques and chemical form of the daughter

Since neither ^{144}Ce nor ^{144}Pr are considered radiopharmaceutical species of interest, the literature on their separation in the form of a radionuclide generator is rather poor. A few publications exist, however, and some of these are summarized in Table 14, although the most recent relevant publication dates from 1987. The various relevant types of information given in the respective references are not comprehensive, and important data on elution yields and mother breakthrough are indicated only in a few cases. Hence, the lifetime of the various generators reported cannot be reliably estimated. A recent thorough search of the available literature has not revealed any more recent attempts to develop and study the ^{144}Ce/^{144}Pr generator. There seems, therefore, to be a need to develop and study new/improved generators based on the ^{144}Ce/^{144}Pr nuclear genetic relationship.

3.2.9.3. Availability and current status

The ^{144}Ce/^{144}Pr radionuclide generator is not commercially available at present. However, radiochemical solutions of pure ^{144}Ce (+ equilibrium activity of ^{144}Pr) can probably be purchased from a few radiochemical companies. Thus, generators can be produced on an individual basis in the various radiochemistry laboratories where published information can be taken as a starting point for further studies and improvements.

The availability of ^{144}Ce depends on reactors producing fission products.

3.2.9.4. Possible applications

Owing to the short half-life of ^{144}Pr (17.3 min), the generator can be used many times per day at maximum activity, allowing repeated studies. Of special interest is the fact that this radionuclide emits three well-spaced gamma photons of 696 keV, 1489 keV and 2185 keV. This enables information of the depth distribution of the ^{144}Pr labelled tracer in extended volumes, and will complement those radiotracers with fewer useful gamma photons.

TABLE 14. THE $^{144}Ce/^{144}Pr$ GENERATORS AND COLUMN SUPPORT MATERIAL USED FOR SELECTIVE SEPARATION OF ^{144}Pr FROM AN INVENTORY OF RADIOCHEMICALLY PURE ^{144}Ce

Generator material	Chemistry involved	Yields
Filter material of 'insoluble' and freshly formed $Ce(IO_3)_4$	To an equilibrium mixture of $^{144}Ce/^{144}Pr$ in 5M HNO_3 is added 1M $NaBrO_3$ to oxidize Ce(III) to Ce(IV)	The solution is percolated through the freshly formed $Ce(IO_3)_4$ and $^{144}Ce(IV)$ is fixed by the mechanism of isotopic exchange. The ^{144}Pr runs unhindered through. The ^{144}Ce labelled $Ce(IO_3)_4$ may be redissolved by adding H_2O_2 and the precipitation process repeated after in-growth
Column support material is Dowex-1 (100–200 mesh) on nitrate form prepared by treating with 6M HNO_3 and 0.5M $NaBrO_3$	To an equilibrium mixture of $^{144}Ce/^{144}Pr$ is added $NaBrO_3$ in 9M HNO_3 to oxidize Ce(III) to Ce(IV)	^{144}Pr is quantitatively eluted by 10 mL 9M HNO_3. Contamination of ^{144}Ce is <0.2%
1–3 mL liquid mixture of $^{144}Ce/^{144}Pr$ in 3M HNO_3	Addition of solid $NaBiO_3$ to oxidize Ce(III) to Ce(IV) Add methyl-isobutyl-ketone (hexone) to extract Ce(IV) into organic phase	^{144}Pr is left in the aqueous phase in a 'pure' form
Filter paper impregnated with MnO_x with Mn having an average oxidation number of +3	A $^{144}Ce/^{144}Pr$ equilibrium mixture is administered on to the centre of the filter paper	^{144}Pr is separated from ^{144}Ce by the ring oven technique: with 0.06–0.07M trichloroacetic acid as wash solution, ^{144}Pr can be concentrated in the ring zone with 80–90% yield and decontamination factors $>10^3$
Column support material of MnO_2	—	—

TABLE 14. THE ^{144}Ce/^{144}Pr GENERATORS AND COLUMN SUPPORT MATERIAL USED FOR SELECTIVE SEPARATION OF ^{144}Pr FROM AN INVENTORY OF RADIOCHEMICALLY PURE ^{144}Ce (cont.)

Generator material	Chemistry involved	Yields
Column support material of MnO_2 prepared by mixing $MnSO_4$ and $KMnO_4$, followed by filtering, washing, drying, grinding and sieving. Fraction of 72–120 mesh is dispersed in 0.1M HNO_3 and filled into a column	Carrier free ^{144}Ce (probably as Ce(IV)?) is loaded on to the column in 0.1M HNO_3	^{144}Pr is eluted with 6 mL of 0.1M HNO_3
Column support material of Al_2O_3 coated with MnO_2	Ce(III) oxidized to Ce(IV) and absorbed on the column as $Ce(IO_3)_4$	^{144}Pr elution with 1% KIO_3 in 1M HNO_3 with an elution yield >60% and a radionuclide purity >99.9%
Column support material of zirconium phosphate (possibly of the form $Zr(HPO_4)_2 \cdot nH_2O$)	Ce(III) oxidized to Ce(IV) by $NaBrO_3$ in H_2SO_4 medium and ^{144}Ce(IV) fixed on column	^{144}Pr eluted with water at pH7 or slightly higher
Column support material of crystalline zirconium arsenate (possibly $Zr_3(AsO_4)_4(H_2O)_2$)	^{144}Ce fixed on column material	Elution of ^{144}Pr with a 'suitable' elution agent
Column of Dowex 1x8, washed with 2M HCl and preconditioned with 6M HNO_3 and 0.3M $KBrO_3$	The ^{144}Ce solution in 6M HNO_3 and 0.2M $KBrO_3$ was applied to the column and washed with a mixture of 1M HNO_3, 6M $NaNO_3$ and 0.2M $KBrO_3$	^{144}Pr was eluted with 3 mL of a mixture of 1M HNO_3, 6M $NaNO_3$ and 0.2M $KBrO_3$ The elution yield was ~75% and the contamination of ^{144}Ce <0.02%

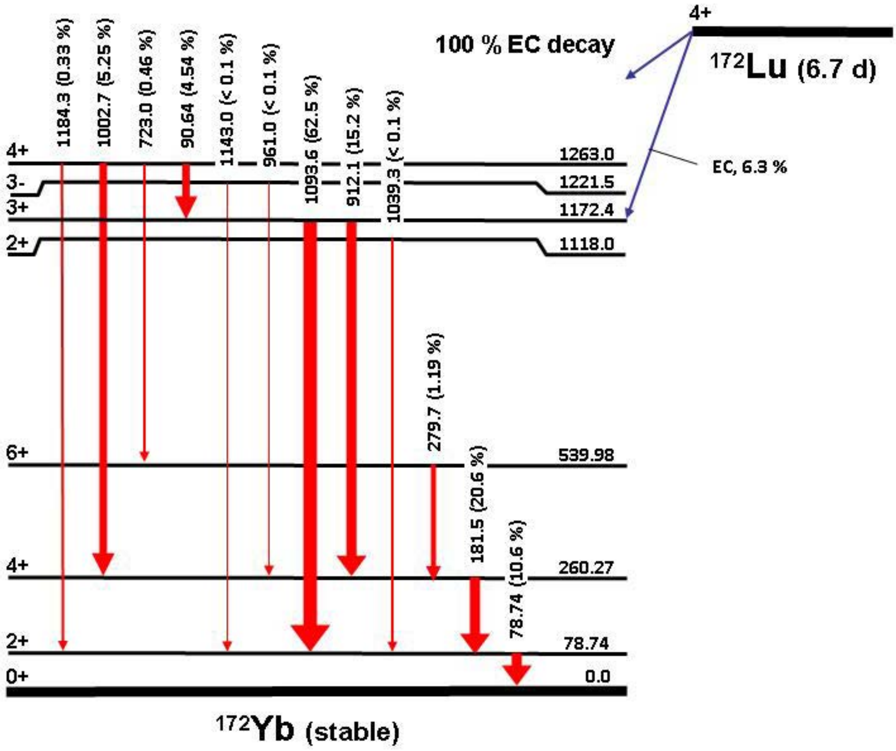


FIG. 66. Partial decay scheme of ^{172}Lu to excited states in ^{172}Yb whose ground state is stable [81].

Post-processing systems to synthesize a variety of industrial ^{144}Pr tracers as aqueous, environmental, organic and solid tracers are possible. However, systematic studies on suitable tracers for aqueous and organic fluids, and mapping of their thermal, chemical and microbial stability have not yet been carried out to a sufficient degree.

3.2.10. The ^{172}Hf/^{172}Lu radionuclide generator

Relevant nuclear data for this radionuclide generator system are given in Table 4, and the simplified and partial decay scheme of ^{172}Lu is given in Fig. 66.

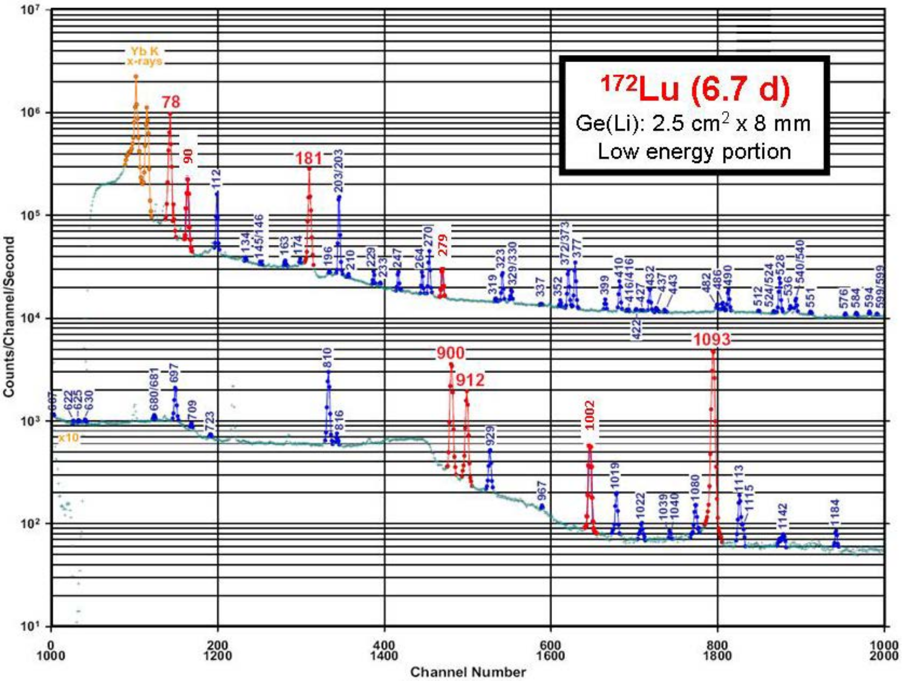


FIG. 67. Gamma spectrum of ^{172}Lu accumulated with a Ge(Li) detector. The most interesting energies are marked in red.

3.2.10.1. Relevant nuclear properties

(a) Gamma energies of the daughter

The decay scheme in Fig. 66 is only a partial decay scheme. The complete scheme is rather complex: there are relatively abundant gamma energies from 79 keV to 1093 keV. A gamma spectrum of ^{172}Lu is given in Fig. 67. It is very complicated with all its gamma peaks, and it would therefore be preferable to use this radionuclide in combination with a detector with high energy resolution.

The high abundance 1093 keV (63%) and 900 keV (30%) photons are optimal for transmission through thick metal walls and detection outside of the vessel or tubing. The low energy gamma rays at 79 keV and 181 keV are sensitive to smaller mass thicknesses. Hence, this radionuclide is suitable for a suite of on-line measurements in the industry.

(b) Half-life of the daughter

The daughter half-life of 6.7 d is adequate for relatively long duration (weeks) investigations, and is the longest lived considered in the present document.

(c) Half-life of the mother

The half-life of the mother radionuclide of 1.87 a offers, in principle, very long time (several years) application without replacing the generator, provided that the generator column keeps up its performance with acceptable quality with respect to elution yield and mother breakthrough.

(d) Main production routes

At moderate beam energies, the mother radionuclide is produced by ^{Nat}Yb(p,xn)^{172}Hf (typically 30–40 MeV), ^{175}Lu(p,4n)^{172}Hf (typically 40–50 MeV) or, for instance, ^{169}Tm(^{7}Li,4n)^{172}Hf (beam energy 35 MeV) reactions. The yields are relatively small at beam intensities of a few particle μA. The highest yields have been achieved in proton induced spallation processes on Ta, W or other high Z targets. Sophisticated separation chemistry to isolate ^{172}Hf from tantalum or spallation targets has been developed. Eventually, cooling periods are required to minimize the amount of other co-produced and more short lived hafnium radionuclides.

An example of the complex chemistry involved in working up the hafnium from an irradiated tantalum target is given in Fig. 68. This is essentially a procedure developed at Los Alamos National Laboratory. For a period of approximately 2 months, a tantalum metal target was irradiated at the Isotope Production Facility (A-6) at LAMPF. The target was irradiated with ~750 MeV protons with an intensity of ~350 μA. The irradiation area received ~10^{22} protons (6×10^5 μA/h) during this 2 month period. The tantalum target consisted of three slabs of the metal. Each slab weighed ~250 g for a total target weight of 770 g. The slabs measured 8 cm^2 and 0.32 cm thick.

Chemistry was started approximately 2 months after the irradiation. At that time, the radiation level of the target was >500 R/h at 1.5 m.

The overall chemical yield for ^{172}Hf was estimated at >90%, for Lu ~90% and for gadolinium and europium 60–70%. The amount of ^{172}Hf at the end of irradiation was estimated to be ~0.25 TBq (~6.7 Ci).

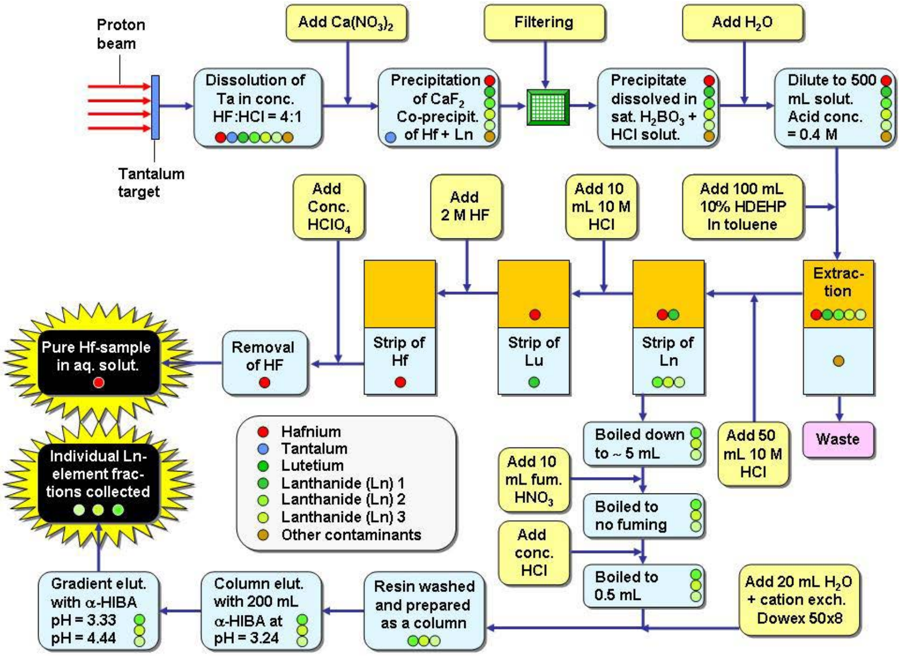


FIG. 68. Schematic diagram of the chemical procedure developed at LAMPF for working up the hafnium (including ^{172}Hf) and heavier lanthanides from the mixture of spallation products after 750 MeV proton irradiation of a tantalum metal target.

3.2.10.2. Generator systems, separation techniques and chemical form of the daughter

While the literature is rich on papers dealing with chemical complexation of Lu isotopes due to the radiopharmaceutical use of ^{177}Lu, it is correspondingly poor on development of ^{172}Hf/^{172}Lu generators. One useful example is the following: a ^{172}Hf/^{172}Lu radionuclide generator based on extraction chromatography with HDEHP as the extractant and Chromosorb-102 as the inert support has been developed in India [163]. A column (0.5 cm × 4.0 cm) was filled with 1.5 g of the inert support Chromosorb-102 (100–200 mesh). Each gram of Chromosorb-102 was impregnated with 100 mg of HDEHP.

The mother ^{172}Hf, dissolved in 5M HNO_3, was loaded on to the column. The ^{172}Lu generated could be eluted quantitatively by using 5M HNO_3. The ^{172}Lu obtained by this method was free from detectable ^{172}Hf activity ($<4 \times 10^{-5}$ of ^{172}Hf).

The characteristics of the column material remained unchanged during 2 years of operation. This has demonstrated the suitability of the technique for availing ^{172}Lu for radiotracer investigation.

3.2.10.3. Availability and current status

At present, there is no commercial availability of this generator. During production, there is a limited batch yield of about 10 mCi, which is, however, sufficient to prepare generator systems.

3.2.10.4. Possible applications

The ^{172}Hf/^{172}Lu generator is an excellent source for availing ^{172}Lu for industrial radiotracer applications. As displayed in Fig. 66, ^{172}Lu emits a multiplicity of gamma rays in its decay, with energies from 79 keV to 1622 keV, including a 63% abundant 1094 keV gamma transition. The high energy γ radiations of ^{172}Lu allow its use as a radiotracer in most types of industrial application, in particular, for investigation of processes inside thick walled pipes and vessels.

The ^{172}Lu exhibits general rare earth chemistry, including complexation with a wide variety of chelating agents and thus permits the preparation of a wide variety of industrial radiotracers. Moreover, Lu exists only in the +3 oxidation state. Any redox complications of its solution chemistry are thus eliminated. Since ^{177}Lu is a popular radiopharmaceutical agent, one has a good starting point for production of industrial radiotracers. However, much work remains to be done to prepare aqueous, organic and solid tracers based on ^{172}Lu. Stress studies on thermal, chemical and microbial stability for various temperatures and chemical compositions are especially important for industrial radiotracers.

The 6.7 d half-life of ^{172}Lu is a valuable asset for real time radiotracer experiments. Moreover, the 6.7 d half-life of ^{172}Lu is convenient for generator post-elution chemical manipulations, tracer shipment to remote sites and experimental utilization.

The long half-life of the parent radionuclide ^{172}Hf ($T_{1/2}$ = 1.87 a) ensures convenient cost effective availability of ^{172}Lu for long periods of time, provided there is sufficient stability of the column support against chemical and radiolytical degradation.

4. GUIDELINES FOR USE OF RADIONUCLIDE GENERATORS FOR RADIOTRACERS FROM EXPERIMENTS

Most tracer studies on industrial process plants are carried out over relatively short time periods, ranging from several seconds to a few hours at maximum. For such studies, radionuclide generators provide a useful means of providing short lived radionuclides on-site. For the effective and safe utilization of radionuclide generators in industry, it is important to ensure that:

- The radionuclide generator is of robust construction, suitable for use in the industrial environment.
- The radionuclide generator is transported safely to and from the worksite.
- Appropriate, written operating instructions and safety procedures for the generator are in place and are adhered to.
- An appropriate procedure is followed to produce the radiotracer from the radionuclide generator eluate.
- The radiotracer is appropriate for the system to be traced.

4.1. PLANNING A RADIOTRACER EXPERIMENT

Prior to carrying out any work on an industrial site with a radionuclide generator, detailed planning is required. The three basic principles of radiation protection as set out by the ICRP, namely, justification, optimization and dose limitation, must be given due consideration.

The proposed work must be discussed with, and approved by, a radiation protection adviser. It is important that a job specific work procedure should be compiled and that this should address the doses that might be received by those carrying out the fieldwork. This should include a description of the controlled and supervised areas that need to be set up. It should also contain a contingency plan describing the actions to be taken in the event of an accident.

Before work can commence, a permit to work must be obtained from the plant supervisor. In carrying out work with radioactive material, it is possible that the radiation from the scanning source may interfere temporarily with the operation of any nuclear level gauge in the near vicinity. The plant operators must be made aware of this possibility and, if necessary, these instruments should be put on manual control for the duration of the measurements.

Prior to the elution taking place, a controlled area will need to be established at the worksite. This is generally done by ‘roping-off’ an area around the injection point. For a controlled area, the dose rate outside the rope barrier should not exceed 2.5 μSv/h. The controlled area should be divided into ‘clean’ and ‘dirty’ areas. Facilities to enable any required decontamination of the field engineer must be available and a gamma radiation dose rate meter and a contamination monitor must be to hand.

4.2. GUIDELINES FOR TRANSPORTATION OF CONTAINERS OF RADIONUCLIDE GENERATORS

The packing and transportation of radioactive material is governed by stringent rules and regulations. The first set of rules and regulations for the safe transport of radioactive material was published by the IAEA in 1961 and based on ICRP recommendations, which forms the basis for all national and international regulations. These regulations are constantly reviewed, revised and updated from time to time by the ICRP and the IAEA. Recently, the IAEA has published revised regulations for the safe handling and transport of radioactive material [164].

4.2.1. The regulatory framework

The transport of radionuclide generators is controlled by the same regulations that apply to the transport of all radioactive material. General guidelines for transport operations are provided in the IAEA safety standard, Regulations for the Safe Transport of Radioactive Material 2009 Edition [164]. The normal modes of transportation are road, air and sea. Transport is rarely carried out by rail.

Road transport is controlled by national regulations. Sea transport is controlled by the International Maritime Dangerous Goods Code. Air transport is controlled by the International Air Transport Association Dangerous Goods Regulations. These regulations are revised annually or every two years, and, therefore, care should be taken to ensure that the current edition is used.

4.2.2. Classification of packages

Packages containing radioactive material are categorized for transport on the basis of their surface dose rate and on their transport index. The transport index is the number obtained by measuring the highest dose rate obtained at a distance of 1 m from any surface of the package in microsieverts per hour divided by 10. This number should be rounded up to one decimal place.

The packaging of radioactive material is tightly controlled. If the quantity of the parent radionuclide in an isotope generator is below 10^{-4} of the A_2 value, then it may be possible to transport the radioisotope generator as an excepted package. In the vast majority of cases, the radioisotope generators used for field studies are in excess of these quantities and must, therefore, be transported as Type A packages. This involves the use of approved, certified Type A transport containers. To be certified as a Type A transport container, the design type must have undergone rigorous tests, including drop tests, penetration tests and temperature cycling.

Depending upon the transport index and the surface dose rate, the package will be designated as I-White, II-Yellow, III-Yellow, as indicated in Table 15.

Where the transport index satisfies the condition for one category but the surface radiation level satisfies the condition for a different category, the package must be assigned to the higher category of the two.

4.2.3. Marking and labelling the package and documentation

The shipper of the radioisotope generator is responsible for ensuring that the correct marks and labels are on the package before it is transported. Any irrelevant markings or labels already on the package should be removed. The package should then be marked with each of the following:

- Proper shipping name: In the case of a radionuclide generator, this will be radioactive material, Type A package.
- United Nations number preceded by the letters ‘UN’. In this case, it will be UN 2915.
- The full name and address of the shipper.
- The full name and address of the consignee.

TABLE 15. TRANSPORT INDEX

Transport index	Maximum radiation level at any point on the external surface	Category
0	Not more than 0.005 mSv/h	I-White
More than 0 but not more than 1	More than 0.005 mSv/h but not more than 0.5 mSv/h	II-Yellow
More than 1 but not more than 10	More than 0.5 mSv/h but not more than 2 mSv/h	III-Yellow

- Type A package identification is in the format: TYPE A/International Vehicle Registration Code/name of the manufacturer or other identification specified by the competent authority of the country of origin or design.
- The package must bear the correct labels and again this is the responsibility of the shipper. There are two types of label: hazard labels, set at an angle of 45°, and handling labels. The hazard labels are either I-White, II-Yellow or III-Yellow.

After the generator has been correctly packaged and labelled the shipment should be documented by completing the appropriate shippers declaration document for either road, sea or air transportation. It is essential that the information given in this document is exactly the same as that on the package.

4.3. GUIDELINES FOR THE EVALUATION OF RADIONUCLIDE GENERATORS

Before using the generator, it is recommended that a measurement of the surface dose rate on the generator containment be carried out and that a swipe test be performed close to the eluate exit to check for any loose contamination.

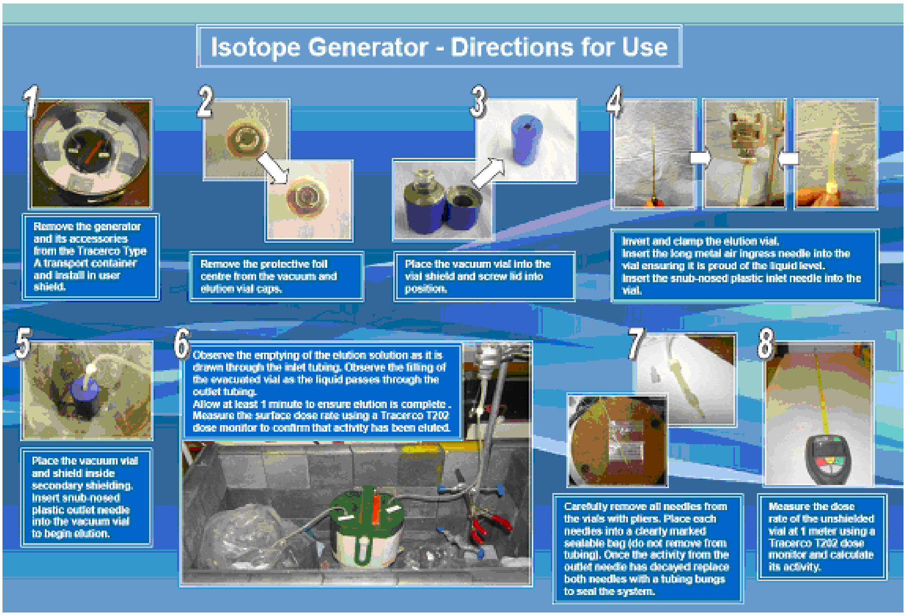
4.3.1. Elution of tracer

The isotope generator should be taken into the controlled 'dirty' area in readiness for elution. While the parent nuclide has a certain amount of shielding so that it may be transported safely, the eluate vessel has none. Some suitable form of shielding is, therefore, required. This can be in the form of a small lead brick wall or lead lined pot.

It is recommended that the daughter radionuclide should be eluted by applying suction only. No pressure should be built up inside the ion exchange column, as it is likely to disrupt the integrity of the column packing material. This may also cause bleeding of any parent radionuclide that is not strongly bound to the adsorbent.

The elution should be carried out in accordance with the elution procedure provided by the producer of the generator. An example is given in Fig. 69.

On completion of work, the field engineer should monitor thoroughly before leaving the controlled area.



Isotope Generator - Directions for Use

1. Remove the generator and its accessories from the Tracerco Type A transport container and install in user shield.
2. Remove the protective foil centre from the vacuum and elution vial caps.
3. Place the vacuum vial into the vial shield and screw lid into position.
4. Invert and clamp the elution vial. Insert the long metal air ingress needle into the vial ensuring it is proud of the liquid level. Insert the snub-nosed plastic inlet needle into the vial.
5. Place the vacuum vial and shield inside secondary shielding. Insert snub-nosed plastic outlet needle into the vacuum vial to begin elution.
6. Observe the emptying of the elution solution as it is drawn through the inlet tubing. Observe the filling of the evacuated vial as the liquid passes through the outlet tubing. Allow at least 1 minute to ensure elution is complete . Measure the surface dose rate using a Tracerco T202 dose monitor to confirm that activity has been eluted.
7. Carefully remove all needles from the vials with pliers. Place each needles into a clearly marked sealable bag (do not remove from tubing). Once the activity from the outlet needle has decayed replace both needles with a tubing bungs to seal the system.
8. Measure the dose rate of the unshielded vial at 1 meter using a Tracerco T202 dose monitor and calculate its activity.

FIG. 69. Procedure for eluting the generators.

4.3.2. Elution efficiency and breakthrough test

When a daughter radionuclide from the generator is eluted, it is possible to elute the mother nuclide from the column material along with the product. This is likely to interfere with the tracer studies and may cause environmental pollution. This must be measured and kept within permissible limits. The presence of excessive metal ions of column material in the eluate indicates lack of stability of the generator column.

Before field application, the following laboratory tests are to be conducted with the eluate of the generator to evaluate its performance:

- Assess physical appearance — whether clear or whether any turbidity is present.
- Measure the activity of the daughter radionuclide by an absolute counting technique to determine the elution efficiency.
- Measure the elution efficiency as a function of time and number of elutions and total eluate volume.
- Check the γ ray spectrum of the decayed eluate to measure the breakthrough of the mother radionuclide.

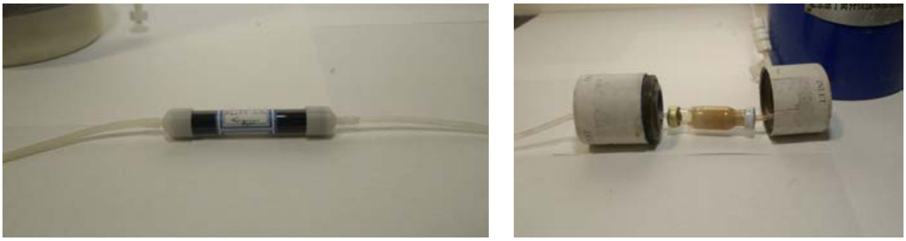


FIG. 70. The catcher columns installed to the Cs/Ba generator and Ge/Ga generator.

4.3.3. Methods to reduce breakthrough and to increase the shelf life of radionuclide generators

For the radionuclide generators such as ^{137}Cs/^{137m}Ba, ^{68}Ge/^{68}Ga and ^{44}Ti/^{44}Sc, and the long half-lives of their parent radionuclides, it is particularly important to avoid breakthrough. However, since the possibility of breakthrough cannot be completely eliminated, the following techniques are used to prevent contamination of the eluate by the long lived mother radionuclide.

4.3.3.1. Catcher columns

Elution and breakthrough tests to study the effect of catcher columns were carried out in the Republic of Korea with the ^{68}Ge/^{68}Ga and ^{137}Cs/^{137m}Ba generators produced in China by the CIAE. The catcher column packing materials are the same as those used in the generators (SnO_2 for the ^{68}Ge/^{68}Ga generator and KCoFC–SiO_2 for the ^{137}Cs/^{137m}Ba generator).

By adding the catcher column (Fig. 70) to the ^{137}Cs/^{137m}Ba generator and to the ^{68}Ge/^{68}Ga generator, the breakthroughs decreased dramatically.

4.3.3.2. Recycling elution

The group at Mainz in Germany has developed efficient and fast on-line protocols for post-processing of the generator eluate. The generator daughter is trapped using cation exchange purification while the rest of the eluate is directed into a waste vial. This results in several advantages:

- The post-processing takes 4 min, with overall recovery yields of 97 ± 2% for ^{68}Ga and 90 ± 2% for ^{44}Sc.
- It results in volume reduction along with chemical and radiochemical purification.

- This procedure leads to almost complete removal of metallic impurities, including the mother breakthrough, down to only becquerel levels.
- Since the content of the mother nuclide in the daughter is negligible, the generators can be used with a significantly extended shelf life.
- The breakthrough is easily collected in waste vials. From these collected waste volumes, the mother nuclide may be chemically processed for recycling.

4.4. GUIDELINES FOR PREPARATION OF RADIOTRACERS FROM RADIONUCLIDE GENERATOR BASED RADIOISOTOPES

It is a requirement of any tracer that it should mimic the behaviour of the material into which it is introduced and that it should faithfully follow that material through the system of interest. Thus, in selecting a tracer for a particular application, care must be taken to ensure its physical and chemical stabilities under the prevailing process conditions.

The generator systems selected in Table 16 provide daughter radionuclides of different half-lives, different gamma energies and different chemical properties. Except for ^{68}Ge/^{68}Ga and ^{99}Mo/^{99m}Tc generators, which can be used with organic eluants, all the other daughters are available in aqueous solutions of acidic or neutral pH in the form of anions (TcO_4^-) or cations (Ba^{2+}, Ga^{3+}, Sc^{3+}, In^{3+}, Pr^{3+}, Lu^{3+}).

Tracers may be required to study aqueous, organic or solid systems. To synthesize chemical species compatible with these systems using the above radioactive cations, several strategies, such as complex formation, sorption, reduction and solvent extraction, are employed.

Aqueous eluates may be used as aqueous radiotracers, provided they are applied to simple aqueous systems. However, as the metal cations tend to be adsorbed on to solid surfaces, it is necessary to make anionic metal complexes using chelating agents (L^a), such as EDTA, DOTA and NOTA. The complexes move with aqueous media without significant absorption on to surrounding solid materials.

Complexes of the cations with ligands (L^b) with a backbone substituted lipophilic group are used as organic radiotracers with low solubility in aqueous solutions and high solubility in organic solutions. Complexes of the cations with bifunctional ligands (L^c) linked via one of their functional groups covalently to a transport vector, such as solid particles, and via the other functional group to metal radionuclides providing strong complex formation are used as solid tracers. This may also be true of nanoparticles.

TABLE 16. CHEMICAL FORM OF DAUGHTER RADIONUCLIDES FROM SELECTED GENERATOR SYSTEMS AND POSSIBILE TRACER SYNTHESIS

Generator	Eluant	Chemical form	Tracer synthesis strategies		
			Aqueous	Organic	Solid
$^{68}Ge/^{68}Ga$	Aqueous (HCl)	$[Ga(H_2O)_6]^{3+}$	Ga–L[a]	Ga–L[b]	Ga–L[c], sorption
	Acetylacetone	Ga(acac)		Ga(acac)	
$^{44}Ti/^{44}Sc$	Aqueous (HCl)	$[Sc(H_2O)_6]^{3+}$	Sc–L[a]	Sc–L[b]	Sc–L[c], sorption
$^{172}Hf/^{172}Lu$	Aqueous (HCl)	$[Lu(H_2O)_n]^{3+}$	Lu–L[a]	Lu–L[b]	Lu–L[c], sorption
$^{144}Ce/^{144}Pr$	Aqueous (HCl)	$[Pr(H_2O)_6]^{3+}$	Pr–L[a]	Pr–L[b]	Pr–L[c], sorption
$^{137}Cs/^{137m}Ba$	Aqueous(NaCl)	$[Ba(H_2O)_n]^{2+}$	Ba–L[a], $BaCl_2$	Ba–L[b]	Ba–[c], sorption
$^{113}Sn/^{113m}In$	Aqueous (HCl)	$[In(H_2O)_6]^{3+}$	In–L[a]	In–L[b]	In–L[c], sorption, reduction
$^{99}Mo/^{99m}Tc$	Aqueous(NaCl)	$[TcO_4]^-$	$[TcO_4]^-$	Solvent extraction	Reduction to TcO_2
	Tri-n-butylphosphate	Tc(TBP)		Tc(TBP)	

[a] Chelating agents.

[b] Ligands.

[c] Bifunctional ligands.

4.5. GUIDELINES FOR VALIDATION OF RADIONUCLIDE GENERATOR BASED RADIOTRACERS

After having produced a radiotracer candidate, this should be subjected to a sequence of laboratory tests to validate its applicability to the industrial investigation in hand. Tracer candidates which do not pass the tests are unlikely to be appropriate and should not be considered further.

Most industrial radiotracer applications involve the use of physical tracers to study the physical dynamics of phases. In these cases, passive tracers are normally used. For radionuclide generator based passive radiotracers, the following items should be considered for validation:

- Flow behaviour similarity (no size exclusion or ion exclusion owing to electric charges);
- Physical stability (no evaporation, segregation or mechanical rupture);
- Chemical stability (no chemical degradation, sorption, partitioning or precipitation);
- Biological stability (not affected by microbial activity in the traced system);
- Thermal survivability (no significant thermal degradation in the system under study);
- Pressure survivability (not affected by increasing or decreasing pressure);
- Labelling efficiency (readily possible to label the intended physical phase);
- Radioactivity (suitable half-life and radiation characteristics);
- Specific radioactivity (may, in some cases, need a high specific radioactivity);
- Radiochemical purity (concentration of other compounds labelled with the same radionuclide within acceptable limits);
- Radionuclide impurity (concentration of other radionuclides present not detrimental for the intended tracing operation);
- Density control (in cases where identical density is required, i.e. for particle tracing);
- Packaging and handling (appropriate procedures have to be implemented in each case);
- Chemical toxicity (has to be as low as possible and data sheets have to be provided).

Detailed guidelines should be developed for validation of specific radionuclide generator based radiotracers for specific applications.

4.5.1. Thermal stability tests

Thermal stability tests are carried out in closed (sealed) vials of polypropylene (<100°C), borosilicate glass or quartz glass containing the pertinent liquid medium. If testing at high pressures is needed, one may have to use steel, hastalloy or titanium vials. Parameters tested may be temperature, pressure, pH, ion composition, concentration of multivalent ions, etc.

The degradation products of a radiotracer normally behave differently from the radiotracer. Removal of the degradation products can normally be done by ion exchange or solvent extraction. In many cases, the degradation products adsorb on to the walls of the vials and are, therefore, removed from the solution. The quotient $R_{x,t}/R_{x,0}$ should be recorded as a function of the parameters tested, where $R_{x,t}$ is counting rate of the radiotracer still in solution after a time t for the parameter value x and $R_{x,0}$ is the corresponding counting rate at time zero. For a good tracer, this value should be constant, with a value of ≈ 1.

4.5.2. Sorption stability tests

For radiotracers that are supposed to come into contact with solid material, sorption tests should be conducted. In nature, they are similar to the thermal stability tests, the only difference being that the vials contain the solid material as a finely divided substrate. Sometimes, it is appropriate to carry out the thermal and sorption tests together in the same vial. The results will then show the composite effect, which may be sufficient for the purpose.

4.5.3. Microbial stability tests

For tracers which have a chance to be biodegraded, corresponding stability tests have to be conducted under aerobic and anaerobic conditions. The experiments are similar to those described above.

4.5.4. Partitioning tests

Tracer molecules, which are non-charged and which contain at least one substituent (ligand) with a lipophilic character, can partition between aqueous and organic phases. For a passive water tracer, for instance, this is undesirable. Therefore, the partition coefficient (distribution ratio), $K_D = [\mathrm{Tr}]_0/[\mathrm{Tr}]_w$, of these molecules should be measured for actual liquid systems (water type and composition and organic phase type and composition) as a function of actual parameters (temperature, salinity, pH, etc.).

Here, $[Tr]_0$ is the concentration of tracer in the organic phase at distribution equilibrium and $[Tr]_w$ is the corresponding concentration in the aqueous phase. The experiments are carried out in simple equipment such as a separation funnel at ambient conditions, and more advanced equipment as temperature and/or pressure increases.

4.5.5. Dynamic flooding tests

Tracer candidates that have passed the preceding tests continue to the final laboratory test, which is a dynamic experiment where the tracer candidate is injected as a sharp low volume pulse, together with a standard reference tracer (mainly HTO for water) into a liquid stream. The stream enters the test system, which should simulate the industrial conditions as closely as possible (except for volume). In-line detectors can be applied on the exit fluid stream or samples can be taken at regular intervals so that the tracer production curves can be recorded. There will be one curve for the standard reference tracer and one for each tracer candidate. The latter production curves should resemble those of the standard reference as closely as possible. From these tests, it is possible to extract quantitative information on, for instance, sorption on to solid substrates or partitioning between fluids. This test is the last test in the laboratory before the systems developed are tested in the field.

4.5.6. Radiochemical purity tests

The radiochemical purity of radiotracer is very important for the accuracy of tracer experiments. For example, any uncomplexed form of ^{113m}In in the ^{113m}In-EDTA tracer will be adsorbed in the system during the process investigation. The radiochemical purity of the generator eluate or radiotracer can be determined by a paper chromatographic technique using standard protocol.

4.6. GUIDELINES FOR RADIOLOGICAL SAFETY IN THE APPLICATION OF RADIONUCLIDE GENERATOR BASED RADIOTRACER

4.6.1. Radiation dose considerations for the injection of radiotracer generators in field tests

Prior to carrying out work with a radionuclide generator based radiotracer, it is necessary to make an estimate of the radiation dose that may potentially be received by the persons carrying out the project, by the workforce on the industrial site and by members of the general public. The method of making this assessment will clearly vary from case to case.

As an example, the radiation dose received by the radiotracer practitioner under the following conditions should be considered:

- Activity of the radionuclide: 3.7 GBq (100 mCi) considered as a point source.
- Distance of the operator to the source: 1 m.
- Duration of the injection operation: 1 min without shielding (elution + transfer to injection system).
- Shielding of the injection system: 3 cm thick lead shielding.

The results of the dose calculations are presented in Table 17. The conclusion is that under normal conditions, the gamma radiation dose to the operator is very low. It is also worth pointing out that in many cases, the activities required for field tests are significantly lower than those used in the above calculations. However, if, for any potential application, the calculation should indicate the potential for a significant radiation dose to be received, then measures must be taken to reduce that dose to a level as low as reasonably achievable (ALARA). Such measures might include the provision of additional shielding, or the use of remote handling equipment.

It can almost always be shown that the potential dose to workers on the industrial site where the project is being carried out will be negligible, because of the dilution of the tracer in the process stream, the shielding provided by the thick metal walls of process vessels, the short half-life of the radionuclide, etc. However, should the potential for a significant dose to workers, or to members of

TABLE 17. DOSE FROM 3.7 GBq RADIOTRACER INJECTION FOR FOUR RADIONUCLIDES

Nuclide activity (3.7 GBq)	Dose rate at 1 m from the unshielded source (μSv/h)	Dose for 1 min operation (μSv)	Dose rate at 1 m from the external walls of the injection system (μSv/h)
^{99m}Tc	70	1.2	4×10^{-20}
^{113m}In	150	2.5	9×10^{-2}
^{68}Ga	470	7.8	8.6
^{137m}Ba	300	5.0	15

the general public exist and should it prove impossible to reduce the dose to an acceptable level by altering the test procedure, then the project must not be undertaken.

Despite the fact that the radiation doses received by the operator and workers on-site are negligible, there is still an obligation on the part of the radioisotope practitioner to keep the worker on-site fully informed about the nature of the work that is being carried out, explaining clearly the potential risk due to radiation dose and the precautions that will be taken to ensure that the work can be carried out safely.

4.6.2. Regulatory framework

The regulatory requirements for the conduct of radiotracer experiments are determined at a national level. These vary from country to country, but are commonly determined with reference to safety standards and guidelines issued by the IAEA and the ICRP. At the highest level, the IAEA's Fundamental Safety Principles [165] are relevant to environmental radiotracer applications in requiring that:

- The benefits of an activity must outweigh the radiation risk (Principle 4).
- Activities should be optimized to ensure exposures are ALARA (Principle 5).
- Radiation exposure to ecosystems should be limited to ensure protection of populations of a species (as distinct from individual organisms) (Principle 7).

These principles are binding on the IAEA for its own operations, and they are also recommended for use by Member States in relation to their own activities.

The IAEA International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources [166] lists the requirements for the use of radiation and radioactive material. An environmental radiotracer application can be considered to be a planned exposure, as defined in this publication, which covers applications of radiation and radioactive material in industry, agriculture and research.

National regulations covering the use of radiotracers in the environment vary from jurisdiction to jurisdiction and it is not appropriate to address these comprehensively here. Generally, the regulation of radiotracer activities is carried out in three ways:

(i) Licensing of an organization to possess and use radioisotopes;
(ii) Licensing of individuals to handle radioisotopes;
(iii) Approval for the release of a radioisotope tracer into the environment.

The injection of a radiotracer into the environment for the purposes of studying environmental processes is, in effect, a radioactive discharge and is considered by many national jurisdictions to be a form of radioactive waste disposal. In some jurisdictions, the limits applied to environmental radiotracer studies are derived from those applied to waste discharges.

In most cases, there is a requirement for pre-approval of each radiotracer experiment either by the national regulator (e.g. France, Indonesia, Sweden, Uruguay) or, in some cases, by a radiation safety body within the applicant's organization (e.g. Australia, Brazil, Republic of Korea) where general licensing for radiotracer studies has already been obtained. The application for approval requires a radiation safety assessment of the release to be carried out, addressing the following aspects:

- Purpose of the experiment and justification for the use of radiotracers;
- Location of the experiment and predicted radioactivity levels in the environment during and following the release;
- Radiological dose calculations for planned radiation exposure of workers and the public and potential unintended exposures;
- Safety measures and monitoring to be employed to minimize radiological dose and risk of injury to workers and to the public during and following the experiment.

In the majority of jurisdictions, a radiological risk assessment is only required for humans potentially exposed to the radiotracer. An underlying assumption is that the environmental impact of radiotracer releases is negligible owing to the rapid dispersion of the radiotracer in the receiving environment. However, in some jurisdictions, particularly Australia, it is also a requirement to address the environmental impact of the radiotracer study in conducting the risk assessment.

General guidelines for the safe handling of radioisotopes have been issued by the ICRP and the IAEA from time to time [165, 167, 168].

Generally, safety regulations include justification of the use of radiotracer, optimization of radiation exposures and the annual dose limits in order to prevent unnecessary exposure. Justification implies that the competent authority should not allow the use of radiation unless a net positive benefit arises from its use.

The design of a radiotracer experiment has to ensure optimization of radiation exposures. All the exposures have to be ALARA. The optimization of radiation exposures depends initially on selecting the most appropriate radioisotope for the project and ensuring that the activity used is kept to a minimum, consistent with achieving the aims of the project. Additionally, distance, time and shielding are used to minimize the dose received:

- The dose rate at a point varies in inversely proportion to the square of the distance between the source and the point. Therefore, radiation workers have to maintain the maximum possible distance between themselves and a radiation source.
- The dose received is directly proportional to the time spent in handling the source. Thus, the time spent handling the source should be minimized.
- The radiation intensity at a point reduces exponentially with the thickness of shielding material. Thus, an optimum thickness of the shielding material has to be used between the source and a radiation worker.

The relevant annual dose limits have to be taken into account and no individual must be exposed to more than the prescribed limit. These dose limits recommended by the ICRP are 1 mSv/a for a member of the public and 20 mSv/a for a radiation worker [169].

5. PREPARATION OF RADIOTRACERS FROM GENERATOR PRODUCED RADIONUCLIDES

5.1. RADIOTRACERS FROM THE ^{68}Ge/^{68}Ga GENERATOR

The high energy photon (1077 keV, abundance: 3.2%) emitted by ^{68}Ga can be detected through thick metal walls and so is suitable for on-line measurements on industrial process plants. Gallium-68 is a positron emitter and so the coincidence detection of annihilation radiation (511 keV, abundance: 178.3%) presents the possibility of determining local radionuclide distributions with high resolution.

There is an increasing demand for ^{68}Ga for PET imaging in medicine, but, in addition, there is good potential for a variety of ^{68}Ga compounds as aqueous, organic and solid phase tracers for industrial applications. The fact that the generator can be used three to four times per day at maximum activity, allowing repeated radiotracer studies, is a particular advantage in this respect. Commercial ^{68}Ge/^{68}Ga generators are produced using modified TiO_2 columns, modified SnO_2

columns, modified (nano-particular) ZrO_2 matrices, as well as organic resins. Common to all systems, hydrochloric acid with concentrations ranging from 0.01N to 1.0N is used for the elution of ^{68}Ga.

In order to avoid any radioactive contamination of the systems under investigation and the environment around them, it is very important to make sure that the concentration of the mother nuclide in the eluate is as low as that specified by the supplier of the generator. Therefore, the eluate needs to be carefully monitored throughout the life of the generator to check for the breakthrough of the mother radionuclide and also to ensure that the elution efficiency continues to be satisfactory.

For preparation of ^{68}Ge/^{68}Ga generators, several radiochemical criteria are relevant, such as effective separation strategies providing high ^{68}Ga elution yields and low ^{68}Ge breakthrough, long term stability and the type of eluate useful for subsequent labelling reactions (low volume, low pH, high purity, etc.). The eluate is obtained as a complex of ^{68}Ga tetrachloride in hydrochloric acid solutions. The ^{68}Ga complex starts to hydrolyze and begins to form hydroxides at pH4. These tend to form colloids or pseudocolloids with unknown physicochemical characteristics and migration behaviour. In consequence, ^{68}Ga should be transformed into species that are chemically stable under environmental conditions. In neutral pH, the ^{68}Ga cation is easily adsorbed on to the surface of solids, creating a stable bonding. Therefore, for the tracing of liquid systems, ^{68}Ga should be transformed into species that are stable in aqueous or organic media.

Thus, the eluate from the ^{68}Ge/^{68}Ga generator can be used either for the labelling of solid particles (by surface adsorption) as solid tracers or for the synthesis of ^{68}Ga–ligand complexes of very high thermodynamic and kinetic stabilities, capable of ‘surviving’ any harsh physical and chemical conditions in the industrial system to be studied.

5.1.1. Aqueous ^{68}Ga tracers

5.1.1.1. The eluate of a ^{68}Ge/^{68}Ga generator

The eluate of a ^{68}Ge/^{68}Ga generator was tested as a radiotracer on a large scale laboratory installation at the INCT in Warsaw, Poland (a model of a rectangular settler on a wastewater plant). A fluorescent dye was used as a reference tracer for the water. RTD analysis demonstrated that the mean residence time of the ^{68}Ga was significantly longer than that of the fluorescent dye. The authors attributed this to the sorption of the Ga^{3+} species on to solids in the system. This indicates that the unmodified eluate from the generator is not a reliable water tracer.

The hydrated Ga(III) ion is only stable under acidic conditions and the insoluble species $Ga(OH)_3$ is formed as the pH increases. Between pH3 and 9.5, insoluble $Ga(OH)_3$ is the predominant species, whereas above pH9.6, the soluble tetrahydroxigallate anion, $Ga(OH)_4^-$, is formed.

$$
\begin{aligned}
&Ga(H_2O)_6^{3+} + H_2O \leftrightarrows Ga(OH)(H_2O)_5^{2+} + H_3O^+ \\
&Ga(H_2O)_6^{3+} + \underset{> 4M}{6HCl} \leftrightarrows GaCl_6^{3-} + 6H_3O^+
\end{aligned}
$$

5.1.1.2. Gallium-68-ligand complexes

As ^{68}Ga exists in trivalent cationic form, its compatibility with aqueous media without significant absorption on to surrounding materials can be achieved by producing a complex compound with the chelating agents such as DOTA (1,4,7,10-tetraazacyclododecane-1,4,7,10-tetraacetic acid), NOTA (1,4,7-triazacyclononane-1,4,7-triacetic acid) and EDTA (ethylene-diamine-tetraacetic acid).

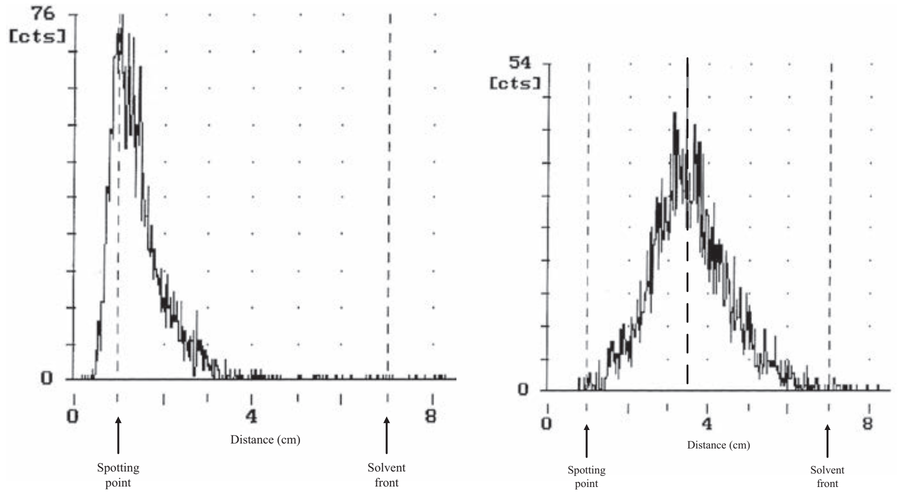
The group at the Korea Atomic Energy Research Institute (KAERI) used DOTA in order to stabilize the Ga^{3+} ion in the aqueous phase. The method of preparing Ga–DOTA is given below:

- Elution of Ga^{3+} ion from the $^{68}Ge/^{68}Ga$ generator;
- Addition of the DOTA solution (2 mg of DOTA in 0.2 mL of 50mM sodium acetate buffer, pH5.5);
- Regulation of pH by using sodium hydroxide and sodium acetate solution (~pH3);
- Heating the solution to ~90°C for 15 min;
- Cooling the solution by ice or cold water.

After the synthesis, thin layer chromatography (TLC) was performed on 0.8 cm × 7.5 cm TLC plates (Kieselgel 60 F-254, Merck) with 75% methanol. The activity distribution on the plates was measured by a TLC radioisotope scanner (TRACER-MASTER20, Berhold). Figure 71 shows the thin layer chromatograms of Ga^{3+} ion and the Ga–DOTA complex, respectively.

Figure 71(a) shows that the Ga ion remained the spotting point of the plate ($R_f = 0$). On the other hand, Fig. 71(b) shows that there is no peak at the origin and that the activity has moved to the centre of the plate ($R_f = 0.45$). This means that all the Ga ions were changed to the Ga–DOTA complex.

The group at Mainz University in Germany developed an efficient and fast on-line protocol based on a cation exchange purification method for on-line



(a) $^{68}Ga^{3+}$ ion ($R_f = 0$) (b) ^{68}Ga-DOTA ($R_f = 0.45$)

FIG. 71. Thin layer chromatograms of ^{68}*Ga ion and* ^{68}*Ga-DOTA.*

radiopharmaceutical syntheses. The key step in this procedure consists of the direct transfer of the initial 0.1N HCl ^{68}Ga eluate to a cation exchanger. Owing to high distribution coefficients, ^{68}Ga is quantitatively adsorbed on to only about 50 mg of the resin directly from the generator eluate.

Low volumes of a mixture of 1.0 mL of 80% acetone/0.15N HCl are applied to purify ^{68}Ga from Ge(IV), Ti(IV), Zn(II) and Fe(III). Admixture of 0.4 mL of a 98% acetone/0.05N HCl is sufficient to completely desorb ^{68}Ga from the resin and to transfer this purified fraction on-line to a labelling vial.

The processing takes 4 min, with overall ^{68}Ga recovery yields of 97 ± 2%. This efficient and simplified system for processing ^{68}Ga eluates results in volume reduction along with chemical and radiochemical purification. This procedure leads to almost complete removal of metallic impurities, including ^{68}Ge breakthrough, thus providing the purified ^{68}Ga in a form useful for direct labelling with acceptable pH, volume and purity.

For subsequent syntheses, the 400 µL fraction of the 80% acetone/0.15N HCl mixture containing the processed ^{68}Ga is transferred to 5 mL of water, containing appropriate amounts of, for example, DOTA–TOC (such as 20 µg). The resulting mixture is pH2.3. Owing to the high purity of the ^{68}Ga fraction and the fact that buffer systems (which may introduce a load of metallic impurities) are not needed, high labelling yields of >95% are achieved with about 5 min of heating to just below 100°C.

The group at the INCT in Poland compared the behaviour of the radiotracers ^{68}Ga–NOTA (Fig. 71) and ^{68}Ga–DOPA (obtained from radiopharmaceutical elution of the generator), with that of fluorescein (classical water tracer) in investigating the settler of a wastewater treatment plant. The RTD curves of these tracers showed very good similarity within the limits of experimental error. Thus, it was concluded that ^{68}Ga–NOTA and ^{68}Ga–DOPA are appropriate tracers of water and water-like systems. The major disadvantage of these two ligands is that they are more expensive than EDTA and DTPA, so their use to trace industrial water systems may not be justified, especially since ^{68}Ga–EDTA is a proven tracer for aqueous systems.

5.1.2. Organic ^{68}Ga tracers

The group in India has developed industrial nano-zirconia based ^{68}Ge/^{68}Ga generators, which can be used to obtain ^{68}Ga eluted in an acetylacetone medium. The ^{68}Ge/^{68}Ga solution containing ^{68}Ge at pH2 was percolated into the column at a flow rate of 0.25 mL/min. The generator column was thoroughly dried by continuous passage of hot dry air (~60°C) through it for 4 h to ensure the sorbent was free from moisture. After drying the column, it is mandatory to cool it for 30 min. followed by washing with 50 mL of acetylacetone. The generator is eluted with 4 mL of acetylacetone. The eluate can be used directly as an organic radiotracer in the oil and petrochemical industries.

The group at Mainz University in Germany uses purified ^{68}Ga adsorbed on a small cation exchange resin as a means of transferring ^{68}Ga into a labelling intermediate applicable for use with anhydrous or organic media. The initial aqueous generator eluate is transferred on-line on to a cation exchange resin. From this resin, ^{68}Ga is eluted with different acetone based, non-aqueous solvent systems.

More than 95% of the generator eluted ^{68}Ga is obtained from the cation exchange resin using a mixture of 600 μL of 98% acetone/2% acetylacetone providing ^{68}Ga(acac)$_3$ as the labelling agent. Water-insoluble macrocyclic polypyrrole derivatives were chosen as model compounds for a proof-of-principle labelling of lipophilic compounds with ^{68}Ga. Labelling of two different porphyrin derivatives was performed in chloroform in a focused microwave synthesis system in yields of up to 97% within 1 min. On the basis of these results, this novel procedure of providing ^{68}Ga(acac)$_3$ offers a wide scope of application for this labelling agent.

Figure 72 illustrates the prototype of such a ligand, which is difficult to dissolve in an aqueous solution. However, even DOTA or NOTA based

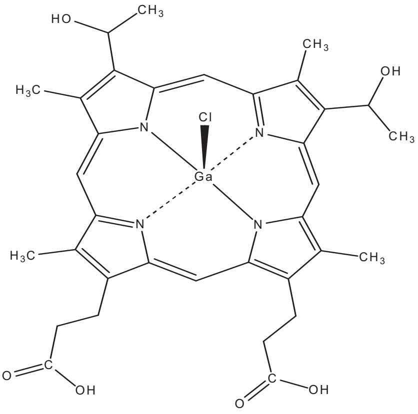


FIG. 72. Gallium-68-hematoporphyrin.

macrocyclic ligands may be applied where those ethyl structures have been conjugated to lipohilic moieties such as aliphatic or aromatic carbohydrates.

5.1.3. Solid ^{68}Ga tracers

The efficient operation of the fluidized catalytic cracking unit (FCCU) is critical to the profitability of a refinery, and radioactive tracers are widely used for troubleshooting operational problems and for process optimization. The ^{68}Ge/^{68}Ga generator can be used to label the FCCU catalyst. If appropriate radioisotope handling facilities are available at the work site, catalyst powder can be soaked in the eluate from the generator and, after drying, injected into the FCCU. Where this is not possible, the eluate from the generator can be injected directly into the FCCU riser as a pulse, to provide in situ labelling. Both of the above approaches have been used successfully to study the behaviour of the catalyst in FCCU risers, primarily to measure its velocity and to study its distribution over the riser cross-section.

An investigation into the sorption of Ga ions eluted from the ^{68}Ge/^{68}Ga generator on to bentonite has been carried out. After about 20 min, the firm and complete sorption of the ^{68}Ga was observed. Rinsing the bentonite showed that approximately 98% of the activity was associated with the solid phase. It is, therefore, feasible that labelled bentonite particles could be used as a tracer of the solid phase.

5.2. RADIOTRACERS FROM THE ^{99}Mo/^{99m}Tc GENERATOR

The ^{99}Mo/^{99m}Tc generator is used extensively in nuclear medicine and is widely available worldwide. For this reason, the somewhat short half-life of ^{99}Mo is not too severe a disadvantage. The 6 h half-life of the daughter ^{99m}Tc is appropriate for a wide range of studies in industry and the environment. Radiotracers from the ^{99}Mo/^{99m}Tc generator have the potential for application in a variety of industrial situations, with tailor-made ^{99m}Tc tracers suitable for aqueous, organic or solid media.

The gamma ray (140 keV) emitted by ^{99m}Tc has a half-thickness of 5 mm of steel and, thus, allows process investigation inside a vessel of wall thickness up to about 20 mm. It is not appropriate for use on high pressure plants owing to the greater wall thickness of the process vessels.

5.2.1. Aqueous ^{99m}Tc tracers

5.2.1.1. Eluate from a ^{99}Mo/^{99m}Tc generator

The eluate of the ^{99}Mo/^{99m}Tc generator is sodium pertechnetate (Na^{99m}TcO$_4$), which is the chemical form preferably used by radiotracer teams to label aqueous systems in which the redox potential of the system is not enough for the chemical reduction of the species TcO^{4-}. In such circumstances, experience has shown that Na^{99m}TcO$_4$ is a good tracer of the water phase, and can be used for:

- Water tracing in wastewater treatment plants for RTD measurement;
- Water tracing in surface bodies (rivers, sea) for the study of the dispersion of effluents and for dispersion coefficient measurements;
- Water tracing for water infiltration coefficient measurements in soil;
- Determination of water channelling in petroleum industry boreholes.

Because of the low gamma energy of ^{99m}Tc gamma ray, detectors submerged in large scale water systems sense only radiotracer in the immediate vicinity, so that localized flow patterns can be observed.

Some authors have reported adsorption of technetium during studies on units of wastewater treatment plants or in very reducing environments. This may be because, in such environments, the TcO^{4-} ion can be metabolized by microorganisms or can be reduced to TcO_2, $TcO(OH)_2$ or Tc^{2+}. In those forms, it can produce radiocolloids and be adsorbed. In such a case, the use of the complexes ^{99m}Tc–EDTA or ^{99m}Tc–DTPA is recommended.

5.2.1.2. *Aquacomplex ion* $[^{99m}Tc(CO)_3(H_2O)_3]^+$

Although the RTD of this ion is similar to that obtained with $K^{82}Br$ when tracing wastewater in a settler, the use of this compound is not recommended because its chemical stability is only about 3 h and its preparation is rather complicated.

5.2.2. Organic ^{99m}Tc tracers

5.2.2.1. *Non-aqueous* ^{99}*Mo/*^{99m}*Tc generator*

The industrial ^{99}Mo/^{99m}Tc generator developed by India with the aid of chromatographic alumina can be used to obtain ^{99m}Tc in a non-aqueous medium (TBP). This generator system is based on the selective extraction of ^{99m}Tc, with appreciable yield and with high radiochemical and radionuclidic purities.

It has also been reported that nearly 85% of the ^{99m}Tc retained in the column is eluted in a volume of 5 mL. The performance of the generator remains consistent over a period of two weeks, which is the normal shelf life of a ^{99}Mo/^{99m}Tc generator. This approach provides ^{99m}Tc at industrial sites in a medium soluble in hydrocarbon solvents, for radiotracer applications. The physical properties of TBP (melting point: –76°C; boiling point: 289°C; flash point: 146°C; water solubility: 6 g/L) and its miscibility with most hydrocarbons make it an attractive tracer.

It has also been reported that MEK is used as an eluant in the ^{99}Mo/^{99m}Tc generator, which is produced by the Indian Radiation and Isotopic Technology Association. Yields greater than 95% have been achieved by extracting the $^{99m}TcO_4^-$ with MEK. However, under conditions of high temperature or salinity, such as those usually found in multiphase systems in the petroleum industry, the thermal and chemical stabilities of this radiotracer have still to be proven.

5.2.2.2. *Solvent extraction method*

The extraction of pertechnetate ($^{99m}TcO_4^-$) with organic solvents is employed for the preparation of organic ^{99m}Tc tracers. TBP and tri-n-octylamine are two of the most used extractants, either on their own or mixed with other solvents.

5.2.3. Solid ^{99m}Tc tracers

For solid phase labelling (particle tracers), the generator eluate ($^{99m}TcO_4^-$) can be reduced to TcO_2 using an aqueous solution of $SnCl_2$. The reduced form can be used to label solid particles, which can be used for radiotracer investigations, such as:

- Sludge tracing in wastewater treatment plants for RTD measurements;
- Mud tracing in surface water (rivers, sea) for particle dispersion in effluent studies and dispersion coefficient measurements;
- Mud tracing in the sea for dredging and dumping studies (dumping efficiency, advection, dilution, dispersion coefficients, etc.).

The eluate ($^{99m}TcO_4^-$) and the particles (always in an aqueous medium) are mixed with a reducing solution of $SnCl_2$ (concentration 20 g/L) so as to achieve a concentration (i.e. the ratio of the mass or particles to the mass of $SnCl_2$) of approximately 3.

After 10–15 min stirring, the particles are labelled with better than 90% efficiency. The stability of the labelled particles must be tested under the conditions of the system to be investigated, especially for high temperature applications. An example of this procedure applied to sand labelling is as follows:

- First dry the sand and take a 200 g sample.
- Take ~20–30 g of $SnCl_2$ and dissolve it in 20 mL of water.
- Add 1 mCi of ^{99m}Tc together with 20 mL of water to the sand and stir until they are mixed homogeneously.
- Then, mix in the $SnCl_2$ solution.
- Wash the sand three times.
- Add sufficient water to make it easy to inject into the system inlet.

5.3. RADIOTRACERS FROM THE $^{113}Sn/^{113m}In$ GENERATOR

The radionuclide ^{113m}In has the potential for application in a variety of industrial situations using tailor-made ^{113m}In tracers as probes for aqueous, organic or solid material. A wide variety of ^{113m}In labelled tracers can be synthesized using information obtained from the radiopharmaceutical literature.

Dual phase systems with solid/liquid are often encountered in various industrial process units that are designed to separate or purify the components. Understanding the dynamic behaviour of each phase is very important in order to investigate the performance of the system. Indium-113m compounds can be used for the selective labelling of both phases.

The gamma energy (391 keV) is sufficiently high for ^{113m}In tracers to be used for on-line measurements in most industrial plant. Indium is extracted from the ^{113}Sn/^{113m}In generator using 0.1N HCl solution to obtain the ^{113m}In^{3+} ion in chloride solution.

5.3.1. Aqueous ^{113m}In tracers

The ^{113m}In^{3+} ion can be complexed with multidentate ligands such as EDTA and DTPA for aqueous tracing in dual phase systems. For many years, the radiotracer pulse velocity technique using the ^{113m}In–EDTA complex has been applied routinely to the measurement of process flow rates on industrial plants. Comparison of the results obtained with the ^{113m}In–EDTA complex with those obtained using K^{82}Br solution (which is known to be a good tracer for water), has shown that the results are identical, within the limits of accuracy of the measurements (±2%).

5.3.2. Organic ^{113m}In tracers

The eluate of the ^{113}Sn/^{113m}In generator was mixed with methanol to be used as an organic tracer for the measurement of velocity and volume flow rate of petrol flowing through a 10 cm diameter pipe at ambient temperature using the pulse velocity technique. The Reynolds number for this flow was 1.7×10^5, indicating that it is fully turbulent. The measured results were in excellent agreement with the results obtained using the well-known organic radiotracer, ^{82}Br labelled paradibromobenzene dissolved in toluene.

Similar studies were carried out on fuel oil flow in a 20 cm diameter pipeline and, again, there was good agreement between the results obtained with the generator eluate and those obtained with the paradibromobenzene tracer.

It is, perhaps, going too far to suggest that on the basis of these two measurements, the eluate from the generator is a good tracer for any fully turbulent flow organic, but these results are encouraging. Further to this, since the chemistry of gallium is very similar to that of indium, it is reasonable to suppose that the eluate from the ^{68}Ge/^{68}Ga would also be a satisfactory tracer for pulse velocity measurements of turbulent flow in organic liquids.

5.3.3. Solid ^{113m}In tracers

Because ^{113m}In is eluted in cationic form, the solid phase can be labelled by mixing the ^{113m}In cation with solid particles, such as sand. In doing this, the pH of the generator eluate should be adjusted to a value where optimum sorption of In^{3+} will occur. For example, ^{113m}InCl$_3$ is a good tracer of sediments. Experience has shown that ^{113m}InCl$_3$ eluted from an indium generator can be used for:

- Sludge tracing in wastewater treatment plants for RTD measurements;
- Mud tracing for dumping site studies;
- Catalyst tracing for RTD measurements.

Indium is easily fixed on fine solid particles by simple adsorption of the hydrolyzed ion without any pretreatment. Although the extraction efficiency decreases from 80% with 0.1N HCl to 70% with 0.08N HCl, it can be the preferred method to prevent mud flocculation, which is low, under 0.1M HCl per kilogram of mud or sludge. The fixation efficiency is always between 95 and 98%.

The natural adsorption capacity of sand particles can vary depending upon the sand origin and grain size distribution, but it is always quite low, typically less than 12%. Thus, it may be necessary to carry out a pretreatment of the sand particles to increase the adsorption efficiency. The pretreatment can be performed as follows:

- Step 1: Treatment with nitric acid to clean the sand of organic matter and impurities, and to create microholes at the surface of the grains.
- Step 2: Treatment with NaOH for the formation of a superficial silica gel is helpful for the adsorption, retention and resistance of the radionuclide deposit.
- Step 3: Tin chloride for its reducing properties.
- Step 4: Silver coating to exchange silver with tracer.

Reactions that are supposed to occur are reduction of silver nitrate.

$$Sn^{2+} + 2Ag^{+} \rightarrow Sn^{4+} + 2Ag^{0}$$

The silver coated sand is stored under distilled water to avoid any oxidation which would transform Ag^0 in AgO. The In^{3+} in the chloride solution is added to the silver coated sand and the following reduction and exchange reactions occur:

- Reduction of In^{3+} by remaining tin: $Sn^{2+} + 2In^{3+} \rightarrow 3Sn^{4+} + 2In^{0}$
- Reduction of In^{3+} by silver: $3Ag^{0} + In^{3+} \rightarrow In^{0} + 3Ag^{+}$

The natural sand labelling method has two main disadvantages: surface labelling and risk of erosion during the transport of the grains. However, following the above process, the fixation efficiency is more than 80% and the resistance to mechanical erosion is quite sufficient.

The $^{113}Sn/^{113m}In$ generator can be used to label the FCCU catalyst. If appropriate radioisotope handling facilities are available at the work site, the catalyst powder can be soaked in the eluate from the generator and, after drying, injected into the FCCU. Where this is not possible, the eluate from the generator can be injected directly into the FCCU riser as a pulse, to provide in situ labelling. Both of the above approaches have been used successfully to study the behaviour of the catalyst in FCCU risers, primarily to measure its velocity and to study its distribution over the riser cross-section.

5.4. RADIOTRACERS FROM THE ^{137}Cs/^{137m}Ba GENERATOR

Commercially available ^{137}Cs/^{137m}Ba generators are of very low activity (typically 370 kBq). They are used in educational establishments for demonstration purposes only. However, the parent radioisotope, ^{137}Cs, may be procured from commercial suppliers and can be used for the construction of generators to meet the needs of individual users.

The main drawback of the ^{137m}Ba as radiotracer is its short half-life ($T_{½}$ = 2.5 min). Even so, it is suitable for many industrial tracer applications. The short half-life of the ^{137m}Ba ensures that there is no possibility of buildup of radioactivity in the system being studied. The half-life of the parent isotope (30 a) facilitates long term storage and the gamma ray energy of the daughter (660 keV) is high enough for most industrial applications.

The long lived parent nuclide (^{137}Cs) provides a renewable source of radiotracer over a period of years. This opens up the possibility of using a ^{137}Cs/^{137m}Ba generator as the source of tracer in permanently installed plant instrumentation. As described in Section 6.1.3, a flowmeter based on this generator operated successfully on the effluent outfall of a major chemical complex to provide semi-continuous monitoring of the effluent flow rate.

5.4.1. Aqueous ^{137m}Ba tracers

There may, sometimes, be a need for chemical treatment/complexation, but in most cases, especially in water and water-like media, the eluted tracer ($^{137m}BaCl_2$) is used directly after milking from the generator. Some typical applications of this generator are:

- Flow rate measurement for flowmeter calibration.
- Mixing studies of reactors which operate with Newtonian low viscosity fluids under turbulent regimes. The homogenization time in the vessel should be comparable with the half-life of ^{137m}Ba.

The $^{137m}BaCl_2$ eluted from the generator was used to characterize the mixing efficiency of a pilot crystallizer. The results confirmed the suitability of the tracer. A further study confirmed that the $^{137m}BaCl_2$ tracer was appropriate for use in RTD measurements in a ceramic membrane module.

Water–oil distribution experiments have been carried out using barium complexes as tracers undert ambient conditions. Aqueous phase Ba complexes of EDTA, DTPA and 18C6 (ethylene oxide cyclic hexamer). The results showed that no oil phase partitioning was observed with Ba–EDTA, Ba–DTPA or Ba–18C6. The degree of sorption to reservoir rock material was acceptably low. Therefore, these complexes may prove applicable to the study of water transport in general, as well as in commingled multiphase fluid handling systems such as separators and scrubbers in the petroleum industry.

5.4.2. Organic ^{137m}Ba tracers

Barium complexes with DC18C6 and HDNNS show promise as tracers for organic phases. They are generally applicable to single phase flow of organic liquids. It is also probable that they could be applied to commingled multiphase flow in oil installations, but further work is needed to confirm this.

5.4.3. Solid ^{137m}Ba tracers

There is no obvious reason why the ^{137}Cs/^{137m}Ba generator should not be used for the in situ labelling of an FCCU catalyst by injecting the eluate from the generator directly into the FCCU riser.

The half-life of the ^{137m}Ba (2.55 min) is certainly sufficiently long for riser studies, since the typical residence time of catalyst in the riser is generally no more than 10 s. The 660 keV gamma rays are sufficiently energetic to penetrate the thick walls and refractory lining of the vessel.

5.5. RADIOTRACERS FROM THE ^{44}Ti/^{44}Sc GENERATOR

The ^{44}Ti/^{44}Sc radionuclide generators may be of interest for medical imaging as well as for industrial applications. The 3.92 h half-life of ^{44}Sc and the high positron branching of 94% may stimulate the application of ^{44}Sc labelled PET radiopharmaceuticals.

In addition, the high energy photon emission of 1.157 MeV appears to be ideal for industrial purposes. However, both ^{44}Ti production and ^{44}Ti/^{44}Sc generator design present challenges for basic radiochemistry.

The ^{44}Ti/^{44}Sc radionuclide generators have been investigated in a few cases only over recent decades. The general concept has been to elute ^{44}Sc from ion exchange based generators. The ^{44}Sc is finally obtained as complex of weak ligands such as halogenides (e.g. chloride) and/or oxalate.

For liquid systems, however, ^{44}Sc should be transformed into species stable in the aqueous or non-aqueous media (i.e. not undergoing hydrolysis in neutral pH or formation of colloids or unspecific surface absorption). Scandium-44 starts to hydrolyze at about pH5 and is already forming hydroxides. These tend to form colloids or pseudo-colloids with unknown physicochemical properties. Therefore, it is essential to transform ^{44}Sc into species that are chemically stable under harsh environmental conditions. In the industrial context, ^{44}Sc could possibly be used for the synthesis of particles for solids tracing or, as ligand complexes, with very high thermodynamic and kinetic stabilities for liquid tracing in harsh industrial environments.

6. CASE STUDIES

Case studies of industrial applications carried out using tracers from radionuclide generators are presented. They have been selected to illustrate important features and advantages of radionuclide generators as used in the industrial context.

6.1. AQUEOUS PHASE TRACING

In most cases, the liquids used for the elution of radionuclide generators are aqueous in nature. For this reason, it is often possible to use the eluants directly, or with minimal modification, as tracers for aqueous systems. In other cases, it has been found necessary to incorporate the radionuclide into a chemical complex to render it appropriate for use as a tracer. The case histories described below are illustrative of both of the above scenarios.

6.1.1. Measuring the water intake to a power station using the ^{99}Mo/^{99m}Tc generator

This case study describes how a commercially available ^{99}Mo/^{99m}Tc generator was used to measure extremely large flows of water through closed conduits.

A coal fired power station received its water supply from a nearby lake. Water was pumped from the lake to the power station via six rising mains, with a total water intake of approximately 120 m^3/s. It was very important that the six water pumps (one on each riser) were properly balanced so that the whole system was operating at optimum efficiency. Thus, to achieve optimal operation, it was necessary to measure accurately the water flow up each riser. This had proved difficult using conventional measurement methods: each main was 2.7 m in diameter, and there were concerns about the accuracy of the installed flow instrumentation.

It was therefore decided to use a radiotracer pulse velocity technique to measure the flow rate up each riser for comparison with the power consumption of the corresponding water pump.

As the power station was located at a significant distance from the nearest nuclear reactor and because a series of flow rate measurements were required over a period of time, it was decided to use a radioisotope generator to provide the tracer. A ^{99}Mo/^{99m}Tc generator of activity 40 GBq was selected, and this was eluted appropriately to provide the 2.6 GBq tracer pulses required for each measurement.

Measurements were carried out using the standard pulse velocity methodology. The tracer was injected as a sharp pulse into the pump suction, as shown in Fig. 73.

A pair of radiation detectors mounted at a measured separation on the riser measured the passage of the tracer pulse so that the flow velocity could be measured. Since the internal cross-sectional area of the main was known, the volume flow rate could be calculated. The accuracy of each measurement was ±1% at 95% confidence limits.

Significant differences in pump efficiency were found, with delivery flow rates varying between 16 and 21 m^3/s.

The measurements are perhaps noteworthy as being among the largest liquid flows that have ever been measured using a pulse velocity technique in a closed conduit.

Owing to the large diameter of the mains, and because detectors mounted externally would only sense radiation that originated within approximately

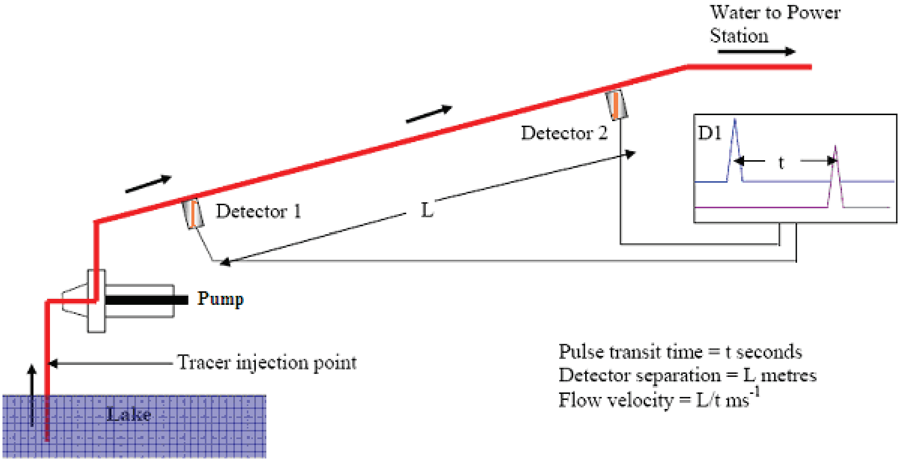


FIG. 73. Water intake flow rate measurement.

150 mm of the pipe wall, there were concerns initially that the flow measurements might be unrepresentative of the true velocity profile across the main.

This problem was resolved by performing a separate test in which, at each detector station, blanked-off tubes were inserted through flanges as far as the centre of the riser.

Velocity measurements were made with detectors positioned at the centreline, at the half-radius position and at the surface of the riser. No significant differences in velocity were observed, thus confirming that the measurements obtained with the detectors mounted external to the risers were reliable.

Since cooling water leaving the power station was returned to the lake, and this was an environmentally sensitive location, it was important to avoid any buildup of the radioactive tracer. Neglecting the very large dilution of tracer in the volume of lake water, which in itself was sufficient to ensure that the tracer concentration was at a negligible level, the 6.02 h half-life of the ^{99m}Tc provided an additional measure of reassurance and facilitated official approval for the measurement.

6.1.2. Studies of the aqueous phase in multiphase flow in a hydrate loop using a ^{137}Cs/^{137m}Ba generator

A 740 MBq ^{137}Cs/^{137m}Ba generator supplied by Atom Hightech Co. Ltd (China) has been used to study the velocity of the aqueous phase inside a hydrate flow loop using the pulse velocity technique described in Section 6.1.1.

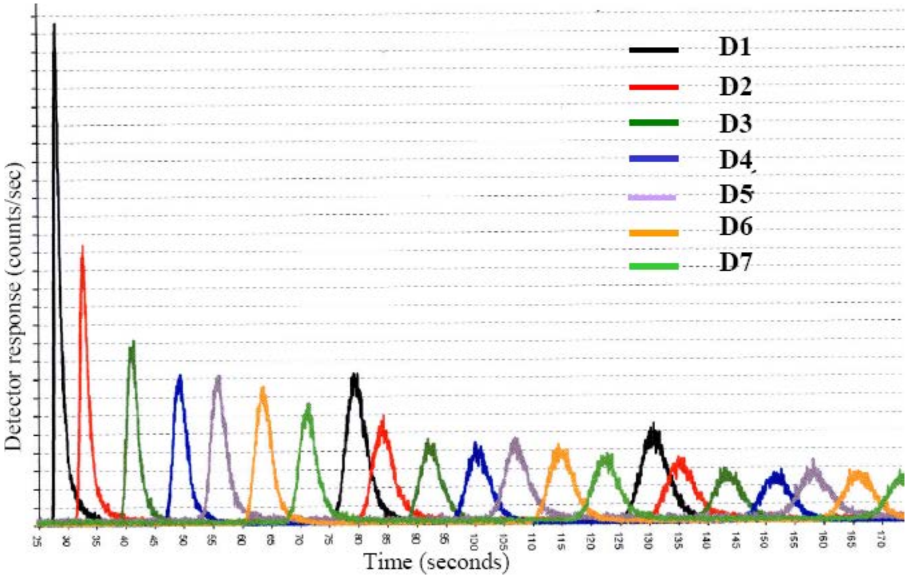


FIG. 74. Aqueous phase velocity measurements in a hydrate flow loop.

Understanding hydrate dynamics in oil production systems is essential for flow assurance in deepwater production fields. In this project, which is part of a much larger study, the velocity in the loop was measured using the eluate from the generator as tracer. Eight detectors were deployed, and from their responses, the velocities in different parts of the loop and under different sets of flow conditions were determined. The detector responses obtained from a single injection of approximately 7 MBq of ^{137m}Ba are shown in Fig. 74.

The study required many successive injections of the tracer, and it was found that the ^{137m}Ba eluate was ideal for this purpose. Its rapid decay (half-life 2.5 min) ensured that background activity did not buildup in the loop, and repeat measurements every 10 min were thereby facilitated. Checks carried out on water drained from the loop confirmed that residual contamination was negligible.

6.1.3. Semi-continuous measurement of effluent flow rate in open channels using a ^{137}Cs/^{137m}Ba generator

This case study describes how a ^{137}Cs/^{137m}Ba generator was used to provide the tracer required by an instrument permanently installed on the effluent outfall of a major chemical complex to provide semi-continuous monitoring of the effluent flow rate. It is perhaps the only example of the use of the radioactive tracer technique automatically, in on-line measurement and control.

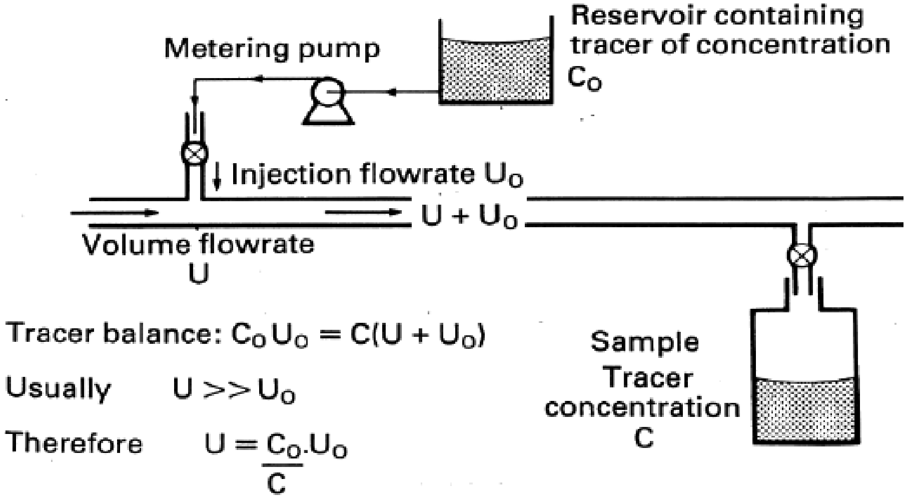


FIG. 75. Principles of the constant rate dilution method of flow rate measurement.

As the requirement for improved environmental management becomes ever more important, there is an increasing need to carefully control all effluent discharges from industrial premises. Before the discharges can be controlled, they must first be monitored. Measurement of liquid effluent flow rate can be particularly difficult as the flows are very often in open conduits of irregular cross-section, such as drainage ditches. The traditional approach to flow measurement is to make use of flumes or weirs and to relate the water head to the flow rate. Such systems are notoriously inaccurate, as they are severely affected by silt buildup, and frequent calibration is needed.

One approach is to use a constant rate radioisotope dilution method to provide measurement of the volume flow rate in the effluent channel for comparison with the readings of the installed meter. The principles of this method are shown in Fig. 75.

This is arguably the most commonly used method of effluent flow rate measurement. It is relatively simple to apply, and provides a direct measurement of the volume flow rate of the effluent, independent of any assumptions about the shape of the conduit, depth or velocity of the liquid.

It is also possible to apply a dilution measurement using pulse injection rather than constant rate injection. This ‘total count’ method is discussed extensively in the literature. Essentially, it can be shown that the flow, Q, in the stream is related to the total activity, A, injected and the total count, N, recorded by a downstream detector by:

$$Q = AF/N \tag{23}$$

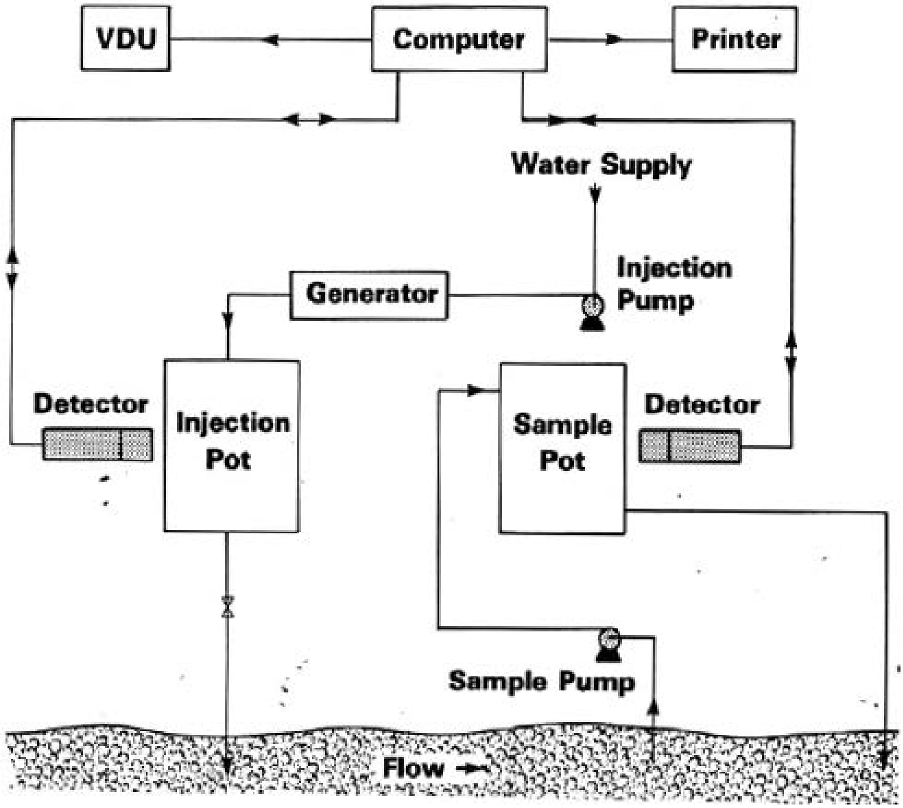


FIG. 76. Installed effluent flow rate meter based on a ^{137}*Cs/*^{137m}*Ba generator.*

where *F* is a constant depending on the efficiency of the detector and the counting geometry. *F* can be determined by calibration.

This approach has advantages over the constant rate injection technique in that it requires a smaller activity of radiotracer. In addition, once the constant *F* has been determined, repeat measurements can be carried out simply and reliably.

The demand for spot checks of effluent flow rates in the main outfall from a major chemical complex increased to a point where a continuous — or at least, semi-continuous — measurement was needed. To satisfy this requirement, an installed flowmeter, based upon the radioisotope 'total count' method was developed. This instrument is shown schematically in Fig. 76.

A water pump and a solenoid valve, operated by relays in the control unit, are used to control the injection of radioactive material from the generator. The water pump feeds clean water to the radioisotope generator, and the eluted ^{137m}Ba is then pumped into the coil at the base of the counting chamber where its radioactivity is measured.

The activity is then pumped into the effluent stream. A sample pump, well downstream of the tracer injection point extracts a sample of effluent over a time interval sufficient to span the transit of the injected tracer pulse, and this sample is also subjected to radioassay in the counting chamber.

The radioactivity counts for the injected water and sampled effluent are fed to a microprocessor that computes the flow rate in the effluent stream using the 'total count' algorithm. The efficiency factor, *F*, of the system is derived from on-line calibration data.

Devices of this type operated successfully for several years on industrial complexes in the UK. Periodic spot checks carried out using a constant rate dilution method indicated that the accuracy of the measurement was ±2.5% at a flow rate of approximately 5000 m^3/h. The ability to install this device without the need for expensive civil engineering — as would be required for the installation of a weir — is clearly a very distinct advantage.

The^{137}Cs/^{137m}Ba generator is crucial, both to the operation of the device and to obtaining necessary safety approvals from the relevant legislative authorities. The long lived ^{137}Cs parent provides a renewable source of radiotracer over a period of years, while the 2.5 min half-life of the daughter ensures that there is no possibility of buildup of radioactivity in natural water systems.

6.1.4. RTD studies on a hot water sterilizer using a ^{99}Mo/^{99m}Tc generator

A manufacturer of domestic water heaters was developing a new type of heater for hospitals and surgeries, the purpose of which was to give a supply of sterilized water, on demand. Calculations had shown that if ordinary tap water were contacted with superheated water for 2 min, all known pathogens would be destroyed. It was necessary to conduct tests on the prototype heater to ensure that it complied with this specification. The most direct way of doing this was to measure the mean residence time and RTD of the water flowing through the heater.

The factory producing the heater was in the middle of an urban industrial site, with a drainage system shared with surrounding public housing. Clearly, disposal of water contaminated with radioactive tracer was a matter to which consideration had to be given in planning the measurements. Two different radioisotope generators were available: ^{99}Mo/^{99m}Tc and ^{137}Cs/^{137m}Ba.

The ^{137}Cs/^{137m}Ba generator was considered first, on the grounds that the short half-life of the ^{137m}Ba (2.55 min) would ensure that the small amount of radioactive material entering the drain would have decayed to a negligible level before it passed out of the site boundary. However, the prototype heater was of

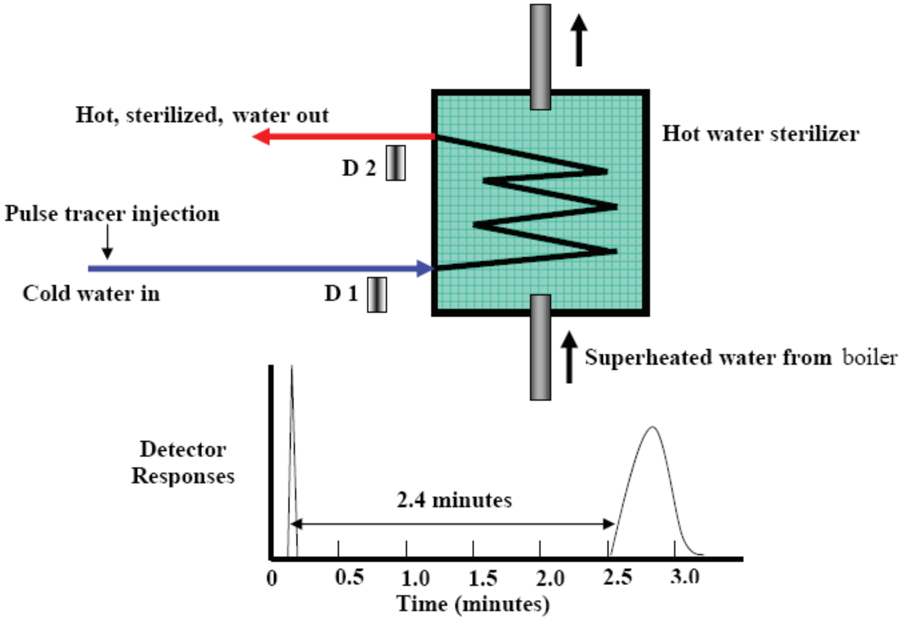


FIG. 77. Residence time distribution studies to prove the design of a hot water sterilizer using a ^{99}Mo/^{99m}Tc generator.

small volume (approximately 0.5 m^3), and the associated pipe work was correspondingly small. It was found that the heavy shielding needed to collimate the radiation detectors and to screen out extraneous radiation from the tracer flowing round the system was too cumbersome. It was therefore decided to use the ^{99}Mo/^{99m}Tc generator. The low energy 140 keV gamma rays could be easily shielded and collimated using thin lead sheet, which made positioning of the radiation detectors on the compact pipe work relatively easy. The contaminated water from the system was simply collected in a container and stored securely for 3 d, after which time, the activity was reduced to a negligible level by radioactive decay. It was then disposed of to drain.

The measurement set-up is shown in Fig. 77. The results showed that though there was some back mixing of the water, the flow through the heater was essentially plug flow in character. The shortest residence time of any element of the fluid was 2.4 min, which was deemed to be acceptable by the heater designers.

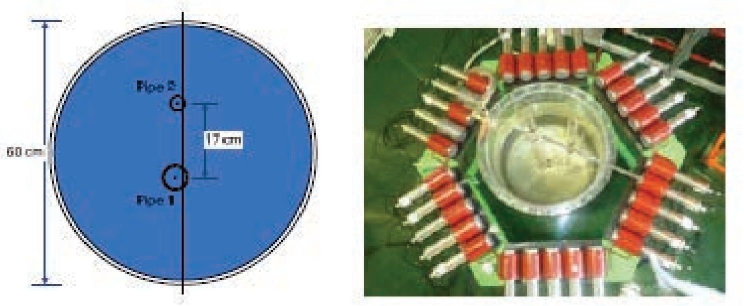


FIG. 78. The In-SPECT system.

6.1.5. Industrial emission tomography experiments with ^{68}Ga and ^{137m}Ba

The eluates from two radioisotope generators, ^{137}Cs/^{137m}Ba and ^{68}Ge/^{68}Ga, were used as tracers in industrial emission tomographic measurement experiments to visualize the spatial distribution of radioisotope as a function of time and hence to investigate the flow dynamics of a laboratory scale process vessel.

The industrial single photon emission computed tomography (In–SPECT) system consists of 30 50 mm × 50 mm NaI(Tl) detectors connected to 30 single channel analysers.

Initially, the equipment was tested under static conditions (Fig. 78). Two cylindrical pipes with different diameter were installed inside a water filled vessel of diameter 60 cm. The pipes were separated by 17 cm from each other and filled with various combinations of the radioisotopes eluted from the generators, as shown in Table 18, which also lists the data accumulation mode of the detectors.

TABLE 18. EXPERIMENTAL DATA FOR TESTS UNDER STATIC CONDITIONS

No.	Nuclide (keV)	Counting mode	Specific activity of the radioisotope (mCi/L)	
			Pipe 1	Pipe 2
1	^{68}Ga (511)	Gross	2.2	—
2	^{68}Ga (511)	Gross	1.1	3.0
3	^{137m}Cs (662)	Gross	6.2	10.4
4	^{137m}Cs (662)	Photopeak	4.0	11.0

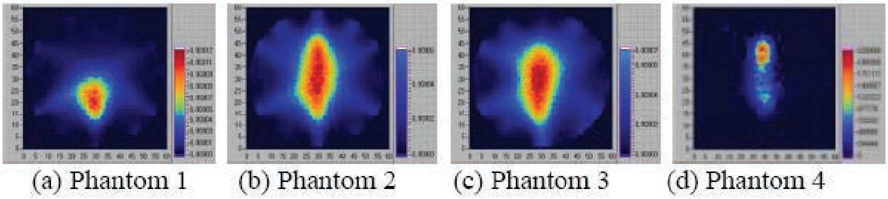

(a) Phantom 1 (b) Phantom 2 (c) Phantom 3 (d) Phantom 4

FIG. 79. Reconstructed images from the measurements in the static condition.

‘Gross’ means that the detector counts all radiation energies above a threshold set at a low energy level. In this mode, not only the primary gamma rays, but also Compton scattered radiation is detected. ‘Photopeak’ means that only energies in a narrow band about the photopeak are detected, thus discriminating against the Compton scattered radiation.

The expectation maximization algorithm was applied to the image reconstruction. Figure 79 shows the reconstructed images from the four cases:

(a) Pipe 1 is filled with ^{68}Ga;
(b) Both pipes are filled with ^{68}Ga;
(c) Both pipes are filled with ^{137m}Cs;
(d) Both pipes are filled with ^{137m}Cs and only the photopeak was recorded.

Image (a) provides only a rough indication of the location of the pipe. The low quality of the image is certainly due to the small number of detectors. In images (b) and (c), the images corresponding to the two pipes are not resolved because of interference from scattered radiation. In (d), the two pipes are clearly resolved, and it is also possible to observe the difference in the specific activities of the radioisotopes they contain. The above measurements clearly show that the performance of the In–SPECT system is significantly impaired by the scattered photons. However, counting on the photopeak implies that far fewer gamma ray photons will be detected than when gross counting is employed, and this clearly impacts on the counting statistics. For this reason, it is preferable to use a radioisotope with high gamma ray energy to minimize signal loss from attenuation in the walls of the vessel.

The experimental arrangement for the measurements under dynamic conditions is shown in Fig. 80. During the experiment, the flow rate of water was kept constant at 8.7 L/min, and 2.2 mCi of ^{68}Ga and 9.7 mCi of ^{137m}Ba were injected into the vessel inlet.

Figures 81 and 82 show the pattern of tracer movement for ^{68}Ga and ^{137m}Ba, respectively. The patterns observed with the two tracers are of similar form, but

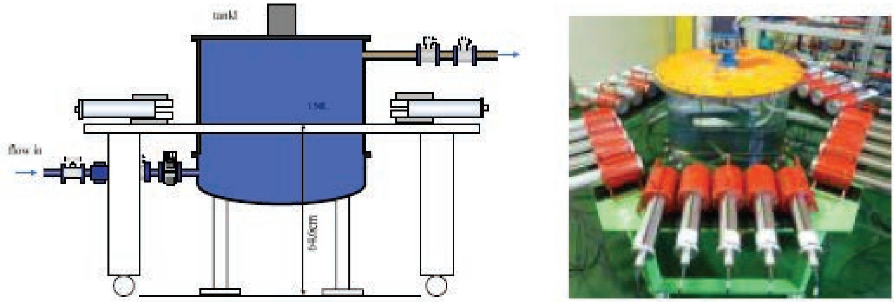
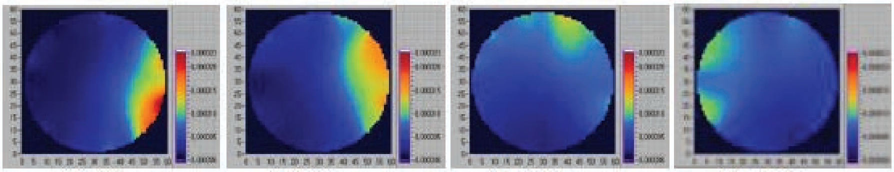
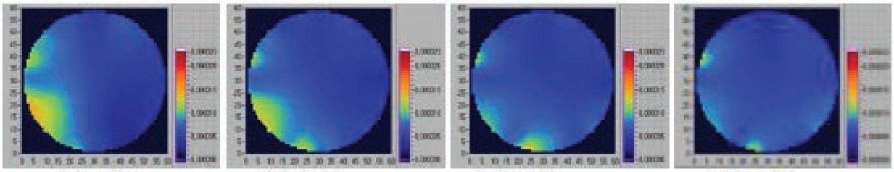


FIG. 80. Experimental set-up for the measurements under dynamic flow conditions.

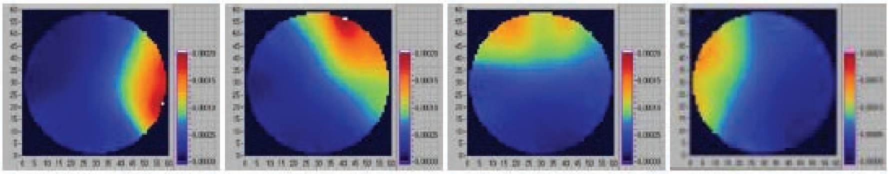


(a) 80sec (b) 90sec (c) 100sec (d) 110sec

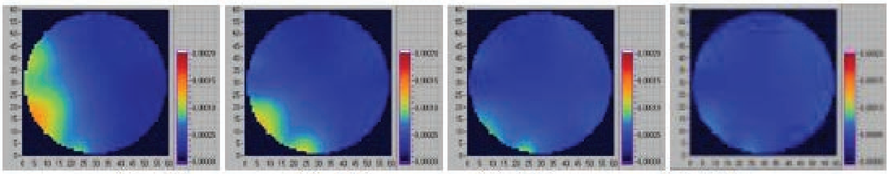


(e) 120sec (f) 130sec (g) 140sec (h) 150sec

FIG. 81. The spatial distribution of ^{68}Ga as a function of time.



(a) 80sec (b) 90sec (c) 100sec (d) 110sec



(e) 120sec (f) 130sec (g) 140sec (h) 150sec

FIG. 82. The spatial distribution of ^{137m}Ba as a function of time.

those obtained with ^{137m}Ba are clearer because of its higher gamma energy and the resulting improvement in counting statistics.

6.1.6. Investigation of the performance of a sand filter on a wastewater treatment plant

The ^{113m}In eluate from a ^{113}Sn/^{113m}In generator was used to investigate the performance of a sand filter on a wastewater treatment plant. The filter contained beds of sand and gravel. The objective of the study was to determine the wastewater flow model through the system and hence measure the volume occupied by the sand and gravel.

The ^{113m}In in the eluate from the generator is of the form of positively charged ions that tend to be absorbed on to particles. For this reason, the untreated eluate is a good tracer for flocks and suspended particles. However, by chelating with EDTA, ^{113m}In forms a complex compound (^{113m}In–EDTA), which is very stable and can be used as a water tracer. This was used as the water tracer to investigate the flow of wastewater through the sand filter system.

As shown in Fig. 83, the tracer was injected just before the pump upstream of the filter. Two radiation detectors (D-1 and D-6) were placed between the pump and the sand filter at a measured separation, so that the flow rate of the wastewater could be measured using the pulse velocity technique. The RTD of the wastewater in the filter was measured by detector D-2, placed on the outlet line of the sand filter. This is shown in Fig. 84. A two-stirred-tanks-in-series model provided a good fit to the experimental data. The experimental mean residence time was found to be 747 s.

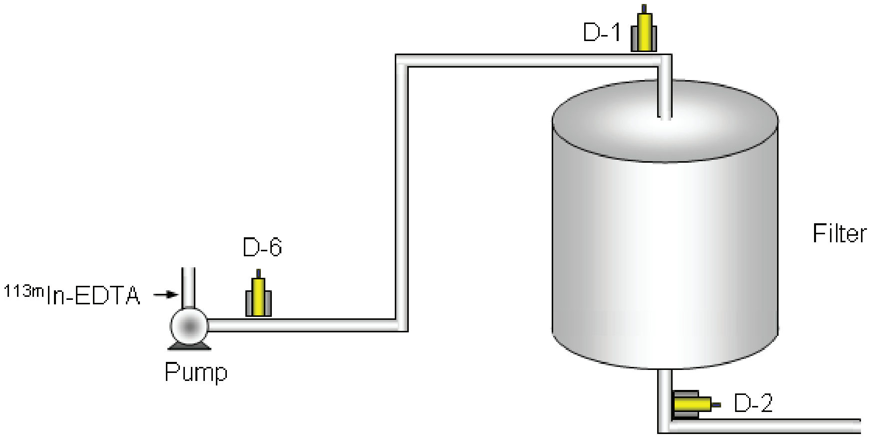
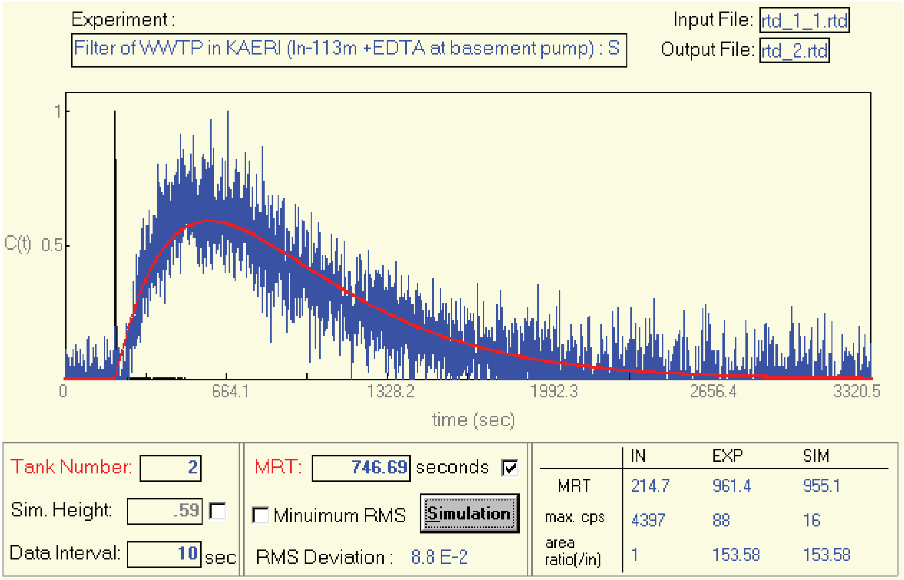


FIG. 83. Experimental arrangement for sand filter measurements.



Experiment : Filter of WWTP in KAERI (In-113m +EDTA at basement pump) : S

Input File: rtd_1_1.rtd

Output File: rtd_2.rtd

Tank Number: 2

Sim. Height: .59 ☐

Data Interval: 10 sec

MRT: 746.69 seconds ☑

☐ Minuimum RMS

RMS Deviation : 8.8 E-2

	IN	EXP	SIM
MRT	214.7	961.4	955.1
max. cps	4397	88	16
area ratio(/in)	1	153.58	153.58

FIG. 84. RTD of wastewater through the sand filter.

From the measured wastewater flow rate and the known overall volume of the sand filter, the theoretical mean residence time was calculated to be 1065 s. Comparing this figure with the measured mean residence time of 747 s, it was concluded that 70% of the volume of the filter was occupied by water. The result showed that the filter was operating satisfactorily. There was no evidence of any liquid channelling through the system.

6.2. ORGANIC PHASE TRACING

6.2.1. Measurement of organic flows using a ^{113}Sn/^{113m}In generator

The velocity and volume flow rate of petrol flowing through a 10 cm diameter pipe at ambient temperature were measured using the integrated pulse velocity technique. The measurement procedure is identical to that of the normal pulse velocity technique: a pulse injection of radiotracer, with two downstream radiation detectors to measure the transit time of the tracer. However, in this variant of the technique, the detector response curves are integrated. The detector outputs are then similar to those shown in Fig. 85. From these, the transit time of the tracer is easily and accurately determined as the time separation between the two half-heights of the integrated response curves.

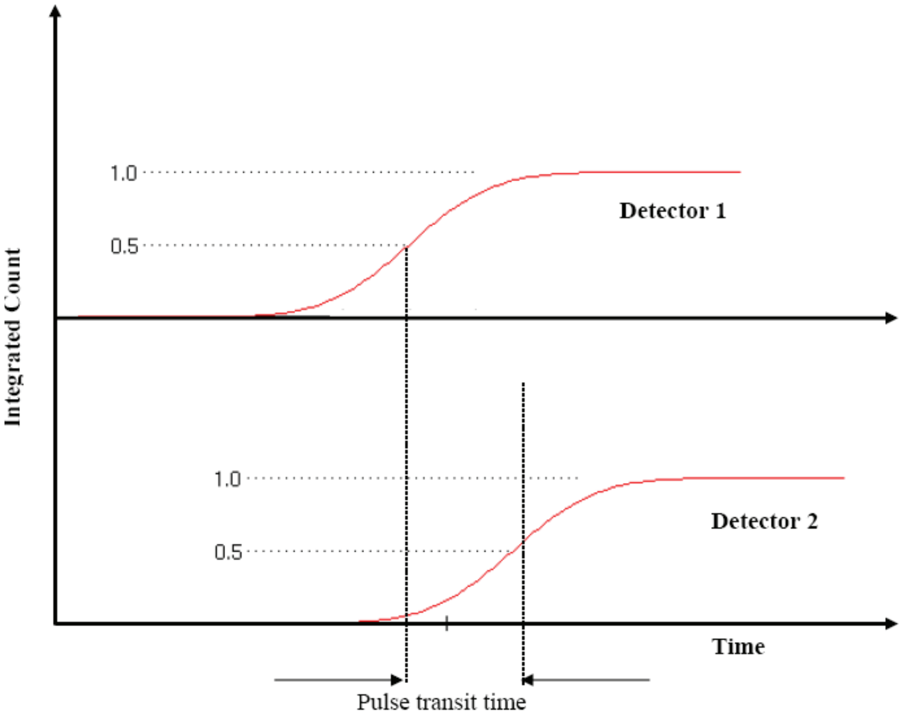


FIG. 85. Detector outputs in the integrated pulse velocity technique.

Two tracers were used:

(i) Paradibromobenzene, labelled with ^{82}Br, dissolved in toluene. This is known to be a reliable tracer for the measurement of the flow rate of organic liquids.
(ii) The ^{113m}In eluate from a ^{113}Sn/^{113m}In generator, supplied by Polatom. The eluate was mixed with methanol prior to injection.

The results of the tracer injections with paradibromobenzene are shown in Table 19, while the results obtained with the eluate injections are shown in Table 20. The detector separation was 7.19 m. The calculations of the mean transit time and mean flow velocity for radiotracer ^{82}Br give:

- Mean transit time = 5.12 ± 0.09 s;
- Mean flow velocity = 1.40 ± 0.03 m/s.

TABLE 19. RESULTS OF INTEGRATED PULSE VELOCITY MEASUREMENTS USING THE PARADIBROMOBENZENE TRACER

Run number	Transit time (s)
1	5.13
2	5.26
3	5.16
4	5.11
5	5.01
6	5.07

TABLE 20. RESULTS OF INTEGRATED PULSE VELOCITY MEASUREMENTS USING THE ^{113m}In ELUATE AS A TRACER

Run number	Transit time (s)
1	5.10
2	5.01
3	5.44
4	5.01
5	4.87
6	5.10
7	5.25
8	5.10
9	5.27
10	4.92

The calculations of mean transit time and mean flow velocity for radiotracer ^{113m}In give:

- Mean transit time = 5.11 ± 0.11 s;
- Mean flow velocity = 1.41 ± 0.03 m/s.

Note that the uncertainty in the mean velocities includes the uncertainty associated with the measurement of the detector separation (±5 cm).

It is clear that the two sets of results are in excellent agreement with one another, from which it is concluded that the aqueous eluate from the ^{113}Sn/^{113m}In generator is an appropriate tracer for this measurement. The Reynolds number for this flow is 1.7×10^5, indicating that it is fully turbulent.

Similar studies were carried out on a fuel oil flow in a 20 cm pipeline and again, there was good agreement between the results obtained with the generator eluate and with the paradibromobenzene tracer. It is, perhaps, going too far to suggest that on the basis of these two measurements, the eluate from the generator is a good tracer for any fully turbulent flow, aqueous or organic, but these results are encouraging.

Further to this, since the chemistry of gallium is very similar to that of indium, it is reasonable to suppose that the eluate from the ^{68}Ge/^{68}Ga would also be a satisfactory tracer for pulse velocity measurements of turbulent flows of organic liquids.

6.2.2. Tracer studies on a multiphase reactor in a pilot plant

The objective of these experiments was to study the flow patterns of gas, liquid and solid phases in the reactor. The experiment involved breakthrough studies of all three phases together with flow pattern visualization using the SPECT technique. However, concern here is confined to the liquid phase and, to a lesser extent, the solid phase, RTD measurements.

The RTD of the liquid phase (heptane) was studied using two tracers: ^{82}Br as a solution of paradibromobenzene in heptane (well known as a reliable tracer for organic liquids) and an ^{113m}In–EDTA complex. The ^{113m}In was eluted from a ^{113}Sn/^{113m}In generator and mixed with EDTA to create a stable complex. The purpose of using two tracers for the liquid phase was to compare their performance. If ^{113m}In–EDTA could be validated as a suitable tracer for this application, it would be possible to use the generator in subsequent tests, thereby simplifying the logistics of tracer supply.

Tracer was injected as a pulse into the inlet line to the reactor. A collimated NaI(Tl) detector, 38 mm in height and 50 mm in diameter was placed on the outlet line. Its response curves for the ^{82}Br and ^{113m}In tracers are shown in Fig. 86.

The response curves were similar, and it was therefore concluded that ^{113m}In–EDTA is suitable for tracing organic liquid flows on pilot plants. However, it must be remembered that the reactor was operating at ambient temperature. Further validation work is required before its use under high temperature conditions can be considered to be validated.

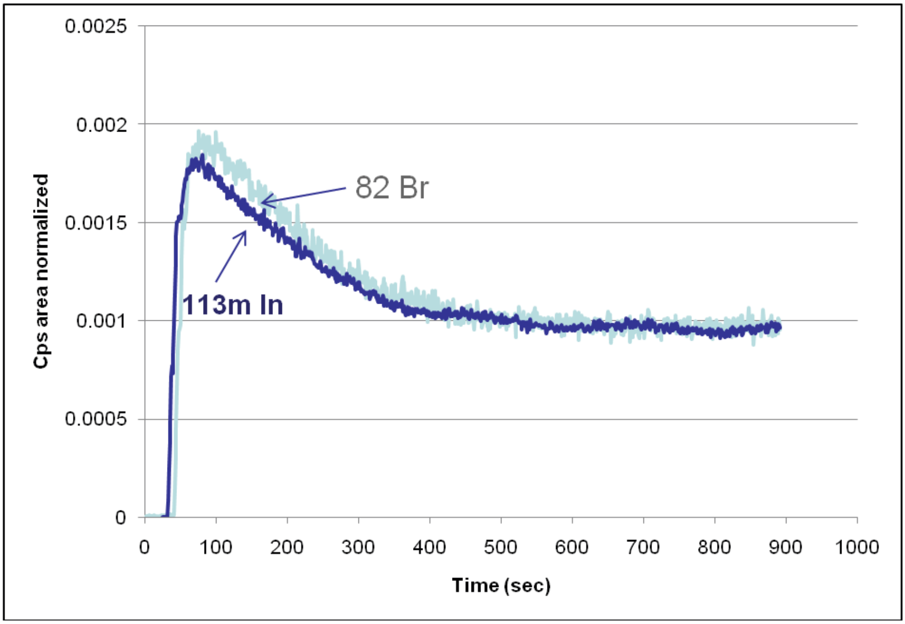


FIG. 86. Comparison of the detector response curves (area normalized) obtained from the ^{82}*Br and* ^{113m}*In–EDTA liquid phase tracers.*

The flow of the solid phase (catalyst) through the vessel was studied using the catalyst labelled with ^{113m}In. The tracer was prepared by stirring 5 g of catalyst powder with 10 cm^3 of the eluate from the ^{113}Sn/^{113m}In generator for approximately 1 min, after which 99.9% of the indium was adsorbed on to the catalyst. The labelled catalyst was injected into the inlet line to the reactor using nitrogen backing pressure. Examples of the response curve of the detector on the outlet line are presented in Fig. 87.

6.3. SOLID TRACING

6.3.1. RTD studies on attritioners in the silica sand industry using a ^{99}Mo/^{99m}Tc generator

Attritioners are used in silica sand processing plants to remove impurities from the silica particles. The silica particles are suspended in water, which is agitated by a propeller in the attritioner so that the particles impact upon one another at high velocity and suffer abrasion. Impurities are scrubbed from the

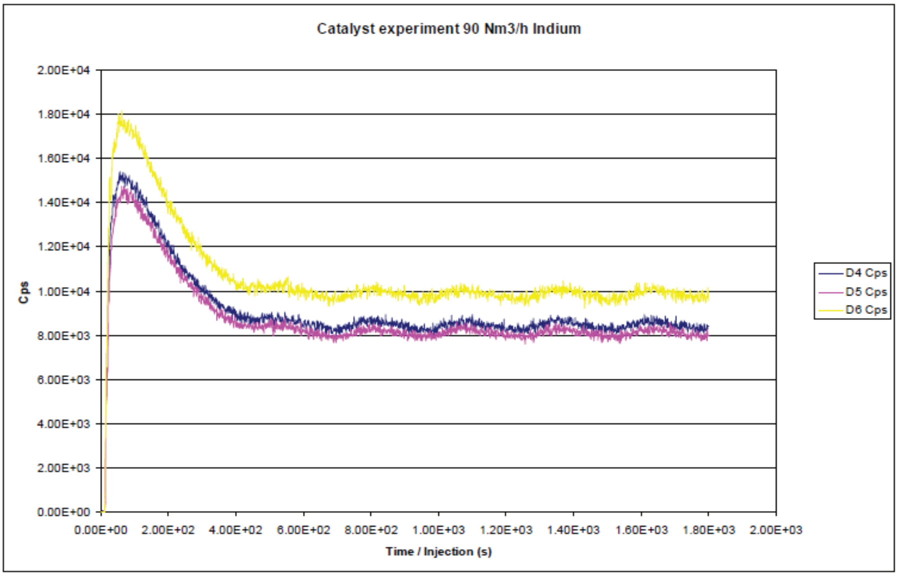


FIG. 87. Examples of detector response curves obtained using the In labelled catalyst as the tracer.

surface of the particles in this process, and are removed by water washing immediately downstream of the attritioner.

The objective of these studies was to study the RTDs of sand and water through the attritioner with a view to improving the efficiency of the attrition process.

The tracer selected to trace the liquid was the pertechnetate eluate from a ^{99}Mo/^{99m}Tc generator. This was known to be a reliable tracer for the water and was used without any chemical modification.

The solid phase was also labelled with ^{99m}Tc. This was done by stirring samples of sand with the eluate from the ^{99}Mo/^{99m}Tc generator, together with stannous chloride solution, which acted as a reducing agent. This causes the ^{99m}Tc to become fixed to the solid particles in the form of TcO_2.

Examples of RTD curves obtained for the liquid and solid phases are given in Figs 88 and 89, respectively.

The sand passes through the attritioner significantly more quickly than the liquid. This is not apparent from visual observation, which suggests that the two phases flow together through the system.

The long tails on the RTD curves are indicative of the presence of dead zones in the attritioner. This is due to the fact that there are significant gaps between the attritioner's propeller and the walls and bottom of the vessel. As a

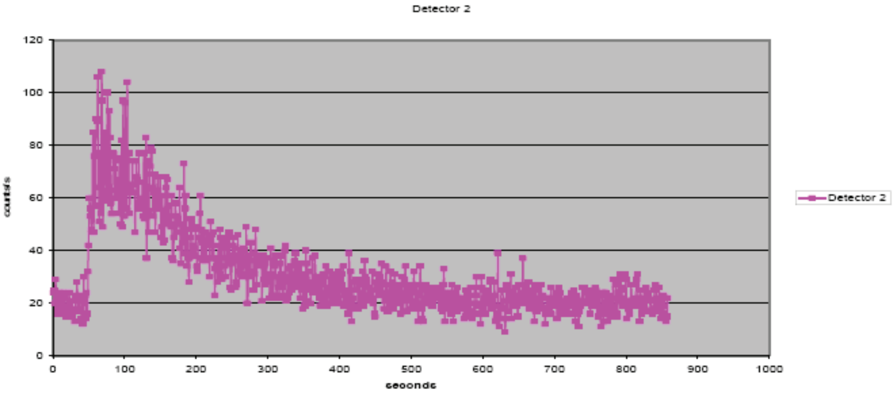


FIG. 88. Liquid RTD in the attritioner.

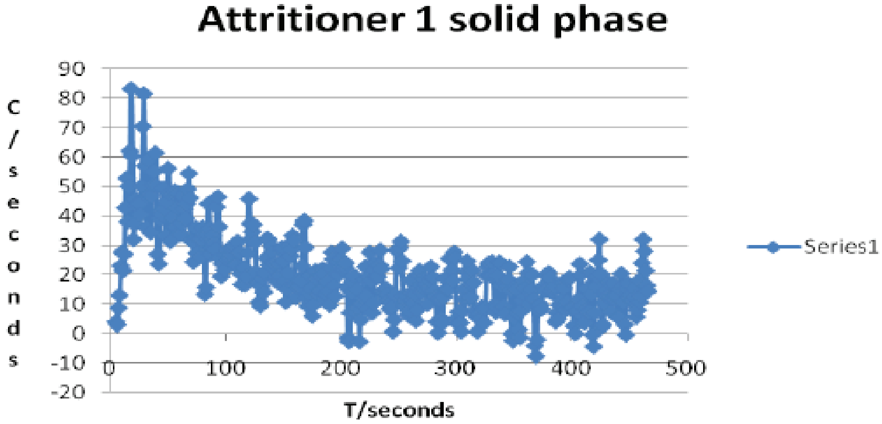


FIG. 89. Solid RTD in the attritioner.

result, a significant amount of sand is not subjected to the attrition process. Design improvements are necessary to reduce the dead zones. This is especially relevant as the higher quality sand reserves are becoming depleted, making it necessary to process material with higher levels of impurities. This, in turn, means that the efficiency of the attrition process must be improved.

6.3.2. Visualization of water and sludge flows through a clarifier using a ^{113}Sn/^{113m}In generator

Indium-113m from a ^{113}Sn/^{113m}In generator was used to monitor the flow behaviour of both the water and the sludge in a laboratory scale clarifier.

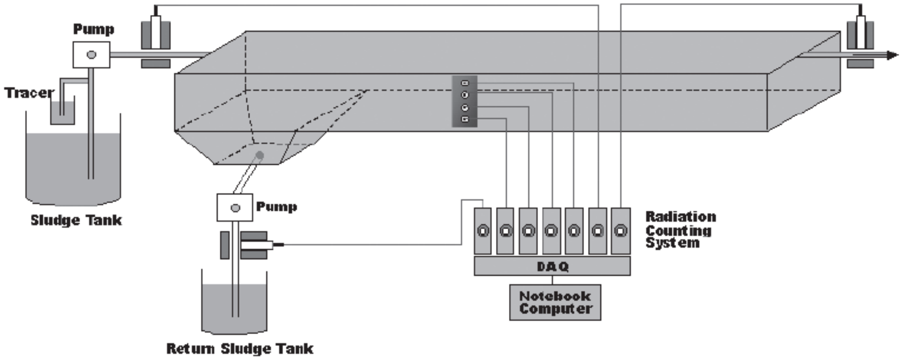


FIG. 90. Experimental arrangement for the clarifier studies.

There were several reasons for the choice of tracer:

- Firstly, the 0.392 keV gamma rays of ^{113m}In was very suitable for this study: this energy was sufficiently high to be detected effectively through the walls of the clarifier, yet it was still easy to provide shielding using relatively small thicknesses of lead.
- Secondly, the short half-life of 99.5 min ensured that contamination of the equipment with radioactivity and disposal of the contaminated water and sludge would not present problems.
- Thirdly, ^{113m}In can be used to trace both the water and the sludge. Because indium exists as a cation of $InCl_3$ in 0.05M HCl solution, it was possible to label the sludge in situ, simply by adding the eluate from the generator to the water containing the sludge to be monitored. In order to trace water flow, however, the indium cation was first reacted with EDTA to produce a complex compound that is very stable in the water and traces the water flow faithfully.

The clarifier is shown in Fig. 90.

NaI(Tl) scintillation detectors were installed at the inlet and at the outlet of the clarifier to monitor the inflow and outflow. The tracer inside the clarifier was monitored by four collimated 12.5 mm × 12.5 mm NaI(Tl) scintillation detectors, installed in a vertical line. This four detector arrangement was designed to move sequentially to five preselected measurement positions. The detector response signals were transmitted to a multichannel data acquisition system and recorded on a notebook computer. Approximately 5 mCi of radiotracer was used for each experimental run. This was injected as a pulse into the system just before the rotary pump upstream of the clarifier.

The results of the water tracing study are presented in Fig. 91 as a series of five tracer concentration contour maps, while the results of the sludge tracing are presented in Fig. 92. In contrast to the water, which was observed to move progressively through the clarifier before being discharged from its outlet, the sludge remained concentrated on the bottom of the vessel near the inlet.

It is interesting to compare these contour maps with a photograph of the clarifier taken during the study (Fig. 93). The dark region on the bottom of the clarifier is the settled sludge.

6.4. GAS TRACING

6.4.1. A radiotracer pig test for leak inspection in an underground gas pipeline

Bromine-82, labelled CH_3Br, produced by the portable methyl bromide generator was used in conjunction with a pipeline pig to check for the presence of

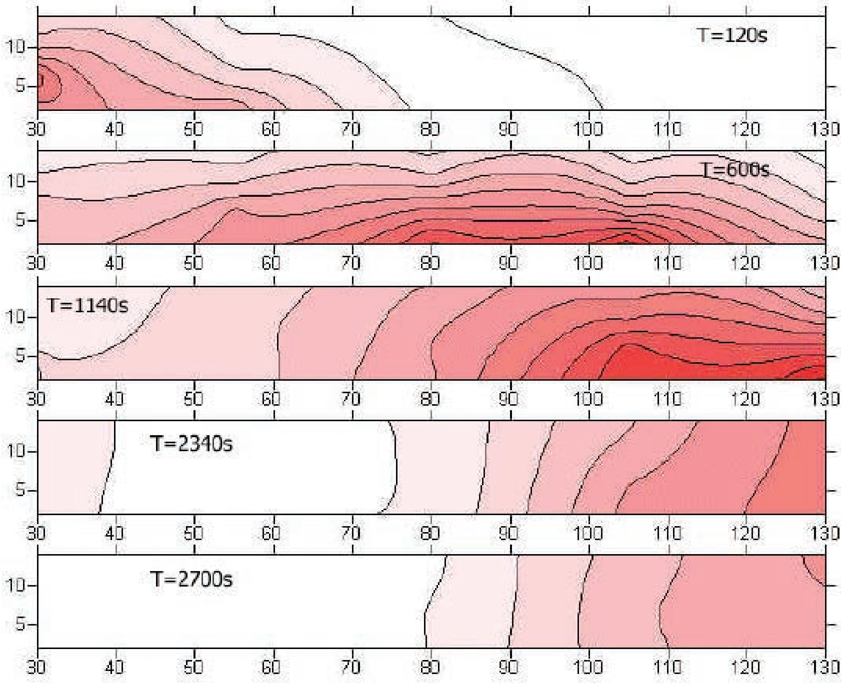


FIG. 91. Time variation of the water tracer concentration along the clarifier.

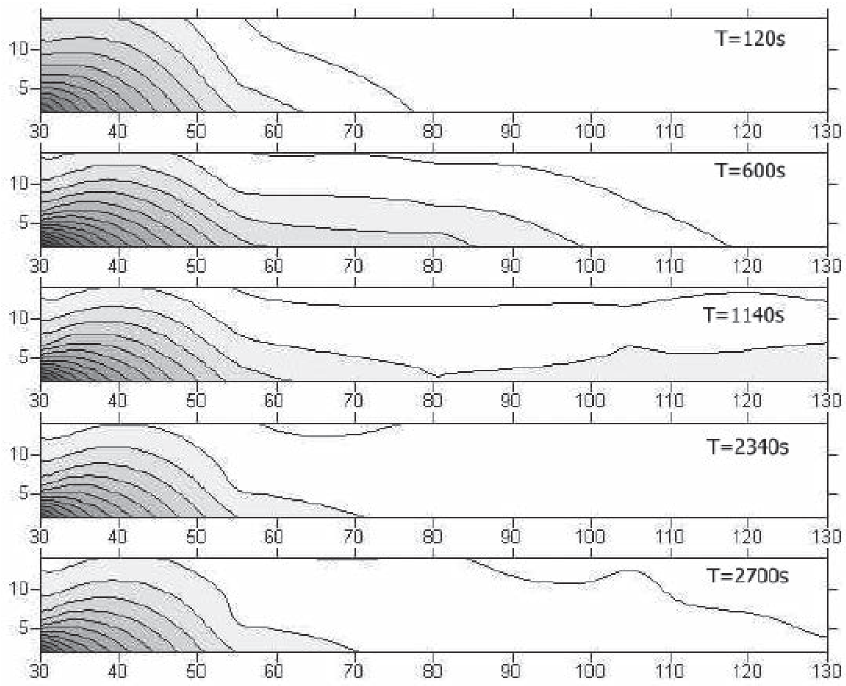


FIG. 92. Time variation of the sludge tracer concentration along the clarifier.

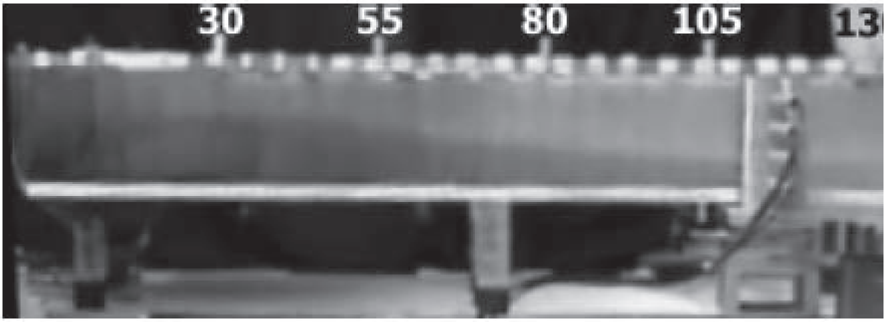


FIG. 93. Photograph of the clarifier.

leaks in an underground gas pipeline. The pipeline, of diameter 400 mm and length 40 km, was buried 2–3 m under the soil surface.

The pig, which was of a rugged, shockproof design, contained a 76 mm × 76 mm NaI scintillation detector, together with a data recorder. It was capable of 100 h continuous operation, the power for the detector and recording

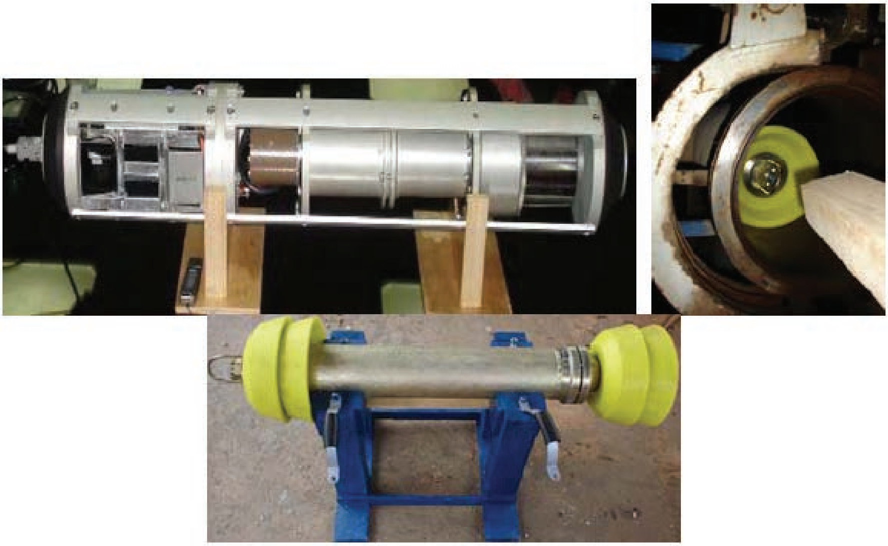


FIG. 94. The pipeline pig showing the pig internals, the assembled pig and the pig being launched into the pipe.

FIG. 95. Injection of the radiolabelled methyl bromide gas from the generator to the pipeline using a manual air pump.

units being supplied from batteries. Figure 94 shows the pig in its housing, ready for launching into the pipeline.

Ten curies of ^{82}Br methyl bromide gas provided by the generator was used as the radiotracer. The radiotracer gas was injected directly into the operational pipeline, as illustrated in Fig. 95.

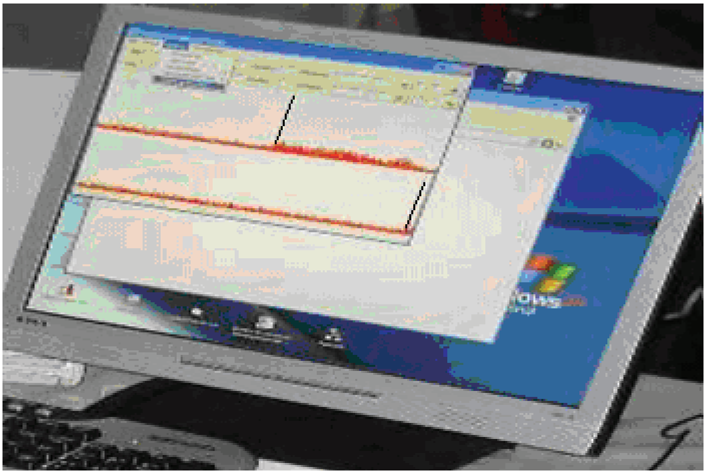


FIG. 96. Data downloaded from the pig.

A manual air pump was connected through a flexible metallic tube to the gaseous methyl bromide generator on one side and to the injection valve installed at the pipeline launcher on the other. By operating the air pump, the radiotracer was injected instantaneously, from the generator into the pipeline. After 2 h, the radiotracer pig was inserted into the pipeline through the cleaning chamber, as shown in Fig. 94.

The pig moved downstream under the influence of the flowing gas, taking approximately 6 h to transit the pipeline. After retrieving the pig at the other end of the pipeline, the data stored by it were downloaded to a PC. A section of the data covering a 1 h time interval is shown in Fig. 96.

The red traces shown in Fig. 96 comprise the background counts recorded by the detector, while the two sharp black peaks correspond to radiation received by the detector in transiting two small ^{60}Co sources that had been placed near the pipeline to act as distance markers. Had the radioactive tracer leaked from the pipeline into the soil, additional peaks would have been observed in the detector response at locations corresponding to the leak positions. No such peaks were present, from which it was concluded that the pipeline was not leaking.

REFERENCES

[1] FRIEDLANDER, G., KENNEDY, J.W., MACIAS, E.S., MILLER, J.M., Nuclear and Radiochemistry, John Wiley & Sons, New York (1981).

[2] KNAPP, F.F. (Ed.), Radionuclide Generators: New Systems for Nuclear Medicine Applications, ACS Symposium Series 241, American Chemical Society, Washington, DC (1984).

[3] VÉRTES, A., NAGY, S., KLENCSÁR, Z., LOVAS, R.G., RÖSCH, F. (Eds), Handbook of Nuclear Chemistry, Vol. 1: Basics of Nuclear Science, Springer Verlag, New York (2010).

[4] FIRESTONE, R.B., SHIRLEY, V.S. (Eds), Table of Isotopes, 8th edn, Lawrence Berkeley Laboratory, John Wiley & Sons, New York (1999).

[5] LEBOWITZ, E., RICHARDS, P., Radionuclide generator systems, Semin. Nucl. Med. **4** 3 (1974) 257–268.

[6] FINN, R., MOLINSKI, V., HUPF, H., KRAMER, H., Radionuclide Generators for Biomedicals Applications, Oak Ridge Technical Information Ctr (1983).

[7] SAHA, G.B., Physics and Radiobiology of Nuclear Medicine, 3rd edn, Springer (1993).

[8] BJØRNSTAD, T., “Industrial radionuclide generators — status and perspectives”, paper presented at Conf. on Tracer and Tracing Methods, Ciechocinek, Poland, 2004.

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Report of Consultants Meeting on Radionuclide Generators for Industrial Radiotracer Technology, 5–8 December 2005, IAEA, Vienna (2005).

[10] BJØRNSTAD, T., HOLTEBEKK, T., RUUD, A., The Production of ^{81m}Kr Generators at the Oslo Cyclotron, Technical Research Report 84-02, Institute of Physics Report Series, University of Oslo (1984).

[11] CHAKRAVARTY, R., DASH, A., KOTHARI, K., PILLAI, M.R.A.,VENKATESH, M., A novel ^{188}W/^{188}Re electrochemical generator with potential for medical applications, Radiochim. Acta **97** 6 (2009) 309–317.

[12] BOYD, R.E., The gel generator: A viable alternative source of ^{99m}Tc for nuclear medicine, Appl. Radiat. Isot. **48** 8 (1997) 1027–1033.

[13] SARASWATHY, P., et al., “Tc-99m generators for clinical use based on zirconium molybdate gel and (n,γ) produced Mo-99: Indian experience in the development and deployment of indigenous technology and processing facilities”, paper presented at International RERTR Meeting, Prague, 2007.

[14] RÖSCH, F., KNAPP, F., Jr., “Radionuclide generators”, Handbook of Nuclear Chemistry (VÉRTES, A., NAGY, S., KLENCSÁR, Z., RÖSCH, F., Eds), Vol. 4, Klüwer Academic Publishers, Dordrecht (2004) 81–118.

[15] DIETZ, M.L., HORWITZ, E.P., Applications of extraction chromatography in the development of radionuclide generator systems for nuclear medicine, Ind. Eng. Chem. Res. **39** 9 (2000) 3181–3188.

[16] SKURIDIN, V.S., CHIBISOV, E.V., Development of a small-size extractor for separation of the ^{99}Mo/^{99m}Tc couple, Radiochem. **52** 1 (2010) 90–94.

[17] BOYD, R.E., Technetium-99m generators — the available options, Int. J. Appl. Radiat. Isot. **33** 10 (1982) 801–809.

[18] NORONHA, O.P.D., Solvent extraction technology of ^{99}Mo-^{99m}Tc generator system 1: An Indian experience — process design considerations, Isot. Environ. Health Stud. **22** 2 (1986) 53–57.

[19] CHAKRAVARTY, R., DASH, A., VENKATESH, M., A novel electrochemical technique for the production of clinical grade ^{99m}Tc using (n, γ)^{99}Mo, Nucl. Med. Biol. **37** 1 (2010) 21–28.

[20] MOLINSKI, V.J., A review of ^{99m}Tc generator technology, Int. J. Appl. Radiat. Isot. **33** 10 (1982) 811–819.

[21] MILLER, J., "Procedure for production of ^{99m}Tc sublimation generators based on eutectic SiC–MoO_3–V_2O_5", Alternative Technologies for ^{99m}Tc Generators, IAEA-TECDOC-852, IAEA, Vienna (1995) 44–46.

[22] JIN XIAO-HAI, et al., "Development of ^{90}Sr/^{90}Y generator technology", Therapeutic Radionuclide Generators: ^{90}Sr/^{90}Y and ^{188}W/^{188}Re Generators, Technical Reports Series No. 470, IAEA, Vienna (2009).

[23] FIGGINS, F.E., The Radiochemistry of Polonium, Rep. NAS-NS 3037, National Academy of Sciences, US Atomic Energy Commission, Washington, DC (1961).

[24] MATTHEWS, K.M., KIM, C.-K., MARTIN, P., Determination of 210Po in environmental materials: A review of analytical methodology, Appl. Radiat. Isot. **65** (2007) 267–279.

[25] CHAKRAVARTY, R., et al., Development of an electrochemical ^{90}Sr–^{90}Y generator for separation of ^{90}Y suitable for targeted therapy, Nucl. Med. Biol. **35** 2 (2008) 245–253.

[26] CHAKRAVARTY, R., VENKATESH, M., DASH, A., A novel electrochemical ^{99}Mo/^{99}mTc generator, J. Radioanal. Nucl. Chem. **290** 1 (2011) 45–51.

[27] RAMANUJAM, A., et al., Separation of carrier-free 90Y from high level waste by supported liquid membrane using KSM-17, J. Radioanal. Nucl. Chem. **247** 1 (2001) 185–191.

[28] NAIK, P.W., et al., Separation of carrier-free 90Y from 90Sr by SLM technique using D2EHPA in n-dodecane as carrier, Sep. Sci. Technol. **45** 4 (2010) 554–561.

[29] HAPPEL, S., STRENG, R., VATER, P., ENSINGER, W., Sr/Y separation by supported liquid membranes based on nuclear track micro filters, Radiat. Meas. **36** (2003) 761–766.

[30] EISWIRTH, M., SCHWANKNER, R., WEIGEL, F., WISHNEVSKY, V., Electrostatic radionuclide separation — a new version of Rutherford's "thorium cow", J. Chem. Educ. **59** (1982) 608–611.

[31] MELICHER, F., SVOBODA, K., PROKOP, J., BUDSKY, F., "Experience in utilisation of WR-S reactor with medium neutron flux intensity for ^{99m}Tc production in Czechoslovakia", paper presented at IAEA Sem. on Radionuclide Generator Technology, Vienna, 1986.

[32] JENNEWEIN, M., SCHMIDT, A., NOVGORODOV, A.F., QAIM, S.M., RÖSCH, F., A no-carrier-added ^{72}Se/^{72}As radionuclide generator based on distillation, Radiochim. Acta **92** (2004) 245–249.

[33] BORROTO, J.I., DOMINGUEZ, J., GRIFFITH, J., FICK, M., LECLERC, J.P., Technetium-99m as a tracer for liquid RTD measurement in opaque anaerobic digester: Application in a sugar wastewater treatment plant, Chem. Eng. Process **42** 11 (2003) 857–865.

[34] PALIGE, J., MAJKOWSKA, A., HERDZIK, I., PTASZEK, S., ^{68}Ge/^{68}Ga radioisotope generator as a source of radiotracers for water flow investigations, Nukleonika **56** 1 (2011) 77−80.

[35] STAVSETRA, L., FURE, K., HAUGAN, A., BJØRNSTAD, T., “^{137m}Ba oil tracer — for applications in multiphase separators and scrubbers”, paper presented at Tracer 6, Oslo, 2011.

[36] HUBBELL, J.H., SELTZER, S.M., Tables of X-Ray Mass Attenuation Coefficients and Mass Energy-Absorption Coefficients from 1 keV to 20 MeV for Elements Z = 1 to 92 and 48 Additional Substances of Dosimetric Interest, NISTIR 5632, The National Institute of Standards and Technology (NIST), on-line 1996, last update 2004, http://www.nist.gov/pml/data/xraycoef/index.cfm

[37] BÉ, M.M., et al., Table of Radionuclides (Vol. 1-A = 1 to 150), Monograph BIPM-5, Bureau International des Poids et Mesures, Sèvres, France (2004) 311.

[38] GREENE, M.W., HILLMAN, M., A scandium generator, Int. J. Appl. Radiat. Isot. **18** (1967) 540–541.

[39] SEIDL, E., LIESER, K.H., Die Radionuklidgeneratoren ^{113}Sn/^{113m}In, ^{68}Ge/^{68}Ga und ^{44}Ti/^{44}Sc, Radiochim. Acta **19** (1973) 196–198.

[40] MIRZA, M.Y., AZIZ, A., A scandium generator, Radiochim. Acta **11** (1969) 43–44.

[41] CUI, A., O’BRIEN, H.A., Jr., STEINKRUGER, F.J., TAYLOR, W.A., “A new inorganic-based ^{44}Ti/^{44}Sc radioisotope generator”, Am. Chem. Soc. Nat. Meeting, 13–18 April 1986, American Chemical Society, Paper Nucl. 91 **53** (1990) 26–27.

[42] LOKTIONOVA, N., Development and Evaluation of a ^{44}Ti/^{44}Sc Radionuclide Generator and Labeling of Biomolecules with ^{44}Sc and ^{68}Ga for PET Imaging, PhD Thesis, Johannes Gutenberg-Universität (2010).

[43] PRUSZYNSKI, M., LOKTIONOVA, N., MAJKOWSKA, A., ROESCH, F., Radiolabeling of DOTATOC with ^{44}T/^{44}Sc generator-derived ^{44}Sc, J. Nucl. Med. **51** Suppl. 2 (2010) 1534.

[44] ENSDF Decay Data in the MIRD (Medical Internal Radiation Dose) Format for ^{68}Ga (1999),
http://www.orau.org/ptp/PTP%20Library/library/DOE/bnl/nuclidedata/MIRGa68.htm

[45] PAO, P.J., SILVESTER, D.J., WATERS, S.L., A new method for the preparation of ^{68}Ga-generators following proton bombardment of gallium oxide targets, J. Radioanal. Chem. **64** 1–2 (1981) 267–272.

[46] INTERNATIONAL ATOMIC ENERGY AGENCY, Cyclotron Produced Radionuclides: Physical Characteristics and Production Methods, Technical Reports Series No. 468, IAEA, Vienna (2009).

[47] GREENE, M.W., TUCKER, W.D., An improved gallium-68 cow, Int. J. Appl. Radiat. Isot. **12** (1961) 62–63.

[48] GIRARDI, F., PIETRA, R., SABBIONI, E., Radiochemical separations by retention on ionic precipitate adsorption tests on 11 materials, J. Radioanal. Chem. **5** (1970) 141–171.

[49] KOPECKY, P., MUDROVA, B., SVOBODA, K., The study of conditions for the preparation and utilization of a ^{68}Ge–^{68}Ga generator, Int. J. Appl. Radiat. Isot. **24** (1973) 73–80.

[50] KOPECKY, P., MUDROVA, B., ^{68}Ge-^{68}Ga generator for the production of ^{68}Ga in an ionic form, Int. J. Appl. Radiat. Isot. **25** (1974) 263–268.

[51] MALYSHEV, K.V., SMIRNOV, V.V., Gallium-68 yield from hydrated zirconium oxide based generators, Sov. Radiochim. **17** (1975) 137–140.

[52] EHRHARDT, G.J., WELCH, M.J., A new germanium-68/gallium-68 generator, J. Nucl. Med. **19** 8 (1978) 925–929.

[53] ARINO, H., SKRABA, W.J., KRAMER, H.H., A new ^{68}Ge/^{68}Ga radioisotope generator system, Int. J. Appl. Radiat. Nuclides **29** (1978) 117–120.

[54] LOC'H, C., MAZIERE, B., COMAR, D., A new generator for ionic gallium-68, J. Nucl. Med. **21** 2 (1980) 171–173.

[55] NEIRINCKX, R.D., DAVIS, M.A., Potential column chromatography for ionic Ga-68. II: Organic ion exchangers as chromatographic supports, J. Nucl. Med. **21** (1980) 81–83.

[56] SCHUMACHER, J., MAIER-BORST, W., A new ^{68}Ge/^{68}Ga radioisotope generator system for production of ^{68}Ga in dilute HCl, Int. J. Appl. Radiat. Isot. **32** (1981) 31–36.

[57] DAVIS, M.A., NEIRINCKX, R.D., Generator for Ionic Gallium-68 Based on Column Chromatography, US Pat. Doc. 4,288,424/A/, United States Patent and Trademark Office, Alexandria, VA (1981).

[58] NEIRINCKX, R.D., LAYNE, W.W., DAVIS, M.A., SAWAN, S.P., Development of an ionic ^{68}Ge-^{68}Ga generator III. Chelate resins as chromatographic substrates for germanium, Int. J. Appl. Radiat. Isot. **33** 4 (1982) 259–266.

[59] WELCH, M.J., Preparation of Gallium-68 Radiopharmaceuticals for Positron Tomography, Progress Report, 1 November 1980–31 December 1983, Rep. DOE/EV/04318-5, National Technical Information Service, Alexandria, VA (1983).

[60] McELVANY, K.D., HOPKINS, K.T., WELCH, M.J., Comparison of ^{68}Ge/^{68}Ga generator systems for radiopharmaceutical production, Int. J. Appl. Radiat. Isot. **35** 6 (1984) 521–524.

[61] YANO, Y, et al., Platelet labelling and positron emission tomography in the detection of sites of vascular injury, J. Nucl. Med. **26** 5 (1985) 21, Abstract No. 78.

[62] AMBE, S., ^{68}Ge/^{68}Ga generator with alpha-ferric oxide support, Appl. Radiat. Isot. **39** (1988) 49–51.

[63] SONG, M., BAO, B., A new (68)Ge-(68)Ga generator based on CeO_2, Nucl. Technol. **18** 7 (1995) 435–439.

[64] BAO, B., SONG, M., A new ^{68}Ge/^{68}Ga generator based on CeO_2, J. Radioanal. Nucl. Chem. **213** 4 (1996) 233–238.

[65] TANG, Q., ZHAO, G., LUO, W., LI, D., Preparation and study of a ^{68}Ge-^{68}Ga generator, J. Nucl. Radiochem. **19** 2 (1997) 29–33.

[66] CAO, B., LI, Z., WANG, Y., ^{68}Ge-^{68}Ga generator with alpha-ferric oxide support in trigonal structure, J. Radioanal. Nucl. Chem. **238** 1–2 (1998) 175–178.

[67] LIU, Z., JIA., C., The effect of SnO_2xH_2O on adsorption characteristics for ^{68}Ge-^{68}Ga generator, J. Isotopes **12** 1 (1999) 27–32.

[68] CHENG, W.-L., YAO, J., LEE, C.-S., LO, A.R., Preparation of ^{68}Ge/^{68}Ga generator with a binary Ga/Ag electrodepositions as solid target, J. Radioanal. Nucl. Chem. **245** 1 (2000) 25–30.

[69] EGAMEDIEV, S.K.H., KHUJAEV, S., MAMATKAZINA, A.K.H., Influence of preliminary treatment of aluminum oxide on the separation of ^{68}Ge-^{68}Ga radionuclide chain, J. Radioanal. Nucl. Chem. **246** 3 (2000) 593–596.

[70] NAKAYAMA, M., et al., A new ^{68}Ge/^{68}Ga generator system using an organic polymer containing N-methylglucamine groups as adsorbent for ^{68}Ge, Appl. Radiat. Isot. **58** 1 (2003) 9–14.

[71] AARDANEH, K., VAN DER WALT, T.N., Ga_2O for target, solvent extraction for radiochemical separation and SnO_2 for the preparation of a ^{68}Ge/^{68}Ga generator, J. Radioanal. Nucl. Chem. **268** 1 (2006) 25–32.

[72] LEWIS, R.E., CAMIN, L.L., Germanium-68/gallium-68 generator for the one step elution of ionic gallium-68, J. Labelled Compd. Radiopharm. **18** (1981) 164.

[73] BJØRNSTAD, T., FURE, K., STAVSETRA, L., HAUGAN, A., "Observations on the performance of the ^{68}Ge-^{68}Ga test generator produced at CIAE, China: Status per August 2007", paper presented at IAEA Mtg on Evaluation and Validation of Radionuclide Generator-based Radiotracers for Industrial Applications, Vienna, 2011.

[74] CHARLTON, J.S., FIRBY, B., JENKINS, D., "AUSTRALIA/UK: Evaluation of the ^{68}Ge-^{68}Ga and ^{137}Cs-^{137m}Ba generators from CIAE, China – final report", ibid.

[75] JUNG, S.-H., "Evaluation and improvement of the applicability of IRGs for industrial process diagnosis tests", ibid.

[76] RÖSCH, F., KNAPP, F.F., "Radionuclide generators", Handbook of Nuclear Chemistry (VÉRTES, A., NAGY, S., KLENCSAR, Z., LOVAS, R.G., RÖSCH, F., Eds), Vol. 4, Springer, New York (2011) 1935–1976.

[77] MIRZADEH, S., Generator-produced alpha-emitters, Appl. Radiat. Isot. **49** (1998) 345–349.

[78] RÖSCH, F., FILOSOFOV, D.V., Production, radiochemical processing and quality evaluation of Ge-68, IAEA Radioisotopes and Radiopharmaceuticals Series No. 2, IAEA, Vienna (2010) 11–28.

[79] FANI, M., ANDRÉ, J.P., MAECKE, H., ^{68}Ga-PET: A powerful generator-based alternative to cyclotron-based PET radiopharmaceuticals, Contr. Media Mol. Imaging **3** (2008) 67–77.

[80] ROESCH, F., RISS, P.J., The renaissance of the ^{68}Ge/^{68}Ga radionuclide generator initiates new developments in ^{68}Ga radiopharmaceutical chemistry, Curr. Top. Med. Chem. **10** 16 (2010) 1633–1668.

[81] LEDERER, C.M., SHIRLEY, V.S., Table of Isotopes, 7th edn, Wiley, New York (1978).

[82] QAIM, S.M., et al., Yield and purity of ^{82}Sr produced via the ^{nat}Rb(p,xn)^{82}Sr process, Appl. Radiat. Isot. **65** 2 (2007) 247–252.

[83] JOHN, K.D., MAUSNER, L., RUTH, T.J., Production and processing of ^{83}Sr, IAEA Radioisotopes and Radiopharmaceuticals Series No. 2, IAEA, Vienna (2010) 31–48.

[84] TARKANYI, F., QAIM, S.M., STOECKLIN, G., Excitation-functions of He-3-particle and alpha-particle induced nuclear-reactions on natural krypton: Production of Sr-82 at a compact cyclotron, Appl. Radiat. Isot. **39** (1988) 135–143.

[85] TARKANYI, F., QAIM, S.M., STOECKLIN, G., Excitation-functions of He-3-particle and alpha-particle induced nuclear-reactions on natural krypton with special reference to the production of Sr-82, Appl. Radiat. Isot. A **41** (1990) 91.

[86] GRANT, P.M., ERDAL, B.R., O'BRIEN, H.A., Jr., A ^{82}Sr-^{82}Rb isotope generator for use in nuclear medicine, J. Nucl. Med. **16** 4 (1975) 300.

[87] YANO, Y., et al., Rubidium-82 generators for imaging studies, J. Nucl. Med. **18** 1 (1977) 46–50.

[88] YANO, Y., CAHOON, J.L., BUDINGER, T.F., A precision flow-controlled Rb-82 generator for bolus or constant-infusion studies of the heart and brain, J. Nucl. Med. **22** 11 (1981) 1006–1010.

[89] HORLOCK, P.L., et al., The preparation of a rubidium-82 radionuclide generator, J. Radioanal. Chem. **64** 1–2 (1981) 257–265.

[90] GENNARO, G.P., et al., “A radionuclide generator and infusion system for pharmaceutical quality Rb-82”, Radionuclide Generators: New Systems for Nuclear Medicine Applications (KNAPP, F.F., Ed.), American Chemical Society, Washington, DC (1984) 135–150.

[91] GENNARO, G.P., HANEY, P.S., Strontium-82/Rubidium-82 Generator, US Patent 4,597,951, United States Patent and Trademark Office, Alexandria, VA (1986).

[92] BRIHAYE, C., et al., Preparation and evaluation of a hydrous tin(IV) oxide ^{82}Sr/^{82}Rb medical generator system for continuous elution, Int. J. Rad. Appl. Instrum. A **38** 3 (1987) 213–217.

[93] DEKEMP, R., ALVAREZ-DIEZ, T.M., BEANLANDS, R., VINCENT, J., Manufacture of strontium-82/rubidium-82 generators and quality control of rubidium-82 chloride for myocardial perfusion imaging in patients using positron emission tomography, Appl. Radiat. Isot. **50** 6 (1999) 1015–1023.

[94] SYLVESTER, P., MÖLLER, T., ADAMS, T.W., CISAR, A., New ion exchange materials for use in a ^{82}Sr/^{82}Rb generator, Appl. Radiat. Isot. **61** (2004) 1139–1145.

[95] BILEWICZ, A., BARTOŚ, B., MISIAK, R., PETELENZ, B., Separation of ^{82}Sr from rubidium target for preparation of ^{82}Sr/^{82}Rb generator, J. Radioanal. Nucl. Chem. **268** 3 (2006) 485–487.

[96] INTERNATIONAL ATOMIC ENERGY AGENCY, Manual for Reactor Produced Radioisotopes, IAEA-TECDOC-1340, IAEA, Vienna (2003).

[97] BOYD, R.E., “The gel generator option”, Production Technologies for Molybdenum-99 and Technetium-99m, IAEA-TECDOC 1065, IAEA, Vienna (1999) 148–156.

[98] ECKELMAN, W.C., COURSEY, B.M. (Eds), Special issue on “^{99m}Tc generators, chemistry and preparation of radiopharmaceuticals”, Int. J. Appl. Radiat. Isot. **33** 10 (1982).

[99] ARGONNE, L.N., Mo-99 Production Using LEU Targets, Link to related papers presented in the RERTR Mtg in Brazil (1998), http://www.rertr.anl.gov/MO99.html

[100] ADLOFF, J.P., LIESER, K.H., Special issue on “Radionuclide generator technology”, Radiochim. Acta **41** 2/3 (1987) 74.

[101] LEE, J.S., et al., “Sol-gel processed ceramics for ^{99}Mo/^{99m}Tc generator materials”, 5ICI (Proc. 5th Int. Conf. Isotopes 2005, Brussels), Medimond, Bologna (2005) 21–26.

[102] BAKEL, A.J., et al., “Progress in technology development for conversion of ^{99}Mo production from HEU targets — development of new separation schemes, and new processing equipment”, paper presented at 2006 RERTR Mtg, Cape Town, 2006.

[103] RAMAMOORTHY, N., GOLDMAN, I., ADELFANG, P., Enhanced prospects of ^{99m}Tc gel generator usage with the aid of post-elution concentration of pertechnetate in ISRS-2007, J. Labelled Compd. Radiopharm. **50** Suppl. 1 (2007) 896.

[104] RAMAMOORTHY, N., “Production and availability of ^{99m}Tc generators and of ^{99}Mo: Technology considerations and the IAEA contributions”, paper presented at NUCAR-2007 Symp., Mumbai, 2007.

[105] ZOLLE, I. (Ed.), Technetium-99 Pharmaceuticals: Preparation and Quality Control in Nuclear Medicine, Springer (2007).

[106] CHAKRAVARTY, R., et al., Polymer embedded nanocrystalline titania sorbent for ^{99}Mo-^{99m}Tc generator, J. Nanosci. Nanotechnol. **8** (2008) 4447–4452.

[107] NATIONAL RESEARCH COUNCIL OF THE NATIONAL ACADEMIES, Committee on Medical Isotope Production Without Highly Enriched Uranium: Medical Isotope Production Without Highly Enriched Uranium (Nuclear and Radiation Studies Board), The National Academies Press, Washington, DC (2009) 202.

[108] CHATTOPADHYAY, S., DAS, S.S., BARUA, L., A simple and rapid technique for recovery of ^{99m}Tc from low specific activity (n,γ)^{99}Mo based on solvent extraction and column chromatography, Appl. Radiat. Isot. **68** (2010) 1–4.

[109] YAMAURA, M., et al., "Studies on the separation of ^{99}Mo from nitric acid medium by alumina", RERTR 2010 32nd International Meeting on Reduced Enrichment for Research and Test Reactors,10–14 October 2010, Lisbon, Portugal (2010).

[110] CHAKRAVARTY, R., DASH, A., VENKATESH, M., A novel electrochemical technique for the production of clinical grade ^{99m}Tc using (n,γ)^{99}Mo, Nucl. Med. Biol. **37** (2010) 21–28.

[111] MONROY-GUZMAN, F., JIMENEZ MARTINEZ, T.S., ARRIOLA, H., LONGORIA GANDARA, L.C., Magnesium-molybdate compounds as matrix for ^{99}Mo/^{99m}Tc generators, Pharmaceuticals **4** (2011) 215–232.

[112] BORROTO, J., et al., "Development of a ^{137}Cs/^{137m}Ba radioisotope generator for educational and industrial radiotracer applications", paper presented at IAEA Mtg on Evaluation and Validation of Radionuclide Generator-based Radiotracers for Industrial Applications, Vienna, 2011.

[113] CHAKRAVARTY, R., et al., Practicality of tetragonal nano-zirconia as a prospective sorbent in the preparation of ^{99}Mo/^{99m}Tc generator for biomedical applications, Chromatographia **72** 9 (2010) 875–884.

[114] SERRANO GÓMEZ, J., GRANADOS CORREA, F., ^{99m}Tc generator with hydrated MnO_2 as adsorbent of ^{99}Mo, J. Radioanal. Nucl. Chem. **254** 3 (2002) 625–628.

[115] MUSHTAQ, A., MANSOOR, M.S., KARIM, H.M.A., KHAN, M.A., Hydrated titanium dioxide as an adsorbent for ^{99}Mo-^{99m}Tc generator, J. Radioanal. Nucl. Chem. **147** 2 (1991) 257–261.

[116] EL-KOLALY, M.T., A ^{99}Mo-^{99m}Tc generator based on the use of zirconium molybdophosphate-^{99}Mo gel, J. Radioanal. Nucl. Chem. **170** 2 (1993) 293–298.

[117] MILLER, J., "Procedure for production of ^{99}Tc^{m} sublimation generators based on eutectic SiC-MoO_3-V_2O_5", Alternative Technologies for ^{99}Tc^{m} Generators, IAEA-TECDOC-852, IAEA, Vienna (1995) 44–46.

[118] HANSEN, H.H., DE ROOST, E., MOUCHEL, D., VANINBROUKX, R., Internal conversion coefficients of the 392 keV γ-ray and half-life of the isomeric state in ^{113}In, Int. J. Appl. Radiat. Isot. **22** 1 (1971) 1–8.

[119] BĚTÁK, E., MIKOŁAJCZAK, R., STANISZEWSKA, J., MIKOŁAJEWSKI, S., RURARZ, E., Activation cross sections for reactions induced by 14 MeV neutrons on natural tin and enriched ^{112}Sn targets with reference to ^{111}In production via radioisotope generator ^{112}Sn(n,2n)^{111}Sn → ^{111}In, Radiochim. Acta **93** (2005) 311–326.

[120] KOPECKY, J. (Ed.), Atlas of Neutron Capture Cross Sections, INDC(NDS)-362, IAEA, Vienna (1997) 389.

[121] BRODOVITCH, J.C., Excitation Functions and Isomeric Yield Ratios of (p,xn) Reactions Induced in ^{75}As and ^{115}In by 20–85 MeV Protons, PhD Thesis, McGill Univ. (1973).

[122] NEVES, M., KLING, A., OLIVEIRA, A., Radionuclides used for therapy and suggestion for new candidates, J. Radioanal. Nucl. Chem. **266** 3 (2005) 377–384.

[123] GANDARIAS-CRUZ, D., OKAMOTO, K., Status of the Compilation of Nuclear Data for Medical Radioisotopes Produced by Accelerators, Technical Report INDC(NDS)-209/GZ, IAEA, Vienna (1988) 121.

[124] RAMAMOORTHY, N., NARASIMHAN, D.V.S., MANI, R.S., Studies on the preparation of ^{113}Sn-^{113m}In generators, Isot. Environ. Health Stud. **11** 7 (1975) 246–249.

[125] TERANISHI, K., YAMAASHI, Y., MARUYAMA, Y., ^{113}Sn-^{113m}In generator with a glass beads column, J. Radioanal. Nucl. Chem. **254** 2 (2002) 369–371.

[126] ALLAN, K.F., ALI, M.M.S., HANAFI, H.A., EL-AZONY, K.M., Separation of ^{113m}In from ^{113}Sn based on activated carbon used as column matrix, J. Radioanal. Nucl. Chem. **284** (2010) 297–301.

[127] EL-SAID, H., MSc Dissertation, Alexandria Univ. (1998).

[128] ABDEL-RAHMAN, H.E.R., Comparative Studies on Tin-113/Indium-113m Radioisotope Generators based on Tungstocerate (IV) Column Matrices, PhD Thesis, Helwan Univ. (2002).

[129] ABDEL-HALIM, A.A., Study on ^{113}Sn-^{113m}In Generator of the Chromatographic Column Elution Mode, PhD Thesis, Cairo Univ. (2002).

[130] MUSHTAQ, A., MANSUR, M.S., MUHAMMAD, A., Separation of radioindium (N.C.A.) from neutron irradiated tin and cadmium on hydrated antimony pentoxide, J. Radioanal. Nucl. Chem. **212** 2 (1996) 107–115.

[131] ARINO, H., SKRABA, W.J., KRAMER, H.H., A new ^{137}Cs/^{137}Ba radioisotope generator system, Int. J. Appl. Radiat. Isot. **24** 4 (1973) 197–200.

[132] ONICIU, L., VEGLIA, A., Étude des générateurs d'indium-113m, J. Radioanal. Chem. **24** (1975) 99–114.

[133] INTERNATIONAL ATOMIC ENERGY AGENCY, Manual for Reactor Produced Radioisotopes, IAEA-TECDOC-1340, IAEA, Vienna (2003).

[134] SUN, W.H., SONG, D.H., HUANG, Y., "Application of ^{113m}In radiotracer to well logging", Isotope Production and Applications in the 21st Century (STEVENSON, N.R., Ed.) (Proc. 3rd Int. Conf. Vancouver, 1999), World Scientific Publishing (1999) 226–228.

[135] ARINO, H., REARDON, V.D., Radioactive Barium-137m, CA Patent Document 907287, US Patent 3663177, United States Patent and Trademark Office, Alexandria, VA (1972) 11.

[136] ARINO, H., KRAMER, H.H., ^{113}Sn/^{113}In^{m} radioisotope generator systems, Int. J. Appl. Radiat. Isot. **25** 11–12 (1974) 493–496.

[137] RAMAMOORTHY, N., KRISHNAMOORTHY, M., MANI, R.S., Studies on the preparation of a ^{137}Cs-^{137m}Ba generator, Radiochem. Radioanal. Lett. **31** 4–5 (1977) 251–264.

[138] MATSUDA, H.T., ABRÃO, A., New Technique for Synthesizing AMP: Precipitation Inside an Ion-Exchange Resin and its Application to Separation of Cesium from Fission-products and to a ^{137m}Ba Generator, Rep. IPEN-Pub-13, IPEN, São Paulo (1980).

[139] KOPRDA, V., SCASNAR, V., Extraction — chromatographic separation of ^{137m}Ba from ^{137}Cs, J. Radioanal. Nucl. Chem. **121** 2 (1988) 415–428.

[140] WANG, Y., LEE, C., CHENG, W., “Study on the preparation of a Cs-137/Ba-137m radionuclide generator”, Proc. International Topical Conf. on Methods and Applications of Radioanalytical Chemistry II (MARC-2), Kona, HI (1991).

[141] WANG, Y.M., LEE, C.S., CHENG, W.L., Study on the preparation of a Cs-137/Ba-137m radionuclide generator, J. Radioanal. Nucl. Chem. **160** 2 (1992) 517–527.

[142] ROY, K., PAL, D.K., BASU, S., NAYAK, D., LAHIRI, S., Synthesis of a new ion exchanger, zirconium vanadate, and its application to the separation of barium and cesium radionuclides at tracer levels, Appl. Radiat. Isot. **57** (2002) 471–474.

[143] EL-ABSY, M.A., et al., Cs-137/Ba-137m radioisotope generator based on 6-tungstocerate(IV) column matrix, J. Radioanal. Nucl. Chem. **267** 1 (2005) 161–167.

[144] DHARA, S., DEY, S., BASU, S., DREW, M.G.B., CHATTOPADHYAY, P., Separation of ^{137m}Ba from ^{137}Cs using new ion exchanger $Na_2(H_2O)_4(H_3O)[Al(OH)_6Mo_6O_{18}]$, Radiochim. Acta **95** (2007) 297–301.

[145] LI, J., YANG, G., WANG, G., “Development and evaluation of ^{68}Ge-^{68}Ga and ^{137}Cs-^{137m}Ba radionuclide generators for industrial tracer application”, paper presented at IAEA Mtg on Evaluation and Validation of Radionuclide Generator-based Radiotracers for Industrial Applications, Vienna, 2011.

[146] CHAKRAVARTY, R., et al., Ammonium molybdophosphate impregnated alumina microspheres as a new generation sorbent for chromatographic ^{137}Cs/^{137m}Ba generator, J. Chromatogr. **A 1220** (2012) 82–91.

[147] BORTUN, A.I., et al., Modified titanium phosphates as cesium selective ion exchangers, Solv. Extr. Ion Exch. **15** 5 (1997) 895–907.

[148] MARAGEH, M.G., HUSAINA, S.W., KHANCHIB, A.R., Selective sorption of radioactive cesium and strontium on stannic molybdophosphate ion exchanger, Appl. Radiat. Isot. **50** (1999) 459–465.

[149] SATYANARAYANA, J., MURTHY, G.S., SASIDHAR, P., Adsorption studies of cesium on a new inorganic exchanger ammonium molybdophosphate-alumina (AMP-Al_2O_3), J. Radioanal. Nucl. Chem. **242** 1 (1999) 11–16.

[150] MARDAN, A., AJAZ, R., MEHMOOD, A., RAZA, S.M., GHAFFAR, A., Preparation of silica potassium cobalt hexacyanoferrate composite ion exchanger and its uptake behavior for cesium, Sep. Purif. Technol. **16** 2 (1999) 147–158.

[151] HASSAN, N.M., KING, W.D., McCABE, D.J., HAMM, L.L., JOHNSON, M.E., SuperLig®644 equilibrium sorption data for cesium from Hanford tank waste supernates, J. Radioanal. Nucl. Chem. **253** 3 (2002) 361–367.

[152] ISHFAQ, M.M., KARIM, H.M.A., KHAN, M.A., Preparation and characterization of potassium copper nickel hexacyanoferrate(II) as ion exchanger for cesium, J. Radioanal. Nucl. Chem. **159** 2 (1992) 335–342.

[153] NILCHI, A., MALEK, B., MARAGHEH, M.G., KHANCHI, A., Exchange properties of cyanide complexes — Part I. Ion exchange of cesium on ferrocyanides, J. Radioanal. Nucl. Chem. **258** 3 (2003) 457–462.

[154] NILCHI, A., MALEK, B., MARAGHEH, M.G., KHANCHI, A., Investigation of the resistance of the potassium copper nickel hexacyanoferrate(II) ion exchanger against gamma radiation, Radiat. Phys. Chem. **68** 5 (2003) 837–842.

[155] NILCHI, A., ATASHI, H., JAVID, A.H., SABERI, R., Preparation of PAN-based adsorbers for separation of cesium and cobalt from radioactive waste, Appl. Radiat. Isot. **65** (2007) 482–487.

[156] FENG, M., et al., Synthesis and characterization of a new activated carbon supported ammonium molybdophosphate composite and its cesium-selective adsorption properties, Radiochim. Acta **98** 1 (2010) 39–44.

[157] ORTH, R.J., BROOKS, K.P., KURATH, D.E., Review and Assessment of Technologies for the Separation of Cesium from Acidic Media, Rep. PNL-9874/UC-721, Batelle Pacific Northwest Laboratory, Richland, WA (1994).

[158] GAUR, S., Determination of Cs-137 in environmental water by ion-exchange chromatography, J. Chromatogr. A **733** (1996) 57–71.

[159] MUSHTAQ, A., Inorganic ion-exchangers: Their role in chromatographic radionuclide generators for the decade 1993–2002, J. Radioanal. Nucl. Chem. **262** 3 (2004) 797–810.

[160] MAJI, S., BASU, S., Separation of ^{137m}Ba from its parent ^{137}Cs from an equilibrium mixture using amide incorporated Amberlite IRC-50, Radiochim. Acta **95** (2007) 183–188.

[161] BJØRNSTAD, T., HOLTEBEKK, T., RUUD, A., The Production of ^{81m}Kr Generators at the Oslo Cyclotron, Technical Research Rep. 84-02, Institute of Physics Report Series, University of Oslo (1984).

[162] STAVSETRA, L., FURE, K., HAUGAN, A., BJØRNSTAD, T., “Development of an oil tracer labelled with ^{137m}Ba”, paper presented at 6th International Conf. Tracers and Tracing Methods (Tracer 6), Oslo (2011).

[163] CHAKRAVARTY, R., et al., Nanoceria-PAN composite-based advanced sorbent material: A major step forward in the field of clinical-grade ^{68}Ge/^{68}Ga generator, ACS Appl. Mater. Interfaces **2** 7 (2010) 2069–2075.

[164] INTERNATIONAL ATOMIC ENERGY AGENCY, Regulations for the Safe Transport of Radioactive Material 2009 Edition, IAEA Safety Standards Series No. TS-R-1, IAEA, Vienna (2009).

[165] INTERNATIONAL ATOMIC ENERGY AGENCY, Fundamental Safety Principles, IAEA Safety Standards Series No. SF-1, IAEA, Vienna (2006).

[166] INTERNATIONAL ATOMIC ENERGY AGENCY, International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, IAEA, Vienna (in preparation).

[167] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Dose Coefficients for Intakes of Radionuclides by Workers, Publication 68, Pergamon Press, Oxford and New York (1994).

[168] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, General Principles for the Radiation Protection of Workers, Publication 75, Pergamon Press, Oxford and New York (1997).

[169] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, 1990 Recommendations of the International Commission on Radiological Protection, Publication 60, Pergamon Press, Oxford and New York (1990).

BIBLIOGRAPHY

BJØRNSTAD, T., Selection of Tracers for Oil and Gas Reservoir Evaluation, Tech. Res. Rep. IFE/KR/E-91/009, Institute for Energy Technology, Kjeller, Norway (1991).

BJØRNSTAD, T., “Recent and current oil field tracer development for interwell application”, Proc. 2nd Tracer Workshop, University of Texas at Austin, 14–15 November 1994 (BJØRNSTAD, T., POPE, G. (Eds)), Res. Rep. IFE/KR/E-95/002, Institute for Energy Technology, Kjeller, Norway (1995).

BJØRNSTAD, T., New Development in Tracer Technology for Reservoir Description, Res. Rep. IFE/KR/E-98/006, Institute for Energy Technology, Kjeller, Norway (1998).

BJØRNSTAD, T., HAUGEN, O.B., HUNDERE, I.A., Dynamic behavior of radio-labelled water tracer candidates for chalk reservoirs, J. Petrol. Sci. Eng. **10** (1994) 223.

BJØRNSTAD, T., DUGSTAD, O., GALDIGA, C., SAGEN, J., “Interwell tracer technology in oil reservoirs: State-of-the-art”, paper presented at Proc. 2nd International Congress on Tracers and Tracing Methods (TRACER 2), Nancy, France, 2001.

BLET, V., BERNE, Ph., TOLA, F., VITART, X., CHAUSSY, C., Recent developments in radioactive tracers methodology, Appl. Rad. Isot. **51** (1999) 615–624.

BLET, V., BERNE, Ph., LEGOUPIL, S., VITART, X., Radioactive tracing as aid for diagnosing chemical reactors, Oil Gas Sci. Technol. **55** (2000) 171.

CHARLTON, J.S., Radioisotope Techniques for Problem-solving in Industrial Process Plants, Leonard Hill, Glasgow and London, 1986.

CHARLTON, J.S., HURST, J.A., Radioisotope techniques for problem solving in the offshore oil and gas, J. Pet. Technol. (1992).

CHMIELEWSKI, A.G., et al., “Application of radioisotope techniques”, paper presented at 4th Conf. on Radioisotope Application and Radiation Processing in Industry, Leipzig, 1989.

DUDUKOVIC, M.P., Opaque multiphase flows: Experiments and modeling, Exp. Thermal Fluid Sci. **26** (2002) 747–761.

FRIES, B.A., “Training the engineer for radiotracer applications in industry”, Radiation Engineering in the Academic Curriculum (Proc. Study Group Mtg Haifa, 1973), IAEA, Vienna (1975).

GARDNER, R.P., ELY, J.R., Radioisotope Measurement Applications in Engineering, Reinhold Publishing Corporation, New York (1967).

INTERNATIONAL ATOMIC ENERGY AGENCY, Guidebook on Radioisotope Tracers in Industry, Technical Reports Series No. 316, IAEA, Vienna (1990).

INTERNATIONAL ATOMIC ENERGY AGENCY, Radiotracer Technology as Applied to Industry, IAEA-TECDOC-1262, IAEA, Vienna (2001).

INTERNATIONAL ATOMIC ENERGY AGENCY, Tracer Applications in Oil Field Investigations, IAEA, Vienna (2003).

INTERNATIONAL ATOMIC ENERGY AGENCY, Radiotracer Applications in Industry — A Guidebook, Technical Reports Series No. 423, IAEA, Vienna (2004).

INTERNATIONAL ATOMIC ENERGY AGENCY, Industrial Process Gamma Tomography, IAEA-TECDOC-1589, IAEA, Vienna (2008).

INTERNATIONAL ATOMIC ENERGY AGENCY, Leak Detection in Heat Exchangers and Underground Pipelines Using Radiotracers, IAEA Training Course Series No. 38, IAEA, Vienna (2009).

JIN, J.-H., et al., Radiotracers and Labelling Compounds for Applications in Industry and Environment, Report of the CM, Warsaw (2004).

JOHANSEN, G.A., Nuclear tomography methods in industry, Nucl. Phys. A **752** (2005) 696c–705c.

KRAS, J., et al., “New tracer and adsorbents for checking the tightness of technological units in petroleum industry”, paper presented at 4th Conf. on Radioisotope Application and Radiation Processing in Industry, Leipzig, 1989.

PLASARI, E., THERESKA, J., LECLERC, J.P., VILLERMAUX, J., Tracer experiments and residence-time distributions in the analysis of industrial units: Case studies, Nukleonika **44** (1999) 39–58.

THYN, J., et al., Analysis and Diagnostics of Industrial Processes by Radiotracers and Radioisotope Sealed Sources, Czech Technical University, Prague (2000).

TRACERCO, Diagnostics^{TM} FCC Study, Brochure, Tracerco, Billingham, United Kingdom (1998).

TRACERCO, Using Radioisotope Technology to Maximise Efficiency, Petromin (1999) 62–69.

ZEMEL, B., Tracers in the Oil Field, Elsevier Science, Amsterdam (1995).

CONTRIBUTORS TO DRAFTING AND REVIEW

Bjørnstad, T.	Institute for Energy Technology, Norway
Borroto Portela, J.I.	Agencia de Energía Nuclear y Tecnologías de Avanzada, Cuba
Brisset, P.	International Atomic Energy Agency
Chankow, N.	Chulalongkorn University, Thailand
Charlton, J.S.	Oatley, Australia
Conejo Solis, M.A.	Instituto Tecnológico de Costa Rica, Costa Rica
Dagadu, C.P.K.	Ghana Atomic Energy Commission, Ghana
Dash, A.	Bhabha Atomic Research Centre, India
Diaz Vargas, F.	Comisión Chilena de Energía Nuclear, Chile
Ghiyas-Ud-Din	Pakistan Institute of Nuclear Science and Technology, Pakistan
Jin, J.-H.	International Atomic Energy Agency
Li Jinfu	China Institute of Atomic Energy, China
Muñoz Anrique, L.	Comisión Chilena de Energía Nuclear, Chile
Pinto Ferreira, A.	Comissão Nacional de Energia Nuclear, Brazil
Palige, J.M.	Institute of Nuclear Chemistry and Technology, Poland
Roesch, F.	Institute of Nuclear Chemistry, Germany
Sung-Hee Jung	Korea Atomic Energy Research Institute, Republic of Korea
Vargas, C.	Instituto Tecnológico de Costa Rica, Costa Rica

No. 22

IAEA
International Atomic Energy Agency

Where to order IAEA publications

In the following countries IAEA publications may be purchased from the sources listed below, or from major local booksellers. Payment may be made in local currency or with UNESCO coupons.

AUSTRALIA
DA Information Services, 648 Whitehorse Road, MITCHAM 3132
Telephone: +61 3 9210 7777 • Fax: +61 3 9210 7788
Email: service@dadirect.com.au • Web site: http://www.dadirect.com.au

BELGIUM
Jean de Lannoy, avenue du Roi 202, B-1190 Brussels
Telephone: +32 2 538 43 08 • Fax: +32 2 538 08 41
Email: jean.de.lannoy@infoboard.be • Web site: http://www.jean-de-lannoy.be

CANADA
Bernan Associates, 4501 Forbes Blvd, Suite 200, Lanham, MD 20706-4346, USA
Telephone: 1-800-865-3457 • Fax: 1-800-865-3450
Email: customercare@bernan.com • Web site: http://www.bernan.com

Renouf Publishing Company Ltd., 1-5369 Canotek Rd., Ottawa, Ontario, K1J 9J3
Telephone: +613 745 2665 • Fax: +613 745 7660
Email: order.dept@renoufbooks.com • Web site: http://www.renoufbooks.com

CHINA
IAEA Publications in Chinese: China Nuclear Energy Industry Corporation, Translation Section, P.O. Box 2103, Beijing

CZECH REPUBLIC
Suweco CZ, S.R.O., Klecakova 347, 180 21 Praha 9
Telephone: +420 26603 5364 • Fax: +420 28482 1646
Email: nakup@suweco.cz • Web site: http://www.suweco.cz

FINLAND
Akateeminen Kirjakauppa, PO BOX 128 (Keskuskatu 1), FIN-00101 Helsinki
Telephone: +358 9 121 41 • Fax: +358 9 121 4450
Email: akatilaus@akateeminen.com • Web site: http://www.akateeminen.com

FRANCE
Form-Edit, 5, rue Janssen, P.O. Box 25, F-75921 Paris Cedex 19
Telephone: +33 1 42 01 49 49 • Fax: +33 1 42 01 90 90
Email: formedit@formedit.fr • Web site: http://www. formedit.fr

Lavoisier SAS, 145 rue de Provigny, 94236 Cachan Cedex
Telephone: + 33 1 47 40 67 02 • Fax +33 1 47 40 67 02
Email: romuald.verrier@lavoisier.fr • Web site: http://www.lavoisier.fr

GERMANY
UNO-Verlag, Vertriebs- und Verlags GmbH, Am Hofgarten 10, D-53113 Bonn
Telephone: + 49 228 94 90 20 • Fax: +49 228 94 90 20 or +49 228 94 90 222
Email: bestellung@uno-verlag.de • Web site: http://www.uno-verlag.de

HUNGARY
Librotrade Ltd., Book Import, P.O. Box 126, H-1656 Budapest
Telephone: +36 1 257 7777 • Fax: +36 1 257 7472 • Email: books@librotrade.hu

INDIA
Allied Publishers Group, 1st Floor, Dubash House, 15, J. N. Heredia Marg, Ballard Estate, Mumbai 400 001,
Telephone: +91 22 22617926/27 • Fax: +91 22 22617928
Email: alliedpl@vsnl.com • Web site: http://www.alliedpublishers.com

Bookwell, 2/72, Nirankari Colony, Delhi 110009
Telephone: +91 11 23268786, +91 11 23257264 • Fax: +91 11 23281315
Email: bookwell@vsnl.net

ITALY
Libreria Scientifica Dott. Lucio di Biasio “AEIOU”, Via Coronelli 6, I-20146 Milan
Telephone: +39 02 48 95 45 52 or 48 95 45 62 • Fax: +39 02 48 95 45 48
Email: info@libreriaaeiou.eu • Website: www.libreriaaeiou.eu

JAPAN
Maruzen Company Ltd, 1-9-18, Kaigan, Minato-ku, Tokyo, 105-0022
Telephone: +81 3 6367 6079 • Fax: +81 3 6367 6207
Email: journal@maruzen.co.jp • Web site: http://www.maruzen.co.jp

REPUBLIC OF KOREA
KINS Inc., Information Business Dept. Samho Bldg. 2nd Floor, 275-1 Yang Jae-dong SeoCho-G, Seoul 137-130
Telephone: +02 589 1740 • Fax: +02 589 1746 • Web site: http://www.kins.re.kr

NETHERLANDS
De Lindeboom Internationale Publicaties B.V., M.A. de Ruyterstraat 20A, NL-7482 BZ Haaksbergen
Telephone: +31 (0) 53 5740004 • Fax: +31 (0) 53 5729296
Email: books@delindeboom.com • Web site: http://www.delindeboom.com

Martinus Nijhoff International, Koraalrood 50, P.O. Box 1853, 2700 CZ Zoetermeer
Telephone: +31 793 684 400 • Fax: +31 793 615 698
Email: info@nijhoff.nl • Web site: http://www.nijhoff.nl

Swets and Zeitlinger b.v., P.O. Box 830, 2160 SZ Lisse
Telephone: +31 252 435 111 • Fax: +31 252 415 888
Email: infoho@swets.nl • Web site: http://www.swets.nl

NEW ZEALAND
DA Information Services, 648 Whitehorse Road, MITCHAM 3132, Australia
Telephone: +61 3 9210 7777 • Fax: +61 3 9210 7788
Email: service@dadirect.com.au • Web site: http://www.dadirect.com.au

SLOVENIA
Cankarjeva Zalozba d.d., Kopitarjeva 2, SI-1512 Ljubljana
Telephone: +386 1 432 31 44 • Fax: +386 1 230 14 35
Email: import.books@cankarjeva-z.si • Web site: http://www.cankarjeva-z.si/uvoz

SPAIN
Díaz de Santos, S.A., c/ Juan Bravo, 3A, E-28006 Madrid
Telephone: +34 91 781 94 80 • Fax: +34 91 575 55 63
Email: compras@diazdesantos.es, carmela@diazdesantos.es, barcelona@diazdesantos.es, julio@diazdesantos.es
Web site: http://www.diazdesantos.es

UNITED KINGDOM
The Stationery Office Ltd, International Sales Agency, PO Box 29, Norwich, NR3 1 GN
Telephone (orders): +44 870 600 5552 • (enquiries): +44 207 873 8372 • Fax: +44 207 873 8203
Email (orders): book.orders@tso.co.uk • (enquiries): book.enquiries@tso.co.uk • Web site: http://www.tso.co.uk

On-line orders
DELTA Int. Book Wholesalers Ltd., 39 Alexandra Road, Addlestone, Surrey, KT15 2PQ
Email: info@profbooks.com • Web site: http://www.profbooks.com

Books on the Environment
Earthprint Ltd., P.O. Box 119, Stevenage SG1 4TP
Telephone: +44 1438748111 • Fax: +44 1438748844
Email: orders@earthprint.com • Web site: http://www.earthprint.com

UNITED NATIONS
Dept. I004, Room DC2-0853, First Avenue at 46th Street, New York, N.Y. 10017, USA
(UN) Telephone: +800 253-9646 or +212 963-8302 • Fax: +212 963-3489
Email: publications@un.org • Web site: http://www.un.org

UNITED STATES OF AMERICA
Bernan Associates, 4501 Forbes Blvd., Suite 200, Lanham, MD 20706-4346
Telephone: 1-800-865-3457 • Fax: 1-800-865-3450
Email: customercare@bernan.com · Web site: http://www.bernan.com

Renouf Publishing Company Ltd., 812 Proctor Ave., Ogdensburg, NY, 13669
Telephone: +888 551 7470 (toll-free) • Fax: +888 568 8546 (toll-free)
Email: order.dept@renoufbooks.com • Web site: http://www.renoufbooks.com

Orders and requests for information may also be addressed directly to:

Marketing and Sales Unit, International Atomic Energy Agency
Vienna International Centre, PO Box 100, 1400 Vienna, Austria
Telephone: +43 1 2600 22529 (or 22530) • Fax: +43 1 2600 29302
Email: sales.publications@iaea.org • Web site: http://www.iaea.org/books

12-46311

RELATED PUBLICATIONS

RADIOTRACER APPLICATIONS IN INDUSTRY – A GUIDEBOOK
Technical Reports Series No. 423
STI/DOC/010/423 (281 pp.; 2004)
ISBN 92–0–114503–9 Price: €45.00

CYCLOTRON PRODUCED RADIONUCLIDES: PHYSICAL CHARACTERISTICS AND PRODUCTION METHODS
Technical Reports Series No. 468
STI/DOC/010/468 (266 pp.; 2009)
ISBN 978–92–0–106908–5 Price: €52.00

RADIOTRACER RESIDENCE TIME DISTRIBUTION METHOD FOR INDUSTRIAL AND ENVIRONMENTAL APPLICATIONS
Training Course Series No. 31
(2008)
Price: €0.0

Radionuclide generators represent a cost effective strategy to ensure a continuous and reliable supply of short lived radionuclides for radiotracer investigations. Making use of radiotracers from radionuclide generators can alleviate the difficulties associated with radioisotope supply. Another advantage is the ease with which the daughter product can be separated on demand and converted into a radiotracer by some tailor-made labelling procedure at the site. This publication contains guidelines for the selection and preparation of radiotracers derived from radionuclide generators. It describes the results of research undertaken in the characterization of ^{68}Ge/^{68}Ga, ^{137}Cs/^{137m}Ba, ^{99}Mo/^{99m}Tc and ^{113}Sn/^{113m}In radiotracer generators and their validation in industrial process investigations. Radiotracers have been widely used throughout industry to optimize processes, solve problems, improve product quality, save energy and reduce pollution. Recent trends in the industrialization process of developing countries suggest that radiotracer techniques will continue to play an important role in industry for many years to come.

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA
ISBN 978–92–0–135410–5
ISSN 2220–7341