FACULTY
OF MATHEMATICS
AND PHYSICS
Charles University

MASTER THESIS

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Medical radioisotopes production capabilities at the LVR-15 reactor

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Study programme: Physics

Study branch: Nuclear and subnuclear physics

Prague 2021

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Abstract: This thesis is focused on medical radioisotopes production capabilities at the LVR-15 research reactor at Řež, near Prague in the Czech Republic. Firstly, a calculation prediction of activities for the chosen target materials is provided as well as the determination of the suitable irradiation time for the experiment. These target materials in the form of activation detectors are then irradiated in the reactor. Their activities are measured by methods of γ-spectrometry and the results are compared with the predictions. This validates those calculation predictions for the use of activity predictions for the process of medical radioisotope production at the LVR-15 reactor and for the subsequent shipment of these radioisotopes to the hospitals and/or final processors.

Contents

Introduction

Radioisotopes are radioactive isotopes of an element. Different isotopes of the same element have the same number of protons in their atomic nuclei, but they differ in the number of neutrons. They can be found in nature or can be artificially produced.

Artificially produced radioisotopes have a wide ranging area of applications in various fields such as industry, research, medicine and agriculture. These may include industrial radiography,where radioactive isotopes of various kinds are used for measuring the thickness of metal or plastic sheets,or can be used for irradiation and sterilization of food. Radiopharmaceutical usage is of essential importance for example in nuclear imaging or scanning methods such as Positron Emission Tomography (PET) scans, Computed Tomography (CT) scans. Furthermore, numerous diseases especially cancers are directly treated using radioisotopes.

Today, there are 2 major methods of radioisotope production. They can be produced in a nuclear reactor environment (most of them) or accelerator production is also possible. Usually, this is done by exposing a target material to particles, such as neutrons or protons, followed by different chemical processing to bring them into the required chemical form. Accelerator production is used for isotopes with special properties that cannot be achieved in the reactor, for instance isotopes used in treatment with a half-life of minutes, hours which can be considered a short half-life. These are mostly produced in the accelerator in a facility near the location of the final user.

Reactor radioisotope production started on a more significant scale in several countries with the commissioning of reactors designed specifically for this purpose and also for research starting from the late 1950s. Between the 50s and the 70s the number of new research reactors was growing rapidly resulting in an increase of new radioisotopes being produced with the advance of the technology as well.

This thesis is focused on reactor-produced radioisotopes used in medicine and its main aim is to assess the current production capabilities of the LVR-15 reactor for irradiation of new radioisotopes. The first two chapters discuss the most common medicinal radioisotopes for treatment or for imaging purposes and their properties. The third chapter gives details on the reactor facility itself. Chapter 4 presents some theoretical background on activation analysis and related phenomena which is strongly connected to the production procedure of reactor produced radioisotopes. Chapters 5,6,7 give details on experimental part of this thesis, results of the experiment and the production capabilities of medicinal radioisotopes at the LVR-15 reactor respectively.

1. Nuclear medicine

As it was mentioned, radioisotopes are useful in various fields, one of which is medicine. They are used routinely in the clinics for various diseases, including some of the most important and frequent ones, like cancers and cardiovascular diseases. Nowadays, there is plenty of radioisotopes with a specific use to help the patient.

Nuclear medicine plays a growing role in diagnosis and therapy. Radioisotopes are a crucial component of the radiopharmaceuticals that are used for the non-invasive diagnosis and treatment. Radiopharmaceuticals are specific biological molecules tagged (or "labelled") with medical radioisotopes. They are also called "tracers", because they only need to be administered in very small quantities (traces) thanks to the high sensitivity provided by nuclear radiations. They allow to trace biological processes[3]. Radioisotopes can be used in 2 forms: as sealed sources (i.e. radioisotopes incorporated in solid substances or sealed in an inactive capsule such as Co-60,Kr-85,) or as unsealed sources (e.g. I-131, Ra-223).

A sealed radioactive source is radioactive material that is permanently sealed in a capsule or bonded and in a solid form (see fig.1.1). The capsule of a sealed radioactive source is designed to prevent the radioactive material from escaping or being released from encapsulation under normal usage and probable accident conditions. In most practices, a sealed radioactive source is installed in a device that is designed either to allow the source to move safely out of the shielded device to where the radiation beam is used and to be returned to the shielded device after the operation is complete, or to allow a beam of radiation to be released from the device while maintaining shielding around the source. The beam of radiation may then be used for treatment of cancer in medical patients[20].

Unsealed sources are used in radiotherapy to treat medical conditions, mainly cancer. These radioactive substances are introduced into the body by various means (injection or ingestion) and localise to specific organs or tissues depending on their properties and administration routes.

1.1 Nuclear radioisotopes used in medicine

Nuclear medicine is non-invasive and includes two main aspects: diagnostics and treatment.

1.1.1 Diagnostic analyses

For diagnosis using radiopharmaceuticals, a radioactive dose is given to the patient and the activity in the organ can then be studied either as a two dimensional picture or, using tomography, as a three dimensional picture. Diagnostic techniques in nuclear medicine use radioactive tracers which emit gamma rays from within the body. These tracers are generally short-lived isotopes (of the order of hours such as F-18 with half- life of 110 minutes) linked to chemical compounds which permit specific physiological processes to be scrutinised. They can be given by injection, inhalation, or orally. The earliest technique developed uses single photons detected by a gamma camera which can view organs from many different

Figure 1.1: A sealed source [20]

angles. The camera builds up an image from the points from which radiation is emitted, this image is enhanced by a computer and viewed on a monitor for indications of abnormal conditions. Single photon emission computed tomography (SPECT) is a major scanning technology to diagnose and monitor a wide range of medical conditions. Radioisotopes that are used in SPECT include iodine-123, technetium-99m, xenon-133, thallium-201, and fluorine-18 in order to label tracers.

A more recent development is positron emission tomography (PET) which is a more precise and sophisticated technique using isotopes produced in a cyclotron. A positron-emitting radionuclide is introduced, usually by injection, and accumulates in the target tissue. As it decays it emits a positron, which promptly combines with a nearby electron resulting in the simultaneous emission of two identifiable gamma rays in opposite directions. These are detected by a PET camera and give very precise indications of their origin. PET's most important clinical role is in oncology, with fluorine-18 as the tracer, since it has proven to be the most accurate non-invasive method of detecting and evaluating most forms of cancers. It is also well used in cardiac and brain imaging. New procedures combine PET with computed X-ray tomography (CT) scans to give co-registration of the two images (PET-CT), enabling 30% better diagnosis than with a traditional gamma camera alone[11].

A distinct advantage of nuclear imaging over X-ray techniques is that both bone and soft tissue can be imaged very successfully. Diagnostic radiopharmaceuticals can be used to examine blood flow to the brain, functioning of the liver, lungs, heart, or kidneys, to assess bone growth, and to confirm other diagnostic procedures.

1.1.2   Therapy and treatment

The usage of radioisotopes in therapy is also important. Cancerous cells are sensitive to damage by radiation. For this reason, some cancerous cells can be controlled or eliminated by irradiating the area containing the growth. There are 2 ways of treatment using radiation, i.e. external source or internal radionuclide therapy.

External irradiation (sometimes called teletherapy) can be carried out using a gamma beam from a radioactive cobalt-60 source, though in developed countries the much more versatile linear accelerators are now being used as high-energy X-ray sources (gamma and X-rays are much the same). An external radiation procedure is known as gamma knife radiosurgery, and involves focusing gamma radiation from 201 sources of Co-60 on a precise area of the brain with a cancerous tumour.

Internal radionuclide therapy is administered by planting a small radiation source, usually a gamma or beta emitter, in the target area. Short-range radiotherapy is known as brachytherapy, and this is becoming the main means of treatment. Iodine-131 is commonly used to treat thyroid cancer, probably the most successful kind of cancer treatment. It is also used to treat non-malignant thyroid disorders. Iridium-192 implants are used especially in the head and the breasts. They are produced in wire form and are introduced through a catheter to the target area. After administering the correct dose, the implant wire is removed to shielded storage. Brachytherapy procedures give less overall radiation to the body, are more localized to the target tumour, and are cost-effective.

1.1.3   Production of radioisotopes

Many radioisotopes are made in nuclear reactors, some in cyclotrons. Generally neutron-rich ones and those resulting from nuclear fission need to be made in reactors. On the other hand neutron-depleted ones are made in cyclotrons. Some of the most used radioisotopes with their properties are listed below with their properties and specific use. The radioisotopes produced in the reactor are then discussed in the following chapter in more detail.

Isotope	Half-life	Main use
Caesium-131	9.7 days	brachytherapy
Caesium-137	30 years	low-intensity sterilisation of blood
Chromium-51	28 days	quantifying gastro-intestinal protein loss
Cobalt-60	5.27 years	brain cancer treatment, sterilising
Dysprosium-165	2 hours	synovectomy treatment of arthritis
Erbium-169	9.4 days	relieving arthritis pain in synovial joints
Holmium-166	26 hours	diagnosis and treatment of liver tumours
Iodine-125	60 days	brachytherapy (prostate and brain)
Iodine-131	8 days	treating thyroid cancer and imaging
Iridium-192	74 days	internal radiotherapy source
Iron-59	46 days	studies of iron metabolism in the spleen
Lead-212	10.6 hours	melanoma, breast cancer and ovarian cancer
Lutetium-177	6.7 days	small (e.g. endocrine) tumours treatment
Molybdenum-99	66 hours	a generator to produce technetium-99m
Palladium-103	17 days	brachytherapy
Phosphorus-32	14 days	treatment of excessive red blood cells count
Potassium-42	12 hours	determination of potassium in blood flow
Radium-223	11.4 days	TAT (targeted alpha therapy) brachytherapy
Rhenium-186	3.8 days	pain relief in bone cancer
Rhenium-188	17 hours	beta irradiation of coronary arteries
Samarium-153	47 hours	prostate and breast cancer treatment
Scandium-47	4.5 days	therapy or diagnosis
Selenium-75	120 days	digestive enzymes production study
Sodium-24	15 hours	studies of electrolytes within the body
Strontium-89	50 days	pain reduction of prostate and bone cancer
Technetium-99m	6 hours	most common radioisotope for diagnosis
Thorium-227	18.7 days	TAT
Xenon-133	5 days	pulmonary (lung) ventilation studies
Ytterbium-169	32 days	cerebrospinal fluid studies in the brain
Ytterbium-177	1.9 hours	progenitor of Lu-177
Yttrium-90	64 hours	brachytherapy

Table 1.1: Reactor-produced radioisotopes[11]

Isotope	Half-life	Main use
Actinium-225	10 days	TAT
Astatine-211	7.2 hours	TAT
Bismuth-213	46 min	TAT
Carbon-11	20 min	positron emitter used in PET
Nitrogen-13	10 min	positron emitter used in PET
Oxygen-15	2 min	positron emitter used in PET
Fluorine-18	110 min	tracer with PET, imaging tumours
Cobalt-57	272 days	marker to estimate organ size
Copper-64	13 hours	cancer therapy, imaging of tumours
Copper-67	2.6 days	beta emitter, used in therapy
Gallium-67	78 hours	tumour imaging locating lesions
Gallium-68	68 min	positron emitter used in PET and PET-CT
Germanium-68	271 days	parent in a generator to produce Ga-68
Indium-111	2.8 days	brain, infection and colon transit studies
Iodine-123	13 hours	diagnosis of thyroid function
Iodine-124	4.2 days	imaging of the thyroid using PET
Krypton-81m	13 sec	pulmonary ventilation imaging
Rubidium-82	1.26 min	PET agent in myocardial perfusion imaging
Strontium-82	25 days	parent in a generator to produce Rb-82
Thallium-201	73 hours	diagnosis of heart conditions and diseases

Table 1.2: Cyclotron-produced radioisotopes[11]

2. Reactor-produced radioisotopes

There are many element isotopes that can be irradiated in a reactor and later used for medicinal purposes. Each radioisotope has different characteristics suitable for different usage. In this chapter some of the most common or promising radioisotopes used in medicine are listed with their respective properties.

2.1 Molybdenum-99

Mo-99 is a very important isotope in nuclear medicine for diagnostic imaging. It decays into Tc-99m which is discussed below. Furthermore, other medical isotopes such as I-131 or Xe-133 are by-products of the fission production of the Mo-99 and its half-life of 65.94 hours[2] is very good concerning the transporting options.

2.1.1 Mo-99 production process

There are 2 main approaches for Mo-99 production:

- **Fission of U-235**

- **Neutron capture by Mo-98**

Majority of the isotope is produced in the fission process using highly enriched (90%) uranium-235 targets. This method includes fabrication of the uranium targets, their irradiation in the reactor and consequent dissolution of the targets and purification of Mo-99.

The target used for Mo-99 production is a material containing uranium-235 that is designed to be irradiated. It needs to satisfy several criteria to be used. First, it has to have defined dimensions to fit into the irradiation position inside the reactor. Second, it must contain a sufficient amount of U-235 to produce the required amount of Mo-99 when it is irradiated. Third, it must have good heat transfer properties to prevent overheating (which could result in target failure) during irradiation. Fourth, the target must provide a barrier to the release of radioactive products, especially fission gases, during and after irradiation. Fifth, the target materials must be compatible with the chemical processing steps that will be used to recover and purify Mo-99 after the target is irradiated. To meet these criteria, targets are fabricated in a wide variety of shapes and compositions. Targets may be shaped as plates, pins, or cylinders.

Irradiation of the targets in a reactor proceeds with thermal neutrons. The fission of the U-235 nucleus produces two but sometimes three lighter nuclei known as fission fragments. Approximately 6% of these fission fragments are Mo-99 atoms[1].

After the irradiation, the targets need to be cooled which is done by using water. Once the cooling period is over the targets are transported to the final facility where they are chemically processed and the Mo-99 is extracted.

In the last 10 years the tendencies are to replace the targets made of HEU (highly-enriched uranium) with LEU (low-enriched uranium max 20% of U-235) targets given the potential missuse of HEU in weapons.

Majority of the radioisotope production at the LVR-15 reactor is the Mo-99 fission production from uranium targets (both HEU and LEU).

Fission production yields are higher than in the case of Mo-98 neutron capture method and that is the reason why the fission method is preferred.

2.2   Technetium-99m

Technetium-99m is the most common radioisotope for nuclear medicine. It offers many advantages with respect to many other radionuclides, due to its very good physical and chemical characteristics. Among other things, the gamma radiation emitted has the appropriate energy to provide a good image while keeping low radiation dose to the patient. Its 6-hour half-life is appropriate (neither too long nor too short) for a medical examination and a good image quality. It also allows the patient to leave shortly after the examination.

It is a decay product of Molybdenum-99 which decays into excited state of technetium via β^- decay. This state is metastable and represents a nuclear isomer of technetium-99. A nuclear isomer is a long-lived excited nuclear state. Eventually, this state de-excites to its ground state by emission of a photon. In case of technetium-99m, the energy of the photon is 140 keV[2]. This is well-detectable by standard medical equipment in hospitals.

2.2.1   Use of Technetium-99m

Technetium-99m is the isotope used for over 80% of all nuclear medicine procedures performed around the world[3]. It is used as a radioactive tracer which is administered to the patient. Depending on the procedure, the Tc-99m is bound to a pharmaceutical that transports it to its required location.

It has a versatile and well-known chemistry. It can easily and quickly be bound to many different chemical compounds, and so can be used to label various molecules of interest for many diagnostic applications. Other isotopes tend to have one or more drawbacks compared to it, like higher radiation dose to the patient, non-ideal gamma energy, lability of the chemical bonds, more difficult production and/or manipulation, logistics (e.g. need of a cyclotron at reasonable distance, due to short half-life) and/or higher cost[3].

Main use of this isotope is imaging and functional analyses of the brain, lungs, liver, myocardium, thyroid, kidneys, skeleton, blood, and tumors. As mentioned before, the de-excitation photons are then detected by the hospital equipment.

2.3   Lutetium-177

Lutetium is another element widely used in medicine. Lutetium-177 decays to stable hafnium (Hf-177) with a half-life of 6.65 days, by emitting beta radiation with a maximum energy of 0.498 MeV and photonic gamma radiation of 0.208 MeV and

0.113 MeV[4]. It can be produced directly in reactors due to its large cross section for neutron capture. Because of the relatively low 2.6% natural abundance, enriched Lu-176 (more than 70%) target material is required for this production route[2]. The specific activity of Lu-177 directly produced from enriched Lu-176 may be limited by the irradiation periods required to minimize the production of the long-lived Lu-177m produced by the competing Lu-176(n,γ)Lu-177m reaction. For these reasons the indirect route provides the highest radiochemically pure Lu-177. As an alternative to the direct production route, Lu-177 can be obtained from ytterbium-177 via β^- decay which is also produced by neutron capture of enriched ytterbium-176.

Radioactive Lu-177 is produced by the Lu-176 (n,γ) Lu-177 reaction in a Lu_2O_3 target and is prepared as $LuCl_3$ in HCl solution. After irradiation, the target is decayed for 3 days for reduction of activity of Lu-176m (half-life 3.66 hours) produced by the side reaction. Target material used must be guaranteed more than 99% for purity of the product by activation analysis. Radioactivity of target in ampoule is measured with a calibrated ionization chamber. Final product is colourless and clear aqueous solution[2].

2.3.1   Use of Lutetium-177

Lutetium-177 is a radioisotope used in nuclear medicine for cancer treatment or it can be reacted with a dotatate (an amino acid peptide) to form radiopharmaceuticals for positron emission tomography (PET) imaging and/or radionuclide therapy.

Lu-177 dotatate is a radioactive medicine that binds itself to a specific part of certain tumor cells, allowing the radiation to enter and destroy those cells. It is administered as an infusion into the vein. Lutetium is relatively new element in medicine that can be used to cure various types of cancer diseases including stomach, pancreas, intestines and prostate cancers. Lu-177 is a radioisotope with promising future. It is expected that the consumption of Lu-177 will increase four times in the near future.

2.4   Holmium-166

Over the years, a broad spectrum of applications of the radionuclide holmium-166 as a medical isotope has been established. The isotope holmium-166 is attractive as it emits high-energy beta radiation which can be used for a therapeutic effect and gamma radiation which can be used for nuclear imaging purposes.

Furthermore, holmium-165 can be visualized by MRI because of its paramagnetic properties and by CT because of its high density. Since holmium-165 has a natural abundance of 100%, the only by-product is metastable holmium-166 and no costly chemical purification steps are necessary for production of nuclear reactor derived holmium-166.

Several compounds labelled with holmium-166 are now used in patients, such Ho-166-labelled microspheres for liver malignancies, Ho-166-labelled chitosan for hepatocellular carcinoma (HCC) and Holmium DOTMP for bone metastases.

Holmium is one of the 15 rare earth elements called lanthanides, a group of elements that has become an established source of radionuclides for nuclear

diagnostic and therapeutic applications. Holmium-166 (Ho-166) can be produced by two methods; neutron activation by (n,γ) irradiation in a nuclear reactor or by neutron activation of dysprosium-164 (Dy-164)[5].

Because holmium-165 (Ho-165 has a natural abundance of 100% and a cross section of 64 b), it can be neutron activated in a relatively short neutron activation time resulting in Ho-166 with a high purity of the isotope. The only by-product is metastable holmium-166 (Ho-166m), many times less than Ho-166. Ho-166m has a half-life of 1200 years and emits beta radiation and a number of gamma rays between 80 keV and 1563 keV. The cross section of the Ho-165(n, γ) Ho-166 reaction is 64 b and the cross section of the Ho-165(n, γ)Ho-166m reaction is around 3.4 b for thermal neutrons[5].

The other production option is via neutron activation of Dy-164 by two neutrons following a (2n,γ) reaction forming dysprosium-166 (Dy-166). The cross section of Dy-164 is extremely high (2650 b). A second neutron irradiation on this unstable nuclide is necessary to result in Dy-166. The radionuclide Dy-166 will decay with a half-life of 81.5 hours into Ho-166, during which two beta particles up to 481 keV are emitted[5].

The high energetic beta particles are responsible for the therapeutic effect and the gamma ray of 80.57 keV can be used for nuclear imaging purposes. Furthermore, Ho-165 can be visualized by MRI because of its paramagnetic properties.

Since the production of Ho-166 has become more and more standardized, the number of clinical applications has been growing and several compounds are now used in patients. Thus, it can be expected that the significance of the use of this radioisotope will continue to grow rapidly.

Apart from the dominant Mo-99 production is Ho-166 the only radioisotope routinely produced at the LVR-15 reactor for nuclear medicine.

2.5   Gadolinium-160

Gadolinium is a colorless or light yellow lustrous metal. Its stable isotopes include Gd-152, Gd-154, Gd-155, Gd-156, Gd-157, Gd-158 and Gd-160 with Gd-158 and Gd-160 having the highest natural abundances of 24.84% and 21.86%[19].

The most important application of this metal is as control rod material for shielding in nuclear power reactors. This is because gadolinium has the highest thermal neutron capture cross section from stable elements (48890 $\pm$ 104 b[18]). Other uses are in thermoelectric generating devices, as a thermoionic emitter, in yttrium-iron garnets in microwave filters to detect low-intensity signals, as an activator in many phosphors, for deoxidation of molten titanium, and as a catalyst.

In nuclear medicine, gadolinium is used as an injectable contrast agent for MRI. The gadolinium used in the MRI contrast agent enhances the quality of MRI pictures by showing a clear distinction of areas of the body where the dye collects showing the doctor possible site of abnormalities. Another use for gadolinium is for production of Tb-161 (described in the following section) from neutron irradiation of gadolinium enriched targets.

2.6 Terbium-161

Terbium is a silvery-white rare earth metal that is malleable, ductile and soft enough to be cut with a knife. It can be recovered from the minerals monazite and bastnaesite by ion exchange and solvent extraction. It is also obtained from euxenite, a complex oxide containing 1% or more of terbium. The metal is usually produced commercially by reducing the anhydrous fluoride or chloride with calcium metal, under a vacuum. It is also possible to produce the metal by the electrolysis of terbium oxide in molten calcium chloride[6]. It is a relatively stable element on air compared with other lanthanides. Naturally occurring terbium is composed of only one stable isotope and that is terbium-159 (Tb-159). Thirty-six radioisotopes have been characterized, with the most stable being Tb-158 with a half-life of 180 years. However, most of the radioactive isotopes of terbium have very short half-lives in order of seconds.

It belongs to lesser used radioisotopes used in treatment. Nevertheless, there have been studies to investigate Tb-161 in combination with PSMA-617 (PSMA stands for prostate-specific membrane antigen) as a potentially more effective therapeutic alternative to Lu-177 combined with PSMA-617 due to the abundant co-emission of conversion and Auger electrons, resulting in an improved absorbed dose profile in radionuclide therapy of prostate cancer[7].

2.7 Iodine-131

Iodine-131 is a radioisotope of iodine with half-life of 8.04 days[2]. It is used in diagnostic and treatment procedures, especially thyroid cancer. These are carried out by oral administration of radioactive iodine in the form of liquid or capsules. Use of I-131 in capsule form simplifies procedures of handling compared to liquid form of I-131. The liquid form of I-131 has higher risk factors such as vaporization, spillage and need for management of higher activity wastes.

Iodine-131 is produced by the neutron irradiation of a TeO_2 target where Te-130 (n, γ)Te-131 reaction takes place and subsequently Te-131 undergoes β^- decay into I-131. The iodine is separated from the target material by dry distillation above 750 ° C and absorbed in dilute sodium hydroxide solution. This is the so-called dry distillation method for separating the I-131 from the target TeO_2. Final product should have an appearance of a colourless clear solution.

2.8 Rhenium-188

Rhenium-188 has a half-life of approximately 17 hours. It is a beta emitter with decay energies of 2.120 MeV (71.1%) and 1.965 MeV (25.6%)[2]. It is sufficient to penetrate and destroy targeted abnormal tissues. In addition, the low-abundant gamma emission of 155 keV is efficient for imaging and for dosimetric calculations. Re-187 (n, γ) Re-188 is its production scheme.

It is a very attractive radioisotope for a variety of therapeutic applications in nuclear medicine, oncology and in cardiology.

Although carrier-free rhenium-188 is conveniently and inexpensively available from the tungsten-188/rhenium-188 generator system, production of tungsten-188

from irradiation of enriched tungsten-186 requires a very high thermal neutron flux, which for practical purposes to provide tungsten-188 with a specific activity over 18.5 MBq/mg must exceed about $5 \cdot 10^{14} n/cm^2.s$. However, neutron activation of enriched rhenium-187 can also provide rhenium-188 with relatively high specific activity. Thus, even research reactors with relatively modest thermal neutron flux can be used to directly produce this important therapeutic radioisotope. Highly enriched rhenium-187 (97%) metallic targets are usually used for production of rhenium-188[2].

2.9   Iridium-192

It is a radioactive isotope of iridium with the half-life of 73.83 days [12]. It mainly decays through β^- decay into Pt-192 (95.13%) or by neutron capture into Os-192 (4.87%)[12] with a photon released subsequently in the process. It is produced in reactors via the scheme Ir-191 (n, γ) Ir-192. As a target material Na_2IrCl_6 may be used[2].

It has its applications in industrial radiography. It is used to capture X-ray images of heavy metal objects. However, its main application is in medicine. It is used in brachytherapy for cancer treatment. Iridium-192 is used in implants that have also medical uses. These implants are manufactured in a form of a wire and are introduced into the target through a catheter. The implant wire is then removed after delivering the required dose. This procedure is very effective at providing localised radiation to tumor site while minimizing the whole body dose for the patient.

2.10   Samarium-153

Samarium-153 is a radioisotope used in radiotherapy for the treatment of pain from bone metastases. It is produced in nuclear reactors, by neutron irradiation of samarium-152. Samarium-153 decays by emitting both beta minus particles and gamma photons. Its half-life is 46.27 hours[2]. With a gamma decay energy of 0.103 MeV occurring in 32.2% [2] of the time, it can be used for imaging as well.

The target material is made of Sm_2O_3 with purity exceeding 99%. The carrier molecule used is the ethylene diamine tetramethylene phosphonate acid (EDTMP) and the final product for use is a limpid, colourless water solution [2]. It is injected into the vein and distributed throughout the body, where it is preferentially absorbed in areas where cancer has invaded the bone.It is commonly used in lung cancer, prostate cancer, breast cancer, and osteosarcoma.

2.11   Mercury-203

This is a radioisotope with a half-life of 46.61 days [2] which decays through β^- to an excited state of Tl-203 which in turn undergoes γ decay. Target material is a HgO powder.

It was formerly used as a diagnostic aid in determination of renal function (renal cell carcinoma is one of the most common types of kidney cancer). It is

a radioactive imaging agent used for brain scans contained in a chemical called neohydrin-203.

2.12 Xenon-133

Xenon-133 is formed during the fission reaction of U-235 among the other fission products when uranium-235 contained targets are irradiated by thermal neutrons. Xenon-133 is a gas. Then it is purified from iodine radioactive isotopes and other volatile elements. Its half-life is 5.243 days and it decays through β^- emission[2].

In medicine, it is mainly used for lung perfusion tests, which is a form of diagnostics for lung problems. The patients inhale the gas, which then spreads through the lungs. Using a gamma camera, an image of the lungs can be created to evaluate the condition of the lungs. The distribution of xenon in the lungs can be measured. This image can help to determine how well the lungs are functioning. The patient does not suffer from the gas, as the xenon leaves the body when exhaling.

This method is used in North America only, in Europe, different methods are commonly used. In 2019, approximately 250,000 patients have been scanned using this technique, primarily in the United States[13].

3. LVR-15 reactor

3.1 Description

Reactor LVR-15 is a research reactor located at Řež near Prague. It started its operation in 1957 in the Institute of Nuclear Physics of Czechoslovak Academy of Sciences. The reactor's thermal power was 2 MW. It was operational until 1987. It underwent a reconstruction during 1988 and 1989 with the goal of a power uprate and an increase of safety. The reactor has been operating on full power since 1995.

Its strategic location in the middle of Europe and its high irradiation parameters make the reactor a very good choice for being a central-European supplier of radioisotopes.

LVR-15 is a light water tank-type research reactor placed in a stainless steel vessel under a shielding cover. Its operational power is 10 MW. Reactor operations run in campaigns. Usually the campaign lasts for 30 days, followed by an outage lasting for 20 days for maintenance and fuel reloading. Technical parameters of the reactor are listed below.

Diameter	2300 mm
Height	5760 mm
Wall thickness	15 mm
Bottom thickness	20 mm
Water volume in the vessel	22 m^3
Weight (without water)	7900 kg

Table 3.1: Reactor Vessel

Maximal thermal power	10 MW
Maximal thermal neutron flux	$10^{14} n.cm^{-2}.s^{-1}$
Pressure	atmospheric
Maximal temperature	56 ° C
Control rods	12
Absorber	B_4C

Table 3.2: Operational parameters

Type	IRT-4M – tube sandwich-like type
Active length	600 mm
Cladding	Al
Fuel core	$UO_2 + Al$
Enrichment	19.75% U-235

Table 3.3: Fuel
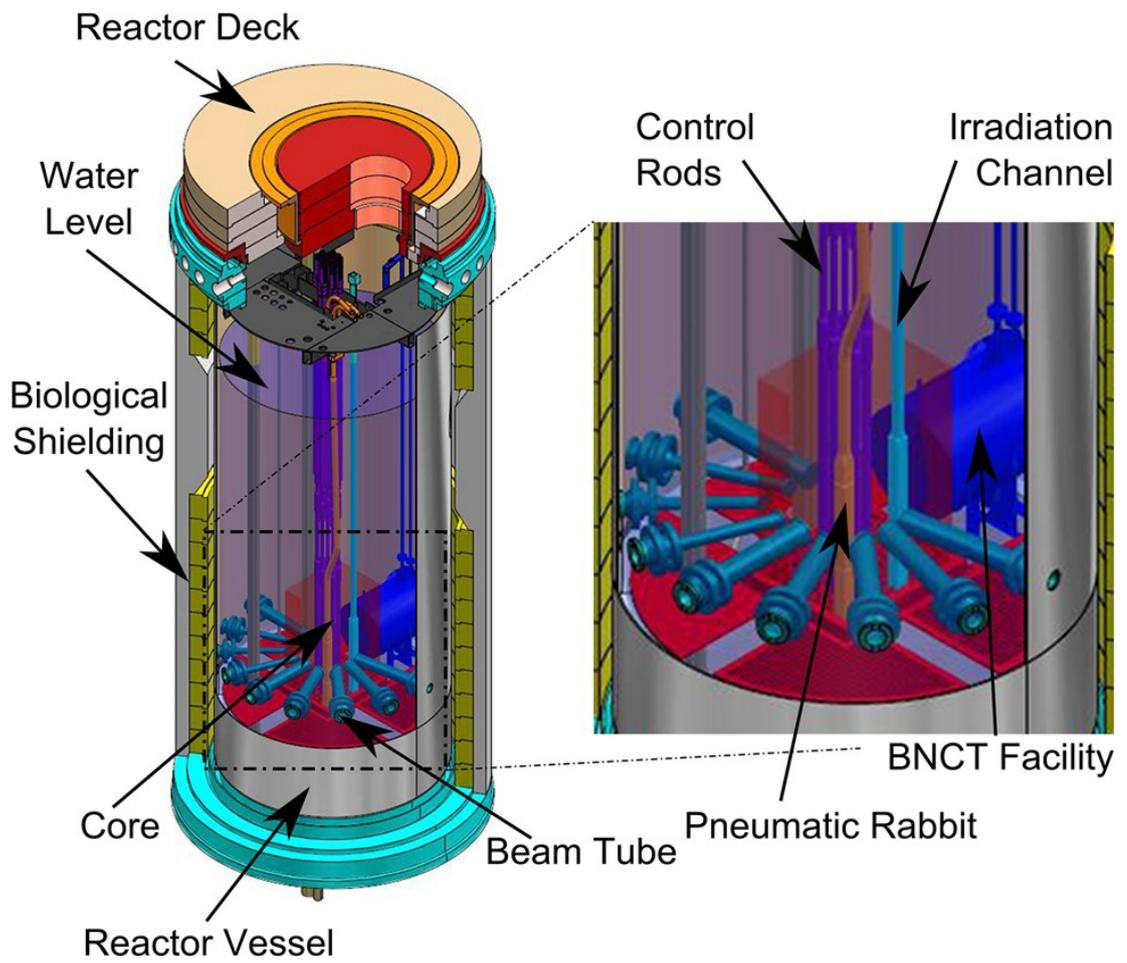
Figure 3.1: LVR-15 reactor vessel cross-section[10]

Figure 3.2: Drilled irradiation cask (upper) with a special purpose sample holder (lower)[10]

3.2 Utilization and services

The reactor is used for research and variety of services including

- Neutron activation analysis (used to determine materials composition)

- Development and production of new radioisotopes

- Production of medical and industrial radioisotopes

- Irradiation services

- Neutron capture therapy (irradiation of patients with glioblastoma-type brain tumours)

- Neutron optics and neutron topography

- Irradiation experiments focused on material behaviour research and influence of radiation and chemical parameters

- Dose rate measurements at a defined distance from the ionizing radiation sources

- Neutron measurement of structures and textures at room and helium temperatures

- Deep neutron profiling and the study of immediate gamma radiation from neutron radiation capture

3.3 Production of radioisotopes

As it was mentioned above, the reactor is used in development and production of radioisotopes. Main isotopes produced in the reactor are Mo-99 (fission production), Sm-153, Ir-192, Lu-177, Hg-203 and Xe-127[10]. Apart from Mo-99 the other isotope production represent a one-time experimental production.

Prepared target materials are mostly sealed into quartz ampoules, compact matrix of the irradiation target or a special sample holders. In case of short time irradiation, not exceeding 1 h, encapsulation into polyethylene plastic ampoules is also used for on-site prepared samples. The ampoule is then sealed into an aluminium cask 3.2 (upper) compatible with handling tools for loading and unloading of the cask. The cask might be either drilled or watertight with respect to the requirement for ampoule vicinity and cooling. For certain applications, special purpose casks may be also developed. For the measurement, standard drilled irradiation capsules 3.2, filled with water during irradiation, are used[10].

Special ampoules are used for the irradiation of samples in the form of powder or in the case of fragile samples which would be otherwise destroyed during the process. For the purposes of this particular experiment discussed in more detail in the following chapters, the detector samples were wrapped in the aluminium foil and were placed in a watertight cask.

During the production process, the cask with the target material is lowered down into the irradiation position. After the irradiation, the cask is taken out of

the irradiation channel and relocated into the hot cells which are situated under the reactor. Here the irradiated target material with the required radioisotope is relocated from the cask to a transporting container and shipped to the final processing companies or hospitals (see figure 3.3).

A hot cell is a large volume shielded enclosure in which operations with radioactive isotopes are carried out[2]. Hot cells are used in both the nuclear-energy and the nuclear-medicines industries. They protect individuals from radioactive isotopes by providing a safe containment box in which they can control and manipulate the equipment required. A hot cell is shown in the figure 3.4.

A layout of the reactor is shown in the figure 3.5. To help with the orientation, a sort of coordinate system resembling a chess board is used with positions being assigned a number (1 to 10) in the vertical direction and a letter (A to H) in the horizontal direction. Two positions marked with a red cross (H5 and H6) represent the positions where the irradiations regarding the radioisotope production take place. From all of the available positions, H5 and H6 are the most favourable ones.
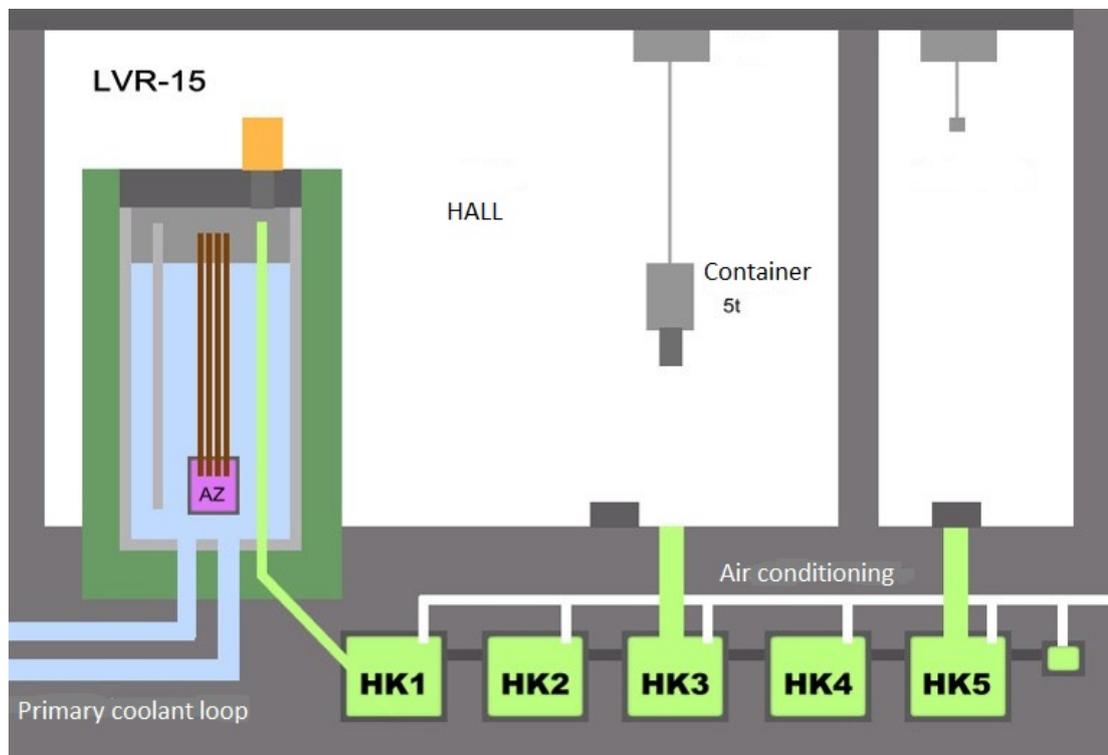
Figure 3.3: LVR-15 reactor hall[21]

Figure 3.4: Hot cell [21]
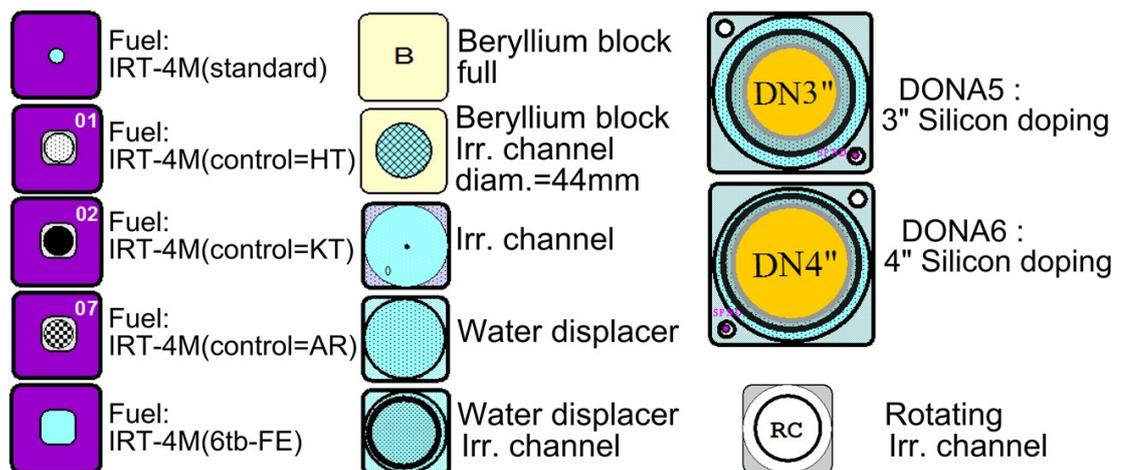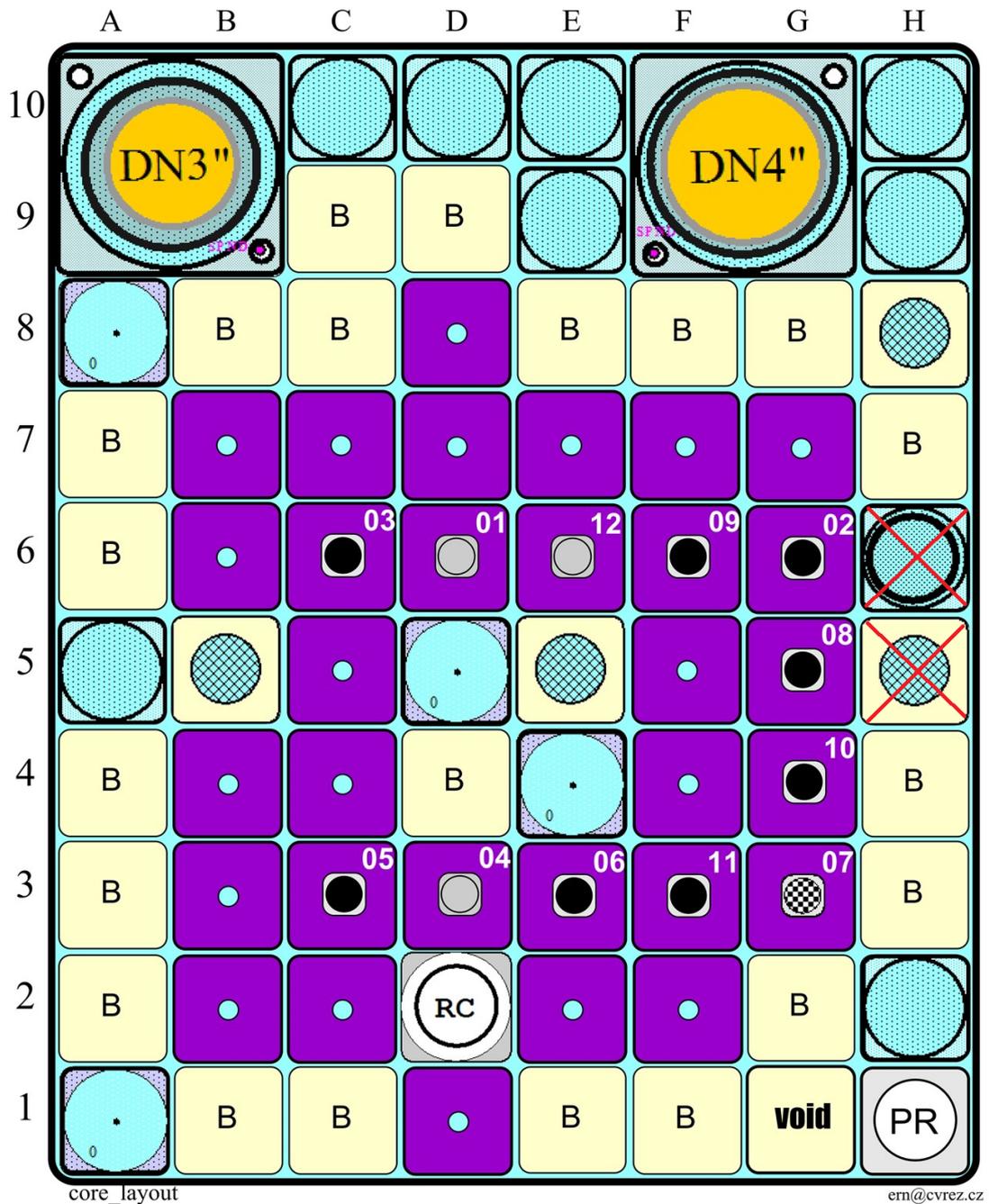
Figure 3.5: Layout of the LVR-15 reactor[10]

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4. Activation analysis

Activation analysis methods are methods based on activation of the sample and subsequent measurements of the quantities that are of interest. Samples can be bombarded by neutrons, protons, photons or other particles. The name of the activation method is derived from the bombarding particles, so if the sample is bombarded by neutrons, then it is called neutron activation.

Most of the radioactive isotopes emit γ-radiation of different intensities and energies. This radiation can be detected and analysed resulting in a γ-spectrum. This method is called γ-spectrometry. It is used in determining the identity of the γ-emitter and/or the intensity if the radiation. This is useful because every nuclide emitting γ-radiation has a specific spectrum.

For clarity, let us make a distinction between spectroscopy and spectrometry. In general, spectroscopy is the science of studying the interaction between matter and radiated energy while spectrometry is the method used to acquire a quantitative measurement of the spectrum. Spectroscopy (scopy means observation) does not generate any results. It is the theoretical approach of science. Spectrometry (metry means measurement) is the practical application where the results are generated. It is the measurement of the intensity of the radiation using an electronic device[14].

4.1 Neutron activation

The reactor is used for irradiation of a sample in order to create a radioisotope that can be used for various purposes. Neutrons are used as projectile particles in this case. Once the sample is placed in the reactor, neutrons can interact with the sample, most commonly via elastic scattering, inelastic scattering or can be absorbed. If the reaction proceeds via elastic scattering, the total kinetic energy of the system is conserved. In the case of inelastic scattering, the neutron interacts with the nucleus and the kinetic energy of the system is changed. If the neutron is captured by the nucleus, a new, heavier nuclide is formed. The nucleus is in an excited state which may be unstable. This then results in an emission of radiation bringing the nucleus to the state with lower energy.

When the sample is placed in the flux of neutrons, neutron capture occurs and a subsequent γ-radiation is emitted. Detection of this radiation is the subject of γ-spectrometry. The entire process is depicted and can be seen in figure 4.1.

Neutron activation can be initiated by slow neutrons as well as with fast neutrons. However, the most important are reactions with thermal neutrons because capture of these neutrons leads to activation in almost all stable elements. This is because these reactions have a large activation cross section. An example of neutron capture cross section depending on incident neutron energy is shown in fig.4.3.

Possible disadvantage using neutrons for activation is formation of a radioactive isotope of the same element that cannot be chemically separated from the sample. However, during the short irradiation time with low fluence of neutrons it is diluted by stable isotopes of the substance which results in low specific activity activity.
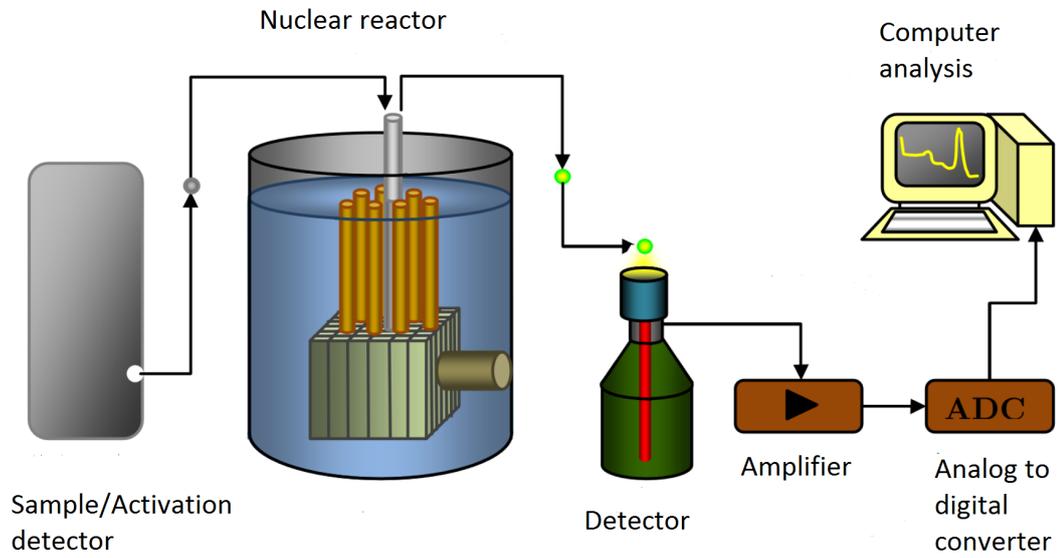
Figure 4.1: Activation analysis and measurement process [15]
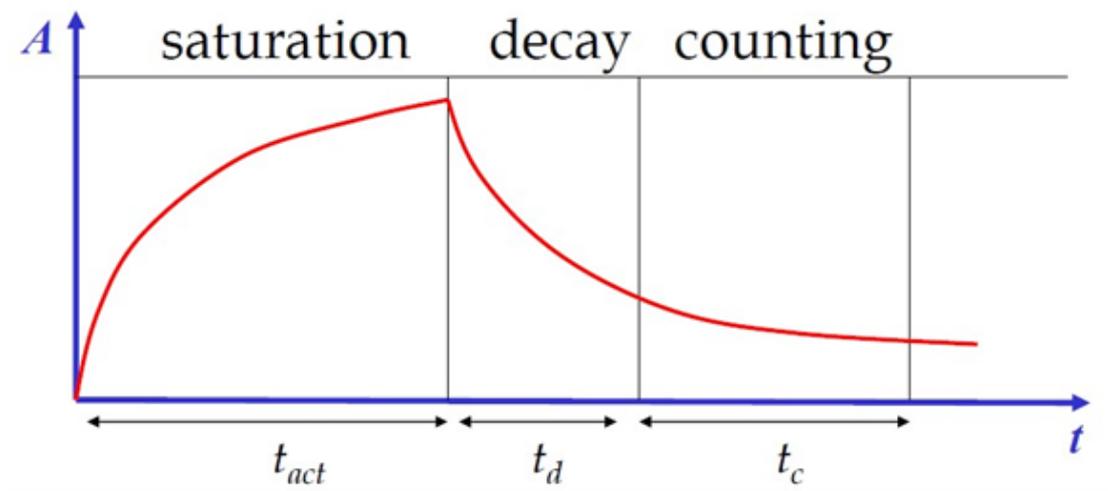
Figure 4.2: Activation and decay for the irradiation process
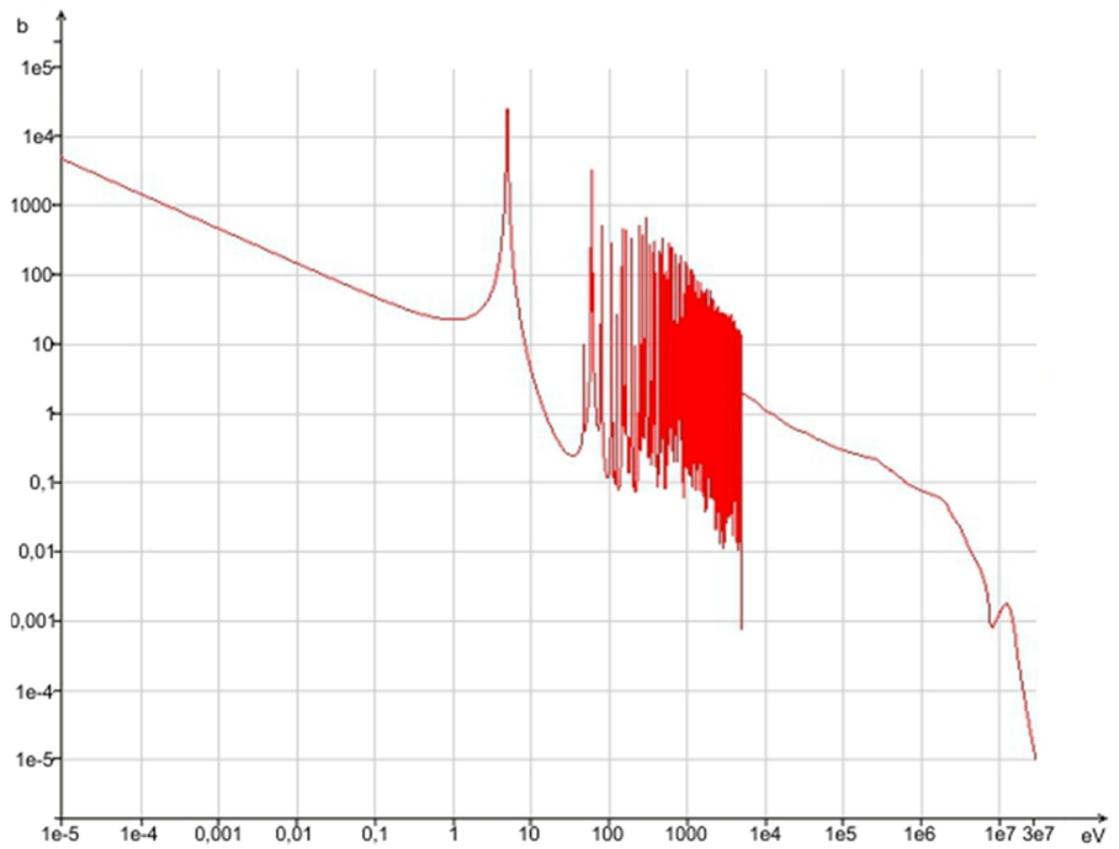
Figure 4.3: Neutron capture cross section of Au-197 nuclei as a function of neutron energy [22]

The activation analysis consists of the following steps[15]:

- Sample activation by neutrons (t_a-activation time)

- Transport of the irradiated sample from the reactor to the spectroscopic laboratory

- Cooling and decay (including the time of transport)of the sample(t_c-cooling time).

- Measurement of the spectrum (during the cooling time,t_m-measurement time)

- Quantitative analysis of the measured spectrum

4.2 Methods of activation analysis

Absolute method: This method allows to calculate the value of activity induced in the sample according to the following formula

$$A = N_0 f \phi \sigma_a (1 - e^{-\lambda t_a}) e^{-\lambda t_d} \tag{4.1}$$

where N_0 is the number of constituent particles in the substance, f is relative isotopic abundance in the nature, λ is the decay constant, t_a, t_d are the activation and decay times respectively , σ_a is the activation cross section and ϕ represents the density of neutron flux. Due to the difficulties in evaluating the absolute activities using the formula for this method, the so called relative(comparison) method is used more often.

Relative (comparison) method: This method is based on comparison of the activity of the sample that is being measured and a prepared standard sample of known properties that was irradiated by particles from the same source and in the identical conditions as the measured sample. The formula used in this case is the following

$$m_s = m_{st} \frac{A_s}{A_{st}} \tag{4.2}$$

where m_s, A_s are the mass and activity of the sample,m_{st}, A_{st} are the mass and the activity of the prepared standard.

4.3 Activation detectors

Neutron detection with activation detectors is a common method. Advantages of these detectors include small dimensions, low price, large energetic range of detection and insensitivity to γ-radiation. Choice of the detectors for the experiment depends on the quantity of interest(e.g. flux or spectrum measurement), but also on physical and chemical properties of the detector such as mechanical durability, chemical purity, temperature sensitivity. In terms of nuclear properties decay half-time, energy of emitted γ-radiation, occurrence of possible secondary reactions and cross section of the primary detection reaction. Activation detectors are usually chemically pure substances or they can be enriched by certain isotopes of given element according to the requirements of the experiment.

Activation methods are based on the reaction of neutrons with the nuclei of the target material. In a constant neutron flux $\phi(E)$, the time-dependent relation for radionuclide production and decay is[16]

$$\frac{dN}{dt} = N_0\phi(E)\sigma_a(E) - \lambda N(t) = N_0 R_R - \lambda N(t) \qquad (4.3)$$

where N_0 is the number activation detector's nuclei before the irradiation, $\phi(E)$ is the neutron flux, $\sigma_a(E)$ is the cross section of the specific activation reaction, λ is the decay constant and R_R is the reaction rate per one nucleus of the detector. The first term of the equation 4.3 represents the production of the radioactive isotope in the volume of the detector per unit time and the second term represents its decay.

Multiplying the reaction rate with the number of nuclei in the detector, saturated activity A_{sat} can be obtained. This is given in the equation 4.4 [16].

$$A_{sat} = N_0 R_R = N_0\phi(E)\sigma_a(E) \qquad (4.4)$$

The figure 4.2 shows the qualitative behaviour of activity depending on time during the irradiation and decay in the sample.

4.3.1 Self-shielding

When using activation detectors to get the desired radioisotope, neutron self-shielding phenomenon that is present in nuclear reactors needs to be taken into account here as well.

The amount of absorption reactions can be reduced despite the unchanged microscopic cross section of the material. This is known as resonance self-shielding. There are two types of self-shielding: **Energy self-shielding** and **spatial self-shielding**.

In a nuclear reactor, there is a characteristic neutron energy spectrum that, in case of thermal reactors, covers also the resonance region of various materials. The vicinity of the resonance causes an increase in the neutron absorption probability, when a neutron has an energy near the resonance. This results in a reduction of the effective absorption per nucleus due to the depression of the energy dependent flux $\phi(E)$ near the resonance in comparison to the flat flux. At energies just below the resonance, where $\sigma_a(E)$ becomes small again, the neutron flux reaches almost to the same value just above the resonance. This reduction in the energy dependent neutron flux near the resonance energy is known as energy self-shielding.

The basic fuel element of light water reactors is a fuel rod which contains fuel pellets made of uranium dioxide. The reactor contains fuel assemblies in which the fuel rods made of the fuel pellets are stored. The neutron moderator surrounds these fuel rods. The spatial self-shielding is phenomenon primarily connected with this heterogeneity of the reactor core.

The fission neutrons are born in the fuel, but they are primarily moderated (slowed down) in the moderator. Thermal and epithermal neutrons return back into the fuel to be absorbed and to cause further fission. With the aid of energy self-shielding, neutrons are more likely absorbed by the nuclei near the fuel surface. In this case the surface layers of the fuel geometrically shield the inner

layers from neutron flux, leading to a relatively lower neutron flux inside the fuel rod. For neutron energies near resonances (energy self-shielding) the depression of the energy dependent flux is dramatical[17].

A very similar situation occurs in the case of activation detectors made of materials with high cross section for neutron absorption (see table 6.3). Neutrons are absorbed by the nuclei near the detector surface. The surface layers of the detector material shield the inner layers from neutron flux, leading to a relatively lower neutron flux inside the whole volume of the detector. This can then result in a lower total activity of the particular detector.

5. Description of the experiment

Experimental part of this thesis was focused on production capabilities of medical radioisotopes at the LVR-15 reactor. The whole procedure consisted of several steps. Firstly, a number of isotopes that were thought to be suitable for radioisotope production were chosen. These were mainly objects of research in several projects currently running at the LVR-15. The isotopes had to be prepared in a form of an activation detector for the irradiation procedure in the reactor. Before the irradiation itself a simulation of given conditions in the reactor was calculated to give us a sense of possible resulting activities for the samples. Two software programs were used for the simulations of activities of the nuclides and neutron spectrum in the reactor. These are briefly described in the following sections of the chapter as well as the experiment and the nuclides that were used.

The next step is the irradiation procedure itself where the detectors are irradiated under given conditions (see table 6.1). Their activities are then measured and compared with calculations that take into account the conditions of the experiment. If the measured and calculated activities do not deviate from each other significantly, then it can be concluded that these activity calculations represent a good method for activity predictions for radioisotope production and other experiments.

The final step is the evaluation of the isotope production capabilities of the LVR-15 research reactor for medical radioisotopes in the form of chemical compounds that are delivered to the patients or the final processors. The entire process is also shown in the 5.1 flowchart for easier imagination.

Calculation   Experiment

Older neutron spectra for the LVR-15 irr. position

Chemical composition determinationof the detector material

Fispact calculation estimate for activation detectors activities

Production of activation detectors

Weight measurements

General MCNP model of the LVR-15 reactor core

Irradiation planning

MCNP model adjustments and neutron spectra calculation

Irradiation of the activation detectors

Activity measurements

Fispact calculation of activities for activation detectors after irradiation

Evaluation of the results for the activated radioisotopes

Comparison of measured and calculated activities

Validated method of activity calculation

Determination of chemical composition of target substances

Fispact activity calculation of examined radioisotopes

Validated radioisotope production capabilities at the LVR-15 reactor

Figure 5.1: validation process for radioisotope production capabilities at the LVR-15 reactor

5.1   FISPACT-II

FISPACT-II is an enhanced multiphysics, inventory and source-term code system providing a wide variety of advanced, predictive, spectral and temporal simulation methods employing the most up-to-date and complete nuclear data forms for both neutron and charged-particle interactions. FISPACT-II has been developed and is maintained by the United Kingdom Atomic Energy Authority at Culham.

Fispact-II is a practical activation-transmutation engineering prediction tool. The four principal tasks that it undertakes are[8]:

1. extraction, reduction and storage of nuclear and radiological data from the ENDF or legacy EAF library files

2. construction and solution of the rate equations to determine the time evolution of the inventory in response to different irradiation scenarios. These scenarios include:

    * a cooling-only calculation

    * a single irradiation pulse followed by cooling

    * multiple irradiation pulses where only flux amplitudes change, followed by cooling

    * multi-step irradiation where flux amplitude, flux spectra and cross-sections may change, followed by cooling

3. computation and output of derived radiological quantities

4. subsidiary calculations to identify the key reactions and decays, and to assess the quality of the predictions. The four main subsidiary items are

    * pathways analysis

    * uncertainty calculations from pathways

    * reduced model calculations

    * Monte-Carlo sensitivity and uncertainty calculations

In the experiment of this thesis, irradiation and cooling calculations were simulated.

5.2   MCNP

MCNP is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. Specific areas of application include[9]:

* Reactor design

* Nuclear criticality safety

* Nuclear safeguards

* Medical physics, especially proton and neutron therapy

- Design of accelerator spallation targets, particularly for neutron scattering facilities

- Investigations for accelerator isotope production and destruction programs, including the transmutation of nuclear waste

- Research into accelerator-driven energy sources

- Accelerator based imaging technology such as neutron and proton radiography

- Detection technology using charged particles via active interrogation

- Design of shielding in accelerator facilities

- Activation of accelerator components and surrounding groundwater and air

- High-energy dosimetry and neutron detection

- Investigations of cosmic-ray radiation backgrounds and shielding for high altitude aircraft and spacecraft

- Single-event upset in semiconductors from cosmic rays in spacecraft or from the neutron component on the earth's surface

- Analysis of cosmo-chemistry experiments, such as Mars Odyssey

- Charged-particle propulsion concepts for spaceflight

- Investigation of fully coupled neutron and charged-particle transport for lower-energy applications

- Transmutation, activation, and burnup in reactor and other systems

- Nuclear material detection

- Design of neutrino experiments

5.2.1 Structure of the input

MCNP uses an input text-file which consists of so-called blocks. These blocks contain information about

- the geometry specification,

- the description of problem materials and selection of cross-section evaluations,

- the location and characteristics of the particle source,

- the type of answers or tallies desired, and

- any variance reduction techniques used to improve efficiency.

5.2.2 Output

After the calculation MCNP creates an output file with results based on parameters of the calculation. Given that the MCNP is a statistical code, it also provides the statistical uncertainty. This then allows to asses the the reliability of the simulation. This uncertainty is affected by the number of simulated particles, distribution etc[16]. Typically, the goal is to get the statistical error at the end of run is below 10%.

5.3 Experiment simulations

Table 5.1 gives nuclides in the form of detectors that were chosen for the experiment. They are also shown in the fig. 5.2.

The iron and gold detectors were chosen as reference detectors. Detectors from these materials are used in radiation dosimetry of reactors. The gold detector consisted of an aluminium matrix containing 1% of gold due to the self-shielding phenomenon. Comparison of calculated activities and measured activities of these detectors should determine whether the calculation method can be used for valid prediction of experimental activities. The rest of the detector materials were chosen for specific radionuclide production for medicinal purposes. Ho-166 and Mo-99 are routinely produced, whereas the production of others is only experimental. Isotopes of lutetium and rhenium are used directly in treatment. Irradiation of ytterbium and molybdenum leads to their activation and their subsequent decay produces lutetium and technetium. Gadolinium is used also in order to obtain the Tb-161 isotope production capabilities. This isotope has a potentially promising future in nuclear medicine. More details on use of each of these radionuclides are in chapter two.

Using FISPACT-II software, irradiation of the samples from 5.1 in the reactor was simulated and activities for every produced radioisotope were calculated. Figures show the activities of the samples as a function of irradiation time and also the behaviour of activities depending on cooling time after specific time of irradiation i.e. the decay of the irradiated samples with no more radiation present. Specific irradiation times for activity calculations were 1 hour, 6 hours, 1 day, 3 days and 7 days. These calculational estimates are important for the planning of the irradiation time for the experiment. The resulting activities should not be very high (ideally up to 1 MBq) so that the detection apparatus is not over-saturated and safe handling from the radiation protection point of view is ensured. A summary table 5.5 shows the activities for given detectors after each time interval. Figures 5.7,5.8 and 5.9 show the corresponding plots. Specific activities after 1 hour of irradiation and subsequent cooling are given in the 5.6 table with the corresponding plots depicted in figures 5.10,5.11 and 5.12. It can be seen from these plots that the isotopes with shorter half-lives reach a state of saturated activity after a few hours or days of irradiation whereas the isotopes with longer half-lives tend to increase their activity in a linear fashion even after a week of irradiation.

These results can be compared with the activities obtained from the experiment. To perform the simulation in FISPACT-II a value of neutron spectrum representing the neutron conditions in the actual irradiation position must be

Element	Label	Weight[mg]
Fe	1	7.48
Au(1%)	2	2.82
Mo	3	10.27
Ho	4	27.31
Yb	5	21.18
Gd	6	20.97
Lu	7	3.57
Re	8	20.88

Table 5.1: Samples used for the experiment

specified. The exact value used in the calculation was the value in the same position in the reactor during a period in the previous campaign. The specific irradiation and decay times mentioned in the previous paragraph illustrate possible outcomes for activity calculation and predictions for future experiments.

The next step of the experimental validation is to calculate the neutron spectrum using the MCNP software for given reactor conditions during the time of the experiment. The resulting neutron spectrum is then used in the FISPACT-II software for the determination of the activities.

Simulations of the irradiation experiments in the reactor were carried out and the resulting neutron spectra in the positions H6 (where the samples were actually placed and irradiated) and H5 (for possible future reference as this position can also be used for irradiation experiments) are given in the table 5.2. The plot depicting the neutron spectra in the H5 and H6 positions as a function of neutron energy is also shown in figure 5.5. Further, the same plot with energy axis in logarithmic scale is shown in figure 5.6. In this case the spectra in the lower neutron energy interval is visible. A dominant peak in the figure 5.5 can be seen around the energy of about 2 MeV. This is because neutrons created from fission have maximum energy around 2 MeV. Other peaks in the thermal neutron energy interval are visible in figure 5.6. This is caused by neutron moderation that takes place in the reactor during the reaction and it increases the probability of the capture. Table 5.3 gives fluxes in the thermal neutron energy region and in table 5.4 total neutron fluxes at the H5 and H6 positions are also given.

MCNP allows to plot a geometrical model of the environment in which the simulations take place. Figure 5.4 shows a model of the active zone of the reactor from the day of the experiment (November 5, 2020).

After the irradiation, the samples are analysed using γ-spectrometry that allows us to measure the activities of the targeted radionuclides.
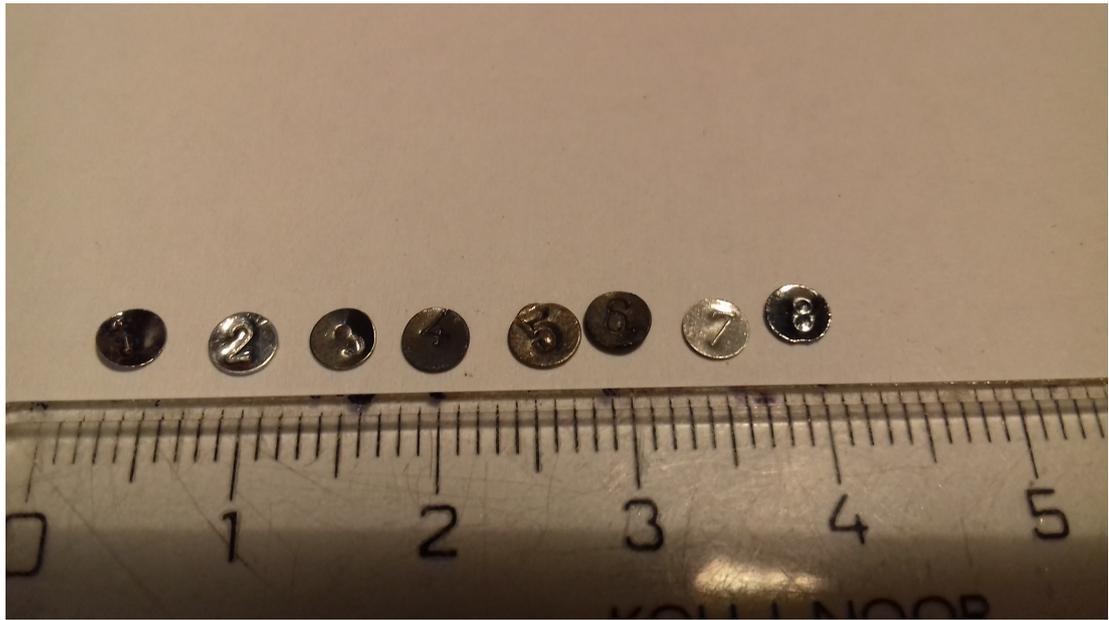
Figure 5.2: Activation detectors used in the experiment
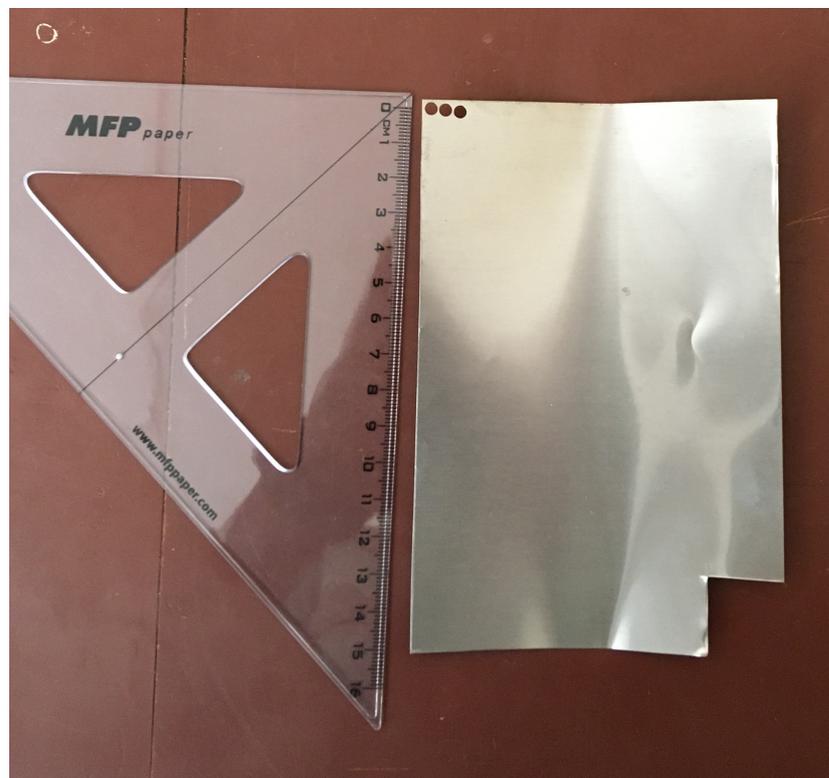
Figure 5.3: A sheet used for the preparation of the activation detectors

E[eV]	Neutron spectrum in H6 $[n.cm^{-2}.s^{-1}]$	Neutron spectrum in H5 $[n.cm^{-2}.s^{-1}]$
5.00E-03	1.11E+12	1.19E+12
1.00E-02	2.79E+12	3.04E+12
1.50E-02	3.93E+12	4.32E+12
2.00E-02	4.55E+12	4.91E+12
2.50E-02	4.83E+12	5.25E+12
3.00E-02	4.78E+12	5.21E+12
3.50E-02	4.76E+12	5.19E+12
4.20E-02	6.22E+12	6.77E+12
5.00E-02	6.43E+12	7.06E+12
5.80E-02	5.52E+12	6.09E+12
6.70E-02	5.22E+12	5.81E+12
8.00E-02	5.92E+12	6.40E+12
1.00E-01	6.24E+12	6.94E+12
1.40E-01	6.07E+12	6.74E+12
1.80E-01	2.38E+12	2.76E+12
2.20E-01	1.21E+12	1.43E+12
2.50E-01	5.92E+11	6.91E+11
2.80E-01	4.71E+11	5.61E+11
3.00E-01	2.81E+11	3.39E+11
3.20E-01	2.51E+11	2.93E+11
3.50E-01	3.34E+11	3.92E+11
4.00E-01	5.17E+11	5.97E+11
5.00E-01	8.51E+11	9.62E+11
6.25E-01	8.11E+11	9.39E+11
7.80E-01	7.81E+11	9.33E+11
8.50E-01	3.05E+11	3.54E+11
9.10E-01	2.37E+11	2.64E+11
9.50E-01	1.45E+11	1.71E+11
9.72E-01	7.76E+10	9.34E+10
9.96E-01	8.75E+10	9.94E+10
1.02E+00	7.83E+10	9.48E+10
1.05E+00	8.04E+10	9.27E+10
1.07E+00	7.97E+10	1.01E+11
1.10E+00	7.76E+10	1.07E+11
1.12E+00	8.65E+10	8.97E+10
1.15E+00	7.72E+10	9.02E+10
1.30E+00	4.29E+11	4.69E+11
1.50E+00	4.66E+11	5.92E+11
2.10E+00	1.10E+12	1.32E+12
2.60E+00	6.85E+11	8.45E+11
3.30E+00	7.69E+11	9.34E+11
4.00E+00	6.21E+11	7.33E+11
9.88E+00	2.94E+12	3.38E+12
1.60E+01	1.54E+12	1.83E+12
2.77E+01	1.77E+12	2.04E+12
4.81E+01	1.77E+12	2.06E+12
7.55E+01	1.49E+12	1.77E+12

E[eV]	Neutron spectrum in H6 $[n.cm^{-2}.s^{-1}]$	Neutron spectrum in H5 $[n.cm^{-2}.s^{-1}]$
1.49E+02	2.23E+12	2.54E+12
3.67E+02	2.94E+12	3.42E+12
9.07E+02	3.01E+12	3.43E+12
1.43E+03	1.48E+12	1.71E+12
2.24E+03	1.49E+12	1.75E+12
3.52E+03	1.51E+12	1.74E+12
5.53E+03	1.50E+12	1.74E+12
9.12E+03	1.69E+12	1.96E+12
1.50E+04	1.72E+12	1.97E+12
2.48E+04	1.86E+12	2.05E+12
4.09E+04	1.92E+12	2.20E+12
6.73E+04	2.11E+12	2.39E+12
1.00E+05	1.85E+12	2.13E+12
1.83E+05	3.38E+12	3.86E+12
3.03E+05	3.60E+12	4.07E+12
5.00E+05	4.39E+12	4.77E+12
8.21E+05	5.64E+12	5.59E+12
1.00E+06	2.25E+12	2.30E+12
2.23E+06	1.06E+13	1.08E+13
3.68E+06	5.52E+12	4.80E+12
6.07E+06	2.61E+12	2.42E+12
1.00E+07	6.05E+11	5.54E+11
2.00E+07	3.54E+10	3.51E+10

Table 5.2: Neutron spectra in H5 and H6 positions

E[eV]	Neutron spectrum in H6 $[n.cm^{-2}.s^{-1}]$	Neutron spectrum in H5 $[n.cm^{-2}.s^{-1}]$
5.00E-03	1.11E+12	1.19E+12
1.00E-02	2.79E+12	3.04E+12
1.50E-02	3.93E+12	4.32E+12
2.00E-02	4.55E+12	4.91E+12
2.50E-02	4.83E+12	5.25E+12
3.00E-02	4.78E+12	5.21E+12
3.50E-02	4.76E+12	5.19E+12
4.20E-02	6.22E+12	6.77E+12
5.00E-02	6.43E+12	7.06E+12
5.80E-02	5.52E+12	6.09E+12
6.70E-02	5.22E+12	5.81E+12
8.00E-02	5.92E+12	6.40E+12
1.00E-01	6.24E+12	6.94E+12
1.40E-01	6.07E+12	6.74E+12
1.80E-01	2.38E+12	2.76E+12
2.20E-01	1.21E+12	1.43E+12
2.50E-01	5.92E+11	6.91E+11
2.80E-01	4.71E+11	5.61E+11
3.00E-01	2.81E+11	3.39E+11
3.20E-01	2.51E+11	2.93E+11
3.50E-01	3.34E+11	3.92E+11
4.00E-01	5.17E+11	5.97E+11
5.00E-01	8.51E+11	9.62E+11
Total flux of thermal neutrons	7.53E+13	8.29E+13

Table 5.3: Thermal neutron fluxes

Total neutron flux	H6 $[n.cm^{-2}.s^{-1}]$	H5 $[n.cm^{-2}.s^{-1}]$
	1.55736E+14	1.70627E+14

Table 5.4: Total neutron fluxes
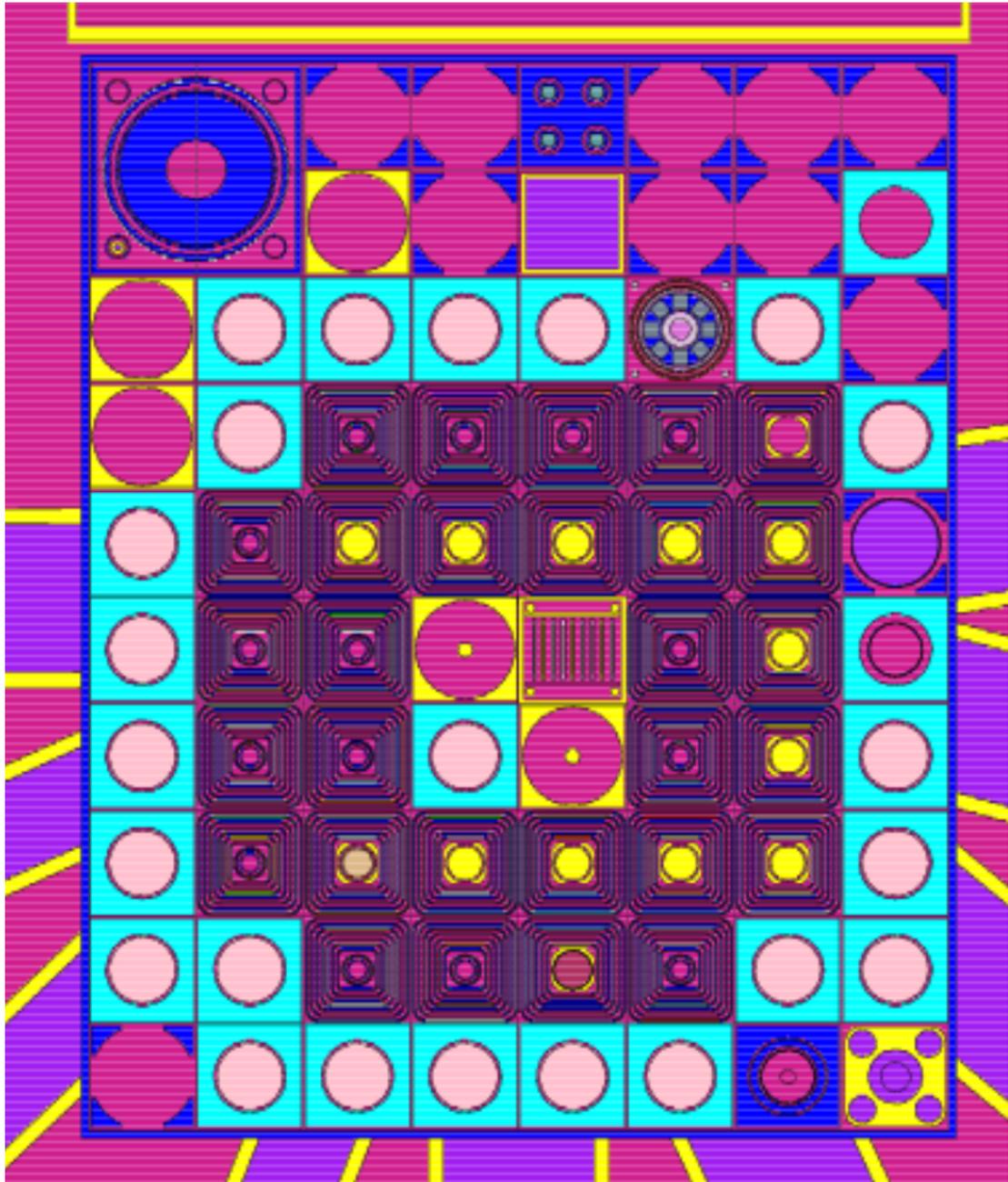
Figure 5.4: MCNP geometrical model of the reactor active zone from November 5,2020
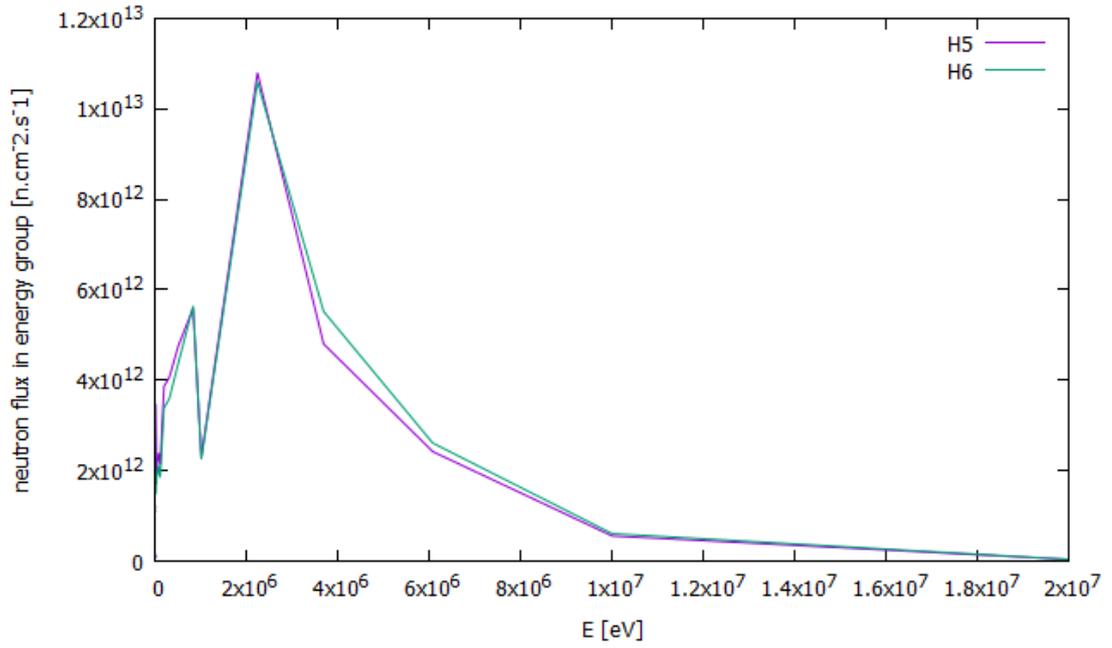
Figure 5.5: Energy beam based neutron spectra in H5 and H6 positions
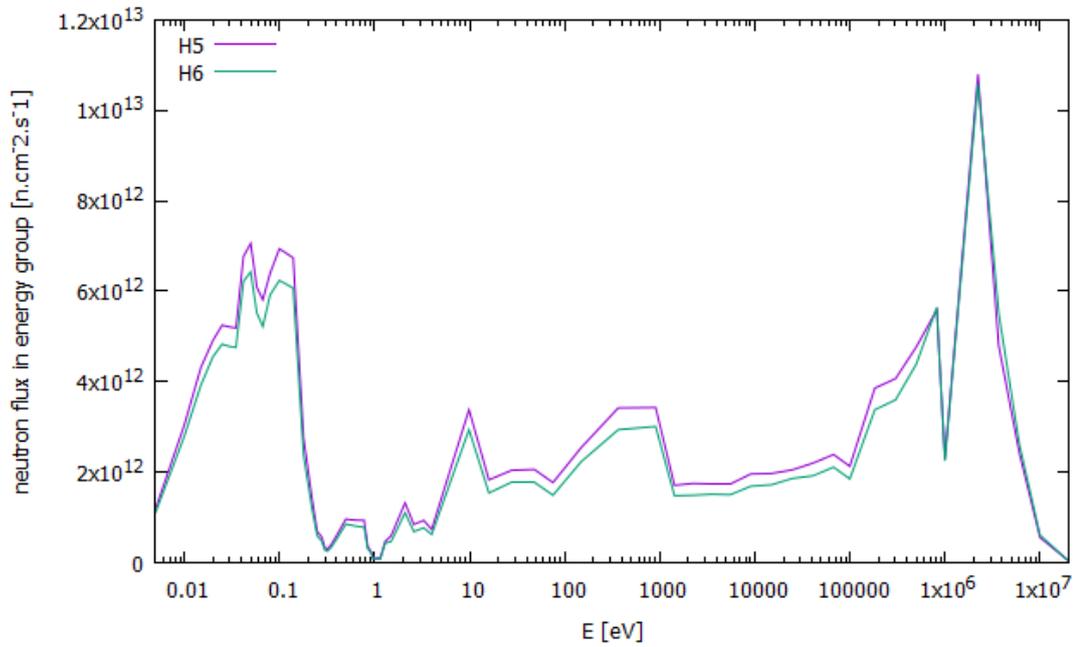
Figure 5.6: Energy beam based neutron spectra in H5 and H6 positions (x-axis in logarithmic scale)

| | IRRADIATION TIME | | | | |
Isotope	1 hour	6 hours	1 day	3 days	7 days
Au-198	9.47E+07	5.47E+07	2.01E+08	4.77E+08	7.38E+08
Gd-161	1.75E+09	1.75E+09	1.75E+09	1.75E+09	1.75E+09
Ho-166	1.42E+10	7.97E+10	2.56E+11	4.67E+11	5.44E+11
Lu-177	1.41E+07	8.38E+07	3.21E+08	8.62E+08	1.63E+09
Mo-101	1.32E+08	1.39E+08	1.39E+08	1.39E+08	1.39E+08
Re-188	8.42E+09	4.58E+10	1.32E+11	1.99E+11	2.10E+11
Fe-59	9.87E+03	5.92E+04	2.35E+05	6.95E+05	1.57E+06
Mo-99	4.37E+06	2.56E+07	9.32E+07	2.22E+08	3.46E+08
Re-186	2.30E+09	1.35E+10	5.04E+10	1.27E+11	2.15E+11
Tc-99m	2.14E+05	6.28E+06	5.57E+07	1.78E+08	2.98E+08
Tb-161	6.67E+06	4.28E+07	1.67E+08	4.55E+08	8.85E+08
Lu-177(Yb)	1.91E+04	1.13E+05	4.33E+05	1.16E+06	1.06E+06
	ACTIVITY [Bq]				

Table 5.5: Summary of calculation prediction for the validation experiment planning
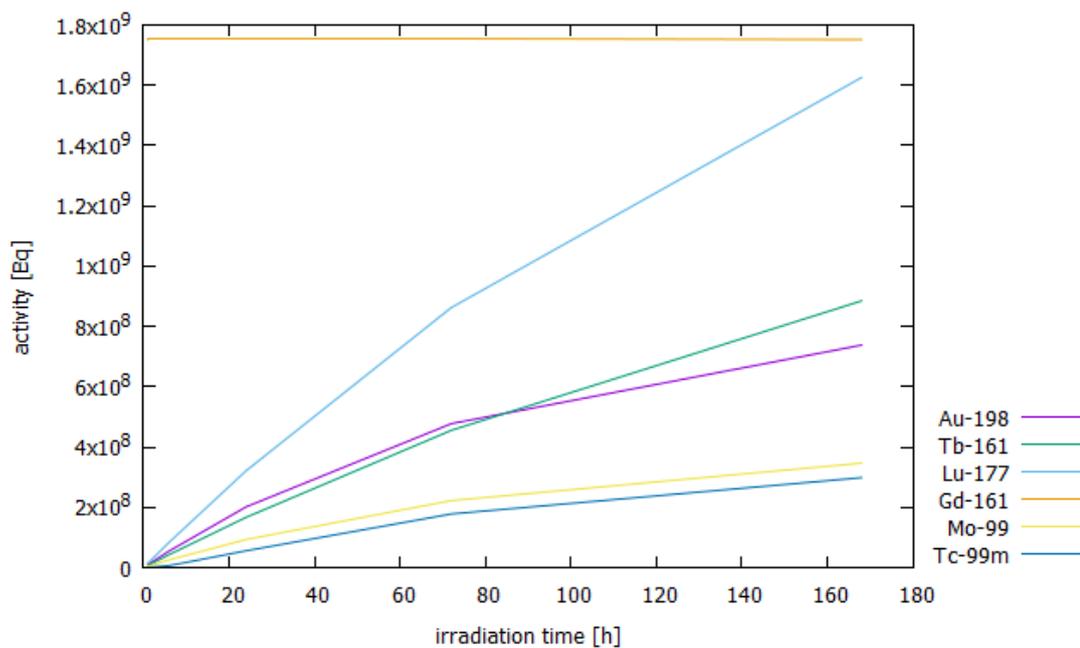
Figure 5.7: Calculated activity as a function of irradiation time
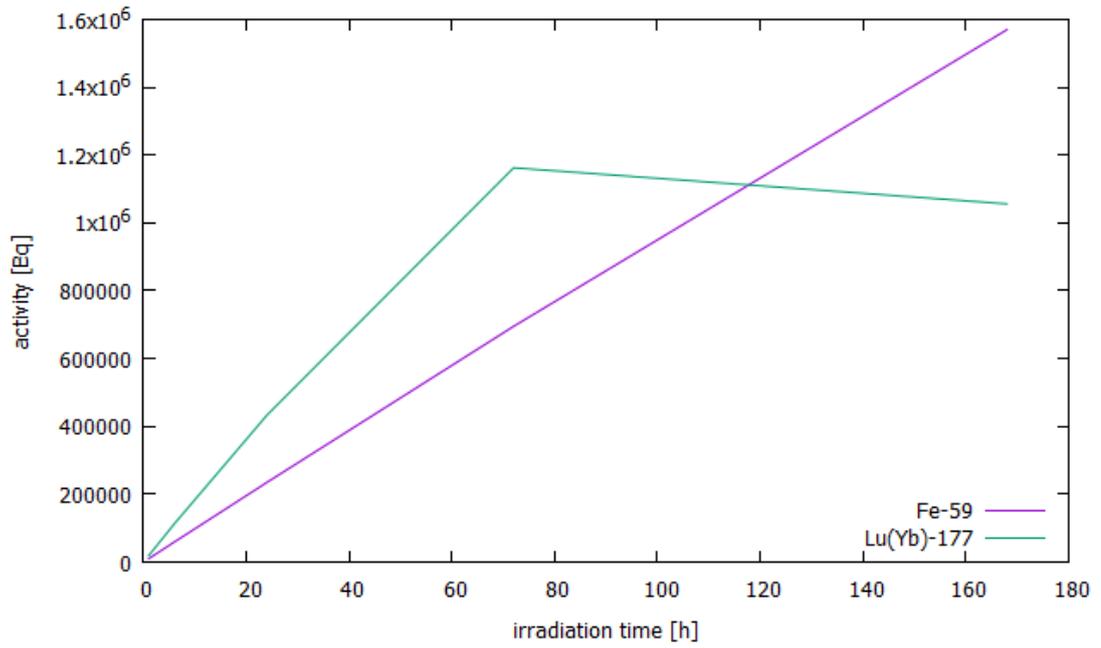
Figure 5.8: Calculated activity as a function of irradiation time
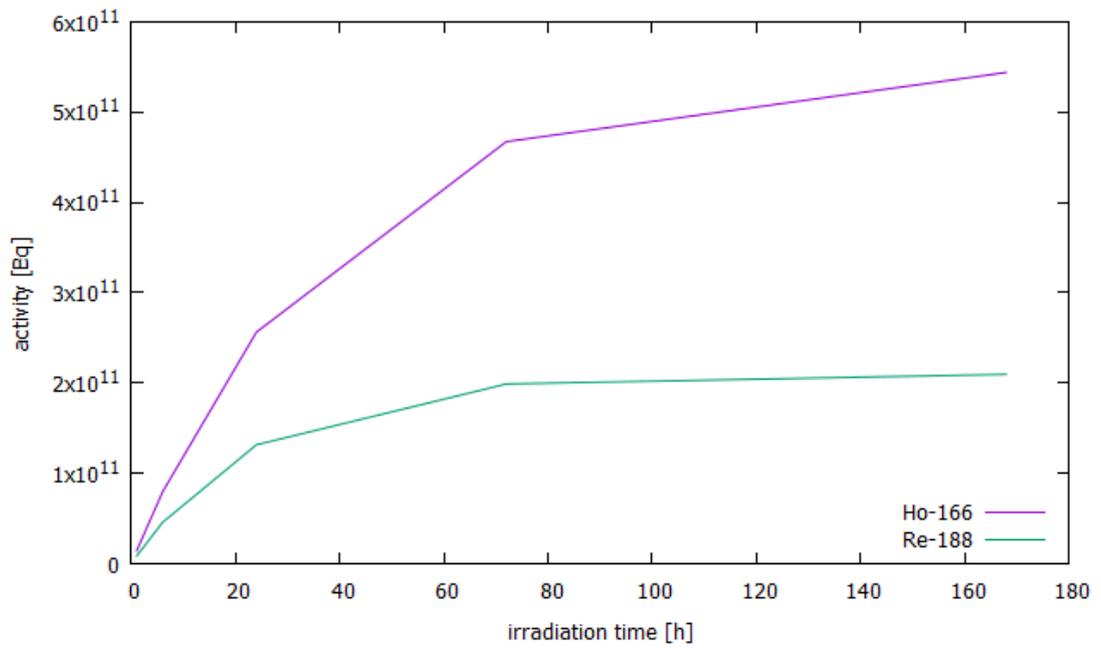
Figure 5.9: Calculated activity as a function of irradiation time

IRR. Time 1 hour	COOLING TIME				
	1 hour	6 hours	1 day	3 days	7 days
Isotope					
Au-198	9.37E+06	8.88E+06	7.32E+06	4.37E+06	1.56E+06
Ho-166	1.38E+10	1.21E+10	7.61E+09	2.19E+09	1.87E+08
Lu-177	1.41E+07	1.38E+07	1.27E+07	1.03E+07	6.81E+06
Mo-101	8.22E+06	3.93E+01			
Re-188	8.09E+09	6.63E+09	3.30E+09	3.98E+08	9.84E+06
Fe-59	9.87E+03	9.84E+03	9.72E+03	9.42E+03	8.87E+03
Mo-99	4.33E+06	4.10E+06	3.40E+06	2.05E+06	7.47E+05
Re-186	2.28E+09	2.19E+09	1.91E+09	1.31E+09	6.23E+08
Tc-99m	6.06E+05	1.96E+06	3.03E+06	1.98E+06	7.22E+05
Tb-161	7.28E+06	7.13E+06	6.61E+06	5.41E+06	3.62E+06
Lu-177(Yb)	1.90E+04	1.86E+04	1.72E+04	1.40E+04	9.23E+03
			ACTIVITY [Bq]		

Table 5.6: Summary of calculated activities after 1 hour of irradiation and subsequent cooling
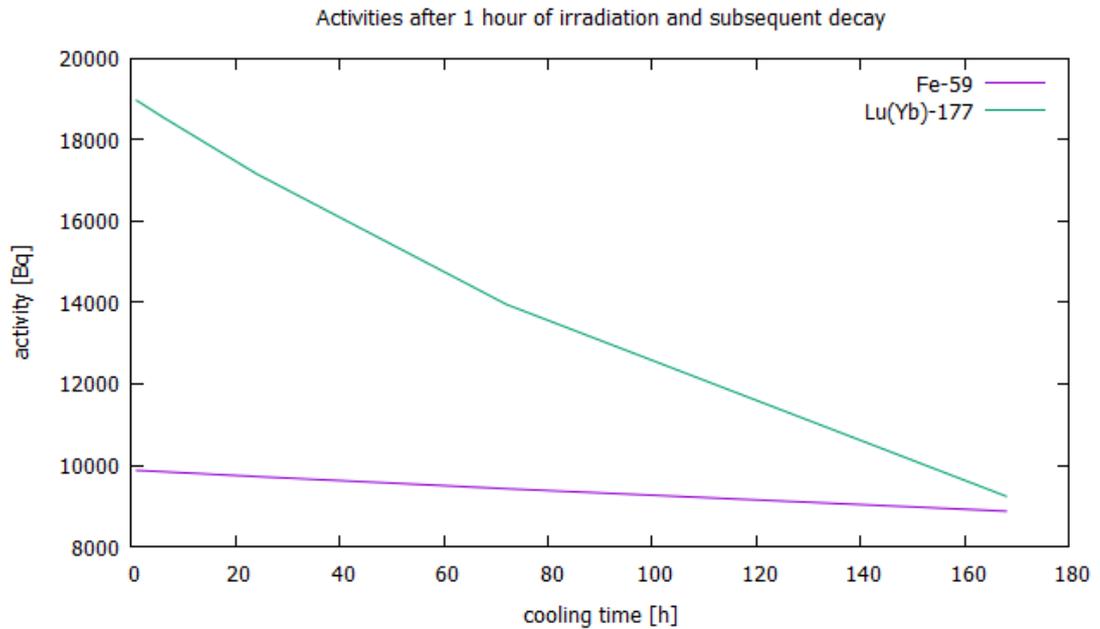
Figure 5.10: Calculated activity after 1 hour of irradiation and its decay
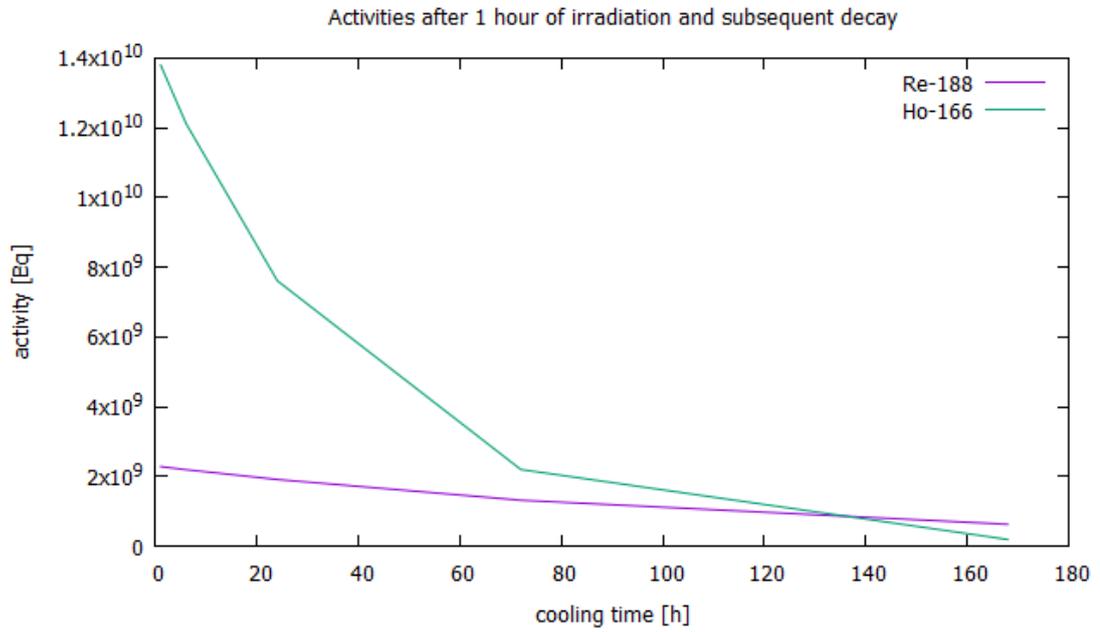
Figure 5.11: Calculated activity after 1 hour of irradiation and its decay
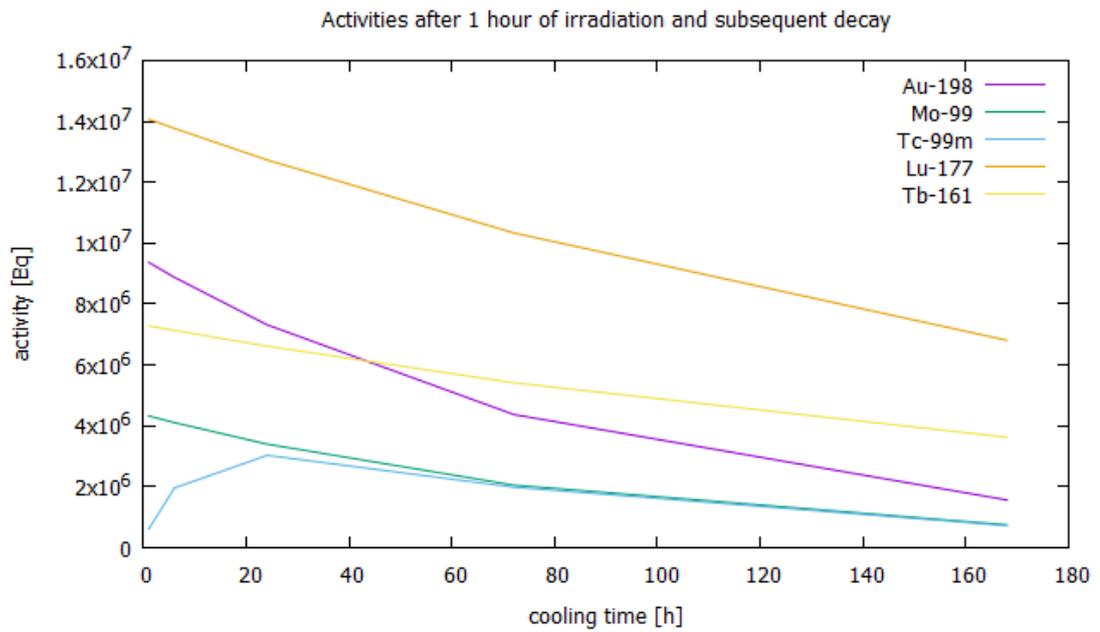
Figure 5.12: Calculated activity after 1 hour of irradiation and its decay

6. Experimental results

6.1 Experiment and simulation comparison

The detector samples were irradiated in the reactor on November 5, 2020. Details of the experiment are given below in the 6.1 table. The reactor layout for the experiment is also shown in the figure 6.1. The samples from the experiment were irradiated in the H6 position (see 6.1). Given the fact that the calculation estimates in many cases predicted high activities even for 1 hour of irradiation, 3 minutes of irradiation time was chosen in order to prevent over-saturation of the detectors and also to ensure radiation safety when manipulating with the activation detectors after the irradiation.

After the irradiation, the activity of the samples was measured at the spectroscopy laboratory. Results are given in the 6.2 table. The set up of the γ-spectrometer used for the measurement is shown on 6.3, 6.2 figures.

6.2 Evaluation of the results

One of the goals of this experiment was to determine whether the FISPACT-II software can be used as a reliable means for prediction of the sample activities. For this reason detectors from iron and gold were included in the experiment among the possible medical radioisotopes. They were used as referential detectors. This is because these detector materials are commonly used for purposes of reactor dosimetry. As it can be seen from the 6.2 table, results for the calculated and measured activities differ by approximately 1.8% in the case of iron and by approximately 0.6% in the case of gold for reaction caused by thermal neutrons. This is a very good match concluding that the software can be used as a possible prediction for activity calculation.

In case of some other samples, the calculated value is higher than the measured value. These differences, sometimes significant might be caused by several factors. The main ones include

- Strong self-shielding of the detector material, which is not accounted by the chosen calculation method of the gamma spectrum in MCNP

- Automatic evaluation of the gamma spectrum, where some isotopes may emit gamma lines with similar energies, which may merge into one peak in the final gamma spectrum

- Some differences may result from the chosen nuclear data library in FISPACT

- A simple model of the irradiation cask in the form of a cylinder instead of its exact shape in MCNP may lead to a different value of the neutron flux. A calculation with an exact shape of the detectors would be very time-consuming.

Strong self-shielding phenomenon occurs in the case of materials with high cross section for neutron capture. The neutrons are captured and absorbed by the

material in the top layers of the material preventing the activation in the whole volume of the detector. This does not occur for materials with low cross section for neutron capture. The cross sections for some of the materials are given in the 6.3 table.

Sample materials	Au, Fe, Ho, Re, Lu, Yb, Gd, Mo
Irradiation time	3 min
Reactor power	9.40 MW
Sample positions	H6/2
Cask material	Al (shell), Pb (weight)

Table 6.1: Experiment details

Detector	Activated Isotope	Activation scheme	Measured Activity [Bq] (±10%)	Calculated Activity [Bq]	Calculated /Measured
Fe	Mn-54	Fe-54(n,p)Mn-54	5.43E+01	4.51E+01	0.83
	Fe-59	Fe-58(n,γ)Fe-59	5.54E+02	5.44E+02	0.98
Au	Au-198	Au-197(n,γ)Au-198	5.09E+05	5.12E+05	1.01
Mo	Mo-99	Mo-98(n,γ)Mo-99	2.65E+05	2.83E+05	1.07
Gd	Gd-153	Gd-152(n,γ)Gd-153	3.32E+03	4.34E+04	13.07
	Gd-159	Gd-158(n,γ)Gd-159	6.96E+06	1.32E+07	1.90
	Tb-161	Gd-160(n,γ)Gd-161$\rightarrow$ Tb-161	6.37E+04	9.67E+04	1.52
Lu	Lu-177	Lu-176(n,γ)Lu-177	6.19E+05	7.53E+05	1.22
Yb	Yb-169	Yb-168(n,γ)Yb-169	7.94E+05	7.69E+05	0.97
	Yb-175	Yb-174(n,γ)Yb-175	5.61E+07	3.12E+07	0.56
	Lu-177	Yb-176(n,γ)Yb-177$\rightarrow$ Lu-177	3.14E+05	3.38E+03	0.01
Ho	Ho-166	Ho-165(n,γ)Ho-166	3.07E+08	7.81E+08	2.54
Re	Re-188	Re-187(n,γ)Re-188	3.55E+08	4.76E+08	1.34
	Re-186	Re-185(n,γ)Re-186	6.73E+07	1.22E+08	1.81

Table 6.2: Experimental results and calculated results from simulations

Material	σ_{nc}[b]
Fe	2.56 ± 0.03
Mo	2.55 ± 0.05
Gd	48890 ± 104
Lu	76.40 ± 21.00
Au	98.65 ± 0.09
Re	89.7 ± 1.7
Ho	64.7 ± 1.2

Table 6.3: Neutron capture cross sections (isotopic abundance of the elements corresponds to natural abundance) [18]
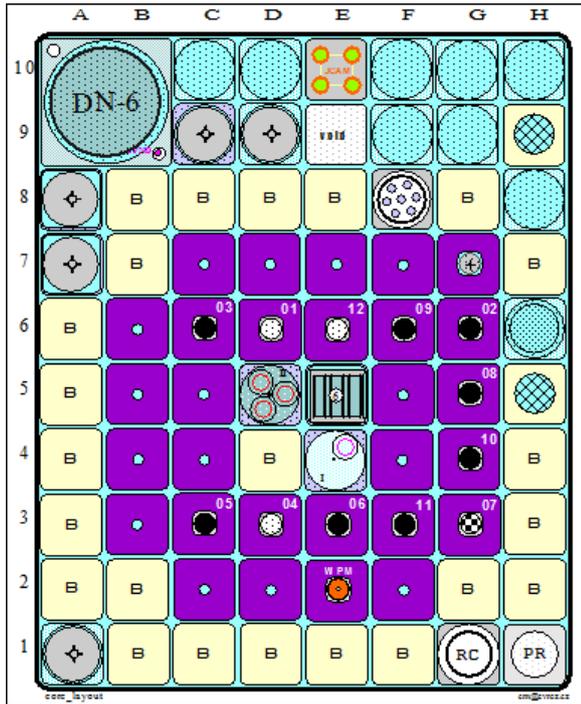
Figure 6.1: Reactor layout for the experiment

Figure 6.2: γ-spectrometer set up with electronics, shielding box and liquid nitrogen tank

Figure 6.3: High purity Ge-crystal detector inside the shielding box with a sample handler

7. Medical radioisotopes production capabilities at the LVR-15 reactor

This final chapter presents results for activity calculations of substances or compounds containing the radioisotopes that are typically irradiated and then delivered to hospitals or to the final processors. These results are based on the calculations of the experimentally validated method described in the previous chapters. The main differences are in the definitions of chemical compositions of the detectors and their masses. The activation detectors are usually chemically pure with a small percentage of impurities. That was the case of the experiment described above. Each of the detectors had a different mass and the same diameter of 4 mm, see table 5.1.

In the case of the following results the chemical composition of the irradiated material was a compound (usually a certain type of oxide) containing enriched isotopes of a given element. Enrichment is used in order to increase the yields and to decrease the presence of parasitical nuclides. The masses of the substances were modelled to be 1 mg in each case. Details are given in the 7.1 table.

The simulations were performed for the H5 and H6 positions in the reactor (these are the positions where most of the irradiation experiments take place see, figure 6.1 or 3.5) with neutron conditions from the day of the experiment i.e. November 5, 2020.

The final results are given in the tables 7.3 and 7.2. As it can be seen, the activities in the H5 position were higher than those in H6. This is caused by the fact that the neutron flux and especially the thermal part of the neutron spectrum flux in the H5 was higher (see tables 5.4 and 5.3) and this lead to higher values of obtained activities. The reason for the higher flux in the H5 position compared to H6 position is that there is a reflector made from beryllium which reflects the neutrons back increasing the flux and also acts as a moderator. This way the neutrons are also slowed down increasing their probability of interaction. One additional reason is the larger diameter of the irradiation capsule (see fig.7.1)in the H6 position compared to the H5 position which allows less neutron moderation and thus less thermal neutrons in H6 compared to H5.

The results in the tables 7.2 and 7.3 represent a valid prediction for the medical radioisotope production at the LVR-15 research reactor for the case of 1 mg substance masses. Calculation predictions for the irradiation of higher masses of substances need to be adjusted by a correction factor determined experimentally. Another possibility to obtain better prediction would be to model the neutron spectrum in MCNP for the exact volume of the irradiated substance to account for the self shielding phenomenon.

Several observations can be seen from these results. For the case of Re-188, doubling or tripling the time of irradiation does not prove to increase the activities in comparison with the 1 day of irradiation time. For Lu-177, Yb-175, the state of saturated activity is achieved after 2 weeks of irradiation. Linear increase of activity can be seen in the cases of Mo-99, Tc-99m, Re-186, Ho-166 implying that

the saturation was not yet achieved.

In general, this has to do with different half-lives of the isotopes. Calculations show that irradiation times exceeding two half-lives of the radionuclide are ineffective in further activity increase.

Isotope	Compound	Enrichment
Gd-160	Gd_2O_3	98.20%
Ho-165	Ho_2O_3	100%
Lu-176	Lu_2O_3	82%
Yb-176	Yb_2O_3	99.33%
Mo-98	MoO_3	95.20%
Re-187	metal	95.20%

Table 7.1: Properties of the substances used in the simulations

IRRADIATION TIME

Isotope		1day	2 days	3 days	6 days	9 days	1 week	2 weeks	3 weeks
Gd-161							3.61E+08	3.61E+08	3.61E+08
Gd-153							2.65E-01	2.20E-01	1.63E-01
Gd-159							1.55E+07	1.55E+07	1.55E+07
Tb-161							1.82E+08	2.72E+08	3.17E+08
Yb-175	(Yb)						6.83E+07	8.91E+07	9.52E+07
Yb-177	(Yb)						5.72E+08	5.72E+08	5.71E+08
Lu-177	(Yb)						2.87E+08	4.22E+08	4.83E+08
Lu-177	(Lu)						2.92E+11	3.92E+11	4.06E+11
Yb-175	(Lu)						5.35E+03	7.02E+03	7.53E+03
Mo-99				6.35E+07	9.32E+07	1.07E+08			
Tc-99m				5.09E+07	7.96E+07	9.29E+07			
Re-188		1.06E+10	1.46E+10	1.61E+10					
Re-186		3.30E+08	6.04E+08	8.30E+08					
Ho-166		8.97E+09	1.38E+10	1.64E+10					

ACTIVITY [Bq]

Table 7.2: Activities calculated for the H6 position

				IRRADIATION TIME				
Isotope	1day	2 days	3 days	6 days	9 days	1 week	2 weeks	3 weeks
Gd-161						4.00E+08	4.00E+08	4.00E+08
Gd-153						2.40E-01	1.88E-01	1.36E-01
Gd-159						1.73E+07	1.73E+07	1.73E+07
Tb-161						2.02E+08	3.02E+08	3.51E+08
Yb-175 (Yb)						7.48E+07	9.75E+07	1.04E+08
Yb-177 (Yb)						6.30E+08	6.30E+08	6.30E+08
Lu-177 (Yb)						3.16E+08	4.63E+08	5.31E+08
Lu-177 (Lu)						3.20E+11	4.26E+11	4.36E+11
Yb-175 (Lu)						5.07E+03	6.64E+03	7.13E+03
Mo-99			7.21E+07	1.06E+08	1.21E+08			
Tc-99m			5.78E+07	9.04E+07	1.06E+08			
Re-188	1.17E+10	1.61E+10	1.78E+10					
Re-186	3.79E+08	6.94E+08	9.54E+08					
Ho-166	1.01E+10	1.55E+10	1.84E+10					

ACTIVITY [Bq]

Table 7.3: Activities calculated for the H5 position

Isotope	Irradiation time	Activity in H5 [Bq]	Activity in H6 [Bq]	Activity in H5 /Activity in H6
Tb-161	1 week	2.02E+08	1.82E+08	1.10
Lu-177(Yb)	2 weeks	4.63E+08	4.22E+08	1.10
Lu-177(Lu)	2 weeks	4.26E+11	3.92E+11	1.09
Tc-99m	3 days	5.78E+07	5.09E+07	1.16
Re-188	3 days	1.78E+10	1.61E+10	1.11
Ho-166	3 days	1.84E+10	1.64E+10	1.12

Table 7.4: Activity comparison of radioisotopes for H5 and H6 positions at a specific irradiation time

Figure 7.1: Irradiation capsules used in the experiment (larger in H6 pos., smaller in H5 pos.)

Conclusion

Radioisotopes commonly used in nuclear medicine were discussed in this thesis with emphasis on those that can be produced in research nuclear reactor environment. Most importantly, production capabilities of these radioisotopes at the LVR-15 research reactor facility at Řež near Prague were discussed as well as the reactor facility itself.

Experimental part of this thesis was therefore focused on the following aspects. Firstly, it was a calculation prediction for activities of the target materials that were later on used in the experiment. This provided estimates of the activities for the materials according to which a suitable irradiation time for the experiment was determined. These materials in the form of activation detectors were then irradiated in the reactor.

Secondly, the activities of the detectors after the irradiation were measured. These results were compared with the calculation predictions confirming the calculations as a good validation method for the experiments.

Lastly, using this validated calculation method, activities of final chemical compounds containing the radioisotopes that can be shipped to hospitals or final processors from the LVR-15 reactor were determined. The method described in this thesis is limited to low amounts of material. In case of higher amounts of materials (100 mg and more), self-shielding plays even more significant role and this has to be taken into account in calculations for the MCNP model. More precisely, the neutron spectrum has to be calculated in the cell with the exact volume and weight of the target material rather than the approach used in calculations above, where the neutron spectrum was calculated in a cylinder filled with vacuum that represented the whole irradiation cask. Such calculation is extremely time- consuming, but it can provide a more exact prediction of the activity of a given radionuclide.

Analyses show that irradiation times exceeding two half-lives of the radionuclide are ineffective due to low activity increase of the required radionuclides.

The highest activities could be obtained in the central positions of the reactor (such as D5, D4, E5, E4) as these positions have the highest fluxes of thermal neutrons available. However, they are reserved for the Mo-99 production by fission in the foreseeable future.

Bibliography

[1] National Center for Biotecnology Information *Molybdenum-99 Production and Use.*[online], avilable from
`https://www.ncbi.nlm.nih.gov/books/NBK215133/,`, cited 22.8.2020

[2] International Atomic Energy Agency *Manual for reactor produced radioisotopes* Vienna, 2003

[3] EUROPEAN COMMISSION HEALTH AND CONSUMERS DIRECTORATE-GENERAL *Preliminary Report On Supply Of Radioisotopes For Medical Use And Current Developments In Nuclear Medicine.*[online], available from
`https://ec.europa.eu/euratom/docs/2009_radioisotopes_report.pdf`, cited 23.8.2020

[4] National Center for Biotecnology Information *Lutetium Lu 177 Dotatate* [online], avilable from
`https://www.ncbi.nlm.nih.gov/books/NBK500964/`, cited 2.10.2020

[5] Klaassen, N.J.M. et al. *The various therapeutic applications of the medical isotope holmium-166: a narrative review. EJNMMI radiopharm. chem. 4, 19 (2019)* [online], avilable from
`https://ejnmmipharmchem.springeropen.com/articles/10.1186/-s41181-019-0066-3`, cited 22.11.2020

[6] Periodic table- Terbium [online], avilable from
`https://www.rsc.org/periodic-table/element/65/terbium` cited 11.2.2021

[7] Terbium-161 for PSMA-targeted radionuclide therapy of prostate cancer [online], avilable from
`https://pubmed.ncbi.nlm.nih.gov/31134301/`, cited 11.2.2021

[8] J.C.Sublet et al. *The FISPACT-II User Manual.* UK Atomic Energy Authority, UKAEA-R(11)11 Issue 7 September 2015

[9] C.J.Werner *MCNP Users Manual-Code Version 6.2.* LA-UR-17-29981 (2017)

[10] Michal Koleška et al. *Capabilities of the LVR-15 research reactor for production of medical and industrial radioisotopes.* Journal of Radioanalytical and Nuclear Chemistry An International Journal Dealing with All Aspects and Applications of Nuclear Chemistry ISSN 0236-5731 Volume 305 Number 1 J Radioanal Nucl Chem (2015) 305:51-59 DOI 10.1007/s10967-015-4025-5

[11] RADIOISOTOPES IN MEDICINE [online], available from
`https://www.world-nuclear.org/information-library/non-power-nuclear-applications/radioisotopes-research/radioisotopes-in-medicine.aspx`, cited 6.3.2021.

[12] Iridium-192 [online], available from
`https://www.chemistrylearner.com/iridium-192.html`, cited 12.3.2021.

[13] About Xenon-133 [online], available from
https://www.advancingnuclearmedicine.com/en/-
products/xenon-133, cited 2.4.2021.

[14] Spectroscopy and spectrometry differences [online], available from
https://www.nuclear-power.net/nuclear-engi-
neering/radiation-detection/gammaspectroscopy/ cited 12.4.2021.

[15] Štefánik M. *Neutrónová aktivačná analýza na horizontálnom radiálnom
kanále školského reaktora VR-1* Diploma thesis, CTU in Prague, 2008

[16] Šoltés J. *Vývoj filtra neutrónových zväzkov* Disertation thesis, CTU in
Prague, 2017

[17] Self-shielding [online], available from
https://www.nuclear-power.net/nuclear-power/reactor-physics/-
nuclear-engineering-fundamentals/neutron-nuclear-reactions/-
self-shielding/, cited 26.4.2021.

[18] THERMAL NEUTRON CAPTURE CROSS SECTIONS RESONANCE
INTEGRALS AND G-FACTORS [online], available from
https://www.osti.gov/etdeweb/servlets/purl/20332542, cited
28.4.2021

[19] Stable isotopes of gadolinium [online], available from
https://www.isoflex.com/gadolinium-gd, cited 30.4.2021

[20] International Atomic Energy Agency *Sealed radioactive sources* Vienna, 2013

[21] Jan Vít *Research reactor LVR-15 Irradiation capabilities* Power Point pre-
sentation, October 2018

[22] Šoltés J. *Stanovenie tepelného výkonu reaktora VR-1 pomocou aktivačných
detektorov* Diploma thesis, CTU in Prague, 2011

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List of Abbreviations

CT   Computed Tomography
EDTMP   Ethylene Diamine Tetramethylene Phosphanate Acid
HCC   Hepatocellular Carcinoma
HEU   Highly Enriched Uranium
LEU   Low Enriched Uranium
LVR-15   Lehkovodní reaktor (15 MW power)
MCNP   Monte Carlo Neutral Particle
MRI   Magnetic Resonance Imaging
PET   Positron Emission Tomography
PSMA   Prostate Specific Membrane Antigen
SPECT   Single Photon Emission Computed Tomography
TAT   Targeted Alpha Therapy