TOMSK POLYTECHNIC UNIVERSITY
ТОМСКИЙ ПОЛИТЕХНИЧЕСКИЙ УНИВЕРСИТЕТ

Министерство науки и высшего образования Российской Федерации
федеральное государственное автономное
образовательное учреждение высшего образования
«Национальный исследовательский Томский политехнический университет» (ТПУ)

Школа Инженерная школа ядерных технологий
Направление подготовки 14.04.02 Ядерные физика и технологии
Отделение школы (НОЦ) Ядерно-топливного цикла

МАГИСТЕРСКАЯ ДИССЕРТАЦИЯ

Тема работы
Расчёт нейтронного и гамма полей горизонтального экспериментального канала ГЭК-1 и оценочный расчёт биологической защиты

УДК 621.039.512:621.039.511

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Томск – 2019 г.

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School Nuclear Science & Engineering
Field of training (Specialty) 14.04.02 Nuclear Physics and Technology
Division Nuclear Fuel Cycle

MASTER'S GRADUATION THESIS

Topic of research work
The calculation of the neutron and gamma fields of the horizontal experimental channel HEC-1 and the estimated calculation of biological protection

UDC 621.039.512:621.039.511

Student

Group	Full name	Signature	Date
0АМ7И	Zinner Vsevolod Olegovich		

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Position	Full name	Academic degree, academic rank	Signature	Date
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Section "Social Responsibility"

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Senior Lecturer	D.A.Verigin	PhD		

ADMITTED TO DEFENSE:

Director of the programme	Full name	Academic degree, academic rank	Signature	Date
Nuclear power installations operation	V.V.Verhoturova	PhD		

Tomsk – 2019

Expected learning outcomes

Learning outcome (LO) code	Learning outcome (a graduate should be ready)	Requirements of the FSES HE, criteria and / or interested parties
Professional competencies		
LO1	To apply deep mathematical, scientific, socio-economic and professional knowledge for conducting theoretical and experimental research in the field of the use of nuclear science and technology/	FSES HE Requirements (PC-1,2, 3, 6, UC-1,3), Criterion 5 RAEE (p 1.1)
LO2	To demonstrate ability to define, formulate, and solve interdisciplinary engineering tasks in the nuclear field using professional knowledge and modern research methods.	FSES HE Requirements (PC-2,6,9,10,14, UC-2,3,4, BPC1,2), Criterion 5 RAEE (p 1.2)
LO3	To plan and conduct analytical, simulation and experimental studies in complex and uncertain conditions using modern technologies, and to evaluate critically research results.	FSES HE Requirements (PC-4,5,6,9,22, UC-1,2,5,6), Criterion 5 RAEE (p 1.3)
LO4	To use basic and special approaches, skills and methods for identification, analysis, and solution of technical problems in the field of nuclear science and technology.	FSES HE Requirements (PC-7,10,11,12,13, UC-1-3,BPC1,3), Criterion 5 RAEE (p 1.4)
LO5	To operate modern physical equipment and instruments, to master technological processes in the course of preparation for the production of new materials, instruments, installations, and systems.	FSES HE Requirements (PC-8,11,14,15, BPC-1), Criterion 5 RAEE (p 1.3)
LO6	To demonstrate ability to develop multi-option schemes for achieving production goals with the effective use of available technical means and resources.	FSES HE Requirements (PC-12,13,14,16, BPC-2), Criterion 5 RAEE (p 1.3)
Cultural competencies		
LO7	To demonstrate ability to use a creative approach to develop new ideas and methods for designing nuclear facilities, as well as to modernize and improve the applied technologies of nuclear production.	FSES HE Requirements (PC-2,6,9,10,14, UC-1,2,3), Criterion 5 RAEE (p 1.2,2.4,2.5)
Basic professional competencies		
LO8	To demonstrate skills of independent learning and readiness for continuous self-development within the whole period of professional activity.	FSES HE Requirements (PC-16,17,21, UC-5,6, BPC-1), Criterion 5 RAEE (p 2.6) coordinated with the requirements of the international standard EURACE & FEANI

LO9	To use a foreign language at a level that enables a graduate to function successfully in the international environment, to develop documentation, and to introduce the results of their professional activity.	FSES HE Requirements (BPC-3, UC-2,4), Criterion 5 RAEE (p 2.2)
LO10	To demonstrate independent thinking, to function efficiently in command-oriented tasks and to have a high level of productivity in the professional (sectoral), ethical and social environments, to lead professional teams, to set tasks, to assign responsibilities and bear liability for the results of work.	FSES HE Requirements (PC-18,20,21,22,23, UC-1,4, BPC-2), Criterion 5 RAEE (p 1.6,2.3) coordinated with the requirements of the international standard EUR-ACE & FEANI

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School Nuclear Science & Engineering

Field of training (specialty) 14.04.02 Nuclear Physics and Technology

Division Nuclear Fuel Cycle

APPROVED BY:
Director of the programme
_____________ Verkhoturova V.V.
«____» ____________ 2019

ASSIGNMENT
for the Graduation Thesis completion

In the form:

Magister's dissertation

For a student:

Group	**Full name**
0AM7И	Zinner Vsevolod Olegovich

Topic of research work:

The calculation of the neutron and gamma fields of the horizontal experimental channel HEC-1 and the estimated calculation of biological protection	
Approved by the order of the Director of School of Nuclear Science & Engineering (date, number):	№ 1711c at 05.03.2019

Deadline for completion of Master's Graduation Thesis:	07.06.2019

TERMS OF REFERENCE:

Initial date for research work: *(the name of the object of research or design; performance or load; mode of operation (continuous, periodic, cyclic, etc.); type of raw material or material of the product; requirements for the product, product or process; special requirements to the features of the operation of the object or product in terms of operational safety, environmental impact, energy costs; economic analysis, etc.)*	- The calculation of the neutron and gamma fields of horizontal experimental channel HEC-1 and the estimated calculation of biological protection; - Literary sources containing information about Boron Neutron Capture Therapy;

	- Literary sources containing information about calculation biological sources for reactors; - Software for building model of reactor and calculation of neutron and gamma fields.
List of the issues to be investigated, designed and developed *(analytical review of literary sources with the purpose to study global scientific and technological achievements in the target field, formulation of the research purpose, design, construction, determination of the procedure for research, design, and construction, discussion of the research work results, formulation of additional sections to be developed; conclusions).*	Tasks: - Create a model of an IRT-T reactor with changed biological protection of horizontal experimental channel HEC-1; - Calculate the neutron and gamma fields of HEC-1 at the exit of the channel; - Conduct an estimated calculation of biological protection for new geometry. - Show the necessaryty of upgrading existing biological protection of the horizontal experimental channel HEC-1.
List of graphic material *(with an exact indication of mandatory drawings)*	**N/A**

Advisors to the sections of the Master's Graduation Thesis
(with indication of sections)

Section	Advisor
One: Literature Review	Yemets E.G.
Two: Methodology	Yemets E.G.
Three: Results	Yemets E.G.
Four: Financial management, resource efficiency and conservation	Menshikova E.V.
Five: Social Responsibilities	Verigin D.A.

Date of issuance of the assignment for Master's Graduation Thesis completion according to the schedule	07.06.2019

Assignment issued by a scientific supervisor / advisor (if any):

Position	Full name	Academic degree, academic status	Signature	Date
Docent	Yemets E.G.	PhD		

Assignment accepted for execution by a student:

Group	Full name	Signature	Date
0АМ7И	Zinner Vsevolod Olegovich		

TASK FOR SECTION

«FINANCIAL MANAGEMENT, RESOURCE EFFICIENCY AND RESOURCE SAVING»

To the student:

Group	Full name
0АМ7И	Zinner Vsevolod Olegovich

School	Nuclear Science & Engineering	**Division**	Nuclear-Fuel Cycle
Degree	Master	**Educational Program**	14.04.02 Nuclear physics and technologies

Input data to the section «Financial management, resource efficiency and resource saving»:	
1. Resource cost of scientific and technical research (STR): material and technical, energetic, financial and human	– Salary costs – 157192.70 rub. – STR budget – 194422.00 rub.
2. Expenditure rates and expenditure standards for resources	– Electricity costs – 5,8 rub per 1 kW
3. Current tax system, tax rates, charges rates, discounting rates and interest rates	– Labor tax – 27,1 %; – Overhead costs – 30%;
The list of subjects to study, design and develop:	
1. Assessment of commercial and innovative potential of STR	– comparative analysis with other researches in this field;
2. Development of charter for scientific-research project	– SWOT-analysis;
3. Scheduling of STR management process: structure and timeline, budget, risk management	– calculation of working hours for project; – creation of the time schedule of the project; – calculation of scientific and technical research budget;
4. Resource efficiency	– integral indicator of resource efficiency for the developed project.
A list of graphic material *(with list of mandatory blueprints):*	
1. Competitiveness analysis *2. SWOT- analysis* *3. Gantt chart and budget of scientific research* *4. Assessment of resource, financial and economic efficiency of STR* *5. Potential risks*	

Date of issue of the task for the section according to the schedule	05.03.2019

Task issued by adviser:

Position	Full name	Scientific degree, rank	Signature	Date
Associate professor	E.V. Menshikova	PhD		

The task was accepted by the student:

Group	Full name	Signature	Date
0АМ7И	Zinner Vsevolod Olegovich		

TASK FOR SECTION
"SOCIAL RESPONSIBILITY"

To the student:

Group	Full name
0АМ7И	Zinner Vsevolod Olegovich

School	**School of Nuclear Science & Engineering**	**Department**	**Department of Nuclear Fuel Cycle**
Degree	Master	**Specialization**	14.04.02 Nuclear Power Installation Operation

Input data to the "social responsibility":	
1. *Describe workplace (work area) for occurrence of:*	– Harmful factors of the environment: microclimate, illumination, noise, vibration, electromagnetic fields, ionizing radiation; – dangerous factors of environment: electrical, fire and explosive nature.
2. *Acquaintance and selection of legislative and normative documents on the topic*	– electrical safety; – fire and explosion safety; – labor protection requirements when working on a PC.
The list of subjects to study, design and develop:	
1. *Analysis of the identified harmful factors of the environment in the following sequence:*	– The effect of the factor on the human body; – Reduction of permissible standards with the required dimensionality (with reference to the relevant normative and technical document); – Proposed remedies (collective and individual).
2. *Analysis of identified hazards of the environment:*	– Electrical safety (including static electricity, protective equipment); – fire and explosion safety (causes, preventive measures, primary fire extinguishing agents).

Date of issue of the task for the section according to the schedule	

Task issued by consultant:

Position	Full name	Scientific degree, rank	Signature	date
Senior lecturer	Verigin D.A.	PhD		

The task was accepted by the student:

Group	Full name	Signature	date
0АМ7И	Zinner Vsevolod Olegovich		

Content

Introduction

In connection with the design of the channel for neutron-capture therapy (NCT), it is necessary to calculate the parameters of biological protection around the channel to ensure safe working conditions for personnel.

For the calculations, the mcu-5 software package will be used. MCU (Monte Carlo Universal) is a project on development and practical use of a universal computer code for simulation of particle transport (neutrons, photons, electrons, positrons) in three-dimensional systems by means of the Monte Carlo method.

The main advantage of the Monte Carlo method is its ability to simulate the interaction of radiation with substance on the basis of the information from files of the evaluated nuclear data (i.e. the most exact data without additional assumptions is used). Besides, this method practically does not impose restrictions on the geometry of considered systems.

At the reactor IRT-T, it is planned to create a technology of neutron capture therapy on channel HEC-1.

To increase the flux density of thermal and resonant neutrons, it is possible to approach the core by removing the retractable concrete insert. In this regard, it is necessary to conduct an estimated calculation of the neutron and gamma components falling on the defense. This is necessary in order to show whether an upgrade of the existing protection will be necessary.

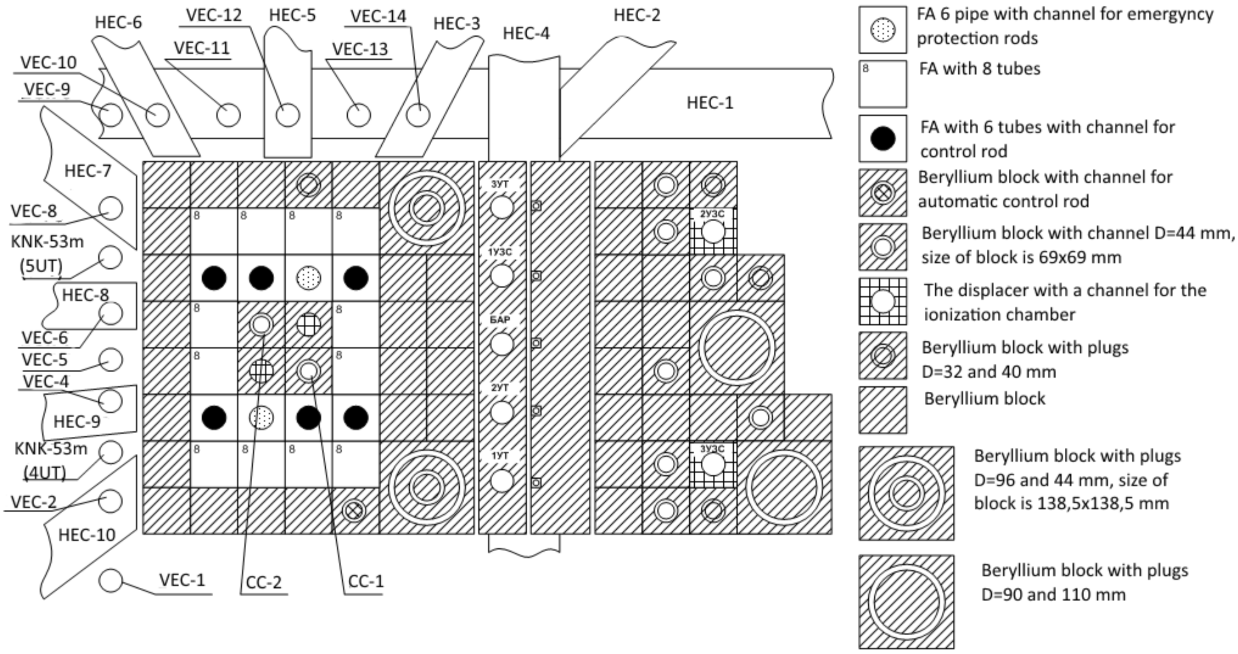

Figure 1 – Cartogram of reactor core and experimental devices

The reactor power is set to 6 MW. As a protective material, it is advisable to choose ordinary concrete.

1.1 Reactor overview

A nuclear reactor is a powerful source of neutron and gamma radiation. The reactor is designed to carry out research work on solid state physics, neutron activation analysis of the elemental composition of substances, production of radionuclides, silicon doping, neutron radiography and other works using reactor radiation.

The reactor core is assembled from fuel assemblies (FA) of the type IRT-3M with a high multiplication factor and a small migration length, which allows to obtain geometrically small zone sizes and a large neutron leakage into the reflector.

Beryllium having a long neutron migration length is used as a reflector, which makes it possible to ensure a broad maximum of the thermal neutron flux density and a high level of neutron flux density in the experimental channels.

Despite the small size of the core, it has 14 vertical experimental channels (VEC) and 10 horizontal channels (HEC), which is significantly more than on more powerful research reactors. All this allows simultaneous irradiation of a large number of targets. Accordingly, the cost of their exposure is reduced significantly. In addition, a unique feature of the IRT-T reactor is that its neutron spectrum contains many resonant neutrons due to the use of a beryllium moderator and a successful arrangement of beryllium traps in the zone of the central channels.

The main characteristics of the reactor are presented in the following table 1.

Table 1 – The main characteristics of the reactor

Power, MW	6
Number of fuel assemblies 8^{th}-pipe 6^{th}-pipe	 12 8
Core heat transfer surface, m^2	29.6
Core volume, l	59.3
Reactivity margin, % dk/k	7.2
Working bodies of management and protection - emergency protection - compensating - automatic control	 2 6 1
Total organ efficiency EP, % dk/k	3.5
Total organ efficiency compensating and automatic control, % dk/k	10.0
Coefficient and uniformity of energy release - over the horizontal section of the core - according to the height of the core	 1.78 1.26
Maximum power density, kW/l	227
Maximum heat flux density, kW/m^2	427
Pressure drop across the core, mm water	3.4
Average coolant velocity in the gaps of FA, m/s	2.88
Coolant flow through the core and reflector, t/h	900
Active water inlet temperature, ˚C	45
Maximum design temperature of the fuel rod surface, ˚C	77
Surface boiling point, ˚C	123
Maximum undisturbed thermal neutron density, $(n/cm^2 \cdot s)/MW$ - in reflector - in the core For fast neutrons - in reflector - in the core	 $1.76 \cdot 10^{13}$ $1.50 \cdot 10^{13}$ $0.34 \cdot 10^{13}$ $1.12 \cdot 10^{13}$

The reactor core is located in the lower part of the pool filled with water. Its center is located at a depth of 6.5 m, and the body is made of aluminum brand AD-1. Its support grid is made of a CAB-1 alloy in the form of a rectangular plate with dimensions of 940x721x85 mm^3, the perforated central part of which is located above the rectangular opening in the flange of 600x530 mm^2.

The core body is attached to the supporting spacer grid installed on the flange of stainless steel with a thickness of 29 mm. It is welded to sheets of built-in retention capacity and relies additionally on 6 racks made of tubes with a diameter of 108 mm and a thickness of 5 mm attached to the bottom of the stainless steel base plate, which is fixed to the bottom of the tank by welding. The top grid for the CPS channels with a thickness of 30 mm is attached to it from above. A titanium gasket is located between the supporting distance grid of the core and the stainless steel flange. At the contact points of aluminum alloy and stainless steel parts, titanium gaskets are installed to prevent corrosion of aluminum.

The base plate and the lower core grille protect the concrete under the bottom of the tank from radiation heating. A fuel assembly, beryllium reflector blocks and neutron traps, propellants are placed into the core of the core on the supporting spacing grating. At the top, these devices are fixed with special protrusions on their end parts, at the bottom they are fixed on the spacer grid with the help of slots in the lower tips. Fuel assemblies and beryllium blocks are mounted with some effort, i.e. their sleeves provide them a tight fit. The coolant, passing through the active zone from top to bottom, eliminates the possibility of moving its elements and the reflector. In the zone housing there are 56 cells for the installation of fuel assemblies and beryllium blocks. Four central ones are occupied by beryllium blocks, forming a neutron trap.

From the core through the gates, located in the array at elevation +0,9 m, neutron beams can be output to the physical hall radially.

The permissible fluence of fast neutrons (with E> 0,821 MeV) on the surface of the reference grid (in terms of the possibility of its brittle destruction) is $1{,}6 \cdot 10^{22}$ $n \cdot cm^{-2}$. The maximum differential flux density of fast neutrons on the support grid is $0{,}87 \cdot 10^{12}$ $n \cdot cm^{-2} \cdot s^{-1} \cdot MW^{-1}$. Thus, the maximum permissible value of the fluence of these particles on it will be reached in approximately 35 years.

The core of a nuclear reactor of IRT is a parallelepiped with sides of 670x560x500 mm^3. From the sides, it could be surrounded by a beryllium or graphite reflector 71,5 mm thick. When designing the irradiation zone, a number of geometric and technological constraints were imposed on the design of the thermal assembly.

When choosing the material of the reflector, only the use of beryllium, graphite, and light tank water was allowed. Geometric constraints were imposed on the position of the channel relative to the core and the size of the thermal assembly, which was a continuation of its faces and ended with the wall of the reactor tank.

The design of the nuclear installation made it possible to change the distance between the channel wall and heat-generating cassettes by shifting the cassettes by an amount multiple of the step of the lower carrier grid of the active zone. The grid spacing is 71,5 mm. Moreover, only the following options were possible when this distance is zero: cassettes with fissile material are adjacent to the canal wall or when this distance is 71,5 and 143 mm.

In the latter two versions, you can use beryllium or graphite cassettes, or leave the tank water layer. By selecting the type of moderator and its thickness, one can influence the neutron field in the channel. Moreover, the spatial distribution of the neutron field in it, the spectrum and the absolute values of the fluxes change. Consequently, in our case, the variable parameters in the formation of the irradiation zone are the type of moderator and its thickness between the channel wall and the neutron source.

Structurally, the core is made in the form of a cubic lattice of 56 cells in a horizontal section. Twenty of them are fuel assemblies (FA). Four central cells are occupied by a beryllium propellant with two central channels in it. In the remaining three peripheral cells, beryllium reflector blocks are installed. The apparatus uses FA of the type IRT-3M [1], consisting of tubular fuel elements based on uranium-aluminum alloy with an enrichment of 90% for uranium 235.

The IRT-T reactor has 10 horizontal experimental channels: eight radial diameters of 100 mm, two tangent channels (HEC-1 and HEC-4) with a diameter of 150 mm. HEC-3, -5, -8 are equipped with pneumatic transport systems with a set of equipment for carrying out neutron activation analysis. There are 14 vertical channels in the core. Two of them are installed in a beryllium "neutron trap" in the center of the core. Four are equipped with a pneumatic post, the site of irradiation of one of these channels is surrounded by a boron-cadmium filter. Vertical channels, except for VTC, VEC-11, VEC-5, CC-4-4, are bent to exclude direct neutron and gamma radiation. They are "dry" and do not have protective plugs.

Materials with well-studied properties and high radiation resistance, tested in research reactor building: metal ceramics, metallic beryllium, stainless steel, and aluminum alloys were used in various reactor assemblies.

1.2 Nuclear fuel characteristics

Research reactors are simpler than power reactors and operate at lower temperatures. They need far less fuel, and far less fission products build up as the fuel is used. On the other hand, their fuel requires more highly enriched uranium, typically up to 20% U-235, although some use 93% U-235; while 20% enrichment is not generally considered usable in nuclear weapons, 93% is commonly referred to as "weapons grade".

For the IRT-T reactor, fuel of the type IRT-3M is used. Its characteristics are presented in table 2.

Table 2 – the characteristics of the fuel assembly IRT-3M

Enrichment (U235)	36%
The number of fuel rods in FA	8 (6)
Mass of uranium 235, g	352 (309)
Shell Material	Aluminium alloy
Weight, kg	4.3 (3.7)
Fuel cladding material	Metal ceramics

It is used in a water-water research nuclear research reactor. The IRT-3M fuel assembly was developed as a new generation fuel assembly instead of the IRT-2M fuel assembly.

The fuel assembly of IRT-3M in comparison with the fuel assembly of IRT-2M has an increased heat removal surface from 0.785 m^2 to 1.56 m^2 in the same volume of fuel assembly and is 1.53 times the mass of uranium-235.

There are two modifications of the fuel assembly IRT-3M with four and six fuel elements, providing an additional cavity, which can be used to house experimental devices or rods of the reactor control system. In this case, standard assemblies with eight and modification with six fuel rods for CPS will be used.

1.3 Overview of the software package MCU-5

To calculate the flux densities of gamma radiation and neutrons, the software package MCU-5 will be used. MCU (Monte Carlo Universal) is a project on development and practical use of a universal computer code for simulation of particle transport (neutrons, photons, electrons, positrons) in three-dimensional systems by means of the Monte Carlo method. [2]

The MCU-5 software package is continuation of the MCU-4 one which development was finished in 2006. Since then the software package was largely rewritten and many important capabilities and improvements were included: dynamic memory, parallel calculation, translation from Fortran-77 standard to Fortran-90/95 one, nuclear data updating, new modules were developed (photon and electron–positron transport, uncertainty analysis, feedback), existing modules were rewritten and expanded, and etc. All of those features allow simulating particles transport in the large models like 3D reactor core with detailed power distribution.

1.3.1 Particle transport

In solving problems of criticality by means of the Monte Carlo method one usually uses generations with a fixed total weight of neutron sources in one generation. In this scheme phase coordinates of neutrons of the initial (zero) generation can be chosen arbitrarily. To calculate neutron flux tally, such as the effective multiplication factor and reaction rates in tally areas a random variable ξ is used. Its sample values are elementary estimates xn, calculated for the so-called series, each of which includes a user-defined number of generations NBAT (with NBAT = 1 notions of series and generation coincide). This approach is realized reasoning from the idea that the correlation between series consisted of several generations is less than the correlations between generations. It provides more reliable estimates of result's uncertainty.

Each generation consists of NTOT neutrons. NTOT quantity and the spatial and energy distribution of the first generation neutrons are defined by the user in input data.[3]

A history of each neutron is modeled as the sequence of collisions with nuclei from the birth point to the point of absorption or escape from the system. Phase coordinates of collision, absorption, or escape points are defined by subroutines of the geometrical and physical modules. Reaction rates and others neutron flux tally are calculated by the tally module.

For each fission point the number νn of the secondary fission neutrons and their energy En is defined. These neutrons after normalization are used to make the next generation of NTOT particles. Secondary photons, electrons and positron are also taken into account, if it is required. Simulation of those particles is carried out at the end of each generation.

1.3.2 Fixed source problem

Formally, there are still such notions as series and generations, but this is only for the unification of the code's work. User may set one generation consisting of a single particle per one series.

In the beginning the source module provides NTOT particles for each generation. It can be a generation of any type of particles that is supported by MCU-5 or even a mixed one. These particles create a queue in the bank, the so-called main line. [4]

Further, a trajectory is modeled for every such particle. Secondary particles appearing as a result of the modeling are placed at the end of the same queue. After the main line is finished the code starts working with the secondary particles. This may cause additional particles to be placed into at end of the main queue.

It is possible to turn off modeling of the secondary particles and only do the transport of the main line.

1.3.3 Variance reduction techniques

Nonanalog Monte Carlo or variance reduction techniques allow to focus the particles in regions of interest (e.g. with a small volume) that substantially decreases the number of histories that is necessary to achieve required statistics.

Traditional variance reduction techniques such as a weight window, energy cutoff, energy and geometry splitting with Russian roulette, ring and point detectors are implemented in the transport module of MCU-5.

When simulating collisions of neutrons with nuclei in different energy regions, it is possible to combine data libraries listed in Table 1 by applying the corresponding submodules of the physical module. In other words, it is possible to build physical model from pure multigroup to pure point-wise approximation. Intermediate models are also allowed.

Table 3 – Composition of the MCUDB50 data bank. [4]

Library	Description
ACE/MCU	Library of cross-sections of neutron interaction with nuclei in the epithermal energy region in a point-wise representation obtained from ENDF/B-VII.0 files and other source
BNAB/MCU	Expanded and modified version of the BNAB-93 26-group system of constants
LIPAR	Parameters of nuclide cross-sections in the region of resolved resonances
MULTIC	301-Group library containing the data on the temperature dependence of subgroup parameters of nuclides in the region of unresolved resonances
KORT	Library of neutron physical constants in a point-wise representation for the energy range from 10−5 to 5 eV
TEPCON	Library of 40-group cross-sections for the thermalization region (up to 1 eV)
VESTA	Library for simulating neutron collisions with the nuclei of moderators taking continuous variations in the neutron energy in the thermalization region into account; it is represented in the form of probabilistic tables obtained from the S(α, β) scattering laws
BOFS	Library of generalized phonon spectra of moderators
DOSIM	Library of activation cross-sections in a point-wise representation
ABBNL	Library of 63-group cross-sections used for obtaining “summarized isotope” cross-sections
PHOTONS	Library of multigroup cross-sections of photon generation in neutron interaction with matter based on the data of DLC-41/VITAMIN-C and DLC-184/VITAMIN-B6 libraries
PHOTONT	Multigroup cross section of the photon interaction with matter based on the data of DLC-41/VITAMIN-C and DLC-184/VITAMIN-B6 libraries
BURN5	Data for depletion calculation: half-lives of nuclei, yields of fission fragments, chains of transformations, etc.

Continuation of table 3

SHELLDATA	Library of atomic transitions (LLNL EADL)
PHOTDATA	Library of point-wise cross-sections of photon interaction with matter in the energy range from 100 eV to 100 MeV (LLNL EPDL)
ELECDATA	Library of point-wise cross-sections of electron interaction with matter in a point-wise representation for the energy range from 100 eV to 100 MeV (LLNL EEDL)
POSIDATA	Library of point-wise cross-sections of positron interaction with matter in a point-wise representation for the energy range from 100 eV to 100 MeV (LLNL EEDL)
NEUTRONK	Library of neutron heating cross sections in a point-wise representation for the energy range from 10−5 eV to 20 MeV
PHOTONK	Library of photon heating cross sections in a point-wise representation for the energy range from 10−5 eV to 20 MeV

In the fast energy region, it is possible to use either the constants of the point-wise ACE/MCU library or the 26-group BNAB/MCU library.

In the unresolved resonance region, the cross-sections are calculated using subgroup parameters, Bondarenko's f-factors or probability tables.

In the region of the fully resolved resonances both subgroup and point-wise representation of the cross-sections may be used. If point-wise representation is used, the cross-sections of the most important nuclides are described by "infinite" number of points, because they are calculated using resonance parameters in each energy point during simulation. Such scheme allows calculation using the data on resonance parameters without preliminary preparation of tables of cross-sections and to evaluate temperature effects through analytical dependences of cross-sections on temperature.

Modeling of collisions in the thermal area is carried out either in multigroup transport approximation, or using the model of continuous change of energy considering correlations between changes of energy and scattering angles. In both cases chemical bounds, thermal movement of nucleus and coherent effects for elastic scattering are taken into account.

It is possible to take into account both prompt and delayed neutrons in the fission spectrum.

Generation of photons in the result of neutron reactions is simulated using multigroup approximation.

1.3.4 Photon physics treatment

The interaction of photons with matter can be simulated using both multigroup and point-wise representations of the cross-sections. The following processes are simulated: coherent and incoherent scattering, photoelectric effect and production of electron–positron pairs with the possibility of generation of secondary photons, electrons and positrons. [5]

The cross-sections and the spectra for photoneutron generation are obtained from the analytical dependences for two isotopes characterized by low reaction thresholds and important for reactor applications, namely, deuterium and beryllium.

3.3. Electron and positron physics treatment

The following processes of electron–positron interaction with matter are simulated by the electron–positron submodule (Kulakov, 2010): elastic interaction with the Coulomb field of nuclei of the medium, inelastic scattering on bound atomic electrons with ionization and excitation of atoms, inelastic interaction with the Coulomb field of nuclei and atomic electrons accompanied by bremsstrahlung, and positron–electron annihilation.

It is possible to use the scheme of individual collisions or condensed collision model.

1.3.5 Nuclear data

Two data banks have been developed for the MCU-5 package using different evaluated nuclear data files and data libraries.

Main sources for the data bank development are as follows: BNAB (Manturov et al., 1995), LIPAR (Abagyan et al., 1995); ENDF/B (Section of Nuclear Constants MAGATE, 2013), JENDL (JENDL-4.0, 2013), JEFF (The JEFF Nuclear Data Library, 2013), IRDF (IRDF-90, 1993), RRDF98 (Badikov et al., 1996), VITAMIN/C (VITAMIN-C, 1984), VITAMIN/B6 (VITAMIN-B6, 1996), EPDL97 (Cullen et al., 1997), EADL (Cullen et al., 1991a), EEDL (Cullen et al., 1991b).

The libraries of constants forming the data banks are processed using different widely-used codes: NJOY (MacFarlane and Muir, 1994), PREPRO (PREPRO 2012, 2013), GRUCON (Sinitsa and Rineiskiy, 1993) and our own processing codes. [6]

The main data bank MCUDB50 includes libraries listed in Table 1. This bank contains data for 375 isotopes. The additional MCUDB50RF data bank is based on the RUSFOND (Ignatyuk et al., 2007) library and contains data for 460 isotopes.

1.3.6 Geometry

Geometry description uses the Cartesian coordinate system. The geometry module allows modeling of three-dimensional systems with geometry of arbitrary shape by using combinatorial approach. The code provides about 20 simple bodies (spheres, parallelepipeds, cylinders and etc.), half-spaces and planes that may be combined using intersection, union, and complement operations.

Geometrical zones are constructed from the bodies by means of the combinatorial geometry mentioned above. Any point of a system belongs only to one geometrical zone. The geometrical zone is considered homogeneous. This in particular means that any such zone consists of a single material. [7]

For each geometrical zone user defines three attributes: material, tally zone and tally object numbers. Thus, on the mesh of geometric zones the user defines a system physically, i.e. consisting of materials, and statistically, i.e. consisting of zones and objects for tally purposes. This allows one to obtain tallies in materials, zones and objects separately. Though all the three tally types are based on the mesh of geometric zones they may be rather different shapes. There are no restrictions on the number of materials, geometrical zones and tallies in MCU-5. E.g. in reactor calculations zones may refer to fuel pins and objects to fuel assemblies.

There are means to simplify the description of the geometries by repeating elements. Such elements can be defined by means of lattices and nets. Lattices are created from elements and nets are created from cells. Nets fill a part of space by the cells having the same external form. The cells are closely adjoined to each other. The trajectory inside the cell is calculated using the local coordinate system. The transformation of the coordinates reduces to shifts and do not slow the calculation. One may think of nets as of a regular array of cells. Lattices allow placing any number of copies of the elements in any point of space. Both nets and lattices use prototypes for their cells and elements. The main difference between nets and lattices is their inner presentation in the memory. When the cell prototype is placed into the net no copy of it is created, as is the case with the element prototype. So, the use of nets saves memory and accelerates the calculation a bit. However, the use of lattices has fewer restrictions, because it does not require regularity. In MCU-5 only two levels of geometry are possible. General geometry, including lattices, is the first level and nets are the second one, i.e. element of lattice may contain a nets' description. [8]

A boundary condition is assigned to each external boundary of the system. The condition specifies symmetry or physical properties of the system.

The available boundary conditions are mirror symmetry, rotation symmetry, translation symmetry, black absorption surface, white reflection and boundary condition corresponding to the problem with leakage given by buckling.

Besides the leakage and reflection conditions their combination is allowed. In this case a positive albedo $\alpha < 1$ is set for the outer surface of the system. When the particle's path crosses the surface, α is the probability of reflection and $(1 - \alpha)$ is the probability of absorption.

The special algorithm of the geometry module allows considering effects of double heterogeneity when the fuel elements contain tens of thousands of the fuel kerns.

Moreover, the Woodcock method (Woodcock et al., 1965) gives an opportunity to carry out calculations of complex geometrical objects, which surfaces are not described by planes or surfaces of the second order. Such geometrical objects are, e.g. screw fuel rods or deformed fuel assemblies.

1.3.7 Source

Initial data about spatial, energy and angle particle's distribution are set by source module. There are two types of source definition: simple and complex.

The simple point source is always a point source with isotropic angle distribution. Its energy spectrum can be both delta-function or step function.

The complex source is based on five bodies: rectangular parallelepiped, cylinder or its sector, sphere or its sector, rectilinear hexahedral vertical prism or its sector, and cylinder with internal distribution of birth point probabilities.

Each body may be used to define the so-called primitive source. Any number of primitive sources may be used to model the necessary distribution of particles. For each primitive source the user defines its intensity and probability together with energy and angle distributions. It is possible to specify sources with correlated energy and angular distributions.

Tally objects may be used for geometrically complex source modeling, since source module has no possibilities for combinatorial geometry description. The numbers specify those tally objects, described previously in the geometry module, where particle birth is allowed for the particular primitive source. If the coordinates of the new particle are not inside at least one of the specified objects, they are rejected and the new point of birth is sampled. [9]

It is possible to use lattices to ease the task of defining sources complicated in shape and with some regularity (for example, pin-by-pin distribution in a fuel assembly is shown in Fig. 3). The lattice defines a set of source positions and provides means for distributing intensities and probabilities inside the lattice.

In addition, a surface source is implemented. It allows particles crossing a surface in one problem to be used as the source for a subsequent problem. The decoupling of a calculation into several parts allows detailed design or analysis of certain geometrical regions without having to rerun the entire problem from the beginning each time.

1.3.8 Tally capabilities

MCU-5 allows calculating different tallies of the neutron flux for arbitrary energy ranges and isotopes. The contributions to the tallies are obtained on all particle trajectories of NBAT generations on the basis of collision, absorption and track-length estimators usually used in Monte Carlo particle transport codes. These types of estimators are well-suited for calculating integral quantities of the form:

$$F = \int_{\delta V} dr \int_{\delta \Omega} \mathrm{d}\Omega \int_{\delta E} dE \varphi(r, \Omega, \mathrm{E}) \Phi(r, \Omega, \mathrm{E})$$

Here drdΩdE is the phase space volume where coordinates of a particle are determined by radius-vector r, velocity direction Ω, and particle's energy E; φ is an arbitrary function and Φ is the angular flux distribution function of position, direction, and energy. [10] By selecting an appropriate function φ and adjusting the range of integration, one can produce many useful tallies for nuclear design and analysis. For example, if φ is defined as product of the macroscopic fission cross section and average number of neutrons released per fission event, then Eq. (1) gives the tally of effective multiplication factor rate of the considered model.

Thus the tally module of the MCU-5 code allows calculating the following global tallies for the system as a whole:

— effective multiplication factor;
— effective fraction of delayed neutrons;
— neutron lifetime;
— number of absorptions;
— average lethargy causing fission;
— leakage.

The following tallies are available for arbitrary energy ranges and isotopes in tally areas (tally in materials, zones, objects):

— fluxes over the volume or surface,
— surface current,
— various reaction rates and their macroscopic cross-sections,
— dosimetry reactions rate,
— nuclear heating,
— flux at a point using two approaches: point detector and ring detector,
— neutrons and photons dose characteristics,
— few-group constants of fuel pins and fuel assemblies, scattering and fission matrixes, diffusion coefficient,
— and etc.

For photons, electrons and positrons it is possible to estimate the fluxes, reaction rates, the charge, and the energy to be absorbed by material.

An important condition is consistency and unbiasedness of the estimates. In addition to flexibility, the tally module contains software tools which make it possible to substantiate consistency and unbiasedness of the estimates and to correctly calculate the confidence intervals.

For reliable calculation of statistical errors, the possibility to take into account the correlation of separate generations' contributions into tallies is realized in tally submodule.

Codes created on the base of modules from the MCU-5 software package may be used to solve the following problems:

— nuclear reactor design;

— evaluation of criticality and nuclear safety of objects where nuclear energy is used;
— calculation of the power distribution in the reactor core;
— nuclear heating in the fuel and structural elements;
— simulation of the operation of different types of nuclear reactors;
— evaluation of the special effects that cannot be calculated with the required accuracy on the basis of design codes;
— simulation of radiation protection, evaluation of radiation safety;
— reactor dosimetry;
— calculation of signals from the in-core detectors responsible for controlling the energy release distribution;
— calculations of signals from out-of-core control detectors;
— uncertainty analysis;
— evaluation of radiation characteristics of irradiated fuel;
— simulation of transmutations of actinides and fission products;
— evaluation of the quality of nuclear data libraries;
— evaluation of the quality of experiments;
— calculation of group libraries of macroscopic cross-sections for design codes;
— verification of design codes for nuclear reactor calculations.

1.4 Boron Neutron Capture Therapy

Neutron capture therapy (NCT) is a noninvasive therapeutic modality for treating locally invasive malignant tumors such as primary brain tumors, recurrent head and neck cancer, and cutaneous and extracutaneous melanomas. It is a two-step procedure: first, the patient is injected with a tumor-localizing drug containing the non-radioactive isotope boron-10 (10B), which has a high propensity to capture thermal neutrons. The cross section of the 10B (3,837 barns) is many times greater than that of the other elements present in tissues such as hydrogen, oxygen, and nitrogen. In the second step, the patient is radiated with epithermal neutrons, the source of which is either a nuclear reactor or, more recently, an accelerator. After losing energy as they penetrate tissue, the neutrons are captured by the 10B, which subsequently emits high-energy alpha particles that can selectively kill those tumor cells that have taken up sufficient quantities of ^{10}B. All of the clinical experience to date with NCT is with the non-radioactive isotope boron-10, and this is known as boron neutron capture therapy (BNCT). At this time, the use of other non-radioactive isotopes, such as gadolinium, has been limited to experimental studies, and to date, it has not been used clinically. BNCT has been evaluated clinically as an alternative to conventional radiation therapy for the treatment of high grade gliomas, meningiomas, and recurrent, locally advanced cancers of the head and neck region and superficial cutaneous and extracutaneous melanomas.

BNCT is based on the nuclear capture and fission reactions that occur when non-radioactive boron-10, which makes up approximately 20% of natural elemental boron, is irradiated with neutrons of the appropriate energy to yield excited boron-11 ($^{11}B^*$). This undergoes instantaneous nuclear fission to produce high energy alpha particles (^4He nuclei) and high energy lithium-7 (7Li) nuclei. The nuclear reaction is:

$$^{10}B + n_{th} \rightarrow [11B]^* \rightarrow \alpha + {}^{7}Li + 2{,}31\ MeV$$

Both the alpha particles and the lithium nuclei produce closely spaced ionizations in the immediate vicinity of the reaction, with a range of 5–9 μm, which is approximately the diameter of the target cell. The lethality of the capture reaction is limited to boron containing cells. BNCT, therefore, can be regarded as both a biologically and a physically targeted type of radiation therapy. The success of BNCT is dependent upon the selective delivery of sufficient amounts of 10B to the tumor with only small amounts localized in the surrounding normal tissues. Thus, normal tissues, if they have not taken up sufficient amounts of boron-10, can be spared from the nuclear capture and fission reactions. Normal tissue tolerance is determined by the nuclear capture reactions that occur with normal tissue hydrogen and nitrogen.

A wide variety of boron delivery agents have been synthesized, but only two of these currently are being used in clinical trials. The first, which has been used primarily in Japan, is a polyhedral borane anion, sodium borocaptate or BSH (Na2B12H11SH), and the second is a dihydroxyboryl derivative of phenylalanine, referred to as boronophenylalanine or BPA.

There are two approaches to the implementation of boron neutron capture therapy - the use of research reactors, or the use of accelerators.

Accelerator-based neutron sources are an attractive alternative to nuclear reactors for providing epithermal neutron beams for BNCT [1]. One of the possible reactions to produce neutrons is the 7Li(p,n)7Be. Although this reaction has important difficulties regarding the target construction, its relative high neutron yield, and the fact that it is an endothermic reaction, makes protons on lithium the optimal choice from a neutronic point of view.

The common neutron-producing reaction 7Li(p,n)7Be for accelerator-based BNCT, having a reaction threshold of 1880.4 keV, was considered as the primary source of neutrons.

Also as a source of neutron radiation for BNCT research reactors are used. There are two basic techniques which are used to obtain the appropriate neutron flux: Spectrum shifting and Filtering. In the spectrum shifting technique, the spectrum shifter “or moderator”; placed close to the core; moderates the fast neutrons to the epithermal energies. In the filtering technique, the filters placed near to the beam exit, transmit neutrons of the desired energies and block neutrons of other energies. The spectrum shifting technique gives a much higher flux-to-power ratio, so it is preferable than the filtering technique. A typical spectrum-shifting BNCT beam includes: spectrum shifter, thermal neutron filter, gamma filter, reflector, collimator and shield.

The main challenge in epithermal BNCT beam design is that the desired energy of neutrons is an intermediate energy between fast and thermal energies, both of which are undesired. Therefore, it is difficult to reduce the fast and thermal spectrum components without considerable decrease in the epithermal component.

Any material that could be used to reduce the fast neutrons will reduce epithermal neutrons to some limit and increase gamma radiation which is undesired. Gamma radiation is either emitted directly from the nuclear reactor core or indirectly due to neutron absorption in the beam materials. This gamma radiation needs to be reduced to the acceptable level.

The spectrum shifting materials should have:

1. Scattering cross sections for fast neutrons higher than total cross sections for epithermal neutrons to limit the fast neutron dose;
2. Low absorption cross sections to reduce neutron loss;
3. Low (n, gamma) cross sections to minimize gamma production;
4. Low mass numbers to increase the average energy loss per interaction and to maintain the angular streaming of neutrons in the forward direction.

The neutron flux characteristics at the beam outlet depend mainly on the neutron cross section data of the materials used. Resonance scatterers are promising materials for epithermal neutron beams, but the cross sections change strongly with energy in the resonance region. Thus, there is no single material that could be efficient enough to be used alone as a spectrum shifter for epithermal BNCT beam. When using two or more materials in a multi-layered spectrum shifter, it will be important to determine the order in which the materials are placed. This needs to be based on a robust basis. It could be concluded from the literature that the exchange of materials' positions in the spectrum shifter significantly affects the flux intensity and/or quality. The used method in previous spectrum shifting design studies implies comparing the results of various arrangements to select the most optimum configuration among them. The materials: Al, Mg, C, F, O, H-2, Li-7 and Ni-60 as well as some of their combinations have been used for spectrum shifter design.

1.5 Reactor TRIGA Mark II (Italy)

University of Pavia is equipped with a TRIGA Mark II research nuclear reactor, operating at a maximum steady state power of 250 kW. It has been used for many years to support Boron Neutron Capture Therapy (BNCT) research. An irradiation facility was constructed inside the thermal column of the reactor to produce a sufficient thermal neutron flux with low epithermal and fast neutron components, and low gamma dose. In this irradiation position, the liver of two patients affected by hepatic metastases from colon carcinoma were irradiated after borated drug administration. The facility is currently used for cell cultures and small animal irradiation.

The TRIGA Mark-II reactor was installed by General Atomic (San Diego, California, U.S.A.) in the years 1959 through 1962, and went critical for the first time on march 7, 1962. Operation of the reactor since that time has averaged 220 days per year, without any long outages. The TRIGA reactor is purely a research reactor of the swimming-pool type that is used for training, research and isotope production (Training, Research, Isotope Production, General Atomic = TRIGA). Throughout the world there are more than 50 TRIGA-reactors in operation, Europe alone accounting for 10 of them.

The reactor core consists of 80 fuel elements (3,75 cm in diameter and 72,24 cm in length), which are arranged in an annular lattice. Two fuel elements have thermocouples implemented in the fuel meat which allow to measure the fuel temperature during reactor operation. At nominal power (250 kW), the center fuel temperature is about 200 °C. Because of the low reactor power level, the burn-up of the fuel is very small and most of the fuel elements loaded into the core in 1962 are still there. Should these fuel elements ever become unserviceable, they will be sent back to the United States. Inside the fuel element cladding (aluminum or steel), the fuel is in the form of a uniform mixture of 8 % uranium, 1 % hydrogen and 91 t% zirconium, the zirconium-hydride, being the main moderator. Since the moderator has the special property of moderating less efficiently at high temperatures, the TRIGA-reactor Vienna can also be operated in a pulsed mode (with a rapid power rise to 250 MW for roughly 40 milliseconds). The power rise is accompanied by an increase in the maximum neutron flux density from $1x10^{13}$ $cm^{-2}s^{-1}$ (at 250 kW) to $1x10^{16}$ cm^{-1} (at 250 MW). This negative temperature coefficient of reactivity, as it is called, brings the power level back to approximately 250 kW after the excursion, the maximal pulse rate is 12 per hour, since the temperature of the fuel elements rises to about 360 °C during the pulse and, therefore, the fuel is subjected to strong thermal stress. The reactor is controlled by three control rods which contain boron carbide as absorber material. When these rods are fully inserted into the reactor core, the neutrons continuously emitted from a start-up source (Sb-Be photoneutron source) are absorbed by the rods and the reactor remains sub-critical. If the absorber rods are withdrawn from the core (two of them by an electric motor and one pneumatically), the number of fissions in the core and the power level increases. The start-up process takes roughly one minute for the reactor to reach a power level of 250 kW from the sub-critical state. The reactor can be shut-down either manually or automatically by the safety system. It takes about 1/10 of a second for the control rods to fall into the core. [11]

The thermal column was originally modified to obtain an air channel for organ irradiation in a clinical trial of Boron Neutron Capture Therapy of explanted liver. Since then, the facility has been used for preclinical studies with small animal models and cell cultures. A simulation of the reactor was set-up, using MCNP5 with the neutron source based on criticality calculation. A method to measure the entire neutron spectrum in the animal irradiation position, consisting in an unfolding algorithm using both multi-foil activation results and a priori simulated values demonstrated that the new model is able to reproduce the neutron spectrum reliably at least up to 10 keV, which covers the large majority of the neutron flux in the facility. For gamma dose, a measurement campaign has been conducted with alanine dosimeters, using the same MCNP5 reactor model to separate the different dose components producing a signal in the detectors. Also in this case, the reactor model proved to be robust and suitable for treatment planning and dose evaluations, using the RE factor suggested by literature. Presently, the thermal column of the TRIGA reactor operating in Pavia hosts a large irradiation facility (1 m by 40 cm by 20 cm) well characterized in terms of neutron spectrum and gamma dose, which can be used for different kinds of measurements requiring thermal neutron irradiation with flux values between 10^9 and 10^{10} cm^{-2} s^{-1} and with low gamma contamination. [12]

1.6 Reactor TRR (Iran)

The theoretical and practical researches shows that TRR could be used for clinical BNCT for brain tumors treatment based on the use of the TRR medical room. There is some technical challenge that should be solved for the purpose. The main challenges are the operation of the reactor core in the open pool position and construction of an in-pool beam tube. The result shows that TRR has a good potential to consider it as a pilot facility for BNCT research in the Middle East

The Tehran Research Reactor (TRR) was supplied by the United States under the Atoms for Peace program. The 5-megawatt pool-type nuclear research reactor became operational in 1967 and initially used highly enriched uranium fuel. Light water is used as moderator, coolant and shielding. The TRR core lattice is a 9×6 array containing Standard Fuel Elements (SFEs), Control Fuel Elements (CFEs), irradiation boxes (as vertical tubes provided within the core lattice configuration for long term irradiation of samples and radioisotope production) and graphite boxes (as reflectors). [13]

After the Iranian Revolution the United States cut off the supply of highly enriched uranium (HEU) fuel for the TRR, which forced the reactor to be shut down for a number of years. Due to the nuclear proliferation concerns caused by the use of HEUs and following Reduced Enrichment Research and Test Reactor (RERTR) Programs, Iran signed agreements with Argentina's National Atomic Energy Commission to convert the TRR from highly enriched uranium fuel to low-enriched uranium, and to supply the low-enriched uranium to Iran in 1987–88. TRR core was converted to use Low Enriched Uranium (LEU) fuels in 1993. Fuel elements of TRR are now plate-type U_3O_8-Al with approximately 20% enrichment. In February 2012, Iran loaded the first domestically produced fuel element into the Tehran Research Reactor.

Standard fuel elements of TRR have 19 fuel plates, while CFEs have only 14 fuel plates to accommodate the fork-type control rods. Control of the reactor is accomplished by the insertion or removal of safety and regulating absorber plates, which contain Ag–In–Cd alloy and stainless steel, respectively. Additional control is provided by the inherent negative temperature coefficient of reactivity of the system. [14]

The reactor core is immersed in either section of a two-section, concrete pool filled with water. One of the sections of the pool contains an experimental stall into which beam tubes and other experimental facilities converge. The other section is an open area for bulk irradiation studies. The reactor can be operated in either section.

The reactor experimental facilities in the stall end are as follow:

— Two pneumatic rabbit tubes (for short term irradiation of samples);
— One graphite thermal column;
— One 12"×12" beam tube;
— Four 6" diameter beam tubes;
— One 8" diameter beam tube;
— One 6" diameter through tube.

The TRR core cooling is accomplished by gravity flow of pool water at nominal rate of 500 m^3/hr through the reactor core, grid plate, plenum and into the hold-up tank from where it is pumped through the shell of the heat exchanger and then back into the pool.

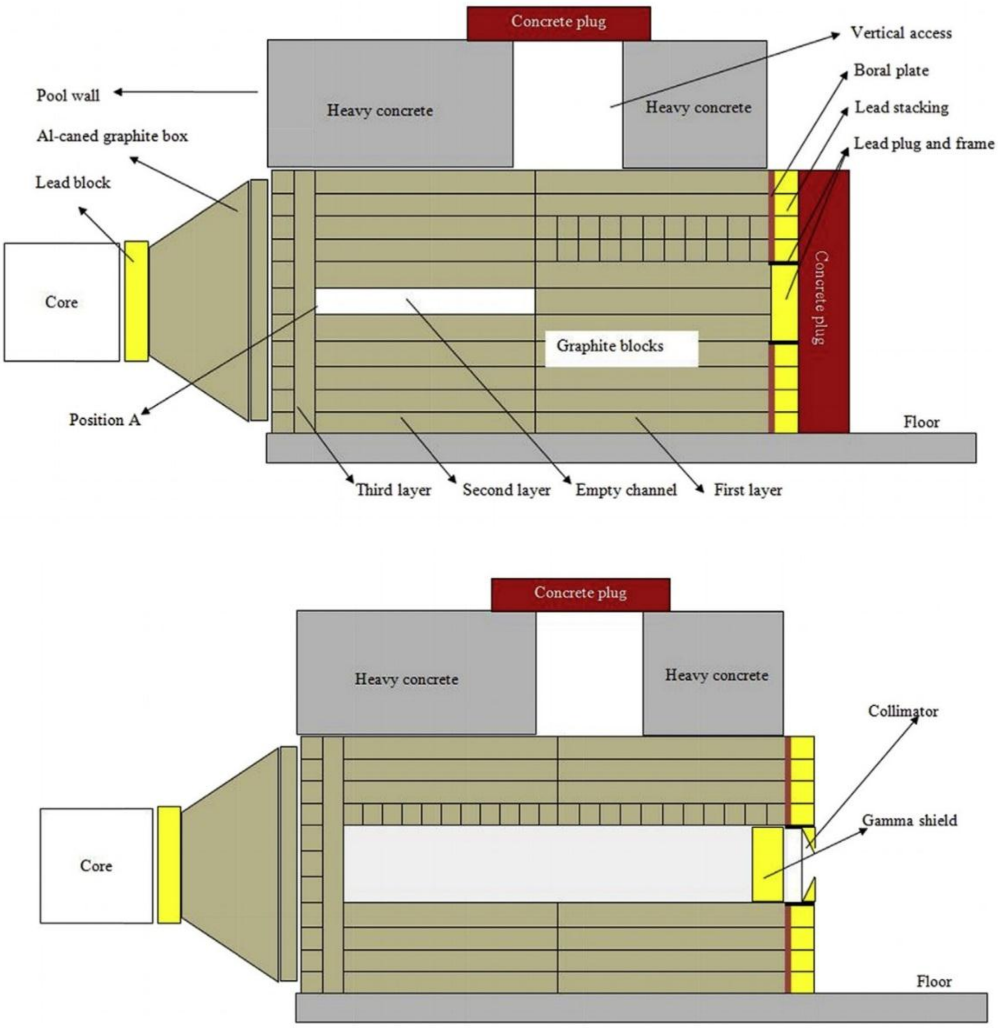

Figure 2 – (Top) thermal column structure, (bottom) new arrangement of graphite blocks in the thermal column

TRR offers a variety of education and exposure services and production of radioisotopes for medical, scientific and industrial centers. One of the primary objectives of the facility is to render services to scientists, engineers and graduate students in nuclear techniques. Tehran research reactor can be utilized for laboratory work involving studies of the reactor core and experiments on neutron diffusion, neutron diffraction, shielding, gamma spectroscopy, boron neutron capture therapy, neutron radiography and Neutron Activation Analysis.

1.7 Reactor IRT MEPhI (Russia)

The first facility in Russia for researching NCT of malignancies at IRT MEPhI nuclear reactor. The established medical irradiation unit will be used to continue the preclinical and clinical trials. This facility is to be used for the treatment of patients with malignant tumors that are resistant to traditional method of therapy.

The IRT reactor refers to pool type reactors in which the core is immersed in an open tank (pool) filled with (normal or heavy) water. In this reactor design, water serves as both a moderator and a neutron reflector, a core cooler and a biological defense. This design has several advantages. These include, above all, the simplicity of the design, due to the fact that all elements of the core and the first cooling circuit are under atmospheric pressure, as well as ease of access to the reactor elements and ease of maintenance.

Table 4 – General information of IRT reactor

Power	6 MW
Core volume	59,3 l
Core heat transfer surface	21,5 m2
The mass of uranium-235 in the core	3,5 kg
Maximum reactivity margin	9
Experimental channels: – horizontal: – vertical:	 10 4

To improve the performance of the reconstruction beam-channel devices HEC-4 and to obtain the characteristics of the neutron beam, which are the most suitable for NCT, the calculations and experimental research on optimization-channel devices were carried out. The following parameters of HEC-4 channel equipment were optimized:

1. Position of the graphite scatterer in the channel that provides maximum thermal neutron flux and minimum dose rate of concomitant secondary photon radiation and fast neutrons;

2. Position and thickness of the bismuth filter in the channel as a function of the ratio between the neutron and photon fluxes at the channel outlet; 3. Arrangement and thickness of the lead collimators installed in the channel; 4. Material, position, and size of neutron moderators in HEC-4 channel;

5. Effect of the core reflector composition on the characteristics of radiation fields at the outlet of HEC-4.

6. Influence of the walls of the irradiation room and the object under irradiation on the characteristics of radiation fields at the outlet of HEC-4. In order to form a neutron beam, a graphite scatter of 150 mm in diameter and 400 mm long was installed in the tangential horizontal channel against the core. In the initial phases of work, the optimum position of the graphite scatter was determined experimentally. To reduce the photon dose rate from the thermal protection of the reactor, two conic lead collimators were set in the fifth section of the shutter. The neutron beam was guided from the HEC-4 channel through the lead diaphragm, where inserts with output diameters of 30, 40, and 60 mm can be set. A shield of powdered 6 Li-enriched Li2CO3 was installed on the lead diaphragm in order to decrease the thermal neutron flux outside the beam.

Thus, on the basis of the tangent HEC-4 channel of IRT MEPhI reactor, the first facility for irradiation experiments and preclinical testing of NCT was established. Experiments using the created irradiation facility were conducted on small dogs having spontaneous tumors. All dogs were subjected to thorough clinical examination and had their peripheral blood tested. To detect the dose of photon and neutron radiation, several sensors were placed on the dog's body surface. As a result of NCT method testing, all dogs demonstrated the regression of tumors. However, there were such specific effects of radiation as radiation burns, alopecia, which indicated the significant fraction of fast neutrons in the beam spectrum. [15]

Currently, the development of technologies for neutron capture therapy is carried out on the basis of IRT-T TPU research nuclear reactor. The operation lifetime of the IRT-T research nuclear reactor is approved up to year 2034. Moderator, coolant and biological protection of reactor is represented by distilled water. Core reflector is beryllium. Reactor capability is 6 MW. There are 10 horizontal channels and 4 vertical channels. Horizontal channel number 1 is suitable for the study of NCT technology. This channel already provides for the necessary neutron flux and energy.

2.1 Integral flux densities of neutrons and γ-quanta in the horizontal tangential channel of the reactor

At present, many papers have been published on the study of the neutron spectrum of fission of ^{235}U on thermal neutrons. To describe them Watt function is widely used [16, 17]:

$$N(E) = \sqrt{\frac{2}{\pi e}} \cdot \mathrm{sh}\sqrt{2E} \cdot e^{-E}, \tag{1}$$

where *N(E)* – the number of neutrons with energy from *E* to *E*+Δ*E* (MeV) per emitted neutron; $\pi = 3{,}1415$, $e = 2{,}7183$, sh – hyperbolic sine. This formula is in good agreement with measurements in the energy range from 0.075 to 17 MeV. There is another approximation of the fission spectrum according to the Cranberg formula [16]:

$$N(E) = 0{,}459 e^{-E/0.965} \mathrm{sh}\sqrt{2{,}29E}. \tag{2}$$

There is a simpler formula proposed in [2]:

$$N(E) = 0{,}770\sqrt{E} e^{-0.776E}. \tag{3}$$

At high neutron energies ($E \geq 5$ MeV), expression (1) gives too high a value *N(E)*. In the energy range from 4 to 14 MeV, the fission spectrum can be approximated by a simple exponential [3]:

$$N(E) = 1{,}75 e^{-0.776E}. \tag{4}$$

Here the error is 7%. Retarded neutrons occupy such a small fraction of the total number of neutrons (<1%) that, for the purposes of calculating protection, the spectrum described by formulas (1 - 4) is essentially the spectrum of all fission neutrons. Table 1 shows the neutron spectrum of ^{235}U fission by thermal neutrons, calculated by the Cranberg formula.

Knowing the distribution of reactor power through the core, it is easy to estimate the intensity of neutron and γ-radiation. If the specific power is *P(r)* W/cm^3, then the distribution of the intensity of the instantaneous emission with respect to energies during fission can be written in the following form [17]:

$$N_{prompt\ neut.}\ (r,E) = 7{,}75 \cdot 10^{10} N(E)P(r)\ prompt\ neut./(cm^3 \cdot s) \qquad (5)$$

$$N_{pr.\gamma}\ (r,E) = 3{,}1 \cdot 10^{10} N(E)P(r), \qquad (6)$$

where *N(E)* –the spectrum of neutrons or γ-quanta normalized to one particle, r – is the radius vector. The spectrum of instantaneous γ-radiation is given in Table 5.

Table 5 – The neutron spectrum in the fission of ^{235}U by thermal neutrons.

E, MeV	Fraction of neutrons with energy E per interval 1 *MeV*.	Fraction of neutrons with energy higher *E*	*E, MeV*	Fraction of neutrons with energy E per interval 1 *MeV*.	Fraction of neutrons with energy higher *E*
0,00	0,000	1,000	6,50	0,0127	0,01669
0,25	0,290	0,948	7,00	$8{,}77 \cdot 10^{-3}$	0,01138
0,50	0,347	0,867	7,50	$6{,}01 \cdot 10^{-3}$	$7{,}72 \cdot 10^{-3}$
0,75	0,358	0,779	8,00	$4{,}10 \cdot 10^{-3}$	$5{,}22 \cdot 10^{-3}$
1,00	0,347	0,690	8,50	$2{,}79 \cdot 10^{-3}$	$3{,}52 \cdot 10^{-3}$
1,25	0,325	0,606	9,00	$1{,}88 \cdot 10^{-3}$	$2{,}36 \cdot 10^{-3}$
1,50	0,298	0,528	9,50	$1{,}27 \cdot 10^{-3}$	$1{,}58 \cdot 10^{-3}$
1,75	0,268	0,457	10,00	$8{,}56 \cdot 10^{-4}$	$1{,}06 \cdot 10^{-3}$
2,00	0,239	0,394	10,50	$5{,}74 \cdot 10^{-4}$	$7{,}05 \cdot 10^{-4}$
2,50	0.184	0,2884	11,00	$3{,}84 \cdot 10^{-4}$	$4{,}69 \cdot 10^{-4}$
3,00	0,138	0,2082	11,50	$2{,}56 \cdot 10^{-4}$	$3{,}11 \cdot 10^{-4}$
3,50	0,102	0,1486	12,00	$1{,}70 \cdot 10^{-4}$	$2{,}06 \cdot 10^{-4}$
4,00	0,0738	0,1050	13,00	$7{,}48 \cdot 10^{-5}$	$8{,}97 \cdot 10^{-5}$
4,50	0,0528	0,0736	14,00	$3{,}26 \cdot 10^{-5}$	$3{,}88 \cdot 10^{-5}$
5.00	0,0375	0,0512	15,00	$1{,}42 \cdot 10^{-5}$	$1{,}67 \cdot 10^{-5}$
5.50	0,0263	0,0354	16,00	$6{,}07 \cdot 10^{-6}$	$7{,}10 \cdot 10^{-6}$
6,00	0,0184	0,0244	17,00	$2{,}59 \cdot 10^{-6}$	$3{,}00 \cdot 10^{-6}$

If the average value of the specific power of the reactor *W/V* is substituted for *P(r)* in (2) and (3) (W is the power, W, V is the volume of the core, cm^3), then the obtained intensities of the neutron and γ–radiation will be higher , than actually existing at the edge of the core of the reactor. This is due to the fact that instead of fuel rods there are blocks of a neutron reflector made of beryllium or graphite.

Table 6 – Characteristics of instantaneous γ-radiation, which arises in fission in different energy intervals.

Energy spacing, *MeV*.	Number of gamma–quanta per division in the indicated energy interval	The number of gamma quanta per fission, having an energy of E MeV/quantum
0,25 – 0,75	3,1	
0,75 – 1,25	1,9	3,2 at 1 MeV
1,25 – 1,75	0,84	0,8 at 1,5 MeV
1,75 – 2,25	0,55	
2,25 – 2,75	0,29	0,85 at 2,3 MeV
2,75 – 3,25	0,15	0,15 at 3, 0 MeV
3,25 – 3,75	0,062	
3,75 – 4,25	0,065	
4,25 – 4,75	0,024	
4,75 – 5,25	0,019	0,2 at 5,0 MeV
5,25 – 5,75	0,017	
5,75 – 6,25	0,007	
6,25 – 6,75	0,004	

Table 7 – Integral densities of neutron fluxes at the tangential of the channel of the reactor adjacent to the core.

Interval of neutron energies, *MeV*.	Integral flux density of neutrons, $n{\cdot}cm^{-2}{\cdot}s^{-1}$.	Interval of neutron energies, *MeV*.	Integral flux density of neutrons, $n{\cdot}cm^{-2}{\cdot}s^{-1}$.
Thermal neutrons	$1{,}41{\cdot}10^{13}$	2,0 – 3,0	$2{,}29{\cdot}10^{11}$
0,1 – 0,25	$2{,}91{\cdot}10^{11}$	3,0 – 4,0	$9{,}12{\cdot}10^{10}$
0,25 – 0,50	$2{,}95{\cdot}10^{11}$	4,0 – 5,0	$8{,}24{\cdot}10^{10}$
0,50 – 1,00	$3{,}78{\cdot}10^{11}$	5,0 – 6,0	$3{,}93{\cdot}10^{10}$
1,00 – 1,50	$2{,}61{\cdot}10^{11}$	6,0 – 7,0	$2{,}12{\cdot}10^{10}$
1,50 – 2,00	$2{,}41{\cdot}10^{11}$	7,0	$2{,}25{\cdot}10^{10}$

In the tables 6 and 7 (3) and (4), taking into account their spectra, normalized to one particle, with the average specific power of the reactor (reactor power 6 MW, reactor core volume 60.0), the intensity distributions of the instant neutrons and gamma quanta, 60,0×57,2×42,9 см^3).

As noted above, during the operation of the apparatus the main neutron source in the core is the instantaneous neutrons accompanying the fission. Investigation of γ-radiation sources is more difficult. It can be divided into two types: instant and late. The most important is the first species that accompanies the division. However, the total energy liberated by the second type, which emits fission products, as well as those arising from the capture of neutrons by the nuclei of structural materials, can have the same order as the energy of the first. The total energy released by the γ-radiation that accompanies fission is $2.4 \cdot 10^{14}$ $MeV \cdot kW^{-1} \cdot s^{-1}$ [18]. The total energy released by the retarded type is $2,3 \cdot 10^{14} \cdot 0.93\text{-}t^{-0,2}$ $MeV \cdot kW^{-1} \cdot s^{-1}$ (t – is the time, s), which in the equilibrium state approximately corresponds to the energy released instant γ-quanta. [18, 19]

The value of the intensity of the γ-radiation, equal to two-thirds of the equilibrium, is reached in just a few minutes of operation of the reactor. On the other hand, when the total energies of these two types are equal, the average energy of the retarded γ-quanta is 0.7 MeV, which is much lower than the instantaneous mean energy (2.5 MeV) [20]. Thus, the γ-radiation produced by fission is the dominant factor in the calculation of the protection.

Nevertheless, for calculations it is necessary to use the resulting spectrum and intensity of both types. In work [21] the spectrum of γ-radiation of fission products during the operation of the reactor is given in Table 4.

The value of the intensity of the γ-radiation, equal to two-thirds of the equilibrium, is achieved in a few minutes of operation of the reactor. On the other hand, the average energy of the delayed γ-quanta is 0.7 MeV, which is much lower than the internal energy (2.5 MeV) [21]. Thus, the γ-radiation produced by fission is the dominant factor in the calculation of the protection.

Nevertheless, for calculations it is necessary to use the resulting spectrum and intensity of both types. In work [18] the spectrum of γ-radiation of fission products during the operation of the reactor is given in Table 7.

Table 8 – Spectrum of γ-radiation of fission products during reactor operation.

Interval of energies, MeV.	$E_{ef.}$, MeV.	N_γ
0,10 – 0,40	0,4	0,202
0,40 – 0,90	0,8	0,605
0,90 – 1,35	1,3	0,062
1,35 – 1,80	1,7	0,078
1,80 – 2,20	2,3	0,037
2,20 – 2,60	2,5	0,015
2,60	2,8	0,001

When calculating the integral density of γ-quantum fluxes (Table 8), The data of Table 8 and the expression (7) with a factor of 2 on the right-hand side were used:

$$N_{quanta}(E) = 6{,}2 \cdot 10^{10} \cdot N_\gamma \cdot W/V. \tag{7}$$

Table 8 – Integral flux density of γ-quanta at the tangential of the channel of the reactor adjacent to the core

Interval of energies, MeV.	*$E_{ef.}$, MeV.*	Integral flux density of γ-quanta, *quantum/cm² s.*
0,10 – 0,40	0,4	$4{,}19 \cdot 10^{13}$
0,40 – 0,90	0,8	$1{,}56 \cdot 10^{13}$
0,90 – 1,35	1,3	$5{,}77 \cdot 10^{12}$
1,35 – 1,80	1,7	$3{,}54 \cdot 10^{12}$
1,80 – 2,20	2,3	$3{,}82 \cdot 10^{12}$
2,20 – 2,60	2,5	$4{,}22 \cdot 10^{12}$
2,60	2,8	$5{,}43 \cdot 10^{12}$

$E_{ef.}$ – effective energy of the considered energy interval, N_γ – normalized to 1 quantum - the output of γ-quanta.

The data in Table 5 is in good agreement with the experimental results obtained on high-flow horizontal channels [19].

2.2 Calculation of neutron and gamma flux densities

In order to calculate the neutron and gamma flux densities, an IRT-T reactor model was built using the MCU software package. The HEC-1 horizontal experimental channel in the model is divided into 89 separate zones, in which the value of flux densities is assumed to be equal throughout the zone.

In the software package MCU-5, the flux densities are represented as functional. The functional is the sum of the lengths of the segments in the considered volume of a separate zone. To convert to absolute values, you must use the following formula. [22]

$$\Phi = f \cdot P \cdot v_f / \left(E_f \cdot \pi \cdot R^2 \cdot l\right) \quad (8)$$

when f – functional, cm-1,

P – power of the reactor IRT-T,

vf – number of secondary neutrons on the fission of a uranium nucleus 235,

Ef – energy of one fission,

R – radius of HEC-1,

l – length of one zone.

The values of the spectra normalized to the unit are presented in the table (9) below.

Table 10 – The values of the spectra normalized to the unit

Energy, eV	functional, cm	Energy, eV	functional, cm
0,00E+00	0,000551	6,30E+04	0,001680
1,00E-04	0,000000	8,00E+04	0,001693
1,28E-04	0,000000	1,00E+05	0,002528
1,60E-04	0,004222	1,15E+05	0,004215
2,00E-04	0,000007	1,35E+05	0,002118
2,55E-04	0,000841	1,60E+05	0,000000
3,20E-04	0,000000	1,90E+05	0,001681
4,00E-04	0,000000	2,20E+05	0,000006
5,00E-04	0,000000	2,55E+05	0,000840
6,30E-04	0,000846	3,00E+05	0,000841
8,00E-04	0,000000	3,60E+05	0,002528
1,00E-03	0,000915	4,25E+05	0,000011
1,28E-03	0,001901	5,00E+05	0,004209
1,60E-03	0,002389	5,75E+05	0,001682
2,00E-03	0,000016	6,60E+05	0,003364
2,55E-03	0,003028	7,60E+05	0,004214
3,20E-03	0,003841	8,80E+05	0,004411
4,00E-03	0,009942	1,00E+06	0,000841

5,00E-03	0,011963	1,10E+06	0,000840
6,30E-03	0,012382	1,20E+06	0,002522
8,00E-03	0,015474	1,30E+06	0,001681
1,00E-02	0,026269	1,40E+06	0,003255
1,28E-02	0,025400	1,50E+06	0,000841
1,60E-02	0,039731	1,60E+06	0,002522
2,00E-02	0,070937	1,70E+06	0,002148
2,55E-02	0,059459	1,80E+06	0,001683
3,20E-02	0,082835	1,90E+06	0,000000
4,00E-02	0,084689	2,00E+06	0,000000
5,00E-02	0,094450	2,10E+06	0,001743
6,30E-02	0,096023	2,20E+06	0,000000
8,00E-02	0,068539	2,30E+06	0,001681
1,00E-01	0,058000	2,40E+06	0,000000
1,28E-01	0,027440	2,50E+06	0,000000
1,60E-01	0,009141	2,60E+06	0,000000
2,00E-01	0,003641	2,70E+06	0,000000
2,55E-01	0,004005	2,80E+06	0,000000
3,20E-01	0,003422	2,90E+06	0,000841
4,00E-01	0,002383	3,00E+06	0,000841
5,00E-01	0,001699	3,20E+06	0,000000
6,30E-01	0,006747	3,40E+06	0,000000
8,00E-01	0,001688	3,60E+06	0,000840
1,00E+00	0,001693	3,80E+06	0,000841
1,28E+00	0,004075	4,00E+06	0,000000
1,60E+00	0,000802	4,20E+06	0,000000
2,00E+00	0,000847	4,40E+06	0,000000
2,55E+00	0,002628	4,60E+06	0,000000
3,20E+00	0,001688	4,80E+06	0,000000
4,00E+00	0,004170	5,00E+06	0,000439
5,00E+00	0,002539	5,20E+06	0,000000
6,30E+00	0,000840	5,40E+06	0,000000
8,00E+00	0,004202	5,60E+06	0,000840
1,00E+01	0,007855	5,80E+06	0,000000
1,28E+01	0,001701	6,00E+06	0,000000
1,60E+01	0,002539	6,20E+06	0,000000
2,00E+01	0,002582	6,40E+06	0,000000
2,55E+01	0,000000	6,60E+06	0,000000
3,20E+01	0,003369	6,80E+06	0,000000
4,00E+01	0,001978	7,00E+06	0,000000
5,00E+01	0,003369	7,20E+06	0,000000
6,30E+01	0,000852	7,40E+06	0,000000
8,00E+01	0,001682	7,60E+06	0,000000
1,00E+02	0,001687	7,80E+06	0,000000
1,28E+02	0,004216	8,00E+06	0,000000
1,60E+02	0,003260	8,20E+06	0,000000

2,00E+02	0,000841	8,40E+06	0,000000
2,55E+02	0,000000	8,60E+06	0,000000
3,20E+02	0,002534	8,80E+06	0,000000
4,00E+02	0,002539	9,00E+06	0,000000
5,00E+02	0,000852	9,20E+06	0,000000
6,30E+02	0,000841	9,40E+06	0,000000
8,00E+02	0,002522	9,60E+06	0,000000
1,00E+03	0,003691	9,80E+06	0,000000
1,28E+03	0,000000	1,00E+07	0,000000
1,60E+03	0,002522	1,05E+07	0,000000
2,00E+03	0,001699	1,10E+07	0,000000
2,55E+03	0,003344	1,15E+07	0,000000
3,20E+03	0,003577	1,20E+07	0,000000
4,00E+03	0,001701	1,25E+07	0,000000
5,00E+03	0,001682	1,30E+07	0,000000
6,30E+03	0,002593	1,35E+07	0,000000
8,00E+03	0,001682	1,40E+07	0,000000
1,00E+04	0,000000	1,45E+07	0,000000
1,28E+04	0,000006	1,50E+07	0,000000
1,60E+04	0,002522	1,55E+07	0,000000
2,00E+04	0,001763	1,60E+07	0,000000
2,55E+04	0,001705	1,65E+07	0,000000
3,20E+04	0,001687	1,70E+07	0,000000
4,00E+04	0,001682	1,75E+07	0,000000
5,00E+04	0,004203	1,80E+07	0,000000

The total neutron flux density at the channel exit is equal to – 2,35·109 n/cm^{2}s.

The figure 2 shows the dependence of the flux density on the neutron energy at the channel exit.

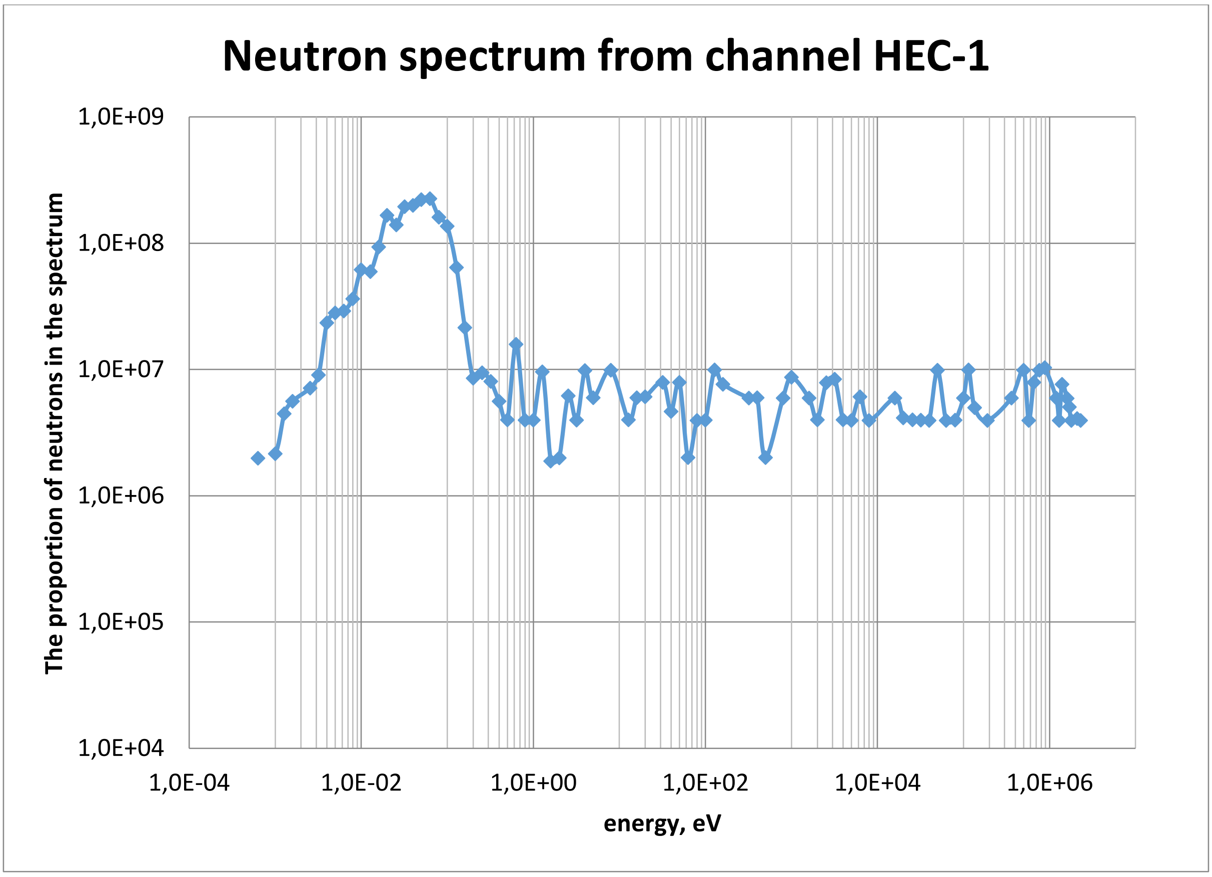

Figure 2 – total neutron flux density at the channel exit

The channel exit is rather far relative to the reactor core, therefore the values of the flow densities will be small. In this case, in order to increase the "smoothness" of the distribution of the neutron flux density, it is necessary to increase the simulation time of the processes in MCU-5, or the computational powers. Due to the limitations of the project, this is not possible. Nevertheless, for estimated calculations, the data obtained at the input of the simulation have an acceptable error.

2.3 Spectrum of gamma radiation at the channel exit

The translation of functionals for gamma radiation is carried out similarly to the translation for neutrons. The simulation results are presented in the table.

Table 10 – Spectrum of gamma radiation

Energy, eV·10^6	ϕ-tional, cm·10^{-6}	Vg	gamma flux density, *quantum/cm^2 s.*
1,00	7,38	9,31	$2{,}12\cdot10^{10}$
3,00	1,02	0,75	$2{,}36\cdot10^{8}$
5,00	0,341	0,099	$1{,}04\cdot10^{7}$
7,00	0,0720	0,0154	$3{,}42\cdot10^{5}$
9,00	0,124	0,0029	$1{,}11\cdot10^{5}$

2.4 Estimated calculation of biological protection

For safe work of the discussing personnel and patients for boron-neutron capture therapy, it is required to apply protective materials that reduce the level of ionizing radiation to the values required by radiation safety standards.

Protection against ionizing radiation is classified according to the following characteristics: by purpose, type, layout, shape and geometry.

Purpose of protection:

— Biological protection should ensure a reduction in the radiation dose of the personnel servicing the installation to the maximum permissible levels;

— Radiation protection should provide an acceptable level of radiation damage (change in strength characteristics, destruction of organic compounds, radiolysis of water) of structural and protective materials exposed to radiation;

— Thermal protection should provide an acceptable level of radiation energy release and temperature distribution in protective compositions.

— By type of protection:

— Solid - completely surrounds the radiation sources;

— Separate - consists of primary, surrounding the most powerful sources of radiation (for example, primary protection of the reactor core), and secondary, designed to protect against radiation sources between it and the primary protection (for example, the coolant system of a nuclear reactor);

— Shadow - set between the radiation source and the protected area. The size of this area is limited by the “shadow” cast by the protection. Such protection is used with limitations of its mass and dimensions;

— Partial - weakened protection in areas with elevated permissible exposure levels (for example, for areas of limited personnel access).

Protection Layout:

— Homogeneous - protection consists of one material;

— Heterogeneous - protection consists of a set of different materials.

In order to solve specific problems of calculating protection, in general, the following initial data should be known:

— maximum permissible radiation levels for this category of persons and adopted dose rate or dose tolerance factors in designing protection for a certain type of premises and category of persons (this information is contained in the currently valid radiation safety standards and basic sanitary rules for ensuring radiation safety);

— the duration of the working time and distance from the radiation source to the detector;

— radiation characteristics of all radiation sources that are protected from (specific and total activities, gamma equivalents, the spectral composition of the primary radiation, the spatial – energy distribution of the scattered radiation at the boundary of the medium);

— the size and geometric shapes of the radiation sources, the source material (its chemical composition and density), the distribution of activity by source, the presence of filter screens (walls) of the source;

— location (layout) of sources relative to each other and this service area;

— the presence in the workplace of other sources and types of ionizing radiation to ensure that the sum of exposure to a worker of all types of external and internal exposure does not exceed one dose limit (PD);

— protective shields material, constructive implementation of protection (full shadow, flat. spherical, homogeneous, heterogeneous, etc.);

— the presence in the irradiation zone of radiation-sensitive equipment, photographic materials, etc.

the method of calculating the protection, making it possible to quickly and with sufficient accuracy to determine its thickness, taking into account any combination of the listed initial parameters.

The tasks of calculating the protection against ionizing radiation are reduced to the calculation of the detector readings at the point of detection inside or outside the protective environment. In the general case, to solve this problem, it is necessary to know the space-time energy-angular dependence of the particle flux density. [23]

For effective protection against γ - radiation, materials with a high-Z and high density are used, for example: lead, concrete, steel, etc. In number of these materials, the best absorption rate of γ - rays has lead, the application of which, as a protection against exposure to γ-radiation is limited by its low melting temperature (327 0 C), so the metals using in hot zones are: tungsten (atomic number Z=74; density ρ=19.3 g / cm3), tantalum (atomic number Z=73; density ρ=16.6 g / cm3), titanium (atomic number Z=22; density ρ=4.54 g/cm3), etc. Its high atomic number and high specific gravity make it possible to use the materials to create a space-saving protection. An important means of ensuring individual radiation safety of employees working in the area of radiation sources or radioactive contamination is protective clothing, which, depending on the application should provide a high level of biological protection of personnel. At present day, many modern materials used for protective suits are high-filled composite materials based on a polymer (rubber, rubber resin, plastic, etc.) with special filler of absorber compounds. In the capacity of fillers such metals as lead, barium, cadmium, molybdenum, tungsten, their carbides, lanthanides, and various other compounds are used.

2.5 Initial data

The obtained values of gamma-radiation and neutron fluxes at the channel exit will be the initial data for the estimated calculation of protection.

According to the standards of radiation safety, it is necessary to ensure the dose received by personnel within 1.4 mR(roentgen) per hour.

The protection material is plain concrete.

2.6 Calculation of protection against gamma radiation

To begin, we obtain the dose rate values for all groups.

In table 12 we write the main values for the calculation.

Table 12 – values for the calculation of biological protection.

Energy, MeV	gamma flux density, *quantum/cm² s.*	dose rate, P/h	$\mu_{ma}, cm^2/kg$	k
1,00	$2{,}12 \cdot 10^{10}$	2280,59	63,5	16300
3,00	$2{,}36 \cdot 10^{8}$	76,18	63,5	5440
5,00	$1{,}04 \cdot 10^{7}$	5,60	63,5	400
7,00	$3{,}42 \cdot 10^{5}$	0,26	63,5	18,4
9,00	$1{,}11 \cdot 10^{5}$	0,11	63,5	7,65

To calculate the thickness of protection we use the method of competing lines. The method of competing lines allows to move from the calculation of protection from non-mono-energy sources to the calculation of protection of mono-energy sources using universal tables. It is necessary to allocate energy intervals with a certain value of energy and the corresponding percentage.

The greatest thickness of protection will correspond to the main line of the spectrum, which we denote by xm. The spectrum line corresponding to the next largest protection thickness is called the competing spectrum line. Denote this thickness of protection xk. Protection thickness is defined as:

$$x = x_k + \Delta_{1/2} \tag{9}$$

when $\Delta_{1/2}$ – the largest value of the half-attenuation layers for the main and competing lines.

As a result of calculations, the thickness of ordinary concrete required for protection against gamma radiation is 215.57 cm.

2.7 Calculation of protection against neutron radiation

Table 13 – values for the calculation of biological protection.

Energy, MeV	Flux, $n \cdot cm^{-2} \cdot s^{-1}$	k
thermal	$2{,}17 \cdot 10^9$	$3{,}10 \cdot 10^6$
0,005	$1{,}27 \cdot 10^8$	$2{,}95 \cdot 10^5$
1	$5{,}74 \cdot 10^7$	$3{,}02 \cdot 10^6$

To calculate the attenuation of the absorbed dose rate of fast neutrons, we use the data from [10]. For concrete with $\rho = 2{,}3$ t/m^3, the elimination cross-section is 0,08 cm^{-1}, respectively. The attenuation rate for fast neutrons should be $2{,}45 \cdot 10^7$. The thickness of the concrete layers in this case is 147 cm.

The absorbed dose rate of intermediate neutrons is determined by their shares in the spectrum of the particles falling on the defense and the accumulation factor, which for concrete with $\rho = 2{,}3$ t/m^3 is 1,6. The thickness of the protection of concrete to attenuate such neutrons will be equal, respectively, and 154 cm. These layers will weaken thermal neutrons to significantly lower doses than the maximum allowable.

3 Social responsibility

Nowadays one of the main ways to radical improvement of all prophylactic work referred to reduce Total Incidents Rate and occupational morbidity is the widespread implementation of an integrated Occupational Safety and Health management system. That means combining isolated activities into a single system of targeted actions at all levels and stages of the production process.

Occupational safety is a system of legislative, socio-economic, organizational, technological, hygienic and therapeutic and prophylactic measures and tools that ensure the safety, preservation of health and human performance in the work process [30].

Rules for labor protection and safety measures are introduced in order to prevent accidents, ensure safe working conditions for workers and are mandatory for workers, managers, engineers and technicians.

A dangerous factor or industrial hazard is a factor whose impact under certain conditions leads to trauma or other sudden, severe deterioration of health of the worker [30].

A harmful factor or industrial health hazard is a factor, the effect of which on a worker under certain conditions leads to a disease or a decrease in working capacity.

3.1 Analysis of hazardous and harmful factors

The working conditions in the workplace are characterized by the presence of hazardous and harmful factors, which are classified by groups of elements: physical, chemical, biological, psychophysiological.

The main elements of the production process that form dangerous and harmful factors are presented in Table 1.

Table 14 – The main elements of the production process, forming hazardous and harmful factors

Name of the types of work and the parameters of the working process	FACTORS GOST 12.0.003-74 Occupational safety standards system		Normative documentation
	Harmful	Dangerous	
Work on PC, Division for Nuclear-Fuel Cycle	The impact of radiation (HF, UHF, SHF, etc.)	–	SanPiN 2.2.2 / 2.4.1340-03 Sanitary-epidemiological rules and regulations. "Hygienic requirements for personal computers and organization of work"
	–	Electricity	GOST 12.1.038-82 Occupational safety standards system.electrical safety
	–	Fire	Fire and explosion safety of industrial installations GOST R12.1.004-85 SSBT

The following factors effect on person working on a computer.

Physical:

— temperature and humidity;
— noise;
— static electricity;
— electromagnetic field of low purity;
— illumination;
— presence of radiation.

Psycho physiological dangerous and harmful factors are divided into:

— physical overload (static, dynamic);
— mental stress (mental overstrain, monotony of work, emotional overload).

The master thesis was written for a personal computer, using the MCU-5 program for calculations.

The masters working on a computer are affected by the following factors: temperature, noise, low-purity electromagnetic field, light intensity, mental overvoltage, and work monotony.

The layout of the workplace is made with the foreseeing of these impacts. Organized debugging of ventilation and air conditioning. The heating system provides constant and uniform heating of the air. The noise level on the PC at the workplace does not exceed 50 dB. What is required for work is located in the zone of easy reach of the working space.

The height of the working surface of the table and height of the working surface with the keyboard is 630 mm. The width of the table is 700 mm and the length is 1440 mm. The legroom has a height of 600 mm, a width of 800 mm, a depth at the level of the knees of 550 mm and at the level of the elongated legs of 680 mm.

The work chair is liftable and adjustable in height and angle of inclination of the seat and backrest, as well as the distance of the backrest to the front edge of the seat. Seat height above floor level is 420 mm. The design of the working chair provides a width and depth of the seat surface of 400 mm.

The monitor is located at a distance of 700 mm from the user. The keyboard is located on the table surface at a distance of 100 mm from the edge. The keys have a concave surface, quadrangular in shape with rounded corners. Since the keyboard is mechanical, the key design provides the operator with a click sensation. The key color is black with white symbols.

3.2 Justification and development of measures to reduce the levels of hazardous and harmful effects, and eliminate their influence

3.2.1 Organizational arrangements

All personnel are required to know and strictly observe the safety rules. The training of personnel in occupational safety and industrial sanitation consists of introductory briefing and briefing at the workplace by the responsible person.

The qualification commission or by the person responsible for the workplace check the knowledge of safety rules after training at the workplace. After that, commission assign the safety qualification group corresponding to the employee's knowledge and experience of work and issue a special certificate.

Persons serving electrical installations must not have injuries and illnesses that interfere with manufacturing activity. The state of health is established by medical examination before being employed.

3.2.2 Technical Activities

The rational layout of the workplace provides for a clear order and permanent placement of objects, means of labor and documentation. Object, what is required to perform the work more often, should be located in the easy reach of the workspace, as shown in Figure 4.

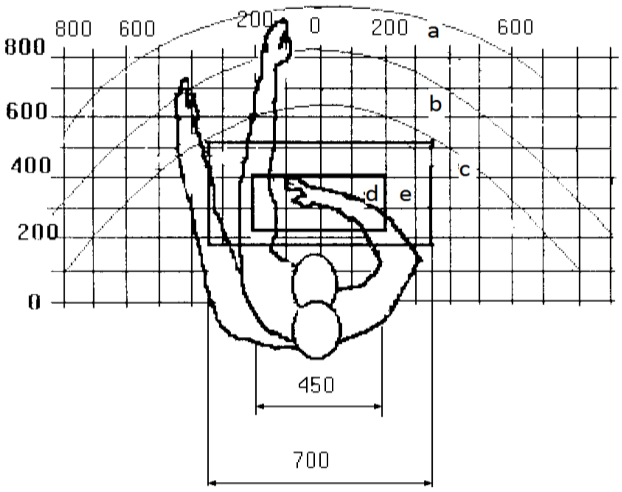

Figure 4 – Hand reach zones in the horizontal plane:
a – zone of maximum reach of hands; b – reach zone of fingers with outstretched arm;
c – easy reach zone of the palm; d – Optimum space for fine handmade work;
e – the optimum space for rough manual work

Optimal placement of objects of labor and documentation in the reach of hands:

- the display is located in zone a (in the center);
- the keyboard is located in the area of e / d;
- the system unit is located in zone b (on the left);
- the printer is in zone a (right).

The documentation is placed in the easy reach of the palm - in (left) - literature and documentation necessary for work; in the drawers of the table - literature that is not used constantly.

When designing a desk, the following requirements must be taken into account.

The height of the working surface of the table should be within 680-800 mm. The height of the working surface with the keyboard should be 650 mm. The working table must be at least 700 mm wide and at least 1400 mm long. There should be a legroom of not less than 600 mm in height, a width of at least 500 mm, a depth at the knee level of at least 450 mm and at the level of elongated legs - not less than 650 mm.

The work chair must be liftable and adjustable in height and angle of inclination of the seat and backrest, as well as the distance of the backrest to the front edge of the seat. It is recommended that the height of the seat be above the floor level of 420 to 550 mm. The design of the working chair should ensure: the width and depth of the seat surface is not less than 400 mm; Seat surface with recessed front edge.

The monitor should be located at the eye level of the operator at a distance of 500 - 600 mm. According to the norms, the viewing angle in the horizontal plane should be no more than 45º to the normal of the screen. It is better if the viewing angle is 30º. In addition, it should be possible to select the level of contrast and brightness of the image on the screen.

It should be possible to adjust the screen:

- height +3 cm;
- slope from 10 to 20 degrees with respect to the vertical;
- in the left and right directions.

The keyboard should be placed on the surface of the table at a distance of 100 - 300 mm from the edge. The normal position of the keyboard is at the elbow level of the operator with an angle of inclination to the horizontal plane of 15º. It is more convenient to work with keys that have a concave surface, a quadrangular shape with rounded corners. The key design should provide the operator with a click sensation. The color of the keys should contrast with the color of the panel.

It is recommended to choose soft, low-contrast floral shades that do not disperse attention (low-saturated shades of cold green or blue colors) in the case of monotonous mental work requiring considerable nervous tension and great concentration. Shades of warm tones are recommended at work, which requires intense mental or physical tension, due to excitation of human activity.

3.3 Safe work conditions

The main parameters characterizing the working conditions are microclimate, noise, vibration, electromagnetic field, radiation, illumination.

The air of the working area (microclimate) is determined by the following parameters: temperature, relative humidity, air speed. The optimum and permissible values of the microclimate characteristics are established in accordance with [31] and are given in Table 2.

Table 15 – Optimal and permissible parameters of the microclimate

Period of the year	Temperature, °C	Relative humidity, %	Speed of air movement, m/s
Cold and changing of seasons	23-25	40-60	0.1
Warm	23-25	40	0.1

The measures for improving the air environment in the production room include: the correct organization of ventilation and air conditioning, heating of room. Ventilation can be realized naturally and mechanically. In the room, the following volumes of outside air must be delivered:

- at least 30 m^3 per hour per person for the volume of the room up to 20 m^3 per person;
- natural ventilation is allowed for the volume of the room more than 40 m^3 per person and if there is no emission of harmful substances.

The heating system must provide sufficient, constant and uniform heating of the air. Water heating should be used in rooms with increased requirements for clean air.

The parameters of the microclimate in the laboratory regulated by the central heating system, have the following values: humidity 40%, air speed 0.1 m/s, summer temperature 20-25 °C, in winter 13-15 °C. Natural ventilation is provided in the laboratory. Air enters and leaves through the cracks, windows, doors. The main disadvantage of such ventilation is that the fresh air enters the room without preliminary cleaning and heating.

Noise and vibration worsen working conditions, have a harmful effect on the human body, namely, the organs of hearing and the whole body through the central nervous system. It results in weakened attention, deteriorated memory, decreased response, and increased number of errors in work. Noise can be generated by operating equipment, air conditioning units, daylight illuminating devices, as well as spread from the outside. When working on a PC, the noise level in the workplace should not exceed 50 dB.

The screen and system blocks produce electromagnetic radiation. Its main part comes from the system unit and the video cable. According to [31], the intensity of the electromagnetic field at a distance of 50 cm around the screen along the electrical component should be no more than:

- in the frequency range 5 Hz - 2 kHz: 25 V/m;
- in the frequency range 2 kHz - 400 kHz: 2.5 V/m.

The magnetic flux density should be no more than:

- in the frequency range 5 Hz - 2 kHz: 250 nT;
- in the frequency range 2 kHz - 400 kHz: 25 nT.

There are the following ways to protect against EMF:

- increase the distance from the source (the screen should be at least 50 cm from the user);
- the use of pre-screen filters, special screens and other personal protective equipment.

When working with a computer, the ionizing radiation source is a display. Under the influence of ionizing radiation in the body, there may be a violation of normal blood coagulability, an increase in the fragility of blood vessels, a decrease in immunity, etc. The dose of irradiation at a distance of 20 cm to the display is 50 μrem/hr. According to the norms [31], the design of the computer should provide the power of the exposure dose of x-rays at any point at a distance of 0.05 m from the screen no more than 100 μR/h.

Fatigue of the organs of vision can be associated with both insufficient illumination and excessive illumination, as well as with the wrong direction of light.

3.4 Electrical safety

Depending on the conditions in the room, the risk of electric shock to a person increases or decreases. Do not operate the electronic device in conditions of high humidity (relative air humidity exceeds 75% for a long time), high temperature (more than 35 °C), the presence of conductive dust, conductive floors and the possibility of simultaneous contact with metal components connected to the ground and the metal casing of electrical equipment. The operator works with electrical devices: a computer (display, system unit, etc.) and peripheral devices. There is a risk of electric shock in the following cases:

- with direct contact with current-carrying parts during computer repair;
- when touched by non-live parts that are under voltage (in case of violation of insulation of current-carrying parts of the computer);
- when touched with the floor, walls that are under voltage;
- short-circuited in high-voltage units: power supply and display unit.

Measures to ensure the electrical safety of electrical installations:

- disconnection of voltage from live parts, on which or near to which work will be carried out, and taking measures to ensure the impossibility of applying voltage to the workplace;
- posting of posters indicating the place of work;
- electrical grounding of the housings of all installations through a neutral wire;
- coating of metal surfaces of tools with reliable insulation;

- inaccessibility of current-carrying parts of equipment (the conclusion in the case of electroporating elements, the conclusion in the body of current-carrying parts) [32].

3.5 Fire and explosive safety

According to [33], depending on the characteristics of the substances used in the production and their quantity, for fire and explosion hazard, the premises are divided into categories A, B, C, D, E.

The room belongs to category B according to the degree of fire and explosion hazard. It is necessary to provide a number of preventive measures.

Possible causes of fire:

- malfunction of current-carrying parts of installations;
- work with open electrical equipment;
- short circuits in the power supply;
- non-compliance with fire safety regulations;
- presence of combustible components: documents, doors, tables, cable insulation, etc.

Activities on fire prevention are divided into: organizational, technical, operational and regime.

Organizational measures provide for correct operation of equipment, proper maintenance of buildings and territories, fire instruction for workers and employees, training of production personnel for fire safety rules, issuing instructions, posters, the existence of an evacuation plan.

The technical measures include: compliance with fire regulations, norms for the design of buildings, the installation of electrical wires and equipment, heating, ventilation, lighting, the correct placement of equipment.

The regime measures include the establishment of rules for the organization of work, and compliance with fire-fighting measures. To prevent fire from short circuits, overloads, etc., the following fire safety rules must be observed:

- elimination of the formation of a flammable environment (sealing equipment, control of the air, working and emergency ventilation);
- use in the construction and decoration of buildings of non-combustible or difficultly combustible materials;
- the correct operation of the equipment (proper inclusion of equipment in the electrical supply network, monitoring of heating equipment);
- correct maintenance of buildings and territories (exclusion of the source of ignition - prevention of spontaneous combustion of substances, restriction of fireworks);
- training of production personnel in fire safety rules;
- the publication of instructions, posters, the existence of an evacuation plan;
- compliance with fire regulations, norms in the design of buildings, in the organization of electrical wires and equipment, heating, ventilation, lighting;
- the correct placement of equipment;
- well-time preventive inspection, repair and testing of equipment.

In the case of an emergency, it is necessary to:

- inform the management (duty officer);
- call the Emergency Service or the Ministry of Emergency Situations - tel. 112;
- take measures to eliminate the accident in accordance with the instructions.

3.6 Workplace health and safety compliance

In order to make sure that the workplace complies with sanitary-epidemiological rules and regulations, it is necessary to compare its characteristics with existing standards. This comparison is shown in Table 15.

Table 16 – Comparison of conditions with standards

	Working conditions	Requirements
The height of the working surface of table	690 mm	from 680 to 800 mm
The height of the working surface with keyboard	650 mm	650 mm
The width of the table	700 mm	> 700 mm
The length of the table	1420 mm	> 1400 mm
The height of legroom	600 mm	> 600 mm
The width of legroom	800 mm	> 500 mm
A depth at the knee level	550 mm	> 450 mm
The level of elongated legs	680 mm	> 650 mm
The height of the seat above the floor level	420 mm	from 420 to 550 mm
The width of the seat surface	400 mm	> 400 mm
The distance from the eye level to the monitor	550 mm	from 500 to 600 mm
The distance from the edge of the table to the keyboard	120 mm	from 100 to 300 mm
Temperature	23 ^{o}C	from 23 to 25 ^{o}C
Noise level	≈ 35 dB	< 50 dB

As can be seen from table 3, all conditions are met and the workplace fully complies with sanitary-epidemiological rules and regulations.

4 Financial management, resource efficiency and resource saving

4.1 Potential consumers of research results

It is obvious that at the moment there is an urgent need to plan and organize research projects. Note that it is important not only to develop this or that scientific topic, but also to analyze it from the point of view of resource efficiency and resource saving, in other words, it is necessary to determine the costs of research and the duration of work. [34]

In the course of this work, the protection parameters of the horizontal experimental channel 1 of the IRT-T reactor for the implementation of neutron capture therapy were determined. Development by virtue of its specificity will have as its target market medical organizations involved in cancer treatment.

4.2 Analysis of competitive technical solutions

To protect the staff and patients, concrete was chosen as a protection material. For analysis, it makes sense to compare different protection materials. In this case, we compare concrete (P_d) with lead (competitor 1, P_{c1}) and steel (competitor, P_{c2}).

The evaluation analysis map is presented in Table 1. The position of the development and competitors is evaluated for each indicator by an expert on a five-point scale, where 1 is the weakest position and 5 is the strongest. The weights of indicators determined by an expert in the amount should be 1.

Analysis of competitive technical solutions is determined by the formula:

$$C = \Sigma_i^n W_i P_i \tag{10}$$

when C – is the competitiveness of scientific research or a competitor;

W_i – is the weight of the indicator (in unit fractions);

P_i – is points of i-th indicator.

Table 17 - Evaluation map for comparison of competitive technical solutions (developments)

Criteria for evaluation	Weight criterion	Points			Competitive ability		
		P_d	P_{c1}	P_{c2}	C_d	C_{c1}	C_{c2}
1	2	3	4	5	6	7	8
Technical criteria for evaluating resource efficiency							
1. Material cost	0,2	5	3	3	1	0,6	0,6
2. The length of the particle in the protective material	0,3	4	5	4	1,2	1,5	1,2
3. Physiological effect of material	0,3	5	2	5	1,5	0,6	1,5
4. Resistance to external influences (temperature, humidity, etc.)	0,1	4	5	5	0,4	0,5	0,5
5. Durability	0,1	3	5	5	0,3	0,5	0,5
Total	1	21	20	22	4,4	3,7	4,3

Five criteria for evaluation were highlighted. The most important, when choosing a material, is cost, retarding ability and biological effect on people. As a result, the competitive ability of concrete was the highest among the competitors. Concrete has good performance in the determining criteria and, despite the shortcomings in the area of strength, therefore concrete was chosen for the study.

4.3 SWOT analysis

SWOT - Strengths (Strengths), Weaknesses (Weaknesses), Opportunities (Threats) and Threats (Threats) - is a comprehensive analysis of a research project. SWOT analysis is used to study the external and internal environment of the project. [35]

Strengths are the factors that characterize the competitive side of a research project. Strengths indicate that the project has a distinctive advantage or special resources that are particularly competitive. In other words, strengths are the resources or opportunities that the project management has and which can be effectively used to achieve the goals. At the same time, it is important to consider the strengths from the point of view of the project management, and from the point of view of those who are still involved in it. It is recommended to ask the following questions:

— What technical advantages do you have compared to competitors?
— What can your project members do best?
— How close is your project to completion compared to competitors?

Weakness is the lack, omission or limitation of a research project that impedes the achievement of its goals. This is something that is bad at the project or where it has insufficient capabilities or resources compared to its competitors. To clarify which aspects of your competitors may be superior to, you should ask:

— What can be improved?
— What is being done badly?
— What should be avoided?

Opportunities include any predictive present or future situation arising in the project environment, such as a trend, change or perceived need that supports the demand for project results and allows the project management to improve its competitive position. The formulation of project capabilities can be simplified by answering the following questions [36]:

— What opportunities do you see in the market? Search for free niches, but remember that they do not remain free for long. The opportunity seen today may cease to exist within three months. Opportunities may arise due to the following factors:

- changes in the technological sphere and in the market - both world-wide and regional scale;
- changes in government policy regarding the industry where research is being conducted;
- changes in social standards, population profile, lifestyle, etc.

— What are the favorable market opportunities?
— What interesting trends are noted?
— What needs, wishes have a buyer, but are not satisfied by competitors?

A threat is any undesirable situation, tendency or change in the environmental conditions of a project that are destructive or threatening to its competitiveness in the present or future. A threat can be a barrier, restriction, or anything else that may cause problems, destruction, harm or damage to the project. To identify threats to the project, it is recommended to answer the following questions:

— What trends do you see that can destroy your research project or make its results steady?

— What are competitors doing?

— What are the obstacles facing your project (for example, changes in legislation, reduction of budget funding for a project, delay in financing a project, etc.)?

— Do the required specifications or standards change for research results?

— Does technology change threaten your project?

— Does the project management have logistical problems?

Table 18 presents a SWOT analysis of a table.

Table 18 - SWOT Analysis

	Strengths of a research project: S1. The reactor is in operation for a long time, modernized, it is typical, as a result, it is reliable. S2. The protection material - the concrete chosen for the project performs several functions: protection against IR, forms the structure of the room. S3. It is possible to add various boron-containing compounds to concrete, or other elements with a high neutron absorption cross section, to protect against neutron radiation. S4. Concrete is a fairly cheap material. S5. Concrete is biologically harmless.	**Weaknesses of a research project:** W1 A relatively large amount of material to reduce the level of AI to the required standards of radiation safety. W2 Concrete is less durable than lead, tungsten, titanium, steel, etc. W3 The large amount of protection material complicates its subsequent disposal at decommissioning. W4 Concrete is less durable than metal alternatives to protect against gamma radiation.
Opportunities: O1. Improving the method of manufacturing concrete to protect against IR O2. Replacing concrete blocks if necessary O3. Reducing the cost of protection materials	No additional personal protective equipment is required when using concrete. The availability of concrete reduces the time to replace it.	In case of damage / loss of properties, it is possible to replace concrete. Improving the method of making concrete will increase its strength
Threats: T1. Natural or man-made disaster T2. Loss of demand for boron neutron capture therapy T3 Shortage of funds for re-equipment of the canal and the technical zone behind the reactor for BNCT.	In case of loss of demand financial losses for the purchase of concrete will be insignificant, due to the low cost relative to other materials and its use for other purposes.	The lack of funding in combination with a large amount of materials will play a negative role in the project. Concrete's vulnerability to natural disasters relative to other materials.

4.4 Project Initiation

The initiation process group consists of processes that are performed to define a new project or a new phase of an existing one. Within the framework of initiation processes, initial goals and content are determined and initial financial resources are recorded. The internal and external stakeholders of the project who will interact and influence the overall result of the research project are determined. This information is fixed in the Charter of the project.

4.5 Objectives and outcome of the project

Stakeholders of a project are persons or organizations that are actively involved in a project or whose interests may be affected both positively and negatively during the execution or as a result of the completion of the project. These can be customers, sponsors, the public, etc. Information on project stakeholders should be provided in the table 19. [36]

Table 19 – Project stakeholders

Project stakeholders	**Stakeholder expectations**
Medical institutions	Ensuring working conditions according to radiation protection standards
TPU	Calculation of protection satisfying the principles of radiation safety.

Table 19 should provide information on the hierarchy of project objectives and criteria for achieving goals. The objectives of the project should include goals in the area of resource efficiency and resource conservation.

Table 20 – Objectives and results of the project

Project Goals:	Calculate neutron and gamma fields of the horizontal experimental channel HEC-1 and estimate the biological protection
Expected results of the project:	Numerical value of neutron and gamma fields, calculated biological protection.
Criteria for acceptance of the project result:	Distributions of neutron energy and gamma radiation at the exit of the channel, doses of radiation protection.
Requirements for the project result:	**Requirements:**
	Obtaining distributions of neutron energy and gamma radiation at the exit of the channel
	Reducing the doses of radiation to the required values
	Justification of the correctness of the data

4.6 Project Organization Structure

At this stage of work, it is necessary to solve the following questions: who will be part of the working group of this project, determine the role of each participant in this project, and also prescribe the functions performed by each of the participants and their labor costs in the project. This information should be presented in tabular form (Table 21). [37]

Table 21 – Project Working Group

№	FULL NAME, position	Role in the project	Functions	Labor hours
1	Emets Evgeny Gennadyevich, Assistant professor	Supervisor	Guide, check results	30
2	Zinner Vsevolod Olegovich, Master student	Engineer	completing of the scientific work	366
Total:				386

4.7 Limitations and assumptions of the project

Project limitations are all factors that can serve as a restriction on the degree of freedom of the project team members, as well as “project boundaries” —the parameters of the project or its product that will not be implemented under this project. [38]

Table 22 – Project Restrictions

Factor	**Limitations / Assumptions**
Project's budget	139691,05
Source of financing	TPU
Dates of the project:	05.03.19-20.05.19
Date of approval of the project management plan	05.03.19
Project completion date	20.05.19

4.7 Project plan

As part of the planning of a research project, it is necessary to build a project calendar and network schedules.

Line graph is presented in the form of a table (Table 23).

Table 23 – Project Schedule

№	Name	**Duration, working days**	Start date	Date of completion	List of participants
1	Drafting and approval of technical task	1	05.03	06.03	S, M
2	Examine software package documentation MCU	8	06.03	16.03	M
3	Creating a model in the software package MCU to calculate	12	16.03	30.03	S, M
4	The choice of parameters for calculation in MCU	2	30.03	2.04	S, M
5	MCU calculation	25	02.04	2.05	M
6	Analysis of the neutron and gamma fluxes obtained	1	02.05	5.05	S, M
7	Calculation of protection against ionizing radiation	4	05.05	10.05	S, M
8	Registration of work in accordance with GOST	8	10.05	20.05	M
Total:		61			

M-Master student, P – Supervisor

Table 24 – The calendar schedule of the work will be presented in a table.

Calendar schedule of work on the topic											
№	Job title	Workers	Days	Duration of work, days							
				March			April			May	
				1	2	3	1	2	3	1	2
1	Drafting and approval of technical task Examine software package documentation MCU	M S	2								
2	Creating a model in the software package MCU to calculate	M	10								
3	Drafting and approval of technical task	M S	14								
4	The choice of parameters for calculation in MCU	M S	3								

Continuation of table 24.

5	MCU calculation	M S	30						
6	Analysis of the neutron and gamma fluxes obtained	M	3						
7	Calculation of protection against ionizing radiation	M P	5						
8	Registration of work in accordance with GOST	M	10						

Supervisor | Master Student

4.8 Scientific and technical research budget

The amount of costs associated with the implementation of this work is the basis for the formation of the project budget. This budget will be presented as the lower limit of project costs when forming a contract with the customer.

To form the final cost value, all calculated costs for individual items related to the manager and the student are summed.

In the process of budgeting, the following grouping of costs by items is used:

- Material costs of scientific and technical research;
- costs of special equipment for scientific work (Depreciation of equipment used for design);
- basic salary;

- additional salary;
- labor tax;
- overhead.

4.8.1 Calculation of material costs

The calculation of material costs is carried out according to the following formula:

$$З_{м} = (1 + k_T) \cdot \sum_{i=1}^{m} Ц_i \cdot N_{\text{расх}i}$$

when m – the number of types of material resources consumed during scientific research;

$N_{\text{расх}i}$ – the amount of material resources of the i-th species planned to be used when performing scientific research (pcs, kg, m, m^2, etc.);

$Ц_i$ – the acquisition price of a unit of the i-th type of material resources consumed (rubles / units, rubles / kg, rubles / m, rubles / m2, etc.);

k_T – coefficient taking into account transportation and procurement costs.

Values of prices for material resources can be set according to data posted on relevant websites on the Internet by manufacturers (or supplier organizations). [39]

The material costs required for this development are recorded in table 9.

The main work for the thesis was carried out at a personal computer in a room in a residential building. The time spent working at the computer, we assume equal to 36x6 = 216 hours. 36 working days' minus calculation period in MCU. By this time, we should add the simulation time on the computer, which took about 600 hours. The simulation in the software package MCU took 30 calendar days, on average, the computer worked 20 hours a day, so the operation of the computer the total work time is 216+30x20=816 hours. Computer power: 0.5 kW.

Energy costs are calculated by the formula:

$$C = P_{el} \cdot P \cdot F_{tot} = 0{,}5 \cdot 816 \cdot 5{,}8 = 2366{,}4 \text{ Rub}$$

where F is the tariff for industrial electricity (5.8 rubles per 1 kWh);

P_{el} – equipment power, kW;

P– time of equipment use, h.

The cost of electricity amounted to 2366.4 rubles.

Table - 25 Material costs

Name	unit of measurement	amount	Price per one., rub.	The cost of materials, rub.
1 Electricity	kWh	408	5,8	2366,4
Total				2366,4

4.8.2 The basic salary of the performers of the topic

This point includes the basic salary of participants directly involved in the implementation of work on this research. The value of salary costs is determined based on the labor intensity of the work performed and the current salary system

The basic salary (S_b) is calculated according to the following formula:

$$S_b = S_a \cdot T_w,$$

when S_b – basic salary per participant;

T_w – the duration of the work performed by the scientific and technical worker, working days;

S_a - the average daily salary of an participant, rub.

The average daily salary is calculated by the formula:

$$S_d = \frac{S_m \cdot M}{F_v},$$

when S_m – monthly salary of an participant, rub .;

M – the number of months of work without leave during the year:
at holiday in 48 days, M = 11.2 months, 6 day per week;

F_v – valid annual fund of working time of scientific and technical personnel (251 days).

Monthly salary of an employee:

$$S_m = S_t(1 + k_{pr} + k_{ad})k_d$$

где S_t – salary at the tariff rate, rub.;

k_{pr} – premium rate equal to 0,3 (i.e. 30%);

k_{ad} – the coefficient of additional payments and surcharges is approximately 0,2–0,5 (in scientific research institutes and at industrial enterprises — for the expansion of service industries, for professional skills, for harmful conditions: 15–20%);

k_d – district coefficient equal to 1,3 of Tomsk.

Example of calculation for a supervisor:

$$S_m = 33664(1 + 0{,}3 + 0{,}2) \cdot 1{,}3 = 65644{,}8$$

$$S_d = \frac{65644{,}8 \cdot 10{,}4}{251} = 1879{,}73 \text{ Rub}$$

$$S_b = S_d \cdot T = 1879{,}73 \cdot \frac{26}{6} = 8145{,}48 \text{ Rub}$$

Table 26 - Balance of working time

Working time	**Supervisor**
Calendar number of days	365
The number of non-working days - weekend - holidays	66
Loss of working time - vacation - absences due to illness	48
Valid annual working time fund	251

Table 27 - the calculation of the basic wage

Performers	S_t, rub	k_{pr}	k_{ad}	k_d	S_m, руб	S_d, rub	T, days	S_b, rub
Supervisor	33664	0,2	0,3	1,3	65644,8	2719,94	5	13599,72
Master student	12663	0,2	0,3	1,3	24692	1023,13	61	62410,93
TOTAL								76010,65

4.8.3 Additional salary

The costs of additional wages for the executors of the topic take into account the amount of surcharges provided for by the Labor Code of the Russian Federation for deviations from normal working conditions, as well as payments related to the provision of guarantees and compensations (when performing state and public duties, when combining work with training, when granting annual paid leave etc.).

The calculation of additional wages is carried out according to the following formula:

$$S_{add} = S_b \cdot k_{add}$$

when k_{add} – the ratio of additional wages (at the design stage is assumed to be 0,15).

Table 28 - Calculation of additional wages

Performers	Basic salary, rub	Additional salary, rub
Supervisor (Associate Professor)	13599,72	2039,96
Master student	62410,93	9361,64
Total		87412,25

4.8.4 Labor tax

Labor tax is compulsory according to the norms established by the legislation of the Russian Federation to the state social insurance (FSS), pension fund (PF) and medical insurance (FFOMS) from the costs of workers.

The amount of contributions to extra-budgetary funds is determined on the basis of the following formula:

$$S_{soc} = k_{soc}(S_b + S_{add})$$

when k_{soc} – coefficient of deductions for payment for social needs (pension fund, compulsory medical insurance fund, etc.).

Table 29 – Labor tax

Performers	Basic salary, rub	Additional salary, rub
Supervisor (Associate Professor)	13599,72	2039,96
Master student	62410,93	9361,64
The rate of deductions to extrabudgetary funds	0,271	
Total		23688,72

4.8.5 Overhead expenses

Overhead costs take into account other expenses of the organization that are not included in the previous cost items: printing and photocopying of research materials, payment of communication services, electricity, postal and telegraph expenses, reproduction of materials, etc. Their value is determined by the following formula:

$$С_{over} = k_{н} \cdot (S_m + S_{ad})$$

when $k_{нр}$ – overhead factor.

$$С_{over} = 0{,}3 \cdot (87412{,}25) = 26223{,}68 \text{ Rub}$$

4.8.6 Budget formation costs

The calculated amount of research costs (themes) is the basis for budgeting project costs. Determining the budget for the cost of a research project for each variant is given in Table 30.

Table 30 - the calculation of the budget costs

Name	Amount, rub.
Material costs	2366,40
Basic salary of the performers	76010,65
Additional salary	11401,60
Extrabudgetary funds	23688,72
Overhead costs	26223,68
Budget costs	139691,05

4.9 Determination of resource (resource saving), financial, budget, social and economic efficiency of research

Determination of efficiency is based on the calculation of the integral indicator of the effectiveness of scientific research. Its finding is associated with the definition of two weighted average values: financial efficiency and resource efficiency.

The integral indicator of the financial efficiency of a scientific study is obtained in the course of estimating the budget for the costs of three (or more) variants of the execution of a scientific study. To do this, the largest integral indicator of the implementation of the technical problem is taken as the calculation base (as the denominator), with which the financial values for all the options are correlated.

The integrated financial measure of development is defined as:

$$I_{\text{финр}}^{\text{исп.}i} = \frac{\Phi_{\text{р}i}}{\Phi_{\max}}$$

when $I_{\text{финр}}^{\text{исп.i}}$ – integral financial index of development;

$\Phi_{\text{р}i}$ – cost of i-th version;

$\Phi_{\max}$ – the maximum cost of execution of a research project (including analogues).

The obtained value of the integral financial indicator of development reflects the corresponding numerical increase in the budget of development costs in times (the value is greater than one), or the corresponding numerical reduction in the cost of development in times (the value is less than one, but greater than zero).

Since the development has one execution,

$$I_{fin}^{e1} = \frac{\Phi_{w1}}{\Phi_{max}} = \frac{139691{,}05}{139691{,}05} = 1.$$

The integral indicator of the resource efficiency of the variants of the research object can be determined as follows:

$$I_{pi} = \sum a_i \cdot b_i$$

when I_{pi} – integral indicator of resource efficiency for the i-th version of the development;

a_i – the weighting factor of the i-th version of the development;

b_i^a, b_i^p – the ball rating of the i-th version of the development, is established by an expert on the selected rating scale;

n – number of comparison parameters.

The calculation of the integral indicator of resource efficiency is presented in the form of a table (table 31).

Table 31 - Evaluation of the performance of the project

Criteria	Object of study	
	Parameter weight	Points
1. Material cost	0,2	5
2. The length of the particle in the protective material	0,3	4
3. Physiological effect of material	0,3	5
4. Resistance to external influences (temperature, humidity, etc.)	0,1	4
5. Durability	0,1	3
Total	1	21

$$I_{r-1} = 5 \cdot 0{,}2 + 4 \cdot 0{,}3 + 5 \cdot 0{,}3 + 4 \cdot 0{,}1 + 3 \cdot 0{,}1 = 4{,}4$$

The integral indicator of the effectiveness of development options ($I_{ucni.}$) is determined on the basis of the integral indicator of resource efficiency and the integral financial indicator using the formula [40]:

$$I_{.1} = \frac{I_{r1}}{I_{fin}^{r.1}}, \quad I_{2} = \frac{I_{r2}}{I_{fin}^{r2}} \text{ and etc.}$$

Comparison of the integral indicator of the effectiveness of the development options will determine the relative effectiveness of the project and select the most appropriate option from the proposed ones. Comparative project effectiveness (C_{pe}):

$$C_{[e} = \frac{I_{e.1}}{I_{e.2}}$$

Table 32 - Development Efficiency

№	Indicators	Value
1	Integrated financial development indicator	1
2	Integral indicator of development resource efficiency	4,4
3	Integral Performance Indicator	4,4

Comparison of the values of integral performance indicators allows us to understand and choose a more effective solution to the technical problem from the standpoint of financial and resource efficiency. But since the task has rather strict conditions, the solution has only one option.

Conclusion

A model of an IRT-T reactor with a modified geometry was created at the protection of the HEC-1 channel (a large insert of concrete was not included in the calculation) using the MCU5TPU software package, which simulated the distribution of neutron flux densities and spectra along the axis of the HEC-1 channel. As a result of the simulation, integral densities of neutron and γ-ray fluxes incident on the channel protection and directly on the dimensions of the protection itself were obtained.

We chose concrete as a material of protection because it is cheap, reliable, and easy to manufacture. From a practical point of view, the thickness of the protection is important, which depends on the dose rate of γ-radiation, and not on the neutron flux density, even for high-density concrete. Its thickness can be reduced in proportion to the density of concrete.

According to the results of the work, it was obtained that the thickness of the protection when increasing the concrete insert should be increased to 215 cm when using conventional concrete or up to 107 cm when using heavy concrete. Fulfillment of radiation safety requirements when working in the physical hall of the IRT-T reactor.

The calculations can be used to upgrade the biological protection of the HEC-1 channel.

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