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)

Инженерная школа ядерных технологий
Направление подготовки 14.04.02 Ядерные физика и технологии
Отделение ядерно-топливного цикла

МАГИСТЕРСКАЯ ДИССЕРТАЦИЯ

Тема работы
Оптимизация конфигурации активной зоны водоводяного реактора типа ВВЭР

УДК 621.039.534:004.42

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Томск – 2020 г.

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School of Nuclear Science and Engineering
Field of training (specialty) 14.04.02 Nuclear Science and Technology, Nuclear Power Installation Operation
Nuclear Fuel Cycle Division

MASTER THESIS

Topic of research work
Advanced core design of VVER type nuclear reactor

UDC 621.039.534:004.42

Student

Group	**Full name**	**Signature**	**Date**
0АМ8И1	Li Chuanbin		

Scientific supervisor

Position	**Full name**	**Academic degree, academic rank**	**Signature**	**Date**
Associate Professor	Naymushin Artem	PhD		

ADVISERS:

Section “Financial Management, Resource Efficiency and Resource Saving”

Position	**Full name**	**Academic degree, academic rank**	**Signature**	**Date**
Associate Professor	Ekaterina V. Menshikova	PhD		

Section “Social Responsibility”

Position	**Full name**	**Academic degree, academic rank**	**Signature**	**Date**
Associate Professor	Dan A. Verigin	PhD		

ADMITTED TO DEFENSE:

Programme Director	**Full name**	**Academic degree, academic rank**	**Signature**	**Date**
Nuclear Power Installation Operation	Vera V. Verkhoturova	PhD		

Tomsk – 2020

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 multioption 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 of Nuclear Science and Engineering
Field of training (specialty) 14.04.02 Nuclear Science and Technology, Nuclear Power Installation Operation
Nuclear Fuel Cycle Division

APPROVED BY:
Programme Director
_____________ Verkhoturova V.V.
«____» ____________ 2020

ASSIGNMENT
for the Graduation Thesis completion

In the form:

Master Thesis

For a student:

Group	Full name
0АМ8И1	Li Chuanbin

Topic of research work:

Advanced core design of VVER type nuclear reactor	
Approved by the order of the Director of School of Nuclear Science & Engineering (date, number):	№129-4/c at 08.05.2020

Deadline for completion of Master Thesis:	01.06.2020

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.)*	-Advanced core design of VVER type reactor； -Data from WIMS-ANL code software； -List of recommended literature；
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).*	1. Analysis of relationship between Keff and lattice size; 2. Determination of optimized lattice size; 3. Comparison of results in fuel cycle for both sizes; 4. Comparison of results in temperature effect for both sizes; 5.Comparison of results in poisoning part for both sizes; 6. Evaluation of the cost of research, determination of economic aspect and social responsibility; 7. Analysis of factors affecting safety during the study.
List of graphic material *(with an exact indication of mandatory drawings)*	General view of the VVER-1000 FA and lattice, the main results and a graph of the calculation, the basic characteristics and performance.

Advisors to the sections of the Master Thesis
(with indication of sections)

Section	Advisor
Literature review, methodology and results	Naymushin Artem
Financial Management, Resource Efficiency and Resource Saving	Menshikova E.V.
Social Responsibility	Verigin D.A.

Date of issuance of the assignment for Master Thesis completion according to the schedule	

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

Position	Full name	Academic degree, academic status	Signature	Date
Associate Professor	Naymushin Artem	PhD		

Assignment accepted for execution by a student:

Group	Full name	Signature	Date
0АМ8И1	Li Chuanbin		

TOMSK POLYTECHNIC UNIVERSITY — ТОМСКИЙ ПОЛИТЕХНИЧЕСКИЙ УНИВЕРСИТЕТ

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«Национальный исследовательский Томский политехнический университет» (ТПУ)

School of Nuclear Science and Engineering
Field of training (specialty) 14.04.02 Nuclear Science and Technology, Nuclear Power Installation Operation
Level of education: Master degree programme
Nuclear Fuel Cycle Division
Period of completion: spring semester 2019/2020 academic year

Form of presenting the work:

Master Thesis

SCHEDULED ASSESSMENT CALENDAR
for the Master Thesis completion

Deadline for completion of Master's Graduation Thesis:	01.06.2020

Assessment date	Title of section (module) / type of work (research)	Maximum score for the section (module)
19.01.2020	*Obtaining the assignment*	…
19.02.2020	*Analytical review of literature sources*	…
10.03.2020	Analysis of relationship between Keff and lattice size and determination of optimized lattice size	
20.04.2020	Comparison of results in fuel cycle, temperature effect and poisoning part for both sizes	
30.04.2020	Writing the master's thesis report	

COMPILED BY:
Scientific supervisor:

Position	Full name	Academic degree, academic status	Signature	Date
Associate Professor	Naymushin Artem	PhD		

AGREED BY:

Programme Director	Full name	Academic degree, academic status	Signature	Date
Nuclear Power Installation Operation	Vera V. Verkhoturova	PhD		

TASK FOR SECTION
«FINANCIAL MANAGEMENT, RESOURCE EFFICIENCY AND RESOURCE SAVING»

To the student:

Group	Full name
0АМ8И1	Li Chuanbin

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 –163619.8rub – Lab Tax-48775.06 rub – Overhead-53994.54 rub – STR budget – 290990.88 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	19.02.2020

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
0АМ8И1	Li Chuanbin		

Task for section
«Social responsibility»

To student:

group	Full name
0АМ8И1	Li Chuanbin

School	Nuclear Science and Engineering	Department	Nuclear fuel cycle
Degree	Master programme	Specialization	Nuclear Power Operation Installation

Title of graduation thesis:

Advanced core design of VVER type nuclear reactor	
Initial data for section «Social Responsibility»:	
1. Information about object of investigation (matter, material, device, algorithm, procedure, workplace) and area of its application	Data from WIMS-ANL code software. Application area: development of nuclear reactor
List of items to be investigated and to be developed:	
1. Legal and organizational issues to provide safety: – Special (specific for operation of objects of investigation, designed workplace) legal rules of labor legislation; – Organizational activities for layout of workplace.	– Labour code of Russian Federation #197 from 30/12/2001 GOST 12.2.032-78 SSBT – Sanitary Rules 2.2.2/2.4.1340-03. Hygienic requirements for PC and work with it
2. Work Safety: 2.1. Analysis of identified harmful and dangerous factors 2.2. Justification of measures to reduce probability of harmful and dangerous factors	– Enhanced electromagnetic radiation level – Insufficient illumination of workplace – Excessive noise – Deviation of microclimate indicators – Electric shock
3. Ecological safety:	– Indicate impact of nuclear power plant on hydrosphere, atmosphere and lithosphere
4. Safety in emergency situations:	– Fire safety;

Assignment date for section according to schedule	15.04.2020

The task was issued by consultant:

Position	Full name	Scientific degree, rank	Signature	date
assistant professor	Verigin D.A.	Cand.of Sc.		

The task was accepted by the student:

Group	Full name	Signature	date
0АМ8И1	Li Chuanbin		

Summary

The masters’ thesis consists of 96 pages, 34figures, 26 tables, 13references, 2 appendix..

Key words: PWR, VVER-1000, WIMS-ANL, lattice cell, fuel assembly, neutronic characteristics, reactivity, effective multiplication factor, temperature effect, reactivity temperature and power coefficient, Xenon poisoning, Samarium poisoning.

The purpose of this study is putting forward new size design of lattice for VVER type reactor and proving that it has better neutron parameters for VVER-1000 reactor core using WIMS-ANL code. In this study, calculations and simulations were performed by the nuclear reactor lattice cell calculation code WIMS-ANL and its 69 library package for investigating temperature effect on the VVER-1000 reactor core reactivity for nuclear fuel which has different enrichment.

The results of this work shows optimized size improve the reactivity coefficient and ideally reduce the insertion of negative reactivity of VVER-1000 reactor core. And whole process happened without control rods and burnable poisons.

Basic design, technological, technical and operational characteristics: the experiment is carried on the heterogeneous type VVER-1000 reactor. It is one type of light water reactor which uses light water as both coolant and moderator. Fuel assembly of this type reactor is hexagonal structure and VVER-1000 has 163 fuel assemblies and 312 fuel elements in each fuel assembly.

Applied areas: The research can be successfully applied to the field of nuclear physics and engineering. The results provide tables and diagrams that can provide useful information to nuclear engineers in designing nuclear reactor cores.

CONTENT

List of symbols and abbreviations

VVER	water-water energetic reactor
PWR	pressurized water reactors
LWR	light water reactors
Keff	effective multiplication coefficient
FA	fuel assembly
Zr	Zirconium alloy
Nb	Niobium
MW	megawatts
UO_2	uranium dioxide fuel
GWd	gigawatts per day
U-235	uranium 235
U-238	uranium 238
Pu-239	plutonium239
Pu-240	plutonium240
Pu-241	plutonium241
Pu-242	plutonium242
Xe-135	xenon 135
Sm-149	samarium149
ρ	reactivity
η	reproduction factor

Abstract: In order to solve the shortcomings of the previous third-generation reactors which could not fully realize the potential of nuclear power, we conducted a series of experiments on the core design of VVER reactors. In this experiment, we will use the WIMS-ANL code to study the relationship between the real lattice size and reactivity of reactor vver-1000. And then we get one optimized design of lattice from results got from first step research. Third step is do some experiments both for optimized size and real size lattice. Fuel cycle, temperature effect and poisoning are included. After comparison of results got from above experiments, we proved that optimized size has better neutronic characteristic than real size of VVER-1000 and the experiment is successfully completed.

Topic: This master's thesis focuses on the study of neutron physical parameters that contribute to changes in core reactivity. In addition, the effects of fission product poisons on core reactivity were studied. To some extent, almost all fission products can be considered reactor poisons because they capture neutrons to the extent that they reduce the effective multiplication factor. As the fuel is consumed, the product is slowly formed, and its impact is the main contribution of the decrease in fuel reactivity over time and neutron flux. Therefore, it is important to understand their stacking sequence and analyze their contribution to reactor core reactivity. Since the successful design will eventually be applied to the reactor, the fuel cycle part has also been studied. In addition, mathematical equations that depend on multiplication factors are calculated and analyzed visually using charts and tables of data.

In this master's thesis, the core is simulated using the WIMS-ANL code, which uses transport theory to calculate the neutron flux as a function of energy and space position in one-dimensional cells. The first version of the code, WIMS-D4, was originally developed in 1980 and acquired by Oak Ridge national laboratory (RSICC) in 1992. There are many versions of WIMS in use, but the most recent is WIMS-ANL. It is revised version of WIMS-D4. The version has improved error checking, output control, and can accommodate more than 69 energy groups (172 energy groups) in a cross section library for reactor physics calculations. The simulated results received are acceptable and show relative design values for the

VVER 1000 reactor core. We will take VVER-1000 as example to see that there are some parameters can be changed to improve reactor.

Purpose of this work: putting forward new size design of lattice for VVER type reactor and proving that it has better neutron parameters for VVER-1000 reactor core using WIMS-ANL code.

Objective of work:

1. Study the relationship between reactivity and size of lattice parameters
2. Get new design of size of lattice
3. Comparison of results between optimized size and real size in fuel cycle
4. Comparison of results between optimized size and real size in temperature effect
5. Study the reactivity of the optimized design and some properties about poisoning of the reactor VVER-1000

Introduction

1.1 VVER type reactor

1.1.1 VVER overview

The water - water reactor VVER is a series of pressurized water reactor designs originally developed by OKB Gidropress in the Soviet Union and Russia. The reactor idea was proposed by Savely Moiseevich Feinberg at the Kurchatov Institute. VVER type reactor was originally developed before the 1970s and has been updated ever since. Thus, the name VVER is associated with a variety of reactor designs ranging from first generation reactors to modern third generation reactor designs. The power output ranges from 70 to 1300 MWe, with designs under development up to 1700 MWe. The first prototype VVER-210 was built at the Novovoronezh nuclear power plant.

VVER plants are installed mainly in Russia and the former Soviet union, and also in China, Finland, Germany, Hungary, Slovakia, Bulgaria, India and Iran. Countries planning to introduce VVER reactors include Bangladesh, Egypt, Jordan and Turkey.

The earliest VVER was built before 1970s. The VVER-440 V230 is the most common design and can provide 440 megawatts of electricity. The V230 USES six main coolant loops, each with a horizontal steam generator. A modified version of the VVER-440, V213, was the first product of the nuclear safety standard adopted by Soviet designers. The model includes an additional emergency core cooling and auxiliary water supply system and an upgraded incident location system.

The larger vver-1000, developed after 1975, is a four-loop system installed in a closed structure with a spray steam suppression system (emergency core cooling system). VVER reactor design passes elaborate design, automatic control associated with western third-generation reactors, passive safety and containment system. VVER-1200 is currently provided to build version, is the improved version VVER - 1000, the power output increased to about 1200 MWe and provide other passive safety features. In 2018 it began to build the first VVER - 1300 (1300 MWe VVER - TOI).

1.1.2 Features of VVER

The design is a pressurized water reactor (PWR). The main distinguishing characteristics of VVER from other PWRS are:

1.Horizontal steam generators

2.Hexagonal fuel assemblies

3.No bottom penetrations in the pressure vessel

4.High-capacity pressurizers providing a large reactor coolant inventory

1.1.3 Versions of VVER

a.VVER-440

One of the earliest versions of the VVER type showed some problems in its container architectural design. Because of V - 230 and earlier models not initially for resistance to design and manufacture of large pipe burst, so manufacturers on the V - 213 type adds a so-called bubble column condenser, it with additional volume and quantity of water - is the purpose of the containment leak in no inhibition rapidly under the condition of the reaction between the steam to escape. As a result, all member states with factories with VVER-440 V-230 and older designs have been forced to close by eu politicians. The Bohunice nuclear power plant and the Kozloduy nuclear power plant must shut down four of the two units. With the greifswald plant, German regulators had already made the same decision after the fall of the Berlin wall.

b. VVER-1000

The original VVER design was built to run for 35 years. After that, a midlife overhaul was deemed necessary, including a complete replacement of key parts, such as fuel and lever passages. Since the RBMK reactor had been designated as a major replacement program in 35 years, the designers initially thought that this also needed to be done in the VVER version, although their design was more robust than the

RBMK version. Most VVER factories in Russia are now 35 years old. Recent design studies have shown that replacing equipment can extend its service life to 50 years. The new VVER will have a longer service life. In 2010, the oldest vver-1000 in Novovoronezh was closed for modernisation to extend its service life by 20 years; The first company to extend its service life. The work includes the modernization of management, protection and emergency systems, as well as the improvement of safety and radiation safety systems. Rosatom announced in 2018 that it had developed a thermal annealing technology for reactor pressure vessels that could improve radiation damage and extend service life by 15 to 30 years. This has been demonstrated at unit 1 of the barakovo nuclear power plant.

c.VVER-1200

The VVER-1200 (or NPP-2006 or AES-2006) is an improved version of the vver-1000 for both domestic and export use. The reactor design has been optimized to optimize fuel efficiency. The specifications include an electrical capital cost of $1,200 per kilowatt, 54 months of planned construction time, 60 percent design life (90 percent capacity factor) and 35 percent fewer operators than the vver-1000. The total thermal efficiency and net thermal efficiency of the VVER 1200 were 37.5% and 34.8%, respectively. VVER 1200 will generate 1,198 MWe power. The first two units are being built at the Leningrad no. 2 nuclear power station and novovolonezh no. 2 nuclear power station. As with the Leningrad II design, more vver-1200/491 reactors are planned (kaliningrad and NPP nnovgorod) and under construction. Seversk, Zentral and south-urals NPP also chose the vver-1200/392m installed on Novovoronezh npp-ii. According to the VVER-TOI (vver-1300/510) design, the standard version was developed as vver-1200/513.

In July 2012, the company agreed to build the contract for two aes-2006 plants in belarus, in Ostrovets, at a cost of approximately $10 billion (note that this price is equivalent to $4,200 per kilowatt of capital cost, compared to the initial $1,200 per kilowatt mentioned above). Aes-2006 is bidding for Finland's Hanhikivi nuclear power plant. On November 30, 2017 for Bangladesh Rooppur two V - 392 - m device of the first nuclear bottom of V - 392 - m units pad pouring the concrete. In August 2016, the first grid Novovoronezh II VVER - 1200-1 and start commercial operations on February 27, 2017. 2015-2017, Egypt and Russia to reach an agreement in the El Dabaa nuclear power plant construction with four VVER - 1200 units. On March 7, 2019, China national nuclear corporation (CNNC) signed a detailed contract with Atomstroyexport. Four vver-1200 units, two for each, were built at tianwan nuclear power station and xudaobao nuclear power station. Construction will begin in May 2021 and commercial operation of all units is expected to take place between 2026 and 2028. Starting in 2020, an 18-month refuelling cycle will be trialled, increasing capacity utilisation compared with the previous 12-month cycle. The nuclear portion of the plant is located in a building that serves as a containment and missile shield. In addition to the reactor and steam generators, this includes an improved tanker and a computerized reactor control system. Also protected in the same building are emergency systems, including emergency core cooling, emergency backup diesel power and emergency water supply. A passive deheating system has been added to the existing active system of the aes-92 version of the VVER 1000 used at India's Kudankulam nuclear power station. This issue has been reserved for newer VVER 1200 and future designs. The system is based on a cooling system and a water tank built on top of the containment vessel. Passive systems can handle all security functions for 24 hours and core security for 72 hours. Other new safety systems include aircraft crash protection, a hydrogen reformer and a core collector to hold molten reactor cores in the event of a serious accident. The core catcher will be deployed at the construction of the El Dabaa nuclear power plant.

d.VVER-TOI

VVER-TOI was developed from vver-1200. It aims to develop a new generation III + power plant based on VVER technology that typically optimizes information for advanced projects that use modern information and management techniques to meet many goal-oriented parameters.There are several improvement from VVER-1200.First, power of reactor is increased to 1300 MWe, passive safety system and cooling system are improved. VVER-TOI uses low-speed turbines and has lower construction and operating costs.

Table 1-technical information of VVER types reactor

Specifications	VVER-210	VVER-365	VVER-440	VVER-1000	VVER-1200	VVER-1300
Thermal output,MW	760	1325	1375	3000	3212	3300
Efficiency,net %	25.5	25.7	29.7	31.7	35.7	37.9
Vapor pressure in 100KPa						
In front of turbine	29	29	44	60	70	-
In the first circuit	100	105	125	160	165.1	165.2
Water temperature,°C						
Core coolant inlet	250	250	269	289	298.2	297.2
Core coolant outlet	269	275	300	319	328.6	328.8
Equivalent core diameter,m	2.88	2.88	2.88	3.12	-	
Active core height,m	2.5	2.5	2.5	3.5	-	3.73
Outer diameter of fuel rods,mm	10.2	9.1	9.1	9.1	9.1	9.1
Number of fuel rods in assembly	90	126	126	312	312	312
Uranium loading，tons	38	40	42	66	76-85.5	87.3

1.2 Core design

1.2.1 reactor design

The design content of the reactor includes 7 parts. First, the first part is the nuclear design. Specifically, it includes the basic design of the reactor core physical design and radiation shielding. The second part is the thermal and hydraulic design, which specifically includes the reactor core and the thermal analysis of fuel elements, the design of the coolant system of the primary circuit. The third part is the reactor control and dynamics analysis, including the design of the reactor control system, and the fourth part is the mechanical design, including related to nuclear analysis and thermal analysis. The design of the fuel element and the design of the stack structure and internal components. The fifth part is the thermodynamic analysis, which is the analysis and design of the thermodynamic cycle used to generate electrical energy. The sixth part is the safety analysis, which is based on various assumptions. The final part of the analysis of reactor performance under accident conditions is economic analysis, which is the investment and cost of nuclear power and its evaluation. As can be seen from the above, the reactor engineering is a comprehensive technical engineering that requires the cooperation of multi-disciplinary technicians in a wide range of fields.

1.2.2 Procedure of reactor design

The steps of reactor design are divided into three parts. The first part is scheme design, also known as conceptual design. The content is to compare different combinations of various parameters within the allowable change range and select the best scheme under the conditions of safety restrictions. The basic principle of choosing the best plan is to obtain the best economic benefits under the premise of ensuring safety. In the process of selecting the best solution, we need to obtain the main parameters, such as the size of the fuel assembly, the size of the core, and also the degree of fuel enrichment, composition, and coolant temperature.

After these main parameters are determined, detailed design analysis is carried out, specifically nuclear design, thermal hydraulic design, fuel assembly design, control system design, thermal cycle design and safety and economic analysis and evaluation. After the design plan is completed, the next step is the preliminary design. The preliminary design is carried out on the determined parameters of the completion of the plan, including the determination of the reactor operation mode, the detailed static analysis and preliminary safety analysis of the core, and the revision of the plan. Design parameters, determine the specific functions and flowcharts of each system. And then use standard specifications, design guidelines to reasonably select equipment materials, instruments and make preliminary layout drawings. For unsure technical problems, we must carry out the necessary experiments to obtain reliable data, and repeatedly coordinate among various majors to solve major technical problems. The final step is the core design. It is a careful and dynamic analysis of the core, completes the preliminary safety analysis report and the detailed design of all the primary and secondary loop systems, and draws all detailed layout drawings and the structure, parts and installation drawings of the equipment. Prepare the necessary technical requirements, commissioning outline and operation outline.

The above are the basic steps of reactor design. Now we can see what part of the task we undertake in reactor design for this topic, which is the main task of core design. The main part of the core physical design can still be divided into three parts. The first part is the design calculation of the core grid and power distribution. The main content of this part is to calculate the effective neutron multiplication coefficient and neutron flux density distribution, that is, power. distributed. Power distribution is the basis for thermal design analysis, core burnup analysis, and fuel management in the core. There are many factors that affect it, such as fuel enrichment, moderator-fuel volume ratio, reactivity control methods, and fuel Component design, etc. On the other hand, with the operation of the reactor, the nuclides U235 in the reactor are constantly being consumed, while other nuclides such as plutonium Pu239 and some fission readings are being produced and accumulated, they will affect the reactivity and power distribution of the reactor, so The power distribution is constantly changing with time and space. Another common concern is the uneven power distribution coefficient of the core, that is, the hot channel factor, which is the ratio of the peak power density of the core to the average value. We can use it to determine whether the core design exceeds the thermal limit . The other main task of the physical design of the core is the reactive control design calculation. Its main content is the reactive distribution of various control methods, such as movable control rods, chemical control agents can dissolve boron, and combustible poisons are also Need to design the control rod layout and insertion sequence in detail. The other content of the reactivity design calculation is to calculate various reactivity feedback coefficients, firstly the coolant reactivity temperature coefficient, and secondly the fuel reactivity temperature coefficient, in addition to the reactivity effect caused by the accumulation of fission poisons. The final part of the physical design of the core is burnup analysis and in-core fuel management. Its content is to determine the supervision and determination of fission throughout the core by solving the burnup equation of the main nuclide and the neutron flux density equation. The consumption of nuclides and the production and accumulation of fission products. The goal of fuel management is to optimize fuel loading, layout and refueling schemes within the design limits specified for reactor operation in order to produce the most economical electricity. One of the core physical design criteria is the requirement of the reactivity temperature coefficient. Both the reactor fuel temperature coefficient and the moderator temperature coefficient must be negative values, so as to ensure the inherent safety of the reactor.

1.3 WIMS-ANL code

The WIMS-ANL (Winfrith Improved Multigroup Scheme) is a code widely used for power and lattice physical analysis. The program was originally developed to use transport theory to calculate neutron flux as a function of the energy space position in a one-dimensional unit. The program USES its own set of 69 constant libraries and ENDF/BV libraries, which are based on the ENDF format and are prepared for a variety of materials, power, and temperatures.

There are two main modes of transport, known as DSN (discrete coordinates) and PERSUS (collision probability). The transport solution can be executed using any specified intermediate group structure up to the number of library groups. Before the main transfer, one or two STECTROX flux spectra are calculated for a small number of spatial regions in a small number of regional library group structures, and then the spectral calculations are performed using the spatial regions in the total energy group of the library. The spectrum condenses the basic cross section into several groups.

Upon completion of the main transport solution, the cross section of the intermediate group is folded into a broad group structure (⩽20 groups) and can be written in microscopic or macroscopic ISOTXS format for subsequent transport or diffusion theoretical codes. The microscopic ISOTXS cross section contains the complete Po and P1 scattering matrices for transport calculations, but their primary use will be in the multi-group diffusion theory analysis. The cross section is also burnup and space dependent.

The main purpose of this code is to develop algorithms for the WIMS code, create master data and prelude data to compute the spectra of a few spatial regions and rated regions in a uniform medium, and create libraries for these programs. The WIMS code and its 69 libraries were found to be one of the appropriate predictors of cell reactivity, burnup processes, and flux spectrum modeling. The input data model includes the prelude data (two transport solutions, PERSUS and DSN) and the main data (geometry, composition, battery characteristics, burn up and reaction rate editing).

In order to calculate neutrons in the reactor core, a reactor simulation is required. In this work, a unit calculation was performed to stimulate fuel assembly in the reactor core and the output was used in the core calculation, which clearly determined the neutron parameters of the reactor. First, the fuel assembly is simulated using the wims-anl code. It is important to remember that the processes that occur in the simulation correspond to the actual physical processes. Neutrons are absorbed by the nucleus, splitting themselves and releasing energy to sustain the fission chain reaction. This happens at the end of the simulation, where the particles are called neutron fluxes (neutron clusters). Internally, the WIMS code also generates a region average cross-section in an intermediate group structure that takes advantage of the maximum number of fine groups in the library. Currently, fine group libraries with 69 and 172 groups are used in the ANL RERTR program.

The WIMS-ANL code uses transport theory to calculate neutron flux as a function of energy and cell location. It begins by performing spectral calculations on several regions of the entire energy group of 69 spectral libraries, and then USES these spectra to condense the basic cross section into several groups. The obtained pass value is then expanded by using formal spectral calculations to calculate the rate of arrival in each library structure. Different geometries were evaluated, for example (basic shapes are uniform, circular, cylindrical geometry of rod clusters and r-z geometry of cylinders). In addition to the main battery calculation, the program is also used to calculate the burn up using time step intervals or power values.

In addition, the program reads the basic macro cross section from its tape library. It then calculates the macroscopic cross section of each material, screens the initial spectrum by automatically calculating the resonance, and assesses it using the collision probability method. In this work, fuel assemblies with different concentrations have been modeled using WDSN (to perform transport calculations in one dimension) and the main transport options for finding the problem Keff (homogeneous) in geometric media. In modeling, it is recommended to convert the hexagonal structure of the fuel module (with triangular lattice spacing) to a circular fuel module model, since the WIMS code can only perform cell calculations in the circular model. When the hexagonal structure is transformed into a circular model, the volume remains the same. Figure 1 and figure 2 below show the hexagon model and the circular model for cell computation, respectively.

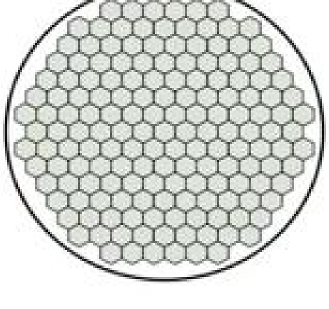


Figure1–Hexagonal model

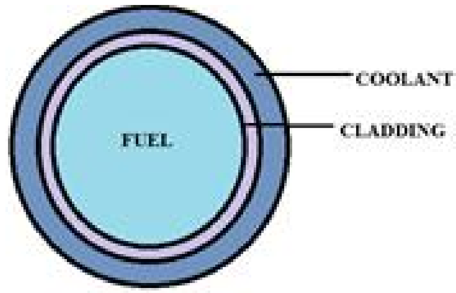


Figure2–circular model for a unit cell

The WIMS-ANL code need the following parameters:

1. material of each part of the lattice
2. the geometry of the lattice
3. burnup parameters
4.temperature and specific power used and other information depending on the method used for the calculation

At the end of the simulation, the values of the effective multiplication coefficients obtained were used to evaluate the optimal size of the three regions of the lattice cell and were further used to optimize the neutron parameters related to reactor dynamics.

2 Description of research

In this experiment we will study relationship between reactivity and the size of real lattice of the reactor VVER-1000 using WIMS-ANL code. And then we get one optimized design of lattice from results got from first step research. Third step is do some experiments both for optimized size and real size lattice. Fuel cycle, temperature effect and poisoning are included.

2.1.1 VVER-1000 type reactor

The VVER 1000 is a water-water energy reactor, originally developed by the Soviet union as a pressurized light water reactor, and now developed by OKB Gidropress as Russia. It is a third-generation type, built after 1975 but launched in May 1980. It is a four-ring system in a container-type structure. It is an heterogeneous vessel reactor that USES light water as a coolant and a moderator. Fission reactions are caused by collisions between thermal neutrons and the parent nucleus. This special vver-1000 reactor is designed to produce thermal energy with an installed capacity of 3,000 MW and a capacity rating of 1,000 MW. The reactor core consists of 163 fuel assemblies of the same design, hexagonal on a hexagonal grid with constant spacing between 200 and 240mm, while some pressurized water reactors consist of square fuel assemblies on a square grid with 193 FA. The FA of the VVER 1000 contains 312 fuel elements, but the fuel enrichment is variable.

2.1.2 Structure of fuel assembly type

The reactor core is occupied by the basic TVSA and the alternative TVS-2 fuel assembly design. The TVSA fuel assembly (FA) is considered the basic version of the fuel component (FA) design and TVS-2 is the alternative. Both versions of FA are interchangeable and have reference features.

The reactor core consists of 163 fuel assemblies. The design is identical to 312 fuel elements. Each FA consists of the following components, top nozzle, fuel rod bundle, bottom rod, guide channel and spacer gate, as shown in figures 4 and 5. The fuel rods are cylindrical and Zr with 1% Nb. Each rod contains fuel pellets with inner and outer diameters of 0.15cm and 0,757cm. In addition, the fuel pellets are covered with materials with an diameter of 0,91cm.

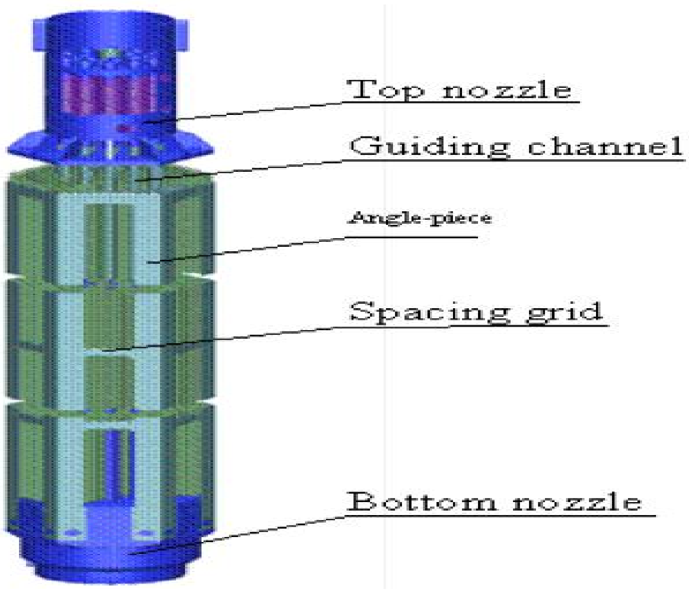


Figure3– TVSA general view

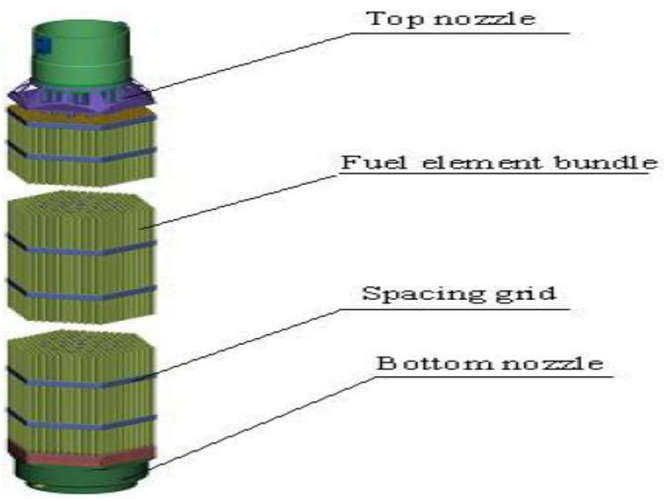


Figure4– TVS-2 general view

During the operation of the reactor, the fuel rods shall be fully immersed in water with a nominal pressure of 15MPa to prevent water from boiling at normal operating temperatures (220°C to 300°C). The fuel is low enriched UO2 (between 1.6% and 4.4%).

Since boiling on the surface of the tray is prohibited, the temperature of the cladding depends on the type of fuel used. The inserted fuel assembly consists of 312 fuel elements, 18 tubes, a central tube and a gauge tube, all arranged in a hexagonal lattice with a pitch of 1,275cm. The lattice pitch of the assembly unit is 23.6cm.

The structure and configuration of the fuel assemblies in the core remain unchanged. Water is mainly used as a reflector. The VVER-1000 reactor is required for large-scale power generation, and the performance of the reactor vver-1000 can be greatly enhanced by increasing the size or length of the fuel assembly or the propellant fuel (UO2) without changing the core volume.

Table 2 – Main parameters of the VVER-1000 reactor

Main parameter, unit	Values
Nominal thermal capacity, MW	3000
Rated Electrical capacity, MW	1000
Fuel assembly quantity	163
Cladding material(Alloy)	Zr + 1%Nb
Fuel rod effective height (cm)	353
Fuel material	UO_2
Density of fuel, g/cm^3	10.5
Fuel temperature, K	500
Fuel pellet diameter, cm	
outer	0.757
inner	0.15
Cladding diameter, cm	0.91
Moderator-coolant	H2O
Pressure, MPa	14
Temperature, °C	300
Fuel assembly form	Hexagonal
Lattice, cm	1.257
Maximum linear heat rate of the fuel rod(W/cm)	448
Average enrichment	4.4%

2.1.3 Configuration of the core and FAs of VVER-1000

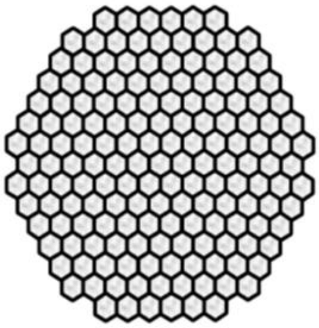


Figure6 – Core (163 FAs)

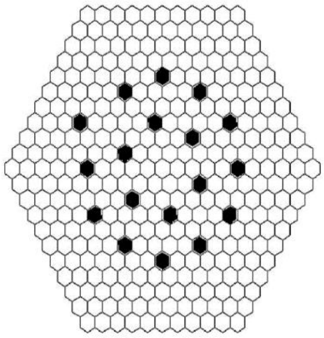


Figure7– FAs (312 FE)

Figure8- VVER-1000 fuel assembly

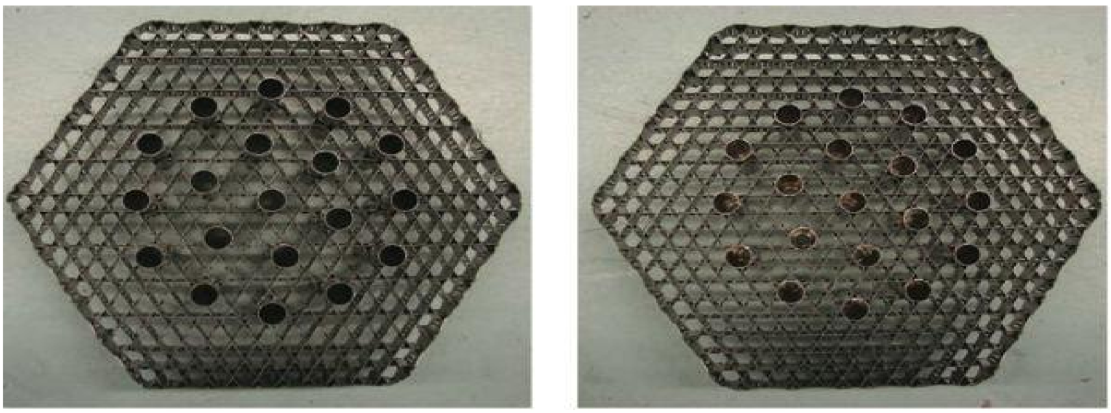


Figure9-VVER-1000 fuel assembly

2.2 Simulating the VVER-1000 reactor core using WIMS-ANL codes

The neutron characteristics associated with reactor VVER-1000 design are analyzed using the WIMS-ANL code. In the simulation, two main inputs are used, namely the main input data and the prelude input data.

A prelude to the input options include solutions, including SEQUENCE, which defines the lattice calculation procedures, mainly used in the transportation and the CELL card is used to select CELL type, in this case, the CELL is a single homogeneous unit, 2) the accuracy of the solution, which contains NGROUP card, the card is set up by using the number of the main transportation group. Note that the more groups you select, the more accurate flux and unit reactivity calculations will generally result, and the NMESH card is set up in the main transport calculation to calculate the number of grid points.

The description of core geometry (NREGION) and component (NMATERIAL) is also defined in the prelude data. The NREGION card is used to set the number of regions in a given lattice, while the material card defines the number of material components.

For one dimensional geometry auxiliary input options, ANNULUS card is used to define the unit cell, whereas NMATERIAL card is used to specify the set lattice density of the material/component, temperature and spectral type. The other is the main data POWERC (according to the time step calculation of fuel consumption), the reaction rate card and ISOTXS (for micro and macro written into the output data file cross section), BEGINC card and will terminate the master data input PRTOPT card is set to 1 in order to complete the output editing results.

In this study, whether the reactor causing the increase or decrease in the motion length has been evaluated, and the neutron properties related to dynamics have been determined. Parameters such as reactivity and reactivity feedback are calculated. The reactivity associated with xenon transients due to changes in power levels was also calculated. The energy contribution of each fuel component in the battery was also calculated. When inserting a fuel assembly into the reactor core, it is important to know that the fuel element contains a composition of fresh fuel isotopes that have not yet been irradiated by thermal neutrons. In order to obtain an accurate crystal cell optimal lattice and optimize fuel consumption in the reactor, the general idea is to calculate the contribution of each isotope based on the enrichment degree. According to the given enrichment degree, the isotopic concentrations of U-235, U-238, O_2 and other fuels were calculated comprehensively, and their respective reactivity was analyzed according to the Keff values obtained from WIMS-ANL. The diagram below shows the effect of reactivity on fuel enrichment.

2.3 Calculating reactivity and reactivity feedback using WIMS-ANL Theory about reactivity and reactivity coefficient

The amount of reactivity (ρ) in the reactor core determines the number of neutrons at any given time, which in turn determines the thermal power of the reactor. Reactivity is affected by several parameters of the instance. Fuel consumption, temperature, pressure or poison. The purpose of this calculation is to simulate the factors that affect reactivity and how they can be used to control or predict the behavior of the reactor. However, during the operation of the reactor, the temperature of the reactor will change as the number of neutron (or their power) changes, so the Keff and ρ values will change. These changes affect reactor power. In order to quantify the effect of parameter changes on nuclear reactivity, reactivity coefficients were used.

The reactivity coefficient is the amount of reactivity that will change for a given parameter change.The two temperature coefficients which have the greatest influence on the thermal reactor are temperature and reactivity power coefficients. These reaction coefficients, which are associated with reactor core design and combined with the physical properties of neutrons, have been extensively studied in the Coon’s age and have recently been applied in the research field and introduced into reactor physics textbooks.

2.3.1 Reactivity

Reactivity is a parameter that characterizes the degree of deviation from the critical state of the chain reaction system, which can be expressed quantitatively as:

$$\rho = 1 - \frac{1}{Keff}$$

Where, keff is the effective multiplication coefficient. The degree of reactivity depends mainly on the amount of fuel loaded and fuel enrichment also depends on the type and structure of the stack.

When the reactor is critical, keff=1, reactivity = zero; When the reactor is supercritical, keff>1, reactive ρ is greater than zero, which is positive. When the reactor is subcritical, keff<1, reactivity is less than zero and negative.

2.3.2 Reactivity coefficient

Reactivity coefficient refers to the temperature change of reactivity relative to fuel, structure and coolant. They depend primarily on the uranium, p, and burnout characteristics in the reactor core. In pressurized water reactors, changes in fuel temperature have the greatest impact on reactivity and changes in nuclear coolant/regulator temperatures. These dependencies define the dynamic characteristics of the reactor and its stability.

Therefore, it is absolutely necessary to understand this dependency of each reactor. The two main temperature coefficients usually specified for the thermal reactor are fuel temperature coefficient and reducer temperature coefficient. The reactivity coefficient is defined as the reactivity change per unit change in some operating parameters of the reactor. Such as:

$$\alpha = \frac{d\rho}{dT} = \frac{\rho_2 - \rho_1}{T_2 - T_1},$$

2.3.3 Reactivity temperature effect and coefficient

The change of reactivity with temperature is called the reactivity temperature coefficient. During the operation of a reactor, different materials can have multiple effects on reactivity at different temperatures. Because of the uneven temperature change in the reactor, the increase in the reactor power will result in an increase in the fuel temperature (the area where the power is generated) and then, after the transfer of heat, a change in the temperature of the regulator and coolant. From the fuel temperature reactivity is measured as reactivity. The temperature reactivity of a particular reactor is the change of its reactivity from 300K temperature to operating temperature with heating. The temperature reactivity coefficient is generally defined as:

$$\propto_T = \frac{d\rho}{dT} = \frac{\rho_2 - \rho_1}{T_2 - T_2}$$

or replacing the reactivity equation we can obtain:

$$\rho = 1 - \frac{1}{k_{eff}} \rightarrow \quad \propto_T = \frac{1}{k_{eff}^2} \frac{dk_{eff}}{dT}$$

The reactor feedback is based on the following algebraic notation for the reactivity temperature coefficient:

$\alpha_T > 0$: is positive because the multiplication factor is always positive. Therefore, the increase in temperature leads to an increase in the number of neutrons.

The reactor power increases as a result of temperature variations, further increasing the effective multiple,if temperature goes down, then the effective multiple will go down:

$\alpha_T < 0$: since the multiplication factor is always positive, it is always negative. In this case, an increase in temperature would result in a decrease in the neutron multiplication factor.

And from the point of view of operation, it is better to divide ρ_T and α_T into two components - slowly changing in time with the change of temperature and operating mode (isothermal ρ_t and α_t) and fast tracking the change of the nuclear reactor power (dynamic, power ρ_N and α_N) :

$$\rho_T = \rho_t + \rho_N;$$

$$\alpha_T = \alpha_t + \alpha_N.$$

We can also use the formula of four factors to determine the temperature of the separate influence degree of temperature reaction coefficient (α_T). In this case, it is best to represent Keff as $K_\infty P$ product:

$$(P = R_{mod} \cdot R_{dif})$$

where P – the probability for neutron to avoid leakage). Then:

$$(\alpha T = \frac{dk_{eff}}{k_{eff}dT} = \frac{dk_{\infty}}{k_{\infty}dt} + \frac{dP}{PdT}d)$$

representing K_{∞} as 4 factors. Therefore, probability for a neutron to avoid leakage will be calculated as

$$P = \frac{1}{(1 + B^2 \cdot M^2)}$$

where : B^2 – geometrical parameter of the reactor;

M^2 – neutron migration length

2.3.4 Reactivity power effect and coefficient (Fuel temperature coefficient)

The fuel temperature coefficient is another reactivity coefficient. The reactivity coefficient is the reactivity change per degree change of fuel temperature. Because an increase in reactor power causes an immediate change in fuel temperature, it is also known as the immediate temperature coefficient. Because of the immediate increase in reactor power, the fuel temperature coefficient is negative than the moderator temperature.

The time that the heat is transferred to the host is measured in seconds. When a large amount of positive reactivity is inserted, the fuel temperature coefficient immediately begins to increase the negative reactivity. Fuel temperature reactivity coefficient is also known as fuel doppler reactivity coefficient. The name is used because in a typical mild, low-enrichment thermal reactor, the fuel temperature reactivity coefficient is negative, which results in a Doppler effect, also known as Doppler broadening. The Doppler coefficient is as follows:

1. By expressing the multiplication factor as k_∞ in the four formula factors, resonance escape probability can be determined in the following formula:

$$\ln k_{\infty} = \ln(\eta \varepsilon f P) + \ln p_{esc},$$

2.Differentiating with respect to temperature and assuming all parameters to be constant except the escape probability results in a simple expression for the Doppler coefficient

$$\frac{d}{dT}(\ln p_{esc}) = \frac{1}{p_{esc}} \cdot \frac{dP}{dT} = \frac{d}{dT}(\ln k_{eff}) = \frac{1}{k_{eff}} \cdot \frac{dk_{eff}}{dT},$$

Hence,

$$\propto_p = \frac{d\rho}{dT(fuel)} = \frac{\rho_2 - \rho_1}{T_2 - T_2}$$

2.3.5 Reactivity coefficient as a function of reactor core parameter and fuel enrichment

Similarly, the reaction effect and reaction coefficient are simulated by using WIMS-ANL code. The influence of fuel temperature and coolant/regulator temperature on reactivity was analyzed.

For values ranging from 300K (zero power) to 700K, the optimal reactivity Keff is calculated at intervals of 50 between 1.1 and 1.4, and the reaction temperature coefficient is calculated as a function of the coolant or regulator temperature. WIMS ANL used its 69 library codes for calculation and analysis, and used the Keff and K ∞ result values given after modeling to calculate the coefficient values. The coefficient is estimated as the change of reactivity with temperature.

$$\alpha_T = (k_{\infty 1} - k_{\infty 2})/\partial T$$

In the same way, the behavior of the power reactivity coefficient is studied. The power coefficient of reactivity is calculated as a function of fuel temperature and fuel enrichment. Unlike the reactivity temperature coefficient, the fuel density remains constant with the change of fuel temperature. The calculation analysis is then modeled in the same way and the power coefficient values are estimated.

2.4 Reproduction factor

Factors η is defined as the number of fast neutron thermal fission and fuel thermal neutron absorption in the ratio of the number. The reproductive factors are shown below. The capture fission ratio can be used as an indicator of the "mass" of fission isotopes. This ratio is largely dependent on the incident neutron energy. This coefficient is determined by multiplying the probability of a fission reaction by the average number of neutrons produced per fission reaction. In the case of using fresh uranium fuel, we only consider a kind of U-235 fission isotope, η value given by type:

$$\eta = \frac{\upsilon.\Sigma_f}{\Sigma_f + \Sigma_c} = \upsilon\frac{N_5.\sigma_f^5}{N_5.\sigma_f^5 + N_5.\sigma_c^5 + N_8.\sigma_c^8}$$

in which ν is the average neutrons production of 235U, N5 and N8 are the atomic number densities of the isotopes 235U and 238U (when using other uranium isotopes or plutonium the equation is modified in a trivial way). This equation can be also written in formula of uranium enrichment:

$$\eta = \upsilon\frac{e.\sigma_f^5}{e.\sigma_a^5 + (1-e)\sigma_c^8}$$

This is relationship between reproduction factors and uranium enrichment. Where e is the fuel enrichment degree e = N5 / (N5 + N8). The propagation factor is determined by the composition of the nuclear fuel and largely depends on the neutron flux spectrum in the nuclear core.

2.5 Xenon effects on the reactor core reactivity

Another factor leading to reduced reactivity is the accumulation of poisoning in the reactor core. During operation, most fission products are produced slowly with fuel consumption, and fission is one of the major contributions to fuel reactivity that decreases with time and exposure to neutron flux. Due to the control and operation of the reactor, the two fission products xenon-135 and samarium-149 have attracted much attention due to their large neutron absorption cross sections and quantities. The main effect of these fission product poisons is to reduce the heat utilization coefficient (f), and they are therefore considered to be a source of negative reactivity. But one of the most important is xenon poisoning. As shown in figure 10, the accumulation of xenon is directly due to fission or decay of iodine nuclides. ^{135}Xe has a very high heat absorption cross section ($\sigma_{a,Xe} \approx 2{,}6 * 10^6$) for VVER-1000 and a cumulative yield rate with a half-life of 9.2h has a profound effect on flux.

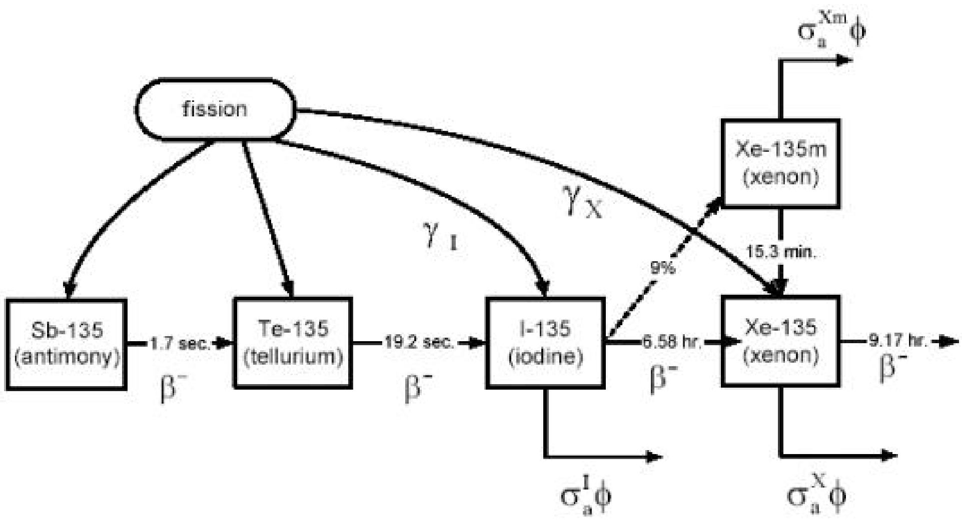


Figure 10- Production and decay of ^{135}I and ^{135}Xe

The flux change after iodine decays into xenon will have a direct influence on the core reactivity. In regions of high flux, reactivity increases and xenon gas decays, and vice versa. This condition helps the flux tilt, leading to greater reactivity. Therefore, the behavior of xenon poisoning is very special under non-stationary operating conditions and under certain conditions under stable loads.

2.5.1 Equation for xenon contribution due to the changes in reactivity

$$
{}^{235}_{92}U + n \rightarrow {}^{135}_{52}Te \xrightarrow[t_{1/2}=19s]{\beta,\gamma} {}^{135}_{53}I \xrightarrow[t_{1/2}=6{,}7h]{\beta,\gamma} {}^{135}_{54}Xe \xrightarrow[t_{1/2}=9{,}10]{\beta,\gamma} {}^{135}_{55}Ce
$$

The accumulation of xenon poisoning in fuels tends to reduce Keff, thereby reducing reactivity. However, based on reactor physics simulations, xenon accumulation can be derived directly from differences in reactivity, and vice versa, since the highest absorption cross section occurs at thermal energy. And we think about reactivity

$$
\rho = \frac{k_{eff} - 1}{k_{eff}}
$$

Where ρ – reactivity. Knowing that the highest absorption cross section occur at thermal energies and that the poisons normally affect the thermal process, then it is easier to know that the main impact is on the thermal utilization factor within the four factor formula ($k_\infty = \varepsilon\eta f p$). Considering the thermal utilization in the absence of poison

$$f_0 = \frac{\bar{\Sigma}_{aF}}{\bar{\Sigma}_{aF} + \bar{\Sigma}_{aM}}$$

where – $\boldsymbol{f_0}$ is the thermal utilization factor in the absence of poisons

$\bar{\boldsymbol{\Sigma}}_{\boldsymbol{aF}}$ – the average microscopic absorption cross section of fuel

$\bar{\boldsymbol{\Sigma}}_{\boldsymbol{aM}}$ – the average microscopic absorption cross section of moderator, therefore modifying the impact to account for poisons

$$f = \frac{\bar{\Sigma}_{aF}}{\bar{\Sigma}_{aF} + \bar{\Sigma}_{aM} + \bar{\Sigma}_{aP}}$$

where $\bar{\boldsymbol{\Sigma}}_{\boldsymbol{aM}}$ – the average microscopic absorption cross section of moderator

If $\boldsymbol{\rho_0}$ is the initial reactivity in the absence of poison then the change in reactivity, $\Delta\boldsymbol{\rho}$ in the presence of poison will be achieved by

$$\Delta\rho = \rho - \rho_0 = \frac{k_{eff} - 1}{k_{eff}} \approx \frac{f - f_0}{f} \cong - \frac{\bar{\Sigma}_{aP}}{\bar{\Sigma}_{aF} + \bar{\Sigma}_{aM}}$$

Therefore, all conditions being equal, the reactivity difference from the initial state to a certain period will be equal to the change in the concentration of the poison and its aggregate absorption cross section.

2.5.2 Determining the xenon accumulation as a function of reactor core parameter and fuel enrichment using WIMS-ANL codes

The WIMS-ANL codes were used to simulate the effect of xenon core reactivity and different power levels were used. The ISOXS card were used in order to write either micro or macroscopic ISOTXS-formatted cross-sections to an output data file. During the rapid buildup of equilibrium xenon, short time steps were used. Six days cycle duration was used as referenced and one fuel assembly was replaced at the end of each cycle. The POWERC card in the main data was used to define the depletion time steps ranging from 0,5 days to 4 days(period taken for xenon to reach its maximum equilibrium) each for a specific power 45.45MW/tU. Number of burn steps to perform, including one step for initial calculation.

3 Optimization of cell (size means)

In this result, the neutron physics behavior that contributes to the transient behavior of the reactor in the absence of burnable poison and control rods is calculated. In this section, the results of the calculation are described, compared and discussed with the values of the VVER-1000 designer. Calculations were made not to establish the limiting contour safety margin but to demonstrate the relative impact of certain phenomena on core modeling and to assess the neutron physics of the vver-1000 reactor. Each parameter was analyzed independently and the reactivity effect on the reactor was determined jointly. Neutron parameters examined in this study included power and temperature activity feedback, reactivity associated with xenon transients due to changes in power levels, and energy contributions of specific isotopes. The reactivity effects of each phenomenon are usually evaluated according to the effective multiplication coefficient $\rho = {(k_{eff} - 1)}/{k_{eff}}$, in which keff-1 is the condition affecting the change and Keff is the basis or main source of the neutron population.

Through simulation, the reaction temperature effect and coefficient (as a function of coolant/moderator temperature and density), the reaction power effect and coefficient (as a function of fuel temperature), and the negative insertion of reactivity due to the presence of xenon are calculated. Toxicity and the contribution of isotopes present in cells to energy. These calculations are based on valid product values obtained from the output of the wims-anl program and are used to determine reactivity. The reactivity effects in this case were evaluated as differences in reactivity, and their coefficients were calculated as differences between their current and initial reactivity at their respective temperatures. Calculations of reactivity associated with xenon transients due to power changes were also evaluated, because of the reactivity differences obtained as xenon approached its maximum equilibrium to its initial reactivity (reactivity in the absence of xenon) over a given period of time. All calculations were performed using Microsoft Excel.

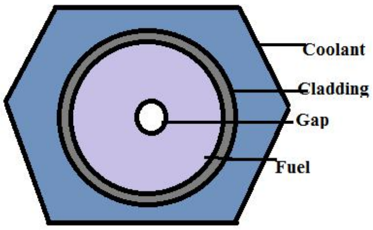
From the neutronic point of view, the optimum sizes occurred where Keff is maximum in the cold zero power reactor condition (i.e. T = 300K). The VVER-1000 reactor core was firstly simulated, using the WIMS-ANL program with various allowable values for each annulus of lattice.

ANNULUS 1: region of gap in fuel pellet

ANNULUS 2: fuel region

ANNULUS 3: cladding

ANNULUS 4 coolant



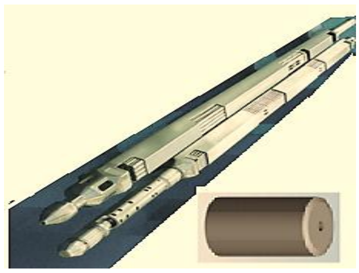


Figure 11-VVER lattice Figure12 – lattice cell structure and a fuel assembly

3.1 Dependence between Keff and radius of lattice parameters

In order to obtain a good design, we need to do experiments to get relationship between Keff and radius of gap, pellets and cladding. Now we will VVER-1000 with 4.4% enrichment fuel as sample to get diagram of relationship between Keff and radius of lattice parameters. Because if we change annulu 4,design of FA and may others have to be changed so in this master's thesis, only annulus 1,2 and 3 are changed.

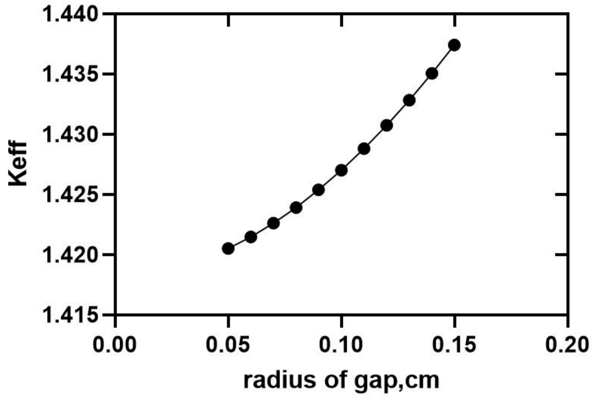


Figure13-Dependence between Keff and radius of gap

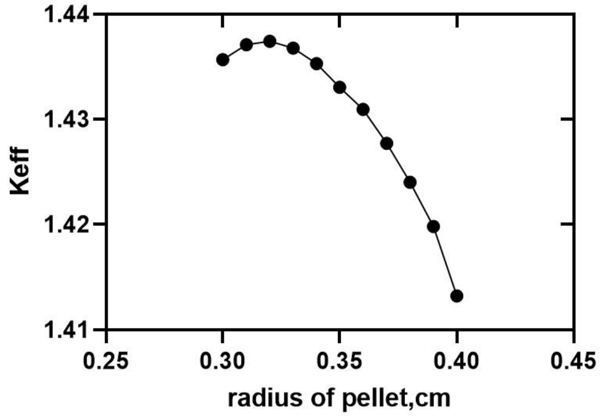


Figure14-Dependence between Keffand radius of pellets

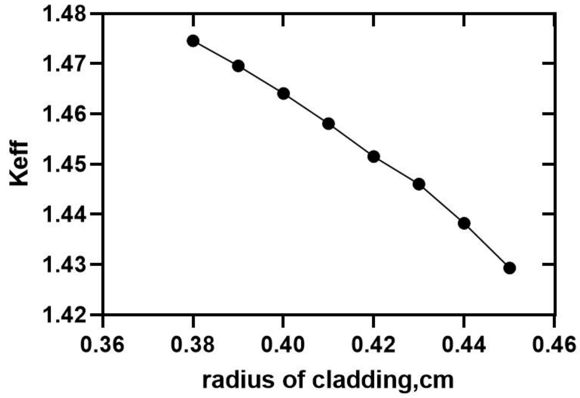


Figure15-Dependence between Keff and radius of cladding

From above diagrams, we can get a lot of useful information. In figure 13 Keff will increase due to increase in radius of gap in range 0.05 cm to 0.15 cm when radius of pellets and cladding do not be changed. It is because that moderator to fuel ratio will increase when radius of gap is increased, fast neutron can be moderated sufficiently and stay in region of resonance less time. As a result resonance escape probability will increase. However it will show peak in following range of radius of gap. The reason is same with peak in figure 14. To the opposite, Keff will decrease due to increase in radius of cladding when radius of pellets and gap do not be changed in figure 15. It is for nearly same reason with figure 13. Moderator to fuel ratio decrease when radius of cladding is increased, as a result resonance escape probability will decrease. In addition, decrease of moderator will decrease P_{mod}. So Keff will decrease with increase of radius of cladding. But we know that increase of moderator to fuel ratio will not only increase resonance escape probability but also decrease thermal utilization factor. This is reason why there is one peak in figure 14. Because we didn't change radius of coolant region, so radius of coolant will be 0.67 cm as real size. After considering, we chose gap radius 0.15cm，pellet radius 0.35cm，cladding 0.41cm and coolant radius 0.67cm as new design of sizes of lattice.

Table 3-Comparison of Keff of two design of lattice

Enrichment	real lattice design	Optimized lattice design
2.4%	1.287	1.306
3.3%	1.365	1.401
4.4%	1.423	1.471

Table above is result of comparison about Keff of two size in enrichment 2.4%,3.3% and 4.4%.From table 10,we can see that new lattice design have better Keff than real lattice design for VVER-1000 for each enrichment fuel. It shows that new lattice design has advantages in this part. Then we did study fuel cycle parts of two lattice sizes.

3.2 Comparison of results of real size and optimized size

An optimized design should have longer length of fuel cycle and higher reactivity than real size while reactor work normally. Condition is set at fuel temperature 1000K,temperature of coolant 570K and cladding at 600K at whole process. According to table of VVER-1000 in IAEA database, average enrichment of VVER-1000 is 4.4% and mass of fuel is 66t.New lattice is 1.37 times lower than volume of real size, so mass of fuel in new design is equal to about 48.18t. For this reason, enrichment of fuel in new design will at 6.0% to get equal mass of U-235 in two design.

4 Reactor fuel cycle calculations

4.1 Fuel cycle

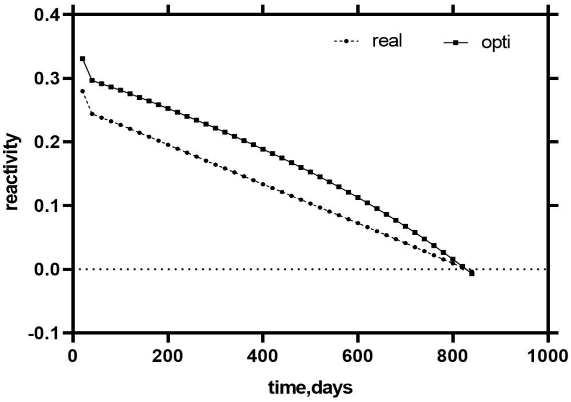


Figure16-change of reactivity with time for optimized and real size

From above diagrams we can see that optimized has a little longer length of fuel cycle. Time intervals in above diagram is equal to 20 days and real size get to negative reactivity one interval earlier than optimized size. Reactivity shows a sharp decrease in first time interval due to poisoning. And in whole process, reactivity of optimized size is higher than real size all the time. Because experiment is done in single condition and many parameters are not considered so result may not correspond with data got from real work of reactor, but it is enough to get some conclusion according to comparison of result above. After calculation, reproduction factor η of optimized size is 2.0187 and 1.985 for real size. Averaged reactivity lost is $3.84*10^{-4}$/day for optimized size and equal to $3.93*10^{-4}$/day for reactor with real size. Burnup of fuel is equal to 52.3GWD/tU for optimized size. It is obviously higher than number of burnup of fuel for real size which is at 40.5GWD/tU. It increase worth of each ton fuel.

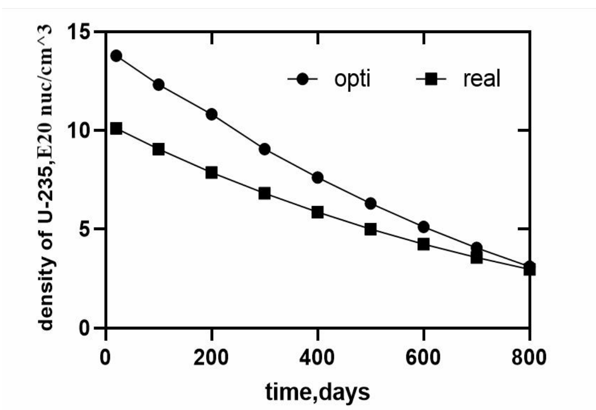


Figure17-change of density of U-235 with time for optimized and real size

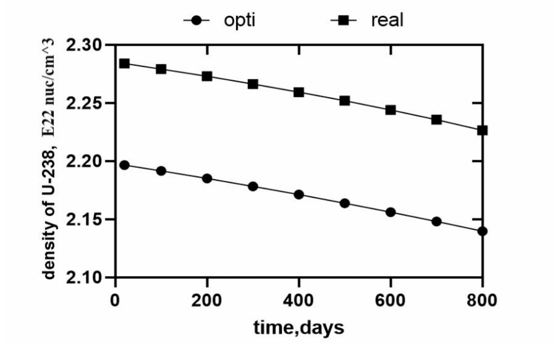


Figure18-change of density of U-238 with time for optimized and real size

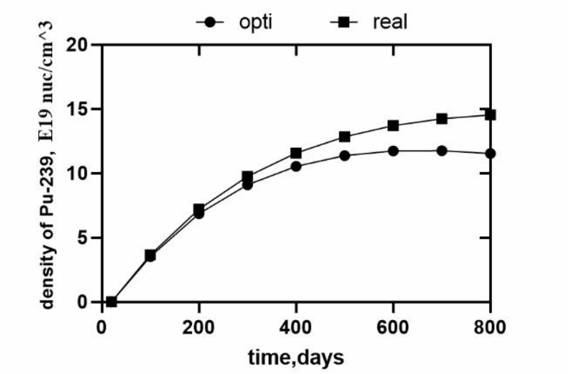


Figure19-change of density of Pu-239 with time for optimized and real size

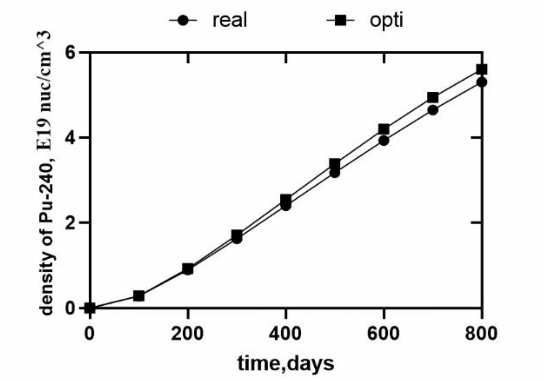


Figure20-change of density of Pu-240 with time for optimized and real size

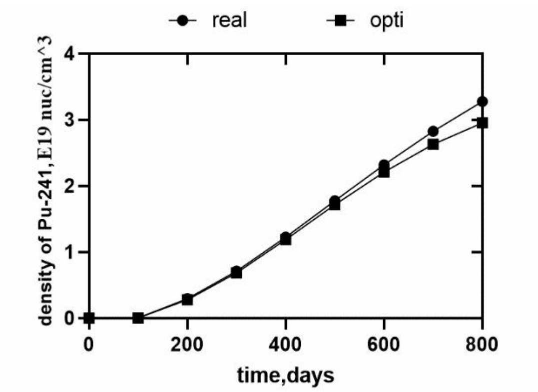


Figure21-change of density of Pu-241 with time for optimized and real size

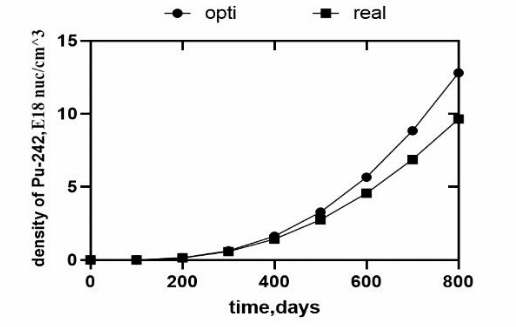


Figure22-change of density of Pu-242 with time for optimized and real size

Diagrams above is comparison of results of two size in density of elements U-235, U-238,Pu-239,Pu-240,Pu-241 and Pu-242.Interval of diagram is equal to 200 days. From above diagrams we can get to know that density of U-235 decrease in process of work of reactor. Density of U-235 in reactor with optimized size drops faster than real size but higher than real size all the time because of initial higher enrichment of fuel and higher power density. Density of U-238 in reactor with optimized size seems drop at same speed with real size but lower than real size all the time because initial higher enrichment of fuel lead to initial density of U-238 is less than real size. Density of Pu-239 in reactor with optimized size rises slower than real size and lower than real size all the time. It is because reactor with real size has higher concentration of U-238. And Pu-239 is generated from decay of U-239 generated by U-238 absorption. Density of U-238 in real size is higher than optimized lattice and lead to higher density of Pu-239.Density of Pu-240 in reactor with optimized size shows nearly curve with real size but it is a little higher in optimized size. About 62% to 73% of the time when Pu-239_captures a neutron, it undergoes fission. In the remainder of the time, it forms Pu-240. The isotope Pu-240 has about the same thermal neutron capture cross section as Pu-239 but only a tiny thermal neutron fission cross section. When the isotope Pu-240 captures a neutron, it is about 4500 times more likely to be become Pu-241 than to fission. Density of Pu-241 in reactor with optimized size rises slower than real size and lower than real size all the time.to the opposite, density of Pu-242 in reactor with optimized size rises faster than real size and higher than real size all the time. Pu-241 has a neutron absorption cross section about 1/3 bigger than Pu-239. When Pu-241 absorbs neutron and will not fission, it will produce Pu-242.

4.2 Radial flux and temperature distribution

In order to study the radial distribution of neutron flux, we divided the fuel pellet into several small parts and studied independently.Power level is set at 3000MW for both sizes. Research points are set as values in table 4. Real size is divided to 6 parts by same radius difference and optimzied size is 4 parts.

We choose 0.1,0.15,0.2,0.25,0.3,0.35cm as research points in real size and 0.175,0.225,0.275,0.325cm as research points in optimzied size.

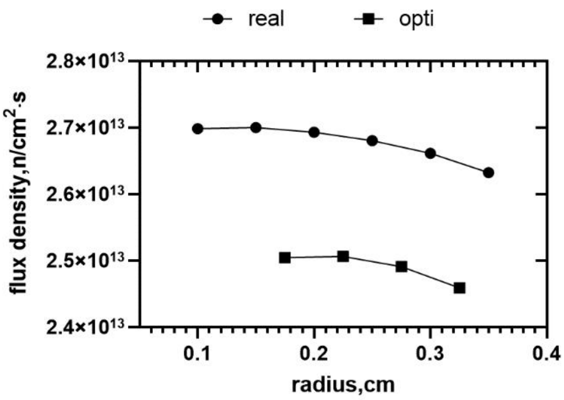


Figure23-distribution of fast neutron flux density with radius

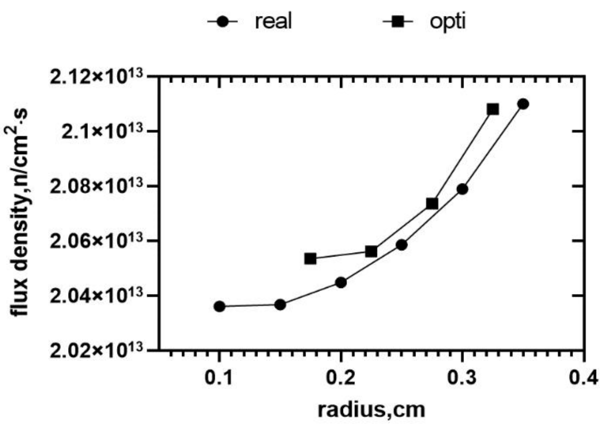


Figure24-distribution of thermal neutron flux density with radius

Figures 23 and 24 are diagrams, which show relationship between radius and fast and thermal neutron flux density. Fast neutron is higher than flux density of thermal neutron at all radius for both sizes. Because power level for both sizes are equal to 3000 MW, so average thermal neutron flux density is same and real size has higher total amount of neutron flux according to formula RR $=\sum_f \cdot \Phi$ and total amount of reaction is RR•Volume. From figure 23 and 24 we can get to know flux density of neutron have same trend. Flux density of thermal neutron in real size is a little lower and flux density of fast neutron a little higher. At same time, flux density of optimized size seems evenly distributed than real size lattice.

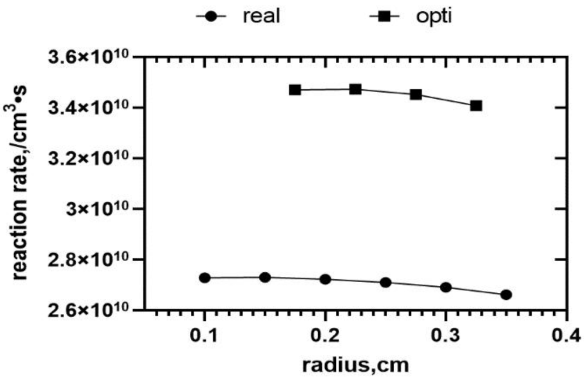


Figure25-distribution of reaction rate of fast neutron with radius

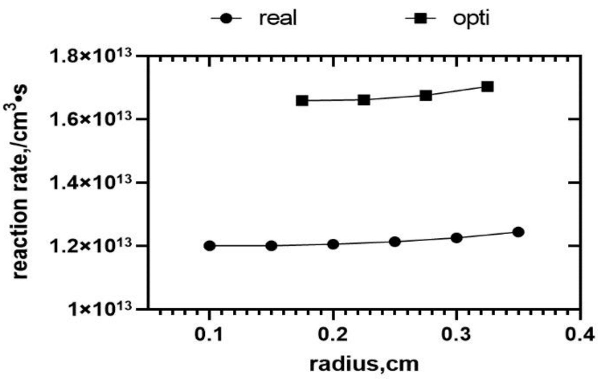


Figure26-distribution of reaction rate of thermal neutron with radius

Figures 25 and 26 are diagrams, which show relationship between radius and reaction rate of fast and thermal neutron. Different with flux density, thermal neutron becomes main in reaction rate. In addition, reaction rate of fast neutron has a unit at about $10^{10}/cm^3 \cdot s$ when unit of rate of thermal neutron is $10^{13}/cm^3 \cdot s$. It comes from difference of fission cross section for fast and thermal neutron. But for real size and optimized size have same power level so integrated values of reaction are same. On the other hand, optimized size has higher average reaction rate for smaller volume.

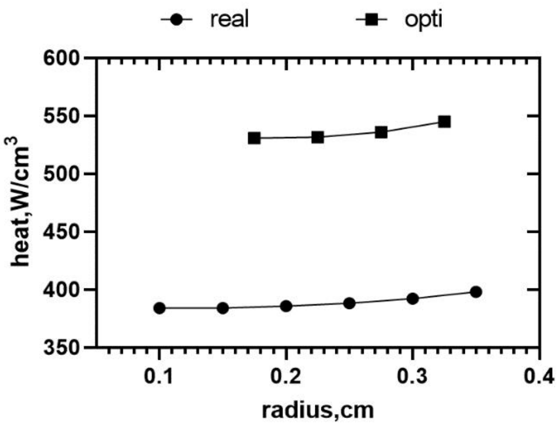


Figure27-distribution of heat generated by thermal neutron with radius

Figures 27 is diagram, which show relationship between radius and heat generated by thermal neutron. Here we only take fission of U-235 into account and fission cross section of fast neutron is too small compared with thermal neutron. This part is nearly same with reaction rate for formula. Volume of optimized size is a bit smaller than real size and they have same power level. So optimized size must have higher specific volume power to obtain same power level with real size. And for same reason, integrated power in two diagrams is equal.

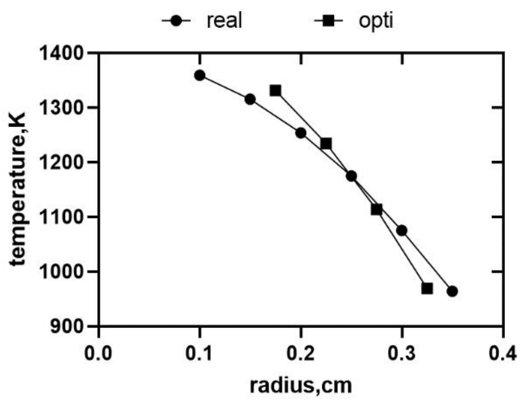


Figure28-distribution of temperature with radius for optimized and real size

Figures 28 show radial temperature distribution for real and optimized size. Real size has different range of radius with optimized size. We assume that it is maximum temperature along the fuel rod. Temperature also decrease with increasement of radius. Because longer radius, decrease of real size is flatter and optimized size is sharper. At end of pellet, it has nearly same temperature with cladding. We know that maximum heat density of optimized size is less than real size and has evenly distribution of heat along radius from figure 27. Optimized size shows a little lower maximum temperature than real size. It declares that optimized size will not have problem about thermal conduction in pellets for higher specific volume power.

4.3 Temperature effects

Table 4-Comparison of results of power effect in reactivity for two sizes

Keff	ρ	Δρ	T，K	Keff	ρ	Δρ
1.524	0.344	0	300	1.425	0.298	0
1.519	0.342	-0.006	400	1.418	0.295	-0.01
1.515	0.34	-0.012	500	1.413	0.292	-0.02
1.510	0.338	-0.017	600	1.408	0.29	-0.028
1.507	0.336	-0.022	700	1.403	0.288	-0.036
1.503	0.335	-0.027	800	1.399	0.285	-0.044
1.5	0.333	-0.031	900	1.394	0.283	-0.051
1.496	0.332	-0.035	1000	1.39	0.281	-0.058
1.494	0.33	-0.04	1100	1.387	0.279	-0.065
1.49	0.329	-0.044	1200	1.383	0.277	-0.071
1.487	0.328	-0.047	1300	1.379	0.275	-0.077
1.484	0.326	-0.051	1400	1.376	0.273	-0.083
1.482	0.325	-0.055	1500	1.373	0.271	-0.089
1.479	0.324	-0.058	1600	1.369	0.27	-0.095

Table 5-Comparison of results of coolant temperature effect in reactivity for two sizes

Keff	ρ	Δρ	T，K	Keff	ρ	Δρ
1.524	0.34386	0	300	1.425	0.29804	0
1.523	0.343599	-0.0007	400	1.423	0.29752	-0.001763693
1.522	0.343	-0.0025	500	1.422	0.29673	-0.004399253
1.521	0.34262	-0.0045	600	1.42	0.29592	-0.007120549
1.519	0.34164	-0.0064	700	1.419	0.29513	-0.009769777

Table 4 is comparison of results of power effect in reactivity for two sizes. Data in left part is result of optimized size and right part is result of real size. Range of temperature is from 300K to 1600K.Keff, reactivity and rate of change are included in table. Rate of change is calculated at formula $\frac{\rho_2-\rho_1}{\rho_1}$.

Table 5 is comparison of results of coolant temperature effect in reactivity for two sizes. Data in left part is result of optimized size and right part is result of real size. Range of temperature is from 300K to 700K.Keff, reactivity and rate of change are included in table.

From 300K to 600K, reactivity loss 0.005918 for fuel and 0.001531 for coolant when reactivity for two condition of optimized size is both equal to 0.343855. Reactivity decrease 1.72% in first condition and decrease 0.45% in coolant temperature effect.

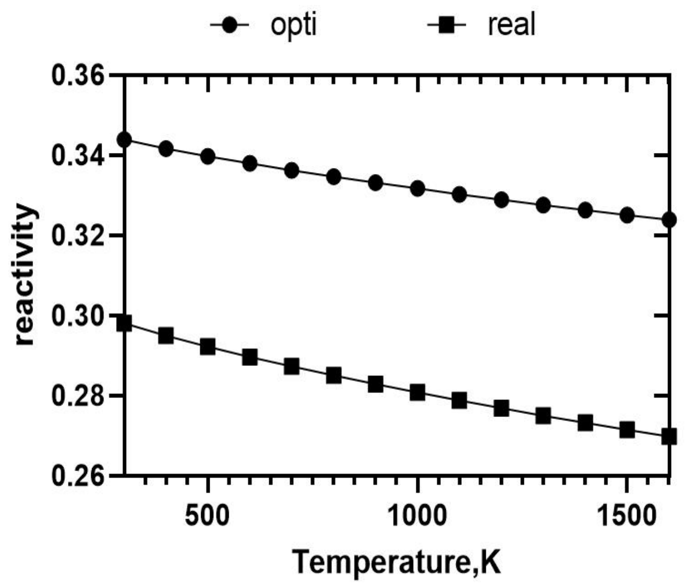


Figure29-change of reactivity with temperature of fuel for optimized and real size

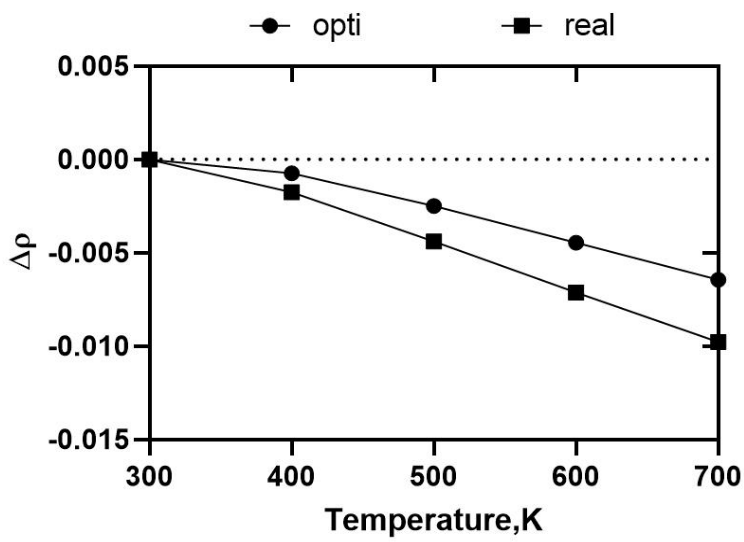


Figure30-change rate of reactivity with temperature of coolant for optimized and real size

Figure 30 is diagram to show change of reactivity with temperature of fuel for optimized and real size for power effect. And figure 31 is change rate of reactivity for optimized size and real size for coolant temperature effect.

In the case of the power reactivity effect and coefficient, only the temperature of fuel were changed. Condition is set as temperature of cladding and coolant at 300K.The multiplication factor obtained from the computer output were used in evaluating the reactivity power effect. Similarly, the effect was evaluated for fuel temperatures ranging from 300K to 1600K. The calculated results for the power effect are presented on diagram and in table. The number of reactivity plotted on the vertical axis with fuel temperature on the horizontal axis. The diagram presented dependence between reactivity and the fuel temperature and interval is equal to 50K.

As shown in diagram, power effect on reactivity decreases linearly with increase in temperature. Note that at temperature 300K a zero effect was obtained since the reactivity effect was calculated based on the initial value for reactivity ρ1. Reactivity coefficient for optimized size is equal to - $1.54*10^{-5}$/K and reactivity coefficient for real size is equal to - $2.17*10^{-5}$/K. Results shows that optimized size has lower power effect coefficient. It is because optimized size has higher moderator to fuel ratio and shorter fuel rod diameter which lead to higher effective resonance integral, so less changes in resonance escape probability when fuel temperature changes.

The fuel temperature coefficient is mainly caused by the Doppler effect (see neutron nuclear reaction cross section) of the resonance absorption of the fuel core, so it is also called the Doppler reactivity coefficient. The increase in fuel temperature will broaden the U resonance absorption peak, so for reactors with low-enrichment nuclear fuel, the Doppler reactivity coefficient is negative. As a result, the thermal expansion of nuclear fuel due to temperature changes leads to nuclear fuel density changes, thereby dating reactive changes, this mechanism is more important for metallic uranium fuel.

Doppler effect is a law of acoustic physics, which is caused by the radiation source relative to the observer. The motion causes a change in the observed radiation wavelength. Doppler broadening refers to the broadening of spectral lines caused by the thermal movement of molecules, atoms or nuclei.

The probability that the atom of a substance absorbs a neutron is expressed in terms of the microscopic absorption cross section, but the cross section depends on the kinetic energy of the neutron relative to the target. In fact, the target is not stationary, the atoms that are bound to a lattice are vibrating. This kind of vibration moving means that the target core has a certain velocity, and now the relative kinetic energy of the neutron is measured in a vibrating core. First, consider the resonance peak with a low U-238 (at a relative energy of 6.7ev). If the target core does not vibrate, the neutron with 6.7ev kinetic energy will be absorbed with a high probability. With the increase of fuel temperature, the vibration of U-238 atoms increases dramatically, and the probability of the neutron with 6.7ev kinetic energy entering the fuel hitting the target nucleus of U-238 core decreases.

However, the likelihood of neutrons reacting with the target nucleus increases at about 6.7eV, at 6.7eV the neutron absorption cross section deviating from the energy of the formant (nearby) increases. It represents the lowering and broadening of the formant. This effect is called Doppler broadening. The total absorption probability of the region, which is the area under these energy cross sections, is not expanded. If it changes, the average cross section of this energy region is not going to change. If the neutrons of all energies are evenly distributed in the fuel, then the absorption probability and the escape resonance probability don't change, whereas the fuel inside the reactor is. The entire core is not uniformly distributed, and several pellets are stacked inside the fuel rods. Near the combustion of the cladding. The material can encounter neutrons of various energies, while the fuel inside the fuel pellet, because it is shielded, it doesn't have a neutron that has a resonant energy.

In the case of the coolant temperature reactivity effect and coefficient, only the coolant temperature changes. Set the condition to 300K for the fuel and cladding. The multiplication coefficient obtained from the computer output was used to evaluate the temperature effect of reactive coolant. Similarly, the effect of fuel temperature from 300K to 700K was evaluated. Calculations of the power effect are shown in diagrams and tables. The number of reactivity is plotted on the vertical axis, and the temperature of the fuel is plotted on the horizontal axis. This diagram shows the dependence between reactivity and the temperature of the fuel. The interval is 100K.

Similar to the power effect, for the power reactivity effect, the reactivity value obtained is negative. As shown in the table, the influence of coolant temperature on reactivity decreases linearly with the increase of temperature. Note that at 300K, zero impact is obtained because the reactivity effect is calculated based on the reactivity initial value ρ_1. The reactivity coefficient of the optimized size is equal to - 5.54 * 10^{-6} / K, while the reactivity coefficient of the actual size is equal to - 7.28 * 10^{-6} / K.

The results show that the optimized size has lower coolant temperature coefficient. It is because optimized size has higher moderator to fuel ratio so less changes in resonance escape probability and non-leakage probability in moderation.

Due to the heat transfer process from the fuel to the reducer, the temperature change of the reducer lags behind the power change in a period of time, so the temperature effect of the reducer is the hysteresis effect. , the density of the moderator (especially the liquid one) and its microscopic neutron cross section will change, which will weaken the regulation ability and harden the neutron spectrum. Due to the decrease of the moderating capacity, the possibility of neutrons being absorbed by nuclear resonance increases before they decelerate to thermal energy, resulting in a decrease in reactivity. In addition, due to the hardening of the neutron energy spectrum, the average fast neutron absorbs thermal neutrons from each fuel, resulting in a decrease in the atomic number caused by fission, resulting in a decrease in reactivity. The negative coolant temperature coefficient is preferred because it is self-regulating. It is known that an increase in reactivity produces high power, thus increasing the coolant temperature will increase negative reactivity, which will slow or decrease the increase in power.

4.4 Poisoning

4.4.1 Reactivity associated with the xenon transients in power level 1000MW and 3000MW

This section will also introduce and discuss the calculation results of the influence of xenon on the reactivity of the system. Although the results in larger cores vary with flux power, negative reactivity is always inserted due to the accumulation of xenon. In the calculations, it was assumed that the reactor had been running for a sufficiently long duration between 0 and 4 days (approximately 50 hours longer than it would have taken xenon to reach equilibrium t> for 40 hours) before being shut down, so that iodine and xenon would have reached maximum equilibrium.

On this basis, the power levels of 1000MW and 3000MW are calculated respectively. Multiplication factor from each time step in the computer output are used to evaluate the reactivity of the system. The negative reactivity was calculated independently of the initial reactivity, and the rate of change was calculated according to the formula $\frac{\rho_2 - \rho_1}{\rho_1}$. The following figure shows the results.

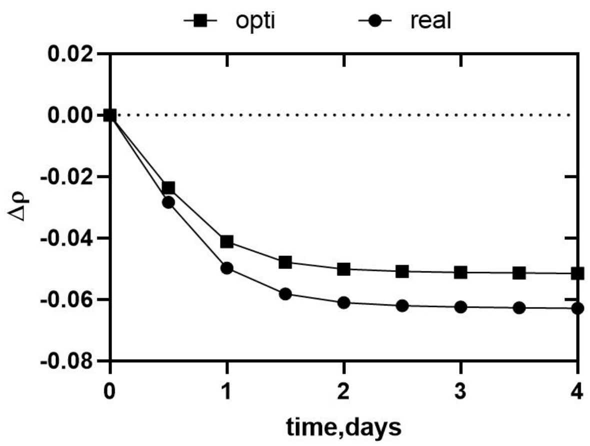


Figure31-change of negative reactivity due to the build-up of Xenon with time for optimized and real size for power level 1000MW

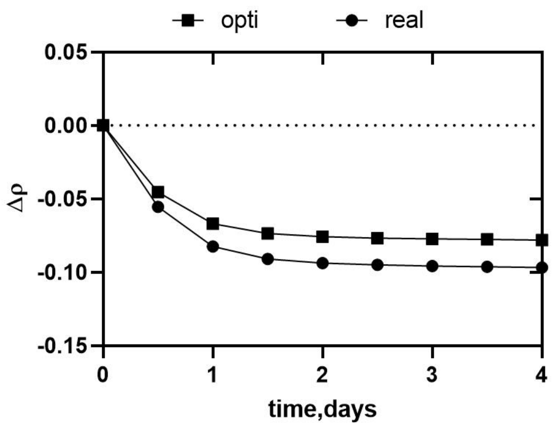


Figure32-change of negative reactivity due to the build-up of Xenon with time for optimized and real size for power level 3000MW

The diagrams show the accumulation of xenon concentrations and the negative reactivity of the corresponding vver-1000. The toxic reactivity of xenon accumulates on the Y-axis and assumes the time (days) required for xenon to reach the maximum equilibrium on the X-axis. These diagrams show the poisoning reactivity accumulation levels of xenon at different power levels of 1000MW and 3000MW respectively. The simulation time was extended above the equilibrium level (t> 40 hours) to address the effects of xenon on the poisons after maximum equilibrium. Reactivity was also observed at 0-1 day (24 h). However, the decrease was relatively small between day 1 and day 1.5 and decreased after flattening. The results can be used for two power levels, and optimization of the lattice size than the actual lattice size have lower $\Delta\rho$. It is because optimized size has higher enrichment and macroscopic fission cross section which lead to less change in thermal utilization factor, this will help to reduce the power consumption of the reactor. In addition, higher power level shows higher reactivity lost. The reason comes from concentration of xenon when it get to equilibrium. Concentration of xenon depends on flux, and higher concentration when reactor has higher flux. So in 3000MW shows higher reactivity lost than 1000MW.

Based on the data provided in these diagrams and based on the analytical experience derived from the observations. Between 0.5 and 1 day, when there is fresh fuel in the core or the reactor is in its initial operating state, the concentration of xenon in the core is therefore zero, leading to a sharp increase in reactivity.

Immediately after activation, xenon concentrations began to increase and reactivity to decrease (xenon nuclei absorbed thermal neutrons, thereby affecting the thermal utilization factor f, k_∞ which in turn affected reactivity). At 1.5 to 2 days, xenon has reached its maximum equilibrium, where xenon generation and attenuation are equal, and the rate of change of xenon accumulation becomes zero. In this case, the effect becomes constant. After the equilibrium time (1.5-2 days) when the neutron flux was essentially zero, the production of fission iodine, xenon, and the production of xenon from decay continued, resulting in a decrease between the second and fourth days. In different core regions, the power density is different, and the xenon accumulation changes accordingly. As can be seen from the figure, the accumulation of xenon depends largely on the reactor power before the shutdown. Since xenon-induced reactivity is negative, but this symbol may not be used every time, and the accumulation of xenon reactivity actually means an increase in negative reactivity.

4.4.2 Reactivity associated with the Samarium transients in power level 1000MW and 3000MW

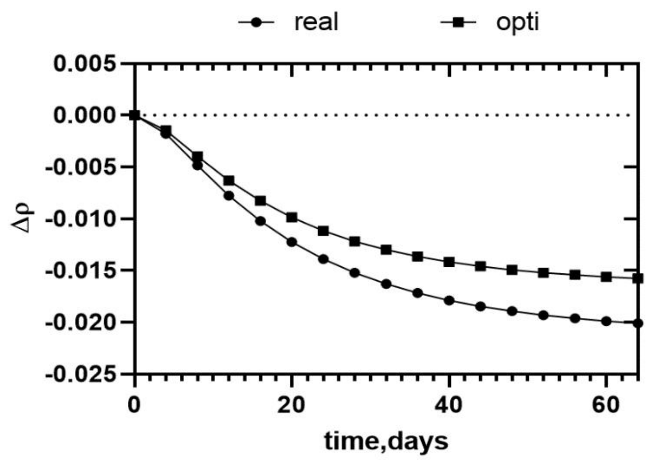


Figure33-change of negative reactivity due to the build-up of Samarium with time for optimized and real size for power level 1000MW

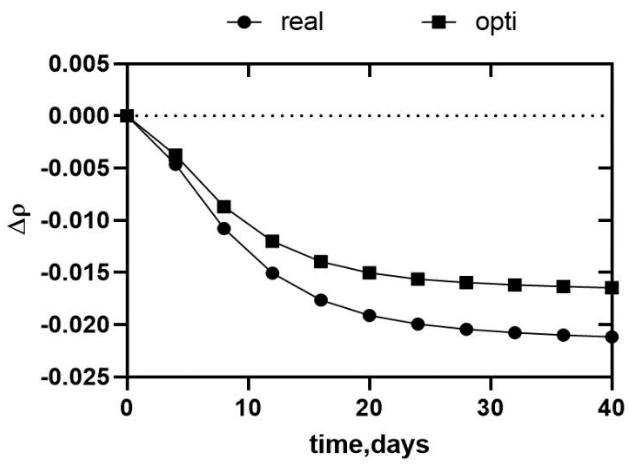


Figure34-change of negative reactivity due to the build-up of Samarium with time for optimized and real size for power level 3000MW

Results for calculations performed for the effect of samarium on the systems reactivity are also presented and discussed in this section. In calculation, it was assumed that prior to shut-down the reactor was operating for sufficiently long duration between 0-40 days because Sm-149 has long half-life so long-term experiment will increase obviousness of effect.

Based on that calculation were performed in power level 1000MW and 3000MW. The multiplication factor obtained for each time-steps from the computer output were used in the evaluating the reactivity of the system. The negative reactivity were calculated independently on the initial reactivity and rate of change is calculated at formula $\frac{\rho_2 - \rho_1}{\rho_1}$. The results of the calculated are shown in the diagrams below. It can also be observed that reactivity effect has a faster decrease between 0-15 days for power level 3000MW. And after it the decrease becomes slower. Compare it with diagrams for Xe transients, interval in the diagram is 10 days and 40 days is enough to make curve flatten. So another experiment is done in power level 1000 MW. Interval of time in figure 34 is 2 days and we can observe Sm poisoning effect. The diagram shows samarium will affect reactivity of reactor at longer time and same reactivity lost with 3000MW. The reason comes from concentration of samarium will not affected by flux. However, time needed to get equilibrium depends on power level. Sm-149 comes from decay of Pm-149 and concentration of Pm-149 will affected by power level when it get to equilibrium. So power level will affect time cost to get equilibrium by affect concentration of Pm. And optimized size also shows lower influence by effect of Sm than real size. It is because that optimized size has higher enrichment and less changes in thermal utilization factor in same value of increasement of absorption cross section. Sm-149 absorbs a large number of neutrons in the reactor, and its fine length affects thermal reduction and nuclear reactivity. In addition, because Sm-149 is not radioactive and will not be consumed by decay, causing the different problem with Xe-135. After a few weeks of reactor operation, the concentration of Sm-149 will reach an equilibrium and then remain unchanged during operation. Compared with Xe-135, reactivity lost is less. Neither the cross section nor the yield is as big as for Xe-135, lead to reactivity effect of samarium is less than xenon.

5 Financial management, resource efficiency and resource saving

The purpose of this section discusses the issues of competitiveness, resource efficiency and resource saving, as well as financial costs regarding the object of study of Master's thesis. Competitiveness analysis is carried out for this purpose. SWOT analysis helps to identify strengths, weaknesses, opportunities and threats associated with the project, and give an idea of working with them in each case. For the development of the project requires funds that go to the salaries of project participants and the necessary equipment, a complete list is given in the relevant section. The calculation of the resource efficiency indicator helps to make a final assessment of the technical decision on individual criteria and in general.

5.1 Competitiveness analysis of technical solutions.

It is important to realistically assess the strengths and weaknesses of the development of competitors. The analysis of competitive technical solutions from the standpoint of resource efficiency and resource saving makes it possible to evaluate the comparative effectiveness of scientific development and determine the directions for its future enhancement. This analysis was carried out using the evaluation map and three competitive developments have been selected. Criteria for comparison and assessment of resource efficiency and resource saving, given in Table 1. In the study of advanced core design of VVER type reactor, making the usage of WIMS-ANL code to calculate reactor core parameters. The cost of the research on the physical installation is too high as well as accuracy of the analysed data is low, therefore different technological approaches has been implemented in order to lower cost of undertaking such studies.

With this research the three technical solution includes the use:

- WIMS-ANL -P_f
- RELAP5-3D -P_{i1}
- Experimental analysis-P_{i2}

First of all, it is necessary to analyze possible technical solutions and choose the best one based on the considered technical and economic criteria. Evaluation map analysis presented in Table 1.4.1 The position of my research and competitors has been evaluated for each indicator based on a five-point scale, where 1 is the weakest position and 5 is the strongest. The weights of indicators determined in the amount 1. Analysis of competitive technical solutions is determined by the formula:

$$C = \sum W_i \cdot P_i,$$

(1)

C - the competitiveness of research or a competitor;
Wi– criterion weight;
Pi – point of i-th criteria.

Table 7 – Comparison of competitive technical solutions.

Evaluation criteria example	**Criterion weight**	**Points**			**Competitiveness**		
		P_f	P_{i1}	P_{i2}	C_f	C_{i1}	C_{i2}
1	2	3	4	5	6	7	8
Technical criteria for evaluating resource efficiency							
1. Energy efficiency	0.1	3	3	4	0.3	0.3	0.4
2. Reliability	0.2	4	5	5	0.8	1.0	1.0
3. Safety	0.2	5	5	4	1.0	1.0	0.8
4. Functional capacity	0.1	3	4	5	0.3	0.4	0.5
Economic criteria for performance evaluation							
1. Development cost	0.2	5	3	4	1.0	0.6	0.8
2. Market penetration rate	0.1	3	4	3	0.3	0.4	0.3
3. Expected lifecycle	0.1	4	4	5	0.4	0.4	0.5
Total	**1**	**29**	**27**	**32**	**4.1**	**4.2**	**4.3**

Analysis of the results shows that the competitiveness is acceptable for the suggested model. It is not feasible to make use of software or get all data by experiments.

5.1.1 SWOT analysis

The SWOT analysis is a compact method to show the results obtained by this study in a strategic way. The Strengths and Weaknesses of the PWRs (VVER-1000) nuclear power plant in the Russian scenario are reported to internal factors evaluation. Indeed the Opportunities and the Threats are reported from external factors evaluation. A collected strengths of considerable importance for the competitiveness and profitability of an investment. As for the weaknesses there is the negative effect of economies of scale and the direct consequence of this factor: the lack of competitiveness PWRs as compared to other power plants. These opportunities are very relevant and although they cannot be quantified and valued, provide a strategic advantage that adds competitiveness to a possible deployment of the. PWR reactors."Strength" are understood here as internal factor which positively impact the relative competitiveness of nuclear in the future."Weakness" are understood here as internal factor which positively impact the relative competitiveness of nuclear in the future.Opportunities" are understood here as external factor which positively impact the relative competitiveness of nuclear in the future. "Threat" here are understood as the external factors that could threaten or negatively impact the relative competitiveness of nuclear in the future.

Table 8 – SWOT Analysis.

	Strengths: S1. Nuclear power generation is much less sensitive to fuel price increase than fossil fuels. S2. Nuclear power plants do not emit CO2, and the use of nuclear power across its lifecycle results in only very small amounts of greenhouse gas emissions. S3. Satisfy demand of consumers since its used as baseload power production .	**Weaknesses:** W1. Uranium resources are limited as compared to unlimited availabilities of renewable energy resources. W2. The low utilization ratio of nuclear fuel is a major problem in the process of nuclear power development.
Opportunities: O1.High fossil fuel prices O2.Energy supply stability as compared to wind power, solar, hydro which have a characteristics of intermittency, randomness or shortage. O3.Lower energy generating cost as compared to wind, solar and so on.	**Strategy which based on strengths and opportunities:** Increase the proportion of nuclear power in the country's total power generation	**Strategy which based on weaknesses and opportunities:** Develop technology to increase the utilization of nuclear fuel
Threats: T1. Risk of accident during plants' operation, and corresponding risk perception following bad accident management.	**Strategy which based on strengths and threats:** Increase the ability to prevent and respond to nuclear accidents	**Strategy which based on weaknesses and threats:** Maintain a stable nuclear fuel reserve and increase investment in the exploration of nuclear

T2. The occurrence of nuclear accidents will affect people's attitude towards nuclear power		fuel mines

5.2 Project Initiation and Organizational Structure.

The initiation process group consists of processes that are performed to define a new project or a new phase of an existing one. In the initiation processes, the initial purpose and content are determined, and the initial financial resources are fixed. The internal and external stakeholders of the project who will interact and influence the overall result of the research project are determined.

Table 9– Stakeholders of the project.

Project stakeholders	Stakeholder expectations
TPU	Deepen the understanding of the parameters and properties of the reactor

Table 10– Purpose and results of the project.

Purpose of project:	Get dependence between reactivity and sizes of each annulus of lattice
Expected results of the project:	Get a new design of lattice
Criteria for acceptance of the project result:	New design shows better neutron characteristic than standard size
Requirements for the project result:	Shows better reactivity power effect
	Shows better reactivity coolant temperature effect
	Shows better neutronic characteristic in poisoning
	Advantages above can apply in not only one enrichment fuel,for core has several enrichment fuel exist

Table 1 – The organizational structure of the project.

№	Participant	Role in the project	Functions	Labor time, hours.
1	Engineer	Executor	Read literature, and conduct data collection, analysis and processing	150
2	Supervisor	Head of project	Formulation of research topic and direction of research Verification work through weekly meetings Control of deadlines and objectives in the research.	20

Table 11 – Project limitations.

Factors	Limitations / Assumptions
3.1. Project's budget	50000 Rub
3.1.1. Source of financing	Possibility of cutback in research investment
3.2. Project timeline:	10.01.2020-12.05.2020
3.2.1. Date of approval of plan of project	10.01.2020
3.2.2. Completion date	20.05.2020

Table 12–Project Schedule.

Job title	Duration, days	Start date	Date of completion	Participants
Development of technical specifications	2	10.01.2020	12.01.2020	Scientific supervisor
Drafting and approval of the Terms of Reference	2	12.01.2020	14.01.2020	Scientific supervisor
Research Direction	5	14.01.2020	19.01.2020	Scientific supervisor, Engineer
Collection and study scientific technical literature	30	19.01.2020	19.02.2020	Engineer
Study reactivity power effect and coolant temperature effect of standard design lattice	10	19.02.2020	10.03.2020	Engineer

Analysis on relationship between sizes of standard design and reactivity	5	10.03.2020	15.03.2020	Engineer
Get new design and do comparison with standard design lattice	30	15.03.2020	15.04.2020	Engineer
Summary and assessment of results	5	15.04.2020	20.04.2020	Engineer Scientific supervisor
Compilation of results for report preparation	10	20.04.2020	30.04.2020	Engineer
Defense preparation	20	30.04.2020	20.05.2020	Engineer

A Gantt chart is a type of bar chart that illustrates a project schedule. This chart lists the tasks to be performed on the vertical axis, and time intervals on the horizontal axis. The width of the horizontal bars in the graph shows the duration of each activity.

Table 13 – Gantt chart of Project Schedule

№	Activities	Participants	T_c, days	Duration of the project													
				February			March			April			May			June	
				1	2	3	1	2	3	1	2	3	1	2	3	1	2
1	Development of technical specifications	Scientific supervisor	2														
2	Drafting and approval of the Terms of Reference	Scientific supervisor	2														
3	Research Direction	Scientific supervisor, Engineer	5														
4	Collection and study scientific technical literature	Engineer	30														
5	Study reactivity power effect and coolant temperature effect of standard design lattice	Engineer	10														

6	Analysis on relationship between sizes of standard design and reactivity	Engineer	5
7	Get new design and do comparison with standard design lattice	Engineer	30
10	Summary and assessment of results	Scientific supervisor, Engineer	5
11	Compilation of results for report preparation	Engineer	10
12	Defense preparation	Engineer	20

– Scientific supervisor, Engineer

5.3 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 Technician 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.

Calculation of material costs

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

$$C_m = (1 + k_T) \cdot \sum_{i=1}^{m} P_i \cdot N_{consi} ,$$

where m – the number of types of material resources consumed in the performance of scientific research;

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

P_i – the acquisition price of a unit of the i-th type of material resources consumed (rub./units, rub./kg, rub./m, rub./m^2, etc.);

k_T – coefficient taking into account transportation costs.

Prices for material resources can be set according to data posted on relevant websites on the Internet by manufacturers (or supplier organizations).

Table 14– Material costs

Name	Unit	Amount	Price per unit, rub.	Material costs, rub.
notebooks	-	2	200	400
pens	-	5	159	795
notebooks				1195

Calculation of the depreciation

If you use available equipment, then you need to calculate depreciation:

$$A = \frac{C_{\text{перв}} * \text{Н}_{\text{а}}}{100}$$

A - annual amount of depreciation;

$C_{перв}$ - initial cost of the equipment;

$\text{Н}_{\text{а}} = \frac{100}{\text{Т}_{\text{сл}}}$ - rate of depreciation;

$T_{сл}$ - life expectancy.

$$C_{dep} = C_{eq} / T$$

In this research work, the special equipment necessary for conducting simulation computer which cost 35000rubles and life time expectancy of 7years

$$Cdp = Ceq/T$$

$$Cdp = 35000/7 * 365$$

$$Cdp = 13.7 rubbles/day$$

The equipment was used for 70 days, the cost of equipment:

$$Ceq = 13.7 rubbles/day * 80 day = 1095.9 rubbles$$

Table 15 – Depreciation

№	Equipment identification	Quantity of equipment	Total cost of equipment, rub.	Life expectancy, year	Depreciation for the duration of the project, rub.
1.	computer	1	45999	5	2520.5
Total					2520.5

Basic salary

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:

$$S_b = S_a \cdot T_w \text{ ,} \qquad ($$

where 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:

$$S_d = \frac{S_m \cdot M}{F_v},$$ 3.4)

где 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).

The valid annual fund of working time.

Table 16 – Work time balance

Working time indicators	Scientific supervisor
Calendar number of days	365
The number of non-working days - weekend - holidays	 52 14
Loss of working time - vacation - isolation period - sick absence	 48 7
The valid annual fund of working time	244

Monthly salary is calculated by formula:

$$S_{month} = S_{base} \cdot (k_{premium} + k_{bonus}) \cdot k_{reg},$$ x)

where S_{base} – base salary, rubles;

$k_{premium}$ – premium rate;

k_{bonus} – bonus rate;

k_{reg} – regional rate.

Table 17 – Calculation of the base salaries

Performers	S_{base}, rubles	$k_{premium}$	k_{bonus}	k_{reg}	S_{month}, rub.	W_d, rub.	T_p, work days	W_{base}, rub.
Supervisor	35120	–	–	1.3	45656	2095.7	14	29339.6
Engineer	17310				22503	1032.9	130	134280.2
Total								163619.8

Additional salary

This point includes the amount of payments stipulated by the legislation on labor, for example, payment of regular and additional holidays; payment of time associated with state and public duties; payment for work experience, etc.

Additional salaries are calculated on the basis of 10-15% of the base salary of workers:

$$W_{add} = k_{extra} \cdot W_{base} \,, \qquad \text{x)}$$

where W_{add} – additional salary, rubles;

k_{extra} – additional salary coefficient (10%);

W_{base} – base salary, rubles.

Table 18– Additional Salary

Participant	Additional Salary, rubles
Supervisor	2933.96
Engineer	13428.02
Total	16361.98

Labor tax

Tax to extra-budgetary funds are compulsory according to the norms established by the legislation of the Russian Federation to the state social insurance (SIF), pension fund (PF) and medical insurance (FCMIF) from the costs of workers.

Payment to extra-budgetary funds is determined of the formula:

$$P_{social} = k_b \cdot (W_{base} + W_{add})$$
x)

where k_b – coefficient of deductions for labor tax.

In accordance with the Federal law of July 24, 2009 No. 212-FL, the amount of insurance contributions is set at 30%. Institutions conducting educational and scientific activities have rate - 27.1%.

Table19 – Labor tax

	Project leader	**Engineer**
Coefficient of deductions	0.271	
Salary, rubles	32273.56	147708.22
Labor tax, rubles	8746.13	40028.93
Total	48775.06	

Overhead costs

Overhead costs include other management and maintenance costs that can be allocated directly to the project. In addition, this includes expenses for the maintenance, operation and repair of equipment, production tools and equipment, buildings, structures, etc.

Overhead costs account from 30% to 90% of the amount of base and additional salary of employees.

Overhead is calculated according to the formula:

$$C_{ov} = k_{ov} \cdot (W_{base} + W_{add})$$
x)

where k_{ov} =50% – overhead rate.

Table 20 – Overhead

	Project leader	**Engineer**
Overhead rate	0.3	
Salary, rubles	32273.56	147708.22
Overhead, rubles	9682.07	44312.47
Total	53994.54	

Other direct costs

Energy costs are calculated by the formula:

$$C = P_{el} \cdot P \cdot F_{eq},$$

where P_{el} – power rates (5.8 rubles per 1 kWh);

P – power of equipment, kW;

F_{eq} – equipment usage time, hours.

Table 21– Other direct costs

Name	Power of equipment, kW	Amount,hours	Price per unit, rub.	Material costs, rub.
Energy costs	0.6	1300	5.8	4524
Total				4524

Formation of budget costs

The calculated cost of research is the basis for budgeting project costs.

Determining the budget for the scientific research is given in the table .

Table 17– Items expenses grouping

Name	Cost, rubles
1. Material costs	1195
2. Depreciation	2520.5
3. Basic salary	163619.8
4. Additional salary	16361.98
5. Labor tax	48775.06
6. Overhead	53994.54
7. Other direct cost	4524
Total planned cost	**290990.88**

Evaluation of the comparative effectiveness of the project

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. For 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 integral financial measure of development is defined as:

$$I_{\phi}^{p} = \frac{\Phi_{pi}}{\Phi_{\max}}, \quad \text{(x)}$$

where I_{ϕ}^{p} – integral financial measure of development;

Φ_{pi} – the cost of the i-th version;

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

The obtained value of the integral financial measure 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 performance, then $I_{\phi}^{p} = 1$.

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

$$I_{m}^{a} = \sum_{i=1}^{n} a_i b_i^a, \quad I_{m}^{p} = \sum_{i=1}^{n} a_i b_i^p \quad \text{(-)}$$

where I_m – 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 – score 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 table 22.

Table 22 – Evaluation of the performance of the project

Criteria	**Weight criterion**	**Points**
1. 1. Energy efficiency	0.1	3
2. Reliability	0.2	4
3. Safety	0.2	5
4. Functional capacity	0.1	3
Economic criteria for performance evaluation		
1. The cost of development	0.2	5
2. Market penetration rate	0.1	3
3. Expected life	0.1	4
Total	1	4.1

The integral indicator of the development efficiency (I_e^p) is determined on the basis of the integral indicator of resource efficiency and the integral financial indicator using the formula:

$$I_e^p = \frac{I_m^p}{I_f^d}, \quad I_e^a = \frac{I_m^a}{I_f^a} \quad \text{and etc.} \qquad (=)$$

Comparison of the integral indicator of the current project efficiency and analogues will determine the comparative efficiency. Comparative effectiveness of the project:

$$E_c = \frac{I_e^p}{I_e^a}. \qquad (=)$$

Thus, the effectiveness of the development is presented in table 23.

Table 23 – Efficiency of development

№	**Indicators**	Points
1.	Integrated Financial Development Indicator	4
2.	Integral indicator of resource efficiency of development	3
3.	Integral indicator of the development efficiency	5

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.

Therefore, this section was developed stages for design and create competitive development that has the requirements in the resource and energy saving. Hope we have a better tomorrow.

6 Social responsibility

6.1 Introduction

Main work of the master's thesis is obtaining some new design of lattice and completing experiments to prove it has better neutronic characteristics. Most of work is completed using software WIMS-ANL and data processing is done by excel. The main purpose of work is improvement of reactor economy and competitiveness. Because of effect of coronavirus, the workplace of experiment is mostly at dormitory. The higher economics of nuclear power will also meet the needs of the people and at the same time lower costs.

6.2 Legal and organizational items in providing safety

Nowadays one of the main way 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.

According to the Labor Code of the Russian Federation, every employee has the right:

- to have a workplace that meets Occupational safety requirements;

- to have a compulsory social insurance against accidents at manufacturing and occupational diseases;

- to receive reliable information from the employer, relevant government bodies and public organizations on conditions and Occupational safety at the workplace, about the existing risk of damage to health, as well as measures to protect against harmful and (or) hazardous factors;

- to refuse carrying out work in case of danger to his life and health due to violation of Occupational safety requirements;

- be provided with personal and collective protective equipment in compliance with Occupational safety requirements at the expense of the employer;

- for training in safe work methods and techniques at the expense of the employer;

- for personal participation or participation through their representatives in consideration of issues related to ensuring safe working conditions in his workplace, and in the investigation of the accident with him at work or occupational disease;

- for extraordinary medical examination in accordance with medical recommendations with preservation of his place of work (position) and secondary earnings during the passage of the specified medical examination;

- for warranties and compensation established in accordance with this Code, collective agreement, agreement, local regulatory an act, an employment contract, if he is engaged in work with harmful and (or) hazardous working conditions.

The labor code of the Russian Federation states that normal working hours may not exceed 40 hours per week, The employer must keep track of the time worked by each employee.

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.

6.3 Basic ergonomic requirements for the correct location and arrangement of researcher's workplace

The workplace when working with a PC should be at least 6 square meters. The legroom should correspond to the following parameters: the legroom height is at least 600 mm, the seat distance to the lower edge of the working surface is at least 150 mm, and the seat height is 420 mm. It is worth noting that the height of the table should depend on the growth of the operator.

The following requirements are also provided for the organization of the workplace of the PC user: The design of the working chair should ensure the maintenance of a rational working posture while working on the PC and allow the posture to be changed in order to reduce the static tension of the neck and shoulder muscles and back to prevent the development of fatigue.

The type of working chair should be selected taking into account the growth of the user, the nature and duration of work with the PC. The working chair should be lifting and swivel, adjustable in height and angle of inclination of the seat and back, as well as the distance of the back from the front edge of the seat, while the adjustment of each parameter should be independent, easy to carry out and have a secure fit.

6.4 Occupational safety

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.

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.

6.4.1 Analysis of harmful and dangerous factors that can create object of investigation

The object of investigation is “Advanced core design of VVER type nuclear reactor”. Most of work is completed on PC. Therefore object of investigation itself cannot cause harmful and dangerous factors.

6.4.2 Analysis of harmful and dangerous factors that can arise at workplace during investigation

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 24.

Table 24 - Possible hazardous and harmful factors

Factors (GOST 12.0.003-2015)	Work stages			Legal documents
	Development	Manufacture	Exploitation	
1. Deviation of microclimate indicators	+	+	+	Sanitary rules 2.2.2 / 2.4.1340–03. Sanitary and epidemiological rules and regulations "Hygienic
2. Excessive noise		+	+	
3.Increased level of electromagnetic	+	+	+	

radiation				requirements for personal electronic computers and work organization."
4.Insufficient illumination of the working area		+	+	Sanitary rules 2.2.1 / 2.1.1.1278–03. Hygienic requirements for natural, artificial and combined lighting of residential and public buildings. Sanitary rules 2.2.4 / 2.1.8.562–96. Noise at workplaces, in premises of residential, public buildings and in the construction area. Sanitary rules 2.2.4.548–96. Hygienic requirements for the microclimate of industrial premises.
5. Abnormally high voltage value in the circuit, the closure which	+	+	+	Sanitary rules GOST 12.1.038-82 SSBT. Electrical safety.

may occur through the human body				Maximum permissible levels of touch voltages and currents.

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;

psychophysiological:

- psychophysiological dangerous and harmful factors are divided into:
  - physical overload (static, dynamic)
  - mental stress (mental overstrain, monotony of work, emotional overload).

Deviation of microclimate indicators

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 and are given in Table 25.

Table 25 - 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

Excessive noise

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 result 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.

Increased level of electromagnetic radiation

The screen and system blocks produce electromagnetic radiation. Its main part comes from the system unit and the video cable. According to [2], 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.

Abnormally high voltage value in the circuit

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.

Table 26-Upper limits for values of contact current and voltage

	Voltage, V	Current, mA
Alternate, 50 Hz	2	0.3
Alternate, 400 Hz	3	0.4
Direct	8	1.0

Insufficient illumination of the working area

Light sources can be both natural and artificial. The natural source of the light in the room is the sun, artificial light are lamps. With long work in low illumination conditions and in violation of other parameters of the illumination, visual perception decreases, myopia, eye disease develops, and headaches appear.

According to the standard, the illumination on the table surface in the area of the working document should be 300-500 lux. Lighting should not create glare on the surface of the monitor. Illumination of the monitor surface should not be more than 300 lux.

The brightness of the lamps of common light in the area with radiation angles from 50 to 90° should be no more than 200 cd/m, the protective angle of the lamps should be at least 40°. The safety factor for lamps of common light should be assumed to be 1.4. The ripple coefficient should not exceed 5%.

6.4.3 Justification of measures to reduce the levels of exposure to hazardous and harmful factors on the researcher

Deviation of microclimate indicators

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.

Excessive noise

In research audiences, there are various kinds of noises that are generated by both internal and external noise sources. The internal sources of noise are working equipment, personal computer, printer, ventilation system, as well as computer equipment of other engineers in the audience. If the maximum permissible conditions are exceeded, it is sufficient to use sound-absorbing materials in the room (sound-absorbing wall and ceiling cladding, window curtains). To reduce the noise penetrating outside the premises, install seals around the perimeter of the doors and windows.

Increased level of electromagnetic radiation

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, 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.

Abnormally high voltage value in the circuit

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).

Insufficient illumination of the working area

Desktops should be placed in such a way that the monitors are oriented sideways to the light openings, so that natural light falls mainly on the left.

Also, as a means of protection to minimize the impact of the factor, local lighting should be installed due to insufficient lighting, window openings should be equipped with adjustable devices such as blinds, curtains, external visors, etc.

6.5 Ecological safety

6.5.1 Analysis of the impact of the research object on the environment

Most nuclear power plants release gaseous and liquid radiological effluents into the environment, which must be monitored. Civilians living within 80 km of a nuclear power plant typically receive about 0.1 μSv per year.

All reactors are required to have a containment building in according to international requirements. The walls of containment buildings are several feet thick and made of concrete and therefore can stop the release of any radiation emitted by the reactor into the environment

Large volumes of water are used during the process of nuclear power generation. The uranium fuel inside reactors undergoes induced nuclear fission which releases great amounts of energy that is used to heat water. The water turns into steam and rotates a turbine, creating electricity. Nuclear plants are built near bodies of water.

All possible impact of nuclear power plant on environment is greatly reduced in operating regime by many safety precautions means. The most danger of nuclear energy come because of different sorts of disaster

6.5.2 Analysis of the environmental impact of the research process

Process of investigation itself in the thesis do not have essential effect on environment. One of hazardous waste is fluorescent lamps. Mercury in fluorescent lamps is a hazardous substance and its improper disposal greatly poisons the environment.

Outdated devices goes to an enterprise that has the right to process wastes. It is possible to isolate precious metals with a purity in the range of 99.95–99.99% from computer components. A closed production cycle consists of the following stages: primary sorting of equipment; the allocation of precious, ferrous and non-ferrous metals and other materials; melting; refining and processing of metals. Thus, there is an effective disposal of computer devices.

6.5.3 Justification of environmental protection measures

Pollution reduction is possible due to the improvement of devices that produces electricity, the use of more economical and efficient technologies, the use of new methods for generating electricity and the introduction of modern methods and methods for cleaning and neutralizing industrial waste. In addition, this problem should be solved by efficient and economical use of electricity by consumers themselves. This is the use of more economical devices, as well as efficient regimes of these devices. This also includes compliance with production discipline in the framework of the proper use of electricity.

Simple conclusion is that it is necessary to strive to reduce energy consumption, to develop and implement systems with low energy consumption. In modern computers, modes with reduced power consumption during long-term idle are widely used.

6.6 Safety in emergency

6.6.1 Analysis of probable emergencies that may occur at the workplace during research

The fire is the most probable emergency in our life. 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.

6.6.2 Substantiation of measures for the prevention of emergencies and the development of procedures in case of emergencies

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, and 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 fire works);
- 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.

In this section about social responsibility the hazardous and harmful factors were revealed. All necessary safety measures and precaution to minimize probability of accidents and traumas during investigation are given. Possible negative effect on environment were given in compact form describing main ecological problem of using nuclear energy. It could be stated that with respect to all regulations and standards, investigation itself and object of investigation do not pose special risks to personnel, other equipment and environment.

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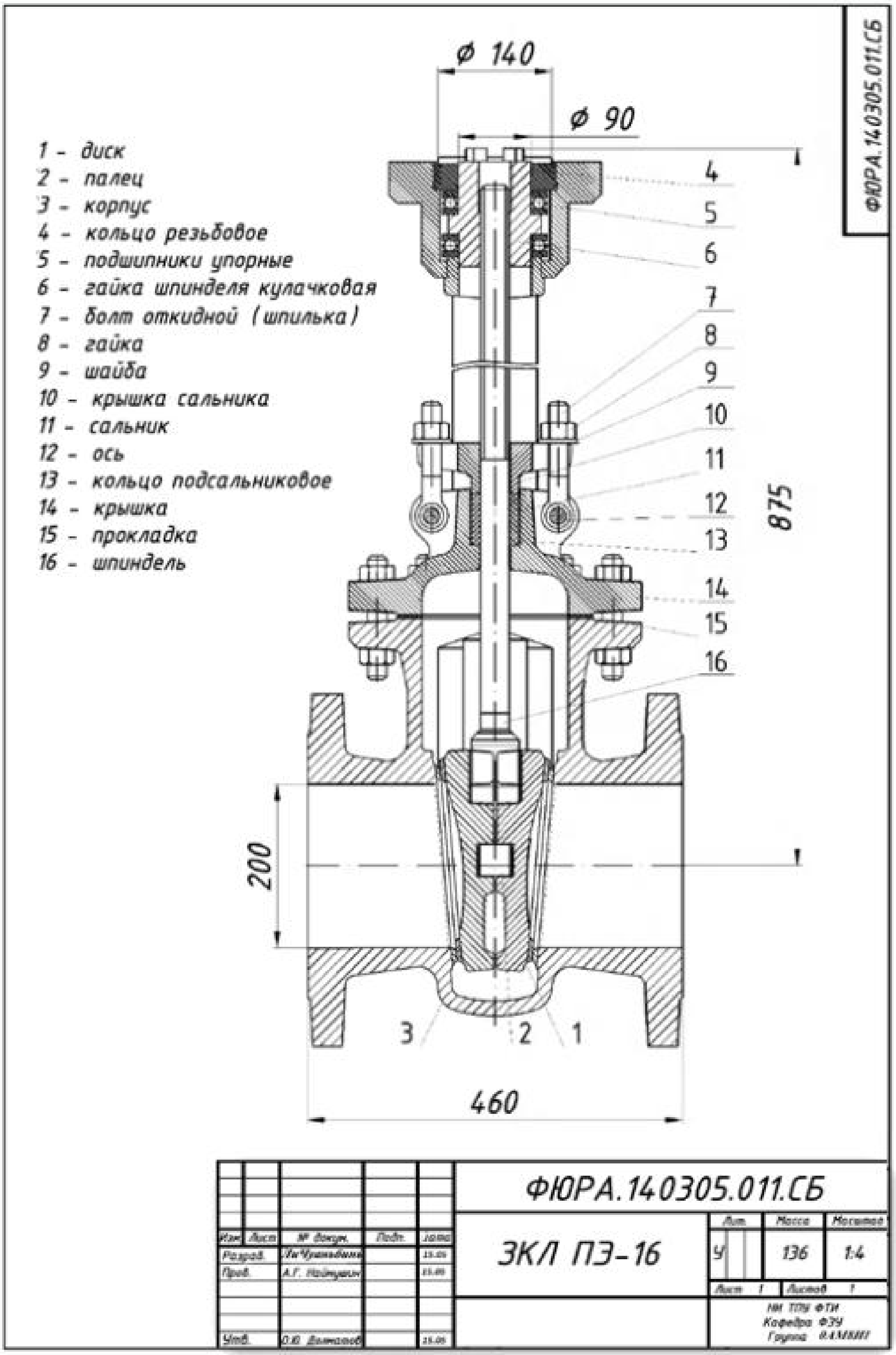
Appendix



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ϕ 140
ϕ 90
1 - диск
2 - палец
3 - корпус
4 - кольцо резьбовое
5 - подшипники упорные
6 - гайка шпинделя кулачковая
7 - болт откидной (шпилька)
8 - гайка
9 - шайба
10 - крышка сальника
11 - сальник
12 - ось
13 - кольцо подсальниковое
14 - крышка
15 - прокладка
16 - шпиндель
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Подп.
Дата
Разраб.
Пров.
А.Г. Найтушин
Утв.
Лист 1
Листов 1
НИ ТПУ ФТИ
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Формат	Зона	Поз.	Обозначение	Наименование	Кол.	Примечание
				<u>Документация</u>		
А3			ФЮРА.140305.011.СБ	Задвижка клиновая	1	
				под электропривод		
				<u>Детали</u>		
		1	ФЮРА.140305.011.01	Диск	1	
		2	ФЮРА.140305.011.02	Палец	1	
		3	ФЮРА.140305.011.03	Корпус	1	
		4	ФЮРА.140305.011.04	Кольцо резьбовое	1	
		5	ФЮРА.140305.011.05	Подшипники упорные	2	
		6	ФЮРА.140305.011.06	Гайка шпинделя кулачковая	1	
		7	ФЮРА.140305.011.07	Болт откидной (шпилька)	4	
		8	ФЮРА.140305.011.08	Гайка	6	
		9	ФЮРА.140305.011.09	Шайба	1	
		10	ФЮРА.140305.011.10	Крышка сальника	1	
		11	ФЮРА.140305.011.11	Сальник	1	
		12	ФЮРА.140305.011.12	Ось	1	
		13	ФЮРА.140305.011.13	Кольцо подсальниковое	1	
		14	ФЮРА.140305.011.14	Крышка	1	
		15	ФЮРА.140305.011.15	Прокладка	1	
		16	ФЮРА.140305.011.16	Шпиндель	1	

Изм.	Лист	№ докум.	Подпись	Дата	ФЮРА.140305.011.СБ			
Разраб.		Ли Чуаньбинь		15.05	Спецификация	Лит.	Лист	Листов
Пров.		А.Г. Наймушин		15.05		У	1	1
						НИ ТПУ ФТИ Группа 04М8И1		
Утвердил		О.Ю. Доматов		15.05				

Conclusion

In order to achieve better fuel economy, we redesigned the lattice according to the effect of each annulus size on the reactivity. In the calculation, new design sizes for each region (gap, fuel, cladding) of the lattice cell were investigated. Diagrams of reactivity varies according to radius were plotted for each annulus. After optimized size is determined, a series of experiments are done using WIMS-ANL code. There are four parts including fuel cycle, radial flux and temperature distribution, power effect, coolant temperature effect and poison part. And result of comparison between optimized size and real size are shown in tables and diagrams. In most aspects, optimized size designed by the master's thesis shows better characteristics than real size of VVER-1000.This master's thesis completed purpose and work of objective, new core design shows better neutron characteristic properties.