LAPPEENRANTA-LAHTI UNIVERSITY OF TECHNOLOGY LUT

School of Energy Systems

Energy Technology

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SAFETY AND ECONOMY OF FLOATING POWER PLANTS

Examiners: Professor Juhani Hyvärinen
Assistant Professor Heikki Suikkanen

Supervisors: Professor Juhani Hyvärinen
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ABSTRACT

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Keywords: KLT-40S, RITM-200, SMR, FPP, safety, economy, floating power plant.

In this dissertation Russian floating power plant "Akademik Lomonosov" will be observed and its safety parameters will be assessed as well as its economic feasibility. The information on the plant design and concept will be gathered and analyzed to obtain the perspective of the facility. Also, two Russian SMRs will be observed – KLT-40S and RITM-200, which are typically designed for floating power plants and icebreakers. For each reactor, safety-oriented values will be calculated regarding next parameters: 1) thermal hydraulics – temperatures, heat fluxes and natural circulation rates, 2) reactor physics characteristics – burnup and neutron economy index. Economic feasibility will be evaluated for floating power plant by use of simplified calculations of payback time and levelized cost of electricity, which can show the viability of the project. All economic parameters will be compared to the Finnish electricity prices as well as to the Russian stationary nuclear power plant tariffs.

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Lappeenranta, 29.05.2021

Dmitrii Dziadevich

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LIST OF SYMBOLS

Latin:		
B	Fuel burnup	$\mathrm{MWd/kgU}$
C	Electricity price; Heat supply price	$€/(\mathrm{MW \cdot h})$
c_p	Isobaric specific heat	$\mathrm{kJ/(kg \cdot K)}$
D	Hydraulic diameter; Reactor core diameter	m
d	Fuel element diameter; Control rod diameter	m
E	Energy	$\mathrm{J;\ MeV}$
F	Coolant flow area in one fuel assembly	$\mathrm{m^2}$
H	Reactor core height	m
K	Capital building costs	$€$
k	Form factor	$-$
L	Fuel assembly across flats size	m
m	Heavy metal mass; Uranium mass	kg
N	Amount of fuel assemblies	pcs.
NEI	Neutron Economy Index	$-$
n	Amount of fuel elements	pcs.
n	Exponential parameter	$-$
P	Installed electric capacity; Heating capacity	$\mathrm{MWe; Gcal/h}$
p	Pressure	MPa
Q	Thermal power	MW
q'	Linear power	$\mathrm{W/m}$
q''	Heat flux	$\mathrm{W/m^2}$
q'''	Power density	$\mathrm{W/m^3}$
R	Resistance constant	$-$
S	Fuel assembly surface	$\mathrm{m^2}$
T	Temperature	$\mathrm{℃, K}$
V	Volume	$\mathrm{m^3}$
w	Coolant velocity	$\mathrm{m/s}$
z	Position	$-$

Greek:		
α	Heat transfer coefficient	$\mathrm{W/(m^2 \cdot K)}$
β	Thermal expansion coefficient	$1/\mathrm{°C}$
Δ	Difference of two values	–
δ	Extrapolated distance	m
δ	Wall thickness	m
λ	Thermal conductivity coefficient	$\mathrm{W/(m \cdot K)}$
ν	Kinematic viscosity	$\mathrm{m^2/s}$
ρ	Density	$\mathrm{kg/m^3}$
φ	Neutron flux distribution	–

Dimensionless numbers:		
Nu	Nusselt number	–
Pr	Prandtl number	–
Re	Reynolds number	–

Constants:		
e	Electron volt energy $= 1.602 \cdot 10^{-19}$	J/MeV
g	Acceleration of gravity $= 9.81$	$\mathrm{m/s^2}$
N_A	Avogadro number $= 6.022 \cdot 10^{23}$	$1/\mathrm{mol}$
π	Pi number $= 3.14$	–

Indices:	
ave	Average
cl	Cold leg
cld	Cladding
CL	Center line
cr	Critical
el	Electrical
FA, fa	Fuel assembly
FE, fe	Fuel element

fr	Fuel rod
g	Gravity
gg	Gas gap
h	Hydraulic
ic	Inner cladding
in	Inlet
l	Loop
max	Maximum
mean	Mean
nc	Natural circulation
oc	Outer cladding
out	Outlet
p	Pump
r	Radial
rod	Control rod
th	Thermal
w	Coolant
xtr	Extrapolated
z	Axial direction

LIST OF ABBREVIATIONS

ECCS	Emergency Core Cooling System
FPP	Floating Power Plant
FPU	Floating Power Unit
IAEA	International Atomic Energy Agency
IMO	International Maritime Organization
KTZ	Kaluzhsky Turbinniy Zavod (Turbine factory)
LCOE	Levelized Cost of Electricity
MCP	Main Coolant Pump
NEI	Neutron Economy Index
NPP	Nuclear Power Plant
NSSS	Nuclear Steam Supply System
RHRS	Residual Heat Removal System
RP	Reactor Plant
SG	Steam Generator
SIS	Safety Injection System

1. INTRODUCTION

Nowadays, mankind brings a claim to energy technology pioneers and its sympathizers to develop such power plants, that produced energy would be eco-friendly, and utilized fuel would be NO_x- and CO_2-free. Although the demands for electricity and heat increase annually as well.

Nuclear power plants (NPP) do not produce traditional energy, such as gas and coal power plants, nor do they produce so-called renewable energy, such as solar, wind, and so on. Even if nuclear fuel doesn't emit air pollutions as traditional ones, it is still an important problem to utilize the spent fuel, because it is able to cause severe damage anyway.

Nuclear energy was used for naval purposes since 60's, at submarines and icebreakers, and not so long time ago floating nuclear power plant (FPP) has been launched in Russia. Its purpose is to provide with heat and electricity remote and hard to reach towns and areas. Moreover, in some areas power plant can't be built due to such problems as infrastructure and transport issues, which ruin all the reasons to build a plant.

There have been a lot of controversial debates around FPP's economical parameters and safety characteristics. On the one hand, small modular reactors (SMR) are less expensive to build than traditional reactors; however, this SMR is surrounded by a massive and costly barge with a displacement of 21,500 tons [9], and if a critical unit of the FPP had critically failed, something tragic would have happened in the middle of the sea.

1.1. Thesis objectives

Finishing the introduction, the aim of this work can be divided in several parts:

1) To gather, structure and analyze actual information about FPP and two reactors: KLT-40S – actually used one, and RITM-200 – the one that is designed to work at future FPPs;
2) To calculate significant parameters of reactors, that have a straight impact on nuclear safety;
3) To form an estimate about economic viability of this facility, its profitableness.

2. FLOATING POWER PLANT CONCEPT

The most recent topic in the Russian Federation is the use of atomic energy in the area of district heating, which is the most significant sector of fuel and energy resource consumption [8].

The regions of the North and remote regions equated to them occupy more than half of Russia's territory and are home to 20 million people. These locations are distinguished by their isolation from year-round water transport routes and railways. The richest mineral reserves were discovered and developed here. Two-thirds of the country's natural resource potential is concentrated in Russia's north, requiring significant energy capacities to implement [8].

2.1. The first ever floating power plant…

… was designed and constructed in USA and operated in 1968-1975 years. MH-1A – Sturgis (named after general Samuel D. Sturgis Jr.) – a one-loop pressurized water reactor; was used to produce 10 MWe. Power station purpose was to supply an inaccessible site with electricity – the Panama Canal Zone [27].

Figure 2.1. The first floating power plant MH-1A (Sturgis) [27].

In 1976 the plant was retired from service, due to high operation costs and the land-based power plant was built already.

2.2. Project 20870

In 2007, project 20870 – Russian FPP "Akademik Lomonosov" – has been started. Background of this project is the same as of Sturgis – to supply remote areas with electricity. So, the FPP should annually and reliably operate in Arctic and Far East. Except supplying the town Pevek (fig. 2.2 – highlighted with a red dot), FPP can be used for powering the main mining companies located in Chukotka in the Chaun-Bilibino energy center – a large ore-metal cluster, including gold mining companies and projects related to the development of the Baim ore zone [15].

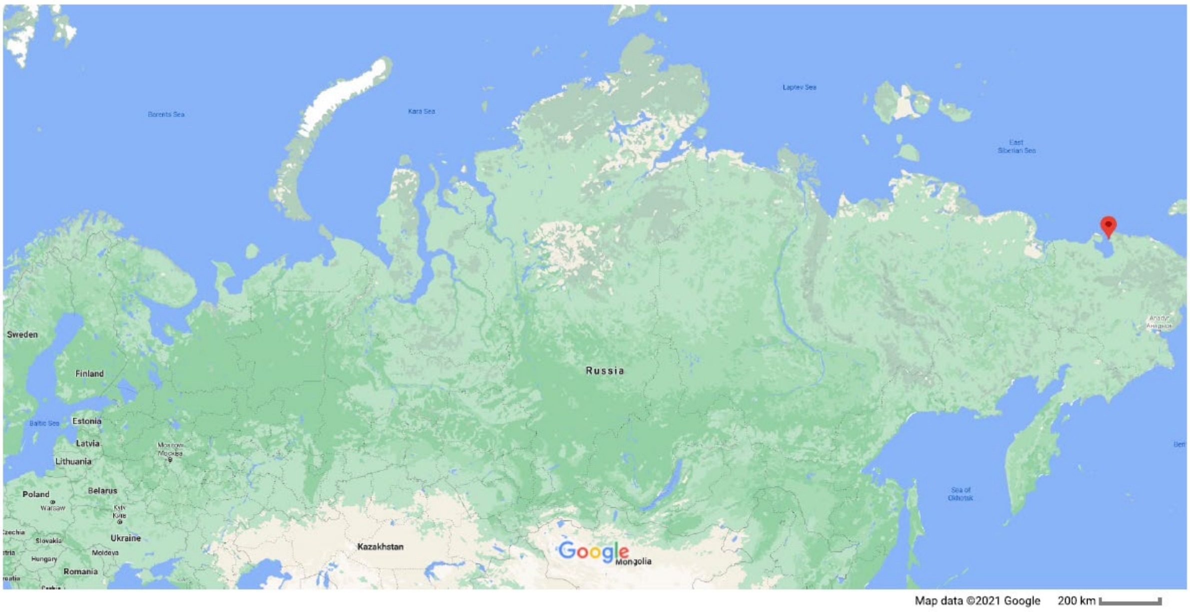


Figure 2.2. Pevek on Google Maps.

2.3. "Akademik Lomonosov"

FPP is a non-self-propelled power barge (i.e. it should be anchored in a long-term) of KE*(2) A2 class, where [23]:

1) KE* – for non-self-propelled ships and floating facilities with total power output of prime movers 100 kW and more;
2) (2) – flooding of two adjacent compartments won't affect floating of the vessel;
3) A2 – automation extent is sufficient for the machinery installation operation by one operator at the main machinery control room with unattended machinery spaces.

General information on ship parameters presented in the table 2.1.

Table 2.1. FPP vessel main characteristics [9].

	Name	Value
1.	Water displacement, t	21,500
2.	Dimensions:	
	Length, m	140
	Width, m	30
	Draught, m	5.56
	Side wall height, m	10
	Superstructure, m	~30
3.	Reliability parameters:	
	Life time, years	35-40
	Period between maintenance, years	10-12
	Maintenance period, years	1
	Main equipment life time, thousand hr	240-300
4.	Required resources:	
	Irretrievable water intake, m^3/year	3650
	Drinking water, m^3/day	18
	Water volume for sewer system, m^3/day	25
	Electricity for own needs, MWe	9.3

Figure 2.3. FPP vessel [5].

As it is seen from the figure 2.3, the barge is quite large – the vehicles look minute compared to the vessel.

2.4. FPP turbine characteristics

There are two steam turbines, one for each reactor. TK-35/38-3.4 is a steam turbine of Kaluzhsky Turbinniy Zavod (KTZ) – turbine factory. The marking of the turbine means next [1]:

1) TK means that steam turbine contains controllable extraction;
2) 35/38 stands for nominal and maximum power output consequently, MW;
3) 3.4 is for a live steam pressure at the head of the turbine, MPa.

Turbine has three extractions along its axis: 1st and 3rd are for heating the feedwater in regenerative system heat-exchangers, and 2nd is a controllable extraction, which is designed for the district heating. [1]

Table 2.2. TK-35/38-3.4 turbine parameters [8].

	Name	Value
1.	Power output:	
	Nominal electrical, MWe	2x35
	Maximum electrical, MWe	2x38.5
	Nominal heating, Gcal/h	2x25
	Maximum heating, Gcal/h	2x73
2.	Live steam parameters:	
	Pressure, MPa	3.43
	Temperature, °C	285

2.5. Storage

Power barge designed in a way, that spent nuclear fuel will be stored onboard of the FPP, so no other vessels are needed to store it. FPP contains storages for spent fuel assemblies (FA). Firstly, the spent fuel follows to a wet storage, where leak-tight tanks are used, and where decay heat removal is performed. In total, there are three independent wet storage tanks, capacity of each one is enough to store in there spent FA's of one reactor. Afterwards, fuel transferred to a dry storage, where leak-tight canisters are used. [9]

In wet storage decay heat removal is performed with one of two active heat removal channels through three circuits: cooling circuit – intermediate circuit – seawater. In addition, passive heat removing is going on because of evaporating of the water. [9]

In dry storage heat removal proceeds with an open-loop ventilation. [9]

2.6. Safety characteristics

According to Bellona community’s report, for the past sixty years there were more than 40 atomic accidents at submarines, and a few at atomic icebreakers, only in Russia [1]. Due to the fuel nature, that is utilized at these vessels, and to the fact, that these vessels placed at the ships that surf the oceans and the seas, nuclear safety should be performed at the extremely high level.

FPP’s safety solutions are combined of passive and active systems, according to worldwide trends, including International Atomic Energy Agency (IAEA) safety standards and Russian codes and standards. [1]

At the FPP used defense-in-depth principle and safety measures are divided in 5 levels. In the overview document of the KLT-40S in the IAEA report, according to NP-022-2000 these levels are explained. But this Code is inactive, and nowadays it is recommended to use NP-022-17 [3] instead. In actual documentation these levels defined as [3]:

Level 1: Prevention of failures while normal operation

- Development of the design documentation for the vessel, based on conservative approach with a developed inherent safety protection of the reactor plant and measures, aimed at eliminating the threshold effect;
- Ensuring the required quality of the systems and components of the ship important for safety work, performed in the field of atomic energy use;
- Operation of the vessel, according to the requirements of guidelines and operating instructions;
- Keeping systems in a working state and safety-important elements, determining the defects, using preventive measures, controlling the resource, organizing efficient maintenance system, managing the work documentation;
- Selecting and providing the required-skill level of the ship personnel to carry out work in the area of use atomic energy, during normal operation and in case of

abnormal ones, including pre-emergency situations and accidents, building a safety culture.

Level 2: Prevention of abnormal accidents with systems of normal operation

- Early detection of operation deviations and their elimination;
- Safety control in abnormal operation.

Level 3: Prevention of severe accidents

- Preventing the escalation of initial events into abnormal conditions, and of abnormal conditions into severe ones;
- Mitigating the consequences of accidents, that could not be prevented, by localization the radioactive substances.

Level 4: Control of severe accidents

- Return of the RP to a controllable state, in which fission mitigates, and constant cooling of nuclear fuel is provided and the radioactive substances are held within the vessel boundaries;
- Preventing the development of severe accidents and mitigating the consequences, with use of special technical means for management of such accidents, as well as any technical means, capable of performing the required functions under the prevailing conditions;
- Protection against destruction of the protective shell and (or) protective fence in case of severe accidents and maintaining their performance.

Level 5: Emergency planning

- Preparation and implementation of work to protect the working personnel in the event of a severe accident on board, and measures to protect the population, providing assistance to the ship personnel and (or) special personnel of the ship with attracting additional forces and means.

Another Russian code for nuclear safety on ships is ND 2-020101-112 [22], which was also used in the design of the FPP. This code correlates with the resolution A.491(XII) – code of safety for nuclear merchant ships by International Maritime Organization (IMO) [7]. These are having the same aims and reasons in their contents.

For example, in the resolution A.491(XII), Chapter 4 – NSSS, can be found such statements:

"4.3.1.1. The likelihood of events resulting in unplanned reactivity increases should be remote, as defined in Chapter 1, and should not lead to situations which pose a hazard to the public, crew or environment greater than that defined in Chapters 1 and 6" [7, p.52] – meaning of this statement correlates with the one from ND 2-020101-112, Chapter XVIII, item 19.11.1 [22].

Point about the pressure vessel, which is presented below, can be referred to the item 13.3 of Chapter XVIII [22] in the Russian Code.

"4.6.2. The primary pressure boundary should be designed with sufficient margin so that, when stressed under operation, maintenance, testing and postulated accident conditions, the boundary behaves in a ductile manner. The design should reflect consideration of service temperatures and other conditions affecting the boundary material under these conditions, as well as the uncertainties in determining:

.1 material properties;

.2 effects of irradiation on material properties;

.3 residual, steady-state and transient stresses; and

.4 sensitivity of non-destructive test equipment and test frequency." [7, p.55]

2.6.1. Main passive systems

First of all, inherent ones – feedback on reactivity insertions – doppler effect, moderator temperature, voids. Also, as inherent safety features following are declared [9]:

1) Thermal conductivity effectiveness; fuel stored energy is relatively low;
2) Natural circulation;
3) Compact design excludes pipework of a large diameter;
4) Use of burnable absorber.

Passive systems are high of importance. In case of blackouts, abnormal transients or other unseen circumstances, once passive safety systems are launched, the situation will be regulated to normal condition or a handicap to the personnel to search for a solution of a problem will be given.

Passive systems are [9]:

1) Rods insertion due to gravity force in case of locking electromagnets demagnetized;
2) Hydroaccumulators;
3) Reactor vessel cooldown system;
4) Containment cooling system and containment itself.

2.6.2. Main active systems

Active systems are taking place, when electricity production is stable, and personnel launches the system or safety algorithm itself works out. Active systems are [9]:

1) Reactor shutdown with control and emergency rods, having used the drives;
2) Emergency cooldown through the SG;
3) Emergency cooldown through the purification system heat exchanger;
4) ECCS.

Active and passive systems are shown in the figure 2.4.

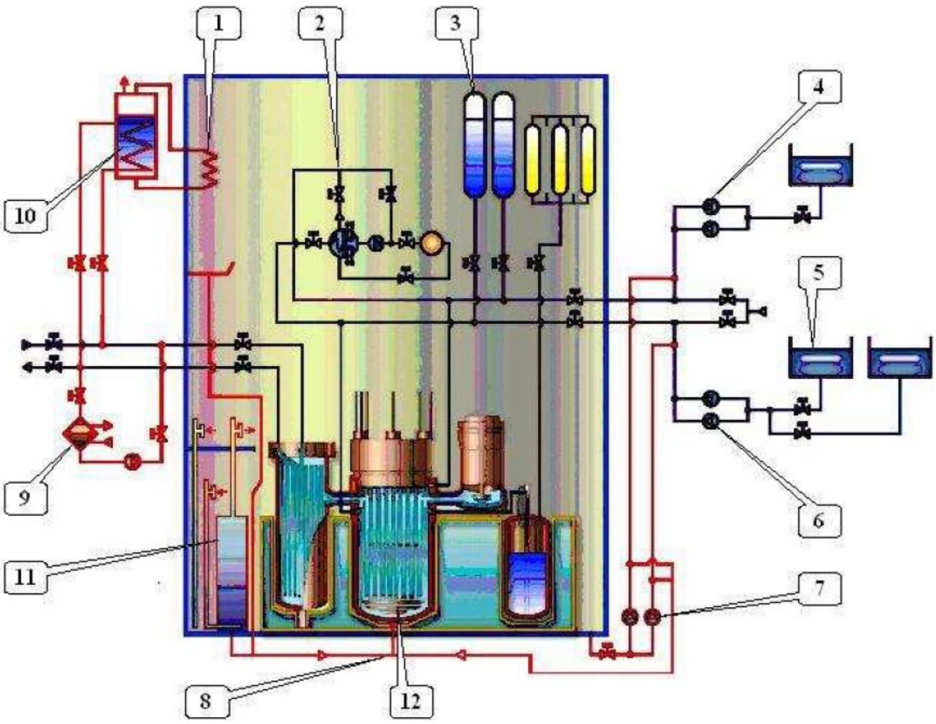


Figure 2.4. KLT-40S Containment [9].

1 – Containment cooling system; 2 – Purification system; 3 – ECCS accumulators; 4, 5, 6 – Active ECCS; 7 – Recirculation system; 8 – Reactor vessel cooling system; 9 – Active emergency heat removal system; 10 – Passive emergency heat removal system; 11 – Bubbling system; 12 – Reactor.

2.7. Economy goals

In chapter 2.1, there was a statement about the cancellation of project “Sturgis” – US FPP – and the statement pointed out that this very first and unique facility could not achieve a high profit, from an economic standpoint.

Table 2.3 shows the cost proposals for electricity, power and heat supply for early 2021. Firstly, costs for FPP are revealed, and next to it for Kalininskaya NPP, where 4 VVER-1000 are located. This NPP supplies the region of town Tver with 70% of overall electricity demand [17].

Table 2.3. Electricity and heat supply prices [19 and 20].

Power plant	Tariff	
	Rub	€
1. FPP “Akademik Lomonosov”:		
Electricity, MW·h	23881.71	259.02
Power, MW (monthly)	5587061.69	60597.20
Heating, Gcal	5962.40	64.67
Heating, MW·h	5126.74	55.61
2. Kalininskaya NPP:		
Electricity, MW·h	270.36	3.05
Power, MW (monthly)	343212.14	3588.82
Heating, Gcal	235.37	2.46
Heating, MW·h	202.38	2.12

Because there is a lack of data on investments and other expenses, it is difficult to know the capital expenses; therefore, the value will be assumed using information from open sources.

According to Bellona Foundation, it was estimated 6 billion rubles (€65 million) for building, but in the end of building and constructing process it became 37 billion rubles (€401 million).

For instance, the average electricity cost in Finland is €46 as of early 2021 [4]. It is clear that FPP electricity price is not even close to the Finnish one, but in scope of regular stationary Russian NPP price it is extremely large.

Figure 2.5 shows balancing energy costs in the early 2021.

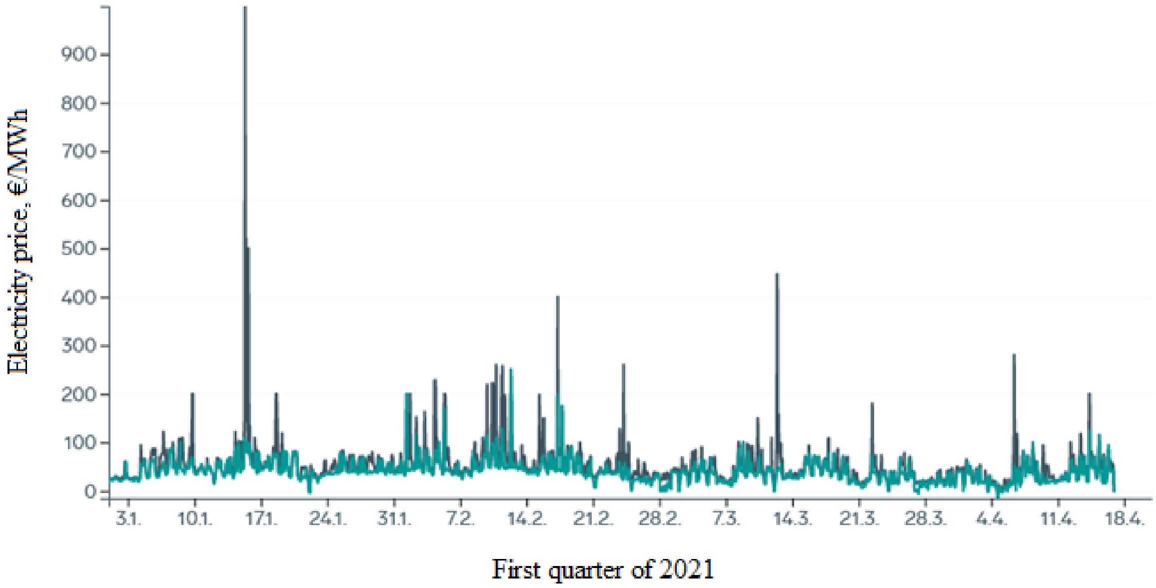


Figure 2.5. Balancing energy price in first quarter of 2021 in Finland [4].

Dark line – Upregulating price; Light line – Downregulating price.

2.8. Critics

Bellona community published in 2011 the report, in which FPP project was absolutely criticized. Main points were the safety issues of a nuclear plant, dislocated at the ship, and economy challenges that will appear due to specificity of the facility. [1]

General safety questions were asked to the KLT-40S reactor, while the report had been written, there weren't any exhaustive and detailed information about reactor's emergencies, accidents and some major and minor design concerns. [1]

Also, it was claimed that natural circulation flow rate equals to 3÷5% of nominal for this type of reactor construction. And if the emergency shutdown had taken the place, this 5% wouldn't have been enough to compensate ~7% of nominal thermal power decay heat. [1]

Another statement was referred to fuel enrichment value. At the time of writing that report, information about the enrichment was that its value is 18.5% (and in the IAEA report it was 14.1%, which is still a high value), as a consequence, it was mentioned that any reactor is a convertor to some extent, and KLT-40S is able to produce 60 kg of Plutonium at 80% of power in one year. [1]

These arguments will be discussed and negated further in the discussion chapter.

3. KLT-40S

This chapter contains details on the KLT-40S reactor. The Nuclear Steam Supply System (NSSS) and all of the main equipment, such as the reactor vessel, Steam Generator (SG), and Main Coolant Pump (MCP), will be discussed in this section. The flow of coolant inside the reactor will also be explained.

3.1. Nuclear Steam Supply System

3D model of the primary circuit main equipment showed at figure 3.1. It can be seen that NSSS composition is similar to that of a standard Pressurized Water Reactor (PWR). One reactor core, four SGs, four MCPs, four pressurizers.

Figure 3.1. Primary circuit model [8].

1 – Valve; 2 – Hot leg; 3 – Rod drives; 4 – MCP; 5 – SG; 6 – Reactor core;

7 – Pressurizer; 8 – Hydroaccumaltors.

3.1.1. Reactor vessel

The site dimensions impose limitations on all pipework and large heat exchangers, so they should be avoided. As a result, the pipework problem was solved by designing a 'pipe-in-pipe' construction. Figure 3.2 shows that the reactor has co-axial pipes of different radius, so the inner pipe directs the coolant into the steam generator (i.e. hot leg), and the outer pipe returns the flow back to the reactor core (i.e. cold leg).

Reactor head and vessel are made of heat-resistant high-strength pearlitic steel with anticorrosive surfacing. KLT-40S contains three actuators of emergency rods, and nine of control rods [8].

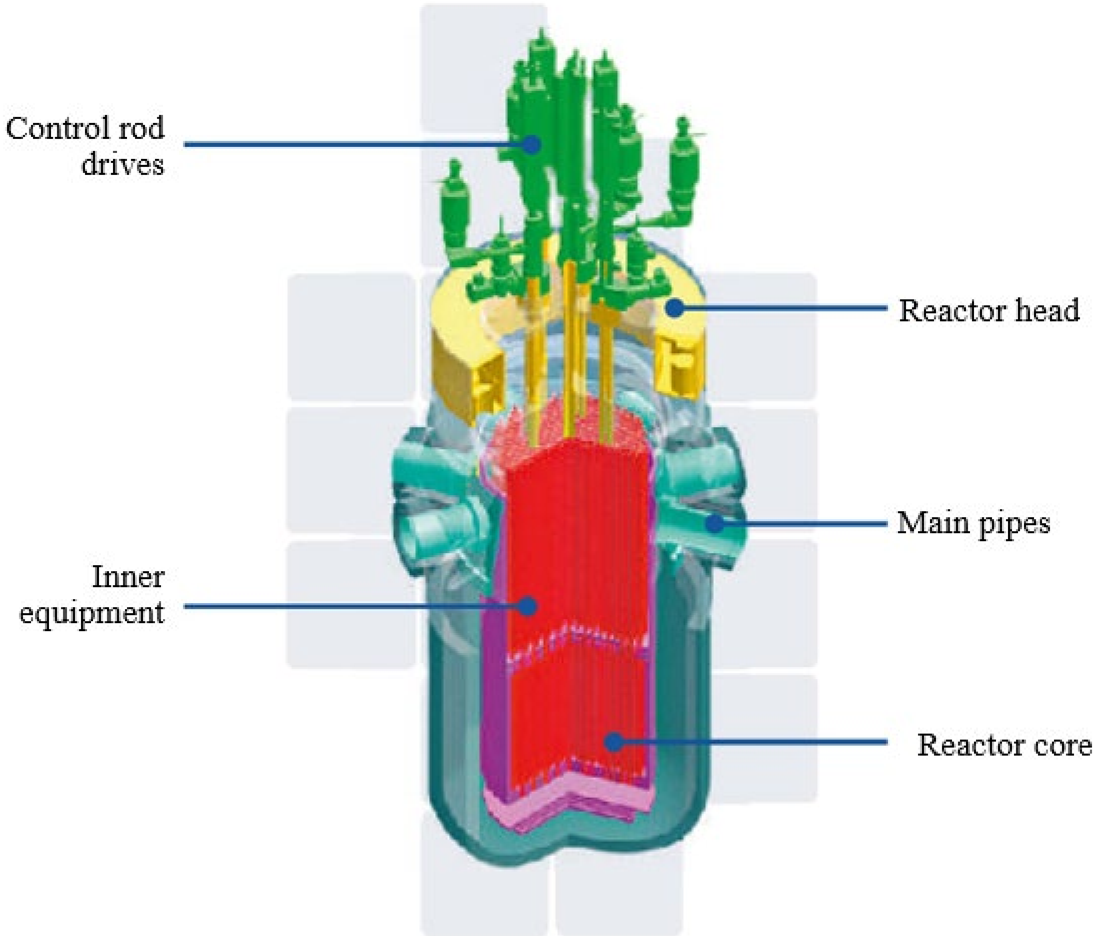


Figure 3.2. Reactor vessel 3D section [8].

3.1.2. Steam generator

PG-28 is a vertical once-through steam generator with coiled tubing system of a secondary circuit. Tubing system material contains titanium alloy, and SG vessel is made of low-doped steel with anticorrosive surfacing [8]. Figure 3.3 shows tubing system condition after a long-term operation. Also, remarkably, the size of SG relatively to a size of a man, it's quite compact.

Figure 3.3. PG-28 (after 140,000 hours of operating in 25 years) [14].

Table 3.1 holds main operating parameters.

Table 3.1. PG-28 SG main parameters [8].

Name	Value
Steam pressure, MPa	3.82
Superheat, °C	42
Steam temperature, °C	290
Feedwater temperature, °C	170
Coolant pressure, MPa	12.7
Coolant mass flow, tn/h	680

3.1.3. Main coolant pump

MCP is a sealed centrifugal one-staged powered by two-speed asynchronous drive. Pump vessel constructions are made of austenite stainless steel, rotor – of ferritic stainless steel. Lubrication and cooling are performed with a coolant of primary circuit in an autonomous circuit, which is cooled by other cooling water [8]. Table 1.3 shows main MCP parameters.

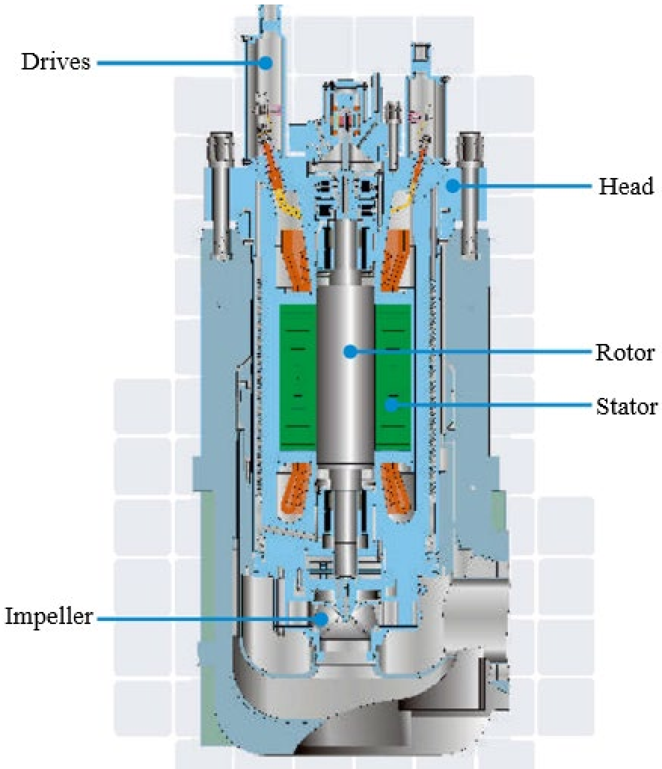


Figure 3.4. Main coolant pump section [8].

Table 3.2. MCP operating parameters [8].

Name	Value
Head, MPa	0.38
Mass flow, m^3/h	870

3.2. Coolant flow

Figure 3.5 represents the section of KLT-40S reactor prototype. The direction of the coolant flow is indicated by arrows.

This type of composition provides compactness of primary circuit, its placing into containment and also maintainability of main equipment and enough flow for natural circulation in emergency cases.

So, constructively this unit consists of interconnected high-pressure vessels with main equipment inside.

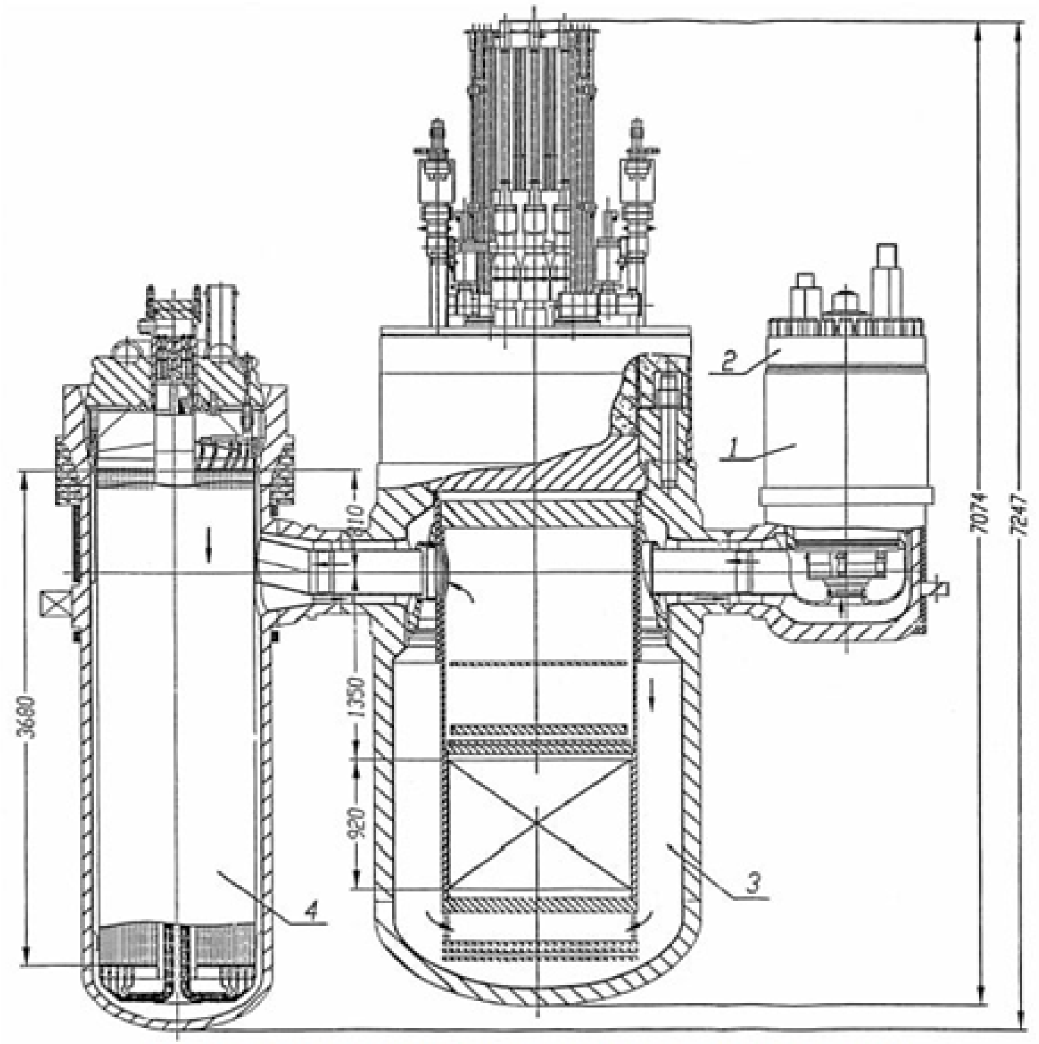


Figure 3.5. Reactor unit prototype section [24].

1 – Pump containment; 2 – Pump containment head; 3 – Reactor; 4 – SG.

The pressurized water flows through four inner connecting pipes to the enter chamber and afterwards to the reactor core, where it removes the heat from the fuel. Then again it goes through four inner connecting pipes to four steam generators, where heat is transferred to the second circuit. [24]

Afterwards, from each SG water flows back through a ring or annular channel between inner and external pipes to one of four ring or annular chambers, that are formed of cone-shaped shell and the vessel. [24]

The chambers are separated from each other by vertical baffles (or walls) and are sucking chambers for the pumps. From there coolant heads to the pumps, and circulation goes again. [24]

To have generators with disabled pumps in hot condition there is still some flow, that gets through special holes in separating baffles (or walls) between reactor chambers. [24]

3.3. Fuel cartogram

121 fuel assembly is distributed in a hexagonal lattice with average U-235 enrichment of 14.1%, and fuel elements (FE) in a triangular lattice. Fuel assemblies with higher enrichment are placed in the center of the core, whereas lower-enriched ones are on the periphery, to decrease neutron leakage. [9]

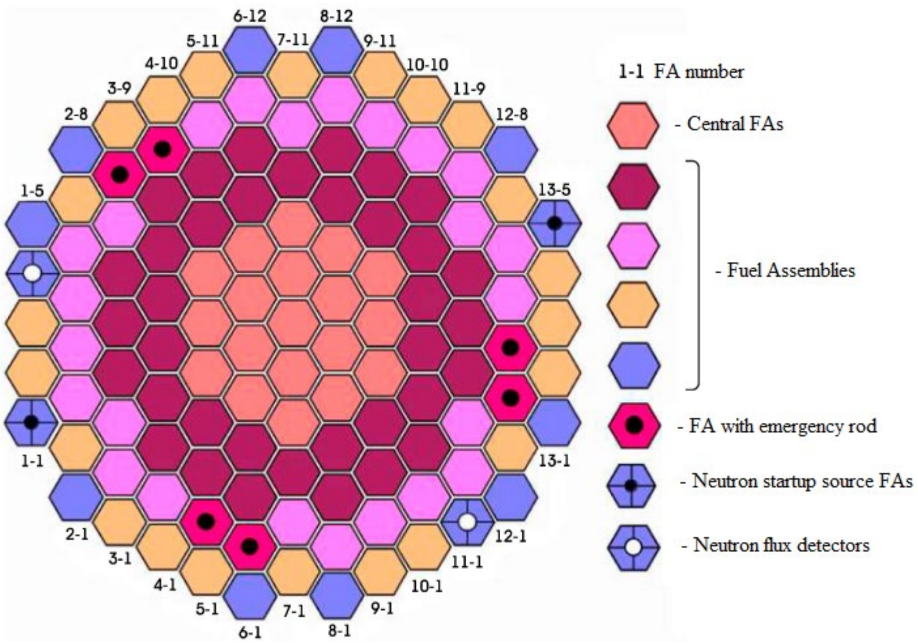


Figure 3.6. KLT-40S cartogram [8].

3.4. Technical data

In the table 3.3 actual physical parameters of the reactor operating at FPP "Akademik Lomonosov" can be found and they will be used as initial values for the ongoing calculations.

Table 3.3. KLT-40S main parameters [8 and 9].

Name	Value
Thermal power, MW	150
Number of fuel assemblies	121
FA across flats size, mm	98.5
Lattice pitch, mm	100
Core diameter, mm	1220
Core height, mm	1200
Fuel element width, Ø×δ, mm	6.8×0.5
Fuel cladding material	Zirc. alloy

4. RITM-200

For 2021, there is no much information about RITM-200 reactor. Nowadays, it is used at atomic icebreaker (LK-60Ya), but future FPPs will contain this reactor as a heat source. The main difference from KLT-40S is that steam generator is integrated into reactor's vessel [10].

4.1. Primary circuit

The integration of SGs into a single vessel with a reactor reduces the risk of LOCA-accidents. Constructively, the steam generator includes a straight-tube system through which feedwater flows. The vessel contains a total of four SGs. The reactor unit is made up of a vessel, a head, removable equipment (tubes, instalments), a core, control rods (12 pieces), and emergency rods (6 pcs.). [10]

The pressurized coolant flows into the reactor core via the ring clearance between the vessel and the reactor shell. After that, heated up coolant enters the SGs and flows counter-currently with the feedwater of secondary circuit to the sucking chamber above the cone-shaped shell.

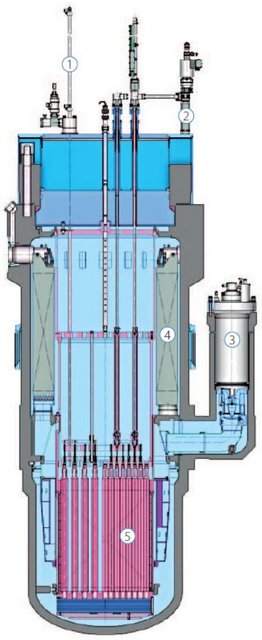


Figure 4.1. RITM-200 Section (left) and vessel (right) [10].

1 – Emergency rods drives; 2 – Control rods drives; 3 – MCP; 4 – Steam generator; 5 – Reactor core.

4.2. Reactor core

Reactor core fulfilled with 199 FAs dispersed in a hexagonal lattice, reactor core active height is 1200 mm, and diameter is 1650 mm [10].

4.3. Safety systems

Nuclear safety principle is defense-in-depth. Figure 4.2 represents main active and passive systems, where RHRS stands for Residual Heat Removal System, and SIS – Safety Injection System.

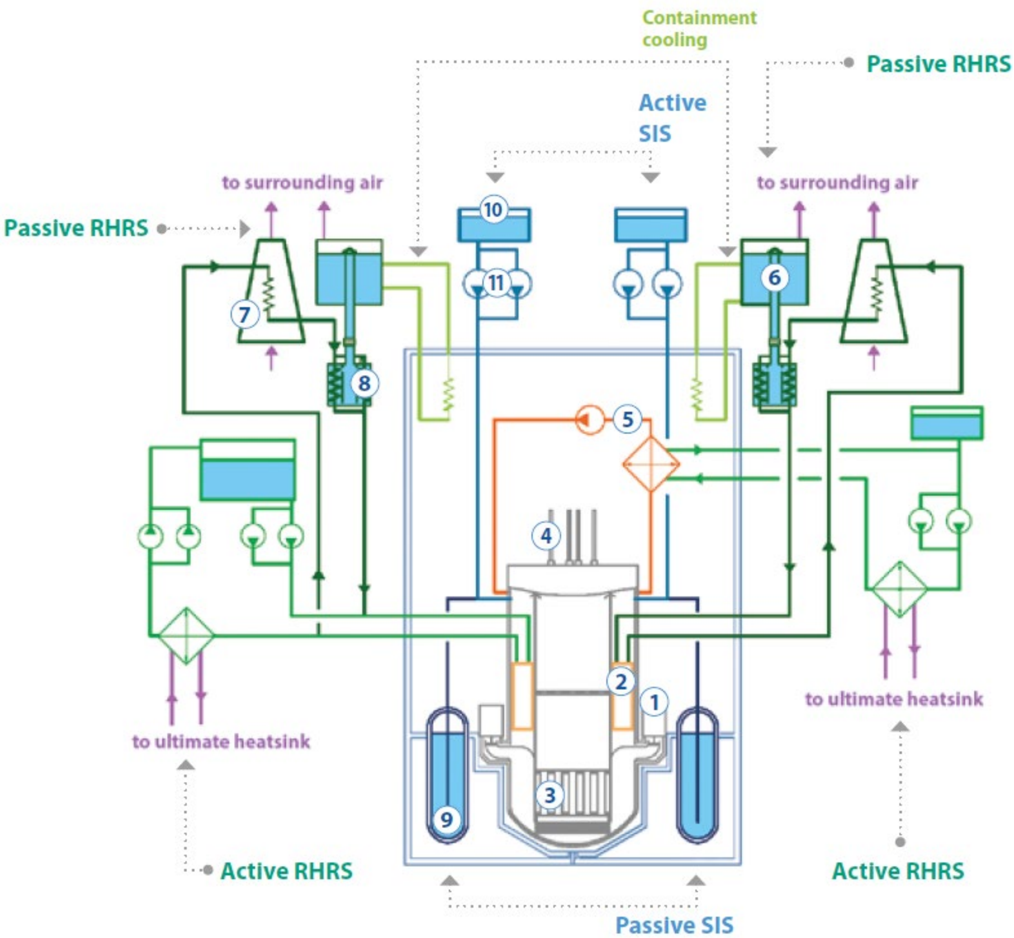


Figure 4.2. RITM-200 safety systems [18].

1 – MCP; 2 – SG; 3 – Reactor core; 4 – Control rod drive; 5 – Coolant purification circuit; 6 – Water tank; 7 – Air-to-water HX; 8 – Water HX; 9 – Hydroaccumulator; 10 – Water tank; 11 – ECCS pumps.

4.4. Technical data

Technical parameters of RITM-200 can be seen in table 4.1 and will be used as initials for further calculations.

Table 4.1. RITM-200 main characteristics [10 and 28].

Name	Value
1. Fuel management:	
Number of fuel assemblies, pcs.	199
Number of fuel rods, pcs.	13692
U-235 mass load, kg	438
Average fuel enrichment, %	19
2. Reactor core:	
Core diameter, mm	1600
Core height, mm	1200
Thermal power, MW	175
Coolant pressure, MPa	15.7
Cold leg temperature, °C	277
Hot leg temperature, °C	313
Coolant mass flow, tn/h	3250
3. Steam Generator:	
Steam pressure, MPa	3.82
Steam temperature, °C	295
Steam generation, tn/h	62

5. METHODOLOGY

In this chapter will be observed the methodology of calculating thermal hydraulics, physics and economics of the reactors.

5.1. Thermal hydraulics

Values from tables 3.1, 3.2, and 3.3 will be used as starting points for KLT-40S calculations, and values from table 4.1 will be used for RITM-200 calculations.

Firstly, fuel and coolant temperatures will be evaluated, afterwards critical heat flux, and in the end natural circulation rate.

5.1.1. Temperature distribution

Here the temperatures of interest will be evaluated. These temperatures are:

1) Coolant temperature;
2) Fuel cladding temperature;
3) Fuel rod temperature;
4) Center line temperature.

Geometry calculation

Firstly, volume of the reactor core should be found. Reactor core is assumed to be cylindrical shaped; thus, it has two significant dimensions – diameter and height:

$$V = \pi \cdot \frac{D^2}{4} \cdot H \tag{5.1}$$

Fuel assembly has a right hexagon profile; thus, the surface of the FA will be:

$$S_{\text{fa}} = \frac{3 \cdot \sqrt{3}}{2} \cdot \left(\frac{L}{\sqrt{3}}\right)^2 \tag{5.2}$$

Hydraulic diameter is a significant parameter, used for finding Reynolds number. Formula (5.3) shows the D_{h} for triangular flow area [6]:

$$D_{\text{h}} = \frac{2 \cdot \sqrt{3} \cdot p^2}{\pi \cdot d_{\text{fe}}} - d_{\text{fe}} \tag{5.3}$$

Flow area in one FA – a difference between the surface of a hexagon and the surface occupied by fuel rods and control rod:

$$F_{\mathrm{fa}} = S_{\mathrm{fa}} - \left[n \cdot \frac{\pi \cdot d_{\mathrm{fe}}^2}{4} + \frac{\pi \cdot d_{\mathrm{rod}}^2}{4} \right] \tag{5.4}$$

Extrapolated dimensions are needed to find form factor value, as it can tell about unevenness in power distribution (extrapolated distance assumed 31 mm for both reactors [12]):

$$H_{\mathrm{xtr}} = H + 2 \cdot \delta \tag{5.5}$$

$$R_{\mathrm{xtr}} = R + \delta \tag{5.6}$$

Form factors, axial, radial and full, respectively:

$$k_{\mathrm{z}} = \frac{\frac{\pi}{2} \cdot \frac{H}{H_{\mathrm{xtr}}}}{\sin\left(\frac{\pi}{2} \cdot \frac{H}{H_{\mathrm{xtr}}}\right)} \tag{5.7}$$

$$k_{\mathrm{r}} = 1.2$$

$$k_{\mathrm{v}} = k_{\mathrm{r}} \cdot k_{\mathrm{z}} \tag{5.8}$$

Power distribution

Linear power and heat flux are of interest. These two characteristics has an impact on overall safety. Center line temperature of a fuel rod is fully depending on a linear power, whereas inadequately large heat flux can cause drying out of flow area, a situation that is not much of a desire.

Firstly, absolute power values for assembly and fuel rod should be found.

Average power of one FA:

$$Q_{\mathrm{FA_{ave}}} = \frac{Q}{N} \tag{5.9}$$

Full power of one FA:

$$Q_{\mathrm{FA_{max}}} = Q_{\mathrm{FA_{ave}}} \cdot k_{\mathrm{r}} \tag{5.10}$$

The same for one fuel element:

$$Q_{\mathrm{FE_{ave}}} = \frac{Q_{\mathrm{FA_{ave}}}}{n} \tag{5.11}$$

$$Q_{\mathrm{FE}_{\mathrm{max}}} = \frac{Q_{\mathrm{FA}_{\mathrm{max}}}}{n} \tag{5.12}$$

Linear power, average and maximum, respectively:

$$q'_{\mathrm{ave}} = \frac{Q_{\mathrm{FE}_{\mathrm{ave}}}}{H} \tag{5.13}$$

$$q'_{\mathrm{max}} = \frac{Q_{\mathrm{FE}_{\mathrm{max}}}}{H} \tag{5.14}$$

Heat flux is an amount of heat that flows through a surface, so it can be found in a similar way with linear power:

$$q''_{\mathrm{ave}} = \frac{Q_{\mathrm{FE}_{\mathrm{ave}}}}{\pi \cdot d_{\mathrm{fe}} \cdot H} \tag{5.15}$$

$$q''_{\mathrm{max}} = \frac{Q_{\mathrm{FE}_{\mathrm{max}}}}{\pi \cdot d_{\mathrm{fe}} \cdot H} \tag{5.16}$$

With form factor effect maximum values can be obtained for medium and fully loaded rods:

$$q'_{\mathrm{ave}} = q'_{\mathrm{ave}} \cdot k_{\mathrm{z}} \tag{5.17}$$

$$q'_{\mathrm{max}} = q'_{\mathrm{max}} \cdot k_{\mathrm{z}} \tag{5.18}$$

$$q''_{\mathrm{ave}} = q''_{\mathrm{ave}} \cdot k_{\mathrm{z}} \tag{5.19}$$

$$q''_{\mathrm{max}} = q''_{\mathrm{max}} \cdot k_{\mathrm{z}} \tag{5.20}$$

Introducing the flux distribution φ in axial direction helps getting the distribution of linear power and heat flux over a fuel rod length, and diagrams of power distribution can be received, finally.

$$\varphi = \cos\left(\frac{\pi}{H_{\mathrm{xtr}}} \cdot \left(z_i - \frac{H}{2}\right)\right) \tag{5.21}$$

Coolant

Temperature of coolant should be known in the sense of safety (and effectiveness). Heat removal should be performed constantly and properly, as of it both primary and secondary circuits are dependent of.

Coolant mass flow, interacting with one FE:

$$q_{\mathrm{m_{fe}}} = \frac{Q_{\mathrm{FE_{ave}}}}{h_{\mathrm{out}} - h_{\mathrm{in}}} \tag{5.22}$$

Where $h = f(p, t)$ – enthalpy defined, using WaterSteamPro software.

Obviously, mass flow for a whole FA:

$$q_{\mathrm{m_{fa}}} = q_{\mathrm{m_{fe}}} \cdot n \tag{5.23}$$

Where n is amount of fuel rods.

To find the change of temperature of coolant, specific heat is needed. Specific heat strongly depends on a temperature $\mathrm{c_p} = f(T)$, it will be changing along the fuel rod. But for the first iteration, average value can be used.

$$\mathrm{c_{p\_mean}} = f\left(T_{\mathrm{p\_mean}}\right)$$

$$T_{\mathrm{p\_mean}} = \frac{T_{\mathrm{in}} + T_{\mathrm{out}}}{2} \tag{5.24}$$

Now, temperature distribution for medium and fully loaded fuel rod can be found, integrating over the rod height:

$$T = T_{\mathrm{in}} + \frac{1}{q_{\mathrm{m_{fe}}} \cdot \mathrm{c_p}} \cdot \int_0^z q' \, dz \tag{5.25}$$

After simplifications, equations will be [6]:

$$T_{\mathrm{w\_ave}} = \ T_{\mathrm{in}} + \frac{q'_{\mathrm{ave}} \cdot H_{\mathrm{xtr}}}{q_{\mathrm{m_{fe}}} \cdot c_{\mathrm{p\_mean}} \cdot \pi} \cdot \left(\sin\left(\frac{\pi \cdot \mathrm{z}_i}{H_{\mathrm{xtr}}}\right) + \sin\left(\frac{\pi \cdot H}{2 \cdot H_{\mathrm{xtr}}}\right) \right) \tag{5.26}$$

$$T_{\mathrm{w\_max}} = \ T_{\mathrm{in}} + \frac{q'_{\mathrm{max}} \cdot H_{\mathrm{xtr}}}{q_{\mathrm{m_{fe}}} \cdot c_{\mathrm{p\_mean}} \cdot \pi} \cdot \left(\sin\left(\frac{\pi \cdot \mathrm{z}_i}{H_{\mathrm{xtr}}}\right) + \sin\left(\frac{\pi \cdot H}{2 \cdot H_{\mathrm{xtr}}}\right) \right) \tag{5.27}$$

For the second iteration, can be used just found temperatures and respective specific heat.

Fuel cladding

As was mentioned, center line temperature is an important parameter, but along with it a temperature of a fuel cladding is significant too. Fuel cladding material is made of zirconium alloy. Zirconium is resistant to gases at room temperatures, but at high temperatures, it readily interacts with oxygen, hydrogen and other gases [12]. Thus, cladding temperature should remain, or not exceed, critical one, to prevent cladding destroying.

Cladding temperature can be found using coolant one, from previous paragraph:

$$T_{\mathrm{cld}} = T_{\mathrm{w}} + \Delta T_4 \tag{5.28}$$

The nature of heat transfer from cladding to coolant is convective. The main parameter, explaining this transfer is α – heat transfer coefficient. It shows what amount of heat flows through a surface in one second [6]:

$$\Delta T_4 = \frac{q''}{\alpha} \tag{5.29}$$

This coefficient depends on Nusselt number:

$$\alpha = \frac{Nu \cdot \lambda}{D_{\mathrm{h}}} \tag{5.30}$$

In its turn, Nusselt number characterizes relation between convective heat transfer rate and conductive one. The formula below is used for next intervals: $Re = 5 \cdot 10^3 \div 5 \cdot 10^5$; $Pr = 0{,}7 \div 20$; $x = 1{,}1 \div 1{,}8$.

$$Nu = A \cdot Re^{0.8} \cdot Pr^{0.4} \tag{5.31}$$

Where $A = 0.0165 + 0.02 \cdot (1 - 0.91 \cdot x^{-2}) \cdot x^{0.15}$ – coefficient depended on fuel rod relative pitch.

Reynolds number explains whether the flow turbulent ($Re > 10{,}000$) or laminar ($Re < 2{,}300$). To have as intense heat transfer as possible, turbulent flow is needed. Thus, Reynolds number should be larger than 10,000.

$$Re = \frac{w \cdot D_h}{\nu} \tag{5.32}$$

$$w = \frac{q_{\mathrm{m_{fa}}}}{\rho \cdot F_{\mathrm{fa}}} \tag{5.33}$$

Prandtl number shows how physical character (kinematic viscosity) of the coolant affect the thermal diffusivity.

$$Pr = \frac{\nu}{a} \tag{5.34}$$

$$a = \frac{\lambda}{\rho \cdot c_{\mathrm{p\_mean}}} \tag{5.35}$$

After perceiving the cladding temperature, the center line temperature difference can be iterated.

But firstly, temperatures of inner cladding wall and of fuel rod itself should be found.

Heat transfer through the cladding is the heat conduction. It depends on the cladding material parameters: thermal resistance, heat conductivity, etc. Equation will be [6]:

$$\Delta T_3 = \frac{q'}{2 \cdot \pi \cdot \lambda_{\mathrm{cld}}} \cdot \ln\left(\frac{d_{\mathrm{oc}}}{d_{\mathrm{ic}}}\right) \tag{5.36}$$

Where heat conductivity is chosen $\lambda_{\mathrm{cld}} = 20 \frac{\mathrm{W}}{\mathrm{m \cdot K}}$ [12].

Inner cladding (or gas gap temperature from another side) wall temperature will be:

$$T_{\mathrm{gg}} = T_{\mathrm{w}} + \Delta T_4 + \Delta T_3 \tag{5.37}$$

Fuel rod

Fuel rod temperature is not similar with fuel inner cladding wall, due to a small gas gap (0,001 m) between them, and convection takes place at that small gap.

Temperature difference will be [6]:

$$\Delta T_2 = \frac{q''}{\alpha_{\mathrm{gg}}} \tag{5.38}$$

Where $\alpha_{\mathrm{gg}} = 10^4 \frac{\mathrm{W}}{\mathrm{m^2 \cdot K}}$ [12].

Fuel rod temperature, subsequently:

$$T_{\mathrm{fr}} = T_{\mathrm{w}} + \Delta T_4 + \Delta T_3 + \Delta T_2 \tag{5.39}$$

Fuel center line

Now, the center line temperature difference can be iterated:

$$\Delta T_1 = \frac{q'}{4 \cdot \pi \cdot \lambda_{\mathrm{fuel}}} \tag{5.40}$$

Heat conductivity of a fuel rod is extremely sensitive to temperature difference, so it will be approximated with an empirical formula [6]:

$$\lambda_{\text{fuel}} = \left(\frac{38.24}{402.4 + T} + 6.1256 \cdot 10^{-13} \cdot (T + 273)^2\right) \cdot 10^2 \ \left[\frac{\text{W}}{\text{m} \cdot \text{K}}\right] \tag{5.41}$$

To find the temperature of center line will be used method of iterating [24]:

1) Temperature of center line T_{CL} is equal to temperature of cladding T_{cld};
2) Heat conductivity λ_{fuel} is a function of temperature;
3) Iteration equation:

$$0 = (T_{\text{CL}} - T_{\text{cld}}) - \frac{q'}{4 \cdot \pi \cdot \lambda_{\text{fuel}}(t)} \tag{5.42}$$

4) Solution is achieved when equation equals zero.

Center line temperature will be [6]:

$$T_{\text{CL}} = T_{\text{w}} + \Delta T_4 + \Delta T_3 + \Delta T_2 + \Delta T_1 \tag{5.43}$$

Using all the significant temperatures, diagram of temperature distribution can be formed.

5.1.2. Critical Heat flux

This phenomenon that should be always avoided, as if the critical heat flux is obtained, then the wetted surface will start to dry out and heat conduction will be spoiled.

To estimate the critical value Bezrukov’s correlation will be used [2]:

$$q''_{\text{cr}} = 0.795 \cdot (1 - x)^{(0.105 \cdot p - 0.5)} \cdot (\rho w)^{(0.184 - 0.311 \cdot x)} \cdot (1 - 0.0185 \cdot p) \ \left[\frac{\text{MW}}{\text{m}^2}\right] \tag{5.44}$$

Where: p – pressure of coolant [MPa]; ρw – coolant mass flux [kg/(m^2·s)]; x – quality.

Quality can be obtained as a function of enthalpy in WaterSteamPro software.

$$x = \frac{h - h'}{h'' - h'} \tag{5.45}$$

Bezrukov correlation is applicable in following cases [2]:

- Quality $x = -0.07 \div 0.4$;
- Pressure $p = 7.5 \div 16.5$ [MPa];
- Mass flux $\rho w = 700 \div 3500$ [kg/(m^2·s)];
- Fuel bundle length $l = 1.7 \div 3.5$ [m];
- Rod diameter $r = 9$ [mm];
- Rod pitch to rod diameter ratio $p_{\mathrm{r}} = 1.35 \div 1.385$ [m];

5.1.3. Natural circulation

Natural circulation is reliable and effective method of decay heat removal in normal shutdowns, transients and in emergency cases.

To understand the natural circulation direction and heat sources/sinks, simplified models of coolant flow in each reactor are shown in figures 5.1 and 5.2.

To calculate the natural circulation mass flow, difference between thermal center of steam generator and of reactor core should be found.

Reactor core geometrical center can be assumed as the thermal center, because power distribution is symmetrical in axial direction.

To find the center of SG, t,Q-diagram will be plotted. Using this diagram, SG can be divided in three imaginary parts: first – where feedwater is heated up to saturation point, second – where boiling appears and steam quality rises; third – where saturated steam is superheated.

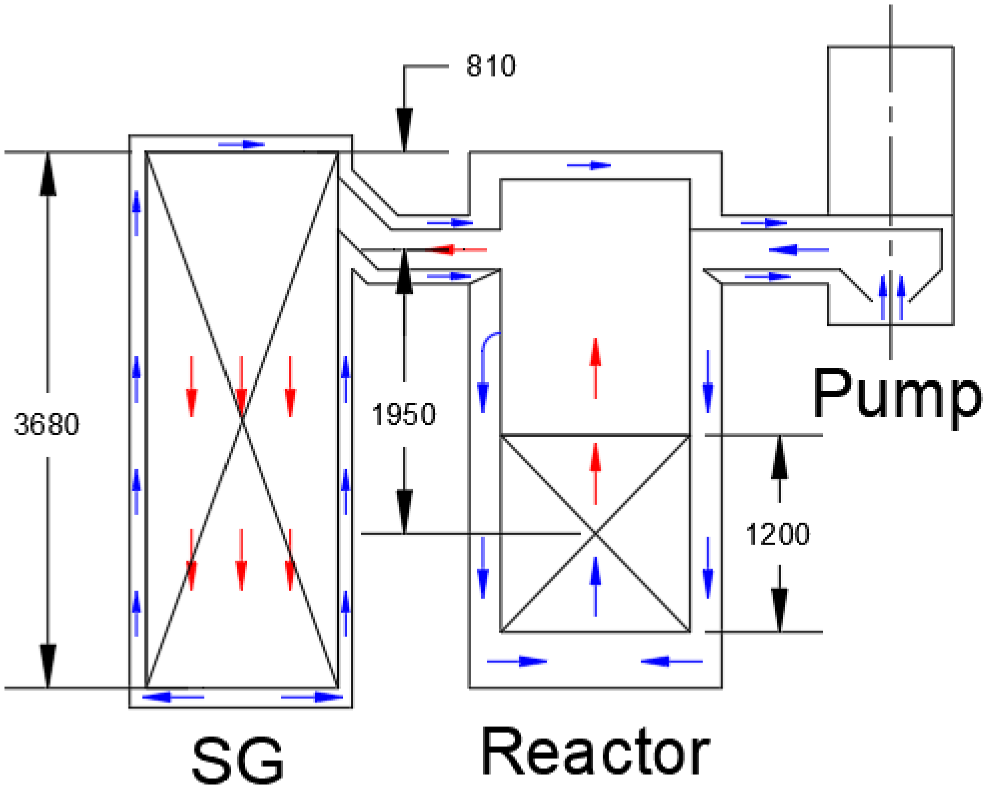


Figure 5.1. KLT-40S coolant flow model.

Afterwards, pressure drop in the loop should be found. It will be divided in pressure head, driven by the gravity, that appears because of density difference along the reactor core. Second component is pressure head performed by the circulating pump. Hence:

$$\Delta P_{\mathrm{l}} = \Delta P_{\mathrm{g}} + \Delta P_{\mathrm{p}} \tag{5.46}$$

Where gravity component is:

$$\Delta P_{\mathrm{g}} = (\rho_{\mathrm{cl}} - \rho_{\mathrm{hl}}) \cdot g \cdot \Delta H \tag{5.47}$$

Next, hydraulic resistance constant should be evaluated, using the pressure drop in the loop:

$$\Delta P_{\mathrm{l}} = \frac{1}{2} \cdot \frac{q_{\mathrm{m_l}}^{2-\mathrm{n}}}{\rho_{\mathrm{cl}}} \cdot R \tag{5.48}$$

Where $n = 0.2$ for high-turbulent flow.

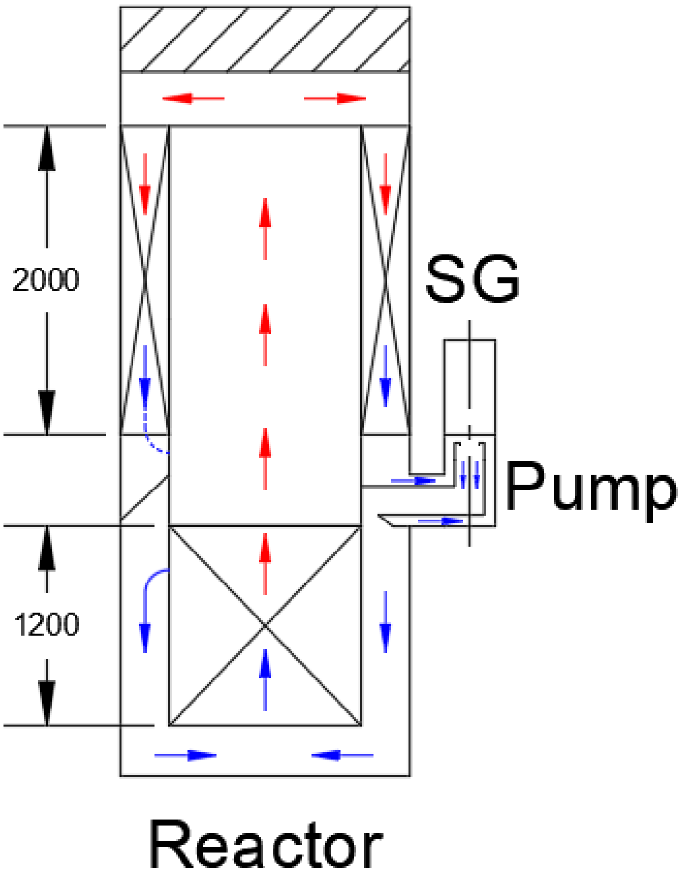


Figure 5.2. RITM-200 coolant flow model.

To find the mass flow when natural circulation takes place, it is assumed that pump component in eq. (5.46) is zero (pump is turned off), and hydraulic resistance constant remains the same.

$$q_{\mathrm{m_{nc}}}^{2-\mathrm{n}} = \frac{2 \cdot \beta \cdot \Delta T \cdot g \cdot \Delta H}{R} \cdot \rho_{\mathrm{cl}}^{2} \tag{5.49}$$

Where β is a thermal expansion coefficient:

$$\beta = -\frac{1}{\rho} \cdot \left(\frac{d\rho}{dT}\right) \left[\frac{1}{^{\circ}\mathrm{C}}\right] \tag{5.50}$$

To obtain a qualitative perspective, and due to a lack of dimensional information, a range of effective height differences between thermal centers will be assumed.

As initial guess for temperature difference between hot and cold leg nominal values from technical documentation will be used; it is equal 36 °C for both reactors.

Table 5.1. Initial parameters for natural circulation calculations.

	Name	Value
1.	KLT-40S:	
	Inlet temperature, °C	280
	Outlet temperature, °C	316
	Coolant pressure, MPa	12.7
	Pump head, MPa	0.38
	Mass flow through the SG, kg/s	188.9
	SG capacity, MW	37.5
	Effective height difference, m	0.2; 0.5; 0.9; 1.2
2.	RITM-200:	
	Inlet temperature, °C	277
	Outlet temperature, °C	313
	Coolant pressure, MPa	15.7
	Pump head, MPa	0.38
	Mass flow through the SG, kg/s	225.7
	SG capacity, MW	43.75
	Effective height difference, m	1.6; 1.8; 2.0; 2.2

5.2. Reactor physics

This chapter includes burnup and NEI evaluations, which are important factors in terms of fuel economy and safety.

5.2.1. Burnup

Fuel discharge burnup can reveal information about fuel economy and safety.

On the one hand, the larger the burnup, the more efficiently it is used, and thus fewer fuel bundles are required. Large discharge burnup, on the other hand, alters the isotopic composition of the fuel, making it radiotoxic, and both the fuel and the cladding structure are threatened with demolition throughout the cycle time. [13]

One way to find the fuel burnup is to divide the total energy produced in the cycle with the uranium mass. Thus:

$$B = \frac{E_{\mathrm{cycle}}}{m_{\mathrm{u\_tot}}} \tag{5.51}$$

Where $E_{cycle} = P_{th} \cdot t$, and m_{u_tot} is the total mass of natural uranium fueled into reactor core.

5.2.2. Neutron Economy Index

Another significant value is NEI. It shows the effectiveness of in-core fuel management.

$$NEI = \frac{m_{\mathrm{HM}}}{m_{\mathrm{u235\_tot}}} \tag{5.52}$$

To evaluate this index, total mass of heavy metals that is utilized in energy generation and total mass of U-235 [16]. The second term can be found from technical data of the reactor of interest.

To estimate m_{HM} the energy that is produced in the whole cycle will be used too. Further, the amount of fission reactions can be found with assumption, that 200 MeV is released per one fission.

$$N = \frac{E_{\mathrm{cycle}}}{e \cdot E_{\mathrm{f}}} \tag{5.53}$$

Subsequently, the amount of heavy metals can be found:

$$n = \frac{N}{N_{\mathrm{A}}} \tag{5.54}$$

Next, it should be assumed that average heavy metal atom molar mass is 237 g/mol.

$$m_{\mathrm{HM}} = n \cdot M \tag{5.55}$$

Thus, if NEI is equal to one, bred heavy metals are producing the same amount of energy with wasting U-235 fuel.

Table 5.2. Initial parameters for physics calculations.

	Name	Value
1.	KLT-40S:	
	U-235 mass load, kg	179
	Average fuel enrichment, %	14.1
	Thermal power, MW	150
	Campaign time, d	850
2.	RITM-200:	
	U-235 mass load, kg	438
	Average fuel enrichment, %	19
	Thermal power, MW	175
	Campaign time, d	850

5.3. Economy

Simplified calculation of economy parameters will be conducted in this chapter. The payback time of FPP and LCOE will be evaluated.

Prices will be converted with the exchange rate: 1€ = 92.2 rub (as of early 2021).

5.3.1. FPP and NPP payback time

Table 5.3. FPP and Kalininskaya NPP product prices and capital costs.

Name	Value
1. FPP:	
Capital costs, €	$401 \cdot 10^6$
Electricity price, €/(MW·h)	259.02
Heating, €/Gcal	64.67
Heating price, €/(MW·h)	55.61
2. Kalininskaya NPP:	
Capital costs, €	$748 \cdot 10^6$ [21]
Electricity price, €/(MW·h)	3.05
Heating, €/Gcal	2.46
Heating price, €/(MW·h)	2.12

Installed capacities and heat generation of the plants:

FPP:

Installed capacity – 70 MWe

Heating capacity – 146 Gcal/h

NPP:

Installed capacity – 1000 MWe

Heating capacity – 200 Gcal/h

Heating capacity for Kalininskaya NPP will be assumed to be 200 Gcal/h, due to the lack of information on NPP technical data.

Payback time will be evaluated as:

$$T = \frac{K_{\text{inv}}}{(C_{\text{el}} \cdot P_{\text{el}} + C_{\text{th}} \cdot P_{\text{th}}) \cdot 24 \cdot 365} \text{ [year]} \qquad (5.56)$$

5.3.2. LCOE value

Levelized Cost of Electricity is usually used for predicting the economic viability of a plant. It is the discounted average of the electricity price over the economic lifetime of the plant. To find this value equation (5.57) will be used [11]:

$$LCOE = \frac{K_{inv} + \sum_{t=1}^{n} \frac{M_{\mathrm{t}} + F_{\mathrm{t}}}{(1+r)^{\mathrm{t}}}}{\sum_{t=1}^{n} \frac{E_{\mathrm{t,el}}}{(1+r)^{\mathrm{t}}}} \left[\frac{€}{\mathrm{kW \cdot h}}\right] \tag{5.57}$$

Where:

1) K_{inv} – Capital expenditures, €;
2) F_{t} – Fuel expenses, $\frac{€}{\mathrm{year}}$;
3) M_{t} – Maintenance expenses, $\frac{€}{\mathrm{year}}$;
4) $E_{\mathrm{t,el}}$ – Annual electricity generation, kWh;
5) r – discount rate, $\frac{\%}{\mathrm{year}}$;
6) t – year;
7) n – lifetime of the project.

Table 5.4. Initial values for LCOE calculation.

Name	Value
Capital expenditures, €	$401 \cdot 10^6$
Fuel expenses, €/year	$7.78 \cdot 10^6$
Maintenance expenses, €/year	$1.56 \cdot 10^6$
Economic lifetime, year	20
Discount rate, %/year	3; 7; 10

Maintenance expenses are defined as 20% of fuel expenses, as fuel ones are usually much higher; fuel expenses are taken from [19] for half-year of 2021. And for discount rate a range of values is used, to receive results for upper and lower LCOE. Also, economic lifetime is assumed to be a half of the technical lifetime of the FPP [11].

6. RESULTS

This chapter includes the results of the above-mentioned calculations. Results of KLT-40S will be shown at the top of the sheet, and RITM-200 at the bottom, except where will be specified else.

6.1. Hydraulics

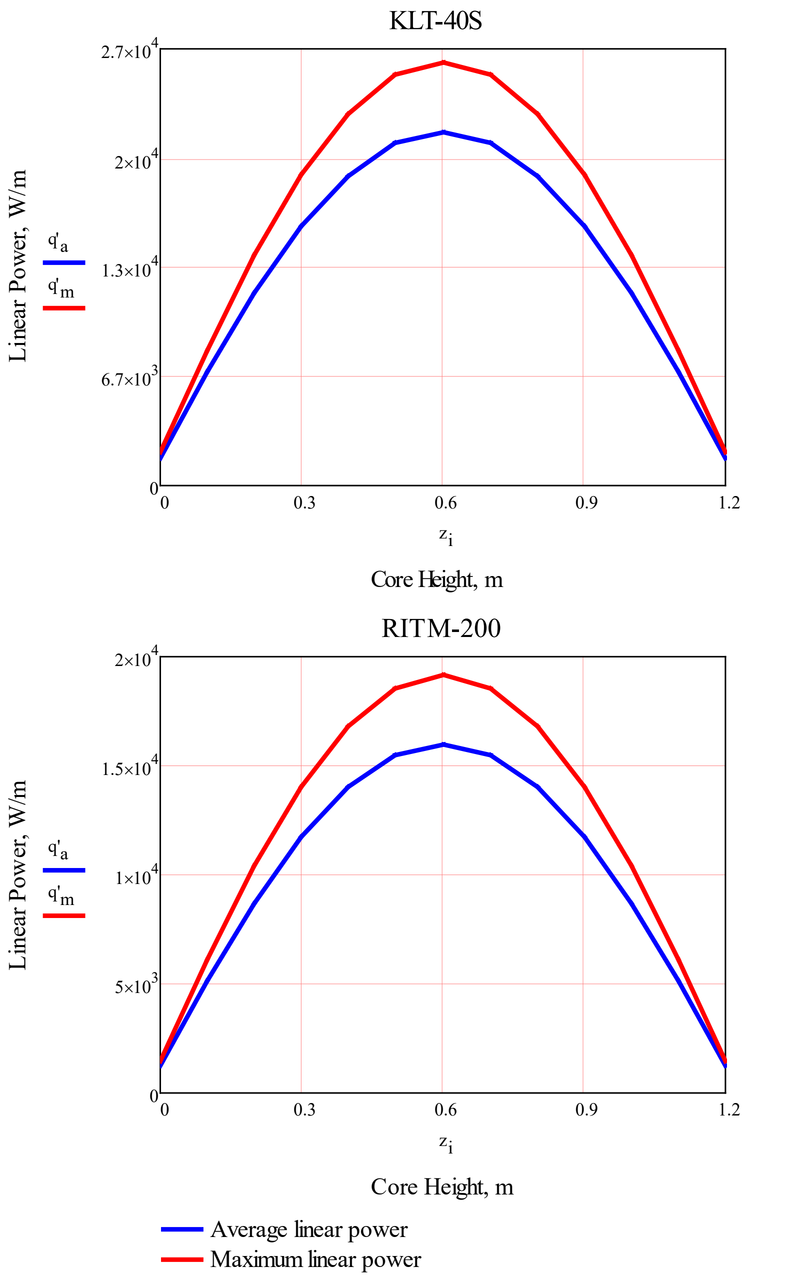


Figure 6.1. Linear power distribution for KLT-40S and RITM-200.

Note Y-axis scale!

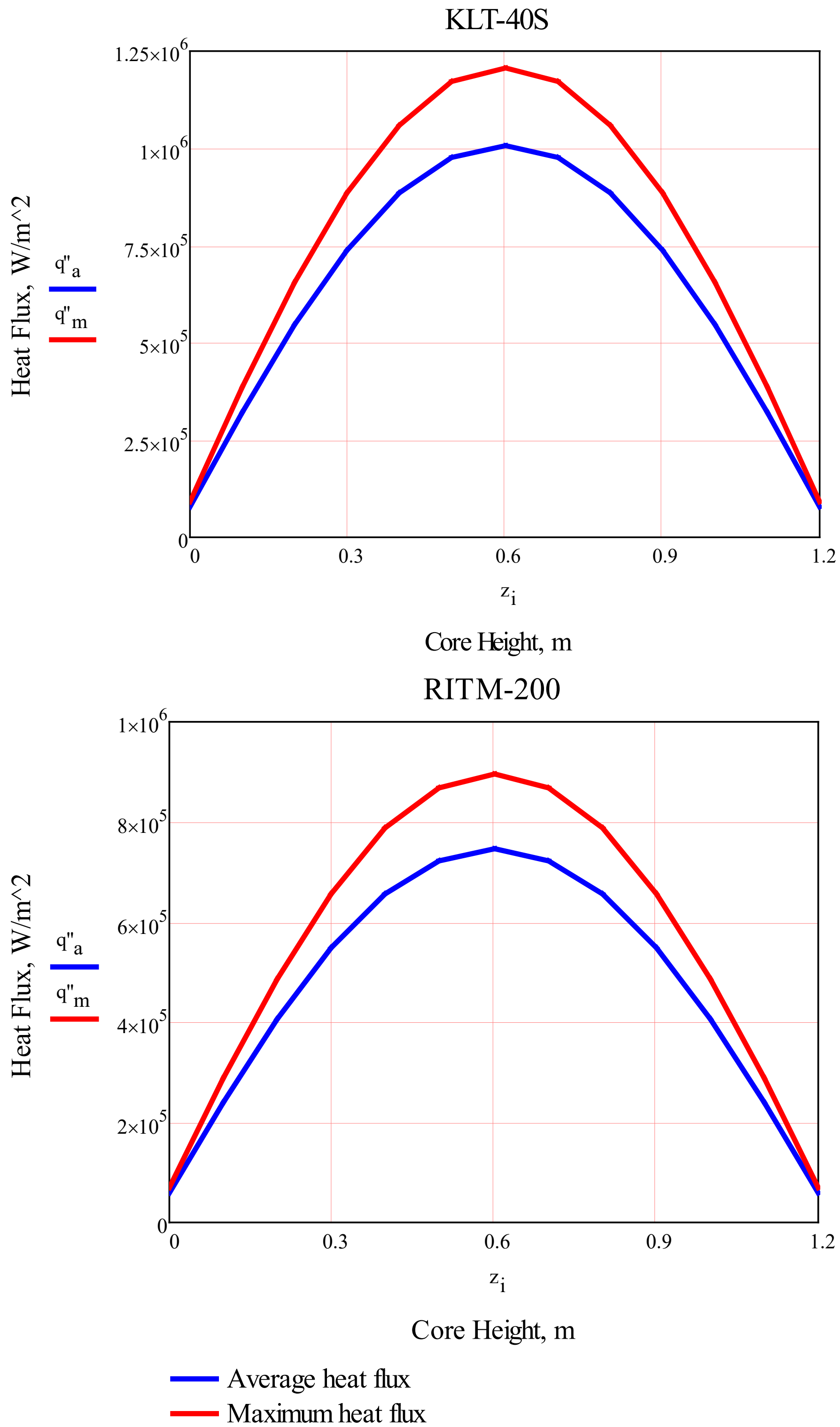


Figure 6.2. Heat flux distribution for KLT-40S and RITM-200.

Note Y-axis scale!

Figures 6.1 and 6.2 show the distribution of both average and maximum linear power and heat flux for KLT and RITM.

Difference in coolant main thermodynamic parameters can be seen from tables below.

Tables show the parameters that are correspond to maximum linear power distribution in a fuel rod, for several positions. They are of interest in scope of safety estimation.

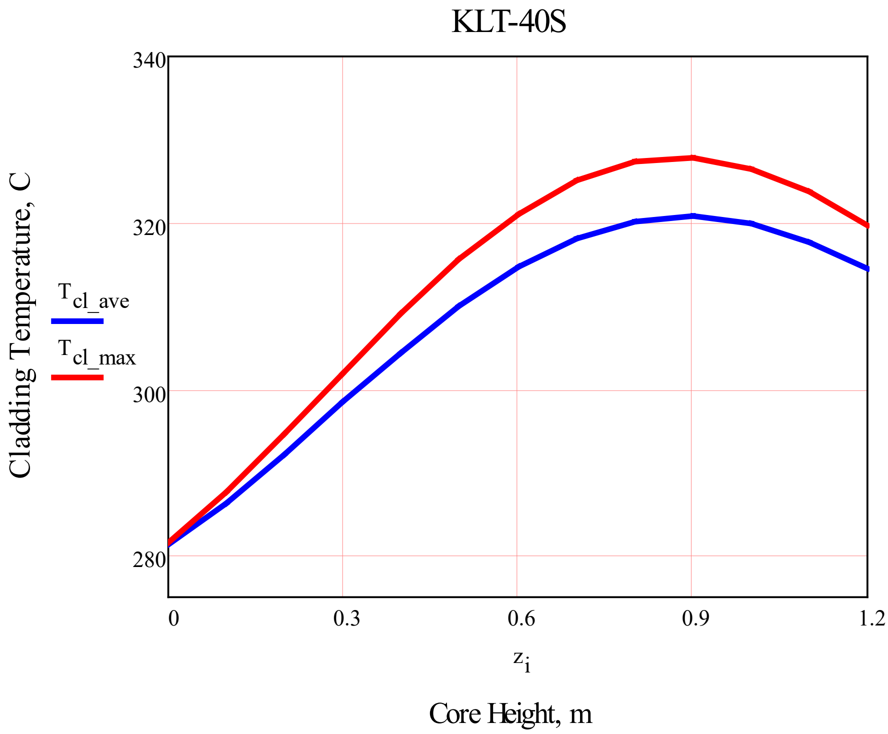
Table 6.1. Coolant thermodynamic parameters for $q'_{\max}$ (KLT-40S).

Position	Temperature, °C	Density, kg/m³	Velocity, m/s	Reynolds number, ×10⁵
1.	280.00	760.191	0.970	2.424
2.	281.16	758.099	0.973	2.439
3.	283.66	753.538	0.979	2.471
6.	291.67	728.373	1.013	2.623
7.	301.41	717.917	1.028	2.674
8.	306.02	707.530	1.043	2.719
11.	315.86	683.146	1.080	2.803
12.	317.46	678.853	1.087	2.815
13.	318.17	676.904	1.090	2.820

Table 6.2 Coolant thermodynamic parameters for $q'_{\max}$ (RITM-200).

Position	Temperature, °C	Density, kg/m³	Velocity, m/s	Reynolds number, ×10⁵
1.	277.00	769.719	0.701	1.733
2.	278.15	767.756	0.703	1.744
3.	280.63	763.476	0.707	1.768
6.	293.51	739.829	0.729	1.883
7.	298.49	729.969	0.739	1.923
8.	303.23	720.137	0.749	1.958
11.	313.58	696.820	0.774	2.026
12.	315.30	692.654	0.779	2.036
13.	316.08	690.751	0.781	2.040

Cladding temperature distribution is significant, due to characteristics of this material. As thermal tenses and loads affect the material lattice on the structural level.



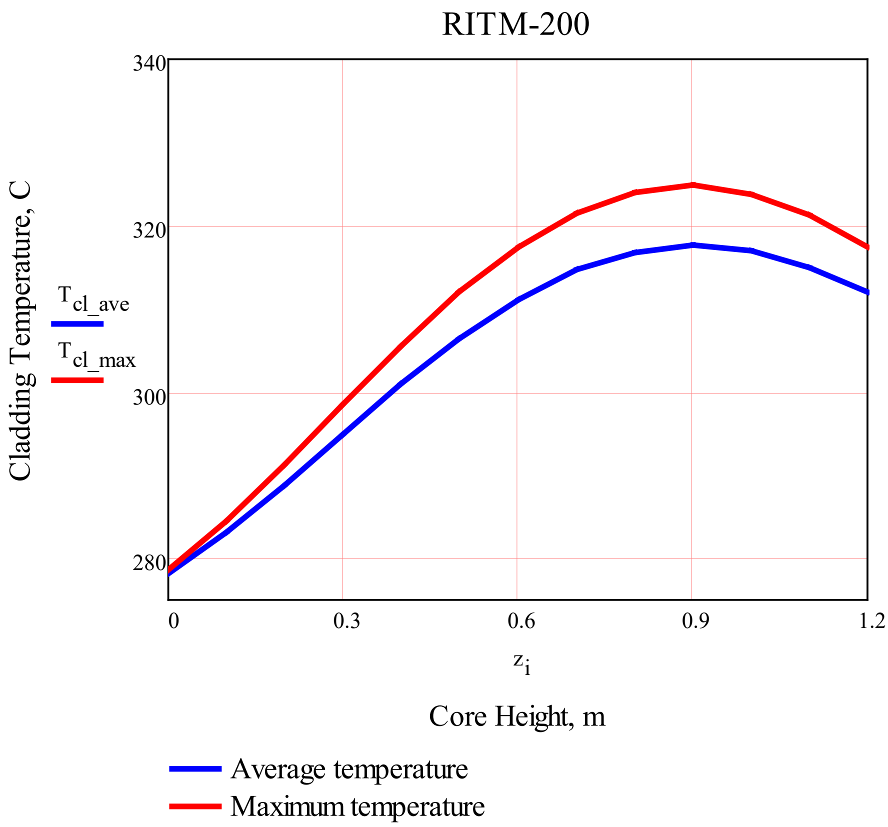


Figure 6.3. Cladding temperature distribution for KLT-40S and RITM-200.

Center line temperature distribution presented in figure 6.4.

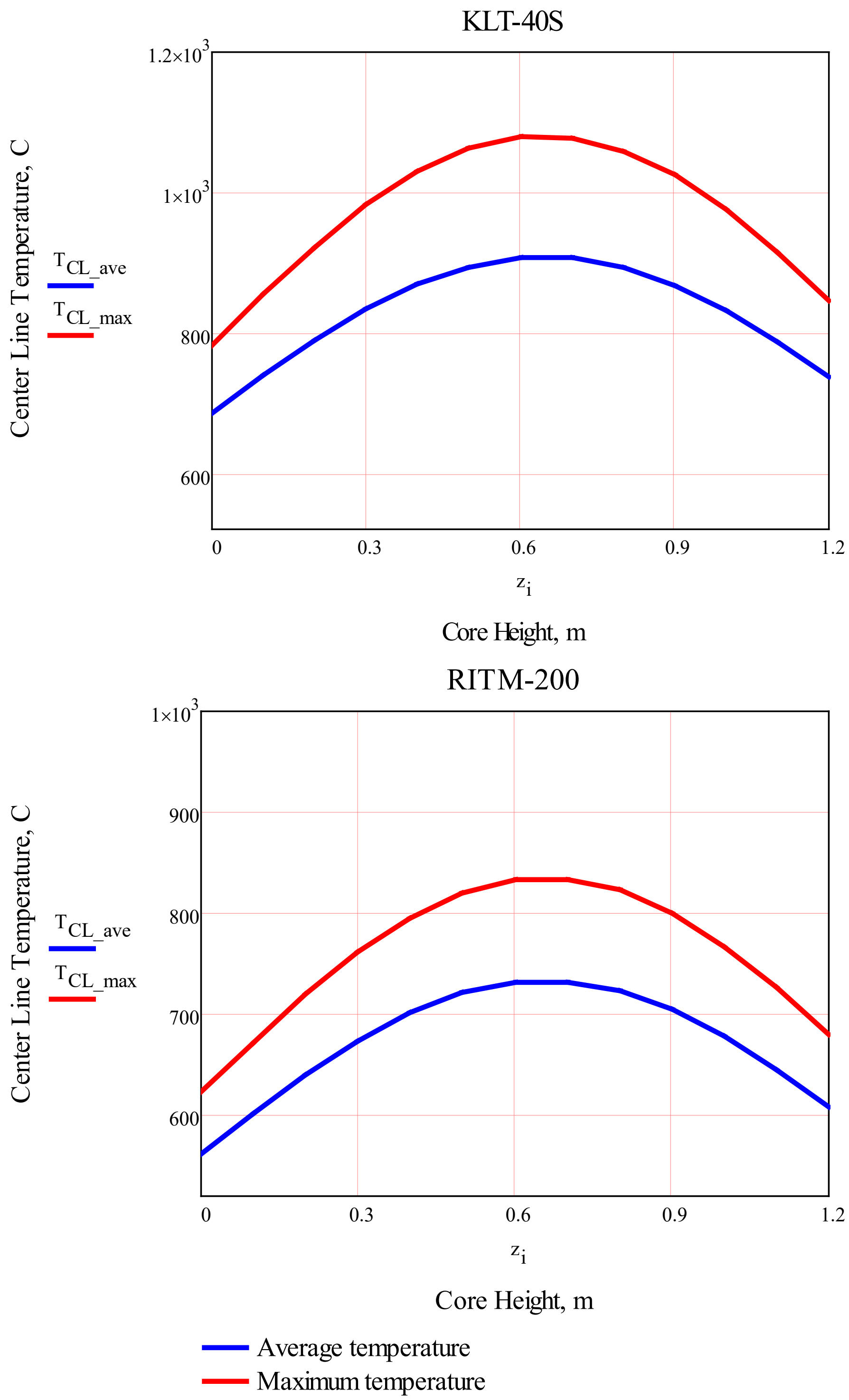


Figure 6.4. Center line temperature distribution for KLT-40S and RITM-200.

Note Y-axis scale!

Overall distribution in the maximum heat-loaded fuel rod is shown in figure 6.5.

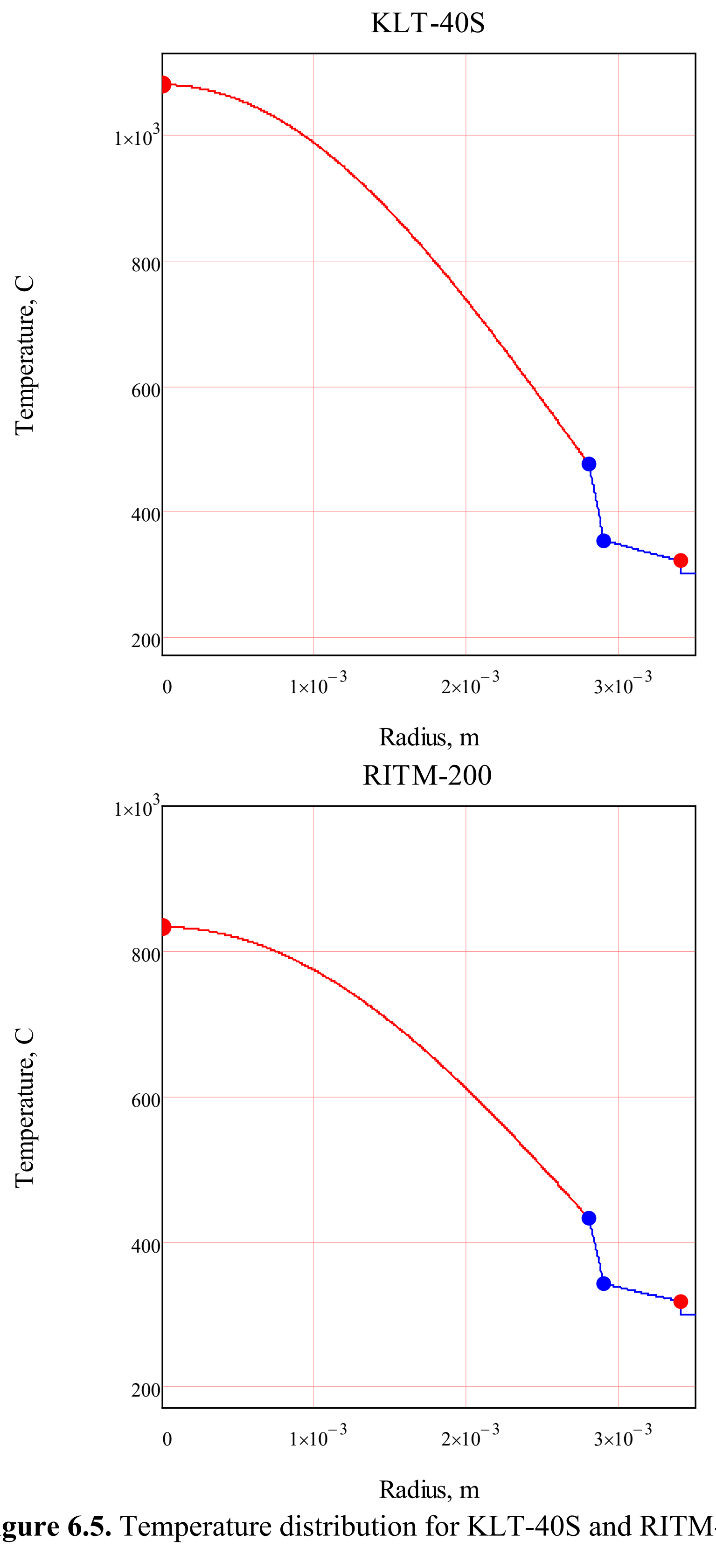


Figure 6.5. Temperature distribution for KLT-40S and RITM-200.

Note Y-axis scale!

Temperature from the center line (radius is equal to zero) distributes firstly in the fuel rod itself – parabolic line, after which a steep temperature drop can be observed in the tiny gas gap (width not more than 1 mm), due to Helium heat transfer parameters. Next line is a heat conduction through the fuel cladding.

In tables 6.3 and 6.4 column "Ratio" represents the ratio of critical heat flux and maximum heat flux, which is $q''_{\mathrm{max}} = 1.207\ \frac{\mathrm{MW}}{\mathrm{m}^2}$ for KLT and $q''_{\mathrm{max}} = 0.896\ \frac{\mathrm{MW}}{\mathrm{m}^2}$ for RITM.

Table 6.3. Critical heat flux for fuel rod (KLT-40S).

Position	Critical Heat Flux, MW/m^2	Ratio	Quality
1.	4.11	3.40	-0.25
2.	4.05	3.36	-0.24
3.	3.93	3.25	-0.23
6.	3.34	2.76	-0.17
7.	3.13	2.58	-0.15
8.	2.93	2.42	-0.13
11.	2.54	2.11	-0.08
12.	2.48	2.06	-0.07
13.	2.45	2.03	-0.06

Table 6.4. Critical heat flux for fuel rod (RITM-200).

Position	Critical Heat Flux, MW/m^2	Ratio	Quality
1.	2.27	2.54	-0.44
2.	2.26	2.53	-0.44
3.	2.24	2.50	-0.42
6.	2.11	2.36	-0.35
7.	2.06	2.30	-0.32
8.	2.01	2.25	-0.30
11.	1.90	2.13	-0.23
12.	1.89	2.10	-0.22
13.	1.88	2.09	-0.22

Natural circulation results are closing hydraulics part of the chapter. t,Q-diagrams are plotted to understand where the largest part of heat transfer takes place. And as it can be seen, boiling holds the most of the area. Hence, it can be assumed that thermal center of SG coincides with its geometrical one.

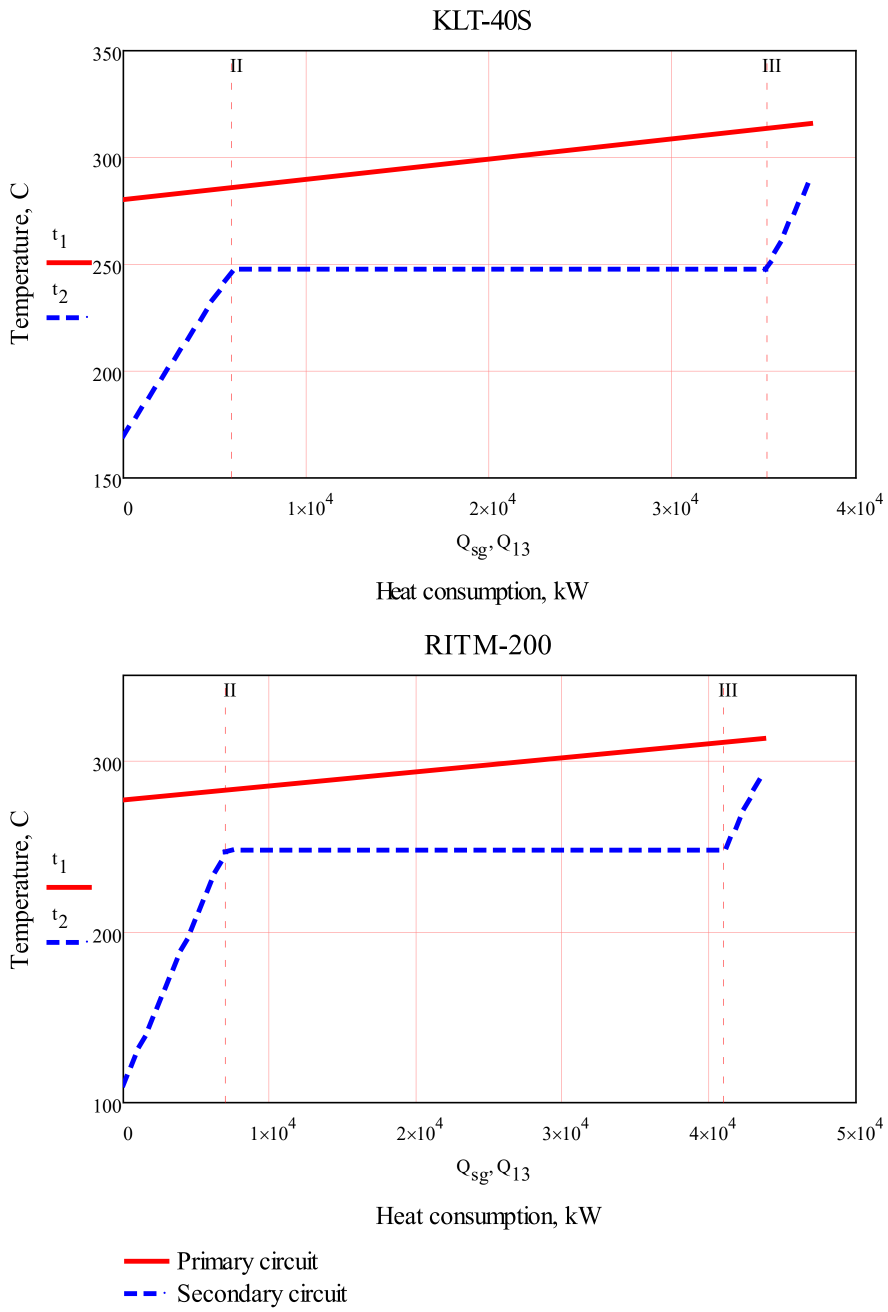


Figure 6.6. Heat transfer in SG for KLT-40S and RITM-200.

Note axis scale!

Numerical values of the heat fluxes in each heat-transfer "zone":

KLT-40S:

$$Q_{\mathrm{SG}} = 37500\ \mathrm{kW};$$

$$Q_{\mathrm{I}} = 5976\ \mathrm{kW};$$

$$Q_{\mathrm{II}} = 29203\ \mathrm{kW};$$

$$Q_{\mathrm{III}} = 2332\ \mathrm{kW}.$$

RITM-200:

$$Q_{\mathrm{SG}} = 43750\ \mathrm{kW};$$

$$Q_{\mathrm{I}} = 6961\ \mathrm{kW};$$

$$Q_{\mathrm{II}} = 34068\ \mathrm{kW};$$

$$Q_{\mathrm{III}} = 2721\ \mathrm{kW}.$$

Table 6.5 holds the results for both KLT-40S and RITM-200 about normal condition operation, and natural circulation one. Percentage of mass flow for each reactor presented below as a ratio of natural circulation flow rate to a normal one.

Table 6.5. Natural circulation results.

Effective height difference, m	**Buoyancy, Pa**	**Resistance factor, $\times 10^4$**	**Mass flow, kg/s**	**Ratio, %**
		KLT-40S		
0.2	151.86	4.621	2.5	1.3
0.5	379.65	4.623	4.2	2.2
0.9	683.37	4.627	5.8	3.1
1.2	911.15	4.630	6.8	3.6
		RITM-200		
1.6	1122.264	3.405	9.1	4
1.8	1262.547	3.406	9.7	4.3
2.0	1402.83	3.407	10.3	4.6
2.2	1543.113	3.408	10.9	4.8

6.2. Physics

Results for burnup and NEI are shown in table 6.6.

Table 6.6. Burnup and NEI.

Parameter	Value
KLT-40S	
U-235 mass, kg	179
Total fuel mass, kg	1273.0
Heavy metal mass, kg	135.3
Cycle lifetime, d	850
Cycle energy, MWd	127,500
Burnup, MWd/kgU	100.2
NEI	0.754
RITM-200	
U-235 mass, kg	438
Total fuel mass, kg	2305.3
Cycle lifetime, d	850
Heavy metal mass, kg	135.3
Cycle energy, MWd	148,750
Burnup, MWd/kgU	64.5
NEI	0.416

6.3. Economics

This part shows results of simplified evaluations of payback times for FPP, and also for Kalininskaya NPP to see the difference.

FPP: $$T = 1.7 \text{ years}$$

NPP: $$T = 24.8 \text{ years}$$

Also, inserting in the equation (5.56), prices for electricity and heat supply of Kalininskaya NPP will give the next result for the floating power plant:

$$T = 79.3 \text{ years}$$

Table 6.7 holds the results of LCOE evaluations according to different discount rates.

Table 6.7. LCOE results for FPP.

Discount rate, %/year	**LCOE**	
	€/kWh	**€/MWh**
3	0.145	145
7	0.189	189
10	0.226	226

The difference in prices can be seen in table 6.8.

Table 6.8. Comparison of LCOE with electricity prices (chapter 2.7).

Parameter	**Price, €/MW·h**
FPP Electricity price proposal	259.02
LCOE (for 7% discount rate)	**189**
Kalininskya NPP electricity price proposal	3.05
Finnish average electricity price	46

7. DISCUSSION

This chapter consists of discussion about achieved results and suggestions on future developments and improvements.

7.1. Reactors' parameters difference

Generally, these two reactors are designed to be used on atomic icebreakers and floating power units. The main difference is the composition of the reactor vessel, it is traditional for KLT-40S, and integrated for RITM-200. Also, RITM-200 has greater capacity, due to use of more fuel assemblies and larger fuel enrichment.

They both are to be used to provide remote areas with heat and electricity, so consequently the refueling periods should be decreased to minimum. Thus, it is established that one fuel campaign will be approximately 2.3 years.

Linear power is higher for KLT-40S on the average as well as power density ($q''' = 106.9\,\frac{\mathrm{MW}}{\mathrm{m}^3}$ for KLT and $q''' = 72.5\,\frac{\mathrm{MW}}{\mathrm{m}^3}$ for RITM) rather than for the second reactor. Maximum values for KLT and RITM, respectively are $q'_{\mathrm{KLT\_max}} = 257.9\,\frac{\mathrm{W}}{\mathrm{cm}}$ and $q'_{\mathrm{RITM\_max}} = 191.5\,\frac{\mathrm{W}}{\mathrm{cm}}$. It is less than in usual LWRs, but due to the safety concerns it can't be disadvantageous (typically, for LWRs $q'_{\mathrm{max}} \approx 400\,\frac{\mathrm{W}}{\mathrm{cm}}$ [6]).

Consequently, maximum heat fluxes will have the similar relation: $q''_{\mathrm{KLT\_max}} = 120.7\,\frac{\mathrm{W}}{\mathrm{cm}^2}$ and $q''_{\mathrm{RITM\_max}} = 89.6\,\frac{\mathrm{W}}{\mathrm{cm}^2}$ (for LWRs typical value for heat flux is $q''_{\mathrm{max}} \approx 135\,\frac{\mathrm{W}}{\mathrm{cm}^2}$ [6]).

Coolant temperature depends on secondary side parameters (temperature, pressure and quality of secondary steam for turbine) and coolant nature itself. As these are the PWRs, coolant must not be heated to a saturated condition and should be subcooled to some extent. Temperature distribution curves both for KLT and RITM seem to be correct as at the outlet average coolant temperature is a bit less than nominal.

Reynolds number has a large value, so it can be said that a strong turbulence takes place, which increases heat transfer efficiency.

Center line temperatures are not higher than 1200 °C, which gives a good margin before reaching the melting point temperature, which is between 2760 °C and 2865 °C, typically [12].

Critical heat flux values for both reactors look reliable. Minimum CHF ratios are slightly above 2, and even 3 at some positions. It should be disclaimed, that even if both reactor cores have triangular lattices, but some other parameters are not corresponding with the range of use of the Bezrukov correlation for finding the CHF. For instance, rod diameter and length are not matching the conditional ones. Thus, these values should be treated with the certain caution.

And CHF values for RITM-200 seem to be lower, rather than for KLT-40S. It can be explained with less mass flux, which is $\rho w_{\text{KLT}} = 737.7$ and $\rho w_{\text{RITM}} = 549.5$ kg/(m^2·s), thus less coolant for a fuel rod to transfer the heat to.

Also, quality values of RITM look too low, comparing with KLT ones. It can be explained with operating pressures. 12.7 MPa for KLT and 15.7 MPa for RITM, quite large difference. But hot leg temperatures are almost the same. Therefore, coolant in RITM is highly subcooled. From one point of view, it keeps the coolant away from boiling conditions, which is good for safety reasons. But from the other point, it seems that potential of the coolant is not fully developed, subsequently, it can affect the economic and thermal efficiency of the reactor.

Natural circulation is a strong and straight-forward mean of removing decay heat in both normal and abnormal transients, as well as emergency cases. Natural circulation flow rate has low values both for KLT and RITM (1.3% ÷ 3.6% and 4% ÷ 4.8%, respectively), but again due to the lack of technical documentation it is hard to get reliable and accurate values.

The Bellona community's statement that natural circulation flow rate of 3÷5%, is not enough to compensate the SCRAM of full-powered reactor appears to be invalid as natural circulation flow rate must not be 7% constantly. Even if the flow rate is insufficient to remove all the heat, the temperature of the coolant is highly dependent of the feedwater temperature. Therefore, the coolant temperature will rise up until some new equilibrium state is obtained.

In addition, there are different correlations for decay heat removal, which show that already after 1000 seconds (approx. 16 min) of shutdown there is only about 2 ÷ 3% of heat to be removed.

Figure 7.1 shows correlation for ratio of remained power to the full one after the shutdown.

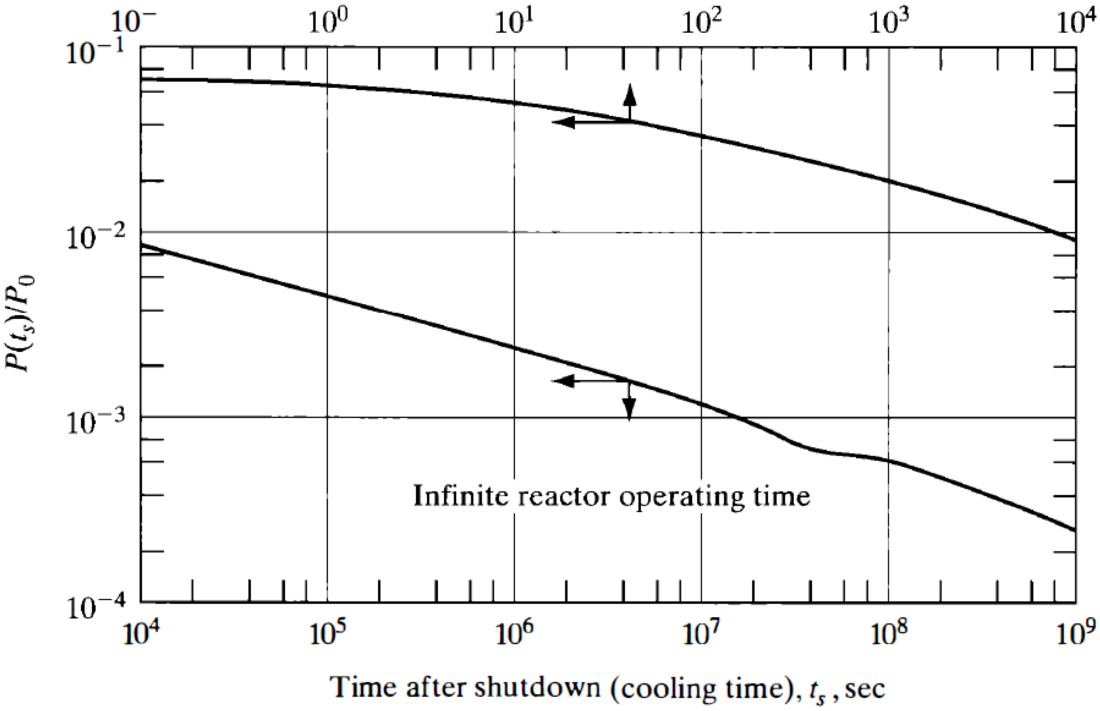


Figure 7.1. Decay heat removal correlations [12].

Furthermore, the statement about Plutonium generation appears to be incorrect, as this generation is a well-known nature of reactor, and proven technologies for tracking such generations are already in place. And in the latest IAEA report [9] it was stated that in KLT-40S Uranium is enriched to 14.1%, not to 18.5% as in Bellona report. As a result, it provides additional margin for exceeding the Low Enriched Uranium boundary – 20%.

NEI indexes are poor for both reactors. Obviously, refueling the whole reactor core affects the efficiency of fuel cycle. Where Uranium could be utilized more deeply, it is withdrawn. For example, the change of neutron economy index throughout the cycle at Olkiluoto 1 can be seen at figure 7.2.

It can be clearly seen that this value doesn't decrease less than one almost nowhere, whereas the index for KLT and RITM is 0.754 and 0.416, respectively. Yet, there have been assumptions made, such as average heavy metal mass generated, but nevertheless, values are quite small.

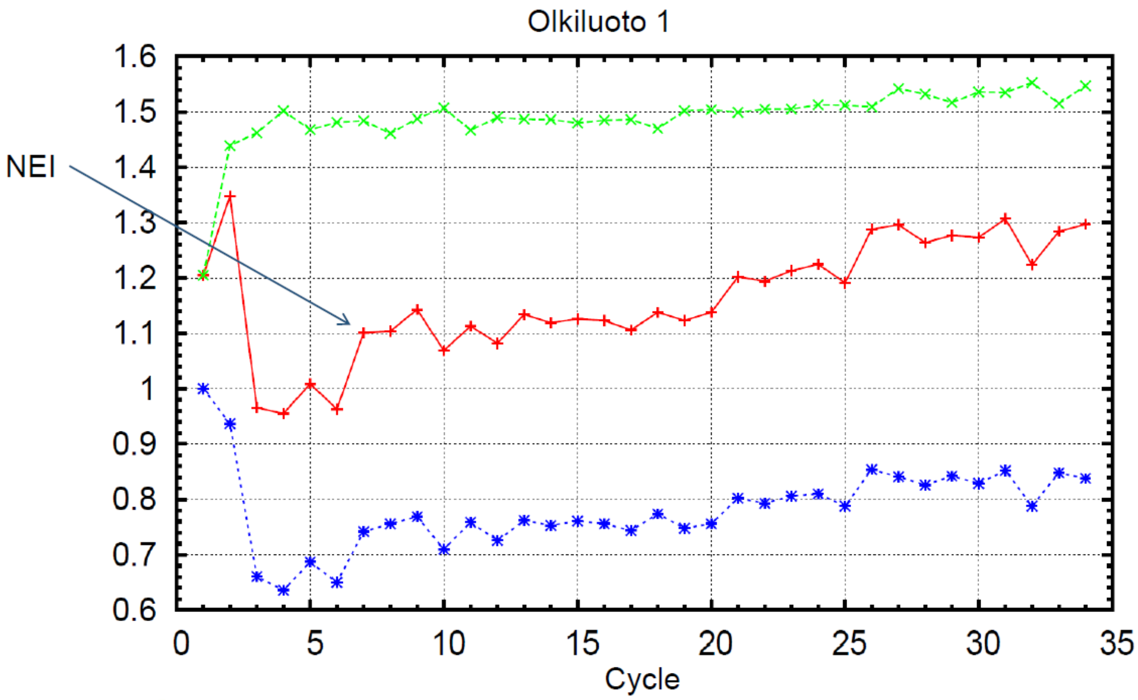


Figure 7.2. Neutron economy index at Olkiluoto 1 [16].

7.2. FPP parameters

Floating power plant is a complex and large facility. Wrong control and regulation can cause harmful circumstances. One must have requirements twice as strong for an ordinary power plant. So, while designing such project, it should be clear that safety comes first.

KLT-40S is a reactor that starts the way from soviet times, yet there is no full technical documentation on its parameters in the open share even today.

Results that were obtained from calculations in previous chapters have good safety characteristics. Therefore, one can say that using this type of reactor is acceptable. But even if it is a proven by time technology, there is still some probability of an accident to occur.

Economy goals look strange in scope of Russian average tariffs for electricity and heat supply. The project cost is way too large, to hold the product prices low. For example, the fourth unit of NPP Kalininskaya generates 1000 MWe, but the capital cost is equal to 748 million euros [20], whereas FPP cost equals to 401 million euros, with 70 MWe generated.

Hence, it is clear that FPP is an extremely expensive facility. Comparing the costs of one MWe for each power plant, the large difference can be seen:

FPP:

$$\frac{401 \cdot 10^6 \text{ €}}{70 \text{ MWe}} = 5.73 \frac{\text{€ million}}{\text{MWe}}$$

NPP:

$$\frac{748 \cdot 10^6 \text{ €}}{1000 \text{ MWe}} = 0.75 \frac{\text{€ million}}{\text{MWe}}$$

Also, LCOE value seem to be too high even for European costs. Table 7.1 contains LCOE for Finland [26] compared to FPP values with 3%, 7% and 10% discount rate. US dollars from the reference were converted to euros (1 $ = 0.82 €).

It is clear that levelized costs are not comparable to Finnish ones.

Table 7.1. LCOE comparison (€/MWh).

Country	**Discount rate, %/year**		
	3	**7**	**10**
Finland	37.8	63.6	89.4
Russia (FPP)	145	189	226

Therefore, it is hard to say whether such prices are applicable, in Russia at least.

Nevertheless, such a power plant will be used in areas where any other means of generating electricity are expensive as well.

7.3. Future suggestions

Overall, the project is advantageous, due to maneuverability of the FPP. It can be transported to any location, where enough water level can be found. Right now, FPP is anchored in Pevek and supplies this town with heat and electricity. Pevek is located in the far east of Russia. For example, distance between Moscow and Pevek is larger than 5,500 km and even Google Maps return error, when one's trying to get the route by car. However, if there'd been a coal or gas power plant, the fuel would have been delivered not from Moscow. The closest coal deposit is in Zyryansk region, and it even has infrastructure connection with Pevek.

Anyway, in some years the future of such project will be seen. The purposes of the FPP look decently, but the use of nuclear energy in regions with so harsh climate like the one, where FPP dislocates is questionable.

It could be a good step to prepare some research facilities in the areas of interest, firstly. After conducting all the necessary researches and retrieving the data and analyzing it, one might think of transferring such design to larger scale.

8. CONCLUSION

The floating power plant is a project that introduces new ways of delivering energy; in addition to producing power in areas where it is difficult to do so traditionally, it can also be used to supply water desalination plants, where the most vital resource on Earth can be produced.

The full disclosure of technical and safety values would have helped independent experts and curious students in analyzing this one-of-a-kind project, discovering new ways to improve effectiveness, and gaining trust in the design. Nonetheless, there is some information in open sources that was used in this dissertation, though it is not complete. Therefore, all results should be interpreted with caution.

On paper, both reactors appear to have good safety conditions:

KLT-40S:	**RITM-200:**
$q'_{\text{KLT\_max}} = 257.9\,\frac{\text{W}}{\text{cm}}$	$q'_{\text{RITM\_max}} = 191.5\,\frac{\text{W}}{\text{cm}}$
$q''_{\text{KLT\_CHF\_min}} = 2.45\,\frac{\text{MW}}{\text{m}^2}$	$q''_{\text{RITM\_CHF\_min}} = 1.88\,\frac{\text{MW}}{\text{m}^2}$
$q''_{\text{KLT\_max}} = 1.207\,\frac{\text{MW}}{\text{m}^2}$	$q''_{\text{RITM\_max}} = 0.896\,\frac{\text{MW}}{\text{m}^2}$
$T_{\text{KLT\_CL\_max}} = 1080\ ℃$	$T_{\text{RITM\_CL\_max}} = 835\ ℃$
$T_{\text{KLT\_cld\_max}} = 328\ ℃$	$T_{\text{RITM\_cld\_max}} = 325\ ℃$

This is a significant advantage, as it is critical in the sea to maintain constant control, provide high-level safety maintenance, and meet all requirements.

Economical values seem to be good only if either the prices are too high in scope of Russian economy or exporting takes place. Anyway, use of such expensive power is a controversial mean of utilizing nuclear energy, which is treated as a cheap one. Although, there are no any alternatives so far.

The main reason for utilizing FPP would be in areas with insufficient infrastructure and high electricity demand, and there are such locations, obviously. In particular, coastal or island states in the north and south (Latin America, Africa, Asia, etc.).

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