LAPPEENRANTA-LAHTI UNIVERSITY OF TECHNOLOGY LUT
LUT School of Energy Systems
Degree Programme in Energy Technology–Nuclear Engineering

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# REACTOR CORE CONCEPTUAL DESIGN FOR A SCALABLE HEATING EXPERIMENTAL REACTOR, LUTHER

Examiners: Professor D.Sc. (Tech.) Juhani Hyvärinen
Assistant Professor D.Sc. (Tech.) Heikki Suikkanen

## ABSTRACT

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**Reactor core conceptual design for a scalable heating experimental reactor, LUTHER**

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Keywords: conceptual design, nuclear district heating reactor, small modular reactor, pressure-channel reactor, moving fuel assemblies

In this thesis, the first conceptual design and a preliminary study of LUT heating experimental reactor (LUTHER) for a 2 $MW_{th}$ power are presented. Additionally, commercial-sized reactor designs for 24 $MW_{th}$ and 120 $MW_{th}$ powers are also studied and discussed. LUTHER is a scalable light-water pressure-channel reactor designed to operate at low temperature, low pressure and low core power density. The LUTHER core utilizes low enriched uranium (LEU) to produce low-temperature output, targeting specifically the district heating demand in Finland. LUTHER is developed to contribute to decarbonizing the heating and cooling sector, which is a more significant greenhouse gas emitter than electricity production in the Nordic countries.

The main principle in the development of LUTHER is to simplify core design and safety systems, which, along with using commercially available reactor components, would lead to lower fabrication costs and enhanced safety. LUTHER also features a unique design with moving fuel assemblies used for reactivity control, fuel burnup compensation and reactor shutdown. The 2 $MW_{th}$ LUTHER core is designed to experiment and demonstrate the novel means of reactivity control and feasibility of a pressure-channel district heating reactor. However, the 2 $MW_{th}$ core seems too small to be feasible as an operating operator. Recommendation for increasing the core power of the demonstration reactor to 6 $MW_{th}$ is proposed.

2-dimensional (2D) and 3-dimensional (3D) fuel channels with fuel assemblies inside and reactor cores are modeled with the Serpent Monte Carlo reactor physics code. Different reactor design parameters and safety configurations are calculated and assessed, regards the core's basic thermal hydraulics and reactor physics. Preliminary results show an optimal basic core design, a good neutronic performance and feasibility of controlling reactivity by moving fuel assemblies, eliminating the use of conventional control rods and soluble poisons, such as boron.

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Thank you, Lord, for always being there for me.

Lappeenranta, 25 November 2019

Thinh Truong

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# NOMENCLATURE

## Latin alphabet

| | | |
|---|---|---|
| $A$ | area / surface area | $\mathrm{m^2}$ |
| $c_p$ | specific heat capacity | $\mathrm{J/(kg \cdot K)}$ |
| $d, D$ | diameter | $\mathrm{m}$ |
| $f$ | mass fraction | $-$ |
| $Fr$ | radial power peaking factor for the whole core | $-$ |
| $h$ | convective heat transfer coefficient | $\mathrm{W/(m^2 \cdot K)}$ |
| $k$ | multiplication factor | $-$ |
| $L$ | length | $\mathrm{m}$ |
| $m$ | mass | $\mathrm{kg}$ |
| $M$ | molecular weight | $\mathrm{g/mol}$ |
| $\dot{m}$ | mass flow rate | $\mathrm{kg/s}$ |
| $n$ | number of fuel rods in a fuel assembly | $-$ |
| $N$ | number of fuel assemblies in a reactor core | $-$ |
| $p$ | fuel pin lattice pitch | $\mathrm{m}$ |
| $Q$ | thermal power | $\mathrm{W}$ |
| $\dot{Q}$ | rate of heat transfer or heat loss | $\mathrm{W}$ |
| $Q'$ | linear heat rate | $\mathrm{W/m}$ |
| $Q''$ | heat flux | $\mathrm{W/m^2}$ |
| $Q'''$ | power density | $\mathrm{W/m^3, W/l}$ |
| $r$ | radius | $\mathrm{m}$ |
| $R$ | thermal resistance | $\mathrm{K/W}$ |
| $T$ | temperature | $\mathrm{^\circ C, K}$ |
| $v$ | flow velocity | $\mathrm{m/s}$ |
| $V$ | volume | $\mathrm{m^3}$ |

## Greek alphabet

| | | |
|---|---|---|
| $\alpha$ | fuel channel lattice pitch | $\mathrm{m}$ |
| $\gamma$ | fraction of recoverable energy from fission reaction | $-$ |
| $\delta$ | spacing clearance | $\mathrm{m}$ |

| | | |
|---|---|---|
| $\Delta$ | change in the following variable | – |
| $\kappa$ | thermal conductivity | $\mathrm{W/(m \cdot K)}$ |
| $\mu$ | dynamic viscosity | $\mathrm{Pa \cdot s}$ |
| $\rho$ | density | $\mathrm{kg/m^3}$ |
| $\omega$ | weight fraction | – |

**Dimensionless numbers**

| | | |
|---|---|---|
| $Nu$ | Nusselt number | – |
| $Pr$ | Prandtl number | – |
| $Re$ | Reynolds number | – |
| $\rho$ | reactivity | – |

**Supscripts**

| | |
|---|---|
| $1, 2, 3$ | numbered item |
| c | fuel clad |
| ci | inner surface of the fuel clad |
| co | outer surface of the fuel clad |
| cool | reactor coolant |
| core | reactor core |
| eff | effective |
| f | fuel pellet |
| fa | fuel assembly |
| fi | inner surface of the fuel pellet |
| fo | outer surface of the fuel pellet |
| g | gas gap |
| h | hydraulic |
| mod | reactor moderator |
| O | oxygen |
| p | pressure tube |
| r | reactor channel |
| t | thermal insulator |
| total | total |

| | |
|---|---|
| th | thermal |
| U | uranium |
| U235 | uranium-235 |
| U238 | uranium-238 |
| ∞ | infinite |

**Abbreviations**

| | |
|---|---|
| 2D | two dimensional |
| 3D | three dimensional |
| ACR | Advanced CANDU Reactor |
| AGS | Annulus Gas System |
| ATWS | Anticipated Transient Without Scram |
| BOC | beginning-of-cycle |
| BWR | Boiling Water Reactor |
| CANDU | CANada Deuterium Uranium |
| CHP | Combined Heat and Power |
| DHR | District Heating Reactor |
| DPR | Deep Pool Reactor |
| EOC | end-of-cycle |
| EPR | European Pressurized Reactor |
| EPZ | Emergency Planning Zone |
| EU | European Union |
| GHG | Greenhouse Gas |
| H/HM | Hydrogen-to-heavy-metal ratio |
| HWR | Heavy Water Reactor |
| IAEA | International Atomic Energy Agency |
| INET | Institute of Nuclear Energy Technology |
| INSAG | International Nuclear Safety Advisory Group |
| iPWR | integrated Pressurized Water Reactor |
| KAERI | Korea Atomic Energy Research Institute |
| LEU | Low Enriched Uranium |
| LOCA | Loss of Coolant Accident |

| | |
|---|---|
| LTNR | Low-Temperature Nuclear Reactor |
| LUT | Lappeenranta-Lahti University of Technology |
| LUTHER | LUT Heat Experimental Reactor |
| LWR | Light Water Reactor |
| MOC | middle-of-cycle |
| NHR | Nuclear Heat Reactor |
| NIMBY | not in my backyard |
| NPP | Nuclear Power Plant |
| PIUS | Process Inherent Ultimate Safety |
| PRHRS | Passive Residual Heat Removal System |
| PWR | Pressurized Water Reactor |
| RCS | Reactivity Control System |
| RPV | Reactor Pressure Vessel |
| RWFA | Robust Westinghouse Fuel Assembly |
| SCWR | Supercritical Water-cooled Reactor |
| SECURE | Safe Environmentally Clean Urban REactor |
| SHR | Swiss Heating Reactor |
| SLOWPOKE | Safe LOW POwer Critical Experiment |
| SMART | System-integrated Modular Advanced ReacTor |
| SMR | Small Modular Reactor |
| VVER | Water-Water Energetic Reactor |
| YSZ | Yttria-Stabilized Zirconia |
| ZIRLO$^{TM}$ | Zirconium Low Oxidation |
| ZYC | Zirconium Oxide Cylinder |

# 1 INTRODUCTION

## 1.1 Background

In colder climate regions, such as the Nordic countries, heating plays an essential role in energy markets and is one of the dominant sectors of the final energy use. In the European Union (EU), heating and cooling take up approximately 50% of the total final energy consumption, approximately 6600 TWh in 2012. Of these, 75% of the heating and cooling supply is still generated by the direct use of fossil fuels, namely coal, gas and oil. More specifically, the space heating had a share of about 50% of the total final energy demand for heating and cooling, which contributes significantly to the total annual greenhouse gas (GHG) emissions in the EU. Figure 1.1 provides the different shares of heating and cooling end-uses in the EU, as well as in the Nordics. (Patronen et al. 2017)

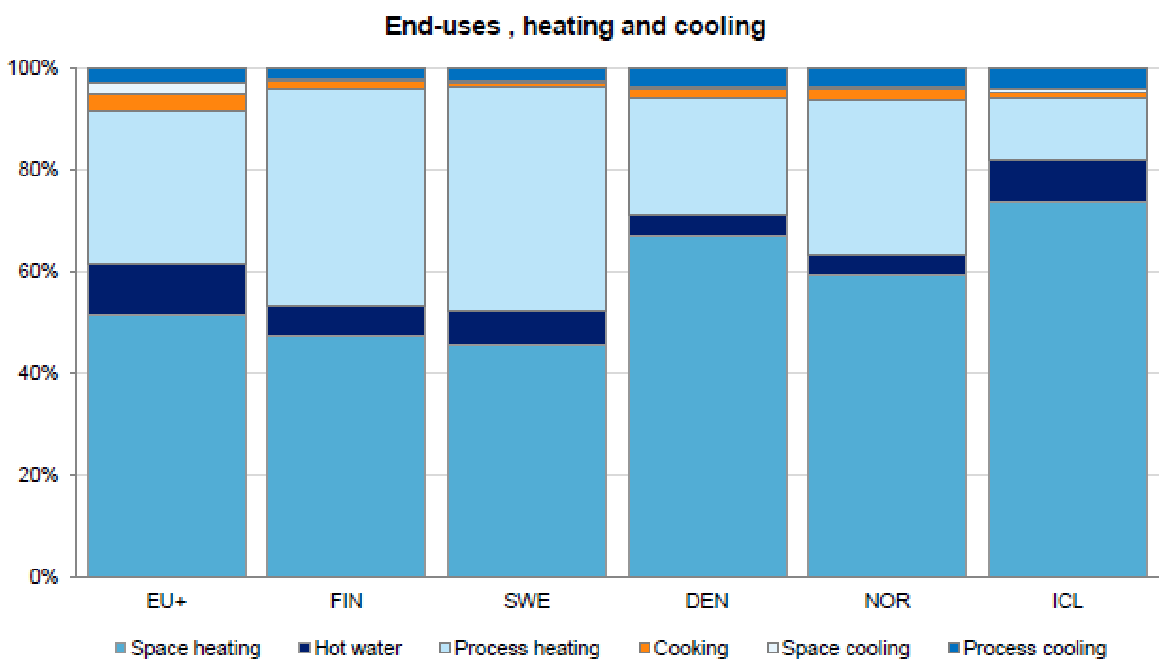



Figure 1.1: Heating and cooling end-uses in the EU (including Norway and Iceland) and in the Nordic countries (Patronen et al. 2017).

In particular, the majority of heating (in other words, space heating or district heating) in Finland is still depending significantly on the use of fossil fuels, which is a more significant $CO_2$ emitter than electricity production. District heating in Finland had a share of about 46% of the national heat market in 2016, as shown in Figure 1.1, along with other end-uses (Paiho and Saastamoinen 2018). Figure 1.2 shows the break-down share of different energy sources for the district heat supply in Finland with a total of 37.1 TWh for 2018. It can be seen clearly that fossil fuels, mainly coal and gas, and peat, are still the primary source of fuels for district heat production in Finland (Energiateollisuus ry 2019). Consequently, the district heating sector still contributes significantly to the total GHG emissions in Finland and is in need of emission-free and reliable sources of energy in replacing current sources.

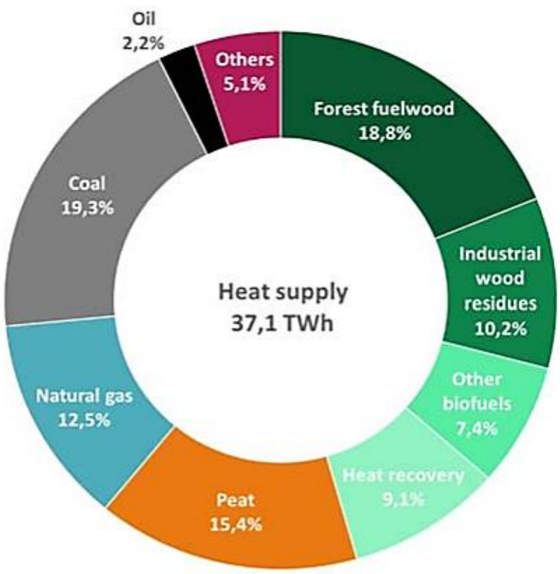



Figure 1.2: Energy sources of district heat supply in Finland for 2018 (Energiateollisuus ry 2019).

Due to the current trend of consumption and production of energy, the EU established the heating and cooling policy and strategy in 2016 to reduce GHG emissions by 2030 (Patronen et al. 2017). The EU's climate and energy goals aim to decarbonize by reducing the use of fossil fuels and increase energy efficiency in the heating and cooling sector. Furthermore, Finland, in particular, has ambitious long-term goals of becoming a carbon-neutral country while securing the national energy supply, as well as improving the current energy systems and technology by 2035 (Valtioneuvosto 2019); initially, it was set for 2050 (Patronen et al. 2017). Finland’s long-term energy and climate goals focus on reducing the use of fossil fuels while increasing the use of emission-free energy, eliminating the use of coal in energy production after 2030 and achieving an 80-95% reduction in GHG emissions (Patronen et al. 2017).

In order to transition energy entirely away from fossil fuels, the security of energy supply is one of the priorities in the heating and cooling sector. As shown in Figure 1.2, fossil fuels and peat still constitute a considerable share of about 50% in the district heat supply in Finland. To achieve carbon neutrality and make the energy and climate goals achievable by 2035, dependency on fossil fuels should be reduced and replaced with emission-free and reliable sources. Increasing renewable energy is an option; however, Paiho and Saastamoinen (2018) assessed that renewable energy sources have seasonal and daily variation, which affects the end-users of district heating. Renewable energy sources also have a problem with producing lower water temperatures than the required temperatures that are currently utilized in the Finnish district heating network (Paiho and Saastamoinen 2018). Therefore, these limitations urge the need for a stable and reliable source of clean heat production, especially during the peak winter season, which also meets the technical requirements of the current Finnish district heating networks.

These ambitious decarbonization plans from the EU and Finland’s reaching carbon neutrality, along with the need for secure and reliable energy supply, make nuclear heating an attractive option. Additionally, due to the current trend towards de-centralized energy systems and recent difficulties in the construction of large units, there is a keen interest in small reactors, in other words, small modular reactors (SMRs). Furthermore, the cost-effective production of low-temperature heat with dedicated small reactor units calls for a reactor design with simplified reactor core and safety systems. It also needs to be easy to manufacture, for instance, in serial production and should utilize off-the-shelf components as far as possible to help to keep unit cost low and competitive. Therefore, LUT University is motivated to start the conceptual designing of a dedicated district heating reactor with the aims of cost-effectiveness, modularity, simplification and safety.

## 1.2 Objectives and scope of the study

The primary objective of this research is to develop a conceptual design of a small modular light water reactor (LWR) with a simple reactor core and minimal dedicated safety systems. The proposed reactor is aimed for the district heating supply in Finland, compatible with the current district heating networks, thereby replacing the current fossil-fueled plants, and enabling serial production with associated cost and time savings. A simple and robust reactor system is necessary because district heating reactors have to be sited relatively close to consumers (in other words, urban areas). Simplification will lead to an easily understood safety justification and lower infrastructure costs, thereby improving both societal acceptance and the economy of nuclear power.

LUT heating experimental reactor (LUTHER) is a scalable light-water pressure-channel reactor designed to operate at a low temperature, low pressure and low core power density. The process of conceptual designing LUTHER starts with a small core of 2 $MW_{th}$ power and follows by the commercially sized versions of 24 $MW_{th}$ and 120 $MW_{th}$ powers. The 2 $MW_{th}$ LUTHER design is aimed to experiment and demonstrate the novel means of reactivity control and the feasibility of a pressure-channel district heating reactor.

In this research, the pressure-channel based design was selected for the LUTHER core concept due to two main reasons. The first reason that makes the pressure-channel based design favorable is the elimination of the reactor pressure vessel (RPV) in LUTHER. Secondly, the proposed design also allows for the reactor core to be scaled up with ease by simply adding more pressure channels, containing fuel assemblies inside, to increase the thermal power output. Hence, the pressure-channel based design keeps the LUTHER core concept simple and cost-effective, which contrasts with conventional pressurized water reactors (PWRs).

Furthermore, the development of LUTHER core concept is based on the past and on-going development and designs of low-temperature nuclear reactors (LTNRs) or heat-only reactors. Some of these reactors are SECURE (Safe Environmentally Clean Urban Reactor), NHR (Nuclear Heating Reactor) and DPR (Deep Pool Reactor). Some features used in pressure-channel reactors, such as Canada Deuterium Uranium (CANDU), Advanced CANDU reactor

(ACR) and Supercritical Water-cooled Reactor (SCWR), are considered and implemented in the design proposal.

The aims of this study are:

- To design a pressure-channel LWR core for supplying district heat in Finland, which operates at low temperature, low pressure and a low core power density,
- To calculate and assess for optimal dimensions and design parameters for the proposed reactor concept,
- To conceptually design a fuel assembly that can move in a pressure tube, providing a primary means to control reactivity, replacing conventional control rods and soluble boron,
- To develop an alternate diverse shutdown mechanism of the reactor, without introducing control rods or dissolved boron, and
- To assess the feasibility of a pressure-channel district heating reactor and the novel means of reactivity control by moving fuel assemblies.

The development of LUTHER concept should also be established under the following criteria:

- The design should be utterly simple for low cost, simple regulation and highly enhanced safety;
- The design should be based on proven conventional technology and uses commercially available reactor components as far as possible;
- The design should follow and satisfy the safety standards of the International Atomic Energy Agency (IAEA), stated in the Safety Guide No. NS-G-1.12 that are relevant to the reactor core design of a low-temperature nuclear reactor (LTNR) (IAEA 2005).

In this thesis, the scopes of the research are to conceptually design the LUTHER core and study the feasibility of a pressure-channel district heating reactor and a unique feature of moving fuel assemblies for reactivity control. Basic reactor physics and heat transfer calculations are necessary for determining design dimensions and parameters featuring in the design. In-depth calculations and assessments regarding the proposed designs are beyond the scope of the study.

## 1.3 Research methodology

This thesis work was performed by using a combination of a diverse literature review, trials and errors in designing, numerical calculations and computational simulations of different proposed design parameters. The aim of the background and literature review was to understand the current situation of energy systems (mainly district heating in Finland), the development of nuclear district heating reactors or LTNRs and referenced reactors used for the proposal of LUTHER core concept. Furthermore, the available design methodology and considerations used in designing and developing a nuclear reactor core were also reviewed. A flow chart of an engineering design process for the LUTHER core concept is also presented and used as a designing guidance for the study. Referenced pressure-channel reactors (for example, CANDU, ACR and SCWR) and previous and on-going developing LTNR designs (for example, SECURE, NHR and DPR) were reviewed and considered in the development of the LUTHER design.

Furthermore, basic heat transfer calculations and reactor physics simulations were performed to assess and optimize different proposed design parameters of the LUTHER core. Thermal-hydraulic calculations in reactor core were done by Microsoft Excel to acquire basic heat transfer parameters and average temperatures of different reactor components (for example, fuel elements, coolant, thermal insulation, pressure tube and moderator) for reactor modeling. In addition, a computational tool called Serpent Monte Carlo reactor physics code developed by the VTT Technical Research Centre of Finland Ltd. was used to model the proposed design in two-dimensional (2D) and three-dimensional (3D) simulations (Leppänen et al. 2015). Serpent code is used in this study for calculating the multiplication factor, power distribution, fuel burnup and reactivity control of the core.

## 1.4 Organization of the study

The thesis is divided into eight chapters, beginning with background introduction, objectives and scopes of the study and methodology behind this novel research and development. Chapter 2 presents a brief background information and literature review of district heating networks in

Finland, low-temperature nuclear reactors and referenced pressure-channel reactors used in designing the LUTHER core.

Chapter 3 describes the safety and considerations in designing a nuclear reactor core, which complies with the IAEA safety standards of nuclear reactor design. In addition, a methodology of reactor core design is presented and serves as a guideline in designing the LUTHER core.

Chapter 4 provides an overview of the reactor core conceptual design and considerations in designing LUTHER. Different reactor core components and selection of materials used in this current study are also presented and implemented in the design. Different options for the design features and materials are also discussed and compared in order to optimize the performance of the proposed reactor while maintaining its simplified concept of the reactor core. In addition, thermal-hydraulic calculations were performed to acquire basic heat transfer parameters and the average temperatures of different reactor components used in modeling by the Serpent Monte Carlo code.

Chapter 5 covers the LUTHER modeling and reactor physics calculations in the Serpent code by using proposed design parameters and selected material for reactor components. Results obtained from the calculations are presented in this chapter, which describes characteristics of the current designs regarding the fuel assembly and fuel channel, reactor core and the reactivity control system. Modeling methodology and assessment logics are also included in the chapter.

Chapter 6 covers the discussions and analyses of the results, which consist of proposed design parameters for LUTHER core, reactor thermal hydraulics and reactor physics. Feasibility of the LUTHER core concept and the use of moving fuel assemblies for reactivity control are also discussed in this chapter.

Chapters 7 and 8 summarize this research along with preliminary results and provide concluding remarks regarding the development of the novel design and recommendations for future work of the development of LUTHER.

# 2 LITERATURE REVIEW

## 2.1 District heating in Finland

In Finland, spacing heating or district heating has a share of about 46% of the national heat market; the biggest end-use of energy in the heating and cooling sector, following by the processing heat (Patronen et al. 2017; Paiho and Saastamoinen 2018). In 2018, the total annual district heat supply was 37.1 TWh, of which approximately 50% of the total energy supply came from the direct use of fossil fuels, mainly coal and gas, and peat (Energiateollisuus ry 2019). The break-down share of energy sources of district heating supply in 2018 can be seen in Figure 1.2. Even though the share of renewable energy, such as bio-based fuels, has been growing and constitutes more than one-third of the district heat supply, reducing the use of fossil fuels is still essential by increasing the use of emission-free energy sources.

The Finnish government has set long-term energy and climate goals to becoming a carbon-neutral country by 2035 (Valtioneuvosto 2019); initially, it was set for 2050 (Patronen et al. 2017). The aims of the energy and climate goals in Finland are to increase the use of emission-free energy and ban energy production from coal in 2029 (Leppänen 2019). The plan is ambitious and challenging yet possible to achieve carbon neutrality with the use of emission-free nuclear energy.

In 2018, there were 107 power plants delivering district heat to about 200 district heating networks (in other words, municipalities) (Energiateollisuus ry 2019), most of which are wholly or partially off-grids (Partanen 2019). Depending on the season, weather and peak demands, in Finland district heating networks are operated at a temperature range of 65-120$^{o}$C (Leppänen 2019). Large cogeneration power plants are typically used to provide a baseload in heat supply. Therefore, nuclear energy can be used to provide a reliable baseload in heat supply throughout the year while contributing to decarbonizing of the district heating networks in Finland.

## 2.2 Low-temperature nuclear reactors

### 2.2.1 Background information

Since the oil crisis of 1973 when the price of oil rose significantly, nuclear energy became an attractive source for heating applications. Besides the economic aspect, the concerns of environment, security of energy supply and worldwide trading during that time were also factors imposed on the reduction of the use of oil (Nilsson and Hannus 1978). Thereby, the crisis brought a significant concern on the availability of cheap fuels for heating purposes to several countries, especially for the Nordic countries.

In addition to the combined-heat-and-power (CHP) technology from existing nuclear power plants, from the past, various concepts and designs dedicated to low-temperature heat production were introduced and demonstrated. Low-temperature nuclear reactors are mainly an LWR or a heavy water reactor (HWR) type that uses nuclear fission energy to heat water to a desired low-temperature output. LTNRs are designed to operate at low temperature and low pressure, conceptually ranging from 110-224$^{o}$C and 0.3-2.5 MPa, respectively. Meanwhile, conventional LWRs currently operate at higher temperature and pressure, 286-345$^{o}$C and 7-15.5 MPa, respectively. (Leppänen 2019)

### 2.2.2 Advantages of low-temperature nuclear reactors

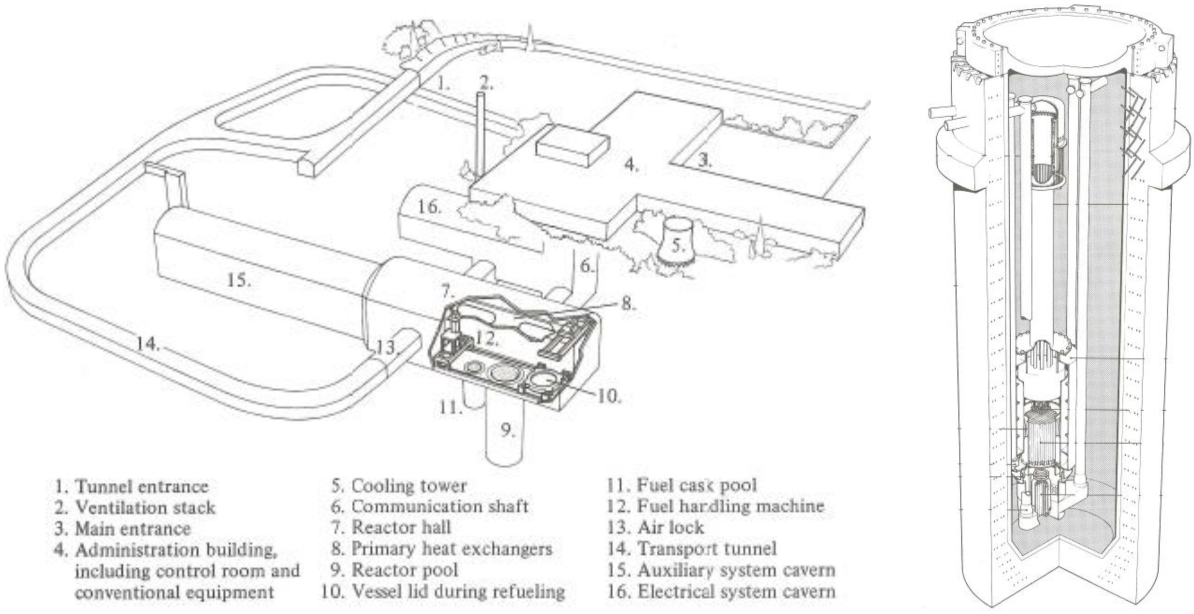
LTNRs are designed to operate at significantly lower thermal parameters, compared to large nuclear power plants (NPPs), in order to be compatible with the district heating networks. Consequently, the core design and its safety systems are becoming much simpler, thereby simplifying the operation of the reactor during normal or any abnormal condition. Owing to its distinct characteristics, LTNRs have a high potential to minimize the emergency planning zones (EPZ) requirement, thus, making the reactors to be possible to be sited near the customers. (Leppänen 2019)

### 2.2.3 SECURE reactor

One of the early low-temperature nuclear reactors or heat-only reactors developed for district heating in the Nordics was the Safe Environmentally Clean Urban REactor (SECURE). SECURE was developed as a result of a Swedish-Finnish collaboration in the 1970s (Bento and Mankamo 1978). The SECURE reactor was designed as a dedicated baseload heat-only reactor for moderate- sized district heating networks (Nilsson and Hannus 1978). The reactor features a 200 $MW_{th}$ or a 400 $MW_{th}$ output with an operating pressure of 0.7 MPa and a temperature of 115$^{o}$C (Leppänen 2019). The heat generated by the fission chain of reactions is transferred to the district heating networks via an immediate water loop. 8×8 standard boiling water reactor (BWR) fuel assemblies from the ASEA-ATOM with four different low enrichments were used for the SECURE core (Gransell and Höglund 1978).

The reactivity of the SECURE core is controlled by the concentration of soluble boron material in the moderator, which replaces the use of control rods in the system. In any transient or an emergency accident or during an annual outage, the reactor is ensured in a subcritical state by dropping boron steel balls as a neutron absorber into the water channels of the fuel assemblies. (Lemmetty 2012a)

Unlike conventional NPPs, the SECURE concept depicted a unique feature of using soluble absorber for reactivity control, eliminating control rods. It also approached the urban siting problem of any reactor faces nowadays. In the primary design criteria of SECURE, the need for large EPZs was aimed to be eliminated. The reactor aimed to simplify the system by minimizing the use of active safety systems and relying on inherent passive safety systems (for example, gravity and natural circulation) that are based on the Process Inherent Ultimate Safety principle (PIUS), which was proposed at that time (Leppänen 2019). Furthermore, the reactor concept was designed to be situated below-grade level, which can be served as physical protection of the power plant and a primary containment of the reactor. Figure 2.1 depicts schematic views of the SECURE design concept, which comprises an underground reactor plant layout (figure a) and reactor vessel with its components in a vertical view (figure b).




(a) (b)

Figure 2.1: Schematic views of the SECURE design concept: an underground reactor plant layout (a); a reactor vessel with internal components in a vertical view (b). Adapted from Bento and Mankamo (1978).

Moreover, the SECURE concept had several flaws in the design that can pose problems concerning the safety of the reactor. Some of the flaws included the lack of heat removal mechanism from the containment (in other words, spray system), weak retention of radioactive substances, and unqualified auxiliary systems and automation. Although SECURE was only conceptually designed with a few flaws, some developed ideas of the reactor were beneficial and relevant in many existing SMRs. Some of which is the idea of eliminating control rods for reactivity control and reactor siting closer to the consumers, in other words, city, and other densely populated areas. (Lemmetty 2012b)

### 2.2.4 Nuclear heating reactor

The development of heat-only LWRs in China has started promptly in the early 1980s due to the need for a reliable and clean energy source in the energy sector (Dazhong 1993). In the past, Chinese heating consumption was supplied by mainly burning coal. To replace the use of coal as a primary source, the Institute of Nuclear Energy Technology (INET) in Beijing has started the development of heat-only reactors, aiming to supply district heating to the cities. The Chinese development of low-temperature heat-only reactors has focused on two different technologies: nuclear heating reactor (NHR) and deep pool reactor (DPR) (Leppänen 2019). A Schematic configuration of the NHR reactor design is presented in Figure 2.2.

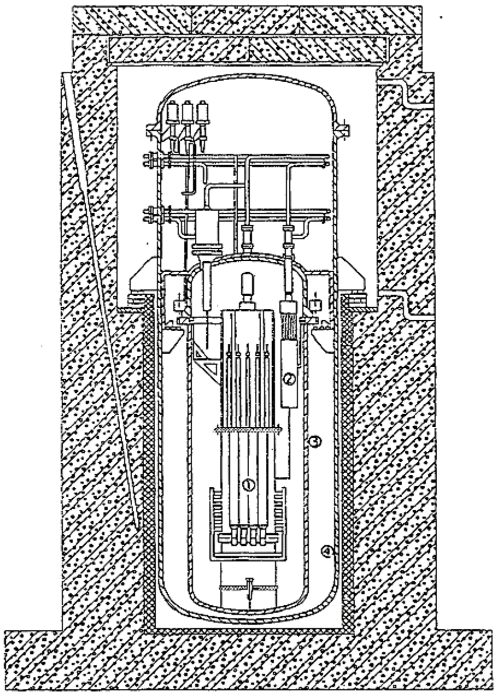



Figure 2.2: Schematic configuration of NHR-5 reactor; referenced from Dafang et al. (1997). NHR-5 is a demonstration reactor for district heating supply in China.

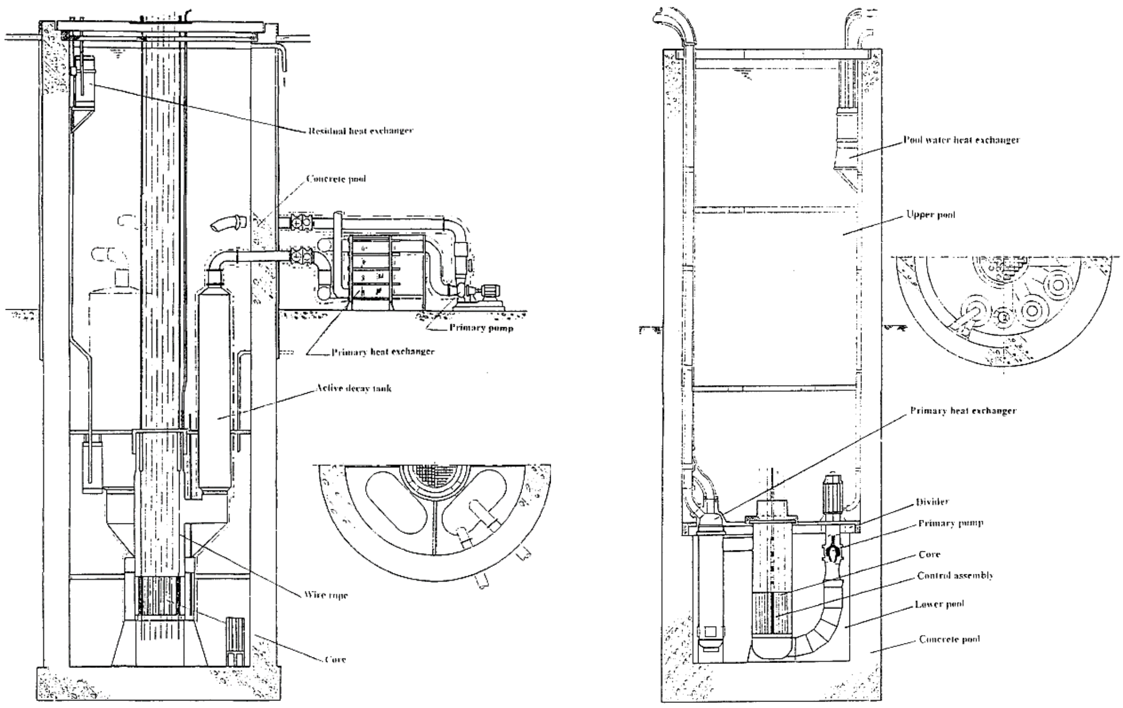
NHR design is based on an integrated pressurized water reactor (iPWR) where the core and primary system of the reactor are housed within an RPV (Dafang et al. 1997). The NHR is designed with a dual pressure vessel and features low temperature, low pressure and low core power density. The designers of the NHR claimed that the dual vessel, enclosing the primary system, ensures the coolant flooding of the reactor core without relying on any emergency cooling system in the case of a large loss of coolant accident (LOCA) (Yajun et al. 2003). In addition to the iPWR's features, the primary coolant system relies on the natural circulation at the full-power operation to transfer heat to the secondary side (for example, district heating grid) via an intermediate loop (Dong et al. 2018). For reactivity control, the NHR reactor uses hydraulic-driven control rods along with burnable poison in the fuel (Dazhong 1993). Also, a boric acid injection is used as a secondary standby shutdown system during the event of anticipated transient without scram (ATWS).

A prototype of an experimental NHR reactor with the thermal power of 5 MW (NHR-5) was constructed and demonstrated the feasibility of the design as a district heating reactor since 1989 (Dafang et al. 1997). The NHR-5 was designed to operate at a pressure of 1.37 MPa and a temperature between 146$^{o}$C and 186$^{o}$C. Later on, the NHR-200II with thermal power of 200 MW has been developed on the experience gained from the design, construction and operation of NHR-5. Similar safety features from NHR-5 have been adapted by the NHR-200II. Slight modifications to the operating conditions of the reactor that includes an increase in primary pressure from 1.37 to 8 MPa, with core inlet and outlet temperatures of 232$^{o}$C and 280$^{o}$C, respectively. The NHR-200II is designed for electricity generation, district heating and seawater desalination. (Dong et al. 2018)

### 2.2.5 Pool-type reactor

On the other hand, INET also developed a pool-type reactor (in other words, DPR), similar to a typical nuclear research reactor. The distinct difference of the DPR design is the use of hydrostatic pressure from a deep pool to obtain outlet temperature compatible with the district heating networks in China (Leppänen 2019). The DPR is designed to operate at low temperatures and atmospheric pressure, eliminating the need for the RPV, thereby removing the possibility of a LOCA caused by depressurization (Jiafu et al. 1998). The primary coolant system of DPR relies on forced circulation by pumps. The residual heat removal system is depending on a natural circulation driven by the temperature difference between the upper and lower pools.

Two DPR design concepts were developed, which are DPR-3 with 120 $MW_{th}$ by using 205 8×8 fuel assemblies, and a larger DPR-6 with 200 $MW_{th}$ by using 81 standard 15×15 PWR fuel assemblies (Jiafu et al. 1998). Figure 2.3 shows the schematic configurations of DPR-3 and DPR-6 reactors. Both reactors are designed to supply heated water at the temperature at 90$^{o}$C to meet the requirement of Chinese district heating networks, and DPR-6 can supply 120°C water in a short time. For Finland, the constant output temperature of DPR-6 is quite low in order to meet its district heating networks, especially during the winter season (Partanen 2019). In DPR-3 design, a 25-m deep pool is used to submerge the reactor core, allowing the reactor to operate at 0.29 MPa and 110°C. Meanwhile, DPR-6 design operates at 132°C by applying additional pressurization using the primary coolant pumps.




(a) (b)

Figure 2.3: Schematic configurations of DPR-3 reactor (a) and DPR-6 reactor (b); referenced from Jiafu et al. (1998); DPR-3 and DPR-6 are demonstration reactors for district heating supply in China.

With the successful study, license and demonstration of DPRs, the DHR400 (District Heating Reactor) has also been developed with thermal power of 400 MW, operating at low temperature (between 68°C and 98°C) and atmospheric pressure (IAEA 2018, 19). The design uses standard 17×17 PWR fuel assemblies, and its safety features are based on the previous DPR designs.

The DHR400 is designed for district heating, seawater desalination and radioisotope production.

### 2.2.6 Other low-temperature heat-only reactor designs

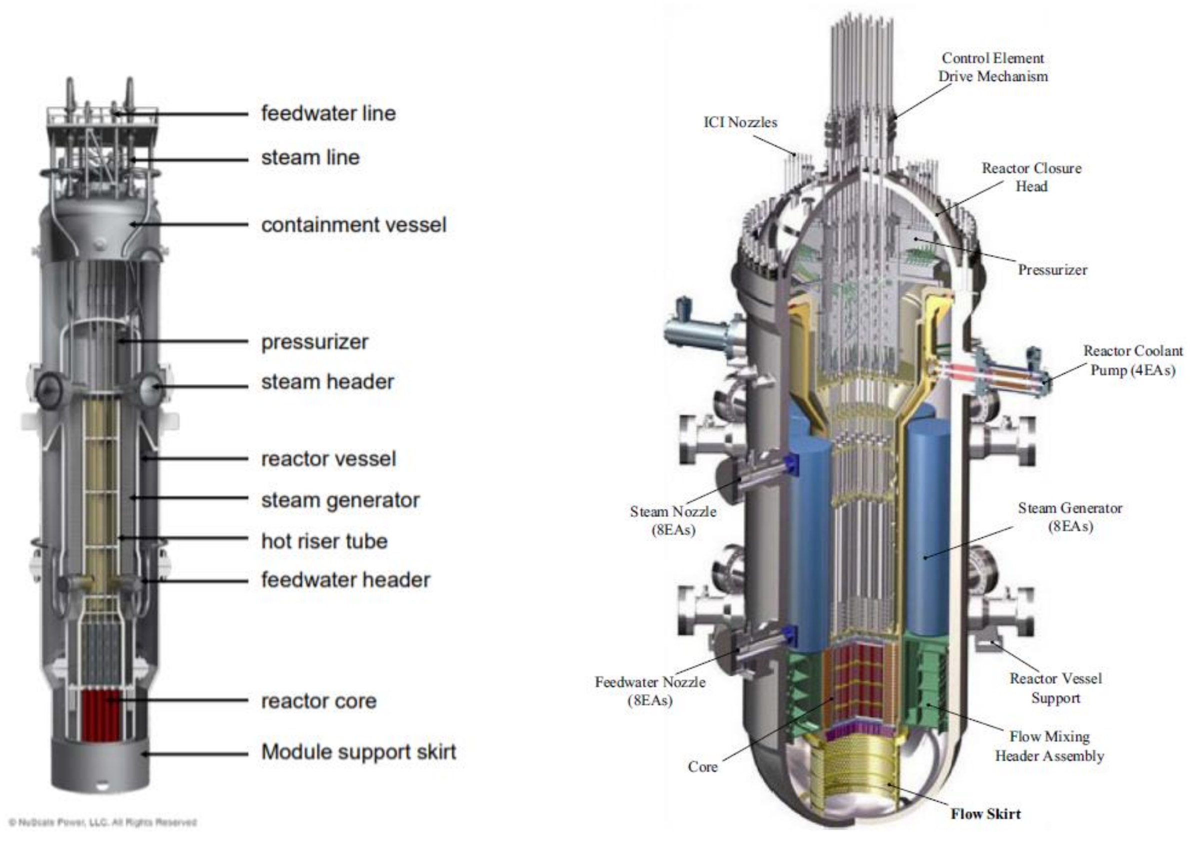
In addition to the SECURE, NHR and DPR reactors, there are existing several heat-only reactor designs around the world, including LWR and non-LWR types, which are not discussed in this thesis. Some of which interesting designs that might be useful to the LUTHER development consist of (IAEA 1987; 1988):

- The AST-500 reactor with a power of 500 $MW_{th}$ from USSR;
- The RUTA-70 pool-type reactor with a power of 70 $MW_{th}$ from Russia;
- The SLOWPOKE (Safe LOW POwer Critical Experiment) research and demonstration reactor from Canada, specifically SLOWPOKE-3 with 2-10 $MW_{th}$ power for district heating;
- The THERMOS reactor with 100-200 $MW_{th}$ power from France;
- The SHR (Swiss Heating Reactor) with 10 MWth power from Switzerland.

## 2.3 Small modular reactors

Another trend in the new nuclear development nowadays is small modular reactors, which is a smaller scaled version (up to 300 $MW_e$) of conventional large NPPs (IAEA 2018, 1). SMRs can be used for electricity, heat-only or CHP (cogeneration). One of the few highlighted features of the SMR designs is the incorporating of advanced or inherent safety features, such as the passive residual heat removal system (PRHRS). The PRHRS system is used to maintain the reactor core within adequate safety margins, obviating the dependence on active safety systems that are used in previous NPPs (Kim et al. 2016). Another feature is the modularity and flexibility of those SMR designs. This unique feature allows for manufacturing and assembling at a factory, lessening on-site construction; unmanned or remotely operating, reducing the staff required; power scalable by coupling multiple modules together; ability to work remotely without relying on existing power grids. Thereby, SMRs not only use necessary enhanced safety functions but also offer better economic affordability during construction than large NPPs. (Vujić et al. 2012; Ingersoll et al. 2015)

Among several existing designs, the most promising commercial light-water SMRs are the NuScale reactor and the SMART (System-integrated Modular Advanced ReacTor). The NuScale, providing 50 $MW_e$, is developed by NuScale Power Inc. in the United States and the SMART, providing 100 $MW_e$, is developed by Korea Atomic Energy Research Institute (KAERI) (IAEA 2018, 35 and 75). Figure 2.4 depicts a whole reactor concept for NuScale and SMART reactor designs.




(a) (b)

Figure 2.4: Schematic configurations of a 50 $MW_e$ NuScale reactor (a) and a 100 $MW_e$ SMART reactor (b); referenced from Ingersoll et al. (2015) and Kim et al. (2016), respectively.

Both designs are based on the integrated pressurized water reactor concept. In other words, it means the entire primary system pressure is contained in a containment vessel where its pressure is controlled by an in-vessel pressurizer, enhancing its robustness by eliminating major accidents such as pipe breaks (Vujić et al. 2012; Ingersoll et al. 2015). In NuScale SMR, the primary coolant system relies on natural circulation. In contrast, SMART's coolant system relies on reactor coolant pumps. Both reactor core designs are composed of conventional PWR low enriched uranium (LEU) fuel assemblies (17×17 square of $UO_2$ ceramic fuels with enrichment of less than 5%) with a shorter active length of fuel elements (about 2 meters long), along with other off-the-shelf reactor components used in PWRs. In addition, they also use conventional control rods and soluble boron for reactivity control and reactor shutdown. (IAEA 2018, 35-38 and 75-78)

# 3 REACTOR CORE DESIGN

## 3.1 Background overview

Safety is the key to the success of a reactor core design and operation for a nuclear power plant. The IAEA has established recommended safety standards in reactor core design for a conventional NPP. The IAEA's safety standards, stated in the Safety Guide No. NS-G-1.12, provide a guideline in designing a new nuclear reactor, which is useful in the conceptual designing of LUTHER (IAEA 2005). In this chapter, relevant safety requirements to the LUTHER core conceptual design in this study are presented, which serves as a guide in designing a nuclear district heating reactor.

## 3.2 Safety objectives

For any nuclear reactor design, it is essential that the system is able to demonstrate its designed functions and meet the safety objectives required during the operation. There are three fundamental safety functions as follow that the International Nuclear Safety Advisory Group (INSAG) and IAEA advised (1999, 42):

- Reactor core reactivity can be controlled;
- The fuel is adequately cooled;
- Radioactive material is securely confined.

These basic functions are the fundaments in assuring the safety of the nuclear reactor, which is highlighted by the IAEA, as presented in Figure 3.1.

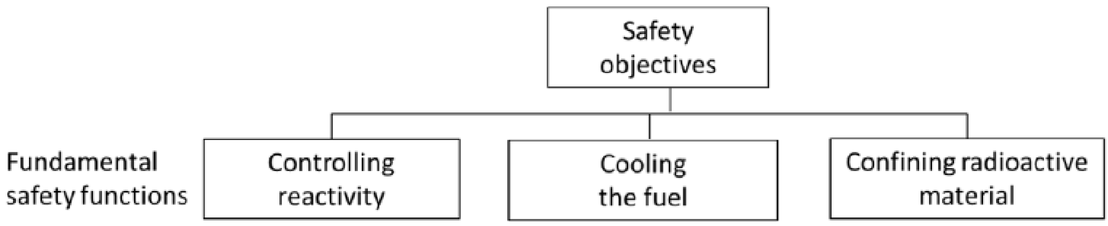



Figure 3.1: Basic representation of fundamental safety functions in reactor safety as expressed by the IAEA (Peakman et al. 2018).

### 3.2.1 Controlling reactivity

Based on the geometry and the design of the reactor core, neutronic performance, such as core reactivity, is an essential parameter to study and fully understand. The design choice in the core composition also affects the distributions of neutron flux and of the power and the core neutronic characteristics that make up a nuclear reactor. In a nuclear plant, two important features in counteracting a change in the core reactivity are the inherent reactivity feedbacks of the core design and the external systems which affect the core reactivity, for example, neutron absorbers (INSAG 1999, 52).

Under all operating conditions, the design of a reactor core relies on both inherent safety features and reactivity control systems to prevent reactivity induced accidents. To maintain within safe operating limits, the reactivity control systems are designed to enable the power change and compensate for changes in reactivity (IAEA 2005, 18-19). In addition, safety shutdown systems are designed independently from the reactivity control system, minimizing any system failures if used as a multi-system. The objective of these shutdown systems is to timely and effectively suppress the reactivity induced power transients and prevent damage to the reactor core (INSAG 1999, 52).

### 3.2.2 Cooling the fuel

The most critical design choice that impacts heat removal safety function in a nuclear reactor is the selected coolant medium (Peakman et al. 2018). The selected coolant is vital in the primary coolant system that provides a reliable means of cooling the core in normal operation. Any impairment of the ability to cool the fuel could lead to severe core damage, in extreme cases, which potentially propagates to loss of confinement of the radioactive material (INSAG 1999, 56).

The primary coolant system can also serve as a means for decay heat removal after an abnormal condition or accident (INSAG 1999, 54). As a precautionary measure, residual heat removal systems (RHRS), emergency core cooling systems (ECCS) and emergency feedwater systems are designed to protect the reactor coolant system integrity, preventing any arising conditions that could lead to a rupture of the primary coolant system boundary.

### 3.2.3 Confining radioactive material

The primary design purpose of multi-engineering barriers in a nuclear plant is to confine radioactive material against the possibility of its release from the fuel into the environment. As part of the defense-in-depth principle, the multi-barrier system is implemented to protect humans and the environment during an abnormal condition (INSAG 1999, 19). The confinement capability must be able to demonstrate its function such that the design would limit the leakage of any radioactive material (INSAG 1999, 58). The main objective of placing multiple barriers between radioactive materials and the environment is to provide redundant means to ensure several successive levels of protection. Its specific design of engineering barriers may be varied with different designs of NPPs.

## 3.3 Reactor safety functions

Reactor safety functions are introduced and implemented in a nuclear reactor design to assure the safety and integrity of the systems. In this section, further discussions concerning reactor safety functions and possible safety systems used for each function are presented. The discussions consist of reactivity control, reactor shutdown, removal of heat and radioactivity confinement, respectively.

### 3.3.1 Reactivity control

In addition to inherent reactivity feedback features of the design, the reactivity control system (RCS) is used to maintain the reactor core within an adequate safety margin during normal operation. RCS also takes into account possible design basis accidents and their consequences, providing the capability to reinstate the stable operating condition of the core. Various types of the system used for regulating the core reactivity and the power distribution which are relevant to the LWR design are listed as follow (IAEA 2005, 19):

- Use of solid neutron absorber rods and blades;
- Use of soluble absorber in the moderator and coolant;
- Control of the coolant flow;
- Use of fuel with distributed or discrete burnable poison;

- Control of the moderator temperature and height;
- Use of a batch refueling and loading pattern.

### 3.3.2 Reactor shutdown

In any abnormal or emergency or any temporarily disabling condition such as maintenance or refueling, a reactor core is needed to be shut down timely and effectively. The safety shutdown system is designed to quickly suppress the core reactivity induced power transients and prevent damages to the reactor core from such a cause (INSAG 1999, 52). IAEA (2005, 6) advised that there should be at least two independent and diverse shutdown systems available to secure the subcritical state of the reactor core. Among those systems, at least one shutdown system has the capability to quickly render the reactor subcritical, given the other systems operate as a redundant safety function (IAEA 2005, 23).

Different means of inserting negative reactivity into the core area used for different LWR designs consist of (IAEA 2005, 24):

- Injection of neutron poisons (for example, boron, gadolinium) into the moderator;
- Draining of the moderator;
- Insertion of solid control rod absorbers (for example, boron and stainless steel rods).

### 3.3.3 Removal of heat

Coolant is an essential means to protect the reactor core from overheating (in other words, meltdown) fuels resulted from accumulating fission energy. A selected coolant used in the reactor core should exhibit specific characteristics and meet the requirements in a nuclear environment (Peakman et al. 2018):

- High volumetric heat capacity;
- Good thermal conductivity;
- Low neutron absorption;
- High neutron scattering cross-section;
- Operating a low pressure at operational temperatures;

- Exhibiting limited activation in the presence of neutrons;
- Chemically compatible with the core and structural materials.

### 3.3.4 Reactor confinement

Reactor confinement is designed to mainly retain radioactive material release from the fuel during any abnormal condition. The design principle of multi-engineering barriers is implemented in a nuclear plant to ensure several successive levels of protection. For an LWR design, typical barriers confining the fission products are:

- The fuel matrix;
- The fuel cladding.

In addition, design precautions are also taken to prevent radioactivity from the primary loop into the district heating networks for a nuclear district heating reactor such as LUTHER. An implementation of maintaining a higher pressure in an intermediate heat transfer loop than that in the primary coolant loop is considered (IAEA 1998, 16).

## 3.4 Reactor siting

The current siting requirement of reactors intended for nuclear heat applications is a critical issue to the economic feasibility of the plant. An important factor affecting site selection is the NIMBY, which stands for "not in my back yard," syndrome. This syndrome affects decision-makers to choose remote locations to avoid potential conflicts and public opposition (IAEA 1998, 13). Newly designed small reactors or SMRs nowadays are facing regulatory challenges of urban siting requirements. Currently, the plants are required to be situated far away from the densely populated areas due to the EPZ requirements as part of the defense-in-depth principle (Leppänen 2019). Nuclear Regulatory Commission (NRC) and INSAG proposed that the siting decision of a reactor is affected by four main factors, which are summarized as follow (Lamarsh and Baratta 2014, 670-672; INSAG 1999, 40-41):

- Reactor design characteristics and its operation mode;
- Population density and characteristics of the environments of the site;
- Physical characteristics of the site;
- Safeguards of the reactor.

For nuclear district heating reactors, siting as close as possible to the customers is favorable due to economic feasibility. It is practically a necessary condition to be fulfilled since it is costly to transport heat to end-users in a long-distance, as of the current situation. Simple yet highly safe, nuclear district heating reactor, such as LUTHER, with robust inherent safety features, can be perceived as acceptable for close siting by the public. Thus, it would allow the reactor to sit relatively close to population centers and thereby keep heat transmission costs at reasonable levels.

## 3.5 Reactor core design methodology

In this study, the basic core designing of a nuclear district heating reactor LUTHER is performed to determine the feasibility of the conceptual design and the use of moving fuel assemblies as a primary means to control core reactivity. The process of conceptually designing the LUTHER core in this research is based on an iterated engineering design process. The engineering design process is presented in a flow chart presented in Figure 3.2, which covers six major steps of LUTHER core conceptual design and its feasibility studies. During whichever step, redesigning and optimizing are also performed iteratively if necessary. Throughout the designing process, safety objectives and reactor safety functions, as mentioned previously, are considered and implemented, in order to obtain a prototypical feasible conceptual design.

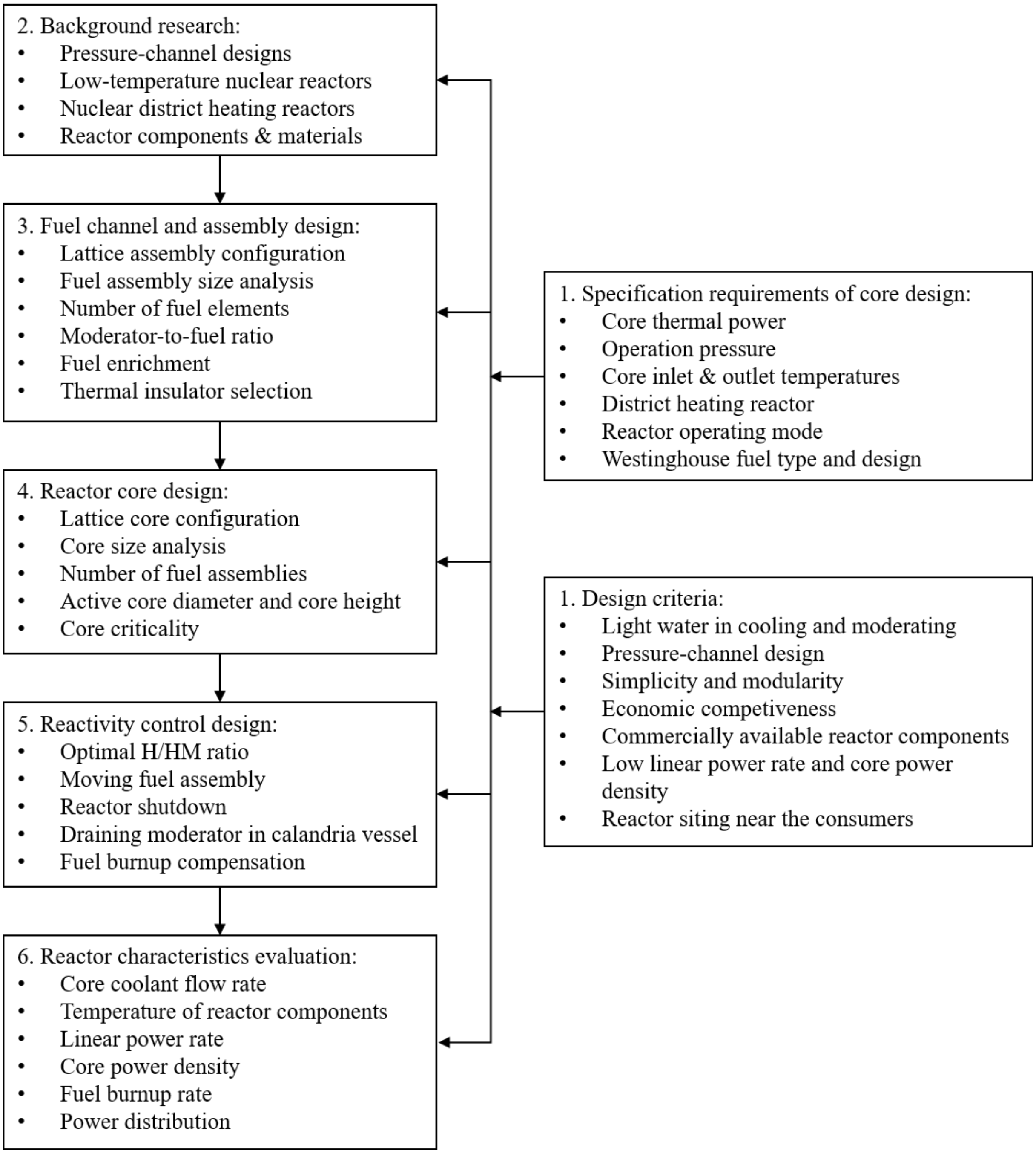



Figure 3.2: Flow chart of an engineering design process for a reactor core conceptual design of a LUT heating experimental reactor, LUTHER.

The designing process of the LUTHER conceptual core starts with defining design criteria and specification requirements. During this first step, different requirements, expectations and constraints regarding the conceptual design are identified and considered. Then, the second step covers the background research and study of the previous and on-going developments of relevant reactor designs (for example, low-temperature LWRs and pressure-channel designs) and other related literature concerning the design of LUTHER (for example, commercial fuel and materials used for reactor components).

The third step is dedicated to the design of the fuel channel and fuel assembly, which is a crucial step for which determines the feasibility of the design. In this stage, several explorations and assessments of different design parameters and features are performed to fulfill the design criteria and specification requirements as part of the LUTHER's engineering design process according to the flow chart.

The fourth step is focused on the development of the LUTHER core by implementing the proposed fuel channel and fuel assembly design from the previous stage. In this stage, core size assessments are carried out to determine an optimal number of fuel assemblies, the lattice pitch of fuel channels, active fuel height and core diameter, which affect the criticality of the core. Additionally, any issues or problems that arise from the proposed design are also identified and considered to improve and modify by iteratively repeating previous stages.

The fifth step is focused on the preliminary study and evaluation of reactivity control system options, as well as reactor shutdown mechanism, for the prototypical core. Here, the concept of moving fuel assemblies to control reactor reactivity is explored and assessed for its feasibility. In addition, the reactor shutdown mechanism by draining calandria and fuel assembly burnup calculations are performed.

Lastly, once the prototypical LUTHER core design is satisfied, the final step is to evaluate the neutronic performance and thermal characteristics of the reactor core. Core coolant flow rate and basic heat transfer calculations, such as linear heat rate and core power density, are performed in order to acquire the basic parameters of the reactor. In addition, core power distribution is calculated for the current analysis and further assessment.

# 4 CONCEPTUAL CORE DESIGN OF LUTHER

## 4.1 Overview of the design

The objective of this research is to propose an SMR or a heat-only low-temperature reactor with a simple and robust reactor system and inherent safety features. The proposed reactor is aimed for the district heating supply in Finland, replacing current fossil-fueled plants, and enabling serial production with associated cost and time savings. The simplifications of the reactor systems are necessary because district heating reactors have to be sited relatively close to the consumers (in other words, urban areas, geographically distributed all over the country). Simplification will lead to an easily understood safety justification and lower infrastructure costs, thereby improving both societal acceptance and the economy of nuclear power. The reactor is proposed to be situated in a below-grade level bedrock or rock cavern. This design choice can be used both as a physical protection barrier against external threats or radioactivity release and as a passive heat sink for decay heat removal.

In keeping with the simplified design concept, LUTHER uses movable fuel assemblies, enclosing by individual pressure tubes, to control the reactor power (in other words, core reactivity) and to compensate for fuel burnup during operation. This concept eliminates the use of conventional control rods and soluble boron in the systems, giving materials and equipment savings. Furthermore, the design approach with pressure tubes allows for eliminating the need for RPV and features their benefits in the scalability of the reactor core.

For the district heating networks in Finland, the outlet temperature range of the facility should be 90-120$^{o}$C, depending on the network structure, reactor operation mode and peak demands between seasons (Leppänen 2019). LUTHER cooling system is a pressurized water loop with an intermediate loop coupling the reactor circuit to the district heating network. LUTHER's conceptual core is designed to operate at the temperature of 150-180$^{o}$C and pressure of 1.25 MPa in the primary circuit. Thus, the manufacturing costs are expected to be significantly lower, and safety systems are considerably simpler than the systems in a reactor design using a conventional pressure vessel.

## 4.2 LUTHER fuel channel and fuel assembly design

### 4.2.1 Fuel assembly

The design of LUTHER is aimed to utilize many of its features from proven LWR technology, which ensures the enhanced safety and reliability. A schematic view of the LUTHER fuel channel with a fuel assembly inside is presented in Figure 4.1.

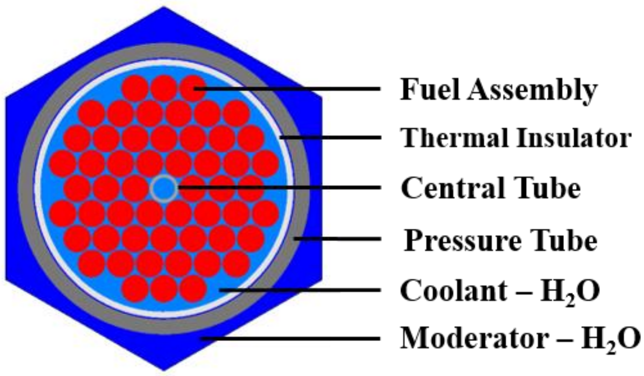



Figure 4.1: Schematic view of the LUTHER fuel channel with a fuel assembly inside.

The fuel assembly design selected for the LUTHER core is based on the VVER-1000 (Water-Water Energetic Reactor) Robust Westinghouse Fuel Assembly (RWFA) with modifications to the lattice pitch and the number and length of fuel elements. The RWFA, designed and manufactured by the Westinghouse Electric Company, has been used as a standard fuel product for the VVER-1000 units in Ukraine (2019). According to Westinghouse's report (2019), the RWFA design has been a reliable and excellent product in performance for the VVER-1000 units.

The RWFA's fuel pins comprise LEU ceramic pellets coated with ZIRLO™ (zirconium low oxidation) cladding. The original RWFA design consists of 312 fuel elements with a lattice pitch of 1.275 cm, 18 guide tubes for control rods and one instrument tube. For the LUTHER fuel assembly, the first design was modified and comprised of 54 fuel elements arranged in a hexagonal lattice. Additionally, the assembly is also designed with a central tube used for mechanical moving support and instrumentation. In addition, the pin lattice pitch was also modified for a 54-element fuel assembly fitting inside of a conventional size of pressure tubes, which is used typically in CANDU-type reactors. The selection of the optimal pitch was determined as a compromise between the mechanical design and the neutronic performance of the assembly. For the present design proposal, the lattice pitch of 0.96 cm was selected for the LUTHER fuel assembly.

### 4.2.2 Coolant and moderator

The selection of coolant and moderator materials is vital to the neutronic performance and heat removal capability in nuclear reactor design. In the design of LUTHER, light water is selected as both reactor coolant and moderator because it features

- A great heat-transfer medium,
- Highest neutron macroscopic slowing down power (in other words, macroscopic scattering cross-section) among common moderator materials, and
- Its abundancy, along with a low cost of production (U.S. Department of Energy 1993, 23-28).

In addition, light water also serves as an effective neutron shielding in both radial and axial directions of the core.

LUTHER's primary coolant system is designed to operate at the temperature of 150-180$^{o}$C and pressure of 1.25 MPa; the boiling point of water at this pressure is about 190$^{o}$C, as shown in Figure 4.2. Meanwhile, conventional boiling water reactor (BWR) and pressurized water reactor (PWR) operate at pressures of 7 and 12-15.5 MPa, respectively (Todreas and Kazimi 2001, 5). The saturation point of these pressures are 286$^{o}$C and 325-345$^{o}$C, respectively.

Additionally, LUTHER fuel channels are surrounded by the atmospheric-pressure light-water moderator, which is contained in a low-pressure calandria vessel (in other words, moderator tank). The light-water moderator is maintained at 40$^{o}$C, well below the saturation temperature of water at one atmospheric pressure or 0.101325 MPa (in other words, 100$^{o}$C). At this temperature, the moderator's nucleate boiling is avoided during normal operation. Thus, the coolant level in the calandria vessel is safely maintained for neutron moderation and passive cooling of fuel channels. Additionally, the chosen moderator temperature also allows for flexibility in increasing the operating temperature up to its saturation point without having to pressurize the calandria vessel if desired.

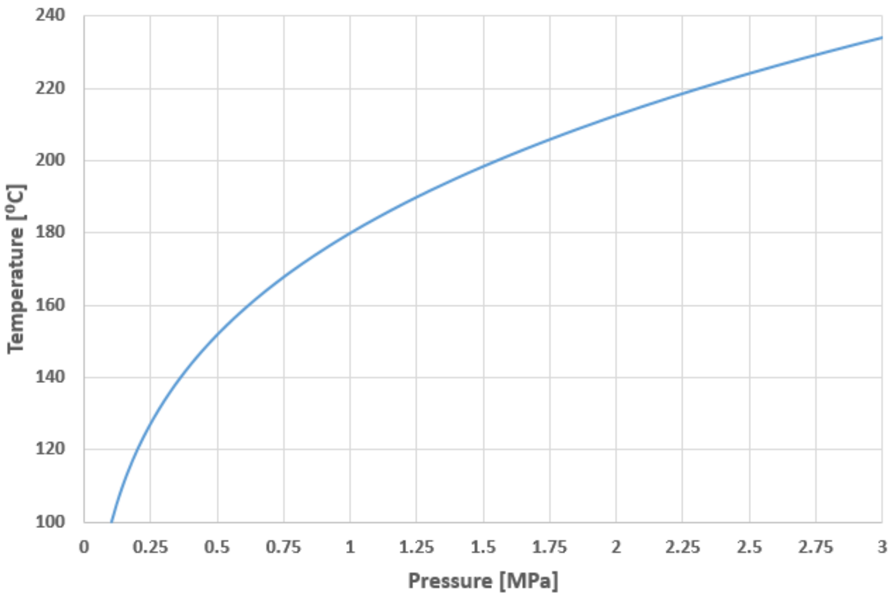



Figure 4.2: Saturation point of light water as a function of pressure. The data was obtained from the Indian Institute of Technology Bombay (2016).

### 4.2.3 Central tube

The central tube is a 1.2-mm thick annular cylinder with an inner diameter of 7.2 mm, and it is made of the same material as the fuel cladding in the fuel assembly. The central tube, as part of the assembly, is attached to the fuel assembly drive mechanism, similar to a conventional control rod drive mechanism in typical NPPs. However, in this case, the whole fuel assembly inside the pressure tube is raised or lowered by a simple drive mechanism, for example, electromagnetic drive or the magnetically coupled electric motor drive. The capability to move selected fuel assemblies serves as a primary means for reactivity control, fuel burnup optimization and as a shutdown mechanism of the reactor, thus obviating the need for control rods and soluble boron. Additionally, the annular configuration allows instrumentation to be inserted into the fuel assembly for measurements, such as neutron flux, core temperature and core pressure or irradiation of materials.

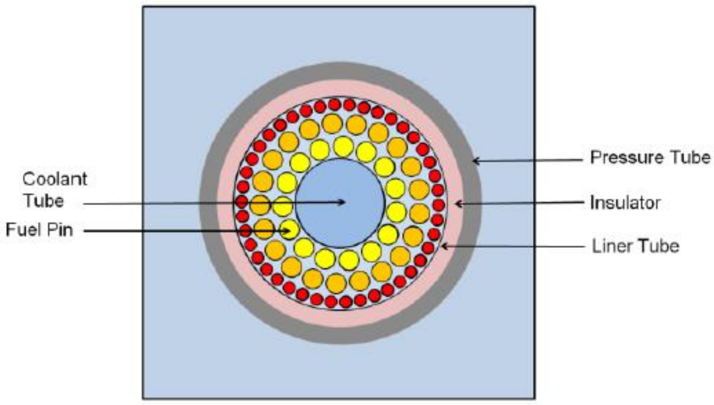
### 4.2.4 Thermal insulator

To maximize the economy of generated nuclear heat and assure the safety of the core, thermal insulation is implemented in the design of LUTHER's fuel channels. The main purpose of having a thermal insulator in a fuel channel is to minimize heat losses from the pressure tubes to the moderator and thermal stresses on the pressure tube caused by the temperature difference between the inner and outer surfaces (Yetisir et al. 2018). Hence, selecting appropriate material and technology for thermal insulation is essential in the LUTHER conceptual core. According to Lo et al. (2016), thermal insulator used in a nuclear reactor environment must possess these following characteristics:

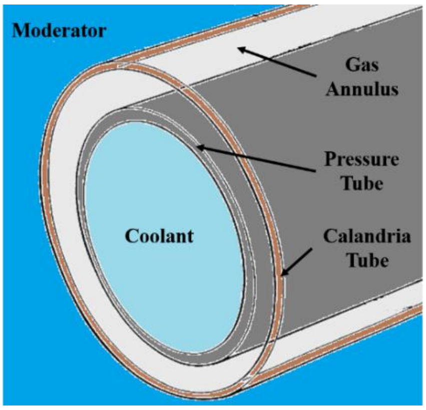
- Low neutron absorption cross-section;
- Neutron irradiation resistance;
- Low thermal conductivity;
- Acceptable mechanical and corrosion-resistant properties;
- High thermal expansion coefficient;
- High strength and high fracture toughness under neutron irradiation;
- Ease of fabrication.

In an SCWR, ceramic yttria-stabilized zirconia (YSZ) is used as a thermal insulator in the fuel channel, as depicted in Figure 4.3a. The YSZ material features an ideal insulator in a high-temperature in-core environment. This material possesses "low neutron absorption, good thermal resistance, moderate mechanical stability under neutron irradiation and a low corrosion rate" (Yetisir et al. 2016). Lo et al. (2016) also affirmed that, among different ceramic materials, YSZ material has the lowest thermal conductivity while satisfying other requirements in order to qualify for an in-core use in a nuclear reactor.

Alternatively, thermal insulation of a fuel channel can also be done by surrounding the pressure tube with a calandria tube and filling with an insulating gas annulus in between, which is typically used in the CANDU fuel channel design. Figure 4.3b shows a schematic view of the design with a calandria tube and gas annulus. In this case, the material selected for the calandria tube can be, for example, Zircaloy-2 in CANDU-6 or Zircaloy-4 in ACR-700, and $CO_2$ gas is used to thermally insulate the hot pressure tube from the cold calandria tube (Dimitrov 2002). For this approach, an additional annulus gas system (AGS) is needed to supply and circulate dry $CO_2$ gas and monitor the moisture content of the gas for leakage from pressure tube or calandria tube (Atomic Energy of Canada Limited 2003).




(a)




(b)

Figure 4.3: Thermal insulation options for the LUTHER pressure-channel design. (a) A ceramic thermal insulator is added inside the pressure tube, referenced from a Canadian SCWR fuel channel (Yetisir et al. 2013). (b) Thermal insulation is done by a calandria tube and a gas annulus, referenced from an ACR fuel channel, adapted from Dimitrov (2002).

Furthermore, having an individual calandria tube surrounding each fuel channel would require more radially spacing clearance between pressure tubes. The only way to fulfill this technical requirement is to increase the fuel channel lattice pitch. Thus, it would result in an additional moderation in the reactor core and, consequently, a bigger size of the calandria vessel. Since the LUTHER core is designed to operate near the optimal H/HM ratio for the maximum of neutron multiplication factor or slightly under-moderated, the additional moderator in the reactor core would result in an over moderation. Therefore, a decrease in $k_\infty$ and positive moderator temperature coefficient are expected (Oka 2014).

As a result of the comparison, the choice of using the ceramic thermal insulator over a calandria tube filled with insulating gas is favored in order to keep the reactor core dimensions as small as possible and keeping the simplified concept in LUTHER design. Thus, a 2-mm ceramic thermal insulator is proposed to be added inside LUTHER's fuel channels in keeping with the simplified design concept, as shown in Figure 4.1. A ceramic silica bonded yttria-stabilized zirconia, also known as zirconium oxide cylinder (ZYC), manufactured by Zircar Zirconia, Inc., is chosen as a material for thermal insulation at this first stage of the study. The material is already available commercially and ready to be used off-the-shelf. Figure 4.4 shows a sample of the ZYC product from the company. According to Zircar Zirconia (2019), the ZYC material features:

- Excellent thermal resistance (0.08 W/mK at 400°C),
- Good dimensional stability and hot strength,
- Low mass ($0.48\ \mathrm{g/cm^3}$ with a porosity of 91%),
- Low heat storage, and
- Machinability to any intricate shapes with tight tolerances.

Figure 4.4: A sample of the zirconium oxide cylinders used as a thermal insulator for high-temperature applications (Zircar Zirconia 2019).

The thickness of the thermal insulator was determined based on simple heat conduction in cylinders, as Cengel (2009, 422) described for a multi-layered cylinder. In this case, there are two layers: ceramic thermal insulator (t) and pressure tube (p). The rate of heat transfer or heat loss ($\dot{Q}$) through the two-layered fuel channel of length $L$ shown in Figure 4.5 with convection on both sides can be expressed as:

$$\dot{Q} = \frac{T_{\infty,1} - T_{\infty,2}}{R_{total}}. \tag{4.1}$$

$R_{total}$ is the total thermal resistance that comprises of coolant, thermal insulator, pressure tube and moderator, which is expressed as:

$$R_{total} = R_{cool} + R_t + R_p + R_{mod} = \frac{1}{h_1 A_1} + \frac{\ln(r_2/r_1)}{2\pi L_r \kappa_1} + \frac{\ln(r_3/r_2)}{2\pi L_r \kappa_2} + \frac{1}{h_2 A_3}, \tag{4.2}$$

where $A_1 = 2\pi r_1 L_r$ and $A_3 = 2\pi r_3 L_r$ are the surface areas of the inner thermal insulator (1) and outer pressure tube (3), respectively. $h_1$ and $h_2$ represent the convective heat transfer of coolant and moderator and $r_1$, $r_2$ and $r_3$ are the radii of the inner and outer thermal insulator and outer pressure tube, respectively. $L_r$ is the length or height of the reactor channel and $\kappa_1$ and $\kappa_2$ are the thermal conductivity of the insulator and pressure tube in the respective order. The thickness of the thermal insulator is obtained as a result of the difference of $r_1$ and $r_2$ by minimizing the heat losses from the pressure tubes into the moderator from equation (4.1).

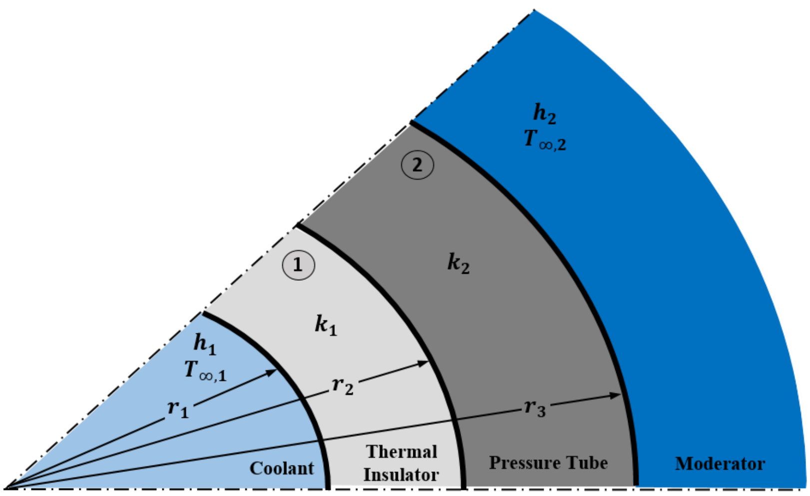



Figure 4.5: Heat transfer through a two-layered composite cylinder (thermal insulator (1) and pressure tube (2)) subjected to convection on both sides (coolant and moderator). Note that the figure is not drawn to scale.

### 4.2.5 Pressure tube

The pressure tube designed for the LUTHER reactor core is made of a zirconium 2.5-wt.% Niobium alloy (Zr-2.5 wt.% Nb), a common material used in the CANDU-type pressure tubes (Dimitrov 2002). The pressure tube is a 5-mm thick cylinder designed to form a pressure boundary to contain the low-pressure coolant at 1.25 MPa and a 54-element fuel assembly with an annular ceramic thermal insulator inside. The given thickness was chosen as an average value based on various literature reviews to assure the integrity of the channel during any transients and to provide adequate strength for end-fitting plugs of the pressure tube.

Unlike CANDU-type reactors, LUTHER pressure tubes are oriented vertically. Each pressure tube is individually connected to thermal collectors, positioned above and below the core, where heat from the primary circuit is transferred to the district heating network via an intermediate water loop. This design choice is made to allow access to an individual fuel assembly in the fuel channel for maintenance and refueling. Additionally, it is also to ensure coolant availability to the core during any accidents, for example, fuel channel rupture and LOCA.

Since the pressure tube forms a pressure boundary for each fuel assembly, this configuration allows the calandria vessel or the moderator tank to be designed for low temperature and low pressure (in other words, atmospheric pressure). Consequently, the need for a high-strength pressure vessel is eliminated (Atomic Energy of Canada Limited 2003). Therefore, the reactor vessel’s wall thickness is reduced, which lowers the costs of fabrication and manufacture and simplifies component quality control (Leppänen 2019).

Furthermore, another highlighted benefit of the pressure-tube-based design is the scalability in the reactor core itself. In the case of an RPV-based reactor, scaling up its thermal power design would require an upgrade for a larger RPV in order to accommodate additional fuel assemblies in the core. This approach could be significantly costly and result in lower material quality for the larger RPV. In contrast, scaling up the LUTHER pressure-channel reactor would simply require additional pressure tubes, enclosing fuel assemblies inside, to be installed in addition to the current ones in the calandria vessel. Hence, the pressure-tube-based design enables a simple upgrade process with associated cost and time savings.

### 4.2.6 Fuel assembly driving mechanism

To ensure the reliability of the fuel assembly driving mechanism, LUTHER is proposed to use a simple conventional control rod driving mechanism, such as electromagnetic drive or the magnetically coupled electric motor drive, which has been used in conventional NPPs. For the case of LUTHER, instead of control rods being driven into the core, the whole fuel assembly is raised or lowered inside the pressure tube at a small increment using electromagnets. In the case of an emergency shutdown, LUTHER design can utilize an inherent passive safety system, for example, gravity, to quickly drop moving fuel assemblies out of the main core by demagnetizing the holding magnets from the system. Thus, an amount of negative reactivity is inserted into the core, which shuts down the reactor. Similarly, in the case of a power cut-off, all moving fuel assemblies are dropped at once by free fall, ensuring the safety of the reactor. In this study, a proposal of a specific driving mechanism design for LUTHER has not been made and is recommended in future studies.

## 4.3 LUTHER reactor core design

### 4.3.1 Calandria vessel

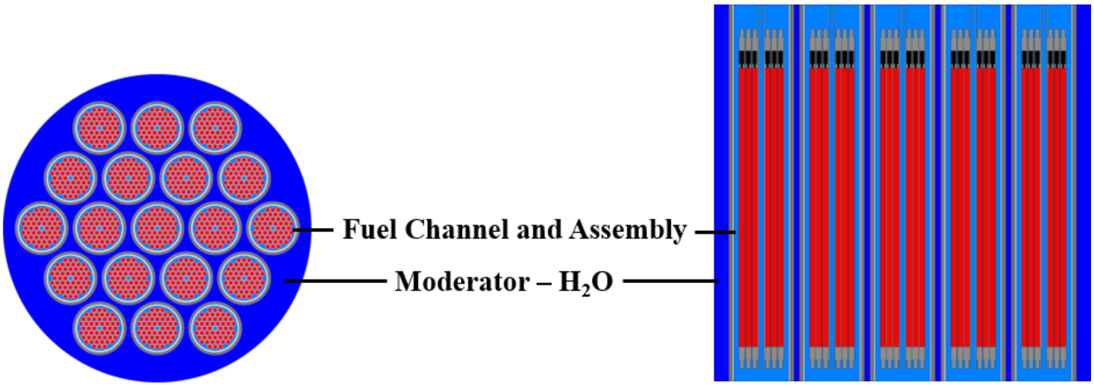
Calandria vessel is a low-pressure moderator tank that provides both a pressurized environment for moderator and a safety barrier for protection against any nuclear accident, radiation or external damage. A typical vessel contains fuel channels, light-water moderator, reactivity control systems and emergency shutdown mechanisms (Yetisir et al. 2016). In a conventional LWR that operates at high pressures and temperatures (7-15.5 MPa and 286-345$^{o}$C, respectively), the thickness of the pressure vessel shell ranges from 152 mm for BWR (Todreas and Kazimi 1990, 31) to 246 mm for European pressurized reactor (EPR) (Leppänen 2019).

Since the operating pressure of the reactor moderator is one atmospheric pressure or 0.101325 MPa, the vessel wall can be a few millimeters thick to hold this moderator inside the tank, in other words, the hydrostatic pressure of the moderator. Leppänen (2019) reported that the required thickness for a maximum design pressure of 1.6 MPa could be 10-11 mm, which is significantly thinner than conventional LWR's RPVs. Therefore, the LUTHER's calandria vessel can theoretically be as thick as 10-11 mm or even thinner. However, the proposal of the final vessel thickness is still needed to be studied and assessed for any additional protections, such as against external threats. Hence, the reduction of calandria vessel thickness implies "significantly lower manufacturing costs, simplified quality control and a wider supply of industrial companies capable of performing the work" (Leppänen 2019).

The calandria vessel of the LUTHER conceptual core is designed to be an open vessel operating at one atmospheric pressure. Figure 4.6 shows both radial and axial schematic views of the calandria vessel for a 2 $MW_{th}$ LUTHER core. The 2 $MW_{th}$ LUTHER core design contains 19 vertically oriented fuel channels, enclosing movable fuel assemblies inside, arranged in a hexagonal lattice. These fuel channels are surrounded by the atmospheric-pressure light-water moderator, which is maintained at 40$^{o}$C for the current analysis. To maintain the moderator temperature, some ground source cooling loops such as a ground-coupled heat exchanger are considered to dissipate unavoidable heat loss from the fuel channels to the nearby ground. Moreover, the choice of 40$^{o}$C moderator in the current design allows for flexibility in manipulating the moderator-to-fuel ratio without pressurizing the calandria vessel, thereby influencing the core reactivity and reactor power.

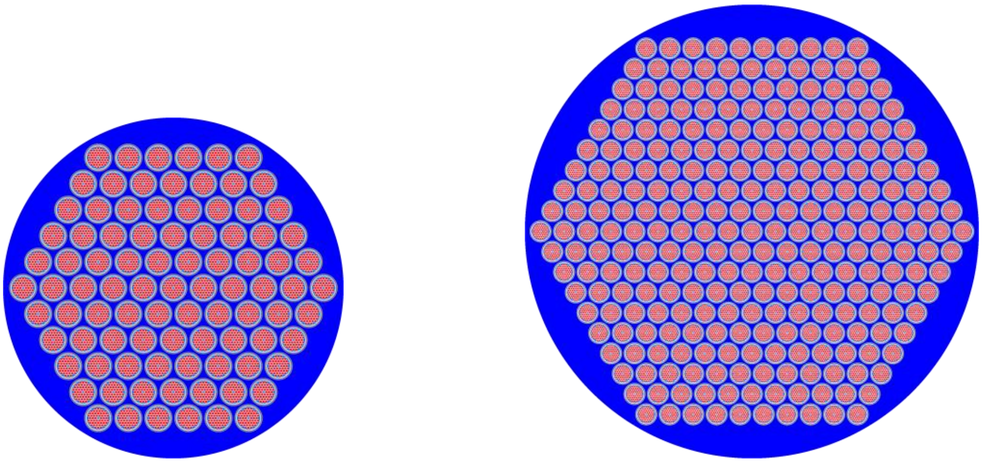
Light water in the calandria vessel also serves as a neutron moderator and a passive heat removal medium. In addition, the calandria vessel is also designed with the capability to drain the moderator, which can act as an alternate means to shut down the reactor, should the selected fuel assemblies fail to move on demand.

The lattice pitch for the fuel channels is determined through iterative reactor physics analyses to obtain optimal neutronic performance. For the present analysis, a 10.5-cm lattice pitch for fuel channels is selected as a compromise between optimal infinite multiplication factor and sufficient radial spacing for the mechanical design of the. Furthermore, 24 $MW_{th}$ and 120 $MW_{th}$ reactor cores are designed with a similar configuration, with each comprises 91 and 271 fuel assemblies, respectively, as shown in Figure 4.7.




(a) (b)

Figure 4.6: Schematic views of the LUTHER 2 $MW_{th}$ core that comprises 19 fuel channels surrounded by light-water moderator: radial view (a) and axial view (b). Note that the reflector region is not included in this study.



(a) (b)

Figure 4.7: Schematic view of the LUTHER 24 $MW_{th}$ core (a) and 120 $MW_{th}$ core (b), each comprises 91 and 271 fuel channels. Note that the reflector region is not included in this study.

### 4.3.2 Neutron reflector

In a thermal reactor, a neutron reflector is desired to improve the neutron economy near the boundary of the core. The reason is in a bare LWR, the migration area is minimal, and thereby there is a significant leakage of neutrons, mainly fast energy neutrons (Reuss 2008, 540). By having a neutron reflector surrounding the core, fast neutrons escaping from the core thermalize in the reflector. Since absorption cross-section in the reflector is smaller than as in the core, thermal neutrons tend to accumulate in the reflector. The behavior of thermal flux in a reflected thermal core can be seen in Figure 4.8. The accumulation of thermal flux in the reflector helps to reduce neutron leakage, flatten the thermal flux distribution in the core, and consequently reduce the critical size and mass of the reactor. (Lamarsh and Baratta 2014, 303-304)

In this current study, the present LUTHER core design does not include a radial or axial reflector region. Further studies on neutron reflector and material selection are recommended in future research in improving the neutronic performance of the LUTHER core.

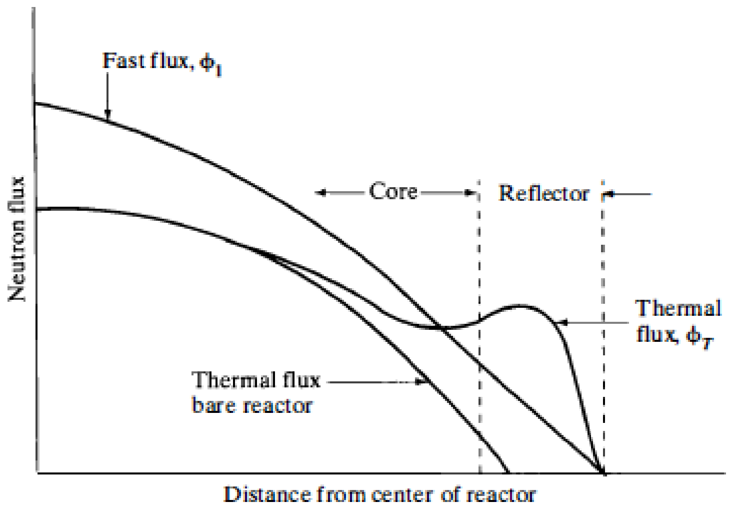



Figure 4.8: Fast and thermal fluxes in a reflected thermal reactor and a bare thermal reactor (Lamarsh and Baratta 2014, 304).

## 4.4 Basic thermal and hydraulic designs of LUTHER

In this section, basic calculations of reactor thermal performance and heat transfers are performed in order to acquire parameters for the design assessment and reactor modeling in the Serpent code. Additionally, the primary coolant flow rate and temperature distributions inside the LUTHER fuel channel and fuel assembly are also calculated. Equations used in the calculations for these analyses are presented.

### 4.4.1 Reactor core thermal performance

Reactor core thermal performance is important when designing and selecting the overall characteristics and suitable safety features for the LUTHER core concept. This thermal performance is dictated by the selected operating temperature and pressure of the reactor. The smallest LUTHER core is designed to produce a 2 $MW_{th}$ power for experiment and demonstration purposes. The thermal performance of commercially deployable powers of 24 $MW_{th}$ and 120 $MW_{th}$ are also studied and calculated for performance and safety assessments. The core thermal performance of LUTHER is described by a variety of terms as follow: linear heat rate $(Q')$, surface heat flux $(Q'')$, core power density $(Q''')$ and core specific power $\left(\frac{Q}{m_U}\right)$. Also, equations used in this study are presented and referenced from Todreas and Kazimi (1990, 22-23; 47-51).

The average linear heat generation rate $(Q')$ in a fuel rod can be obtained as:

$$Q' = \frac{\gamma Q}{nNL_{core}}, \tag{4.3}$$

where

$\gamma$: fraction of recoverable energy from fission reaction deposited in the fuel

$Q$: design thermal power [$kW_{th}$]

$n$: number of fuel elements in a fuel assembly

$N$: number of fuel assemblies in a reactor

$L_{core}$: active length or height of the core [m]

In the present analysis, the fraction of recoverable energy from the fission reaction deposited in the fuel $(\gamma)$ is assumed to be 95%, which is an average fraction in a conventional LWR (Todreas and Kazimi 1990, 42). The other 5% of the fission energy is deposited outside of the fuel, for example, part of that directly heat the moderator water and surrounding structures due to neutrons and gamma radiation.

The average heat flux $(Q'')$ at the interface between a fuel rod and the coolant is obtained as:

$$Q'' = \frac{\gamma Q}{nNL_{core}2\pi r_{co}} = \frac{\gamma Q}{nNL_{core}\pi d_{co}} = \frac{Q'}{\pi d_{co}}, \tag{4.4}$$

where $r_{co}$ and $d_{co}$ are the outer radius and diameter of the fuel clad, respectively, as shown in Figure 4.9 .

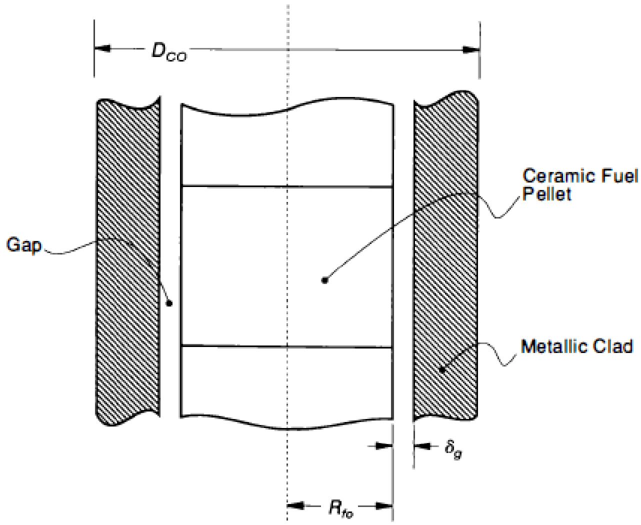



Figure 4.9: Schematic layout of a typical ceramic fuel used in a NPP; referenced from Todreas and Kazimi (1990, 33).

In the LUTHER core, the power density is used as a figure of merit for core thermal performance, ensuring high safety thermal margins of the design. The average power density $(Q''')$ in the core is expressed as:

$$Q''' = \frac{Q}{V_{core}}, \tag{4.5}$$

where $V_{core}$ is the total active core volume of the reactor, which is calculated as follow:

$$V_{core} = \pi r_{core}^2 L_{core} = \frac{1}{4}\pi d_{core}^2 L_{core}. \tag{4.6}$$

To optimize the neutronic performance of the core, the ratio between the equivalent core diameter ($d_{core}$) and active length of the core ($L_{core}$) is kept to be one. Thus, the equivalent core diameter and active length of the core for the LUTHER's hexagonal lattice are computed as follow:

$$d_{core} = L_{core} = \sqrt{\frac{4}{\pi} A_{core}}, \tag{4.7}$$

where $A_{core}$ is the active area of LUTHER core, expressed as:

$$A_{core} = A_{fa} N = \frac{\sqrt{3}}{2} \alpha^2 N. \tag{4.8}$$

$A_{fa}$ is denoted as the active area of one fuel assembly and $\alpha$ is the lattice pitch of the fuel channel, which is shown in Figure 4.10. Lastly, the core specific power $\left(\frac{Q}{m_U}\right)$ representing the amount of energy generated per unit mass of fuel material ($UO_2$) is calculated as:

$$\frac{Q}{m_U} = \frac{Q'}{\pi r_f^2 \rho_{UO_2} f_{UO_2}}, \tag{4.9}$$

where

$r_f$: radius of the outer surface of the ceramic fuel pellet [cm]

$\rho_{UO_2}$: density of the fuel pellet [g/cm$^3$]

$f_{UO2}$: mass fraction of the $UO_2$ fuel

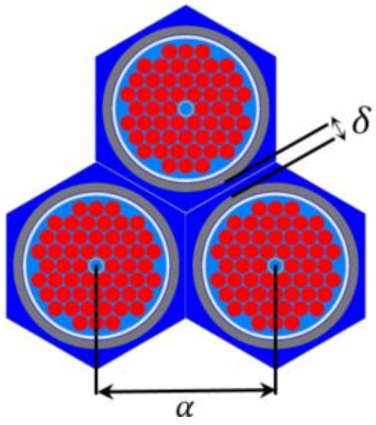



Figure 4.10: Hexagonal lattice fuel channel and assembly configuration of LUTHER.

The mass fraction of $UO_2$ fuel is expressed as:

$$f_{UO_2} = \frac{\omega_{U235}M_{U235} + (1-\omega_{U235})M_{U238}}{\omega_{U235}M_{U235} + (1-\omega_{U235})M_{U238} + 2M_O}, \tag{4.10}$$

where

$\omega_{U235}$: weight fraction of uranium-235 enrichment

$M_{U235}$: molecular weight of uranium-235 [g/mol]

$M_{U238}$: molecular weight of uranium-238 [g/mol]

$M_O$: molecular weight of oxygen [g/mol]

### 4.4.2 Reactor primary coolant system

In this section, characteristics of the primary coolant system inside the LUTHER fuel channel is calculated based on the selected operating conditions of the reactor (in other words, temperature and pressure). Equations used for characterizing the primary coolant systems are presented, which consists of the heat transfer coefficient $(h)$, hydraulic diameter $(D_h)$, Nusselt number $(Nu)$, Reynolds number $(Re)$, Prandlt number $(Pr)$ and mass flow rate $(\dot{m})$.

The heat transfer coefficient of the reactor coolant can be evaluated using the definition of the Nusselt number and the Dittus-Boelter correlation since the LUTHER fuel channel operates in a single-phase turbulent flow (Todreas and Kazimi 1990, 443). The turbulent flow profile can be checked with the obtained Reynolds number of the reactor coolant.

$$Nu = 0.023 Re^{0.8} Pr^{0.4} = \frac{hD_h}{\kappa}, \tag{4.11}$$

where $\kappa$ is the thermal conductivity of the fluid and $D_h$ is the hydraulic diameter. The hydraulic diameter for fuel rods in a triangular lattice configuration can be calculated as:

$$D_h = d\left[\frac{2\sqrt{3}}{\pi}\left(\frac{p}{d}\right)^2 - 1\right], \tag{4.12}$$

where
$d$: diameter of the fuel rod [m]
$p$: lattice pitch of the fuel rods [m]

The Reynolds number and Prandtl number are expressed as:

$$Re = \frac{\rho v D_h}{\mu}, \tag{4.13}$$

$$Pr = \frac{\mu c_p}{\kappa}, \tag{4.14}$$

where
$\rho$: density of the fluid [kg/m$^3$]
$v$: flow velocity of the fluid [m/s]
$\mu$: dynamic viscosity of the fluid [Pa·s]
$c_p$: specific heat capacity of the fluid [J/(kg·K)]

Finally, the mass flow rate of the coolant is calculated as follow:

$$\dot{m} = \frac{Q_r Fr}{c_p \Delta T}, \tag{4.15}$$

where

$Q_r$: thermal power generated from a reactor channel [W]

$Fr$: radial power peaking factor for the whole core

$\Delta T$: temperature difference between the inlet and outlet temperatures [K]

For the current analysis, the highest radial power peaking factor for the whole core is 1.96 that is obtained from the reactor physics calculation in the Serpent code.

### 4.4.3 Temperature distribution in the LUTHER fuel channel

To acquire an accurate neutronic performance of the LUTHER core, the temperature distributions inside the LUTHER fuel element and fuel channel are calculated. Conduction and convection heat transfer calculations were performed for each layer that is made up the LUTHER fuel channel and fuel assembly (a total of seven layers), listed as follow, from the inner-most to the outer-most layer:

- Ceramic fuel pellet,
- Gas gap,
- Fuel clad,
- Coolant,
- Thermal insulator,
- Pressure tube, and
- Moderator.

Figure 4.11 shows different layers inside the LUTHER fuel element and fuel channel in the respective order, as listed above. Table 4.1 tabulates the thermal conductivity and convective heat transfer parameters used for different reactor regions in the LUTHER fuel element and fuel channel.

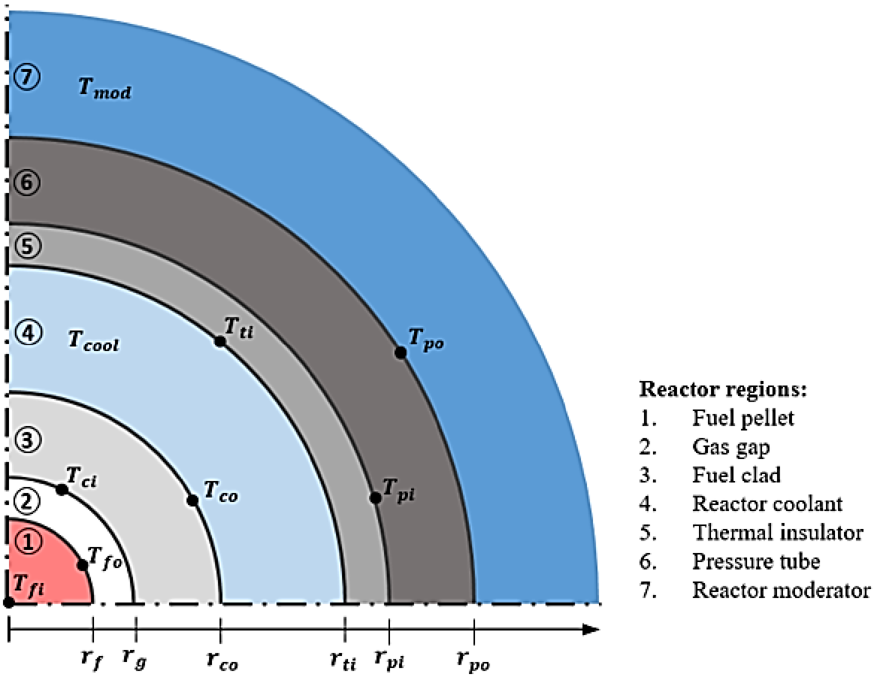



Figure 4.11: Reactor regions of the LUTHER fuel channel and fuel assembly.

Using Newton's law of cooling and Fourier's law, the temperature differences in the coolant $(\Delta T_{cool,1})$, cladding $(\Delta T_c)$, gas gap$(\Delta T_g)$ and the fuel pellet $(\Delta T_f)$ are expressed in term of the core power density as follow, respectively (Akimoto et al. 2016, 259-262):

$$\Delta T_{cool,1} = T_{co} - T_{cool} = \frac{Q'''}{2h_{cool}}\frac{r_f}{r_{co}}; \tag{4.16}$$

$$\Delta T_c = T_{ci} - T_{co} = \frac{Q'''}{2\kappa_c} r_f^2 \ln\left(\frac{r_{co}}{r_{ci}}\right); \tag{4.17}$$

$$\Delta T_g = T_{fo} - T_{ci} = \frac{Q'''}{2\kappa_g} r_f^2 \ln\left(\frac{r_{ci}}{r_f}\right); \tag{4.18}$$

$$\Delta T_f = T_{fi} - T_{fo} = \frac{Q'''}{4\kappa_f} r_f^2. \tag{4.19}$$

The temperature differences in the coolant $\left(\Delta T_{cool,2}\right)$, thermal insulator$(\Delta T_t)$, pressure tube $\left(\Delta T_p\right)$ and the moderator $(\Delta T_{mod})$ are obtained as following in the respective order (Cengel 2009, 422-423):

$$\Delta T_{cool,2} = T_{cool} - T_{ti} = \dot{Q}R_{cool}; \tag{4.20}$$

$$\Delta T_t = T_{ti} - T_{pi} = \dot{Q}R_t; \tag{4.21}$$

$$\Delta T_p = T_{pi} - T_{po} = \dot{Q}R_p; \tag{4.22}$$

$$\Delta T_{mod} = T_{po} - T_{mod} = \dot{Q}R_{mod}. \tag{4.23}$$

Here $\dot{Q}$ is the rate of heat transfer from the coolant to the moderator and $R_{cool}$, $R_t$, $R_p$ and $R_{mod}$ are thermal resistance of coolant, thermal insulator, pressure tube and moderator, respectively, which are calculated from the equations (4.1) and (4.2).

Table 4.1: Thermal conductivity and convective heat transfer parameters for different reactor components used in the LUTHER fuel element and fuel channel.

| **Reactor region** | **Unit** | **Value** |
|---|---|---|
| Fuel ($\kappa_f$) | W/(m·K) | 5.4[a] |
| Gas gap ($\kappa_g$) | W/(m·K) | 0.3 |
| Cladding ($\kappa_c$) | W/(m·K) | 14.8[b] |
| Thermal insulator ($\kappa_{ti}$) | W/(m·K) | 0.08[c] |
| Pressure tube ($\kappa_{pt}$) | W/(m·K) | 17.4[d] |
| Reactor coolant[e] ($h_{cool}$) | kW/(m$^2$·K) | 15.47 / 32.26 / 48.83 |
| Reactor moderator ($h_{mod}$) | W/(m$^2$·K) | 1522.76 |

[a]The fuel thermal conductivity is obtained by using the Westinghouse's temperature-dependent correlation (Todreas and Kazimi 1990, 301). The value was evaluated at an average temperature of 550K.
[b]The thermal conductivity of the ZIRLO fuel cladding is obtained by using the cladding thermal conductivity (CTHCON) correlation given in the Luscher and Geelhood's report (2011, 3.4). The value was evaluated at an average temperature of 460K.
[c]The thermal conductivity of the ceramic ZYC thermal insulator is obtained directly from the manufacture's specification report (Zircar Zirconia 2019). The value was evaluated at a temperature of 400$^o$C.
[d]The thermal conductivity of the pressure tube, made of Zr-2.5 wt.% Nb, is obtained by using Pade's approximation as presented in IAEA's technical report (2006, 293). The value was evaluated at an average temperature of 320K.
[e]The convective heat transfer coefficient for reactor coolant was computed for each design thermal power of LUTHER: 2 $MW_{th}$, 24 $MW_{th}$ and 120 $MW_{th}$, respectively.

# 5 LUTHER MODELING CALCULATIONS AND RESULTS

In this study, reactor modeling and reactor physics calculations for LUTHER core are performed using the Serpent Monte Carlo code. Serpent code is used in this research for calculating core criticality (in other words, multiplication factor), power distribution, fuel burnup and reactivity control.

## 5.1 Serpent Monte Carlo reactor physics code

The Serpent is a multi-purpose Monte Carlo reactor physics code that was developed by the VTT Technical Research Centre of Finland Ltd. (Leppänen et al. 2015). Serpent code is well known for reactor core modeling and traditional reactor physics calculations, including the reactor criticality and burnup calculation, reactivity control and safety analyses. In addition, Serpent code is also capable of performing coupled multi-physics calculations, such as reactor physics simulations coupling with thermal hydraulics code, which is not used in this current study.

## 5.2 Modeling methodology

With the proposed design parameters tabulated in Appendix I, the first LUTHER core design is modeled by the Serpent reactor physics code. The modeling process for LUTHER core is performed in four steps for which there are assessments conducted after each step regards the proposed design and obtained results:

- Preparing material data and design parameters,
- Modeling a general fuel assembly and fuel channel design,
- Modeling a whole reactor core for different design thermal powers, and
- Calculating basic reactor physics parameters and neutronic characteristics of the core, for example, core criticality and reactivity worth of moving fuel assemblies.

The design is analyzed on two levels: 2D single fuel assembly (1) and 2D and 3D reactor core (2). The first level analysis is aimed at optimizing the design parameters concerning the reactivity of the fuel assembly and mechanical design of the channel. Additionally, the power distribution calculation was performed in a fuel assembly. For the second level, the primary objective of this analysis is to determine the feasibility of controlling reactivity by moving selected fuel assemblies and the criticality of the proposed core design for each thermal power.

In these calculations, 100 000 – 500 000 neutrons per cycle were used for a total of 2000 cycles with 50 – 100 inactive cycles. Analyses of the LUTHER designs in this study were calculated at the "hot" operating condition with average temperatures for different reactor core components obtained from thermal-hydraulic calculations and predetermined temperatures for reactor coolant and moderator. The average temperatures used in the first LUTHER model for different reactor components are listed in Appendix VI.

In addition, material data used in the LUTHER modeling is presented in Appendix IV that lists the isotopic compositions for each reactor component. Also, the design configuration and dimension of the referenced VVER fuel rod were used in the modeling of the LUTHER fuel rod, and an illustration of the design is shown in Appendix V.

## 5.3 Fuel channel and fuel assembly design

### 5.3.1 Infinite multiplication factor in a fuel assembly

The infinite multiplication factor of a LUTHER fuel channel containing a fuel assembly is calculated at the design operating temperatures for reactor coolant and moderator. The result is presented as a function of hydrogen-to-heavy-metal (H/HM) ratio (in other words, hydrogen-to-uranium or H/U) in Figure 5.1. In this calculation, the lattice pitch of fuel pins is fixed at 0.96 cm, and the lattice pitch of fuel channels is varied that corresponds to the H/HM ratio. As the lattice pitch of fuel channels enlarges, the H/HM ratio increases (in other words, the amount of moderator increases), as well as the space clearance $\delta$ between pressure tubes, as shown in Figure 4.10. At this first stage of the study, the H/HM ratio in the fuel channel was optimized to approximately 4.06, achieving the maximum $k_{\infty}$. This selection results in a fuel channel lattice pitch of 10.5 cm and an 8-mm spacing clearance between pressure tubes.

In addition, Figure 5.1 also shows the reactivity effect when the moderator of a fuel channel lattice was drained completely, which yields the reactivity change of approximately -9.16 $\%\Delta k/k$ and the H/HM ratio of 1.70. Therefore, the calandria draining looks a promising diverse means of reactor shutdown of the LUTHER, in addition to the primary means of dropping selected fuel assemblies out of the core.

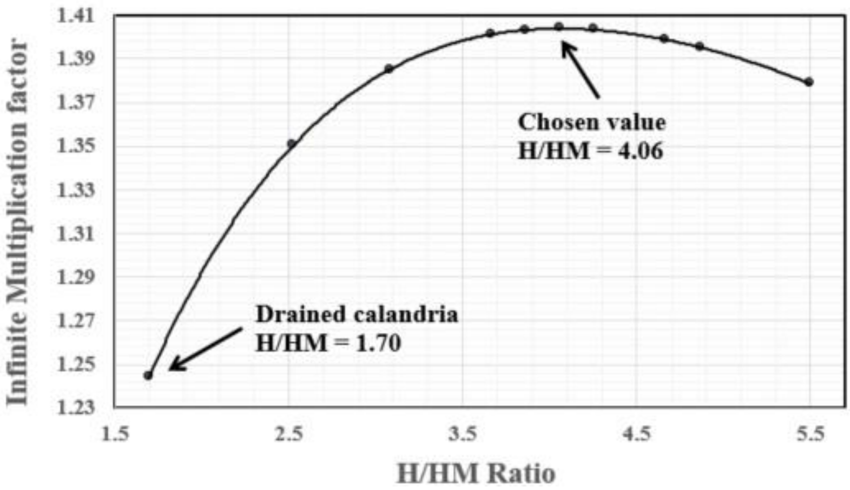


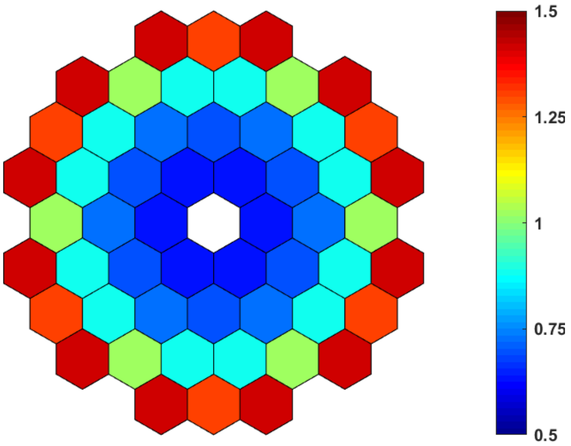

Figure 5.1: $k_\infty$ of a fuel channel as a function of H/HM ratio with a fuel pin lattice pitch of 0.96 cm. The average relative statistical error of the calculation is ±3.00 E-5.

### 5.3.2 Power distribution in a fuel assembly

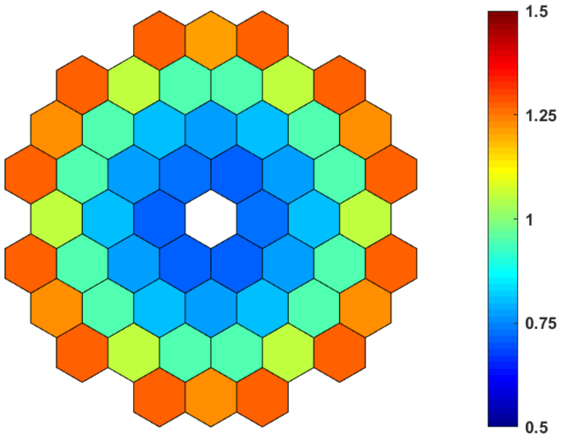
The power distributions in the LUTHER fuel assembly at each fuel burnup step are calculated in the analysis of fuel utilization in the assembly. The normalized power distributions at the beginning-of-cycle (BOC), middle-of-cycle (MOC) and the end-of-cycle (EOC) are presented in Figure 5.2.

The fuel assembly design at this preliminary design phase consisted of identical fuel pins with the same uranium enrichment of 4.95 wt.% and without burnable absorbers (for example, gadolinium). As a result, the power distribution in the LUTHER fuel assembly is non-uniform, and power peaks occur on the fuel pins located at the outer ring of the lattice, which can be seen in all three burnup steps. The highest pin power peak, occurring at the BOC burnup step, is 1.42

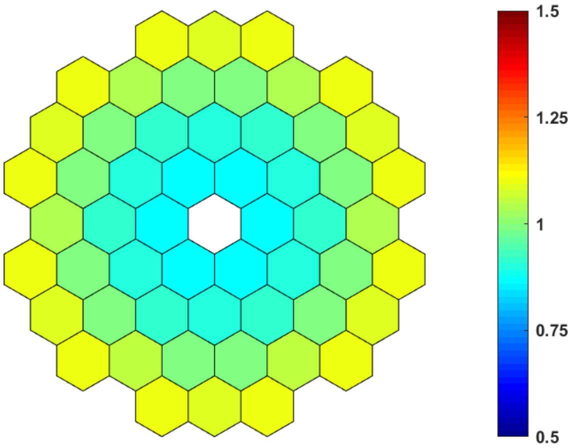
in the current design. Additionally, it is noticed that the power distributions in the LUTHER fuel assembly are becoming flattered along with the burnup.




BOC (0 MWd/kgU)




MOC (18 MWd/kgU)




EOC (38 MWd/kgU)

Figure 5.2: Normalized power distributions in a LUTHER fuel assembly with identical 4.95 wt.% uranium enriched fuel pins at BOC, MOC and EOC. The average relative statistical error of the calculations is ±4.90 E-4.

### 5.3.3 Burnup of a fuel assembly

A fuel burnup calculation was performed to determine the average fuel utilization in a fuel assembly or how often the core needs to be refueled. In the current design, a fuel burnup was calculated with a fuel assembly containing identical pure fuel pins of 4.95 wt.% uranium enrichment without burnable absorbers. The normalization power of a fuel assembly was set at 105 kW in the Serpent code. As a result, shown in Figure 5.3, the infinite multiplication factor of the assembly is presented as a function of fuel burnup. The current fuel burnup at EOC is estimated to be 36 MW/kgU, assuming a single-batch loading, before the reactor core needed to be reloaded with fresh fuel assemblies.

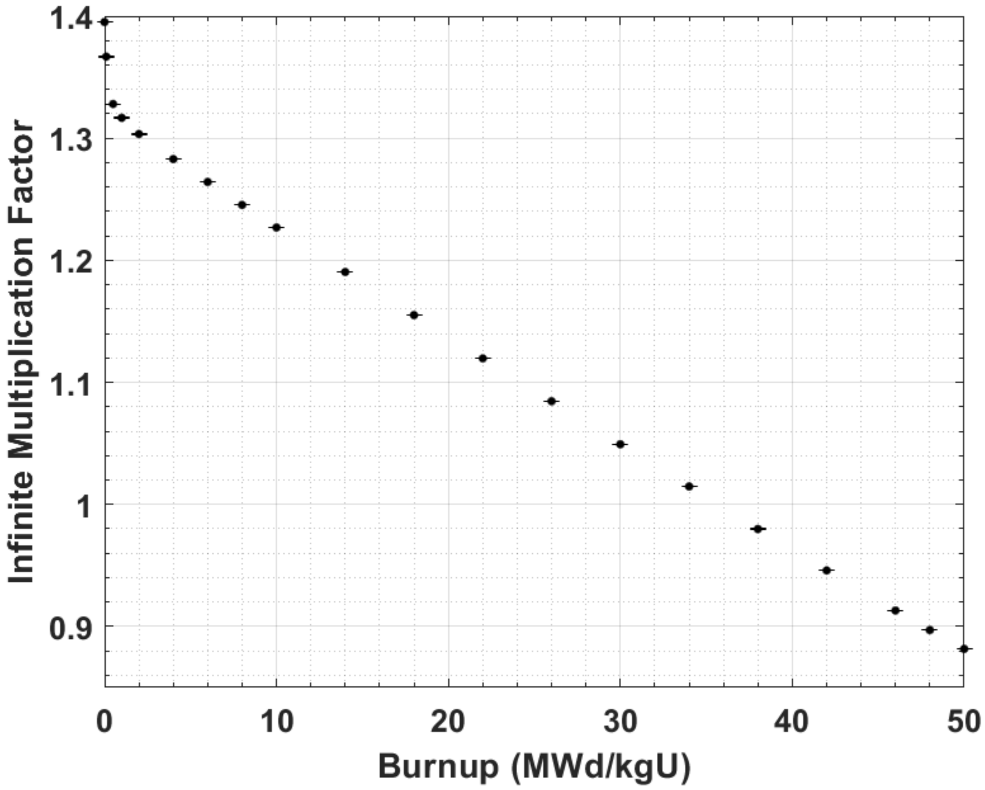



Figure 5.3: $k_\infty$ of a fuel channel with a fuel assembly without burnable absorbers inside as a function of fuel burnup. The average absolute error of the calculation is ±1.84 E-4.

## 5.4 Reactor core design

### 5.4.1 Core criticality

In this section, the effective multiplication factor of the core at BOC is calculated to assess for the LUTHER reactor core criticality of different thermal powers. For the current analysis, identical fuel assemblies with 4.95 wt.% uranium enrichment and without burnable absorbers are used to determine the criticality of each core thermal power. The result of the calculations is tabulated and presented in Table 5.1. As a result, the 2 $MW_{th}$ LUTHER core is subcritical with $k_{eff}$ of about 0.91, and the other two commercial-sized cores are well above the criticality (in other words, supercritical) with $k_{eff}$ of 1.23 and 1.33, respectively.

Additionally, the reactivity of the core, which represents the change in the criticality state of the reactor core, is calculated and presented in Table 5.1. The reactivity $(\rho)$ is defined in terms of $k_{eff}$ as follow:

$$\rho = \frac{k_{eff} - 1}{k_{eff}}. \quad (5.1)$$

Table 5.1: Effective multiplication factor at BOC of LUTHER for 2 $MW_{th}$, 24 $MW_{th}$ and 120 $MW_{th}$ using identical fuel assemblies with 4.95 wt.% uranium enrichment and without burnable absorbers. Absolute errors of the calculations are also presented.

| **Design thermal power [$MW_{th}$]** | 2 | 24 | 120 |
|---|---|---|---|
| **Effective multiplication factor $(k_{eff})$** | 0.910 ± 1.13E-4 | 1.23 ± 1.11E-4 | 1.33 ± 1.14E-4 |
| **Reactivity $(\rho)$ [pcm]** | -9880.21 | 18 890.42 | 24 817.68 |

### 5.4.2 Power distribution in a LUTHER reactor core

The normalized power distributions of LUTHER reactor core at 2 $MW_{th}$, 24 $MW_{th}$ and 120 $MW_{th}$ powers are calculated and presented in Figure 5.4 in the respective order from left to right. The LUTHER core design at this stage consisted of identical fuel assemblies with the same average uranium enrichment of 4.95 wt.% and without burnable absorbers. As a result, the power distribution in the LUTHER core for three thermal powers is also non-uniform. The power peak occurs at the center of the core, and the highest assembly power peak is 1.96.

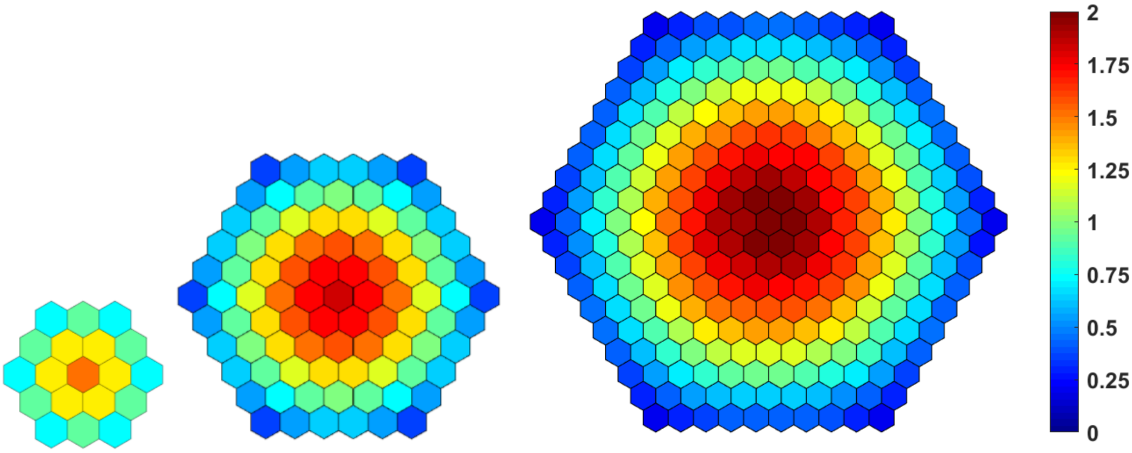



Figure 5.4: Normalized power distributions in 2 $MW_{th}$, 24 $MW_{th}$ and 120 $MW_{th}$ LUTHER cores with identical fuel assemblies from left to right, respectively. The average relative statistical errors of the calculations are ±1.52 E-4, ±3.17 E-4 and ±8.00 E-4 respectively.

## 5.5 Reactor reactivity control by moving fuel assemblies

LUTHER is proposed to utilize moving fuel assemblies for reactivity control and also for fuel burnup compensation and reactor shutdown, replacing conventional control rods and soluble poisons. Three possible configurations of moving fuel assemblies are presented and shown in a 2 $MW_{th}$ reactor core in Figure 5.5. The moving fuel assemblies are highlighted with a light blue color in the fuel channels. The selection of moving fuel assemblies in configuration A is every third of the fuel channels starting at the center of the core. Similarly, the selection in configuration B is also every third of the fuel channels, but starting at off-center (for example, one channel to the right of the center). Configuration C is different among all, and the selection of fuel assemblies is every odd ring of fuel channels in a hexagonal lattice arrangement. A similar configuration is also applied in 24 $MW_{th}$ and 120 $MW_{th}$ cores, which are not shown in this paper.

To calculate for the reactivity effect of moving fuel assemblies (in other words, withdrawing from or inserting into the core region), the selected fuel assemblies were withdrawn below the core at 10% of the increments, corresponding to the active height of the fuel elements. The reactivity worth of selected moving fuel assemblies is calculated at a steady-state condition of the core in the Serpent code. As a result, Figure 5.6 presents the reactivity worth of moving fuel assemblies with a polynomial fit for configuration A in different design thermal powers. The total reactivity worth of those moving fuel assemblies for the configuration A is approximately 17 000 pcm, 12 500 pcm and 12 000 pcm for 2 $MW_{th}$, 24 $MW_{th}$ and 120 $MW_{th}$ cores, respectively.

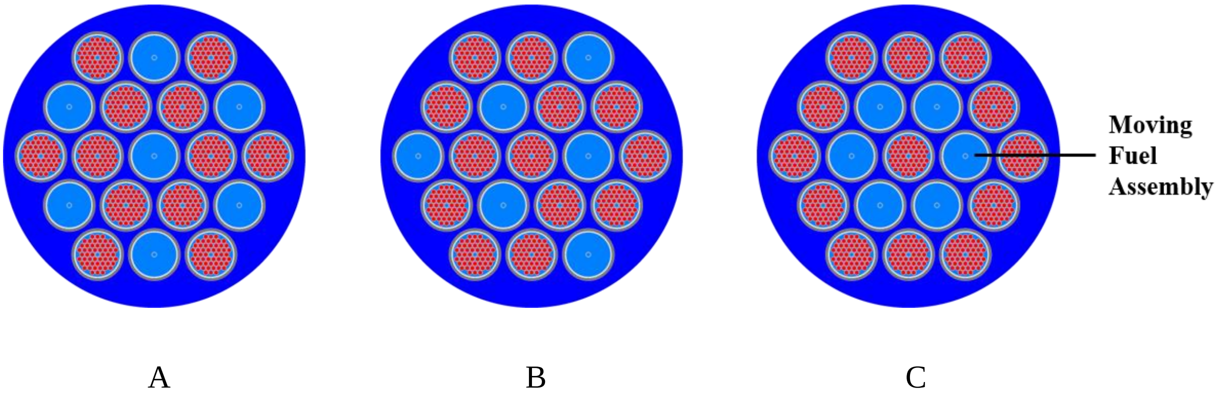



Figure 5.5: Three possible configurations (A, B and C) of moving fuel assemblies shown in a 2 $MW_{th}$ LUTHER reactor core.

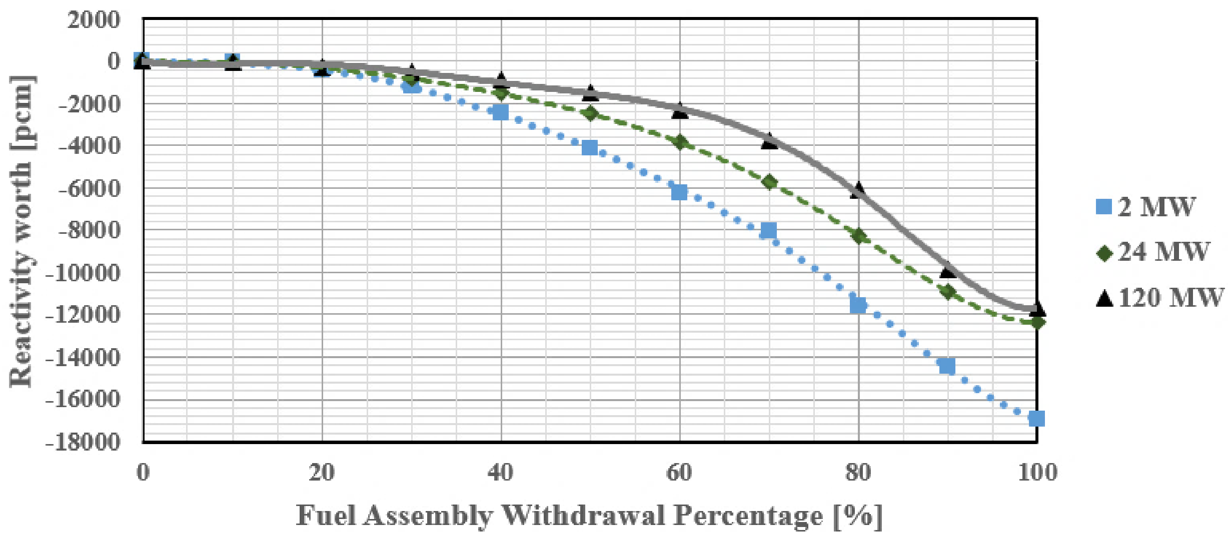



Figure 5.6: Fuel assembly reactivity worth at various thermal powers for the configuration A of moving fuel assemblies. The average relative statistical error of the calculation is ±5.25 E-5.

# 6 DISCUSSIONS OF THE RESULTS

In this chapter, discussions about the proposed design and calculation results are presented, concerning the first conceptual core design of the LUT heating experimental reactor. Further suggestions and recommendations beyond the scopes of the thesis are also presented for future improvement and development of the LUTHER core concept.

## 6.1 Fuel channel and assembly design

In the current study and analysis, the lattice pitches of fuel pins and fuel channels are chosen to be 0.96 cm and 10.5 cm, respectively. The result presented in Figure 5.1 shows a tight optimum range of H/HM ratio, which induces a problem of choosing possible options for a fuel channel lattice pitch. Due to the high moderation power and large neutron absorption characteristics of "cold" light-water moderator, the current LUTHER design is limited in the selection of the fuel channel lattice pitch. Thus, the minimum spacing clearance between pressure tubes is also limited by the selected lattice pitch. Consequently, the current spacing clearance between the pressure tubes is 8 mm, which is potentially inducing problems for the mechanical design and fuel channel assembling.

One possibility to increase the spacing clearance between pressure tubes while maintaining the optimal neutronic performance of the core is installing separated moderator tubes in between the fuel channels. These moderator tubes can be filled with the light-water moderator or air to manipulate the hydrogen-to-fuel ratio of the fuel assembly. Thus, fuel channels can be theoretically placed slightly further away from each other. However, additional materials and operating systems required for these moderator tubes would be needed; thus, contrasting with the simple concept of LUTHER reactor core.

Another possible solution to this spacing problem would be to optimize the current configuration of the fuel assembly lattice. Fuel elements can be rearranged in a tighter configuration (for example, circular lattice). The tighter configuration would reduce the excess amount of neutron moderator inside the channel, thereby enlarging the optimum range of H/HM ratio. Further explorations and studies are recommended for considering these options.

In addition, the reactivity coefficient is an important parameter for the safety and stability of the reactor operation. The current LUTHER fuel assembly is designed to operate at the optimum moderator-to-fuel ratio at its “hot” operating condition. Consequently, it might result in positive reactivity feedbacks at the “cold” operating condition of the core due to changes in material densities (mainly in light-water moderator and coolant) as proportional to temperatures, thereby inducing over-moderation. To provide the safety analysis of the LUTHER fuel assembly, calculations of changes in reactivity caused by temperature are necessary and recommended in the future development.

Furthermore, the fuel assembly power distribution was calculated at different burnup steps and presented in Figure 5.2. The result shows an ununiformed power distribution in the fuel assembly. The causes are believed to be due to the semi-symmetrical geometry of the assembly lattice and the tighter fuel pin configuration from the modified RWFA design. Hence, there is a substantial lack of neutron moderation (in other words, light-water coolant) in the assembly’s flow sub-channels, compared to the outer edges of the assembly. Thereby, the current design results in power peaking at the outer ring of the lattice. Consequently, the current fuel burnup or fuel utilization is not optimal. Therefore, the design is needed to be optimized in the future, using either fuel pins with different enrichments or with gadolinium doped fuel pins to flatten the power distribution in the fuel assembly.

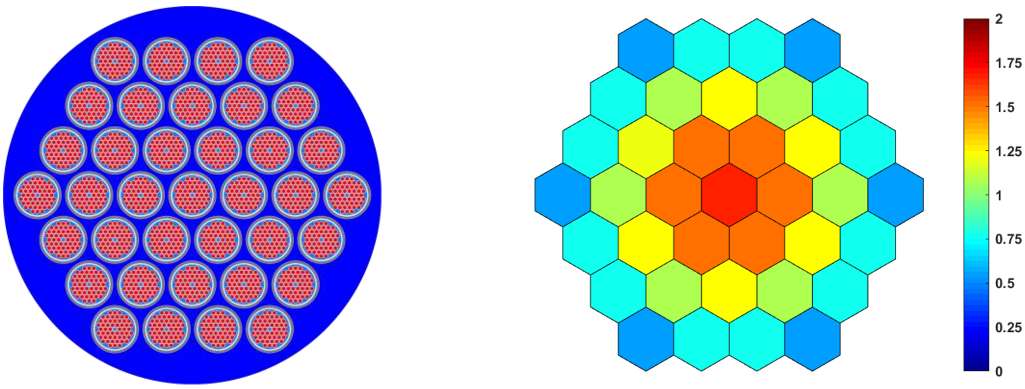
Preliminary fuel assembly burnup calculations were also performed in the assessment of fuel utilization and the fuel cycle length. The preliminary result, presented in Figure 5.3, shows a fuel assembly burnup of 36 MWd/kgU at EOC, which is quite moderate compared to large reactors (for example, 60 MWd/kgU). Moderate burnup in the LUTHER core may be acceptable because it leads to safer spent fuel due to smaller activity inventory present. However, longer fuel cycle length might be obtained with optimized fuel shuffling during refueling, for example. Further studies and assessments on fuel cycle length are recommended in the future.

## 6.2 Reactor core design

In the design of LUTHER reactor cores, the effective multiplication factor was calculated for different thermal powers to determine the reactor criticality. Using the same fuel assemblies with an average fuel enrichment of 4.95 wt.% and no burnable absorbers present, reactor criticalities of three design thermal powers were obtained and presented in Table 5.1. As a result, the 24 $MW_{th}$ and 120 $MW_{th}$ reactor cores show promising operable designs with an adequate amount of excess reactivity required for the reactor to operate over a period of time. Nevertheless, the demonstration-sized reactor of 2 $MW_{th}$ results in a subcritical condition (in other words, $k_{eff} < 1$), which means that modifications in the core design are needed in order to make the reactor critical. Some of the possible ways to make the 2 $MW_{th}$ core reaching criticality that can be implemented in future studies are:

- Using thick neutron reflector (for example, light water, aluminum or stainless steel),
- Increasing the fuel enrichment up to 20 wt.%,
- Increasing the size of the core lattice and core power rating, and
- Redesigning the current design of the fuel assembly lattice.

Among those possible options, it is recommended to increase the core power to 6 $MW_{th}$ for the LUTHER demonstration reactor, while keeping the current design of the fuel assembly. Thus, the LUTHER demonstration reactor contains 37 fuel assemblies, as shown in Figure 6.1a, which makes the core size slightly larger. With the same fuel assembly design used in the core, as described before, the effective multiplication factor of the 6 $MW_{th}$ core was calculated and resulted in $k_{eff} = 1.08 \pm 1.17\text{E-}4$, which is operable for the demonstration reactor. A rough calculation of the core power distribution was performed and presented in Figure 6.1b, exhibiting a similar power peaking distribution as other cores do.




(a) (b)

Figure 6.1: (a) Schematic view of the 6 $MW_{th}$ LUTHER core with 37 fuel assemblies and no reflector region. (b) Normalized power distribution in a 6 $MW_{th}$ LUTHER core with identical fuel assemblies; the average relative statistical error of the calculation is ±2.10 E-3.

Furthermore, the core power distribution on the same design was also calculated for different thermal powers, including 2 $MW_{th}$, 24 $MW_{th}$ and 120 $MW_{th}$. The results are presented in Figure 5.4, showing a concentrated power peaking in the center of the core, with the highest-peaking value of 1.96. For the best economy and optimal design of the reactor core, the flattened shape of the power distribution in the core is desired. At this preliminary design phase, the optimizing phase for the first design proposal is not within the scope of the study. Nevertheless, possible considerations are presented in order to achieve optimal reactor performance and economy. Some possible ways to flatten the power in the core are:

- A multi-batching fuel loading pattern (in other words, three-batch loading),
- Core neutron reflector,
- Gadolinium doped fuel matrix, and
- Fuel channels with different average uranium enrichments.

Moreover, reactor physics calculations concerning reactivity feedbacks (for example, fuel and coolant temperature feedbacks) and other reactor core behaviors such as core burnup were not calculated and assessed in this current analysis. It is highly recommended to investigate and study in-depth in future studies in order to ensure the safety of the LUTHER reactor.

In addition, the first core design dimensions and performance parameters, as presented in Appendices I and II, show a promising feasible design of a low-temperature nuclear reactor for supplying district heating in Finland. Other similar LTNRs' design parameters that are relevant to the development of LUTHER are presented in Appendix III, which can be used for design comparison purposes.

## 6.3 Reactor reactivity control

One of the few unique features in the LUTHER core design concept is the use of movable fuel assemblies as a primary means to control reactivity, replacing the use of control rods and soluble poisons in a typical NPP. The use of movable fuel assemblies is aimed not only for reactivity control but also for fuel burnup compensation and reactor shutdown. To assess for the feasibility of this feature, several calculations in the Serpent code were made to determine the reactivity worth of selected moving fuel assemblies, as discussed previously. Reactivity calculations were done for the configuration A's moving fuel assemblies for all three design thermal powers. Selected moving fuel assemblies in configuration A can be seen in Figure 5.5. As a result, shown in Figure 5.6, the reactivity worth of moving fuel assemblies exhibits a similar reactivity effect as control rods would provide. Additionally, the total reactivity worth of the moving fuel assemblies in configuration A shows promising feasibility of controlling the core reactivity by moving those assemblies inside the pressure tubes.

As the primary means of controlling reactivity, it is essential to study and understand more about the neutronic behavior and reactor dynamics of moving fuel assemblies in various configurations. Also, fuel behavior analyses for such operation would be needed, ensuring the overall safety of the fuel in operation. Further investigations on other different configurations and criteria in selecting moving fuel assemblies are recommended for the normal operation of the reactor. In addition, analyses of potential accidents or failures of the reactivity control system are also recommended for future studies.

Furthermore, movable fuel assemblies can also be used for fuel burnup compensation and also for reactor shutdown during a scheduled off-line period or for an emergency. These multi-functions of moving fuel assemblies are the key to simplifying the operating reactor systems required for an LTNR or a nuclear district heating reactor, thereby lowering the costs of materials and minimizing the probability of system failures. Thus, for example, moving fuel assemblies can be categorized into three main groups: the safety assemblies, the shim assemblies and the regulating assemblies. The proposed classification is based on the purpose of the moving fuel assemblies, which is similar to a typical control rod classification in conventional NPPs.

On another hand, controlling the level of the moderator (in other words, the height of the moderator) in the calandria vessel also can be used as a secondary means of reactivity control and for a reactor shutdown. As presented in Figure 5.1, the preliminary calculation shows the reactivity effect of a complete draining of the moderator in a fuel assembly lattice, exhibiting a promising diverse means of reactor shutdown for LUTHER. Therefore, varying the height of the moderator in the calandria vessel also can be theoretically used to control core reactivity if desired. Further investigations would be recommended to ensure the reliability of the systems and also for the thermal safety margins of the fuel assemblies.

# 7 CONCLUSIONS

In conclusion, the LUTHER pressure-channel reactor is a feasible concept. Scalable and modular designs have considerable potential in decarbonizing the district heating sector and are needed to meet the EU and Finland's ambitious climate goals by 2030 and 2035, respectively. The preliminary proposed design shows reasonable dimensions and design parameters for a nuclear district heating reactor. This work provides an early conceptual understanding for a light-water pressure-channel reactor with the unique feasible feature of moving fuel assemblies, replacing control rods and soluble boron in reactivity control. Further studies and assessments are needed to understand and improve the first LUTHER design, and to complete the thermal-hydraulic system design to implement robust inherent safety characteristics.

Future studies include improvement and optimization of the fuel assembly design, as well as the fuel channels, in order to maximize the neutronic economy while providing an adequate spacing clearance required for mechanical design and thermal hydraulics. From the preliminary analysis results, redesigning the fuel assembly is likely considered with few possible changes, namely, lattice configuration (for example, circular vs. hexagonal), design dimensions and the number of fuel pins per assembly. Additionally, flattening assembly power distribution is also desirable, and implementations of possibly with gadolinium doped fuel pins and pins with different enrichments are considered.

In addition to optimizing the fuel assembly and fuel channel design, investigations and studies concerning reactivity feedbacks and reactor core behavior (for example, fuel burnup, power distribution, reactivity control and shutdown mechanisms) are necessary to develop the LUTHER conceptual core further. Also, flattening the power distribution in the core would be essential, and few possible considerations are a multi-batch loading pattern (for example, three batches), core reflector, gadolinium doped fuel matrix, and fuel channels with different average uranium enrichments. Furthermore, for future work, thorough studies and understanding the neutronic behavior and reactor dynamics of moving fuel assemblies for reactivity control, reactor operation and reactor shutdown are essential in the development of the LUTHER core, assuring the overall feasibility and safety of the reactor.

# 8 SUMMARY

Due to the current trend of consumption and production of energy in the heating and cooling sector in the EU and Finland, there is a keen demand for emission-free heating energy. LUT has started conceptual designing of a dedicated district heating reactor with the aims of cost-effectiveness, modularity, simplification and safety. LUT heating experimental reactor (LUTHER) is a scalable light-water pressure-channel reactor designed to supply district heating in Finland. In the LUTHER concept, reactor power control is achieved by moving selected fuel assemblies in and out of the core. This allows complete elimination of control rods and soluble poisons, yielding a major simplification of the reactor concept.

A small LUTHER core of 2 $MW_{th}$ power is designed to experiment and demonstrate the novel means of reactivity control and feasibility of a pressure-channel district heating reactor. LUTHER core can be scaled up to 24 $MW_{th}$ and 120 $MW_{th}$ for commercial uses by simply adding additional fuel channels, thereby maintaining the cost competitiveness of nuclear heating. Furthermore, the cycle length of fuel is a significant cost factor in the operation of LUTHER. The fuel cycle length obtained from the current design is quite moderate, and a longer fuel cycle than in large power reactors also might be achieved.

The proposed design and features of LUTHER are presented in the paper, which is supported by the first core design calculations that show the feasibility of the new core design for a nuclear district heating reactor and its reactivity control by moving fuel assemblies. However, the 2 $MW_{th}$ core seems too small to be feasible as an operating reactor; the LUTHER demonstration version needs to be somewhat larger to be properly operable. Recommendation of increasing the core power to 6 $MW_{th}$ is proposed to make the LUTHER demonstration reactor operable.

Scalable and modular LUTHER pressure-channel designs have considerable potential in decarbonizing the district heating sector in Finland. This work provides an early conceptual understanding of the LUTHER core design concept and shows the feasibility of its unique feature of movable fuel assemblies. Furthermore, assessments and recommendations in different aspects, including both the design and the reactor physics calculations, are also presented for the future development of the LUTHER conceptual core design.

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**Basic LUTHER core, fuel channel and fuel assembly design parameters for design powers of 2 $MW_{th}$, 6 $MW_{th}$, 24 $MW_{th}$ and 120 $MW_{th}$**

| | |
|---|---|
| ***Reactor core*** | |
| Design thermal power [$MW_{th}$] | 2 / 6 / 24 / 120 |
| Equivalent core diameter [m] | 0.48 / 0.67 / 1.05 / 1.82 |
| Active core height [m] | 0.48 / 0.67 / 1.05 / 1.82 |
| Number of fuel assemblies | 19 / 37 / 91 / 271 |
| Linear heat rate (average) [kW/m] | 3.853 / 4.254 / 4.411 / 4.292 |
| Surface heat flux (average) [$kW/m^2$] | 134.1 / 148.1 / 153.6 / 149.4 |
| Core power density (average) [kW/l] | 22.94 / 25.32 / 26.26 / 25.55 |
| Core specific power (average) [kW/kgU] | 8.688 / 9.591 / 9.947 / 9.677 |
| Mass inventory of $UO_2$ fuel [tons] | 0.25 / 0.65 / 2.60 / 13.36 |
| ***Heat transport system*** | |
| Reactor coolant pressure [MPa] | 1.25 |
| Reactor coolant inlet / outlet temperature [$^oC$] | 150 / 180 |
| Reactor moderator pressure [MPa] | 0.101325 |
| Reactor moderator temperature [$^oC$] | 40 |
| Single channel flow rate (average) [kg/s] | 0.807 / 1.24 / 2.02 / 3.39 |
| ***Fuel channel*** | |
| Pressure tube inner diameter [cm] | 8.7 |
| Pressure tube thickness [mm] | 5 |
| Thermal insulator inner diameter [cm] | 8.2 |
| Thermal insulator thickness [mm] | 2 |
| Fuel channel pitch [cm] | 10.5 |
| Thermal power output (average) [kW/channel] | 105.3 / 162.2 / 263.7 / 442.8 |
| ***Fuel assembly*** | |
| Number of fuel rods | 54 |
| Fuel pellet diameter [mm] | 7.844 |
| Fuel cladding thickness [mm] | 0.5715 |
| Fuel rod outer diameter [mm] | 9.144 |
| Fuel rod lattice pitch [cm] | 0.96 |
| Enrichment of the fuel (95% TD) [%] | 4.95 |
| Number of central tubes | 1 |
| Central tube inner / outer diameter [mm] | 7.2 / 9.6 |

**Basic thermal-hydraulic parameters of the LUTHER conceptual core for design powers of 2 $MW_{th}$, 6 $MW_{th}$, 24 $MW_{th}$ and 120 $MW_{th}$**

| Parameter | LUTHER | | | |
|---|---|---|---|---|
| ***Reactor core*** | | | | |
| Thermal power [$MW_{th}$] | 2 | 6 | 24 | 120 |
| No. of fuel assemblies / fuel rods | 19 | 37 | 91 | 271 |
| No. of fuel rods in a fuel assembly | 54 | 54 | 54 | 54 |
| Active core height [m] | 0.4806 | 0.6707 | 1.0518 | 1.8151 |
| ***Primary coolant system*** | | | | |
| Ave. thermal power / channel [kW] | 105.26 | 162.16 | 263.74 | 442.80 |
| Hydraulic diameter of fuel bundles [cm] | 0.19694 | 0.19694 | 0.19694 | 0.19694 |
| Coolant mass flow rate / channel [kg/s] | 0.807 | 1.24 | 2.02 | 3.39 |
| Coolant flow area of a fuel bundle [$cm^2$] | 7.6376 | 7.6376 | 7.6376 | 7.6376 |
| Ave. coolant flow velocity [m/s] | 1.170 | 1.802 | 2.932 | 4.922 |
| Prandtl number | 1.05593 | 1.05593 | 1.05593 | 1.05593 |
| Reynolds number | 12 619 | 19 440 | 31 617 | 53 084 |
| Nusselt number[a] | 44.88 | 63.41 | 93.57 | 141.63 |
| Coolant heat transfer coefficient [$W/(m^2\cdot K)$] | 15 471.8 | 21 861.5 | 32 259.4 | 48 830.4 |
| ***Moderator system*** | | | | |
| Ave. heat loss to moderator / channel [kW] | 1.15 | 1.15 | 1.15 | 1.15 |
| Hydraulic diameter of fuel channels [cm] | 2.8328 | 2.8328 | 2.8328 | 2.8328 |
| Moderator mass flow rate [kg/s] | 1.05 | 2.04 | 5.01 | 14.91 |
| Moderator flow area of a fuel channel [$cm^2$] | 410.0 | 798.5 | 1963.9 | 14 913.6 |
| Ave. moderator flow velocity [m/s] | 0.0257 | 0.0257 | 0.0257 | 0.0257 |
| Prandtl number | 4.3275 | 4.3275 | 4.3275 | 4.3275 |
| Reynolds number | 1106.25 | 1106.25 | 1106.25 | 1106.25 |
| Grashof number ($\times 10^7$) | 3.98 | 3.98 | 3.98 | 3.98 |
| Rayleigh number ($\times 10^8$) | 1.72 | 1.72 | 1.72 | 1.72 |
| Nusselt number[b] | 68.4 | 68.4 | 68.4 | 68.4 |
| Moderator heat transfer coefficient [$W/(m^2\cdot K)$] | 1522.76 | 1522.76 | 1522.76 | 1522.76 |

[a]The Dittus-Boelter correlation was used to approximate the Nusselt number of the coolant.
[b]The Nusselt number for the moderator was approximated by using the Churchill and Chu correlation.

**Main design parameters of the SECURE, NHR-5, NHR-200II, DPR-3 and DPR-6 reactors**

| Parameter | SECURE[a] | NHR-5[b] | NHR-200II[c] | DPR-3[d] | DPR-6[d] |
|---|---|---|---|---|---|
| Reactor type | PWR | iPWR | iPWR | Pool-type | Pool-type |
| Thermal power [MW] | 200 | 5 | 200 | 120 | 200 |
| Primary system pressure [MPa] | 0.7 | 1.37 | 8 | 0.29 | 0.44 |
| Core inlet / outlet temperature [$^{o}$C] | 90 / 115 | 146 / 186 | 232 / 280 | 80 / 110 | 123 / 132 |
| Number of fuel assemblies | 144 | 12 (A) + 4 (B) | 208 | 205 | 81 |
| Number of fuel rods in assemblies | 60 | 96 (A) + 35 (B) | 77 | 60 | 208 |
| Active core height / diameter [m] | 1.94 / 1.8 | 0.69 / 0.57 | 2.1 / 2.2 | 1.1 / 1.74 | 1.4 / 2.034 |
| Fuel rod lattice pitch [mm] | 15.0 | - | 13.3 | 13.4 | 13.3 |
| Enrichment of initial core [wt.%] | 2.58 | 3.0 | 1.8 / 2.67 / 3.4 | 1.3 / 1.8 / 2.4 / 3.0 | 1.8 / 2.4 / 3.0 |
| Core flow rate [t/h] | 6840 | 107 | - | 3420 | 18800 |
| Inventory of $UO_2$ [t] | 13 | 0.51 | 16.87 | 7.715 | 13.45 |
| Ave. linear heat rate [kW/m] | 2.7 | 5.6 | - | 8.87 | 8.48 |
| Core power density [kW/l] | - | 26 | 16.87 | 45.6 | 44 |
| Power density in fuel [kW/kgU] | 15 | - | 22.52 | - | - |
| Reactivity control / reactor shutdown systems[e] | BA, SB, BSS | BA, CR, SB | BA, CR, SB | CR, BA | CR, BA |
| Primary coolant system | Forced circulation | Forced circulation | Natural circulation | Forced circulation | Forced circulation |
| Residual heat removal system | Natural circulation | Natural circulation | Natural circulation | Natural circulation | Natural circulation |

[a]Referenced from Nilsson and Hannus (1978), Gransell and Höglund (1978) and Leppänen (2019).
[b]Referenced from Dafang et al. (1997) and Yajun et al. (2003).
[c]Referenced from Dong et al. (2018) and Li et al. (2019).
[d]Referenced from Jiafu et al. (1998) and Leppänen (2019).
[e]BA stands for burnable absorbers; CR stands for control rods; SB stands for soluble boron; BSS stands for boron steep spheres.

**Material data used in the Serpent Monte Carlo code modeling of LUTHER**

Isotopic compositions for 2.5-4.95 wt.%-enriched $UO_2$ fuel pellet; referenced from McConn Jr. et al. (2011)

| **Isotope** | **Wt.%** | | | | | | **Remarks** |
|---|---|---|---|---|---|---|---|
| | **2.5%** | **3.0%** | **3.5%** | **4.0%** | **4.5%** | **4.95%** | |
| $^{16}O$ | 11.8529 | 11.8536 | 11.8543 | 11.8551 | 11.8558 | 11.8564 | ρ = 10.412 g/cm$^3$ (95% theoretical density) |
| $^{235}U$ | 2.2037 | 2.6444 | 3.0851 | 3.5258 | 3.9665 | 4.3631 | |
| $^{238}U$ | 85.9434 | 8.5020 | 85.0606 | 84.6191 | 84.1777 | 83.7805 | |

Isotopic composition for ZIRLO$^{TM}$ alloy material (fuel cladding and central tube); referenced from Stuckert et al. (2011)

| **Element** | **Wt.%** | **Remarks** |
|---|---|---|
| O | 0.112 | ρ = 6.55 g/cm$^3$ |
| Cr | 0.0055 | |
| Fe | 0.11 | |
| Zr | 97.7675 | |
| Nb | 1.0 | |
| Sn | 1.0 | |
| Hf | 0.005 | |

Isotopic composition for light water ($H_2O$) (reactor coolant and moderator); referenced from McConn Jr. et al. (2011)

| **Element** | **Wt.%** | **Remarks** |
|---|---|---|
| H | 11.1894 | ρ = 0.90284 g/cm$^3$ (165$^o$C, 1.25 MPa)<br>ρ = 0.99222 g/cm$^3$ (40$^o$C, 1 atm) |
| O | 88.8106 | |

Isotopic composition for Zircaloy-2.5 wt.% Nb alloy material (pressure tube); referenced from IAEA (2008)

| Element | Wt.% | Remarks |
|---|---|---|
| C | 0.02 | $\rho = 6.57\ g/cm^3$ |
| N | 0.003 | |
| O | 0.05 | |
| Al | 0.004 | |
| Si | 0.004 | |
| Ti | 0.003 | |
| Fe | 0.015 | |
| Ni | 0.007 | |
| Zr | 97.3940 | |
| Nb | 2.5 | |

Isotopic composition for silica bonded yttria stabilized zirconia (YSZ) material (thermal insulator); referenced from Zircar Zirconia (2019)

| Chemical Composition | Wt.% | Remarks |
|---|---|---|
| $ZrO_2$ | 84.0 | $\rho = 0.48\ g/cm^3$ |
| $Y_2O_3$ | 10.0 | |
| $SiO_2$ | 5.0 | |
| $HfO_2$ | 1.0 | |
| **Element** | | |
| O | 26.753798 | |
| Si | 2.337152444 | |
| Y | 7.874397701 | |
| Zr | 62.18667325 | |
| Hf | 0.84797861 | |

Isotopic composition for air; referenced from McConn Jr. et al. (2011)

| Element | Wt.% | Remarks |
|---|---|---|
| C | 0.0124 | ρ = 0.001205 g/cm$^3$ |
| N | 75.5268 | |
| O | 23.1781 | |
| Ar | 1.2827 | |

Isotopic composition for Zircaloy-4 material (fuel end plugs, fuel nozzles and gap and spring plugs); referenced from McConn Jr. et al. (2011)

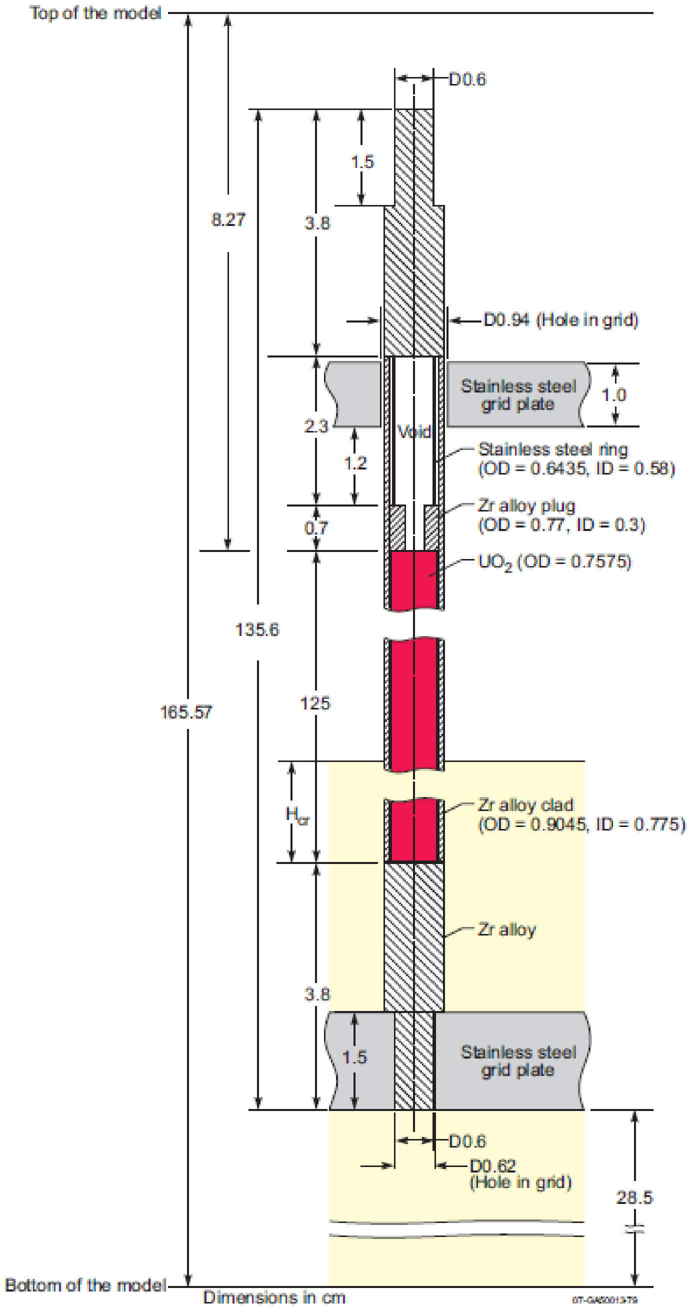
| Element | Wt.% | Remarks |
|---|---|---|
| O | 0.1196 | ρ = 6.56 g/cm$^3$ |
| Cr | 0.0997 | |
| Fe | 0.1994 | |
| Zr | 98.1858 | |
| Sn | 1.3955 | |

Isotopic composition for Stainless Steel 302 material (internal fuel rod spring); referenced from McConn Jr. et al. (2011)

| Element | Wt.% | Remarks |
|---|---|---|
| C | 0.14 | ρ = 7.86 g/cm$^3$ |
| Si | 0.93 | |
| P | 0.042 | |
| S | 0.028 | |
| Cr | 18.0 | |
| Mn | 1.86 | |
| Fe | 70.0 | |
| Ni | 9.0 | |

**Referenced design configuration of the VVER fuel rod used for LUTHER modeling in the Serpent code**

Note: The active height of the fuel rod in the LUTHER core was modified and is not shown in the illustration. Otherwise, the rest of the dimensions is conserved. The design configuration is referenced from NEA (2012).



**Average temperatures used in the first LUTHER model for different reactor core components at hot operating condition**

| Reactor component | Temperature [°C] | Temperature [K] |
|---|---|---|
| Fuel pellet ($UO_2$) | 276.85 | 550 |
| Fuel clad (ZIRLO$^{TM}$) | 186.85 | 460 |
| Fuel end plugs (Zircaloy-4) | 146.85 | 420 |
| Internal fuel rod spring (SST-302) | 146.85 | 420 |
| Fuel nozzles (Zircaloy-4) | 146.85 | 420 |
| Gap and spring plugs (Zircaloy-4) | 146.85 | 420 |
| Central tube (ZIRLO$^{TM}$) | 186.85 | 460 |
| Thermal insulator (YSZ) | 111.85 | 385 |
| Pressure tube (Zr-2.5wt.% Nb) | 46.85 | 320 |
| Reactor coolant ($H_2O$) | 165 | 438.15 |
| Reactor moderator ($H_2O$) | 40 | 313.15 |
| Ambient air | 26.85 | 300 |