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DISSERTATION THESIS

Optimization of spent nuclear fuel storage

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Declaration

I declare that I have elaborated the dissertation thesis individually, and have used sources which I cite and list in the bibliography.

Pilsen, 28 October 2022

. . . .

signature

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Abstract

Legislative and criticality safety margins are commonly achieved by placing neutron absorbers in the cask basket design. Currently, boron content in steel or aluminium alloy is exclusively used in spent nuclear fuel transport and storage facilities absorber components as the absorber material. The reason is that the mechanical and chemical properties of light boron nuclei can be added directly to basket tube materials or placed in extra sheets between the tubes, and boron is a very good neutron absorber. Nevertheless, with higher fuel enrichment and a limit on boron content in alloys, criticality safety criteria are not easily met.

A unique solution presented in this dissertation thesis is based on special fixed neutron absorbers placed directly within the fuel assembly. A neutron absorber, permanently connected in specially designed tubes, decreases system reactivity more efficiently than absorber sheets between the assemblies. This solution is more efficient than absorber tubes even with a neutron flux trap. Hence, it allows significant basket design changes (e.g., lowering boron content in steel or decreasing fuel assembly pitch in the basket resulting in lower cask wall diameter and total cask mass). It is even possible to use absorbers to increase the capacity of the spent fuel pool. Absorbers allow safe storage in a more compact rack with increased capacity.

Keywords

spent nuclear fuel, nuclear safety, spent fuel cask, neutron absorbers, nuclear fuel cycle

Abstrakt

Legislativní a bezpečnostní limity jsou v současné době dosahovány pomocí neutronového absorbátoru umístěného v obalových souborech. Díky svým mechanickým, chemickým a hlavně neutronickým vlastnostem je používána směs oceli a neutronového absorbátoru v podobě bóru. Výsledná slitina je používána pro transportní i skladovací komponenty. Bór je možné přidávat přímo do obalových souborů nebo zvlášť do plátů mezi trubky. Nicméně, se zvyšujícím se obohacením paliva je stále složitější bór ve slitině mechanicky zpracovávat, zároveň však dochází ke zmenšování bezpečnostních rezerv.

Představené unikátní řešení je založeno na speciálních fixních neutronových absorbátorech vložených přímo do palivového souboru. Neutronový absorbátor je trvale spojen ve speciálně navržených pouzdrech. Prezentované řešení efektivněji snižuje reaktivitu systému než standardně používané absorbční pláty vložené mezi soubory i s použitím neutronové pasti. Nový přístup umožňuje výrazně změnit konstrukci obalových souborů (kontejnerů na vyhořelé palivo). Například snížením obsahu bóru ve slitině nebo zmenšením rozteče mezi palivovými soubory, což umožňuje snížit tloušťku materiálu kontejneru a tím i jeho celkovou hmotnost. Navíc je možné použít absorbátory i na zvětšení kapacity bazénu použitého paliva. Absorbátory umožňují výrobu kompaktnější mříže skladování s větší kapacitou.

Klíčová slova

vyhořelé jaderné palivo, jaderná bezpečnost, obalový soubor na vyhořelé palivo, neutronové absorbátory, jaderný palivový cyklus

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List of Acronyms

AR .... At Reactor
AVLIS .... Atomic vapor laser isotope separation
BA .... Burnable absorber
BU .... Burnup [MWd/MTU]
CFR .... Code of Federal Regulations
CTU .... Czech Technical University
DGR .... Deep geological repository
EDF .... Électricité de France
FA .... Fuel assembly
FR .... Fast reactor
FP .... Fuel pin
GNS .... Gesellschaft für Nuklear-Service
GWd .... Gigawatt day
HLW .... High-level waste
IAEA .... International Atomic Energy Agency
INFCIS .... Integrated Nuclear Fuel Cycle Information Systems
ISL .... In-situ leaching
ISR .... In-situ recovery
k-eff, k_{eff} .... Effective multiplication factor [-]
LWR .... Light water reactor
MOX .... Mix-oxide fuel
MTU .... Metric ton of uranium
MTHM .... Metric tons of heavy metal
MVDS .... Modular vault dry storage
MWd .... Megawatt day
NEA .... Nuclear Energy Agency
NPP .... Nuclear power plant
NRC .... Nuclear Regulatory Commission
NRI .... Nuclear Research Institute
NFC .... Nuclear fuel cycle
OECD .... Organisation for Economic Co-operation and Development

PHWR	Pressurized heavy water reactor
PRIS	Power Reactor Information System
PWR	Pressurized water reactor
SF, SNF	Spent nuclear fuel
SÚJB	State Office for Nuclear Safety
SÚRAO	Czech Radioactive Waste Repository
SWU	Separative work unit
UOS	Waste disposal packages, cask
UWB	University of West Bohemia
VVER, WWER	Water-Water Energetic Reactor
wt%	Mass fraction [-]
ρ	Reactivity [-]

1

Introduction

Nuclear energy plays a very significant role in the world's energy mix. Despite some mixed opinions, fission energy will still be an essential part of our modern civilization. It is important to realize that if humankind aims to achieve climate goals such as carbon neutrality (e.g., Paris Agreement), then people currently do not have a suitable stable low-emission energy source. Likewise, we are not yet able to store enough energy.

Once again, we are observing a boom in nuclear power. While this is mainly driven by countries in the East, several new units are also planned or under construction in the European Union. There is no need to go far beyond the borders. Even in the Czech Republic, new nuclear units are planned. Additionally, the energy crisis in 2022 makes the need for construction even more urgent.

Whether with or without the expansion of nuclear energy, we can be sure of three premises:

- Nuclear fuel is constantly increasingly adapted for higher utilization.
- It is necessary to effectively manage nuclear fuel.
- The amount of spent nuclear fuel is increasing, and it will not change in the following decades. To give an idea, just nuclear power plant Dukovany and Temelín in the Czech Republic more than 300 spent fuel assemblies are produced per year. This is about 100 tonnes of highly radioactive nuclear waste.

This thesis addresses an existing problem which the nuclear sector is facing. It is not just about current generations, but many generations in the future. The solution must be ready for thousands of years, which is a great challenge for research teams worldwide. No one who uses or will use nuclear energy can avoid it.

1.1 Main contributions of the thesis

The research is focused on a unique approach to improve the optimization of medium and long-term storage, and final disposal of spent nuclear fuel. The solution is aimed at improving nuclear safety. It is also optimized in terms of economic perspective so that the final product can be transferred into the industry. This thesis endeavours to connect theoretical research with industrial demands.

1.2 Thesis outline

The thesis is divided into four main parts: introduction, theoretical, practical (production) and experimental verification in the research reactor.

- The first part is devoted to a general overview of the nuclear fuel cycle and the current status of spent nuclear fuel management. Furthermore, this section includes detailed research on the cask history used in the Czech Republic.
- The theoretical part presents a new approach to optimization of the storage of spent nuclear fuel. This unique approach is verified on many created models, and analyzed in detail. The proposed solution was investigated to optimize fuel storage in western and eastern-type fuel casks, approach the use of spent fuel pools, and investigate the possibility of increasing their capacity. Furthermore, the use of neutronic absorbers for the final disposal cask in the Czech Republic was also evaluated.
- Based on a detailed theoretical study, the production of a prototype was prepared in order to allow experiments for the purpose of verification. This chapter is also dedicated to designing an inseparable connection between the absorber and the fuel assembly.
- The results presented in last chapter describe experimental verification of the unique designed neutron absorbers in the LR-0 reactor operated by the Research Center Řež in the Czech Republic.

2

Storage of Spent Nuclear Fuel

The main goal of this work is chiefly focused on the back-end of the nuclear fuel cycle (NFC). However, in reviewing the current state of the world, it is important to understand the whole process involved in the nuclear fuel journey. This chapter describes each stage of the NFC, and provides an introduction and a background to this thesis. The introduction to the NFC is devoted mainly to uranium fuel, because in the last 60 years uranium has become the world's most important nuclear fuel.

2.1 Nuclear fuel cycle

The nuclear fuel cycle may be defined as a set of operations and processes needed to manufacture nuclear fuel. The process describes the whole journey of nuclear fuel through the main nodes. There are several nuclear fuel cycle designs depending on their scheme. However, the main division is based on the characteristic of the fuel cycle:

a) open or once-through fuel cycle,

b) closed fuel cycle with reuse of nuclear fuel.

Also, there is a third approach called "Wait and See", which is currently used as a policy in the Czech Republic. It means that the spent nuclear fuel is stored in medium-term and awaiting the final decision [1]. The policy of nuclear fuel management is entirely up to each country.

Open fuel cycle

This approach is based on one pass through the reactor, and the fuel will not be returned to the reactor in any form. It means that the fuel will be put into the final repository after the operation in the reactor. Nowadays, this strategy is applied in most countries with nuclear power. On the other hand, this process of final storage is long-term and many countries have not established the final repository yet, see Fig. 2.1.

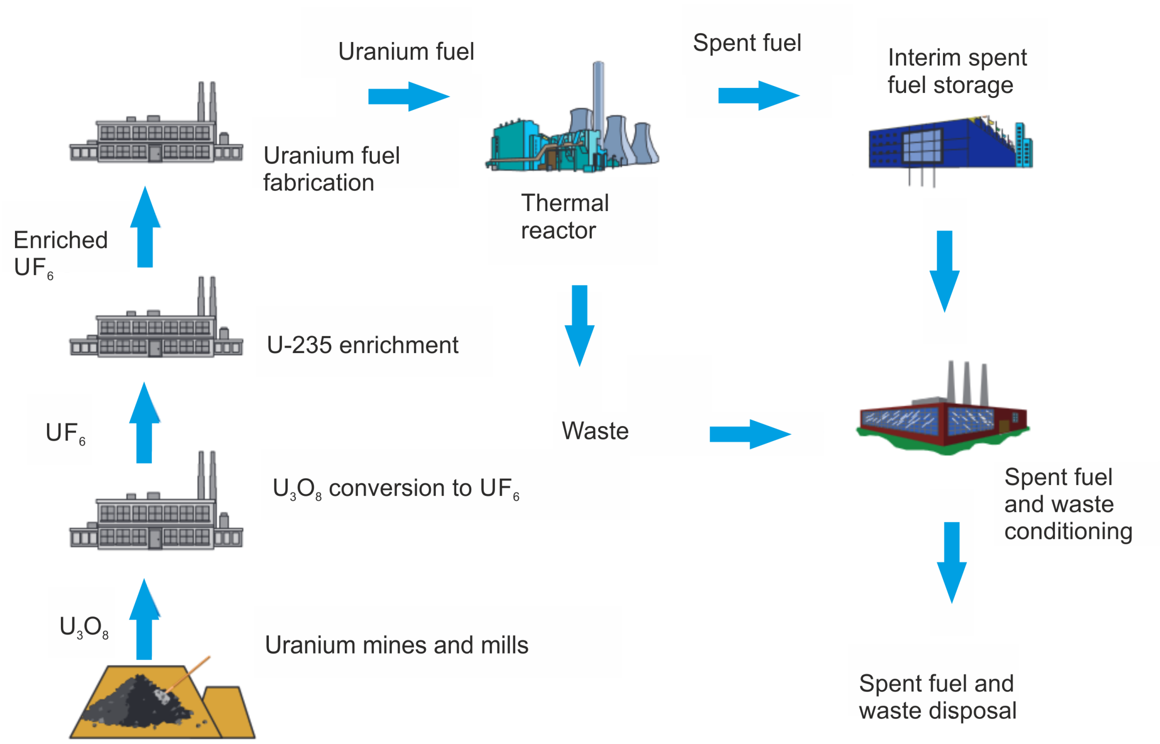

Fig. 2.1: Open nuclear fuel cycle [2].

Closed fuel cycle

This approach is based on fuel reprocessing. The nuclear fuel is further utilized after irradiation. The remaining plutonium and uranium from the spent fuel, is separated from fission products and higher actinides. This recycled fuel in the form of mixed oxide (MOX) is reused in the reactor, mainly in light water reactors (LWR) or fast reactors (FR). Due to economic reasons, only a few countries can use a closed fuel cycle. According to [3], these are: China, France, Germany, India, Japan, the Russian Federation, and the United Kingdom.

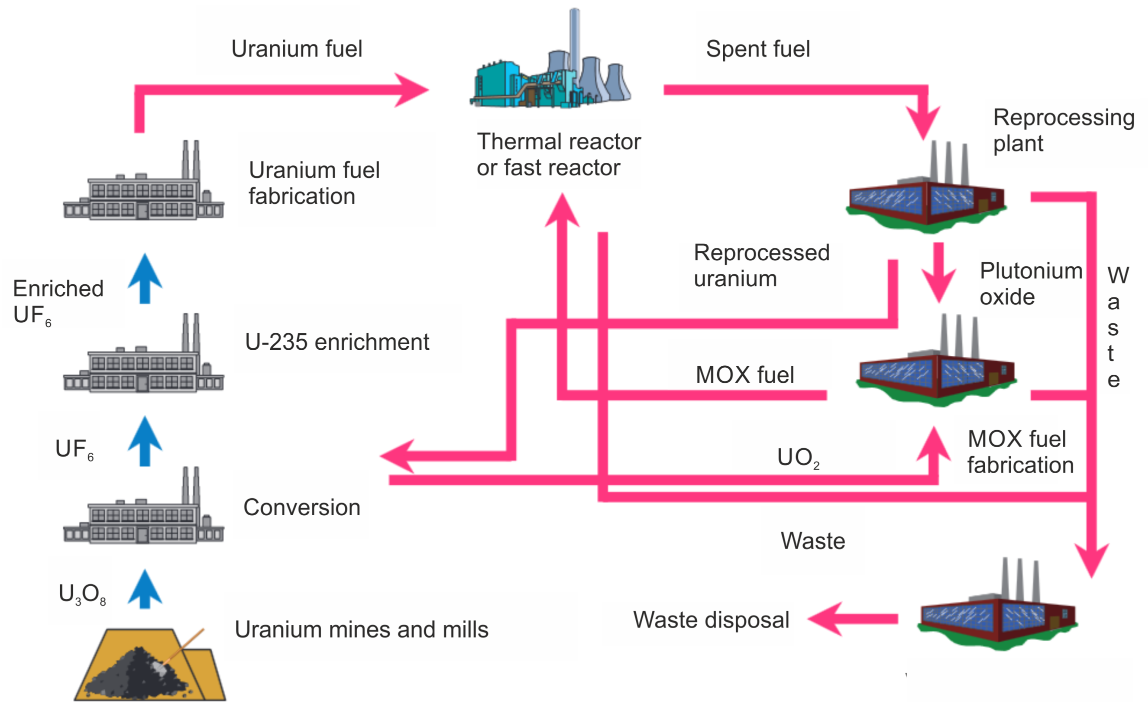

Fig. 2.2: Closed nuclear fuel cycle [2].

2.1.1 Stages of the nuclear fuel cycle

The nuclear fuel cycle is divided into three main stages: the front-end, irradiation/reactor operation, and the back-end. In the following sub-chapters, all parts in each stage will be described.

2.1.1.1 Front-end

Uranium as an element is widely distributed within the Earth's crust. It is even more abundant than gold (about 500 times more), silver or mercury [4]. It is slightly less abundant than cobalt or lead. Uranium has to be extracted from the ore and converted into a form that can be utilized and manufactured to the fuel according to requirements. Naturally, the abundance of the uranium element is mainly from two isotopes; U-235 (99.28%) and U-233 (0.711%), and traces of U-234 (0.0054%). The less abundant fissile isotope U-235 is crucial for most nuclear reactors.

Fig. 2.3 - 2.4 and Tab. 3.2 show reasonably assured uranium resources by country, with the production method, and global uranium production from 1945 to 2015 [1].

Fig. 2.3: Global distribution of identified resources (<USD 130/kgU as of 1st January 2017) [4].

Tab. 2.1: Reasonable assured resources by production method [cite: 1.1 2017] [4].

Production method	<$ 40/kgU	<$ 80/kgU	<$ 130/kgU	<$ 2600/kgU
Open-pit mining	18 089	96 787	908 839	1 078 486
Underground mining	320 784	449 777	1 002 018	1 464 394
In situ leaching acid	283 173	428 108	524 479	586 705
In situ leaching alkaline	20 300	27 720	30 100	70 704
Co-product/by-product	71 050	256 704	1 308 131	1 537 926
Unspecified	-	20 822	91 336	76 664
Total	713 396	1 279 918	3 864 903	4 814 879

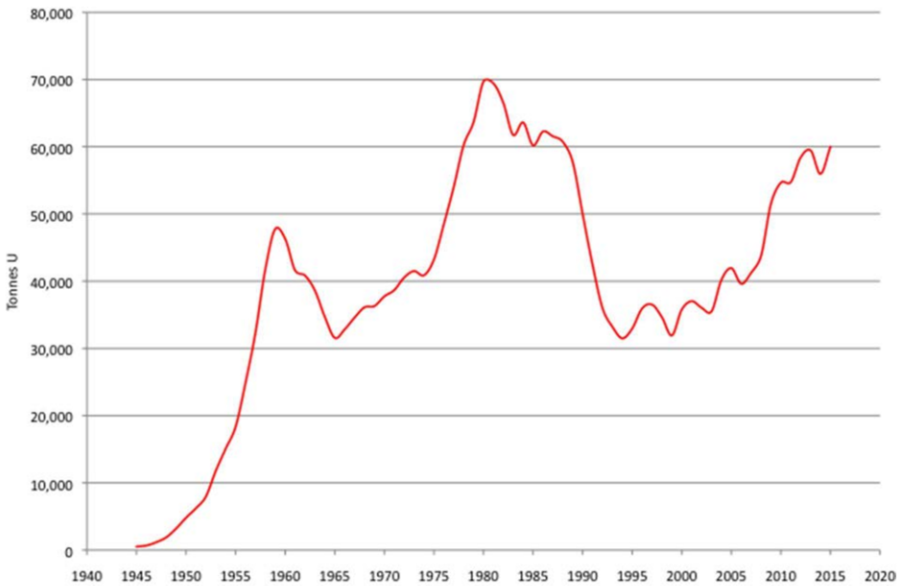

Fig. 2.4: Global uranium production (1945 – 2015) [1].

According to the IAEA statement, the global reserves of uranium are highly dependent on the cost of extraction. More details can be found in the IAEA Red Book. The newest Red Book from 2017 states that reasonably assured resources of uranium with a recovery cost of <40 USD/t were 713 thousand tonnes. The global production of uranium is driven by a range of factors, mainly economics (the uranium price) [6].

The primary stage is finding (by geological research, air-surveillance, geological probes, etc.) a suitable location to explore uranium ore if the uranium yield in the ore is profitable. In the next stage, uranium is extracted by several processes, Underground mining, a traditional process which uses a system of mine shafts, has been known for many years. Open-pit mining uses surface mining techniques to extract the ore. Both types of mining (underground or open-pit) require a further milling process in which the uranium is separated from the ore by grinding and other techniques. This relative disadvantage is not present for the extracting process called in-situ leaching (ISL) also known as in-situ recovery (ISR). Nowadays, it is the most used method for uranium mining in the world (see Tab. 2.2). This method can be mostly used to extract uranium from sandstone deposits at a depth of up to 750 m [6]. An ISL operation consists of a wellfield with associated infrastructure to pump and extract lixiviant (alkaline or acidic solution) in and out of the mineralized zone, and a processing facility to extract uranium from the lixiviant to produce the desired final uranium product [6].

The above is the most used approach, but there are also other less used methods such as: heap leaching and non-conventional uranium extraction. Heap leaching is an alternative method of extracting uranium-rich liquor from the mined ore. Mining of the ore is conventional (either underground or surface), and the ore is placed on surface pads where extractive liquors (acid or alkaline) are pumped over and through the material. This process may be repeated until liquor of sufficient uranium content is transferred for further processing to extract the uranium. The method of non-conventional uranium extraction is based on by-products because uranium can occur in association with other minerals such as gold and copper, and is often mined just as a by-product of these materials.

The final product of uranium extraction is the yellowcake. It is a material that has a high concentration of U_3O_8, and it is the input to the next stage of NFC.

Tab. 2.2: Percentage distribution of world production by production method [6].

Production method	2013	2014	2015	2016	2017
Open-pit mining	17.6	13.9	12.6	12.9	13.7
Underground mining	28.6	27.3	32.2	30.8	31.9
ISL	44.5	27.3	32.2	30.8	31.9
In-place leaching	7.1	-	-	-	-
Co-product/by-product	1.5	7.2	6.0	6.1	5.8
Heap leaching	0.7	0.5	0.4	0.4	0.5
Other	-	0.1	-	0.1	0.1

The next stage is conversion. The mentioned yellowcake is converted to the form of uranium hexafluoride (UF_6), which is usable in the next steps for different cases: enriched uranium fuel, natural uranium fuel, metal uranium fuel or metallic uranium alloy. The choice of UF_6 will be described later.

The reactor is not designed to operate with natural fuel enrichment, such as pressurized heavy-water reactor (PHWR). Thus, the fuel must be enriched. The enrichment of uranium is a process used to increase the concentration of the U-235 isotope, usually by isotope separation. It is a very complicated process with many stages. The reason is that each stage has a low separation factor. It is necessary to have many stages in series, and it is very energy-consuming. Only a few countries in the world have met this technological challenge, and of course, it is associated with non-proliferation treatments. Enrichment can be achieved by several methods: centrifugation, chemical separation, gas diffusion, AVLIS, etc.

The most common methods are gas diffusion and centrifugal enrichment. In gaseous diffusion the separation is achieved by diffusion through a porous membrane, and the most commonly used centrifuge enrichment uses high rotational speed to separate the lighter isotope. Both methods use uranium in the form of gas UF_6. The UF_6 has unique properties: low sublimation temperature – around 56.4 °C, fluorine has only one isotope, and it has a low atomic weight [6]. It is ideal for the enrichment process to separate very similar isotopes.

The final level of enrichment is divided into these categories [7]:

- $<$ **20%** U-235 – Low-enriched uranium (LEU)
- $>$ **20%** U-235 – Highly-enriched uranium (HEU)
- $>$ **90%** U-235 – Weapon-grade (WG)

The slightly enriched uranium (SEU) category is also specified in some sources. It is about 0.9-2% enrichment, and is related to Canadian PHWRs [5]. However, in a standard energy power reactor the enrichment is up to 5%. In general, the limit is set at 4.95%, and 0.05% is given as a margin for production uncertainty.

The unit which defines the effort required in the enrichment process is called the Separative Work Unit (SWU). It is measured in units of kg, and can then be manipulated to determine the cost per SWU. An example is shown in Fig. 2.5. The graph shows how one tonne of natural uranium feed might end up: 120-130 kg of uranium for power reactor fuel, as 26 kg of typical research reactor fuel, or conceivably as 5.6 kg of weapon-grade material. The curve flattens out because the mass of material being enriched progressively diminishes to these amounts, from the original one tonne, so it requires less effort relative to what has already been applied to progress a lot further in percentage enrichment. The relatively small increment of effort needed to achieve the increase from normal levels is the reason why enrichment plants are considered a sensitive technology in relation to preventing nuclear weapons proliferation, and are very tightly supervised under international agreements [8].

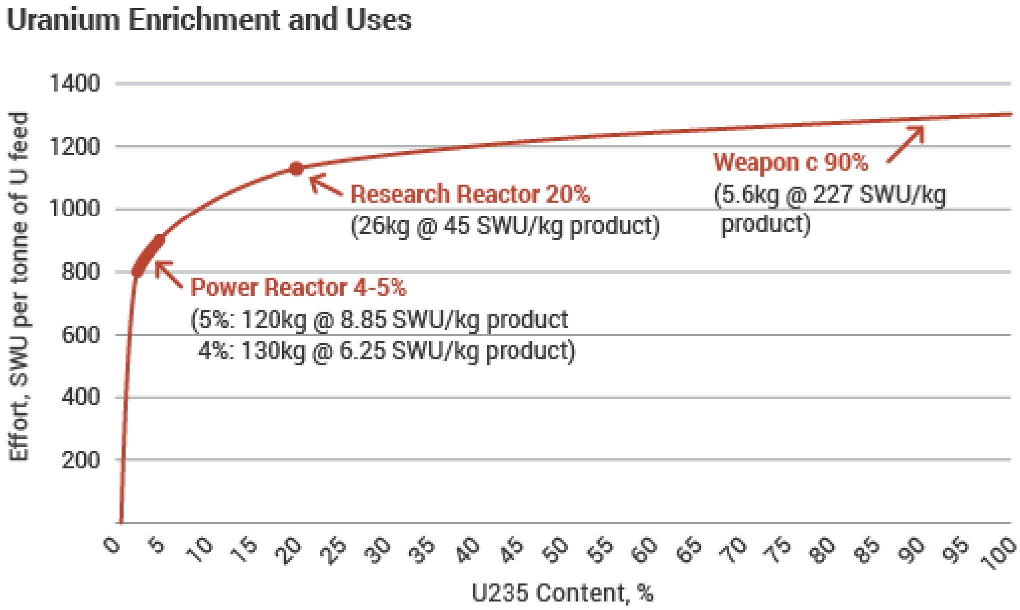

Fig. 2.5: Graph showing increase in SWU with increase in U-235 concentration. [8].

The last step of front-end NFC is manufacturing the nuclear fuel in the form of fuel assembly with uranium dioxide (UO_2) pellets in order to be utilized in the nuclear power reactors. The first step in the enriched fuel fabrication is UF_6 re-conversion into UO_2 powder by the dry or wet way. The following process is pelletizing. This process includes blending, pre-pressing, granulation, sintering, grinding, and inspection. The product is the standard ceramic UO_2 fuel pellet, with an optimal ratio between porosity and density. These pellets are inserted into fuel rods, and rods (fuel pins) are put into the assembly. The final dimensions and lattice shape of fuel assembly depend on the type of reactor.

2.1.1.2 Reactor operation

The nuclear assemblies are loaded into the core in the reactor. For a long time, the fuel is irradiated and produces a lot of energy. In the case of the LWR reactor, it is from 3 to 6 years. In the case of Temelín NPP with VVER-1000 reactor, the maximum limit for nuclear fuel in the core is 6 years. But the time of nuclear fuel in the core highly depends on the strategy of core loading (legislative and operational limits, utilization, etc.).

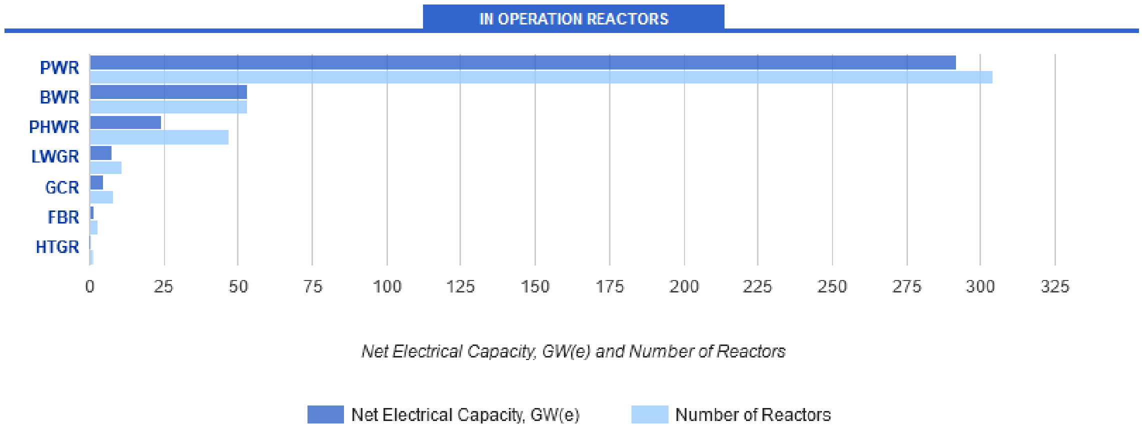

Fig. 2.6: Types of currently operating nuclear power reactors by reactor type [cit. 2022-10-22] [9].

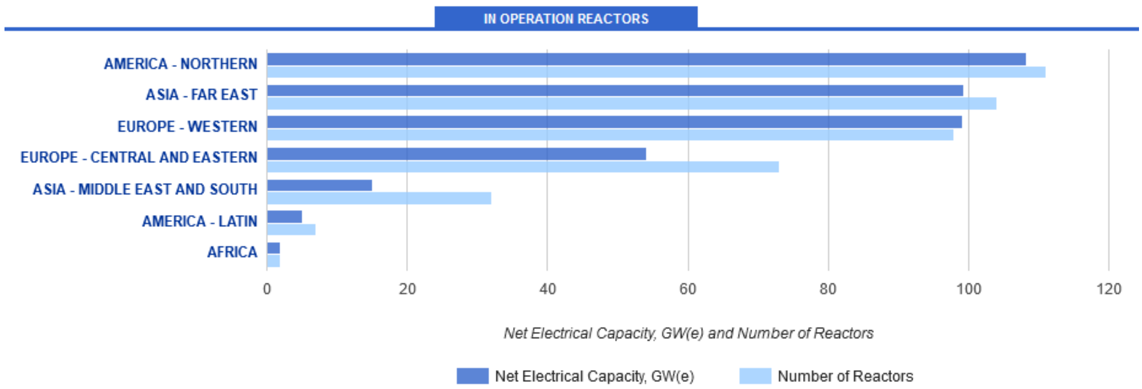

Fig. 2.7: Types of currently operating nuclear power reactors by region [cit. 2022-10-22] [9].

According to the IAEA PRIS database, there are 427 nuclear power reactors in operation with 382 796 MW total net electrical capacity and 56 under construction [cit. 2022-10-22].

2.1.1.3 Back-end

The nuclear fuel which has been irradiated in the nuclear reactor has to be removed from the reactor after the irradiation period. This used fuel is called spent fuel or irradiated fuel. After discharge, the fuel still has a large amount of decay heat as a result of radioactive decay, and must be stored in At-Reactor (AR) pools because it is necessary to immediately cool the fuel. These facilities deal with a large amount of decay heat. As an approximation for thermal power generation, we can use Decay – Heat Wigner-Way formula see Eq. 2.1, [11]. For example, the thermal power after shut down is more than 1% after 24 hours [10].

$$P_d(t) = 0.0622 \times P_0 \times [t^{-0.2} - (t_0 + t)^{-0.2}] \tag{2.1}$$

where

- P_d(t) = thermal power generation due to decays,
- P_0 = thermal power before shutdown,
- t_0 = time, in seconds, of thermal power level before shutdown,
- t = time, in seconds, elapsed since shutdown.

The fuel is stored in the AR pool for up to 10 years. After that, it is moved to medium-term storage, where it is stored temporarily from several years to several decades for future use, or into permanent disposal (deep geological disposal). Both approaches will be described in the next chapters [11].

Medium-term storage

After discharge from the AR pool, the irradiated fuel is stored in pools (wet type storage), or in casks or vaults (dry type storage) for many years. In the Czech Republic, it is licensed for 60 years. However, the duration of the storage and the method of storage depends on the fuel management of each country.

For example, in the Temelín NPP it is now around 2–3 casks per year, and each cask contains 19 FAs. In total it is 38–57 FAs per year [12].

Fig. 2.8: Medium-term storage in Temelin NPP [12].

Long-term storage

Long-term or permanent storage is the final solution for spent nuclear fuel. This strategy is based on the isolation of waste in a final geological disposal. It is a location that is expected to be stable over a very long period of time. In some references, long-term storage may be used for extended storage up to 100 or 300 years as part of a "Wait and See" strategy.

2.1.2 Summary

There are several ways to utilize nuclear fuel, depending on the selected fuel cycle. In the previous sub-chapter, we briefly described all of the main parts. The conclusion of this introduction to the nuclear fuel journey is that it does not matter which scenario we choose. All paths have the same result sooner or later. At the end of the cycle, we must always deal with nuclear fuel storage, as you can see in Fig. 2.9 below. It is an unquestionable conclusion that we must take into account.

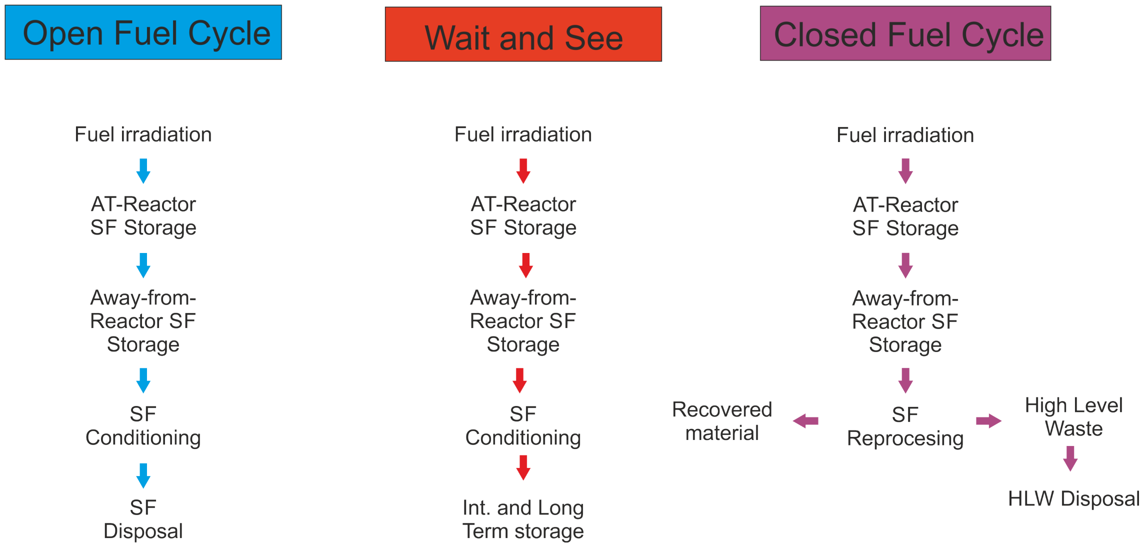

Fig. 2.9: Different fuel management options.

In Fig. 2.10 and Fig. 2.11, the trend of storage of nuclear fuel increases linearly. Due to IAEA INFCIS [13], the number of spent fuel storages in the World is 171 [cit. 2022-10-26], and the number of spent fuel storage facilities is rising.

However, these statistics are only for medium-term storage. Most of these fuel assemblies will be finally disposed in the geological depository. These are thousands of fuel assemblies, which must be stored in some disposal cask/canister. There is a huge potential for research on how to improve economics and nuclear safety. In the case of the final disposal, it is a potentially dangerous waste as the SNF, and it needs to be isolated for thousands and thousands of years. It is an area of research with great potential and responsibility.

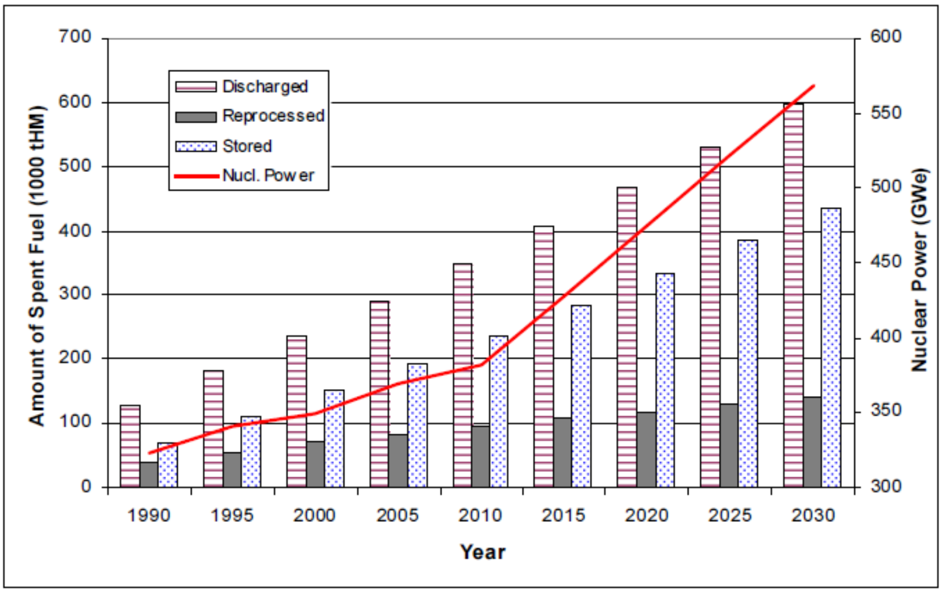

Fig. 2.10: Cumulative spent fuel discharged, stored and reprocessed from 1990 to 2030 [11].

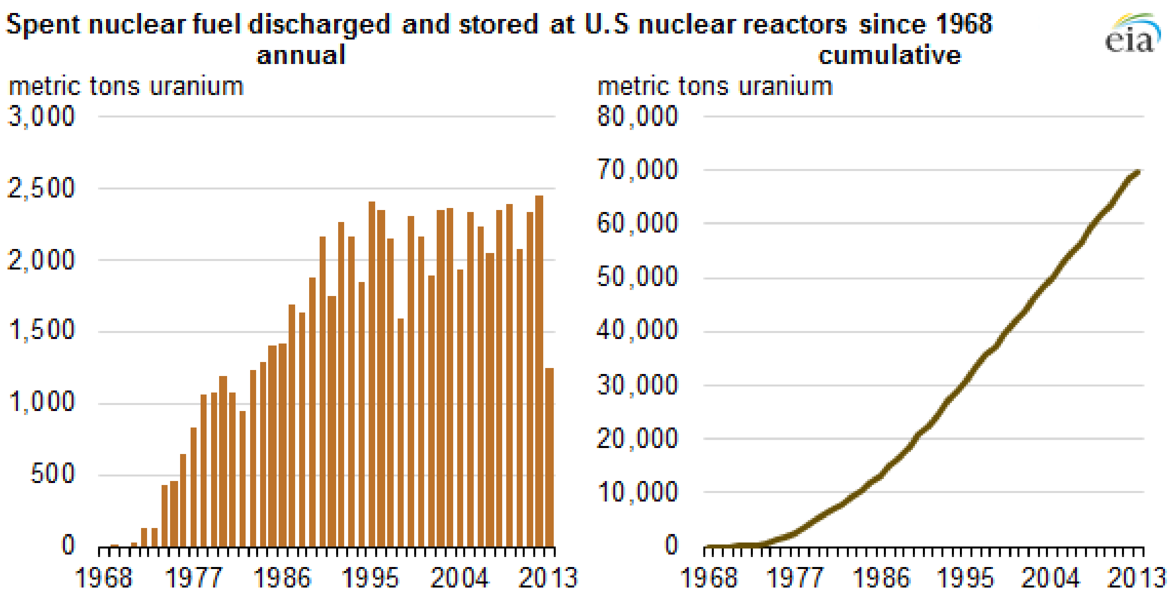

Fig. 2.11: Cumulative spent fuel discharged [14].

2.2 Spent nuclear fuel management

Each country has specific views on spent fuel management, witch can be observed in the national arrangements for ensuring spent fuel and radioactive waste. The national legislature is usually responsible for approving legislation covering the safe management of spent fuel and radioactive waste. This legislation generally includes the establishment of a nuclear regulatory body and an authority for spent fuel and radioactive waste management, as well as defining the essential elements of the national policy [15],[16].

However, in the international policy there are very strict laws regarding spent fuel and radioactive waste disposal. All nuclear waste must be disposed in the country where the waste is generated. There are a few special exceptions where it is possible to transfer spent nuclear fuel for reprocessing in another country, but thereafter all the residual waste is generally returned to the originating country. This is a legal requirement for the EU member states, in line with Article 2(3)b of the Euratom Waste Directive [15]. There are only a few countries (such as France, the Russian Federation and the United Kingdom) which allow the import of spent fuel from other countries for reprocessing and recycling of plutonium (and other actinides), usually then returning reprocessed materials and any radioactive waste to the country of origin [15],[17].

International fundamental principle

The basic international principle defined by the IAEA states that "Radioactive waste must be managed in such a way as to avoid imposing an undue burden on future generations" (principle 7, para. 3.29). The management of spent requires the process to be sustainable for thousands of years. The producers of nuclear waste must provide adequate guarantees and measures: economical, technical, organizational, safety, and legal [18].

2.2.1 Spent fuel

The nuclear fuel which has completed a reactor operation, is called spent fuel. This is a slightly inaccurate general definition, as there is still a large amount of energy in the fuel. Most nuclear reactors use fuel in the form of fuel elements consisting of pellets made of ceramic uranium dioxide. At the beginning of reactor operation, the fuel contains only isotopes of U-238, U-235 (3-5%), and a small amount of U-234 and oxygen. At the end of the nuclear operation, the original enrichment of U-235 has been reduced to about 0.8%, but the exact value highly depends on initial enrichment and reactor operation. However, the content of newly created elements is about 5% (including around 1% of plutonium isotopes). Thanks to unstable elements, the fuel becomes very radioactive for a long time (see Fig. 2.12), and must be isolated in a stable environment [18].

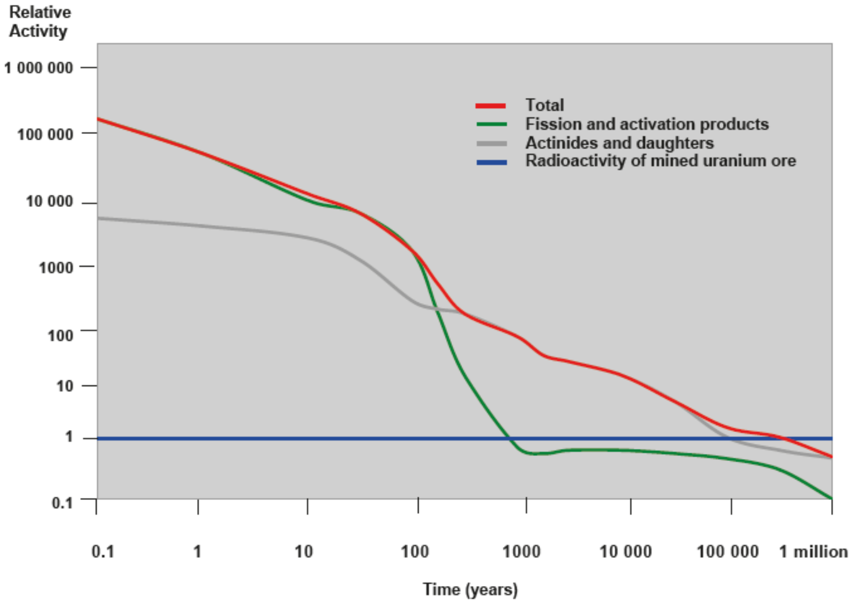

Fig. 2.12: Decay of activity of spent fuel relative to the activity of the uranium ore, which is above the health limits [18].

According to [18], the modern LWRs with a capacity of 1 000 MWe, availability of 90%, efficiency of 35%, and a burnup of around 45 GWd/tU will generate around 25 t of spent fuel per year. IAEA estimates that there are around 250 000 tons of spent fuel in storage worldwide [19].

2.2.2 Storage of spent fuel

As mentioned in the overview, there are two basic types of storage according to time of storage: medium-term and long-term. This subpart describes the options of storage in more detail. For medium-term storage, there is wet storage in storage pools or dry storage in the cask, with several decades of operating experience. In the case of a deep geological repository for long-term storage, it is more complicated, because the first DGR is still under construction and we have just a few proposals which have been shown [19].

2.2.2.1 Wet storage of spent fuel

This type of storage is based on over 50 years of experience. The use of water provides efficient cooling and shielding after fuel discharge. Over the years there have been many technological improvements in terms of instrumentation, improvements in storage density or remote handling, etc. This is the oldest approach in medium-term storage. The wet storage method was the only licensed technology in the USA until 1985 [19],[20].

A typical medium-term wet storage facility requires:

- Transport cask, decontamination, unloading, maintenance and dispatch.
- Auxiliary services: radiation monitoring, solid radioactive waste handling, water cooling and purification, ventilation, power supply etc.

The storage pool is a reinforced concrete structure, usually built above ground or at least on the ground. This approach of wet storage has some disadvantages, such as financial requirements, continuous cooling of spent fuel and possible leakage of fission products.

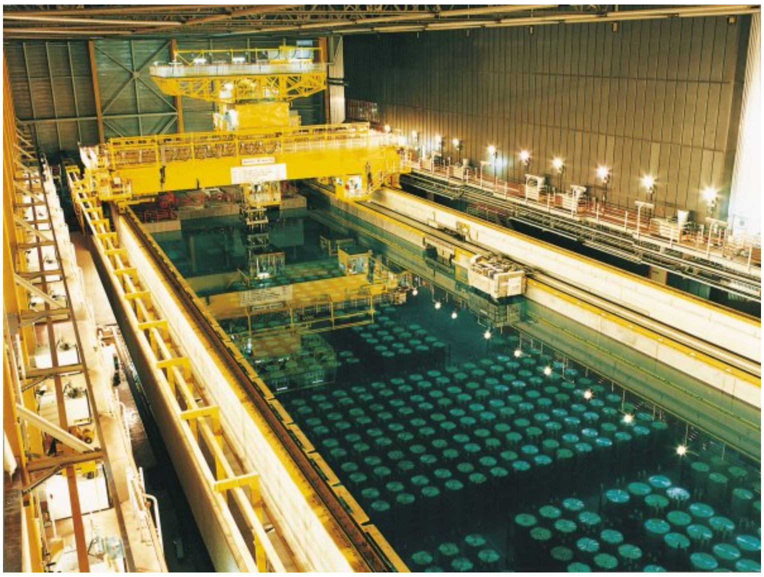

Fig. 2.13: Wet storage in Ontario, Canada [21].

2.2.2.2 Dry storage of spent fuel

For dry storage of the spent nuclear fuel two concepts are used: canisters and casks[1]. Further, it can be divided according to the material from which they are made.

Silos and concrete canisters

The first technology for dry storing spent fuel was silos, or concrete canisters. It was developed by Atomic Energy Canada Limited in the 1970s, with a prototype system deployed in 1975. The original design relies upon natural convection from the concrete shielding, and it leads to a restricted heat loading limit. The later silo systems incorporate vents to aid thermal dissipation, and the technology consists of a fixed monolithic or modular concrete structure. This design also includes an integral inner metal liner, which can be sealed after fuel loading, and fuel being sealed into a basket or by a separate sealed metal canister.

[1] In Paks NPP with four VVER-440/213 reactor type, the SNF are taken to storage called Modular Vault Dry Storage (MVDS).

Metal and concrete casks

The dry casks in the base are specially designed canisters made of different materials. Metal and concrete casks are variations of containers for dry storage of spent fuel. They are either thick walled containers, or a thin walled container. The casks are made of forged steel, ductile cast iron, steel/lead, or a forged steel/resin/steel shell sandwich structure. Concrete casks are steel/concrete/steel sandwich structures, or a thin walled steel canister which is then placed inside an overpack. The casks are usually transferred directly from the fuel loading area to the storage site. The designs were improved over 30 years. First metal cask studied for the storage and transportation of spent fuel was designed in 1977 in Germany, and the concept of using concrete was studied a little bit later in 1988 by Ontario Hydro (Canada) [20]. Technology for dry storage is constantly developing. In recent years, the casks could perform multiple functions depending on to their desired purpose. The newest casks allow storage and transport to and from a storage facility without change of casks. An appropriate way would be to reasonably implement final purpose, which may be used to determine final disposal [20],[19].

In general, the methodology of medium-term storage for both approaches independent of the design has the same safety principles which have to be taken into account [22]:

- Subcriticality of the spent fuel has to be maintained under normal and accidental conditions.
- Residual heat of the spent fuel should be removed effectively.
- Radiological shielding of the spent fuel should protect plant operators, the public and the environment from receiving radiation doses exceeding the regulatory limits.
- Fuel cladding integrity should be maintained during handling and exposure to corrosion effects of the storage environment.
- Fuel degradation during storage should be prevented by providing adequate cooling in order not to exceed fuel temperature limits.
- Environmental protection should be assured by minimizing the release of radioisotopes.
- Fuel retrievability should always be available.

Tab. 2.3: Storage options for away-from-reactor storage of spent fuel [18].

Type	Option	Heat transfer	Shielding
Wet	Pool	Water	Water
Dry	Metal cask	Conduction through cask wall	Metallic wall
	Concrete cask/silo	Air convection around canister	Concrete and steel
	Concrete module	Air convection around canister	Concrete wall
	Vault	Air convection around thimble tube	Concrete wall
	Drywell/tunnel	Heat conduction through earth	Earth

Fig. 2.14: Dry storage in Dukovany NPP, Czech Republic [23].

Fig. 2.15: Dry storage in Temelín NPP, Czech Republic [24].

2.2.2.3 Deep geological repository

The DGR isolates the SNF and HLW over thousands of years through the robust engineered barriers and properties of the host rock, located hundreds of meters below the surface, that provides a stable and safe environment. The idea of DGR following well-known natural laws is based on multiple safety designed engineered barriers, which protect humans and the environment without requiring any maintenance or remedial action by future generations (defence-in-depth). See Fig. 2.16 for an example of the five barriers system.

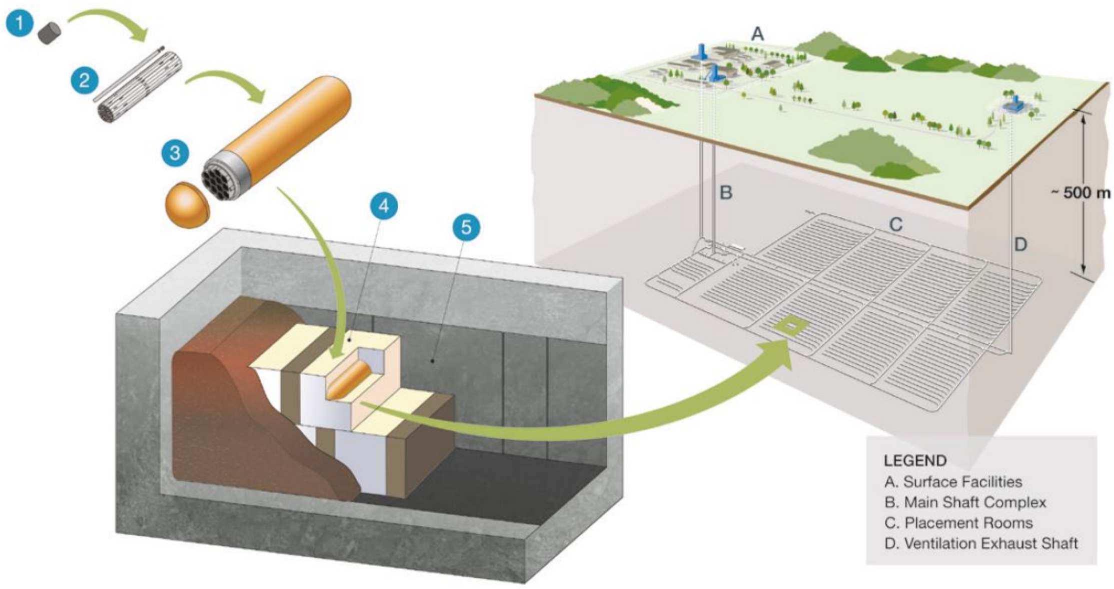

Fig. 2.16: The five barriers design, Nuclear Waste Management Organization, Canada [19].

The most recent guidance by OECD/Nuclear Energy Agency with respect to geological disposal was provided in 2013, publication: Radiological Protection in Geological Disposal of Long-Lived Solid Radioactive Waste [19]. The publication recommends to address the long-term hazards, such as:

- In the optimization principle, the radiological criterion for the design of a waste disposal facility recommended by the International Commission on Radiological Protection is an annual dose constraint for the population of 0.3 mSv per year for people living near the DGR.
- Optimizing protection has to be understood in the broadest sense as an iterative, systematic and transparent evaluation of protective options, including the best available techniques for engineering design, for enhancing the safety function of a DGR.
- Considering three main time frames: time of direct oversight, time of indirect oversight and even for times in the far future when the knowledge of the disposal facility might have been lost.

International organizations such as the International Atomic Energy Agency, OECD Nuclear Energy Agency, and even the World Health Organisation and the European Union use these principles as a key basis for the protection of individuals and the environment. Also, on the level of national policies (e.g. regulatory requirements and guidance, recommendations etc.), these principles and approaches to be used as part of a safe and responsible DGR are essentially the same in every national context [19].

2.2.3 Summary of the current global situation

2.2.3.1 Short-term management

Since the beginning of nuclear power, it is estimated more than approximately than 400 000 tHM of spent nuclear fuel has been discharged by the nuclear reactors [25]. The majority of waste has been produced from nuclear power plants (more than 98%), with the rest coming from research and small modular reactor or isotope production. It needs to be remembered that part of the spent nuclear fuel comes from military reactors, but this data is generally classified [15].

About one third has been reprocessed, resulting in the separation of fissile material or residual waste. This means that the remaining two thirds (more than 260 000 tHM) is currently being stored, and waiting for the final solution of a disposal or reprocessing [15]. In the table below, the spent nuclear fuel discharged until 2013 can be seen in Tab. 2.4.

Tab. 2.4: Spent fuel discharged form nuclear power plants (tHM), as of 31 December 2013 [15].

	Wet storage	Dry storage	Reprocessed	Total
Africa	850	50	n/a	900
Eastern Europe	28 600	2 700	3 200	40 000
Western Europe	37 000	4 600	108 400	154 100
Far East	32 100	5 700	8 600	46 400
North America	79 300	41 900	n/a	121 200
Latin America	3 000	2 000	n/a	5 000
Global total	180 800	56 900	120 300	367 600

In the next figure, one can see the location of spent fuel. Most of the spent fuel is still located at nuclear power plant sites in wet storage in the reactor pools, because it is necessary to cool down the residual power of discharged nuclear fuel. The second option is a location in dry storage, and the third is wet storage.

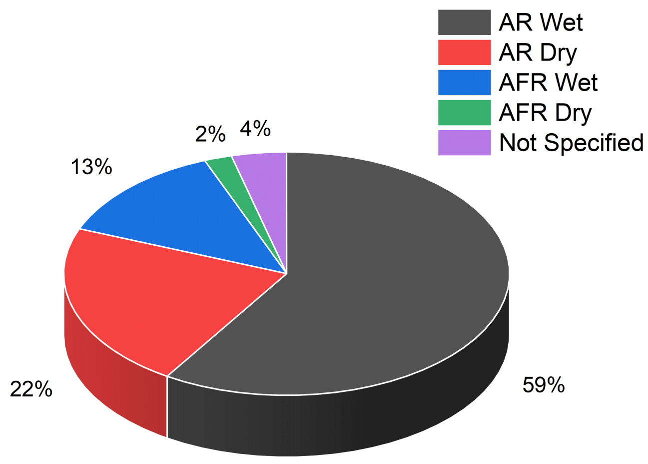

Fig. 2.17: Nuclear power plant spent fuel storage by type [15].

2.2.3.2 Long-term management

In many countries, the final decision on long term management has not yet been made. This decision is still open, and country policy is to place the fuel in medium-term storage and wait. Many states face delays in their programs for the DGR, for a variety of technical and socio-political reasons. This has resulted in the need for more and longer storage, and more challenges for spent fuel. The length of the storage time could be for many decades. The challenge is thus to ensure long term safety and integrity of the storage facilities and fuel for many decades to come, which is the main goal of this thesis.

According to a study [15] from 2018, one can observe the different approaches for spent fuel by planned disposition. But it is possible that it this could change year by year.

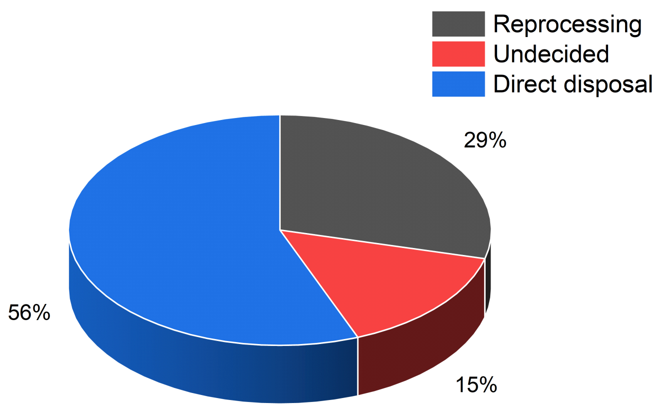

Fig. 2.18: Nuclear power plant spent fuel storage by approach [15].

2.2.3.3 GDR in the World

However, in a few countries the decision has been made, and now these countries are leaders in the development of disposal facilities for spent fuel and HLW. Here is a list of the most advanced countries compared with the Czech Republic, according to [19].

Finland (Olkiluoto)
Preparation of the spent fuel disposal started in 1983 by government decision. Since that time, site investigations have been on-going at Olkiluoto. Knowledge of the site has increased significantly since the Decision-in-Principle stage. The construction of the disposal facility transportation and tunnels began in 2016. The DGR is now under construction, with an expected commissioning date of 2023.

Sweden (Forsmark)
In Sweden, following an extensive research and development program, the license application for the construction of a DGR was submitted in 2011, and a government decision is expected soon. Swedish Nuclear Safety Regulatory issued a positive decision for final disposal in 2018. There was a small issue with copper canister documentation, but in 2019 it was defended against by the Swedish Nuclear Fuel and Waste Management Company.

France (Cigéo)
France started development of the GDR from 1991 to 2005. The design of the Cigeo facility had been in development since 2010, and was delivered in 2016 to the French Nuclear Safety Authority for review.

France is the only country on this list that practices fuel reprocessing. According to the newest NEA document, every year about 120 MTHM of MOX fuel is produced to be used in the EDF nuclear fleet [26]. This fuel is produced by the Melox plant in Marcoule. The plutonium from the nuclear fuel is separated at La Hague plant, which reprocesses around 1 100 MTHM of used nuclear annually fuel to meet the demand for MOX. This level of reprocessing corresponds to used nuclear fuel discharged annually by the French nuclear reactors in operation.

The existing industrial facilities already enable a significant improvement in nuclear waste management: fission products and minor actinides are encapsulated in a glass matrix, and structural pieces (hulls and end-piece) are also conditioned in standardised canisters suitable for transport, storage and final disposal. The volume reduction of conditioned waste pending final disposal is significant (about five times less than for open-cycle).

Czech Republic

In the Czech Republic, back-end strategy follows "National Strategy on Radioactive Waste and Spent Nuclear Fuel Management", rev. 1 (approved 2014, rev. 0, 2002). SNF is stored in interim storage at the sites of both power plants. The fuel is stored in dry-type containers, designed for 60 years of service life, and the final solution will be storage in the GDR.

The current situation is in the phase of site selection for the deep geological repository. In 2020 there was a reduction from nine candidate sites to the preferred four (i.e., Brezovy potok, Horka, Hradek, Janoch), which the remaining five as back-up solutions. In the next years, research and exploration work will be carried out on these sites [27]. Two final locations will be selected in 2025 – main and backup. As a host rock, only granite is considered and investigated. The repository is designed to dispose of together about 9 900 tonnes of heavy metal and 4 300 tonnes of radioactive waste unacceptable for inclusion in near-surface repositories [26]. There is also an underground research laboratory in which a generic experimental programme has been launched.

The final decision should be confirmed by 2030, according to the Czech government [27]. The operation of the GDR is expected to start in 2065 [68].

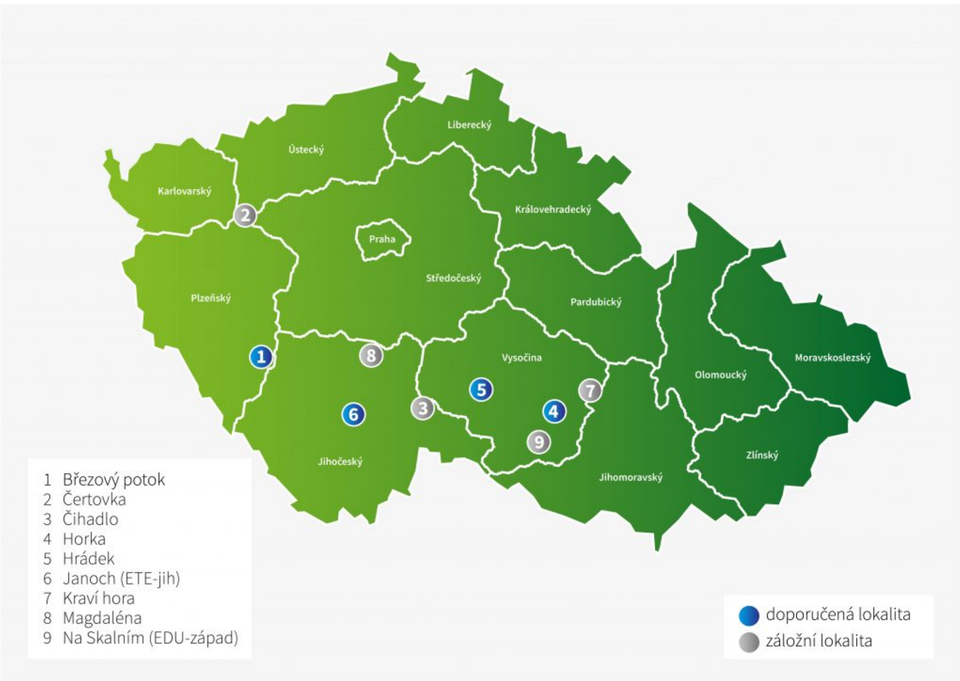

Fig. 2.19: Selection of considered sites, blue colour (recommended site), grey (back-up side) [68].

Site evaluation is a very complex task. For this reason, a methodology has been proposed by The Czech Radioactive Waste Repository (SÚRAO) to evaluate the suitability area to build a deep geological repository, as seen in [28]. As an example, the basic elimination criteria used in the Czech Republic are listed below. Each of these main criteria has a number of additional sub-criteria with different importance.

1. Exclusionary design criteria - the size of the usable rock mass, hydrogeological conditions, ensuring the stability of structures, quantity and complexity of conflicts of interest

2. Exclusion criteria for long-term safety - geological and hydraulic characteristics, site stability, characteristics that could lead to disruptions of the repository by future human activities

3. Exclusion criteria for the siting of a nuclear installation in terms of operational safety - natural phenomena, factors affecting emergency management

4. Environmental exclusion criteria - occurrence of specially protected nature areas

Defined responsibility in the Czech Republic: The NPP operator is the spent nuclear fuel owner, and responsible operator of the spent nuclear storages. The SÚRAO becomes the owner of the SNF at the moment when the operator declares that nuclear fuel is waste (spent fuel).

The Czech Radioactive Waste Repository is the owner and operator of the existing repositories, and is responsible for development and construction of the deep geological repository. Repository operations and development of DGR are covered by a nuclear account, to which the operators of nuclear power plants and radioactive waste producers contribute [26].

2.3 Spent fuel casks in the Czech Republic

Regarding the topic of this dissertation, it is desirable to analyse in detail the types of cask used in the Czech Republic. Containers used in Czech nuclear power plants will be presented. Several of them have been modelled and used for research purposes in the framework of the dissertation.

The first part is devoted to the casks for the Dukovany nuclear power plant, where the four presurized VVER-440/V-213 reactors are located. NPP Dukovany is the first nuclear power plant built on Czech territory.

- CASTOR-440/84 and modernized CASTOR-440/84M
- ŠKODA 440/84

The second part is devoted to the younger Temelín NPP with two units VVER-1000/V-320. A particular part of the spent fuel casks research is devoted to this type.

- CASTOR-1000/19
- ŠKODA 1000/19 and modernized ŠKODA 1000/19M

Furthermore, the final part is dedicated to the considered cask for the deep geological repository. It is based on the final technical report published by the Radioactive Waste Repository Authority in 2021 [29]. The crucial finding is shown in Tab. 2.5, which predicts the number of casks needed in future years [29].

- UOS ŠKODA 440/7
- UOS ŠKODA 1000/3

Tab. 2.5: Concept of Radioactive Waste and Spent Nuclear Fuel Management in the Czech Republic [30].

Reactor type	VVER-440	VVER-1000	New unit
Max. burn-up [MWd.kg^{-1}]	60	60	70
Max. operating time [year]	60	60	60
Min. cooling time [year]	65	65	65
Number of FA [pcs.]	21 700	5 400	8 100
Number of FA in cask [pcs.]	7	7	3
Number of casks [pcs.]	3 100	1 800	2 700

2.3.1 CASTOR-440/84

Information is based on the decision of the SÚJB see [31]. The CASTOR-440/84 cask is designed for the transport and storage of 84 fuel assemblies, with fuel from VVER-440 nuclear reactors. The manufacturer of the cask is GNB Gesellschaft für Nuklear-Behälter. The body of the cask is made of a single piece of thick-walled cylindrical casting. The material used is cast iron with ball graphite. The cask has radial fins on the outside.

In the inner space, there is a basket consisting of 84 hexagonal tubes made of steel, into which the FAs is loaded. In the space between these tubes, there are aluminium plates. The cask is closed by a system of two lids fitted with seals and bolted to the body. The total thickness of the lid is 315 mm. The lid is fixed to the cask body with 48 screws. The secondary lid is fixed to the cask body with 48 M 36 screws. The lid is made of stainless steel. Its thickness is 120 mm. The internal space between the primary and the secondary lid is used to monitor the tightness of both lids.

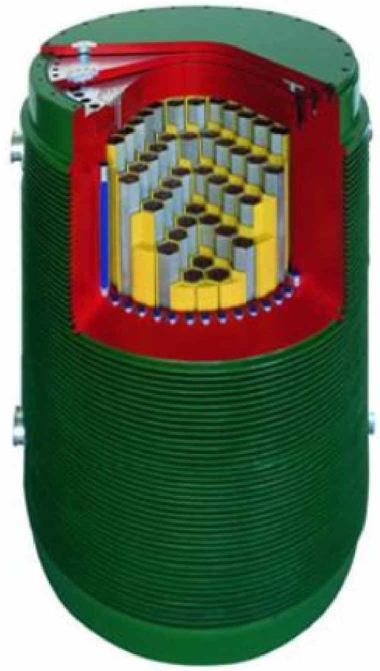

Fig. 2.20: CASTOR-440/84 [69].

2.3.2 CASTOR-440/84M

CASTOR-440/84M has several improvements due to the use of higher enrichment (in the first phase, 3.87 (3.82 + 0.05 uncertainty) wt% U-235 initial enrichment. In the second phase, it was increased for GD-2+ fuel to 4.43 (4.38 + 0.05 uncertainty)wt% U-235 initial enrichment) [32]. The changes from the original design relate to the increased tightness requirements between the primary and secondary lids. For this reason, each lid is fitted with a profiled elastomeric seal in addition to the sealing metal ring. Two rows of overlapping polyethylene bars contribute to the improved shielding properties of the cask. Another change is the new support basket developed by Škoda JS's design. The basket is made of 85 hexagonal tubes produced by the extrusion method. The basket is made of aluminium alloy. The gaps between the walls of the tubes are filled with reinforcing absorption steel plates made of boron-impregnated steel. The last change from the previous cask is the modification of the radial bars [33].

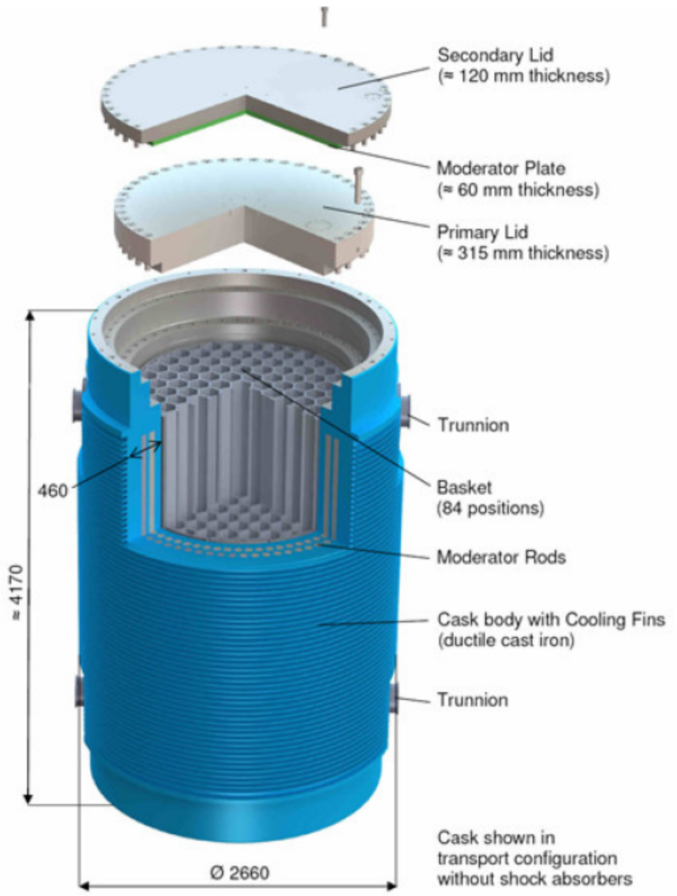

Fig. 2.21: CASTOR-440/84M [32].

2.3.3 ŠKODA 440/84

Information is based on the decision of the SÚJB see [34]. The ŠKODA 440/84 cask consists of a thick-walled cylindrical body, with a two-lid restraint system and a support basket. The cylindrical wall of the cask is 410 mm thick. In order to improve neutron moderation, 60 axial bores filled with high-density polyethylene moderator bars are located at regular intervals on the two pitch circles. These bars are inserted into the cask body from the bottom side, with sufficient clearance due to thermal expansion, and held in place by coil springs. The boreholes are closed by a 30 mm thick bottom closure plate. In addition, a 45 mm thick polyethylene plate is located under the secondary lid. A second 40 mm thick polyethylene plate for neutron moderation is located between the bottom of the cask and the bottom closing plate. On the outer surface are 213 radial fins of 30 mm in height to improve heat transfer. The bottom of the envelope is 340 mm thick.

The containment system in a cask consists of the primary lid system and the secondary lid system (double barrier requirement), together with their sealing components. The pressure level set in the space between the lids is permanently monitored by a pressure sensor integrated into the secondary lid. A protective plate covers the cask leak barriers. It protects the secondary lid from mechanical and weathering influences. The protection plate is made of carbon steel and is 80 mm thick. It is fixed to the cask body with 24 pcs. M36 screws. A passage is formed in the protection plate to the pressure monitoring sensors in the secondary lid.

The support basket is designed to accommodate 84 FAs of VVER-440 type. It consists of 85 hexagonal aluminium alloy tubes (the central tube is blinded), and between them are arranged boron stainless steel plates. Edge profiles made of aluminium alloy are placed around the perimeter to ensure the cylindrical shape of the support basket. Heat transfer in the support basket is ensured by the use of aluminium alloy in the hexagonal tubes and edge profiles. Two support pins are constantly screwed to the cask on the side for handling purposes.

Fig. 2.22: ŠKODA 440/84 [35].

2.3.4 CASTOR 1000/19

Information is based on information from the GNS producer, see description in [36]. The CASTOR 1000/19 cask is designed for the transport and storage of spent fuel assemblies of VVER-1000 reactors. The cask consists of a monolithic body made of ductile cast iron, a basket for accommodating the FAs, and a double-lid system (primary and secondary lid) arranged one above the other, as well as a protection plate.

On the outside wall, radial cooling fins are machined to improve the passive heat removal. The double-lid system made of stainless steel is tightly bolted to the cask body, guaranteeing a safe long-term containment of the fuel assemblies. During interim storage, the lid system consisting of the two barriers is permanently being monitored for leak-tightness. Monitoring is performed by a pressure switch integrated into the secondary lid.

For neutron moderation, axial boreholes are drilled into the cask wall and filled with polyethylene moderator bars. In addition, there are plates of polyethylene at the bottom end, and on the underside of the secondary lid. At the bottom and lid end of the cask body, four trunnions are bolted for attachment of handling equipment. For transport on public routes, the cask can be equipped with shock absorbers.

Fig. 2.23: CASTOR 1000/19 [36].

2.3.5 ŠKODA 1000/19

The request for a new type of cask resulted from the change of nuclear fuel supplier. The VVANTAGE-6 type from Westinghouse has been replaced with TVSA-T type from TVEL. Information is based on the decision of the SÚJB see [37]. The ŠKODA 1000/19 spent nuclear fuel cask consists of a thick-walled cylindrical body, with a closing system of two lids and a support basket in the cask's inner part. The cylindrical wall of the cask is 410 mm thick. On the outer surface of the envelope shell, 185 radial fins of 20 mm height are reamed to improve heat transfer to the surroundings.

The bottom of the cask is 338 mm thick. The two primary and secondary lids forming the cask containment system are made of corrosion-resistant forged steel, and are fixed to the cask body, made of forged carbon steel, by bolts and closed nuts or cap screws. The two sealing barriers are independent of each other.

In storage, the two independent leak barriers of the cask define an enclosed space which, when pressurised against both the internal space and the external environment, and in conjunction with the pressure monitoring device, allows the leak tightness of the package to be checked during storage.

For protection against mechanical and weathering influences, the sealing device is located above the protective plate, and is installed over the lid sealing system. If the primary lid of the ŠKODA 1000/19 is no longer available during storage and does not meet the tightness requirements, and the corresponding repair is not immediately possible it will be used to restore the lid if further storage is required.

In order to improve neutron shielding, the cylindrical wall of the cask is fitted with 42 axial boreholes of 97 mm diameter placed at regular intervals, filled with moderator bars made of high-density polyethylene. The holes for rods are closed at the bottom of the body by a closing plate. The polyethene bars are pushed into the ends of the individual bores by springs. Polyethene plates are also placed under the secondary lid at the top of the OS housing, and under the closure plate at the bottom of the housing. There are also two shut-off valves mounted in the bottom closure plate. This equalizes the pressure in the moderator bar area, and moderator plates under the bottom closure plate.

Radial fins on the outer surface of the cask increase the surface area for heat transfer. For handling the cask, four support pins are bolted to the cask body on the lid side. The two pins are located on the bottom side, to rotate the cask from the vertical to the horizontal position and the reverse. On the bottom side, there are also pins for fixing the cask to the transport device.

In order to reduce the impact stress under accident conditions during transport, shock absorbers are installed at both ends.

The support basket is designed to load nineteen FAs with VVER-1000 fuel type, and consists of 19 hexagonal aluminium profiles with boron carbide content for their positioning. On the outer edges of these tubes are spacers made of aluminium alloy. The bottom of the support basket consists of a plate made of corrosion-resistant sheet metal.

Fig. 2.24: ŠKODA 1000/19 [35].

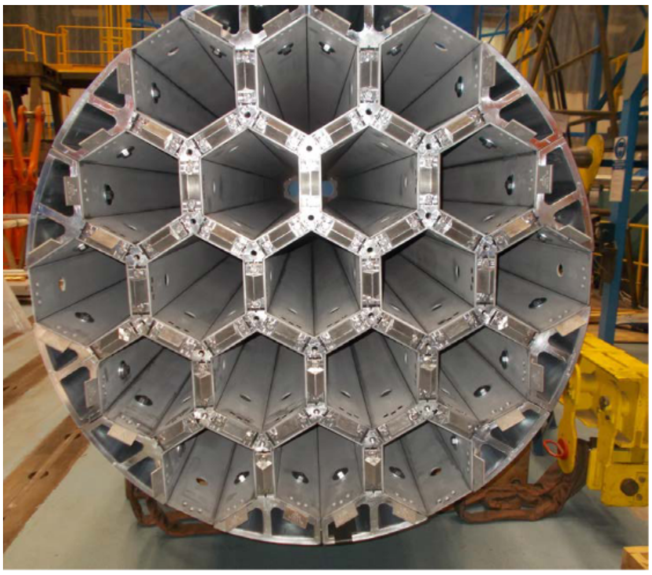

Fig. 2.25: Assembly of the basket [35].

ŠKODA 1000/19M cask is identical, only instead of aluminium tubes with B_4C, there are steel tubes and aluminium joints

2.3.6 UOS ŠKODA 440/7

The cask consists of a carbon steel outer casing 65 mm thick, seven stainless steel inner sleeves 36 mm thick, and a built-in assembly to ensure the position of the inner sleeves relative to each other. The semi-finished material of the inner sleeves is a standardised tube. The inner casings are corrosion-resistant, and have a strength resistance function. The strength of the inner casings is conservatively proven to a uniform external overpressure of 20 MPa [29],[38].

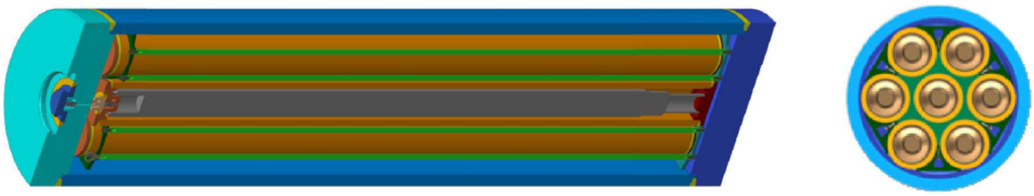

Fig. 2.26: UOS ŠKODA 440/7 (total length of UOS - 3790 mm; total diameter of UOS - 914 mm) [29].

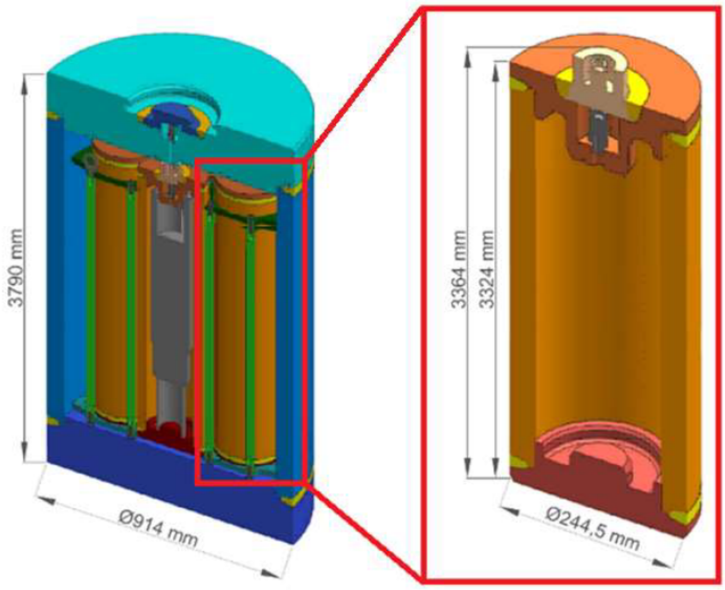

Fig. 2.27: 3D model UOS ŠKODA 440/7 (dimensions without scale) [29].

2.3.7 UOS ŠKODA 1000/3

The ŠKODA VVER 1000/3 cask is designed for three reactor fuel assemblies from a VVER-1000 reactor. It consists of a 65 mm thick carbon steel outer casing with a welded lid, three 36 mm thick stainless steel inner sleeves, and an internal build-up to ensure that the inner sleeves are positioned in relation to each other [29], [38].

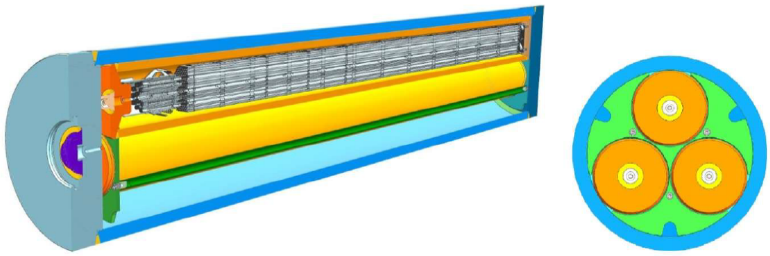

Fig. 2.28: UOS ŠKODA 1000/3 (total length of UOS - 5205 mm; total diameter of UOS - 914 mm) [29].

Tab. 2.6: Parameters of cask UOS ŠKODA 440/7 and UOS ŠKODA 1000/3 [29],[30].

	UOS ŠKODA 1000/3	UOS ŠKODA 440/7
Cask length [cm]	521	379
Fuel assemblies [pcs.]	3	7
Cask weight with fuel [kg]	17 379	12 135
Maximum heat power [W]	1 125	655
Life time [years]	100 000	100 000

3

Neutron Absorber Concept

In the previous chapter, the whole nuclear cycle was presented with a focus on the back-end part, especial on spent fuel casks. Now it is apparent that the amount of spent nuclear fuel discharged from the reactor and stored is constantly growing. In the near future this growing trend will continue, even in the event of an unexpected boom in fast reactor technologies.

However, one fact that was not mentioned in the previous chapter is the phenomenon of increasing demand for better utilization of nuclear fuel. In practice, this means the requirement of higher-enriched uranium fuels (high reactivity), which are a challenge for spent fuel handling facilities in all countries with nuclear power plants. The use of higher-enriched fuels leads to reduced reserve on legislative and safety limits of spent fuel transport, storage, and consequently, for final disposal facilities. In some cases, the required boron amount in the absorber plates or tubes can be higher than the current metallurgy process allows.

This research addresses the new fixed neutron absorber solution, with significantly increased nuclear safety and improved economics, where a new concept of inseparable neutron absorber is introduced to achieve fuel reactivity decrease. Same or better criticality safety is reached, with significantly lower or even no boron content in the cask basket absorber. Alternatively, it is possible to reduce fuel assembly pitch with the same boron amount, and subsequently decrease overall cask dimensions and cost. Because of less strict requirements for absorber material compared to active core environment, and better spatial localization inside a spent fuel handling facility, the choice of absorber material expands the currently used boron element [39]. This will be explained in more detail.

The presented research is based on several publications and activities:

- ZÁVORKA, J., LOVECKÝ, M., JIŘIČKOVÁ, J., ŠKODA, R., *Enhanced nuclear safety of spent fuel*, In Proceedings of the 27th International Conference on Nuclear Engineering (ICONE27), New York: American Society of Mechanical Engineers (ASME), 2019, p. 1-4., ISBN 978-4-88898-305-1.
- LOVECKÝ, M., ZÁVORKA, J., ŠKODA, R., *Neutron absorber concept in spent fuel casks aiming at improved nuclear safety and better economics*, In Proceedings of the 19th International Symposium on the Packaging and Transportation of Radioactive Materials (PATRAM 2019), New Jersey: Institute of Nuclear Materials Management, 2019, p. 1-7.
- LOVECKÝ, M., ZÁVORKA, J., JIŘIČKOVÁ, J., ŠKODA, R., *Increasing efficiency of nuclear fuel using burnable absorbers*, Progress in Nuclear Energy, 2020, volume 118, 2020, p. 1-12., ISSN: 0149-1970.
- LOVECKÝ, M., ZÁVORKA, J., JIŘIČKOVÁ, J., ŠKODA, R., *Criticality safety analysis of GBC-32 spent fuel cask with improved neutron absorber concept*, In Proceedings of the PHYSOR 2020, La Grande Park: American Nuclear Society, 2020, p. 1-8., ISBN 978-1-5272-6447-2.
- LOVECKÝ, M., ZÁVORKA, J., JIŘIČKOVÁ, J., ŠKODA, R., *Neutron absorber for VVER-1000 final disposal cask*, In Proceedings : 29th International Conference Nuclear Energy for New Europe (NENE 2020), Ljubljana: Nuclear Society of Slovenia, 2020, p. 1502.1-1502.8, ISBN 978-961-6207-49-2.
- ZÁVORKA, J., LOVECKÝ, M., JIŘIČKOVÁ, J., ŠKODA, R., *New neutron absorber in spent fuel casks aiming at improved nuclear safety and better economics*, 2021 International Congress on Advances in Nuclear Power Plants (ICAPP 2021), Abu Dhabi, UAE, 2021.
- ZÁVORKA, J., LOVECKÝ, M., JIŘIČKOVÁ, J., ŠKODA, R., *Experimental verification of new neutron absorbers concepts*, Progress in Nuclear Energy, 2022, (preprint).

R&D project:

- TK02010102 - Optimization of Dry Storage for Spent Nuclear Fuel, Technology Agency of the Czech Republic. Realized in years 2019 - 2022

3.1 Current situation

The criticality safety analysis evaluates the reactivity of the fuel during transport or in a storage facility. Reactivity is limited by IAEA recommendation, and subsequently by national regulatory decrees (e.g. 10 CFR 50.68 in the United States), to a neutron multiplication factor value below 0.95 with all uncertainties.

Keeping reactivity in transport and storage facilities (e.g., pool racks, cask baskets) within the legislative limit has been done by inserting absorber sheets between fuel assemblies in the past. Today, modern designs are made up of absorber tubes for each position of the fuel assemblies. Space between two adjacent assemblies, separated by an absorber sheet on each side, creates a neutron trap. From a neutron transport point of view, neutrons are initially slowed down within neutron traps to thermal energies. Then, after the moderation, the thermal neutron returning to the fuel assembly is absorbed in the tube absorber materials with a significantly higher probability.

The current solution has several disadvantages. The space between the fuel assemblies is not optimized for the fuel amount capacity, but for the neutron trap volume. Another disadvantage is the material of the absorber itself. Currently only boron is used, which is technologically compatible with steel and aluminum alloys. The boron content in the absorber tubes required to achieve safe subcriticality increases with higher enriched fuels, up to the maximum absorber content from the technological point of view (i.e., problems with sheet rolling, extrusion of profile, and welding of high boron content sheets).

3.2 New Fixed Neutron Absorber Concept

The improved neutron absorber concept is based on placing neutron absorbers directly into the fuel assembly (the most effective location). This solution is more efficient than absorber tubes, even with a neutron flux trap, and allows significant basket design changes. The main changes are lowering the boron content in absorber tubes, and decreasing fuel assembly pitch in the basket, resulting in lower cask wall diameter and total cask mass. The absorber is placed in steel cladding, and fixed inside guide tubes. The temperature, radiation, chemical compatibility and pressure parameters are not limiting, since the absorber would not be exposed to reactor operation environment (much higher temperature and pressure), so material selection analysis is performed to optimize improved neutron absorber concept. This concept was initially published in proceedings of the 27th International Conference on Nuclear Engineering (ICONE) [58].

For analysis carried out in 2019, commonly used materials have been chosen as absorbers. Boron carbide (B_4C) as material for control rods for VVERs or PWR, and gadolinium oxide (Gd_2O_3) which is added to UO_2 rods as burnable absorber. Both selected materials are widely available, low-cost, and have high absorption properties. Boron was selected as a dominant material used in fuel transport and storage facilities. Gadolinium is characterized by an excessive thermal absorber cross section that can be utilized in spent fuel neutron spectra. Both elements are stable in the spent fuel environment, since the neutron source strength is distinctly lower than in the reactor and the transmutation processes change absorber material composition only in ppm level.

In this concept, the absorbers are directly placed into guide tubes (purple colour, see Fig. 3.1). This solution is similar to standard control rods, with the difference being that the absorber is firmly and inseparably connected to the guide tubes in spent fuel casks as it can be required by the regulatory body (this part is devoted to the experimental part of the work). In the figure, one can observe the example of use (see Fig. 3.2), introduced in the first study in 2019. The example shown is for reference cask designed as a concept for the Czech deep geological repository, with planned operation starting in 2065.

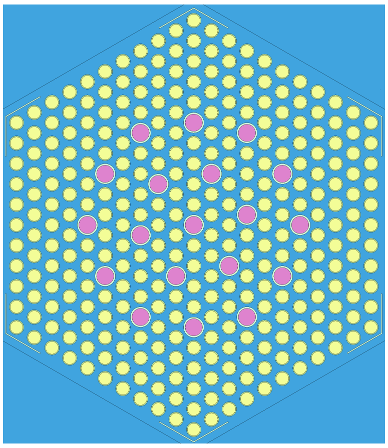

Fig. 3.1: Neutron fixed absorber concept in nuclear fuel.

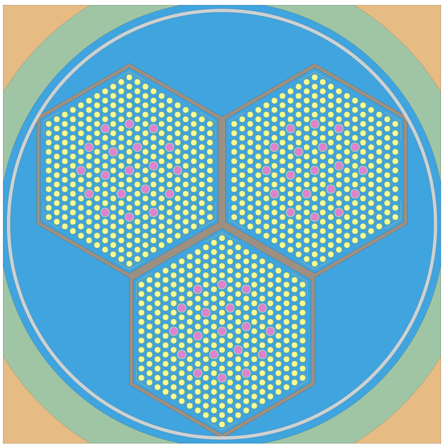

Fig. 3.2: Reference disposal cask model with 3 fuel assemblies, VVER-1000-type with hexagonal lattice.

Tab. 3.1: Calculation model material description [29].

Material	Purpose	Color
UO_2	fuel	yellow
E-110	cladding	green
E-635	guide tube	green
air/water	moderator	blue
B_4C/Gd_2O_3	absorber rod	purple
steel EN 1.4306	distance tube	brown
steel EN 1.4404	inner casing	grey
steel CSN 12 022	outer casing	brown
bentonite	fission product	orange

The results of our first studies can be divided into two scenarios – cask in wet conditions (i.e., neutron multiplication factor in accidentally flooded cask for criticality safety analysis), and cask in dry conditions (i.e., normal cask operation, neutron source depends on neutron multiplication factor due to subcritical multiplication), see Fig. 3.3 and Fig. 3.4.

Boron content around 0.5 wt% B-nat is required to maintain k-eff under 0.95 for fresh fuel. That is the way current casks are designed. In our concept, borated steel can be replaced by absorbers placed in guide tubes (either B or Gd compound). Boron is more absorbing than gadolinium in the chosen chemical compounds. Fuel burnup dependency is depicted in the case where the burnup credit approach is used. When using the newly proposed concept, there is no need for steel tubes to be borated. On the other hand, standard approach requires around 0.5 wt% B-nat or burnup credit. Moreover, cask basket redesign can only be achieved with absorbers fixed in guide tubes because of high criticality reserve to 0.95 limit.

Similar conclusions can be drawn for dry conditions of the final disposal cask, see Fig. 3.3 with k-eff values around 0.2 lead to neutron source multiplication by 25% according to equation 3.1:

$$\frac{\text{source}}{1 - k_{eff}} \tag{3.1}$$

Graph lines are similar with Fig. 3.4. Absorber material placed in guide tubes decreases total neutron source and resulting fluence by 5%, hence the neutron dose rate on cask surface will also be decreased by 5%.

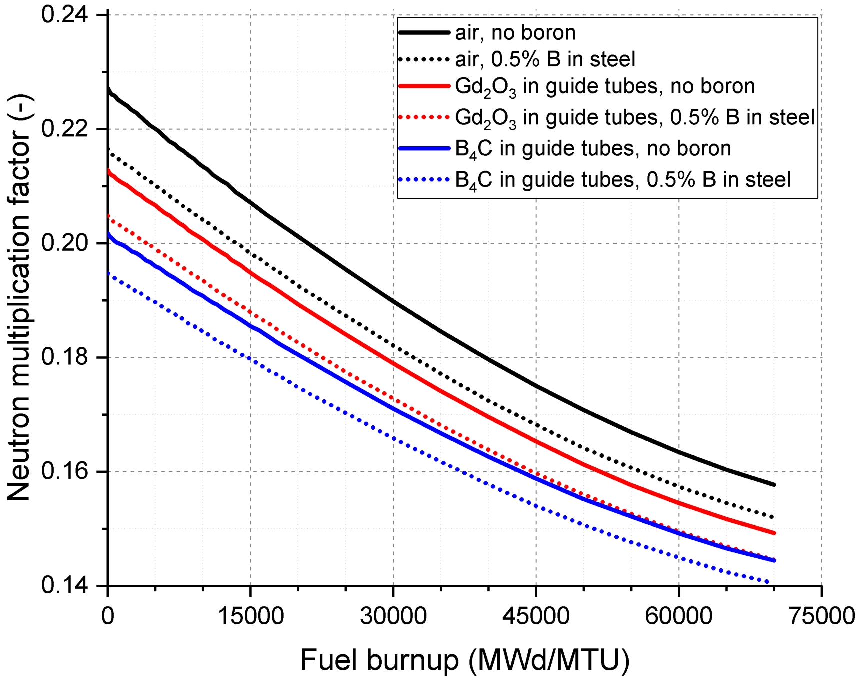

Fig. 3.3: Results for disposal cask in dry conditions.

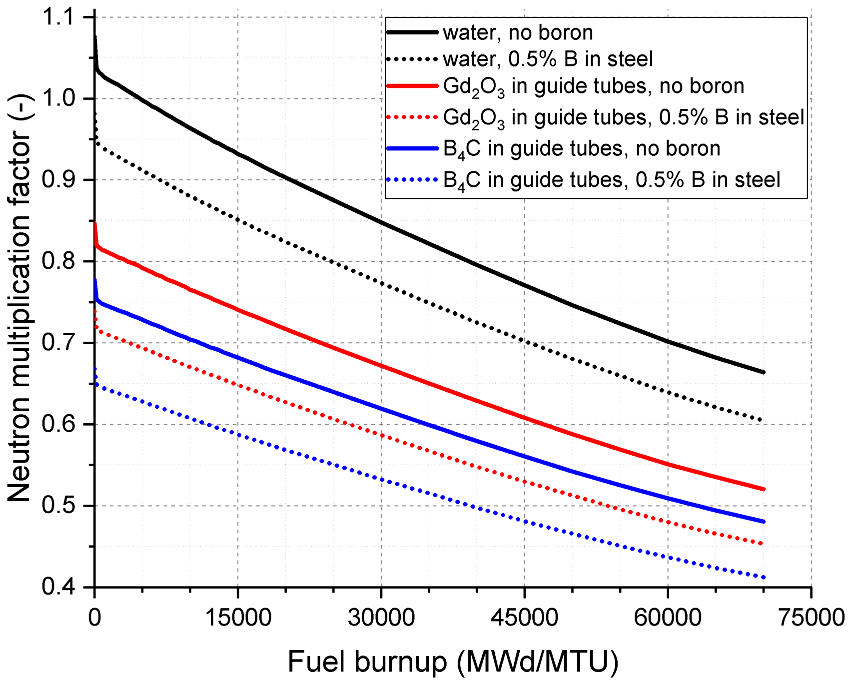

Fig. 3.4: Results for disposal cask in wet conditions.

Summary of the fixed absorber concept for spent fuel transport, storage and disposal in the first study:

- The decrease of the reactivity in the disposal cask in both wet and dry conditions was compared to standard approach with absorber tubes. The resulting effect is the possibility to reduce the cost of the cask, for example by reducing the fuel assembly pitch that leads to reduced cask dimensions.
- Tested suitable materials are B_4C or Gd_2O_3.
- The concept decreases neutron fluence by 5% due to subcritical multiplication. Consequently, the associated dose is decreased.
- The possibility of replacing expensive borated steel within the cask construction. Cask basket redesign would allow smaller cask wall diameter and lower wall mass.

The results show the basic functionality of the neutron absorber approach. However, a number of follow-up studies are presented in the next chapter in deep detail:

1. Theoretical part
   (a) Optimization of suitable material
   (b) Applications for Spent Fuel Cask
   (c) Application for Final Disposal Cask
2. Practical part - prototype production
   (a) Solution of inseparable joint of fixed neutron absorber
   (b) Absorber material form
3. Experiment verification at LR-0 research reactor
   (a) Experimental arrangement
   (b) Evaluation of results

3.3 Theoretical part

This chapter is devoted to detailed material optimization, using complex matrix calculations, followed by the application of absorbers for spent fuel cask. The optimization of nuclear fuel storage considered the storage of the eastern type - VVER, and the western type - PWR. Subsequently, the application for the spent fuel pool is also performed. And finally, the solution for the final disposal cask is presented.

Models in the following theoretical part were created by neutronic transport code Serpent (version 2.1.31 or 2.1.32) [41] with ENDF/B-VIII.0 continuous energy nuclear data library [59]. Fuel composition was calculated with TRITON code sequence from SCALE-6.2.3 code package [40]. And several analyses were made by UWB_1, which is a fast nuclear depletion code developed by the Czech universities (UWB and CTU in Prague) [42].

Several complex scripts were created in-house for this study, with the primary objective of an autonomous process or maximizing the optimization of the computation time, while preserving the required calculation accuracy. In many cases, it was a combination of several neutronic solvers.

3.3.1 Optimization of Suitable Material

In introduction to fixed neutron absorbers, commonly used materials such as boron carbide and gadolinium oxides were tested. However, from a practical point of view, it is required to analyse all suitable materials which could theoretically be good neutron absorbers. For this reason, extensive (also calculation-time consuming) mapping analysis was prepared for all standard elements in the periodic table [61].

This is a very complex and valuable study, where all the essential elements of the periodic table are compared. A script written in Fortran has been developed for automatic calculation, which can select predefined models such as ŠKODA 1000/3, GBC-32 etc. It allows the user to automatically add groups of elements to these models, and change the simulation conditions wet/dry. The results are computed with the required accuracy from the user. The results are then automatically plotted using the script in the OriginPro graphics program [43]. One can see results of reactivity effect for 92 elements compared to the standard subcritical limit 0.95 (red dotted line), and a reference without absorber (green dotted line).

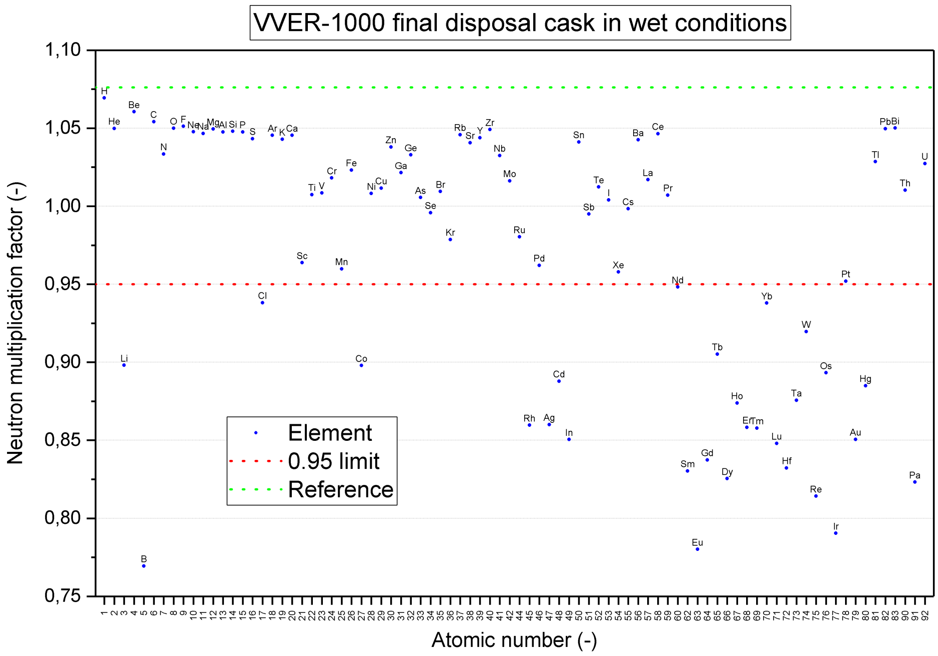

Fig. 3.5: Results of material optimization for model of ŠKODA 1000/3 final disposal cask in wet conditions.

Based on this global study, where neutron absorption properties of many elements can be observed, the most suitable elements were divided into three main groups according to the amount of reactivity decrease in the presented model see Fig. 3.6.

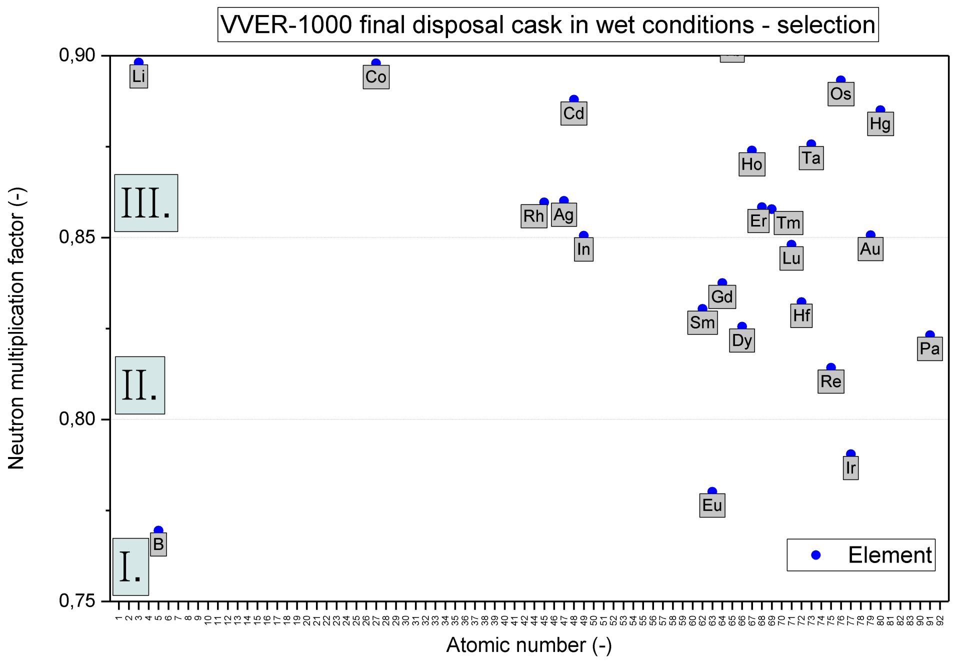

Fig. 3.6: Selected elements divided into three groups.

The selected elements and groups can be seen in tabulated values in Tab. 3.2. In the conservative approach, the highest possible absorption is required. Only group I. for k-eff ≤ 0.80 and II. for k-eff = (0.80-0.85> are further considered. Last group III. for k-eff = (0.85-0.90> containing the most elements (13) is not considered for further studies.

Tab. 3.2: Selected elements (I. and II. group) for further studies.

Group	Element	Atomic number	k-eff
I	B	5	0.76943
I	Eu	63	0.78013
I	Ir	77	0.79041
II	Re	75	0.81420
II	Pa	91	0.82316
II	Dy	66	0.82548
II	Sm	62	0.83038
II	Hf	72	0.83226
II	Gd	64	0.83745
II	Lu	71	0.84805

This presented study of selected elements is essential background for further experimental research. Materials in categories I. and II. were used for prototype production. These two categories are input data for further evaluation.

The following requirements play a crucial part in prototype production, required for the experimental verification:

- Physical properties (melting point, density, chemical resistance etc.) of selected elements.
- Price of selected elements. The final product should be suitable from an economic point of view.

3.3.2 Application for Spent Fuel Cask - PWR type

The efficiency of the concept is demonstrated on the criticality safety analysis of the GBC-32 spent fuel cask. The GBC-32 benchmark cask is a simplified cask for burnup credit benchmark purposes [44]. For neutron absorber concept feasibility study, a 2-D model of the cask was analysed. The fuel was assumed as a uniform material. Uncertainties were not taken into account, and it is assumed that they are at the same level as 2-D simplification. Therefore a 0.95 limit is used in the analysis. Fuel composition was calculated by TRITON code. The criticality limit is recommended by IAEA, and subsequently restricted by national regulatory decrees (e.g. 10 CFR 50.68 in the US). Initial analysis with boron and gadolinium was studied in [44].

Spent fuel composition for burnup credit was calculated with an actinide and fission product level with U.S. Nuclear Regulatory Commission (NRC) approved set of 28 nuclides [63]. Isotopic correction factors were not applied, since the absorber reactivity worth is around 10 times larger.

GBC-32 geometry model in Fig. 3.7 is comprised of 32 fuel assemblies of Westinghouse OFA 17x17 design. Fixed neutron absorbers were placed in all 25 guide tubes of selected fuel assemblies. Filling only a fraction of guide tubes in all assemblies can be slightly more effective with the same amount of absorbers. However, filling all guide tube positions in selected fuel assemblies results in cost savings during reactor outages when fixed absorbers are being installed. Fuel assemblies are placed in aluminum tubes with absorbing BORAL[2] panel inserted between adjacent tubes. The BORAL panel is 0.2057 cm thick with boron density of 0.0225 g B-10/cm^2. The cask is flooded with unborated water.

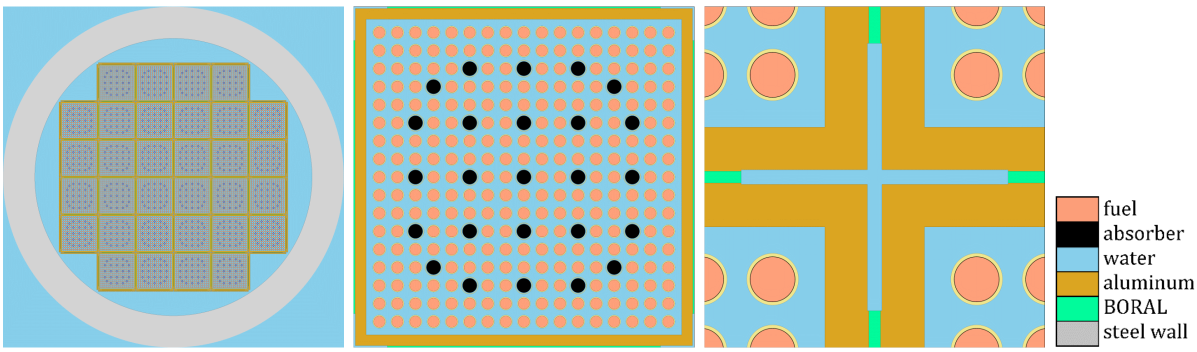

Fig. 3.7: Model of GBC-32 spent fuel cask in Serpent 2. From left to right: fuel cask, fuel assemblie 17x17 design, BORAL position.

[2]Neutron absorber composite, known as BORAL, is used as a absorber material in dry storage casks or spent fuel pool storage racks. Boral is made by mixing boron carbide granules and aluminum powder [45].

GBC-32 results without neutron absorber

A comparison of the current solution with the newly proposed neutron absorber concept is analysed only for zero cooling time for wet conditions, with a maximum of 5 wt% U-235 enrichment for different burnups. Results for different enrichment and cooling times (0, 5, 10 and 20 years) without neutron absorbers are depicted in Fig. 3.8. The standard criterion for reactivity (k-eff = 0.95) is not met even for 3 wt% U-235 for fresh fuel, or 5 wt% U-235 up to 35 000 MWd/MTU [46].

Due to the increasing demands on nuclear fuel and its increasing enrichment, this is a complication for nuclear fuel storage. The required fuel burnup level is not always achieved. In exceptional cases, small breaks in the integrity of the nuclear fuel may occur (so-called leaking fuel). This fuel assembly cannot continue to operate in the reactor. Furthermore, this fuel is classified as used fuel. Historically, these leaks have also occurred after only one cycle of operation. This phenomenon is primarily caused by mechanical damage to the fuel pin (fuel cladding).

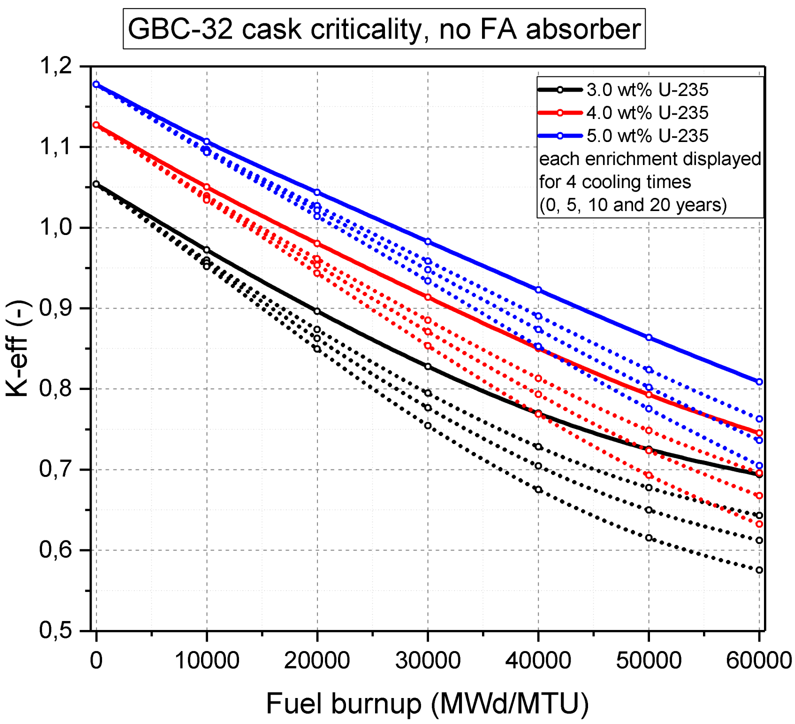

Fig. 3.8: Spent fuel cask criticality without neutron absorber concept.

GBC-32 results with neutron absorber

Absorber materials were selected by the most common chemical composition, in these forms: one carbide, 4 oxides and 3 metals. Also, this was based on the results in Tab. 3.2.

Tab. 3.3: Selected elements for GBC-32 studies [46].

Element	Density [g/cm^3]
B_4C	2.52
Sm_2O_3	8.35
Eu_2O_3	7.42
Gd_2O_3	7.07
Dy_2O_3	7.80
Hf	13.31
Re	21.02
Ir	22.56

All 8 selected absorber materials significantly decrease system reactivity, and 0.95 criticality limit is achieved with a significant margin even for fresh fuel. Absorber material with 0.45 cm radius was placed inside 0.5 cm steel cladding tube. The possibility to save absorber material by using it with an inner hole was analyzed. However, only 1/6 of absorber mass can be saved, and the added manufacturing cost are not justified. The results favor full absorber design.

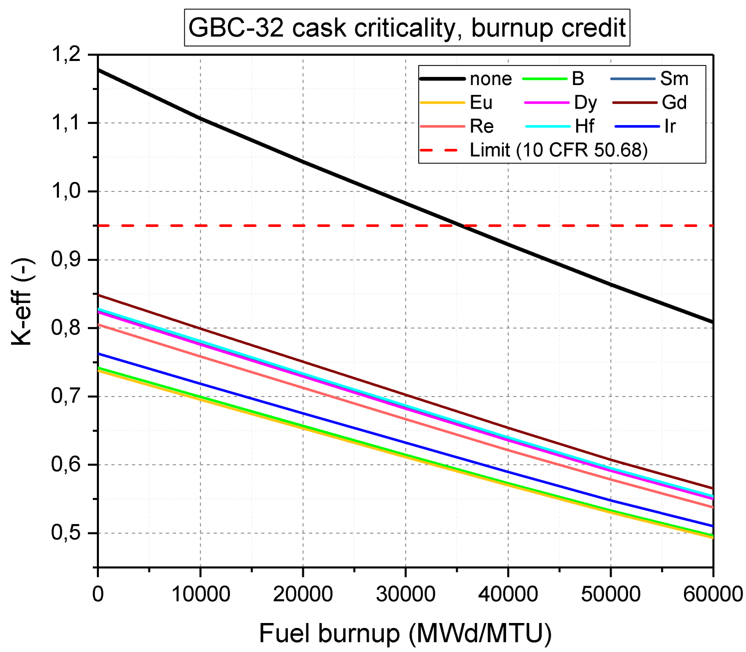

Fig. 3.9: Model of GBC-32 spent fuel cask criticality with selected materials.

Neutron absorber loading in GBC-32 cask

Criticality safety of GBC-32 cask is achieved by two measures: using BORAL absorber sheets, and burnup credit. Each of the criticality measures can be replaced by using fixed absorbers.

In the first case, criticality is achieved by using BORAL absorber sheets and fixed absorbers. In this case, BORAL content can be lowered by to to 2.2% of the original BORAL content for the most effective absorber (europium). The least effective absorber from the analysed batch (gadolinium) can lower BORAL content to 17% of the original BORAL content.

In the second case, criticality is achieved using fixed absorbers and burnup credit without changes in minimum burnup. BORAL sheets are removed, and fuel assembly pitch is decreased even by removing aluminum tubes. In order to securely place fuel assemblies, 2 mm aluminum plate between assemblies was assumed to remain in the cask. The inner cask wall radius decreased by almost 7.5 cm, and consequently, cask wall mass decreased by 8%. Moreover, system reactivity still has margins. The multiplication factor varies between 0.80 and 0.89 for various absorber materials. The least effective absorber was analyzed for lowering the number of fuel assemblies loaded with fixed absorbers. Various absorber loading schemes summarized in Fig. 3.10 were analyzed. Both number of absorber-loaded fuel assemblies, as well as their positions in the cask influence cask criticality, see Fig. 3.11. The higher number of fuel assemblies loaded with the absorber generally improves safety margins. Recommended absorber loading scheme is shown in Fig. 3.12, grey = neutron absorber in FA, yellow = without absorber in FA.

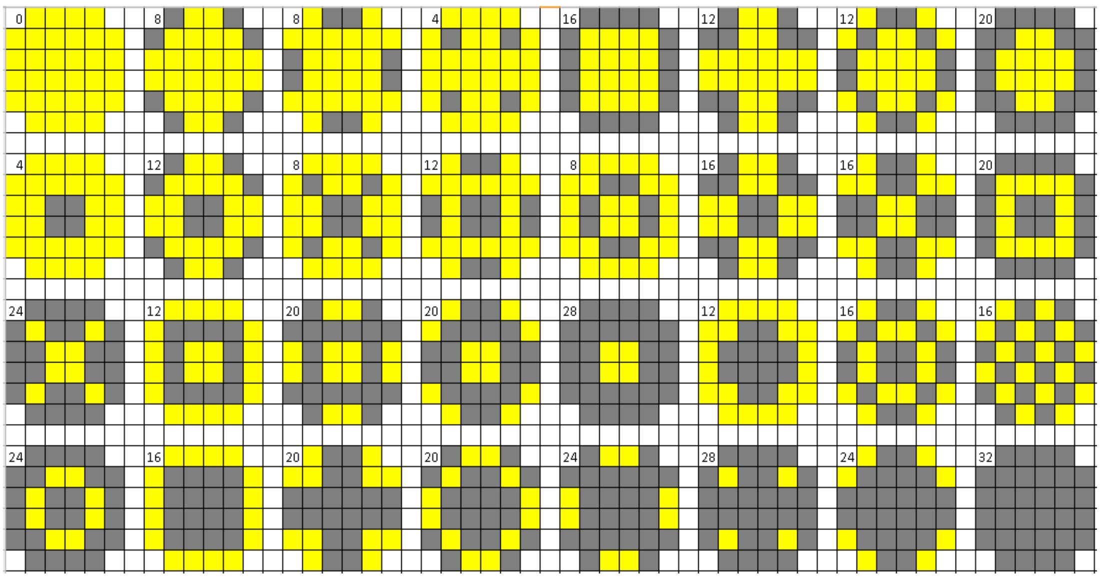

Fig. 3.10: GBC-32 absorber loading scheme.

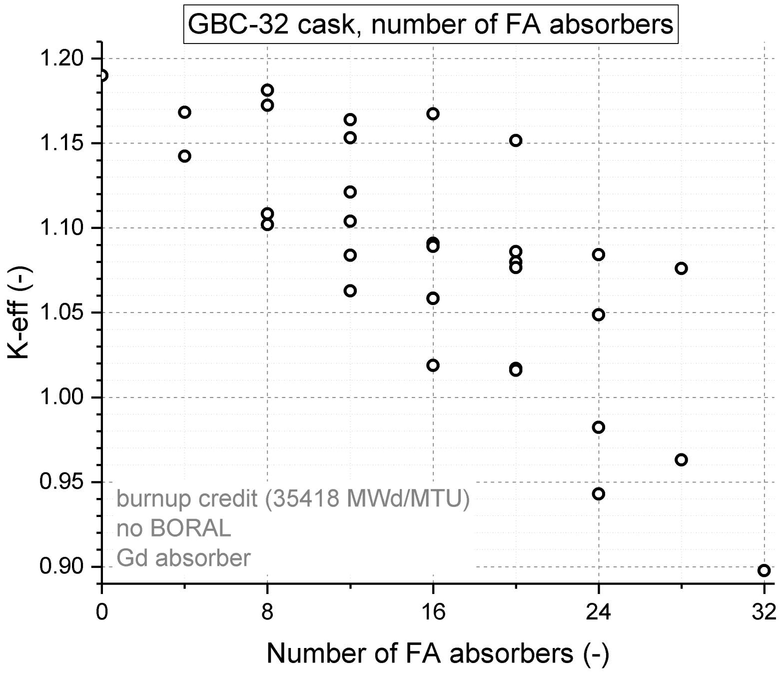

Fig. 3.11: GBC-32 criticality with partially loaded fixed gadolinium absorbers.

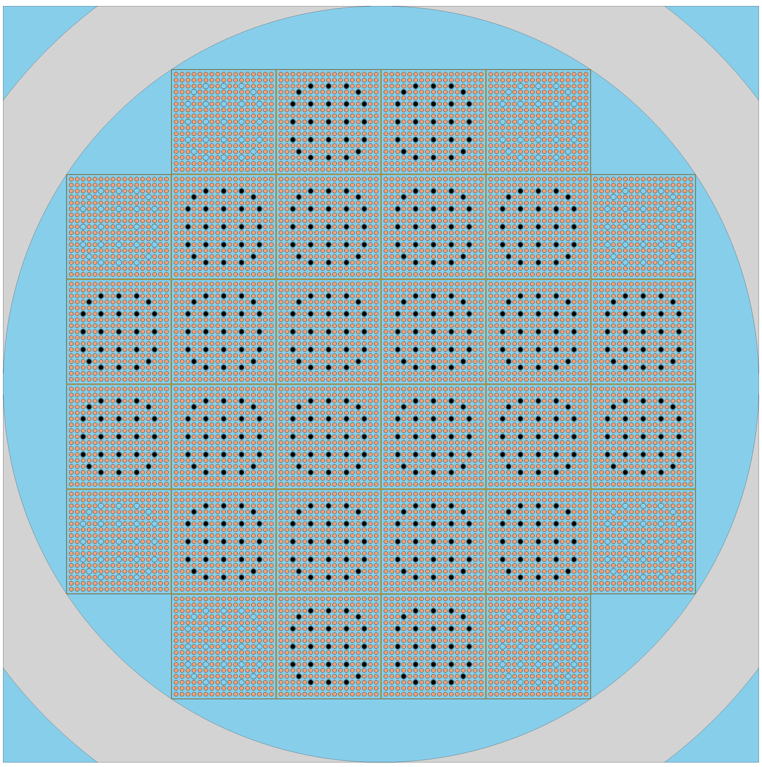

Fig. 3.12: GBC-32 cask design with fixed absorbers in 24 FA (black dots) from 32 FA.

Neutron absorber in steel

As was shown, placing absorber rods in guide tubes directly into the fuel assembly significantly decreases cask reactivity. However, it can be more economical to use the absorbers as a steel alloy component. These results in Fig. 3.9 show the case when the absorber material is an alloy mixture with Stainless steel 304. For future research associated with industrial mass production, it is necessary to investigate possibilities when the absorber material is in a form of a steel alloy component.

This analysis presents the economics of the solution. Using burnup credit (35 418 MWd/MTU) and a new fixed absorber concept allows a complete removal of BORAL sheets, and also allows decreasing the fuel assembly pitch. It also allowed decreasing the inner cask wall (by more than 7 cm), and the mass of the cask wall diameter (by more than 7.5%). Even the minimal content of an absorber material within steel has the same effect as usage of BORAL.

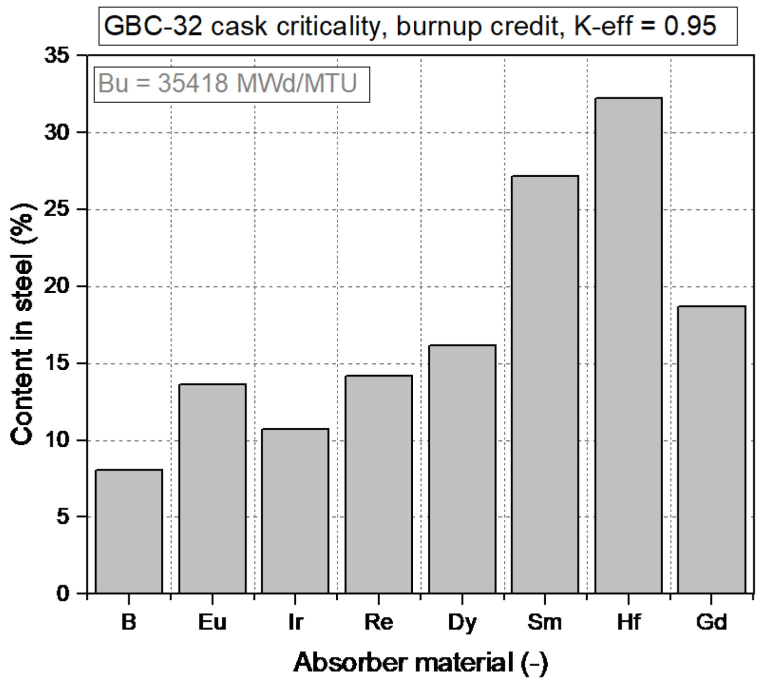

Fig. 3.13: GBC-32 criticality 0.95 with burnup credit for different content of absorber materials in steel.

3.3.3 Application for Spent Fuel Cask - VVER type

Nuclear fuel is being cooled in the spent fuel pool near reactor for up to 10 years, and then transported for long-term storage. There are two common solutions, wet and dry storage. A wet storage pool is very similar to a spent fuel pool, only with capacity typically for the whole country. Using fixed neutron absorbers for wet storage is, therefore very similarly efficient as in the spent fuel pools. Dry storage is based on spent fuel casks that can be constructed and licensed both for transport and storage. CASTOR 1000/19 cask is a typical dual-purpose cask licensed for transport and storage of spent nuclear fuel.

Two-dimensional model of CASTOR cask was implemented based on data from [62]. Loading factor of fixed neutron absorbers is 68%, intentionally slightly less than in the spent fuel pool see Fig. 3.14. Absorbers are loaded in 13 central assemblies out of 19 assemblies in the cask. Criticality safety is secured by fresh fuel assumption and steel tube borated to 1.0 wt%. Regular lattice with 297 mm assembly pitch was adopted for the calculations (description in subchapter 2.3.4).

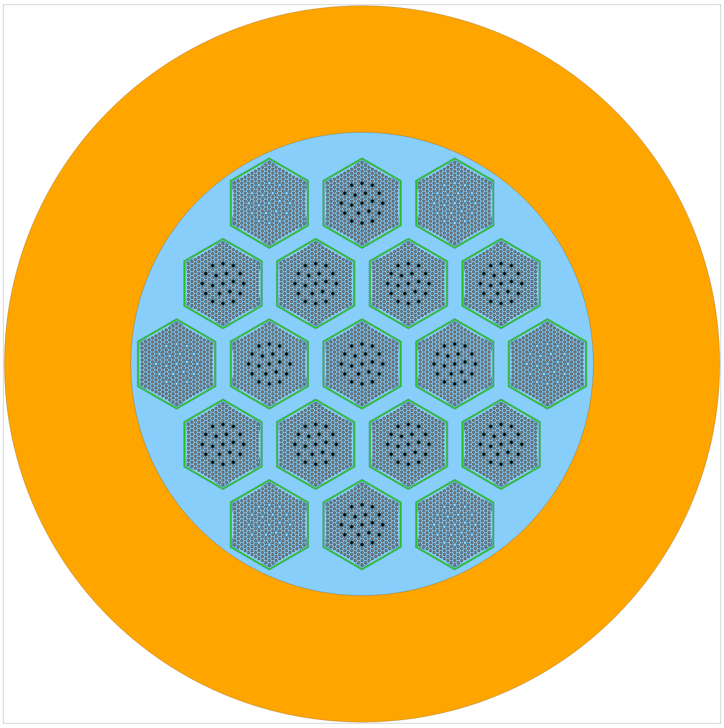

Fig. 3.14: CASTOR spent fuel cask criticality model in Serpent 2.

Cask criticality with fixed neutron absorbers is summarized in Fig. 3.15. Burnup credit is not required, since neutron multiplication factor is around 0.92 without uncertainties. Similarly to spent fuel pool, fixed neutron absorbers can be combined with design changes. In the case of spent fuel casks the benefit of using fixed neutron absorbers is cask wall mass reduction rather than capacity increase. Smaller cask size is achieved by decreased pitch, from 259 mm to 265 mm from original 297 mm. Regular assembly cell volume decreases to between 81% and 84%, and related cask wall inner diameter decreases while the shielding wall thickness of 405 mm remains unchanged. Cask wall mass savings are listed in Tab. 3.4. Neutron absorbers can save around 10% of cask costs of CASTOR 1000/19 casks, comparable to 8% cost savings for PWR-fueled GBC-32 casks.

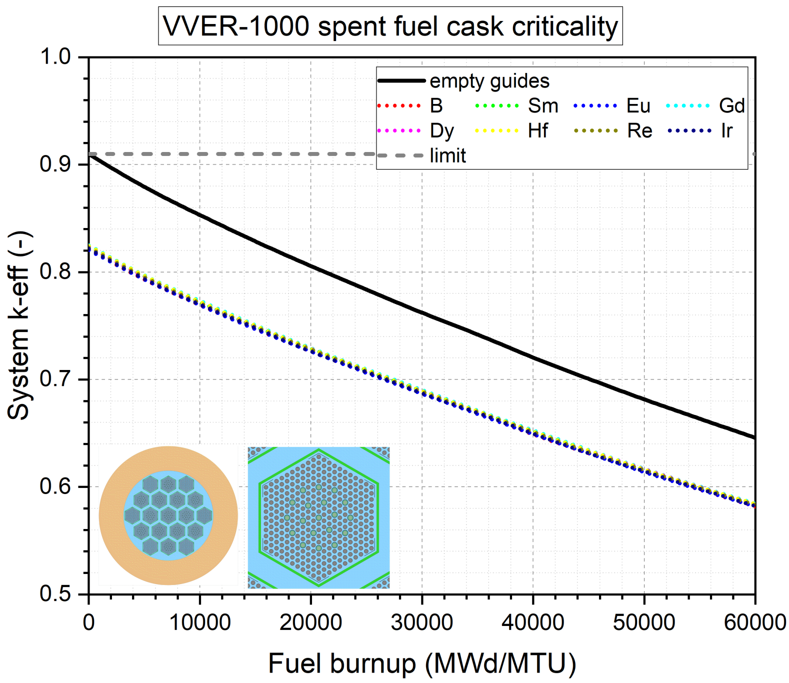

Fig. 3.15: CASTOR spent fuel cask criticality with fixed neutron absorbers.

Tab. 3.4: CASTOR spent fuel cask wall mass savings while using fixed neutron absorbers.

Absorber	Relative cask wall volume	Absorber	Relative cask wall volume
empty	1.000	empty	1.000
Gd	0.915	Re	0.908
Hf	0.912	Ir	0.902
Sm	0.911	Eu	0.899
Dy	0.911	B	0.899

3.3.4 Application for Spent Fuel Pool

The efficiency of fixed neutron absorbers in used spent fuel handling facilities was analyzed in [39]. VVER-1000 nuclear fuel was chosen for the analysis. The first spent fuel facility is where the spent fuel is stored in the spent fuel pool.

Generic VVER-1000 spent fuel storage pool for V-320 reactor specification was modelled in Serpent 2 code, with ENDF/B-VIII.0 nuclear data library. Uniform loading of 5.0 wt% U-235 enriched fuel in the pool was modelled in a 3-D infinite array of 12 fuel assemblies with 9 selected fuel assemblies loaded with fixed absorbers. Therefore, the pool capacity is equipped with fixed absorbers by 75%, which defines the loading limit for subsequent fuel handling facilities. Criticality safety is maintained by placing fuel assemblies in absorbing steel tubes in 288 mm pitch and 1.0 wt% boron content in the steel, see Fig. 3.16.

Fuel depletion and burnup credit methodology for the VVER-1000 model was consistent with the initial GBC-32 analysis. All 18 guide tube positions were loaded with fixed absorbers. Results are summarized in Fig. 3.13. Burnup credit is not required since neutron multiplication factor without assuming uncertainties is slightly below 0.92.

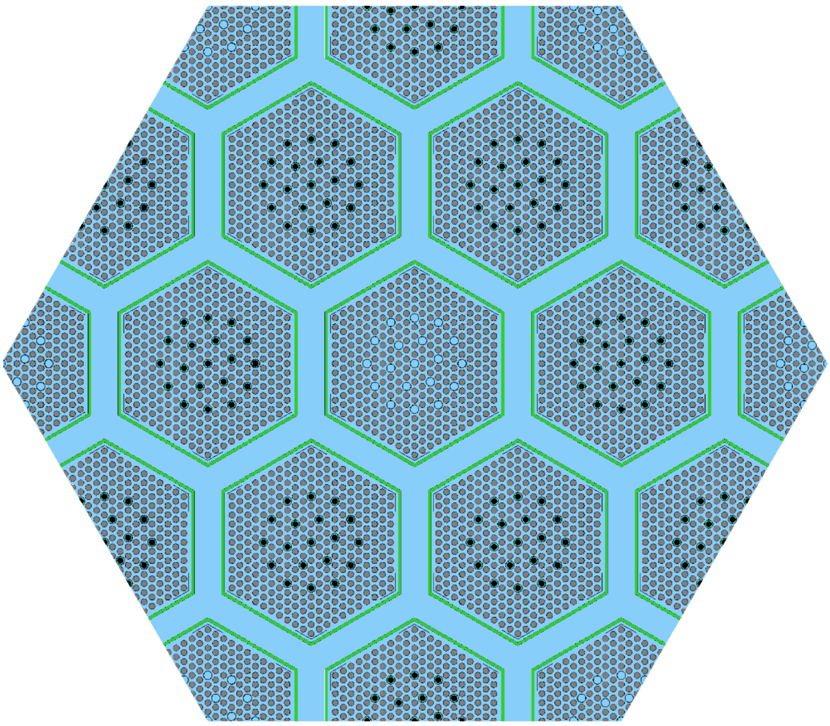

Fig. 3.16: VVER-1000 spent fuel pool criticality model in Serpent 2.

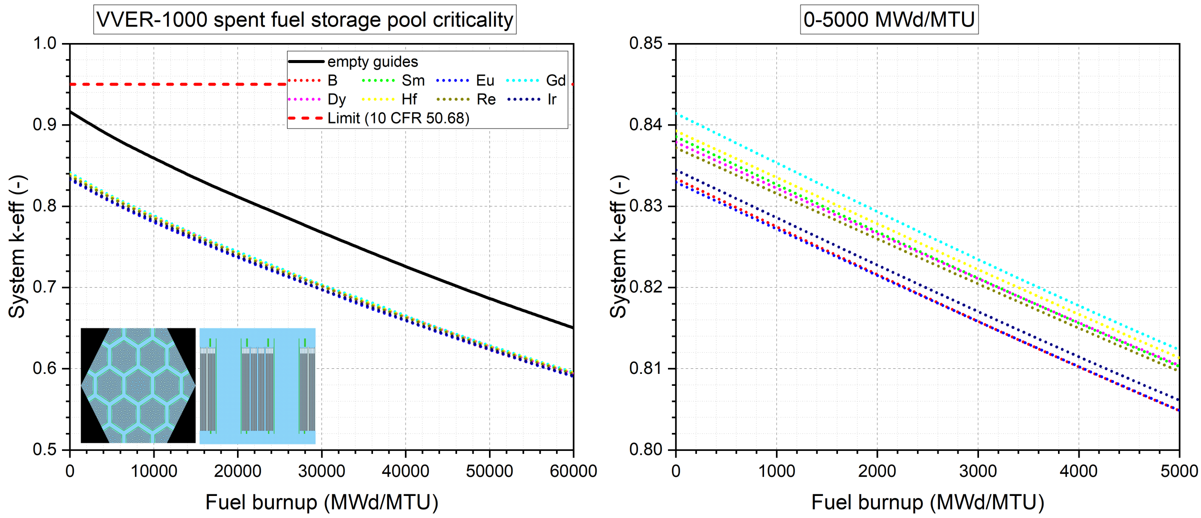

Fig. 3.17: VVER-1000 spent fuel pool criticality with fixed neutron absorbers, zoom 0-5000 MWd/MTU (right).

This subcriticality level can be maintained by simultaneously introducing fixed neutron absorbers and decreased assembly pitch. The first measure decreases system criticality, while the latter increases it. More compact racks consequently reduce regular cell volume that can be used to enlarge pool capacity. For the strongest absorber (boron), the regular cell volume reduction is 82%, while for the least effective absorber (gadolinium), the reduction stands at 85% (see Table 3.5 for comparison of pool capacity). Associated fuel pitch varies from 260 mm to 264 mm, from a reference value of 288 mm.

Tab. 3.5: VVER-1000 spent fuel pool capacity increase while using fixed neutron absorbers.

Absorber	Relative pool capacity
empty	1.000
Gd	1.180
Hf	1.188
Sm	1.191
Dy	1.192
Re	1.195
Ir	1.212
B	1.219
Eu	1.221

3.3.5 Application for Final Disposal Cask

Application for final disposal cask was shown for an updated ŠKODA 1000/3 cask, which is the cask used for Czech deep geological repositories. It is a cask designed with three VVER-1000 fuel assemblies see in Fig. 3.18. For the neutron absorber concept, a 3-D model of the cask was analysed. The fuel was assumed uniform in all fuel rods as one material with 5.0 wt% U-235 fuel enrichment. Uncertainties were not taken into account. Therefore, the legislative limit of 0.95 on neutron multiplication factor (system k-eff) was replaced by a calculation limit of 0.88, which takes into account uncertainties in fuel composition (conservatively 0.05), fuel assembly manufacturing uncertainties, and nuclear data bias (conservatively 0.02). More about the computation model is in [39].

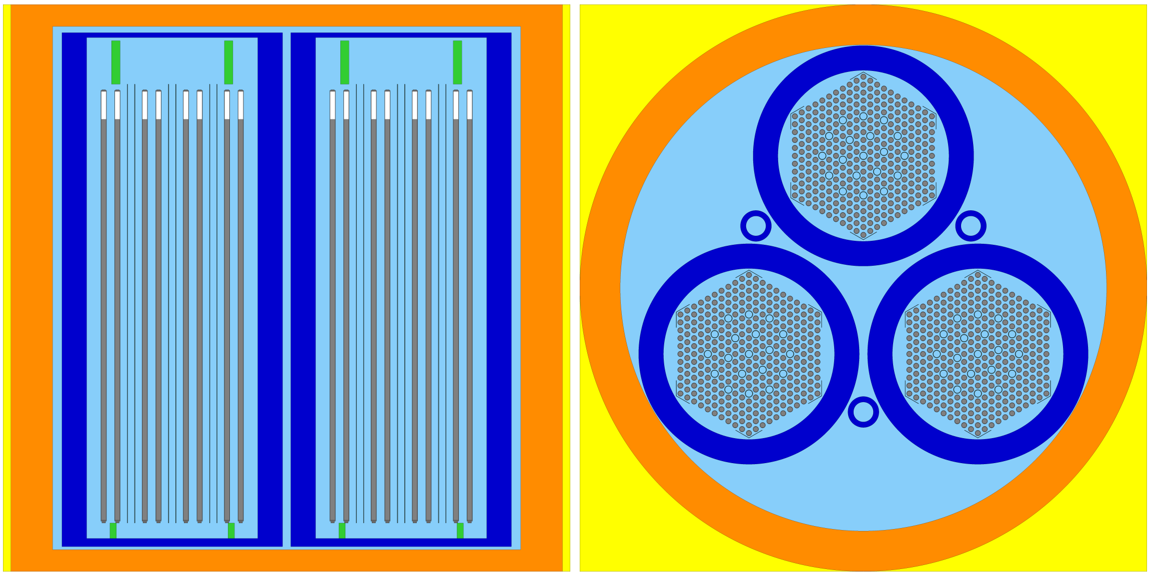

Fig. 3.18: ŠKODA 1000/3 final disposal cask, 3D model in Serpent neutronic code.

ŠKODA 1000/3 final disposal cask criticality for current design specifications is achieved with stainless steel tubes around each fuel assembly. It should be noted that the design of the tubes is the result of required cask lifetime and strength properties from structural analysis rather than criticality design. However, with burnup credit it can be easily shown that the cask is safely subcritical. Minimal fuel burnup is 12 745 MWd/MTU, which is achieved by all fuel assemblies in their first fuel cycle in the nuclear reactor operation, see Fig. 3.19.

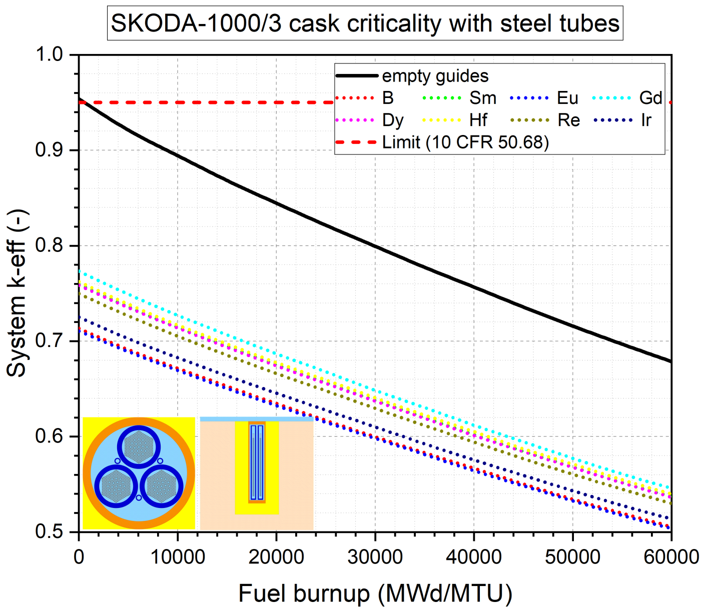

Fig. 3.19: Final disposal cask criticality with steel tubes.

Neutron absorbers were placed in all 18 guide tube positions, see Fig 3.19. Neutron absorber materials have the same diameter as the fuel pellet, and it is placed inside its own steel cylinder cladding. The thickness of stainless steel tubes around fuel assembly can be decreased, since the subcriticality can be achieved by neutron absorbers, see Fig. 3.20. Tube thickness can be decreased from the original 40 mm to 14 mm, but further decrease of tube thickness is not feasible since the tube creates a neutron trap between tubes.

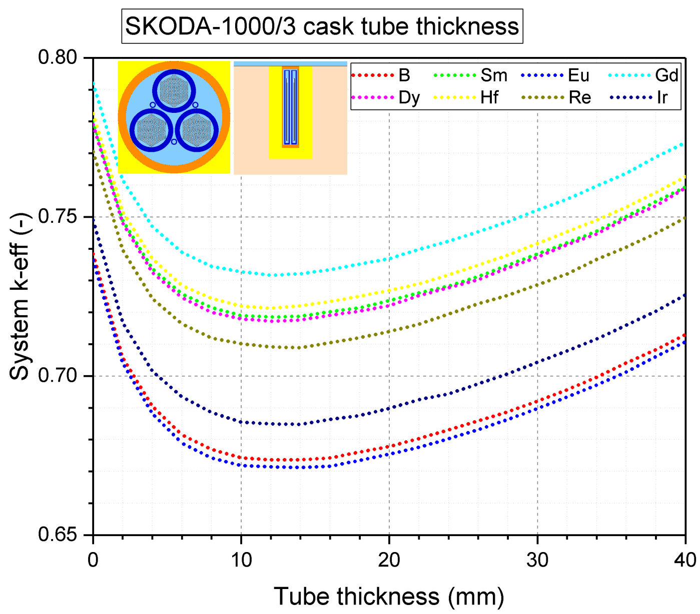

Fig. 3.20: Final disposal cask tube thickness.

Although the final disposal cask criticality can be achieved without the neutron absorbers, their use significantly reduces cask criticality and allows for other cask designs. In addition, the neutron absorber can be placed inside the fuel assembly in early storage stages (for example, in the spent fuel pool), and reduce back-end cost by using the neutron absorbers in more facilities:

- **Storage** – spent fuel pools and casks
- **Transport** – spent fuel casks
- **Disposal** – final disposal casks

3.4 Practical part - prototype production

The basic premise of the whole concept is the inseparability of the fixed neutron absorber. This practical part of the research is focused on development of a fixed connection between absorber and fuel assembly. Obviously, the concept is only for selected types of fuel assemblies. And for our new prototype, we focused on the VVER-1000 fuel assembly. This part of the research is joint research with the company TES a.s.

In this part, two approaches of joint development are presented: one conservative mechanical solution, and one progressive solution using chemical method.

At the end of this part, the final form of the absorbent material used is presented.

3.4.1 Mechanical junction

The primary focus is on the traditional mechanical method. First of all, the analysis of top of the fuel assembly was performed. The junction location is at the top nozzle, since it is the place where standard control rod cluster is inserted, as one can see in Fig. 3.21.

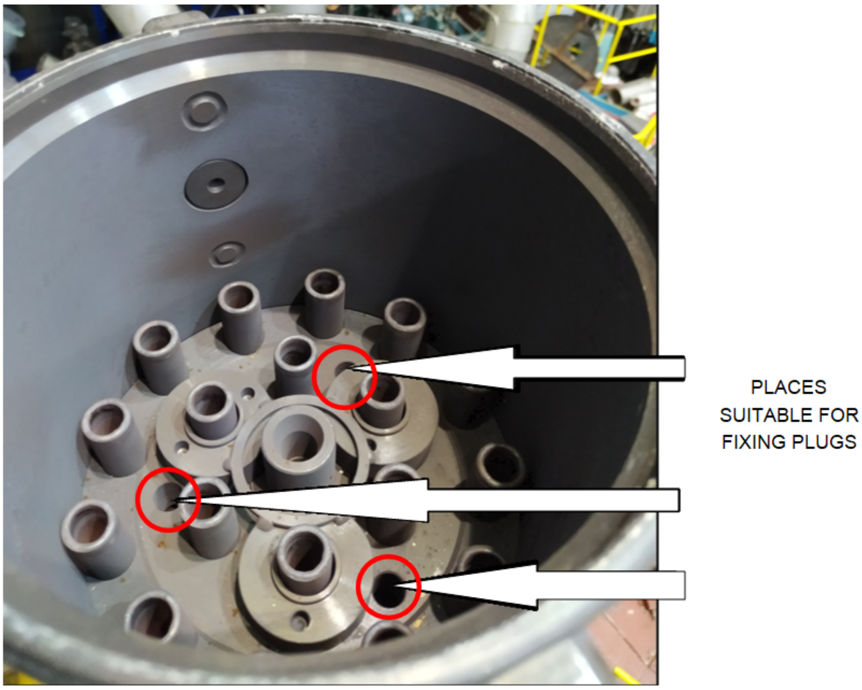

Fig. 3.21: Top of the VVER-1000 fuel assembly, location of the inseparable joint.

There are three holes in the fuel assembly top nozzle, which are used to lock the plug after it is inserted into the nozzle as the basis of mechanical-based junction. The plug in Fig. 3.22 itself is a stainless steel disc with the following elements:

- 18 holes with threads for mounting fixed neutron absorbers
- locks for handling the plug
- 3 threaded holes for mounting locking elements
- groove for securing the cover cap after locking the plug

The mechanical junction locking mechanism can be manufactured by either push thorns or pull thorns. The thorns and fixed neutron absorbers are inserted into the holes of the plug locking mechanism, and locked after pressing the cover cap against the thorns in the plug, see Fig. 3.22. The cover cap installation subsequently makes the junction permanent without the possibility of unlocking it.

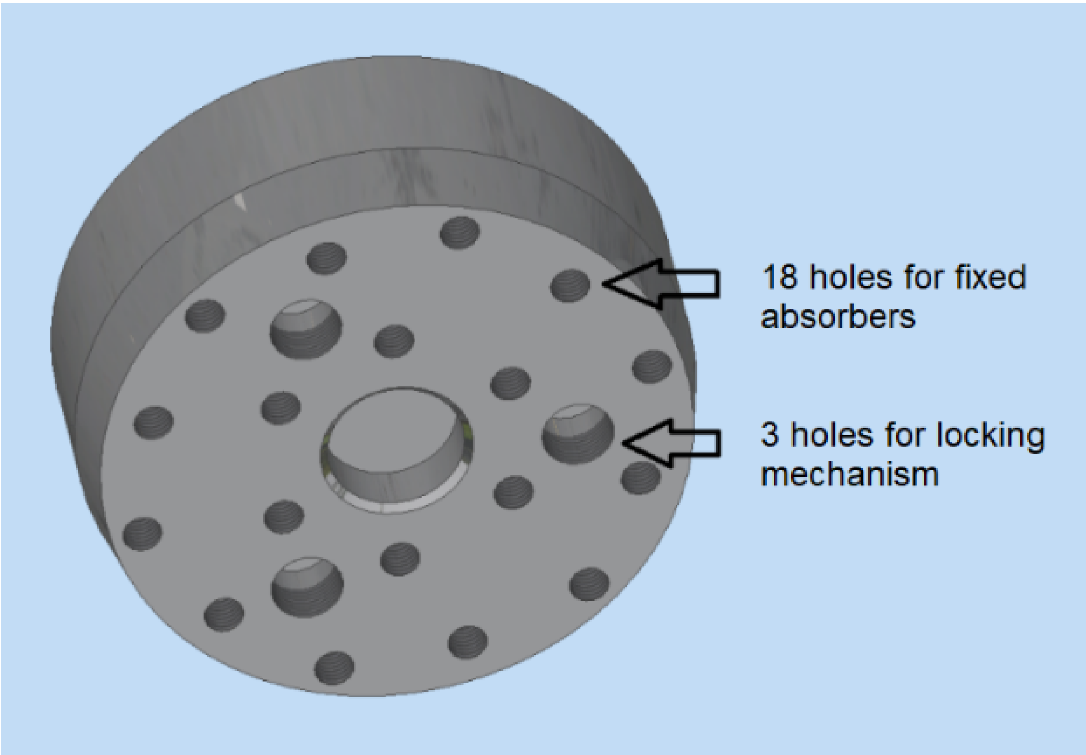

Fig. 3.22: Mechanical junction – plug with holes for absorbers and lock thorns.

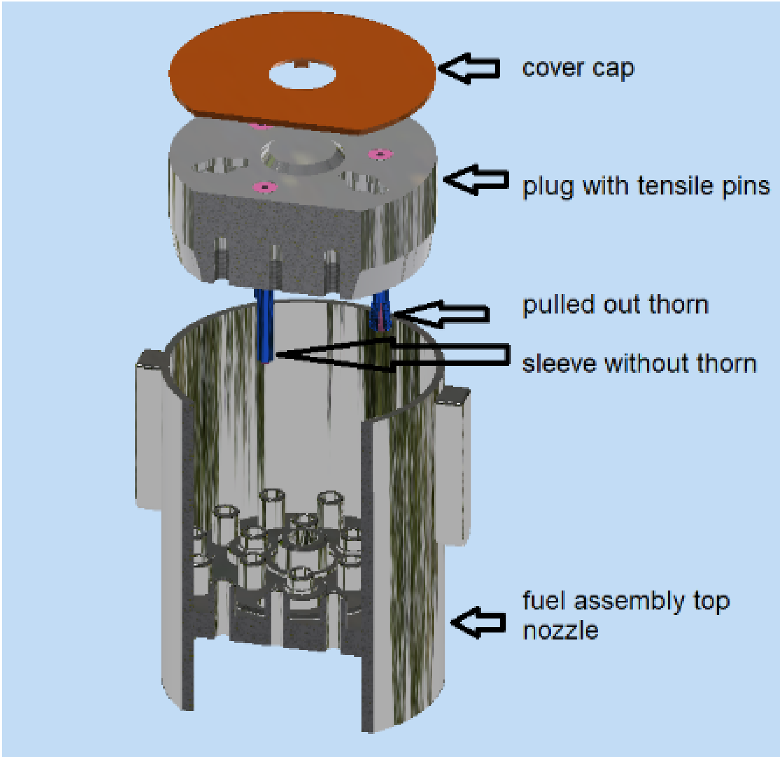

Fig. 3.23: Mechanical junction – fuel assembly top nozzle and mechanical plug.

In this variant, it is assumed that the plug of the fuel assembly will be secured with three towing pins. The pins will be subjected to tensile stress when securing the plug. After pulling the pins into the pin capsule, the ends of the pin will expand to the wall of the pin capsule and become fixed. The ends of the locking pins will be subjected to tensile stress see in Fig. 3.24.

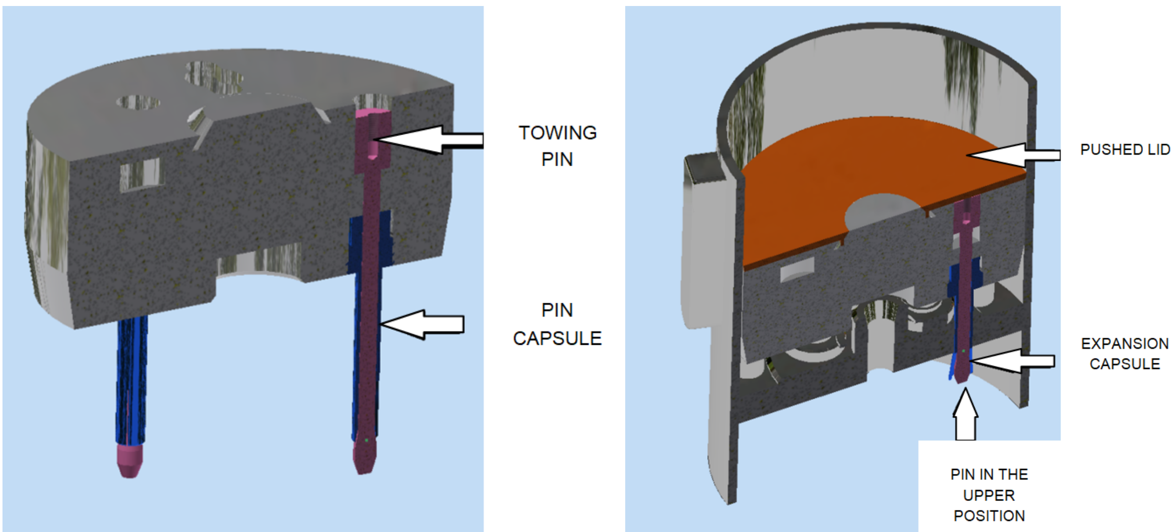

Fig. 3.24: Proposed solution with throwing pins.

The following Figs. 3.25, 3.26, 3.27 show the individual parts of the fixing system in detail.

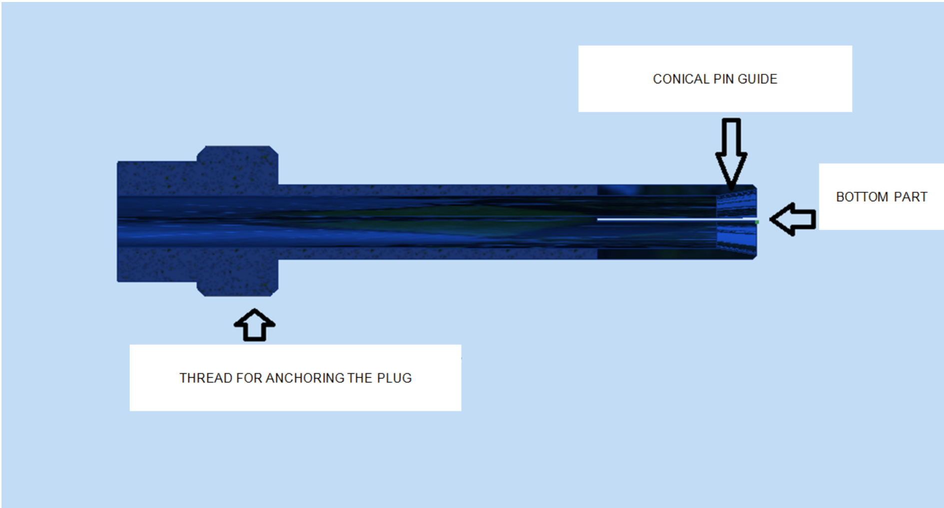

Fig. 3.25: Pin capsule.

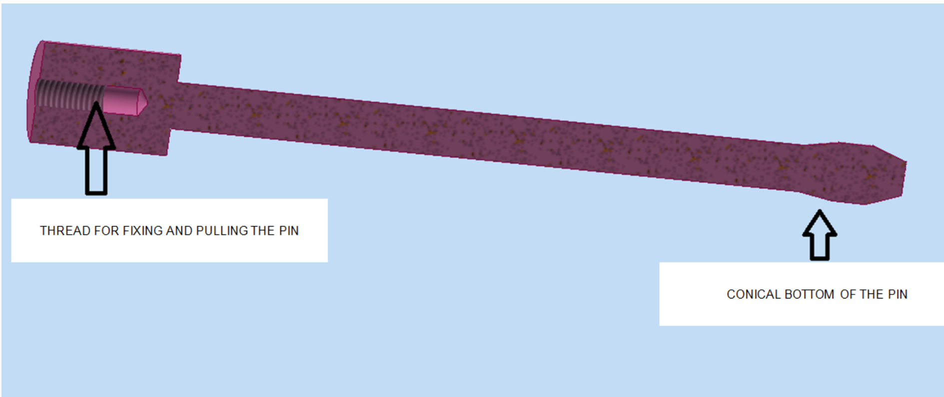

Fig. 3.26: Throwing pin.

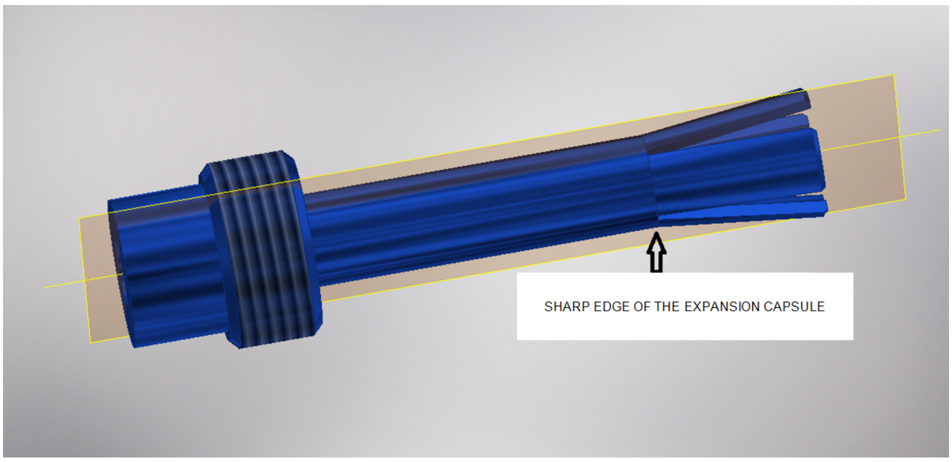

Fig. 3.27: Fixed throwing pin in pin capsule.

3.4.2 Chemical junction

This is a very progressive method, both unique and atypical for the nuclear industry. As an alternative, a possible proposal for the use of two-component chemical technology of adhesives and potting compounds was selected and tested.

These are modern technologies used in the entire spectrum of industry and services (e.g., automotive industry, the transport technology industry, electronics, etc.). Three types of material with very high mechanical properties were selected, including environmental resistance, and various mixtures of chemicals:

1. Sika Power 1277 - very tough two-component epoxy compound hardened by polyaddition reaction [47].
2. Sika Power 1200 - high strength two-component epoxy compound hardened by polyaddition reaction [48].
3. Sika Biresin RG53 - solid two-component polyurethane mixture hardened by polyaddition reaction [49].

Several experiments were performed in compliance with EN 1465 (Determination of tensile lap-shear strength of bonded assemblies.), ISO 9142 (Adhesives — Guide to the selection of standard laboratory ageing conditions for testing bonded joints.), and DVS 1618 (Determined procedure of artificial aging in laboratory conditions.). The test results showed that the best results were achieved with a polyurethane-based material. The adhesion was at an excellent level and the strength was sufficient, so further experimental tests will be conducted with Sika Biresin RG53. Also, there will be simulation deposition at a depth of 10 m below the water. The application equipment is now being prepared, and currently the installation of a robotic arm is under way that will apply the filling of the required space, including mixing of 2K polyurethane.

Summary

Potting of the two-component chemical technology compound utilizes resins based on epoxy and polyurethane. For the fixed neutron absorbers, biresin was chosen as the material. Its density is 1.23 g/cm^3, chemical composition $C_{15}H_{10}N_2O_2$, and thermal stability up to 65 °C [47]. Composition of irradiated biresin was calculated by Serpent 2 code, and only 0.16% of the material changed, mainly due to the production of C-14 radioactive carbon. Potting was experimentally verified in 10 m water depth pressure.

The described chemical approach has never been used in the nuclear industry. However, it is a unique and modern technology which is worth exploring.

Fig. 3.28: Chemical junction – potting inside the plug and absorbers holder.

3.4.3 Absorber material form

A complex task is the choice of the material itself and its form. Several forms of absorber material were considered for our experiments (detailed in Czech utility model n. 34 636 - Neutron absorber). For economic reasons, Gd_2O_3 and Sm_2O_3 in powder form were the form and materials chosen.

This is the most economical solution. The price of this element is around 50 USD/kg for samarium oxide and 100 USD/kg for gadolinium oxide. The price depends strongly on the purity of the product that is required for electronics, but not for the neutron absorber application. For our applications, 99.99% purity is sufficient. Potential impurities were also considered in the study, but lower purities can be considered due to their similarity to other absorbent materials, such as Eu_2O_3 [50].

- **Gd_2O_3 powder**

Tab. 3.6: Gd_2O_3 properties [51].

Density	7.07 g/cm^3
Melting point	2 420 °C
Solubility in water	insoluble

Fig. 3.29: Gd_2O_3 in powder form [52].

- **Sm_2O_3 powder**

Tab. 3.7: Sm_2O_3 properties [51].

Density	8.35 g/cm^3
Melting point	2 335 °C
Solubility in water	insoluble

Fig. 3.30: Sm_2O_3 in powder form.

4

Experimental verification at LR-0 Research Reactor

The main goal of the this part of dissertation is the experimental neutronic verification of the fixed neutron absorber for VVER-1000 fuel assemblies, at the LR-0 zero-power reactor, within the Open Acces program at the Nuclear Research Institute (NRI). The goal is to determine the critical state parameters for fuel without absorbers and for fuel with absorbers, and to compare them with the calculation. The choice of the position of absorbers (inside the guide tubes instead of absorber tubes outside the fuel assembly), and the choice of the absorber material (Gd_2O_3 and Sm_2O_3 in powder form based on subchapter 3.4.3), are innovative.

4.1 LR-0 reactor

The LR-0 reactor is a pool-type, zero-power light water reactor operated by the Research Center Řež in the Czech Republic. The mock-up core was performed with axially 1/3 shorter VVER-1000 type fuel assemblies, as described in the section below. The control of the reactor is archived by a number of safety rods, or by changing the level of the moderator, as was the case in the setup for absorber verification [53]. Continuous nominal power is 1 kW, with a thermal neutron flux of about 10^9 $cm^{-2}.s^{-1}$, and a fast neutron flux of 2 x 10^8 $cm^{-2}.s^{-1}$ [54], see reactor scheme in Fig. 4.1.

For the experiment, a well-known reference neutron field designed by the Research Center Řež in the Czech Republic was chosen. This neutron field is defined as a permanent and reproducible neutron field, which is acceptable as a measurement reference by a community of users [55],[56].

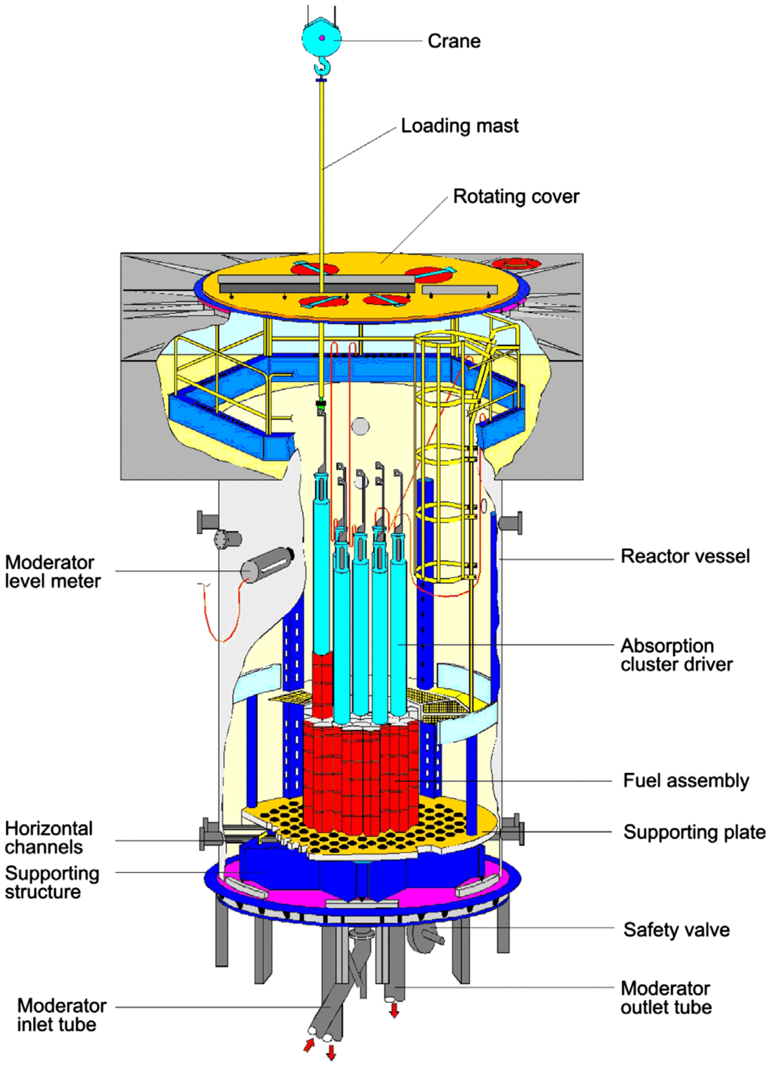

Fig. 4.1: LR-0 reactor scheme [55].

4.2 Experimental arrangement

For the experiment, a reference neutron field with a special core configuration described in detail in [54] was chosen. This configuration consists of six fuel assemblies with around 3.3% nominal enrichment, surrounded by a central experimental dry channel in the LR-0 core. This dry channel in central position was replaced by a special fuel assembly with or without experimental neutron absorbers. The goal of the experimental setup was to experimentally determine critical parameters, namely the moderator level of each variant of cases from which the reactivity of the absorbers can be determined.

As mentioned, the reactor's criticality is reached by adjusting the moderator level to the critical level H_{cr}. The critical level of moderator is defined by the core loading, and specially prepared neutron absorbers inserted in the central fuel assembly.

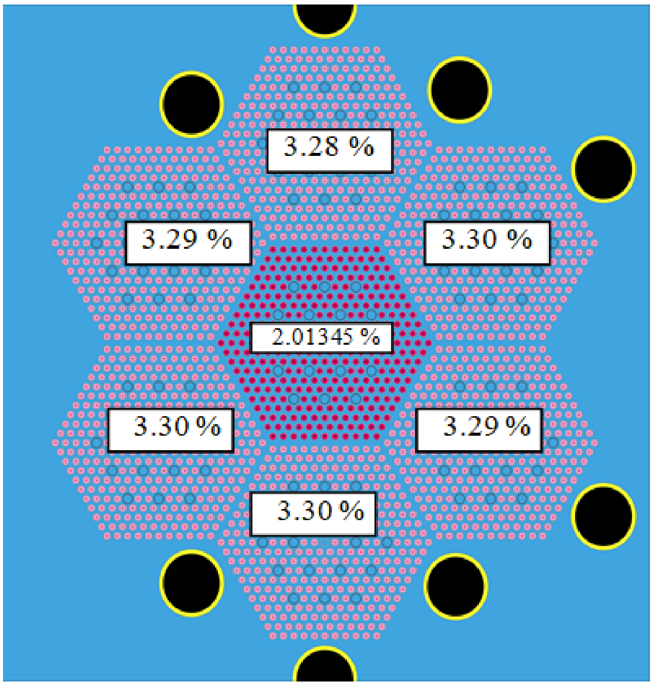

Fig. 4.2: The LR-0 reference core configuration pattern, avg. enrichment of U-235 in the labes.

In order to verify the effectiveness of neutron absorbers, it is advisable to insert as many absorbers as possible into the core. At the same time, it is necessary that the reactivity worth of the inserted absorbers does not exceed the allowed operating values, given by their weight in the fraction of delayed neutrons, and that the reactivity worth of the control clusters at the inserted absorbers is sufficient. The experiment was designed for two absorber materials aligned at the same positions in the core. Next, comparing the condition without absorbers is necessary. The insertion of both absorber materials into the core is the last condition for experimental verification. A total of 4 cases with gadolinium and samarium oxide were measured.

The core configuration chosen for the experiment is a modified GK-07 core, with low-enriched FA at the central position with enrichment of 2.01345 wt% U-235. The other six fuel assemblies have an enrichment seen in Fig. 2. The Monte Carlo code Serpent 2.1.32 with the ENDF/B-VIII.0 library was used to determine the basic parameters of the core. The reason for the choice of the core configuration is its similarity to the LR-0 reference core [54]. A low-enriched FA was placed in the central position to increase the reactivity worth of the measured absorbers.

The neutron absorbers with samarium and gadolinium are based on an oxide powder form placed inside a steel tube (pin). The inner diameter of the steel tube is 9 mm, the outer diameter of the steel tube is 10 mm, and the height is 140 cm. The absorber steel tube is then inserted into guide tubes with an inner diameter of 11 mm.

The process of preparing the absorber powder into the steel tube is summarized in Fig. 3 and Fig. 4. Contrary to the assumptions, it was impossible to fill the pin with the powder with the expected 85% volume fraction. The actual volume fraction used in the experiment was about a half of the expected value, due to a very fine powder structure of the two oxides.

Details of the prepared samples are given in the Table 4.1 below.

Tab. 4.1: Parameters of absorber materials in tubes.

Absorber pin (material/ID)	**Weight of empty tube (g)**	**Weight of filled tube [g]**	**Weight of absorber in tube [g]**	**Height of absorber in tube [mm]**
Gadolinium 1G	335	492	157	1185
Gadolinium 2G	335	492	157	1207
Gadolinium 3G	335	482	147	1350
Gadolinium 4G	335	488	153	1358
Gadolinium 5G	335	496	161	1111
Gadolinium 6G	335	498	163	1337
Samarium 1S	335	462	127	1333
Samarium 2S	335	498	163	1341
Samarium 3S	335	440	105	1338
Samarium 4S	335	494	159	1351
Samarium 5S	335	488	153	1356
Samarium 6S	335	482	147	1358
Samarium 7S	335	482	147	1354
Samarium 8S	335	490	155	1357
Samarium 9S	335	500	165	1357
Samarium 10S	335	422	87	1352
Samarium 11S	335	492	157	1353
Samarium 12S	335	494	159	1355

The bottom “plug” height is 38 mm, the beginning of the column of absorbing material.

Fig. 4.3: The absorber insertion into the steel tube, view of the Sm_2O_3 powder.

Fig. 4.4: A view of the end plugs.

The objective of the experimental verification of the neutron absorber for the VVER-1000 fuel in the LR-0 is to determine the critical water level for four different absorber distribution states:

- Case 1 (reference): core configuration with seven fuel assemblies according to Fig. 4.2, low enriched fuel in the central position is ready for absorbers insertion in the steel tubes, see Fig. 4.5

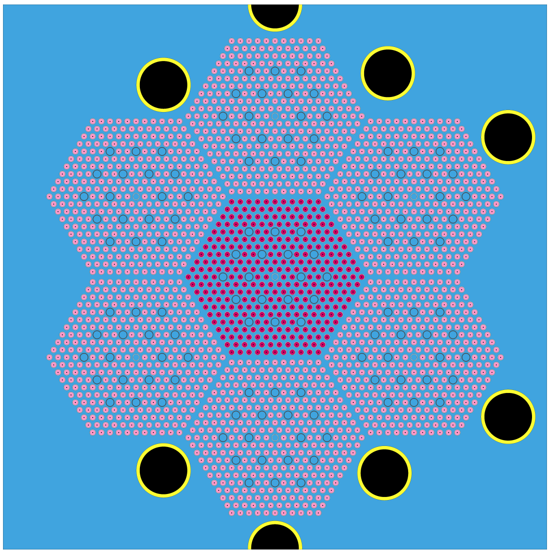

Fig. 4.5: Case 1: core in LR-0 reactor (without absorbers).

- Case 2 (Sm): Case 1 with 6 pcs. samarium absorbers inserted into the inner row of guide tubes of the central fuel assembly, see Fig. 4.6.

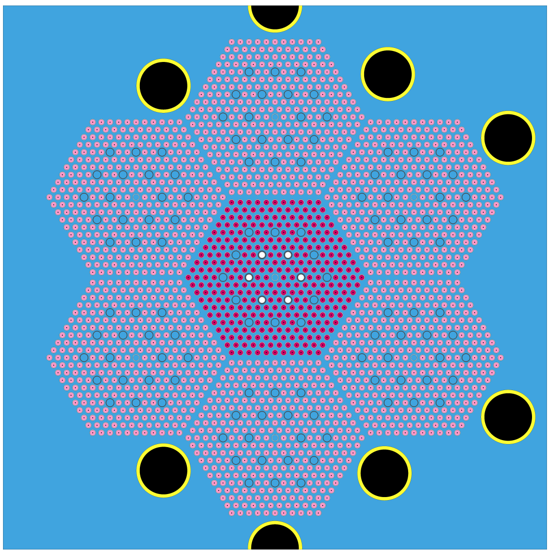

Fig. 4.6: Case 2: core in LR-0 reactor (6 pcs. Sm absorbers – white dots).

- Case 3 (Gd): Case 1 with 6 pcs. gadolinium absorbers inserted into the inner row of guide tubes of the central fuel assembly, see Fig. 4.7.

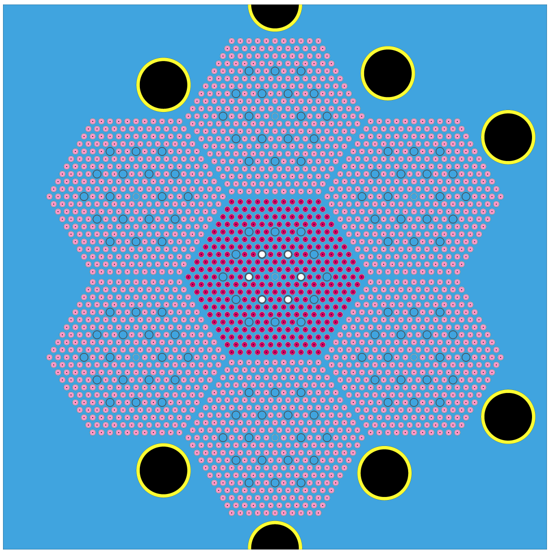

Fig. 4.7: Case 3: core in LR-0 reactor (6 pcs. Gd absorbers – white dots).

- Case 4 (Sm + Gd): Case 1 with 6 pcs. gadolinium absorbers inserted into the inner row of guide tubes of the central fuel assembly and 12 pcs. samarium absorber inserted into the outer row, see Fig. 4.8

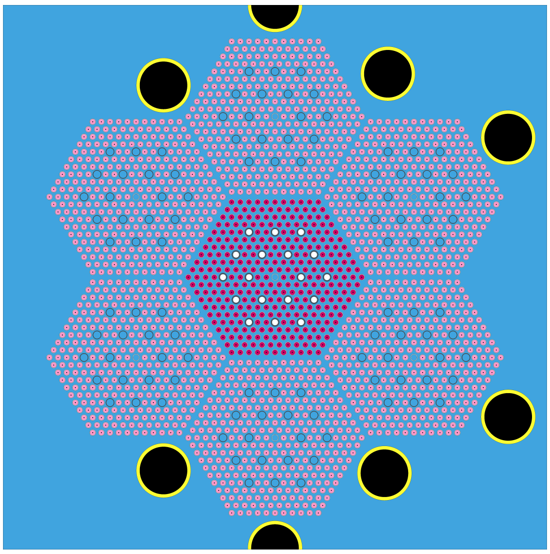

Fig. 4.8: Case 4: core in LR-0 reactor (6 pcs. Gd + 12 pcs. Sm absorbers – white dots).

Inserting 6 samarium absorbers in the core (Case 1 and Case 2) was designed for determination of samarium absorber reactivity worth. Similarly, Case 1 and Case 3 are used to evaluated gadolinium absorber reactivity worth. Finally, Case 4 combines two different absorber materials.

4.3 Simulation

The criticality simulations were performed using the Serpent 2.1.32 neutronic code [41], with the ENDF/B-VIII.0 [59] and ENDF/B-VII.1 [60] nuclear data libraries. Two approaches are provided, the first where the same H_{cr} as in the experimental configuration is set to evaluate reactivity difference. The second approach determines the critical moderator level based on a calculation. This value is compared with the experimental H_{cr}. Uncertainty based on the nuclear data library was not included. The simulations were performed with 100 000 neutrons per generation, 2 600 active cycles, and 100 non-active cycles with statistical errors up to 6 pcm.

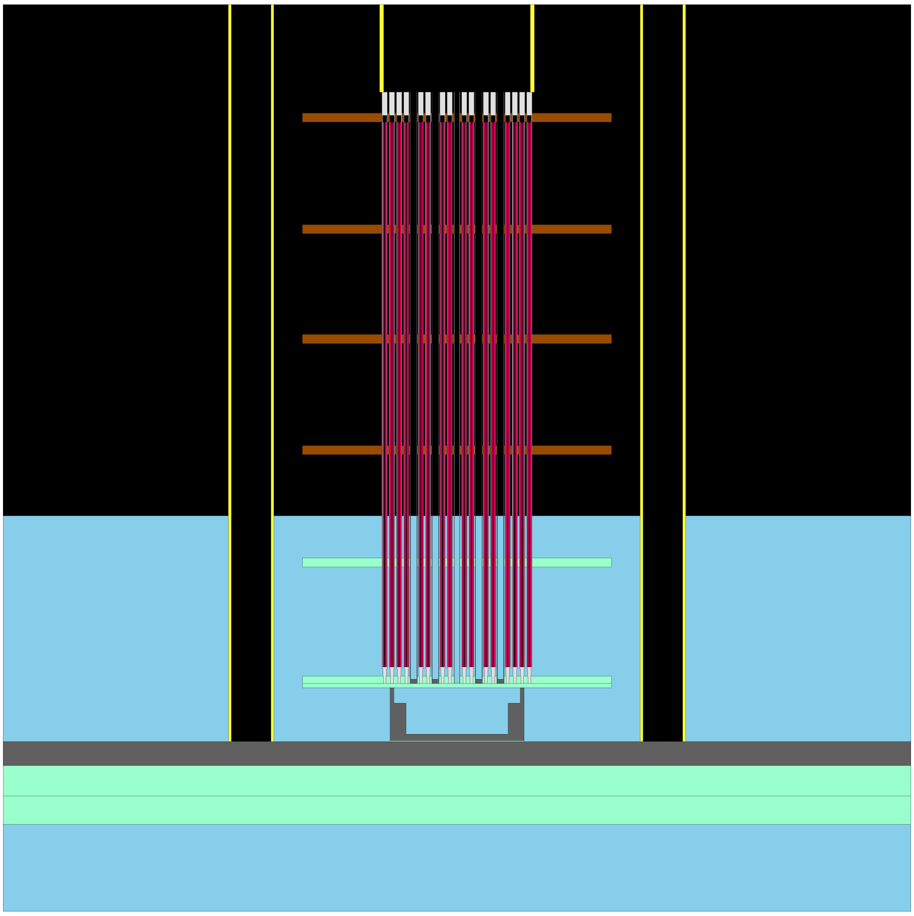

Fig. 4.9: Axial cut of the core model, Case 1 (H_{cr} = 34.63 cm).

4.4 Results

For each of the four measured cases, two measurements of the critical moderator level were made (for total of 8 measurements), due to the necessity of filling and draining the reactor vessel, moving positions with filled absorbers and the rate of filling the reactor vessel with the moderator.

The insertion of the absorbers into the guide tubes was carried out manually; see Fig. 4.10 and Fig. 4.11.

Fig. 4.10: Insertion of absorber tubes into the inner row of the fuel assembly.

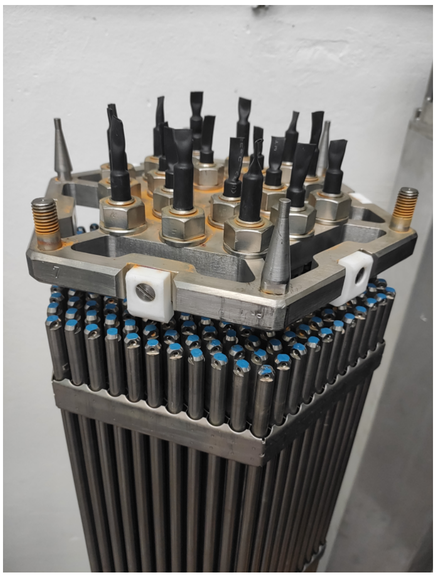

Fig. 4.11: Fuel assembly filled with absorber tubes, Case 4.

A summary of all four experimental data is given in Table 4.2 below. As expected, with an increasing number of inserted absorbers, the critical moderator level increases to compensate for the inserted negative reactivity of the absorbers.

Tab. 4.2: Data for experimentally determined critical state set by moderator level.

Case [-]	1	2	3	4
Number of Gd absorbers [pcs.]	0	0	6	6
Number of Sm absorbers [pcs.]	0	6	0	12
Moderator critical level - H_{cr} [cm] (k-eff = 1)	34.63	37.39	37.41	42.63
Absorber weight [pcm]	**0**	**1 481**	**1 497**	**3 866**

The critical parameter, multiplication factor (k-eff) with measured H_{cr}, was determined for each studied case by neutronic simulation. Comparisons between simulation and experimental data are in Tables 4.3 and 4.4. The newest ENDF/B-VIII.0 nuclear data library is, on average, closer to measurements than the older ENDF/B-VII.1 library.

Tab. 4.3: Evaluation of the calculated multiplication factor with measured moderator critical level for Case 1 and 2.

Nuclear	Case 1		Case 2	
data library	k-eff[-]	Δ[pcm]	k-eff[-]	Δ[pcm]
ENDF/B-VIII.0	0.99978	22	0.99994	6
ENDF/B-VII.1	1.00027	-27	1.00036	-36

Tab. 4.4: Evaluation of the calculated multiplication factor with measured moderator critical level for Case 3 and 4.

Nuclear	Case 3		Case 4	
data library	k-eff[-]	Δ[pcm]	k-eff[-]	Δ[pcm]
ENDF/B-VIII.0	1.00028	-28	1.00003	-3
ENDF/B-VII.1	1.00072	-72	1.00043	-43

For all four cases, a moderator critical level was calculated. Gadolinium absorber is a slightly more absorbent material than samarium, which requires a smaller moderator volume. However, this measurement conclusion was not confirmed by the calculations. Although the direct calculation results would indicate the opposite, the differences are smaller than calculation uncertainties.

4.5 Summary

The results presented in this paper describe experimental verification of the newly designed neutron absorbers as Gd_2O_3 and Sm_2O_3 in powder form designed in special steel tubes. There were four different core configurations in the well-known reference neutron field in reactor LR-0, operated by the Research Center Řež in the Czech Republic.

The different absorber distributions were designed to be verified sequentially, and the critical water level was determined. Case 1 was considered a reference without neutron absorbers with H_{cr} = 34.63 cm. In Case 2, six pcs. samarium absorbers were inserted into the inner row of guide tubes of the central fuel assembly, with H_{cr} = 37.39 cm and absorber weight 1 481 pcm. In Case 3, six pcs. gadolinium absorbers were inserted into the inner row of guide tubes of the central fuel assembly, with H_{cr} = 37.41 cm and absorber weight 1 497 pcm. Moreover, for Case 4 six pcs. gadolinium absorbers were inserted into the inner row of guide tubes of the central fuel assembly, and twelve pcs. samarium absorber inserted into the outer row with H_{cr} = 42.63 cm and absorber weight 3 866 pcm.

The results show good neutron absorption properties for all tested cases, even for Case 2 with samarium (1 481 pcm), as compared to the standardly used control B_4C absorber in LR-0, where the absorber weight is 1 014 pcm [57].

A very good agreement was achieved between experiment and calculations, and a reactivity difference of up to 28 pcm with the newest ENDF/B-VIII.0 nuclear data library was observed.

In conclusion, this experiment verified the potential of the new neutron absorbers from a neutronic point of view, with sufficient negative reactivity worth even for the absorber in powder form. Furthermore, from an economic point of view, this is a viable option compared to absorber materials in a metal form.

5

Conclusions

The presented research of neutron absorbers placed directly within a fuel assembly was shown and investigated. Several absorber materials and forms were analysed and optimised. Based on complex studies and experimental verification, it was proved that this concept allows:

- Improved safety – decrease in cask reactivity, possibility to remove the requirement to use burnup credit methodology, and criticality safety limits are met even with fresh fuel.
- Better economics – system/facility redesign is possible due to improved neutronic properties. For example: design with no stainless steel tubes between fuel assemblies in the final disposal casks, and significantly reduced cask wall inner diameter.

The usage of fixed neutron absorbers in nuclear fuel directly after reactor operation could reduce back-end stage costs in more facilities:

- storage – spent fuel pools, spent fuel casks
- transport – spent fuel casks
- disposal – final disposal casks

Apart from theoretical research, the work also contains a practical/experimental part, with the production of a prototype, and verification and proper validation in the research reactor. The introduced concept is unique, and on December 8, 2020, a patent (utility model) was obtained with number 34 636 (see attachment A.1), and more than six publications.

As declared at the beginning of the thesis, the presented solution must be sustainable for thousands of years, which is a big challenge for research. Nuclear research is always a long journey of simulation, verification and experimentation. This dissertation thesis can serve as a stepping stone for further research.

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Appendix A

List of Student's Publications and Activities

A.1 Patents and utility models

[A1] **Utility model - Neutron absorber**
Application number: 2020-38121
Registration number: 34636
Filing date: 16.10.2020
Title:
EN: Neutron absorber
CS: Neutronový absorbátor
Inventor:
Ing. Martin Lovecký, Ph.D., Kralovice, Czech Republic
Ing. Jiří Závorka, Líbeznice, Czech Republic
Ing. Jana Jiříčková, Ph.D., Hlohovice, Czech Republic
doc. Ing. Radek Škoda, Ph.D., Liberec, Liberec II-Nové Město, Czech Republic

[A2] **Patent - A coating of a zirconium cover of nuclear fuel**
Application number: 2016-269
Registration number: 307396
Filing date: 10.05.2016
Title:
EN: A coating of a zirconium cover of nuclear fuel
CS: Povlak zirkonového pokrytí jaderného paliva
Inventor:
Bc. Jiří Závorka, Líbeznice, Czech Republic
doc. Ing. Radek Škoda, Ph.D., Liberec, Liberec II-Nové Město, Czech Republic

A.2 Articles in impacted journals

[A3] ZÁVORKA, J., LOVECKÝ, M., JIŘIČKOVÁ, J., ŠKODA, R., *Experimental verification of new neutron absorbers concepts*, Progress in Nuclear Energy, 2022, (preprint).

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co-author

[A11] LOVECKÝ, M., ZÁVORKA, J., GINCELOVÁ, K., JIŘIČKOVÁ, J., ŠKODA, R., *Neutron Absorber for VVER-1000 Storage, Transport and Final Disposal Facilities*, Ljubljana: Nuclear Society of Slovenia, 2021, p. 1012.1-1012.8, ISBN: 978-961-6207-51-5.

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A.5 Awards

[A38] 1st place, Nuclear Days 2022 at the UWB in Pilsen, Student poster competition, 2022.

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[A40] 3rd place, Becquerel prize 2020, organizer: French Embassy in the Czech Republic, EDF, ATMEA, 2020.

[A41] 1st place, Winner in the ICONE27 Student Best Poster, The International Conference On Nuclear Engineering, Tsukuba, Japan, 2019.

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A.6 Other research publications

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