UNIVERSITY OF WEST BOHEMIA IN PILSEN

Faculty of Electrical Engineering
Department of Electrical Power Engineering

MASTER'S THESIS

Spent nuclear fuel from Small Modular Reactors

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2024

Declaration

I declare that I have prepared this master's thesis individually and that all cited and paraphrased sources are listed in the bibliography. I also declare that all software used is legally owned.

Pilsen, 23 May 2024

Bc. Bůžek Ondřej

ZÁPADOČESKÁ UNIVERZITA V PLZNI
Fakulta elektrotechnická
Akademický rok: 2023/2024

ZADÁNÍ DIPLOMOVÉ PRÁCE
(projektu, uměleckého díla, uměleckého výkonu)

Jméno a příjmení:	**Bc. Ondřej BŮŽEK**
Osobní číslo:	**E22N0018P**
Studijní program:	**N0713A060013 Výkonové systémy a elektroenergetika**
Specializace:	**Elektroenergetika**
Téma práce:	**Vyhořelé jaderné palivo z malých modulárních reaktorů**
Zadávající katedra:	**Katedra elektroenergetiky**

Zásady pro vypracování

Malé modulární reaktory mohou mít vysoký měrný výkon a z pohledu zadní části palivového cyklu (transport a skladování vyhořelého jaderného paliva, ukládání v hlubinném úložišti) mohou na jednotku vyrobené energie představovat vyšší zátěž než klasické jaderné reaktory s velkým výkonem. Cílem práce je srovnat parametry vyhořelého jaderného paliva malých jaderných reaktorů s parametry bloku VVER--1000.

1. Proveďte rešerši na téma malé modulární reaktory (SMR). Zaměřte se pouze na SMR s potenciálem výstavby v Česku.
2. Seznamte se s metodikou výpočtů parametrů vyhořelého paliva v kódu SCALE/TRITON.
3. Proveďte výpočet parametrů vyhořelého jaderného paliva pro vybrané SMR na základě charakteristik paliva z databáze Advanced Reactors Information System (ARIS) IAEA a vyhořelého jaderného paliva VVER-1000.
4. Srovnejte zbytkový výkon a radiotocixitu SMR a VVER-1000 na jednotku vyrobené energie.

Rozsah diplomové práce: **40-60**
Rozsah grafických prací:
Forma zpracování diplomové práce: **elektronická**

Seznam doporučené literatury:

1. B. T. Rearden, M.A. Jessee: SCALE Code System, ORNL/TM-2005/39, Version 6.2.4, Oak Ridge National Laboratory, Oak Ridge, Tennessee (2018).
2. L. M. Krall, A. M. Macfarlane, R. C. Ewing: Nuclear waste from small modular reactors. Proceedings of the National Academy of Sciences 119 (23) e2111833119, 2022.
3. T. K. Kim, L. Boing, W. Halsey, B. Dixon: Nuclear Waste Attributes of SMRs Scheduled for Near-Term Deployment. ANL/NSE-22/98, Revision 1, 2022.

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Termín odevzdání diplomové práce: **24. května 2024**

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Abstract

This master's thesis focuses on the back-end of the fuel cycle of small modular reactors, which are expected to play a significant role in the transition to emission-free energy sources according to the current energy concept. The main aim of this work is to assess the possibilities and challenges associated with the storage of spent nuclear fuel from these reactors, considering the potential for higher levels of fuel burnup. The thesis includes a review section examining the back end of the fuel cycle and potential small modular reactors suitable for the needs of the Czech Republic. The crucial part of the thesis involves calculating the parameters of spent nuclear fuel from small modular reactors using the SCALE/TRITON code and comparing it with fuel from VVER-1000 reactors.

Keywords

Decay heat, nuclear spent fuel, radiotoxicity, small modular reactors, VVER-1000.

Abstrakt

Tato diplomová práce se zaměřuje na zadní část palivového cyklu malých modulárních reaktorů, které podle současné energetické koncepce budou hrát významnou roli v přechodu na bezemisní zdroje energie. Hlavním cílem této práce je posoudit možnosti a výzvy spojené s ukládáním vyhořelého jaderného paliva z těchto reaktorů, s ohledem na možný vyšší stupeň vyhořívání paliva. Práce obsahuje rešeršní část zabývající se zadní částí palivového cyklu a potenciálními malými modulárními reaktory vhodnými pro potřeby České republiky. Klíčovou částí práce je výpočet parametrů vyhořelého jaderného paliva z malých modulárních reaktorů v kódu SCALE/TRITON a jeho porovnání s palivem z VVER-1000.

Klíčová slova

Zbytkový výkon, vyhořelé jaderné paliva, radiotoxicita, malé modulární reaktory, VVER-1000.

Acknowledgments

I would like to thank my supervisor Ing. Martin Lovecký, Ph.D. for his excellent guidance, valuable advice, and patience while writing the thesis. I would also like to thank Prof. Ing. Radek Škoda, Ph.D. and Ing. Jiří Závorka, Ph.D. for the unique opportunities they provided me throughout my studies.

Contents

List of acronyms

AFR	Away-from-reactor	
AN	Actinide	
AR	At-reactor	
B	Burnup	(GWd/t_U)
B_4C	Boron carbide	
BWR	Boiling water reactor	
CANDU	Canada deuterium uranium	
CEUS	Coastal and estuarine unifying study	
CZE	Czech Republic	
FA	Fuel assembly	
FBR	Fast breeder reactor	
FP	Fission product	
GCR	Gas-cooled reactor	
Gd_2O_3	Gadolinium oxide	
HEU	Highly-enriched uranium	
HLW	High level waste	
HTGR	High-temperature gas-cooled reactor	
IAEA	International Atomic Energy Agency	
IPHWR	Indian pressurized heavy water reactor	
iPWR	Integral pressurized water reactor	
LEU	Low-enriched uranium	
LWGR	Light water graphite-moderated reactor	
LWR	Light water reactor	
MA	Minor actinide	
MOX	Mixed oxide fuel	
MPC	Multi-Purpose Canisters	
M_{SNFe}	SNF fuel mass	(t/GW_e-yr)
M_{SNFth}	SNF fuel mass	(t/GW_{th}-yr)
NPP	Nuclear power plant	
NRC	Nuclear Regulatory Commission	

LIST OF ACRONYMS

NSSS	Nuclear steam supply system	
PHWR	Pressurized heavy water reactor	
PSAR	Preliminary safety analysis report	
PWR	Pressurized water reactor	
RPV	Reactor pressure vessel	
SMR	Small modular reactor	
SNF	Spent nuclear fuel	
SWU	Separation work unit	
UO_2	Uranium dioxide	
U_3O_8	Triuranium octoxide	
US	United States	
WG	Weapon-grade	
wt%	Mass fraction	(%)
WWER, VVER	Water-water energy reactor	
η	Thermal efficiency	(%)

Introduction

In the current global energy context, nuclear energy represents one of the promising options for replacing outdated fossil fuel-burning power plants. The most debated topic today are small modular reactors (SMRs), which should fulfill these ambitions. With the increase in world population and the growing need for energy devices, humanity faces the difficulty of ensuring the energy supply's sustainability, security, and cost-effectiveness. In this context, SMRs could offer an innovative solution that could spark an energy revolution across the globe.

One of the main reasons why SMRs might gain importance is their theoretical modularity and flexibility. Unlike traditional full-scale nuclear power plants (NPPs), SMRs might be built in locations unsuitable for large nuclear units. Due to their smaller size and modular design, SMRs could better meet a specific location's conditions and requirements. Another significant factor is that SMRs do not have to pay emission allowances, reducing the energy cost.

However, although much attention is paid to SMR design and construction, the question of spent nuclear fuel (SNF) management needs to be addressed. It is an inherent obligation of any nuclear operator. While most debates focus on the technical and economic aspects of SMR deployment, more attention should be paid to the back-end of the fuel cycle of these reactors.

SMRs could find their place in the Czech energy mix, which is suggested by CEZ Group's interest in building and operating such a reactor. CEZ Group is currently considering seven designs from all over the world. Additionally, CEZ Group already operates two NPPs, water-water power reactor (VVER)-1000 and VVER-440, which could provide valuable experience incorporating SMR technologies into the country's energy infrastructure.

This thesis focuses on the potential use of SMRs in the Czech Republic (CZE), considering the CEZ Group strategy. It analyses the SNF parameters of potential SMRs and compares them with traditional VVER-1000-type nuclear units. Specifically, it compares residual decay heat and radiotoxicity of the SNF over a time horizon of one million years. This work evaluates whether the SNF from SMRs represents a greater burden on the back-end of the fuel cycle compared to the VVER-1000. The benefits and risks of implementing SMRs in CZE are crucial topics that this thesis aims to shed light on and thus contribute to an informed debate on the future of nuclear power in the country.

1 Nuclear fuel cycle

The nuclear fuel cycle is a set of complicated and challenging processes that is a critical element of the use of nuclear energy. It describes the total fuel cycle from when the uranium ore is extracted to the final processing of the SNF. The uranium fuel cycle is the most widespread in the world, and there are potential SMRs for the CZE; therefore, in my master's thesis, I focus only on it. The fuel cycle is divided into [1]:

- front-end of fuel cycle,
- reactor operation,
- back-end of fuel cycle.

The back-end of the fuel cycle can be divided from the time the fuel assembly (FA) is removed from the reactor into [1]:

- closed fuel cycle,
- open fuel cycle,
- wait and see.

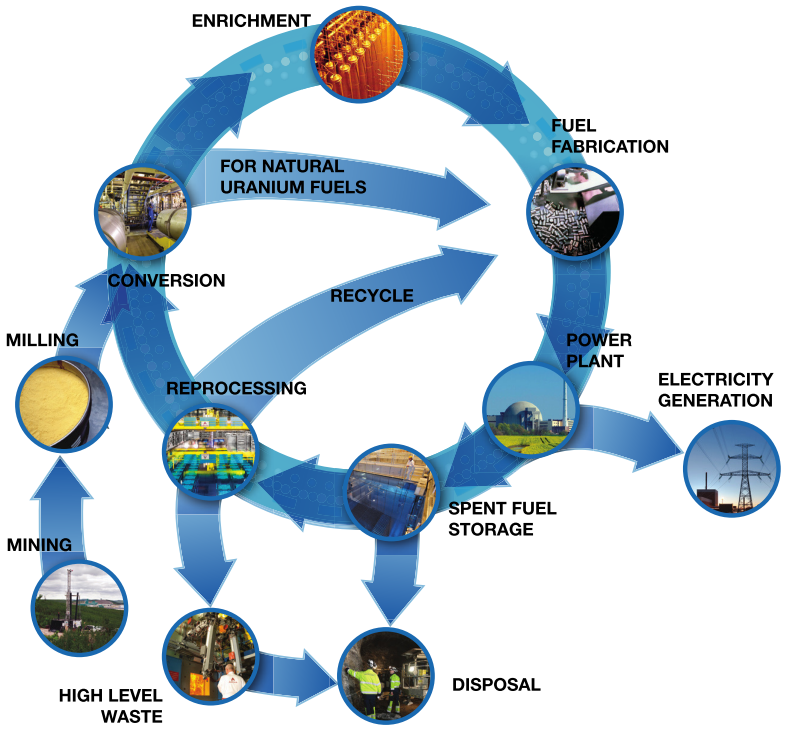

Fig. 1.1 Nuclear fuel cycle [2]

1.1 Front-end of fuel cycle

The front-end of the fuel cycle describes all the processes required to produce a FA, which is one of the necessary elements for transforming nuclear energy.

1.1.1 Mining

Uranium is a common ore. It is found worldwide in most rocks, soils, rivers, and seawater. Uranium is about 500 times more abundant than gold and about the same as tin. There are three methods of uranium mining: open pit mines, underground mines, and in situ leaching, where the uranium is leached directly from the ore. The largest producers of uranium are Kazakhstan, Canada, Namibia, Australia, and Uzbekistan, as shown in Table 1.1. [2, 3]

Uranium mining began in the 19th century in the CZE. At that time, uranium was employed to produce uranium paint, which was utilized for coloring glass and porcelain. In the second half of the 20th century, the CZE became a world leader in uranium mining. The Czech mines were of strategic importance to the then-Soviet Union. Currently, uranium is mined only in Dolní Rožínka, and uranium is extracted as a by-product of mining by the chemical route in Stráží pod Ralskem. [1, 4]

Tab. 1.1 World uranium mining production [3]

	Production (t_U)								
Country	2014	2015	2016	2017	2018	2019	2020	2021	2022
Kazakhstan	23,127	23,607	24,689	23,321	21,705	22,808	19,477	21,819	21,227
Canada	9,124	13,325	14,039	13,116	7,001	6,938	3,885	4,693	7,351
Namibia	3,255	2,993	3,654	4,224	5,525	5,476	5,413	5,753	5,613
Australia	5,001	5,654	6,315	5,882	6,517	6,613	6,203	4,192	4,553
Uzbekistan	2,400	2,385	3,325	3,400	3,450	3,500	3,500	3,520	3,300
Russia	2,990	3,055	3,004	2,917	2,904	2,911	2,846	2,635	2,508
Niger	4,057	4,116	3,479	3,449	2,911	2,983	2,991	2,248	2,020
China	1,500	1,616	1,616	1,692	1,885	1,885	1,885	1,600	1,700
India	285	385	385	421	423	308	400	600	600
South Africa	573	393	490	308	346	346	250	192	200
Ukraine	926	1,200	808	707	790	800	744	455	100
USA	1,919	1,256	1,125	940	582	58	6	8	75
Pakistan	45	45	45	45	45	45	45	45	45
Brazil	55	40	44	0	0	0	15	29	43
Iran	0	38	0	40	71	71	71	21	20
Czech Republic	193	155	138	0	0	0	0	0	0
Romania	77	77	50	0	0	0	0	0	0
France	3	2	0	0	0	0	0	0	0
Germany	33	0	0	0	0	0	0	0	0
Malawi	369	0	0	0	0	0	0	0	0
Total world	56,041	60,304	63,207	60,514	54,154	54,742	47,731	47,808	49,355
Tonnes U_3O_8	66,087	71,113	74,357	71,361	63,861	64,554	56,287	56,377	58,201
World demand (%)	85	98	96	93	80	81	74	76	74

1.1.2 Milling

The next stage of production is milling. It is usually carried out close to the uranium mine. The mined uranium ore is crushed and chemically treated to separate the uranium. The resulting product is the so-called yellow cake, a yellow powder with a uranium concentration of more than 80 wt%. After milling, the yellow cake is transported to the conversion facility. [2]

Fig. 1.2 Yellow cake [5]

1.1.3 Conversion

Natural uranium consists of three isotopes, uranium U-238 (99.28 wt%), U-235 (0.7 wt%) and U-234 (0.05 wt%). NPPs utilize uranium enrichment within the range of 3 wt% to 5 wt%. To increase the ratio of U-235 to U-238, uranium must be enriched. Firstly, the yellow cake must be converted to uranium hexafluoride gas (UF_6) at a conversion facility to be subsequently enriched. The UF_6 is filled into large cylinders where it solidifies. The cylinders are stored in metal containers for subsequent transport to the enrichment plant. [2]

1.1.4 Enrichemnt

The enrichment process is possibly the most complicated in the entire fuel cycle. All known enrichment methods are based on different molecular weights. In terms of molecular weight, U-238 has heavier molecules than U-235, and therefore, it is possible to separate the two isotopes. Enrichment methods are [1]:

- diffusion,
- centrifugal method,
- electromagnetic separation,

– laser enrichment.

Nuclear material can be divided on the enrichment of U-235 into [6]:

– <20 wt% U-235 - low-enriched uranium (LEU),

– >20 wt% U-235 - highly-enriched uranium (HEU),

– >90 wt% U-235 - weapon-grade (WG).

Most nuclear reactors are light water reactors (LWRs) of the pressurized water reactors (PWRs) or boiling water reactors (BWRs) type; these reactors require enrichment of 0.7 to 5 wt%. The nuclear fuel for these reactors falls into the LEU category. There is undoubtedly an interest in enrichment of up to 10 wt% for power reactors and even up to 20 wt% for some special fuels. Pressurized heavy water reactors (PHWRs) use naturally enriched uranium as fuel. Currently, centrifugal technology is the most widely used enrichment method, which provides a lower cost per separation work unit (SWU), allowing operations to be sustained even when demand declines. SWU is a unit that indicates the energy input concerning the amount of uranium processed, the degree of enrichment, and the rate of depletion of the remainder. SWU measures the amount of separation work done to enrich a certain amount of uranium.

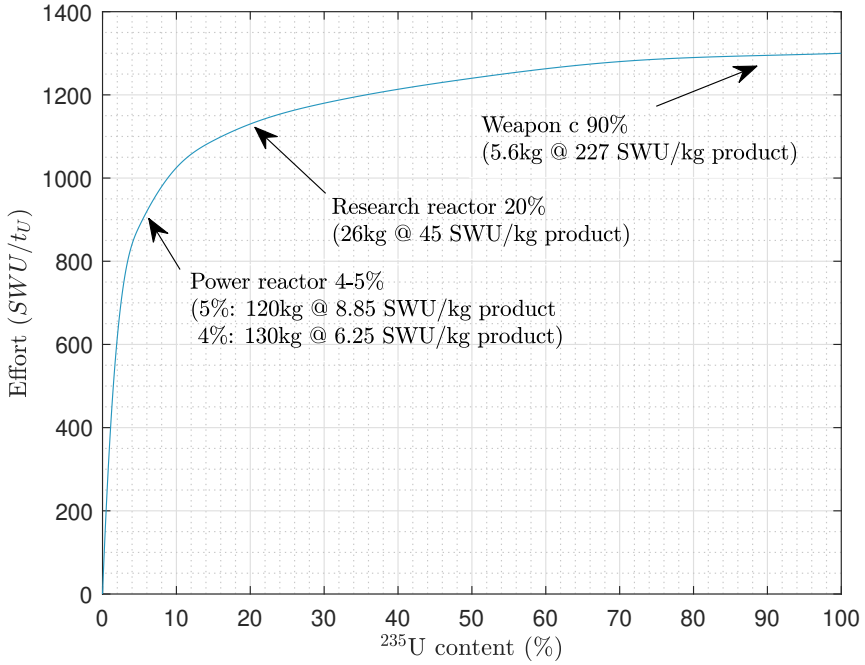

Fig. 1.3 Uranium enrichment and uses [6]

Figure 1.3 shows the transformation of one tonne of natural uranium into 120-130 kg of uranium for power reactors. The exact amount of uranium would be enough to produce 26 kg of fuel for research reactors or 5.6 kg of material for WG. Figure 1.3 also shows that due to the non-linear nature of the curve, less effort is required

to enrich uranium from 20 wt% to 90 wt% than to enrich natural uranium to 20 wt% enrichment. It explains why enrichment facilities are considered sensitive technology concerning weapons proliferation prevention and are under rigorous supervision under international agreements.

As indicated in Table 1.2, the Russians had the largest enrichment capacity in 2022, which is unsurprising because they have always been among the world leaders in nuclear engineering. The table also shows that the China National Nuclear Corporation (CNNC) plans to increase its enrichment capacity, which is evident from China's energy strategy because China is constantly building new NNPs. [6]

Tab. 1.2 World enrichment capacity – operational in 2020 and planned [6]

	Capacity (kSWU/yr)		
Operator	2020	2025	2030
CNNC	6,300	11,000	17,000
Orano	7,500	7,500	7,500
Rosatom	27,700	26,200	24,800
Urenco	18,600	17,300	16,300
Other	66	375	525
Total	60,166	62,375	66,125

1.1.5 Fuel fabrication

Solidified enriched UF_6 is converted into uranium dioxide (UO_2) pellets by pressing and sintering at temperatures over 1,400 °C to achieve high density and stability. The cylindrical UO_2 pellets are 8-15 mm in diameter and 10-15 mm long. These pellets are then placed in long metal tubes called fuel rods (pins), which must resist the temperatures and pressures in the nuclear reactor. The fuel rods are then assembled into FAs that serve as nuclear fuel. The same process is used to fabricate mixed oxide fuel (MOX), which consists of uranium and plutonium. [2]

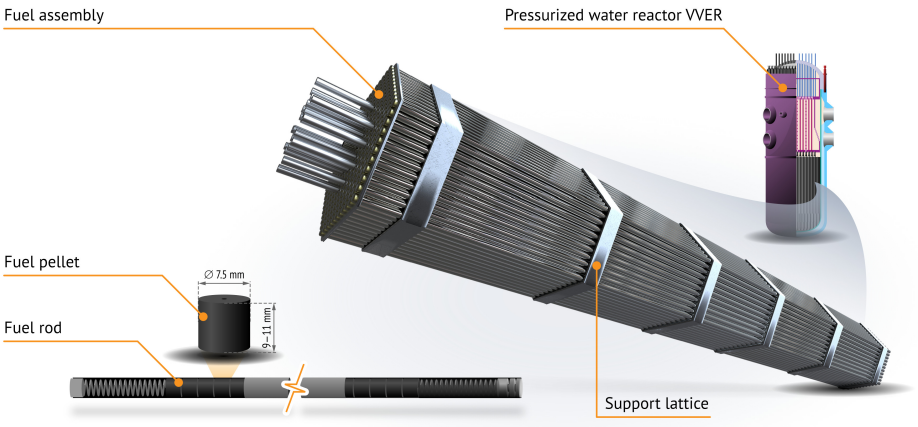

Fig. 1.4 VVER FA [7]

1.2 Reactor operation

The assembled FAs are loaded into the reactor core, where nuclear energy is transformed into thermal energy. The FAs are kept in the reactors for 3-6 years, a period strongly dependent on the core loading strategy, which is mainly influenced by legislation, the type of facility, physical limits, and economics. In the case of the Temelín[1] NNP, approximately one-third of the FAs are replaced with new ones every 18 months during shutdown. The rest of the FAs are rearranged to ensure a balanced loading of the reactor to allow for the long-term operation of the NPP. [8]

From Figure 1.5, it can be inferred that the most populated reactor types in the world are LWRs. By the end of 2022, 301 PWRs were in operation with a total capacity of 298.1 GW_e, which makes it the most numerous reactor type. The second most common reactor type is PHWRs, with a total of 46 facilities, which are subdivided into Canada deuterium uranium (CANDU) reactors and Indian pressurized heavy water reactors (IPHWRs). The third most numerous type is BWRs, which have 42 facilities. [9]

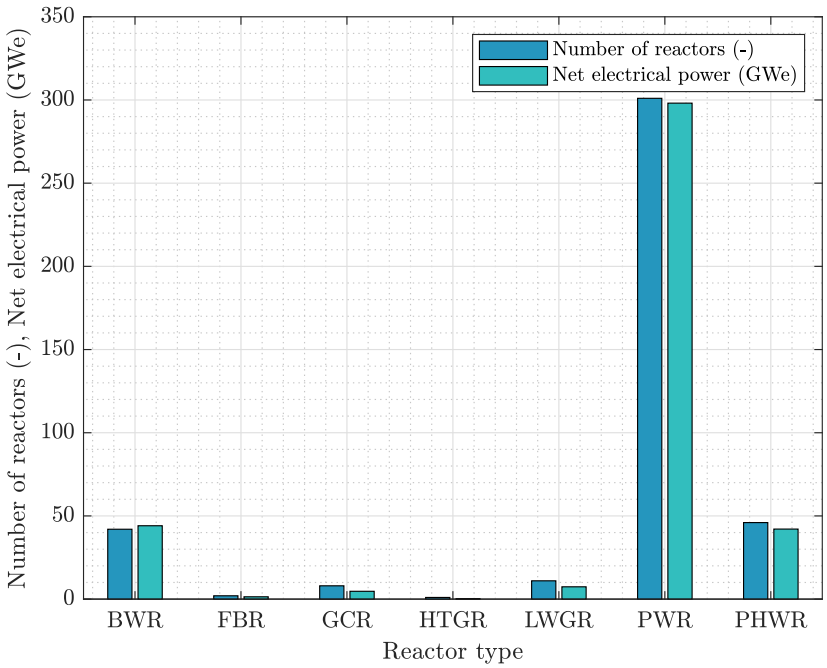

Fig. 1.5 Total operational reactors categorized by type and net electrical power (as of 31 Dec. 2022) [9]

One possible reason why PWRs are the most numerous may be the size of the reactor pressure vessel (RPV), as this significantly affects the size of the containment, which has a significant impact on building permitting and planning, and also reduces the cost of construction. As can be seen in Figure 1.6, PWRs achieve the highest power density compared to other types; this is achieved due to the higher operating pressure in the RPV, which

[1] VVER-1000/V-320 - Russian version of PWR with installed capacity of 1,000 MW_e

can be, for example, 15.7 MPa in the case of Temelín NPP. CANDU reactors operate at an RPV pressure of approximately 11 MPa and BWRs at approximately 7.5 MPa.

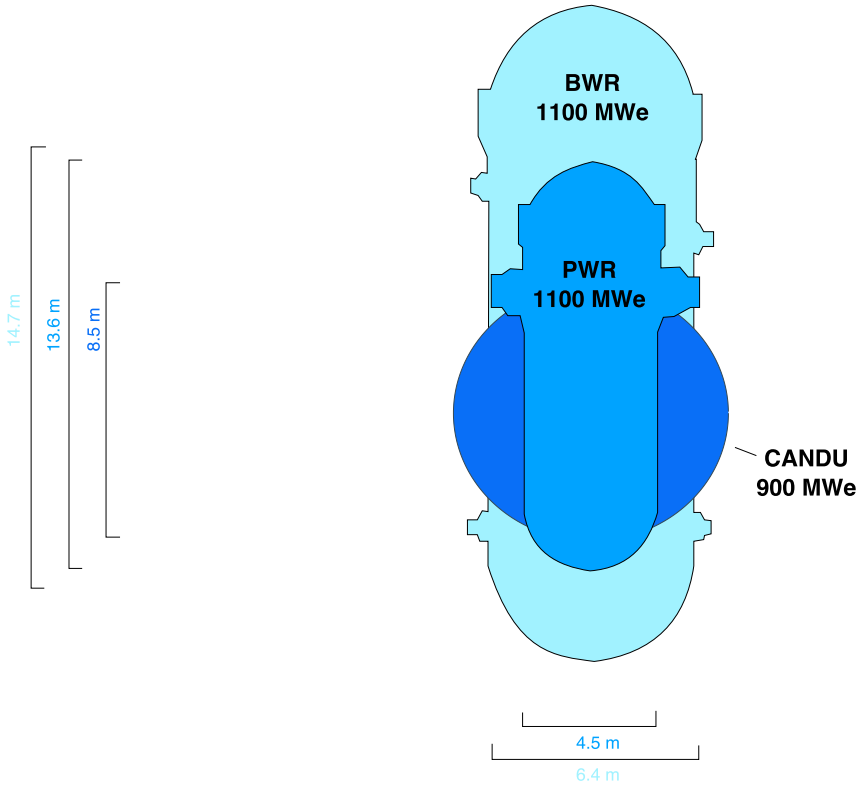

Fig. 1.6 Comparison of RPV sizes [10]

Other reasons may be, for example, that PWRs and CANDU reactors have two systems compared to BWRs, which prevents turbine contamination. Another advantage of PWRs and CANDUs is that the control drive rods, located at the top of the RPV, can be inserted into the reactor by gravity during reactor shutdown, thus increasing the proportion of passive safety systems. Additionally, these reactors use boric acid, which allows a homogeneous power distribution in the reactor. LWRs utilize light water as a coolant and moderator, offering a cost advantage over the heavy water used in PHWRs. Moreover, light water's physical and chemical properties have been thoroughly researched.

1.3 Back-end of fuel cycle

After several years, the SNF must be removed from the reactor because it contains suboptimal concentrations of fission fragments and heavy elements, resulting in its low reactivity value. SNF usually contains 1 wt% U-235, 0.6 wt% fissile plutonium (almost 1 wt% Pu in total), 95 wt% U-238, and almost 3 wt% fission products (FPs) and minor actinides (MAs). Once the fuel is removed from the reactor, it emits radiation and heat, placing it in an at-reactor (AR) storage pool. The purpose of this storage is to reduce the fuel's activity level. The water in the pool serves to shield the radiation and removes the decay heat through natural or forced circulation

through external heat exchangers. SNF is stored AR pools for several months and sometimes for several years. The following process depends on the legislation and strategy of each country. After removal from the storage pools, SNF is transported to medium-term storage, which can be dry or wet. [11]

Fig. 1.7 Back-end of fuel cycle [2]

1.3.1 Open fuel cycle

In this cycle, the fuel passes through the reactor only once. After removing the SNF from the medium-term storage, the fuel is transported to the final storage, called long-term storage. Most countries practice this method of dealing with SNF using nuclear energy. Only a few countries, such as Finland, Sweden, or Canada, have long-term storage or plan to do so. [12]

1.3.2 Closed fuel cycle

The closed fuel cycle is based on a reprocessing process. The uranium produced by reprocessing usually contains a slightly higher concentration of U-235 than natural uranium. The resulting uranium can be reused as fuel after conversion and enrichment. Plutonium can be made into MOX fuel, which consists of uranium and plutonium oxides. Plutonium replaces U-235 in reactors using MOX fuel. According to Areva, approximately eight reprocessed FAs can produce one MOX FA, two-thirds of the FA from enriched uranium, approximately 3 t of depleted uranium, and approximately 150 kg of waste. It can save the cost of purchasing 12 t of natural uranium. Another option to close the fuel cycle are fast neutron reactors, which are still the music of the future.

Countries using this cycle include China, Germany, India, Japan, the Netherlands, and the United Kingdom. [11]

1.3.3 Wait and see

The wait and see method is based on the fact that SNF is stored in medium or long-term storage. The fuel will be there until a decision is made on how to deal with the SNF. Technically, the CZE is currently following an open fuel cycle strategy with a wait and see approach. [13]

2 Spent nuclear fuel

SNF consists of FPs and ANs, which are radioactive and, therefore, require special handling. Certain countries may classify SNF as radioactive waste, which falls under the International Atomic Energy Agency (IAEA) classification of high-level waste (HLW). In the case of the CZE, nuclear fuel is considered radioactive waste only when the owner (i.e. CEZ GROUP) or the State Office for Nuclear Safety declares it waste. HLW contains such high concentrations of short- and long-lived radionuclides, requiring more containment, heat transfer, and isolation from the accessible environment than intermediate-level waste. These conditions are usually met in deep geological repositories with engineered barriers. Activity is a fundamental metric within SNF analysis that provides an overall view of waste hazard. It serves as a benchmark for determining specific waste attributes essential for safety, such as dose and radiotoxicity, and attributes important for handling and disposal, such as decay heat. The activity value ranges from 10^4 to 10^6 TBq/m^3 for HLW. [14, 15]

2.1 Fission products

The most critical FPs produced during the operation of thermal-neutron power reactors are listed in Table 2.1. The most significant FPs in terms of SNF disposal are Sr-90 and Cs-137.

The isotope Sr-90 undergoes β^- decay to Y-90 with a decay energy of 0.546 MeV and a half-life of approximately 28.8 years. It originates from nuclear reactors and explosions. There are studies that, because of its similarity to Ca-20, it accumulates in bones and teeth, increasing the risk of bone cancer and leukemia with exposure. Its presence in the environment has major public health implications. [16, 17]

Cs-137 is a radioactive isotope of cesium with a half-life of 30 years. Like Sr-90, Cs-137 are produced in nuclear reactors during the fission of U-235 and other fissile isotopes and the explosion of a nuclear bomb. Cs-137 is a source of β^- radiation with an energy of 0.514 MeV and γ radiation with an energy of 0.662 MeV. Cs-137 can enter the human body through the food chain because it is deposited in the top layers of the soil where plant and mushroom roots draw water. Exposure to this isotope increases the risk of cancer. [18, 19, 20]

Tab. 2.1 FPs produced during operation of PWRs [21]

Nuclide	Half-life (yr)
Se-79	6.5x10^4
Kr-85	10.7
Sr-90	28.8
Zr-90	1.5x10^6
Tc-90	2.1x10^5
Pd-107	10.5x10^6
Sn-126	10^5
I-129	1.6x10^7
Cs-135	3x10^6
Cs-137	30
Sm-151	90

Figure 2.1 shows the progression of FPs decay in SNF from VVER-1000. The figure shows that initially, the isotopes Sr-90 and Cs-137 contribute most to the decay heat and radiotoxicity. After about 700 years, the decay heat of these isotopes decreases by almost three orders of magnitude. After that, most of the residual heat is supplied by the long-lived FPs Tc-90 and Sn-126.

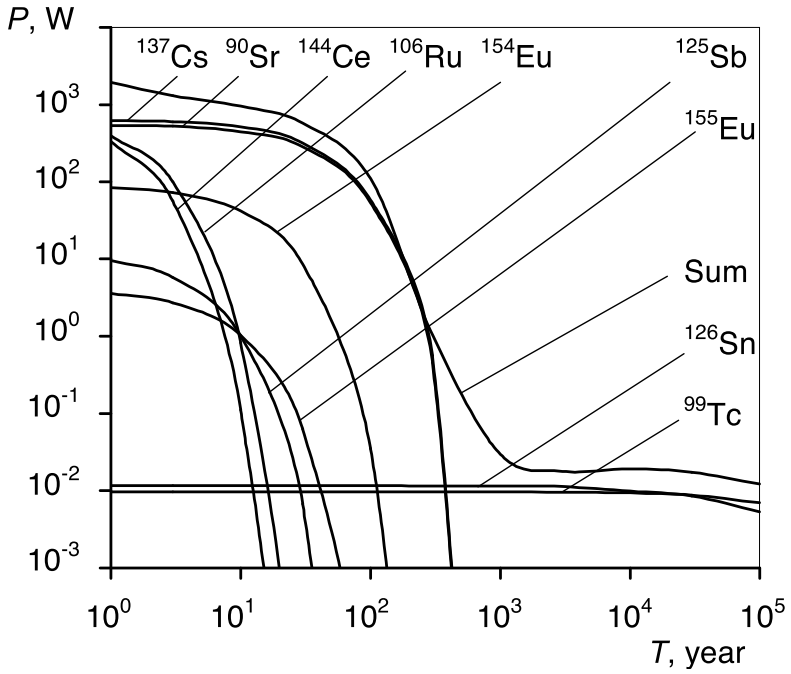

Fig. 2.1 Decay heat of FPs from 1 ton of SNF of VVER-1000 reactor for fuel burnup 40 MWd/kg [22]

2.2 Actinides

ANs are the group of isotopes that present the most significant challenge in the long-term storage of SNF. It is mainly due to their radioactivity to which isotopes such as Pu-238, Pu-239, Pu-240, Am-241 and Cm-244 contribute most.

Pu-238 is one of the first isotopes of plutonium ever discovered. Pu-238 is a strong α particle source with a half-life of 88 years. Since plutonium emits α radiation, indoor exposure poses a health risk. It can remain in the body for decades, exposing organs and other tissues to radiation and increasing the risk of cancer. The same characteristics also apply to Pu-239 and Pu-240. [23]

Am-241 is one of the MAs. It is the source of α particles with an energy of 5.486 MeV and γ radiation with an energy of 0.06 MeV. When Am-241 is consumed, it accumulates in the bones and liver for decades. It accumulates permanently in the testes and ovaries. All these organs promote the development of cancer cells. [24]

Cm-244 is one of the MAs with a very short half-life of approximately 18 years compared to other MAs. Like previous ANs, it is a source of α particles and an important source of neutrons, contributing to the increased

cancer risk from exposure to α particles. [25]

Tab. 2.2 ANs produced during operation of PWR [21]

Nuclide	Half-life (yr)
Np-237	2.1×10^6
Pu-238	88
Pu-239	2.4×10^4
Pu-240	6.6×10^3
Pu-241	14.4
Pu-242	3.8×10^5
Am-241	423
Am-242m	141
Am-243	7.4×10^3
Cm-243	28.5
Cm-244	18.1

From Figure 2.2, the enormous contribution during the first ten years to the decay heat is mainly Cm-244 due to its very short half-life. The isotope Pu-238 also significantly contributes to the heat output in the storage period, which exceeds 100 years after removal from the reactor. In the 30-1300 years, the primary isotope is Am-241. In the period after 10,000, the primary sources of decay heat are the plutonium isotopes 239 and 240.

Comparing Figure 2.1 and Figure 2.2 with each other, it is clear that for the first 150 years, the dominant sources of decay heat are primarily FPs, which are up to an order of magnitude higher than ANs. After 150 years, on the other hand, the dominant sources of decay heat are ANs, which continue until the end of the lifetime of the SNF.

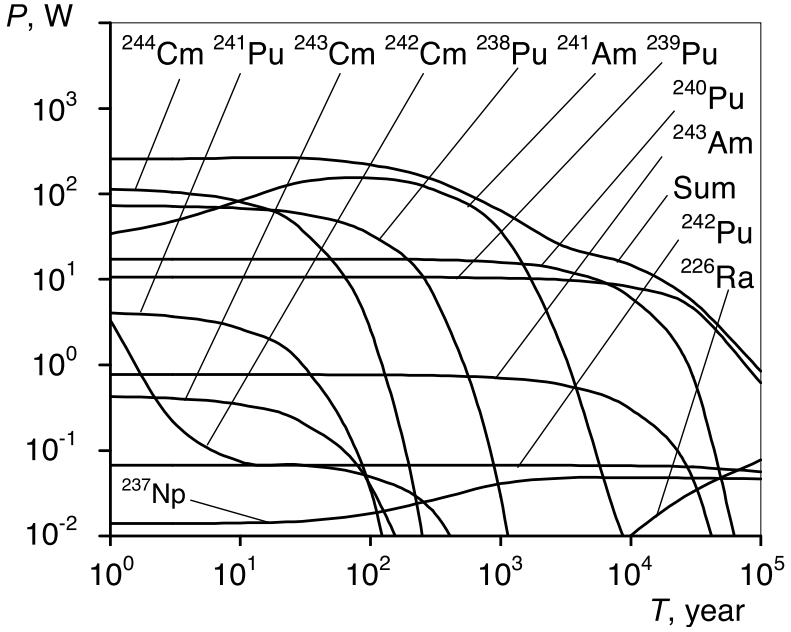

Fig. 2.2 Decay heat of ANs from 1 ton of SNF of VVER-1000 reactor for fuel burnup 40 MWd/kg [22]

2.3 Medium and long-term storage

The storage facility should be designed to fulfill the main safety functions such as maintaining subcritical, heat removal, isolation of radioactive materials, shielding from radiation, and the ability to remove fuel or spent fuel. Storage time can be divided into medium-term and long-term storage facilities. Medium-term storage methods include wet and dry storage. Long-term storage means final deep geological repository. [26]

2.4 Medium-term wet storage

The wet storage method is based on more than 50 years of experience. Wet storage can be divided into:

- AR storage pools,
- Away-from-reactor (AFR) storage pools.

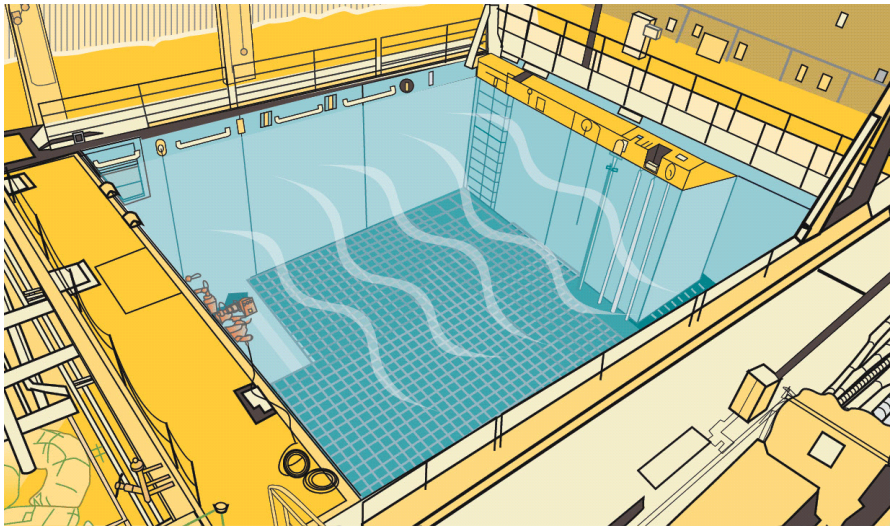

Fig. 2.3 Storage pool [27]

AR facilities are essentially storage pools described in Chapter 1.3. SNF is transported in AFR facilities in transport casks. Storage pools are usually reinforced concrete and may be lined with stainless steel. Some of the first pools were constructed without a roof, which led to a deterioration in water purity, so all pools are now covered. The activity level in the water is kept as low as reasonably achievable. It can be achieved by in-pool ion exchange systems, external ion exchange systems, or by limiting activity release into the pool water. Currently, racks or baskets made of materials such as boron stainless steel, boral[1], or boraflex[2] are used. It is to achieve a higher storage density and subcritical of the SNF. [28, 29, 30]

[1] Boron carbide aluminum alloy composites
[2] Silicone polymer and boron carbide powder

2.5 Medium-term dry storage

The dry storage system is based on more than 30 years of operating experience. Dry casks include specially designed casks that must withstand degradation mechanisms such as exposure to heat and radiation. The storage facilities are designed to maintain their safety function in a wide range of accident scenarios and SNF handling activities.

The price of a dry storage system depends on the requirements placed on it. The requirements result from the specific application of the system. Therefore, the dry storage systems are divided to make the system more economical. In terms of application, dry storage systems are divided into [28, 31]:

- single-purpose storage systems,
- single-purpose transport systems,
- single-purpose disposal packages,
- dual-purpose systems,
- multi-purpose systems.

2.5.1 Single-purpose storage systems

Single-purpose dry storage systems are used for AFR applications. The systems are based on cask-based designs, canisters in vaults and overpacks. In the past, metal casks were widely used, whereas nowadays canisters with vaults or cask-like overpacks are more commonly used. Unlike metal casks, both of these solutions are made of reinforced concrete. The advantage of concrete is its low cost and the ability to fabricate the product on site. According to a rough cost estimate, canister systems account for 20-25 % of the cost of metal casks. At the time of decommissioning, both variants may require decontamination and disposal of low-level waste. [31]

2.5.2 Single-purpose transport systems

These systems transport fuel from the reactor to storage or reprocessing facilities. The cost of the equipment for storage and transportation is comparable, although the SNF to be stored is much older than the fuel to be reprocessed. Transport casks can be more expensive than metal storage casks (anywhere from 50 % to 300 %) because they have to meet more strict transport requirements. Transport casks are needed significantly less than storage casks because transport casks are typically designed to last at least 25 years. During this lifetime, 10-25 shipments may be made per year. [31]

2.5.3 Single-purpose disposal packages

There is little experience with this method of dry storage at present. Disposal packages are expected to be constructed of concentric layers to prevent the spread of radioactive materials. Part of the layers will provide

insulation, and the remaining parts will be sacrificial. The purpose of the sacrificial layers is to delay radioactivity penetration into the near-surface environment. An example of a single-purpose disposal package is illustrated in Figure 2.5. The outer layer of the canister or container will be copper, and the interior will be a cast iron insert. Once the canister/container is placed properly, it will be encased in bentonite clay. Because of the copper, this dry storage method will be costly. [12, 31]

2.5.4 Dual-purpose systems

Dual-purpose systems are intended for transport and storage. The oldest dual-purpose technologies were based on casks. Nowadays, canister-based technologies are more commonly used. Dual-purpose systems allow to avoid some of the activities associated with decontamination, low-level waste disposal and associated costs that occur with single-purpose technology.

Cask-based dual-purpose systems are metal casks constructed of ductile cast iron or forged steel. The same unit provides monitored dry storage AFR and eventual transport to an AFR facility or repository. The cost of a metal drum for dual-purpose will be approximately the same as a metal drum for transportation only.

Canister-based dual-purpose transfer systems are usually at the reactor site and are stored in low-cost concrete vaults or overpacks. During transport, the fuel is transferred to a transport overpack, similar to a single-purpose transport cask, except that the canister provides a basket, estimated to cost 5-15 % of the cost of a single-purpose transport cask. [31]

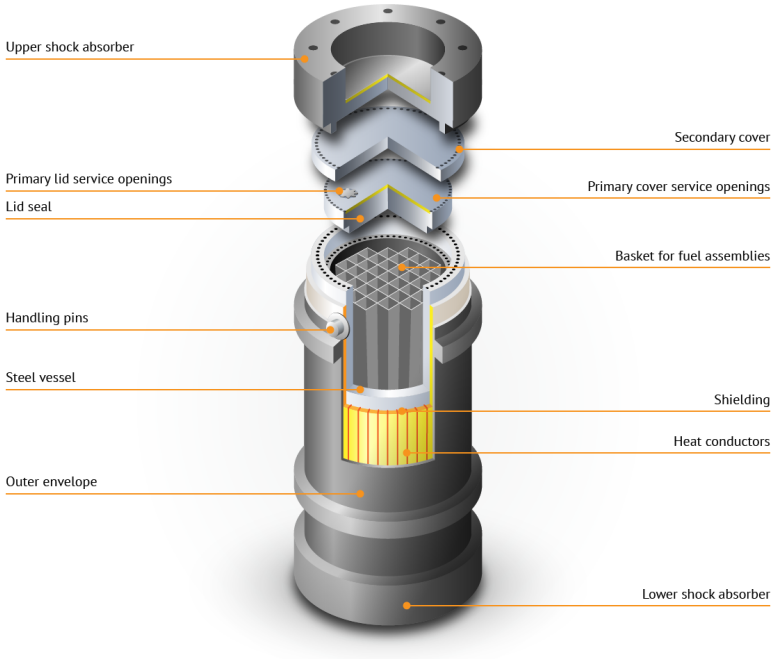

Fig. 2.4 Dual-purpose cask [32]

2.5.5 Multi-purpose systems

It is an extension of the dual-purpose system described in Chapter 2.5.4. It can be based on canisters or casks. An important factor when considering this technology is that it is not yet fully developed. A consequence of the immaturity of this technology is the economic risk, which is compounded by the fact that the design life of the storage and waste package is significantly longer than the design life of the storage or transport systems. [31]

2.6 Deep geological repository

A deep geological repository represents long-term storage. It is a sophisticated technological facility where SNF from nuclear reactors can be stored. It can also store HLW generated by the nuclear power, research, and medical industries. A deep repository is usually constructed in a suitable crystalline rock mass at a depth of around 500 m below the Earth's surface. Research activities, a subsurface investigation, and a public opinion study precede the construction. SNF and HLW are safely isolated from the environment for hundreds of thousands of years in the repository. Figure 2.5 illustrates the principle of a deep geological repository. [33]

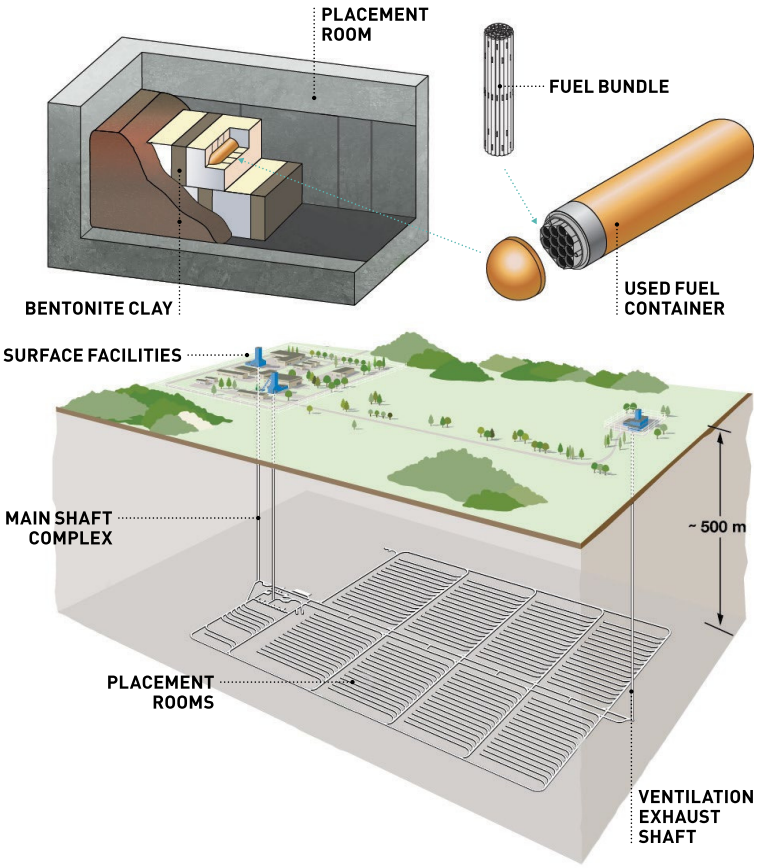

Fig. 2.5 Deep geological repository [34]

2.7 Storage of SNF from VVER-1000

This chapter examines the medium-term storage of SNF by countries that have successfully operated the VVER-1000. The chapter is based on information that is publicly available on the Internet.

Tab. 2.3 List of VVER-1000 and SNF storage type [35, 36, 37, 38, 39, 40, 41, 42, 43]

State	NNP	Storage facility	Storage type	Producer
India	Kudankulam	On-site	Wet	Holtec International
Ukraine	South Ukraine, Rivne Khmelnytskyi	Chernobyl	Dry	Holtec International
	Zaporizhzhia	On-site	Dry	Holtec International
Czech Republic	Temelín	On-site	Dry	ŠKODA JS
Bulgaria	Kozloduy	On-site	Wet	Rosatom
China	Tianwan	On-site	Dry	Orano SA
Iran	Bushehr	Russia	Reprocessing	Rosatom
Russia	Novovoronezh, Kalinin Rostov, Balakovo	Zheleznogorsk	Dry,Wet	Rosatom

2.8 Holtec International

US-based Holtec International is a global supplier of equipment and services essential to nuclear, solar, geothermal and fossil power plants in the energy and petrochemical industries. Holtec is also a developer of SMRs potentially suitable for the needs of the CZE. The detailed parameters of the SMR are described in Chapter 3.4.6. [44]

2.8.1 Wet storage system

A more detailed description of the principle of wet storage is explained in Chapter 2.4. The Indian NNP Kudankulam uses high-density storage racks for wet storage AFR. The high-density racks are of the honeycomb type and use Metamic's patented Metamic neutron absorption technology. Metamic technology is based on metal matrix composites with no porosity. The patented detuned honeycomb technology minimizes the kinematic response of the modules during earthquake shaking and maintains a large subcriticality reserve for all design accidents. Holtec also manufactures DREAM inserts that serve as neutron absorbers without the need to replace the entire rack. For operations between two spent fuel pools, Holtec supplies a single-purpose transport cask called the Shielded Transport Cask. [45]

2.8.2 Dry storage system

Holtec is providing Ukraine with a Multi-purpose canister (MPC) that can hold up to 31 FAs from the VVER-1000. The MPC is designed as a versatile package for on-site storage, transport, and even permanent storage in a deep repository. The entire structure is made of stainless steel. The honeycomb construction ensures a robust

design suitable for accident events. Complete edge-to-edge continuity between the continuous cells inside the basket allows an intermittent path for efficient heat dissipation. [36, 46]

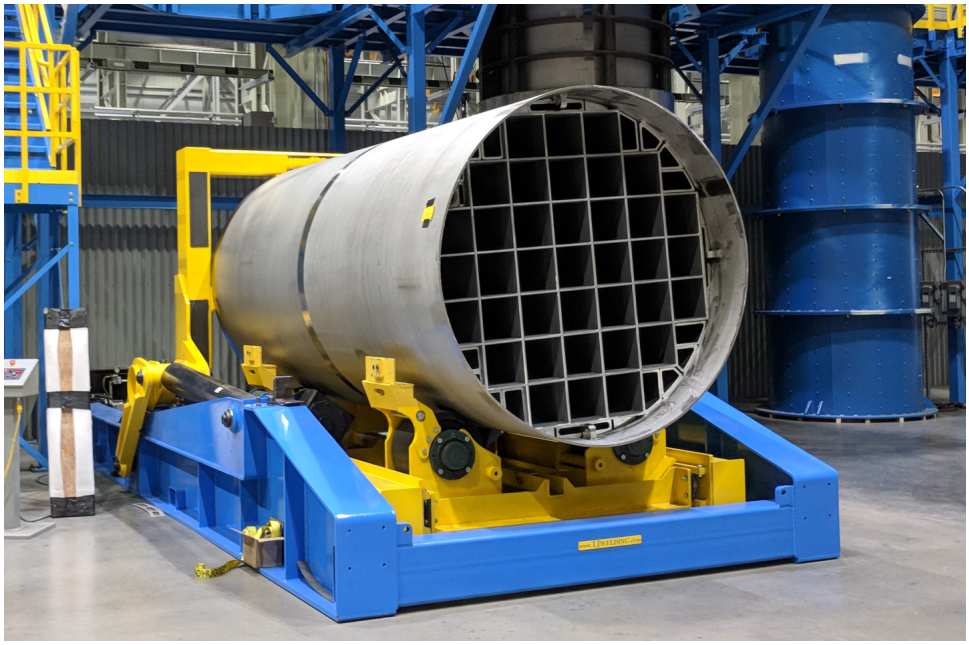

Fig. 2.6 Multi-purpose canisters [46]

The MPC is then loaded into the HI-STAR transport cask, which transports the SNF to the storage facility. The HI-STAR system is designed for storage in the vertical position and for transport in the horizontal position by using AL-STAR Impact Limiters. These overpacks combine the role of a RPV and a robust transport device. HI-STAR consists of three shells:

- innermost shell - containment boundary,
- series of thick steel shells - gamma shielding,
- outer shell - neutron shielding.

Fig. 2.7 HI-STAR [46]

In the last stage, the MPC is unloaded from the HI-STAR transport cask and then loaded into the HI-STORM storage cask. It is a vertical, passively ventilated, cylindrical overpack designed to minimize local dose and provide a robust enclosure for the MPC. The steel cask is designed to withstand high winds, tornadoes, lightning, and other natural phenomena easily. The HI-STORM comes in two configurations, namely above ground and underground. [35, 45]

Fig. 2.8 HI-STORM cask with handling device [46]

2.9 ŠKODA JS

ŠKODA JS is a major Czech engineering company based in the CZE, focusing on constructing and servicing NPPs at home and abroad. Since 2022, the company has become part of CEZ GROUP.

The NPP Temelín stores SNF in a ŠKODA 1000/19 and ŠKODA 1000/19M casks. The dual-purpose cask is designed for transport and storage. In total, the cask is capable of holding 19 FAs from VVER-1000. The cask consists of a thick-walled cylindrical body with a closing system of two lids and an inner basket of aluminum with boron content. The cask also has 185 radial fins to increase the heat transfer surface. The cask containment system consists of two primary and secondary lids of corrosion-resistant forged steel. These lids are attached to the forged carbon steel body. The body is equipped with a leak monitoring system and an anti-mechanical and weatherproof system. To improve neutron shielding, the cask has 2x42 moderator high-density polyethylene rods and polyethylene plates under the secondary lid and the closure plate. At the bottom of the body are two shut-off valves to equalize the pressure in the area of the moderator plates and rods. To limit mechanical shock stresses, the body is equipped with shock absorbers. [13]

Fig. 2.9 Škoda 1000/19 [47]

2.10 Orano SA

ORANO SA delivered around 31 NUHOMS 31VTH canisters to China in 2016, similar to the existing 32PTH1 canisters. Due to slight design changes, the 31VTH canister can hold slightly fewer FAs than the 32PTH1. From a design point of view, it is a high-temperature dual-purpose canister suitable for storage and transport. Compared to standard NUHOMS canisters, the 31VTH canister has a larger diameter of around 8 cm, which allows for a higher payload and better heat dissipation. The basket is made of boron and aluminum. The shell and cover plates are stainless steel.

The NUHOMS 31VTH canister, inserted into the NUHOMS EOS HSM (Horizontal Storage Module), is made of reinforced concrete and structural steel. The reinforced concrete shields the system from neutrons and gamma radiation. Decay heat is removed from the system by all types of heat transfer. Outside air enters the system through ventilation inlets and circulates the canisters and heat shields. The air is then exhausted through the outlet vents. The passive cooling system is designed to maintain the integrity of the fuel cladding. [40, 48, 49]

2.11 Rosatom

Rosatom is a Russian state corporation focused on nuclear energy. Its operations are spread all over the world. It is also evident in Table 2.3, where we can see that Rosatom takes care of the management of SNF from VVER-1000 in Bulgaria, China, and obviously in Russia.

The only source of SNF in Bulgaria is the Kozloduy NNP. SNF from VVER-1000 is stored in a wet storage facility. Each VVER-1000 basket contains 12 FAs with a maximum permitted residual heat capacity of 20 kW and an activity of 0.41 EBq. At the same time, some of the fuel was sent to Russia for reprocessing before the military conflict between Russia and Ukraine. The transport was carried out using a TUK-141O transport cask, which weighs over 100 t and can hold 18 FAs. [39, 50, 51]

In 2015, Iran concluded a deal with Russia to send SNF back to Russia for reprocessing. It is unclear if this contract is still in place due to the military conflict between Russia and Ukraine. In any case, the Bushehr NPP is built with a spent fuel pool, so there is temporary short-term storage capacity. [41]

For transport from the plant, the Russians use the TUK-141O transport cask, which is suitable for the increased initial enrichment (up to 4.92 wt% U-235) and increased burnup (up to 68 GWd/t_U). The cask body is made of high-strength cast iron with spheroidal graphite, the spacer grid is stainless steel with boron, and the body is filled with solid neutron shielding. The capacity of the TUK-141O cask is 18 FAs, capable of withstanding 18 loadings. The Russians also use TUK-13/1V, TUK-10V-1, and TUK-13V transport casks.

After transportation, the SNF is stored in a basket within a wet storage facility (see Chapter 2.4), where the fuel spends 10-15 years cooling before it can be transferred to dry storage. The fuel is then loaded into a canister and stored in the dry long-term storage of the SNF.

The Russian portfolio also includes the dual-purpose TUK-137D cask for transport and long-term storage

of SNF from the VVER-1000/1200. The cask can hold up to 20 FAs with initial enrichment up to 5 wt% and increased burnup up to 70 GWd/t_U. The fuel must be cooled for a minimum of 9 years before insertion into the cask to ensure that it is capable of 50 years of operation. [52, 53, 54]

3 Small modular reactors

According to the IAEA, SMRs are advanced nuclear reactors that have a power capacity of up to 300 MW_e per unit, which is approximately one-third of the generating capacity of full-scale nuclear reactors. They use the same physical principles as full-scale NNPs. [55]

Tab. 3.1 Classification of fission reactors by IAEA [55]

Reactor size	Electrical capacity (MW_e)
Large, conventional reactor	700>
Small modular reactor	<300
Microreactor	<10

3.1 iPWR

The development of SMRs is inseparably linked to the integral pressurized water reactor (iPWR) concept. If we compare a classical PWR with an iPWR, we find that from a physical point of view, there is no difference between them, but they differ only in design. The structural difference between the PWR and the iPWR is that the iPWR has a steam generator, pressurizer, and reactor coolant pumps installed inside the RPV. It means the iPWR has the entire primary circuit integrated into the RPV. This solution brings simpler designs, the possibility of passive systems, and smaller footprints, which are especially important for SMRs.

3.2 Advantages

Based on their nature, SMRs can be placed where full-scale NNPs can not. This is possible due to the smaller footprint that is characteristic of SMRs. Another critical feature of SMRs is modularity. Prefabricated units of SMRs can be made in a manufacturing factory and then transported and installed at the site of demand. This process can significantly reduce manufacturing and construction costs in the future, as well as construction time. SMRs require a smaller initial capital investment than full-scale nuclear reactors, making them more accessible to a wider range of investors and utilities. Compared with conventional NNPs, SMRs are more straightforward in design and often rely on passive safety systems and inherent reactor safety characteristics such as low power and operating pressure. It means that no human intervention or external energy or force is required to shut down the reactor because SMRs rely on physical phenomena such as natural circulation, convection, gravity, and self-pressurization, thanks to passive systems. These safety features significantly reduce the risk of radioactivity leaking into the environment in the event of an accident. A final essential feature is a reduced requirement for nuclear fuel. SMRs may require a reduced refueling intensity every 3 to 7 years. Conventional NPPs must refuel every 1 to 2 years. This fact plays to the advantage of SMRs because NPPs lose most of their earnings during reactor shutdowns. Some SMR designs allow for fuel changes even after 30 years. [55]

3.3 Disadvantages

In general, one of the most essential factors in any construction is finance. This fact is a disadvantage for the first SMR built. Illustrative Figure 3.1 shows that the construction of SMRs is significantly less profitable than the construction of full-scale NNPs. Factors affecting construction costs include simplifying design, standardization, harmonization, modularisation, and factory building. NPPs have gone through this process; unsurprisingly, this is the same today with SMRs. Nuclear power is one of the options for eliminating carbon dependency, so we must consider that research and development of new industrial technologies take time. [56]

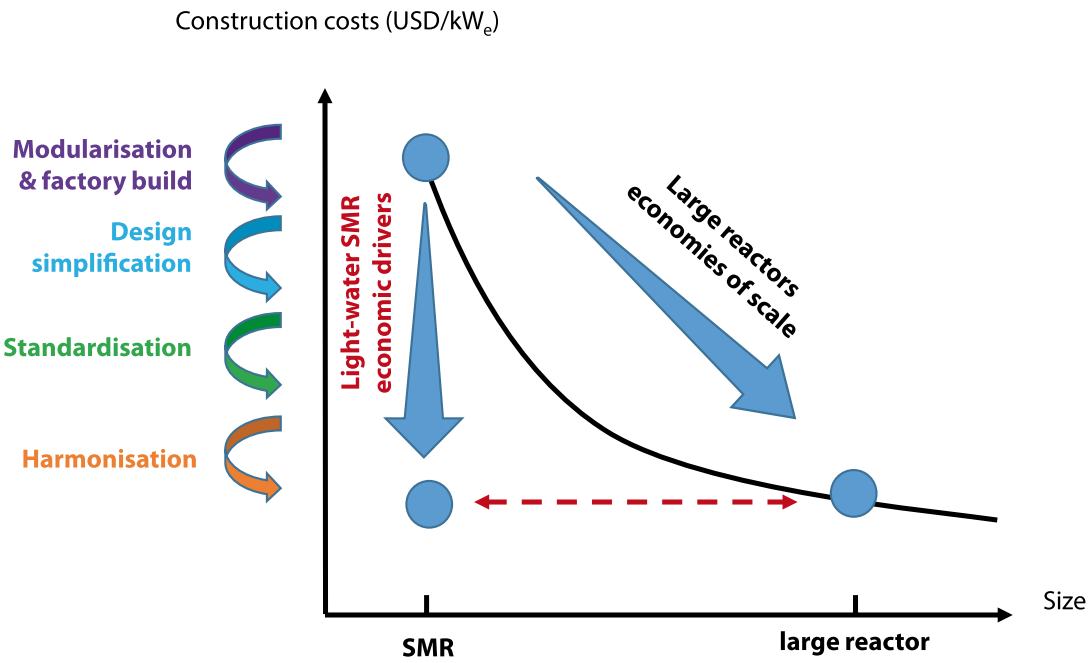

Fig. 3.1 SMR key economic drivers to compensate for diseconomies of scale [56]

SMRs operate with the same control systems as conventional NNPs, which is not beneficial from a fixed cost point of view because SMRs are expected to reduce fixed costs as much as possible to achieve a lower purchase price.

Another problem is that, in terms of legislation, SMRs are seen as nuclear facilities with the exact requirements as full-scale NPPs. For instance, SMRs and large NPPs encounter similar conditions in safety and fire services. Moreover, the need for an emergency planning zone remains the same for SMRs and large NPPs under current legislation.

3.4 Potential SMRs for the CZE

This chapter focuses on the potential SMRs which, according to CEZ GROUP, are most suitable for the conditions of the CZE. CEZ GROUP is not the only company considering SMRs, but it is the furthest in the process of applying SMRs. It has already successfully operated two NPPs, NPP Dukovany and Temelín. CEZ GROUP has already signed a memorandum of cooperation with global SMR developers NuScale, GE Hitachi, Rolls Royce, EDF, Westinghouse, KAERI, and Holtec. All the reactors mentioned are LWRs operating with thermal neutron spectrum. It has not yet been decided which design CEZ GROUP will choose; therefore, this master's thesis will be dedicated to all potential designs valid at the beginning of 2024. [57]

3.4.1 NuScale, VOYGR

The US SMR from Nuscale Power Corporation uses a design based on proven PWR technologies but in a new integral design. The SMR VOYGR was developed to supply energy for electrical generation, district heating, desalination, commercial-scale hydrogen production, and other heat processing applications. It is the first SMR design to receive design approval from the US Nuclear Regulatory Commission (NRC). VOYGR uses natural circulation. VOYGR will be produced in 3 configurations: VOYGR-4, VOYGR-6, and VOYGR-12, with the number at the end indicating the number of reactors, i.e., VOYGR-12 has a capacity of 924 MW_e. [58, 59]

Tab. 3.2 VOYGR major technical parameters [58, 59]

Parameter	Value
Technology developer, country of origin	NuScale Power Corporation, USA
Reactor type	iPWR
Coolant/moderator	Light water
Thermal/electrical capacity (MW_{th}, MW_e)	250/77
Primary circulation	Natural circulation
Primary system pressure (MPa)	13.8
Secondary system pressure (MPa)	4.3
Core inlet/exit temperatures (°C)	249/316
Fuel type/assembly array	UO_2 pellet/17x17 square array
Number of fuel assemblies in the core	37
Fuel enrichment (wt%)	4.95 (max)
Core discharge burnup (GWd/t_U)	$\geq$45
Refueling cycle (mo)	18
Reactivity control mechanism	Control rod drive mechanism, soluble boron
Approach to safety systems	Passive
Design life (yr)	60
Plant footprint (m^2)	140,000 (VOYGR-12)
RPV height/diameter (m)	17.7/2.7
Module weight (t)	700
Seismic design (SSE)	0.5 g
Fuel cycle requirements/approach	Nominal three-stage in-out refueling scheme
Distinguishing features	Unlimited time for core cooling without power, or water addition, or operator action
Design status	Equipment manufacturing in progress

3.4.2 GE Hitachi, BWRX-300

The BWRX-300, designed by GE Hitachi Nuclear Energy, is a water-cooled reactor based on natural light water circulation. This design uses simple safety systems controlled by natural phenomena. It is the tenth generation of BWR and represents the most straightforward BWR design since GE began manufacturing BWRs in 1955. The BWRX-300 has an installed electrical capacity of 300 MW_e, which is suitable for base load, load following electrical generation within a range of 50-100 % power, district heating, synthetic fuel production, hydrogen production, and other process heat applications. The BWRX-300 leverages the US NRC-approved economic simplified BWR design. An exciting feature of the BWRX-300 is that the reactivity of conventional BWRs is also controlled by varying the coolant flow through the reactor, which is impossible with the BWRX-300 because it does not include a reactor coolant pump.

The BWRX-300 is the only BWR representative considered for the CZE and would also be the first BWR technology in the country. This fact could significantly influence the decision on the selected design, as the remaining SMRs considered for the CZE are PWRs with considerable experience in the CZE. [58, 60]

Tab. 3.3 BWRX-300 major technical parameters [58, 60]

Parameter	Value
Technology developer, country of origin	GE Hitachi Nuclear Energy, United States and GE Hitachi Nuclear Energy, Japan
Reactor type	BWR
Coolant/moderator	Light water
Thermal/electrical capacity (MW_{th}, MW_e)	870/300
Primary circulation	Natural circulation
Primary system pressure (MPa)	7.2
Secondary system pressure (MPa)	Direct cycle
Core inlet/exit temperatures (°C)	270/288
Fuel type/assembly array	UO_2 pellet/10×10 square array
Number of fuel assemblies	240
Fuel enrichment (wt%)	3.81 (avg)/4.95 (max)
Core discharge burnup (GWd/t_U)	49.6
Refueling cycle (mo)	12-24
Reactivity control	Control rod drive mechanism, solid burnable absorber (B_4C, Hf, Gd_2O_3)
Approach to safety systems	Passive
Design life (yr)	60
Plant footprint (m^2)	9,800
RPV height/diameter (m)	26/4
Module weight (t)	485
Seismic design	0.3 g
Fuel cycle requirements/approach	Open fuel cycle utilizing standard BWR fuel
Distinguishing features	Natural circulation BWR, integral RPV isolation valves, isolation condenser
Design status	Detailed design

3.4.3 Rolls-Royce Ltd, Rolls-Royce SMR

Rolls-Royce is coming to market with its SMR, which aims to provide affordable, low-emission power generation. It relies on fast and repeatable construction, which is enhanced by optimizing site layout and maximizing modular construction. Rolls-Royce's pressurized SMR is primarily designed to deliver baseload power to both onshore and offshore sites. The SMR can be designed to support other industrial applications requiring heat or cogeneration. The modules for construction will be divided into three categories: heavy pressure vessels, mechanical electrical and plumbing, and civil engineering.

The Rolls-Royce SMR uses standard industrial fuel (17x17 square array). The installed capacity is 470 MW_e, which, according to the IAEA (see Chapters 3), does not fit the definition of an SMR. This capacity is almost comparable to one unit of the Dukovany[1] NPP. As mentioned in Chapter 3.3, the disadvantage of the first SMRs is that their construction will be considerably more expensive than full-scale NPPs if the price-performance ratio is evaluated. This relationship is illustrated in Figure 3.1. So, for this SMR, the main question will be whether to build one or two units of full-scale NPPs instead. [58, 61]

Tab. 3.4 Rolls-Royce SMR major technical parameters [58, 61]

Parameter	Value
Technology developer, country of origin	Rolls-Royce SMR Ltd, United Kingdom
Reactor type	3-loop PWR
Coolant/moderator	Light water
Thermal/electrical capacity (MW_{th}, MW_e)	1,358/470
Primary circulation	Forced circulation
Primary system pressure (MPa)	15.5
Secondary system pressure (MPa)	7.8
Core inlet/exit temperatures (°C)	295/325
Fuel type/assembly array	UO_2 pellet/17x17 square array
Number of fuel assemblies	121
Fuel enrichment (wt%)	4.95 (max)
Refueling cycle (mo)	18
Core discharge burnup (GWd/t_U)	60
Reactivity control	Control rod drive mechanisms
Approach to safety systems	Passive and active
Design life (yr)	60
Plant footprint (m^2)	40,000
RPV height/diameter (m)	7.9/4.2
Module weight (t)	150
Seismic design	>0.3g
Fuel cycle requirements/approach	Open cycle
Distinguishing features	Modular approach facilitating rapid and cost-effective build
Design status	Detailed design

[1] VVER-440/V-213 with an installed capacity of 440 MW_e

3.4.4 EDF, NUWARD

SMR NUWARD is a collaboration between several French companies, where each company brings its know-how to the project. NUWARD SMR is based on proven PWR technologies and critical technological innovations. The reactor consists of two fully integrated reactors housed in one shared building, hence the 2x170 MW_e indicated in Table 3.5. Each reactor offers independent control and flexible operation. NUWARD is also designed to support other energy-intensive applications such as cogeneration, hydrogen production, district heating, and water desalination. NUWARD aims to replace coal-fired power plants with a capacity of 300-400 MW_e, which in the case of the CZE means that NUWARD can replace up to two power plant units because the most typical capacity of one power plant unit in the CZE is 200 MW_e. NUWARD can be safely located near city centers or industrial sites thanks to the reduced need for an emergency planning zone.

As with all SMRs, it is no exception that NUWARD will use conventional nuclear fuel 17x17 from UO_2. The reactor uses forced circulation of light water. The reactivity is controlled by control rods. NUWARD does not use boron as a neutron absorber in normal operations but only in rare cases. [58, 62]

Tab. 3.5 NUWARD major technical parameters [58]

Parameter	Value
Technology developer, country of origin	EDF, France with major contributions from with CEA,Naval Group, Framatome, TechnicAtome, and Tractebel-Engie, France
Reactor type	iPWR
Coolant/moderator	Light water
Thermal/electrical capacity (MW_{th}, MW_e)	2x540/2x170
Primary circulation	Forced circulation
Primary system pressure (MPa)	15
Secondary system pressure (MPa)	4.5
Core inlet/exit temperatures (°C)	280/307
Fuel type/assembly array	UO_2 pellet/17x17 square array
Number of fuel assemblies	76
Fuel enrichment (wt%)	4.95 (max)
Refueling cycle (mo)	24
Reactivity control	Control rod drive mechanism, solid burnable poisons
Approach to safety systems	Passive
Design life (yr)	60
Plant footprint (m^2)	3,500, nuclear island including fuel storage pool
RPV height/diameter (m)	15/5
Module weight (t)	310
Seismic design	0.3 g
Distinguishing features	Integrated NSSS with pool submerged containment boron-free in normal operation and in all Design Basis Conditions, semi-buried nuclear island
Design status	Conceptual design

3.4.5 KAERI and K.A.CARE, SMART

Under the name SMART is hidden System-integrated Modular Advanced ReacTor. It is a collaborative project between the Republic of Korea and the Kingdom of Saudi Arabia to develop a pressure-integrated SMR. The significance of this collaboration is in establishing a strategic partnership. The SMART project aims to improve economics through system simplification, modular component design, reduced construction time, and high SMR availability.

It is a LWR with a thermal capacity of 365 MW_{th}, which can be transformed into up to 107 MW_e of electricity. The value of the electrical output should be sufficient to meet the electricity requirements for a city with a population of 100,000 inhabitants. SMART is a multi-purpose SMR suitable for power generation, desalination, district heating, and industrial process heat. This performance is achieved using 57 FAs made of UO_2 pellets that form a classic 17x17 square array. Designers indicate a fuel enrichment below 5 wt%, which fulfills the criterion for LEU fuel typical of today's power reactors.

SMART is equipped with a reactor coolant pump, so it is not one of the SMRs using natural circulation. The use of control rods and soluble boron achieves an effort to distribute power evenly throughout the reactor. [58]

Tab. 3.6 SMART major technical parameters [58]

Parameter	Value
Technology developer, country of origin	KAERI, Republic of Korea and K.A.CARE, Kingdom of Saudi Arabia
Reactor type	iPWR
Coolant/moderator	Light water
Thermal/electrical capacity (MW_{th}, MW_e)	365/107
Primary circulation	Forced circulation
Primary system pressure (MPa)	15
Secondary system pressure (MPa)	5.8
Core inlet/exit temperatures (°C)	296/322
Fuel type/ assembly array	UO_2 pellet/17x17 square array
Number of fuel assemblies	57
Fuel enrichment (wt%)	4.95 (max)
Core discharge burnup (GWd/t_U)	<54
Refueling cycle (mo)	30
Reactivity control	Control rod drive mechanism and soluble boron
Approach to safety systems	Passive
Design life (yr)	60
Plant footprint (m^2)	90,000
RPV height/diameter (m)	18.5/6.5
Module weight (t)	1,070 (including coolant)
Seismic design	> 0.3 g with 0.18 g of automatic shutdown
Fuel cycle requirements/approach	Conventional LWR requirements applie (spent fuel storage capacity: 30 yr)
Distinguishing features	Coupling with desalination and process heat application, integrated primary system
Design status	Detailed design

3.4.6 Holtec, SMR-160

SMR-160 is developed by Holtec International. It is an advanced PWR-type SMR in a classic design. The SMR-160 has a thermal capacity of 525 MW_{th} or an electrical capacity of 160 MW_e. According to Holtec International's philosophy, the SMR-160 is designed as a "walk-away safe" power plant, meaning that human intervention is not required to handle design accidents and safely remove decay heat. The SMR is greatly simplified from conventional NPPs to improve its manufacturability, constructability, and maintainability, facilitated by including fully passive natural circulation safety systems. The estimated construction time per unit is 24 months.

The primary application of the SMR-160 is power generation with optional cogeneration equipment, which can be hydrogen generation, thermal energy storage, district heating, and seawater desalination. The SMR-160 is also suitable for locations where water is scarce due to its patented air-cooled condenser system. The SMR also has black-start and islanded operation capabilities, suitable for areas with unstable power grids. [58]

Tab. 3.7 SMR-160 major technical parameters [58]

Parameter	Value
Technology developer, country of origin	Holtec International, USA
Reactor type	PWR
Coolant/moderator	Light water
Thermal/electrical capacity (MW_{th}, MW_e)	525/160
Primary circulation	Natural circulation
Primary system pressure (MPa)	15.5
Secondary system pressure (MPa)	3.4
Core inlet/exit temperatures (°C)	243/321
Fuel type/assembly array	UO_2 pellet/17x17 square array
Number of fuel assemblies	57
Fuel enrichment (wt%)	4.0 (avg)
Core discharge burnup (GWd/t_U)	45
Refueling cycle (mo)	24
Reactivity control	Control rod drive mechanism, soluble boron
Approach to safety systems	Fully Passive; utilize varied phenomena and redundancy.
Design life (yr)	80
Plant footprint (m^2)	28,000
RPV height/diameter (m)	15/3
Module weight (t)	295 (with fuel and internals)
Seismic design	0.5 g, derived from the NRC Regulatory Guide 1.60
Distinguishing features	Passive safety cooling systems and active non-safety systems; critical components below grade. Integrated dry SNF storage and transportation system
Design status	PSAR for commercial project by 2023, with detailed design to complete by 2025

3.4.7 Westinghouse, AP300

The last is the American SMR from Westinghouse. This company draws on more than 70 years of experience developing and deploying new nuclear technologies. The operational experience and tens of millions of hours of development of the AP1000 reactor serve as the foundation for the AP300. The AP300, with an electrical capacity of 300 MW_e (990 MW_{th}), is a PWR in integral design.

The AP300 meets the expectations associated with SMRs, which are smaller scale, modular design, and the most advanced passive safety systems. The developer boldly states an SMR lifetime of more than 80 years. The AP300 targets grids with large amounts of installed renewable energy capacity because of its ability to change its power output as demand requires. SMR also supports district heating, desalination, and hydrogen production.

Westinghouse is one of the world leaders in nuclear fuel supply. The AP300 uses a robust fuel design incorporating several proven and advanced fuel elements and a four-year refueling cycle. [58, 63]

Tab. 3.8 AP300 major technical parameters [58, 63]

Parameter	Value
Technology developer, country of origin	Westinghouse Electric Company LLC, USA
Reactor type	iPWR
Coolant/moderator	Light water
Thermal/electrical capacity (MW_{th}, MW_e)	990/300
Primary circulation	Forced circulation
Primary system pressure (MPa)	15.5
Secondary system pressure (MPa)	-
Core inlet/exit temperatures (°C)	294/324
Fuel type/assembly array	UO_2 pellet/17x17 square array
Number of fuel assemblies	89
Fuel enrichment (wt%)	4.95 (max)
Core discharge burnup (GWd/t_U)	>62
Refueling cycle (mo)	24
Reactivity control	Control rod drive mechanism, soluble boron
Approach to safety systems	Passive
Design life (yr)	>80
Plant footprint (m^2)	65,000
RPV height/diameter (m)	28/3.7
Seismic design	Based on CEUS sites
Distinguishing features	Incorporates passive safety systems and proven components of the AP1000 plant and earlier Westinghouse designs
Design status	Conceptual design completed

3.5 Places of application

According to CEZ GROUP, the most prominent Czech power plant operator, three suitable locations for applying SMRs have already been selected. The first SMR should be built at the site of NPP Temelín because this location is suitable for building other nuclear facilities, and thousands of qualified people are already employed there. After the first SMR, further SMRs could be built on the locations of the current coal-fired power plants Dětmarovice and Tušimice. These two locations will undergo many additional intensive exploration and monitoring activities before it is finally determined whether they are suitable for SMRs. Exploration work began in February 2023, dealing with tectonic faults, assessing the area's hydrogeology, and analyzing the underlying bedrock. Transformation of non-nuclear locations requires a series of exploration activities taking 3-5 years. CEZ Group has committed itself in its Vision 2030 to prepare to construct SMRs with a total capacity of over 1 GW_e after 2040.

In addition to the locations mentioned above, CEZ GROUP considers other locations suitable for the construction of SMRs. The analyses relate to the locations of the power plants Prunéřov, Ledvice, Mělník and Dukovany. In addition to Dukovany, these are locations of existing coal-fired power plants where the infrastructure for central heat supply already exists. [57]

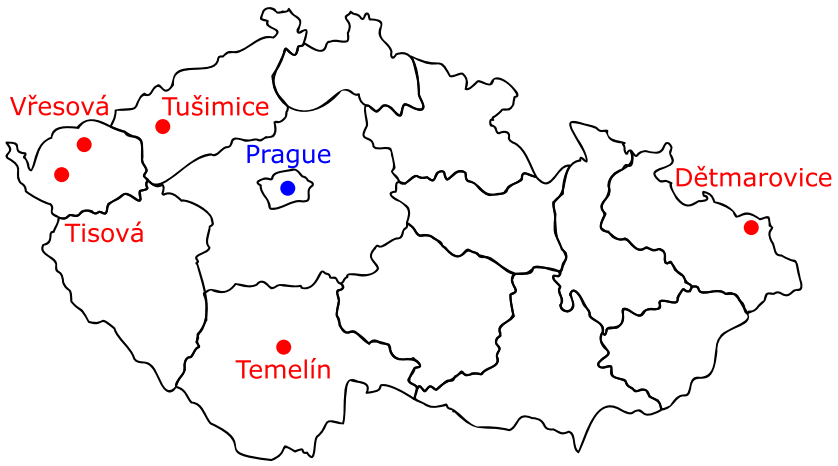

Fig. 3.2 Potential locations for SMRs in the CZE [57, 64]

According to Sokolovská uhelná and SUAS GROUP, these two companies successfully applied for a grant in the US Phoenix program for a feasibility study to locate the SMR. The United States Department of State aims to support the transition from coal-fired power and heat generation to neutral nuclear energy sources. The

cost of feasibility can reach a maximum of USD 2 million. The companies are considering applying SMR to the Tisova power plant site or the industrial complex in Vřesová. Geological exploration of the selected sites is currently in progress. Based on the exploration, it will be proven whether these areas are suitable for the location of SMR. Currently, companies consider PWRs with an electrical capacity of 400-500 MW, which corresponds to the Karlovy Vary Region's requirements. [64]

4 Computational models

All simulations in the master's thesis were created using the SCALE/TRITON computer code. The TRITON reactor physics sequence in SCALE allows the simulation of time-dependent material depletion in 1D, 2D, or 3D depletion models by coupling the neutron transport solvers with the ORIGEN depletion and decay code.

Table 4.1 lists the reactors that are the object of the simulations. These are the VVER-1000 and the SMRs. Additionally, the table delineates the fuel types, assembly arrays, and the appropriate fuel supplier for each specific reactor. All reactors operate with fuel in the form of UO_2 pellets. Notably, the VVER-1000 is a solitary reactor employing hexagonal FAs, a characteristic feature of Russian reactor designs. Among SMRs, only one BWR uses 10x10 square FAs, which are comparatively smaller than the conventional 17x17 FAs used by the rest of the SMRs and represent the world's prevailing standard.

Tab. 4.1 Basic fuel information [65, 66, 67, 68, 69]

Reactor	Fuel type/assembly array	Supplier
VOYGR	UO_2 pellet/17x17 square array	Framatome
BWRX-300	UO_2 pellet/10×10 square array	GE Hitachi
Rolls-Royce SMR	UO_2 pellet/17×17 square array	Westinghouse
NUWARD	UO_2 pellet/17x17 square array	Framatome
SMART	UO_2 pellet/17x17 square array	KAERI
SMR-160	UO_2 pellet/17x17 square array	Framatome
AP300	UO_2 pellet/17x17 square array	Westinghouse
VVER-1000	UO_2 pellet/VVER-1000 hexagonal array	Westinghouse

4.1 Composition

The models were based on publicly available information. This information is strongly dependent on the development stage of the specific reactor designs. Regarding the state of development, the BWRX-300 and SMR-160 reactor designs are the most advanced because, ignoring the VVER-1000, they are the only reactor designs that give an average fuel enrichment value. Other developers typically report a fuel enrichment of 4.95 wt%, which is the maximum feasible fuel enrichment value when respecting inaccuracies during enrichment. In models where the average enrichment value is known, the average value was included, while where only the maximum enrichment is known, this maximum value was considered. This fact introduces some inaccuracies in the results of the depletion simulation that need to be considered when evaluating the results.

Tab. 4.2 Fuel composition [70, 71, 72, 73, 74]

Reactor	Fuel enrichment (wt%)	Cladding materials
VOYGR	4.95 (max)	M5
BWRX-300	3.81 (avg)	Zircaloy 2
Rolls-Royce SMR	4.95 (max)	Optimized ZIRLO
NUWARD	4.95 (max)	M5
SMART	4.95 (max)	HANA-4
SMR-160	4.0 (avg)	M5
AP300	4.95 (max)	Optimized ZIRLO
VVER-1000	4.45 (avg)	Optimized ZIRLO

Table 4.3 presents the weight percentages of fuel cladding materials categorized by fuel supplier. Currently, zirconium-based materials are primarily applied as cladding materials for their fundamental properties, such as high corrosion resistance, low neutron absorption, high strength, and thermal stability. Additionally, I utilized the cladding material for constructing guide tubes, instrumentation tubes, central water rods, and channel boxes in my model. These materials are usually very similar to fuel cladding materials in practical applications.

Tab. 4.3 Cladding compositions [75]

	Chemical element composition (wt%)						
Cladding materials	Zr	Nb	Sn	Fe	Cr	Ni	Cu
M5	99	1	-	-	-	-	-
Zircaloy 2	98.72	-	1.2	-	0.05	0.03	-
Optimized ZIRLO	98.2	1	0.7	0.1	-	-	-
HANA-4	97.8	1.5	0.4	0.2	0.1	-	-

In all analyses of PWRs, I inserted a timetable data block that respects the change in the density of soluble boron in the moderator caused by the removal of poisons during fuel cycles. The change in density is performed in the SCALE/TRITON by a linear interpolation function between the data pair.

4.2 Geometry

I have analyzed all models in the 2D domain only, using reflective boundary conditions on their boundaries. In the case of the VVER-1000 FA, I used isotropic boundary conditions on all edges except the top and bottom edges, where I applied reflex boundary conditions.

4.2.1 PWR fuels

I chose the Westinghouse design as the reference geometry for the PWR FAs because it was more publicly available than other designs. Again, this inaccuracy may present a bias in the results of the depletion simulation. Table 4.4 illustrates the material composition, including the inner radius of the material that defines the interface between adjacent materials. The table also specifies the inner and outer radius for the guide and central instrumentation tubes, enabling a complete geometry description.

Tab. 4.4 PWR lattice cell calculation information [76, 77]

Description	Parameter (cm)	Material
Fuel radius	0.409575	UO_2
Gap radius	0.41783	^4He
Fuel clad radius	0.47498	Cladding
Full pitch	1.25984	H_2O/B
Inner guide tube radius	0.561	H_2O/B
Outer guide tube radius	0.602	Cladding
Inner instrument tube radius	0.559	H_2O/B
Outer instrument tube radius	0.605	Cladding

In the model design process, I opted to simulate just one-quarter of the FA due to its symmetrical geometry. This decision aimed to speed up the computing processes and thus reduce the computational demands of the device. The final geometry is illustrated in Figure 4.1. Similarly, it is possible to model one-eighth of the FA using this approach. The actual model comprises 264 fuel rods, 24 guide tubes, and one instrumentation tube positioned at the center of the assembly itself.

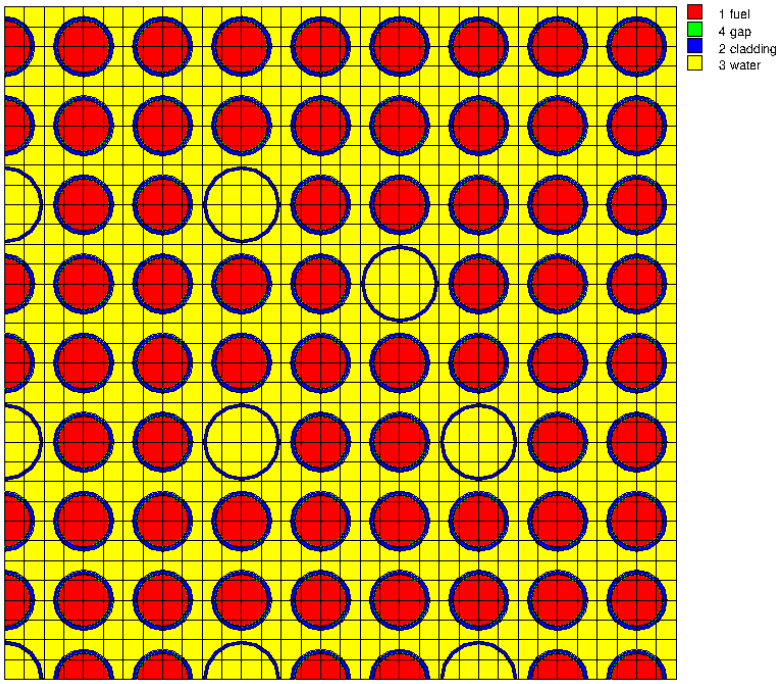

Fig. 4.1 PWR 17x17 FA

4.2.2 BWR fuel

For the BWRX-300, I created a geometry based on the GE Hitachi design to reflect the actual configuration truly. It utilizes a 10x10-8 fuel configuration, where ”-8” represents the presence of two water rods instead of eight fuel rods. In this case, I could no longer rely on the axial symmetry of the fuel because the two water rods disturb the symmetry of the fuel. All radii and materials are visible in Table 4.5, including the inner and outer

radii of the water rods. In contrast to PWRs, BWRs do not utilize boric acid to control reactivity. Therefore, only water without soluble boron was used in the model.

Tab. 4.5 BWR lattice cell calculation information [78]

Description	Parameter (cm)	Material
Fuel radius	0.438	UO_2
Gap radius	0.447	^4He
Fuel clad radius	0.513	Zircaloy 2
Full pitch	1.295	H_2O
Inner water rod radius	1.185	H_2O
Outer water rod radius	1.285	Zircaloy 2

Throughout the development of the geometry, I aimed to be as approximate to the actual system as possible and, therefore, created a detailed model containing two large central water rods and a channel box with rounded corners. The FA comprises 92 fuel rods, as seen in the visualization in Figure 4.2. It can also be seen from this visualization that no guide or instrumentation tubes are located within the FA. The absence of guide tubes is caused by the specifics of the BWR type, where control rod blades are used to control reactivity, unlike the PWR.

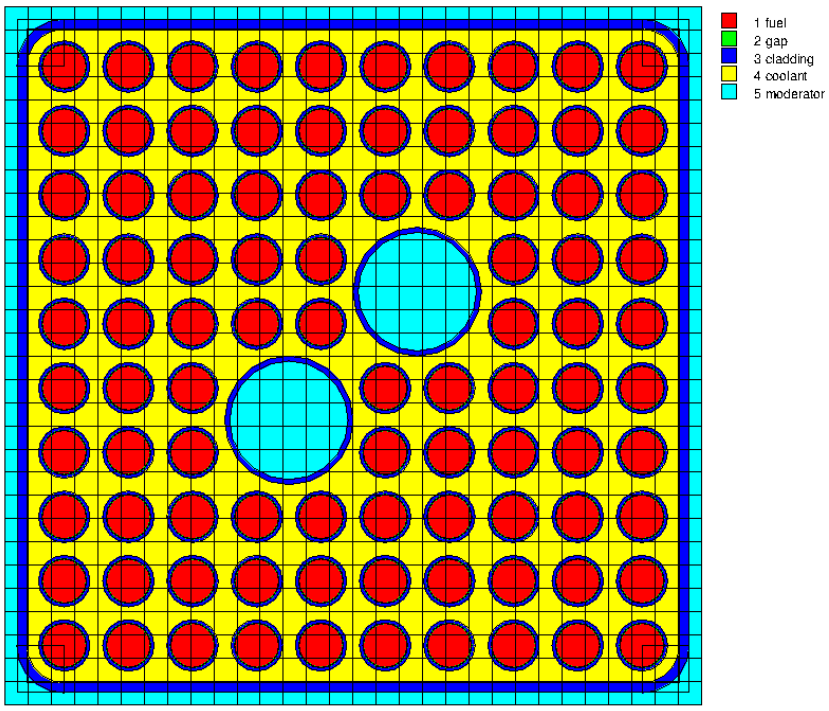

Fig. 4.2 BWR 10x10-8 FA

4.2.3 VVER fuel

The model of the VVER-1000 FA that I designed based on information provided by the American company Westinghouse, which became one of the suppliers of nuclear fuel for the Temelín NNP. Since my thesis deals

with potential SMR for the CZE, I chose the Temelín power plant for the same reason. I have modeled the FA as a whole, but it could also be solved as one-sixth because the fuel is symmetrical. The appropriate parameters and materials are entered in Table 4.6 as with the previous geometries.

Tab. 4.6 VVER-1000 lattice cell calculation information [76]

Description	Parameter (cm)	Material
Fuel radius	0.3922	UO_2
Gap radius	0.4005	^4He
Fuel clad radius	0.4572	Optimized ZIRLO
Full pitch	1.275	H_2O/B
Inner instrument/guide tube radius	0.545	H_2O/B
Outer instrument/guide tube radius	0.6323	Optimized ZIRLO

The only example of a hexagonal FA is the VVER-1000 fuel type, visible in Figure 5.6. I constructed the model using an array embedded in a global hexagon. Subsequently, I filled each array cell with a defined unit containing three concentric circles and one hexagon rotated concerning the global hexagon. I applied SCALE's pre-defined unit 0 to fill in the empty cells in the array. The fuel file contains 312 fuel rods, 18 guide tubes, and one central instrument tube.

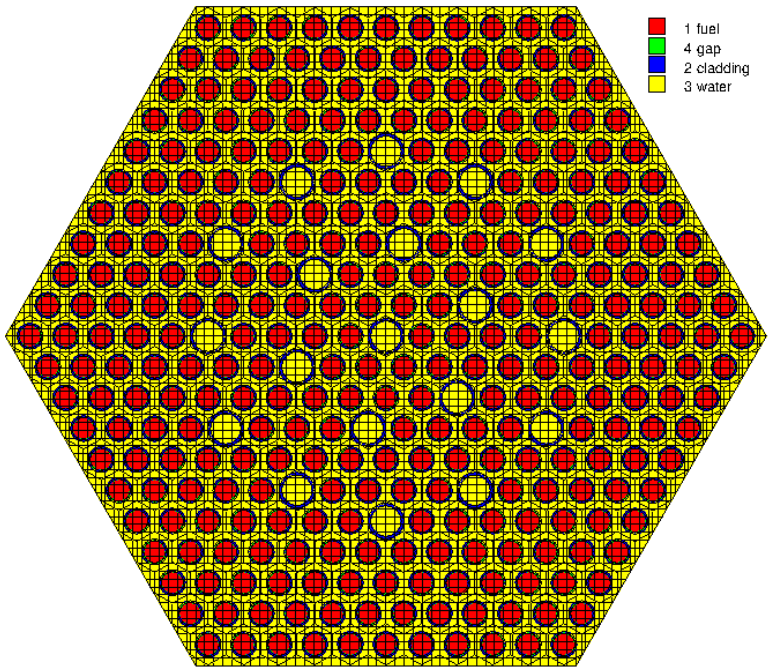

Fig. 4.3 VVER-1000 FA

4.3 Depletion

In the depletion simulations, I used the computational sequence T-DEPL, which allows the computation of 2D multigroup deterministic transport in NEWT code coupled with ORIGEN code depletion. The T-DEPL computational sequence uses pre-tabulated physical lattice data for the neutron calculations, i.e., homogenized cross-sections of a few groups with appropriate discontinuity factors, pin powers, and kinetic parameters, functionalized in terms of burnup and other system conditions including, for example, fuel temperature and moderator density. I chose the cross-section library ENDF/B-VII.1 with 252 neutron groups; the latest version 6.2.4. [79]

Since reactors typically operate at nearly constant power levels, the possibility of depletion due to power is closer to reality, so I applied this method to calculate UO_2 depletion for all fuel types. This depletion method internally determines the corresponding flux from the specific power input and internal fission tables and captures energy releases for the nuclides present and the macroscopic cross-sections of those nuclides. At each time step, a recalculation of the power density of the material is also performed according to the changing nuclide inventory. For non-fuel materials such as cladding, which do not contribute to the total power, I have considered depletion by constant flux. [80]

4.4 Burn data

For the burn data block, I first specified the volume of a single fuel rod. It was determined using the formula for calculating the volume of a cylinder, where the input variables were the active length and the fuel rod radius (see Chapter 4.2). By multiplying the volume of the rod by the density of UO_2, I obtained the mass of a single fuel rod. Then, I multiplied this mass by the number of fuel rods to obtain the total mass of UO_2 per FA. In the last step, I multiplied the total mass by 0.88 because 88 wt% of UO_2 is uranium, which was necessary for further calculations. The results of these mathematical processes are presented in Table 4.7 below. It is essential to mention that the active length for NUWARD is not yet publicly known. Therefore, I have used the same value as SMR-160 and VOYGR because NUWARD will be operated with the same fuel. At the same time, the active length for SMR-160 and VOYGR is already known as 2 m.

Tab. 4.7 FA geometry and mass [58, 74, 81]

Reactor	Active length (m)	Mass of uranium in the FA (kg)	NO. of FAs (-)
VOYGR	2	254.83	37
BWRX-300	3.8	180.84	240
Rolls-Royce SMR	2.8	356.76	121
NUWARD	2	254.83	76
SMART	2	254.83	57
SMR-160	3.65	465.06	57
AP300	2.4	305.79	89
VVER-1000	3.65	501.25	163

I calculated the total mass of uranium in the core as the product of the mass of the FA and its total number.

Similarly, I have determined the specific reactor power as the ratio of the thermal capacity to the total mass of uranium. Its value indicates how much heat a tonne of uranium generates in the reactor. In other words, it is a quantitative parameter that assesses the reactor's heat conversion efficiency. The VVER-1000 achieved the highest specific power value, and the SMR-160 achieved the lowest.

Tab. 4.8 Reactor core information [58, 74]

Reactor	Total mass of uranium (t)	Thermal capacity (MW_{th})	Specific power (MW/t_U)
VOYGR	9.43	250	26.51
BWRX-300	43.4	870	20.05
Rolls-Royce SMR	43.17	1,358	31.46
NUWARD	19.37	540	27.88
SMART	14.53	365	25.13
SMR-160	26.51	525	19.8
AP300	27.22	990	36.38
VVER-1000	81.7	3,120	38.19

The last step was to evaluate the irradiation time, which I evaluated based on the burnup, which indicates how much energy can be extracted from the fuel per unit of initial mass of uranium. The fuel burnup for the reactors under study is stored in Table 4.9. Again, it should be mentioned that the burnup for the NUWARD reactor is not yet publicly known, so I chose the same burnup for the VOYGR and SMR-160 reactors for the same reason as the active length. These three reactors, NUWARD, VOYGR, and SMR-160, operate with the lowest fuel burnup. On the other hand, the highest values are presented for the AP300 and Rolls-Royce SMR reactors. In order to simulate the SNF as precisely as possible, I have respected the design refueling cycle lengths, which are 18 months for four reactors, 24 months for three reactors, and even up to 30 months for the SMART reactor. For the VVER-1000, I have already assumed refueling after 18 months, which has recently been finally approved for the Temelín power plant. I also considered a refueling outage for refueling. For the SMR, the designers specify a refueling outage of 10 to 15 days. For the VVER-1000, I chose a 30-day refueling outage period. The outage time does not affect the simulation results, but only short-lived isotopes are affected.

Tab. 4.9 Irradiation time [58]

Reactor	Burnup (GWd/t_U)	Refueling cycle (mo)	Refueling outages (d)
VOYGR	45	18	10
BWRX-300	49.6	18	15
Rolls-Royce SMR	60	18	15
NUWARD	45	24	10
SMART	54	30	10
SMR-160	45	24	10
AP300	62	24	10
VVER-1000	52.8	18	30

I determined the irradiation time as a fraction of burnup and specific power. This time significantly affects the number of FPs and ANs, a significant challenge in SNF disposal. To better understand, I also calculated how many refueling cycles a single FA can go through. According to my calculations, the BWRX-300 reactor

offers the longest irradiation time of approximately 2,474 days, allowing the fuel to reach approximately 4.52 refueling cycles. This time is caused by the BWR-300 running at the second lowest specific power of 20.05 MW/t_U and a burnup of 49.6 GWd/t_U. In comparing the VVER-1000 to the BWRX-300, the BWR-300 is designed with almost twice the irradiation time. Conversely, the VVER-1000, according to my calculations, is operated with the shortest irradiation time because of its highest specific power among all the analyzed reactors, which is 38.19 MW/t_U, making its operation the most efficient in terms of energy gain per ton of uranium.

In order to better analyze the results, I also created additional models that all operate the same burnup value, which is 45 GWd/t_U. The VOYGR, NUWARD, and SMR-160 reactors are already designed with this value, so this change is not reflected in the results of these reactors. The change in the burnup value impacted the irradiation times for the BWRX-300, Rolls-Royce SMR, SMART, AP300, and VVER-1000 reactors. However, the Rolls-Royce SMR and AP300 reactors registered the most considerable difference in irradiation time, with a difference of about 470 days for both. All values I described and calculated are in Table 4.10.

Tab. 4.10 Calculated length of fuel irradiation

	Projected burnup		Burnup 45 GWd/t_U	
Reactor	Irradiation time (d)	No. of refueling cycles (-)	Irradiation time (d)	No. of refueling cycles (-)
VOYGR	1,697.47	3.1	1,697.47	3.1
BWRX-300	2,473.82	4.52	2,244.39	4.1
Rolls-Royce SMR	1,907.18	3.48	1,430.39	2.61
NUWARD	1,614.06	2.21	1,614.06	2.21
SMART	2,148.83	2.35	1,790.69	1.96
SMR-160	2,272.73	3.11	2,272.73	3.11
AP300	1,704.23	2.33	1,236.94	1.69
VVER-1000	1,382.56	2.53	1,178.32	2.15

5 Results

In this chapter, all the results of my thesis are presented. Firstly, I started to carry out simulations with a projected burnup. Then, I allowed the fuel to reach the same burnup level to analyze the effect of this burnup on the parameters of the SNF. In this thesis, I mainly focused on the period of 5 years after removal from the reactor, when the SNF is transferred to medium-term storage. I also focused on the 65 years when the fuel is moved from medium-term to long-term deep geological repository.

5.1 Projected burnup

5.1.1 Decay heat

The decay heat curves for all reactors exhibit similar characteristic curves but differ significantly in values. The initial value in Figure 5.1 corresponds to one-hundredth of a year. I have observed up to twice the difference in initial values between the VVER-1000 and SMR-160 reactors. The initial decay heat value equals 144,100 W/t_U for the VVER-1000 reactor and 76,450 W/t_U for the SMR-160. The magnitude of the initial decay heat value is strongly dependent on the specific power of the reactor, which affects the amount of short-lived isotopes that contribute most to the heat generation at this time.

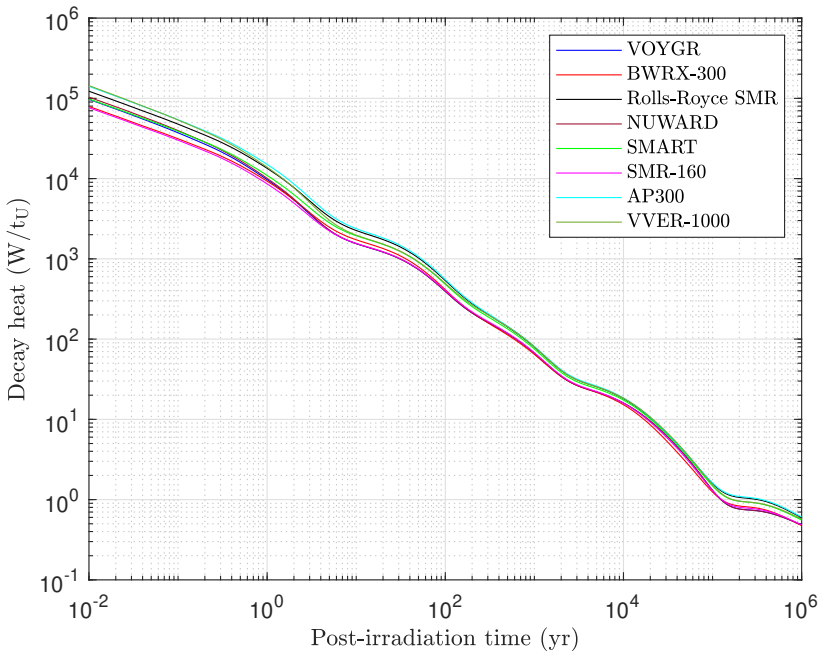

Fig. 5.1 Decay heat for projected burnup

During the first five years, the SNF is stored AR in a storage pool. After five years, the fuel is handled, removed from the storage pool, and placed in a medium-term storage facility; therefore, SNF management

is essential. After five years, the AP300 fuel reaches its highest absolute decay heat value of 3,593 W/t_U, corresponding to 39 times lower than its value after four days of removal from the reactor. At this time, the absolute magnitude of the decay heat is most influenced by the fuel burnup because it determines the irradiation time and, thus, the generation time of FPs and ANs. At this time, the SNF might be ranked by its burnup and, in the case of reactors with the same burnup, by specific power (see Tables 4.7 and 4.9).

Another critical milestone is year 65, where the fuel might be transferred from a medium-term storage facility to a deep geological long-term storage facility. According to my calculations, the lowest decay heat value should belong to SMR VOYGR with a value of 593.5 W/t_U. In contrast, the highest value is achieved by SMR AP300 with a value of 849.7 W/t_U, corresponding to 166 times lower than its value after four days of removal from the reactor. In this period, actinides are the dominant sources of decay heat, according to my simulations.

To compare the decay heat results between the SNF from VVER-1000 and SMRs, I normalized all data at a given time to the VVER-1000 value at that time to simplify the evaluation. Based on my calculations, I determined that SNF from AP300 and Rolls-Royce SMR reactors should represent a more significant burden on the back-end of the fuel cycle. Analysis of Figure 5.2 reveals that the decay heat values for the AP300 significantly surpass those of the VVER-1000 reactor for most of the time examined. Similarly, the Rolls-Royce SMR reactor achieves higher values than the VVER-1000 but lower values than the AP300. The rest of the SMRs studied did not exceed the VVER-1000 values over the entire span and, therefore, did not represent a significant burden.

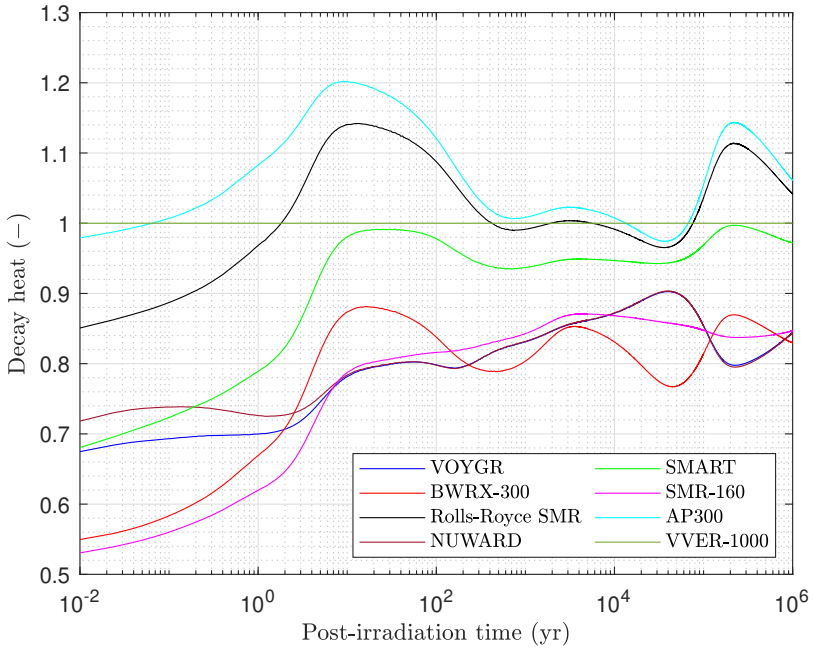

Fig. 5.2 Decay heat normalized by VVER-1000 for projected burnup

In Table 5.1, I have entered the normalized values for the 5th and 65th year. As I mentioned earlier, only the SNF from the AP300 and Rolls-Royce SMR exceed the VVER-1000 values. Specifically, AP300 by 18 % in the 5th year and 15 % in the 65th year. The AP300 1.202 reaches its maximum decay heat value around year 10. Also, I would like to mention that SMART is significantly closer to VVER-1000 values since the 10th year.

Tab. 5.1 Decay heat normalized by VVER-1000 for projected burnup

	Decay heat (-)		
	5th yr	65th yr	Maximum value
VOYGR	0.7458	0.8022	0.9029 (40,200 yr)
BWRX-300	0.8180	0.8569	0.8813 (16 yr)
Rolls-Royce SMR	1.1039	1.1087	1.1421 (13 yr)
NUWARD	0.7534	0.8024	0.9035 (40,700 yr)
SMART	0.9271	0.9866	0.9971 (227,000 yr)
SMR-160	0.7339	0.8135	0.8712 (4,480 yr)
AP300	1.1846	1.1486	1.202 (9.801 yr)

5.1.2 Radiotoxicity

The SCALE/TRITON calculation code uses the unit of $m^3_{H_2O}/t_U$ to evaluate radiotoxicity results. This volumetric radiotoxicity expresses the amount of water required to dilute the ingestion of potentially radioactive material below the threshold for general public safety.

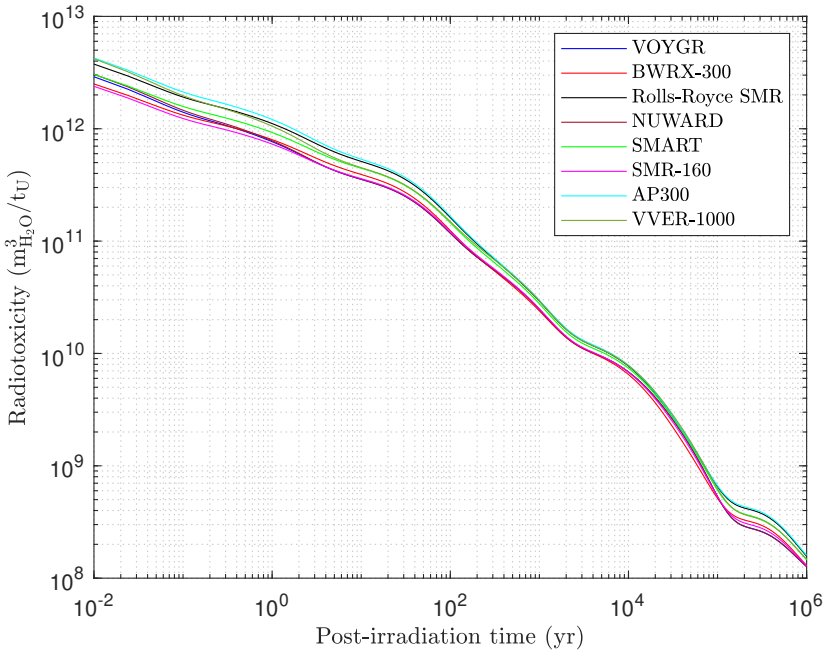

Fig. 5.3 Radiotoxicity for projected burnup

The dominant source of radiotoxicity initially appears to be SNF from AP300 with a rate of 4.26×10^{12}

$m^3_{H_2O}/t_U$. Compared to the least radiotoxic fuel from SMR-160, fuel from AP300 is almost twice as radiotoxic.

Based on the output data, I determined that the average radiotoxicity decreases by a factor of 6.34 after five years of cooling in the storage pool. Absolute radiotoxicity values can be ranked by burnup, except for reactors with a burnup of 45 GWd/t_U. Among these reactors, SMR-160 shows the highest value, followed by NUWARD, and VOYGR is the last. This ranking does not agree with the hypothesis that reactors with identical burnups can be ordered by specific power after a particular time, as in the case of decay heat. The different compositions of FPs and ANs may cause this fact.

After 60 years of storage in the medium-term storage facility, the radiotoxicity of SNF has been significantly reduced. The average reduction in radiotoxicity is approximately 17 times compared to the initial value and 2.69 times compared to the previous study period. According to my calculations, the radiotoxicity value of the AP300 fuel after that period is 2.319x10^{11} $m^3_{H_2O}/t_U$, which is 14 % higher than that of the fuel from the VVER-1000 reactor.

The radiotoxicity values, normalized by the VVER-1000 fuel parameters, are plotted in Figure 5.4. In evaluating the radiotoxicity, I concluded that the SNF from AP300 and Rolls-Royce SMR represents a higher burden than the fuel originating from VVER-1000, as is the case for decay heat. Although SMART fuel was irradiated at a lower specific power and reached a slightly higher burnup than VVER-1000 fuel, it might still represent a comparable burden to VVER-1000 fuel regarding decay heat and radiotoxicity.

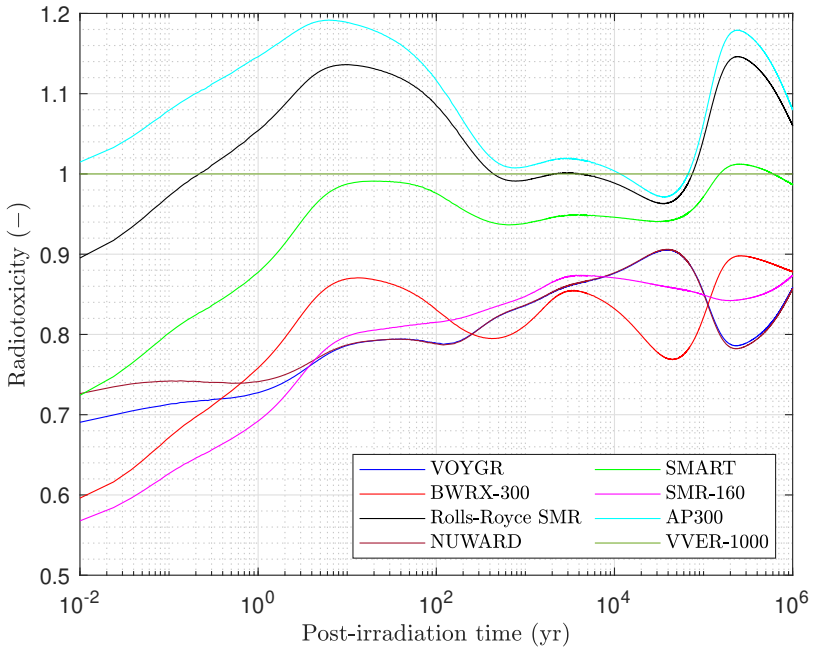

Fig. 5.4 Radiotoxicity normalized by VVER-1000 for projected burnup

The results presented in Table 5.2 demonstrate the previously mentioned facts, namely that the higher burnup fuels exceed the VVER-1000 values. The SNF from AP300 reaches its maximum at year 6. This value

is approximately 19 % higher than the VVER-1000 value and reaches it in year 5. Another threat regarding the back-end of the fuel cycle is the SNF originating from the Roll-Royce SMR, which exceeds the VVER-1000 value by 13 % in year 5 and by 10 % in year 60. The table quantifies that the SMART fuel is nearly equal to the VVER-1000 fuel, as mentioned earlier.

Tab. 5.2 Radiotoxicity normalized by VVER-1000 for projected burnup

	Radiotoxicity (-)		
	5th yr	65th yr	Maximum value
VOYGR	0.7698	0.7923	0.9056 (38,900 yr)
BWRX-300	0.8515	0.8463	0.8981 (266,000 yr)
Rolls-Royce SMR	1.1282	1.1046	1.1463 (243,000 yr)
NUWARD	0.7725	0.7918	0.9062 (39,000 yr)
SMART	0.9683	0.9844	1.0124 (256,000 yr)
SMR-160	0.7756	0.8127	0.8739 (4,220 yr)
AP300	1.1905	1.1411	1.1917 (6.1 yr)

5.2 Burnup 45 GWd/tU

After realizing fuel depletion models with projected burnup, I developed models with a burnup level of 45 GWd/t_U. I achieved this by adapting the existing fuel models and recalculating the irradiation times (see Table 4.10) to achieve the desired burnup.

In this Chapter, I have presented only the normalized decay heat and radiotoxicity curves because the non-normalized curves are almost identical to those presented in the previous Chapter 5.1. Moreover, in the present format, they are not as readable and thus not as informative as the normalized curves.

5.2.1 Decay heat

Based on the output data, I can say that the residual heat of decay is lower for SNF from SMRs than for VVER-1000 reactors for almost the whole period under study. It supports the hypothesis that higher levels of fuel burnup have an inherent effect on the back-end of the nuclear fuel cycle, thereby increasing the cost of storing and disposing of such fuel.

Two interesting phenomena can be observed in Figure 5.5, namely that the AP300 SNF exceeded the residual heat value for the VVER-1000 by less than 3 % after about one month and the SMR-160 fuel by about 1 %. For example, these interesting phenomena may be explained by the higher enrichment level than VVER-1000 and the different production of FPs and ANs during irradiation in the reactor. It is not a significant overshoot but should be considered in the design of the storage facility. These exceedances do not represent a significant risk but should be considered when evaluating the back-end of the fuel cycle. These short-term anomalies do not appear elsewhere or reach significantly higher values.

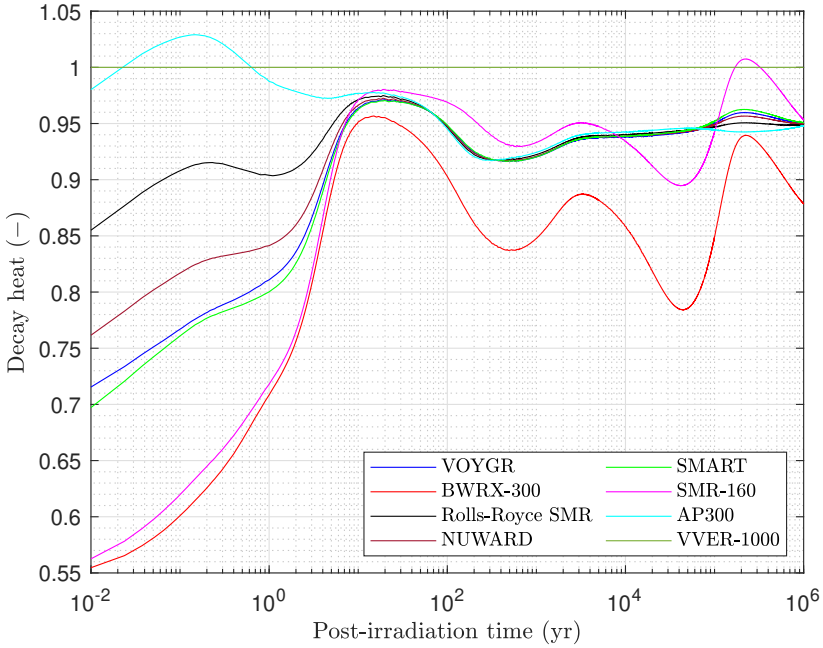

Fig. 5.5 Decay heat normalized by VVER-1000 for burnup 45 GWd/t_U

Table 5.3 numerically demonstrates that at times I examined, the values of SNF from potential SMRs for the CZE at a burnup of 45 GWd/t_U are lower than for the VVER-1000.

Tab. 5.3 Decay heat normalized by VVER-1000 for burnup 45 GWd/t_U

	Decay heat (-)		
	5th yr	65th yr	Maximum value
VOYGR	0.9188	0.9526	0.9712 (20 yr)
BWRX-300	0.8895	0.9152	0.9567 (15 yr)
Rolls-Royce SMR	0.9476	0.9509	0.9749 (19 yr)
NUWARD	0.9281	0.9520	0.9725 (19 yr)
SMART	0.9147	0.9538	0.9704 (20 yr)
SMR-160	0.9041	0.9772	1.0075 (226,000 yr)
AP300	0.9724	0.9485	1.029 (0.1481 yr)

5.2.2 Radiotoxicity

From the radiotoxicity output data, I conclude that the hypothesis formulated in the previous Chapter 5.2.1 of higher fuel burnup rates might be true because the normalized output radiotoxicity values from the depletion models are lower over almost the entire studied span.

Also, in the case of radiotoxicity, there is a slight overshoot of the VVER-1000 spent fuel value at 245,000 years after irradiation by 4 % for the SMR-160 spent fuel. A significant approach to the VVER-1000 spent fuel value occurred again for the AP300 spent fuel. As I have already mentioned, both of these phenomena could

be due to the higher enrichment and the different production of FPs and ANs.

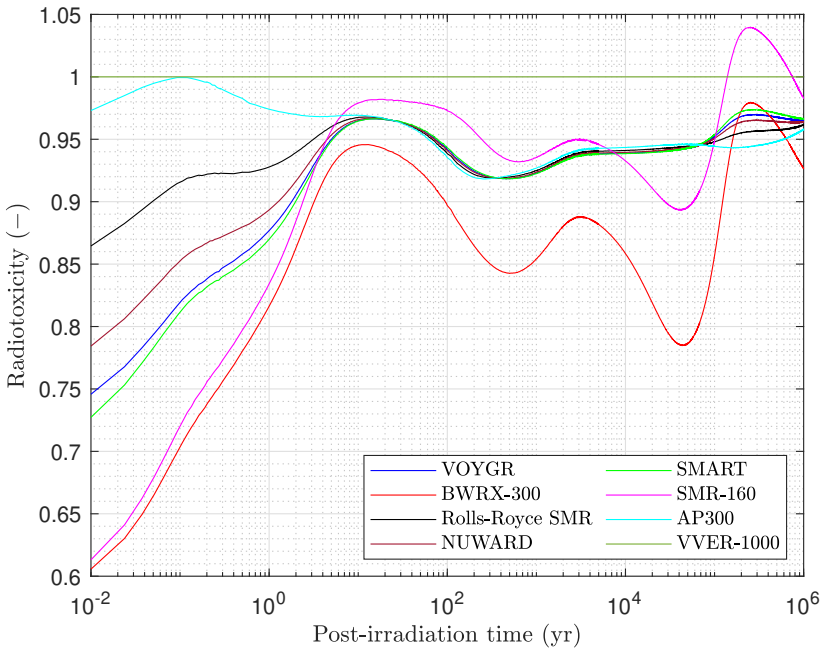

Fig. 5.6 Radiotoxicity normalized by VVER-1000 for burnup 45 GWd/t_U

The table below confirms numerically that during the years under study, there were no exceedances of the radiotoxicity values achieved for the VVER-1000 spent fuel, except for a slight exceedance for SMR-160. However, this exceedance is not so significant as it is a difference of units of percent from a value that is no longer as high as, for example, the initial radiotoxicity value.

Tab. 5.4 Radiotoxicity normalized by VVER-1000 for burnup 45 GWd/t_U

	Radiotoxicity (-)		
	5th yr	65th yr	Maximum value
VOYGR	0.9499	0.9526	0.9699 (294,000 yr)
BWRX-300	0.9282	0.9152	0.9795 (255,000 yr)
Rolls-Royce SMR	0.9600	0.9509	0.9678 (12 yr)
NUWARD	0.9532	0.9520	0.9669 (13 yr)
SMART	0.9485	0.9538	0.9739 (266,000 yr)
SMR-160	0.9570	0.9772	1.0397 (245,000 yr)
AP300	0.9685	0.9485	0.9994 (0.09286 yr)

5.3 Discussion

To check the validity of my results, I decided to compare the results of two contradicting studies on the same topic. The first of these studies is by Lindsay M. Krall, entitled ”Nuclear waste from small modular reactors.”

Another analysis, prepared for the US Department of Energy, is by T.K. Kim and is entitled "Nuclear Waste Attributes of SMRs Scheduled for Near-Term Deployment." [82, 83]

5.3.1 Nuclear waste from small modular reactors

The first study focused on three SMR designs, of which only one structure was relevant to the analysis in my thesis - namely, the VOYGR reactor. Differences in the modeling approach and variance in input parameters were also identified. While I chose a thermal capacity of 250 MW_{th}, fuel enrichment of 4.95 wt%, and burnup of 45 GWd/t_U, the first study opted for a thermal capacity of 160 MW_{th}, fuel enrichment of 5 wt%, and burnup of only 34 GWd/t_U. The study highlighted that the gaps were filled through explicit assumptions, references to similarly analyzed designs, or by using known design parameters.

The differences in the VOYGR output data are likely due to different input values, as my analysis considers significantly higher specific power and fuel burnup levels. The author of the first study chose a PWR reactor with thermal power of 3,400 MW_{th} and a burnup of 55 GWd/t_U as a comparative reference, while I chose a VVER-1000 reactor with thermal power of 3,120 MW_{th} and a burnup of 52.8 GWd/t_U.

A different approach to the modeling may have been that I modeled the FAs on an infinite grid, meaning that the assembly is surrounded by other identical assemblies and the neutron flux is the same throughout. In contrast, the first study's author may have modeled the entire core, including the geometry and reflector, which varied the neutron spectrum within the system. The author probably used a probabilistic calculation, while I took a deterministic approach.

However, the main idea of this study is to escape neutrons from the active zone. According to Lindsay M. Krall [82]*"For thermal-spectrum reactors, the neutrons undergo elastic scattering with the water or graphite moderator, leading to neutron diffusion lengths that are short relative to the core dimensions. Here, leakage grows quadratically with decreasing core radius and reactor size... Small increases in neutron leakage have a significant effect on core criticality and power output and will lead to reduced SNF burnup unless compensated for by design changes to the reactor and/or fuel."* It means that if the burnup value of the fuel is reduced, there can be a significant increase in the SNF production, according to Equation 5.1.

As I mentioned, I took a different modeling approach, so I decided to try at least to compare the results of the simulations with the projected burnup and with a burnup of 45 GWd/t_U, taking into account the mass of SNF generated when generating one gigawatt of thermal per year.

$$M_{SNFth} = \frac{365}{B} \tag{5.1}$$

where

- M_{SNFth} = SNF fuel mass (t/GW_{th}-yr),
- B = burnup (GWd/t_U).

Tab. 5.5 Mass of SNF per gigawatt thermal-year

	SNF fuel mass (t/GW_{th}-yr)
VOYGR	8.1111
BWRX-300	7.3589
Rolls-Royce SMR	6.0833
NUWARD	8.1111
SMART	6.7593
SMR-160	8.1111
AP300	5.8871
VVER-1000	6.9129

As previously reported, the VOYGR, NUWARD, and SMR-160 reactors have already been modeled to achieve a 45 GWd/t_U burnup. Therefore, these reactors will not experience a change in burnup, unlike the other SMR designs. The values in both columns of Table 5.6 are normalized by values for the projected burnup of the VVER-1000. It means that the values in column 45 GWd/t_U are also normalized by values for the projected burnup of the VVER-1000 to demonstrate the consequence of reducing the fuel burnup compared to the projected burnup. When comparing the decay heat at projected and lower burnup, it is clear that there is a reduction of less than 2 % in the values. On the other hand, in the case of radiotoxicity, there is an increase in the range of 5 to 18 %.

Tab. 5.6 SNF parameters normalized by VVER-1000 per gigawatt thermal-year

	Projected burnup		45 GWd/t_U	
	Decay heat (-)	Radiotoxicity (-)	Decay heat (-)	Radiotoxicity (-)
	100th yr	10,000th yr	100th yr	10,000th yr
VOYGR	0.9368	1.0281	0.9368	1.0281
BWRX-300	0.8918	0.8857	0.8923	0.9400
Rolls-Royce SMR	0.9571	0.8700	0.9353	1.0305
NUWARD	0.9363	1.0288	0.9363	1.0288
SMART	0.9562	0.9250	0.9375	1.0278
SMR-160	0.9575	1.0213	0.9575	1.0213
AP300	0.9546	0.8548	0.9339	1.0332
VVER-1000	1.0000	1.0000	1.0000	1.0000

The author presents radiotoxicity results 10,000 years after irradiation and claims that the radiotoxicity of SNF from a VOYGR reactor (with a burnup of 34 GWd/t_U) is 50 % higher than that of PWRs (with a burnup of 50 GWd/t_U). The decay heat results 100 years after irradiation indicate that SNF from a PWR produces 30 % more heat than fuel from a VOYGR reactor. Compared with our own results, it is impossible to determine conclusively whether our values would be similar or identical. However, I observed similar trends for reactors with higher projected burnup where radiotoxicity began to increase markedly, and decay heat decreased compared to lower burnup.

Based on the available results, it is possible to confirm the hypothesis of the first study discussed, namely that an increase in neutron leakage from the core could lead to a deterioration of the SNF management process. At the same time, further studies will be needed to investigate this issue in more detail.

5.3.2 Nuclear Waste Attributes of SMRs Scheduled for Near-Term Deployment

In the second study, three SMR designs were re-examined, including VOYGR, as in the first study. While examining this study, I also identified differences in the input values that indicated a 4.5 GWd/t_U higher fuel burnup than I predicted. However, when comparing the mass of uranium in the FA, I found a deviation of 0.18 kg between my calculated value and the value obtained in this study. This deviation could confirm the correctness of my assumed active FA length.

As in the first study, a reference PWR with a capacity of 3,500 MW_{th}/1,175 MW_e and a fuel enrichment of 4.5 wt% was considered in this second analysis. The enrichment value differs by only 0.05 wt% from my value. The burnup level for the reference PWR was assumed to be 50 GWd/t_U, which again does not match my model.

This study utilized metrics from the US Department of Energy's existing "Nuclear Fuel Cycle Evaluation and Screening" analysis from 2014. This analysis developed technology-neutral metrics for a wide range of potential fuel cycles. These metrics were enhanced with derived decay heat and radiotoxicity values calculated using ORIGEN 2.2 for one ton of SNF. To benchmark the results, the author uses a normalized measure of SNF decay heat and radiotoxicity relative to the proportion of electricity generated by a given reactor per year. All results of this study are presented in Table 5.7.

Tab. 5.7 Comparison of nuclear waste metrics [83]

	SNF decay heat (kW/GW_e-yr)		SNF radiotoxicity (MSv/GW_e-yr)	
	10th yr	100th yr	10,000th yr	100,000th yr
VOYGR	42.2 (1.04)	10.3 (1.05)	127 (1.06)	9.12 (1.06)
Ref. PWR	40.6 (1.00)	9.76 (1.00)	121 (1.00)	8.5 (1.00)

To convert my results to the same quantities, I had to calculate the mass of SNF produced in generating a gigawatt of electricity per year, according to Equation 5.2. This equation shows that the mass of SNF produced per year decreases with increasing fuel burnup and reactor thermal efficiency.

$$M_{SNFe} = \frac{365}{B \times \eta} \tag{5.2}$$

where

- M_{SNFe} = SNF fuel mass (t/GW_e-yr),
- B = burnup (GWd/t_U),
- η = thermal efficiency (%).

It is remarkable that reactors with higher levels of burnup, such as the AP300 and Rolls-Royce SMR, as discussed in Chapter 5.1, appeared to be the most burdensome in terms of the back-end of the fuel cycle. Conversely, after accounting for the generated mass of SNF, these reactors appear to be the most favorable for storage and disposal compared to the other SMRs with lower burnup. The BWRX-300 reactor results are also very low compared to the other SMRs.

Tab. 5.8 Mass of SNF per gigawatt electric-year

	SNF fuel mass (t/GW_e-yr)
VOYGR	26.3348
BWRX-300	21.3407
Rolls-Royce SMR	17.5770
NUWARD	25.7647
SMART	23.0573
SMR-160	26.6146
AP300	17.6613
VVER-1000	21.5682

When comparing the decay heat results from my study with the results of the second study, it can be seen that the values are very similar in magnitude. During the period studied, it was observed that the SMART reactor always exceeded the reference value by 5 %. The SMR-160 reactor achieved values almost comparable to those of the VVER-1000 reactor. Thus, the potential SMRs investigated represent a smaller or similar load to the full-scale VVER-1000 reactor in terms of decay heat.

The radiotoxicity results show a slightly more unfavorable trend than the decay heat. Reactors with higher burnup, such as BWRX-300, Rolls-Royce SMR, and AP300 SMRs, achieved the best results, as in the case of radiotoxicity. The other SMRs with lower burnup show a higher radiotoxic burden on the back-end of the fuel cycle in units of percent higher than the VVER-1000 reactor.

Tab. 5.9 SNF parameters for projected burnup per gigawatt electric-year

	SNF decay heat (kW/GW_e-yr)		SNF radiotoxicity ($Gm^3_{H_2O}/t_U$)	
	10th yr	100th yr	10,000th yr	100,000th yr
VOYGR	40.71 (0.95)	10.37 (0.97)	180.08 (1.07)	14.02 (1.04)
BWRX-300	36.88 (0.86)	8.81 (0.83)	138.57 (0.82)	10.95 (0.81)
Rolls-Royce SMR	39.65 (0.93)	9.42 (0.89)	135.61 (0.81)	11.46 (0.85)
NUWARD	39.94 (0.94)	10.14 (0.95)	176.31 (1.05)	13.69 (1.02)
SMART	44.73 (1.05)	11.12 (1.05)	170.23 (1.01)	13.99 (1.04)
SMR-160	41.47 (0.97)	10.71 (1.01)	180.79 (1.07)	14.12 (1.05)
AP300	41.98 (0.98)	9.76 (0.92)	138.34 (0.82)	11.73 (0.87)
Ref. VVER-1000	42.66 (1.00)	10.63 (1.00)	168.32 (1.00)	13.47 (1.00)

Overall, our results are very similar despite the differences in the input data. The result of the second study is that SMRs will not pose a major challenge regarding SNF management. I do not fully agree with this statement, as Table 5.9 shows lower burnup reactors have increased storage and disposal requirements compared to the VVER-1000 after accounting for the mass of SNF. On the other hand, reactors with higher burnup levels have reduced their parameters compared to the VVER-1000.

Conclusions

The aim of this work was to analyze in detail the back-end of the fuel cycle of small modular reactors, especially those that show potential for implementation in the Czech Republic. Subsequently, the results of the spent nuclear fuel from small modular reactors were compared with data from the Temelín nuclear power plant, which has extensive operational experience in the Czech Republic and covers a wide range of technologies. The main purpose of this study was to determine whether these reactors would represent a higher burden per unit of energy produced than conventional reactors.

The results of my analysis show that specific small modular reactors with higher burnup levels may pose an increased risk in the context of spent nuclear fuel management. These reactors show an increase in decay heat and radiotoxicity up to twenty percent per tonne of initial heavy metal. I have also found that none of the reactors examined at comparable burnup levels show significant increases in residual decay heat and radiotoxicity requirements.

I came to some remarkable findings when comparing my data with other scientific studies, considering the mass of spent nuclear fuel produced per gigawatt of energy per year. As the author claims, I have confirmed numerically that if burnup is reduced for these reactors, the radiotoxicity rate will increase, and the decay heat will decrease. However, I consider very crucial the finding that reactors with higher burnup levels will place less burden on the back-end of the fuel cycle in terms of both decay heat and radiotoxicity. Conversely, reactors with lower levels will present comparable or even greater burdens.

As mentioned in the introduction, there are countless studies on small modular reactor's technical and economic aspects. However, there are noticeably fewer studies analyzing the back-end of the fuel cycle. This work may one day play an important role in the selection of the final design of a small modular reactor in the Czech Republic, as the study provides important information on the impact of these reactors on the back-end of the fuel cycle and informs the decision-making processes in the nuclear power industry in this country.

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