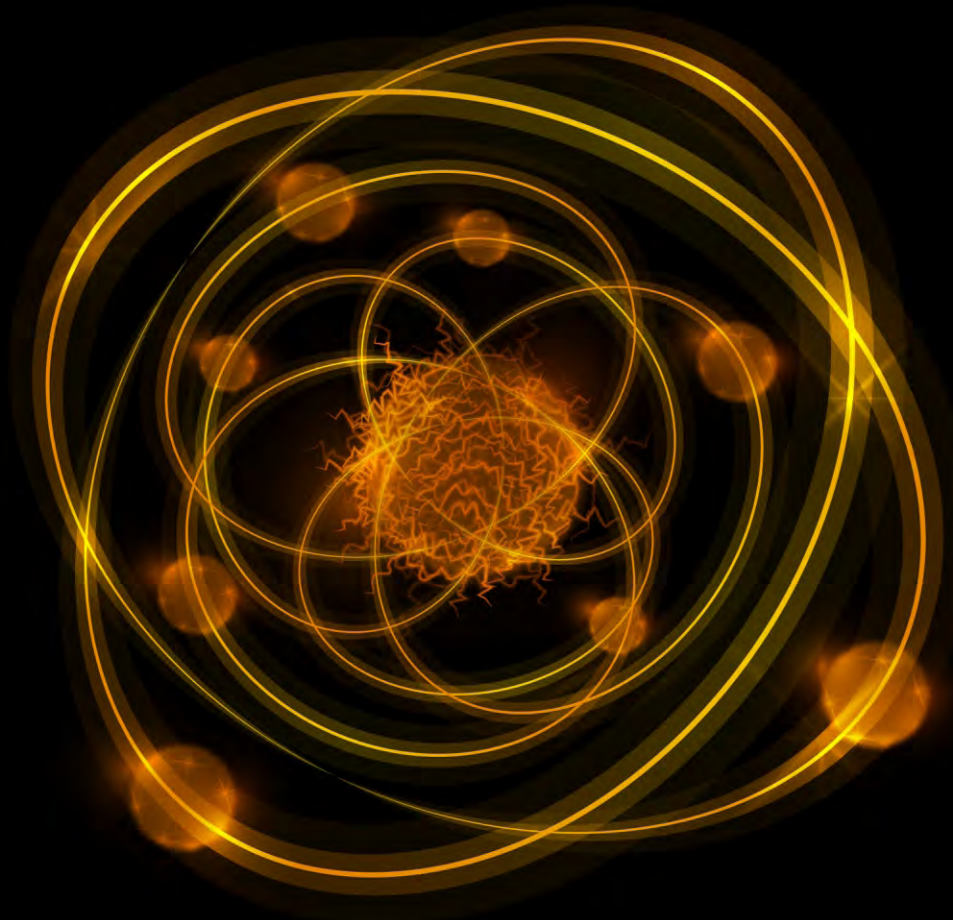


ISSN: 2161-6795

Volume 10, Number 3, July 2020



# World Journal of Nuclear Science and Technology



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ISSN 2161-6795 (Print) ISSN 2161-6809 (Online)

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# World Journal of Nuclear Science and Technology (WJNST)

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The *World Journal of Nuclear Science and Technology* (Online at Scientific Research Publishing, <https://www.scirp.org/>) is published quarterly by Scientific Research Publishing, Inc., USA.

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Print: \$79 per issue.

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# Towards Safer and More Sustainable Ways for Exploiting Nuclear Power

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**How to cite this paper:** Kröger, W., Sornette, D. and Ayoub, A. (2020) Towards Safer and More Sustainable Ways for Exploiting Nuclear Power. *World Journal of Nuclear Science and Technology*, 10, 91-115. <https://doi.org/10.4236/wjnst.2020.103010>

**Received:** May 15, 2020

**Accepted:** June 14, 2020

**Published:** June 17, 2020

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## Abstract

Future electricity systems are challenged by deep decarbonization and concurrently increasing demand and there are growing concerns that renewables cannot shoulder this alone. Starting from the proven principle of diversity, we argue for keeping the nuclear option open or even for expanding its use. However, the perspectives are dim for the current technology as safety concerns and social aversion remain as fundamental problems. While looking for future revolutionary safe and more sustainable nuclear concepts we first review the main characteristics of civil nuclear energy, as well as its safety records and technical progress. We then list the key requirements for innovative nuclear systems designs which are less dependent on active safety systems and human performance as well as social stability. This allows us to provide a concept by concept comparison and assessment of existing and novel technologies and designs including different coolants and neutron spectra. The results indicate a high potential for far-reaching improvements compared to most advanced LWRs, although none of the candidate concepts meets all requirements convincingly, yet, helium cooled, small modular reactors (HTR-PM) come closest. We end by stressing the need for future research and development, and keeping human capital and know-how in nuclear energy; we call for an urgent increase in government and international RD&D funding by the order of a few hundreds of billions of USD per year, which will likely lead to breakthroughs that will restart productivity growth in severely affected stagnating modern economies.

## Keywords

Low-Carbon Energy Systems, Sustainability, Nuclear Safety, Safety Requirements, Advanced Reactor Concepts, SMR, RD&D Needs

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## 1. Introduction

Availability of energy is key for the well-being of our societies and economies. The intensive use of electricity, in particular, has enabled the 3<sup>rd</sup> and 4<sup>th</sup> industrial revolutions; the latter is ongoing with the progressive fusion of the natural and digital world. By the end of 2018 [1] the worldwide primary energy consumption totaled to 166378.8 TWh, has doubled within the last 50 years and increased by 2.9% compared to 2017. Almost 40% are converted into electricity which corresponds to a share 16% of the final energy demand. The annual energy consumption per capita varies strongly by country and region; 20% of the global population consume 80% of the global energy, and roughly 1.3 billion people have no access to electricity. Scenario analyses predict a massive growth of primary energy, mainly driven by developing countries, to cope with the expected increase of world population and to expand energy access and economic opportunities to billions of people. The electricity sector is expected to grow disproportionately, e.g. by a factor of 2.5 till 2050 [2], in particular to penetrate domains other than the traditional ones, *i.e.* e-mobility, digitalization, buildings, industries. Moreover, electricity will be needed in concentrated forms to power future mega-cities—another mega-trend [3].

This challenging trajectory is simultaneously confronted to the consensus that anthropogenic CO<sub>2</sub> emissions must be drastically reduced, requiring the decarbonization of the energy systems which are currently relying on fossil fuels with about 84.7% [1]. The electricity production, currently based at about 38% on coal, 23.2% on gas and 2.8% on oil, contributes almost 30% to the global carbon emissions of 33,891 million tons (which increased by 2% compared to 2017). Therefore, the electricity sector needs to play a central role in any transition to a deeply de-carbonized energy system; a new mix and roughly doubled share of low-carbon electricity generation assets by 2050 is required to meet the “2°C target” for global warming, while other sustainability indicators like use of land and other resources, affordability, waste production, factorial and perceived risks, and so on, must be kept in mind.

Most scenario-based projections (see also [4], Chapter 1.3.1) and strategies of countries focus on expanded use of renewables. Besides hydroelectricity with a share of roughly 15%, wind and solar contributed 9.3% of the global power production (18.7% in Europe) in 2018, with a 14.5% annual growth, slightly below its historical average; China contributed 45% of the global growth in renewable power generation. However, there are growing concerns about whether 1) renewable generation will grow sufficiently fast, with the need for it to grow more than twice as quickly than it actually did [1], 2) variable energy sources alone, depending on weather, daytime and season, will be adequate and sufficiently secure and 3) the required infrastructure including storage, upgraded grids and flexible backups can be provided. Diversification seems to be a prudent principle.

Nuclear power is regarded as a promising asset in a de-carbonized, more sustainable energy system. Currently, nuclear power contributes 10.15% to global elec-

tricity production—varying by countries/regions, 23% in Europe. The share increased from about 2% in 1971 to 18% in 1998, decreased afterwards but grew in 2018 by 2.4%, the fastest growth since 2010, to which China contributed almost three quarters [1]. However, the prospects of nuclear power are dim in many parts of the world, with costs [5], lack of public acceptance and some unresolved issues including disposal of high-radioactive waste as key problems [4]. Prominent studies regarding future shares of electricity production by nuclear energy are ambiguous and vary significantly—from zero ([6], by input requirement) to a grow by 28% till 2040 [2]—the latter corresponds to additional 510 GWe, and raises questions of commercial deployment readiness as well as industrial and regulatory capabilities.

In a nutshell, a transition to a deep de-carbonized and more sustainable electricity sector will require of a mix of generation assets, including nuclear energy while advanced technology options are necessary to overcome existing barriers against its extended, even expanded use.

## 2. Characteristics of Nuclear Power

The use of nuclear power has proven to be a mature technology. In 2019 [7], there was a fleet of 450 reactor units with 398.9 GWe total net installed capacity in operation, distributed throughout 31 countries. Experience accumulated to roughly 17,000 reactor-years. The mean capacity factor of the operating units was 80% [8], which supplied 2 701.4 TWh of electricity or a share of 10.15% of worldwide electricity production [1]. Currently, there are 54 new units under construction with 57.4 GWe capacity in 20 countries, with China taking the lead with 11 units (11) [9]. In 2019, 6 new reactors were commissioned while 13 reactors were permanently shut down, including 5 units in Japan. New builds in the Western world are rare and, like the EPR in Finland and France, confronted with serious costs and construction time overruns (tripled) while projects in Asia tend to stay within basic conditions.

Currently, 80% of all operating nuclear reactors are light water reactors (LWR), which use low enriched uranium (3% - 5% U235). Uranium has incomparably high energy density compared to other energy carriers; the energy density of a mix of natural non-fissile U238 and Pu239 used in breeder reactors is approximately 80,620,000 MJ/kg, or 3.55 million times higher than black coal, meaning that, while undergoing full breeding and fission, one kg of uranium is the equivalent of burning 3,500 tons of black coal [4]. This high energy density also simplifies the storage and transport of nuclear fuel.

An essential aspect is the continuous availability of uranium and long-lasting confirmed reserves. According to [10], the total yearly consumption of uranium is approximately 63,000 tons. Considering that the current proven reserves in the low-cost range extraction (up to US\$ 130/kgU) are about 6.1 million tons and higher-cost range extraction (up to US\$ 260/kgU) total at 7.9 million tons, the world can produce enough uranium for the next 125 years. When consider-

ing the inferred and reasonably assured resources, the total reserves are estimated to be around 15.9 million tons, enough for 254 years of operation, all with the current rate of consumption. Additionally, assured reserves for uranium are higher than what is considered normal for most minerals, indicating that further exploration will most likely lead to new deposits, albeit at a higher cost. Furthermore, increased energy efficiency and higher burn-up, moving to advanced nuclear options including breeder reactors, using thorium (three to four times more abundant in the Earth’s crust), application of new uranium mining and extraction technologies, and so on, could theoretically place nuclear as a practically unlimited resource.

Currently, nuclear power is the second largest source of global low-carbon energy, behind hydro. Broadening the view to the “three pillars of sustainability”, *i.e.* the environmental, economic and social dimension, a detailed study was done in the EU-Project NEEDS [11], where 26 advanced electricity generation technologies were analyzed and compared using 36 technology-specific evaluation criteria and indicators. Nuclear options were ranked top, given equal weighting of the three dimensions. In particular, current nuclear technology has very low greenhouse gas emissions, comparable to the renewable energy sources (considering the whole life cycle). By 2050, with the deployment of generation IV reactors, the emissions are estimated to decrease further, potentially making nuclear power the cleanest form of energy, see Figure 1. On the other side, giving higher weight on the social dimension and focusing on radioactive wastes, land contamination due to hypothetical accidents, risk aversion and perception issues, terrorist threats and conflict potential, the ranking changes to the disadvantage of nuclear energy, see also [4].

Nuclear power is not without its drawbacks, however, both in the physical process and current technologies. The physical process of nuclear fission generates a surplus of neutrons, radioactive decay heat producing fission products

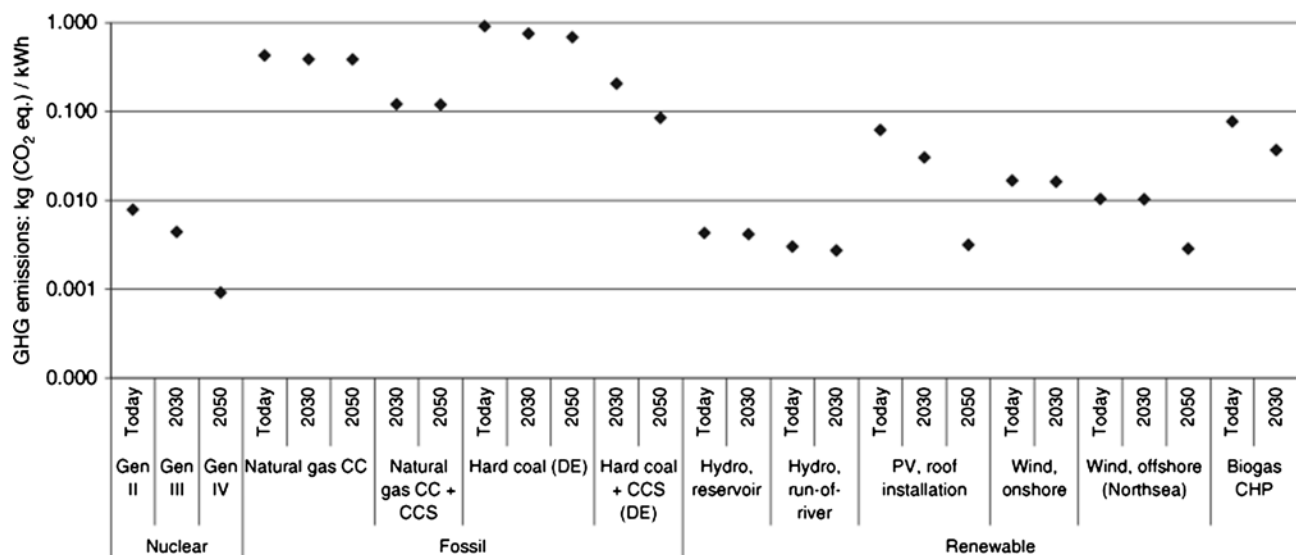


Figure 1. Greenhouse gas emissions of selected technologies at different time periods [12].

and high-level nuclear waste. This leads to major design challenges, implementation of safety functions and development of demanding strategies for operation and decommissioning of nuclear power plants as well as for management and storage of nuclear waste. Current technologies address these issues successfully. However, certain aspects are still problematic, such as meltable fuel cladding, reliance on active safety systems and early operator actions, vulnerable structural material, and little grace time in case safety systems fail. Large radioactive releases cannot be excluded despite their rarity, would a beyond-a-design-basis accident happen in today's large-scale nuclear power plants with high power density.

An assessment of various technologies with regard to operational risk and maximum potential fatalities in the event of a severe accident was also performed as part of the NEEDS project [11]. The results show fossil technologies with the highest operational risks, while nuclear is close to the PV and wind. However, nuclear has the highest number of potential fatalities in the event of a severe accident, in addition to the long-term contamination of the affected area, leading to a very high level of risk aversion by the general public and strong opposition, especially in the Western world. As a consequence, some western projects have been delayed or cancelled, while some countries have decided to completely phase out nuclear power due to the public pressure and reduced cost-effectiveness. Contrary to this, the public opposition is far less pronounced in Asia and nuclear power is largely accepted as a viable option.

Many developing countries would greatly benefit from the use of nuclear energy (especially in Africa). However, they face some fundamental problems on their path of adopting nuclear technology. The development of small modular reactors might provide a solution to some of the problems, considering the potential safety and cost-effectiveness of these reactors.

Nuclear energy has the potential to play a major role in a diversified, de-carbonized/more sustainable future energy mix. However, safety concerns, social acceptance and costs remain fundamental problems.

### 3. Safety Records and Technical Progress

Throughout its long operational history, the civil nuclear industry has stood as a unique, reliable, and clean source of energy. Nevertheless, it had witnessed numerous operational disruptions ranging from anomalies, incidents up to near-misses, and even major accidents, that have all been well recorded by regulators and international organizations. Herewith, we will focus on three core disruptive events [4], for a comprehensive list and analysis of safety-relevant events see [13].

- 1) The first was the Three Mile Island PWR unit 2 (TMI-2) partial meltdown in the United States on March 23, 1979, initiated by a loss of main feed water transient during full power operation. Emergency feed water system was unavailable due to a testing and maintenance error. To relieve the increasing reac-

tor pressure, the pilot-operated relief valves (PORVs) opened, yet failed to reclose, leading to large losses of primary coolant. Operators were not aware of what was going on due to inadequate valve instrumentation; they tripped the automatic safety injection systems, actuated on the low reactor water level, thinking they were inadvertently actuated. The core was already uncovered by the time the next-shift operators realized the situation and re-established the coolant injection. Nevertheless, they managed to minimize the offsite consequences and contained the fission product within the containment.

2) The second was the accident at Chernobyl RBMK unit 4 in the former Soviet Union on April 26, 1986. Operators disabled several safety systems (including several control rods withdrawal) in order to perform a safety test on the capacity of the turbine generator to supply emergency power during its rundown. Several operator errors in addition to inherent design flaws (such as positive reactivity coefficients and a positive scram effects due to control rod graphite tip) resulted in unstable power conditions and ultimately led to a power excursion followed by destruction/meltdown of the core, an uncontrollable graphite burning and large release of fuel and radioactive substances; the plant had no reinforced containment building. 28 workers/firemen died directly after the accident and several further deaths were recorded, 4000 people are estimated to have prematurely died from radiation in the neighboring countries.

3) The third accident happened on March 11, 2011 in Japan. A beyond design 9.0 - 9.2 seaquake hit the Fukushima Daiichi nuclear power plant, consisting of six BWR units, causing a reactor SCRAM and total loss of offsite power. While the structural damage to the plant was very limited, the following 14 - 14.5 m tsunami flooded the onsite emergency diesel generators causing an extended blackout at units 1 - 4. Three units experienced core meltdown with hydrogen buildups and explosions. Large amounts of radioactive substances were released and tens of thousands of people evacuated. Health effects were psychological rather than physical, and the huge number of casualties (more than 18,500 people) is due to the natural disaster caused by the quake and tsunami.

The accidents were not specific to a single nation or a particular reactor type but have stressed several common safety deficits and concerns. In addition to random equipment failures and occurrence of initiating events, the non-technical contributions (human, organizational, societal, safety culture) have been clearly present at multiple levels in all the serious nuclear events. At the plant level, operator and testing & maintenance errors have been evident, either by creating/aggravating the scenario, or by failing to mitigate the consequences. At the operating company level, there was a tendency to hide safety deficits and inherent design flaws, in addition to overlooking potential beyond design situations. At the regulatory and political level, there were institutional deficits and conflicts of interests. Moreover, the accidents have demonstrated the importance of an adequate safety culture, training and communication and knowledge transfer across all levels. Other practical lessons were realizing the importance of post

core-damage provisions in minimizing the accident consequences, namely, containment filtered venting systems and offsite emergency measures.

These major accidents, even though they did not result in many direct casualties, yet have had huge economic and financial consequences, including health costs and physical losses, as well as contamination, decommissioning, evacuations, and others, see ([4], Chapter 5.3) for more details. This created a public view that nuclear energy is dreadful, amplified by the association with nuclear bombs and wars, the fear of invisible radiations, and the problem of nuclear waste [14]. This “dread factor” is rooted deep down in the human mindset of overestimating the risk of “low probability-high consequence events”. Nevertheless, for a genuine risk appraisal, one has to have a correct entanglement of both probability of an accident and its consequences. Probabilistic Safety Analysis (PSA) is the standard framework used for analyzing risk in nuclear power plants at three sequential levels. It helps identifying accident scenarios, estimating their probabilities/frequencies, as well as their consequences. PSA level 1 studies and quantifies core damage frequency (per reactor-year, CDF). PSA level 2 describes the containment response and quantifies the radioactive release frequencies (large early release frequency, LERF). PSA level 3 characterizes the environmental and public consequences that are important for evacuation and emergency planning [15].

The most developed level is level 1, being concerned with modeling reliability of in-plant engineered safety systems and their demands during postulated initiating events, hence faced by less uncertainties and beyond-plant unknowns. CDF estimates for operating (Generation II) LWRs are between  $10^{-4}$  to  $10^{-5}$  per reactor-year and can be as low as  $10^{-6}$  for Gen III and some retro-fitted Gen II plants [4]. To see how these model results represent reality, a first estimate could be using core meltdown events in the nuclear power sector. Five core melt-down events have occurred during roughly 17,000 reactor-years of operation (counting Fukushima multi-unit meltdown as 3 occurrences). Therefore, an empirical estimate for core damage frequency of operating LWR will be around  $3 \times 10^{-4}$  per reactor-year, which is not so far from the PSA CDF results. However, this estimate is based on a very limited sample size due to the—fortunately—rare occurrence of core damage events, and does not account for safety improvements over time. A better estimate could be calculated while expanding the sample size and considering accident precursors that are quantified by a conditional core damage probability (CCDP), *i.e.* probability of core damage knowing that this precursor has happened, and this is calculated using the relevant part (chain) of the PSA models. To have an idea of the operational fleet safety performance, the USNRC Accident Sequence Precursor (ASP) program [16] have trended the so-called “integrated ASP index” for the US fleet over time, which serves as an empirical proxy of CDF. It is defined as the sum of precursors CCDPs identified in a given year, normalized by the total operating years for all US reactors in that year. **Figure 2** shows that a more recent and a more representative statistical CDF

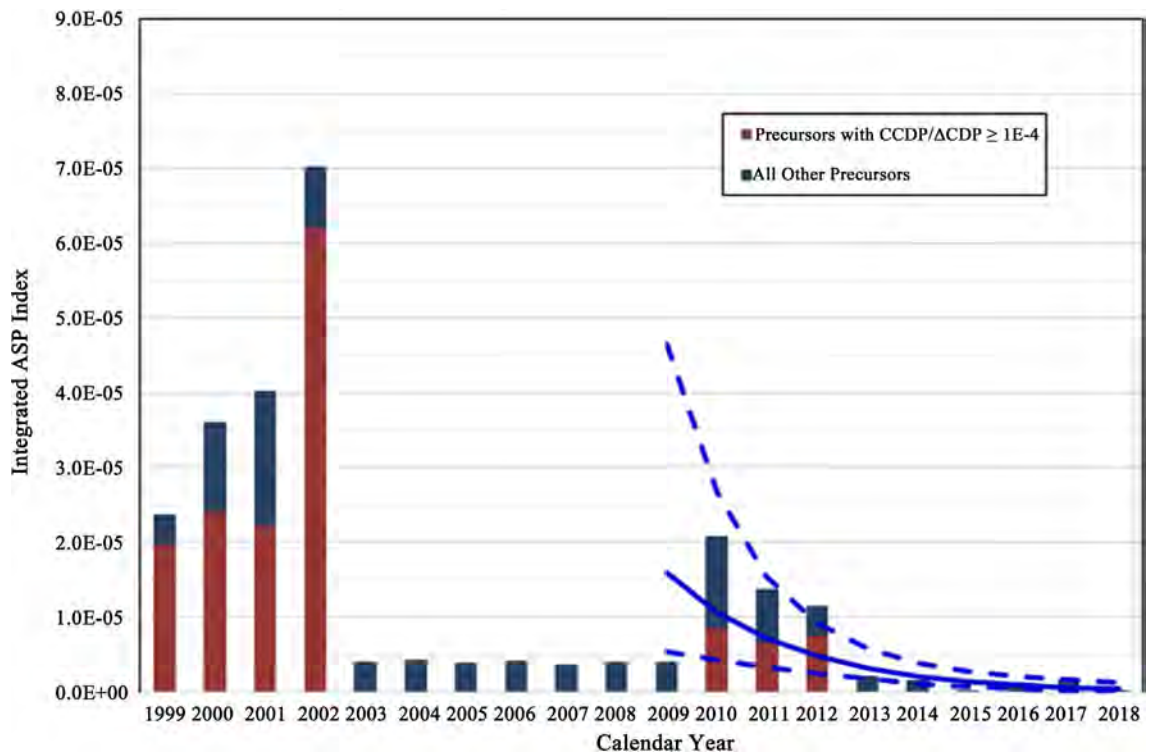


Figure 2. USNRC integrated ASP index trend [16].

estimate for the US fleet is in the order of magnitude of  $10^{-5}$  per reactor-year, which is similar to the PSA estimates. Moreover, it shows a decreasing trend representing plant retrofits and improved safety considerations. Some researchers [17] [18] are developing methods to generalize these calculations to worldwide operating fleets and compute representative CDF estimate, using generic PSA models applied to a sufficiently large sample of worldwide precursors and safety significant events.

The other issue of concern, when discussing the safe operation of nuclear power plants, is radioactive waste burden. To start with, LWRs follow one of three fuel cycle concepts, “once-through”, “partially closed” and “fully closed” [4]. All start with uranium mining, enrichment to 3% - 5%, and fabrication of uranium dioxide fuel pellets to be loaded in fuel rods and elements. The spent fuel (SF) is then unloaded and stored in a water pool for several months to cool down to sufficiently low levels. In the once-through cycle, SF is sent for extended storage, pending conditioning and emplacement in permanent disposal facilities such as deep geological repositories. On the other hand, SF can be reprocessed to extract fissile and other usable material such as uranium and plutonium before disposal (partially closed cycle). In the fully-closed fuel cycle, uranium, plutonium, and other minor actinides (long-lived radionuclides) are extracted and used as fuel in advanced fast reactors, hence significantly reducing the amount of long-lived radioactive waste to be disposed.

The different fuel cycles concepts have advantages and disadvantages, notably in terms of non-proliferation issues; the once-through fuel cycle is the most fa-

avorable as no separation of fissile material, Pu in particular, takes place. However, this means that large volumes of high-level radioactive wastes will have to be disposed. In contrast, closed fuel cycle concepts are more vulnerable in terms of proliferation. However, they allow for better exploitation of fuel reserves, and bring down nuclear waste volumes to manageable amounts with low radiotoxicity levels [4].

In all fuel cycle concepts, a safe and long-term disposal of radioactive wastes is required, and the deep underground storage—mainly for high and intermediate level waste—seems to be the most promising and favorable option by countries and the public [19]. However, due to the inherent uncertainties, strong opposition, and strict regulatory and safety requirements, the advancement in licensing and operation of repositories are still slow, and there is no operating deep geological repository around the world yet; nuclear waste is currently stored on site or in dedicated interim storage facilities. Nevertheless, enormous work and research are ongoing to address the underlying uncertainties over the large geological time scales and push forward the development of safe and passive deep geological repositories [20] [21] [22] [23]. Finland is in the lead, granting the license and starting the construction at Olkiluoto site in 2015 with the disposal process expected to start by 2024. Sweden has submitted its license application in 2011 and is waiting for a final approval by the authorities. Other countries like France, Switzerland, Canada, and US are also pushing forward in this direction, investigating appropriate sites and preparing their license applications [24].

Major accidents and public concerns have triggered significant improvements, either by retrofitting existing plants, or by evolutionary new reactor design concepts. Retrofits such as additional emergency power, mobile pumps, bunkered equipment for flooding protection, and others have been introduced. Furthermore, with the advanced Gen III reactors, improvements in the lifetime, fuel technology, thermal efficiency, and safety features are achieved. A prominent example is the European Pressurized Reactor EPR, having advanced active safety features and severe accident mitigation systems such as [25]:

- 1) increased reliability (four 100% redundant safety systems, CDF  $2 \times 10^{-6}$  per reactor-year),
- 2) fully digitalized instrumentation and control systems,
- 3) extended grace periods,
- 4) core catcher, advanced containment systems (passive hydrogen recombiners, dedicated containment heat removal system).

Moreover, the so-called revolutionary Gen III+ reactors such as the AP-1000 have pushed systems' reliability forward while incorporating the concept of passive safety. Some of these advanced reactor designs have already been in operating since a few years, and others are under construction with some delays. Further safety improvements and innovative design concepts are required—and some are on the way—to ensure that the nuclear civil industry avoids high-consequence accidents and extremely long stewardship times of long-lived fission products and

actinides.

#### 4. Key Requirements for Innovative, Less Vulnerable Nuclear Systems Design

We should not overburden the use of nuclear energy with an impossible “zero risk” expectation. However, there are major barriers to make its future, potentially expanded, use acceptable to the public. These barriers include the unequal treatment of (i) probabilities of the extraordinarily high potential consequences, should a severe accident happen, and (ii) the perceived cancer dread of even low doses of radiation, in particular. To overcome these barriers, we recommend a fundamental shift from reactor designs that depend on properly designed (active) safety systems, requiring AC power and reliable actuation mechanisms, towards designs that incorporate passive and inherent safety features. Furthermore, nuclear plants should be less sensitive to adequate protection against natural events and malicious man-made physical and cyber-based attacks and should warrant higher tolerability to human errors, lack of safety culture and socio-political instability within the operational environment. The following more specific requirements, aiming as far as possible at a deterministic exclusion of serious conditions and states, are put forward (see [4], p. 187-8), all where appropriate:

1) Control of nuclear reactivity and elimination of potential reactivity induced accidents by reactor core design or at least controllability by passive means; this can be achieved by:

- a) weak, negative reactivity coefficients (graceful reaction on increasing fuel temperature, power, void fraction, burn-up),
- b) small reactivity surplus at startup with fresh fuel,
- c) fail-safe design of shutdown absorber rods.

2) Assurance of heat removal to ultimate heat sink and retention of fission products, *i.e.* forgiveness against loss of active core cooling; this can be achieved by:

- a) low power density and power size (to avoid exceeding critical temperature limits),
- b) strategies to avoid high fission product inventory, e.g. by dispersed fuel,
- c) temperature resistant fuel cladding and structural material that will not melt or burn, if adverse conditions occur,
- d) sufficient heat storage capability and inherent/passive heat transfer mechanisms in case of loss of normal (forced) cooling/control of coolant inventory (depressurization)/total loss of power,
- e) passive decay heat removal systems.

3) Securing structural integrity to avoid geometric disorders (e.g. losing core cooling capability) or loss of confinement of radioactive inventory; this can be obtained by:

- a) low primary circuit pressure or leak/rupture proof components (reactor

pressure vessel),

b) radiation resistant and chemically and physically robust core structures,  
 c) underground siting for protection against extreme external impact, including conventional weapons' attack.

4) Use of chemically non-reactive, non-toxic materials and fluids or avoid direct contact of reacting substances; this can be achieved by intermediate cycles, if necessary.

5) Avoidance/incineration of long-lived radioisotopes (actinides) by fuel cycle designs allowing for reduced long-term stewardship (husbandry times); this can be achieved by:

a) a switch to thorium with drastically smaller generation of long-lived minor actinides,

b) waste burner core designs,

c) striving for enhanced closed fuel cycles or for long-term stable, high burn-up spent fuel as an open fuel cycle option.

6) Enhanced intrinsic proliferation resistance characteristics of the fuel, entire fuel cycle and related processes; based on the discourse on proliferation issues, the following principles, means and strategies can be assigned and should be applied:

a) avoid use of highly enriched uranium (HEU),

b) configure nuclear reactors to enable maximum burn-up of fuel and thereby decrease the amount of plutonium in spent fuel that could be used for weapons,

c) avoid high-grade plutonium generation, e.g. by employed blankets,

d) reprocess spent fuel only if there is a clear plan to minimize the time during which weapons-grade material, notably plutonium, is in separated form and to reuse it as soon as feasible, to avoid accumulating a stockpile,

e) strive for online reprocessing including fuel fabrication at the reactor location and avoid transportation of sensitive material,

f) implement protective measures throughout the entire fuel cycle.

## 5. Candidate Design Features and Revolutionary New Technology Options

In this section, we will discuss and analyze crucial design features for advanced reactors that aim at meeting the key requirements, outlined before. **Table 1** presents advantages and disadvantages of the different operating neutron spectra and different potential coolants.

Other design variables that could be also revisited are power densities and power levels, different fuel material and designs, and fuel cladding. Power densities can range from about 70 MW/m<sup>3</sup> for typical LWR, and can be as high as 290 MW/m<sup>3</sup> for sodium cooled fast reactors, depending on different heat transfer capabilities, coolant properties, structural materials, and others. Besides, power levels can range from few tens of MWe for small-modular reactors (SMR) and can reach more than a thousand MWe (e.g. 1600 MWe for large size LWR). The

**Table 1.** Advantages and disadvantages of different designs features [4].

Design Features		Advantages	Disadvantages
<b>Neutron spectrum</b>	<b>Fast</b>	<ul style="list-style-type: none"> <li>• large fission to absorption ratio</li> <li>• minimize radioactive waste formation</li> <li>• burnup extension</li> <li>• sustainability (breeding)</li> </ul>	<ul style="list-style-type: none"> <li>• potential for disruptive power excursions</li> <li>• low margin to prompt criticality (Pu-239 compared to U-235)</li> </ul>
	<b>Thermal</b>	<ul style="list-style-type: none"> <li>• robust reactivity against fluctuations in physical parameters</li> <li>• safer margin to prompt criticality</li> <li>• long operational experience</li> </ul>	<ul style="list-style-type: none"> <li>• inefficient fuel exploitation</li> <li>• larger radioactive waste formation</li> </ul>
<b>Coolant</b>	<b>Sodium</b>	<ul style="list-style-type: none"> <li>• superior thermal hydraulic properties/heat transfer characteristics</li> <li>• excellent neutronic properties and economy</li> <li>• good compatibility with structural materials</li> </ul>	<ul style="list-style-type: none"> <li>• significant reactivity insertion issues (sodium boiling, large coolant temperature coefficient)</li> <li>• high chemical activity with water, steam, and air (explosion risk)</li> <li>• optical opacity</li> </ul>
	<b>Lead</b>	<ul style="list-style-type: none"> <li>• good natural circulation and heat transfer properties</li> <li>• superior neutronic characteristics and performance</li> <li>• chemically inactive</li> <li>• low cost (lead is abundant)</li> </ul>	<ul style="list-style-type: none"> <li>• high melting point (freezing potential)</li> <li>• erosion and corrosion potentials (need for coating)</li> <li>• Polonium-210 activity build up</li> <li>• optical opacity</li> </ul>
	<b>Molten Salt (fluorides or chlorides)</b>	<ul style="list-style-type: none"> <li>• high density-specific heat product (large grace period)</li> <li>• high boiling temperatures (no void reactivity insertions)</li> <li>• chemically inactive</li> <li>• optical transparency</li> </ul>	<ul style="list-style-type: none"> <li>• high melting point (freezing potential)</li> <li>• neutronicallly challenging</li> <li>• small thermal conductivity limiting the power density</li> <li>• pumping constraints (large viscosity)</li> </ul>
	<b>Inert Gas (e.g. Helium) [26]</b>	<ul style="list-style-type: none"> <li>• high breeding ratio</li> <li>• small void reactivity coefficient</li> <li>• chemically inactive (inert)</li> <li>• no corrosion/material challenges</li> <li>• optical transparency</li> <li>• potential for direct Brayton cycle (lower capital costs)</li> </ul>	<ul style="list-style-type: none"> <li>• high neutron leakage</li> <li>• relatively poor heat transfer characteristics</li> <li>• water ingress concerns (positive reactivity insertions)</li> </ul>

advantages of small power densities and small power ratings mainly lie in the increased robustness against loss of decay heat removal accidents and having more grace periods while staying away from critical temperatures. On the other hand, large power levels and densities benefit from the economy of scale, and have less environmental footprints, as they require less land use per MWe. Moreover, large reactors suit better the needs of mega-cities and centralized hubs. However, from the downside, higher density units are more susceptible to loss of decay heat removal accidents.

When it comes to the fuel, uranium (U-235) is the mostly utilized fissile element in commercial nuclear power plants. As said before, total reserves are estimated to be around 15.9 million tons, enough for 254 years of operation with the current rate of consumption. Taking all possibilities into account, the availability of nuclear energy could be extended for thousands of years [10]. Moreover,

thorium (namely Th-232) is becoming a fashionable promising fuel option, for which all uranium fuel cycles apply, and having many attractive properties. Thorium is three to four times more abundant than uranium and has superior physical properties in metallic and oxide states (high melting points, high thermal conductivity, small expansion coefficient). It has more specific energy (200 times more than natural uranium), and produces less nuclear wastes with shorter lifetimes [27]. Th-232 is fertile, and has no naturally fissile isotope, it does not undergo fission itself but, on capturing a neutron, it leads to uranium-233 as final product of the reaction chain, which is fissile. It could be misused for weapon production and, as its forerunner Pa-233 can be separated effectively, the Th-232 fuel is not proliferation proof. Thorium-based technologies are still at early phases with little commercial experience, therefore significant investments are still needed (testing, licensing).

Current commercial fuels are based on metal oxide ceramics (UO<sub>2</sub>), which enjoy high melting point (2850°C for UO<sub>2</sub>), yet suffer low thermal conductivities. Pure metal fuels include pure uranium or uranium alloys and have very high—if not the highest—fissile atom density, and high heat conductivity, but suffer from low melting points (1133°C). Non-oxide ceramic fuels have the advantage of high heat conductivities and melting points (2700°C - 2800°C), however, they are more prone to swelling than oxide fuels. Uranium carbide fuel have high thermal conductivity and high melting point, and are considered interesting candidates for some Gen IV reactors. Dissolved liquid fuels offer an inherently stable self-adjusting reactor dynamics and rapid drain ability into dump-tanks [4]. Finally, fuel cladding is another design variable that can be optimized. A huge program on accident tolerant fuel (ATFs), coordinated by Westinghouse, is working on high temperature resistant fuel (enriched U<sub>3</sub>SiC fuel pellets), and focus on protecting claddings from oxidation by coating.

In what follows, we introduce some new/exotic reactor designs that are at different stages of development. One of the prominent reactor concepts is Generation IV fast breeder reactors. They enjoy a high neutron economy that allows them to breed more fissile fuel than they consume using fast neutrons. Most of their proposed designs can use thorium, natural uranium, or spent fuel from LWRs, hence closing the fuel cycle and increases the utilization of uranium by at least 60 times compared to current LWR [28]. The new concepts claim to be inherently safe and highly resistant to proliferation. Fast breeder projects, which are followed with great interest, include: PRISM, BREST-OD-300, SWaB as well as a novel sub-critical accelerator-driven system (ADS) named MYRRHA [4]. PRISM (Power Reactor Innovative Small Module) is a sodium-cooled fast breeder reactor designed by GE Hitachi Nuclear Energy, and uses spent nuclear fuel from LWR [29]. The design comes with two reactor modules, each having a 311 MWe power output. PRISM is in an advanced stage of development with deployment aims by 2026 [30]. BREST-OD-300 is a 300 MWe Russian lead-cooled

fast breeder reactor. It is a pool-type reactor equipped with passive safety systems and uses spent nuclear fuel. The reactor is in an advanced stage and construction was approved in 2016, with the plant expected to be finished by 2026 [31]. SWaB (Seaborg Waste Burner) is a proposed molten salt reactor design by the Danish company Seaborg Technologies [32]. The molten salt used in this modular reactor will be a combination of spent nuclear fuel and thorium. The design is still in a very early stage of development. A novel concept currently in development by the Belgian Centre for Nuclear Research (SCK CEN) is the sub-critical accelerator-driven system (ADS) named MYRRHA (Multi-purpose Hybrid Research Reactor for High-tech Applications). This unique design couples a lead-bismuth fast reactor with a proton accelerator and is intended to function as an actinide burner (transmutation process). The reactor has a budget of 1.6 billion Euros and is scheduled to be commissioned by 2036 [33]. More details regarding these reactor concepts, characteristics, and key parameters are presented in **Table 2**.

Moreover, there is a revival of interest in small and simpler units for electricity production and other purposes. The incentive to develop advanced small (up to 300 MWe) modular reactors (SMR) comes from different sources. There is a strong belief [34] [35] [36] that SMRs would:

- open additional market sectors, e.g. heat production for chemical processes or seawater desalination, and, based on enhanced safety and security characteristics, allow for site flexibility, e.g. sites closer to consumers which are not accessible to large reactors;
- better adapt to low growth rates of energy demand and provide power away from large grid systems, often found in small or developing countries, and are more suitable to replace aging fossil-fired plants;
- lower requirement for access to cooling water and land, the latter comparing favorably with respect to wind and solar energy;
- lower upfront capital cost and lead to easier financing and earlier revenues;
- better meet specific user requirements, mostly in relation to safety by design incorporating reduced core inventory/potential source term, enhance safety performance through a high level of passive and/or inherent safety features and allow for underground construction for enhanced security and protection against external hazards including seismic, all that would hopefully help to improve public acceptance;
- allow for greater simplicity of design and modularization, enabling economy of serial production largely in factories, shipping to utilities and site-assembling, going along with potential usage of domestic resources;
- enable implementation of higher quality standards and shorter construction times;
- add ability to remove reactor modules or facilitate in-situ decommissioning at the end of lifetime.

Some argue that SMR facilitate a wider spread of nuclear energy, *i.e.* to newcomer

Table 2. Comparative survey of concepts with potentials to achieve key requirements. All lines include Small Modular Reactors (SMR) that reduce challenges and increase possibility to achieve outlined requirements, compared to large reactors.

Aspects	Reactor Lines					
	Pressurized Water Reactor (PWR)	Sodium Cooled (SR)	Molten Salt Cooled (MSR)	High Temperature, Gas-Cooled (HTGR)	Lead/Lead Bismuth (LR/LBR)	Accelerator-Driven Systems (ADS)
<b>Key Reactor Design Features</b>						
<b>Neutron spectrum, moderator; reactivity coefficients; excess reactivity</b>	thermal, light water; all negative; needed	fast, -; pos. CTC <sup>(5)</sup> ; small	semi-thermal to fast <sup>(6)</sup> , -; most challenging, CTC positive to different degrees; small	thermal, graphite; all negative; not needed	fast; -; CTC small positive/negative; small	subcritical fast reactor; -; additional neutrons from outside source; not needed
<b>Coolant; primary pressure (MPa)</b>	light water; high (15,5)	sodium; unpressurized	molten salt <b>fluorides<sup>(9)</sup></b> or chlorides <sup>(8)</sup> ; unpressurized	helium; moderate (7)	Pb/Pb-Bi; unpressurized	Pb-Bi used as target and reactor coolant
<b>i. melting/boiling temperature (°C)</b>	i. reasonable (0/345 <sup>(2)</sup> ) ii. moderate (330) iii. absorptive, scattering	i. low (98)/high (892) ii. elevated (510) iii. excellent	i. reasonable (fluorides: 450/chlorides 396) ii. high to very high (750 <sup>(8)</sup> to 900) iii. neutral iv. non-stable; corrode metallic alloys v. good, high storage capability	i. high (750 -1000 envisaged) iii. neutral iv. inert gas v. poor	i. reasonable/high (327/1737; spallation target, as 125/1670) ii. high (573) iii. neutral/Po-210 build-up iv. chemically inactive v. excellent thermodynamic properties	i-v: see LBR
<b>ii. core outlet temperature (°C)</b>	iv. stable, low corrosive, non-toxic	iv. highly reactive with water and air				
<b>iii. neutronic characteristics</b>	v. good	v. excellent				
<b>iv. chemical characteristics</b>						
<b>v. heat transfer/storage capabilities</b>						
<b>Fuel: enrichment and cladding</b>	UO <sub>2</sub> , (U, Pu)O <sub>2</sub> solid; low (3.7); zircaloy	(U, TRU)O <sub>2</sub> or metal (U-Pu-Zr), solid; elevated (15.2); high-strength stainless steel	U, Th, TRU, dissolved/dispersed in salt; elevated (15, 7 <sup>(8)</sup> ); -	UO <sub>2</sub> or ThO <sub>2</sub> solid; elevated (8, 5); ceramic coating	(U,TRU)O <sub>2</sub> or metal, solid; elevated (16, 7); stabilized stainless steel	see LBR
<b>Power, density (MW/m<sup>3</sup>), size(MWt); burnup (GWd/tHM)</b>	high (80), large (4800); moderate (45 - 50)	extremely high (290), small (200) to large (2400);elevated (72)	high (70) to extremely high (300 <sup>(8)</sup> ); small (50) to large (2400 <sup>(8)</sup> ); elevated (67 <sup>(8)</sup> )	small (8), small (250); high (100)	high(130), small (70) to large (2400);elevated (77)	high, small (100 critical mode);
<b>General layout; structural materials; siting</b>	loop type; metallics; above ground	pool type; metallics; n.a.	pool type with overflow tank; metallics (special alloys); above ground/underground	loop type; ceramic core, steel vessel; <b>underground</b>	pool or loop type; metallics-steel or prestressed concrete vessel; above ground/ n.a.	<b>pool type;</b>
<b>Start up; refueling, cycle length (years)</b>	no startup fuel; periodical, 3 - 5	no startup fuel; periodical, 3, 5	startup fuel; continuous, 5 - 6 <sup>(8)</sup>	fuel continuously added; continuous, 15 cycles	needed in case of Th fuel; periodical, 5.4 to 8	see LBR

**Continued**

**Fuel Cycle/Waste Management**

<b>Reactor characteristics;</b> <b>Basic fuel cycle concept</b>	fuel burner; open or partially closed (MOX fuel)	fuel breeder, waste burner; closed cycle	fuel burner, breeder <sup>1)</sup> , waste burner <sup>2)</sup> ; fully closed	fuel burner (power & heat production); open	self/fuel breeder, waste burner; fully closed	waste burner; fully closed
<b>Use of energy content;</b> amount of waste	Small (2/3); high	efficient; minimized	efficient; minimized	small-moderate; high volume	efficient; minimized	see LBR
<b>Radiotoxicity of disposed waste;</b> husbandry times	high; extremely long <sup>3)</sup>	lowered <sup>4)</sup> ; reduced to historical scale (as waste burner)	lowered; reduced to historical scale (as waste burner)	high; extremely long (but leach proof coated particle)	lowered; reduced to historical scale (as waste burner)	see LBR

**Proliferation resistance**

moderate (smaller in  
case of reprocessing)

concerns due to elevated,  
almost high enrichment  
and reprocessing (reduced in case of thorium)

moderate to small (due to  
elevated enrichment)

concerns due to elevated  
enrichment and reprocessing  
(reduced in case of Th fuel)

see LBR

**Reactor Safety**

<b>Basic approach;</b> target values for accidental states; decay heat removal	defenseindepth; CDF <sup>4)</sup> $2 \times 10^{-6}/a$ ; active	defenseindepth; ?; passive (premature)	defenseindepth; no target values; passive incl. dumping system (all premature)	inherently safe design; no target values; inherent/passive	defense indepth, inherently safe design; no target values; inherent/passive (all premature)	see LBR
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**Core fission product  
inventory (Bq)**

extremely high ( $10^{16}$  -  
 $10^{19}$ )

low (continuous extraction of  
some fission products)

**moderate** due to power  
size (about  $10^{17}$ )

moderate to high (depending  
on scale)

see LBR

**Sensitivity to unprotected  
scenarios**

i. small; ii. high; iii. non-full power/ startup; notable	i. notable, self-controllability might be ensured	i. distinctive, self-controllability hard to ensure	i. very small, self-controllability	i. very small, self-controllability	i. very small, self-controllability	i. eliminated ii-iii: see LBR
i. reactivity induced accidents; ii. loss of cooling accidents; iii. others (concept specific)	ii. narrowed down, hard to fully ensure self-control	ii. narrowed down, decay heat removal by passive means	ii. eliminated	ii. narrowed down, decay heat removal by passive means	ii. narrowed down, decay heat removal by passive means	
	iii. sodium fire/explosion following leaks	iii. flawed fuel addition/chemical reprocessing, overcooling/freezing	iii. potential air/water ingress scenarios	iii. overcooling/freezing (LBE)	iii. overcooling/freezing (LBE)	

**Importance of  
containment/protection  
against external impact**

high/high

high/high

minor/minor

minor/n.a.

(see LBR)

**Necessity of off-site  
protective measures**

not accounted for

n.a.

regarded not necessary

n.a./probably not

(see LBR)

**State of Development**

<b>Design state/maturity</b>	commercially deployed	feasibility partly proven, use of real experiments/demos	early research state	feasibility proven, <b>advanced stage of construction</b>	early design state of new concepts, reactors operated in Russia	early design state, under <b>development</b>
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## Continued

<b>Technology approach;</b> development risk	proven; small	renewed, non-standard; high	exotic; high	non-standard/innovative; moderate	evolutionary; high	revolutionary; extremely high
<b>Show stoppers;</b> major barriers	none; economics, lack of acceptance	mastering sodium technology, passive system design; public resistance	insufficient stability of salts and material resistance; long term reliability of components	no/fuel and materials qualification for very high temperatures; economic	insufficient corrosion control of steels; fast creep of hanging, heavy loaded vessel	lack of high-intensity proton accelerator
<b>Regulatory issues</b>	certified, licensed	partially new approach and methods needed, can base on experience	new approach and methods needed	certified, <b>licensed in PR China</b>	new approach and methods needed, can base on experience	new approach and methods needed
<b>Costs/Construction Times</b>						
<b>Capital/Investment costs</b>	high (8.5 - 10.5 billion €)	presumably high	n.a.	lowered <sup>6)</sup> (modular design)	n.a.	high (about <b>1 billion €</b> )
<b>Construction time (years)</b>	on-site; long (8 - 11 Flamanville, 14 Olkioluto)	on-site; presumably long	n.a.	on-site; lowered	n.a.	on-site; long (>10)
<b>Electricity generating costs</b>	competitive	n.a.	n.a.	probably not competitive	n.a.	n.a.
<b>Overarching, Contextual Issues</b>						
<b>Complexity, novelty</b>	moderate, evolutionary standard	high, innovative (can base on experience)	high, highly innovative (but past and ongoing R&D)	small, innovative	high, innovative	very high, innovative
<b>Dependance on socio-political stability, safety culture</b>	up to major (waste); to be ensured	high-lowered for minimized waste; to be ensured	n.a.-lowered for minimized waste; n.a.	less <sup>6)</sup> ; less	n.a.-lowered for minimized waste; n.a.	(see LBR)
<b>Need of intricate infrastructure</b>	moderate (depending on fuel cycle)	moderate, secondary sodium cycle	high, on-line processing, secondary salt coolant cycle	small	small, no intermediate cooling cycle	high, high intensity proton accelerator and spallation source
<b>Overall uncertainties</b>	moderate (mainly management, cost/construction time overruns)	high-public resistance	high, barely assessable	moderate (first of its kind problems)	high-moderate, barely assessable	very high, barely assessable

**Concrete benchmarks by column:** PWR: European Pressurized Water Reactor (EPR); SR: bold numbers for MIT Sodium Fast Reactor concept; MSR: bold numbers for Seaburg Waste Burner (SaWB), a) for Eur. Molten Salt Fast Reactor concept (MSR), b) for MIT Molten Salt Fast Reactor concept; HTGR: bold numbers for Chinese pebble-bed, modular HTR (HTR-PM); LBR/LRR: bold numbers for MIT Lead/LBE cooled Fast Reactor concept; ADS: bold numbers for MYRRHA, Mol/Belgium. **Annotations:** <sup>1)</sup>indicating loss of core cooling, loss of heat sink (transients), station blackout; <sup>2)</sup>at system pressure; <sup>3)</sup>beyond historical time frame; <sup>4)</sup>core damage frequency; <sup>5)</sup>coolant temperature coefficient; <sup>6)</sup>compared to EPR benchmark; "n.a.": not assessable due to premature design status.

States, and make easier the build up a domestic nuclear infrastructure and capabilities, and even foresee a hybrid energy system with integration of SMR and renewables.

All types of large reactors currently in use or being developed are represented in the SMR lines, including thermal neutron spectrum water-cooled reactors, various kinds of fast neutron spectrum reactors (FR) including liquid metal and temperature molten salt cooled reactors, and gas-cooled, graphite moderated high reactors (Figure 3). LWR-based SMR has the lowest technological and regulatory risk, while some fast SMR concepts enable longer operation before refueling, which is regarded advantageous, where appropriate. However, some question the economic competitiveness of SMR and raise concerns regarding adequacy of the current regulatory system and licensability of some (first-of-its-kind) designs.

A promising SMR reactor design that enjoys interesting inherent safety features is the HTR-PM, or the high-temperature gas-cooled pebble-bed modular reactor. HTR-PM is a generation IV thermal reactor utilizing multi-layered spherical fuel elements (TRISO coated particles) which act as a barrier against fission product escape. The design comes with two reactor modules each having 250 MW thermal power, connected to a single 210 MWe turbine. The HTR-PM demonstration plant is in advanced stage of construction in the Shindao Bay, Shandong area in China, and is expected to be operational soon [37].

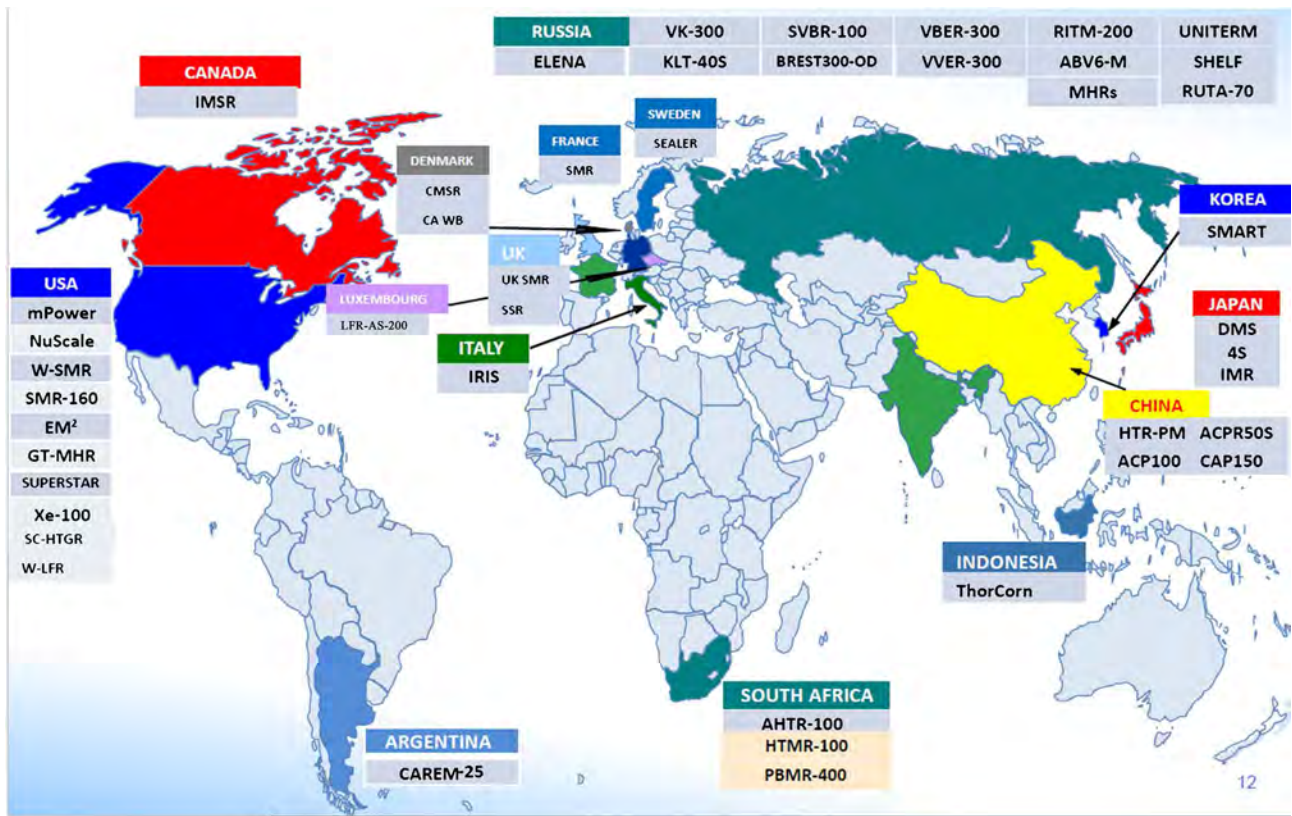


Figure 3. Worldwide development of small and medium sized modular reactors [36].

After commissioning, the HTR-PM will be the first operable generation IV reactor in the world.

Other interesting concepts include the floating SMRs, which can be built at shipbuilding facilities and towed to the designated areas where they could provide electricity, district heating and seawater desalination. These features could be notably important for power-hungry developing countries undergoing industrialization. Currently, there is one operating floating nuclear power plant in the world, the Akademik Lomonosov, which was commissioned in December 2019 in Russia [38]. The plant consists of two 35 MWe PWRs, which are based on the KLT-40 marine propulsion reactors. These reactors operate in the thermal neutron spectrum using 14.1% enriched uranium and employ a combination of passive and modernized active safety systems [39].

## 6. Concept by Concept Comparison and Assessment against Key Requirements

After presenting different novel reactor concepts that try to fulfill the key requirements of Section 4, we try here to make a concept-by-concept comparison between the different technologies. The results indicate a high potential for far-reaching improvements compared to the most advanced existing LWRs. **Table 3** shows a ranking of the presented Gen IV reactors against key safety criteria detailed in section 4, compared with the Gen III+ large EPR. As can be

**Table 3.** Ranking from excellent (5) to neutral to very poor (1) of Gen IV reactors against key safety criteria—with the Gen III EPR as the benchmark.

<i>Candidate reactor concepts—varying coolant, selected designs in brackets</i>					
Key requirements	Water-thermal (large EPR)	Sodium-fast (PRISM)	Molten Salt-fast (SaWB)	Helium-thermal (HTR-PM)	Lead-fast (BREST-OD-300)
<b>Elimination of Reactivity Induced Accidents</b>	4	2	1	5	2 - 3
<b>Resistance to Loss of Active Core Cooling</b>	1	2	3	5	2 - 3
- avoid exceeding critical temperatures	1	n.a.	n.a.	5	n.a.
- avoid high fission product inventory	1	4 <sup>1</sup>	5 <sup>2</sup>	4 <sup>1</sup>	4 <sup>1</sup>
- provide sufficient heat storage & transfer capacity	4	5	4	4	5
<b>Structural Integrity</b>	2	4	4	5	4
- avoid high operating pressure [suitability of underground siting]	1 [2]	4 <sup>3</sup> [5]	5 [5] <sup>4</sup>	4 [5] <sup>4</sup>	4 <sup>3</sup> [4]
<b>Use Non-chemically Reactive/Non-Toxic Materials</b>	4	1 <sup>5</sup>	2 <sup>5</sup> (non-stable)	5	4
<b>Avoid Long-lived Radioisotopes</b>	1	4	5	4	5
<b>Enhance Proliferation Resistance</b>	4	2	2	3	2
- avoid high enriched uranium	5	2 <sup>6</sup>	2 <sup>6</sup>	2 - 3	2 <sup>6</sup>

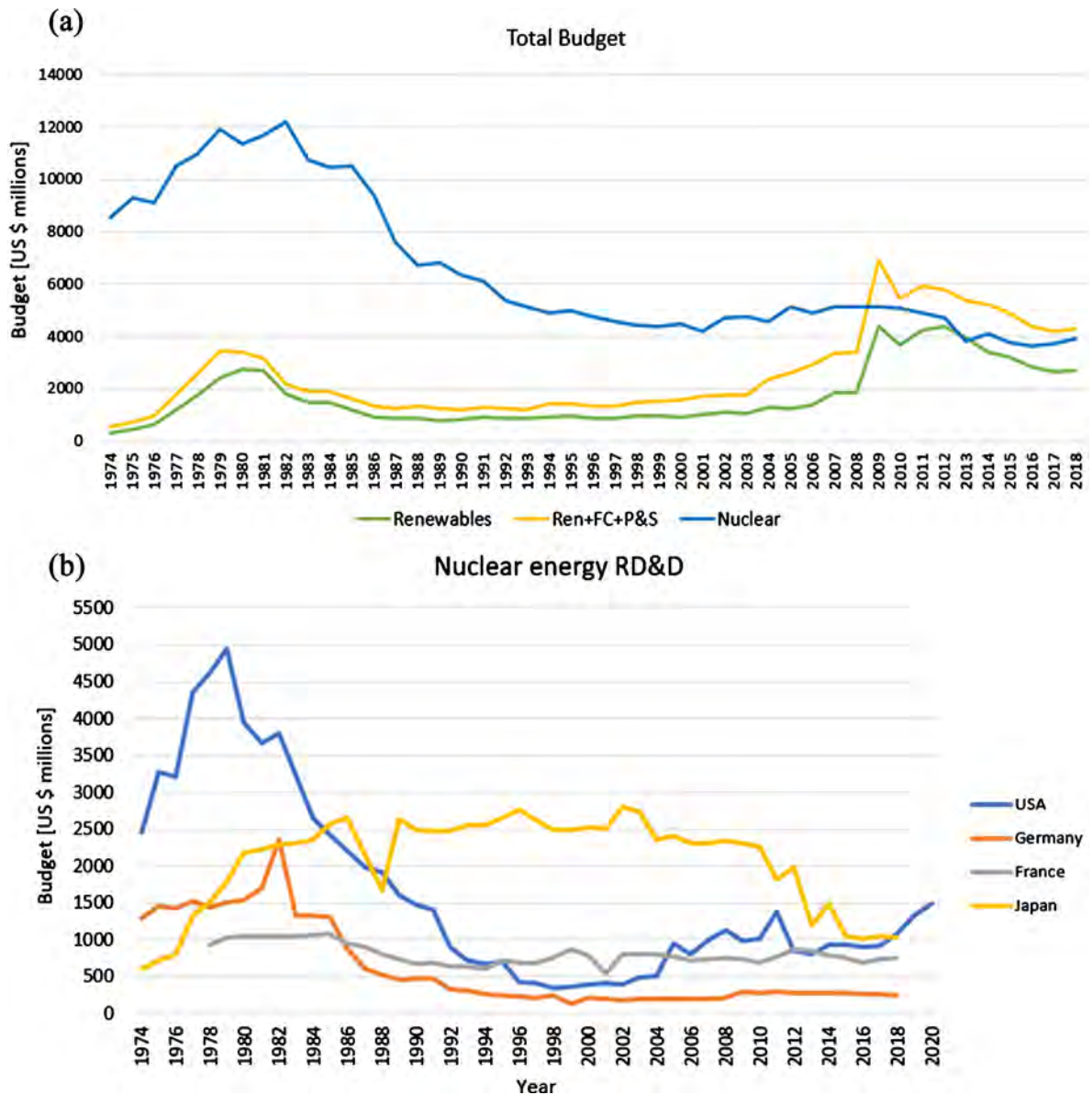
<sup>1</sup>Due to small power size; <sup>2</sup>in case of dispersed fuel & due to small power size; <sup>3</sup>not pressurized but high static load; <sup>4</sup>foreseen; <sup>5</sup>intermediate cycle (IH)X foreseen; <sup>6</sup>close to HEU lower limit.

seen, none of the best versions, *i.e.* small sized in general, of the candidate concepts meet all requirements convincingly, yet. Thermal helium cooled reactors (HTR-PM) come closest, promising inherent robustness against classical severe accidents and largely avoiding long-lived radioisotopes when using thorium fuel; they are however not capable of burning waste. With respect to burning waste, molten salt fast reactors promise to do best but appear most susceptible to reactivity-induced accidents, as are all liquid metal cooled fast reactors, albeit to different degrees. There is also a potential of new concept specific accidents, such as overcooling/ freezing of coolant, chemical reactions following coolant out-flows after leaks or air/water ingress into hot graphite cores, which deserve special attention. Thus, future research and development appear necessary, aiming at further improving some essential characteristics and features and/or combining design elements in a radically new and innovative way. All concepts seem to have limited capabilities to achieve the goal of reducing proliferation risk or even to maintain the current level, mainly due to partially elevated and/or significantly increased enrichment needed for reprocessing.

It is also important to note that revolutionary designs and technologies often introduce new man-machine interfaces, lack experience, and tend to represent a jump in complexity. The molten salt cooled systems with dissolved fuel, fission products, and off-gas systems may serve as example; some features of coolants, e.g. production of activation products, chemical toxicity, non-transparency, freezing at high temperatures, may require complex operations and maintenance procedures ([5], p. 69). All this may also present regulatory barriers since the regulators need to assess unfamiliar technologies. This calls for unprecedented excellence in research, development, design, and for adequate funding.

Historically, public funding for civil nuclear research, development, and demonstration (RD&D) has contained the highest share of the total budget for energy RD&D in OECD countries until 2009 (Figure 4(a), Figure 4(b)). Since then, it was overtaken by the combined budget for renewable energy sources (wind, solar, small hydro), research in fuel cells, hydrogen and other power and storage technologies (Ren + FC + P & S respectively) [40]. The share has been on the decline for 30 years, from having 75% of the total energy RD&D funding in 1974, to approximately 20% in 2013. In the more recent years, the investments in civil nuclear research in OECD countries are slowly increasing, stabilizing their share at 22% of the total energy technology RD&D budget. This increase is mainly due to a revised strategic view in the USA on nuclear power as a viable and affordable low carbon footprint power, especially with the future application of advanced reactor designs.

In the USA, the 2019 RD&D budget for nuclear energy was \$1.3 billion, including \$100 million for the research of advanced SMR and \$112 million for Advanced Reactor Technologies [41]. The budget for 2020 was increased by 13% to \$1.5 billion, out of which \$230 million are allocated for a program focused on the “construction of real demonstration reactors that are safe and affordable (to



**Figure 4.** Total budget for RD&D over time per type of technology (a) and countries (b), in millions of US dollars in their value in 2018 (adjusted for inflation).

build and operate) in the near and mid-term” [42]. Two demonstration projects are set to begin this year, where the program will support at least 50% of the costs, the remainder to be provided from the private sector. This program will also support the Nuclear Regulatory Commission (NRC) in evaluating safety issues and technical challenges unique to advanced reactors and their licensing. Other OECD countries (e.g. France, UK) have also increased their research in nuclear power in recent years, as well as Russia and China, which have invested heavily in the development of advanced reactor designs.

## 7. Concluding Remarks

There is a growing demand in electricity production due to its increasing use in oth-

er sectors (e.g. e-mobility, heat, and digitalization) and expanding access to billions of people while simultaneously drastically enlarging the share of de-carbonized generation assets, in particular. With such high stakes, to rely on variable, low-density wind and solar options including necessary infrastructure alone appears a strategic error, disregarding the proven principle of diversity. We argue for keeping the nuclear option open or even for expanding its global use, supported by future revolutionary safe, clean and thus more sustainable nuclear technologies, which would be acceptable to an otherwise presently mostly nuclear-averse society. This proposal is further supported when one acknowledges the real problem of stewardship of already existing high-grade nuclear waste over time scales eclipsing that of stable societies. Novel concepts are portrayed and then assessed against set-up key requirements calling for a shift towards reactor designs that incorporate passive and inherent safety features, use fuel more effectively, avoid or eliminate long-lived actinides, and increase proliferation resistance. The results show a high potential of improvement compared to current designs but further research and development are needed to fully meet those requirements; small modular designs (SMR) of different reactor lines are of special interest.

To realize this vision, substantial ongoing national and international RD&D programs exist, although funding is still at historically low levels. Moreover, in the nuclear industry, there is the risk of stagnation of essential human capital and know-how. The concepts and designs presented provide the impulse to get us over the existing hurdles, but the scope is ambitious, and time delay from RD&D to commercial deployment, in general, appears too long especially in the West, stemming in part from regulatory inertia. Therefore, we call for an urgent increase in government and international RD&D funding by the order of a few hundreds of billions of USD per year, for an international civilian “super-Apollo” program. Such a large-scale public program is not unprecedented in size, and experience indicates that such investments in fundamental technologies are not only of immense public benefit but also enable revolutionary innovations to be spun out that would not otherwise ever have been attained.

## Conflicts of Interest

The authors declare no conflicts of interest regarding the publication of this paper.

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# Americium Transmutation in the SVBR-100 Reactor

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**How to cite this paper:** Gulevich, A.V., Eliseev, V.A., Komlev, O.G., Tormyshev, I.V. and Toshinsky, G.I. (2020) Americium Transmutation in the SVBR-100 Reactor. *World Journal of Nuclear Science and Technology*, 10, 116-128.  
<https://doi.org/10.4236/wjnst.2020.103011>

**Received:** June 2, 2020

**Accepted:** July 10, 2020

**Published:** July 13, 2020

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## Abstract

One of the postponed problems of nuclear power (NP) is the problem of the management of long-lived radioactive waste (RAW), and, first of all, with minor actinides (MA), of which americium-241 is the most difficult. The aim of this work is to study the efficiency of americium transmutation in a fast reactor with a heavy liquid metal coolant lead-bismuth eutectic alloy. The article presents the results of calculations of the transmutation of americium in the SVBR-100 reactor using standard uranium oxide fuel with the addition of americium-241. The obtained values of the rate of transmutation of americium are compared with similar values for the SVBR-100 reactors on MOX-fuel and in the BN-800 reactor.

## Keywords

SVBR-100, Fast Reactor, Nuclear Power, Lead-Bismuth Eutectic, Minor Actinides, Americium, Nuclear Fuel Cycle, Neutron Spectrum, Core

## 1. Introduction

One of the postponed problems of nuclear power (NP) is the problem of the treatment of long-lived radioactive waste (RAW), and, first of all, with minor actinides (MA). Of the minor actinides (Neptunium, Americium, Curium),  $^{237}\text{Np}$  poses the least radioecological hazard due to the very long half-life ( $T_{1/2} = 2.1 \times 10^6$  years). Therefore, its transmutation should be considered last. At the same time, when storing  $^{237}\text{Np}$  in large quantities, it is necessary to take into account nuclear safety problems, since its critical mass for the fast neutron spectrum is only 60 kg (Ref. [1]) (in the thermal spectrum,  $^{237}\text{Np}$  does not have a critical mass, since its fission cross section has a threshold character).

Curium transmutation is also impractical, since the main part of its isotopes

$^{243}\text{Cm}$  and  $^{244}\text{Cm}$ , which are alpha emitters, on the contrary, have relatively short half-lives (29 years and 18 years), and it is expediently to keep them in controlled storage for about 100 years before decaying into plutonium isotopes that return to the closed nuclear fuel cycle (CNFC) of the future large-scale nuclear power. In this case, of course, it is necessary to take into account the high specific heat release of curium.

The greatest difficulty in handling is represented by americium isotopes ( $^{241}\text{Am}$ ,  $^{242\text{m}}\text{Am}$ ,  $^{243}\text{Am}$ ), due to their orders of magnitude higher specific radioactivity compared to  $^{237}\text{Np}$  and high gamma radiation  $^{241}\text{Am}$ , formed during  $^{241}\text{Pu}$  beta decay. The americium isotopes have long half-lives in comparison with the curium isotopes, which makes it unpromising for their long-term storage until they decay into curium isotopes. Their half-life equals 432, 141 and 7370 years, respectively. The  $^{242\text{m}}\text{Am}$  and  $^{243}\text{Am}$  isotopes are formed upon neutron capture by the  $^{241}\text{Pu}$  and  $^{242}\text{Pu}$  isotopes.

Fast reactors (FR), working in the CNFC, are able to solve the postponed problem of handling MA by their multiple recycling. However, at the same time, the complexity of manufacturing fresh recycled fuel and subsequent handling (before loading it into the reactor) increases significantly due to the high radioactivity and heat release of fuel rods and fuel assemblies (FAs). In this regard, various methods of nuclear transmutation of americium in the fast neutron spectrum of a critical or subcritical (with an accelerator) specialized transmutation reactor are considered, when americium is fissioned and converted into ordinary relative short-lived fission products, and when neutrons are captured, it becomes isotopes of transuranic elements.

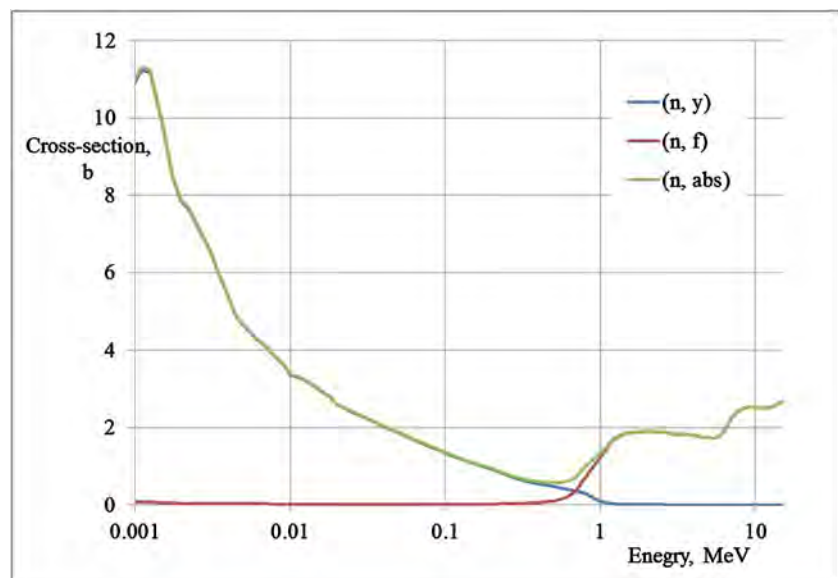
In the earlier work (Ref. [2]), the possibility of transmutation of minor actinides in the SVBR-100 reactor with MOX-fuel was considered. In one of the options considered in this work, core loading was formed from 1360 kg of plutonium ( $^{238}\text{Pu}/^{239}\text{Pu}/^{240}\text{Pu}/^{241}\text{Pu}/^{242}\text{Pu} = 1.86\%/59.81\%/24.85\%/7.67\%/5.81\%$ ) and 218 kg of minor actinides ( $^{237}\text{Np}/^{241}\text{Am}/^{243}\text{Am} = 39.57\%/53.24\%/7.19\%$ ). With a campaign duration of 76,000 eff. h. the mass of MA at the end of the campaign was 165 kg, thus, the decrease in minor actinides was 53 kg.

Since MOX-fuel contains a significant amount of plutonium, during the operation of the reactor with this fuel,  $^{241}\text{Am}$  is produced due to  $\beta$ -decay present in the initial fuel and formed during the operation of the reactor  $^{241}\text{Pu}$ . Thus, the efficiency of MA transmutation in a MOX-fuel reactor is reduced. A more effective option would be to transmute MA in a reactor operating on uranium oxide fuel, in which plutonium is initially absent.

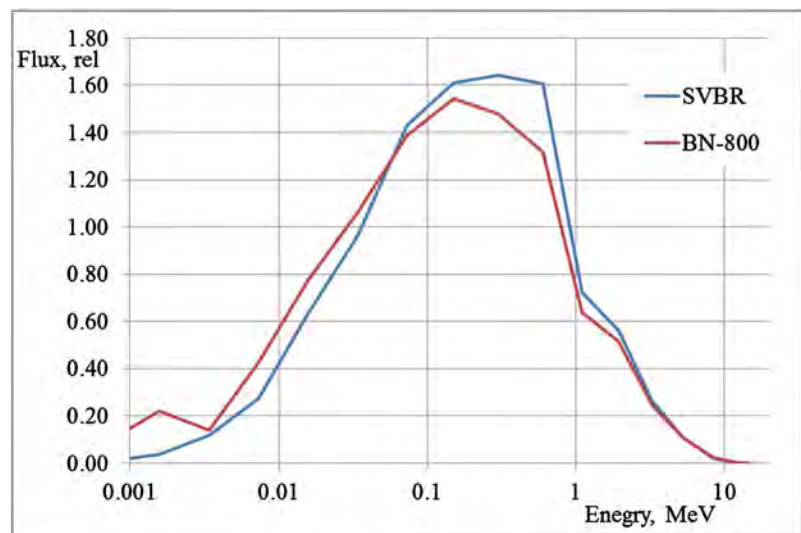
In [3], the transmutation of americium into an FR with a sodium coolant (BN-800) was considered, where the rates of its conversion to both fission products and  $^{242}\text{Cm}$  upon radiative capture of  $^{241}\text{Am}$  neutrons were determined. At a neutron energy lower than 0.8 MeV, its cross section for radiation capture significantly exceeds the cross section for fission (see **Figure 1**). Due to the stronger elastic moderation on sodium nuclei (compared with lead and bismuth), the neu-

tron spectrum in the BN-800 reactor is somewhat softer than in the SVBR-100 (see **Figure 2**). This should also lead to a higher transmutation rate of  $^{241}\text{Am}$  than in the SVBR-100 reactor, *ceteris paribus*.

The influence of the hardness of the neutron spectrum on the ratio of the rates of neutron capture and fission processes can be seen from a comparison of the parameter  $\alpha_{Am}$  (ratio of radiation capture and fission cross sections averaged over the neutron spectrum) of BN-800 reactors (Ref. [3]) and the SVBR-100 reactor (this work), where they are equal to  $\sim 45$  and  $5$ , respectively, due to a sharp decrease the capture cross section of  $^{241}\text{Am}$  at a neutron energy above  $0.8$  MeV and, conversely, by increasing its fission cross section. Therefore, the rate of formation of  $^{242}\text{Cm}$ , having a half-life of about 165 days, with high specific heat release, all else being equal, in the BN-800 reactor will be significantly higher



**Figure 1.** Americium-241 absorption, fission and capture cross-sections energy dependence.



**Figure 2.** Neutron spectrum in SVBR-100 and BN-800 reactors.

than in the reactor SVBR 100.

In this work, we consider the transmutation of americium extracted from spent nuclear fuel (SNF) of VVER-1000 after 15 years of storing, with a typical isotopic composition (Ref. [4])  $^{241}\text{Am}/^{242\text{m}}\text{Am}/^{243}\text{Am} = 85\%/0.1\%/14.9\%$  in an FR with an HLMC, in particular, in the SVBR-100 reactor with uranium oxide fuel.

## 2. Description of Calculated Model

For calculations, the design of the core of the SVBR-100 reactor using uranium oxide fuel was adopted, corresponding to that described in Ref. [2].

The diameter of the core is 1645 mm, the height of the active part of the fuel rods is 900 mm. The total number of fuel rods in the core is  $\sim 12,500$ . The fuel rods are arranged on a triangular lattice with a pitch of 13.6 mm and are combined into fuel assemblies, the total number of which is 61.

Container type fuel elements, the diameter of the cladding along the cylindrical part is  $12 \times 0.4$  mm, the cladding is a tube with 4 spiral spacing ribs, the diameter of the cladding along the ribs is 13.5 mm. Fuel pellets ( $\text{UO}_2$ ) have an average enrichment in uranium-235 of about 16.5%. The density of oxide fuel was taken equal to  $10.6 \text{ g/cm}^3$ , the full loading on heavy atoms (h.a.) is  $\sim 9.1$  tons. The radial power distribution is equalized by physical profiling of the content of fissile nuclei in four radial zones with an increase in the content of fissile nuclei from  $\sim 13\%$  to  $19.5\%$  from the center to the periphery.

Volume fractions of materials in a physical cell corresponding to one fuel element are given below:

- volume fraction of fuel  $\sim 0.615$ ;
- volume fraction of steel  $\sim 0.105$ ;
- volume fraction of the coolant  $\sim 0.28$ .

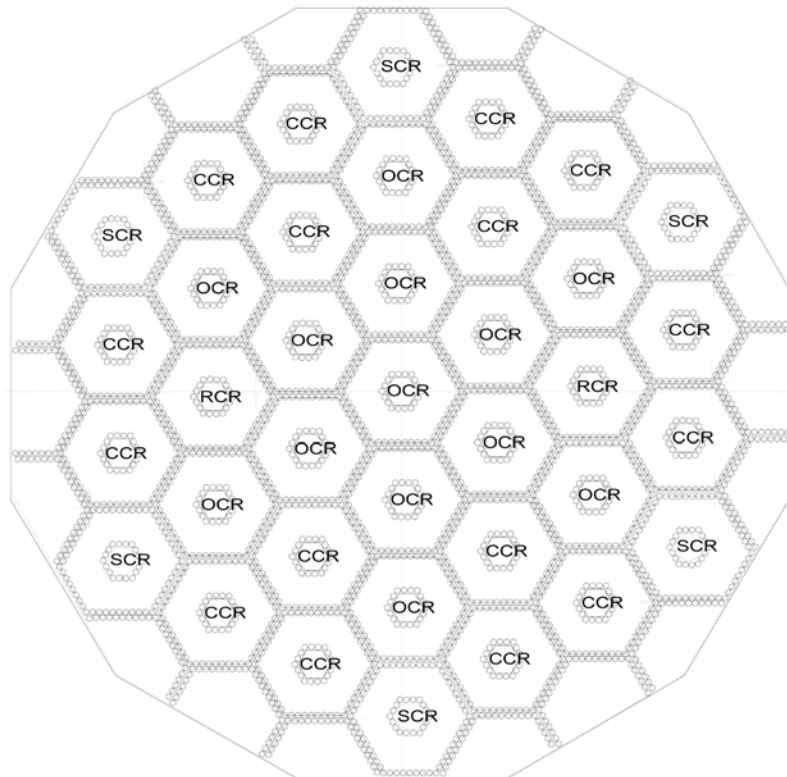
A steel reflector is located in the lower part of the fuel rod, under the steel reflector there is a compensation volume for collecting gaseous fission products (FPs) entering the cladding.

Above and below are the structures of the reactor removing part with the core, which ensure the placement of fuel rods and fuel assemblies in the core and the possibility of their loading and unloading (upper and lower grids of fuel assemblies).

On the sides, the core is surrounded by a reflector, which is a steel box filled with lead-bismuth eutectic (LBE).

The reactivity margin is compensated by a system of control rods placed along the axis of the fuel assembly. The rods form a triangular lattice in the core with a pitch of 224 mm. The number of rods and their design is determined in accordance with the requirements for compensating for changes in reactivity during the campaign. The cross-sectional diagram of the core with the maximum number of rods (37 pcs.) is shown in **Figure 3**.

The calculations were carried out using the REACTOR program package (Ref. [5], used to substantiation the neutron-physical characteristics of fast reactors



OCR – operating control rod      RCR – regulating control rod  
 CCR – compensating control rod    SCR – scram control rod

**Figure 3.** Diagram of the core with the 37 control rods.

with HLMCs. The R-Z model of the core was used in the calculations, such an approximation makes it possible to fairly accurately estimate the reactor characteristics from the campaign—heavy atom balances and reactivity drop.

The starting load of the first campaign was formed as follows.  $^{235}\text{U}$  load left unchanged like Ref. [2]. Americium was added to the fuel, replacing  $^{238}\text{U}$  in the “nucleus to nucleus” ratio. The load with americium was taken to be 218 kg, the total load of  $Np$  and  $Am$  in Ref. [2], which corresponds to its mass content in the fuel of about 2.5%. In this case, the heat release from americium in one fresh fuel assembly (3.6 kg  $Am$ ) will be 0.3 kW, and in the spent fuel assembly one month after the reactor shutdown, taking into account the residual heat and heat generation of minor actinides, ~15 kW, which must be taken into account the technology of refueling of the SVBR-100 reactor, designed today for heat generation in spent fuel assemblies of 9 kW.

Core loads for subsequent campaigns were formed according to the following algorithm. For the second campaign, the fuel load is similar to the start one. The SNF of the first campaign is used to form the load for the third campaign after seven years of storage and reprocessing. In this case, the remaining isotopes of uranium and americium are used, enriched uranium is added instead of removed fission products to restore the reactivity margin, as well as external americium of the accepted isotopic composition from the SNF of the thermal reactors

to make up for its loss.

It is assumed that the extracted Pu will be used in the CNFC of large-scale nuclear power as a fuel for FRs. All MAs are separated and are not used in the calculations. Neptunium and curium go to their storage facilities, *Np*—for long-term storage until they are sent for transmutation after the completion of the transmutation of accumulated americium, and *Cm*—for about 100 years until its isotopes are converted into isotopes of plutonium and returned to the nuclear fuel cycle. According to a similar algorithm, the load of the fourth campaign is formed from the SNF of the second campaign, the load of the fifth from SNF of the third and so on.

Reactor calculated model is presented in **Figure 4**.

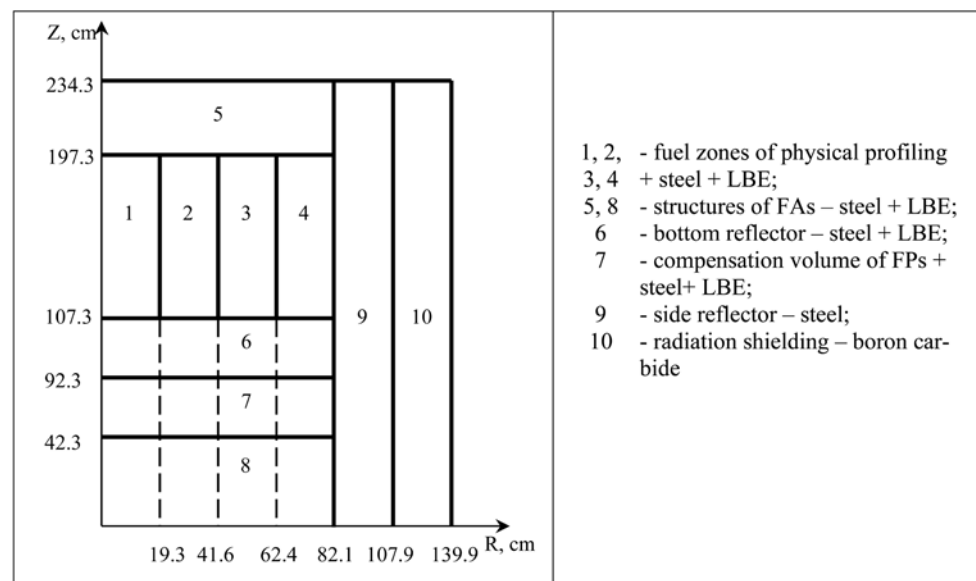
The choice of SNF recycle time for seven years is associated with the features of the fuel cycle of the SVBR-100 reactor. The core of the SVBR-100 reactor is reloaded immediately, but after the end of the campaign. The calendar duration of the campaign in the variant with uranium fuel is 7 years (corresponding to 50,000 eff. hours). Thus, SNF unloaded from the reactor can be used to form new loading of the SVBR-100 core not earlier than after 7 years.

### 3. The Results Obtained and Their Discussion

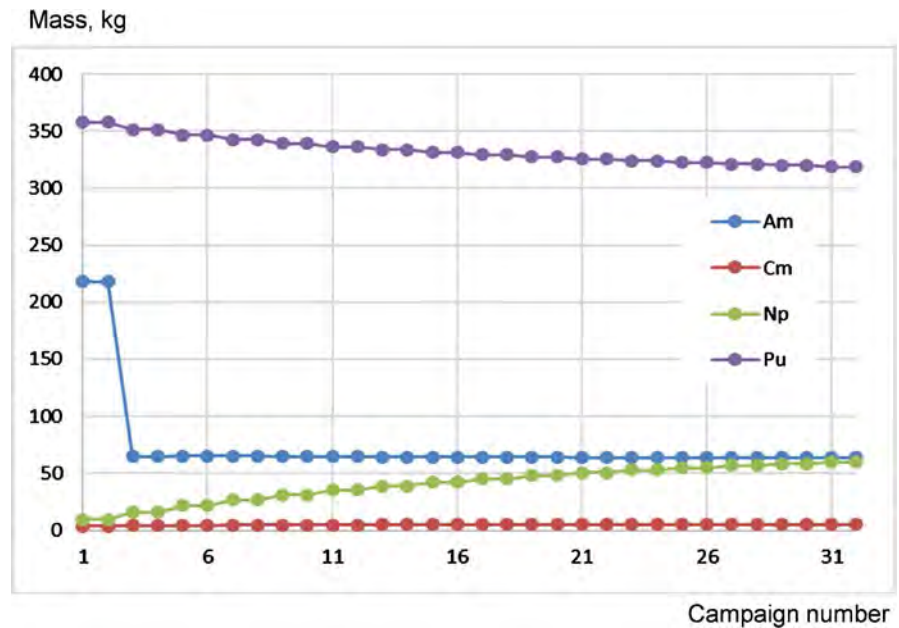
The results of the calculation of the main characteristics of the consumption and production of actinides during the first 32 campaigns of the SVBR-100 reactor are presented in **Figures 5-8** and in **Table 1**.

As can be seen from **Figure 5**, the producing share of *Pu* during fuel recycling is reduced, since  $^{238}\text{U}$  is gradually replaced by the  $^{236}\text{U}$  isotope generated from  $^{235}\text{U}$ . The equilibrium fraction of  $^{236}\text{U}$  in the isotopic vector of uranium is ~13%.

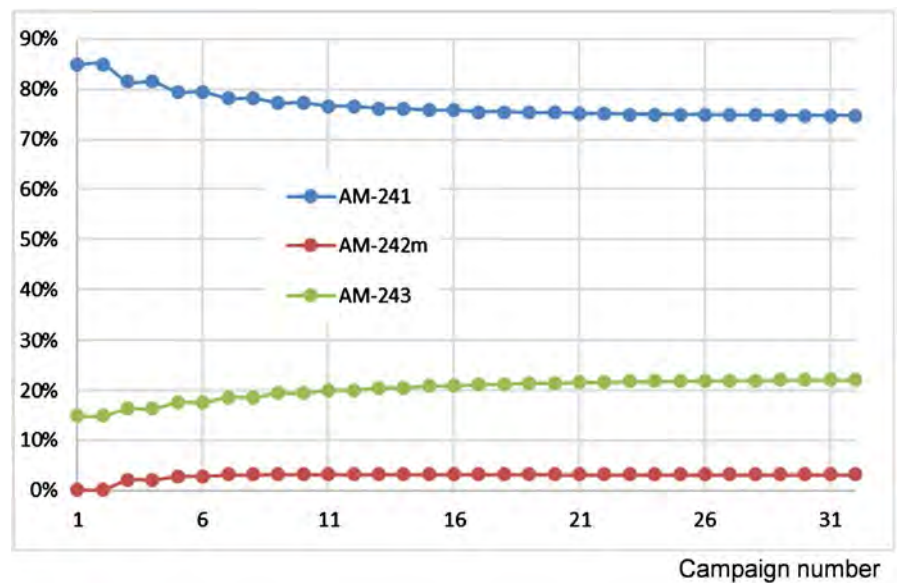
In the steady state, burning (conversion into fission products) and removal (conversion to other isotopes of transuranic elements) of americium (in total



**Figure 4.** Reactor calculated model.



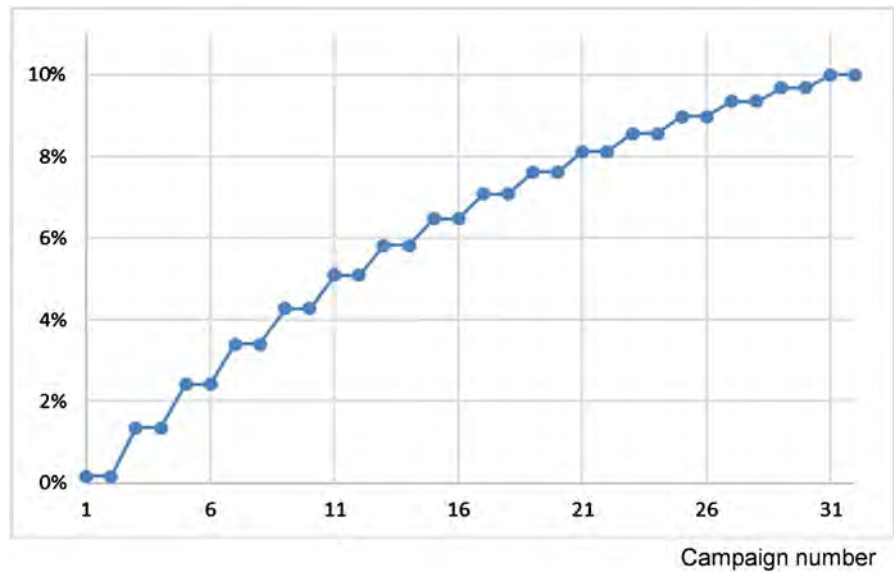
**Figure 5.** Americium consumption and production of curium, neptunium and plutonium in the first 32 reactor campaigns.



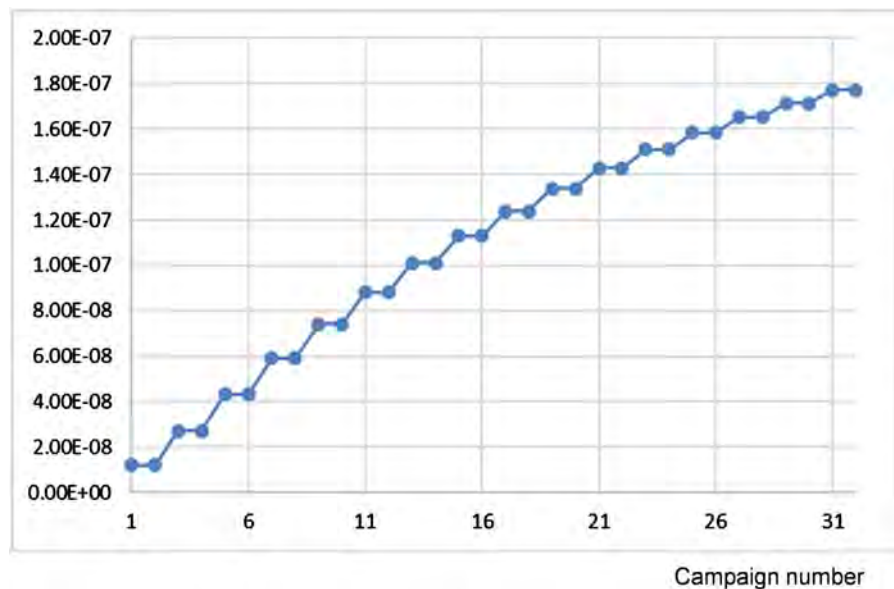
**Figure 6.** The relative content of americium isotopes in the fuel composition in the first 32 reactor campaigns.

transmutation) will amount to ~63 kg per campaign. Taking into account the first loads, into which 218 kg of americium is introduced, it will take about 2000 reactor-years to transmute the accumulated currently in the VVER americium reactors (about 20 tons) with one SVBR-100 reactor.

When is used of a modular NPP based on SVBR-100 reactor facility (RF) with a capacity of 1 GWe for americium transmutation, all the americium accumulated will be used for loading into the cores of SVBR-100 reactors 182 years after the NPP was put into operation (2.18 tons will be used to form the first loads of



**Figure 7.** The proportion of  $^{236}\text{U}$  in the total mass of uranium isotopes in the core loading for the first 32 campaigns.



**Figure 8.** The share of  $^{232}\text{U}$  in the total mass of uranium isotopes in spent nuclear fuel in the first 32 campaigns.

modules, 2.18 tons—to form the second loads, and taking into account the formation of loadings for the next 25 campaigns, the total mass of americium loaded into the cores of the reactors will reach 20.11 tons)—that is, the accumulated amount of americium will be utilized. For transmutation of the indicated amount of americium over the designated service life of the SVBR-100 reactor (60 years), the total capacity of the power units of modular NPPs based on the SVBR-100 reactor will be 3 GWe.

In the equilibrium operating mode, the production of plutonium in one reactor per campaign will be ~307 kg, the share of  $^{238}\text{Pu}$  in the produced plutonium

**Table 1.** Masses of actinides when loading and unloading fuel in the first campaign and in steady state.

Isotope Symbol	Half-life $T_{1/2}$	First campaign			Steady state		
		Loading, kg	Unloading, kg	Difference, kg	Loading, kg	Unloading, kg	Difference, kg
U-232	68.9 years	0.000E+00	3.641E-05	3.641E-05	1.890E-03	1.373E-03	-5.167E-04
U-233	1.592E+5 лет	0.000E+00	4.386E-04	4.386E-04	2.203E-03	1.985E-03	-2.179E-04
U-234	2.455E+5 years	1.406E+01	1.255E+01	-1.502E+00	1.919E+01	1.695E+01	-2.236E+00
U-235	7.04E+8 years	1.529E+03	9.708E+02	-5.578E+02	1.529E+03	9.828E+02	-5.458E+02
U-236	2.342E+7 years	1.406E+01	1.209E+02	1.068E+02	1.173E+03	1.173E+03	3.000E-01
U-238	4.468E+9 years	7.398E+03	6.935E+03	-4.635E+02	6.225E+03	5.838E+03	-3.867E+02
PU-236	2.858 years	0.000E+00	7.571E-05	7.571E-05	0.000E+00	6.330E-04	6.330E-04
PU-237	45.64 days	0.000E+00	5.846E-05	5.846E-05	0.000E+00	7.053E-05	7.053E-05
PU-238	87.7 years	0.000E+00	2.603E+01	2.603E+01	0.000E+00	3.072E+01	3.072E+01
PU-239	24110 years	0.000E+00	3.067E+02	3.067E+02	0.000E+00	2.560E+02	2.560E+02
PU-240	6561 years	0.000E+00	1.519E+01	1.519E+01	0.000E+00	1.256E+01	1.256E+01
PU-241	14.329 years	0.000E+00	4.962E-01	4.962E-01	0.000E+00	3.993E-01	3.993E-01
PU-242	3.75E+50 years	0.000E+00	6.736E+00	6.736E+00	0.000E+00	5.773E+00	5.773E+00
PU-244	8.13E+7 years	0.000E+00	1.815E-03	1.815E-03	0.000E+00	2.617E-03	2.617E-03
NP-235	396.1 days	0.000E+00	5.314E-08	5.314E-08	0.000E+00	4.909E-07	4.909E-07
NP-236	153E+3 years	0.000E+00	5.968E-05	5.968E-05	0.000E+00	5.885E-04	5.885E-04
NP-237	2.144E+6 years	0.000E+00	8.354E+00	8.354E+00	0.000E+00	7.400E+01	7.400E+01
NP-239	2.356 days	0.000E+00	1.197E-04	1.197E-04	0.000E+00	1.152E-04	1.152E-04
AM-241	432.6 years	1.853E+02	1.240E+02	-6.133E+01	1.625E+02	1.099E+02	-5.262E+01
AM242M	141 years	2.180E-01	4.801E+00	4.583E+00	6.916E+00	7.093E+00	1.769E-01
AM-243	7364 years	3.248E+01	2.607E+01	-6.416E+00	4.860E+01	3.921E+01	-9.395E+00
CM-240	27 days	0.000E+00	1.747E-10	1.747E-10	0.000E+00	1.517E-10	1.517E-10
CM-241	32.8 days	0.000E+00	1.948E-07	1.948E-07	0.000E+00	1.694E-07	1.694E-07
CM-242	162.8 days	0.000E+00	3.004E+00	3.004E+00	0.000E+00	2.599E+00	2.599E+00
CM-243	29.1 years	0.000E+00	2.382E-01	2.382E-01	0.000E+00	1.975E-01	1.975E-01
CM-244	18.11 years	0.000E+00	4.327E+00	4.327E+00	0.000E+00	6.239E+00	6.239E+00
CM-245	8423 years	0.000E+00	1.989E-01	1.989E-01	0.000E+00	2.789E-01	2.789E-01
CM-246	4706 years	0.000E+00	5.177E-03	5.177E-03	0.000E+00	7.048E-03	7.048E-03
CM-247	1.56E+7 years	0.000E+00	8.112E-05	8.112E-05	0.000E+00	1.065E-04	1.065E-04
CM-248	3.48E+5 years	0.000E+00	2.299E-06	2.299E-06	0.000E+00	2.927E-06	2.927E-06

will be ~10%, which will determine the heat release in plutonium of 0.56 kW/kg  $^{238}\text{Pu}$ . The mass of curium isotopes in SNF at the end of each campaign will be 9.3 kg (specific heat release is 36 kW/kg for the composition of curium produced in the considered reactor  $^{242}\text{Cm}/^{243}\text{Cm}/^{244}\text{Cm} = 28\%/2\%/67\%$ ), the mass produced for a neptunium per one campaign is ~75 kg.

The fraction of  $^{232}\text{U}$  in the equilibrium vector of uranium isotopes will be  $\sim 2.3 \times 10^{-7}$ . With the operation of the SVBR reactor in the nuclear fuel cycle using MOX-fuel with make-up depleted uranium, the fraction of  $^{232}\text{U}$  after 9 recycles reaches  $3 \times 10^{-7}$ . It should be noted that in both the first and second cases, the content of  $^{232}\text{U}$  significantly exceeds the limit content given in Ref. [6] (with  $^{232}\text{U}/^{235}\text{U} < 1.1 \times 10^{-7}$ ), which determines the possibility of removing  $^{232}\text{U}$  from the regenerate uranium during gas centrifugal separation of its isotopes.

This limitation is due to the fact that  $^{232}\text{U}$ , which is an  $\alpha$ -emitter with a half-life of about 70 years, has  $^{228}\text{Th}$  as a daughter nucleus with a half-life of 1.9 years, at the end of the chain of radioactive transformations of which, short-lived  $^{208}\text{Tl}$  is formed, without long delay, which is a source of hard gamma radiation with an energy of gamma rays of 2.6 MeV. The same reason makes it difficult to handle fresh recycled fuel in the CNFC of  $\text{Th}$ - $^{233}\text{U}$ , in which  $^{232}\text{U}$  is accumulated in large quantities. The importance of  $^{232}\text{U}$  for its removal from uranium regenerate can be significantly reduced if the storage time between its chemical separation from SNF and its transfer to the separation is reduced to a value significantly less than 1.9 years.

The rate of production of americium by thermal reactors (TR) and, accordingly, the number of FRs for its transmutation can be significantly reduced by decreasing the time of storing SNF before reprocessing, since  $^{241}\text{Am}$  is a daughter product of  $^{241}\text{Pu}$ , which has a half-life about 14 years. In this case, the separated plutonium will be used in the CNFC of two-component NP as fuel component in a mixed fuel of FRs. Estimates show that when the VVER SNF storage time is reduced from 15 to three years, the required number of transmuting reactors is halved. To realize this possibility, it will require the development and industrial mastered of more complex methods for reprocessing SNF due to its higher radioactivity and, accordingly, heat release. At the same time, a reduction in SNF storing time will lead to an improvement in the plutonium isotope vector due to an increase in  $^{241}\text{Pu}$  content.

It should be noted that the rate of americium transmutation in the SVBR-100 reactor can be increased in proportion to the increase in the amount of americium loaded into the core (in Ref. [3], to the BN-800 reactor, the content of americium was considered up to 16%, h.a.). However, the realization of this possibility is associated with the need to solve the problem of cooling spent fuel assemblies when they are unloaded from the reactor and loading fresh fuel assemblies with a high content of americium.

If we define the efficiency coefficient of americium transmutation ( $\text{ECT}_{\text{Am}}$ ) as a percentage ratio of the amount of incinerated (fission plus capture) americium (all isotopes) to the amount of burned out (fission plus capture)  $^{235}\text{U}$ , then the value of  $\text{ECT}_{\text{Am}}$  for the SVBR-100 reactor on uranium oxide fuel will be, as can be obtained from the data listed in the table, about 11 percent. That is, for the transmutation of 20 tons of americium, it is necessary to burn 220 tons of  $^{235}\text{U}$ . At the same time, about 140 tons of plutonium will be returned to the NFC of

large-scale nuclear power over the same period. For FRs operating on MOX-fuel, for which the breeding ratio is close to unity (BN-800) or slightly larger than unity (SVBR-100), such a definition of  $ECT_{Am}$  loses its meaning, since the amount of plutonium is practically does not decrease or even slightly increases.

As the estimates showed, the effective share of delayed neutrons in the SVBR-100 reactor with the addition of americium varies slightly compared to the base case. In the basic version, the effective share of delayed neutrons is 0.71% at the beginning of the campaign and 0.58% at the end of the campaign. For the SVBR-100 supplemented with 218 kg of americium in the first campaign, the effective share of delayed neutrons is 0.69% at the beginning of the campaign and 0.57% at the end of the campaign. In the established regime, the effective share of delayed neutrons at the beginning of the campaign is 0.66%, and at the end of the campaign, 0.56%. It can be expected that other safety characteristics will not change significantly due to the small proportion of americium added to the core.

It should be noted that in order to ensure the operation of transmutation reactors with their on-station CNFC (to reduce the volume of transportation), more complex and “dirty” CNFC, it will be necessary to create a transport infrastructure for transporting separated MA from the CNFC facilities to the transmutation fast reactors sites, and vice versa transportation of plutonium extracted from SNF of transmutation fast reactors to the CNFC enterprises of large-scale nuclear power. These issues require a separate study.

#### 4. Conclusions

It was shown in Ref. [3] that 82 kg of americium can be transmuted in the BN-800 reactor operating on MOX-fuel per year, including 71 kg in the core with its fuel content of about 2.5% and 11 kg in the side breeding zone. Thus, one BN-800 reactor incinerates americium produced by 2.8 VVER-1000 reactors, or in terms of the same power, the BN reactor with a capacity of 1 GWe incinerates americium generated per year from 3.5 VVER-1000 reactors. At the same time, however, a very small part of americium is converted into fission products, and its main share is converted to  $^{242}Cm$  with a specific heat release of 36 kW/kg compared with 0.08 kW/kg for  $^{241}Am$  and 0.9 kW/kg for  $^{244}Cm$ . Bearing in mind that the half-life of  $^{242}Cm$  is about 165 days, its high content in the composition of curium isotopes will make it difficult to reduce the storage time of spent nuclear fuel before its reprocessing with the separation of MA and organize storage of the separated curium until it decays into plutonium isotopes.

In the previously considered variant of SVBR-100 using MOX fuel (Ref. [2]) for the campaign 76,000 eff. hours (10.6 years) 53 kg of MA is transmuted when the MA content in the fuel is 2.5%, which gives a transmutation rate of ~5 kg/year. Thus, SVBR-100 reactors using MOX fuel with a total capacity of 1 GWe can incinerate americium, which is produced per year, from 1.7 VVER-1000 reactors.

The variant of the SVBR-100 reactor based on uranium oxide fuel containing about 2.5% americium considered in this work, while working in the CNFC ac-

According to the described algorithm, allows to incinerate 63 kg of americium for 50,000 effective hours or 9 kg per year. Thus, SVBR-100 reactors with a total capacity of 1 GWe can incinerate americium produced by three VVER-1000 reactors per year. The comparison shows that the transmutation of americium in the core of SVBR-100 with uranium fuel is 1.8 times more efficient than in the core with MOX-fuel and 14 percent lower than in the BN-800 reactor.

Since  $^{241}\text{Am}$  is a product of  $^{241}\text{Pu}$  radioactive decay, with a half-life of about 14 years, the amount of americium generated during storage of SNF of VVER reactors and requiring transmutation can be significantly reduced by reducing the storing time of VVER SNF before reprocessing. Accordingly, the required number of transmutation fast reactors can be reduced.

To realize this opportunity, it will require the development and industrial development of “dry” SNF reprocessing methods, in which there is no radiolysis and hydrogen release inherent in hydro-chemical reprocessing, and restrictions on the amount of fissile material in the reprocessing apparatus are reduced due to the absence of a moderator neutrons.

Since americium differs from other transplutonium elements in the increased volatility that occurs during sintering of fuel pellets, it seems reasonable to use the vibrapack technology of manufacturing of uranium dioxide as a matrix containing the required amount of granular americium oxide (Ref. [7]).

Thus, the problem of handling minor actinides has a multifactorial nature, and its solution requires a systematic approach, taking into account the problems involved, performing technical and economic investigations of various transmutation options in the frame of the development strategy of a two-component nuclear energy system.

## Conflicts of Interest

The authors declare no conflicts of interest regarding the publication of this paper.

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## Abbreviation

BN	sodium fast reactor (in Russia)
CNFC	closed nuclear fuel cycle
ECT	efficiency coefficient transmutation
FPS	fission products
FR	fast reactor
FA	fuel assembly
HLMC	heavy liquid metal coolant
LBE	lead-bismuth eutectic
MA	minor actinides
MOX	mixed oxide uranium plutonium
NP	nuclear power
NPP	nuclear power plant
RAW	radioactive waste
SFR	sodium fast reactor
SNF	spent nuclear fuel
SVBR	lead-bismuth cooled fast reactor
TR	thermal reactor
VVER	pressurized water reactor (in Russia)

# Upward Transition Probabilities $B(E2) \uparrow$ Properties Study of Even-Even $^{104-114}\text{Ru}$ Nuclei

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**How to cite this paper:** Islam, T., Amin, R., Alam, Md.A. and Islam, J. (2020) Upward Transition Probabilities  $B(E2) \uparrow$  Properties Study of Even-Even  $^{104-114}\text{Ru}$  Nuclei. *World Journal of Nuclear Science and Technology*, 10, 129-137.

<https://doi.org/10.4236/wjnst.2020.103012>

**Received:** April 17, 2020

**Accepted:** July 19, 2020

**Published:** July 22, 2020

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## Abstract

In this research work, the upward transition probabilities for the transition levels,  $0^+ \rightarrow 2^+$ ,  $2^+ \rightarrow 4^+$ ,  $4^+ \rightarrow 6^+$  and  $6^+ \rightarrow 8^+$  levels of even-even neutron rich  $^{104-114}\text{Ru}$  isotopes have been calculated by using the Global Best Fit (GBF) method. In addition, the associated parameters such as, Quadrupole moment and Deformation parameter of even-even  $^{104-114}\text{Ru}$  have been calculated. The dependency of these nuclear parameters shows the nuclear magic number tendency.

## Keywords

Upward Transition Probability, Quadrupole Moment, Deformation Parameter, Global Best Fit (GBF) Method

## 1. Introduction

In nuclear physics, the most important part of interest is the shape of nucleus. The ground state shape of nucleus is spherical and it may deviate from this shape, which is closely related to nuclear “magic numbers”. We will consider nuclear “magic numbers” and their evolution along the nuclear chart. In stable nuclei, large gaps exist between nuclear shells when the proton or neutron number is equal to 2, 8, 20, 28, 50, 82 and 126 [1] [2]. These gaps result in large transition energy values between the ground and first excited states, relatively low quadrupole moment and small neutron capture cross sections. The “magic numbers” and their values are not preserved; they evolve for unstable nuclei due to nuclear structure effects. Therefore, nuclear properties of the first excited  $2^+$  states in even-even nuclei provide important information on the evolution of nuclear properties and shell model studies. The electric quadrupole reduced transition probabilities are important for the nuclear structural information. In

recent years, the electric quadrupole reduced transition probabilities of even-even neutron rich nuclei  $^{102-112}\text{Pd}$  [3] [4],  $^{104-112}\text{Cd}$  [5] and  $^{100-102}\text{Ru}$  [6] have been studied using Interacting Boson Model-1 (IBM-1). This model (IBM-1) was developed by Iachello and Arima and the associated quadrupole moment, deformation parameter was studied [3] [4]. All these nuclei studied the downward reduced transition probabilities using the IBM-1. In 1999, excitation energies,  $E2$  transition probabilities, quadrupole excitation properties for even-even  $^{104-114}\text{Ru}$  and  $^{106-110}\text{Pd}$  nuclei have been studied [7]. After this study, Skyrme force SLy4 for even-even  $^{94-110}\text{Ru}$  nuclei has been investigated using the Hartree-Fock-Bogoliubov (HFB) method [8] and the triaxial shapes properties for the even-even neutron-rich  $^{106-108}\text{Mo}$  and  $^{108-112}\text{Ru}$  nuclei have been also investigated [9]. Therefore, we have focused the even-even  $^{104-114}\text{Ru}$  isotopes' properties for the transition levels  $0^+ \rightarrow 2^+$ ,  $2^+ \rightarrow 4^+$ ,  $4^+ \rightarrow 6^+$  and  $6^+ \rightarrow 8^+$ . For this purpose, we have used the GBF model to investigate the basic information of the  $^{104-114}\text{Ru}$  nuclei because this model describes three basic properties: mass and energy dependency with  $\gamma$ -ray transition probability; localization and emphasis for the anchor nucleus; regionalized by the magic number [10].

Using GBF method, we have calculated the upward transition probabilities,  $B(E2) \uparrow$  of neutron-rich even-even  $^{104-114}\text{Ru}$  nuclei. Associated parameters like quadrupole moment ( $Q_0$ ) and deformation parameter ( $\beta_2$ ) have also been estimated. The study also reveals the effects of the estimated parameters on the nuclear structure. This also reveals how the estimated parameters affect the structure of the nucleus. This method presents the near magic number  $N = 50$  region for the  $^{104}\text{Ru}$  nucleus. In this paper, energy and mass dependencies have been showed with the calculated  $B(E2) \uparrow$  values. The relationship among  $B(E2) \uparrow$ ,  $Q_0$  and  $\beta_2$  are also given in graphical. Finally, this paper is arranged as: GBF model has been described in 2.1 Section.  $B(E2) \uparrow$ ,  $Q_0$  and  $\beta_2$  are discussed in Section 2.2, 2.3 and 2.4 respectively.

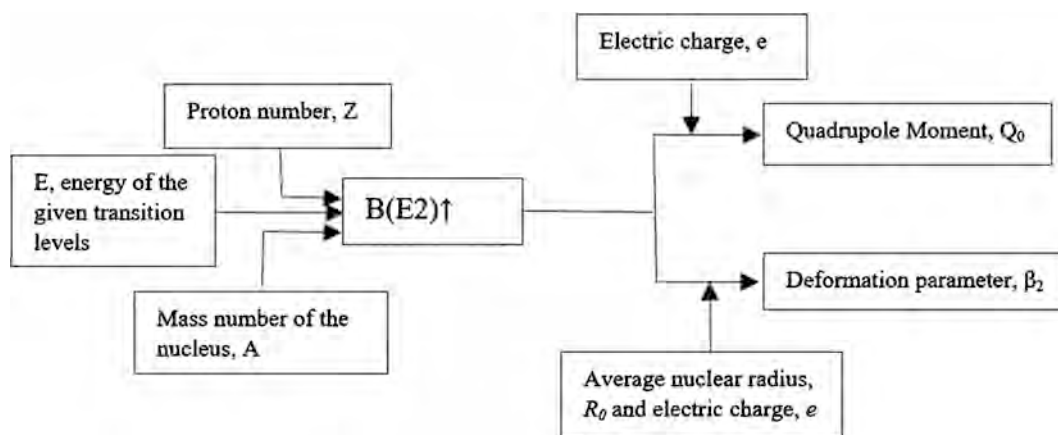
## 2. Theory

In this section we describe the procedure used to compute the electric quadrupole reduced transition probabilities and the corresponding electric quadrupole moment and deformation parameter. The procedure summary of this theory is described in the following flowchart (Figure 1):

### 2.1. Global Best Fit (GBF) Method

According to the Global Best Fit Method, a knowledge of the energy  $E$  (Kev) of the  $2_1^+$  state is all that is required to make a prediction for the corresponding mean life time for the  $\gamma$ -ray,  $\tau_\gamma$  (in ps) and hence, the  $B(E2) \uparrow$  ( $e^2\text{b}^2$ ) value. Within the framework of the hydrodynamic model with irrotational flow, Bohr and Mottelson [11] have derived simple expressions for the  $\tau_\gamma$  value is given by

$$\tau_\gamma \approx 0.6 \times 10^{14} E^{-4} Z^{-2} A^{1/3} \quad (1)$$



**Figure 1.** Flowchart of the given procedure.

For small harmonic vibrations of spherical nucleus, the  $\tau_\gamma$  value is,

$$\tau_\gamma \approx 1.4 \times 10^{14} E^{-4} Z^{-2} A^{1/3} \quad (2)$$

For collective rotations of axially symmetric nuclei. The  $E^{-4}Z^{-2}$  dependence in the above expressions was adopted by Grodzins [12] in his empirical fits (for all even-even nuclei), but he replaced  $A^{1/3}$  with  $A$ . When the exponents of  $E$  and  $A$  were allowed to vary, we found earlier [10] [13] that the best global fit to the data [14] was obtained by

$$\tau_\gamma = 1.25 \times 10^{14} E^{-4} Z^{-2} A^{0.69} \quad (3)$$

Hence,  $\tau_\gamma$  and  $B(E2) \uparrow$  is related by the equation [7],

$$\tau_\gamma = 40.81 \times 10^{13} E^{-5} [B(E2) \uparrow / e^2 b^2]^{-1} \quad (4)$$

When converted to  $B(E2) \uparrow$ , this expression led to

$$B(E2) \uparrow = 3.26 E^{-1} Z^2 A^{-0.69} \quad (5)$$

We also showed that, the  $1/E$  dependence is more important than the exact  $A$  dependence. If the exponent of  $A$  is fixed as  $-2/3$  (instead of  $-0.69$ ), the revised best fit to the data was found [15] to be,

$$B(E2) \uparrow = 2.6 E^{-1} Z^2 A^{-2/3} \quad (6)$$

Here,  $B(E2) \uparrow$  is the electric quadrupole transition probability,  $E$  is the excitation energy,  $Z$  is the atomic number and  $A$  is the mass number.

## 2.2. Electric Quadrupole Reduced Transition Probability

The upward transition probabilities  $B(E2) \uparrow$  is the transition of a particle from lower energy state to higher energy state [10]. It can be calculated by using the following equation obtained from GBF method,

$$B(E2; l_i \rightarrow l_f) \uparrow = 2.6 E^{-1} Z^2 A^{-2/3} \quad (7)$$

Here,  $l_i$  is the lower energy state and  $l_f$  is the higher energy state and the subscript  $i$  and  $f$  indicate the initial and final respectively.

### 2.3. Electric Quadrupole Moment

The nuclear electric intrinsic quadrupole moment is a parameter which describes the nuclear charge distribution. A non-zero intrinsic quadrupole moment  $Q_0$  indicates that the charge distribution is not spherically symmetric. By convention the value of  $Q_0$  is taken to be positive ( $Q_0 > 0$ ) if the ellipsoid is prolate and negative ( $Q_0 < 0$ ) if it is oblate [16]. Intrinsic quadrupole moment,  $Q_0$  is related to the electric quadrupole transition probabilities  $B(E2) \uparrow$ , calculated by the following equation [17]

$$Q_0 = \left[ \frac{16\pi B(E2) \uparrow}{5e^2} \right]^{1/2} \quad (8)$$

Here,  $Q_0$  measured in barn (b) unit.

### 2.4. Deformation Parameter

Deformation Parameter is the parameter which measures the elongation of the axially symmetric shape of a deformed nucleus from its spherical shape. Deformation parameter is denoted by  $\beta_2$  which is, related to  $B(E2) \uparrow$ , calculated by the equation [17]

$$\beta_2 = \left( 4\pi/3 ZR_0^2 \right) \left[ B(E2) \uparrow / e^2 \right]^{1/2} \quad (9)$$

Here,  $R_0$  is the average radius nuclear which can be obtained from the following equation,

$$R_0^2 = 0.0144A^{2/3} \text{ barn (b)}. \quad (10)$$

## 3. Results and Discussion

The values of the  $E$ ,  $R_0$ ,  $B(E2) \uparrow$ ,  $\beta_2$  and  $Q_0$  for the even-even  $^{104-114}\text{Ru}$  nuclei are given in **Table 1**. For the even-even  $^{104-114}\text{Ru}$  nuclei,  $E$  has been obtained from the references [18]-[23] and other values  $B(E2) \uparrow$ ,  $Q_0$ ,  $\beta_2$  and  $R_0$ , have been obtained using the Equations (7)-(10) respectively. Using these values, **Figures 2-6** have been drawn in below where their relations and behaviors have been discussed.

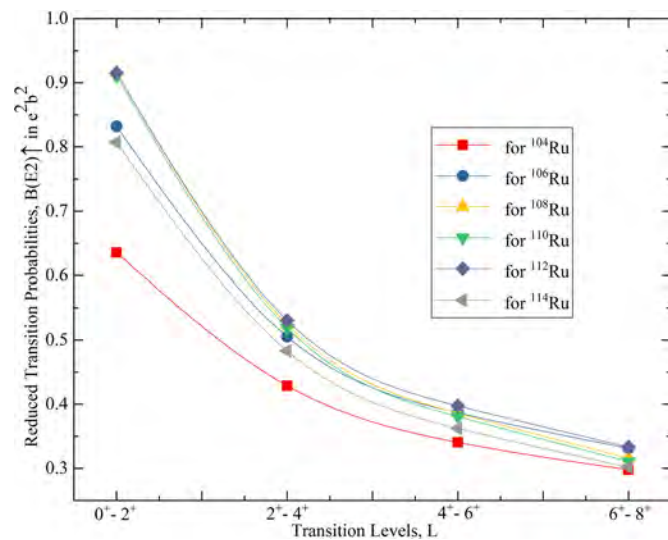
Electric quadrupole transition probabilities  $B(E2) \uparrow$  is drawn as a function of transition levels for even-even  $^{104-114}\text{Ru}$  nuclei in **Figure 2**. This shows the decrease of  $B(E2) \uparrow$  with increasing the transition levels for each nucleus. For the  $0^+ - 2^+$  transition level, transition probability is higher than that of the other transition levels for each nucleus. In this level, transition probability for the  $^{104}\text{Ru}$  is the lowest than the other nuclei.

Deformation parameter is drawn as a function of  $B(E2) \uparrow$  in **Figure 3**. This figure shows the transition probability increasing with the increase of deformation parameter. For  $^{104-114}\text{Ru}$  nuclei, deformation parameter change follows the almost linear relationship with respect to the transition levels.

Quadrupole moment variations with the change of  $B(E2) \uparrow$  are shown in **Figure 4**. **Figure 4** shows that quadrupole moments increase linearly with the

**Table 1.** The electric quadrupole transition probabilities, deformation parameters and quadrupole moments of  $^{104-114}\text{Ru}$  isotope.

Nuclei	Transition Level, $I_i \rightarrow I_f$	Energy, $E$ in KeV [18]-[23]	Average Radius, $R_0^2$ (b)	Upward Transition, $B(E2) \uparrow$ ( $e^2b^2$ )	Deformation parameter, $\beta_2$	Quadrupole Moment, $Q_0$ (b)
$^{104}\text{Ru}$	$0^+ - 2^+$	358.02	0.31845	0.6357	0.2383	2.5279
	$2^+ - 4^+$	530.46		0.4290	0.1958	2.0767
	$4^+ - 6^+$	668.12		0.3407	0.1744	1.8507
	$6^+ - 8^+$	764		0.2979	0.1631	1.7305
$^{106}\text{Ru}$	$0^+ - 2^+$	270.07	0.32252	0.8321	0.2692	2.8922
	$2^+ - 4^+$	444.63		0.5054	0.2098	2.254
	$4^+ - 6^+$	581.1		0.3867	0.1835	1.9716
	$6^+ - 8^+$	677.6		0.3316	0.1699	1.8258
$^{108}\text{Ru}$	$0^+ - 2^+$	242.24	0.32657	0.9162	0.279	3.0349
	$2^+ - 4^+$	422.96		0.5247	0.2111	2.2967
	$4^+ - 6^+$	574.8		0.3861	0.1811	1.9701
	$6^+ - 8^+$	701.6		0.3163	0.1639	1.7831
$^{110}\text{Ru}$	$0^+ - 2^+$	240.73	0.33059	0.9107	0.2748	3.0257
	$2^+ - 4^+$	422.62		0.5188	0.2074	2.2837
	$4^+ - 6^+$	575.75		0.3808	0.1777	1.9565
	$6^+ - 8^+$	705.4		0.3108	0.1605	1.7676
$^{112}\text{Ru}$	$0^+ - 2^+$	236.66	0.33458	0.9153	0.2722	3.0334
	$2^+ - 4^+$	408.24		0.5305	0.2072	2.3093
	$4^+ - 6^+$	545		0.3974	0.1793	1.9987
	$6^+ - 8^+$	649.5		0.3335	0.1643	1.831
$^{114}\text{Ru}$	$0^+ - 2^+$	265.19	0.33855	0.8073	0.2526	2.8488
	$2^+ - 4^+$	443.01		0.4832	0.1954	2.204
	$4^+ - 6^+$	590.6		0.3625	0.1693	1.9089
	$6^+ - 8^+$	709.1		0.3019	0.1545	1.7421

**Figure 2.** The change of reduced transition probabilities with transition levels.

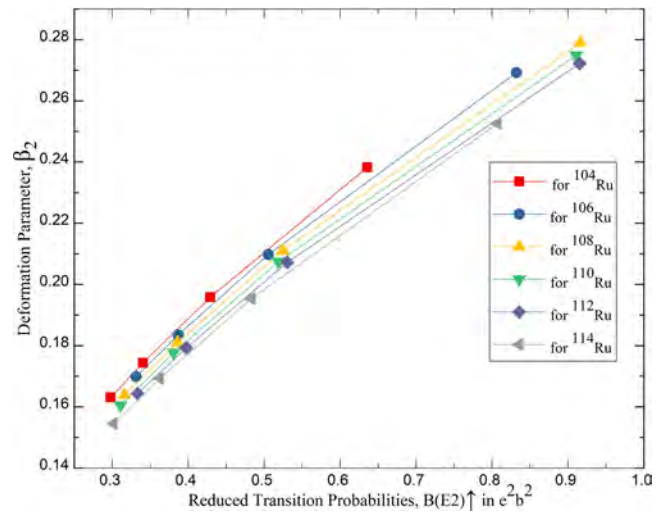


Figure 3. Deformation parameter change with respect to the reduced transition probabilities.

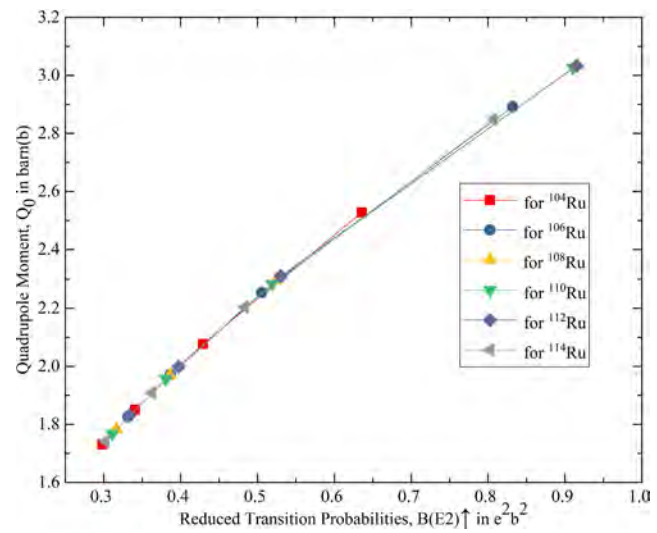


Figure 4. Quadrupole moment variations with the change of reduced transition probabilities.

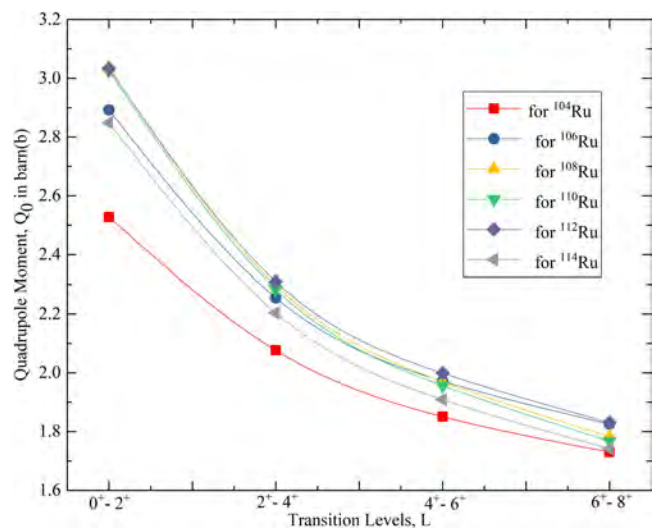
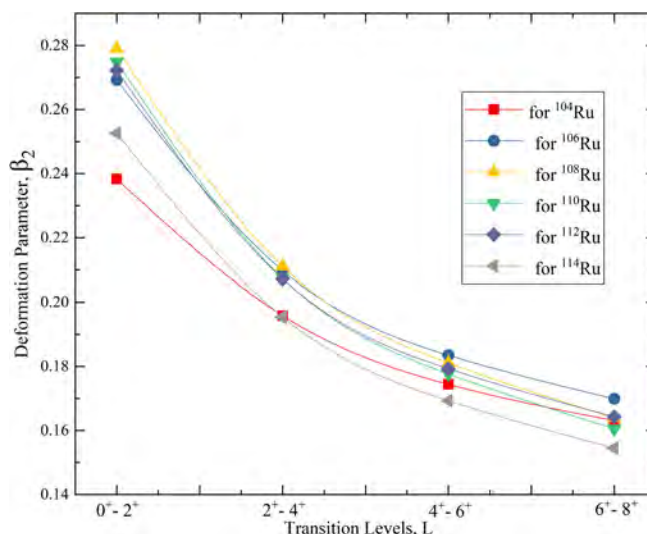


Figure 5. Quadrupole moment change with respect to the transition levels.



**Figure 6.** The change of deformation parameter against the transition levels.

transition probabilities increasing. In the figure, we see that quadrupole moment is lower for the isotope <sup>104</sup>Ru.

Quadrupole moments are represented with the variation of transition levels for even-even <sup>104-114</sup>Ru nuclei in **Figure 5**. In this figure, we see the quadrupole moments decrease with increasing the transition levels for each nucleus and it reaches in the range 1.8 - 3.0 barn (b).

**Figure 6** shows the deformation parameters variation with respect to the transition levels for these even-even nuclei. From this figure, we can conclude that at first transition level 0<sup>+</sup> - 2<sup>+</sup> the deformation of the nucleus shape will be maximum for each nucleus, gradually the deformations decrease for upper transition levels.

#### 4. Conclusion

It is seen from the data and corresponding graphs, when the transition levels of any nuclei increased the electric quadrupole moment and reduced transition probabilities  $B(E2) \uparrow$  of the given nuclei are decreased. The deformation parameters also decrease with increasing transition levels. It concludes from the data and corresponding graph, the transition probabilities, quadrupole moment, and deformation parameters have comparatively lower values for the isotopes which have neutron number close to magic number 50.

#### Conflicts of Interest

The authors declare no conflicts of interest regarding the publication of this paper.

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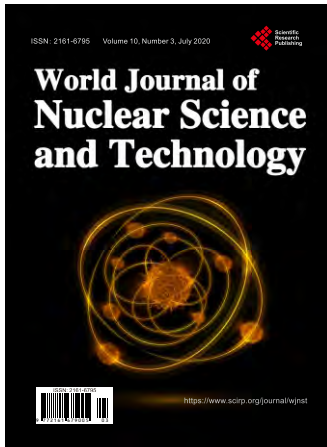
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## Acronyms

### Acronyms' Name

### Acronyms' Symbol

Upward transition probabilities	$B(E2) \uparrow$
Global Best Fit	GBF
Quadrupole moment	$Q_0$
Deformation parameter	$\beta_2$
Interacting Boson Model-1	IBM-1
Mean life time for the $\gamma$ -ray	$\tau_\gamma$
Energy	$E$
Proton number	$Z$
Nuclear mass number	$A$
Initial energy level	$I_i$
Final energy level	$I_f$
Electric charge	$e$
Nuclear average radius	$R_0$



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