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BASE – RESEARCH REPORT ON NUCLEAR SAFETY

Analysis and Evaluation of the Development Status, Safety and Regulatory Framework for So-Called Novel Reactor Concepts

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So-called “novel” reactor concepts

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ADS	accelerator-driven system
AEC	Atomic Energy Commission
AGR	Advanced Gas-cooled Reactor
AMR	Advanced Modular Reactor
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANT	Advanced Nuclear Technology
ARIS	Advanced Reactor Information System
ASME	American Society of Mechanical Engineers
ASN	French regulator (Autorité de sûreté nucléaire)
ASTRID	Advanced Sodium Technological Reactor for Industrial Demonstration
BEIS	Department for Business, Energy and Industrial Strategy
BfS	Federal Office for Radiation Protection (Bundsamt für Strahlenschutz)
BWR	boiling water reactor
CANDU	Canadian pressurised heavy-water reactor (Canada Deuterium Uranium)
CEA	Commission for Atomic Energy and Alternative Energies (Commissariat à l’énergie atomique et aux énergies alternatives)
CGNPC	China General Nuclear Power Group
CNS	Convention on Nuclear Safety
CNSC	Canadian Nuclear Safety Commission
COL	combined building and operating licence (“combined license”)
CP	construction permit
CR	conversion ratio
DBA	design basis accident

DCA	Application for design approval in the USA (design certification application)
DESNZ	Department for Energy Security and Net Zero
DiD	Defense in Depth (Concept of multiple safety layers)
DoD	United States Department of Defense
DoE	United States Department of Energy
EBR	Experimental Breeder Reactor
ENISS	European Nuclear Installations Safety Standards Initiative
EPFL	École Polytechnique Fédérale De Lausanne
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling Water Reactor
ESK	Nuclear Waste Management Commission (Entsorgungskommission)
EVA	external impacts (“Einwirkungen von außen”)
FBR	fast breeder reactor
EFIT	European Facility for Industrial Transmutation of Minor Actinides
FFTF	Fast Flux Test Facility
FHR	fluoride salt-cooled high-temperature reactor
FLiBe	molten salt of the chemical composition LiF-BeF2
FLiNaK	molten salt of the chemical composition LiF-NaF-KF
FOAK	first of a kind
FR	fast reactor
FSAR	final safety analysis report
GA	General Atomics
GBWR	graphite moderated boiling water reactor
GCR	gas-cooled, graphite-moderated reactor
GDA	generic design assessment

GEH	GE Hitachi Nuclear Energy
GFR	gas-cooled fast reactor
GRAPA	Graphite Processing Approaches
GRS	Institute for Radiation Protection and Nuclear Safety (Gesellschaft für Anlagen und Reaktorsicherheit gGmbH)
HALEU	Uranium with an enrichment of < 20%, the internationally defined transition from low enriched uranium to highly enriched uranium “high-assay low enriched uranium)
HEPA	high-efficiency particulate air filter
HEU	highly enriched uranium
HLW	high-level radioactive waste
HTGR	high-temperature gas-cooled reactor
HWR	heavy-water reactor
IAEA	International Atomic Energy Agency
ILW	intermediate-level radioactive waste
IMF	inert matrix fuel
INES	International Nuclear and Radiological Event Scale
INL	Idaho National Laboratory
IRSN	Institute for Radiation Protection and Nuclear Safety (Institut de Radioprotection et de Sûreté Nucléaire)
ISR	Institute of Safety and Risk Sciences at the University of Natural Resources and Life Sciences, Vienna (Institut für Sicherheits- und Risikowissenschaften)
JAEA	Japan Atomic Energy Agency
LCOE	levelised cost of electricity
LEU	low-enriched uranium
LFR	lead-cooled fast reactor
LLFAP	long-lived fission and activation product

LMFR	liquid metal cooled fast reactor
LMR	liquid metal cooled reactor
LWGR	light water-cooled, graphite-moderated reactor
LWR	light-water reactor
MA	minor actinides (transuranium elements without plutonium)
MCSFR	Molten Chloride Salt Fast Reactor
MoU	memorandum of understanding
MOX	mixed uranium-plutonium oxide fuel
MSFR	molten-salt fast reactor
MSR	molten-salt reactor
MSRE	molten-salt reactor experiment
MWe	Megawatts electric
MWth	Megawatts thermal
NASA	National Aeronautics and Space Administration
NASEM	National Academies of Sciences, Engineering, and Medicine
NEA	Nuclear Energy Agency of the OECD
NOAK	Nth of a kind
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
OCC	overnight construction cost
OCRWM	Office of Civilian Radioactive Waste Management of the DOE
OECD	Organisation for Economic Co-operation and Development
ONR	Supervisory authority in the United Kingdom (Office for Nuclear Regulation)
OP	operating license
ORNL	Oak Ridge National Laboratory

P&T	partitioning and transmutation
PBMR	pebble-bed modular reactor
PHWR	pressurised heavy-water reactor
PSA	probabilistic safety assessment
PSAR	preliminary safety analysis report
PUREX	Plutonium URanium EXtraction
PWR	pressurized water reactor
RPV	reactor pressure vessel
REGDOC	regulatory document
SCWR	supercritical water-cooled reactor
SFR	sodium-cooled fast reactor
SKB	Svensk Kärnbränslehantering AB
SMR	“small modular reactor”; also: “small and medium-sized reactor”
SNF	spent nuclear fuel
SNL	Sandia National Laboratories
SNR	So-called “novel” reactor concepts
SSC	systems, structures and components
THOREX	thorium-uranium extraction
TNPP	transportable nuclear power plant
TRISO	triple-coated nuclear fuel pellet (TRistructural-ISOtropic)
TRL	technology readiness level
TRU	transuranium elements
TWR	travelling wave reactor
UCO	uranium oxycarbide
UOX	uranium oxide fuel

VDR	Concept review by the Canadian regulatory authority (vendor design review)
VHTR	very-high-temperature reactor
U.K.	United Kingdom of Great Britain and Northern Ireland
RP	Reprocessing plant
WENRA	Western European Nuclear Regulators’ Association
WIP	TU Berlin Workgroup for Economic and Infrastructure Policy
WNA	World Nuclear Association
WNN	World Nuclear News
WWER	water-cooled, water-moderated power reactor
ZNF	Centre for Natural Science and Peace Research at the University of Hamburg (Zentrum für Naturwissenschaft und Friedensforschung)

Summary

So-called "novel" reactor concepts (SNR) have been discussed, researched and developed for decades worldwide. The focus is often on six technology lines whose development is coordinated by the Generation IV International Forum (GIF), an international research network. There are also other developments that cannot be directly assigned to the GIF technology lines. These include, in particular, accelerator-driven systems. Within the technology lines, a large number of different specific reactor concepts are being pursued by different countries or companies.

According to their developers, SNR will have an advantage over today's operating nuclear power plants in terms of criteria such as fuel utilization, safety and reliability, economic efficiency, and nuclear non-proliferation. Likewise, reduced production of highly radioactive waste is put forward as a possible advantage, as is the use of such reactor concepts to manage radioactive waste (transmutation). In contrast to these announcements, such SNR, despite decades of development in some cases, have either not reached technological maturity or have not become established for commercial or safety reasons.

SNR are often referred to as a "fourth generation" of reactor concepts. In this context, the GIF speaks of a first generation of early prototype reactors, a second generation of large power reactors, and a third generation of advanced power reactors. The six different technology lines pursued in the GIF framework are grouped together as a fourth generation.

However, these technology lines differ in essential characteristics from the nuclear power plants in operation worldwide today (predominantly light-water reactors, LWR). To frame technological generations, classification of development stages within a technology line would be a more reasonable approach: for example, first experimental reactors, first demonstration reactors, first power reactors and further developed power reactors. With such a differentiated view of the research status within specific technology lines, one would have to speak for SNR of first- or second-generation reactor concepts, since typically only first experimental reactors or demonstration reactors have been built so far. For this reason, the “fourth-generation” terminology promoted by the GIF will not be used in the context of the investigation carried out here. Instead, we will continue to speak of so-called "novel" reactor concepts (SNR).[1]

In this study, relevant SNR are analysed and assessed with regard to various criteria. Of particular interest when assessing SNR are the technical development status of the concepts, questions of nuclear safety, questions of fuel supply and waste disposal, proliferation risks and economic aspects. Future developments of today's light and heavy water reactors are not the subject of this study. Furthermore, developments in the field of Small Modular Reactors (SMR) are not considered in depth.

Country-specific consideration

There are numerous SNR research and development activities and pilot projects worldwide. They are presented in detail for selected countries. Technology lines referred to today as "novel" were already researched and developed in the initial phase of nuclear technology developments in the 1940s and 1950s, above all the fast reactors, but also high-temperature reactors and molten salt reactors. Globally, the light-water reactor prevailed, while SNR ended up in the innovation-economic "valley of death", i.e., the inventions were not followed up by subsequent innovations and spread of

[1] Not to be confused with the acronym for the sodium-cooled fast reactor demonstration project SNR-300 developed in Germany in the 1970s and then discarded.

the technology. The path dependency thus produced for the systemic asset of light-water reactors suggests that in the short and medium term this technology line will remain dominant. SNR can be pursued in this setting as a niche technology at most.

Motives of countries to develop SNR include geopolitical and military aspects, decarbonization of the energy system, development of a so-called closed fuel cycle, including with regard to waste management, and innovation competition. In some countries, current SNR efforts represent an opportunity to re-stabilize the nuclear power innovation system after several unsuccessful attempts. This is the case, for example, in the U.S., where SNR have so far been underdeveloped and even light-water reactor construction has now virtually come to a halt. SNR are proving to be too demanding for potential entry-level countries, so that the construction of imported light-water reactors is predominant, e.g., in Poland and the United Arab Emirates.

The three nuclear superpowers (USA, Russia, China) have a common innovation dynamic: In their initial phases of nuclear power development, considerable resources were invested in the development of light-water and non-light-water reactors, but only the light-water reactors became commercially successful in further development.

The idea of the plutonium economy and the "closed fuel cycle" using fast reactors unites the innovation strategies in all countries under consideration. The USA abandoned this development in the 1970s due to a lack of diffusion prospects and the proliferation risks and was able to convince other countries to do so, for example, through bilateral agreements. In China, fast reactors are also being pursued in the SNR field as one prospect among others. Russia today has two fast reactors classified as commercial, but none of them is on the path to series maturity.

In the report the research activities of six selected countries were analysed in more detail, which can be summarized as follows:

The **U.S.** had been a world leader in reactor technology development since the 1950s through the Manhattan-Project. However, diffusion, both in the U.S. and internationally, has occurred only in light water reactors and not, as originally expected, in the other technology lines. With the widespread reduction of orders for light water reactor construction since the 1980s, U.S. nuclear power technology is in decline, which even the Energy Policy Act of 2005 has not been able to halt. The activities observed over the past decade or so to promote SMR and non-light water reactor concepts are an attempt to re-establish a claim to international technology leadership for U.S. nuclear power plant technology. At present, no commercial breakthrough is in sight.

In **Russia**, the focus during the early days of nuclear technology development was on reactors with fast neutron spectrum (SFR, later also LFR) in connection with reprocessing (Mayak pilot plant as well as fuel fabrication plant for uranium-plutonium mixed oxide fuels in Zheleznogorsk). Subsequently, this focus was deepened (BN-600, BN-800). Currently, the Russian innovation system is in a phase where the research infrastructure is aging (BOR-60, in operation since 1969) and projects are postponed (e.g., BN-1200), currently the BREST-OD-300 is prioritized. Russia maintains a long-term strategy of achieving a closed fuel cycle using fast neutron spectrum reactors and, in parallel, advancing the development of light water reactors.

China has advanced its nuclear innovation system through an import strategy since the 1960s. Following military developments in the 1950s, advances have been made in both light water reactors and SNR. SNR are being developed in parallel with the expansion of light water reactors. China has built up a wide range of technology lines, especially fast reactors and high-temperature reactors. Currently, the projects are still in the area of basic research or prototypes, a commercial roll-out is not yet foreseeable.

The Republic of Korea (South Korea) is one of the leading industrialized countries and has become one of the few suppliers of reactor technology, originally with the support of the U.S. the Republic of Korea has its own extensive commercial nuclear power program, which was also able to record exports in the 2000s. The country maintains particularly intensive relations with the U.S. with regard to research and development. In the field of SNR, the Republic of Korea is intensifying its participation in foreign, especially U.S. developments. In addition, the country is pushing ahead with its own developments, e.g., of reprocessing technologies in conjunction with fast reactors. Commercial use of SNR is not foreseeable at present.

Belgium, historically one of the first countries with commercial nuclear power plant use in the 1950s, has developed a small national innovation system since this initial phase. Belgium's activities for the development of SNR are focused on the development and internationalization of the MYRRHA research project, a combination of an accelerator-driven system (ADS) and a lead-bismuth-cooled fast reactor (LFR). Initial schedules and cost estimates have been exceeded and there are difficulties in financing the project.

In **Poland**, the entry into commercial nuclear power has been discussed for several decades, so far without success. Research on reactor technology has been conducted on a small scale since the 1950s, most notably at the MARIA research reactor (in operation since 1974). For SNR, it can be observed that Poland is building knowledge, with Polish scientists participating in European research projects. In particular, there is a focus on the development of high-temperature reactors, including considerations for the construction of a high-temperature gas-cooled research reactor (TeResa).

Overall, the country studies show that a system change from light-water reactors to SNR ready for series production is not foreseeable. Even the further development prospects for light-water reactors are to be assessed as very low due to the breakthrough of low-cost, renewable energy systems in combination with flexibility options.

Technology-specific consideration

For the purpose of this study, the SNR being pursued today are classified into seven different technology lines. These are Sodium-cooled Fast Reactors (SFR), Lead-cooled Fast Reactors (LFR), Gas-cooled Fast Reactors (GFR), Very High Temperature Reactors (VHTR), SuperCritical Water-cooled Reactors (SCWR), Molten Salt Reactors (MSR), and Accelerator-Driven Systems (ADS).

The technology lines studied were assessed in terms of the level of technological maturity achieved to date, security aspects, fuel supply and waste disposal aspects, proliferation issues, and costs.

Concepts for **Sodium-cooled Fast Reactors (SFR)** have been discussed since the beginning of the use of nuclear energy. The main development goal is the production of plutonium from the uranium fuel used and the further use of this plutonium as fissile material for energy production. In addition, its potential for partitioning and transmutation of radioactive waste nuclides (P&T) is often cited today. Therefore, potential use of SFR is always related to the development of plutonium-containing fuels (MOX) and reprocessing facilities for the spent fuel from both LWR and SFR.

As was the case for LWR, the development of SFR took place in three phases, from experimental reactors through prototype and demonstration reactors to commercial plants. The operating experience with SFR to date ranges from the occurrence of serious incidents and accidents, in some cases with years of downtime, to largely trouble-free operation over decades. In addition to a large number of plants that have already been shut down, three experimental SFR and three reactors classified by the International Atomic Energy Agency (IAEA) as commercial are still in operation. SFR, along with VHTR, are the most technically advanced of all SNR technology lines. Research and development efforts are therefore largely focused on optimizing economical and reliable operation and further enhancing safety. Furthermore, developments in the field of reprocessing and fuel fabrication for SFR are required.

With regard to safety, sodium as a coolant has the advantage that during normal operation it only leads to a small degree of corrosion of structural materials or cladding tubes. On the other hand, sodium is an opaque (non-transparent) coolant, so inspections and maintenance of the reactor are more complex and loading and unloading processes cannot be visually checked. Reactivity control is fundamentally more demanding for SFR than for LWR. Especially when plutonium fuels are used, fast reactors react very sensitively; this places special demands on the shutdown system to prevent an uncontrollable power increase. Overall, the higher reactivity control requirements for SFR represent an intrinsic disadvantage compared to today's LWR. Unlike LWR, the primary cooling circuit is not under high pressure. Large loss-of-coolant accidents are therefore less likely than in light-water reactors. On the other hand, a safety disadvantage of SFR compared to LWR is that the sodium coolant reacts exothermically on contact with water and ignites on contact with oxygen at high temperatures. Sodium fires have occurred repeatedly during operation of fast reactors, leading to operational failures.

Most SFR provide for the utilisation of MOX fuels. MOX fuels would also have to be shipped to a repository after use in the reactor. They exhibit increased heat generation and a high proportion of fissile material in the spent fuel compared to uranium fuels from LWR. This has an impact on the space required in the repository and increases the requirements for handling MOX regarding criticality safety and radiation protection. Alternatively, multi-recycling, as originally envisaged in a plutonium economy or contemplated in P&T scenarios, would have to be developed industrially, but this is not to be expected from today's point of view. The use of SFR has only marginal influence on the necessary criteria for a geological repository. The need for a geological repository cannot be avoided by any variant of a closed fuel cycle or P&T strategy.

New fuels for SFR such as carbide and nitride fuels are being researched to achieve high burnups, but also have new characteristics such as the formation of large quantities of radioactive carbon, a long-lived mobile activation product with implications for long-term safety in disposal. SFR concepts contain large quantities of sodium coolant in the primary circuit, which must be cleaned and then conditioned and disposed of as intermediate-level radioactive waste. Coolant residues in the reactor are also problematic when components are replaced and during dismantling. The residues must be removed due to their high chemical reactivity and generate additional sodium-containing waste.

With regard to proliferation, an aspect particularly relevant to SFR is that weapons-grade plutonium is produced in uranium breeder blankets of fast reactors. SFR with breeder blankets are designed to produce additional amounts of fissile material. Together with the envisioned separation of plutonium for reuse in MOX fuel, this makes SFR much more proliferation sensitive than LWR with an open fuel cycle. An economic advantage for SFR was expected in the past if a shortage of uranium resulted in very high prices for fissile materials, making the use of plutonium as an

alternative fissile material attractive. However, this would require extremely high uranium prices. From today's perspective, these are not to be expected over the next decades. The investment costs of SFR are typically estimated to be higher than those of today's LWR. At the same time, the availability of SFR achieved so far is lower than that of LWR. Overall, therefore, SFR can be assumed to be at an intrinsic disadvantage compared with LWR in terms of economic efficiency.

Lead-cooled Fast Reactors (LFR) belong to the class of liquid metal-cooled fast reactors along with SFR, so the advantages and disadvantages of SFR can largely be transferred to LFR. As with SFR, LFR are envisioned to operate in a closed fuel cycle, with the desired advantage of using MOX fuels and thus lower natural uranium consumption. Likewise, LFR are to be used for transmutation of minor actinides. A disadvantage of the LFR compared to the SFR is that much less operating experience is available and no major prototype of an LFR has yet been operated outside of the Soviet program to use lead-cooled submarine reactors. The technological development of LFR is far behind that of SFR.

Compared to SFR, the use of lead or a lead-bismuth alloy has the advantage that the coolant is chemically non-reactive, has a high boiling point and is thermally inert. The power density in the core is lower. This is offset by the fact that lead has a significantly higher potential for corrosion and erosion, and special attention must be paid to seismic effects due to the high mass of lead. In addition to the main role of LFR for power generation, the use of process heat is also considered as an application due to the operating temperatures between 400 and 620°C; in the future, higher temperatures are also to be achieved. Cost advantages over SFR are possible due to a simpler design and no need for an intermediate cooling circuit.

Gas-cooled Fast Reactors (GFR) differ from liquid-metal-cooled fast reactors (SFR, LFR) essentially in the properties of the coolant, with helium being the main coolant under discussion today. A major potential advantage of a GFR is that particularly high operating temperatures in the range of 800-850°C are possible in principle via gas cooling with helium. To date, however, no helium-cooled fast reactor has been built and operated. In order to be able to realize a first experimental reactor, extensive further developments in the field of fuels and high-temperature materials are still required, in particular in order to realize temperatures in the range beyond 800°C. Furthermore, technological developments in the field of systems and components for helium cooling as well as operational and safety systems are still necessary. Before a larger demonstration reactor could be built and operated, the construction and operation of a smaller experimental reactor is a necessary intermediate step. GFR are seen within GIF primarily as a longer-term alternative to liquid-metal-cooled fast reactors.

Helium as a coolant has favourable intrinsic properties with regard to safety. It is chemically inert, optically transparent and does not form radioactive activation products. However, these favourable properties are countered by intrinsic disadvantages with regard to other safety functions, which result primarily from the poorer heat dissipation properties of helium. Unlike VHTR, which are also helium-cooled, GFR exhibit a high power density in the reactor core with only low thermal conductivity and heat capacity of the core structures. Therefore, due to the low density and thermal conductivity of helium, adequate core cooling in a GFR requires high coolant pressure and high coolant flow rate in the core, which must be maintained during all event sequences. A clear overall assessment of the level of safety that can be achieved in the future is not possible today. However, due to problems regarding the assurance of sufficient core cooling, an overall disadvantage in terms of safety can be assumed. Fuel supply and waste disposal aspects as well as proliferation risks depend decisively on the specific design of the fuel cycle for GFR, just as they do for liquid-metal-cooled fast reactors.

In these terms there are no significant technology-specific differences compared to the latter. The higher efficiency of GFR compared to LWR can lead to cost advantages compared to today's LWR technology. However, the extent to which these can be realized depends on the availability of suitable fuels and structural materials and is largely unclear from today's perspective.

The basic idea of a **Molten Salt Reactor (MSR)** is to dissolve fissile and fertile materials in liquid fluoride or chloride salt so that fissioning and breeding of new fissile material occurs in the liquid fuel salt. In principle, a liquid fuel allows flexible handling of nuclear and chemical processes by continuously changing the fuel composition. A wide variety of MSR reactor designs is conceivable, since different salts, fissile materials, moderator materials, and system configurations can be combined. The first experimental plants were operated in the 1950s and 1960s in the USA, but larger demonstration plants have not yet been built. Further development of MSR still requires extensive work in the areas of materials development of fuel salts and structural materials, particularly with respect to corrosion, development of integrated models of reactor behaviour for safety analyses, and systems for production, processing, transport, and storage of radioactive salts. Before a larger demonstration reactor could be built and operated, the renewed construction and operation of a smaller experimental reactor must be considered a necessary intermediate step.

The properties of MSR depend very much on the specific characteristics of a particular reactor concept. MSR can exhibit favourable intrinsic safety-related properties, such as a strong negative feedback of reactivity with temperature. However, MSR are in turn confronted with extensive new safety-related issues, for example in connection with the continuous inflow and outflow of reactivity into and out of the reactor core. Aspects of the fuel cycle of MSR are also still largely open from today's point of view, since extensive questions concerning chemical process control (both with respect to fuel composition and reprocessing technologies) remain unresolved.

The waste streams generated in an MSR differ significantly from those of an LWR. In contrast to LWR, MSR handle much larger quantities of radioactivity relative to the energy produced in completely different process streams. The conditioning of the waste has to be adapted to the different waste streams. From today's perspective, it is unclear whether direct disposal of the fuel salt is possible, whether immobilisation will be necessary and whether the waste can be disposed of together with today's high-level waste. For both chloride and fluoride salts, major gaps remain in the assessment of waste package functionality and separation processes to predict the long-term behaviour of the waste forms in a repository environment. MSR will place high demands on fissile material control because the fuel is in free-flowing form. Due to the very early stage of development of this technology line, it is not yet possible to make any reliable statements regarding costs, and the risks for investors are high. Which of the potential advantages or disadvantages will be realized in a specific reactor concept is unclear today. The extent to which this will result in overall advantages or disadvantages compared with today's LWR therefore cannot be answered today.

The basic idea of a **Supercritical Water-cooled Reactor (SCWR)** is to achieve a higher working temperature compared to today's LWR and thus increase the efficiency of the reactor system. To achieve this, the coolant in the reactor is to be heated to temperatures beyond the critical point of water, which is located at a temperature of 374°C and a pressure of 22.1 MPa. Otherwise, SCWR are very similar to today's LWR and therefore exhibit the least differences to them. Nevertheless, no reactor cooled with supercritical water has been built to date. In order to be able to realize a first experimental reactor, extensive further developments would still be needed, especially in the field of high-temperature materials.

The safety characteristics and the safety concept of SCWR are largely comparable to today's LWR; overall, no significant advantage or disadvantage can be assumed. The higher efficiency of SCWR compared to LWR can lead to a reduction of the mass of spent fuel being produced by about 25-30%. Potential savings in capital costs, as well as the higher efficiency during operation, may result in cost advantages over today's LWR.

The basic idea of the **Very High Temperature Reactor (VHTR)** is to achieve correspondingly high efficiencies for electricity production and/or to utilize process heat due to the high temperatures. To withstand the temperatures, graphite is used both as a structural material and to contain the fuel in spheres or rods. High temperature reactors have been under development for more than 60 years. To date, no research and development program has been able to operate a VHTR at high utilization for an extended period of time. Most of the intensive development programs (Germany, USA, South Africa) have so far stopped at the stage of a demonstration reactor, due to disinterest on the part of the operators because of concerns about reliable and economical operation. Active research and development programs are currently taking place mainly in China, the USA and Japan. For reactor concepts with temperatures between 750°C and 950°C in particular, substantial development work still needs to be done, especially with regard to materials and instrumentation. Depending on the reactor concept, the current maximum permissible fuel temperature of 1600°C is also insufficient and further fuel development is necessary.

VHTR are particularly associated with the term "inherent safety". Due to the low power density in the reactor and due to the inertial temperature behaviour, loss-of-coolant accidents are not the main risk. This is an advantage over LWR. Even in events with a loss of active cooling, the decay heat produced in the reactor can be removed by passive means as long as the reactor power does not exceed a power level of a few hundred Megawatt thermal power. Therefore, the relevant accident scenarios for VHTR are not core meltdown as in LWR, but the ingress of water and/or air into the reactor core. Water above a certain amount can trigger a reactivity accident. The graphite of the fuel and of reactor structures such as the reactor wall can oxidize and decompose on contact with water or burn in combination with oxygen. A massive release of radioactivity is therefore by no means excluded in principle (by inherent properties) in VHTR.

VHTR use coated TRISO fuel particles in a graphite matrix, which contain most of the radioactive inventory. Due to the graphite matrix, significantly higher volumes of spent fuel are produced than with LWR. TRISO fuel particles are robust and in principle suitable for disposal. However, further research is required to determine and demonstrate the effectiveness of the barrier in a repository environment. Separation of the TRISO particles from the graphite to reduce volume requirements is being researched, but there is no generally accepted method for graphite treatment. A conditioning approach for direct disposal would therefore be simpler. In addition to the graphite matrix and particle coating, other graphite waste is generated by structural elements in the reactor core and graphite dust, which contaminates reactor components due to adhering fission products.

With regard to proliferation, the use of uranium fuel in VHTR would require the technology of uranium enrichment, as it does in current LWR. Also closely related to the VHTR is the idea of using thorium fuel. Pebble bed reactors are more vulnerable to fuel diversion than LWR because of the ongoing replacement of fuel. The major potential advantage of a VHTR is the opportunity for cogeneration and for the production of process heat, which could also be used for hydrogen production, depending on the temperature level.

The basic idea of **Accelerator-Driven Systems (ADSs)** is to operate a fast reactor with a subcritical reactor core. Since such a reactor cannot maintain a steady-state (i.e., constant over time) neutron population by itself, it relies on an "external" neutron source. The external neutron source is typically implemented by a proton accelerator whose particle beam impinges on a heavy metal target inside the reactor core and produces high-energy neutrons via a spallation process. The original idea of using large accelerators and neutrons from the spallation process to transmute radionuclides emerged in the 1950s. Since the 1990s, plans to generate energy with ADSs have experienced a resurgence. To date, there is no running prototype ADS plant in the world. Development is currently still in the conceptual and planning stages. In addition to the need to develop a subcritical fast reactor system (lead-cooled systems are planned), there is the need for the development of a reliable spallation neutron source. The accelerator must operate for months without interruption. Accelerators are still expensive and large. In addition, there are technical difficulties with heat removal from the heavy metal target.

ADSs could have significant safety advantages due to the subcritical arrangement. ADSs are also being discussed for transmutation of radioactive waste. They have advantages for transmutation due to the fast neutron spectrum and greater flexibility with respect to fuel composition compared to critical reactors. Since many details of ADS concepts and associated fuel cycles have not been determined today, numerous unanswered questions remain regarding the analysis of the waste inventories that will ultimately be disposed of. There are costs associated with the construction of the accelerator and spallation neutron source and their operation; likewise, a portion of the electricity generated is used to operate the facility. An ADS is therefore more expensive than a comparable LFR.

Conclusions across technologies

For all technology lines considered, extensive **research and development** work has been taking place for several decades, in some cases since the middle of the last century. Depending on the technology line, technical test stands for individual phenomena have been built and operated, and so have smaller experimental reactors (for SFR, for example, the U.S. EBR-I and II plants or the Russian BR-10 and Bor-60) and larger demonstration reactors (for SFR, for example, the French Phoenix and Super-Phoenix plants or the Russian BN-350 or BN-600 plants). Nevertheless, until today no commercially competitive reactor concept exists in the field of SNR.

The most extensive technical experience is available for the SFR and VHTR technology lines. Their technical feasibility in terms of construction and operation of a demonstration reactor for electricity production has been achieved. However, in order to fully realize the advantages associated with these technology lines compared to today's LWR or to exclude possible disadvantages as far as possible, extensive further technological developments and, in particular, proof of reliable operation under economic boundary conditions are still required for SFR and VHTR. For other technology lines, neither has technical feasibility in the form of a demonstration reactor been demonstrated so far (LFR, MSR), nor are more extensive findings from smaller experimental reactors available (GFR, SCWR, ADS).

To plan, license, construct and operate such experimental and demonstration reactors, a period of at least one to two decades must be assumed for each reactor project, probably substantially more based on historical experience. The knowledge gained with these facilities needs to be evaluated and incorporated into the technical design of an eventual prototype reactor.

Extensive programs are required in particular for the (further) development of fuels and structural materials: basic laboratory and research work, testing of the material properties in suitable test rigs and the use of the materials in research reactors with representative technical boundary conditions (temperatures, pressures, coolants, neutron spectrum, etc.) as well as subsequent post-irradiation examinations. Such developments, even with the use of extensive resources, typically also require a period of many years to several decades. Since the demands placed on materials by corrosion, for example, increase with temperature under simultaneous irradiation, quite a few development programs also provide for several stages of development, with a phase that places fewer demands at lower temperatures and later phases with higher temperatures. Higher temperatures are pivotal to economic appraisals of the efficiency of power generation and the use of process heat.

In some cases, developers of specific reactor concepts are planning to forego the intermediate step of building and operating experimental or demonstration reactors for reasons of time. However, this is associated with high risks, both in terms of licenceability and the actual subsequent functionality and reliability of such reactor concepts. It has been shown that developers' schedules are often characterized by overly optimistic assumptions, that developments are delayed by years or even decades, and that in many cases specific approaches are discontinued completely because the underlying technological difficulties could not be overcome.

Thus, the time still required for the development of SNR is probably in the range of several decades. Against this background, it cannot be assumed that such reactor concepts will be used on a relevant scale by the middle of this century. Even the GIF assumes that the introduction of such reactor concepts in addition to today's LWR can only be expected in the second half of this century.

The majority of SNR technology lines envision reactor designs with fast neutron spectra. In these technology lines, there are fundamentally higher requirements for reactivity control in the area of **safety**, since the reactor core is typically not in its most critical arrangement. Thus, unlike in the case of LWR, accident sequences up to a promptly supercritical state are also conceivable. It is essential, however, to practically rule out such accident sequences.

For heat removal, various alternative coolants discussed for SNR offer a number of favourable properties. Helium gas, for example, is chemically and neutronically inert; liquid metals, for their part, have good thermal conductivity and high heat capacity. Furthermore, the primary cooling circuits in systems using liquid-metal-cooling are not under high system pressure, which reduces the risk of large loss-of-coolant events. On the other hand, water is widely used as a coolant and its properties and the necessary technical systems such as pumps and fittings are therefore extremely well known and developed.

The approaches towards confinement of radioactive materials are highly similar in most SNR concepts. Multiple barriers are designed to prevent the release of radioactivity to the environment. Essentially, the systems differ in this respect due to the accident sequences possible in them and the resulting hazards for the barriers designed to contain radioactive substances. While certain event sequences, such as loss-of-coolant accidents, may play only a minor role for the safety of some SNR compared to LWR, other event sequences may decisively determine the achievable safety level. The chemical reactivity of the coolant (such as sodium in the case of SFR) or of the structural materials (such as graphite in the case of VHTR) can play a role, as can the altered requirements upon reactivity control, especially in fast reactors.

While fundamental safety advantages over today's LWR are conceivable for individual technology lines, this is not to be expected for other technology lines. Ultimately, many safety-related questions can only be analysed for a specific reactor concept, since the safety level always depends on the specific detailed technical design of a reactor concept. Therefore, a conclusive safety assessment can only be performed for a completely defined reactor concept.

The technology lines can achieve higher efficiency in power generation compared to today's LWR due to higher operating temperatures. With regard to **fuel supply and waste disposal** aspects, this could reduce uranium demand for power generation and thus also the arisings of highly radioactive waste in the order of several tens of percent.

Higher utilization of the uranium resource is conceivable by means of the production and further use of plutonium through the technologies of reprocessing and MOX fuel production; alternatively, uranium-233 could be obtained as fissile material by using thorium. Only in the case of a very long-term use of nuclear energy or its extreme expansion would such a better utilization of known uranium reserves theoretically be necessary. Under current boundary conditions and those that can be expected for the future, such reprocessing and fuel fabrication will lead to higher costs for the fuels required.

All systems with improved efficiency due to higher operating temperatures have in common that less radioactive fission products are produced in relation to the electric energy produced.

Some SNR use different fuels, coolants and moderators than LWR. This affects the quantities and the chemical and radioactive properties of the waste produced. Due to the targeted separation of different waste streams (gaseous fission products, lanthanides, metals) in some technology lines and the use of different fuel cycle technologies, specific conditioning processes for different waste streams are in use or will be developed.

In particular, the use of different coolants (e.g., sodium) and moderator materials (graphite) in SNR leads to larger quantities of intermediate and high-level waste compared to LWR. Waste treatment for these systems is significantly more difficult than for light water, due to the chemical and radiological characteristics of the coolants and moderator materials. Many waste treatments involve splitting the waste streams as direct disposal is not possible. This requires complex processes with additional secondary waste, which also results in additional costs. In particular, MSR with liquid fuels requires a large number of waste treatment, stabilisation and conditioning processes. Fast reactor designs have an increased volume of activated reactor elements and structural components compared to LWR due to the fast neutron spectrum. The reprocessing of the fuel, which is foreseen in many fuel cycle scenarios for the use of SNR, also generates additional waste streams of intermediate and low-level radioactive waste.

VHTR, but also MSR, are often associated with the use of thorium fuels. The radiotoxicity of thorium-based fuels containing uranium-233 as the fissile material is determined by the accumulation of heavy elements and their decay products and is overall similar to that of uranium-based fuels.

In several technology lines, plutonium or a mixture of plutonium and minor actinides from spent fuel reprocessing is to be used as fissile material instead of enriched uranium. This shifts **proliferation risks** from uranium enrichment facilities to reprocessing and fuel fabrication facilities and fresh fuel shipments. As long as the use of enrichment technologies cannot be completely eliminated, this creates additional diversion pathways and thus new proliferation risks.

Only for individual reactor concepts, such as the Traveling Wave Reactor (TWR), is it a declared goal to be able to dispense with both the enrichment of uranium and the reprocessing of spent fuel by generating and using the fissile material in situ. This could significantly reduce proliferation risks compared to current LWR. However, due to the particularly high technological requirements associated with TWRs, especially for the fuels, the TWR is currently seen only as a longer-term development goal, including by the developers themselves.

Today's LWR are not competitive with today's renewable energy generation technologies in terms of their **levelized cost of electricity (LCOE)**. Moreover, historical cost trends show rising LWR LCOE over time, while the renewable energy sector has seen massively falling costs, especially in the last decade. For the future, there are no apparent reasons why this trend should reverse.

Individual SNR technology lines could achieve certain economic advantages compared to today's LWR. The reasons for this are possible investment cost savings, higher efficiency of power generation through higher operating temperatures and utilization of process heat. The extent to which cost advantages can actually be realized in specific reactor concepts compared to today's LWR remains uncertain. Especially for reactor concepts with high working temperatures, new materials still have to be developed and their applicability has to be proven. Cost estimates are also associated with high uncertainties due to the mostly still very early development stage of SNR. Overall, it cannot be assumed that the cost advantages that can be realized with SNR could compensate for the cost disadvantages of today's LWR compared to other power generation technologies or even turn them into a cost advantage.

Regulation

National and international nuclear safety regulations have been developed mainly on the basis of experience gained in the construction and operation of today's water-cooled reactor designs. On the one hand, they define basic requirements (performance-based regulations), but on the other hand, they also specify concrete technical designs or set requirements with reference to concrete technological solutions (prescriptive regulations). A regulatory regime for SNR, comparable to that for water-cooled reactor designs, does not exist either nationally or internationally. Regulators around the world are therefore reviewing and revising national and international regulations with a view to their applicability to SNR.

The main differences between SNR and today's water-cooled reactor concepts result in particular from the use of different coolants, increased use of passive safety features, different fuel concepts and possibly different fields of application. Based on such differences, numerous problems have been identified in the application of existing regulations for SNR at national and international levels. This affects many key areas of reactor design and construction.

On the one hand, problems with the applicability of the regulations may arise from the fact that the current wording is not technology-neutral, so that an amendment or adaptation of the regulatory texts to a technology-neutral wording may in principle offer a solution. Furthermore, applicability problems have been identified due to innovations in SNR for which there are no corresponding requirements in the existing regulatory texts. Due to the significantly less operating experience of experimental and prototype reactors compared to water-cooled reactor concepts, it can be assumed that considerable time will be needed to develop specific requirements in these areas. Such plants will need to be planned, licensed, constructed, and operated, and the time required for this is likely to be one or more decades rather than a few years.

In the **US**, the Nuclear Regulatory Commission (NRC) is developing a new set of regulations to be published by 2027. This is intended to provide a technology-neutral approach through increased use of risk-informed, performance-based requirements. The NRC is also conducting extensive prelicensing activities for various reactor designs in the area of SNR. As of January 2024, an application for a combined construction and operating licence for a microreactor (Aurora Powerhouse) was submitted, but the process was terminated by the NRC without a final assessment due to a lack of cooperation from the applicant. There have also been three applications to build test reactors, one of which was licensed in December 2023.

Canada has a comprehensive programme for the introduction of SMR at both national and provincial levels. SMR in Canada include both water-cooled concepts and SNR concepts (referred to as ANT in Canada). Several vendor design reviews (VDRs) have been or are being conducted for this purpose by the Canadian Nuclear Regulatory Commission (CNSC). These have typically taken a factor of 1.5 to 3.5 times longer than the time envisaged by the CNSC for such VDRs. Other VDRs continue to be delayed or have been cancelled without result. As a result of the completed VDRs (Phases 1 and 2), extensive outstanding issues have been identified that would need to be resolved before an SNR could be approved in Canada. Despite these outstanding issues, two applications for site preparation for the construction of an SNR have been submitted, but no phase 3 VDR and no application for a construction or operating licence has yet been submitted. The CNSC has formulated a strategy for the successful regulation of new reactor technologies and is pursuing the goal of formulating a technology-neutral regulatory framework.

In the **United Kingdom** (U.K.), research into Advanced Modular Reactors (AMR), i.e., reactor concepts with alternative coolants or fuels, is supported. AMRs in the U.K. are to be understood as small plants with an electrical output of less than 300 MW, and as modular plants. Initially, various technology lines were supported, but from around 2020, support will be limited to concepts in the VHTR technology line. The Office of Nuclear Regulations (ONR) is pursuing its own programme of work and research to strengthen its expertise in AMR and to review the processes and guidelines for licensing new reactors. As a first step, the Generic Design Assessment (GDA) process has been revised and new guidelines have been published. However, to date, no AMR concept has started the GDA process in the U.K. and no applications to build or operate AMRs have been submitted. A review of the basic guidelines by the ONR for their applicability to AMR is planned. Initial research reports are available, but the ONR still sees a significant need for future research in the area of AMR.

Several of the identified applicability issues for SNR are now being addressed at the national and international levels through revision processes of existing regulations or the development of additional regulations, but gaps remain. Overall, it can be assumed that it will take some time for regulators and international organisations to gain operational experience at the first real SNR-plants and to achieve and maintain a high level of confidence in the safety of reactors. This also applies to possible adjustments to nuclear regulations, as such adjustments require a sufficient evidence base, which is not yet available in many areas.

The move from a prescriptive to a more performance-based (technology-neutral) regulation means that rules can be created that are generally applicable to SNR. However, this means that SNR developers will need to demonstrate compliance with the target-oriented requirements for their particular system, without being able to rely on established procedures. At the same time, there is a need for the licensing authority to independently review the system-specific evidence submitted by the developers, which in turn requires the development of appropriate expertise and a clear

knowledge of the relevant safety issues for the respective technology lines. As a result, there is a risk that the relevant authorisation procedures for SNR will take longer.

Partitioning and transmutation

Partitioning and transmutation (P&T) according to the proponents essentially promises the possibility of reducing the requirements for the disposal of highly radioactive nuclear waste. Transmutation involves the targeted separation of transuranic elements from nuclear waste (partitioning). The transuranics must be used to produce new fuel. The transuranics are then converted into shorter-lived or stable isotopes by neutron irradiation in nuclear reactors (transmutation), primarily by fission. As only a small fraction of the transuranic elements is fissioned during a single use in a reactor, multiple recycling is required. Fast reactors (SFR, LFR, GFR, MSR) are a core component of P&T waste management strategies. Accelerator driven subcritical systems (ADS) can also be used. Thermal reactors have limited suitability for transmutation due to the limited amount of TRU in the fuel.

Wet chemical and pyrochemical processes are considered for partitioning. The separation efficiency must be greater than 99.9% in order to limit the residual amounts of transuranium elements in the waste stream. Wet chemical Plutonium separation processes such as PUREX (Plutonium URanium EXtraction) are now fully developed and used on an industrial scale. The separation of minor actinides (MA) with wet chemical processes is still under development as well as pyrochemical processes. Pyro-electrometallurgical processes are probably the most suitable of the pyrochemical processes for MA separation, as a separation efficiency for actinides of up to 99.9% may be achievable.

The production of MOX fuel containing plutonium from partitioning has been demonstrated on an industrial scale but is not commercially competitive with uranium fuels. The production of fuels with MA has not yet been achieved on an industrial scale. The volatility of americium places particular demands on process control and may limit the proportion of americium in the fuel. Due to its high heat generation and neutron emission, curium leads to high costs in fuel production and transport of fresh fuel. If curium is separated, it would have to be irradiated in additional targets or disposed of with the fission products.

Depending on the specific technical design, the required reprocessing and fuel fabrication technologies pose their own significant safety and proliferation risks.

Use plutonium in once-through-MOX for LWR is currently practiced on an industrial scale in a few countries. The amount of plutonium in the fuel can be significantly reduced by irradiation in the reactor. Further or multiple use of spent MOX is not currently practised on an industrial scale, as the heat generation and activity of such MOX fuel makes it difficult to handle and economically unattractive.

For P&T scenarios, a distinction can be made between scenarios for the continued use of nuclear energy and scenarios aimed at reducing an initial stock of radioactive waste (phase-out scenarios). Typical phase-out scenarios have implementation periods of 100 years or more.

The transmutation characteristics of fast reactors (FR) are very similar. Similarly, the choice of fuel (for the same achievable burnup) has little influence on the transmutation properties, only uranium-free fuel would offer advantages. The amount of MA in the fuel is limited by the safety characteristics of the reactor concept. The use of an ADS system would have the advantage that it is subcritical and therefore the safety characteristics are less sensitive to the use of plutonium or MA. Therefore,

uranium-free fuels that do not produce new MA can also be used in ADS. ADS systems therefore have advantages over FR systems, especially in phase-out scenarios. In principle, an advantage of MSR is the flexibility of the liquid reactor concept, which allows a very high actinide reduction to be achieved, and the shortening of the cycle length due to the possibility of reprocessing during operation. A significant limiting factor for the use of plutonium and minor actinides in MSR is the limited solubility of the actinides in reactor designs based on fluoride salts.

In phase-out scenarios with a continuously operated reactor fleet consisting of SFR burners (Pu burners and MA burners), an initial inventory of TRUs could be reduced by a factor of about 80 under ideal conditions (separation efficiency, transmutation rate). This would result in about two and a half times the initial inventory of fission products.

In closed fuel cycles, the volume (mass) of high-level waste can be significantly reduced. However, this is mainly due to the separation of uranium and the allocation of the reprocessed uranium either to another waste stream, e.g., intermediate level waste, or as a recyclable material. Compared to uranium/plutonium fuel cycles, the recycling of MA leads to a further small reduction in the volume of waste.

Taking into account the packaging and conditioning of the waste, the volume of high-level waste from reprocessing and spent fuel is similar in the different scenarios of a closed fuel cycle and in scenarios of continuous use of nuclear energy and amounts to at least about 1/3 of the LWR waste for the same electricity produced. Much of the uranium is no longer part of the high-level waste stream but is classified as intermediate or low-level waste. The volume of intermediate and low-level waste has increased significantly.

P&T treatment of waste is intended to reduce the heat production of high-level waste, as well as the volume, so that the repositories required may be smaller. Sometimes it is even promised that a geological repository can be avoided. P&T is also said to have a positive effect on the risks of disposal in a geological repository.

The heat production of high-level waste determines the required spacing of the waste containers and thus the required storage area in a geological repository and hence its size. In the first 100-200 years, fission products are particularly relevant for heat generation. Separation and separate storage of strontium (and caesium) from high-level waste would significantly reduce the required storage area of a repository, as would interim storage of the waste without partitioning for these periods prior to emplacement in a geological repository. This advantage of a potentially smaller repository has to be weighed against the risks of long-term interim storage of spent fuel or partitioning waste and the additional medium and low-level waste streams generated for partitioning as well as the operational risk of the partitioning facilities. Transmutation of the actinides would have a significant impact only for periods beyond this, with total heat production by the actinides after 300 years being only 10-20% of the initial values of the total waste.

Alternative fuel cycles with SNR therefore have a very limited impact on the heat production of high-level waste and thus on the size of a repository.

The reduction in volume and quantity of HLW by uranium separation is recognised in many studies as an advantage of P&T. This results in some very high reduction factors for HLW. Since the required storage area of a repository is determined by the heat input, a reduction in volume, e.g., by uranium separation, has little effect on the required size of a repository.

Two main criteria are used to assess the potential radiological risks to humans from the disposal of radioactive waste in a geological repository: the radiotoxicity of the waste and the long-term safety of the repository.

Radiotoxicity is only suitable for mapping the very unlikely event sequences of a release of the radioactive inventory from a repository and a subsequent incorporation of radionuclides. P&T treatment using SNR significantly reduces the long-term radiotoxicity of HLW, it thus only has an influence on the reduction of very unlikely risks. This consideration does not take into account the planned development of a repository and the release by mobilisation of radioisotopes and the resulting dose to humans. It therefore makes no statement about the long-term safety relevance of the radionuclides.

The standard methodology for assessing the radiological risks to humans from geological repositories is a long-term safety analysis. This takes into account the release pathways of the nuclides. Actinides are extremely immobile in most repository rocks and are not released. The main contribution to dose estimates in the long-term safety analysis comes from long-lived mobile fission products such as Iodine-129, Technetium-99 and activation products such as Carbon-14. Reducing transuranics does not therefore contribute to the long-term safety analysis of geological repositories; instead, fission product transmutation would be required.

Therefore, in scenarios of longer-term use of nuclear energy, fuel cycles with reprocessing have no advantage in the long-term safety assessment compared to the current use of LWR without reprocessing. In transmutation scenarios with a fixed initial inventory of transuranics (phase-out), the quantity of dose-determining radionuclides would even increase due to the additional fission products produced.

Fission product transmutation would be technically limited to the nuclides Technetium-99 and possibly Iodine-129, due to the technical difficulties of partitioning and target production for irradiation, and the sometimes necessary isotope separation. Overall, fission product transmutation is hardly pursued today.

None of the scenarios for the use of alternative fuel cycles with SNR and P&T treatment of waste can do without a repository for high-level radioactive waste since residual quantities of transuranics and long-lived fission and activation products remain in the waste stream. In addition, the operation and dismantling of the partitioning facilities will generate much larger quantities of intermediate and low-level waste.

General conclusions

From the discussion it becomes clear that individual technology lines – with rigorous design – may deliver advantages over today's LWR in individual evaluation criteria (safety, fuel supply and waste disposal, proliferation, costs). At the same time, however, none of the technology lines can be expected to have an advantage over today's LWR in all areas, and disadvantages compared to today's LWR are possible in individual areas.

Various criteria also compete with each other; improvements in individual areas can lead to disadvantages in other areas. For example, an increase in the safety of a specific reactor concept is often accompanied by additional costs or the need to develop new, improved materials or technologies and thus increased development effort and time required until feasibility. Advances in the area of fuel supply and waste disposal can lead to new safety risks or an increase in proliferation risks.

Developers of SNR repeatedly refer to intrinsic properties of technology lines or their specific reactor concepts to argue definite advantages in individual areas compared to today's LWR. Among the advantages presented are: exclusion of event sequences that are of particular safety relevance for today's LWR; the possibility of producing new fissile material during reactor operation; higher efficiency of electricity production through high coolant temperatures or through utilization of process heat.

It is true that such intrinsic properties can indeed lead to advantages over today's LWR. However, the differences between SNR and today's LWR that are the source of such intrinsic properties typically lead to new technological challenges or disadvantages in other areas. For example, different but equally relevant event sequences may pose safety risks; the recovered fissile material must be suitable for reuse, which may lead to proliferation risks; and higher coolant temperatures require new suitable materials that can withstand the increased demands during operation.

Ultimately, only a detailed analysis of a specific reactor concept can provide a conclusive assessment of the extent to which intrinsic properties of the technology line in interaction with the specific design of a reactor concept will lead to overall advantages or disadvantages in the individual evaluation criteria.

The expectation, often expressed in public discourse and by developers themselves, that SNR concepts can make a significant contribution to solving today's problems in nuclear technology cannot be considered realistic in view of the current state of development of these systems and the actually proven and expected advantages and disadvantages of the individual technology lines.

1 Introduction

Many so-called “novel” reactor concepts (SNR) are being discussed internationally and nationally by experts and in the media debate. In some cases, concrete implementation projects have been announced by operators – in some cases for many years. Individual corresponding experimental plants or prototypes have actually been put into operation in recent years.

The debate often focuses on six technology lines that are summarised under the term “Generation IV” and whose development is being pursued by an international research association, the “Generation IV International Forum” (GIF). The founding of the GIF assumes an initiative of the United States Department of Energy (DoE), which invited government representatives from nine different countries to talks on the joint development of SNR in January 2000. In June 2001, the GIF Charter was signed by Argentina, Brazil, Canada, France, Great Britain, Japan, South Africa, Republic of Korea and the United States. To date, Switzerland (2002), Euratom (2003), China and Russia (2006) and Australia (2016) have also joined the GIF. The GIF Charter was last amended in 2011 and defines the objective of the GIF as the development of concepts for one or more fourth-generation nuclear energy systems that provide a competitive and reliable energy supply for the country in which these systems are deployed while satisfactorily addressing concerns related to nuclear safety, waste, proliferation and public perception. To this end, the GIF pursues, among other things, the identification of research needs and the promotion of joint research projects.

According to their developers, SNR should, therefore, have an advantage over nuclear power plants currently in operation in terms of criteria such as sustainability, safety and reliability, economic efficiency and nuclear non-proliferation (GIF 2002). Proponents also cite a lower fuel demand for fuel and a reduced amount of high-level waste as potential advantages. However, despite decades of development in some cases, such SNR have not yet reached a corresponding readiness level or have not become widely accepted for commercial or safety reasons.

There are also lines of development that are currently being researched and cannot be directly attributed to GIF developments. These include, in particular, so-called “accelerator-driven systems”. Within the SNR technology lines, different countries or companies are pursuing many different specific reactor concepts. Both the various technology lines themselves and the various reactor concepts within a technology line can have relevant differences in terms of their technical design and, thus, also the potential advantages or disadvantages associated with them. An analysis of SNR can, therefore, not only focus on the level of the technology line but must also consider the design in the form of specific reactor concepts.

In this study, the term technology line is used as a generic term for a number of different reactor concepts that are similar in a central characteristic, e.g. that use the same coolant. Within a technology line, different reactor concepts can be distinguished on the basis of other properties such as the moderator used or the fuel used. In turn, one or more specific plants can exist for a reactor concept, which differ again in detail in terms of technology.

Various studies on SNR have already been carried out in the past, which have dealt with such developments to varying degrees (Oeko-Institut e.V.; WIP; PhB 2021; UCS 2021; Oeko-Institut e.V. 2017; IRSN 2015; GRS 2015; IANUS; Oeko-Institut e.V. 1999; Oeko-Institut e.V. 1989a; 1989b; 1986). Since then, new developments have emerged in some technology lines, while previous developments in other areas have been discontinued. This study therefore aims to analyse and evaluate the current readiness level of relevant SNR. This should also help identify relevant trends

regarding such reactor concepts and analyse and evaluate safety and disposal-related issues and risks.

In this study, SNR will be discussed as comprehensively as possible with regard to their various characteristic properties. In principle, the SNR considered include current development projects or those from the recent past with significant research and development activities but which do not yet have a prototype concept (such as molten salt reactors) or whose implementation in a plant design has been delayed for economic or safety reasons (such as sodium-cooled fast reactors). On the other hand, further developments of current light- and heavy-water reactors are not the subject of the investigations carried out here. Furthermore, developments in the field of so-called “Small Modular Reactors” (SMR), in which various reactor concepts can also be assigned to SNR, are not considered in depth. For developments in the area of SMR, please refer to (Oeko-Institut e.V.; WIP; PhB 2021).

Of particular interest for an assessment of SNR are the technical development status of the concepts, concrete plans for their implementation, questions of nuclear safety and possible risks, questions of fuel supply and waste disposal, proliferation risks and economic aspects.

Chapter 2 therefore begins by introducing and systematising the technology lines and reactor concepts classified under the term SNR. It also lays the foundations for the subsequent description and evaluation of SNR. In Chapter 3, international developments in the field of SNR are presented on the basis of selected country studies. Firstly, a general overview of research activities is given, and various categories of countries whose nuclear technology development can be categorised as roughly comparable are introduced. From these categories, the research activities for exemplary countries are then presented in detail.

Chapter 4 presents and evaluates the technology lines selected for this study in detail. For a more in-depth analysis, selected reactor concepts are then presented and evaluated in Chapter **5** based on the overarching findings on the individual technology lines.

Chapter **6** provides a final overarching comparison of the main advantages and disadvantages and draws general conclusions.

Chapter 7 presents developments in national and international regulations with a view to the authorisation of SNR. Chapter 8 concludes by discussing the potential of SNR for partitioning and transmuting high-level nuclear waste to alleviate the disposal problem.

2 So-called “novel” reactor concepts – an introduction

Since the early 2000s, efforts to develop so-called “novel” reactor concepts (SNR) have been coordinated internationally under the term “Generation IV” as part of the “Generation IV Initiative” (GIF) (GIF 2021a; 2018b; 2014; 2002).

Since then, the GIF has been pursuing six technology lines. The GIF technology lines include:

- sodium-cooled fast reactors (SFR),
- lead-cooled fast reactors (LFR),
- gas-cooled fast reactors (GFR),
- molten-salt reactors (MSR),
- supercritical water-cooled reactors (SCWR) and
- very-high-temperature reactors (VHTR).

It is already clear from this list that some of these technology lines, such as sodium-cooled fast breeder reactors or high-temperature reactors, have been under development for many decades, but have not yet brought a commercially competitive reactor concept to market worldwide. For a more detailed description of the history of the various technology lines, please refer to the corresponding sub-chapters in the description of the technology lines in Chapter 4.

For the further investigation, Chapter 2.1 first categorises the term “fourth generation”, which is frequently used for so-called “novel” reactor concepts, in historical and technical terms and rejects the generation term as unsuitable for the further investigation.

Subsequently, Chapter 2.2 introduces and explains key characteristics of nuclear power plants that can be used to differentiate and systematise technology lines and reactor concepts. On this basis, Chapter 2.3 groups and selects the technology lines and reactor concepts on which the further analysis will be based.

To categorise SNR in terms of their readiness level and the possibilities of their (near-term) feasibility, an assessment of the technological maturity level is required. Overarching methodological approaches to this are presented in Chapter 2.4. Furthermore, the future introduction of SNR also depends on its economic feasibility. To this end, Chapter 2.5 discusses fundamental aspects of the economic efficiency of current and future nuclear power plants.

In addition to a systematic recording and presentation of current developments in the field of so-called “novel” reactor concepts, the aim of this study is also a comparative evaluation of such developments based on central evaluation criteria. Chapter 0 introduces the evaluation scheme used for this purpose.

2.1 Temporal and technological generation concept

As already discussed in (Oeko-Institut e.V. 2017), the GIF distinguishes in (GIF 2002) between a first generation of early prototype reactors from the period 1950-1960, a second generation of large power reactors built between 1970 and 1990, some of which are still in operation today, a third generation of advanced power reactors built from around 1990 or currently under construction, and

a fourth generation of so-called “novel” reactor concepts that could be available on the market after 2030.

The GIF thus introduces a generational concept that is based purely on the chronological order of the worldwide introduction of reactor concepts with certain features. At the same time, the GIF itself categorises the reactor concepts classified as fourth generation on the basis of six different technology lines that differ in key characteristics from the nuclear power plants currently in operation worldwide. Such characteristics can include, for example, the coolant used or the average energy of the neutrons in the reactor core, see Chapter 2.2.2. The different technology lines should have advantages over today’s nuclear power plants in key criteria (GIF 2002).

For technology development, however, this means that such reactor concepts cannot build or can only build to a limited extent on the experience gained from previous reactor development in the sense of a technological generation concept. Rather, key building blocks of the technology – materials, components, systems, safety demonstration principles, etc. – must be newly developed or adapted specifically for the respective technology line.

For example, (Stachowski 2021) points out that gas-cooled reactors require a different operating temperature and different pressures in the primary cooling circuit than water-cooled reactors (such as today’s light-water reactors) due to the different density of the coolant. Important components such as heat exchangers or fittings must also be designed differently, and the properties of the required safety systems in gas-cooled reactors also differ from those of water-cooled reactors. In contrast, reactors cooled with liquid metal require different methods for inspecting and maintaining the cooling circuit and for changing the fuel element, for example, as the coolant is not transparent, making visual inspections in the reactor impossible.

Table 2-1 presents examples of different technological development stages for various current and potential future technology lines based on plants that have actually been built or reactor concepts that are currently or in the meantime under development. Here, a distinction is made between

- initial experimental reactors (corresponding to a technological “Generation 0”), which are intended to demonstrate the basic technical feasibility of a technology line,
- initial power reactors (“Generation I”), where the aim is to produce electricity,
- advanced power reactor concepts (“Generation II”), which are aimed at broad global introduction, and
- advanced reactor concepts (“Generation III”), which are intended to offer additional advantages over the advanced power reactor concepts, particularly in the areas of economic efficiency and safety. The reactor concepts referred to in some literature as “Generation III+”, some of which have been further updated compared to so-called “Generation III”, are not further differentiated from “Generation III” in this report.

Table 2-1: Technological development stages of various technology lines

Technology line	Initial experimental reactors	Initial power reactors (Gen I)	Further developed power reactor concepts (Gen II)	Advanced reactor concepts (Gen III)
PWR	MTR, S1W, S2W, MZFR	Shippingport, Obninsk, Obrigheim	Konvoi	AP-1000, VVER-1200, EPR
BWR	BORAX-I to -V, Kahl	Dresden I, Gundremmingen-A	SWR-72	*(KERENA)*, ABWR
PHWR	ZEEP, NRX, NRU	Rolphton	CANDU 500, CANDU 6	*(EC 6, ACR-1000)*
GCR	CP-1, Windscale	Calder Hall, Marcoule	AGR	-
VHTR	Dragon, AVR, HTR-10	Peach Bottom, THTR, HTR-PM, (*VHTR)*	-	-
SFR	Fermi I, Br-10, CEFR, KNK I and II, Rapsodie, TWR	BN-800, Monju, Super-Phoenix	*(BN-1200)*	-
LFR	*(BREST-OD300)*	-	-	-
GFR	*(GFR)*			
MSR	ARE, MSRE	*(LFTR, MCFR)*	-	-
SCWR	HDR	*(CSR1000)*	-	-
ADS	*(MYRRHA)*	-	-	-

Source: (IAEA 2023g; Greenspan 2021; GIF 2002), concepts planned but not yet in operation are written in italics and placed in brackets

A differentiated view of the status of research and development within a technology line makes it clear that a simplified generation concept such as that used in the GIF, which is based solely on a historical sequence of the introduction of certain reactor concepts, must be rejected. Instead, a more realistic generation concept can be identified by comparing the actual technological development status within one of the technology lines in the area of so-called “novel” reactor concepts. It becomes clear that for various SNR, at most only initial experimental reactors have existed to date and only for a few have the first power reactors been built and operated. Therefore, when categorising the technology lines of the GIF with a technologically oriented generation concept, it would not be possible to speak of reactor concepts of a fourth generation, but at most of reactor concepts of a first or second generation. For this reason, the generation concept of the GIF will not be used further in this study. For the possibilities of categorising the technological maturity level of a technology line or a reactor concept, please refer to Chapter 2.4.

2.2 Systematisation of different technology lines and reactor concepts

The principle of power generation in nuclear power plants is basically the same as in conventional thermal power plants: Both generate heat, in conventional power plants by burning coal or gas, for example, and in nuclear power plants by nuclear fission. This heat is transported via a coolant, and steam is usually generated from it directly or in another cooling circuit. This drives a turbine. The turbine in turn drives a generator that produces electricity. Nevertheless, a large number of different technology lines and reactor concepts can be distinguished (Küppers and Pistner 2012).

In the context of this study, the term **technology line** is used as a generic term for a number of different reactor concepts that are similar in a central characteristic, for example that use the same coolant. Within a technology line, different **reactor concepts** can be distinguished on the basis of other properties such as the moderator used or the fuel used. One or more specific **plants** can exist for a reactor concept, which differ again in detail in terms of technology.

A characterisation and description of important reactor concepts can be found, for example, in (Küppers and Pistner 2012). The following presentation updates this characterisation and supplements it with further characteristics of so-called “novel” reactor concepts. To this end, the reactor concepts of power reactors currently in operation are first presented in Chapter 2.2.1 and, building on this, the essential differentiation criteria for technology lines and reactor concepts are explained in Chapter 2.2.2. Chapter 2.2.3 summarises examples of technology lines and reactor concepts. Building on this, Chapter 2.3 presents the categorisation into technology lines and associated reactor concepts undertaken as part of this study.

2.2.1 Current reactor concepts

The power reactors in operation worldwide as of 30.10.2022 include, according to (IAEA 2023g):

- 304 light water-cooled and -moderated pressurised water reactors (PWR),
- 53 light water-cooled and -moderated boiling water reactors (BWR),
- 47 heavy water-cooled and -moderated pressurised water reactors (PHWR),
- 11 light water-cooled, graphite-moderated reactors (LWGR),
- 8 gas-cooled, graphite-moderated reactors (GCR),
- 3 fast breeder reactors (FBR) and
- one high-temperature gas-cooled reactor (HTGR).

As this list shows, the various reactor concepts according to this systematisation differ primarily in terms of:

- the neutron spectrum present in the reactor (fast vs. thermal reactors),
- the moderator used (light water, heavy water, graphite or no moderator) and
- the coolant used (light water, heavy water, gas, sodium).

In addition to these differences, there may be other important differences, e.g. with regard to the temperature and pressure in the primary cooling circuit, for example between PWR and BWR.

Different types of fuel can also be used in the various reactor concepts.

To categorise technology lines and reactor concepts, the respective distinguishing features can be selected in different ways. This results in an extremely large number of potentially distinguishable technology lines or reactor concepts. If all possible combinations are taken, this results in a large number of reactor concepts that already exist or have been discussed as theoretical plant designs, see (Neles and Pistner 2012) and (IAEA 2023d; 2020b; 2020a), for example. When defining a technology line, it makes sense to carry out a suitable categorisation based on key characteristics, which leads to a manageable number of technology lines yet simultaneously ensures sufficient comparability of reactor concepts within a technology line.

2.2.2 Key differentiation criteria for reactor concepts

This chapter explains the key features of nuclear reactors that can be used to characterise a specific reactor concept and to categorise different technology lines. It should be noted that this distinction is not clear-cut, i.e. different definitions of technology lines may exist.

For example, a technology line of fast reactors could be defined within which different reactor concepts differ based on the coolant used. For example, the term “fast breeder reactor” (FBR) used by the International Atomic Energy Agency (IAEA) merely refers to the property of a fast neutron spectrum in the reactor core. Whether such an FBR is cooled with sodium or another coolant such as lead or a gas would still be open in such a definition.

The system chosen in this study to define the technology lines is explained in Chapter 2.3.

The average energy, and thus the velocity of the neutrons used for nuclear fission (neutron spectrum):

Neutrons from nuclear fission have a high energy and therefore velocity. The distribution of the energy of the neutrons in the reactor core is also referred to as the neutron spectrum. If the neutrons from a nuclear fission essentially lead directly to new nuclear fissions, this is referred to as a fast neutron spectrum or **fast reactor**. For reasons related to nuclear physics, however, it can also be advantageous to slow down (moderate) the neutrons before they trigger new fissions. In this case, the neutrons typically have an energy that corresponds to the thermal energy of the material used for deceleration (which is called the moderator). Such neutrons are called thermal neutrons, and reactors based on such a neutron spectrum are therefore called **thermal reactors**. Over 99% of the power reactors in operation today are thermal reactors, see the list in Chapter 2.2.1. In contrast, most of the technology lines being pursued in the GIF are fast reactors.

The moderator:

In fast reactors, the neutrons are not intended to be slowed down, so no moderator is required here. In thermal reactors, however, the neutrons from nuclear fission must be slowed down. Light chemical elements are particularly suitable for this, as they have a similar mass to the neutrons and the neutrons can therefore be slowed down by a few interactions with the atoms of the moderator. Hydrogen or the hydrogen isotope deuterium, as contained in light or heavy water, is particularly suitable as a moderator. Accordingly, such reactors are also referred to as **light- or heavy-water-moderated reactors**. In addition to water, carbon in the form of graphite is also used as a moderator.

The (primary) coolant:

Heat is generated in the fuel of a nuclear power plant through nuclear fission, which leads to the heating of the fuel. This heat is transported from the fuel to the turbine by a coolant, possibly via several stages. The primary coolant is used to cool the fuel directly, while a secondary coolant may be used in an intermediate cooling circuit. Different reactor concepts can differ in terms of the choice of primary coolant. In today’s power reactors, light water is mainly used, which is then used both for cooling and for moderating the neutrons; corresponding reactors are therefore also called **light-water reactors**. However, heavy water or gases such as helium or carbon dioxide are also used for cooling.

As water always contributes to the moderation of neutrons in addition to its cooling function, water cannot be used for cooling in fast reactors. Therefore, either a liquid metal such as sodium (**sodium-cooled fast reactor**) or lead (**lead-cooled fast reactor**), or a gas (**gas-cooled fast reactor**) is used for fast reactors.

Finally, a molten salt can also be used as a coolant (**molten salt reactor**), whereby such reactor concepts can be conceived both with and without an additional moderator.

Pressure and temperature of the (primary) coolant:

If water is used to cool the fuel, reactor concepts can be differentiated according to what temperature and pressure is present in the (primary) cooling circuit. In the most commonly used reactor concept worldwide, the light water in the primary cooling circuit is under high pressure of typically approx. 16 MPa (**pressurised water reactor**), so that the coolant, which is heated to approx. 330 °C as it passes through the reactor core, does not boil. The heat from the primary cooling circuit is transferred via a steam generator to a secondary cooling circuit in which the steam is generated to drive the turbine. In the second most frequently used reactor concept in the world, the pressure in the cooling circuit is lower at approx. 7 MPa, so that the light water boils at temperatures of up to approx. 290 °C as it passes through the reactor core (**boiling water reactor**). The resulting steam is channelled directly to the turbine; this reactor concept has no secondary cooling circuit.

In one of the GIF’s technology lines, the primary cooling circuit will be at an even higher pressure of over 22 MPa. The water used for cooling can then be heated to temperatures of approx. 500 °C, i.e. to a range beyond the critical point of water. Such reactor concepts are accordingly referred to as **supercritical water-cooled reactors**.

Higher temperatures can also be achieved in the cooling circuit when other coolants are used without requiring significantly higher pressure in the cooling system. In certain gas-cooled reactor concepts, temperatures in excess of 600 °C are aimed for (**Very-High-Temperature Reactors**).

The fuel and the fissile material:

The only fissile material occurring in nature to any relevant extent is the isotope uranium-235, which is contained in naturally occurring uranium at approx. 0.72%. A chain reaction with fast neutrons cannot be maintained in such natural uranium. Even if light water is used to moderate the neutrons, no chain reaction is possible with natural uranium, as the water would absorb too many neutrons during the moderation process. To operate a reactor with natural uranium, either heavy water (such as in pressurised heavy-water reactors) or graphite (such as in gas-cooled, graphite-moderated reactors) must thus be used as a moderator.

The proportion of uranium-235 in the fuel must be increased to enable the use of light water as coolant and moderator in a reactor. This requires the use of **enrichment plants**. For today’s light-water reactors, the fuel is typically enriched to a proportion of uranium-235 in the 3-5% range.

For fast reactors, the proportion of fissile material in the fuel must be even higher than for thermal reactors. Today, fuel enrichments at the borderline between low enriched uranium (< 20% uranium-235), for which the term “high assay low enriched uranium” (HALEU) is currently used, and highly enriched uranium are typically discussed.

Plutonium is always produced in uranium fuel during use in the reactor. Plutonium is fissile in both fast and thermal reactors and is, therefore, also used as fissile material. It is first produced from the uranium during the burnup of the fuel and then partially consumed again through nuclear fission. Depending on how long the fuel is used in the reactor, plutonium contributes to energy generation to varying degrees and an increasingly higher concentration of plutonium remains in the spent fuel. In addition to plutonium, other heavy elements, such as americium and curium, are also produced in lower concentrations. These “minor actinides” can also be fissioned and will (also) be used as fissile material in various future reactor concepts.

Thorium is another naturally occurring chemical element that does not contain fissile material, but from which new fissile material can be produced by irradiation in a reactor. When thorium is used in a reactor, the isotope uranium-233 is produced, a good fissile material in both fast and thermal reactors. Since thorium itself does not contain a fissile isotope, another fissile material, either enriched uranium or plutonium, must always be added to the fresh fuel to use thorium as fuel in reactors. For a more detailed discussion of thorium as fuel for future reactor concepts, please refer to (Oeko-Institut e.V. 2017, Kap. 4.1).

In addition to the chemical element used (uranium, thorium) or the main fissile material (uranium-233, uranium-235, plutonium, minor actinides), the chemical form of the fuel can also differ. In current light-water reactors, the fuel is typically used as ceramic fuel in the form of uranium dioxide. However, the fuel can also be used in other chemical forms (carbide, nitride, metal) or as liquid fuel (in molten salts).

To continue using the fissile material produced in the reactor (plutonium and minor actinides in uranium fuel, uranium-233 in thorium fuel) to generate energy, it must be separated from the fuel after reaching its technologically or economically required maximum burnup and processed into new fuel. The separation process is known as **reprocessing**. Ceramic fuels in which separated fissile material is reused are known as mixed oxide fuels (MOX, e.g. uranium-plutonium mixed oxide). Therefore, the utilisation of such fissile materials requires both reprocessing plants and, for reactors with solid fuels, plants for producing mixed oxide fuels. The type of plant and technology required depends on the chemical form of the fuel. Some fast reactor concepts envisage using only natural uranium as fresh fuel and then, in the course of irradiation in the reactor, first producing the plutonium fissile material required to generate energy and then directly consuming it again.

Criticality:

In current nuclear power plants, a continuous chain reaction occurs in which a neutron produced during nuclear fission leads, on average, to a new nuclear fission. Such a reactor is referred to as a **critical reactor**. If the properties of the reactor core change, for example, the composition of the fuel (due to nuclear fission) or the temperature of the coolant (due to changes in heat removal), the average number of new nuclear fissions produced per neutron changes. If the number of neutrons increases continuously over time, the reactor core is described as supercritical; if it decreases, it is described as subcritical.

An independent chain reaction is not possible in a subcritical system. Instead, neutrons must be continuously fed into the system from outside to generate power. For this purpose, particles (typically protons) can be accelerated to high energies in an accelerator and shot at a target. A large number of neutrons per proton are generated in this target through so-called spallation reactions, leading to further nuclear fission in the reactor core. Such a subcritical reactor is therefore referred to as an **accelerator-driven system**.

2.2.3 Exemplary SNR and their properties

Today, many reactor concepts that can be categorised as SNR are being pursued both within the GIF and independently of the GIF. Table 2-2 lists some of these reactor concepts as examples and describes their respective characteristics (GIF 2021a; IAEA 2023d). For further reactor concepts discussed today or in the past, reference is made here to the respective overviews of historical and current developments in the presentation of the technology lines in Chapter 4.

Table 2-2: Exemplary tabular classification of various technology lines and reactor concepts

Technology line / Reactor concept	Moderator			Coolant						Fuel Solid/Liquid					Fissle material				Temp. > 600°C	Pressure		Neutron-spectrum		Trans-mutation		Type	
	Gr	W	D	W	S	L	Fl	Cl	G	M	C	Tr	Fl	Cl	U	Th	Pu	MA		HP	LP	F	T	BR	BU	TL	DS
SFR					X					X	X				X		X	X			X	X		X	X	X	
BN-1200					X						X				X		X				X	X		X			FP
ASTRID					X						X				X		X				X	X		X			DP
CFR-600					X						X				X		X				X	X		X			DP
TWR-P					X					X					X						X	X					FP
VHTR	X								X			X			X	X	X		X		X		X			X	
HTR-PM	X								X			X			X	X			X		X		X				DP
Prismatic HTR	X								X			X			X				X		X		X				DP
GFR									X		X				X		X	X	X		X	X		X	X	X	
ALLEGRO									X		X				X		X				X	X					EX
LFR						X					X				X		X	X			X	X		X	X	X	
BREST-OD-300						X					X				X		X				X	X					DP
ALFRED						X					X				X		X				X	X					DP
SCWR		X	X	X							X				X		X			X		X	X			X	
CSR1000		X		X							X				X					X			X				FP
Canadian SCWR			X	X							X					X			X	X							FP
MSR	X						X	X			X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	
MCFR								X						X	X		X		X		X	X		X	X		DP
DualFluid								X						X	X	X	X	X	X		X	X		X	X		FP
IMSR	X						X						X		X	X	X		X		X		X	X			DP
ADS					X	X				X	X				X		X	X			X	X		X	X	X	
MYRRHA						X					X				X		X				X	X					EX

Moderator - Gr: Graphite, W: Water, D: Heavy Water. Coolant – W: Water, S: Sodium, L: Lead, Fl: Fluoride salt, Cl.: Chloride salt, G: Gas. Fuel – M: Metal, C: Ceramic, TR: TRISO, Fl: Fluoride salt, Cl.: Chloride salt. Fissile material – U: Uranium, Th: Thorium, Pu: Plutonium, MA: Minor actinides. Pressure – HP: High pressure, LP: Low pressure. Neutron-spectrum: F: Fast, T: Thermal, Transmutation – BR: Breeder, BU: Burner. Type – TL: Technology line, DS: Development status, EX: experimental (also less than 300 MW electrical power), DP: Demonstration plant (also less than 300 MW electrical power), FP: First prototype.

2.3 Overview of the technology lines considered and assigned reactor concepts

The aim of the systematisation undertaken here is to obtain a list of technology lines that is as comprehensive as possible and within which all important reactor concepts currently being pursued in the field of SNR could be classified. At the same time, the subdivision is intended to ensure that the fundamental advantages and disadvantages of such reactor concepts resulting from the choice of the respective technology line can be discussed, see Chapter 4.

However, the actual feasibility of fundamental advantages and disadvantages often only emerges at the level of specific reactor concepts. For this reason, the extent to which fundamental advantages and disadvantages can be implemented at the level of the reactor concepts according to the current state of knowledge should be discussed for all technology lines, at least on the basis of a specific reactor concept pursued, see Chapter 5. Within the technology lines, there may be further significant differentiation criteria between different reactor concepts that can be assigned to the technology line. For some technology lines, several reactor concepts are therefore considered in Chapter 5.

For the further investigation, a distinction is made between the following technology lines on the basis of the differentiation criteria discussed in the last chapter, in accordance with the systematisation presented in Table 2-3.

In addition to the purely technical differentiation criteria, other aspects such as the readiness level of the respective reactor concepts and an exemplary treatment of developments in different countries or by different players were also taken into account when selecting the reactor concepts to be examined in more detail within a technology line.

Table 2-3: Systematisation of technology lines and associated reactor concepts in the field of SNR

Technology line	Differentiation criteria				Reactor concept/
	Criticality	Coolant	Moderation	Other characteristics	Plant
ADS	No				MYRRHA
SCWR	Yes	Water			CSR1000
SFR		Sodium		With RP	BN-800
				Without RP	TWR
LFR		Lead			Brest OD-300
GFR		Gas	No		GFR
VHTR			Yes	Spherical FE	HTR-PM
				Prismatic FE	Prismatic HTR
MSR		Salt	No		MCFR
			Yes		LFTR

Source: Own classification, RP: Reprocessing, FE: Fuel element

The technology line of accelerator-driven systems (ADS) is characterised by a subcritical reactor core. Within the technology line, many different basic distinctions are currently still possible with regard to planned coolants or the use of moderators as well as other special features, without clearly defined essential reactor concepts currently emerging. The MYRRHA plant currently planned in Belgium is used as an exemplary reactor concept for ADS.

All other technology lines are critical reactors. The coolant used is taken as a key distinguishing feature in accordance with the subdivision of the GIF. The use of different coolants results in significant technological differences, for example with regard to the materials and components used for the fuel and the cooling circuits.

The technology line of supercritical water-cooled reactors (SCWR) uses water as the coolant. There are concepts based on light water for cooling and as a moderator, but concepts based on heavy water are also under investigation. The variant of the Chinese supercritical water-cooled reactor CSR1000 being pursued in China is used as an exemplary reactor concept.

Various countries are pursuing the introduction of sodium-cooled fast reactors (SFR), and such plants currently exist primarily in Russia. There, the corresponding reactor concepts are being pursued with the aim of implementing a closed fuel cycle with reprocessing and subsequent utilisation of the plutonium contained therein. For this purpose, the Russian BN-800 concept is used as an exemplary reactor concept. An alternative fuel strategy is being pursued by a project of the American company TerraPower. In contrast to other SFR, their travelling wave reactor (TWR) concept is intended to completely dispense with reprocessing.

A demonstration plant for the lead-cooled fast reactor (LFR) technology line is currently under construction in Russia. The underlying reactor concept of the BREST-OD-300 is therefore used as an exemplary reactor concept for this technology line.

A distinction is made between two technology lines in the field of gas-cooled reactors. Gas-cooled fast reactors (GFR) work with a fast neutron spectrum. The GIF reference concept, which is also referred to as GFR, is used as the reactor concept for this.

In contrast, the very-high-temperature reactor (VHTR) technology line works with a thermal neutron spectrum. Graphite is used as the moderator in this technology line. To simultaneously enable a high working temperature of the coolant, a special fuel is used in this technology line in which uranium is used in small particles with a diameter of only about one millimetre, which are covered by two layers of pyrolytic carbon and a layer of silicon carbide (SiC). These particles of fuel and coating layers are called TRISO fuel (tristructural-isotropic) and are surrounded by a graphite coating.

Within this technology line, two main reactor concepts can be distinguished. In so-called pebble-bed reactors, the TRISO particles are embedded in graphite spheres. These spheres represent the fuel elements of a pebble-bed reactor. The HTR-PM pursued in China, for which a demonstration plant went into operation in 2022, is used as an exemplary reactor concept for this. Alternatively, the TRISO particles can also be embedded in graphite blocks as fuel elements. For this, the reactor concept of the Prismatic HTR developed by General Atomics in the U.S. is used as an exemplary reactor concept.

In the molten salt reactors (MSR) technology line, molten salt is used for cooling. A very large number of different reactor concepts are still being discussed within this technology line. An important differentiation criterion is the choice of a fast or thermal neutron spectrum. Accordingly, the MCFR for an MSR with a fast neutron spectrum and the LFTR for an MSR with a thermal neutron spectrum, both of which are being developed by companies in the U.S., are used as exemplary reactor concepts.

Table 2-4 summarises important properties of the exemplary reactor concepts. In addition to the assignment to the technology line based on the features described in Chapter 2.2.2, such as the coolant used, the selected moderator and the resulting neutron spectrum, other characteristic properties such as the planned output of an individual reactor and the intended fuel are also indicated. The table also refers to the country of origin in which the development is being driven forward and the developer of the concept.

Table 2-4: Properties of selected exemplary reactor concepts

Tech-nology line	Name (abbreviation)	Country	Developer	Electrical/ thermal power [MWe/MWth]	Cool-ant	Mode-rator	Fuel	Spec-trum
SFR	BN-800	Russia	Rosen-ergoatom	885 MWe	So-dium	-	MOX	Fast
	Travelling wave reactor (TWR)	USA	Terra Power	1200 MWe	So-dium	-	U-10%Zr	Fast
LFR	Brest-OD-300	Russia	RDIPE	300 MWe	Lead	-	Nit-ride	Fast
GFR	GIF reference concept (GFR)		GIF	1150 MWe	Heli-um	-	Car-bide	Fast
MSR	The Liquid Fluoride Thorium Reactor (LFTR)	USA	Flibe Energy	250 MWe	Fluo-ride salt	Graphi te	U, Th salt	Ther-mal
	Molten chloride fast reactor (MCFR)	USA	Terra Power	800 MWe	Chlo-ride salt	-	U, Pu salt	Fast
SCWR	Chinese supercritical water-cooled reactor (CSR1000)	China	NPIC	1000 MWe	Light-water	Light-water	UO2	Ther-mal
VHTR	High-temperature gas-cooled reactor pebble-bed module (HTR-PM)	China	China Huaneng and China National Nuclear Corp	250 MWth	He-lium	Graphi te	TRISO	Ther-mal
	Prismatic modular high-temperature GCR (Prismatic HTR)	USA	General Atomics	150 MWe	He-lium	Graphi te	TRISO	Ther-mal
ADS	Multi-purpose hYbrid Research Reactor for High-tech Applications (MYRRHA)	Bel-gium	Belgian Nuclear Research Centre (SCK.CEN)	100 MWth	Lead-bis-muth	-	MOX	Fast

Source: Own evaluation, see also the respective chapters for a detailed description of these reactor concepts

2.4 Technological maturity

Technological maturity is an important parameter for assessing the technological risk on the way to the final commercialisation of a reactor concept. It serves as an indicator for risks in the further research and development process. Reactor concepts that only exist on paper have a much higher risk of unexpected problems occurring. This risk is lower for those for which large-scale tests or prototypes have been built and successfully operated. Technological maturity can also be seen as a qualitative indicator of the research and development (R&D) costs still to be incurred.

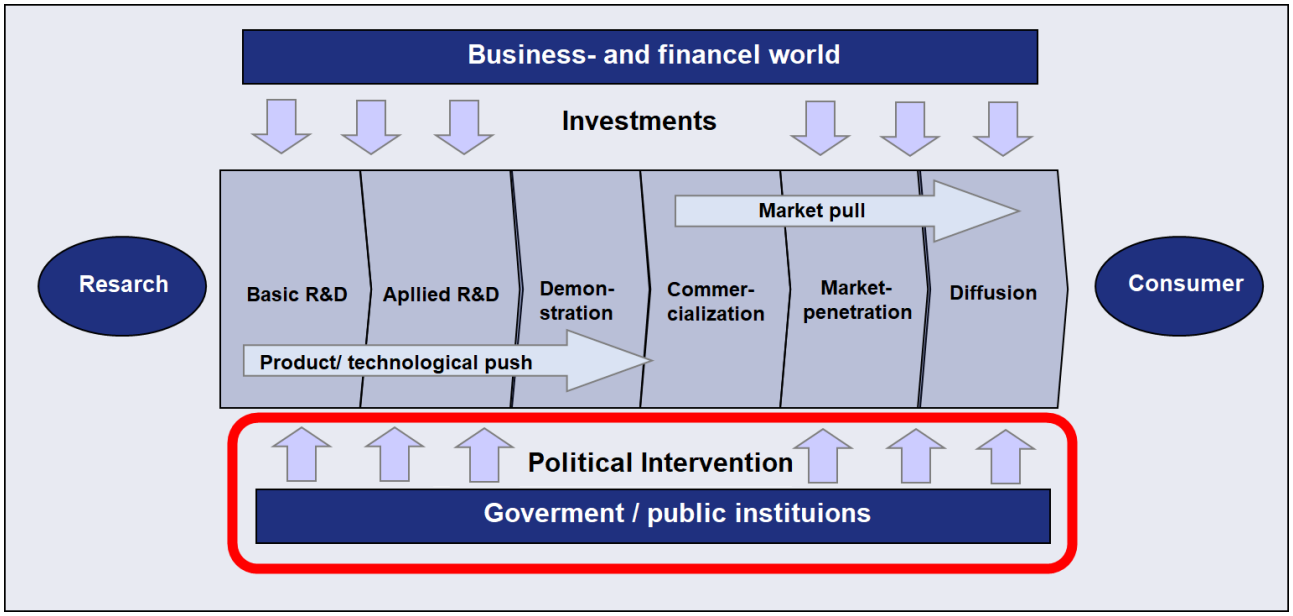
The aim of the following descriptions is to present the state of the literature on innovation and technological maturity in order to enable different reactor concepts to be categorised on this basis. Firstly, the innovation process and the background to the evaluation of innovation activities based on technological maturity, described by so-called technology readiness levels (TRL), are explained. Common TRL scales are then summarised and compared. Two examples are used to illustrate the application of the TRL scales and examples of the TRL evaluation of nuclear energy technologies are presented. Based on this, Chapter 2.6.1 describes the procedure used in this study to evaluate technological maturity.

2.4.1 The structured innovation process (as the basis for the TRL evaluation)

Innovation is the successful utilisation of new ideas. At the interface of technology, society and politics, innovation processes play a key role in the economic efficiency of companies and national economies. A better understanding of the innovation process aids in appreciating the dynamics of the development and application of innovations. Joseph Schumpeter (Schumpeter 1993), an early innovation and development theorist, identified different stages of the innovation process from invention to innovation to diffusion within a national economy. Since then, this process has been depicted in S-curves, among other things. In most cases, however, a technology ends after the invention because the innovation or market diffusion turns out to be infeasible. However, Schumpeter did not differentiate between the driving forces of innovation.

When analysing innovations, a distinction should be made between the supply side and the demand side: the most original innovation may turn out not to be marketable. (Stern 2007) presented the stages of the innovation chain on the basis of (Grubb 2004) in expanded form in his review “Economics of Climate Change”. This process of innovation is illustrated in Figure 2-1.

Figure 2-1: Innovation process between “technology push” and “market pull”



Source: Own illustration based on data from (Stern 2007)

The text distinguishes between research-driven (from the left) and demand-driven (from the right) product development. Research-driven product development can be divided into basic research and development, applied research and development and demonstration. Demand-driven product development (market pull) can be divided into commercialisation, market penetration and market diffusion. Development is driven by government intervention and investment from the private sector via technology push (from research) and market pull (from consumers). (Stern 2007) notes that this is a major simplification of the innovation process: The innovation system is certainly more complex. However, the figure helps to clarify the stages of the innovation process.

2.4.2 Classification of technological maturity (technology readiness level, TRL)

The concept of TRLs has its origin in aerospace. Stan Sadin developed a TRL scale in 1974 for the National Aeronautics and Space Administration (NASA) to categorise innovations in space and rocket research. The motivation was to develop a scale to support targeted research and development programmes to improve technology (Sadin et al. 1989). The advantage of the new 7-level scale was the creation of a structured, criteria-based and documented evaluation of a technology with concrete measures to reduce risks in the development of large-scale facilities of a technology (upscaling, scale-up) and a comparability of the readiness level with other technologies. Each of the stages was provided with a brief definition and explanation. The scope and system boundaries of the analysis differ depending on the TRL definition. In the original NASA definition, systems are also analysed, while in the EU definition the focus is on specific technologies. The TRL scale was finally extended from seven to nine levels in 1995 (NASA 1995 (Edited 2004)). A comprehensive presentation of the history can be found, for example, in (Mankins 2009). Due to the expansion of the system boundary for functional technologies in space, consideration is being given to the extent to which a further level (TRL 10) that also includes applications other than those originally envisaged makes sense (Straub 2015).

While starting in the field of aerospace, the TRL scale is also used in other areas, e.g. in technology assessment in industry (e.g. energy and chemicals). A comparison of different TRL scales and definitions of concepts can be found in the report by (EPRI 2017) or in the detailed descriptions of the currently used TRL scale, its definition and criteria in (DOE 2011). An overview of the TRLs from 1 to 9 and the corresponding description of the stages of invention, validation and market expansion can also be found in (EARTO 2014).

(Olechowski et al. 2015) compile the experiences from interviews with users of TRL scales in a report. In addition to a historical and case study-based analysis, these also include a compilation of the challenges of TRL scales. They can be divided into three categories: complexity of the system, planning and review, and validity of assessment.

The TRL scale is also used to categorise energy technologies. For various funding programmes (e.g. showcase projects of the BMWK [German Federal Ministry for Economic Affairs and Climate] in the field of energy), a classification of the technology using TRL is used to promote technologies at certain stages of their development. The Research Centre of the European Commission has published an overview of the criteria and definitions of technology-specific TRLs and a classification for various technologies in the field of renewable energies (Strazza et al. 2017; Rose et al. 2017). The European Commission’s TRL scale is now being used to evaluate and compare energy innovation projects under the Horizon2020 programme and to include them in funding programmes. The guidelines were developed in a participatory process in which the criteria and principles were reviewed and discussed by stakeholders from science and practice. This TRL scale (see Table 2-5) is based on the scale developed by NASA and ranges from the observation of basic principles to the operational function of the technology.

Table 2-5: Definitions of the technology readiness level (TRL) from the current European Commission guidelines

TRL	Description
TRL 1	Basic principles observed
TRL 2	Technology concept formulated
TRL 3	Experimental proof of concept provided
TRL 4	Technology validated in lab
TRL 5	Technology validated in relevant environment
TRL 6	Technology demonstrated in relevant environment
TRL 7	System prototype demonstration in operational environment
TRL 8	System complete and qualified
TRL 9	Actual system proven in operational environment

Source: (Strazza et al. 2017)

2.4.3 Classification of the TRL literature and alternative evaluation approaches

The evaluation of technologies using the TRL scale has limitations that can be divided into three categories: Limited information availability, linearity and subjectivity of the criteria.

The information required to evaluate a technology is not always (publicly) available or cannot be used for the technology assessment. Different levels of the TRL have different information requirements (see also the comprehensive analysis of information requirements and uncertainty in

(EPRI 2017)). Information from the company and publicly available information can be used for an evaluation. Classification into TRL levels is usually easier with information and data available within the company than a technology assessment based on publicly available information. Aggregated categories are therefore used to summarise several TRLs into one group to simplify matters.

A further limitation of the TRL scale is that, like the schematic innovation process, the TRL scale suggests a linearity in the progress of a technology. However, development depends on numerous factors, including investments, technology development risks and market barriers. In this respect, progress from TRL 1 to 2 is not comparable with a change from TRL 6 to 7.

Thirdly, the selection of criteria and categories used for the evaluation plays a central role. The definitions and criteria catalogues of different TRL scales differ, and the criteria must be evaluated on a technology-specific basis. A good overview of the complexity of the evaluation process and the different levels of evaluation is provided in the handbook of (DOE 2011) and the European Commission (Strazza et al. 2017).

In addition to the TRL scale, there are also various more or less widespread readiness levels. Some relate to the market or the investment environment, such as the innovation readiness level and the market readiness level. Other scales deal with the technology and its environment in the production system. These include the “manufacturing readiness levels” (Ward et al. 2012), “system readiness levels” (Sauser et al.) and “scientific readiness levels” (Baron et al. 2019).

These maturity levels and classifications are usually based on a comprehensive catalogue of criteria used to evaluate the technologies or the market for the technologies. When categorising market maturity levels, the end of the innovation chain is a competitive product available on the market. In contrast to the TRL scales, there is currently no widely used standard or catalogue of criteria for market maturity levels. To analyse the market, the entire market environment must be examined (e.g. with market environment analysis tools such as Porter’s Five Forces and market potential analyses). A study by (Oeko-Institut e.V. 2017) also examines market aspects for new reactors (e.g. utilisation for thorium). This reveals the lack of market demand for nuclear reactors and the resulting lack of a basis for innovation processes motivated by the private sector. Finally, a scientific readiness level is also used. This can be used, for example, to illustrate that engineering and material science problems lead to a low TRL, even though a technology is largely scientifically mature (see e.g. (Baron et al. 2019)).

2.4.4 TRL scales in nuclear energy research and applications

No standard has yet been established in the technological evaluation of nuclear reactor research and innovation status. In particular, there is currently no standardised TRL scale for nuclear energy technologies for commercial power generation that is used to compare different technologies. However, there are various approaches to this. The first question is which technologies should be compared with each other and under which aspects. Reactors such as pressurised water and boiling water reactors should be classified at TRL 9 on such a scale, as they can produce electricity and heat on a large scale. They are also well-advanced in the diffusion of reactor technologies.

According to (Carmack et al. 2017), there are two dimensions that are taken into account in the TRL evaluation of nuclear energy technologies: These include the maturity of the fabrication process and the maturity of fuel performance. The evaluation of the fabrication process includes the quality of the materials used on the one hand and the quantity on the other. (Shepherd et al. 2015) has carried out an approach to analyse and compare different fuels. In the conference paper, the scientists from

the UK National Nuclear Laboratory compare and contrast different fuels for different reactor types and operating temperatures. However, evaluating the fuel itself is not sufficient to assess the safety and long-term provision of electricity and heat. For this, the system must be considered, from the extraction of the raw materials to the safe storage of the fuel. As a result, TRL evaluations are highly specific and only partially transferable between individual technology lines.

An initial attempt to evaluate selected new reactor concepts and the sub-technologies (including heat supply, heat transport, and electricity generation) can be found in the technical report by (INL 2015). A classification and evaluation of the current research funding of the United States Department of Energy (DOE) and the action programmes for nuclear energy technologies can be found in (Dixon et al. 2018). An example of a classification of reactor components into technology and market readiness level can be found in a presentation by (Nygaard 2021), for reprocessing processes as a result of a workshop of the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development (OECD) in (Baron et al. 2019).

The GIF has also distinguished three phases for the development of its favoured systems: a feasibility phase, a development phase and a demonstration phase (GIF 2014). The first viability phase investigates the fundamental feasibility of the reactor concept. In this phase, a rough concept is developed, show-stoppers are identified, and it is shown whether these can be overcome. Basic research is carried out. In the second development phase (GIF calls this phase the “Performance Phase”), the necessary technologies and systems are developed, their interaction is analysed and their viability is demonstrated. Typically, experimental and demonstration plants are built. In the subsequent demonstration phase, if the first phase is completed successfully, the system is licensed and a first fully developed prototype reactor is built.

2.4.5 Interim conclusion: Complexity reduction and alternative technology assessment approaches

In view of the complexity of different technology lines and reactor concepts in the field of nuclear technology, a standardised evaluation based on TRLs is highly complex. Therefore, for the evaluation of complex innovations with limited data and information availability, the question arises as to the extent to which the technologies to be assessed should be classified using other scales. Table 2-6 compares the nine-level TRL scale with various alternative classifications.

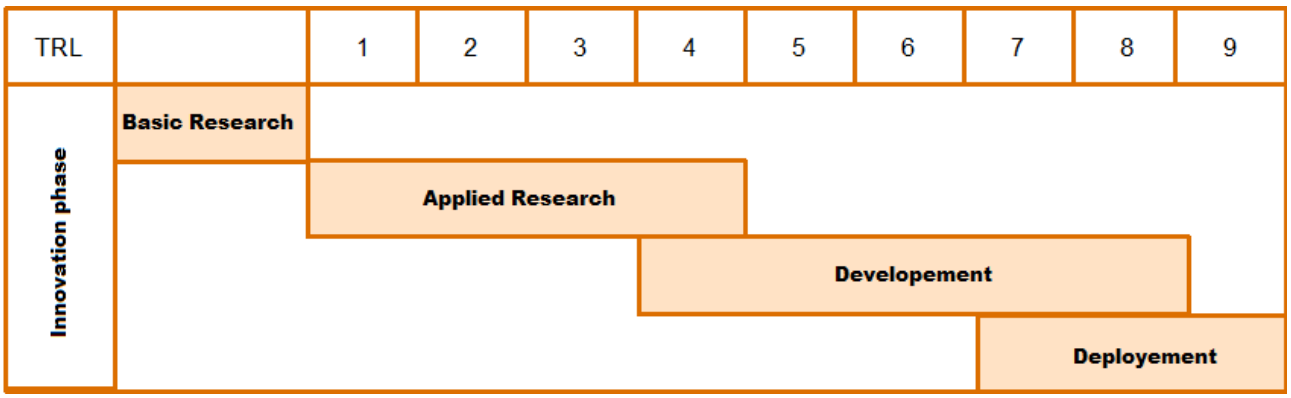
Table 2-6: Comparison of alternative TRL classifications

TRL	Division into phases	Alternative classifications	Research	Fine division
1	“Viability phase” - basic research and concept development	Research phase	Basic research	Functional principles observed and described.
2			Applied research	Technological concept described.
3				Experimental proof of the functional principle.
4	“Performance phase” - proof of viability	Prototype development		Technology confirmed in laboratory tests.
5			Development	Technology confirmed under relevant operating conditions.
6		Demonstration of the prototype		Technology demonstrated under relevant operating conditions.
7	“Demonstration phase” - proof of performance			Prototype demonstrated under operating conditions.
8		Technology confirmed under relevant operating conditions.	Deployment	System complete and qualified.
9				Proven use under operating conditions (commercial).

Source: Own further development on the basis of (Oeko-Institut e.V. 2017)

One possible approach is the four-stage categorisation shown in Figure 2-2. This illustrates the areas of basic research, applied research, development and deployment. Basic research precedes the actual technology development.

Figure 2-2: Comparison of the 9-level TRL scale and a three-level technology classification



Source: Own illustration

The discussion of innovation processes in reactor technology is limited to research-orientated technology development and “ends” with (several) demonstration plants. In contrast, demand-driven market processes have not yet been taken into account in the supply process.

This regularity, which has already been observed with conventional light-water reactors, may be more significant with so-called “novel” reactor concepts. This is a specific feature of nuclear energy: There is no market-driven demand for nuclear reactors; instead, this is based entirely on public supply decisions. This was already the case in the 1950s (Baade 1958) and is still the case today (Davis 2012; Rothwell 2000; 2022). It is therefore also necessary to question whether the principle of a lack of market incentives for the innovation process will change anything. For the second phase, market penetration, a comparison with other technologies that produce the same products as nuclear reactors (heat and electricity) and with the associated system, environmental and disposal costs would be necessary. However, a comprehensive discussion of these aspects is not the subject of the analyses carried out here.

2.5 Economic aspects of nuclear energy

This chapter classifies the economic aspects of nuclear energy. Based on an examination of established light-water reactors, different estimates of the future role of so-called “novel” reactor concepts and the framework conditions for these are discussed. Reference is made to more detailed descriptions in (Zenodo 2021) and (Hirschhausen 2023), among others. Based on this, Chapter 2.6.5 describes the procedure used in this study to evaluate the costs of SNR.

2.5.1 Historical cost development of light-water reactors

One of the main reasons for dealing with so-called “novel” reactor concepts is that the light-water reactor, which has played a pivotal role in the development of nuclear energy, has not become cost-competitive on the electricity markets in recent decades. At the beginning of the nuclear age, commercial nuclear power plants were more expensive than their direct competitors, especially coal-fired power plants. For example, the levelised cost of electricity of the first commercial US nuclear power plant in Shippingport (Pennsylvania) in 1957 was around seven times higher than that of a coal-fired power plant in the region (Baade 1958).

The structural cost disadvantage of nuclear power has persisted since those early days (MIT 2003; University of Chicago 2004) and continues to the present (MIT 2018). Lucas Davis of the University of California at Berkeley concludes that despite “a certain confluence of factors that could theoretically make nuclear power a viable economic option”, there is still no economic case for nuclear power. Davis points to economic efficiency analyses for nuclear power that have shown values in the range of over 10 US cents/kWh compared to around 5 US cents/kWh for electricity from natural gas or coal (Davis 2012).

Private investment in nuclear power plants under competitive conditions has therefore not taken place (DIW 2018). After the early phase of purely military development in the 1940s, a few countries were able to attract private companies to develop, build and operate nuclear power plants, for example in the U.S. and Germany. However, these were always provided with considerable financial security or direct subsidies (Radkau 1983, 178, 199 ff). In most countries, the development, construction and operation of nuclear power remained in the hands of the state, including in the UK and France (Hirschhausen 2023, chapter 4).

Contrary to original expectations, the construction of nuclear power plants has not become cheaper over the decades; on the contrary, the costs (per kilowatt (kW) of installed capacity) have risen continuously, even when adjusted for inflation. The trend towards cost increases initially observed for the U.S. (USA 1978; Mooz 1980; Joskow 1982; DOE 1986) was later confirmed for other

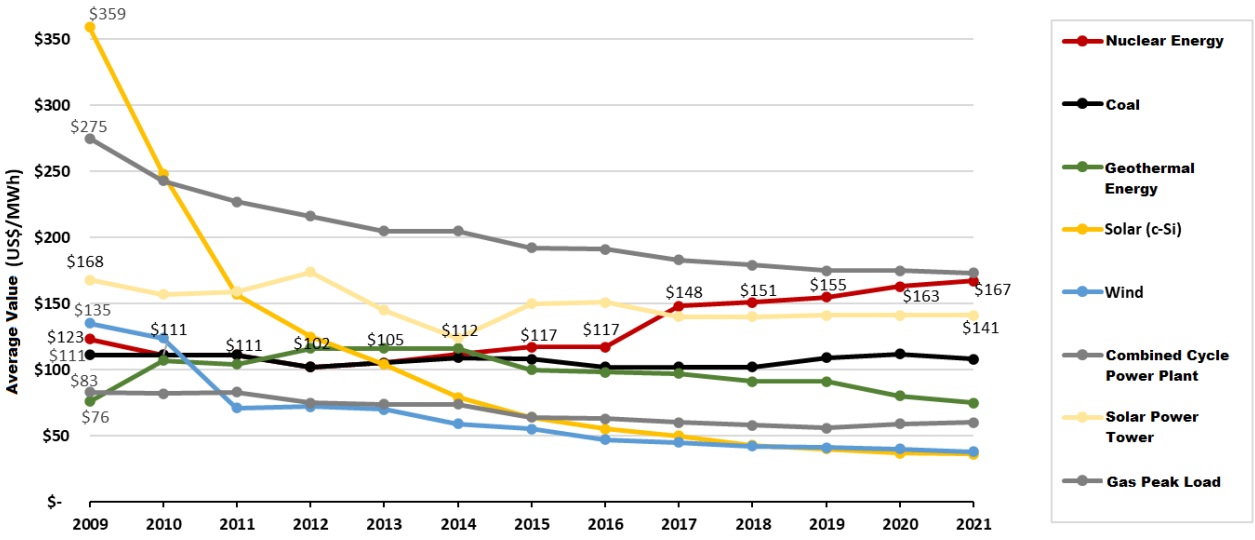
countries, including France (Grubler 2010, pp. 5147–5188; Escobar Rangel and Leveque 2015, pp. 103–126). The French nuclear programme, which took place under stable institutional conditions and with a high degree of standardisation of reactor types, also exhibited cost escalation. Units completed after 1990 were 3.5 times more expensive per installed capacity than the first reactors in the 1970s (Grubler 2010).

2.5.2 Current costs of light-water reactors

The costs of the European Pressurised Reactor (EPR) in Olkiluoto (Finland, on the grid since 2022) and Flammanville (France, commissioning planned for the mid-2020s as of the end of 2022) are still considerably higher than those of the previous generation.[2] Although these are “first-of-a-kind” plants, i.e. pilot projects being implemented for the first time, it is not foreseeable that costs will fall sharply as a result of a massive roll-out; there are currently only three orders (Hinkley Point C, 2 reactors, and Sizewell C with one reactor, both in the UK), which could go online in the 2030s. The same applies to the last two remaining reactor constructions in the U.S. with the “new” AP-1000 model from Westinghouse (Hirschhausen 2023).

Instead of the fossil fuels of the past, renewables, particularly solar and wind, are now cost-effective competitors to nuclear energy (ISE 2018; 2021; Wealer et al. 2021). The investment bank Lazard (Lazard 2022) estimates the levelised cost of electricity from nuclear power at around 16 US cents$_{2021}$/kWh, far higher than solar and wind energy (each around 4 US cents$_{2021}$/kWh), see Figure 2-3.

Figure 2-3: Levelised costs of electricity from various generation technologies (2009-2021)



Source: Own illustration based on data from (Lazard 2022)

[2] For example, the costs of the Olkiluoto-3 nuclear power plant in Finland rose from an original estimate of three billion euros (1995) to over eleven billion euros. This corresponds to around EUR 7200 per kW (as of 2018). Lower cost estimates are available for the two EPRs built in China, which could not be verified as part of this study (Mycle Schneider Consulting 2019, p. 66).

2.5.3 Composition of the costs

The costs of generating electricity in a new power plant are made up of three basic components: The capital costs, the fixed operating costs and the variable operating costs, in particular the cost of fuel (MIT 2018). In the current literature, capital costs are made up of two components: Firstly, there is the “overnight construction cost” (OCC), which includes technical equipment, materials, etc., but is considered independently of the construction period; secondly, there is the cost of interest, which is incurred when funds are borrowed for the construction of the plant (either as a loan or as equity) (MIT 2018). Capital costs account for the largest share of the costs of electricity production in a nuclear power plant at more than 80%, while fuel costs account for around 5% and the remaining operating costs for around 15% (MIT 2018). The costs for dismantling and disposal, on the other hand, are not included in these cost analyses to date.

The annual energy cost analysis published by the U.S. investment bank Lazard provides an illustration of current capital cost trends. It confirms that capital costs not only account for a significant proportion of total costs, but are also on the rise. Capital costs have risen by around 4% annually in recent years and are now in the range of USD 7,800-12,800/kW of installed capacity. Further analyses of cost structures can be found in (Rothwell 2016) and (IAEA 2016g). Despite differences in detail, the studies agree that capital costs account for by far the largest share of the levelised costs of electricity and that the share of fuel costs is low.

In contrast, the trend towards falling capital costs for renewable energies is continuing. In the period between 2010 and 2021, they fell by an average of around 7% per year for photovoltaics and 3% per year for wind power plants (Lazard 2022).

For future costs for nuclear power plants, current energy and climate scenarios in the 6th Assessment Report of the Intergovernmental Panel on Climate Change (IPCC) assume average capital costs of between USD 6778/kW and USD 12,075/kW for the period between 2020 and 2100 (Zenodo 2022). As a result, no cost degression is to be expected. No allocation to light water or so-called “novel” reactor concepts is made in this publication.

2.5.4 Developments in the energy industry

In 2021, around 102 exajoules (EJ) of electricity were generated worldwide, of which around 62% was provided by fossil fuels, around 28% by renewable energy sources (including around 4% PV, 7% wind and 15% hydropower) and around 10% by nuclear power plants (BP 2022). In 2021, 418 nuclear power plants in 33 countries were connected to the grid with an average age of 31 years and a capacity of 369 GWe, an increase of 1.9% on the previous year (Mycle Schneider Consulting 2021).

Estimates of the future importance of nuclear power vary greatly and are based primarily on different assumptions regarding cost trends. There are scenarios with increasing shares of nuclear power and others with sharply declining shares: Scenarios in which electricity generation from nuclear power increases assume that capital costs will fall. In its study “Reduction of Capital Costs of Nuclear Power Plants”, the OECD/NEA (NEA 2000a) described back in 2000 that nuclear energy could continue to be a viable option in the next century, but that the cost of electricity from nuclear power plants would have to be competitive with alternative energy sources. The areas to be considered for fundamental changes are described here as increasing capacity, exploiting new construction methods, shortening construction times, design improvements, improvements in procurement,

improvements in organisation, improvements in contract management, standardisation and series production and improvements in regulation and political measures (NEA 2000a).

On the other hand, taking current cost trends for nuclear energy into account, the share of nuclear energy will decline towards 2050. Model calculations for this included an hourly resolution and a very detailed mapping of sector coupling (Löffler et al. 2017; Teske 2018; Bogdanov et al. 2019). They also allow the system costs of fluctuating renewables to be estimated.[3]

2.5.5 General conditions for the market launch of SNR

From an energy and system economics perspective, SNR would not only have to make up for the increasing competitive disadvantage of light-water reactors compared to renewable energies, but would also have to undergo a complete system change in terms of upstream services and industrial infrastructure. This system change would also be associated with high costs. In this regard, the Generation IV International Forum (GIF 2002) already described 20 years ago in its Technology Roadmap as necessary changes that the problem of nuclear waste disposal would have to be controllable, fuel utilisation would have to be higher and the economic efficiency of the systems would have to be competitive. Progress would also need to be made in the area of safety. In 2018, MIT stated in its study that the use of inherent and passive safety features in SNR could improve overall safety and operation, but that the economic potential has not been proven here either (MIT 2018).

In its study “Unlocking Reductions in the Construction Costs of Nuclear: A Practical Guide for Stakeholders” in 2020, the NEA identifies the maturity of the design, effective project management, stability and predictability of regulation and the utilisation of series effects as cost-influencing factors for the construction of new reactor technologies that are not yet on the market and describes considerable uncertainties regarding the influence of these factors on SNR (NEA 2020a).

2.5.6 Developments in the area of uranium supply

Many SNR rely on improved commodity economics and the assumption that uranium may one day become scarce and very expensive. In the early days of the nuclear industry, it was assumed that uranium would remain permanently scarce. This understanding shaped both the assessment of the reactor technologies to be developed, especially in the direction of a plutonium economy, i.e. the development of fast breeder reactors with high raw material utilisation, as well as the geopolitical positioning between those countries that had access to uranium or enrichment technologies and other countries.

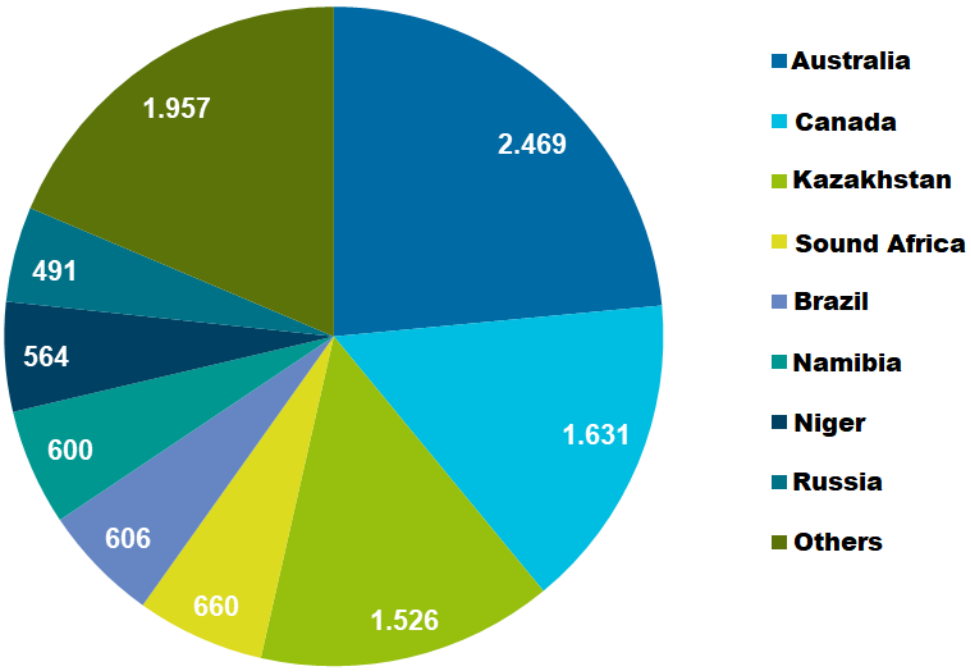
In the early phase, only the U.S., Canada and the Soviet Union had uranium reserves, plus Belgium, whose colony, the Congo, supplied some of the uranium for the Manhattan Project. (Szilard 1947) feared that the U.S. would not be able to import more than 400 tonnes of uranium, which would only be enough to supply two larger nuclear power plants. In Germany, too, which had experimented with heavy water reactors and natural uranium (from the Czech Republic) in the 1940s due to a lack of uranium enrichment technologies, the development of fast breeder reactors and the plutonium economy was the top priority in the post-war period.

[3] These are the costs incurred for the hourly servicing of demand beyond the pure electricity or energy generation costs, including electrolysers for the production of hydrogen, power lines and storage. These are generally around 50% of the direct electricity generation costs, up to a maximum of 100%.

By the 1970s at the latest, however, the situation had turned into the opposite: on the one hand, a large number of producing countries and sites had been developed on the supply side, and on the other hand, the expected boom in nuclear power plant construction had failed to materialise on the demand side (Holdren 1974). This trend has continued to intensify since the 1980s and has led to a situation in which uranium supply from several countries (NEA 2020b) is still faced with constant and relatively low demand. Accordingly, prices for internationally traded uranium have remained moderate. With the exception of a few cyclical outliers, they have remained below USD 100/kg. Due to the sharp rise in the capital costs of nuclear power, the share of resources in total costs has also fallen sharply and is now in the region of a few per cent (Lazard 2022).

Figure 2-4 shows the distribution of uranium reserves by country. From a current perspective, there is no reason to assume a reversal of the trend towards scarce, and therefore very expensive, uranium (NEA 2020b; DIW 2022).

Figure 2-4: Distribution of uranium reserves by country in thousands of tonnes (2019)



Source: Own illustration based on data from (NEA 2020b).

The recoverable uranium reserves currently amount to around 8 million tonnes (NEA 2020b). Based on current demand of around 60,000 tonnes per year, this would be sufficient for over 130 years. Even in the (unlikely) event of a sharp increase in demand, no bottlenecks are to be expected, as a further approx. 7 million tonnes of forecast or speculative resources are still available. In addition, a rapid adjustment of production capacities has been observed historically, i.e. as prices rise, investments in exploration increase sharply and supply is expanded within a few years.

2.6 Evaluation criteria

Essential criteria for the evaluation of nuclear power plant technologies were introduced in (Oeko-Institut e.V. 2017). Some of these were revised and expanded in (Oeko-Institut e.V.; WIP; PhB 2021). Unless otherwise stated, the respective description from (Oeko-Institut e.V.; WIP; PhB 2021; Oeko-Institut e.V. 2017) is used below in the presentation of the evaluation criteria under consideration.

The reference for an evaluation of SNR in the context of the study carried out here are current light-water reactors (LWR). The basic question is therefore to what extent relevant differences are to be expected for a particular criterion compared to LWR of the current generation. It should be noted that many SNR are still at an early stage of development. Only very few prototype or demonstration reactors exist, so many technical details of the reactor concepts will only be finalised and thus evaluated in the near or distant future. Therefore, an evaluation must often remain at a qualitative level or cannot be carried out at all at the present time.

Against this background, the respective evaluation is based on four qualitative levels:

- There is an advantage over today’s LWR
- There is no significant advantage or disadvantage
- There is a disadvantage compared to today’s LWR
- An evaluation is not possible at the present time

Different factors can play a role with regard to an evaluation criterion. These are described in the following descriptions of the selected evaluation criteria. Based on this, representative indicators are named for the factors, on the basis of which the evaluation within a criterion is carried out. If there are deviating evaluations for different indicators within a criterion, an overall evaluation is made on the basis of a qualitative expert judgement.

With regard to the possible implementation in a specific reactor concept, the further question often arises as to the extent to which corresponding evaluations for the technology line can also be transferred to the reactor concepts. This applies if the advantages or disadvantages already arise due to inherent properties of the technology line (e.g. due to the physical or chemical properties of the coolant). In these cases, an evaluation can already be made at the level of the technology line. In many cases, however, the actual feasibility of a corresponding advantage or disadvantage depends on the specific reactor concept design. In such cases, no evaluation can yet be made at the level of the technology line, but this may then be possible at the level of the reactor concept.

It should also be noted that the theoretical advantages of postulated reactor concepts are often compared with the disadvantages of existing systems of other technology lines already in practical use (in this case, LWR). In the actual development to market maturity, however, compromises must always be made, e.g. between safety and costs or improved fuel utilization and proliferation risks, etc., so that theoretical advantages are often not (or cannot be) implemented in the optimised concrete plant design.

Finally, it should be noted that a fully comprehensive evaluation of the technologies, as the following description of the evaluation criteria makes clear, would require comprehensive consideration of a wide range of factors. Such evaluations are practically not available in the literature and cannot be

carried out in the context of this study. The evaluation in this study therefore always focusses on selected properties of SNR that are considered essential for the evaluation.

2.6.1 Technological maturity

When assessing the significance of so-called “novel” reactor concepts, particularly in the context of the energy transition, the question of their realisability in terms of time is of great importance. There are always widely diverging assessments of this, including from the developers of the corresponding technologies themselves.

When the GIF was founded, the goal was formulated that corresponding reactor concepts should be ready for global use by 2030 (GIF 2002, p. 5). In (GIF 2018b), on the other hand, the year 2045 was seen as a possible launch date for the first commercial systems. In its 2021 annual report, GIF again expects the most advanced systems to be launched on the market from 2030, provided that sufficient international political support and ambitious research funding is guaranteed (GIF 2021a, p. 1).

Against this backdrop, technological maturity is an important evaluation criterion for SNR. In contrast to the general assessment of the other criteria (advantage or disadvantage, indifferent), the technological maturity is assessed in three stages, as derived in Chapter 2.4:

- “Applied research”
- “Development”
- “Deployment”

These stages would be followed by the market diffusion stage. An assessment of the respective opportunities for market diffusion cannot currently be made for SNR in view of the great distance to technological feasibility; therefore, at most, comments can be made regarding relevant differences between SNR and LWR with regard to this stage.

The following technology areas, which are essential for the development of a functional nuclear power plant, are used as indicators for assessing technological maturity:

- Fuels and structural materials (e.g. for fuel cladding, structural materials, reactor pressure vessels, cooling circuits)
- Operational system functions (e.g. main coolant pumps, turbines, operational cooling circuits, etc.) as well as inspection and ageing management
- Control and instrumentation technology (operational and safety control technology)
- Safety functions (e.g. reactivity control, cooling, confinement of radioactive materials, with regard to necessary active systems and components)
- Safety demonstration principles (in particular with regard to requirements in authorisation procedures)

The overall technological maturity is categorised in the lowest level of one of the indicators under consideration.

2.6.2 Safety

Fundamental safety requirements for nuclear power plants are developed internationally, in particular by the IAEA as part of the “IAEA Safety Standard Series”, see for example (IAEA 2006b; 2016d). These are subject to continuous development and were revised in particular following the catastrophic accident at the Fukushima-Daiichi nuclear power plant in Japan. The IAEA standards represent an international consensus in the sense of a minimum standard of requirements for the safety of nuclear power plants.

The EU Nuclear Safety Directive 2009/71/EURATOM, last amended by Council Directive 2014/87/EURATOM of 8 July 2014 (Rat der Europäischen Union 2014), formulates in Article 8a as an objective of nuclear safety for nuclear installations that

> *“the national nuclear safety framework requires that nuclear installations are designed, sited, constructed, commissioned, operated and decommissioned with the objective of preventing accidents and, should an accident occur, mitigating its consequences and avoiding:*
>
> *a) early radioactive releases that would require off-site emergency measures but with insufficient time to implement them;*
>
> *b) large radioactive releases that would require protective measures that could not be limited in area or time.”*

This objective for new nuclear power plants in Europe is concretised, for example, in the statement “Safety of new NPP designs” by the Western European Nuclear Regulators’ Association (WENRA), see (WENRA 2013). Requirements for existing nuclear power plants, on the other hand, are formulated in the “WENRA Safety Reference Levels for Existing Reactors” (WENRA 2021b).

The “Bericht der Bundesregierung für die Achte Überprüfungstagung zum Übereinkommen über nukleare Sicherheit im März/April 2020“ (“Report of the Federal Government for the Eighth Review Meeting of the Convention on Nuclear Safety in March/April 2020”) (BMU 2019) provides a good overview of the regulations applicable in Germany; detailed safety requirements are laid down in the “Sicherheitsanforderungen an Kernkraftwerke“ (“Safety Requirements for Nuclear Power Plants”) (BMUB 2015).

The following three fundamental safety functions must be fulfilled to ensure the safety of a nuclear power plant:

- Confinement of radioactive materials: A release of radioactive materials should be prevented by the presence of several sequential barriers and the maintenance of their effectiveness. In current nuclear power plants, these typically include the fuel element cladding, the pressurised enclosure (reactor cooling circuit) and a containment.

- Reactivity control: The release of power in the reactor must be controlled and it must be possible to stop the release of energy at any time so that the integrity of the barriers is not jeopardised. Typically, inherent properties of the reactor (such as negative power feedback when operating temperatures increase or coolant density decreases, etc.) as well as active systems (such as control rods or devices for feeding neutron poisons into the cooling circuit) are used for this purpose.

- Cooling of the fuel elements: the fuel elements must be sufficiently cooled at all times. Even after a reactor is shut down, large amounts of heat are generated by radioactive decay. This must be permanently removed to prevent the fuel from heating up and ultimately causing a core meltdown. A combination of active systems (such as emergency and residual heat removal systems) and largely passive phenomena or devices (such as passive natural circulation in cooling systems or, for example, the largely passive supply of coolant from pressurised storage tanks) is also used for this purpose.

To ensure these safety functions, there are general safety principles such as (BMUB 2015, 3.1 (2)):

- Favouring inherently safe mechanisms in the design
- Use of qualified materials, manufacturing and testing procedures as well as tried and tested or sufficiently tested equipment
- Ensuring and maintaining quality characteristics during manufacture, installation and operation
- Carrying out periodic inspections to the extent necessary for safety reasons

In addition to the safety characteristics of the reactor itself, the possible range of events and incidents must be analysed. Internal events such as pump failure, loss of power supply to equipment, leaks in pipework or internal events such as internal fires can play a significant role. In addition, external events (EVA) must also be taken into account (WENRA 2021b; BMUB 2015). In the case of external events , natural events such as earthquakes (IAEA 2010d), external flooding or extreme weather conditions (IAEA 2011a) must be taken into account. Furthermore, civilisational impacts must be considered, such as an accidental or terrorist-motivated plane crash as well as cyberattacks or sabotage (IAEA 2017a), which in Germany are also recorded under the term “disruptive actions or other interference by third-parties”.

Extensive verification is required to demonstrate the safety of a nuclear power plant, see for example (BMUB 2015, Kapitel 5 sowie Anhang 5). This is based, among other things, on modelling and the use of simulation programs. When applied to SNR, the applicability of these methods and the transferability of the underlying experimental validation of these models and methods must be examined.

Against this background, the following indicators are used to assess safety when discussing technology lines and reactor concepts:

- Differences in normal operation
- Ensuring reactivity control
- Ensuring cooling
- Ensuring the confinement of radioactive materials
- Differences in the event spectrum
- Aspects of verification (especially in the context of authorisation procedures)

In SNR, the term “inherent safety” is often used for the entire reactor concept rather than for individual safety characteristics. However, there are good reasons not to use the term inherent safety without

a corresponding specification for a complete reactor concept. For example, the IAEA initiated a careful review of the definition of safety-related terms in the context of nuclear power plants in 1987, and a technical committee met in 1988 and concluded in the final document (cited in (Englert et al. 2017)):

> *“Potential inherent hazards in a nuclear power plant include radioactive fission products and their associated decay heat, excess reactivity and its associated potential for power excursions, and energy releases due to high temperatures, high pressures and energetic chemical reactions. Elimination of all these hazards is required to make a nuclear power plant inherently safe. For practical power reactor sizes this appears to be impossible. Therefore the unqualified use of ‘inherently safe’ should be avoided for an entire nuclear power plant or its reactor. (IAEA 1991; Martensson 1992)*

At this point, then, the IAEA warns precisely against simplistically applying the term inherent safety to a reactor concept as a whole, since, as a rule, the inherent characteristics of a reactor concept only affect partial aspects of safety.

2.6.3 Fuel supply and waste disposal

Today’s nuclear power plants are operated almost exclusively with uranium or uranium-plutonium fuel. Since the beginning of nuclear energy utilisation, the extent to which natural uranium reserves are sufficient to ensure the long-term use of nuclear energy has been the subject of debate, see also Chapter 2.5.6. Against this background, fuels and reactor concepts have also been discussed for many decades, with which additional fissile material is to be produced. Their contribution in terms of better utilisation of existing uranium reserves or the possible use of thorium as an alternative resource is therefore an important argument in the discussion about SNR.

In addition to the question of resource availability, however, the issue of the necessary industrial infrastructure must always be taken into account. For current nuclear energy utilisation, uranium must be mined, converted, enriched and processed into uranium fuel.

If plutonium produced in uranium fuel is to be utilised further, the spent fuel must be reprocessed and a uranium-plutonium mixed oxide fuel (MOX) must be produced from the plutonium obtained. For SNR, on the other hand, different processing steps or other industrial production processes may have to be developed and made available in corresponding industrial-scale plants.

The operation of a nuclear power plant produces radioactive waste with different properties. The radioactivity of the respective waste is decisive for its handling. In Germany, waste is categorised according to its heat generation, which increases with increasing radioactivity. Waste is classified as heat-generating waste and waste with negligible heat generation. In other countries, other classifications are also used, as properties such as the longevity of a radioactive substance are decisive for its disposal.

Spent fuel elements exhibit a high level of radioactivity and, due to the resulting decay heat power, also a high level of heat generation. Their disposal destination is a repository for heat-generating waste. The regulations of various countries stipulate a reference period for this. In Germany, the reference period according to the Repository Site Selection Act of 2013 is one million years. This type of waste places the highest demands on subsequent final disposal, which is why the quantities arising from nuclear energy utilisation are at the centre of the waste problem.

Estimates of the potential risk resulting from the disposal of high-level waste are currently based on long-term safety analyses in which the processes that determine the potential radiological consequences of such a scenario are modelled in detail, often assuming the failure of technical and/or geological barriers. The impact of SNR on the manageability and outcome of such long-term safety analyses can therefore be used to analyse the possible influence of such concepts on the waste problem.

In contrast, other frequently cited criteria, such as the radiotoxicity of the waste, are not suitable criteria for assessing high-level waste on their own. The radiotoxicity index, for example, calculates the dose caused by the ingestion of radioactive substances. Accordingly, only those nuclides with either a high activity or a high dose factor are identified as relevant by this index, while all processes that determine release rates, retention, mobility, environmental behaviour and uptake into the food chain are not taken into account (Schmidt et al. 2013). Reference is made here to the more detailed discussion in Chapter 4.2 of (Oeko-Institut e.V. 2017).

In addition to the quantities of high-level waste in the form of spent fuel, other relevant waste streams may arise in the context of so-called “novel” reactor concepts, in particular waste from spent fuel reprocessing plants.

Furthermore, the question arises as to whether SNR would also generate fundamentally different types of high-level waste, such as graphite-containing fuels or alternative coolants such as molten salts or liquid metals. A different long-term safety assessment might have to be carried out for such waste, whereby interactions with other high-level waste to be placed in the same repository would also have to be discussed. It may even be necessary to find an additional repository site for other types of waste if, for example, the properties of the other types of waste are not compatible with the properties of the selected repository site.

In individual cases, SNR must also take into account the fact that toxic substances are used or may be produced. Their long-term disposal would also have to be taken into account.

Finally, when considering the quantities of radioactive waste produced, the nuclear power plants themselves must also be considered. For example, radioactive waste will also be produced during the later dismantling of a decommissioned nuclear power plant. For SNR, it must be examined to what extent dismantling concepts for such nuclear power plants are already being discussed and to what extent this could result in special waste streams that pose different requirements for later final disposal than the waste produced to date.

Against this background, the following indicators are used to evaluate the respective supply and waste disposal aspects as part of the discussion of the technology lines and reactor concepts:

- The respective fissile material demand and the need for fuel fabrication or reprocessing facilities.
- The quality of the waste streams produced
- Quantitative properties of the waste inventories (heat generation, activity, volume, mass)
- Effects with regard to the long term safety of a repository

2.6.4 Proliferation

The proliferation of nuclear weapons and the nuclear technologies or fissile materials necessary for their production is referred to as proliferation. Many nuclear technologies are used and researched in the civilian field of nuclear energy production, but can also be used for military purposes as part of nuclear weapons programmes. Actors can therefore exploit this civil-military ambivalence (dual-use). Not all technologies and fissile materials are equally suitable for military use (proliferation resistance). Access to fissile materials is particularly crucial for the production of nuclear weapons. There are two different options for this:

1. Access to existing stocks of fissile materials. These are, in particular, stocks of separated plutonium or highly enriched uranium. The attractiveness level according to (Bathke et al. 2008) can be regarded as a criterion for the proliferation resistance of fissile materials. This takes into account the isotopic composition and the chemical form of the fissile material and enables a quantitative categorisation of fissile material into three categories (unattractive, attractive and preferred). For example, plutonium can be separated or present in spent fuel, or uranium can be enriched to different levels.
2. Access to production technologies that can be misappropriated. One criterion for the proliferation resistance of production technologies is the probability of detecting and preventing the diversion of material or production options within declared production facilities in good time. This applies in particular to the sensitive technologies of uranium enrichment and reprocessing (separation of plutonium from spent fuel). There is also the possibility of setting up secret production facilities.

Historically, it has often been the case that states have utilised the dual-use characteristics of nuclear technologies. The development of a civilian nuclear energy programme establishes a nuclear infrastructure with corresponding facilities, know-how, materials and production processes. This latent potential (latent proliferation) is henceforth available for misuse or use in a parallel or later military nuclear weapons programme. There is also the possibility of theft of fissile material by non-state actors (Feiveson et al. 2014).

The then IAEA Director General Mohamed ElBaradei already stated this in 2004:

> *“Some estimates indicate that 40 countries or more now have the know-how to produce nuclear weapons, which means that if they have the required fissile material - high enriched uranium or plutonium - we are relying primarily on the continued good intentions of these countries, intentions which are in turn based on their sense of security or insecurity, and could therefore be subject to rapid change. Clearly, the margin of security this affords is thin, and worrisome.” (ElBaradei 2004)*

Proliferation is therefore dependent on a number of technical and non-technical factors that depend on the motivation of an actor, its goals and resources; on the technical capabilities and available nuclear means such as the choice of fuel supply and disposal technologies and fissile material. However, national and international monitoring measures also play an essential role. Any attempt to increase proliferation resistance must therefore include a number of intrinsic technical and extrinsic institutional measures.

The IAEA carries out safeguards to monitor the civilian use of nuclear energy; in the European Union, this task is performed by the European Atomic Energy Community (EURATOM) in agreement with the IAEA. All non-nuclear weapon states of the Treaty on the Non-Proliferation of Nuclear Weapons

(NPT) are obliged to accept these safeguards. They are concluded in binding international treaties between the states and the IAEA (IAEA 1972). The IAEA ensures that the use of fissile material is monitored as closely as possible, from fuel production to disposal. It has a range of monitoring technologies at its disposal for this purpose, e.g. radiological methods or visual monitoring through to the use of intelligence methods and remote sensing. The IAEA also analyses the necessity of monitoring which acquisition paths at which frequency to detect a diversion, see e.g. (IAEA 2012a). Safeguards are an essential element of non-proliferation. Safeguards-by-design refers to the consideration of control requirements when designing a nuclear facility.

Natural uranium must be enriched for use in light-water reactors. Today, gas ultracentrifuges are the standard industrial technology used for this purpose. There are proliferation risks from these enrichment plants, especially if states want to build up national enrichment capacities.

Only highly enriched uranium (HEU) with a proportion of > 90% uranium-235, which is present in metallic form, is directly suitable for weapons. Some research reactors use HEU in a different chemical form. LWR are designed for low enrichment < 20% uranium-235, typically in the range of 3-5%. Such low enriched uranium (LEU) would then need to be further enriched for use in a nuclear weapon. However, from an enrichment of approx. 3-5%, two thirds of the enrichment work required to achieve a high enrichment has already been expended, with 19.75% low enriched uranium (also known as HALEU) already over 95% of the separation work. This considerably shortens the time and the plant capacity required for enrichment to > 90% uranium-235.

Plutonium is produced in any reactor that uses uranium in the fuel. When a neutron is absorbed by uranium-238, plutonium-239 is produced. The resulting plutonium-239 can also absorb neutrons. The longer uranium is irradiated with neutrons, i.e. the longer a uranium fuel element is in a reactor, the successively higher plutonium isotopes are produced, i.e. plutonium-240, plutonium-241 etc. or, through further neutron reactions, plutonium-238. The even-numbered plutonium isotopes (plutonium-238, -240 and -242) have rather negative properties for nuclear weapons purposes. They produce neutrons, which can lead to a premature start of the chain reaction during the ignition of a nuclear weapon. They also decay quickly due to their half-life and produce heat and radioactive radiation. These are both properties that have a negative impact on handling, e.g. in production and storage. Depending on the proportion of higher plutonium isotopes, a distinction is often made between weapons plutonium (> 90% plutonium-239), which is ideal for military purposes, and reactor plutonium, which is less suitable for weapons purposes. However, both can be used in nuclear weapons (NPEC 2018; Mark 1993; IANUS 1989).

Fuel elements should be used in a reactor for as long as possible to generate electricity. Therefore, spent fuel elements from nuclear power plants typically contain reactor plutonium. However, if a fuel element is used for a shorter period in the reactor (e.g. during a shortened cycle or when commissioning a reactor), weapons plutonium can also be produced with an LWR fuel element (NPEC 2004).

However, the plutonium produced this way remains contained in the spent high-level radioactive fuel. This is a barrier to military use (radiation barrier). To separate the plutonium, mechanical comminution and dissolution of the fuel element would be necessary, followed by chemical extraction of the plutonium. In the “open fuel cycle”, the plutonium is not separated and is therefore protected from access by the radiation barrier. On the other hand, closed fuel cycles explicitly provide for the reprocessing of spent fuel and the further utilisation of plutonium. Therefore, all reactor concepts that provide for plutonium separation are less proliferation-resistant.

As early as 1977, the IAEA’s Standing Advisory Group on Safeguards Implementations (SAGSI) defined a “significant quantity” as the mass of fissile material that would be sufficient to build a simple nuclear weapon. This definition also takes into account material losses during production. Significant quantities are defined as 8 kg of plutonium, 25 kg of highly enriched uranium (HEU, approx. 90% uranium-235) or 8 kg of uranium-233. However, more advanced weapon designs require less fissile material. A quantity of just 4-5 kg of plutonium or uranium-233 or around 12 kg of HEU is sufficient to build a nuclear warhead (NRDC 1995).

Against this background, the following indicators are used to assess proliferation resistance when discussing technology lines and reactor concepts:

- The respective fissile material or uranium enrichment requirements
- The planned or necessary reprocessing of spent fuel
- The quantities of plutonium or other fissile materials produced and their quality (isotope vector)

2.6.5 Costs

Approximately 65% of the levelised cost of electricity of nuclear power plants is determined by investment costs, approximately 23% by operation, maintenance, dismantling, etc., and approximately 12% by fuel costs (Neles and Pistner 2012). In recent cost studies, capital costs account for more than 80% of the levelised cost of electricity of a nuclear power plant, while operating costs account for 15% and fuel costs for 5% (MIT 2018).

This means that the initial investment costs are of central importance for the levelised costs of electricity. On the one hand, the respective financing conditions are included in the resulting costs, i.e. whether an investor has to take out loans on the financial market at (high) costs or whether he can obtain capital at (lower) costs through state guarantees. Against this background, cost data from countries with a heavily state-regulated energy market are not comparable with those from liberalised electricity markets (University of Chicago 2004).

Investment costs are essentially dependent on the type and capacity of a plant. As investment costs do not increase linearly with a plant’s capacity, the output sizes of new nuclear power plants have risen continuously since the beginning of nuclear energy (“economy of scale”).

Construction times also significantly influence the capital costs associated with the initial investment. The longer it takes to construct a reactor, the later the initial investment will yield profits. In the decade from 2012 to 2021, 62 reactors were commissioned in ten countries. The average construction time of these reactors was 9.2 years, with the shortest construction time being 4.1 years and the longest 42.8 years (Mycle Schneider Consulting 2022). Therefore, most of the completed reactors were significantly longer than their originally planned construction period.

In addition to the question of investment costs, the question of the achievable service life and the availability of the plant in normal operation will also be of central importance for SNR in terms of the attainable levelised cost of electricity.

By 2022, 204 nuclear power plants worldwide had been permanently decommissioned. The average age of these nuclear power plants at the time of decommissioning was 27.7 years. However, the average age of the plants at decommissioning has increased over time. The average age of the

plants decommissioned in the period from 2017 to 2021 was 42.2 years (Mycle Schneider Consulting 2022).

Over the last twenty years, the average global availability of current nuclear reactors has ranged between 72.2% and 82.6% (IAEA 2022g). It should also be taken into account that nuclear power plants have predominantly been used as base load power plants, which has enabled them to achieve high availability rates – well above 90% in Germany, for example. With the breakthrough of cost-effective renewable energies in conjunction with various flexibility options (such as load flexibility, storage, grid infrastructure, etc.), nuclear power plants are losing a considerable number of full-load hours and thus also their economic efficiency.

Against this background, the following indicators are used to assess costs when discussing technology lines and reactor concepts:

- Investment costs
- Operating costs
- Construction times
- Investment risks
- Planned service life/ utilisation of the system

3 Country studies

The following describes the development activities related to so-called “novel” reactor concepts (SNR) using selected country studies. This is followed by the political and economic background to innovation, which describes the innovation process in the nuclear power industry. The various dimensions of national innovation processes are highlighted, forming the basis for the following country portraits.

3.1 Classification in terms of innovation policy

The nuclear power technology and industry have specific characteristics that affect the innovation process. Nuclear power is a complex technology with particular safety risks, and its development requires specific knowledge from different disciplines. Development also requires complex facilities for testing, experimentation, manufacturing and power generation with specific quality requirements in terms of their nuclear safety, non-proliferation of nuclear weapons-grade material and radiation protection (NEA 2007b). The innovation process in the field of nuclear power can therefore generally be characterised as complex, long-term and resource-intensive. The following briefly describes innovation-economic theories, which can help analyse the development of SNR technology based on specific country studies.

3.1.1 Invention – Innovation – Diffusion

All innovation processes are based on the interplay of research-driven “technology push” and demand-driven “market pull” (Figure 2-1). After the invention of a technology, i.e. the invention, further development (innovation) into a product is driven by political (development and funding programmes) and economic factors. For the technology to be used more widely, it must be drawn into the market by an existing demand for it. Otherwise, its diffusion, i.e. the broad utilisation of the technology in the market, will fail to materialise. In this case, the innovation remains in the so-called “valley of death” (Grubb et al. 2014; Stern 2007). This is the case with most inventions.

What is special about the innovation process in nuclear energy are the technical complexity and the necessary safety requirements, which materialize, for example, in complex experimental and research facilities and elaborate authorisation procedures. A deliberate, i.e. intended, technology push in nuclear power is therefore cost-intensive and time-consuming. In addition, demand is characterised exclusively by state-incentivised purchases; this applies to military equipment, for which there is generally no “market”, as well as to commercial power plant technology, which is also beyond market competition. In contrast to marketable innovations such as consumer electronics, there is no market demand for nuclear power due to its lack of economic efficiency, which severely impairs the dynamics of innovation.

It is therefore highly likely that the invention, i.e. the invention of new products or processes, e.g. SNR, could develop into demonstration reactors (innovations), yet their industry-wide implementation, i.e. diffusion, will fail to materialise (Figure 2-1) and these reactor technologies will remain in the “valley of death”. A typical example is the sodium-cooled reactor with a fast neutron spectrum in the last 70 years (DIW 2022). Although it was planned to roll out commercial nuclear power with fast reactors and thus transition to a plutonium economy, this development got stuck in the 1950s and has remained marginalised to this day.

The existing technology of light-water reactors is also based on direct state demand or set framework conditions that are non-competitive. All nuclear power plants in operation today were financed and built on regulated supply markets, securing both the purchase of electricity and an adequate return on investment (DIW 2018). Under these conditions, cost overruns and project delays were covered by higher electricity prices, which customers ultimately paid for (IAEA 2008). In addition, much of the financing for these plants was provided by governments or with government support or guarantees (IAEA 2008). In this way, light water reactor technology has established itself as a technology for generating electricity.

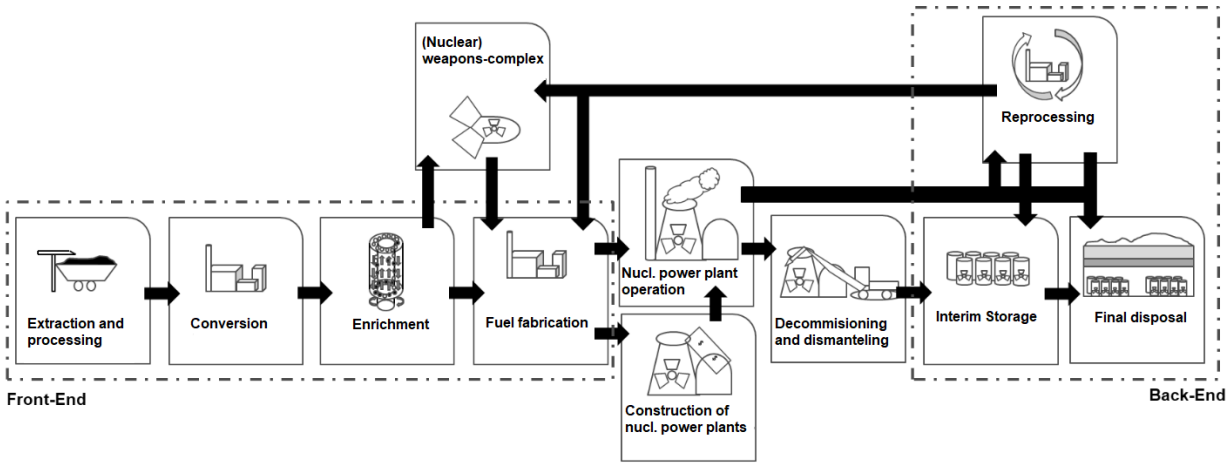
The development of nuclear reactors, especially SNR, is characterised by long periods of time (Oeko-Institut e.V.; WIP; PhB 2021; Pioro 2016; IPFM 2010). Such a lengthy innovation process in a country is characterised by changing framework conditions for the development process. This includes, for example, changes to laws, stakeholders, financing commitments and geopolitical developments.

3.1.2 Presentation and classification scheme for national innovation systems

3.1.2.1 Nuclear power as a system asset and organisational models

In addition to long development periods and multidimensional national innovation processes, nuclear power is also characterised by its system-asset nature. As shown in Figure 3-1, these include value chain elements such as the supply of raw materials, enrichment, fuel element production, power plant construction, reprocessing, military uses and the back end with dismantling, interim storage and final disposal (DIW 2020b). Each national system has special features, such as the availability of specific raw materials (e.g. uranium or thorium), the infrastructure for fuel production, or reprocessing facilities.

Figure 3-1: Nuclear energy as a system asset



Source: Own illustration based on (DIW 2020b)

There are potential differences in the technical system of so-called novel reactors compared to LWR, such as the use of a different fuel or other production infrastructures, for the potential future use of these reactors (Oeko-Institut e.V. 2017). Two aspects are essential for the organisation of innovation processes and the future potential use of SNR technology, which can be analysed in the context of

system goods and organisational models (WIP 2012): The provision (incl. financing) and the production structure.

The provision refers to the decision regarding the so-called “novel” reactor concepts to be developed. This includes both “hard” technology and “soft” production factors such as trained personnel. Given the difficulty of obtaining market-driven funding for SNR, specific subsidy schemes must be created to underpin the state’s deployment decision with state-structured funding streams.

The production structure of the nuclear power plant as a system asset refers to the interaction of in-house and external production (“make-or-buy” decision) in complex value chains involving public and private research and development, e.g. in large research institutes and companies.

3.1.2.2 Development of the national nuclear energy system and motivations

Development of national systems

Due to the long development periods of reactor technologies, the development of national nuclear energy systems must be viewed dynamically over time. It is subject to changes in the socio-technical system, which can hinder but also promote innovation as a “window of opportunity”. In addition, the availability of the research infrastructure is also affected by the long periods of time. This applies, for example, to experimental facilities that were previously built and then decommissioned or had to be replaced at great expense. For example, the Experimental Breeder Reactor II (EBR II) was decommissioned, and today the development of the Versatile Test Reactor is being considered (U.S.). Similarly, the BOR-60 is being replaced by the MBIR (Russia). To describe the national nuclear power innovation system, the following information, among others, should be presented: The age of existing power plants; existing know-how and research infrastructure (national laboratories, universities, companies); export/import of previous specific conventional nuclear power technology and know-how; the historical background of prior research programmes for SNR.

To illustrate the dynamics, it makes sense to present the development of national innovation paths in three phases:

- Firstly, there is a build-up phase (t_{-2}), in which the country establishes its national innovation system with the necessary research infrastructure. The start of technological development can be characterised by in-house developments or the provision of technology by other countries (e.g. through the purchase of research reactors or know-how).
- In the adaptation phase (t_{-1}), the developments achieved in the build-up phase are adjusted, and the strategy is changed.
- The third phase then describes the current status (t_{-0}), taking into account more recent developments (since around the year 2000).

Although the focus here is on the development of SNR, it cannot be considered independently of the development of light water-cooled reactors and other technologies. These are national innovation systems in which the research infrastructure, actors and institutions are partly technology-specific but also have a cross-technology effect.

Discussion of possible motives

SYNERGY BETWEEN COMMERCIAL AND MILITARY USES

As in the case of light water-cooled reactors, SNR must be considered with regard to military interdependencies and geopolitical considerations. The combined nature of military and commercial use is a key motivation for nuclear power activities (University of Sussex 2018; Lovins 2022; DOS 1946; Lévêque 2014). In the U.S., in particular, military utilisation, which manifested itself, for example, in the development of the atomic bomb in the Manhattan Project or nuclear propulsion systems for submarines (S1W, S2W), preceded commercial utilisation (Lévêque 2014; DIW 2017; Reed 2021). China also first developed a military programme, which manifested itself in the nuclear bomb test using a uranium bomb in 1964, before it commissioned the first light water-cooled reactor for commercial power generation in 1994 (Lewis and Xue 1988; CEIP 2018; Zhou 2011).

Due to the technological and historical roots of nuclear power with the military, this link must always be considered as a motive. For SNR, this applies in particular to the development path of reactors with a fast neutron spectrum that can breed plutonium, not only as fuel for generating electricity in nuclear power plants but also potentially for military purposes (development of nuclear weapons) (Oeko-Institut e.V. 2017).

All three nuclear superpowers, i.e. the U.S., Russia and China, are engaged in targeted nuclear diplomacy. This can be understood as strategic technology and foreign trade policy in the areas of weapons and commercial nuclear technology, which attempts to use its existing position of power to involve other countries in its own technology development, pursue geopolitical alliance policy through exports and deter potential opponents (DIW 2017).

Geopolitical significance also plays a role in nuclear power innovation processes. “Nuclear diplomacy”, i.e. the use of specific nuclear expertise in geopolitical strategies, has been an important factor from the outset, both on the part of the U.S. with its partner countries and in the Soviet Union and present-day Russia. Since then, it has become an integral part of foreign and security policy internationally, especially in China (Bracken 2012; Hirschhausen et al. 2018; Gattie 2018; Gattie and Massey 2020). The import of nuclear technology creates long-lasting dependencies. This applies to LWR and is also used for so-called “novel” reactor concepts.[4]

ENERGY SYSTEM DECARBONISATION PRESSURE

In 2015, under the Paris Agreement, also known as the United Nations Framework Convention on Climate Change (UNFCCC), 197 nations set themselves the goal of limiting the increase in the global average temperature to well below 2 °C and pursuing efforts to limit the temperature increase to 1.5 °C above pre-industrial levels (UN 2015). This requires a drastic reduction in emissions (IPCC 2022). As a result, low-emission technologies are being promoted, and nuclear power, which produces less CO_2 during operation than fossil fuels, is gaining new political and social support. Poland, for example, is considering introducing nuclear power, arguing that it will reduce emissions-intensive coal. The U.S. and China also want to use nuclear power as part of their future energy systems, as the following country chapters show. This includes LWR, as well as SNR.

The extent to which nuclear power can significantly contribute to combating the climate crisis is highly controversial (IEA 2019; Zenodo 2021). In particular, the role of nuclear power for potential use in

[4] For a literature summary, see Council on Foreign Relations: https://www.cfr.org/blog/america-risks-missing-out-global-nuclear-power-revival; also: How Russia, China Use Nuclear Reactors To Win Global Influence: https://www.defenseone.com/ideas/2018/07/hina-and-russia-look-dominate-global-nuclear-power/149642/, last checked 29.01.2021.

the future energy system must be compared with other power generation technologies, such as renewable energies (see Chapter 2.5).

WASTE MANAGEMENT AND THE CLOSED FUEL CYCLE

Radioactive waste and decay products, including plutonium, are produced during the operation of nuclear power plants (Hippel et al. 2019). Some of this plutonium is currently separated from spent fuel through reprocessing. A small part of it is reused to produce uranium-plutonium mixed oxide fuel for LWR or SNR, while large parts have so far been stored separately. The worldwide use of nuclear power has, therefore, resulted in a total of 405 tonnes of separated plutonium in the civilian sector, with a further approx. 140 tonnes of plutonium in the nuclear weapons programmes of the nuclear weapons states.[5]

Some countries, particularly Russia and China, strive for a closed fuel cycle in which spent fuel is reprocessed, and the plutonium is reused as MOX fuel. Part of this fuel cycle is reprocessing, in which plutonium could also be separated and used for military purposes (Chen et al. 2018; BCSIA 2003; Pioro 2016). There are different processes (PUREX, pyroprocessing), but none can completely rule out proliferation risks (Oeko-Institut e.V. 2017).

Furthermore, some countries such as the Republic of Korea, Russia and China want to use reactors with a fast neutron spectrum as transmutation reactors to reduce radioactive waste (CEIP 2018; Pioro 2016; Yoo et al. 2017). However, this will not eliminate the need for a geological repository (Oeko-Institut e.V. 2017; ISR 2021). Even if using reactors with a fast neutron spectrum contributes to modifying the radioactive inventory, it also entails a certain proliferation risk (ISR 2021).

INNOVATION COMPETITION, INDUSTRIAL AND SCIENCE POLICY

Another motivation for SNR activities can also be to secure existing production capacities, infrastructure and human capital (Wright et al. 2015). Research and development in nuclear power are also part of industrial and science policy. The economic failure of light-water reactors threatens the development of the commercial nuclear industry. Against this background, SMR and SNR narratives are being developed to create a long-term perspective beyond the lack of techno-economic prospects; on the other hand, attempts are being made to preserve and possibly reorient existing human capital and infrastructure (Thomas and Ramana 2022). This can be seen, for example, in the U.S., which has hardly seen any capacity additions since the 1980s and, at the same time, is striving to keep existing nuclear power plants on the grid for as long as possible via the Civil Credit programme (Thomas and Ramana 2022; Mycle Schneider Consulting 2022; IAEA 2022i).

At the same time, the U.S. is trying to develop SMR and SNR as quickly as possible through billion-dollar development programmes, such as the Advanced Reactor Demonstration Program (ARDP) (DOE 2020a; Infrastructure Investment and Jobs Act (IIJA) 2021). As part of a new bill, the “International Nuclear Energy Act of 2022”, U.S. Senator Manchin argues that the U.S. is motivated by the desire to regain global leadership in the nuclear power sector.[6] Other countries’ adoption of the technology will lead to future contracts and collaborations. This approach also creates technological dependencies, as can be seen, for example, in the role of the U.S. in technology

[5] https://fissilematerials.org/ (last checked on 13.11.22)

[6] The war in Ukraine, in particular, is intended to reduce South Korea’s own dependencies (e.g. fossil fuels) on Russia and China as a means of counteracting geopolitical dangers. https://www.energy.senate.gov/2022/4/manchin-risch-introduce-the-international-nuclear-energy-act-of-2022 (last checked on 08.07.22)

development in the Republic of Korea (South Korea.) South Korea can only use reprocessing technologies with the approval of the U.S. and is granted access to US reactor technology on this condition (CRS 2013; 2015).

The motives are revisited in the following chapters in the context of the specific country portraits, and some examples are discussed based on publicly available information.

3.2 Overview of research activities

The following provides an overview of global research activities relating to the development of SNR. The countries are then divided into three categories, and their core activities in SNR research are briefly described. Based on this, exemplary countries are selected for a more in-depth analysis.

3.2.1 International overview

There are numerous research activities and pilot projects for so-called “novel” reactor concepts (SNR) around the world. The Generation IV International Forum (GIF) was founded in 2001 to promote cooperation between countries in the development of such reactor concepts. The current annual reports of the GIF (available on the website) contain detailed information on current member countries and their (research) activities. The following countries are currently active in the GIF: U.S., China, Russia, Japan, Republic of Korea, France, UK, Canada, Switzerland, South Africa, Australia. EURATOM is also listed as a member of the GIF. Argentina and Brazil have no concrete aspirations and are therefore listed as “non-active members” in the GIF (GIF 2020a). This chapter presents the activities in the individual countries. Countries to be examined in more detail are then selected. The following sources were used for this purpose: (GRS 2015; 2018; 2020; Thomas and Ramana 2022; IAEA 2022h).

For 2022, there were over 60 SNR, which are distributed among the most significant countries as follows, among others: The U.S. with 22 projects, China with 12 projects, Japan and Russia with 7 projects each and the Republic of Korea with 4 projects (Figure 3-2).

Figure 3-2: Global activities in the area of SNR

Source: Own illustration based on Table 3-2

This study identified systems in the most significant countries working on SNR, including: the U.S., China, Japan, Russia, the Republic of Korea, India, Sweden, UK, Canada, France, Belgium, Denmark, Luxembourg, Poland, Romania, Switzerland and South Africa. This list cannot claim to be exhaustive, as new projects are constantly being started and existing projects are being discontinued.

Table 3-1 breaks down the countries and the number of reactor concepts (within the technology lines) to be assigned to these countries and is the result of literature research. The last three columns summarize criteria that can be used to classify these countries into comparable groups: The current use of nuclear power plants for electricity production, the existence of military nuclear technology programmes, and membership in the GIF. Based on this, the countries and their research activities can be divided into three categories.

Table 3-1: Categories for countries with SNR

Country	Number of reactor concepts pursued	Technology lines	Commercial nuclear power plants	Military nuclear technology programme	GIF member/ possibly through EURATOM
Category I: Countries with nuclear activities and military programmes					
USA	22	ADS (1), SFR (3), LFR (2), GFR (1), VHTR (4), MSR (11)	yes	yes	yes
China	12	ADS (3), LFR(2), MSR(2), SCWR(1), SFR(2), VHTR(2)	yes	yes	yes
Russia	7	LFR(2), SFR(5)	yes	yes	yes
UK	2	ADS(1), MSR (1)	yes	yes	yes
France	2	MSR(1), SFR(1)	yes	yes	yes
India	3	ADS(1), SFR(2)	yes	yes	no
Category II: Countries with nuclear activities but without (known) military programmes					
Belgium	1	ADS	yes	no	yes
Republic of Korea	4	ADS(1), LFR(2), SFR(1)	yes	no	yes
Japan	7	GFR(1), MSR(1), SCWR(1), SFR(2), VHTR(2)	yes	no	yes
Sweden	3	ADS(1), LFR(2)	yes	no	yes
Canada	1	MSR	yes	no	yes
Romania	1	LFR	yes	no	yes
South Africa	1	VHTR	yes	no	yes
Category III: Potential entry countries					
Poland	1	VHTR	no	no	yes
Denmark	1	MSR	no	no	yes
Luxembourg	1	LFR	no	no	yes

Source: Own evaluation

3.2.2 Categories of countries with SNR development activities (country overview)

The categories of countries with SNR activities and the criteria for country selection are described in the following chapter. One or more countries are selected from each category for further analysis.

3.2.2.1 Category I: Countries with nuclear activities and military nuclear technology programmes

The first category includes countries that already have a commercial power plant fleet and are also pursuing military programmes.

USA

In the U.S., efforts to develop so-called “novel” reactor concepts have been underway since the 1950s (Pioro 2016). A broad portfolio of SNR technology lines is currently being pursued, in particular molten salt reactors (MSR), very-high-temperature reactors (VHTR) and sodium-cooled fast reactors (SFR). The Advanced Reactor Demonstration Program (ARDP) promotes these three technology lines, which are at different stages of development. For this purpose, two demonstration projects were supported in fiscal year 2020 with USD 80 million each (DOE 2020c): one for Terra Power LLC and the demonstration of a sodium-cooled fast reactor (sodium) with an electrical output of 345 MW and for X-energy with an SMR, the gas-cooled high-temperature reactor (Xe-100) with an electrical output of 80 MW, which can be scaled up to 320 MW.[7] Furthermore, attempts are being made to reduce the development risk within the framework of the ARDP through further financial support, which should enable development up to demonstration maturity (e.g. the Hermes Reduced Scale Reactor concept from Kairos Power was funded here).[8]

CHINA

SNR have been under development in China for many years: Basic research has been conducted in the field of SFR since the late 1960s. In 2011, the sodium-cooled research reactor, the China Experimental Fast Reactor (CEFR), was officially put into operation. The VHTR technology line has been under development since the mid-1970s, starting with the HTR-10 at Tsinghua University (Pioro 2016). There is currently a demonstration project, the Very-High-Temperature Reactor HTR-PM, which has been in operation since December 2021.[9]

RUSSIA

In addition to developments in the field of LWR, Russia has been making efforts for many decades to develop a closed fuel cycle using sodium- and lead-cooled reactors with a fast neutron spectrum. Starting with several test reactors BR-1, -2, -5 and -10 in the late 1950s and the research reactor BOR-60, the construction of the BN-350 followed, which was put into operation in 1973 (Pioro 2016). The successor model, the BN-600, has been in operation since 1980 and was followed by the BN-800 in 2016 to enable further material tests and experiments to be carried out for the closed fuel cycle. The BN-1200 was to be built based on the experience gained.[10] However, construction has been postponed due to the reduction in state funding for Rosatom.[11] The BREST-OD-300 is currently being built as a lead-cooled project.[12] In addition, the multi-purpose research reactor (MBIR) is currently being built to replace the BOR-60 research reactor and is expected to go into operation in 2027.[13]

[7] https://www.energy.gov/ne/articles/us-department-energy-announces-160-million-first-awards-under-advanced-reactor (last checked on 02.08.22)
[8] https://www.energy.gov/ne/articles/energy-departments-advanced-reactor-demonstration-program-awards-30-million-initial (last checked on 02.08.22)
[9] https://www.world-nuclear-news.org/Articles/Demonstration-HTR-PM-connected-to-grid (last checked on 18.03.22)
[10] https://rosatom.ru/en/rosatom-group/engineering-and-construction/modern-reactors-of-russian-design/ (last checked on 21.03.22)
[11] https://www.world-nuclear-news.org/NN-Russia-postpones-BN-1200-in-order-to-improve-fuel-design-16041502.html (last checked on 21.03.22)
[12] https://rosatom-europe.com/press-centre/news/rosatom-will-manufacture-unique-equipment-for-the-brest-od-300-power-unit/?sphrase_id=8141 (last checked on 21.03.22)
[13] https://www.world-nuclear-news.org/Articles/Completion-of-MBIR-reactor-brought-forward (last checked on 01.08.22)

UNITED KINGDOM

The UK has been investing in the development of SFR since the 1950s on the assumption that fast neutron spectrum reactors and reprocessing would quickly become commercially competitive. After the construction of an experimental reactor, the DFR, and a prototype reactor, the PFR, which only became critical for a short time due to recurring technical problems, funding for SFR technology was discontinued at the end of the 1990s after more than four decades of research (IPFM 2010). The United Kingdom also has many years of experience with gas-cooled and graphite-moderated reactors. At the end of 2022, nine reactors will be in operation, all of which, apart from one pressurized water reactor, are gas-cooled and graphite-moderated reactors (AGR) (IAEA 2023g). AGR can be traced back to the earlier Magnox reactors, graphite-moderated and gas-cooled reactors that were built between the 1950s and early 1970s and designed as dual-use reactors (DIW 2018). There were a total of 26 Magnox reactors, all of which have now been shut down (IAEA 2023g). The AGR emerged from the Magnox design and enabled a higher outlet temperature. The higher thermal efficiency and greater power density were intended to reduce the capital costs of the power plant (DIW 2018). A total of 16 AGR were added in the 1970s and 1980s, eight of which have now been closed again and the rest will be shut down in the coming years.

The UK is currently incorporating the development of SNR into its “Net-Zero Strategy”. The plan is to implement a demonstration project focusing on VHTR concepts by the early 2030s. To this end, a budget of £170 million (approx. EUR 200 million) is to be spent on the “AMR Research, Development and Demonstration Program” as part of a £385 million “Advanced Nuclear Fund”.[14]

FRANCE

France has been researching SFR concepts and reprocessing since the 1950s. These efforts were expressed in particular through the Phénix and Superphénix projects, which were discontinued due to technical and economic problems (IPFM 2010). The latest attempt to develop an SFR, the Astrid project, was also discontinued in 2019 for cost reasons and due to limited progress.[15] In addition, further research activities on materials and security research are being funded in isolated cases (GIF 2020a).

INDIA

India is the only country considered here that is not a member of the GIF (GIF 2022a, p. 4). Research on SFR concepts has been pursued as part of a 3-phase development programme since the 1950s in order to circumvent the scarcity of uranium resources in India and use the country’s own thorium reserves (IPFM 2010). This development includes the use of heavy water reactors and natural uranium followed by reprocessing and the use of fast neutron spectrum reactors to utilize thorium in a fuel mixture. In 2004, construction began on a prototype, the PFBR, with an electrical output of 500 MW, cooled by sodium and with a fast neutron spectrum, although its completion is still delayed (Pioro 2016).[16]

[14] https://www.gov.uk/government/publications/advanced-nuclear-technologies/advanced-nuclear-technologies (last checked on 11.03.22)

[15] https://www.neimagazine.com/news/newsfrance-cancels-astrid-fast-reactor-project-7394432 (last checked on 21.04.22)

[16] https://www.world-nuclear-news.org/Articles/Minister-foresees-2022-completion-date-for-Indian last checked on 07.06.22)

3.2.2.2 Category II: Countries with nuclear activities but without military nuclear technology programmes

This category describes countries with a commercial nuclear energy programme without a military nuclear technology programme.

REPUBLIC OF KOREA

The Republic of Korea (South Korea) has been active in SNR research and development for many years and is very actively involved in collaborative R&D activities of the GIF. Research facilities have been built (e.g. Stella-1, Stella-2), various experiments for SFR and VHTR technologies have been carried out and reactor concepts have been developed in several technology lines (e.g. PGSFR, KALIMER-600, NHDD) (Pioro 2016). In addition, various collaborations are being entered into with individual countries worldwide; for example, there is a current agreement between X-Energy from the U.S. and Doosan Industries[17] or agreements with TerraPower in the U.S. or Seaborg in Denmark.[18]

JAPAN

The development of SFR concepts dates back to the 1970s with the construction of the JOYO research reactor, followed by the construction of the Monju prototype in the 1980s (Pioro 2016; Oeko-Institut e.V. 2017). The aim of these activities was to develop a fuel cycle with reprocessing and production of mixed oxide fuels. However, the reactors were shut down after an accident in 1994 and finally decommissioned in 2016 (Oeko-Institut e.V. 2017). Since then, Japan has been trying to advance its research into SFR through collaborations such as the French Astrid project (which has since been discontinued) or the development of an SFR design by the American company Terra Power.[19]

BELGIUM

Belgium has many years of experience in MOX fuel production and produced, among other things, the fuel that was to be used in the fast breeder reactor in Kalkar (which did not go into operation) (SCK CEN 2002). The Belgian nuclear research institute SCK CEN is developing a multi-purpose research reactor (MYRRHA), which is to be operated in both the critical and subcritical range. The aim is to construct a demonstration plant to demonstrate both ADS and LFR technology. The project is therefore also considered a strategic project for the research infrastructure in Europe.

CANADA

In Canada, there are development efforts in the field of SMR concepts, whereby reactor concepts that can be assigned to SNR are also being investigated (GIF 2020a). In 2020, the Canadian government awarded USD 15 million through an innovation fund to Terrestrial Energy’s Integral Molten Salt Reactor (IMSR) to complete the Canadian Nuclear Safety Commission (CNSC) pre-licensing. SMR concepts are expected to play a certain role in Canada’s efforts to achieve climate

[17] https://www.ajudaily.com/view/20210901125057993 (last checked on 02.06.22)
[18] https://www.world-nuclear-news.org/Articles/Seaborg-and-BEES-sign-MOU-relating-to-floating-Com and https://www.world-nuclear-news.org/Articles/Korean-conglomerate-to-cooperate-with-TerraPower (last checked on 02.06.22)
[19] https://www.world-nuclear-news.org/Articles/US,-Japanese-firms-agree-to-cooperate-on-fast-reac (last checked on 12.04.22) and https://www.jaea.go.jp/english/news/press/2020/090102/ (last checked on 07.06.22)

neutrality by 2050.[20] In 2022, two Canadian utilities (Bruce Power and Ontario Power) announced plans to collaborate on the development of SMR technology, drawing on their operating experience from CANDU reactors.[21]

SOUTH AFRICA

South Africa imported the HTR technology developed in Germany and collaborated on the development of the PBMR (ISS 2010). However, this was postponed indefinitely in 2009 due to a lack of public and private funding.[22] The PBMR project’s large helium test facility is still in operation and the fuel development laboratory is still fully equipped. A revival of the project can therefore not be completely ruled out.[23]

SWEDEN

The SEALER reactor concept is currently being developed in Sweden. The SEALER reactor concept is a lead-cooled microreactor with an electrical output of between 3 and 10 MW, which is intended to last 10-30 years without a change of fuel.[24] This design is being developed by the Swedish LeadCold, a spin-off of the Royal Institute of Technology (KTH) in Stockholm, where research into lead-cooled reactor systems has been ongoing since 1996.[25] In 2021, Uniper Sweden, LeadCold and the Royal Institute of Technology (KTH) joined forces to pursue the construction of a demonstration reactor by 2030. In the first stage, they received SEK 125 million (approx. EUR 12 million) in support from the Swedish Energy Agency.[26]

ROMANIA

With regard to the development of so-called “novel” reactor concepts, Romania is participating in the development of the ALFRED project (Advanced Lead-cooled Fast Reactor European Demonstrator), a lead-cooled reactor with a fast neutron spectrum. A consortium (FALCON) was formed in 2013 to develop the project, consisting of the Italian National Agency for New Technologies, Energy and Sustainable Economic Development (ENEA), the Italian company Ansaldo Nuclear and the Romanian nuclear research institute Institutul de Cercetari Nucleare (ICN).[27] For the development of necessary research infrastructure, another consortium was formed in 2021 consisting of the Italian Ansaldo Nucleare and the Romanian Reinvent Energy, which was awarded a contract worth EUR 20 million in 2021 for the design, provision, installation and construction of a research facility ATHENA for the full-scale testing of components for lead-cooled reactors.[28]

[20] https://world-nuclear-news.org/Articles/Canadian-government-invests-in-SMR-technology (last checked on 12.04.22)
[21] https://www.world-nuclear-news.org/Articles/Canadian-companies-collaborate-to-advance-nuclear (last checked on 12.04.22)
[22] https://www.world-nuclear-news.org/NN-PBMR_postponed-1109092.html (last checked on 07.06.22)
[23] https://www.neimagazine.com/news/newssouth-africa-seeks-to-revive-pbmr-project-7765057 (last checked on 07.06.22)
[24] https://www.leadcold.com/sealer.html (last checked on 11.08.22)
[25] https://www.leadcold.com/about-us.html (last checked on 11.08.22)
[26] https://www.world-nuclear-news.org/Articles/Funding-for-demonstration-Swedish-SMR (last checked on 21.04.22)
[27] https://www.world-nuclear-news.org/Articles/Consortium-established-to-build-Alfred (last checked on 22.08.22)
[28] https://www.world-nuclear-news.org/Articles/Contract-for-Romanian-lead-cooled-reactor-research (last checked on 22.08.22)

3.2.2.3 Category III: Potential entry countries

This third category lists countries that do not yet have nuclear power plants but nevertheless have isolated SNR development projects.

POLAND

The entry into nuclear power with the construction of an LWR has been attempted since 1956. However, these attempts have been unsuccessful to date and projects have been discontinued.[29] Discussions are currently underway regarding the purchase of conventional nuclear reactor technology from France, the U.S. and the Republic of Korea.[30] With regard to the development of SNR, there are plans to develop a very-high-temperature reactor to provide industrial process heat. There is an ongoing exchange of knowledge with the Japan Atomic Energy Agency.[31]

DENMARK

As one of the pioneers of nuclear power research, Niels Bohr, who conducted research into nuclear fission in the 1920s and 30s, founded a research centre in Risø. However, not a single nuclear power plant has been built in Denmark to date. The reasons for this are seen in the debate on energy policy triggered by the oil crisis and a lack of political will (University of Copenhagen 2019). Even before the accident in Chernobyl, a law was passed in 1985 prohibiting the production of energy from nuclear power.[32] Seaborg Technologies is currently only working on an MSR concept. Seaborg Technologies was founded in 2014 by the cooperation of three physicists.[33]

LUXEMBOURG

Luxembourg does not currently operate any nuclear power plants or other nuclear facilities.[34] Hydromine Nuclear Energy developed the lead-cooled reactor LFR-AS-200.[35] The company was acquired by Newcleo, a UK company, in 2021.[36]

3.2.3 Selection of country studies

Based on the preliminary considerations regarding the countries and the research activities in the countries, a selection was made in Categories I to III. For each category, one to three countries were selected to be examined in more detail in the country studies.

[29] https://www-pub.iaea.org/MTCD/Publications/PDF/cnpp2018/countryprofiles/Poland/Poland.htm (last checked on 08.06.22)
[30] https://www.world-nuclear-news.org/Articles/Korea-offers-six-reactors-to-Poland (last checked on 08.06.22)
[31] https://www.world-nuclear-news.org/Articles/Poland-plans-next-stage-of-HTGR-development (last checked on 08.06.22)
[32] https://world-nuclear.org/information-library/country-profiles/countries-a-f/denmark.aspx (last checked on 03.06.22)
[33] https://www.seaborg.com/copy-of-about-us (last checked on 03.06.22)
[34] https://www.ensreg.eu/country-profile/Luxembourg (last checked on 03.06.22)
[35] https://www.hydromineinc.com/ (last checked on 03.06.22)
[36] https://www.neimagazine.com/news/newsnuclear-technology-company-newcleo-acquires-hydromine-9051655 (last checked on 03.06.22)

3.2.3.1 Category I: U.S., Russia, China

The countries in Category I are central, as they already have decades of activity in the field of SNR development. The U.S., Russia and China are selected from Category I due to their active research and development efforts. All three countries are striving to further develop one or more technology lines in the field of SNR. In addition, Russia has many years of experience with sodium-cooled fast reactors via the BN-600, BN-800 series and the planned BN-1200. China has state-owned nuclear energy technology development efforts and a broad portfolio of different technology lines. The U.S. has a multi-billion dollar funding programme with the ARDP, which also promotes various technology lines.

In contrast, the UK, France and India from Category I are not considered in more detail due to the lack of diversity in technology lines and lower development activity compared to the U.S., Russia and China. The United Kingdom is mainly active with efforts in the direction of low power reactors (so-called SMR concepts). Following the completion of the Astrid project, France has only minor activities to report. Although India has been active in the development of reactor concepts for many decades, it is less advanced than the U.S., Russia and China and has not yet caught up with international research activities.

3.2.3.2 Category II: Republic of Korea, Belgium

The Republic of Korea (South Korea) and Belgium were selected from Category II. Belgium is developing a pilot project that is considered important in the European context and can be assigned to the area of both accelerator-driven and lead-cooled fast reactors (MYRRHA) (ESFRI 2016). It is of particular interest to take a look at the long-term development history of the project and its financial framework conditions.

The Republic ofKorea is a country that has been actively involved in the research and development of SNR in recent years and has established international collaborations for this purpose. For example, there are current agreements with X-Energy from the U.S. and Doosan Heavy Industries[37] and several memoranda of understanding (MoU), for example with TerraPower in the U.S. or with Seaborg Technologies in Denmark.[38] South Korea is pursuing technology lines without reprocessing, which has been prohibited since 1974 (Korea-U.S. Atomic Energy Agreement).[39] Like the U.S., South Korea is thus oriented towards development without reprocessing and is a more interesting counterpart to Russia and China.

In Category II, Japan also has a broad range of technologies. Similar to France, Russia and China, this is characterized by many years of SFR development and reprocessing. This focus is already adequately covered by the selection of Russia and China. In addition, Japan is primarily concerned with revitalizing the power plant fleet that was shut down after the accident at the Fukushima nuclear power plant. SNR research plays a subordinate role and is characterized by failures (e.g. Monju and participation in the Astrid project in France).

[37] https://www.ajudaily.com/view/20210901125057993 (last checked on 02.06.22)
[38] https://www.world-nuclear-news.org/Articles/Seaborg-and-BEES-sign-MOU-relating-to-floating-Com and https://www.world-nuclear-news.org/Articles/Korean-conglomerate-to-cooperate-with-TerraPower (last checked on 02.06.22)
[39] https://world-nuclear.org/information-library/country-profiles/countries-o-s/south-korea.aspx (last checked on 11.08.22)

The other countries in Category II (South Africa, Canada, Sweden and Romania) are less relevant. South Africa has been working on a very-high-temperature reactor for over a decade, but a lack of public and private funding has postponed the continuation of the project indefinitely and a future revival of the project is unclear. In Canada, the focus is on the development of SMR and is therefore not considered in more detail here. Sweden and Romania are involved in the development of LFR concepts, as is Belgium. However, the MYRRHA project has been classified by Belgium as a key research infrastructure project, meaning that the development activities in Belgium are classified as more relevant than in Sweden. MYRRHA is also seen as a necessary intermediate step for the ALFRED project (Alemberti et al. 2015). Any development delays in the MYRRHA project may therefore also have an impact on developments in Romania and are therefore considered here.

3.2.3.3 Category III: Poland

Poland was selected from Category III. None of the countries in this category operate nuclear power plants or have a known military programme or extensive research programmes for SNR. However, Poland has been planning to move into nuclear power for decades and would like to do so as part of the coal phase-out. In addition, the import of conventional reactors from manufacturers in the U.S., France and the Republic of Korea is currently under discussion, as are the country’s own research activities to develop a very-high-temperature reactor. It is therefore regarded as a “newcomer country”. In contrast, Denmark and Luxembourg also have isolated SNR research projects, but there are no concrete efforts to move into nuclear power. These two countries also took part in a letter to the European Commission to oppose the EU taxonomy and the subsidization of new nuclear power plant construction projects.[40]

3.3 Specific country studies

The development activities for SNR are presented below on the basis of selected country studies. The aim of the following country portraits is to provide an overview of the countries’ innovation systems. Firstly, the electricity markets and the status of the nuclear power programmes are discussed, followed by the dynamics of the respective national innovation process. This includes research infrastructures, constellations of actors, public and private budgets and current developments. The general method and the phases of innovation-economic development are presented at the beginning.

3.3.1 Overview: National innovation paths in reactor technology (with a focus on SNR)

This report analyses the national innovation paths over time. Nuclear technology development is divided into phases in order to show the dynamics. The start and end of each phase varies from country to country. A distinction is made between three central phases for countries with existing nuclear power plants: Build-up phase (“t_{-2}”), adaptation phase (“t_{-1}”) and current status (“t_0”).

Build-up phase (t_{-2}): In the first phase, the build-up phase, the country begins the development of SNR as part of the general development of a nuclear innovation system and with the establishment of the necessary research infrastructure. The start of technological development can be

[40] https://gouvernement.lu/en/actualites/toutes_actualites/communiques/2021/11-novembre/11-nuclear-free-taxonomy.html (last checked on 11.08.22)

characterised by in-house developments or the provision of technology by other countries (e.g. through the purchase of research reactors or know-how).

Adaptation phase (t_{-1}): In the second phase of the innovation process, the adaptation phase, an adaptation to the developments achieved in the build-up phase or a reorientation of the strategy takes place. In some cases, attempts are made to build reactors with a higher output (upscaling) or to focus the initially diverse technology portfolio.

Current status and developments (t_0): This phase describes the current status, sometimes including more recent developments. In some countries, attempts are made in this third phase to reactivate SNR developments, modernise outdated research infrastructures or accelerate development and expand reactor capacities. Table 3-2 summarises the key elements of each of the three phases for the countries under consideration and provides an overview of the national innovation paths.

Table 3-2: Overview of the national innovation paths of selected countries: U.S., Russia, Belgium (as of 2022)

Country	Build-up phase (t_{-2})	Adaptation phase (t_{-1})	Current status (t_{-0})
USA	**1940s – 1970s: Diversification with construction of prototypes** ~ 1950s: Focus on fast reactors: initially with metallic fuels (EBR-I, EBR-II, Fermi-1, reprocessing plant FCF) ~ 1960s: SFR with MOX fuels: Plan for the construction of a Clinch River demonstration reactor; Molten Chloride Experiment	**1970s – 2000s: Project terminations and decommissioning and diffusion of LWR** ~ Discontinuation of fast reactor projects (e.g. Clinch River), decommissioning of fast reactors (e.g. EBR-II, Fermi)	**Since 2000: Reactivation of SNR development and diversification with planning of new demonstration projects** ~ Development push with diverse development portfolio: SFR; VHTR; MSR ~ Attempt to build up missing research infrastructure (VTR) ~ 2020: Focus on two demonstration projects (sodium from TerraPower, Xe-100)
Russia	**1940s – 1970s: First experimental reactors focussing on SFR and the goal of a closed fuel cycle** ~ Development of first fast test reactors (BR-10, later BOR-60) ~ No recognisable focus on other technology lines	**1970s – 2000s: Attempt to upscale SFR** ~ Attempt to upscale fast reactors (BN-350, BN-600, with time delay BN-800) ~ Construction of reprocessing plant RT-1	**Since 2000: Delays and postponements** ~ Continuation of SNR development with focus on fast reactors (SFR, LFR): Commissioning of BN-800 (2016) ~ but: “commercial” reactor concept BN-1200 delayed ~Development and construction of the Brest-OD-300 reactor (LFR)
China	**1950s – 1970s: Establishment of the first elements of an imported nuclear energy innovation system** ~ Completely imported from the Soviet Union ~ 1960s first plutonium reactor ~ Focus on atomic bomb, nuclear missiles and hydrogen bomb ~ late 1960s: SFR research activities began with basic research and experimental facilities ~ No commercial developments yet	**1980s – 2000: Diversification of light-water reactor imports and first experiments with SNR** ~ Extensive imports of LWR (U.S., Russia, France, Republic of Korea) ~ Development of domestic adaptation capacities ~ The aim was to develop one (or more) national LWR (also for export) ~ Initial research work on SNR: ~ 2000: in commissioning HTR-10 ~ 2010: CEFR critical for the first time ~ 2011: Start of MSR development (TMSR-LF1)	**Since 2000: Consolidation of LWR and diversification of SNR** ~ Consolidation of domestic LWR (Hualong 1000) and increasing export attempts (Pakistan, UK) ~ Diversification of SNR: ~ 2021: Commissioning of the demonstration project: HTR-PM ~ 2020: Start of construction of the CFR-600 demonstration project ~2021: Completion of the TMSR-LF1 prototype
Republic of Korea	**1960s – 1990s: Development of an imported infrastructure** ~ Import of U.S. light-water reactors	**1990s – 2010: Supplier of commercial reactor technology and development of research**	**Status quo: Planning of SNR prototypes, problem of proliferation** ~ Planning of prototypes (PGSFR, NHDD)

Country	Build-up phase (t_{-2})	Adaptation phase (t_{-1})	Current status (t_0)
	~ Secret nuclear weapons programme (1970s), aborted after U.S. intervention Concept studies for fast reactors (SFR (Kalimer-150), LFR (Pacer))	**infrastructure regarding SNR** ~ Construction of experimental facilities ~ SFR (Stella-1) ~ LFR (Helios)	~ Active participation in international development activities ~ Problem of pyroprocessing of spent fuel rods (control by the U.S.) ~ Discussion about own nuclear weapons programme (threat from North Korea), currently “nuclear umbrella” provided by the U.S.
Belgium	**1940s –1990s: Development of research and industrial structures focussing on light-water reactors,** ~ e.g. BR-1, BR-2, BR-3, VENUS at the SCK CEN research institute ~ also operation of a Belgonucléaire MOX production plant	**From 1998: Start of MYRRHA design and development work** ~ Establishment of SNR-specific research infrastructure (from 1998, MYRRHA project) ~ 2006: Decommissioning of the MOX plant	**From 2018: Preparations for the construction of MYRRHA** ~ Research, design and authorisation processes of MYRRHA (national project in cooperation with EU) ~ Delay due to financing problems
Poland	**1950s – 1970s: Development of a basic nuclear infrastructure** ~ Research infrastructure for LWR through research reactors ~ EWA (1958-1995) ~ MARIA (1974-present)	**1970s – 2010s: Approaches to the development of a nuclear power plant** ~ until 1980s: Import from Soviet Union planned (VVER) ~ since 1990s: Import from western countries planned	**Since 2010: Continuation of attempts to build an LWR import reactor and planning of an SNR demonstration project** ~ Plans to enter nuclear power with LWR technology ~ Discussions with Westinghouse (AP1000), Framatome (EPR) and KEPCO (APR 1400) ~ Development of first HTR project (TERESA since 2017, GCR) with plans to build the infrastructure ~ Gospostrateg HTR project

Source: Own evaluation

3.3.2 USA

Since the 1940s, the U.S. has been the world’s technological leader in reactor technology with the development of the first nuclear reactors (ORNL 2010). Today, the U.S. has the world’s largest number of nuclear power plants on the grid, but also the oldest nuclear power plants. SNR development took place in parallel with light-water reactors and is characterised by technical problems, project cancellations and decommissioning. Since 2000, government interest in SNR has increased again, particularly on the part of the U.S. Department of Energy (DOE). The aim is to establish technological leadership (Chu 2010; Moniz 2011). In 2005, the Energy Policy Act was introduced for this purpose, but this did not lead to any significant change. Westinghouse is still fighting bankruptcy and the nuclear technology division of General Electrics has now been sold to Japan.

3.3.2.1 Current status of the power plant fleet

In 2021, around 61% of the electricity supply in the U.S. came from fossil fuels, 19% from nuclear energy[41] and 20% from renewable technologies (BP 2022). At the end of 2021, the average age of existing nuclear power plants was 42 years, without any significant addition of new capacity, so that a further decline in the share of the energy mix is expected in the coming years (Mycle Schneider Consulting 2022, p. 147).[42]

As of 31 December 2021, there are 93 reactors with an electrical capacity of around 95 GW in operation in the U.S., including 31 boiling water reactors (BWR) and 61 pressurised water reactors (PWR). It is the country with the largest number of active nuclear power plants in the world (IAEA 2023g, 7-8; Mycle Schneider Consulting 2022, p. 147). The oldest reactor to produce electricity was commissioned on 29 October 1957 in Vallecitos, about 30 miles east of San Francisco, and the most recent reactor was commissioned on 3 June 2016 in Tennessee between Chattanooga and Knoxville (IAEA 2023g).[43]

After a peak in new construction in the 1970s, the construction of new nuclear power plants has declined drastically since the 1980s (Thomas and Ramana 2022; IAEA 2022h, p. 8). In 2016, WAATS BAR-2, a reactor whose construction began in 1973 and was interrupted in 1985 (43 years of construction), was connected to the grid (IAEA 2022h).[44] In 2017, the new construction projects at the V.C. Summer site in South Carolina, which were launched in 2013, were cancelled. Only two reactors are currently under construction, the Vogtle-3 and Vogtle-4 plants in Georgia. Their completion is already significantly behind schedule and their cost estimates have also been exceeded (Wealer et al. 2021; Eash-Gates et al. 2020). Currently, 85% of the plants in operation have already submitted at least one application to the U.S. Nuclear Regulatory Commission (NRC) for an extension of their operating licence.[45] At the same time, the Office of Nuclear Energy of the U.S. Department of Energy (DOE) has stated that 13 commercial nuclear power plants in the U.S. have already been closed prematurely for economic reasons and that further plants could be shut down in the coming years due to a lack of economic efficiency.[46]

With the help of the “Bipartisan Infrastructure Law” and the resulting “Civil Nuclear Credit Program” with a commitment volume of USD 6 billion, owners and operators of commercial US reactors can bid for loans to support their continued operation for four years (117th U.S. Congress 2021, p. 945).[47] To do so, they must first be certified by the DOE, taking into account factors such as the cost of operating the reactor, the air pollution avoided and the reactor’s uranium source. Sealed bids are then submitted based on expected levelised cost of electricity per megawatt hour to be considered for the loan programme (117th U.S. Congress 2021, p. 945).[48]

[41] This corresponds to around 819.1 TWh of electricity generated, assuming a total of 4406.4 TWh (BP 2022).
[42] https://www.eia.gov/todayinenergy/detail.php?id=53459 (last checked on 03.09.22)
[43] (Elhegazy and Kamal 2022, p. 11) describes the X-10 research reactor in Oak Ridge, Tennessee as the first nuclear reactor.
[44] Further information can be found at https://www.latimes.com/business/hiltzik/la-fi-hiltzik-nuclear-shutdown-20170508-story.html (last checked on 02.11.2022)
[45] The current list can be viewed at https://www.nrc.gov/reactors/operating/licensing/renewal/applications.html (last checked on 02.11.2022).
[46] https://www.energy.gov/ne/civil-nuclear-credit-program (last checked on 17.03.22)
[47] https://www.energy.gov/ne/civil-nuclear-credit-program (last checked on 17.03.22).
[48] https://www.energy.gov/ne/civil-nuclear-credit-program (last checked on 17.03.22).

Since the 1990s, U.S. companies such as Westinghouse and General Electric (GE) have not been awarded large export contracts. In the deregulated electricity market, nuclear power is not competitive and the few construction projects have been executed with significant delays (Rothwell 2022). In the context of nuclear power export diplomacy, the U.S. is falling behind China, and both by a considerable margin behind Russia, which has now become the dominant exporter of nuclear power plants (subsidised by Russian state-owned companies) (Minin and Vlček 2018). With the development of SMR and so-called “novel” reactor concepts, the U.S. wants to make a technological leap, among other things to keep the nuclear power industry alive (116th U.S. Congress 2019; U.S. Senate Committee on Energy and Natural Resources 2019).

3.3.2.2 Development of the national innovation system

The following chapter describes the three phases of the build-up phase (1940s to 1970), the adaptation phase (1970 to 2000) and the current status (since 2000) for the U.S.

Build-up phase (t_{-2}): 1940s to 1970: From a military to a military-commercial innovation system and diversification

The origins of the national innovation system are linked to the Manhattan Project and military utilisation. At the Metallurgical Laboratory of the University of Chicago, later the Argonne National Laboratory, the first experimental nuclear chain reaction was produced under the direction of Enrico Fermi on 2 December 1942 at the so-called Chicago Pile 1 (CP-1) (IPFM 2010, p. 89). Subsequently, the pilot project X-10 was initiated for the purpose of plutonium separation. The air-cooled, graphite-moderated experimental reactor X-10, equipped with a thermal power of 1000 kW, went critical for the first time on 4 November 1943 (Reed 2021; ORNL 2010).

The Atomic Energy Law of 1946 stipulated that the previous purely military use of nuclear power was to be extended to include commercial applications. The Atomic Energy Commission (AEC) was founded for this purpose, from which the Department of Energy (DOE) later emerged. The AEC was entrusted with the supervision and control of the plants, equipment and research facilities involved in the development of the atomic bomb (Di Nucci 2019).

Within a few years, an interdisciplinary structure for basic and applied research was created, including research and production facilities that form the basis of the present-day system of national laboratories. Examples include plutonium production in Hanford (Washington) and the development of the first atomic bombs at the Los Alamos Scientific Laboratory (New Mexico) (DOE 2017a; Taylor and Freer 2002). For this purpose, a production apparatus was established with government funding, in which private-sector industrial companies (e.g. DuPont, Combustion Engineering, Babcock & Wilcox, General Electric) and university and public research and development institutions worked together under the leadership of the military. The recruitment and training of personnel was also largely carried out with the direct involvement or support of the army (Groves 1983). In the course of this, national research institutes were entrusted with SNR developments, including Los Alamos National Laboratory (LANL), Argonne National Laboratory (ANL, near Chicago), Oak Ridge National Laboratory (ORNL) and Hanford Engineer Works (HEW) (DOE 2017a).

At the beginning of the development of nuclear power, global uranium deposits were considered scarce. It was unclear whether these would be sufficient for military applications, and especially beyond that for the operation of nuclear power plants.

In August 1946, construction began on the Clementine experimental reactor, a mercury-cooled fast neutron spectrum reactor with 25 kW thermal power at the Omega Site location, the reactor went critical at the end of 1946 (IPFM 2010, p. 91). It was shut down in 1952 after a broken fuel rod caused plutonium to leak into the coolant.

On 19 November 1947, the development and construction of the first sodium-potassium-cooled experimental reactor EBR-I (electrical power 0.2 MW) with a fast neutron spectrum was started at the ANL with the aim of breeding plutonium and generating electricity. The EBR-I reached criticality for the first time in 1951 and was decommissioned in 1963 (IPFM 2010, p. 91).

This was followed by the sodium-cooled EBR-II (electrical output 20 MW, critical in 1963) and the sodium-cooled Fermi 1 (electrical output 66 MW, critical in 1963) at the ANL. The EBR II was also equipped with a reprocessing plant (fuel cycle facility, FCF), which reprocessed the spent fuel from the EBR II from 1965-1969 (IPFM 2010). Although the focus of development was on sodium-cooled fast reactors (SFR), research was also carried out in parallel on other concepts such as molten salt reactors (MSR), e.g. as part of the Molten Salt Reactor Experiment (MSRE) at Oak Ridge National Laboratory; the reactor had a thermal output of 8 MW and was in operation from 1965-1969 (IPFM 2010).

The 1960s also saw the development of alternative fuels. While all fast reactor concepts (EBR-I, EBR-II, Fermi 1) were based on metallic fuels, the development of uranium-plutonium mixed oxide fuels (MOX) began later (e.g. commissioning of the experimental reactor SEFOR in 1969) (IPFM 2010).

During the development of SFR technology line, however, the light water reactor technology prevailed (Cowan 1990). This is mainly due to the early introduction and intensive development of light-water reactors for the propulsion of submarines by the U.S. Navy. When the government massively expanded the use of commercial nuclear power plants, light water reactor technology had a great advantage in terms of experience (Cowan 1990). The corresponding technology was handed over by the U.S. Nuclear Navy, which provided the first commercial reactor in Shippingport (Pennsylvania) (SW2, “Nautilus”). The fast reactor concepts, which were considered superior, did not prevail over the less energy-efficient light-water reactors (Cowan 1990; Di Nucci 2019).

Adaptation phase (t_{-1}): 1970s to 2000: SNR decommissioning and project cancellations

In the late 1960s, plans began to build the SFR technology line with MOX fuel and to increase the capacity of these reactors, as they would be necessary for the long-term use of large-scale nuclear power and uranium, a raw material that was considered scarce at the time (IPFM 2010, 95ff). A commercial reactor was to be built by the mid-1980s. To this end, Congress authorised USD 87.5 million in July 1967 for the construction of a sodium-cooled reactor with a thermal output of 400 MW (Fast Flux Test Facility, FFTF) fuelled by MOX fuel (IPFM 2010, p. 98). This facility was specifically designed to carry out irradiation tests on nuclear reactor fuel and to test materials for liquid metal reactors with a fast neutron spectrum (HEDL 1980). Construction of the facility was completed in 1978 (planned in 1974) and the first “critical” state was reached in 1980 (IPFM 2010, p. 98).

In parallel, the Clinch River Demonstration Project was started in 1969, which was also to be a sodium-cooled fast reactor with MOX fuel with an electrical output of 350 MW based on the FFTF (IPFM 2010). However, the project was subject to increasing public criticism in the early 1970s. This was due to economic reservations as well as proliferation and safety concerns (IPFM 2010). The

nuclear bomb test by India in 1974 using plutonium separated with American help then triggered the decision to postpone commercial reprocessing, including the reprocessing of plutonium, indefinitely in 1977. To this end, the approval process for the operating licence for the Clinch River project was suspended. The Clinch River programme was also characterised by the increase in capital costs. From an original cost estimate of USD 699 million in 1972, the estimated cost of the project rose to over USD 4 billion in 1983. That year, the U.S. Congress cancelled funding for the project and the approval process was halted, marking the end of SNR development in the U.S. at that stage (IPFM 2010).

After the Clinch River project ended in 1983, the EBR-II and the reprocessing plant were converted into research facilities for the Integral Fast Reactor Program at Argonne National Laboratory. In 1994, the U.S. government cancelled the project and Congress cut off funding. The EBR-II was shut down the same year after 30 years of operation. The reprocessing plant was redesigned for the conditioning of radioactive waste (IPFM 2010, 103ff). Despite considerable efforts, the next innovative step in SFR technology failed to materialise.

The plan to build a demonstration reactor for the Molten Salt Breeder Experiment was proposed by Oak Ridge National Laboratory in 1972, but rejected by the Atomic Energy Commission because it wanted to reduce the number of fast neutron spectrum reactor concepts and predicted lower breeder rates compared to SFR. The ORNL’s attempt to continue the project was unsuccessful and was finally cancelled in 1976 for cost reasons (IPFM 2010, p. 99).

Current status (t_0): Since 2000: Attempts at the development of SMR light-water reactors and SNR

At the beginning of the 2000s, attempts were made to halt the decline of U.S. nuclear power plant technology through extensive programmes in various technology lines. In 2005, the Energy Policy Act (EPACT) provided significant financial support for the construction of new light-water reactors. As a result, applications for funding were submitted for 20 nuclear power plant units[49], of which only two reactors (Vogtle, Georgia) are currently under construction with high cost increases (Bade 2017; Amy 2022). At the same time, a number of operating nuclear power plants are experiencing difficulties in ensuring economic efficiency, so that they are being taken off the grid prematurely or secured with new specific subsidies (Lovins 2022; Rothwell 2022). To address the existing problems of LWR, particularly long construction times and high costs, programmes for the development of lower-capacity nuclear power plants (so-called SMR concepts) have been initiated (Chu 2010; Boarin et al. 2021).

During this period, attempts were made to restart the further development of SNR with higher capacity.[50] In the period from 2005 to 2011, the U.S. attempted to push ahead with the construction of a “Next Generation Nuclear Plant Project (NGNP)” in the form of a very-high-temperature reactor (VHTR), which was required as part of the Energy Policy Act of 2005 (109th U.S. Congress 2005). These demonstration projects, as well as commercial application projects, were to be carried out as a cost-shared partnership between the government and the private nuclear power industry (109th U.S. Congress 2005). This approach was not successful in the case of the Next Generation Nuclear

[49] https://world-nuclear.org/information-library/country-profiles/countries-t-z/usa-nuclear-power-policy.aspx (last checked: 03.06.2022).
[50] The founding of the GIF-IV Forum also dates back to this time.

Plant Project. Due to limited government funding and disagreements, the project has not been pursued since 2011 (INL 2011).

In 2019, the DOE initiated the Versatile Test Reactor (VTR) programme with the aim of building a nationally usable test facility for reactors with a fast neutron spectrum by the end of 2025.[51] However, the necessary government funding has not been approved in the required amount, making it difficult for the project to progress. A final decision on the construction of the project is still pending and must be authorised by Congress.[52] The design is to be based on the EBR-II.[53]

In 2020, the DOE initiated the “Advanced Reactor Demonstration Program” (ARDP) with the aim of implementing two demonstration projects with a start of operation from 2035. The programme does not focus on a specific technology line, but these projects are in competition with each other. The award of the contract depends on the technological maturity of the proposed reactor concept and the financial capacities of the companies (DOE 2020c). As a result, new companies dedicated to the development of SNR designs have emerged (e.g. X-Energy and TerraPower). Table 3-3 summarises selected projects of so-called “novel” reactor concepts (SNR) in the U.S. with funding from the current Advanced Reactor Demonstration Program (ARDP).

[51] https://www.energy.gov/articles/secretary-perry-launches-versatile-test-reactor-project-modernize-nuclear-research-and (last checked on 22.06.22)

[52] https://www.powermag.com/does-decision-to-build-versatile-test-reactor-coming-soon/ and https://www.neimagazine.com/news/newsdoe-selects-sodium-cooled-fast-reactor-design-for-versatile-test-reactor-9884005 (last checked on 18.11.22)

[53] https://neutronbytes.com/2022/07/30/doe-decides-to-build-the-versatile-test-reactor-at-inl/ (last checked on 18.11.22)

Table 3-3: Selected SNR projects in the U.S. with funding from the current Advanced Reactor Demonstration Program (ARDP)[54]

Technology line	Reactor concept [Reactor abbreviation]	Developer (company)	Electrical output [MW]	Coolant	Neutron spectrum	Funding programme
SFR	Sodium	TerraPower (Bellevue, Washington) and GE Hitachi	345 MW (~550 MW over 5.5 h from storage)	Sodium	fast	ARDP Category 1 demonstration project, pre-licensing phase
VHTR	Xe-100	X-Energy (Rockville, Maryland)	80 MW	Helium	thermal	ARDP Category 1 demonstration project, pre-licensing phase
VHTR	Hermes Reduced-Scale Test Reactor	Kairos Power, LLC (Alameda, CA)	-	Fluoride salt	thermal	Category 2 in ARDP: Risk reduction for future demonstrations
VHTR	eVinci	Westing-house Electric Company, LLC (Cranberry Township, PA)	7-12 MW	Sodium heat pipes	thermal	Category 2 in ARDP: Risk reduction for future demonstrations
MSR	Molten Chloride Reactor Experiment	Southern Company Services Inc., Birming-ham, AL and TerraPower	-	Molten salt	fast	Category 2 in ARDP: Risk reduction for future demonstrations, design phase
VHTR	BWXT Advanced Nuclear Reactor	BWXT Advanced Technolo-gies, LLC (Lynchburg, VA)	17 MW	Helium	thermal	Category 2 in ARDP: Risk reduction for future demonstrations
SFR	Inherently Safe Advanced SMR for American Nuclear Leadership	Advanced Reactor Concepts LLC	100 MW	Sodium	fast	Category 3: Advanced reactor concepts 2020, conceptual design
GFR	Fast Modular Reactor Conceptual Design	General Atomics	50 MW	Helium	fast	Category 3: Advanced reactor concepts 2020, conceptual design
VHTR	Modular Integrated Gas-Cooled High Temperature Reactor	Massachusetts Institute of Technology	-	-	-	Category 3: Advanced reactor concepts 2020, conceptual design

[54] Data as of June 2022.

Source: Own compilation based on (IAEA 2020; Pistner und Englert 2017)[55]

3.3.2.3 Interim conclusion

The U.S. had been a world leader in reactor technology development since the 1950s through the Manhattan Project. They are pursuing plans to take several so-called “novel” reactor concepts, in particular reactors with a fast neutron spectrum, from invention to innovation and diffusion. However, diffusion, both in the U.S. and internationally, has occurred only in light water reactors and not, as originally expected, in the other technology lines. With the widespread reduction of orders for light water reactor construction since the 1980s, U.S. nuclear power technology is in decline, which even the Energy Policy Act of 2005 has not been able to halt. The activities observed over the last ten years or so to promote both light-water reactors with low outputs (SMR concepts) and SNR are an attempt to re-establish a claim to international technology leadership for US nuclear power plant technology. New companies from the private sector are also active in this area. At present, no commercial breakthrough is in sight.

3.3.3 Russia

Russia owns a large number of reactors, pursues an active export policy and has extensive research programmes in the field of SNR. There have been repeated phases of active research funding in recent decades. After the catastrophic accident in Chernobyl (1986), the construction of new light-water reactors stagnated. Export projects (including Iran, China and India) have increased the importance of the nuclear power industry in Russia in recent decades. The following chapters provide an overview of the current nuclear power plants, the development of the national innovation system and the current status of SNR (including BN-600, BN-800 and BN-1200).

3.3.3.1 Current status of the power plant fleet

In 2021, around 62% of the electricity supply in Russia was generated from fossil fuels, 19% from nuclear energy[56] and 19% from renewable technologies (BP 2022). As of 31 December 2021, there were 37 reactors with an electrical output of around 28 GW in operation in Russia; the average age was 28 years, with eleven reactors already over 41 years old and 13 reactors between 31 and 40 years old. This means that Russia is also facing the challenge of an ageing power plant fleet. Russia currently has 25 light water-cooled and moderated reactors (PWR, VVER design), ten light water-cooled, graphite-moderated reactors (LWGR, RBMK design) and two sodium-cooled fast reactors (BN-600 and BN-800 (IAEA 2023g).[57] Furthermore, three reactors, including two VVER light-water reactors with a total electrical reference power of 3.4 GW and a lead-cooled reactor with a fast neutron spectrum (BREST-OD-300) with an electrical reference power of 300 MW, are under

[55] https://www.terrapower.com/our-work/natriumpower/ und https://www.energy.gov/ne/articles/energy-departments-advanced-reactor-demonstration-program-awards-30-million-initial and https://www.energy.gov/ne/articles/us-department-energy-announces-160-million-first-awards-under-advanced-reactor and https://www.energy.gov/ne/articles/energy-departments-advanced-reactor-demonstration-program-awards-20-million-advanced (last checked on 18.11.22)

[56] This corresponds to around 222.4 TWh of electricity generated, assuming a total of 1157.1 TWh (BP 2022).

[57] The sodium-cooled research reactor with a fast neutron spectrum BOR-60 is not counted here.

construction (IAEA 2023g).[58] In addition, the sodium-cooled research reactor MBIR with a fast neutron spectrum and an electrical output of up to 60 MW is being built to replace the research reactor BOR-60, also an SFR, which has been in operation since 1969.[59]

After the accident in Chernobyl in 1986, the expansion of power generation capacity stagnated. The collapse of the Soviet Union led to a lack of state funding for the development and expansion of nuclear power.[60] Export projects to Iran, China and India at the end of the 1990s gave the Russian nuclear power industry a new economic boost. This resulted in new expansion plans from the year 2000 onwards.[61] Russia is currently pursuing an active export policy with a number of projects (34 projects according to Rosatom) in various countries such as Bangladesh, Belarus, Finland, China, Turkey and Egypt (Thomas 2018).[62] As part of its nuclear diplomacy, Russia is aiming to build up influence in infrastructure projects around the world, but the literature anticipates problems with the fulfilment of orders (Thomas 2018).

In 2000, the Russian government published its long-term strategy (“Strategy for developing nuclear energy in Russia for the XXI century”) and described its plans for nuclear energy up to the year 2050. The strategy envisages the expansion and continued use of water-cooled reactors for electricity generation over the next 20-30 years, combined with a gradual increase in reactors with a fast neutron spectrum for plutonium production and waste disposal (Pioro 2016; Gagarinskiy et al. 2022). Apparently, the realisation of the closed fuel cycle by 2050 is considered possible. Some studies, such as (Gagarinskiy et al. 2022), even speak of a “consensus” in the scientific community and industry to develop a two-component nuclear energy system by the middle of the 21st century. This should therefore consist of the already developed VVER plants and reactors with a fast neutron spectrum (Gagarinskiy et al. 2022).

3.3.3.2 Development of the national innovation system

The following chapter describes the three phases for Russia: the build-up phase (1940s-1970), the adaptation phase (1970s to 2000) and the current status (since 2000).

Build-up phase (t_{-2}): 1940s to 1970

In 1948, the first light water-cooled, graphite-moderated reactor (LWGR) with a thermal output of 100 MW went into operation at the Mayak site for the purpose of plutonium production. In addition to producing plutonium, this type of reactor was also used to generate electricity. In 1954, an LWGR APS-1 (electrical output 5 MW) went into operation in Obninsk and was connected to the power grid.[63] This was followed by further reactors at three locations: At the Mayak Production Association in the Urals (same location as the reprocessing plant), at the Siberian Chemical Combine in Seversk

[58] According to (IAEA 2023g), BALTIC-1 has been under construction since 2012, but it is reported that it has been interrupted since 2013, see https://www.osw.waw.pl/en/publikacje/analyses/2013-06-12/russia-freezes-construction-nuclear-power-plant-kaliningrad (last checked on 17.11.2022). Therefore, it is not included.

[59] See https://www.world-nuclear-news.org/Articles/Completion-of-MBIR-reactor-brought-forward (last checked on 17.11.22)

[60] https://world-nuclear.org/information-library/country-profiles/countries-o-s/russia-nuclear-power.aspx (last checked on 17.11.22)

[61] https://world-nuclear.org/information-library/country-profiles/countries-o-s/russia-nuclear-power.aspx (last checked on 22.3.22)

[62] https://rosatom.ru/en/investors/projects/ (last checked on 01.09.2022)

[63] https://www.energy.gov/ne/articles/9-notable-facts-about-worlds-first-nuclear-power-plant-ebr-i and https://pris.iaea.org/PRIS/CountryStatistics/ReactorDetails.aspx?current=447 (last checked on 18.11.22)

near Tomsk and at the Mining and Chemical Combine near Krasnoyarsk (Diakov 2011). These are the sites that also play a role in some of the fast neutron spectrum reactors and reprocessing (Mayak reprocessing plant, BREST-OD-300) (see below).

At the same time, the development of fast neutron spectrum reactors began in 1949 at the Institute of Physics and Power Engineering (IPPE) in Obninsk on the initiative of Alexander Leypunsky. Given that uranium resources could be used more efficiently with the rapid expansion of this technology, the Russian government initiated a development programme for fast neutron spectrum reactors at the end of 1949 with the aim of developing a closed fuel cycle (IPFM 2010, p. 63). From 1949 to 1959, the institute developed several experimental reactors: in 1955, the BR-1 (Bystry Reactor-1), a plutonium-fuelled critical assembly without coolant, went into operation. The compact plutonium core and the uranium cladding achieved a breeding coefficient of 1.8. One year later, the BR-2 went into operation. Mercury was used as the first coolant. However, this meant that stable operation was not possible even at low temperatures (IPFM 2010, p. 63). In addition, mercury leaks occurred at the pipe connections and corroded the steel panelling. In 1959, the sodium-cooled BR-5 (thermal output 5 MW) went into operation. Its thermal output was gradually increased to 10 MW. It was in operation until 2004. In addition to research purposes, it was also used for the production of medical isotopes (IPFM 2010, p. 64).

The BOR-60 research reactor was built at the SSC-RIAR Institute in Dimitrovgrad for the development of SFR of the BN design, which went into operation in 1969 with an originally planned service life of 20 years. The aim was to test irradiation effects with a fast neutron spectrum on structural materials, fuels and absorption materials (Center for Arms Control, Energy and Environmental Studies 2013). This reactor has been repeatedly extended, is still in operation today and is used internationally.[64]

Adaptation phase (t_{-1}): 1970s to 2000: Attempt at upscaling

From the 1970s onwards, efforts were made to further develop reactors with a fast neutron spectrum of the Russian BN series and to increase their capacity to achieve scaling effects. In 1972, the BN-350 demonstration reactor with an electrical output of 350 MW went into operation. It was operated with uranium oxide, but the first tests with MOX test fuel rods were also carried out (IPFM 2010, p. 64). The reactor was built on the Mangyshlak peninsula on the Caspian Sea with the aim of desalinating water and generating electricity. Like the previous research reactors, it also had to contend with sodium fires and was decommissioned in 1999 (IPFM 2010, pp. 64–65).

Even before the BN-350 went into operation in 1972, the government decided to develop the next SFR with an electrical output of 600 MW. The BN-600 was developed with initial experience made with the BN-350. The BN-600 reached its first criticality at the Beloyarsk power plant on 26 February 1980. (IPFM 2010, p. 64). The steam generators were located in different parts of the plant so that the reactor could continue to run in the event of a fire. In fact, various sodium fires also occurred during the operation of the BN-600. The BN-600 was overhauled in 2010 and its service life was extended until 2025 (IPFM 2010, p. 65).[65]

[64] https://www.neimagazine.com/news/newsrosatom-seeks-foreign-partners-for-mbir-reactor-7986750 (last checked 02.09.22)

[65] https://world-nuclear-news.org/Articles/BN-600-licensed-to-operate-until-2025 (last checked on 24.08.2022)

In the period between 1970 and 1980, the IPPE presented two further designs, the BN-800 (electrical output 800 MW) and the BN-1600 (electrical output 1600 MW), with the aim of developing the BN-1600 as a series-mature commercial reactor (IPFM 2010, p. 65).

The Russian government planned to expand the capacity of SFR of the BN series with five BN-800 reactors in the Ural region. These plans were scrapped after the Chernobyl accident in 1986 and the development programme was curtailed (IPFM 2010; Pioro 2016).

Although construction of the BN-800 started in 1984 as unit 4 of the Beloyarsk power plant, it was interrupted after the accident in Chernobyl. With the decommissioning of the BN-350 in 1999 and the BN-800 still under construction, the BN-600 and BOR-60 remained as research and development reactors for SNR in Russia at the end of the 1990s.

The originally expected uranium shortage was not confirmed by the discovery of large uranium reserves in Kazakhstan in the 1960s and 1970s, which also invalidated the argument of the potential economic competitiveness of reactors with a fast neutron spectrum at very high uranium prices (Pioro 2016).

Current status (t_0): Since 2000: Delays and postponements

Since 2000, expansion plans for nuclear power and the development of so-called “novel” reactor concepts have once again been pursued. In 2010, President Putin announced that, in addition to the construction of new light-water reactors, the focus would be on the development of reactors with a fast neutron spectrum. To this end, the BN series development programme was to be continued, a lead-bismuth-cooled reactor (SVBR-100) with an electrical output of 100 MW was to be built by 2015 and, in addition, a lead-cooled reactor (BREST-OD-300) with an electrical output of 300 MW and a multi-purpose research reactor (MBIR). The total budget for 2020 was around RUB 60 billion (USD 2 billion).[66]

In 2012, the strategy for the development of fast neutron spectrum reactors was continued through the federal programme for advanced nuclear technologies. For this purpose, the construction of three reactors (at that time still planned until 2020) was determined: the BREST-OD-300, the SVBR-100 and the BN-1200 (Mycle Schneider Consulting 2021). The current energy strategy plan up to 2035 stipulates that only the BREST-OD-300 is to be built. The construction of the BN-1200 has been further postponed.[67]

In 2016, after 30 years of construction, the BN-800 was put into operation with the aim of demonstrating the full operation of an SFR with MOX fuel in Russia (Pioro 2016). This began in 2020 and by the following year, the reactor core already contained 30% MOX fuel elements (Rosatom

[66] https://www.world-nuclear-news.org/Articles/Russia-spreads-nuclear-funds-around (last checked on 24.08.22)

[67] https://www.neimagazine.com/news/newsrussia-defers-bn-1200-until-after-2035-7581968 (last checked on 02.09.22)

2019).[68] At the beginning of 2022, a proportion of 60% was reached[69] before full loading with MOX fuel took place in June 2022. [70]

The next prototype in the development series was the BN-1200, a concept with an electrical output of 1220 MW (Shepelev 2013; Pakhomov 2018). However, the project was postponed until the mid-2030s due to budget cuts and the need for further development of the fuel, and other new construction projects were brought forward.[71]

Rosatom states that one challenge of Russian research infrastructure is that much of this infrastructure was commissioned over 35 years ago, leading to a lack of research capacity. However, the BN reactors are capable of being used for research work to a certain extent. There was a lack of replacements for large-scale research programmes.[72] Current research is mostly carried out using the BOR-60 fast neutron spectrum research reactor, which has been in operation since 1969 and has been repeatedly given service life extensions. It continues to be actively used by international research projects in other countries and is to be replaced by the current MBIR (multi-purpose fast neutron research reactor) project.[73]

The MBIR is a sodium-cooled fast reactor (electrical output of 55 MW) and is planned to have a service life of 60 years. It is a key project for maintaining the research infrastructure in Russia and was therefore included in the national strategy “Nuclear Energy technologies of new generation 2010-2015 and till 2020” in 2010.[74] The MBIR has been under construction on the site of the RIAR research institute in Dimitrovgrad since 2015 and is scheduled for completion in 2027, one year earlier than originally planned.[75] AEM Technology, a subsidiary of Atomenergomash (itself a subsidiary of Rosatom),[76] was selected as the supplier of major components for the construction of the research reactor, with the aim of delivering them by 2016.[77] In April 2022, Rosatom reported that the reactor vessel had been delivered.[78] Table 3-4 presents an overview of current SNR projects in Russia.

[68] https://world-nuclear-news.org/Articles/BN-800-fast-reactor-fully-loaded-with-MOX-fuel (last checked on 07.09.22)
[69] https://www.world-nuclear-news.org/Articles/BN-800-running-on-60-MOX (last checked on 07.09.22)
[70] https://www.neimagazine.com/news/newsbn800-fast-reactor-fully-loaded-with-mox-fuel-9795904 (last checked on 07.09.22) and https://www.world-nuclear-news.org/Articles/Beloyarsk-BN-800-fast-reactor-running-on-MOX (last checked on 18.11.22)
[71] https://world-nuclear-news.org/Articles/Rosatom-postpones-fast-reactor-project-report-say (last checked on 19.09.2022)
[72] http://mbir-rosatom.ru/en/about/ (last checked on 19.09.22)
[73] https://www.neimagazine.com/news/newsrosatom-seeks-foreign-partners-for-mbir-reactor-7986750 (last checked 02.09.22)
[74] http://www.niiar.ru/eng/node/4508 (last checked on 24.08.22)
[75] https://www.world-nuclear-news.org/Articles/Completion-of-MBIR-reactor-brought-forward (last checked on 24.08.22)
[76] Also a subsidiary of Rosatom
[77] https://www.world-nuclear-news.org/Articles/MBIR-reactor-supplier-selected (last checked on 24.08.22)
[78] http://mbir-rosatom.ru/en/news/fast-neutron-reactor-vessel-delivered-to-dimitrovgrad/ (last checked on 25.08.22)

Table 3-4: Selection of SNR projects in Russia

Reactor concept	Technology line	Status of the project
BOR-60	SFR	In operation since 1969
BN-600	SFR	In operation since 1980
BN-800	SFR	In operation since 2016
BN-1200	SFR	Construction postponed; commissioning planned for 2036
BREST-OD-300	LFR	Under construction since 2021
SVBR-100	LFR	Design phase
MOSART	MSR	Design phase

Source: Own compilation based on (Pioro 2016; Center for Arms Control, Energy and Environmental Studies 2013; IPFM 2010)

3.3.3.3 Interim conclusion

In the field of SNR, the focus in Russia during the build-up phase was on reactors with fast neutron spectrum (SFR, later also LFR) in conjunction with reprocessing (Mayak, pilot plant and MOX fuel element factory in Zheleznogorsk). During the adaptation phase, this focus was deepened (BN-600, BN-800). Currently, the Russian innovation system with respect to SNR is in a phase where the research infrastructure is ageing (BOR-60, in operation since 1969) and projects are postponed (e.g. BN-1200), currently the BREST-OD-300 is prioritized. With the currently operating BN-800, an operation with 100% MOX fuel is being implemented. Russia maintains a long-term strategy of achieving a closed fuel cycle using fast neutron spectrum reactors and, in parallel, advancing the development of light water reactors.

3.3.4 China

China is a country with a high demand for energy that, in addition to the expansion of fossil fuels and renewable energies, is currently seeing by far the largest expansion of nuclear power. In the context of decarbonisation, the country is aiming to move away from the use of fossil fuels in the energy system and use alternative options. According to current studies, there is sufficient potential for renewable energies to generate electricity for this purpose (Burandt et al. 2019) The share of nuclear energy in electricity generation has so far been very low (approx. 5%).

3.3.4.1 Current status of the power plant fleet

In 2021, around 66% of the electricity supply in China was generated from fossil fuels, 5% from nuclear energy[79] and 29% from renewable technologies (BP 2022). As at 31 December 2021, there are 53 reactors in operation in China with a total electrical reference capacity of 52 GW, with an average age of 9 years (Mycle Schneider Consulting 2022). These reactors are made up of 51 water-cooled reactors, one helium-cooled very-high-temperature reactor (HTR-PM) with an electrical reference capacity of 210 MW and one sodium-cooled experimental reactor with a fast neutron

[79] This corresponds to about 407.5 TWh of electricity generated, assuming a total volume of 8534.3 TWh (BP 2022).

spectrum, the Chinese Experimental Fast Reactor (CEFR), with an electrical reference capacity of 20 MW (Pioro 2016; Oeko-Institut e.V.; WIP; PhB 2021; IAEA 2023g).[80] China has the third largest number of nuclear reactors in the world after the U.S. and France (IAEA 2020b; 2022i; BP 2022).

China is currently expanding its nuclear power capacity. Two additional reactors were commissioned in 2022. As of 31 December 2021, 20 reactors are under construction, with 18 light water-cooled and two sodium-cooled fast reactors (XIAPU-1 and XIAPU-2). This means that China is building by far the highest capacities in the world (IAEA 2022i; Mycle Schneider Consulting 2022).

As a one-party system, the direction of nuclear energy policy in China is under the control of the central government and the Communist Party. Five-year plans are a central control instrument (CEIP 2018). Longer-term developments, such as in nuclear technology, can be covered by several consecutive five-year plans. The 13th five-year plan for 2016-2020 envisaged achieving a reactor capacity of 58 GW in operation and 30 GW under construction by 2020 (NDRC 2016). Up to and including 2020, 50 reactors with an electrical reference capacity of approx. 47 GW were commissioned (IAEA 2021d) and China is therefore behind its five-year plan.

Electricity production from non-hydro renewables (i.e. solar, wind, geothermal and biomass) has increased from 863.2 TWh in 2020 to 1152 TWh in 2021, accounting for around 14% of total electricity generation (BP 2022). However, the largest share of electricity in China is still produced with coal (62%) (BP 2022). The intensive use of fossil fuels means that China has the world’s highest emission levels of carbon dioxide (CO_2), sulphur dioxide (SO_2) and nitrogen oxides (NO_x), which lead to air pollution problems (Jin et al. 2016). President Xi Jingping declared in 2020 that China is aiming for carbon neutrality by 2060.[81] This would mean that China’s massive coal-fired power generation would have to be replaced by lower-emission technologies, which would probably also include further expansion of nuclear power.[82] In contrast to the 13th five-year plan, the 14th five-year plan (2021-2025) adopted in 2021 does not contain a specific target for nuclear power, but aims to further increase electrical capacity to 70 GW. Following the accident in Fukushima in 2011, approvals for new nuclear power projects in China were initially halted, safety checks were carried out on existing nuclear power plants and the construction of a number of already approved nuclear power plants was suspended. Subsequently, further expansion was to be continued step by step (CEIP 2018).

With regard to the country’s innovation activities, the strategic plan “Energy Technology Revolution and Innovation Action Plan (2016-2030)” describes that, in addition to the expansion of water-cooled reactors, there will also be further developments of SNR and SMR technology (in particular very-high-temperature reactors) (Zhan et al. 2021). Since 2013, cooperation agreements have been in place with a large number of countries such as Algeria, Argentina, Bulgaria, Egypt, Pakistan and others as part of the “Belt and Road” (New Silk Road) initiative (Lin et al. 2020). Compared to Russia, France or the U.S., the Chinese nuclear industry is still relatively young and its ability to operate its own reactors safely and reliably has not yet been proven (Thomas 2017).

[80] https://www.world-nuclear-news.org/NN-Chinese_fast_reactor_starts_supplying_electricity-2107114.html (last checked on 25.10.22) and https://www.world-nuclear-news.org/Articles/Demonstration-HTR-PM-grid-connected (last checked on 07.10.22)

[81] https://www.energypartnership.cn/home/events/china-energy-transition-policies-2020/ (last checked on 30.09.22)

[82] https://world-nuclear.org/information-library/country-profiles/countries-a-f/china-nuclear-power.aspx (last checked on 26.08.22)

3.3.4.2 Development of the national innovation system

The following chapter describes the three phases for China: the build-up phase (1950s-1980s), the adaptation phase (1980s to 2000) and the current status (since 2000).

Build-up phase (t_{-2}): 1950s – 1980s: Establishment of the first elements of an imported nuclear energy innovation system

In China, the development of nuclear energy began in the 1950s on the basis of bilateral co-operation with the Soviet Union. In the course of this, the Chinese Academy of Science (CAS) was founded on the Russian model with an institute for nuclear physics, which pursued both military and civilian purposes. In addition to its intentions to utilise nuclear power for peaceful purposes together with the Soviet Union, China also pursued the goal of developing nuclear weapons and built its first reactor to produce plutonium between 1960 and 1967. The reactor was unreliable and is estimated to have had a thermal capacity of between 200 and 600 MW in the early days, with similar specifications to the US plant at Hanford (Wright and Gronlund 2003). The first nuclear bomb test with uranium-235 took place in 1964, the first nuclear-tipped missile was detonated in 1966, the first hydrogen bomb was detonated in 1967 and the first nuclear-powered submarine was commissioned in 1970 (Lewis and Xue 1988; CEIP 2018; Zhou 2011). Shortly afterwards, the Cultural Revolution (1966-1976) initiated by Mao Zedong left a lasting mark on the country. The development of a nuclear industry for power generation was not prioritised, so that the development of a commercial nuclear programme was not carried out in the 1970s (CEIP 2018; Zhou 2011).

In November 1969, the party leadership decided to build a thorium molten salt reactor with an electrical capacity of 50 MW at Tsinghua University (IPFM 2017). An investment of 100,000 yuan was to be made to build the reactor in a cave and supply Tiananmen Square in Beijing with electricity from 1 July 1970 (IPFM 2017, p. 33). The reactor concept was also considered a potential candidate for an advanced power reactor for the navy. However, after considerable effort, the project was abandoned in 1979 (IPFM 2017, p. 33). Further efforts to develop so-called “novel” reactor concepts were made in 1970 through the “Project 728” launched by the Shanghai Institute of Applied Physics and research into various reactor technologies (IPFM 2017). The original goal was to build a reactor with an electrical capacity of 25 MW and liquefied thorium salt as fuel, but this was changed to a molten salt zero power reactor in 1971 due to the state of the art at the time.[83]

The first pressurised water reactor with an electrical capacity of 300 MW (Qinshan-1) was built in China in 1985 and has been in operation since 1991 (Du et al. 2022, p. 3).

With regard to the SNR with fast neutron spectrum, the first phase began with basic research in the field of reactor physics and thermodynamics as well as the construction of test facilities for the sodium cooling cycle and initial model tests from 1965 to 1987 (Cacuci 2010, p. 2348). The first phase of basic research into very-high-temperature reactors began at Tsinghua University in the 1970s (Pioro 2016).

Adaptation phase (t_{-1}): 1980s – 2000: Diversification of light-water reactor imports and first experiments with SNR

From the 1980s, China entered the world of commercial nuclear power with the construction of light water reactor technology (Ramana and Saikawa 2011). China endeavoured to catch up with

[83] Translated from the short introduction on the SINEP homepage, available at http://english.sinap.cas.cn/about_sinap/brief_introduction/ (last checked on 07.10.22).

international developments in order to reduce its use of fossil fuels in the energy system, especially coal, and to meet the increasing demand for energy in areas with a high population density, especially in the large cities on the coast (CEIP 2018). China thus began the system change from a purely military to a dual-purpose utilisation (commercial and military use). To this end, the Ministry of Nuclear Industry (MNI) was founded in 1982 by the Second Ministry of Mechanical Engineering (MMB) and entrusted with the civilian and military sectors. This was subsequently renamed China National Nuclear Corporation (CNNC) in 1989 (Zhou 2011).

When it came to selecting the technology for the first commercial reactor, there were disagreements as to which technology line should be built and whether it should be an in-house development or an import (Zhou 2011). For example, the Ministry of Nuclear Industry was in favour of importing technology, while the Ministry of Mechanical Engineering wanted to push ahead with the development of its own designs based on the nuclear submarines it had developed itself (Zhou 2011). The result of this lack of agreement was that both the company’s own design was sought and parallel negotiations were held on the import of reactors (Ramana and Saikawa 2011).

In 1985, the Shanghai Nuclear Engineering Research and Design Institute (SNERDI) began building its own design of a light-water reactor (CNP-300). The most important components were largely supplied by the Japanese Mitsubishi Corporation. This reactor, designated Qinshan-1, reached its first criticality on 31 October 1991 and went into commercial operation in 1994 together with the two imported light-water reactors (Daya Bay-1 and Daya Bay-2) from Framatome (French design M310). These imported reactors each have an electrical output of 950 MW and reached their first criticality on 28 July 1993 (Daya Bay-1) and 21 January 1994 (Daya Bay-2). This was followed by the import of technologies from Canada (CANDU-600) and Russia (VVER-1000) at the end of the 1990s (Thomas 2017; Ramana and Saikawa 2011; Zhou 2011). However, the projects were not linked to a long-term political development plan or strategy, but were pursued discontinuously and inconsistently, so that China both developed its own design and purchased technologies from other countries. The financing of nuclear power projects in China was based exclusively on state funds (Zhou 2011).

Since the 1980s, a series of development programmes have been introduced to improve China’s competitiveness in science and technology in the 21st century. Three programmes were essential: the “Key Technologies R&D Program”, implemented in 1982, the “863 Program” in 1986 and the “973 Program” in 1998.[84] These programmes contained strategies and objectives for the development of so-called “novel” reactor concepts in the context of nuclear technology development alongside other areas such as biotechnology, space and automation. For example, the 863 programme included the development of sodium-cooled fast reactors and very-high-temperature reactors: The construction of the HTR-10, decided in 1992, was set as a goal for this. China saw the development of very-high-temperature reactors as an opportunity to complement the range of light-water reactors. Passive safety features were intended to increase public acceptance (Xu and Zuo 2002).

The technical developments in the early phase of very-high-temperature reactors were characterised by intensive cooperation between China and Germany (Oeko-Institut e.V. 2017). Germany’s experience with the very-high-temperature reactors AVR and THTR-300 thus contributed to the design development of the HTR-10. Construction of the HTR-10 with an electrical output of 10 MW began in 1994 and the reactor was commissioned in 2003 (Pioro 2016). The aim was to gather

[84] http://www.china.org.cn/english/features/China2004/107131.htm (last checked on 14.09.22)

knowledge for the planning, construction and operation of very-high-temperature reactors, to test inherent safety features of the modular design and to create the possibility for irradiation and testing of fuel elements and materials (Zhang et al. 2009).

Sodium-cooled fast reactors were also included in the 863 programme as a further technology line. One example of this is the construction of the China Experimental Fast Reactor (CEFR), a research reactor with an electrical reference power of 20 MW (Pioro 2016; Oeko-Institut e.V.; WIP; PhB 2021).

With the development of reactors with a fast neutron spectrum, China is pursuing the goal of developing a closed fuel cycle. This goal was first set as a political objective in the 1980s to counteract a shortage of uranium resources with a further increase in capacity (Chen et al. 2018; Zhou 2011). The China National Nuclear Corporation (CNNC) therefore proposed to build a reprocessing plant and fast neutron spectrum reactors so that the spent fuel from the light-water reactors can be reprocessed and then reused for the operation of reactors with a fast neutron spectrum and/or light-water reactors (CEIP 2018). In addition, it is hoped that this will bring advantages in waste management, as the radiotoxicity of the waste will be lower compared to operation without reprocessing (CEIP 2018).

Other SNR technology lines are being pursued in isolated pilot projects. For example, supercritical water-cooled reactors (SCWR) are being supported by the Chinese Ministry of Science as part of the National Key Basic Research Program of China (Project 973) and molten salt reactors (MSR) and lead-cooled fast reactors (LFR) are being developed as part of pilot projects by the Chinese Academy of Sciences (CAS) (Pioro 2016).

Current status (t_0): Since 2000: Consolidation of LWR and diversification of SNR

Work on the CEFR research reactor began in 2000 under bilateral co-operation with Russia on the design and construction of the reactor (BCSIA 2016; CEIP 2018). Since 2007, China has been in negotiations with AREVA (France) for the purchase of a commercial reprocessing plant. In addition, negotiations have been underway with Russia since 2008 for the potential purchase of two BN-800 reactors.

The expansion strategy included both light-water reactors and so-called “novel” reactor concepts. China set itself ambitious goals in a nuclear power-specific strategic plan (“Medium-long term plan for nuclear power 2005-2020”): The aim was to expand the electrical supply capacity of nuclear power plants from 7 GW (2005) to 70 GW (2020) (CEIP 2018). For the accelerated expansion, a well-established state-of-the-art technology was to be used to import some reactors and then gradually transfer the technology so that it could be produced by Chinese companies. The choices were the EPR from AREVA and the AP-1000 from Westinghouse. China decided to import technology from the U.S. and use the Westinghouse design (AP-1000) to develop it into its own design (CAP-1400) through a newly established company (State Nuclear Power Technology Company, SNPTC) (Thomas 2017).

At the same time, the French imported reactor design M-310 was further developed by the Daya Bay project itself (CPR-1000). These developments led to an increasing number of reactor designs being added (Ramana and Saikawa 2011).

The accident in Fukushima in 2011 affected the expansion of nuclear power worldwide and prompted many countries, including China, to adjust their energy policy with regard to the use of nuclear technology (Ming et al. 2016). For example, the rate of expansion of light-water reactors in China has been reduced and safety precautions have been prioritised. This also led to the cancellation of

some projects and ensured that the expansion was only carried out gradually from then on (CEIP 2018). The previously ambitious targets of 70 GW by 2020 were reduced to 58 GW in 2012 as part of the State Council’s energy strategy (“Energy Development Strategy Action Plan 2014-2020”) (BCSIA 2016). As a result, the strategic import strategy was successful: the “ACP-1000” design of the CNNC and the ACPR design of the China General Nuclear Power Group (CGN) were merged into an LWR design of an electrical capacity of 1000 MW (Hualong-One, HPR-1000) (Ramana and Saikawa 2011; Thomas 2017; CEIP 2018). Both players have been building light-water reactors independently since the 1990s and have their own supply chains, so it is possible to use similar but non-identical Hualong-One designs (Nian 2017). Two HPR-1000 reactors are in operation and operated by CNNC.[85]

With regard to the development of so-called “novel” reactor concepts, development is continuing and the first demonstration reactors are being built: since 2012, two high-temperature reactor modules of the HTR-PM (electrical output 210 MW) have been under construction in Shandong Province and reached criticality for the first time in 2021.[86] In 2011, the project for the test reactor TMSR-LF (thermal power 2 MW) was initiated by the China Academy of Science and construction was completed in Gansu Province in 2021 (Pioro 2016).[87]

With regard to the SFR, attempts were also made to import technology from abroad: China had originally planned to base the development of a commercial SFR on the Russian BN-800 concept. In October 2009, CIAE and the China Nuclear Energy Industry Corporation (CNEIC) signed an agreement with the Russian company Atomstroyexport to begin preliminary design and construction work on a commercial nuclear power plant with two BN-800 reactors.[88] These negotiations are still ongoing, as disagreements over the price have meant that no final purchase agreement has yet been concluded (BCSIA 2016). With regard to the closed fuel cycle, China attempted to import technology for reprocessing. Negotiations with France on the purchase of a reprocessing plant have been ongoing since 2007 (BCSIA 2016). In 2016, it was reported that negotiations are ongoing and construction is scheduled to start in 2020 and that the site selection process has been started in potential regions (Jiangsu, Gansu, Fujian, etc.). However, this was suspended by the city administration of Lianyungang in the Chinese province of Jiangsu due to public protests.[89] Further agreements (e.g. memorandum of understanding in 2018 between New Areva and CNNC) are continuing to drive forward cooperation on the construction of a reprocessing plant, but no agreement has yet been reached.[90] Table 3-5 summarises the development status of so-called “novel” reactor concepts in China.

[85] FUQING-5 reached its first criticality on 21.10.2020 and entered commercial operation on 30 January 2021; FUQING-6, reached its first criticality on 12 December 2021 and entered commercial operation on 25.03.2022. https://pris.iaea.org/PRIS/CountryStatistics/ReactorDetails.aspx?current=938 and https://pris.iaea.org/PRIS/CountryStatistics/ReactorDetails.aspx?current=937 (last checked on 16.11.22)

[86] https://www.world-nuclear-news.org/Articles/Demonstration-HTR-PM-grid-connected (last checked on 15.09.22)

[87] https://www.world-nuclear-news.org/Articles/Chinese-molten-salt-reactor-cleared-for-start-up (last checked on 15.09.22)

[88] https://www.neimagazine.com/features/featurea-new-breed-for-china-5919186 (last checked on 06.10.22)

[89] https://www.world-nuclear-news.org/Articles/Reprocessing-plant-siting-work-halted-in-Lianyunga (last checked on 15.09.22)

[90] https://www.world-nuclear-news.org/Articles/France-and-China-to-enhance-nuclear-energy-coopera (last checked on 15.09.22)

Table 3-5: Selection of SNR projects in China

Technology line	Reactor concept	Manufacturer	Electrical output	Coolant	Spectrum	Fuel	Current status
SFR	CEFR	Various e.g. China Institute of Atomic Energy and OKBM Afrikantov, NIKIET, Kurchatov Institute (Russia)	20 MW	Sodium	Fast	Uranium oxide, later MOX	In operation since 2011, with inter-ruptions
SFR	CFR-600/ CDFR-600	China Institute of Atomic Energy	600 MW	Sodium	Fast	Uranium oxide, later MOX	Under construction since 2017 (Xiapu-1) and 2020 (Xiapu-2)
VHTR	HTR-10	Tsinghua University	2.5 MW	Helium	Thermal	TRISO	In operation since 2003
VHTR	HTR-PM	Tsinghua University	211 MW	Helium	Thermal	TRISO	In operation since 2021
MSR	TMSR-LF	China Academy of Science	2 MW (ther-mal)	Flibe	fast	Uranium-thorium mix	Completion of construction in 2021, and operating licence since 2022

Source: Own compilation based on (Oeko-Institut e.V.; WIP; PhB 2021; Mycle Schneider Consulting 2022; IAEA 2023g)

3.3.4.3 Interim conclusion

China has advanced its nuclear innovation system through an import strategy since the 1960s and is now the third nuclear power alongside the U.S. and Russia. Following military developments in the 1950s, advances have been made in both light water reactors and SNR. After the initial phase of basic research, China entered the development of SNR in the 1980s, when countries such as Russia, Germany and the U.S. had gained initial experience and China was able to benefit from this knowledge and initial technological developments. China is pursuing an import strategy that is followed by domestic substitution in order to further develop technologies and expertise and thus build up its own knowledge, design and industrial production. This transfer has worked in the field of light water reactor technology. The so-called “novel” reactor concepts are being developed in parallel with the expansion with light-water reactors. China has built up a wide range of technology lines, especially fast reactors and high-temperature reactors. Currently, the projects are still in the area of basic research or prototypes, a commercial roll-out is not yet foreseeable.

3.3.5 Republic of Korea

The development of nuclear power in the Republic of Korea (South Korea) started in the 1950s and is strongly characterised by agreements with the U.S. on safety and nuclear policy. This applies to both light-water reactors and the development of so-called “novel” reactor concepts. Although South Korea has strategically maintained its innovation system in the field of nuclear technology, it has only

implemented a few new construction projects in recent decades and is therefore heading towards declining capacities. There are research projects on SNR, in particular fast neutron spectrum reactors, very-high-temperature reactors and isolated efforts to develop molten salt reactors.

3.3.5.1 Current status of the power plant fleet

In 2021, around 66% of the electricity supply in South Korea was generated from fossil fuels, 26% from nuclear energy[91] and 7.2% from renewable technologies (BP 2022). As of 31 December 2021, 24 reactors with an electrical reference power of 23 GW were in operation in South Korea, including 21 light water-cooled reactors (PWR) and three heavy water-cooled reactors (PHWR) (IAEA 2023g). In 2022, another light water-cooled reactor (SHIN-HANUL-1) with an electrical output of 1.3 GW went into operation and three more PWR are under construction.[92] This makes South Korea one of the world’s five largest producers of nuclear power (behind the U.S., France, Russia and China) (BP 2022). The average age of the South Korean power plant fleet is 28 years (Mycle Schneider Consulting 2022, p. 116).

Today, the Repblic of Korea is a democratically organised country which, due to its geopolitical position, is located between the spheres of influence of other countries with nuclear technology, such as the U.S., Russia, China and Japan (EPRS 2022). The influence of the U.S. in particular extends to military and economic efficiency. For example, South Korea is under the military “umbrella” of the U.S. as part of the “U.S.-ROK Mutual Defence Treaty”, which plays an important geopolitical role in the area of tension with North Korea’s nuclear weapons programme (CRS 2022). The two countries are linked by a mutual economic partnership, with the U.S. being South Korea’s second-largest trading partner and South Korea being the U.S.’s seventh-largest trading partner (CRS 2022).

South Korea’s energy policy has been focussed on the expansion of nuclear energy since it began using nuclear power commercially in the 1970s (Valentine and Sovacool 2010). However, the change of government in 2017 marked the first change of direction in energy policy. In 2017, then President Moon Jae-in adopted the energy plan the “8th Basic Plan for Long-term Electricity Supply and Demand (2017 - 2031)”, which envisaged a gradual phase-out of the commercial use of nuclear power. To this end, it was planned to shut down ten of the oldest reactors with an operating life of between 30 and 40 years between 2023 and 2029, forgo the construction of six new reactors and increase the proportion of renewable energies in the energy mix (Ministry of Trade, Industry and Energy 2017). [93] The reasons cited for the policy change were public concerns about the safety of nuclear power plants, which had increased following the accident in Fukushima in 2011 and the earthquakes in Gyeongju (2016) and Pohang (2017).

In spring 2022, there was another change of government, whose nuclear policy revised the planned termination of commercial use and once again targeted the expansion of nuclear power. Under the new President Yoon Suk-yeol, the plan is to increase the share of nuclear power in electricity generation to 30% by 2030 to counter the pressure of decarbonisation and ensure energy security.[94]

[91] This corresponds to about 222.4 TWh of electricity generated, assuming a total volume of 600.4 TWh (BP 2022).

[92] https://pris.iaea.org/PRIS/CountryStatistics/CountryDetails.aspx?current=KR (last checked on 14.10.22)

[93] These are the only three heavy water-cooled pressurised water reactors with an operating life of 30 years and 7 light water-cooled pressurised water reactors with an operating life of 40 years.

[94] https://www.world-nuclear-news.org/Articles/New-energy-policy-reverses-Korea-s-nuclear-phase-o (last checked on 14.10.22) and

Potential lifetime extensions for older reactors are also being considered (Mycle Schneider Consulting 2022, p. 117).

Another example of the change in energy policy is the “K-Taxonomy”, a classification system for investments based on six selected environmental goals (Shin & Kim LLC 2021). In the draft of this taxonomy, nuclear power was not listed as a sustainable investment at the beginning of 2022, i.e. before the presidential election. However, the Ministry of the Environment announced in a press release in July 2022 that it would be included in the taxonomy.[95]

In addition, South Korea is actively pursuing export plans, which led to the construction of four APR-1400 reactors with an electrical reference power of 1450 MW each in the United Arab Emirates in 2009. In 2019, a cooperation agreement was reached with Saudi Arabia for the construction of a low-power light water-cooled reactor (SMART).[96] South Korea recently submitted an offer to Poland for the construction of six nuclear power plants with a total electrical output of 8.4 GW and a cost of USD 26.7 billion.[97] According to the press release issued by the Korean Ministry of Trade, Industry and Energy (MOTIE) on 7 July 2022, ten reactors are to be exported by 2030.[98]

3.3.5.2 Development of the national innovation system

Build-up phase (t_{-2}): 1960s to 1990s: Development of an imported infrastructure

The beginnings of the South Korean nuclear programme can be dated back to 1956, when the Department of Atomic Energy was founded in the South Korean Ministry of Education and its head Pak Ch’ol convinced the government to establish a development plan for nuclear energy (Andrews-Speed 2020, p. 48). In 1958, an Atomic Energy Act was passed and the following year the Atomic Energy Bureau was established, which was directly under the supervision of the Presidential Office. In addition, an Institute for Atomic Energy Research was founded to conduct basic research (Valentine and Sovacool 2010; Andrews-Speed 2020, p. 48). The first research reactor (TRIGA MK-II) was imported from the U.S. in the 1960s (Andrews-Speed 2020).

In the 1962 plan to promote nuclear generation, nuclear power was seen as the promising technology that would provide the urgent energy needs for the country’s development and also lead to more energy independence, as South Korea was dependent on imports from other countries due to low fossil resources (Kim and Byrne 1996).

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http://english.motie.go.kr/en/pc/pressreleases/bbs/bbsView.do?bbs_seq_n=1008&bbs_cd_n=2&currentPage=25&search_key_n=&search_val_v=&cate_n= (last checked on 17.08.22)

[95] https://www.lexology.com/library/detail.aspx?g=861eba8d-0fdd-44d8-af20-2dc26ce55fbc (last checked on 18.08.22) and https://eng.me.go.kr/eng/web/board/read.do?pagerOffset=0&maxPageItems=10&maxIndexPages=10&searchKey=titleOrContent&searchValue=taxonomy&menuId=461&orgCd=&boardId=1538750&boardMasterId=522&boardCategoryId=&decorator= (last checked on 18.08.22)

[96] https://world-nuclear-news.org/Articles/Groundbreaking-for-first-UAE-reactor (last checked on 19.08.22) and https://world-nuclear-news.org/Articles/Korea,-Saudi-Arabia-to-cooperate-on-SMART-deployme (last checked on 19.08.22)

[97] https://www.world-nuclear-news.org/Articles/Korea-offers-six-reactors-to-Poland (last checked on 02.06.22) and the current report by Deutsche Welle https://www.dw.com/en/us-south-korean-firms-to-operate-nuclear-plants-in-poland/a-63576093 (last checked on 31.10.2022)

[98] http://english.motie.go.kr/en/pc/pressreleases/bbs/bbsView.do?bbs_seq_n=1008&bbs_cd_n=2&currentPage=25&search_key_n=&search_val_v=&cate_n= (last checked on 17.08.22)

During this time, the country was under the control of a military regime that combined the development of nuclear power with military autonomy and strength and aimed to develop nuclear weapons (Valentine and Sovacool 2010, p. 7975). Since 1953, South Korea has been under the military “umbrella” of the U.S. under the “U.S.-ROK Mutual Defence Treaty”, under which American nuclear weapons were stationed in South Korea and withdrawn again from 1991 (CRS 2022). Nevertheless, North Korea was perceived as a constant threat, which led to national safety being prioritised. (Kim and Byrne 1996, p. 285) argue that the dual-use nature of nuclear technology opened up the possibility of improving the country’s economic efficiency while pursuing the goal of national safety.

Construction of the first light and heavy water-cooled reactors began in 1972. Kori-1 went into operation in 1977.[99] As part of bilateral cooperation with America, the 123 Agreement has been in place for reprocessing and enrichment since 1973. 123 Agreements are bilateral co-operations under Section 123 of the Atomic Energy Act of 1954 between the U.S. and countries that use American reactor technology, components, materials and fuel. No enrichment and reprocessing may be carried out without the prior consent of the U.S. if these are based on supplied American materials and technologies (CRS 2011). Thus, the Republic of Korea has been allowed to acquire U.S. reactor technology, but it requires U.S. consent for any reprocessing or enrichment activities related to U.S.-supplied materials and technology (CRS 2013, p. 1). Parallel to the development of a commercial nuclear power industry, the Republic of Korea signed the Treaty on the Non-Proliferation of Nuclear Weapons in 1975 and renounced the acquisition and development of nuclear weapons.[100] At this time, South Korea attempted to obtain plutonium by building a reprocessing plant in cooperation with France. However, the project was halted in 1976 by the U.S., which threatened to deny export licences and loans, which South Korea was dependent on as they were required for the acquisition of American reactor technology (Kim and Byrne 1996).

Following the cancellation of the South Korean-French agreement on a reprocessing pilot plant, the Korean Energy and Power Corporation (KEPCO) signed a contract with Westinghouse and Atomic Energy of Canada in 1977 for the construction of two nuclear power plants (Kim and Byrne 1996). The Kori-2 light-water reactor and the Wolsong-1 heavy-water reactor went into operation in 1982 and 1983, respectively.[101] The first three reactors (Kori-1, Kori-2, Wolsong-1) were supplied by foreign companies (e.g. Westinghouse) under a “turn-key” contract and constructed on site. For example, Westinghouse supplied the reactor and other components of the nuclear steam generation system and built the Kori-1 and Kori-2 plants. After the first three units, Korean companies took over the construction of all other reactors, starting with Kori-3 (commissioned in 1985). In 1987, South Korea began to develop its own design (OPR-1000, later APR-1400) based on the “System 80” light water-cooled reactor concept from the American company Combustion Engineering (CRS 2013; Andrews-Speed 2020).

Adaptation phase (t_{-1}): 1990s – 2010: Supplier of commercial reactor technology and development of research infrastructure regarding SNR

Since the 1990s, light-water reactors of its own design (OPR-1000) have been built in South Korea, starting with Hanbit-3 (electrical reference power of 986 MW) and its commissioning in 1994.[102]

[99] https://pris.iaea.org/PRIS/CountryStatistics/ReactorDetails.aspx?current=394 (last checked on 07.11.22)
[100] https://treaties.unoda.org/a/npt/republicofkorea/SIG/washington (last checked on 18.10.22)
[101] https://pris.iaea.org/PRIS/CountryStatistics/CountryDetails.aspx?current=KR (last checked on 7.11.22)
[102] https://pris.iaea.org/PRIS/CountryStatistics/ReactorDetails.aspx?current=396 (last checked on 15.10.22)

Furthermore, a fund for research and development in the field of nuclear power was set up in 1996 as part of the Atomic Energy Act to promote nuclear energy in order to ensure stable and continuous funding. The energy supply companies were obliged to contribute a sum of money (1.2 South Korean won, i.e. around EUR 0.001 per kilowatt hour generated) for this purpose (Lee et al. 2018). Since then, there has been initial research into so-called “novel” reactor concepts, in particular reactors with a fast neutron spectrum, high-temperature reactors and isolated efforts to develop molten salt reactors.

The first studies on lead-cooled fast reactors have been conducted at Seoul National University (SNU) since 1996: a group of researchers there conducted a feasibility study for a concept of an accelerator-driven transmutation reactor in the 1990s. The Nuclear Materials Laboratory also conducted experimental studies on lead-bismuth alloy to explore its use as an alternative coolant in transmutation systems. With financial support from the Ministry of Science and Technology, they developed a reactor concept for the proliferation-resistant Environment-friendly Accident-tolerant Continuable Economical Reactor PEACER, a lead-bismuth cooled reactor with a fast neutron spectrum and a planned electrical output of 300 MW (Hwang et al. 2000). R&D activities were expanded in 2002 and the Nuclear Transmutation Energy Research Centre of Korea (NUTRECK) was established at Seoul National University. A test facility (HELIOS) was built in 2005 (Pioro 2016). Building on the previous projects, the SMR concept MircoURANUS was created, a lead-bismuth-cooled fast reactor with an electrical output of 20 MW (IAEA 2023d).

In parallel, SFR technology has been developed at the Korea Atomic Energy Research Institute (KAERI) since 1997 as part of a national research programme with the aim of securing key strategic technologies and developing a conceptual design for a sodium-cooled fast reactor (Yoo et al. 2016, p. 1060). The project aims to develop an SFR technology to secure key strategic technologies and to develop the conceptual design of a salt-cooled reactor. Both would be necessary for an efficient utilisation of uranium resources and a reduction in the volume and radiotoxicity of high-level waste (Yoo et al. 2016, p. 1060).

The results of these initial research projects are the conceptual designs of a sodium-cooled fast reactor KALIMER-150 with an electrical power of 150 MW and the KALIMER-600 with an electrical power of 600 MW and KALIMER-1200 with an electrical power of 1200 MW (Yoo et al. 2016).

Ajou University has also been working on a concept for molten salt reactors since 1998. The AMBIDEXTER-NEC (Advanced Molten-Salt Break-even Inherently safe Dual-function Excellently-Ecological Reactor Nuclear Energy Complex) concept is to be operated with DUPIC (Direct Use of Spent PWR Fuel in CANDU) fuel and used to reduce minor actinides (Pioro 2016, p. 363; KNS 2009).

South Korea is also trying to expand its nuclear expertise by actively participating in the Generation IV International Forum (GIF). South Korea has been an active member of the GIF since the latter was founded in 2001 (GIF 2001). In 2019, the representative of South Korea was appointed as Vice Chair to promote research and development cooperation between GIF members and observer organisations. More than 30 South Korean experts and engineers are involved in various projects carried out as part of the international joint research programme.[103]

[103] https://www-pub.iaea.org/MTCD/publications/PDF/cnpp2020/countryprofiles/KoreaRepublicof/KoreaRepublicof.htm (last checked on 09.08)

In 2005, the advancing climate change and the high dependence on fossil fuel imports prompted the government to develop a long-term strategy for the transformation to a hydrogen-based economy (Pioro 2016). Since 2006, the state research institute KAERI has been preparing the development of very-high-temperature reactors (Pioro 2016). In addition, a programme was initiated at KAERI in 2012 to secure key technologies for the production of hydrogen from nuclear power and to build a demonstration reactor for the Nuclear Hydrogen Development and Demonstration Project (NHDD). According to the organisation, this should be ready for use by 2030 (Pioro 2016). Development gaps were identified for the technical implementation, for example the development of process heat exchangers, the manufacture of components, the analysis tools for VHTR concepts and the development and production of a coated particle fuel. Possible NHDD plant designs based on a reactor with a thermal output of 200 MW were analysed (Chang et al. 2007). However, this plant has not yet been built.

Current status (t_0) (from 2008): Planning of SNR prototypes, problem of proliferation

In 2008, the Korean Atomic Energy Commission authorised a long-term development plan to build a prototype sodium-cooled reactor with a fast neutron spectrum by 2028 to demonstrate transmutation technologies. A safety case for a specific design was to be submitted in 2017 and the licence granted in 2020. To this end, a national project was launched in 2012 with the aim of developing the Prototype Gen IV Sodium cooled Fast Reactor (PGSFR) (Yoo et al. 2016).

In the area of light water-cooled reactors, there was a delay in the construction and operation of plants following the accident in Fukushima in 2011 due to safety upgrades.[104] In addition, the public became more critical of the operation of nuclear power plants (Lee et al. 2020).

Research on SNR continued and the first test facilities were set up at KAERI. In 2014, STELLA-1 (Sodium Test Loop for Safety Simulation and Assessment), a test facility for carrying out safety simulations of components for emergency and residual heat removal from the reactor, went into operation. A second facility for testing the thermal-hydraulic properties of heat transport systems, STELLA-2, has also been operational since 2022 (Pioro 2016; Yoon et al. 2022; Lee et al. 2022).

The bilateral agreement from 1974 (“123 Agreement”), which was concluded between the U.S. and South Korea, expired in 2014 and was renewed in 2015. As part of the negotiations, South Korea requested permission for its own uranium enrichment and reprocessing, both of which were previously only permitted with the consent of the U.S. (CRS 2015; Kang and Hippel 2017). The U.S. once again rejected reprocessing due to proliferation risks and the geopolitical influence of this decision on the desired denuclearisation in North Korea.

In the agreement signed in 2015, the U.S. made concessions and agreed that the spent fuel for reprocessing could be exported from South Korea subject to American approval in order to meet South Korea’s challenge with regard to the increasing quantities of spent fuel reprocessing. With regard to uranium enrichment, the new agreement specifies that enrichment of up to 20% uranium-235 would be possible following consultation with a bilateral commission and written approval from the U.S. These approval conditions were not part of the previous agreement. This means that enrichment is still only possible subject to certain approval conditions (commission, written consent). South Korea therefore does not currently have enrichment capacities, but wanted to include a provision in the new agreement that would open up this possibility for the future (CRS 2015).

[104] https://www.neimagazine.com/features/featuresouth-korea-beefs-up-safety/ (last checked on 16.11.22)

Finally, a historic change in energy policy took place in 2017, which signalled the end of the commercial use of nuclear power (Ministry of Trade, Industry and Energy 2017). The nuclear industry had already been confronted with a decline in new construction projects since the accident in Fukushima. With the lack of new construction projects, companies hoped that exports would be made abroad to maintain the industry, as otherwise they would face an uncertain future with the planned nuclear turnaround (Lee et al. 2020). At that time, funding for nuclear research and development was linked to nuclear power generation, meaning that a decline in the amount of energy generated from nuclear power would have had a direct impact on R&D activities. In 2022, there was another political change and instead of ending expansion and exports, the South Korean nuclear power industry was revitalised, see Chapter 3.3.5.1.

The threat of North Korea’s nuclear weapons programme is still present. In October 2022, North Korea was reported to have fired missiles and artillery shells at its east and west coasts.[105] The development or deployment of nuclear weapons in its own country is the subject of political discussion in South Korea regarding the country’s security policy.[106]

3.3.5.3 Interim conclusion

South Korea is one of the leading industrialised countries and, originally with the support of the U.S., has developed into one of the few suppliers of reactor technology. South Korea has its own extensive commercial nuclear power programme, which was also able to record exports in the 2000s. South Korea is also a member of the Nuclear Non-Proliferation Treaty (NPT). The country maintains particularly close relations with the U.S. with regard to its research and reactor development as part of the “123 Agreement”. In the field of SNR, South Korea is intensifying its participation in foreign, especially U.S. developments. In addition, the country is pushing ahead with its own developments, e.g. of reprocessing technologies in conjunction with fast reactors. The associated ambitions to build a reprocessing plant are still the subject of current discussions. These developments must also be seen in the context of geopolitical tensions with North Korea. Commercial use of SNR is not foreseeable at present.

3.3.6 Belgium

Belgium is an example of a country with decades of experience in the operation of light water-cooled reactors and SNR research. Belgium had already decided in 2003 to end the commercial use of nuclear energy (by 2025). For some years now, however, there have been renewed discussions about extending the operating life of some nuclear power plants. As part of the European research project MYRRHA (Multipurpose hYbrid Research Reactor for High-tech Application), Belgium has been working on the implementation of this project for decades. The Belgian power plant fleet, the development of the national innovation system and research in the field of SNR are presented in this chapter.

[105] https://www.dw.com/en/north-korea-missile-launches-tested-nuclear-wipe-out-of-south/a-63388440 (last checked on 16.11.22)

[106] https://www.dw.com/en/south-korea-eyes-nuclear-option-amid-north-korean-threats/a-63493061 (last checked on 16.11.22)

3.3.6.1 Current status of the power plant fleet

In 2021, around 28% of the electricity supply in Belgium was generated from fossil fuels, 50% from nuclear energy[107] and 22% from renewable technologies (BP 2022). At the beginning of 2022, Belgium was still operating seven light water-cooled reactors (LWR) with an electrical capacity of 5.9 GW and an average age of 41 years, which generated 50.6 TWh of electricity in 2021 (IAEA 2023g; BP 2022; Mycle Schneider Consulting 2021, p. 357). In 2003, it was decided by law to end the commercial use of nuclear power by 2025 and to shut down the reactors after 40 years of operation, which would result in the reactors being shut down between 2015 and 2025. Three reactors (Doel-1, Doel-2 and Tihange-1), which already had a service life of over 40 years, were granted a licence to continue operating until 2025 (Mycle Schneider Consulting 2021, p. 357).[108] In 2023, the Belgian government submitted a request to the operator to maintain operation until 2027.[109]

In spring 2022, the Belgian government decided to extend the service life of the two newest reactors (Doel-4 and Tihange-3, which are around 36 years old) by ten years (IEA 2022, p. 6). A ten-year extension would postpone the shutdown of the last reactor until 2035. However, ENGIE, a subsidiary of Electrabel, which operates the Belgian power plants, sees the extension of the operating life as a major challenge in terms of safety and the implementation of requirements (IEA 2022, p. 108).[110] On 23 September 2022, one of the reactors (Doel-3) was shut down as part of the end of commercial use of nuclear power.[111] This was followed by the shutdown of another reactor (Tihange-2) on 31 January 2023.[112]

In Germany, the Doel-3 and Tihange-2 reactors led to discussions and even a demand for shutdown, as safety deficiencies such as cracks in the reactor pressure vessel had been identified.[113] According to EU regulations, Belgium is obliged to develop a long-term energy and climate strategy that contributes to the EU’s 2050 climate neutrality target. The Belgian strategy (LTS2050) was adopted by the Commission in early 2020 and is based on long-term strategies of the regional governments and focuses on emission reductions in the areas of electricity, industry, buildings, transport, agriculture and waste (IEA 2022). In spring 2022, the Belgian government adopted a programme to accelerate the transformation to climate neutrality. In addition to the expansion of renewable energies and hydrogen technology, the programme also includes so-called “novel” reactor concepts. The SCK CEN research institute has been allocated a budget of EUR 100 million over the next four years to develop these concepts.[114] So far, no technology line is the focus of the planned research,

[107] This corresponds to about 50.6 TWh of electricity generated, assuming a total volume of 100.2 TWh (BP 2022).

[108] https://www.erneuerbareenergien.de/energiemarkt/energiemaerkte-weltweit/belgische-angst-nuklearkonzern-engie-trennt-sich-eilig-von-doel-3 (last checked on 02.08.22)

[109] https://www.world-nuclear-news.org/Articles/Belgium-considers-extended-use-of-older-reactors#:~:text=The%20Belgian%20government%20has%20asked,to%20shut%20down%20in%202025 last checked on 05.03.2023)

[110] https://nuclear.engie-electrabel.be/en/press/release/extension-belgiums-nuclear-power-plants (last checked on 02.08.22)

[111] https://www.world-nuclear-news.org/Articles/First-Belgian-power-reactor-shut-down (last checked on 30.09.22)

[112] https://nuclear.engie-electrabel.be/fr/energie-nucleaire/la-mise-larret-de-nos-centrales-nucleaires/arret-definitif-de-doel-3-et-tihange-2 (last checked on 30.09.22)

[113] https://www.dw.com/de/haarrisse-in-belgischen-akw-tihange-und-doel-schon-seit-der-bauphase/a-40649959 (last checked on 02.08.22)

[114] https://www.premier.be/en/lifetime-extension-doel-4-and-tihange-3-nuclear-power-plants (last checked on 02.08.22)

but reactor designs that are “not water-cooled” are to be developed. Potential synergy effects are seen with the MYRRHA project, which can provide experience for lead-cooled fast reactors.[115]

Despite the planned termination of the commercial utilisation of nuclear power, Belgium intends to maintain and further develop its research activities. Belgium also classifies the maintenance and development of nuclear expertise as a future priority within the framework of the energy and climate strategy, whereby the MYRRHA project is emphasised (Belgian Government 2019).

3.3.6.2 Development of the national innovation system

The following chapter describes the three phases for Belgium: the build-up phase (1940s-1990s), the adaptation phase (1990s to 2018) and the current status (since 2018).

Build-up phase (t_{-2}):1940s to 1990s: Building knowledge and entering nuclear power with LWR technology

Access to nuclear energy expertise in Belgium began as early as the 1940s. Belgium’s uranium reserves in the Congolese mine Shinkolobwe led to an agreement with the U.S. for the development of the Manhattan Project: Belgium pledged uranium supplies in return for access to U.S. nuclear expertise for the commercial, non-military sector (SCK CEN 2002, p. 6).

The development of the national research infrastructure for nuclear power dates back to the 1950s and was expressed in the founding of the research centre “Studiecentrum voor de Toepassingen van de Kernenergie, Centre d’Étude de l’énergie Nucléaire” (SCK CEN) in 1952 in the municipality of Mol. Four nuclear research reactors were put into operation there between 1956 and 1964: the BR-1, an air-cooled and graphite-moderated research reactor with a capacity of 10 W to 1 MW, and the BR-2 a light water-cooled reactor for materials development and for the production of medical radioisotopes with a capacity of 125 MW. The third reactor, BR-3 with a capacity of 10 MW, is Belgium’s first pressurised water reactor, which served as a prototype for the subsequent reactors in Doel and Tihange. It was also used to train personnel. The fourth reactor is the VENUS research reactor, which enabled various studies on reactor core configurations and material embrittlement (SCK CEN 2002). Belgium thus built up its own nuclear research infrastructure with the help of the U.S. at the very beginning of global nuclear power developments. The BR-1, BR-2 and VENUS research reactors are still in operation today.[116]

SCK CEN also has many years of experience in the development and production of uranium-plutonium mixed oxide fuels (MOX) (SCK CEN 2002). In the 1960s, two laboratories were set up and, together with Belgonucléaire, a Belgian company, a plant for processing uranium and plutonium oxide into MOX fuels was developed and built. The plutonium was supplied from the U.S. These fuels were used in light-water reactors, but were also intended for reactors with a fast neutron spectrum, such as the fast breeder reactor in Kalkar, which was built in Germany but never went into operation (SCK CEN 2002, p. 14). Belgonucléaire operated its own production facility for MOX fuel in Dessel until 2006.[117]

[115] https://www.world-nuclear-news.org/Articles/Belgium-government-allocates-funding-for-SMR-resea (last checked on 16.09.22)
[116] https://www.sckcen.be/en/about-sck-cen/corporate-information/infrastructure (last checked on 17.11.22)
[117] https://www-pub.iaea.org/MTCD/publications/PDF/cnpp2020/countryprofiles/Belgium/Belgium.htm (last checked on 09.08.22)

After the accidents at Three Miles Island in 1979 and Chernobyl in 1986, research activities for nuclear safety were prioritised (SCK CEN 2002).

Adaptation phase (t_{-1}): From the 1990s to 2018: Establishment of SNR research structures and initiation of the MYRRHA project

In the 1990s, the development of SNR began with the initiation of the MYRRHA project, see Chapter 5.10. Since 1998, the Belgian nuclear research centre SCK CEN has been pursuing the MYRRHA project, which is intended to replace the BR-2 (SCK CEN 2002, p. 54).

The MYRRHA project was launched with the aim of not only demonstrating the operation of an accelerator-driven fast reactor (ADS), but also to contribute to the development of lead-cooled fast reactors (LFR) in Europe. Furthermore, MYRRHA will act as a European research facility for irradiation with a fast neutron spectrum (NEA 2009). In 2009, the Belgian government commissioned the Nuclear Energy Agency of the Organisation for Economic Cooperation and Development (NEA) to carry out an independent assessment of the MYRRHA project and to advise it on the next steps to be taken (NEA 2009, p. 7).

The MYRRHA project was initially to conduct the necessary research to ensure safe and reliable operation, while demonstrating the viability of a lead-bismuth cooled fast reactor and an accelerator-driven system (NEA 2009, p. 14). Subsequently, MYRRHA should be able to create suitable conditions for testing materials and producing some medium-light radioisotopes, which require a very high neutron flux and can be used as a reserve for the production of other radioisotopes (NEA 2009, p. 14).

The MYRRHA project has been in the process of preparing a licence application (pre-licencing) for many years now. One reason for this is that the design of the project has changed repeatedly and the application has had to be adapted accordingly: The SCK CEN annual report reports that the end of the pre-licencing phase was originally expected in 2014, but was postponed to the end of 2016 due to design changes (SCK CEN 2013, p. 36). The 2021 annual report describes that the further development of the design should fulfil all safety conditions by 2024 in order to submit an application for approval (SCK CEN 2021). According to (SCK CEN 2011, p. 47), it was assumed in 2011 that MYRRHA would be commissioned in 2023. The reactor is now scheduled to be commissioned in 2036.[118]

A further challenge for the licensing of the project was that in 2009 there was not a sufficient number of suitable specialists for the licensing authority who were familiar with subcritical reactor concepts as well as critical reactor systems and liquid metal cooling to carry out the licensing of MYRRHA. In 2001, a project was launched between the Belgian university and the SCK CEN to train nuclear engineering specialists. The OECD NEA team of experts recommended further specialisation so that the Belgian licensing authority would have a sufficient number of specialists available (NEA 2009, p. 21).

Current status (t_{-0}): Since 2018: Planning for approval and construction for the MYRRHA project

[118] https://www.sckcen.be/en/highlights-2021/european-partnerships/myrrha-design-enters-final-phase (last checked on 10.08.22)

The Belgian government approved state funding of EUR 558 million for the MYRRHA project from 2019-2038.[119] In addition, the SCK CEN project will be transformed into a non-profit organisation to attract foreign investors (SCK CEN 2021, p. 49). Initial plans are currently being drawn up for the construction of the particle accelerator buildings.[120]

3.3.6.3 Interim conclusion

Belgium, historically one of the first countries with commercial nuclear power plant use in the 1950s, has developed a small national innovation system since this initial phase. However, so-called “novel” reactor concepts played practically no role in this. Belgium’s activities for the development of SNR began in 1998 and focus on the development and internationalisation of the MYRRHA research project. The project is a combination of an accelerator-driven subcritical reactor (ADS) and a lead-bismuth-cooled fast reactor (LFR). Initial schedules and cost estimates have been exceeded and there are difficulties in financing the project as government funding has fallen short of the promised level in the past and costs have risen at the same time. Other sources of funding include isolated EU subsidised projects, but there are no foreign investors to date. The long-term continuation of the MYRRHA project is therefore uncertain.

3.3.7 Poland

Poland is in the process of transforming its energy system. Around 83% of Polish electricity generation is based on fossil fuels (primarily coal). However, Poland has significant potential for renewables to achieve a rapid transition. According to various Polish governments, the move into nuclear power is intended to reduce dependence on coal; to this end, the import of light-water reactors is planned. Research into very-high-temperature reactors is being conducted in isolated cases.

3.3.7.1 Current status of the power plant fleet

In 2021, around 83% of the electricity supply in Poland was generated from fossil fuels[121] and 17% from renewable energies[122] (BP 2022). This makes Poland the country in the EU that consumes the most coal.[123] Poland has sufficient renewable resources to cover its entire electricity production, as studies on full electricity production in Europe based on renewable energies show (Child et al. 2019; DIW 2020a).

Since the 1950s, there have been repeated discussions in Poland about moving into nuclear power (Gawlikowska-Fyk et al. 2014, p. 13). However, Poland does not yet operate a commercial nuclear power plant. Since 1974, a water and beryllium-moderated research reactor MARIA (thermal output 30 MW) has been operated at the National Research Centre for Nuclear Research (Narodowe

[119] https://world-nuclear-news.org/Articles/Belgian-government-approves-funding-for-Myrrha (last checked on 11.08.22)
[120] https://myrrha.be/news/opinion-issued-european-commission-myrrha-line-euratom-treaty (last checked on 08.09.2022)
[121] At 131.7 TWh, the largest share (73%) of the electricity generated comes from coal, 1% from oil (approx. 1.5 TWh) and 9% from gas (approx. 15.5 TWh)
[122] This corresponds to 30.1 TWh out of a total supply of 180 TWh of electricity
[123] https://ec.europa.eu/eurostat/statistics-explained/index.php?title=Coal_production_and_consumption_statistics#Consumption_and_production_of_hard_coal (last checked on 29.10.22)

Centrum Badań Jądrowych, NBJC) in Otwock (Krzysztoszek 2015). This research centre is currently developing a concept for a helium-cooled research very-high-temperature reactor (thermal power 30-40 MW), called “TeResa” (Dąbrowski 2022, p. 56).

As part of Poland’s current energy policy strategy until 2040, the Ministry of Climate and Environment presented a plan in 2021, which represents Poland’s contribution to the implementation of the goals of the Paris Climate Agreement (Ministerstwa Klimatu i Środowiska 2021). Poland’s energy strategy envisages reducing the share of coal in electricity generation to 56% by 2030 and increasing the share of renewable energy sources (mainly wind and solar) to 32% (Ministerstwa Klimatu i Środowiska 2021). There are also plans to build a Polish nuclear power plant fleet with a capacity of 6 to 9 GW by 2043. To this end, nuclear power is categorised as a “reliable”, “zero-emission” generation technology that should contribute to the stability of the future electricity supply system.

This assessment and targets are based on the strategy paper “Polish Nuclear Power Program” of the National Atomic Energy Authority, which was first adopted by the Polish government in 2014.[124] In the strategy paper, the National Atomic Energy Authority recommends replacing coal-fired power plants with new nuclear power plants. For the addition of 6 to 9 GW, light-water reactors, in particular pressurised water reactors (PWR), are considered by the National Atomic Energy Authority to be a suitable technology line, as they have the most experience in construction and regulation worldwide. As the expansion is to take place through the purchase of technology, a larger number of suppliers could supply these technology lines. Construction of the first reactors is scheduled to begin in 2026 (Ministry of Climate 2020).

Poland received offers from three countries for the selection of a partner country and reactor concept: in 2021, Westinghouse conducted a U.S.-funded study (front-end engineering and design) together with Bechtle and GE Hitachi based on the AP-1000 to advance the nuclear energy programme in Poland while simultaneously seeking consideration as a potential partner.[125] In addition, France (EDF) has offered to build six EPR reactors.[126] There is also an offer from South Korea for the construction of six APR-1400 reactors.[127]

In November 2022, the Polish government proposed using three Westinghouse AP-1000 reactors for the first nuclear power plant in Poland.[128] Westinghouse aims to build the reactor on site together with 22 Polish companies, with the aim of completing the first reactor by 2033.[129] The construction of South Korea’s APR-1400 in south-west Poland (Pątnów) is also still being investigated by the Polish government after the Polish lignite company ZE PAK, the listed energy group PGE (Polska Grupa Energetyczna) and the Korean company Korea Hydro & Nuclear Power signed a letter of

[124] https://www.gov.pl/attachment/4cddd10a-5e8b-414d-bb95-670f6507d73e (last checked on 22.08.22)
[125] https://world-nuclear-news.org/Articles/Polands-president-hails-nuclear-partnership-with-U (last checked on 22.08.22)
[126] https://world-nuclear-news.org/Articles/Poland-expands-cooperation-on-SMR-and-large-react (last checked on 31.08.22)
[127] https://world-nuclear-news.org/Articles/Korea-offers-six-reactors-to-Poland (last checked on 31.08.22)
[128] https://www.world-nuclear-news.org/Articles/Poland-s-government-confirms-Westinghouse-for-nucl (last checked on 22.02.23)
[129] https://www.world-nuclear-news.org/Articles/Poland%E2%80%99s-government-confirms-Westinghouse-for-nucl (last checked on 10.11.22)

intent.[130] In view of unresolved financing issues and lengthy planning processes, the realisation of these plans still appears questionable.

The National Atomic Energy Administration also sees a future role for SMR and so-called “novel” reactor concepts, particularly in high-temperature reactor technology. However, it also notes that these do not currently represent entry-level options and are characterised by major uncertainties, for example in terms of construction, production or costs (Ministry of Climate 2020).

The industry is currently making its first attempts to import light water-cooled SMR concepts from America and obtain an operating licence in 2022. For example, Polish copper and silver producer KGHM (for NuScale’s SMR) and energy producer Orlen Synthos (for GE Hitachi Nuclear Energy’s BWRX-300) requested a general statement from Poland’s national nuclear authority (ństwowa Agencja Atomistyki, PAA) on the use of SMR concepts.[131] The PAA describes the general opinion as an instrument of pre-approval for all solutions planned by the investor, including design, technological and organisational solutions that will have a direct impact on nuclear safety and radiation protection issues.[132]

Discussions are currently underway between the Polish energy-intensive industry, which is seeking alternatives for its emission-intensive production, and SMR providers. For example, the American company Last Energy signed a letter of intent with the Polish Special Economic Zone Legnica (LSEZ) and DB Energy for the construction of a power plant consisting of ten light water-cooled reactors with a total electrical output of 200 MW.[133] KGHM signed an agreement with NuScale in 2022 for the construction of an SMR in Poland.[134]

3.3.7.2 Development of the national innovation system

The following summarises Poland’s historical origins and attempts to enter nuclear power on the basis of (Gawlikowska-Fyk et al. 2014) and other sources.

Build-up phase (t_{-2}): 1950s – 1970s: Development of a basic nuclear infrastructure

Poland’s ambitions to enter nuclear power began in the 1950s in the context of the “Atoms for Peace” speech by Dwight D. Eisenhower, President of the United States of America, at the 470th General Assembly of the United Nations. Poland was part of the Eastern Bloc at the time. After the USSR joined the “Atoms for Peace” programme, the Polish government announced the establishment of the Institute of Nuclear Research (IBJ). The IBJ was founded in 1955. This was followed by the construction of the first research reactor (EWA), a Soviet reactor of the WWR-S type with an electrical capacity of 10 MW, in 1956. This reached its first criticality in 1958 (IBJ 1971) and was in operation until 1995. Since 1974, the MARIA water and beryllium-moderated research reactor (thermal power 30 MW) has been in operation at the National Nuclear Research Centre (Narodowe Centrum Badań

[130] https://www.world-nuclear-news.org/Articles/Poland%E2%80%99s-government-confirms-Westinghouse-for-nucl (last checked on 10.11.22)
[131] https://world-nuclear-news.org/Articles/Applications-for-SMR-submitted-to-Polish-regulato (last checked on 23.11.22)
[132] https://world-nuclear-news.org/Articles/Applications-for-SMR-submitted-to-Polish-regulato (last checked on 31.8.22)
[133] https://www.world-nuclear-news.org/Articles/Last-Energy-agrees-to-build-ten-SMR-for-Polish-in (last checked on 18.08.22)
[134] https://media.kghm.com/en/news-and-press-releases/poland-s-first-application-to-evaluate-the-smr-technology-has-been-submitted-the-kghm-takes-another-significant-step-towards-nuclear-power (last checked on 10.11.2022)

Jądrowych, NBJC) in Otwock and is used for the production of radioactive isotopes and the implementation of nuclear research projects (Gawlikowska-Fyk et al. 2014).

Adaptation phase (t_{-1}): 1970s – 2010s: Approaches to the development of a nuclear power plant

In the early 1970s, Poland wanted to diversify its energy production technologies as it was faced with increasing energy demand and the idea that coal production would soon reach its peak. The Polish government decided to prepare for the construction of nuclear power plants and in 1974 signed a preliminary agreement with the Soviet Union for the construction of Soviet light water-cooled reactor technology (VVER-440) at the Żarnowiec site. Formal approval for four power plant units, including the necessary infrastructure (such as a railway station, workers’ dormitory, warehouses, etc.) was granted by the Council of Ministers in 1982. As a result, the National Atomic Energy Agency was founded and the start of the project was initiated. The reactors were not to be produced and supplied by Russian companies, but were to be manufactured by the Czechoslovak company Škoda and many key elements such as turbines, generators, steam generators etc. were to be produced by Polish manufacturers (Gawlikowska-Fyk et al. 2014).

In the late 1970s, the country was in an economic crisis, resulting in price increases, strike movements and the formation of the “Solidarność” trade union confederation, among other things (Fajfer 1993). With the introduction of martial law from 1981 to 1983, the Polish government intended to restrict the everyday life of the population and counteract the political opposition, in particular the Solidarność movement.[135] Meanwhile, planning for the construction of the nuclear power plants progressed (Gawlikowska-Fyk et al. 2014). The country’s own nuclear physics research was also continued . However, the IBJ was split into three institutes (Institute of Nuclear Studies (IPJ), Institute of Atomic Energy (IEA), Institute of Nuclear Chemistry and Technology (IChT)) in 1982 due to the martial law in force at the time, as the researchers showed solidarity with the strike movement.[136] In addition to the economic and political crisis, the financial difficulties and doubts about the profitability compared to cheap coal, there were national protests against the further construction of the nuclear power plant project after the Chernobyl accident in 1986. The project was finally abandoned in 1990. Some of the equipment was sold and can still be found in reactors in Finland and Hungary (Gawlikowska-Fyk et al. 2014).

In 2011, 30 years after the politically determined division, two of the three institutes (IPJ and IEA) were merged to form the National Nuclear Research Centre (NCBJ) and still operate the MARIA research reactor today.[137]

Current status (t_0): Continuation of attempts to build an LWR import reactor and planning of an SNR demonstration project

While Poland relies on importing technology from abroad in the area of light-water reactors, it is building up knowledge in the area of so-called “novel” reactor concepts. This is reflected in the fact that it is focussing on the development of very-high-temperature reactors. However, there are also activities in which the NCBJ is involved in the development of gas-cooled fast neutron spectrum

[135] https://polskiemiesiace.ipn.gov.pl/mon/all-events/december-1981/history/5296,The-genesis-of-Martial-Law.html (last checked on 16.11.22)
[136] Further information on the origins of the NCBJ can be found at: https://www.ncbj.gov.pl/en/history-ncbj/ncbj-roots-history-institute-nuclear-research-1955-1982 (last checked on 09.11.22)
[137] https://www.ncbj.gov.pl/en/o-nas/history-ncbj (last checked on 09.11.22)

reactors. Since 2012, it has been involved in the ALLEGRO project, which is being developed under the European Sustainable Nuclear Industrial Initiative (ESNII).[138] In 2021, the NBCJ reports that it will continue to be involved in the ALLEGRO project and will be involved in the application of special materials and the optimisation of the reactor core design and cooling system.[139]

However, the focus of Polish research activities is on the development of very-high-temperature reactors. In 2017, the development of a very-high-temperature reactor was included in the government’s economic development programme with the aim of using it for industrial heat generation in the future (Skrzypek et al. 2022). To achieve this, the National Nuclear Research Institute (NCBJ) is building up knowledge for very-high-temperature reactors as part of the GEMINI initiative and the resulting European research projects.[140] The GEMINI initiative is a transatlantic partnership founded in 2014 between the European Nuclear Cogeneration Industrial Initiative (NC2I) and the US Next Generation Nuclear Plant (NGNP) Industrial Alliance, which are working together on the design and regulatory framework of a gas-cooled very-high-temperature reactor (VHTR).[141] The NC2J is chaired by the Director of Development of Very-High-temperature Reactors at the National Centre for Nuclear Research (NCBJ).[142] The GEMINI initiative launched the European research project GEMINI+ (2017-2021) in which a conceptual design for a VHTR with a thermal power of 180 MW was developed (Dąbrowski 2022). Part of the project aimed to investigate the use of very-high-temperature reactors in Poland (Wrochna et al. 2020). Polish companies have also expressed interest in the use of very-high-temperature reactors, for example Tauron Polska Energia, one of Poland’s largest energy companies.[143] In addition, Japan (JAEA research institute) and South Korea (KAERI research institute) have also joined the GEMINI initiative and been involved in the GEMINI+ project (Wrochna et al. 2020).

In addition, the NCBJ is gathering knowledge on very-high-temperature reactors through cooperation with the Japan Atomic Energy Agency (JAEA). To this end, there is a strategic partnership between Poland and Japan, which organises cooperation on the development of VHTR concepts via an action plan (2021 to 2025) (Dąbrowski 2022). There has also been an unspecified agreement between the American company X-Energy and the Polish research institute NCBJ since 2017.[144]

Furthermore, the NOMATEN Centre of Excellence, established by the NCBJ and funded by the Foundation for Polish Science (FNP) and the European Commission, will focus for seven years (2018-2025) on the investigation of novel materials for use in difficult conditions, e.g. high temperatures, radiation and corrosive conditions (Dąbrowski 2022).

As part of the GOSPOSTRATEG national strategy programme, the Polish Ministry of Energy signed funding of EUR 4.5 million to carry out a project with the National Centre for Research and Development (2019-2022) to analyse the legal, organisational and technical instruments necessary for the use of VHTR. In addition, an initial concept for a VHTR research reactor (TeResa) was to be developed (Skrzypek et al. 2022). This is based on the concept that emerged from the GEMINI+ project and envisages a thermal output of 30-40 MW (Dąbrowski 2022).

[138] https://www.ncbj.gov.pl/en/aktualnosci/ncbj-has-joined-allegro-programme-develop-4th-generation-nuclear-reactors (last checked on 10.11.22)
[139] https://www.ncbj.gov.pl/en/seminaria/allegro-gas-cooled-fast-reactor-demonstrator-and-safeg-h2020euratom-project (last checked on 10.11.22)
[140] https://www.ncbj.gov.pl/en/ncbj-brief (last checked on 10.11.22)
[141] https://gemini-initiative.com/ (last checked on 09.11.22)
[142] https://snetp.eu/nc2i/ (last checked on 10.11.22)
[143] https://gemini-initiative.com/geminiplus/ (last checked on 09.11.22)
[144] https://www.ncbj.gov.pl/en/aktualnosci/ncbj-has-reached-agreement-x-energy (last checked on 10.11.22)

Based on this, the new project EUHTER (European High Temperature Experimental Reactor) was launched for the design and construction of an experimental VHTR concept for the TeResa research reactor. The project consists of three phases (Dąbrowski 2022):

- Phase I - basic design + preliminary safety report for authorisation (2021-2024)
- Phase II - detailed design + authorisation + construction + commissioning (2024-2030)
- Phase III - preparation (from 2023) for commercialisation and construction (from 2030)

For Phase I, the National Nuclear Research Institute (NCBJ) and the Polish Ministry of Education and Science signed a contract in 2021. The institute is set to develop a very-high-temperature reactor with a thermal output of 30-40 MW, powered by TRISO fuel and with an initial temperature of 750ºC. The design and concept study will be financed with national funds totalling PLN 60 million (approx. EUR 13 million) (Dąbrowski 2022).

In addition, the Gemini 4.0 project (2022-2025) was launched in June 2022 under the European funding programme “EURATOM-RIA - EURATOM Research and Innovation Actions”. The aim is to further develop the concept from GEMINI+. This includes the preparation for a licence, the development of a plan to develop a European “fuel cycle” for the TRISO fuel and its storage after operation as well as the development of a communication plan to convince politicians, industry and the public of this technology.[145]

3.3.7.3 Interim conclusion

Poland has been discussing the introduction of commercial nuclear energy for several decades. However, this has not yet been implemented. Imports of power plant technology from the Soviet Union were previously planned, but today these are to come from the U.S. and possibly South Korea. Since the 1950s, research on reactor technology has also been conducted on a small scale in Poland itself, most notably at the MARIA research reactor (in operation since 1974). For SNR, it can be observed that Poland is building knowledge, with Polish scientists participating in European research projects (GEMINI+, GEMINI 4.0, ALLEGRO). There is a special focus on the development of very-high-temperature reactors. In this area, there are currently plans to build a gas-cooled high-temperature research reactor (TeResa) based on the GEMINI+ project in collaboration with Japan. This is intended to be used for the development of VHTR in the future.

3.4 Conclusion: Country studies

There are numerous SNR research and development activities and pilot projects worldwide. Technology lines referred to today as “novel” were already researched and developed in the initial phase of nuclear technology developments in the 1940s and 1950s, above all the fast reactors, but also high-temperature reactors and molten salt reactors. Globally, the light-water reactor prevailed, while SNR ended up in the innovation-economic “valley of death”, i.e. the inventions were not followed up by subsequent innovations and spread of the technology.

[145] https://cordis.europa.eu/project/id/101059603 (last checked on 10.11.22)

Motives of countries to develop SNR include geopolitical and military aspects, decarbonization of the energy system, development of a closed fuel cycle, including with regard to waste management, and innovation competition.

The three nuclear superpowers (USA, Russia, China) have a common innovation dynamic: In their initial phases of nuclear power development, considerable resources were invested in the development of light-water and non-light-water reactors, but only the light-water reactors became commercially successful in further development. Overall, the country studies show that a system change from light-water reactors to series-mature SNR is not foreseeable.

4 Technology lines

This chapter discusses the various technology lines introduced in Chapter 2.3. It first outlines the main technical characteristics of each technology line in the form of a system description and explains important differences between reactor concepts within the technology line and in comparison with today’s light-water reactors. This is followed by a review of the historical developments in the technology line and an overview of the various specific reactor concepts currently being pursued within the technology line. Chapter 5 provides a detailed discussion of selected reactor concepts. Finally, the various technology lines are evaluated on the basis of the criteria introduced in Chapter 0.

4.1 Sodium-cooled fast reactors (SFR)

Concepts for lead-cooled fast reactors (SFR) have been discussed since the beginning of the use of nuclear energy. The main development goal is the production of plutonium from spent uranium fuel and the further use of this plutonium as fissile material for energy production. Therefore, potential use of SFR is always related to the development of plutonium-containing fuels and reprocessing facilities for the spent fuel from both LWR and SFR (Ohshima and Kubo 2016; Schulenberg 2020).

In more recent documents, the potential for transmutation of radioactive waste nuclides is cited as an advantage of this technology line in addition to the better yield of uranium resources (IAEA 2012b; Heidet 2021). However, this application requires considerable modifications to the reprocessing procedures, fuel production and the reactor core, as a transmutation reactor should burn plutonium and other actinides instead of breeding them.

In addition to its potential to produce plutonium, the advantages of SFR include the good heat conduction properties of sodium and the unpressurised nature of the primary cooling circuit, while the chemical reactivity of sodium and the higher reactivity control requirements are seen as important disadvantages (Mikityuk 2019).

4.1.1 System description

A distinction can be made between two main SFR designs (Schulenberg 2020). Many of the early SFR were modelled on LWR and were built using the so-called “loop” design. The reactor core is located in a reactor vessel. This is connected to a primary heat exchanger via pipework. In the primary cooling circuit, the coolant (in this case sodium) is pumped into the reactor vessel, flows through the reactor core, heats up in the process, leaves the reactor vessel via pipes and flows through a primary heat exchanger, where it transfers the heat generated to a secondary cooling circuit and is fed back into the reactor vessel (for such a basic design, see Figure 4-3, for example).

In contrast to this so-called “loop” design, newer SFR are typically constructed in a so-called “pool” or basin design (see Figure 4-1). Here, the primary heat exchanger is integrated into the reactor vessel and only the pipework for the secondary cooling circuit is routed out of the reactor vessel. The reactor vessel is typically surrounded by a further protective container, which collects any sodium escaping from the reactor vessel in the event of leaks. As the primary cooling circuit is located entirely in the reactor vessel, this prevents any extensive loss of sodium from the primary cooling circuit. The SFR currently under construction or in operation and classified as commercial are reactors with a basin design, see Table 4-4. This design is described in more detail below.

Reactor system

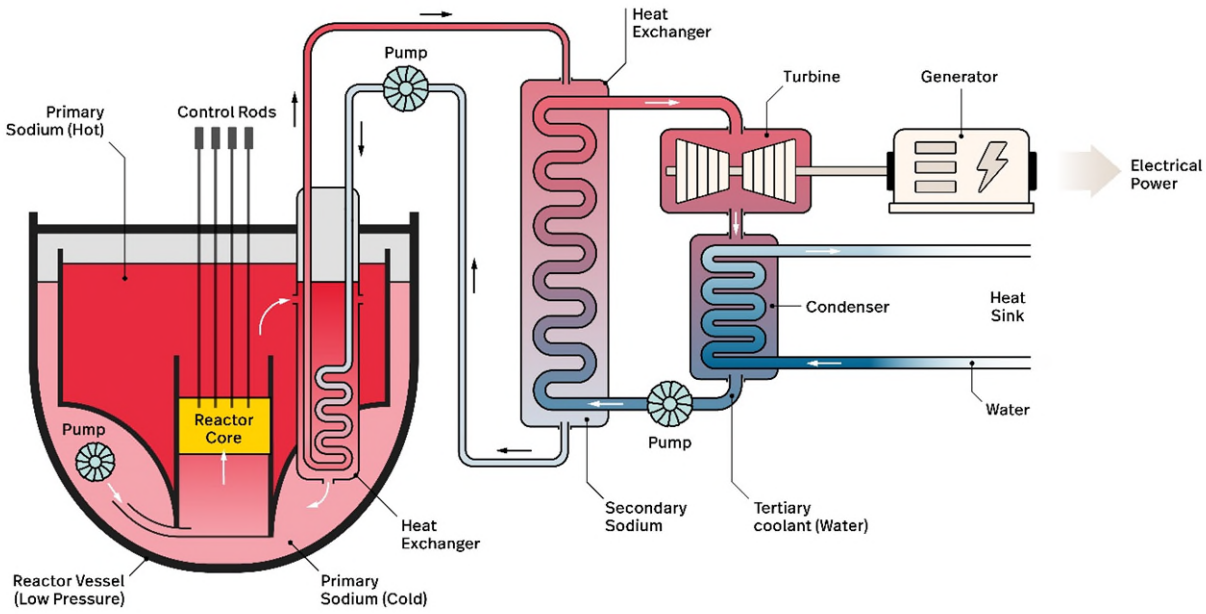
In SFR with a pool design, the reactor core, the primary coolant and the primary heat exchanger are located entirely in one reactor vessel. The boiling temperature of sodium is 883 °C. At the usual operating temperatures in the range of 500-550 °C, the sodium does not boil during operation, so the reactor vessel is not pressurised. The reactor vessel is sealed with a lid in all operating states to prevent sodium from coming into contact with air and to contain any radioactive substances in the coolant. Fuel elements must be loaded and unloaded via airlocks through the lid.

The reactor vessel is filled with liquid sodium. An inert gas is located in the upper area of the reactor vessel to provide a free volume for the expansion of sodium during heating and cooling processes.

The reactor core is located on a support structure in the lower section of the reactor vessel. The sodium is pumped through the reactor core from below and heats up in the process. The hot sodium above the reactor core should not come into direct contact with the walls of the reactor vessel so as not to expose them to the high core outlet temperatures. For this reason, there is a separating structure inside the reactor vessel that separates the colder sodium from the hot sodium.

The hot sodium enters a primary heat exchanger, transfers the heat to a secondary cooling circuit and exits the primary heat exchanger in the area of the colder sodium.

Figure 4-1: Conceptual diagram of a sodium-cooled fast reactor (SFR)



Source: Own illustration

Since no moderator is required in the SFR and sodium is a very good coolant, the fuel can be arranged very densely in the SFR. Unlike in LWR, the fuel rods in an SFR fuel element are therefore typically arranged in a hexagonal lattice. Similar to the LWR, the linear heat generation rate of the fuel rodin the SFR is limited by the maximum permissible temperature in the fuel and is around 40 kW per metre. Together with the denser fuel rod arrangement, this enables a significantly higher power density in the core compared to LWR.

The fuel rods typically have a diameter in the region of 10 mm and a length of one or a few metres. Uranium or uranium-plutonium mixed oxide is typically used as fuel. A fuel column made of depleted or natural uranium is typically located in the lower and upper sections of the fuel rods. This material is intended to reflect neutrons escaping from the reactor core back into the core and capture any remaining neutrons. Above the fuel column is a plenum area which serves to capture the gaseous fission products formed in the fuel and which limits the pressure build-up in the fuel rod.

The fuel rods are combined to fuel elements; metal wires typically serve as spacers between the fuel rods, which wrap around the rods and contribute to mixing of the coolant as it flows through the fuel element. The fuel elements are surrounded by fuel element channels, similar to the fuel elements in today’s boiling water reactors. The flow through the fuel elements can be influenced by orifices in the fuel element base depending on the average power density of the fuel element in the core. The fuel elements are positioned in a support structure in the lower section of the reactor vessel by means of a base section. A support structure is attached to the head of the fuel elements, which is required for changing the fuel elements.

The fuel elements in the reactor core are typically surrounded by reflector elements. These are similarly constructed elements, but are only filled with stainless steel instead of fuel. Their task is to reflect neutrons escaping from the core back into the core. Finally, these reflector elements are surrounded by further absorber elements in which the remaining neutrons are to be absorbed. The aim of this design is to shield the outer structures, in particular the wall of the reactor vessel, from the high neutron flux in the reactor core. The neutrons scattered back into the reactor core from the reflector contribute to new fissions with a time delay and thus also influence the reactor control.

Control and shutdown components made of boron carbide are typically used for power control and shutdown. Alternatively, hafnium is also being discussed as an absorber material. The control elements and the independent shutdown elements can be inserted into the reactor core from above.

The reactor vessel is typically surrounded by a second protective vessel. This is intended to collect the escaping sodium in the event of a leak in the reactor vessel and thus limit the loss of coolant. This ensures that the reactor core is covered with sodium at all times and that a sufficient flow of coolant to the primary heat exchangers is maintained.

In SFR, the core inlet temperature is in the range of 350-400 °C; at the core outlet, the sodium has a temperature of 500-550 °C. The energy generated is transferred via the primary heat exchanger to a secondary circuit, in which sodium is also used as a coolant. This prevents direct contact between the sodium on the primary side and water.

The secondary cooling circuit transfers the energy to a tertiary water-steam circuit, in which the energy is converted into electricity via a turbine and a generator. In the secondary heat exchanger, superheated steam is generated at pressures of 10-20 MPa and temperatures in the 400-450 °C range. In some reactor concepts, a gas circuit, for example with supercritical carbon dioxide, is also being discussed for generating electricity instead of a water-steam circuit. The residual heat is then transferred to an external heat sink via another cooling circuit. The efficiency of an SFR is thus typically around 40%.

For a fuel element change, the fuel elements must first be loaded inside the reactor vessel in a container filled with sodium. They can then be transferred to an external, cooled sodium container via an airlock in the lid of the reactor vessel. There they are stored until their decay power has

decayed. They must then be cleaned of sodium before they can be packed into transport or storage containers.

Fuel

Today, SFR are mainly operated with either enriched uranium or uranium-plutonium mixed oxide fuel (MOX), but various other chemical forms (metals, nitrides, carbides) can also be used as fuel.

Due to the fast neutron spectrum used, the effective cross-sections for neutron reactions in SFR are significantly lower compared to LWR. For this reason, the proportion of fissile material in the fuel must be significantly higher compared to LWR to obtain a critical reactor. The free path length of the neutrons and thus the neutron loss from the reactor core is also greater in SFR than in LWR, which means an additionally increased requirement for fissile material in the fuel, depending on the size of the reactor core. In larger SFR, the proportion of fissile material in the fuel is therefore typically 10-15%, while smaller plants require fissile material proportions in the region of 30%.

Since the spent fissile material is newly produced by neutron capture reactions and the fission products produced in the fuel contribute little to neutron absorption in the fast neutron spectrum, the reactivity of the fuel remains largely constant over the burnup, unlike in LWR. The achievable burnup of the fuel in SFR is therefore essentially limited by the material load of the fuel cladding and the fuel itself and is currently around 100 MWd/kg heavy metal.

Fuel cycle

A very wide range of different fuels, reactor core geometries and associated fuel cycles are possible for SFR.

SFR were originally researched primarily on the basis of the fact that the uranium-235 , which can be fissioned in thermal reactors, contained in naturally occurring uraniumaccounts for only 0.71% of total uranium. The majority of natural uranium is therefore not fissile in thermal reactors. By capturing neutrons, however, this proportion of uranium-238 is converted into heavier nuclides, mainly plutonium. Plutonium, in turn, is a good fissile material in both thermal and fast reactors.

A larger number of neutrons are produced per fission in fast reactors than in thermal reactors. At the same time, the relative probability of these neutrons being absorbed in the resulting fission products or in structural materials decreases. This means that more neutrons can be used in fast reactors to produce plutonium from uranium-238 (or uranium-233 if thorium is used).

SFR with the aim of producing additional fissile material are also known as fast breeder reactors. In such SFR, the reactor core is typically constructed in such a way that fuel elements with a high fissile material content are used in an inner region in which the thermal power of the reactor is generated. This inner region is then surrounded by another region, the so-called breeding blanket, made of fuel elements containing only natural or depleted uranium. The neutrons escaping from the inner reactor region generate new plutonium in this breeding blanket. Reprocessing of the fuel is then required for the later utilisation of this newly produced fissile material. For such a reactor concept, please refer to the description of the BN-800 in Chapter 5.1.

However, as there is currently and for the foreseeable future no recognisable shortage of uranium (see Chapters 2.5.6 and 2.6.3), the reactor core of SFR today is often designed in such a way that the fissile material content either remains constant or even a targeted reduction of actinides from the spent fuel of previous LWR is sought. This requires reprocessing of the fuel from light-water reactors and fuel fabrication with a broader composition of actinides.

In contrast, individual reactor concepts pursue the approach that the fissile material produced in the reactor is directly consumed again. This should make it possible to completely dispense with spent fuel reprocessing, see the description of the travelling-wave reactor (TWR) in Chapter 5.2.

Coolant

The light metal sodium is used as a coolant in SFR. This has various favourable properties as a coolant.

Firstly, it has a very high thermal conductivity of 62 W/m K (at a temperature of 550 °C), which is about a factor of 100 higher than that of water. With a density comparable to water and despite a slightly lower viscosity, the heat transfer from the fuel into the coolant is improved compared to water. This means that even a reactor with a higher power density than LWR can still be cooled.

Sodium melts at 98 °C at normal pressure. This means that it must be kept permanently above this temperature during operation so that it does not solidify. In contrast, its boiling temperature of 883 °C is very high. This means that working temperatures in the range of 500-550 °C are possible during operation without the coolant boiling during normal operation. As a result, the primary cooling circuit can be designed to be unpressurised with only the static pressure of the sodium having to be taken into account.

Sodium is also well compatible with many stainless steels and therefore does not lead to excessive corrosion in the cooling circuits.

The main disadvantage of using sodium is that strong chemical reactions occur when sodium comes into contact with water and, at higher temperatures, with oxygen. Contact with water in the reactor must therefore be avoided. Precautions must also be taken against sodium fires that can occur in the event of leaks in sodium-bearing pipes.

Another disadvantage compared to water is that liquid sodium is not transparent. This means that visual inspections of areas filled with sodium, including the reactor core and the fuel elements in use, are not possible. It is also not possible to visually inspect the positioning of the fuel elements during fuel element changes.

Spectrum (moderator)

SFR are reactors with a fast neutron spectrum. Moderators are therefore not used.

Due to the significantly lower effective cross-sections for the reaction with neutrons in a fast neutron spectrum, the neutron flux in SFR is approx. 500 times higher than in LWR. This means, in particular, that the structural materials used in an SFR are subjected to higher loads.

Pressure and temperature

The temperatures in all sodium cooling circuits must be kept permanently above the melting temperature of sodium. SFR must thus be kept at temperatures above approx. 200 °C during downtime, if necessary with additional heaters. The working temperatures aimed for in SFR are 500-550 °C.

Due to the high boiling temperature of sodium, boiling during normal operation is impossible. In the reactor system, therefore, only the static pressure of the sodium in the range of a few tenths of a megapascal prevails.

Construction materials

Stainless steels can typically be used for the fuel cladding and structural materials in SFR. These still have sufficient strength in the temperature range of 500-550 °C. The reactor vessel itself is typically shielded from the neutron radiation of the core by absorber elements in the outer area of the reactor core. As the cooling circuit itself is not under pressure, the reactor vessel can therefore also typically consist of a stainless steel wall with a thickness of a few centimetres.

4.1.2 Historical developments

The historical development of sodium-cooled fast reactors was summarised in (Oeko-Institut e.V.; ZNF 2015). This presentation is reproduced below essentially unchanged with minor updates. For a more detailed discussion, please refer to (Imel 2021), for example, and specifically to (Guidez 2017) for the experiences from the French plants Phénix and Superphénix.

Ever since nuclear fission was first used for peaceful purposes, concepts for the construction of reactors with a fast neutron spectrum have been developed and realised in prototypes. A major advantage compared to thermal reactors was considered to be the energy yield from the existing uranium reserves, which was about 60 times higher due to the extensive conversion of uranium-238 into fissile plutonium-239 (“breeding”).

The use of this type of reactor thus required the parallel large-scale development of facilities for the reprocessing of spent fuel elements with the aim of separating the plutonium they contain, and of facilities for the production of plutonium-containing (“mixed oxide”) fuel elements for use in fast breeder reactors.

As was the case for light-water reactors, the development of Sodium-cooled Fast Reactors took place in three phases, from experimental reactors (see Table 4-1) through prototype and demonstration reactors (see Table 4-2) to commercial plants (see Table 4-4). For a more detailed description of these reactors, please refer to (IAEA 2006a; 2007a; 2012b; WNA 2021a).

Previous experience from the operation of SFR, for example, is also contained in (NRC 2014).

Table 4-1: Former experimental SFR

Country	Reactor	Start of construction	Output (MWth)	Operation	Remarks
Germany	KNK-II	1975	52	1977-1991	
France	Rapsodie	1962	40	1967-1983	
Great Britain	DFR	1954	60	1959-1977	
India	FBTR	1972	40	since 1985	
Italy	PEC	1974	120		Construction cancelled in 1987
Russia	BR -10	1956	55	1959-2002	Core meltdown accident 1955
	BOR-60	1964	8	since 1958	
USA	EBR-I	1947	1.2	1951-1963	
	EBR-II	1958	62.5	1963-1994	
	Fermi	1956	200	1965-1972	
	FFTF	1970	400	1980-1992	

Source: (Oeko-Institut e.V.; ZNF 2015; Schulenberg 2020)

Table 4-2: Former prototype and demonstration SFR

Country	Reactor	Start of construction	Output (MWth)	Operation	average load factor	Remarks
Germany	SNR-300	1973	762			No operating licence granted
France	Phénix	1968	563	1973-2009	~ 0.50	
	Super-Phénix 1	1976	3420	1985-1996	0.08 [146]	
Great Britain	PFR	1966	650	1974-1983 1984-1994	0.07 0.34	
Japan	MONJU	1985	714	1994-2016		Shut down 1996-2010 after accident
Kazakhstan	BN-350	1964	750	1972-1999	0.85	
USA	CRBRP	1982				Cancelled in 1983 due to increased costs

Source: (Oeko-Institut e.V.; ZNF 2015)

[146] Without an officially ordered two-year shutdown phase to conduct a public hearing procedure.

In addition to the plants that have already been decommissioned, three experimental SFR built in the 1960s and 1970s are still in operation, see Table 4-3.

Table 4-3: Experimental SFR in operation

Country	Reactor	Start of construction	Output (MWth)	Operation	Remarks
India	FBTR	1972	40	since 1985	Pool type
Japan	JOYO	1970	140	1977-2007	Loop type, on standby since 2007
Russia	BOR-60	1964	55	1969	Loop type, partial core meltdown during start-up

Source: (Oeko-Institut e.V.; ZNF 2015; Schulenberg 2020; Heidet 2021)

The main results of operating experience to date are as follows:

a) All of the fast breeder reactors listed use(d) metallic sodium as a coolant.[147] This enables a high power density of the reactor, high coolant temperatures and thus more favourable efficiencies for electricity production than in light-water reactors, but leads to specific accident risks, such as the sodium leaks and fires that occur frequently in some cases.

b) As is to be expected with the development of a new technology, the operating experience was extremely varied. They range from the occurrence of serious incidents and accidents, in some cases with years of downtime, to largely normal operation over decades. For those reactors where the aim was to demonstrate high availability, the average load factors are given where they can be determined. These are calculated as the ratio of actual to theoretically possible power generation, so that they represent a suitable measure of operational reliability.

c) Problems with the sodium cooling circuits are a major cause of the low availability in some cases. As a rule, demonstration and commercial reactors have a primary and secondary circuit with liquid sodium as the coolant. Leaks in the sodium cooling circuits with leakage of the chemically reactive coolant, sometimes combined with sodium fires or – in the case of leaks in the heat exchanger in the secondary circuit – with sodium-water reactions, led to frequent and sometimes prolonged[148] shutdowns in individual reactors.

d) Some of the reactors were operated exclusively or largely with (higher enriched) uranium instead of plutonium. The reasons for this are likely to lie both in the option of using the proven technology of manufacturing uranium fuel elements and in the reactor-physical and safety-related advantages[149] over plutonium fuel.

e) The low load factor of the Superphénix is striking, see Table 4-2. This low availability led to the operator’s decision in 1996 to shut down the reactor for economic reasons (IAEA 2007a).

f) As concepts for the transmutation of long-lived radiotoxic waste isotopes have existed for decades, initial tests on the use of such isotopes in experimental and prototype breeder reactors have been carried out in several countries. These served both to verify the calculated

[147] The only exception: the British DFR used sodium-potassium.
[148] In the case of the Japanese Monju reactor from 1994-2010.
[149] In particular, the higher proportion of delayed neutrons and often more favourable reactivity coefficients should be mentioned here.

conversion rates and to gain experience of the behaviour of actinide-containing fuels when used in reactors. Such transmutation experiments are documented for the EBR-II in the U.S. (Oak Ridge National Laboratory 1982), the PFR in the UK (Oak Ridge National Laboratory 1983), the BOR-60 in Russia (Mayorshin et al. 2002), for Joyo in Japan (IAEA 2012b) and Phénix in France (IAEA 2007a).

4.1.3 Current developments

There are three sodium-cooled fast reactors in operation worldwide today, which are listed as commercial reactors according to (IAEA 2023g), see Table 4-4. These are the two Russian plants BN-600 and BN-800 at the Beloyarsk site and the Chinese CEFR in Tuoli. A CFR-600 plant is also under construction in China. Another prototype reactor, the Indian PFBR, has been under construction at the Madras site since 2004.

Table 4-4: SFR classified as commercial that are under construction or in operation

Country	Reactor	Start of construction	Output (MWth/MWe)	Operation	Note	average load factor
China	CEFR	2000	65/ 20	since 2010	Pool type	2011-2016: 0.0 2017-2021: n/a
	CFR-600	2017	1882/ 642	-	Pool type	-
India	PFBR	2004	1253/ 470	-	Pool type	-
Russia	BN-600	1969	1470/ 560	1980	Pool type	76.4%
	BN-800	2006	2100/ 820	2015	Pool type	65.9%

Source: (IAEA 2023g; Heidet 2021; Schulenberg 2020)

The IAEA ARIS database also lists ten reactor concepts assigned to the technology line of SFR, the 4S, ASTRID, BN-1200, CFR-600, FBR-1 & 2, JSFR, MBIR, PGSFR, PRISM and the TWR-P (IAEA 2023d); for an overview of the developments within the framework of the GIF, see also (Hill 2016).

4.1.3.1 4S

According to the description in (Oeko-Institut e.V.; WIP; PhB 2021, Kap. 6.2.4), the 4S is a simplified design of a sodium-cooled fast reactor of the pool type from the Japanese company Toshiba. The reactor is to have a thermal output of 30 or 135 MW and a corresponding electrical output of 10 or 50 MW.

The reactor core is constructed from metallic fuel made of a uranium-zirconium alloy; a breeding blanket is not required. With an initial enrichment of 17%, a burnup of 34 MWd/kg is to be achieved. The reactor should be able to be operated without loading for 10 to 30 years. The primary cooling circuit is located entirely in the reactor vessel and includes the heat exchangers, electromagnetic pumps, reflectors and a shutdown rod. The movable reflector surrounds the reactor core and compensates for burnup over the life of the plant. An intermediate cooling system transfers the heat to a tertiary cooling circuit. The reactor vessel is surrounded by a protective cover (guard vessel), which ensures core coverage in the event of a sodium leak. The guard vessel is inerted against sodium fires with nitrogen.

According to information in (IAEA 2022a), a preliminary design review for the 4S concept was initiated by Toshiba in 2007. According to this, fourteen technical reports were submitted to the U.S. NRC in 2013. (IAEA 2022a) does not include information on more recent developments.

4.1.3.2 ASTRID

According to (WNA 2021a), the French Commissariat for Atomic Energy and Alternative Energies (CEA) was commissioned by the French government in 2006 to develop an advanced concept for an SFR as part of the GIF collaboration and based on 45 years of experience with sodium-cooled fast reactors. The “Advanced Sodium Technological Reactor for Industrial Demonstration” (ASTRID) was to have an electrical output of 600 MW and serve as a prototype for a series of commercial SFR with an electrical output of 1500 MW, which were to be built from around 2050. Cost estimates from 2010 assumed investment costs of EUR 4.3 billion. (Rodriguez 2018) provides an overview of the ASTRID project.

The fuel to be used would be MOX, comparable to today’s LWR-MOX, but with a fissile material content of 25-35%. The reactor core was designed to achieve a negative void coefficient of reactivity, see Chapter 4.1.5.

With funding from the French government totalling around EUR 650 million for the period from 2010 to 2017, the initial design of the reactor concept was completed by 2015. Work on the concept was continued together with international partners until 2019. In June 2017, the French government decided that the reactor concept should be scaled down to a lower electrical output of 100-200 MW to save costs. In August 2019, the CEA declared that it no longer planned to aim for the construction of ASTRID in the short or medium term (WNA 2021a).

4.1.3.3 BN-600, BN-800 and BN-1200

Compared globally, Russia currently has the most extensive experience with the operation of sodium-cooled (breeder) reactors. Consequently, the intention is to develop commercial reactors (BN-1200) on the basis of the two demonstration reactors BN-600 and BN-800 currently in operation, which are to be used on a large scale from 2040-2050, see also Chapter 5.1.

The BN-600 was commissioned on 26 February 1980 and built on the experience gained from the operation of the BN-350. It is the first SFR built in Russia using the pool design; the predecessor plants were all of the loop design, see Chapter 4.1.1. The reactor has a thermal output of 1470 MW and an electrical output of 600 MW. The plant has three cooling circuits and is operated at a core outlet temperature of 550 °C. Uranium oxide is used as fuel. The average burnup today is 70 MWd/kg of heavy metal, the maximum burnup is 11.1% of the initial heavy metal. This burnup results in a strain of 82 dpa on the fuel cladding. From commissioning until 2017, the reactor was operated with an average availability of just under 75%. Unforeseen operational disruptions led to an average reduction in availability of 1.1%. Until 1994, 27 sodium leaks occurred in the BN-600, 21 of which only led to minor releases, with a maximum of 1000 litres of sodium being released in one event. No further leaks have occurred since 1994. The BN-600 was used in particular to demonstrate the long-term behaviour of components such as pumps and pipelines on an industrial scale. The originally planned service life of the BN-600 of 30 years was reached in 2010. Extensive retrofitting in the period 2005-2010 made it possible to extend the service life to 40 years, and continued operation up to a service life of 60 years is currently being examined (Pakhomov 2018). Most recently, the operating licence was extended in 2020 until 2025 (WNN 2020).

The reprocessing of spent fuel, production of mixed oxide fuel elements and transmutation of minor actinides are planned as part of the future use of sodium-cooled fast reactors in Russia. Since the BN-600 has so far been operated exclusively with higher enriched uranium and Russia had no experience with the industrial production of plutonium-containing mixed oxide fuels until a few years ago, the use of plutonium from Russian nuclear weapons stockpiles (USA; Russia 2010), which was contractually agreed with the U.S., has contributed significantly to the experience gained in the production and use of mixed oxide fuel elements in fast reactors. This is a key purpose of the BN-800. For the BN-800 see Chapter 5.1.

To date, no decision has been made on the construction of the BN-1200 as a successor to the BN-800. With 4 cooling circuits, the BN-1200 differs significantly from its predecessor models with three circuits. Completion of the design was originally scheduled for 2016-2017, but Rosenergoatom postponed the decision to build it in 2015. The construction of a BN-1200 is currently not expected before the mid-2030s, see Chapter 3.3.3.

4.1.3.4 BOR-60/MBIR

The construction of a successor facility, MBIR, is planned at the site of the BOR-60 experimental reactor in Russia, which is now over 50 years old. Research work in the areas of materials, safety, fuels and systems is to be carried out at this facility (GIF 2021a, p. 16). In addition to the MBIR, a radiochemical test facility for carrying out post-irradiation examinations is also to be built at the site. On research in the field of SFR in Russia, see also (Kuzina 2021).

The MBIR is a sodium-cooled fast reactor with a thermal output of 150 MW. It is planned for an operating life of 50 years and is to be operated with MOX fuel. The reactor will enable tests with different coolants (lead, lead-bismuth, gas). As of February 2022, completion is planned for 2027.[150]

4.1.3.5 CEFR

As a research reactor, the China Experimental Fast Reactor (CEFR) is part of the development of large reactors with a fast neutron spectrum in China. Construction of the plant began in 2000.

According to (Oeko-Institut e.V.; WIP; PhB 2021, Anh. 6.2.4.4), various test programmes have been carried out at the CEFR since its commissioning in 2010. Continuous operation of 72 hours was achieved for the first time in 2014. In 2020, the plant was operated continuously for 40 days. The development goal is a commercial plant with an electrical output of 1000-1200 MW.[151]

The IAEA lists a grid feed-in of 0 MWh for the period 2012-2016 and a trial operation for the period between 2017-2020, but without data (IAEA 2023g).

According to (GIF 2021a, p. 10), the reactor reached the planned full output after the planned commissioning test for full load operation in 2020. The planning and construction of the planned successor CFR-600 is progressing as planned.

[150] https://www.world-nuclear-news.org/Articles/Completion-of-MBIR-reactor-brought-forward, last accessed 09.02.2022.

[151] https://www.world-nuclear-news.org/Articles/Chinese-fast-reactor-completes-trial-operating-cyc, last accessed 25.01.2022.

The CEFR was put back into full-load operation in January 2021.[152]

The reactor core, the coolant pumps and the four primary heat exchangers of the CEFR are located in the primary reactor vessel filled with sodium; the heat is transported to a tertiary water circuit via a secondary sodium cooling circuit.

4.1.3.6 CFR-600

The Chinese strategy for introducing fast reactors follows a three-stage plan. Following the construction of the experimental CEFR (see the last chapter), the second step is the construction of the CFR-600 as a demonstration reactor.

The CFR-600 fast reactor XIAPU-1 has been under construction in China since 29.12.2017. The plant is being built by the China National Nuclear Cooperation (CNNC) and will have a net electrical output of 642 MW and a thermal output of 1,882 MW (IAEA 2023g). Construction of a second unit at the site began in December 2020.[153]

The first unit is scheduled to be commissioned in 2023. The Russian fuel manufacturer TVEL has commissioned its own production line for the manufacture of fuel for the CFR-600. The fuel is to consist of MOX and have a target burnup of 100 MWd/kg. At a later date, metallic fuel with a target burnup of 100-120 MWd/kg will also be used. The breeding rate is specified as 1.1. Finally, steam at a temperature of 480 °C is to be provided via two sodium cooling circuits. The design provides for both active and passive safety devices and the planned operating time of the reactor is 40 years.[154]

The CFR-600 is expected to be followed by the construction of a CFR1000 reactor as a commercial plant from December 2028 with a planned commissioning in 2034. This plant will use metallic fuel with a target burnup of 120-150 MWd/kg.[155]

4.1.3.7 Joyo

Japan has also been pursuing the development of sodium-cooled fast reactors for several decades. The Joyo experimental reactor was commissioned for this purpose in 1977. Joyo is an SFR with a loop design whose thermal output was increased in several stages to 140 MW by 2003. Joyo used 100 MOX fuel elements for an operating time of 71,000 hours. The reactor has been shut down since 2003 (WNA 2021a).

According to (GIF 2021a, 13-14), the operator of the Japanese experimental reactor Joyo is planning to recommission it. In October 2018, documents on the planned implementation of new safety requirements following the event in Fukushima were submitted to the Japanese regulator NRA. Furthermore, a new experimental reactor with a capacity of 10 MW is planned at the site of the permanently decommissioned Monju fast reactor, with construction scheduled to begin in 2022.

[152] https://www.world-nuclear-news.org/Articles/Chinese-fast-reactor-begins-high-power-operation, last accessed 25.01.2022.
[153] https://www.world-nuclear-news.org/Articles/China-starts-building-second-CFR-600-fast-reactor, last accessed 25.01.2022.
[154] https://www.world-nuclear-news.org/Articles/TVEL-unit-launches-CFR-600-fuel-manufacturing-site, last accessed 25.01.2022.
[155] https://www.world-nuclear-news.org/Articles/Chinese-fast-reactor-begins-high-power-operation, last accessed 25.01.2022.

4.1.3.8 JSFR

According to (Schulenberg 2020), the Japanese concept of the JSFR is an SFR in a loop design with a thermal output of 3750 MW and an electrical output of 1500 MW. It will have a double-walled reactor vessel with an internal diameter of 10.7 m. The core inlet temperature is 395 °C and the core outlet temperature is 550 °C. The heat is transferred to a secondary sodium cooling circuit via two external primary heat exchangers connected to the reactor vessel via pipework, which also house the coolant pumps on the primary side.

The loop design is intended to contribute to cost savings thanks to its compact construction and also allows the external heat exchangers to be better inspected and, if necessary, repaired thanks to the improved accessibility. An average burnup of 150 MWd/kg of heavy metal is targeted for the fuel elements. The reactor core provides for a breeding blanket in which new fissile material is to be generated in a targeted manner.

The decay power can be transported via natural circulation to an emergency cooling system cooled with passive air cooling. In the event of a core meltdown, a sodium-cooled core catcher is provided in the reactor vessel to prevent the core meltdown from coming into contact with the bottom of the reactor vessel.

4.1.3.9 PFBR and FBR 1&2

India is pursuing the development of sodium-cooled fast breeder reactors with the aim of being able to use its domestic thorium deposits as fuel for its nuclear energy programme. To this end, it has been operating the Fast Breeder Test Reactor (FBTR) with a thermal output of 40 MW since 1985 (WNA 2021a).

The Protoype Fast Breeder Reactor (PFBR) was approved in 2002 as the first SFR prototype reactor and has been under construction since 2004. However, its completion is still delayed. It is intended to achieve an electrical output of 500 MW and be operated with uranium-plutonium mixed oxide fuel with a plutonium concentration of 21% and 27% in two different zones of the reactor core. The plutonium required for this is to be obtained from the reprocessing of spent fuel from the Indian heavy water-moderated pressurised water reactors (PHWR). In a breeding blanket with thorium as breeding material, uranium-233 is to be produced in the PFBR for future use in SFR (WNA 2021a).

Several SFR in a pool design with an electrical output of 600 MW have been announced as successor plants to the PFBR, the first being FBR 1&2. These plants will initially also use oxide fuel, but in future there are plans to switch to metallic fuels, as these can be used to increase the breeding rate for new fissile material. FBR 1&2 are to have a homogeneous reactor core with a breeding blanket and reflector zone. A target burnup of 150 MWd/kg heavy metal is planned (WNA 2021a).

The FBR is to be the design for an SFR to be produced in series. The safety concept of the FBR includes an ultimate shutdown system based on liquid lithium or boron carbide (B_4C), active and passive cooling circuits and a core catcher to handle accident sequences with core destruction (IAEA n.d.b).

4.1.3.10 PGSFR

South Korea has also been pursuing the development of SFR for a number of decades, see also Chapter 3.3.5.

In particular, South Korea is also focussing on the development of electrometallurgical reprocessing technologies to completely separate and reuse transuranic elements produced in nuclear reactors (WNA 2021a). Metallic fuels are also being discussed in particular (Kim et al. 2013).

The Korean Prototype Generation IV sodium-cooled fast reactor (PGSFR) is planned as the next step in the development of SFR in South Korea. Construction of such a plant is scheduled to begin in 2028. This prototype in a pool design is expected to generate a thermal output of 392 MW and an electrical output of 150 MW. The main development goal of this plant is to demonstrate metallic uranium fuels with low enrichment and a 10% zirconium alloy. In future, these fuels are also to be produced using a mixture of transuranic elements from the reprocessing of LWR. The PGSFR is thus intended to demonstrate the potential of SFR for reducing actinide stocks (WNA 2021a).

4.1.3.11 PRISM

According to the description in (Oeko-Institut e.V.; WIP; PhB 2021, Kap. 6.2.4.6), PRISM is the concept of a sodium-cooled fast reactor that has been developed by the US company GE Hitachi Nuclear Energy (GEH) since the 1980s.

The reactor is expected to generate a thermal output of 840 MW and an electrical output of 311 MW. The reactor core of a module consists of 42 fuel elements with a metallic uranium-plutonium-zirconium alloy with a plutonium content of 26%. Furthermore, the reactor core comprises a central blanket with 24 elements, a radial blanket with 33 elements and further reflector and shielding elements. The reactor core also has six control and shutdown elements. The core, four coolant pumps and the primary heat exchangers are located in the sodium-filled primary reactor vessel. The heat is transferred through the primary heat exchangers to a second sodium cooling circuit, also located in the reactor vessel above the reactor core. From there, the secondary cooling circuit transfers the heat to one steam generator per module. The live steam generated in three modules jointly drives a turbine.

4.1.3.12 TWR

The travelling-wave reactor concept (TWR) pursued by the U.S. company TerraPower aims to avoid proliferation risks and make efficient use of natural uranium reserves. At the same time, it aims to minimise the amount of radioactive waste produced and increase reactor safety. This reactor concept is discussed in more detail in Chapter 5.2.

4.1.3.13 VTR

As described in (Oeko-Institut e.V.; WIP; PhB 2021, p. 278), the planned Versatile Test Reactor (VTR) is a scaled-down version of the PRISM reactor concept developed by GE Hitachi with a thermal output of 300 MW. In particular, it is intended to provide fast neutrons for the development of advanced reactor technologies. A consortium consisting of TerraPower and GE Hitachi Nuclear Energy intends to construct the VTR on behalf of the DoE. Battelle Energy Alliance (BEA), the

contracted operator of the Idaho National Laboratory (INL), has entered into negotiations with the alliance of Bechtel National Inc (BNI), TerraPower and GE Hitachi to construct the VTR.

In September 2020, the DoE made an initial decision in principle on the construction of the VTR, allowing it to move into the concrete planning phase. To this end, the DoE has requested a budget of USD 295 million for the 2021 financial year.[156]

In December 2020, the DoE submitted a draft environmental impact assessment for the VTR.[157]

4.1.3.14 Further current developments

The Generation IV International Forum aims to have advanced the R&D work for an advanced sodium-cooled fast reactor with transmutation of minor actinides to such an extent by around 2022 that a demonstration phase of at least 10 years can then follow, during which the approval process, construction and operation of a prototype/demonstration reactor will be carried out together with industry. This development project is therefore a top priority for the GIF. The R&D work is supported by China, EURATOM, France, Japan, Korea, Russia and the U.S., which have signed a corresponding cooperation agreement (“system arrangement”) (GIF 2015).

In addition to the Generation IV International Forum, an industry-led initiative to develop concepts for the sustainable use of nuclear energy, the European Industrial Initiative on Sustainable Nuclear Energy (ESNII), was launched at the instigation of the European Union as part of its SET (European Strategic Energy Technology) plan initiated in 2006 according to (Oeko-Institut e.V.; ZNF 2015). A key goal of this initiative is to achieve commercial availability of fast reactors by 2040 (EU 2015). Supported and funded by the European Union, the aim is to develop the three fast reactor concepts of the Generation IV International Forum to the point where they are ready for use and to demonstrate this with the first corresponding plants. ESNII has chosen the sodium-cooled fast breeder reactor as the reference technology. The timetable envisaged completing the planning, approval process and construction of a prototype for ASTSRID by 2020. Work on ASTRID was discontinued in 2019, see Chapter 4.1.3.2.

The French CEA is pursuing developments in the field of sodium-cooled fast reactors as part of a five-year plan. This includes studies in the field of basic physics, modelling and simulation, in particular the physics of severe accidents, sodium chemistry and risk assessments; an improvement in the fuels and the operating behaviour of structural materials as well as technological developments of some components, in particular monitoring and inspection techniques (GIF 2021a, p. 12).

As reported by (GIF 2021a, p. 14), South Korea has a thermo-hydraulic test facility for the investigation of integral effects in sodium-cooled reactors STELLA-2 in operation.

According to (GIF 2021a, p. 17), the Swiss PSI supports the IAEA working group on fast reactors as part of projects for the American FFTF and the Chinese CEFR. In addition, PSI is coordinating a Horizon 2020 project on safety aspects of sodium-cooled reactors (ESFR-SMART). For findings from ESFR-SMART, see (Guidez 2022), for example.

[156] https://www.world-nuclear-news.org/Articles/US-test-reactor-programme-moves-ahead, last accessed 09.02.2022.

[157] https://www.world-nuclear-news.org/Articles/US-DOE-issues-draft-EIS-for-new-test-reactor, last accessed 09.02.2022.

4.1.4 Technical development status

The following presentation is based on an updated version of (Oeko-Institut e.V. 2017, Kap. 5).

Sodium-cooled Fast Reactors are the most technically advanced of all GIF technology lines. According to (GIF 2014), research and development efforts are therefore largely focused on optimising economical and reliable operation. The focus is on current developments in the Russian programme, see also Chapter 3.3.3.

According to an assessment by the Generation IV International Forum in 2014, development was already in the “performance phase” in 2012, which was essentially due to the commissioning of the BN-800 in Russia and the completion of the ASTRID design in France. Expectations from 2002 that the “performance phase” would start as early as 2006 were therefore pushed back by around half a decade.

The fundamental technical viability of an SFR with uranium fuel can be considered to have been demonstrated by the many years of operation of the BN-600.

(Schulenberg 2020) names the optimisation of reactor cores with regard to a negative or only slightly positive void effect of the coolant (improvement of safety) and the improvement of the economic efficiency of SFR as current research areas for SFR.

(GIF 2017b) defines requirements with a view to further developing the safety level of SFR, see Chapter 4.1.5. There are also research questions on aspects of the thermal hydraulics of SFR and verification with validated calculation programs (Gerschenfeld 2020). (Mikityuk 2019) presents current and historical experimental plants for SFR development. Reference is made to (NRC 2021c) and (NEA 2021b) on questions of fuel qualification. A compilation of material technology issues also for SFR can be found in (Maloy 2018).

For further research on various aspects of SFR, suitable facilities for carrying out experiments and for the development of materials and technologies such as VTR or MBIR are particularly necessary. The requirements needed for SFR development for this were analysed in (NEA 2011a), for example.

In addition to the actual reactor technology, the commercial availability of uranium-plutonium mixed oxide fuels (MOX) and, if necessary, fuels with an actinide mixture is essential for the further introduction of sodium-cooled fast reactors. Table 4-5 provides an overview of commercial plants for the production of MOX fuel for LWR and SFR. Only in France, India and Russia are commercially classified plants for MOX production currently in operation (IAEA 2023c). Other laboratory and pilot plants for MOX production also exist in Italy, Japan, Canada and Russia. Plants for the commercial or large-scale production of fuels with an actinide mixture do not yet exist. (Hayes 2017; Lee 2021) provides an overview of the development of metallic fuels for SFR.

In Japan, a pilot plant for the production of up to 10 tonnes of MOX fuel per year for SFR is in operation in Tokai-Mura, in which fuel with a plutonium content of up to 50% can be produced (Renn 2014). A further plant is currently under construction. Construction was originally scheduled to begin in 2007; according to the operator Japan Nuclear Fuel Limited, the plant has been under construction since October 2010. The plant is scheduled to be commissioned in 2022 (IAEA 2023c). The actual future of this plant is unclear due to the current discussion about nuclear policy in Japan following the accident in Fukushima.

The penultimate commercial MOX production plant to go into operation worldwide was the British Sellafield MOX Plant (SMP) for the production of MOX fuel for LWR. The first application documents

for the construction of this plant were submitted by industry in 1992. Technical commissioning took place in 2002. The originally planned production capacity was around 120 tonnes of fuel per year. In the period from 2002 to 2009, however, only 6.3 tonnes of MOX fuel were actually produced due to technical problems, and only 12 MOX fuel elements were produced and sold during this period. The operating costs of this plant for the period from 2002 to 2009 were stated at GBP 626 million, the construction costs of the plant at GBP 498 million and the planned dismantling costs at GBP 139.4 million (Nuclear Engineering International (NEI) 2009). On 3 August 2011, the British Nuclear Decommissioning Authority (NDA), the operator of the plant, announced the decommissioning at the earliest possible date (Nuclear Engineering and Technology 2011).

The Russian pilot plant RIAR in Dimitrovgrad was modernised to produce MOX fuel for thermal reactors as well as for fast reactors. The first 56 MOX fuel elements for use in the Russian BN-800 were also produced there. Another plant (MCC) for the production of MOX fuel for the Russian BN-800 as part of the planned dismantling of military plutonium stocks is in operation at the Zheleznogorsk site (Krasnoyarsk) (IAEA 2023c). In December 2019, the first 18 MOX fuel elements were installed in the BN-800 and operated normally throughout 2020. A complete reactor core of 169 MOX fuel elements was manufactured and tested at the Mining and Chemical Combine in Zheleznogorsk in July 2020 (GIF 2021a, p. 15). It was deployed in 2022 (WNN 2022).

In future, fuel for the BN-600 is to be produced at the Elemash site in a new production line, which was initially set up for the production of fuel for the Chinese CFR-600.[158]

Finally, further development of reprocessing technologies and corresponding facilities is also required. In the European Union, various hydrochemical partitioning processes have been developed for a programme for the transmutation of plutonium and minor actinides. They represent further developments and additions to the PUREX process. Their suitability has been demonstrated on a laboratory scale. In contrast, pyrometallurgical separation processes are also being investigated internationally. Compared to hydrochemical processes, these offer the major advantage of greater radiological stability of the chemicals used, which is essential for the reprocessing of fuels with a high content of minor actinides, especially when short interim storage times are required. Due to the high temperatures together with the corrosive properties of the chemicals used, special requirements are placed on the structural materials used. Experience with pyrometallurgical separation processes is significantly less than for hydrochemical processes, and the separation factors demonstrated are even lower compared to hydrochemical processes. For a detailed description of the international status of reprocessing technologies, please refer to (Oeko-Institut e.V.; ZNF 2015).

[158] https://www.world-nuclear-news.org/Articles/TVEL-unit-launches-CFR-600-fuel-manufacturing-site, last accessed 25.01.2022.

Table 4-5: Plants for commercial MOX production worldwide

Country	Plant	Fuel type	Status	Design capacity	Start of operation	End of operation
Belgium	Belgonucleaire P0 plant	LWR	Decom-missioned	40 t SM/a	1973	2006
	FBFC International – MOX	LWR	Decom-missioned	100 t SM/a	1997	
Germany	Siemens Fuel Fabrication Plant Hanau, (MOX new)	SFR / LWR	Cancelled	120 t SM/a		
	Siemens Fuel Fabrication Plant Hanau, (MOX old)	SFR / LWR	Decom-missioned	30 t SM/a	1969	1992
France	Areva NC Melox	LWR	In operation	195 t SM/a	1995	
	AREVA NC MOX	SFR / LWR	Decom-missioned	40 t SM/a	1961	2003
Great Britain	NDA MOX	SFR	Decom-missioned	6 t SM/a	1970	1988
	NDA Sellafield MOX Plant (SMP)	LWR	Decom-missioned	120 t SM/a	2006	2012
India	Advanced Fuel Fabrication Facility (AFFF)	LWR / SFR	In operation	20 t SM/a	1993	
Japan	Rokkasho MOX Fuel Fabrication Plant	SFR / LWR	Under con-struction	130 t SM/a	Planned for 2022 (initially 2007)	
	Takeyama		Decom-missioned	10 t SM/a	1972	1973
Russia	Mining and Chemical Combine (MCC)	SFR	In operation	60 t SM/a	2015	
	Mayak MOX plant	SFR	In operation	5 t SM/a	1993	
	MOX Fuel Fabrication Plant (REMIX)		In operation	[159]		
USA	MOX Fuel Fabrication Facility (MFFF)	LWR	Cancelled		2016	
	Nuclear Fuel Services		Decom-missioned		1965	1972

Source: (IAEA 2023c), SM/a: Heavy metal per year

[159] The plant produced MOX pellets for six fuel elements in 2021, see https://infcis.iaea.org/NFCIS/FacilityDetails/1091?Country=All&Status=All&Type=16&Scale=Commercial&SText=, last accessed on 23.01.2023

Conclusion: Technical development status

The operation of sodium-cooled fast reactors, the production of uranium- and plutonium-containing MOX fuel and the reprocessing of spent fuel reprocessing have been demonstrated on an industrial scale and are therefore classified as “operational”.

For commercialisation and diffusion, further developments in the areas of safety and economic efficiency are still required.

The production of plutonium- and actinide-containing nitride or metallic fuels and the reprocessing of spent SFR fuels is currently still classified as “development”.

4.1.5 Safety

The following presentation is essentially based on an updated version of (Oeko-Institut e.V. 2017, Kap. 5).

Liquid sodium is much thinner (less viscous) than water, has a high thermal conductivity and reacts strongly exothermically in contact with water and begins to burn at high temperatures in contact with oxygen.

Sodium is an opaque coolant, so inspections and maintenance of the reactor are more complex and loading and unloading processes cannot be visually checked (Baqué 2021).

As the sodium of the primary circuit is radioactive due to activation, there is an intermediate sodium cooling circuit between the steam generator and the primary circuit to prevent radioactive contamination in the plant in the event of sodium fires; the intermediate cooling circuit is also an important cost factor for the operation of SFR (IPFM 2010).

With regard to reactivity control in SFR, it should first be noted that fissions in a fast neutron spectrum produce more neutrons per fission. However, the delayed neutrons released from the resulting fission products remain more or less unchanged. This means that the relative proportion of delayed neutrons is significantly lower, which leads to a faster reaction of the reactor core to changes in reactivity. This effect is even more pronounced for fuels with a high plutonium content than for uranium fuel.

The change in reactivity over the burnup is lower for SFR than for LWR because spent fissile material is continuously regenerated in the fuel and the resulting fission products absorb significantly fewer neutrons in the fast neutron spectrum of an SFR than in LWR.

The Doppler effect of the fuel is reduced compared to LWR due to the harder neutron spectrum. When sodium boils, the density and thus the neutron absorption in the sodium decreases, which increases the reactivity in the reactor core (positive void effect), which is a significant difference to LWR. In principle, there is a large gap between the operating temperature and the boiling temperature of sodium. Nevertheless, boiling cannot be ruled out under certain boundary conditions, for example if cooling channels become blocked.

However, as the density of the sodium decreases, the mean free path of the neutrons in the reactor core also increases. This increases the neutron leakage from the reactor core. A decreasing sodium density therefore leads on the one hand to less neutron absorption in the sodium itself (positive reactivity effect) and on the other hand to higher neutron losses due to leakage (negative reactivity effect). If the reactor core is therefore designed for a relatively high neutron leakage, the positive

void effect can be reduced or even completely compensated for. However, since the neutrons in SFR are basically intended to be used for the generation of fissile material, heterogeneously designed reactor cores are being discussed for this purpose. In this case, fuel elements for breeding, which also act as neutron absorbers, are arranged between the fuel elements used for power generation. If the sodium density decreases, more neutrons escape the fuel element area and are captured in the breeder elements. However, as the breeder elements also act as absorbers during normal operation, such a reactor core requires fuel elements with a higher fissile material content than a homogeneous reactor core (Schulenberg 2020). Also, isotopically pure plutonium-239 is produced in breeder elements, which can contribute to proliferation risks, see Chapter 0.

Melting of the fuel element cladding tubes can also increase reactivity. In the event of an accident with prompt supercriticality due to a control failure, reactivity only collapses when the core structure is destroyed or altered, for example by a partial core meltdown. However, structural failure can also lead to more critical configurations that are destroyed in a small nuclear explosion. Whether such an explosion can release enough energy to destroy the reactor containment is a cause for concern and is being debated, e.g. in India (IPFM 2010)

The following is stated in (Oeko-Institut e.V.; ZNF 2015):

> *“The introduction of a sodium-filled area above the core, conceived for the first time for the Russian BN-800, can represent a significant safety gain. This significantly reduces the void reactivity coefficient, as the evaporation of the sodium leads to an increase in leakage-related neutron losses in the core when the power is increased (IAEA 2012b). However, design calculations for the European sodium-cooled fast breeder reactor – a concept for a commercial fast breeder reactor promoted by EURATOM – show that even in an optimised core with an upper sodium plenum, the void reactivity coefficient remains positive in large areas of the reactor core, so that a damping reaction only sets in when the boiling of the coolant reaches the plenum (Sun et al. 2011). Coupled neutron-physical/thermo-hydraulic dynamic simulations are therefore required to investigate whether this measure is sufficient to reliably prevent significant core damage.”*

In its System Safety Assessment for SFR,(GIF 2017b) the GIF also comes to the conclusion that the reactor core is not in its most critical configuration for SFR and that an increase in reactivity can therefore occur in the event of boiling of the coolant, cladding damage or fuel compression as a result of accident sequences with core destruction. The GIF therefore requires SFR to have two diverse, independent active shutdown systems, an additional passive shutdown system and design measures to prevent a rapid supercritical plant state. Further requirements with regard to criticality safety are also discussed in (NEA 2021a).

For basic requirements on the design of future SFR, we also refer to (GIF 2017a; 2016b). Options for passive shutdown systems in fast reactors are presented in (GIF 2021c), for example.

Sodium as a coolant has the advantage that the entire primary circuit operates at low pressure due to the high boiling point of sodium (883 °C), in contrast to LWR, where the pressure in the primary circuit can be up to 15 MPa. As a result, large loss-of-coolant accidents are less likely than in LWR, in which the cooling water is forced out of the primary circuit at high pressure in the event of a pipe break and is initially not available for cooling the hot fuel rods.

In principle, residual heat removal from SFR can also be achieved via passive cooling circuits in natural circulation. This requires a corresponding height difference between the heat exchangers of

the intermediate cooling circuit. The atmosphere can be used as the ultimate heat sink. In principle, this is also possible with LWR.

With regard to the necessity of residual heat removal, (GIF 2017b) demands the practical exclusion of a permanent interruption of residual heat removal. In particular, the practical exclusion of a loss of sodium from the primary cooling circuit, a sufficient sodium throughput through the reactor core even under the conditions of a natural circulation and the practical exclusion of a simultaneous failure of all residual heat removal systems, in particular due to jointly caused faults, must be ensured.

Sodium reacts exothermically with water and ignites at high temperatures on contact with oxygen. If there are leaks in sodium-bearing pipework in SFR, the escaping sodium can lead to sodium fires in the system. Furthermore, leaks on heating pipes in heat exchangers between a sodium cooling circuit and a water-steam cooling circuit can lead to sodium-water reactions. Suitable safety precautions must be taken in SFR for both scenarios.

In the event of leaks in pipework, sodium-bearing systems are typically drained and the affected areas flooded with an inert gas. The escape of sodium can either be prevented by double-walled pipework. Alternatively, escaping sodium can be collected using stainless steel trays. If the sodium cools down sufficiently on contact with a collecting tray, the sodium-air reaction comes to a standstill.

In the steam generators, the liquid sodium is only separated from the water-bearing pipes by a thin metal wall; any contact can lead to pipework damage and water-sodium fires. If steam generator pipes leak, the affected cooling circuit must typically be drained. To do this, the sodium contained can be drained into a sump tank and the cooling circuit flooded with an inert gas.

Sodium fires occurred mainly during operation of the BN-350 and BN-600. The BN-600 had 27 sodium leaks between 1980 and 1997, 14 of which led to sodium fires. There were also sodium fires in the Monju reactor in Japan and in the French reactors Rapsodie, Phénix and Superphénix. Sodium leaks also occurred in the British reactors DFR and Prototype Fast Reactor PFR (IPFM 2010).

Many current concepts for SFR have a core catcher, similar to today’s LWR, which is designed to catch and stabilise the molten core mass in the reactor vessel without the hot core meltdown coming into contact with the wall of the reactor vessel. For heterogeneously constructed reactor cores, however, this can lead to a separation of the fuel elements operating at high power and therefore melting first and the colder fuel elements operating at low power. However, this would only concentrate fuel with a high fissile material content on the core catcher, which in turn could lead to reactivity problems and even a possible prompt critical chain reaction (Schulenberg 2020).

(GIF 2017b) also sees challenges with regard to the safe confinement of radioactivity, particularly in accident sequences involving core damage. (GIF 2017b) recognises the need to address the risk of energy release when molten fuel comes into contact with the coolant and to ensure a robust design for a core catcher and the associated systems for the necessary heat removal.

Conclusion: Safety

Sodium as a coolant has the advantage that during normal operation it only leads to a small degree of corrosion of structural materials or cladding tubes. On the other hand, sodium is an opaque coolant, so inspections and maintenance of the reactor are more complex and loading and unloading processes cannot be visually checked.

Reactivity control is fundamentally more demanding for SFR than for LWR. Especially when plutonium fuels are used, fast reactors react very sensitively; this places special demands on the

shutdown system to prevent an uncontrollable power increase. Fast reactors have a partially positive void reactivity coefficient, especially for plutonium fuel. If the coolant becomes too hot and vaporises, the reactivity can increase abruptly due to the onset of density reduction. This is offset by the high boiling point of sodium.

In contrast to the loss of the moderator in light-water reactors, in which the water is also used to transport away the decay heat, the core in a fast reactor does not become subcritical if there is no sodium (moderator) in the core. Even if the fuel rods melt, the reactivity can initially increase further.

Overall, the higher reactivity control requirements for SFR represent an intrinsic disadvantage compared to today’s LWR.

With regard to the required heat removal, redundant and diverse systems are provided for SFR safety concepts. There are also possibilities for purely passive and indefinite heat removal through natural circulation and heat removal to the atmosphere. The extent to which such safety concepts are implemented in specific reactor concepts cannot be assessed at the technology line level.

With regard to possible releases in the event of incidents and accidents, the developers point to good retention of essential radionuclides (iodine, caesium) in the coolant, although there is still a need for further research here too (Kauric 2019). Containment of possible releases by a containment system comparable to present-day LWR is typically envisioned. The extent to which such a safety concept can be implemented in specific reactor concepts cannot be assessed at the technology line level.

Unlike light-water reactors, the primary cooling circuit is not under high pressure. Large loss-of-coolant accidents are therefore less likely than in light-water reactors.

One safety disadvantage of SFR compared to LWR is that the sodium coolant reacts exothermically on contact with water and ignites on contact with oxygen at high temperatures. Sodium fires have occurred repeatedly during operation of fast reactors, leading to operational failures. The opaque, exothermically reacting sodium also makes maintenance of the reactor and repair of damage complex.

With regard to the event spectrum to be considered for SFR, there are therefore fundamental advantages to the fact that the cooling circuits are not under high pressure. This advantage is offset by the disadvantage of possible sodium fires in the event of leaks or water overflow into the intermediate cooling circuit in the event of damage to the steam generator heating tubes. Overall, this does not result in any relevant intrinsic advantage or disadvantage of an SFR compared to current LWR.

In summary, significantly higher requirements must be placed on SFR compared to LWR with regard to reactivity control; in other safety-relevant areas, there are no clear advantages or disadvantages compared to LWR at the level of the technology line.

4.1.6 Fuel supply and waste disposal

The following presentation is essentially based on an updated version of (Oeko-Institut e.V. 2017, Kap. 5).

Fast breeder reactors have been pursued since the beginning of the development of nuclear technology with the aim of providing a solution to the expected problem of scarce uranium resources in the event of a massive expansion of nuclear energy (Ohshima and Kubo 2016).

In the initial phase of their development, fast reactors were primarily conceived as breeder reactors. The idea was to convert uranium-238 into plutonium-239 in the reactors by neutron capture and then separate it. This is possible in fast reactors, as the nuclear fission triggered by faster neutrons releases more neutrons than with slow neutrons. If the reactor core is surrounded by blankets (breeding zones) in which, for example, uranium-238 is irradiated with excess neutrons, new plutonium is produced in such blankets, almost exclusively plutonium-239 due to the neutron spectrum (Kütt et al. 2014). If more plutonium is produced in the blanket than is simultaneously consumed in the reactor core, the reactor operates as a breeder reactor and produces plutonium.

For the initial configuration, either uranium with a higher enrichment in the range of 20% or plutonium must be obtained from the spent fuel elements of a light-water reactor (or from future SFR). The plutonium could then in turn be used in uranium-plutonium mixed oxide fuels (MOX), thus stretching the uranium used. For most of its development, the fast breeder was therefore associated with large-scale reprocessing of the fuel in order to separate the plutonium it contained and produce MOX fuel in fuel element factories (plutonium economy).

To date, however, there are no signs of a shortage of uranium resources, see Chapter 2.5.6. Rather, it has been shown that the production of MOX fuel has repeatedly encountered technical difficulties and that the reprocessing of fuel to separate the plutonium represents a major cost factor, so that the fuel cycle required for breeder reactors would only be economically viable at extremely high uranium prices.

Fast reactors are also a core component of strategies for waste treatment with partitioning and transmutation. In fast reactors, minor actinides that were previously separated from the high-level waste could be split with the help of the fast neutron spectrum. For the advantages and disadvantages of P&T, please refer to (Oeko-Institut e.V. 2023; 2017; Renn 2014). Multi-recycling, as originally envisaged in a plutonium economy or envisaged in P&T scenarios, is economically unattractive today and has not yet been developed industrially. The need for a geological repository cannot be avoided by any variant of a closed fuel cycle or P&T strategy, see (Oeko-Institut e.V.; ZNF 2015) and Chapter 8.

Like all reactors, SFR also generate different waste streams. Most SFR provide for the utilisation of MOX fuels. However, concepts utilising uranium fuel are also being discussed, such as the travelling wave reactor from Terra Power (see Chapter 5.2).

Today, plutonium is obtained from the reprocessing of LWR fuels. During reprocessing, uranium and plutonium are separated from the spent LWR fuel, the plutonium is reused and processed into MOX fuel, the uranium could in principle also be reused, but this is not practised on a large scale today.[160]

[160] This is due to increased concentrations of the uranium isotopes 232U and 236U in the separated uranium, in whose decay series strong gamma emitters occur and make handling more difficult, as well as a necessary compensation of the presence of 234U and 236U by an increased enrichment required compared to natural uranium.

The residual waste is vitrified and the glass canisters are also to be disposed of. There is also waste from the operation and later from the dismantling of the reprocessing plants.

Other waste streams to be expected during SFR operation are the coolant and decommissioning and decontamination waste streams as well as operational waste streams.

4.1.6.1 Fuel

Possible fuels for fast reactors are mainly ceramic MOX fuels (including U-Pu, U-Pu-MA, U-MA) with stainless steel as fuel rod cladding, as used in fast reactors, especially in Russia and France. Very high burnups can be achieved in MOX fuels. MOX fuel for use in fast reactors differs from that used in light-water reactors primarily due to the high proportion of fissile material. In larger SFR, the proportion of fissile material in the fuel is typically over 20%, while smaller plants require fissile material proportions in the region of 30%, compared to only approx. 5-7% in light-water reactors. The increased fissile material content in MOX fuels for fast reactors results in higher requirements for the manufacturing processes (Kumar 2015) and other requirements for disposal. The most experience has been gained with MOX (for use as transmutation fuel using MA only with individual pellets); France, China and Japan want to use MOX fuel in SNR (Rodriguez and Serre 2021).

Fuel development for fast reactors is still taking place worldwide. Metallic fuels, such as U-Zr, were developed and used primarily in the U.S. (J. Kittel et al. 1993) and are still being researched in the U.S., e.g. for the TerraPower sodium reactor or the planned Versatile Test Reactor (NASEM 2023b). However, experience with metallic plutonium-containing fuels (U-Pu-Zr, U-Pu-MA-Zr) is limited. Furthermore, solutions for high burnup and for use in transmutation still need to be developed (Rodriguez and Serre 2021). The fuels of the U.S. development lines are also designed as pure uranium fuels for reasons of proliferation resistance, with enrichments of up to 19.75% (HALEU). South Korea is also planning to use metallic fuel.

U-Mo, U-Cr was used in Great Britain. In addition, carbide and nitride fuels ((U,Pu)C, (U,Pu)N) have been developed, particularly in Russia, which plans to use nitride fuels in the BN-1200. In France, magnesium oxide fuels (a mixture of magnesium oxide and actinide oxide, also known as ceramic-ceramic or CERCER) have also been investigated (Raj 2015; Somers 2015).

The different fuel types differ in important (reactor) physical parameters such as their thermal conductivity and melting point. In addition, in the past, the conversion ratio that could possibly be achieved with these fuels, i.e. the additional production rate of fissile plutonium, and the achievement of higher burnups due to a higher fissile material density than in oxide fuels, was an important selection criterion for pursuing such fuel developments. The conversion ratio of oxide fuels is lower than that of other fuels (IAEA 2003; Somers 2015).

The different fuels and cladding tube concepts alone mean that there are differences with regard to disposal. Added to this are the more complex fuel cycles usually associated with fast reactors with the reuse of plutonium or the burnup of actinides. This is discussed in more detail in Chapter 8.

When plutonium-containing MOX fuel elements are used once in SFR, they typically have a significantly higher activity and heat output after use in the reactor due to higher proportions of heat-producing actinides and higher burnup than spent LWR fuel elements. The resulting higher temperature has an impact on ageing effects during interim storage, and it is expected that the ageing effects of high burnup fuel elements will be more severe and impact the integrity of the inventory (Oeko-Institut e.V. 2023). The fuel element behaviour after extended interim storage is

also important for the assessment of transports. Due to the higher heat output, the space requirement for the storage of SFR fuel elements is also higher compared to spent fuel elements from LWR.

A special feature of nitride fuels is the formation of nitrogen gas and the formation of radioactive carbon-14 by neutron capture in nitrogen ($^{14}N(n,p)^{14}C$) during irradiation. According to (IAEA 2019c), a very large amount of carbon-14 can be produced. Due to its half-life of 5370 years, carbon-14 is problematic in high concentrations when it comes to disposal.

To avoid the formation of carbon-14 and parasitic neutron capture, the fuel should normally be enriched with nitrogen-15, with corresponding cost disadvantages. Nitride fuels have a significantly higher thermal conductivity than oxide fuels. With roughly the same melting point, they therefore have a safety advantage with regard to failure due to melting. They also have a 30-40% higher density. This has a favourable effect on the in-situ brute properties in uranium, but also on their transmutation properties. It also improves the retention properties for fission gases (Ekberg et al. 2018). Disadvantages of the fuel are the instability of the fuel in contact with water, a problem for the use of nitride fuels in water-cooled systems, and the difficulty of production (Ekberg et al. 2018).

Nitride fuels are also being researched in Russia as part of the development of lead-cooled fast reactors using a combination of hydrometallurgical and pyrochemical processes. One advantage of the Russian pyrochemical process is that it is planned to work with unenriched nitrogen, as the carbon-14 does not pass into the gas phase as CO_2 during the reprocessing of the spent fuel, but remains in elemental form as a layer on the surface of the melt and can therefore be easily and compactly removed (Ekberg et al. 2018). A larger amount of carbon-14 is also produced in carbide fuels, which has a similar effect on disposal as with VHTR (see Chapter 4.6.6).

In some fuel elements for fast reactors, such as the U.S. metallic fuels, a sodium bond is created between the fuel (e.g. pellets) and the fuel cladding in order to achieve better heat transport. Experience in the U.S. shows that these metallic fuels can exhibit a number of degradation phenomena that should be better researched (NRC 2019b). Also, due to the exothermic reaction of water with sodium, the fuel elements are not suitable for direct disposal and suitable conditioning for the fuels must be found, e.g. by pyroprocessing and separating the sodium from the waste stream (NASEM 2023b).

Treating the metallic spent fuel elements with pyrochemical processes produces converted salt waste, ceramic waste from the immobilisation of the salt and metallic waste without sodium. The majority of the waste stream is irradiated fuel cladding, which is produced as metallic waste. Although the ceramic and metallic waste forms are in principle suitable for direct disposal (NRC 2019a), the authors also point out the special challenges posed by pyrochemically treated metallic fuels (NRC 2019a): Saline wastes are chloride-containing and soluble, which can lead to pitting and stress corrosion cracking of stainless steel and nickel alloys in a vessel. Immobilisation in ceramics effectively traps radionuclides and dilutes the concentration of plutonium in the fuel, but the salt content in the ceramics can lead to dissolution processes, as ceramic materials containing salt are hygroscopic. Added to this is the influence of radiation embrittlement. Metallic waste could be exposed to oxidation, pitting and galvanic corrosion and the released chloride can increase localised corrosion in the repository environment. Moisture can form uranium oxide and hydrogen, which can form uranium hydrides in the metallic fuel, which in turn are pyrophoric and can affect the waste package (NRC 2019a).

As many SFR are planned in connection with the reprocessing of plutonium, scenarios for the utilisation of SFR with reprocessing will generate waste from the operation and later from the dismantling of the reprocessing facilities. Most of this waste, such as treated sludge or contaminated exchange parts, must also be disposed of. After their use in the reactor, the MOX fuel elements produced with the separated plutonium and burnt down after reactor use have not yet been reprocessed either (see Chapter **8.5.1**).

4.1.6.2 Other waste streams

As with all fast reactors, SFR shields and/or reflectors and other components are used that contain activation products and other contaminants that must be disposed of as low and intermediate-level radioactive waste (see Chapter 4.4.7.3).

In the case of liquid-metal-cooled fast reactors such as SFR and LFR, the influence of the coolant on disposal is a significant difference to LWR. The cooling water of an LWR is itself chemically harmless, with the exception of the contaminants it contains. It can be drained, decanted, stored and cleaned without any major chemical hazards. The coolant sodium used in SFR, on the other hand, is considered a hazardous chemical material in itself and must be handled with care as long as it is in metallic form. The handling of sodium requires special technology and expertise, as well as additional process steps and techniques, such as the use of nitrogen or argon cover gas, preheating of pipework and vessels, smoke and fire detectors, sodium leak detectors, special inspections of components, etc. (Raj 2015). The sodium itself contains a number of activation products such as ^{22}Na, other activation products from the steel structures such as ^{60}Co and ^{54}Mn, as well as contamination from fission products and actinides, but especially ^{3}H, and caesium-137 (NASEM 2023b; IAEA 2007c).

SFR concepts contain large quantities of sodium coolant in the primary circuit. (IAEA 2007c) The order of magnitude is up to 900 tonnes of sodium for research reactors, e.g. 900 metric tonnes for a very large research reactor such as the FFTF (fast flux test facility); 900 to 1700 tonnes for prototype power plants and 1600 to 5500 tonnes for commercial power plants. The sodium must be treated accordingly, for which a number of processes have been developed.

Several hundred kilograms of sodium remain in the reactor, adhering to and in pipework and reactor components, either as a thin film on the surface or in bulges and cavities that could not be drained. Due to the high chemical reactivity of sodium, the residual sodium must be removed. Discharged fuel elements must also be cleaned of residual sodium before being stored in air or water. In addition, sodium-contaminated waste is produced during maintenance and repair. For all waste, contact with air must be avoided during storage due to the potential risk of fire (IAEA 2007c).

(IAEA 2007c) provides a comprehensive overview of SFR dismantling techniques and the generation of sodium waste. Short-lived fission products such as caesium-137 and tritium can be separated in a preliminary stage. Two main processes are used to treat the sodium waste. In the NOAH process, a dosing pump is used to pump small quantities of liquid sodium into a large quantity of aqueous sodium hydroxide to neutralise it. The chemical reaction is moderate and continuously controllable, producing sodium hydroxide and hydrogen gas contaminated with tritium. The Argonne process uses an etching process to convert sodium into crystalline sodium hydroxide monohydrate as the end product, which can be packaged in drums. The process is based on the sodium reaction with aqueous sodium hydroxide solution (IAEA 2007c).

The safe conditioning of metallic sodium into a stable chemical product for storage and disposal also requires specific processes. Several processes are being considered, including dry solidification, cementation, conversion to dry carbonates and conversion to dry sodium salt. Although many of these technologies are technically mature, only a few are currently ready for commercial use (NASEM 2023b).

Conclusion: Fuel supply and waste disposal

MOX fuels would also have to be shipped to a repository after use in the reactor. Alternatively, multi-recycling, as originally envisaged in a plutonium economy or contemplated in P&T scenarios, would have to be developed industrially. The need for a geological repository cannot be avoided by any variant of a closed fuel cycle or P&T strategy.

Most SFR provide for the utilisation of MOX fuels. Problems arise during production due to the higher fissile material content in the fresh fuel elements compared to the production of MOX for LWR. Very high burnups can be achieved in MOX fuel for SFR, which increases the heat generation in the spent fuel and distinguishes them from use in light-water reactors. The increased heat generation has an influence on the short-term decay and interim storage periods and can also increase the space requirements in the repository. In terms of other criteria for safe confinement in a geological repository, however, MOX differs only marginally from LWR fuels. However, there are higher requirements for the handling of MOX in production, transport and interim storage as well as disposal, e.g. with regard to criticality safety and radiation protection. With regard to the proportion of fission products in spent fuel elements from SFR, there is no significant difference in terms of disposal compared to the high-level waste of LWR.

In addition to the ceramic oxide fuels used on an industrial scale, carbide and nitride fuels as well as magnesium oxide fuels are also being developed for use in fast reactors, each with their own cladding tube concepts. This results in differences with regard to disposal. A special feature of nitride and carbide fuels is the formation of radioactive carbon (^{14}C) during irradiation, a long-lived mobile activation product with an impact on long term safety during disposal. Spent carbide and nitride fuels can contain very high amounts of carbon-14, depending on how they are manufactured.

A significant difference between SFR and LWR is the influence of the coolant on disposal. SFR concepts contain large quantities of sodium coolant in the primary circuit, which must be cleaned and then conditioned and disposed of as intermediate-level radioactive waste. Coolant residues in the reactor are also problematic when components are replaced and during dismantling. The residues must be removed due to their high chemical reactivity and generate additional sodium-contaminated waste.

As many SFR are planned in connection with the reprocessing of plutonium, scenarios for the utilisation of SFR with reprocessing will generate waste from the operation and later from the dismantling of the reprocessing facilities.

4.1.7 Proliferation risks

The following presentation is essentially based on an updated version of (Oeko-Institut e.V. 2017, Kap. 5).

All reactors produce plutonium in fuel from uranium. In the case of SFR, however, it is typically planned to operate the reactor with fuel containing plutonium, which requires separated plutonium and thus reprocessing technologies. A basic analysis of proliferation aspects of SFR with a focus on risks of the closed fuel cycle is provided in (GIF 2021c). One reactor concept for SFR that explicitly dispenses with reprocessing is the TWR, which is discussed in more detail in Chapter 5.2.

The fuel elements for fast reactors contain significantly more fissile material (typically approx. 20% plutonium or uranium-235) than the fuel elements for light-water reactors. This means that the production and transport of fresh fuel for SFR already involves higher proliferation risks than the production and transport of fresh uranium fuel for today’s LWR.

An SFR can also be used as a fast breeder reactor at any time by exposing uranium elements (blankets) in or around the reactor core to the neutron flux. Plutonium bred this way is particularly suitable for the construction of nuclear weapons (weapons-grade plutonium), i.e. it has a particularly high proportion of the isotope plutonium-239. The use of plutonium for SFR production is intentional, as it is intended to enable entry into a closed fuel cycle by separating the plutonium produced in the SFR and using it for the production of new fuel elements. Separated plutonium is easier to access than plutonium that is still in spent highly radioactive fuel elements.

India’s first “peaceful nuclear explosion” in 1974 was carried out with plutonium that had been separated for the Indian breeder reactor programme. In France, the Phénix breeder reactor was used to produce weapons-grade plutonium for the Force de Frappe in the blankets (IPFM 2010). The infrastructure of a fast breeder reactor with the associated reprocessing technologies can therefore potentially be used for a military programme at any time. A non-nuclear weapon state that is a member of the Non-Proliferation Treaty would have to leave the treaty and would be exposed to corresponding international consequences if such military use were to become known.

To address the proliferation problem, there are repeated proposals to make the nuclear materials used more proliferation-resistant. For example, the simultaneous separation of other actinides (neptunium, americium) with the plutonium or the separation of the plutonium directly at the reactor site to reduce the risks during transport are being discussed. Such measures certainly improve proliferation resistance in some scenarios (theft of the material by third parties), but state access to weapons-grade fissile material is possible at any time, so that ultimately only the IAEA’s controls to monitor purely civilian use prevent diversion. However, in the case of reprocessing plants in particular, the effort required to achieve a sufficiently accurate material balance is extremely high if an incorrect balance of a few kilograms of plutonium, which is sufficient to build a weapon, is to be detected. The error in material balancing in reprocessing plants today is around 1%.

In principle, fast reactors can also be used to dispose of plutonium. The U.S. and Russia had concluded an agreement to dispose of 34 tonnes of weapons plutonium. Russia uses the BN-800 for this purpose. Plutonium in the form of MOX fuel elements is inserted into the reactors and some of the plutonium is fissioned, or the isotope vector of the plutonium is changed so that it is less suitable for nuclear weapons. The mode of use of the BN-800 for plutonium reduction essentially consists of dispensing with the simultaneous use of breeding elements (blankets) in the reactor jacket or at other positions. In (Kütt et al. 2014), however, it is shown that with a simple

reconfiguration of the reactor core to breeding mode, by using breeding elements in the reactor shell, up to 162 kg of weapons-grade plutonium could be bred per year. The radiation barrier in such blankets would be significantly lower than in spent fuel elements.

A state that operates a fast reactor with appropriate reprocessing technologies possesses all the components for the production of weapons-grade plutonium for the construction of a nuclear weapon. Some of the plutonium will be available for processing in separated form.

Conclusion: Proliferation risks

With regard to proliferation, an aspect particularly relevant to SFR is that weapons-grade plutonium is produced in the uranium breeder blankets of fast reactors. SFR are designed to produce additional amounts of fissile material. Together with the envisioned separation of plutonium for reuse in MOX fuel, this makes SFR much more proliferation sensitive than LWR with an open fuel cycle.

The infrastructure of an SFR with the associated reprocessing technologies can also potentially be used for a military programme. Reprocessing also greatly increases the cost of IAEA safeguards (security measures).

4.1.8 Costs

The following presentation is essentially based on an updated version of (Oeko-Institut e.V. 2017, Kap. 5).

Between 1974 and 2007, around USD 50 billion was spent on research and development for breeder reactors in the OECD countries, USD 15 billion in the U.S., USD 12 billion in Japan, USD 8 billion in the UK, USD 6 billion in Germany and USD 5 billion in Italy. France only reported spending of around USD 1 billion, obviously an incomplete figure, as the construction of the Superphénix alone cost around USD 14 billion. Russia and India also had large programmes to develop breeder reactors, with Russia (Soviet Union) alone spending an estimated USD 12 billion (IPFM 2010).

(IPFM 2010) comes to the conclusion that when looking at the individual national programmes, it becomes clear that breeder reactors are only economically competitive if uranium prices were to rise dramatically, which is unlikely from today’s perspective. Construction costs are more than 25% higher than for light-water reactors. The electricity price per kilowatt hour for the demonstration reactors was about twice as high as for commercial light-water reactors.

A uranium price of less than 130 USD/kg only contributes about 5% to the levelised cost of electricity (LCOE) of today’s light-water reactors. The use of reprocessing technologies and MOX fuel production is not commercially competitive (Holdren et al. 2003).

The use of separated uranium and plutonium from reprocessing is problematic. The recovered uranium contains additional uranium isotopes produced during reactor operation with unfavourable physical properties that make it expensive to reuse as nuclear fuel. This means that a large proportion of the recovered uranium becomes unusable waste. Only the plutonium remains for use in MOX fuel. This is not economically attractive either, as MOX fuel rods are significantly more expensive than those made from natural uranium. Due to the radiation problem and safeguards requirements, the production of MOX fuel is much more expensive. Plutonium is more radioactive than uranium. Extensive automation and shielding of the production process is therefore necessary in order to reduce the impact on employees in MOX production plants. Furthermore, plutonium has a low critical mass. Therefore, only very small quantities of plutonium can be used to safely exclude

unwanted criticality during the manufacturing process. Finally, the handling of plutonium involves considerable expenditure on safeguards, as the diversion of plutonium for military purposes must be ruled out (Oeko-Institut e.V.; ZNF 2015).

A MOX fuel element factory planned in the U.S. for the production of MOX fuel for light-water reactors from 34 tonnes of weapons plutonium to remove the plutonium from the U.S. nuclear weapons programme was ultimately abandoned due to massive cost increases. The original cost estimate of the responsible National Nuclear Security Agency (NNSA) in 2002 for the construction of the plant was USD 1 billion with a planned start of construction in 2004. In 2002, the U.S. Congress also demanded that MOX production begin in 2008 and that the 34 tonnes of plutonium earmarked for disposal at that time be implemented by the end of 2018. The cost estimate had already risen to USD 4.8 billion by the time construction of the plant began in 2007. In 2014, the construction costs of the plant were stated by the responsible consortium of companies at USD 7.78 billion. The total life cycle costs of the plants, which include construction, operation, waste management and dismantling, were estimated by the U.S. Department of Energy at USD 30 billion. For 2015, the responsible U.S. Department of Energy therefore decided to put the plant on “cold standby” until further notice, thereby halting the construction of the plant until a fundamental decision is made on the continuation of the programme (CRS 2014).

A spokesman for Rosenergoatom, Andey Timonov, states in (Nuclear Engineering International (NEI) 2016) that the operating experience of the BN-800 is decisive for the economic feasibility of future fast reactors, as fast reactors perform worse in comparison with commercial Russian-designed light-water reactors of the WWER (water-water-energy reactor) type. Rosenergoatom is hoping for other functions in addition to electricity generation that will make the reactor economically attractive.

(Schulenberg 2020) notes that the need for a secondary sodium cooling circuit increases the investment costs for SFR compared to LWR today. Furthermore, the handling of the spent fuel elements, which have to be stored in sodium for a longer period of time, leads to higher costs.

According to (IPFM 2010), fast reactors would also have to achieve high utilisation rates of at least 80% for power generation like current light-water reactors to be able to compete with them. However, much of the operating experience to date shows that the utilisation of the power plants has not been satisfactory. A significant part of the problem is maintaining and repairing the parts of the reactor that are immersed in sodium (IPFM 2010). Sodium is optically opaque and reacts with atmospheric oxygen at higher temperatures. It is further argued in (IPFM 2010) that, during a repair, the fuel must therefore be removed, the sodium drained and the entire system carefully flushed to rid it of sodium residues without causing fires or explosions. Corresponding preparations can take months or years. This is a big difference compared to light-water reactors, where the reactor lid can be removed and the entire reactor vessel flooded with water. Repairs can be carried out using periscopes and video cameras, while the water shields the radiation from the fuel elements and steel components. With SFR, on the other hand, long interruptions to operation are necessary: the history of the Superphénix, for example, is characterised by long-lasting breaks in operation, as is the case with Japan’s Monju, the British Dounreay and Prototype Fast Reactor and the American Enrico Fermi 1. In the Russian BN-600, higher utilisation rates were achieved because the operator was willing to keep the reactor in operation despite multiple sodium fires (IPFM 2010).

The World Nuclear Association summarises that it has been clear since the 1980s that SFR are not economically competitive with LWR as long as the use of plutonium instead of uranium as fuel does not prove to be economically viable (WNA 2021a):

> *“…significant technical and materials problems were encountered, and also geological exploration showed by the 1970s that uranium scarcity would not be a concern for some time. Due to both factors, by the 1980s it was clear that [SFR] would not be commercially competitive with existing light water reactors for some time.*
>
> *Today there has been progress on the technical front, but the economics of [SFR] still depends on the value of the plutonium fuel which is bred and used, relative to the cost of fresh uranium.”*

Conclusion: Costs

The investment costs of SFR are typically estimated to be higher than those of today’s LWR. At the same time, the availability of SFR achieved so far is lower than that of LWR.

with regard to operating costs, the costs of reprocessing and the higher costs of MOX fuel production compared to pure uranium fuel also result in economic disadvantages for SFR compared to current LWR.

An economic advantage for SFR was expected in the past if a shortage of uranium resulted in very high prices for fissile materials, making the use of plutonium as an alternative fissile material attractive. However, this would require extremely high uranium prices. From today’s perspective, these are not to be expected over the next decades.

Operating experience with SFR to date also points to a high investment risk, as even simple event sequences can trigger longer-term plant shutdowns or even premature decommissioning.

Overall, therefore, SFR can be assumed to be at an intrinsic disadvantage compared with LWR in terms of economic efficiency.

4.2 Lead-cooled fast reactors (LFR)

Lead-cooled fast reactors (LFR) belong to the group of liquid-cooled fast reactors alongside sodium-cooled fast reactors (SFR). As with SFR, operation with a closed fuel cycle is planned for LFR, with the intended advantage compared to LWR of utilising plutonium and MOX fuels and thus achieving lower natural uranium consumption. Likewise, LFR are to be used for transmutation of minor actinides. Like SFR, LFR are operated at low pressure, which can be associated with corresponding safety advantages. Compared to SFR, the use of lead or a lead-bismuth alloy has the advantage that the coolant is chemically non-reactive. The disadvantage is the very high specific mass of the coolant.

In addition to the main role of LFR for power generation, the use of process heat is also considered as an application due to the operating temperatures between 400 and 620 °C, with even higher temperatures are to be achieved in the future (GIF 2020a). The safety of the plants should also be particularly high and the possibility of radioactive releases low, so that plants could also be operated close to smaller towns to provide neighbourhood heating (GIF 2020a).

A disadvantage of the LFR compared to the SFR is that much less operating experience is available and no major prototype of an LFR has yet been operated outside of the Soviet programme to use lead-cooled ship reactors. The technological development of LFR is far behind that of SFR.

4.2.1 System description

Due to the high mass of the coolant, the basic reactor concept is usually a “pool” design (basin principle) so as to avoid problems with seismic stability and statics. The primary circuit is contained within the reactor vessel itself, including the steam generators and the pumps to pump the coolant (lead or lead-bismuth alloy) through the core and the steam generators. In some systems, however, the steam generators are also separated from the primary reactor vessel and supplied with the coolant using, for example, supply pipes set in concrete. The advantage of the “pool” design is lower costs, but there are risks such as clogging of the lead cooling system or steam entering the reactor in the event of a steam generator heating tube leak. Separate steam generators have advantages here; the disadvantage is the more complex structure of the reactor and thus higher requirements for seismic stability and prevention of solidification of the coolant.

An intermediate cooling circuit between the liquid metal-cooled primary circuit and the water-steam circuit is not provided for in the LFR concept. According to (IRSN 2015), however, it is doubtful whether the omission of an intermediate cooling circuit is licenseable due to the transient accident sequences that could result from this and also due to the inaccessibility of the steam generators for maintenance and repairs. In any case, this must be examined more closely depending on the specific reactor concept.

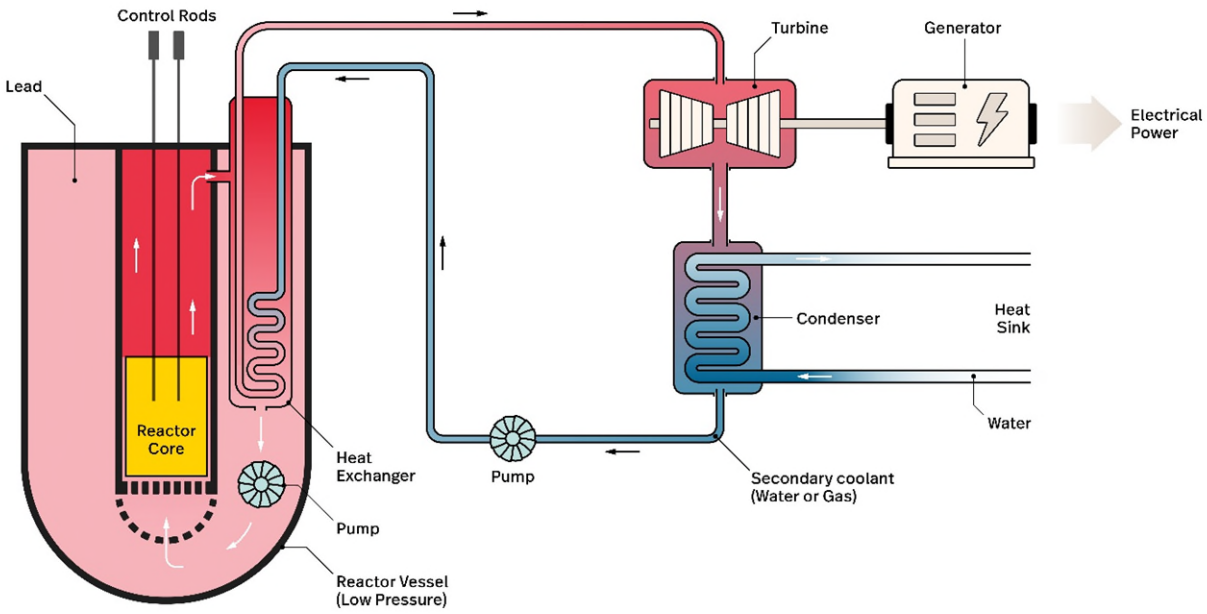
Reactor system

The structure of an LFR in a “pool” design consists of a large reactor vessel (see Figure 4-2). The reactor core with the fuel elements is surrounded by its own cylindrical metal hull. This allows a more stable core geometry and comparatively easy handling of the fuel elements compared to free-standing fuel elements in the reactor pool. The disadvantage of this solution is that the flow of coolant from below into the core can be blocked. Reactor concepts with several coolant accesses from below into the core surrounded by the steel cylinder are intended to prevent a complete blockage of the core in the event of an accident (IRSN 2015).

The spacing of the fuel elements in the core is based on a compromise between neutron velocity and the need to limit the velocity of the coolant flow in the core. If the velocity is too high, the heavy coolant leads to erosion of the fuel rod cladding tubes and structural materials, this effect is intensified by corrosion. The distance between the fuel elements is therefore relatively large and leads to a low overall power density in the core (approx. 100 MW/m3), which makes the LFR less sensitive to local cooling effects than the SFR (IRSN 2015). The low moderation of the coolant has a favourable effect, so that the core does not have to be built very compactly.

Typically, a containment vessel is provided in the LFR to prevent coolant leaks. It consists of a solid concrete shell with an inner steel shell or, due to the high mass of the coolant, a concreted reactor pit with a steel shell. The solid reactor cover typically contains accesses for the steam generators, the primary circuit pump, a system for residual heat removal and other auxiliary systems (IRSN 2015). The reactor vessel is not completely filled with lead-containing coolant, but contains a layer of inert gas, e.g. argon. The inert gas layer plays a crucial role in controlling the oxygen content of the coolant, which is regulated by adding hydrogen or steam to the inert gas (Schulenberg 2020).

Figure 4-2: Conceptual diagram of a lead-cooled fast reactor (LFR)



Source: Own illustration

Fuel

MOX fuel is the typical fuel of an LFR. When used for transmutation, MOX fuel elements with minor actinides can be used. The use of nitride and carbide fuels is also being discussed. Carbide fuels in particular could be used at lower temperatures due to their higher density, which would reduce the formation of oxides in the coolant in the event of cladding tube failure. This could also reduce the loss of reactivity during a cycle and thus the reactivity effectiveness of the control rods. The result would be a reduced risk of reactivity ingress in the event of unintentional extension of the control rods (IRSN 2015).

Fuel cycle

Like the SFR, the LFR is intended for use in a closed fuel cycle. However, most LFR concepts do not provide for use as a breeder reactor. Ideally, after an initial loading with plutonium, the LFR should reach a state of equilibrium in which the fuel composition at the beginning of a cycle is the same as at the end of a cycle and only natural uranium or depleted uranium is added to the fuel after reprocessing. However, such operation is not specific to the LFR and is also the goal of development for some SFR and GFR. The LFR is also frequently discussed for use in transmutation (Alemberti 2021).

Coolant / spectrum (moderator)

Either pure lead or an alloy or eutectic of lead-bismuth in phase equilibrium (55.5% lead and 44.5% bismuth) is intended as the coolant for the LFR. The main advantage of lead or lead-bismuth is that it does not produce a strongly exothermic chemical reaction with air or water; oxidation is very slow. This property is a particular advantage over the sodium-cooled SFR. This property has also led to many reactor concepts providing for the installation of the steam generator directly in the reactor vessel (IRSN 2015; Schulenberg 2020).

Another advantage of lead is the favourable neutron balance due to the low moderation effect of lead and low absorption cross-sections. In principle, this also allows operation as a breeder reactor. The core does not have to be built very compactly, which enables larger distances and thus allows lower flow velocities and also counteracts erosion by the coolant. This also enables low pressure losses in the core and a low power density (approx. 100 MW/m^3). In principle, this in turn allows residual heat removal by natural convection and the use of other passive safety systems. The high density of the coolant is also conducive to natural convection.

The viscosity of lead is relatively low, but higher than that of sodium, while its thermal conductivity is lower than that of sodium. However, despite the significantly poorer specific heat capacity, lead is a better overall coolant than sodium due to its high density at the same flow rate (Schulenberg 2020). Due to the high density and the high volume of coolant, the grace period for loss of coolant is very long and the reactor reacts thermally inert. At low temperatures, the coolant also solidifies, which can limit a possible loss of coolant in the event of a leak. The high density also means that most components experience a strong buoyancy in the coolant due to their lower density. This leads to difficulties when longer objects are immersed in the coolant. The maximum velocity of the coolant is also limited to prevent erosion (Alemberti 2021).

One advantage of lead-bismuth over lead is the lower melting point of 123 °C (lead-bismuth, boiling point 1670 °C) compared to 327 °C (lead, boiling point 1749 °C). This means that the entire reactor does not have to be kept permanently at approx. 400 °C in order to prevent solidification. The disadvantage of lead-bismuth is the production of highly radiating polonium-210 in the coolant.

Several kilograms of polonium can be produced in a reactor with high output, the residual heat of which must be taken into account in the design (Schulenberg 2020). Bismuth is also very expensive. One problem with the lead-bismuth eutectic is that after rapid solidification of the liquid metal due to a change in the crystal lattice, the volume then slowly expands beyond the volume in the liquid phase. This means that solidification must be prevented in any case in order to protect the reactor vessel and the internals (IRSN 2015; Alemberti 2021).

The high boiling point of lead and lead-bismuth has a favourable effect on possible transients due to reactivity input from coolant boiling. Coolant boiling is highly unlikely, even in the event of a station blackout. Structural materials in the core would also fail before the coolant reaches boiling point. Another advantage is the high solubility and retention of volatile fission and activation products such as caesium, iodine and polonium in the coolant, which reduces the source term for releases outside the primary circuit (Alemberti 2021).

Temperature

Typical operating temperatures are between 400 and 620 °C. Due to the very low melting point of the lead-bismuth eutectic, the reactor can be operated at relatively low temperatures (270 °C), thus minimising the risk of corrosion.

Pressure

Like the SFR, the LFR can be operated at low pressure, and the vapour pressure of the lead-containing coolant is also extremely low at normal operating temperatures (Alemberti 2021).

Construction materials

The biggest challenge of using lead and lead-bismuth is corrosion and erosion of the structural materials and fuel rod cladding. The higher the temperatures, the greater the corrosion problems. From a temperature of 450 °C, or 500 °C for pure lead, iron and nickel increasingly dissolve directly in the lead. It is therefore not possible to use typical corrosion-resistant nickel steels; instead, ferritic steels made of iron with a low proportion of chromium are used as structural materials (Schulenberg 2020).

4.2.2 Historical developments

The development of reactors with lead or lead-bismuth as a coolant goes back to a development programme in the U.S., where a research programme was carried out from 1942 onwards, but which was abandoned as early as 1950 due to problems caused by corrosion and contamination of the coolant. It was not until the 2000s that research was resumed with the SSTAR system (see Chapter 4.2.3) (Alemberti 2021; IAEA 2000).

The only other significant research into lead-cooled systems took place in the Soviet Union. In 1953, the Soviet Union began building a prototype LFR with a connected ship propulsion system. The prototype 27/VT was built on land and put into operation in 1959 and led to the construction of ship propulsion systems for submarines in the 1960s, which were in operation until the 1990s. Another land-based prototype was put into operation in 1978 at the A.P. Aleksandrov Scientific Research Technological Institute (NITI) in Sosnovy Bor. The main operating experiences from the use of the reactors were summarised in a report for the International Atomic Energy Agency (IAEA 2000).

The first accident occurred in 1968 with the first prototype, which went into operation on a Soviet submarine in 1963. Due to the lack of a coolant purification system, large quantities of lead oxides in the reactor were not filtered out. The lead oxides probably formed after a leak of air into the cover gas of the primary system during a repair. In addition, lubricating oil from the pump seals entered the primary circuit. The impurities subsequently clogged the core, resulting in localised overheating and a drop in reactivity and reactor performance due to the negative temperature coefficient. Due to a lack of instructions in the operating manual, the full power of the reactor was restored by continuously pulling the control rods and the reactivity reserve of 12 control rods, sufficient for 4000 hours of normal operation, was introduced into the reactor within 30 minutes. The resulting localised overheating caused some of the fuel rods to melt and the slag was flushed out of the core with the coolant (IAEA 2000). Warnings about excessive radiation and the evacuation of the crew from the submarine section were ignored and 9 sailors died from the high radiation dose. The reactor was removed and the submarine was filled with bitumen and furfural and sunk near Navaja Semlja in the Kara Sea in 1982 (IAEA 2000; Schulenberg 2020; Rawool-Sullivan et al. 2002).

The accident showed the importance of appropriate filtration systems for the coolant. The oxygen content in the coolant must be measured and controlled by injecting hydrogen (in a safe mixture with helium). An overpressure in the cover gas must be maintained during repairs. Oil seals should be avoided in favour of other seals such as water and gas seals.

In the early 1970s, the Soviet Union commissioned a new class of submarines with lead-cooled reactor propulsion systems, the Lira class (NATO designation Alfa), of which 7 were built (Rawool-Sullivan et al. 2002). The main part of the Soviet Union’s submarine fleet was equipped with LWR due to safety problems with LFR and advances in light water technology. The biggest disadvantage of LFR was that the reactors had to be kept at temperature, which was only possible with special systems for supplying steam through external pipes, which were only available in the home port. Changing fuel was technically too difficult as the coolant was in danger of freezing and the reactors were therefore never supplied with fresh fuel, nor could maintenance and repair take place. In fact, the LFR ship reactors were left in continuous operation from the 1980s until the end of their service life (1990, except for one submarine that was in operation until 1996), as the steam supply kept failing.

In 1972, damage was discovered on a pipe of the primary circuit on an LFR reactor due to surface corrosion. This was caused by increased humidity in the reactor chamber, which in turn was due to a lack of tightness in the steam generators. This resulted in a leak of radioactive coolant.

Another accident in 1982 involving an LFR submarine reactor was caused by corrosion on the steam generator heating tubes, which was due to incorrect cleaning of the feed water in an ion exchanger with copper, so that the copper entered the secondary circuit and led to corrosion. This allowed steam from the secondary circuit to enter the primary circuit, where it was collected in the emergency condenser provided for this case after separation from the coolant and repeatedly drained by the operating team. However, the draining stopped for unclear reasons and the heat-transferring walls of the condenser were flooded with water, causing the vapour extraction to stop, which in turn led to an increase in pressure in the gas system of the primary circuit. Although the primary circuit is designed to withstand the entire overpressure of the secondary circuit, a faulty open valve caused a manometer to fail and 150 litres of radioactive steam to escape into the crew compartment, exposing 10 sailors to 10% of the permitted dose of polonium-210. The entire reactor was subsequently replaced (Rawool-Sullivan et al. 2002).

According to (Schulenberg 2020), a total of about 80 reactor years of experience with lead-bismuth cooled reactors have been gained in Russia. The experience of the USSR in the military sector is relevant to the current discussion of LFR. Even though three of the eight submarines were decommissioned prematurely due to serious accidents, the IAEA’s overall assessment of the operating experience is positive (IAEA 2007a). However, experimental plants and experiments on LFR technology remain a rarity worldwide. These also tended to be operated at low temperatures. To date, no experiments have been carried out at temperatures above 650 °C, as the corresponding materials are not available (Schulenberg 2020).

4.2.3 Current developments

In the 1990s, LFR attracted renewed interest, mainly due to the discussion about accelerator-driven systems, see Chapter 4.7. More recently, interest has increased due to their inclusion as one of the technology lines of the GIF. There are now several initiatives worldwide to develop an LFR.

The IAEA ARIS database lists ten reactor concepts assigned to the LFR technology line: ALFRED, BREST-OD-300, CLEAR-I, ELECTRA, ELFR, G4M, LFR-AS-200, MYRRHA, PEACER, SEALER, SVBR-100 and the W-LFR (IAEA 2023d).

The three GIF reference systems are the “European Lead Fast Reactor” (ELFR) with an electrical output of 600 MW, the Russian BREST-OD-300 system with an electrical output of 300 MW (see Chapter 5.3) and a small modular system with an electrical output of between 10 and 100 MW, for which the SSTAR formerly developed in the U.S. was used. This means that all power sizes are addressed in GIF, with the idea that synergies are created between the development strands (GIF 2021a).

Among the so-called small modular reactors (SMR), the SSTAR system is worth mentioning. Although it is not being actively pursued, it is nevertheless the reference system for a low-power LFR in the framework of the GIF. The SSTAR should have an electrical output of 20 MW with a thermal output of 45 MW and be transportable in a shipping container. The reactor was to be passively cooled by natural convection, both in operation and at standstill, with a very long service life of 15-30 years (Alemberti 2021). In the U.S., Gen4 Energy Inc, formerly Hyperion Power Generation Inc, is also pursuing a G4M SMR concept with a thermal output of 70 MW and 25 MW of electrical output with active cooling and a service life of 10 years (IAEA 2023d).

With CLEAR-M (China Lead-based Mini-Reactor), China is also pursuing an SMR project with 35 MW thermal and 10 MW electrical output through the Institute of Nuclear Energy Safety Technology (INEST), the Chinese Academy of Sciences (CAS) and Frontier Development of Sciences (FDS) with a core lifetime of 10-20 years and operating temperatures of 375 °C core inlet and 495 °C core outlet temperature. Low-enriched uranium oxide fuel is to be used and the reactor is to be cooled with passive natural circulation cooling.

Other plans include the SEALER from LeadCold (140 MW thermal and 55 MW electrical output) in Sweden and the ELECTRA project, a 0.5 MW training reactor that was planned at the Royal Institute of Technology in Sweden around 2012, but has so far only been presented as a concept study. In China, two companies are proposing a concept for an LFR, the CLFR and BLESS. Also worth mentioning are a reactor concept in South Korea (PASCAR) and in Japan (PBWFR, 100 MW electrical power) (Alemberti 2021).

Table 4-6: Key reactor concepts for the LFR technology line

Reactor concept	Country	Output	Coolant
CLFR	China	100 MWe and 300 MWe	Lead-bismuth / lead
CLEAR-M	China	35 MWth / 14 MWe	Lead-bismuth
BLESS	China	300 MWth / 100 MWe	Lead
ALFRED	EU	300 MWe / 125 MWth	Lead
ELFR	EU	1500 MWth / 630 MWe	Lead
DFR	Canada	3000 MWth / 1500 MWe	Lead
LFR-AS-200	Luxembourg	480 MWth / 280 MWe	Lead
BREST-OD-300	Russia	700 MWth / 300 MWe	Lead
SVBR-100	Russia	280 MWth / 100 MWe	Lead-bismuth
ELECTRA	Sweden	0.5 MW	Lead
SEALER	Sweden	140 MWth / 55 MWe	Lead
PEACER	South Korea	850 MWth / 350 MWe	Lead-bismuth
SSTAR	USA	20 MWe	Lead-bismuth
G4M	USA	70 MWth / 20 MWe	Lead-bismuth
W-LFR	USA	950 MWth / 460 MWe	Lead

Source: (IAEA 2023d)

4.2.3.1 Overview of reactor concepts in the technology line

Table 4-6 provides an overview of several plant concepts.

4.2.3.2 CLFR / BLESS

Two LFR concepts are being developed in China. The reactors of the CLFR series by the China Nuclear Power Technology Research Institute Co. (CNPRI) for the China General Nuclear Power Corporation (CGN). As well as the development of the BLESS (Breeding Lead-based Economical and SafeSystem) by the State Power Investment Corporation Research Institute (SPICRI) and the Research Institute of the State Power Investment Corporation (SPIC) in China.

The CLFR series includes the CLFR-100 and CLFR-300, which were developed to demonstrate the scalability of LFR. The CLFR-100 LFR with 100 MW of electrical power and lead-bismuth as coolant is planned as an SMR and is scheduled to go into operation by 2030. The CLFR-300 is a lead-cooled fast reactor with an electrical output of 300 MW, which is intended to demonstrate its economic efficiency compared to LWR. The reactor concept of the series is planned as a pool design with all primary components in the reactor vessel, which will be removable for maintenance and servicing. The safety systems will be designed to be completely passive (including no batteries or electronic devices).

The BLESS reactor concept is to be a lead-cooled demonstration reactor with a capacity of 300 MW (thermal) and 100 MW (electrical). The concept is also planned as a pool design with all primary components in the reactor vessel, eight steam generators and four pumps as well as passive, redundant, diverse and independent systems for residual heat removal. The core in an inner vessel

is to be equipped with UO_2 and MOX fuel elements. The core inlet and outlet temperatures are 340 and 490 °C. According to the developers, the main aim of BLESS is to cover the energy requirements of remote residential and industrial areas without a national grid connection.

Furthermore, two research projects in China have been funded by the Chinese Ministry of Science and Technology (MOST), which are aimed at small lead-cooled ADS concepts for a flexible energy supply (GIF 2021a, p. 11) (see Chapter 4.7.3.4).

4.2.3.3 ALFRED

The reactor concept of the Advanced LFR European Demonstrator (ALFRED) with 300 MWth was designed in the 7th Framework Programme of the European Commission in the LEADER project from 2010 to 2013. The reactor in a basin design is to be cooled with pure lead. The simple design with a downpipe with a steam generator at the upper end and a riser with the core at the lower end is intended to enable efficient natural convection of the coolant. A torroidal ring with 8 outlet pipes is fitted above the core. Connected to this are 8 vertical units, each with a pump and a vapour generator, which surround the core. The hot lead from the core is channelled upwards by a pump in the vertical pipe before it enters the steam generator through an inlet opening at the upper end and finally returns to the reactor pool at the lower end. The primary circuit pumps are located in the hot region of the reactor between the core outlet and the steam generator. The reactor vessel is surrounded by a safety containment and inside is the inner vessel, which is open at the bottom and contains the core, the torroidal ring, and a fuel element store above the ring. The inner vessel thus separates the hot areas of the reactor from the cold areas (Alemberti 2021).

The reactor concept has not yet been finalised. In the so-called Fostering ALfred CONstruction (FALCON) consortium, an attempt was made to support the relatively rapid construction of the reactor and to identify limitations and problems of older concepts. The consortium consists of Ansaldo Energia (Italy), the National Agency for New Technologies, Energy and Sustainable Economic Development (ENEA, Italy) and the Institute for Nuclear Research (RATEN ICN, Romania). The FALCON consortium renewed its agreement in 2018. The main modifications to ALFRED are an elimination of thermal coolant layers through a new coolant circulation with a hot and cold basin, the use of single-walled bayonet tubes in the steam generator, only three steam generators and three pumps and three additional immersion coolers connected to the isolation condensers as a fully diversified system for residual heat removal that is equipped with a system to prevent the coolant from solidifying. The FALCON consortium also envisages a progressive increase in reactor temperatures with a constant coolant mass flow after commissioning, and a successive qualification of new materials and the corrosion protection strategy. In three phases, there will be a transition from already demonstrated and known technology and materials at low temperatures (100 MWth, 390 °C/430 °C core inlet/outlet temperature) to a second phase with medium temperatures (200 MWth, 400 °C/480 °C) in order to qualify the fuel elements at higher temperatures. In a third phase with high temperatures (300 MWth, 400 °C/520 °C), main components such as the steam generators will also be tested for improved performance under operating conditions that would be typical for an industrial use of LFR. Each phase is to receive another partial licence. (GIF 2020a; Alemberti 2021)

ALFRED is to be built in Romania. In 2019, the Romanian government carried out a project at the RATEN ICN research centre to prepare activities for the development of ALFRED infrastructure in Romania (EUR 2.5 million). Subsequently, projects for supporting infrastructure for test facilities were tendered, the ATHENA project, an experimental facility in basin construction for lead and

ChemLab, a laboratory for the investigation of lead chemistry with a budget of EUR 20 million (GIF 2020a).

4.2.3.4 ELFR

In Europe, the European Commission’s ELSY (European Lead-cooled SYstem) project presented the first studies on the design of a commercial reactor for the European Lead-cooled System (ELFR) with a capacity of 1500 MW (thermal) and 600 MW (electrical) in 2006. The development was continued in the 7th Framework Programme in the LEADER (Lead-cooled European Advanced DEmonstration Reactor) project in 2010. The ALFRED reactor concept is also being researched as part of the LEADER project (see Chapter 4.2.3.2). (IRSN 2015)

The ELFR reactor concept is a further development of ELSY. It is a reactor with a pool design; the reactor vessel was designed to be as compact as possible to minimise the coolant inventory for seismic reasons and at the same time provide space for the reactor internals. These consist of 8 steam generators, 8 primary circuit pumps and 8 immersion coolers for residual heat removal. The area with the hot coolant is enclosed in an inner vessel, which is connected to the primary circuit pumps in the centre of the steam generators via pipes. The coolant inlet temperature is 400 °C and the coolant outlet temperature is 480 °C with a pressure loss in the core of 0.9 bar. The heat source in the core and the heat sinks in the steam generators as well as the simple design of the flow path of the coolant are intended to ensure that natural circulation can occur in the core in the event of an emergency shutdown. The residual heat removal is to be ensured by two independent, redundant, diverse and passive systems.

The aim is to achieve a fuel cycle in which an equilibrium is established and the plutonium content in the fuel always remains the same for proliferation reasons.

4.2.3.5 DFR

In Germany, a plant concept for a dual-fluid molten salt reactor (dual-fluid reactor - DFR) (Huke et al. 2015; He 2016; Wang 2017) was presented, which could also be used for transmutation (Institut für Festkörper-Kernphysik 2019). It is a lead-cooled reactor with a molten chlorine salt with uranium-plutonium as fuel, with the possibility of coupling the reactor directly with pyrochemical processes for reprocessing. The design is to be operated at an average temperature of 1000 °C. The company moved to Canada in 2021 (Dual Fluid 2021).

4.2.3.6 LFR-AS-200

The LFR-AS-200 (Liquid Fast Reactor - Amphora Shaped) with 200 MW electrical power is being developed by Hydromine Nuclear Energy S.à.r.l., Luxembourg. It is a fast reactor in a basin design with lead cooling, in which all primary components are housed in the reactor vessel. Cooling is provided by six steam generators with innovative spiral tubes, each with a pump and 2 diverse, passive, redundant immersion cooler systems for residual heat removal through 6 immersion coolers. Water and steam collectors are provided to prevent water or steam from entering the coolant in the event of a steam generator heater tube leak, and the steam generators are short and protrude from the lead. Reactivity control is carried out via control rods. The non-active end of the fuel elements protrudes beyond the lead and the ends are connected to form a self-supporting core. The fuel element change is thus carried out under visual control by a remote-controlled system; the fuel element storage is outside the reactor vessel. The company states that the compact system will have

a power density of 1 m^3/MWe and therefore a small reactor building, which is an economic efficiency factor. The elimination of a refueling machine and structures above the core, which are immersed in lead, should reduce the need for in-service inspections (ISI). The fuel and the reactor concept are selected to achieve a breed ratio of 0.9 and keep the plutonium requirement low. (Alemberti 2021)

4.2.3.7 BREST-OD-300

Regarding the BREST-OD-300, see Chapter 5.3.

4.2.3.8 SVBR-100

The SVBR-100 with an output of 280 MW (thermal) and 100 MW (electrical) is an SMR and is being conceived by AKME Engineering in Russia. The reactor is a further development of the lead-bismuth cooled reactor line of the Russian submarine programme. The entire reactor with steam generator, core and pumps as well as shielding is to be manufactured as a monoblock with a surrounding protective shell. The reactor is to have passive safety systems using natural convection to prevent a temperature increase for at least 72 hours. The SVBR-100 is to be used as a multi-purpose reactor, for smaller grids to supply electricity and to provide process heat and for desalination. The fuel is uranium oxide with approx. 16% enrichment, but MOX fuel can also be used in a closed fuel cycle. The reactor is to be transportable by rail and manufactured in a centralised factory.

4.2.3.9 PEACER

South Korea is planning an LFR that is to be proliferation-resistant, environmentally friendly, accident-tolerant and economical, the Proliferation-resistant, Environment-friendly, Accident-tolerant, Continual, and Economical Reactor (PEACER). The reactor is to be used primarily for transmutation and has been in planning since 1996. In 2008, however, the Korean Ministry of Science and Technology selected the sodium-cooled fast reactor (SFR) as the technology for the transmutation of long-lived waste. However, there are newer LFR research approaches to an SMR system and ADS concepts. (GIF 2021a; Shin et al. 2015)

4.2.3.10 W-LFR

The Westinghouse Cooperation is also developing an LFR reactor concept, the Westinghouse LFR (W-LFR). Westinghouse is cooperating with several universities, for example to investigate the retention capacity of lead. The Versatile Test Reactor with a fast neutron spectrum, which was cancelled in 2022, was also to be used for development. Westinghouse also endeavoured to make progress on the W-LFR in the UK Advanced Modular Reactor (AMR) Feasibility and Development Project and to demonstrate LFR components and accelerate the development of high-temperature materials and manufacturing processes (GIF 2020a).

The W-LFR is expected to reach an electrical output of 460 MW, small enough to be used in smaller grids and large enough for base load. The reactor is planned as a basin-type design and will be coupled with an air-cooled supercritical CO_2 power generation system. The W-LFR should be economically competitive, flexible to use in future and diverse markets, achieve high safety standards and deliver base load, be load-following capable and have passive safety features. Load-following operation is not a characteristic of the reactor itself, but is achieved through an integrated energy storage system with cost-effective materials. Load-following operation is therefore achieved

without changing the power of the reactor. Due to the air cooling, no large body of water is required and the search for a site is greatly simplified.

The reactor vessel is very compact due to the use of a robust hybrid primary heat exchanger with microchannels, and by utilising a vessel cooling system for residual heat removal in the event of accidents with water as the coolant. In the event of a heat exchanger failure, natural convection is provided, which occurs solely due to heat transport; the heat is transported from the reactor vessel through the safety containment into a surrounding pool of water. After the water has been used up, air circulates outside the safety containment for further cooling. Initially, coolant temperatures of 530 °C are to be used and higher coolant temperatures as soon as innovative materials become available.

4.2.4 Technical development status

In 2015, (IRSN 2015) summarized the key development steps required to create an LFR. A number of development programmes have been conceived since then and some have been started, but no comprehensive development efforts have taken place and the status has therefore not changed significantly. The need for development concerns

- the management of corrosion and erosion risks. Here, the qualification of new steels, e.g. austenitic steels, for the RPV and newer corrosion and erosion-resistant materials for the reactor internals, coated fuel cladding tubes and steam generator heating tubes and for the blades of the primary circuit pumps is necessary.
- the sensitivity of the reactor concept to seismic activity due to the high mass of the coolant. In the SILER project (Seismic-Initiated events risk mitigation in LEad-cooled Reactors, 2011-2014) in the 7th Research Framework Programme of the EU, for example, insulation technologies for seismic decoupling were researched. The study (Alemberti 2021) assumes that shorter reactor vessels and seismic decoupling are ways of counteracting the problem.
- Ongoing inspection of the reactor and containment vessel and the structures and equipment inside the reactor vessel, in the integrated reactor concept without coolant circulation outside the RPV in e.g. pipework. This concerns the development of suitable methods and instruments and the investigation of whether it is possible to replace components of the reactor during its lifetime. The opacity of the coolant makes it difficult to handle components in the RPV and during fuel element changes. The study (Alemberti 2021) assumes that progress could be made here through suitable specific reactor concept properties, for example through innovative core configurations in which the upper, non-active parts of the fuel elements extend into the gas plenum, as would be pursued in some European reactor concepts.
- the filtration of the coolant. Methods need to be developed here to filter the coolant appropriately in order to avoid the formation of aerosols and to monitor and regulate the oxygen content.

In (Alemberti 2021), reference is also made to R&D work required due to polonium production in lead-bismuth, but also in lead due to impurities.

Further development efforts are aimed at understanding the incident behaviour of the LFR according to (IRSN 2015):

- The phenomenology of steam generator tube leakage. Here, for example, experimental tests were carried out at the “Lithium for Fusion” (LIFUS) plant of the “Agenzia nazionale per le nuove tecnologie, l’energia e lo sviluppo economico sostenibile” (ENEA) in Italy. The results were used to validate codes. Further experiments were planned, including the construction of a double-walled steam generator for test purposes.
- The behaviour of a molten core or corium in the coolant.
- The composition and quantity of fission and activation products in the gas plenum after a core meltdown: the problem is one of the major challenges for the concept of an LFR (as well as for an SFR). This requires research into how fission products are transferred from the fuel into the coolant and from there into the cover gas.
- The risks of solidification of the coolant. Here, the experimental analysis and calculation of the damage to the fuel are crucial, but also the development of methods to keep the coolant at a constant temperature even when the reactor is shut down and the external power supply is lost.

However, the main difficulty in developing the LFR into an industrially usable reactor concept is the problem of corrosion and erosion. Lead and lead-bismuth have a highly corrosive effect on steel structures, particularly at higher temperatures, especially on the reactor vessel and fuel cladding tubes, which contain iron, nickel and chromium and can dissolve into lead. (IRSN 2015).

Currently, the only way to minimise corrosion effects is to build up a layer of iron oxide on the structural materials in the reactor that are in contact with the coolant by adding oxygen to the coolant. The addition of oxygen to prevent corrosion by forming an oxide layer on the surfaces of the structural materials was first used in the Soviet submarine programme to mitigate the effects of accidents and was further developed in experimental plants for ADS. Some modifications were made for pure lead as a coolant, but only for small systems.

However, if there is too much oxygen, lead oxide crystals are formed which can lead to clogging. According to (Schulenberg 2020), the oxygen content must be kept between 0.001 and 0.01 ppm by means of precise oxygen measurement and control. This is achieved by mixing oxygen (addition) or hydrogen (removal of oxygen from the coolant) into the cover gas. The problem here is that oxide formation in lead is temperature-dependent and the temperature in the primary circuit varies locally. It is also difficult to keep the oxygen concentration as constant as possible throughout the reactor. The oxide layer must also be monitored. (IRSN 2015; Schulenberg 2020)

> *“The liquid metal must then remain in constant motion and must flush all parts of the reactor evenly. Otherwise, too much oxygen will be trapped in niches that are not flushed and lead oxide crystals will form. If, on the other hand, the flow velocity is too high, it washes the oxygen out of the oxide layer and corrosion holes form. In a large reactor, this oxygen control is anything but trivial. It takes a lot of experience to design a reliable reactor and it is therefore better to start with lower temperatures so that hot spots in the reactor core do not immediately lead to damage.” (Schulenberg 2020)*

Using this process for a small system has been proven, but according to (IRSN 2015) it is extremely complex to transfer the procedure to a large reactor concept. The process would have to be monitored under all operating conditions, including when draining and filling the reactor, in the entire

primary circuit and in areas with different temperatures. If it is not possible to control the effects of corrosion and erosion, this will limit the temperatures that can be reached and the cleaning of the coolant as well as the coolant speed in the reactor. A corresponding proof would also have to be provided for all operating conditions for a large commercial reactor concept. For this reason, (IRSN 2015) concludes that only maximum temperatures of up to 500 °C are possible. (Alemberti 2021) also comes to the conclusion that a relatively low operating temperature would help to minimize the problem, if not avoid it altogether. Another solution would be to pre-treat the corresponding surfaces of the fuel cladding, for example, or to introduce corrosion inhibitors into the coolant itself. The methods are being researched, but industrial proof of viability has not yet been provided (IRSN 2015).

Lead erosion is another problem, as it limits the coolant velocity in the primary circuit. This limits the coolant channel width and accordingly the core design is limited to an electrical output of between 100-600 MW. Erosion corrosion is also a major problem in the primary circuit pumps, which are located in the hot part of the primary circuit. The velocity of the pump blades is also limited and the pump blades cannot be coated with an oxide layer. Research and development programs on suitable materials for the pumps are investigating special materials such as powder ceramics, titanium silicon carbide, ferritic CrMo chilled cast steel or tantalum coatings. The steels can also become brittle, as is the case with T91 steel, which was analysed for the fuel cladding pipes of the ELFR at temperatures between 350 °C-400 °C.

(IRSN 2015) The study therefore concludes that the suitability of the various methods for corrosion and erosion control has not yet been demonstrated and that the combination of corrosion and erosion in particular is very difficult to control. The IRSN concludes that it is not desirable for a GIF reactor concept to entrust the safety of the plant largely to operational solutions that are particularly difficult to manage (IRSN 2015).

(IAEA 2021a) presents a comprehensive overview of recent worldwide research and development work on structural materials in LFR. The authors also conclude that

- there are already materials for low-temperature applications that can be researched and utilised. At high temperatures, however, there are concerns about durability and incomplete material testing standards, among other things.
- New R&D efforts would have to be made to qualify materials for high-temperature operation and to develop corresponding consequences for reactor concepts as well as for evaluation and testing procedures.
- new promising materials and coating technologies based on aluminium oxide have been developed and are available; however, for some new materials and coatings, considerable further research and time is still required for qualification before commercialisation. Aluminium-forming steels in particular have very favourable corrosion properties, while some other properties (e.g. creep) or their behaviour under irradiation would require further investigation;
- it was therefore necessary to address the issue of qualifying new materials for use in new, innovative nuclear systems in a more comprehensive way that would make it possible to harmonise the requirements of the regulators.

As a consequence, it is also assumed in (GIF 2020a) that the development of an LFR should proceed in two steps. Initially with moderate temperatures in a first phase with a demonstration reactor by 2030 and a second phase by 2040 with improved performance.

More recent European research projects include GEMMA (GEneration iv Materials MAturity, 2017-2021, EUR 4 million), M4F (Multiscale Modelling for Fusion and Fission Materials, 2017-2021, EUR 4 million), INSPYRE (Investigations Supporting MOX Fuel Licensing in ESNII Prototype Reactors, 2017-2022, EUR 4 million), and projects are also funded under the US-EU INERI programme (United States-Euratom International Nuclear Energy Research Initiative). Other EURATOM projects include the PIACE project (Passive Isolation Condenser, 2019-2022, EUR 2.5 million from the EU), the PATRICIA project (Partitioning And Transmuter Research Initiative in a Collaborative Innovation Action, 2020-2024, EUR 6.5 million from the EU), which is aimed at the MYRRHA project, and PASCAL (Proof of augmented safety conditions in advanced Liquid-metal-cooled systems, 2020-2024, EUR 3.8 million from the EU). The SESAME project (thermal hydraulics Simulations and Experiments for the Safety Assessment of MEtal cooled reactors, 2015-2019, EUR 5.2 million from the EU) was already completed in 2019 (GIF 2020a).

Conclusion: Technical development status

In the case of the LFR, the challenges for further development include the high melting point of lead, polonium production in lead-bismuth, the opacity of the coolant, seismic problems due to the high mass of the coolant and the potential for corrosion and erosion if the coolant comes into contact with structural materials and fuel cladding tubes. Only limited experience is available from the Soviet submarine programme. The establishment of an experimental infrastructure is necessary for further development.

Due to the unresolved corrosion problem, operation at relatively low temperatures is planned for some of the currently planned experimental and demonstration facilities. Only then will new materials and coatings be used to achieve the higher operating temperatures required for the industrial application of LFR. The research and development programs for the currently preferred lead-cooled ADS complement those for LFR.

Overall, research on LFR can be categorised as “applied research”.

4.2.5 Safety

The properties of the coolant also play a decisive role in the safety features of the LFR. The LFR is similar to the SFR in many properties of a fast reactor. (GIF 2020b; Alemberti 2021) provides a more recent overview of the safety characteristics of the LFR. The assessment according to (IRSN 2015) is still current. Only limited operating experience is available to date and some of the safety features described below have not yet been demonstrated.

One positive property of lead is the lack of chemical reactivity of lead and lead-bismuth in contact with water and air. There is also no chemical reaction with uranium or MOX fuel (Alemberti 2021). However, in the event of air ingress or the ingress of small amounts of water from a steam generator, lead forms oxides in the coolant, which generate and crystallise various lead compounds and can lead to a reduction in heat exchange in the core or to blockage of cooling channels, with the consequences of local overheating and fuel failure. It is therefore necessary to monitor and control the oxide content or filter the coolant (IRSN 2015).

Due to the neutronic properties of lead, the core can be designed with a low power density and large distances between the fuel rods in order to prevent blockages of the core, for example due to lead oxide formation. A design that enables passive residual heat removal by natural convection is also conceivable if the reactor concept has a correspondingly low overall power. In combination with the thermally inert behaviour, the reactor is accordingly independent of active coolant circulation and the external power supply required for this or an emergency power supply on the system. Due to the high thermal inertia of the LFR, the possibilities for a rapid rise in temperatures in the event of a cooling failure are limited and several hours are available to take measures before a material failure of the RPV would occur. In the design, particular attention must also be paid to ensuring that a common-mode failure does not occur in reactor concepts in which both heat exchangers are installed inside the reactor vessel (Alemberti 2021; IRSN 2015).

The high boiling point of lead protects the core from reactivity ingress due to void effects. Void effects can practically not occur. The structural materials would melt before the boiling temperature is reached. Void effects could only occur due to fuel cladding failure and the escape of fission gases. The study by (Alemberti 2021) states that the effect is limited by the amount of gases and the time of formation. In contrast, (IRSN 2015) assumes that overheating and the formation of a cavity can occur if, for example, a fuel element becomes blocked, fission gases escape due to fuel cladding failure or steam enters the coolant after a steam generator tube leak. The IRSN notes that the designers of the reactor concept consider such a scenario (an ELFR concept was analysed, see Chapter 4.2.3.4), which influences the active part of the core through a positive void effect, to be unlikely. Even if the probability were to be assessed as very low, a detailed investigation of all these scenarios would still have to be carried out according to IRSN. For example, the reactor concept can be designed in such a way that the transport of steam bubbles to the core inlet is unlikely due to the flow, or structures can be installed to separate the steam from the lead. However, analysing such phenomena is only possible with a full-scale test facility (IRSN 2015). The LFR must be designed to withstand loss-of-coolant accidents or loss of core cooling through an appropriate neutronic core design.

The lead has a high retention for volatile fission products such as caesium and iodine, so that the release of fissile materials from the primary cooling circuit is unlikely. This could also reduce a possible source term and thus the required planning zones for emergency protection (Alemberti 2021). However, it should be noted that the gas plenum and auxiliary systems can form a possible release pathway for volatile fission products and the behaviour of corium in the coolant and possible releases of fission products from the coolant are not yet well understood.

The high density of lead and the corresponding mass of coolant makes an LFR particularly sensitive to seismic risks and also poses a problem for the internal structures in the reactor vessel. One design goal for LFR is therefore to reduce the mass of the coolant as much as possible without opening up new weak points in terms of seismic design, such as through pipework. This problem also makes the concept of a very large LFR unlikely (IRSN 2015)

Core compaction would lead to reactivity in the core by reducing the spacing in the core, especially as the fuel elements are relatively far apart. The risk can be reduced by using spacers and a diagonal grid and by choosing a hexagonal channel surrounding the core (IRSN 2015)

Safety and reactor vessels must be inspected regularly. The use of lead has disadvantages. Even when the reactor is at a standstill, the high temperature of the coolant is unfavourable for the instruments. It is also difficult to insert instruments into the coolant due to the buoyancy. Optical methods are not possible. Other methods such as ultrasound are also more difficult due to the lower density difference between the coolant and structural materials. The oxide layer provided for corrosion protection prevents direct contact between an instrument and the surface.

Reactivity control is carried out in a similar way to the SFR using redundant and diverse systems with control and shutdown rods. Reactivity control by retracting the control and shutdown rods is comparatively more challenging than in other technology lines due to the buoyancy force in the coolant. Import through compressed air systems is difficult and ballast must be used if the compressed air systems fail. The velocity of the control rods is also a critical aspect (IRSN 2015).

In the event of a core meltdown, the developers assume that the corium floats on the lead and therefore no core catcher would be necessary. In addition, the fuel would spread over the surface. However, this behaviour still needs to be proven, also with regard to the formation of other alloys. Cooling of the molten fuel would also have to be restored as quickly as possible to avoid further damage to the structure. It should be noted that parts of the molten core could have entered the cooling circuit and that the cooling could not be utilised, especially in reactor concepts with cooling tubes. In addition, corium floating on the coolant would transfer heat to the reactor cover. It would therefore have to be shown that the lid can withstand the heat release. It could be difficult to prove this due to the great uncertainties regarding the numerous phenomena. So far, the behaviour of corium is not well understood, even in LWR. It would also have to be demonstrated that no recriticality can occur, either through certain geometric configurations, or through segregation into different phases and the possibility of parts of the corium sinking into the coolant (IRSN 2015).

If a core catcher is not used, the reactor vessel and containment would also have to be assumed to rupture and the interaction with the concrete would have to be investigated (IRSN 2015).

Conclusion: Safety

Due to the coolant lead or lead-bismuth, LFR have an advantage over SFR, as the coolant is less reactive to air and water. The high boiling point is also an advantage of the LFR, as it avoids the effects of boiling due to the coolant. The possibility of passive residual heat removal and the thermally inert behaviour of the coolant also have a positive effect compared to LWR.

On the other hand, the LFR has a number of disadvantages due to the opacity of the coolant and the need to maintain high temperatures even when the reactor vessel is at a standstill in order to prevent the coolant from solidifying. This hinders the possibilities for inspection and maintenance. Regular inspection of the structural materials is necessary due to their susceptibility to corrosion and erosion, and close monitoring of the oxide content in the coolant is required. Reactivity control is more difficult than in SFR or LWR due to the buoyancy of the coolant. The LFR shares with the SFR the same disadvantages of fast reactors with regard to reactivity control compared to LWR.

The LFR technology is still at an early stage of development and it is not yet clear whether a significant advantage compared to LWR can be achieved. At this stage, the LFR has no significant advantages or disadvantages compared to LWR or SFR.

4.2.6 Fuel supply and waste disposal

The LFR has the same advantages and disadvantages in terms of fuel supply and waste disposal as the SFR. The fuels used (MOX) and targeted fuel cycles are similar. The literature discusses its use for transmutation rather than for breeding new fissile material. However, the LFR can also be used as a breeder reactor.

Similar to SFR reactor components and fuel elements, LFR must be cleaned of coolant, in this case lead or lead-bismuth, which has settled on surfaces or remains in bulges and cavities that could not be drained when the coolant was removed from the reactor. Due to the toxicity of lead, special processes are required for this, which have been trialled and need to be further developed for industrial use. When using lead, there is a risk of lead being released into the environment.

As with sodium in SFR, LFR also generate significant amounts of lead waste contaminated with fission products and actinides, as well as containing activation products from structural materials. Due to its high mass, lead also requires appropriate qualification for the transport and storage containers used. Finally, despite the containment capability of lead, the occurrence of polonium, which is produced by the activation of bismuth in lead-bismuth, could entail the need to develop and evaluate special procedures for handling such a radiotoxic volatile substance (IAEA 2019c). Polonium-210 is not a particular problem for the long-term safety of a repository due to its short half-life of 138 days, nor is the activation of lead (^{205}Pb, $t_{1/2}$=1.51*10^{7}y) or the activation of impurities contained in lead (Dehlin and Wallenius 2023). However, the coolant must be disposed of in a repository due to the formation of ^{108m}Ag, with a half-life of 439 years and any other activation products contained in the structural materials (corrosion). In addition, LFR are similar to SFR in terms of disposal and are therefore not dealt with in more detail in order to avoid duplication.

Conclusion: Fuel supply and waste disposal

At the technology line level, the LFR has the same advantages and disadvantages in terms of fuel supply and waste disposal as the SFR (see Chapter 4.1).

With regard to the proportion of fission products in spent fuel elements from LFR, there is no significant difference compared to the high-level waste of light-water reactors. The use of reprocessing technologies also produces radioactive waste.

MOX fuels would also have to be shipped to a repository after use in the reactor. Alternatively, multi-recycling, as originally envisaged in a plutonium economy or contemplated in P&T scenarios, would have to be developed industrially.

The use of fast reactors has only marginal influence on the necessary criteria for a geological repository. The need for a geological repository cannot be avoided by any variant of a closed fuel cycle or P&T strategy.

4.2.7 Proliferation risks

The proliferation risks of the LFR are similar to those of other fast reactors such as the SFR and depend on the fuel cycle used.

Fast reactors are particularly suitable for breeding plutonium as fissile material, even if the use of breeding elements is not envisaged in the current reactor concepts for “regular” operation in accordance with the objectives of the GIF. Due to the compact design of the LFR, use as a breeder reactor is not intended, but individual fuel elements could be replaced by breeder elements depending on the design, although the possibility of breeder reactors is limited by the loss of reactivity in the core.

The use of MOX fuels is envisaged in the LFR and thus the utilisation of reprocessing technologies with their proliferation risks, which also exist when using transmutation fuels with the addition of minor actinides, even if the proliferation resistance of such fuels is increased compared to classic U-Pu-MOX. The use of pure MOX or transmutation fuels would eliminate the need for uranium enrichment and reduce the corresponding proliferation risks.

The GIF presented a more comprehensive overview of LFR proliferation risks in (GIF 2021b). The analysis was carried out using the three reference systems of the GIF, the ELFR, BREST-OD-300 and SSTAR. The authors state that a compact reactor design offers advantages for proliferation resistance.

Conclusion: Proliferation risks

At the level of the technology line, the LFR has the same advantages and disadvantages in terms of proliferation risks as the SFR (see Chapter 4.1).

The possibility of breeding weapons-grade plutonium is particularly relevant. The use of reprocessing increases possible proliferation risks and requires greater effort in fissile material control and monitoring measures. The use of pure MOX or transmutation fuels would eliminate the need for uranium enrichment.

Compared to LWR, the LFR has no significant advantages or disadvantages in terms of proliferation resistance at the technology line level if an open fuel cycle with the use of LEU is assumed.

4.2.8 Costs

So far, no detailed cost estimates for the construction and operation of a commercial LFR are available.

The study (Alemberti 2021) indicates some properties of LFR that could offer a cost advantage in the future. Firstly, it lists the property of the coolant being chemically inert, which would allow simple and efficient designs without complex and expensive intermediate cooling circuits to separate the primary coolant from the steam circuit (Alemberti 2021). However, this simple design requires the steam generators to be placed directly in the reactor vessel, with corresponding effects on reactor safety.

According to (Alemberti 2021), the high boiling point is also an advantage, as a simple design is also possible and the demonstration of safety features is simplified. The low pressure is also a cost advantage.

According to (Alemberti 2021), the use of high burnup fuels could also be a cost advantage by saving resources. However, the costs of reprocessing and MOX fuel fabrication are not taken into account.

With regard to the targeted temperatures, the LFR shares the characteristics of all SNR with high temperatures, increased efficiency and the possibility of commercialising process heat.

In terms of load-following operation, the LFR has the characteristic that the temperature difference between standstill and operation is small due to the need to heat the coolant.

The maximum output of an LFR is limited by the coolant and the desired passive safety features. Corresponding scaling effects to systems with higher output and corresponding cost advantages could not be utilised.

Considerable R&D costs still need to be incurred before a commercial LFR can be successfully developed.

Conclusion: Costs

So far, no detailed cost estimates for the construction and operation of a commercial LFR are available. The risks for investors are high.

Cost advantages over SFR are possible due to a simpler design and no need for an intermediate cooling circuit.

Due to a more compact and simpler design, LFR would have a potential cost advantage compared to LWR when utilising an open fuel cycle.

Overall, however, a significant advantage or disadvantage of LFR compared to LWR is not to be expected.

4.3 Gas-cooled fast reactors (GFR)

Gas-cooled fast reactors (GFR) represent a further technology line of fast reactors alongside liquid-cooled fast reactors (SFR, LFR). They differ from liquid-metal-cooled fast reactors essentially in the properties of the coolant, with helium being the main coolant for GFR under discussion today. The GFR are therefore also similar to the very-high-temperature reactors, which are also helium-cooled but operate with thermal neutrons (Tsvetkov 2016; Schulenberg 2020; Hatala 2021; GIF 2021a).

The potential advantage of a GFR is that particularly high operating temperatures in the range of 800-850 °C are possible in principle via gas cooling with helium (GIF 2021a). As a result, such systems could be both highly efficient and serve to provide high-temperature process heat.

Furthermore, the fast neutron spectrum in the reactor can breed new fissile material and reduce the build-up of minor actinides compared to LWR today. As GFR would enable a somewhat harder neutron spectrum compared to SFR and LFR, they would be fundamentally superior to liquid-metal-cooled fast reactors in this respect.

Disadvantages arise in particular from higher requirements in terms of the necessary materials and the cooling of the reactor core.

Gas-cooled fast reactors are also seen by the GIF primarily as a longer-term alternative to liquid-metal-cooled fast reactors (GIF 2021a).

4.3.1 System description

In GFR, heat removal from the reactor core is not performed by a liquid, but by a gas. There are significant differences within this technology line, particularly with regard to the gas used as the coolant, see below.

Reactor concepts with different power ratings were analysed as part of the GIF. Reactor concepts with lower thermal outputs in the 600 MW range generally offer the possibility of a modular design and advantages with regard to load-following operation. Reactor concepts with higher thermal outputs in the range of 2400 MW have a better neutron balance of the reactor core, so that the goal of fissile material production can be achieved at least for own requirements, but such concepts are more suitable for use in the base load range (Tsvetkov 2016).

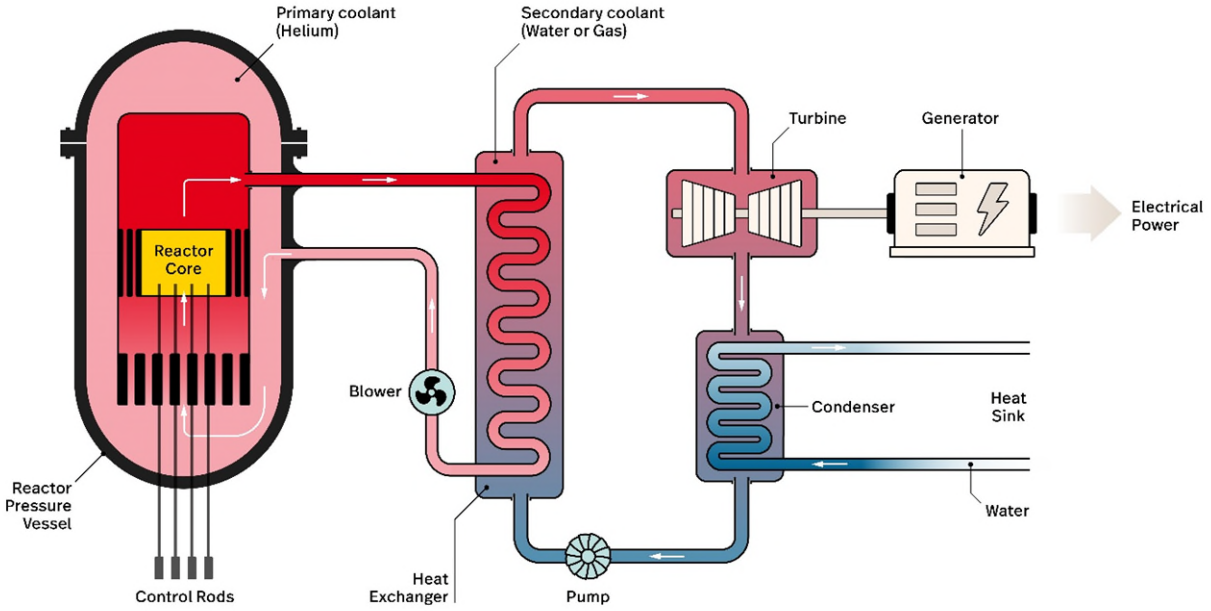
Reactor system

The design of a GFR is similar to today’s pressurised water reactors, see Figure 4-3. For material reasons, the steel of the reactor pressure vessel (RPV) should not be exposed to the high gas temperatures at the core outlet with temperatures above 500 °C. Therefore, the cooling gas must not come into direct contact with the wall of the reactor pressure vessel after passing through the reactor core, which requires appropriate internal structures for flow routing. This prevents the steel wall of the RPV from being exposed to the high core outlet temperatures. In a primary cooling circuit, the cooling gas flows into the RPV and fills the interior completely. The cooling gas flows through the reactor core from below and is heated. Above the reactor core, the hot gas is separated from the outer wall of the RPV by an internal structure and flows out of the RPV through thermally insulated connection lines.

In principle, this cooling gas could also be used directly to drive a gas turbine. However, reactor concepts with a direct cooling circuit and gas turbine are not yet technically feasible and are therefore not currently being actively pursued. Instead, the heat is transferred to a secondary cooling circuit via a heat exchanger. Depending on the reactor concept, this can consist of another gas circuit or a water-steam circuit. However, if water is used in the secondary circuit, the possible effects of a leak between the primary and secondary cooling circuits and the associated ingress of water or steam into the reactor must be taken into account from a safety point of view (reactivity control).

The energy is converted into electricity in the secondary cooling circuit via a turbine and a generator. The residual heat is then transferred to an external heat sink via another cooling circuit. In the primary cooling circuit, fans ensure the required cooling mass flow in the reactor core through active circulation.

Figure 4-3: Conceptual diagram of a gas-cooled fast reactor (GFR)



Source: Own illustration

To ensure that the cooling gas has a sufficiently high density to transport the heat generated in the reactor core, the primary cooling circuit in helium-cooled GFR is typically pressurised to around 7 MPa. If other gases are used, other pressures are required, for example approx. 20 MPa for reactors cooled with carbon dioxide (Tsvetkov 2016). The core outlet temperatures of GFR can be in the range 800-850 °C for helium, while lower core outlet temperatures in the range between 500 and 600 °C are typically discussed for other gases. The maximum possible core exit temperatures also depend on the availability of appropriate structural materials.

As decay heat continues to be generated in the reactor even after shutdown, which must be transported continuously, the reactor cannot simply be depressurised during shutdown in order to open the reactor pressure vessel for fuel element loading and unloading. For this reason, the fuel elements must be inserted into or removed from the reactor pressure vessel via an airlock through the lid. A loading machine, which inserts the fuel elements into their positions in the reactor core, must be located in the upper area of the reactor pressure vessel for this purpose.

For this reason, the control elements provided for power control and shutdown in this reactor concept are inserted into the reactor pressure vessel from below.

Fuel

A wide variety of fuels and fuel cladding materials are being discussed for GFR.

The technologically most advanced variant consists of a uranium-plutonium mixed oxide fuel (MOX), which is enclosed in a stainless steel cladding tube, comparable to the fuels used in SFR. For the planned prototype reactor ALLEGRO, see Chapter 4.3.3.1, such a fuel is planned as the first core load. This fuel would be used in fuel elements with, for example, 169 fuel rods in a hexagonal arrangement. The required plutonium content of the MOX fuel planned for the ALLEGRO is approx. 30%. When using such a fuel, the maximum operating temperature in the reactor is limited to a range between 500 and 600 °C, comparable to the operating temperatures of an SFR, in order to maintain permissible cladding temperatures.

At temperatures above approx. 600 °C, stainless steels lose their stability and the fuel cladding could either swell due to the internal pressure or collapse under the existing external pressure (Schulenberg 2020). This type of fuel is therefore ruled out for future GFR with the aim of achieving high working temperatures.

In order to enable high operating temperatures, TRISO fuels have occasionally been discussed as a possible option for GFR, as they are to be used in very-high-temperature thermal reactors, see Chapter 4.6. However, these are not sufficiently resistant to the fast neutron flux in GFR, as the pyrocarbon they contain would decompose (Schulenberg 2020).

The use of uranium-plutonium carbide is therefore currently being discussed as a further fuel variant. This fuel has a much higher thermal conductivity than MOX fuel (at fuel temperatures of 1000 °C by a factor of about ten), so that this fuel would enable a higher power density. To enable the higher associated cladding temperatures, a fuel cladding made of fibre-reinforced silicon carbide is under discussion (Schulenberg 2020). In the GIF reference concept, see Chapter 5.4, the fissile material content of the fuel should be less than 20%.

Fuel cycle

In principle, GFR can achieve a higher breeding rate for the production of new fissile material with shorter doubling times compared to liquid-metal-cooled fast reactors. While, historically, GFR were therefore also investigated particularly as breeders for the production of additional quantities of fissile material, the focus is currently more on concepts that are intended to contribute to the stabilisation of the existing quantities of actinides (both plutonium and minor actinides) (Tsvetkov 2016).

In principle, fuel cycles with spent fuel reprocessing are always envisaged for GFR. A distinction can be made between concepts with a homogeneous reactor core and those with a breeding blanket in which new fissile material is to be produced in a targeted manner.

Coolant

In principle, different gases can be used for cooling GFR. Today, concepts based primarily on cooling with helium are being discussed, although cooling with air, carbon dioxide or even steam would also be possible in principle.

Helium as a coolant offers various advantages, especially compared to the use of sodium or lead as in the SFR or LFR (Hatala 2021).

Firstly, it does not react chemically with air or water, so that no event sequences based on coolant-water or coolant-air reactions need to be considered. Also, unlike lead, it does not contribute to corrosion of structural materials in the reactor. When the reactor cools down, there is also no risk of the coolant solidifying and blocking cooling channels or causing damage to fuel elements or structural materials.

From the point of view of neutron physics, helium has the advantage that it does not absorb neutrons and, due to its low density, practically does not moderate the neutrons, so that it contributes to a very hard neutron spectrum with low neutron losses. This means that a high breeding rate is also possible in the fuel and no radiologically problematic activation products are produced in the coolant.

Unlike liquid metals, helium is a transparent coolant, which makes it possible to visually inspect the reactor, especially during downtimes and during fuel element loading and unloading processes.

However, these advantages of helium are also associated with disadvantages. This applies in particular to the reactor’s cooling capability. The thermal conductivity of helium is very low compared to liquid metals, which, together with the low density of the coolant, places high demands on the cooling circuit. Therefore, a higher pressure in the range of 7 MPa must be selected in order to ensure a sufficient density of the coolant for cooling. At the same time, a high flow rate must also be guaranteed to ensure sufficient heat transfer from the fuel to the coolant. The higher flow rate can lead to increased vibrations in the fuel elements. The maximum permissible power density in the core of GFR is significantly reduced compared to liquid metal-cooled reactor concepts so that the fuel cladding temperatures remain within the permissible range under all operating conditions.

A sufficiently high coolant flow rate must also be maintained in non power operation to remove the residual heat (Schulenberg 2020).

Air as a coolant would offer the advantage over helium of being easier to replace in the event of loss-of-coolant accidents. However, air would lead to a higher activation and thus a higher radiological inventory in the coolant and would have poorer corrosion properties, so that it is hardly discussed as a coolant for GFR (Tsvetkov 2016).

Carbon dioxide at pressures and temperatures beyond its critical point (see also Chapter 4.5) could also be used as a coolant. However, due to the thermal decomposition of carbon dioxide at higher temperatures beyond approx. 700 °C, the working temperatures for such reactor concepts would be limited to a range below approx. 600 °C, which would rule out high-temperature applications of such reactor concepts in the range above 800 °C (Tsvetkov 2016).

In principle, a gas-cooled fast reactor can also be cooled with steam. For example, concepts with a fast neutron spectrum are also being discussed for supercritical water-cooled reactors, see Chapter 4.5.3. Historically such concepts have also been investigated in Germany (KfK 1966). The coolant concentration in the core must be as low as possible in relation to the fuel concentration in order to avoid excessive moderation of the neutrons. In principle, this can be achieved by a denser arrangement of the fuel rods with a higher fissile material concentration in the fuel However, in these cases the reactor core is not in its most critical arrangement. If the coolant density is increased or water ingress occurs or the fuel configuration is changed, for example in the event of an accident, the reactivity in the reactor core could increase very sharply (Schulenberg 2020). Furthermore, steam would place higher demands on the corrosion behaviour of the fuel cladding (Tsvetkov 2016).

Overall, therefore, the other coolants apart from helium currently play a subordinate role in the development of GFR, so that the following discussions are essentially orientated towards the use of helium as a cooling gas.

Spectrum (moderator)

GFR are reactors with a fast neutron spectrum, so a moderator is not used. Due to the low moderation properties of the coolant, the neutron spectrum of a GFR is even harder compared to liquid-metal-cooled fast reactors and the neutron losses in the fuel and coolant are even lower, so that in principle a high breeding rate in the fuel could be achieved. At the same time, the mean free path of the neutrons in the reactor core is long, so that a relatively high neutron leakage from the reactor core can occur. To improve the neutron balance, large reactor cores and the associated high reactor power are therefore favourable.

However, (van Rooijen 2009) points out that, in view of the large quantities of plutonium available from LWR fuels, the focus of GIF is less on additional production of fissile materials and more on limiting the accumulation of new actinides.

Pressure, temperature and construction materials

Due to the low density and low thermal conductivity of gases, a higher pressure must be present in the primary cooling circuit. Depending on the cooling gas, this is in the range of 7 to approx. 20 MPa.

GFR can achieve a high operating temperature that could allow other industrial applications in addition to the generation of electricity (process heat, hydrogen production). However, with materials currently used in fast reactors and with certain cooling gases, the maximum operating temperatures would be limited to values in the range between 500 and 600 °C, comparable to other fast reactors (SFR, LFR). The GIF is therefore endeavouring to develop new, high-temperature-resistant materials which, with helium as a cooling gas, should then also be able to achieve operating temperatures in the range beyond 800 °C (GIF 2021a).

4.3.2 Historical developments

Developments in the field of GFR took place in the U.S. and Germany in the 1960s and 1970s, in the UK in the 1970s and in Japan in the 1990s. To date, however, no helium-cooled fast reactor has been built and operated (Schulenberg 2020).

In 1962, General Atomics presented a concept for a prototype reactor with a thermal output of 300 MW and a commercial power reactor with an electrical output of 1000 MW in the U.S. The concept for the prototype reactor was further developed from 1968 onwards. It was to be operated at a core inlet temperature of 385 °C, a core outlet temperature of 550 °C and a pressure of 8.5 MPa. The fuel was to be a uranium-plutonium mixture with a stainless steel cladding tube. The core design was very similar to that of sodium-cooled fast breeder concepts. The developments were discontinued in 1981. Safety analyses carried out up to this point in time did not provide a clear picture of the achievable safety level (IRSN 2015).

In Germany, the potential of gas-cooled fast breeder reactors as a fall-back option was investigated in the 1960s in parallel to the work on sodium-cooled fast breeder reactors (KfK 1971; 1967; 1966). It was assumed in (KfK 1971) that developments in the field of gas-cooled thermal reactors could be utilised to a considerable extent for a GFR. A distinction was made between systems with gas turbines and those with steam turbines. Furthermore, oxidic fuel in the form of pellets, carbide fuel

and oxidic fuel in the form of coated particles as well as cladding tube materials made of steel and vanadium were discussed. At the time, clear advantages were seen in favour of a GFR with a steam turbine and oxidic fuel compared to the other variants of a GFR, particularly with regard to timely feasibility. With regard to the fuel elements, a GFR with a steam turbine and oxidised fuel could largely be based on known fuel cladding materials. However, gas venting would be required for the fuel elements during operation in order to reduce the pressure build-up in the fuel rods caused by the resulting fission products. Due to lack of experience, the potential of variants with a gas turbine is more difficult to estimate compared to a system with a steam turbine. Both components for the cooling circuit and, due to the higher operating temperatures, more advanced fuel elements would have to be developed. The developments in Germany were not pursued further in favour of the sodium-cooled fast breeder reactor.

In the EU, a consortium of several manufacturers pursued the development of a GFR in the period from 1970 to the early 1980s. Four different concepts were pursued under the designation GBR 1-4. Both uranium-plutonium mixed oxide fuels and silicon carbide-coated fuel particles similar to TRISO fuel particles in very-high-temperature reactors were discussed (IRSN 2015). According to (Hatala 2021), coated fuel particles were discussed in the GBR-2 and -3 concepts with the aim of achieving a higher core outlet temperature, and thus a higher efficiency. However, this would have meant that various reactor structures would have had to be constructed from high-temperature-resistant ceramic materials, which was considered questionable at the time in terms of technical viability.

From the late 1970s, the UK pursued a programme to develop GFR based on the experience gained from the development of SFR (Dounreay) and carbon dioxide-cooled thermal reactors under the term “Existing Technology Gas Breeder Reactors” (ETGBR). This line of development was pursued into the 1990s. From this point onwards, the concept was redefined as an “Enhanced Gas-Cooled Reactor” with the aim of reducing existing actinide stocks. This concept was to be a reactor with a thermal output of 3600 MW, cooling with carbon dioxide and a nitride fuel (Hatala 2021).

A programme to develop fast breeder reactors was launched in Japan as early as the 1960s, in which gas-cooled reactor concepts were also discussed. Concepts based on steam, carbon dioxide or helium were considered. The developments there were continued into the 1990s and then moved into the context of the GIF developments (Hatala 2021). In the field of fuels, coated particles made from nitride fuel were also investigated in particular (Stainsby 2015).

An early discussion of the safety aspects of thermal and fast gas-cooled reactor concepts can be found, for example, in (IAEA 1980).

(IRSN 2015) notes that in the developments of the 1960s to 1980s, the expectation that a GFR would have advantages over an SFR could not be achieved. In particular, the better neutron balance of a GFR and the possibility of dispensing with an intermediate cooling circuit were seen as advantages, since helium as a coolant has only a low activation. These possible advantages would be offset by higher requirements for cooling the reactor core, and if the fuel cladding materials also discussed for SFR were used, no higher temperatures could be achieved in the reactor, so that there would be no significant efficiency gains compared to SFR.

4.3.3 Current developments

Starting in the 2000s, work on two reactor concepts, a smaller prototype with a thermal output of 75 MW, which is being pursued today as ALLEGRO, and a reference concept with a thermal output of 2,400 MW, which is referred to as a GFR, was carried out within the framework of GIF, primarily driven by France (GIF 2021a).

In addition, various other options based on different fuels (carbide fuel plates, nitride fuels and oxide fuels with silicon carbide cladding), different fuel cladding materials, different cooling circuits and different thermal outputs were considered (Vasile 2017; Stainsby 2015; Poette et al. 2013; Stainsby et al. 2011; van Rooijen 2009). Concepts with a lower thermal power in the range of 600 MW were no longer considered from around the 2010s onwards, as a breeding rate of at least one is not achieved in the associated smaller reactor cores. This means that such a reactor would not be able to generate more fissile material than it consumes. Concepts with direct drive of a gas turbine were also rejected in favour of a system with an intermediate cooling circuit based on a helium-nitrogen mixture in conjunction with a subsequent water-steam circuit. Fuel concepts based on fuel plates were postponed in favour of fuel rod concepts, which are more common today, due to difficulties in the manufacturing process.

The IAEA ARIS database lists three reactor concepts assigned to the GFR technology line, the ALLEGRO, the EM^2 and the KAMADO FBR (IAEA 2023d). The GIF reference concept, which is also listed under the abbreviation GFR, is discussed in detail in Chapter 5.4.

4.3.3.1 ALLEGRO

ALLEGRO was originally designed by the French Commissariat à l’énergie atomique et aux énergies alternatives (CEA) (Hatala 2021).

ALLEGRO is an initial experimental plant designed to test the basic concept of a GFR. The reactor is intended to have a thermal output of 75 MW; electricity production is not planned for this reactor. The reactor is to be cooled with helium at a pressure of 7 MPa and a maximum operating temperature of initially 530 °C (IAEA 2023d).

In a first step, the ALLEGRO reactor core is to be constructed from 81 fuel elements with uranium-plutonium MOX fuel in a stainless steel cladding. Advanced fuel elements are to be tested at six positions in this reactor core. In a second step, the fuel elements with carbide fuel with a ceramic fuel cladding intended for later GFR concepts will then be used. An operating temperature of 850 °C is to be achieved with these fuel elements. The aim is to operate such fuel elements for a burnup of 2000 effective full load days in the reactor under prototypical operating conditions in order to qualify them for later use in the entire reactor core and for future GFR. The reactor is cooled via two helium cooling circuits, which transfer the heat to water-steam circuits via heat exchangers (IAEA n.d.a).

In order to adequately shield the reactor pressure vessel from the fast neutron spectrum inside the core, it is planned to surround the fuel elements with four rows of a total of 174 stainless steel reflector fuel elements and a further three rows of a total of 198 boron carbide absorber fuel elements (Hatala 2021).

To be able to use two different fuel element types in this reactor, they should each have the same dimensions according to (Schulenberg 2020). In the case of carbide fuel elements, the gas inlet into the fuel element would then have to be throttled for higher fuel element temperatures, which in turn would make it necessary to thermally insulate the fuel element channel to limit the maximum temperatures of the structural materials.

The CEA’s work on the development of ALLEGRO has been supported by research institutes from Poland, Hungary, Slovakia and the Czech Republic since 2010 as part of a joint declaration of intent. The V4G4 Centre of Excellence was founded for this purpose in July 2013. Work on ALLEGRO and on general developments in the area of GFR is funded by the EU’s SafeG project, among others (GIF 2021a).

4.3.3.2 EM²

The following presentation is based on (Oeko-Institut e.V.; WIP; PhB 2021). The EM² is a concept from General Atomics for a helium-cooled fast reactor with an electrical output of 265 MW.

The fuel is uranium carbide, which is surrounded by a cladding made of a special silicon carbide composite (SiGA) developed by the U.S. company General Electric. The planned target burnup is 143 MWd/kg of heavy metal. Fission gases produced are to be extracted from the fuel by means of an extraction system and stored. This allows the required high burnup to be achieved and the internal pressure of the fuel rods to always be kept just below the primary circuit pressure of 13.3 MPa.

The reactor core consists of 85 fuel elements. These consist of driver fuel elements with an enrichment of 14.5% and breeder elements with natural uranium. The average enrichment of the core is therefore 7.7%. The fuel elements are surrounded by a heavy metal reflector made of zirconium and another graphite reflector.

Helium gas is used for core cooling at a maximum working temperature of 850 °C, which directly drives a turbine with a connected generator in a closed cooling circuit in a power conversion unit (PCU). The reactor core is located in a reactor pressure vessel, which is connected to the PCU via a cylindrical cross pipe. The reactor pressure vessel, the cross pipe and the PCU are each enclosed in a gas-tight containment. The inside of the reactor pressure vessel is provided with silicon/aluminium insulation in order to keep the wall temperatures below 371 °C. This means that SA-533 grade B steel can be used for the pressure vessel.

4.3.3.3 KAMADO FBR

The KAMADO FBR is a concept for a carbon dioxide-cooled fast reactor with a thermal output of 3000 MW and an electrical output of 1000 MW. The concept was developed by the Japanese Central Research Institute of Electric Power Industry (CRIEPI). Development began in 2008, with the latest information in the ARIS database dating back to 2011. At that time, the concept was at a very early stage of development (ARIS 2011a).

The development goal was formulated as a reactor concept with a negligible probability of severe accidents. At the same time, the concept was intended to address the problem of limited uranium reserves.

Uranium-plutonium mixed oxide with a total plutonium content of 18% is to be used as fuel, and stainless steel is planned for the fuel cladding. This is intended to achieve a burnup of 100 MWd/kg, which would correspond to a fissioned content of 11% of the initial heavy metal. Natural uranium will be used in a breeding area. The fuel is to be reprocessed and the uranium and plutonium it contains reused.

A fuel element is made up of 36 fuel rods of 3.7 metres in length and is located in a pressure tube. The pressure tubes are located in an unpressurised water basin. Supercritical carbon dioxide is used for cooling at an inlet temperature of 200 °C and an outlet temperature of 400 °C at a pressure of approx. 15 MPa. The heat is transferred to a secondary water-steam circuit via a steam generator. The water basin is used for heat removal in the event of active core cooling failures. During operation, water evaporates between the pressure tubes and the resulting steam is also used for heat removal. If the active cooling fails, the steam is displaced by water at low temperatures (< 60 °C), thus ensuring cooling. To compensate for a positive reactivity contribution from the water, gadolinium, which absorbs thermal neutrons very strongly, is to be added to the fuel. The reactor concept completely dispenses with a reactor pressure vessel and sees no need for emergency cooling systems.

4.3.4 Technical development status

The next development step within the GIF is the construction of the ALLEGRO experimental reactor. This reactor will be used to develop and test the fuel, the fuel elements, the required helium technologies and the necessary safety systems, in particular the residual heat removal systems. Furthermore, ALLEGRO will be used to test the provision of process heat for industrial applications and to carry out general research into reactor concepts with fast neutrons (GIF 2021a).

As necessary development steps prior to the construction of ALLEGRO, (IAEA n.d.a) names the qualification of suitable computer programs, in particular for the analysis of transients, as well as technical developments in the field of helium cooling, the verification of core cooling through tests on fuel element sub-bundles, the development of high-temperature resistant materials (1250 °C for one hour), the development of helium blowers that can maintain a constant mass flow in the pressure range between 0.3 and 7 MPa, and the development of core instrumentation for monitoring power distribution and for fuel element handling.

(Hatala 2021) also sees the operation of an experimental reactor, as planned in the form of ALLEGRO, as a necessary next development step for the qualification of high-temperature resistant materials and the demonstration of GFR-specific safety systems. The advanced carbide fuel with ceramic fuel cladding can only be qualified after irradiation and corresponding post-irradiation examinations under realistic operating conditions. Therefore, the ALLEGRO experimental reactor cannot be immediately equipped with such fuel elements.

The fuel for the launch of ALLEGRO is to consist of uranium-plutonium mixed oxide with a fuel cladding made of 15-15 titanium steel. According to (GIF 2021a), further investigations into the behaviour of this fuel cladding material at high temperatures in a helium atmosphere are necessary. Furthermore, irradiation tests of the fuel intended for use in ALLEGRO with this fuel cladding in a test reactor with fast neutrons as well as corresponding post-irradiation examinations are required. For silicon carbide materials, on the other hand, heavy ion irradiation of the material with a subsequent investigation of the material behaviour is to be carried out first. It is also planned to

investigate the oxidation behaviour of these materials at high temperatures (1500 °C). The irradiation behaviour of these materials can then be investigated in ALLEGRO itself.

The further work to be carried out in the short term according to (GIF 2021a) concerns calculations for the reactor core design with uranium-plutonium mixed oxide fuel, the final determination of the reactor power and power density to achieve the research objectives as well as compliance with the safety parameters and the determination of the criteria for the selection of the optimum reactor core. Furthermore, suitable data for calculation programs for fuel development must be determined. The qualification of the MOX fuel for the first operating phase of ALLEGRO must be carried out and a qualification programme for the ceramic fuels in ALLEGRO must be drawn up.

Furthermore, (Hatala 2021) sees the need to carry out tests on fuel elements or partial bundles under representative pressure and temperature conditions before using advanced fuel elements in ALLEGRO. Finally, large-scale tests to demonstrate passive heat removal in ALLEGRO must also be carried out as part of the licensing process.

According to (Tsvetkov 2016), the main development tasks at GFR are generally in the areas of material development, fuel development, control and instrumentation as well as other developments to ensure reliable and safe operation.

(Hatala 2021) The current development goals for fuel development include a system that guarantees a cladding tube temperature of 1000 °C during normal operation and safe confinement of the fission products even at temperatures of 1600 °C for several hours during incidents, and whose coolability remains ensured even at cladding tube temperatures of up to 2000 °C. Appropriate fuel cladding materials and their manufacturing processes still need to be developed. In particular, the requirements for the necessary length, diameter, surface roughness (which is decisive for the heat transfer from the fuel rod into the cooling gas), ductility and tightness, even under the influence of irradiation, must be met. Furthermore, manufacturing processes for uranium-plutonium carbide pellets must be developed and their behaviour tested under irradiation. Current calculation programs for predicting fuel behaviour do not cover the operating conditions of the fuel planned for ALLEGRO. Appropriate measurements are required here, particularly with regard to the release of fission gas from the fuel pellets. This determines the pressure build-up in the fuel rod and thus the question of cladding tube stability. Finally, the nuclear data available in conventional databases must also be supplemented for verification in corresponding calculation programs in accordance with the conditions in GFR.

Conclusion: Technical development status

With regard to the materials required for GFR (in particular the fuels themselves, the cladding, but also the structural materials in the reactor core), a distinction can be made between the requirements for GFR with temperatures in the range of other fast reactors (500 °C-600 °C) and the requirements in the range of higher temperatures (> 800 °C).

In principle, fuels and materials already exist for the temperature range between 500 °C and 600 °C that are suitable for use in fast reactors. However, these must also be adapted for the specific conditions in GFR and further evidence must be provided. The development status for this temperature range can be considered to be on the borderline between “applied research” and the area of “development”. In contrast, neither qualified fuels nor suitable fuel cladding or structural materials exist for the high temperature range above 800 °C that is actually targeted for GFR. Here, developments are still at an early stage of “applied research”. Systems that can be used to

investigate such materials under relevant operating conditions (high temperatures, high pressure, hard neutron spectrum) do not yet exist.

With regard to operational system functions, extensive questions regarding the development of systems and components (core instrumentation, gas blowers, fuel element handling) still need to be clarified for GFR. Here, too, developments are still at an early stage of “applied research”.

There is also still a considerable need for development with regard to key safety functions. This concerns the question of passive or at least diverse shutdown systems, ensuring sufficient residual heat removal in the event of all assumed incidents and accidents and, in particular, the question of the extent to which passive cooling of the reactor can be guaranteed. With regard to the required safety functions, the developments at the GFR are therefore still in the “applied research” phase.

Extensive data on the materials to be used in the reactor, as well as on fuel development under irradiation and thermal-hydraulic verifications, are still required for the verifications required as part of the approval process. The area of safety demonstration principles is therefore also still in the “applied research” phase.

Against this background, the authors of this report estimate that the development status of GFR is currently still in the “applied research” phase.

4.3.5 Safety

Helium has various fundamental advantages as a coolant, which is why it has been discussed for many decades both for thermal gas-cooled reactors (see Chapter 4.6) and for GFR. Helium is chemically inert and does not react with the materials in the reactor (corrosion, hydrogen formation from reactions with metallic fuel cladding at high temperatures) or with other coolants such as water or steam, which could be used in secondary cooling circuits. It also does not absorb neutrons, so that no activation products are formed in the coolant that increase the radiological requirements during normal operation or in the event of an accident. Furthermore, it is a transparent coolant so that, like water but unlike liquid metals, it can be visually inspected in the reactor.

As helium does not have a boiling transition as a coolant, it does not produce a void effect. However, a loss of coolant results in a positive density effect, albeit small, which must be taken into account when designing the core. Due to the higher neutron losses from the reactor core of a GFR compared to liquid-metal-cooled fast reactors, the fissile material content in the fuel is increased compared to such reactor concepts. Together with the harder neutron spectrum, the Doppler coefficient of the fuel is therefore somewhat lower than in SFR or LFR (Tsvetkov 2016).

Reactivity control in GFR is carried out via control and shutdown elements. The ALLEGRO still uses two identical but independent groups of control and shutdown elements to shut down the reactor. According to (Hatala 2021), the development of a completely diverse and purely passive shutdown system is required for GFR in order to increase reactor safety. While a neutron absorber (boron) can be added to the coolant in LWR for this purpose, this is not easily possible in GFR.

Since helium or gases in general do not contribute to the moderation of neutrons in the reactor core, events that can lead to increased moderation in the reactor core must be considered with regard to reactivity control. This would be conceivable, for example, if water or steam were to enter from a secondary cooling circuit or from emergency cooling circuits. The use of water as a coolant in GFR in systems directly adjacent to the primary cooling circuit is therefore problematic.

If a water-steam circuit is selected as the secondary cooling circuit, the pressure in the secondary cooling circuit is typically higher than in the primary cooling circuit. In this case, incidents involving the ingress of water or steam into the reactor core (both during power operation and shutdown phases) must also be taken into account, particularly with regard to the possible introduction of reactivity into the reactor core (Schulenberg 2020).

A major disadvantage of helium and gases in general for cooling a reactor is the low density and thermal conductivity of gases, which results in special requirements for heat removal from the reactor core.

The loss of forced coolant circulation in the primary cooling circuit is an important event sequence with regard to safety in GFR (Schulenberg 2020). While in today’s LWR or liquid-metal-cooled fast reactors with a suitable design, natural circulation occurs in the primary cooling circuit, which can ensure sufficient heat removal from the reactor core to the secondary cooling circuit after the reactor is shut down, a cooling gas without forced circulation is typically not able to do this. An active component (blower or emergency blower) is therefore practically always required here, which must be supplied with energy (electricity or emergency power or steam from the secondary cooling circuit).

(Tsvetkov 2016) also sees a major challenge for GFR in the difficulty of enabling passive heat removal from the reactor core and ensuring sufficiently reliable active heat removal. For reactor concepts with a lower thermal output, such as the ALLEGRO, passive cooling is conceivable in principle, but there would be considerable economic disadvantages for such reactor concepts. (Hatala 2021) notes that the possibility of passive core cooling is a key area of development in current GFR research.

Another important event sequence is the loss of coolant from the primary cooling circuit. In LWR, the highly pressurised coolant initially escapes from the cooling circuit in such cases. It is therefore replenished from pressurised accumulators and via emergency cooling systems to ensure sufficient coolant density and thus the possibility of core cooling. In liquid-metal-cooled fast reactors, the primary cooling circuit is not pressurised, so that the coolant only escapes from the cooling circuit due to its geostatic pressure, depending on the location of a leak. If the reactor vessel in liquid-metal-cooled fast reactors is surrounded by a second vessel that collects the coolant escaping at low pressure, a large drop in the level in such reactors can be avoided. In GFR, on the other hand, the highly pressurised gas would be blown out of the primary cooling circuit, as in LWR. To replace this loss of coolant, very large pressurised storage tanks are therefore required for GFR, which could be used to feed helium or even nitrogen into the cooling circuit (Schulenberg 2020).

The primary cooling circuit and the pressure accumulators required for emergency injection in the event of loss-of-coolant accidents are surrounded by a protective container similar to the containment vessel in today’s LWR, which holds back the escaping coolant to prevent it from being released into the environment. This protective container must be designed for a pressure in the range of approx. 0.5 to 1 MPa so that the coolant continues to have a sufficiently high pressure and thus a sufficiently high density for core cooling.

Conclusion: Safety

An intrinsic advantage of helium-cooled fast reactors compared to today’s LWR is that helium, as a chemically inert gas, does not lead to corrosion or other chemical reactions. Like water, helium is a transparent coolant, which can be seen as an advantage compared to other fast reactors, but not compared to LWR. The fact that no radioactive activation products are formed in helium gives it a further intrinsic advantage over today’s LWR, but also over other fast reactors.

However,these advantages are offset by considerable intrinsic disadvantages with regard to other safety functions, which result primarily from the poorer heat removal properties of helium.

To ensure heat removal, a sufficiently high density and thus a sufficiently high pressure of the coolant must be maintained at all times. This requires a sufficient overpressure to be maintained even during normal operation, i.e. also during plant shutdowns for fuel element changes and inspections. Furthermore, either a sufficient heat transfer from the fuel elements into the coolant must be ensured by a corresponding core design, even under natural circulation conditions, or forced coolant circulation must be maintained with a very high level of reliability, even under fault and accident conditions. This represents an intrinsic disadvantage compared to current LWR.

Particularly with regard to passive cooling of the reactor core, the higher power of a GFR, which is preferable for economic reasons, also conflicts with the safety-related advantages of lower power values, so that there are conflicting objectives here between safety and the expected costs of GFR.

Also with regard to reactivity control, reliable, passive or at least sufficiently diverse shutdown systems have not yet been developed for GFR. The extent to which this is addressed in specific reactor systems cannot be assessed at the technology line level.

It is also not possible to assess at the technology line level the extent to which, for example, event sequences involving the introduction of reactivity (e.g. due to water entering the reactor) need to be taken into account.

At the level of the technology line, it cannot therefore be assumed that GFR have a clear intrinsic advantage or disadvantage compared to LWR today. However, due to extensive unresolved issues regarding the assurance of sufficient core cooling, an overall disadvantage in terms of safety can be assumed.

4.3.6 Fuel supply and waste disposal

GIF’s developments in the field of GFR are aimed at a closed fuel cycle. The actinides contained in the spent fuel are to be completely reused and only fresh breeding material (natural uranium or depleted uranium) is to be added. A reactor should generate its own fissile material requirements from the breeding material contained in the reactor core, but with little or no surplus. Plutonium from the spent fuel of LWR is therefore intended for the first fuel of new GFR reactors. To enable a sufficiently fast deployment of GFR reactors, the reactor core of a GFR should be designed so that it contains no more than approx. 15 tonnes of plutonium per GW of electrical power (GIF 2021a).

Conclusion: Fuel supply and waste disposal

From today’s perspective, the fuels to be used for GFR in the future and the associated fuel cycle are still largely open, with a variety of fuels still being discussed. Against this background, an assessment can currently only be made against the declared goal of the developments and an assessment of the feasibility of these goals.

Accordingly, only natural or depleted uranium would be used to supply a GFR in equilibrium operation. The fact that extensive uranium extraction and the enrichment of uranium as well as the production of new fissile material for the plant’s own needs can thus be dispensed with in principle represents an advantage over today’s LWR. The intrinsic potential of GFR to enable a high breeding rate due to a particularly hard neutron spectrum and thus the additional production of fissile material for further plants is not pursued in the current GIF concepts.

At the same time, the desired closed fuel cycle requires facilities for reprocessing and fuel element production using large quantities of fissile materials. In particular for advanced fuel concepts based on ceramic fuel, neither large-scale production nor reprocessing processes exist to date. Their realisability is therefore largely open at present. This represents a disadvantage of GFR compared to LWR with an open fuel cycle.

If the goal of complete recycling of all actinides from the spent fuel of GFR could be achieved, this would result in a lower input of actinides into a repository. However, the extent to which this goal can be achieved in the medium and long term cannot be assessed at present. It is also currently unclear to what extent the advanced fuels under discussion will result in further changes to the waste inventories that are important for the long-term safety of a repository.

4.3.7 Proliferation risks

The reactor core of GFR generally exhibits higher neutron leakage than the reactor cores of liquid metal-cooled reactors. GFR therefore require a higher proportion of fissile material in the fuel compared to SFR or LFR (Tsvetkov 2016).

GIF is striving for a high burnup of the fuel so that the actinides (especially plutonium) produced in the spent fuel have an unfavourable isotopic composition and diversion during reprocessing is unattractive (GIF 2021a).

(Hatala 2021) , with regard to the planned construction of an experimental reactor (ALLEGRO) in Eastern Europe for the utilisation of MOX fuel, points to considerable uncertainties due to proliferation concerns. An initial reactor core based on enriched uranium (HALEU) is therefore also being examined for ALLEGRO.

A more detailed discussion of the various proliferation aspects of different reactor concepts within the GFR technology line can be found in (GIF 2022b).

Conclusion: Proliferation risks

Also with regard to proliferation risks, it should first be noted that the fuels to be used for GFR in the future and the associated fuel cycle are still largely open from today’s perspective. Against this background, an assessment at the level of the technology line can currently only be made against the declared goal of the developments and an assessment of the realisability of these goals.

A possible abandonment of uranium enrichment facilities represents an advantage over present-day LWR. This is offset by the disadvantage of reprocessing, which is associated with increased proliferation risks and requires greater effort in terms of fissile material control and monitoring measures. If the goal of complete recycling of all actinides from the spent fuel of GFR could be achieved, the proliferation risks are reduced compared to a fuel cycle with separation of pure plutonium.

4.3.8 Costs

(Tsvetkov 2016) sees significant economic efficiency potential for GFR in reactor concepts that can be operated at a high power density without an intermediate cooling circuit.

For reactor concepts with a lower thermal output power, increased safety through the possibility of passive core cooling is conceivable in principle, although this would result in considerable economic disadvantages for such reactor concepts (Tsvetkov 2016).

Furthermore, (Tsvetkov 2016) does not consider the economic viability of GFR with a closed fuel cycle to be realisable in the near future. Only towards the end of the 21st or beginning of the 22nd century could there be an economic incentive for such fuel cycles under conditions of a shortage of fissile materials. By providing high-temperature process heat, (Tsvetkov 2016) sees the potential to bring GFR closer to economic viability.

Conclusion: Costs

The developers of GFR point to improved economics due to the possible higher efficiency of GFR reactors as a major advantage of GFR development. This would also apply in particular if they could be used for high-temperature applications (process heat, hydrogen production). However, the feasibility of this is currently still largely unclear.

No reliable statements can currently be made regarding the investment requirements, necessary construction times, operating costs, service life and capacity utilisation of GFR.

The risks for investors are high, as there is no comparable experience to date.

4.4 Molten-salt reactors (MSR)

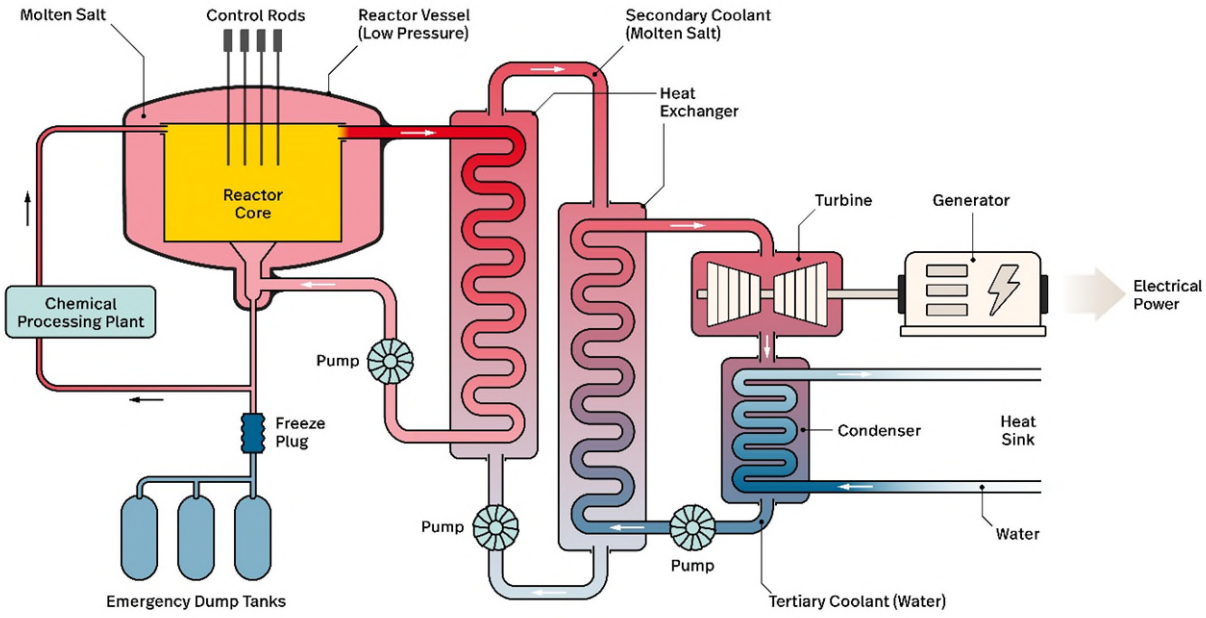
4.4.1 System description

The original idea of a molten salt reactor is to dissolve fissile and fertile materials in liquid fluoride or chloride salt so that fissioning and breeding of new fissile material occurs in the liquid fuel salt. In principle, a liquid fuel allows flexible handling of nuclear and chemical processes by continuously changing the fuel composition without having to shut down the reactor for loading and unloading. This flexibility would enable optimisation strategies, e.g. in terms of fuel utilization. This should also lead to a high operational availability of the reactor.

Reactor system

The molten salt first passes through an active core area in which the chain reaction for energy generation is maintained, see Figure 4-4. Depending on the specific design, the chain reaction in the reactor can be controlled by control rods or stabilised independently due to the thermal feedback effects of the molten salt (as planned for the MSFR).

Figure 4-4: Conceptual diagram of a molten salt reactor (MSR)



Source: Own illustration

Outside the core area, the molten salt is subcritical due to the geometric conditions, so that only the radioactive decay heat (decay power) is released here. The heated molten salt is pumped through a heat exchanger, where the energy is transferred to a secondary cooling circuit. The molten salt in the primary cooling circuit is circulated once within a few seconds. A molten salt is also used in the secondary cooling circuit, which transfers the energy to a third cooling circuit via another heat exchanger. Water can be vaporised in this circuit, for example, to drive a turbine and thus the

generator to produce electricity. Alternatively, helium cooling circuits on the secondary side are also being discussed. The remaining waste heat is released to an external heat sink, such as a river or the sea. Due to the high operating temperature, some concepts also envisage an application for the industrial production of hydrogen.

Molten salt (operating media)

In the most advanced MSR concepts, the fuel is dissolved in molten salt. The molten salt is typically a fluoride salt in which both the fuel and the resulting fission products can dissolve. Alternatively, chloride salts are also discussed, as chloride salt enables a high solubility of uranium and/or transuranium elements and, as a liquid fuel, has a harder neutron spectrum. However, the core of a reactor with chloride salt is considerably larger than with fluoride salt. The thermomechanical properties also differ (Noori-kalkhoran et al. 2022). Lithium, sodium or zirconium fluorides are often favoured as salts, especially the specific salt systems LiF-BeF2 (FLiBe - Flibe) and LiF-NaF-KF (FLiNaK - Flinak). However, the exact chemical composition of a salt for MSR depends on various parameters (neutron spectrum, fuel composition, operating temperatures, structural materials) and therefore on the detailed reactor concept (GIF 2002).

Fuel

The original idea of MSR involved liquid fuels. However, due to the technological challenges, MSR concepts with solid fuel are also being discussed. The molten salt is then only used to cool a reactor core with solid fuel elements containing the fuel, usually in the form of TRISO particles (e.g. TMSR-SF, AHTR, HERMES). There are also concepts in which different molten salts are used for the fuel and for cooling.

With liquid fuels, typical problems associated with the use of fuel elements due to thermal stresses and the accumulation of defects in the fuel, which are associated with a change in performance, are avoided. In terms of safety, MSR with liquid fuels have low operating pressures as the coolant does not boil. Yet the use of liquid fuel means the loss of two essential safety barriers (fuel matrix, fuel cladding) of typical reactor concepts with solid fuel. However, the liquid fuel can be discharged into previously prepared dump tanks with a geometry that makes criticality impossible. However, heat removal must be ensured independently.

The advantage of liquid concepts, in which the fuel is dissolved in the molten salt, is the possibility of integrating pyrochemical reprocessing steps on site and, in principle, the flexible use of different fissile material combinations.

Plutonium, uranium and thorium (U-235/Pu, Th/U-233) are usually used as fissile and breeding materials. The use of transuranic elements (TRU), i.e. plutonium together with minor actinides (Am, Cm), is also being discussed. A thorium-uranium mixture is being discussed as part of advanced fuel cycles.

Thorium-based reactors have been developed since the 1950s. Since thorium itself is not fissile, the reactor must first be started with another fissile material and then re-breed the fissile material uranium-233 during operation. One difference of a Th/U-233 fuel is that orders of magnitude less transuranic elements are produced than in other fuel cycles with uranium or MOX as fuel.

Fuel cycle and chemical fuel processing

Most MSR concepts provide for chemical processing of the fuel. A distinction must be made between chemical processing of the fuel in the sense of filtration and separation of fission products (fission gases but also solids), i.e. clean up or purification of the fuel, and additional pyrochemical treatment to remove and reuse actinides, i.e. classic reprocessing. Some concepts do not provide for either process step (e.g. some concepts with solid fuels). MSR concepts with liquid fuel always include chemical processing to clean the molten fuel salt during operation. In many MSR concepts, reprocessing to extract fissile materials is part of the intended fuel cycle, e.g. for thorium fuels.

Reprocessing can be carried out continuously or in batch operation, i.e. batches of fuel are removed at intervals. In continuous reprocessing, a partial stream of the molten salt is continuously removed in order to extract various fission products in an on-site reprocessing plant and replace the spent fuel, but also, for example, to remove lanthanides and zirconium from the melt, which would negatively affect the solubility of actinides. The purified molten salt is then fed back into the reactor, where new fissile material is also added if necessary. Reprocessing takes place within the plant (onsite) (e.g. Transatomic, Flibe). Other concepts envisage a centralised reprocessing plant for several reactors at different sites (e.g. ThorCon). (Transatomic 2016; IAEA 2016e; 2020e).

To remove gaseous fission products and certain precious metals from the reactor, an inert gas such as helium is also to be continuously added to the cooling circuit, which flows through the molten salt and is then removed from the upper part of the reactor and fed to the on-site reprocessing process (IRSN 2015). MSR with a fast neutron spectrum have the advantage that the requirement for separating fission products for reactivity reasons is lower than for thermal MSR.

Coolant

A melt of fluoride or chloride salt is used as the primary coolant in an MSR (LeBlanc 2010; Riley et al. 2019). The coolant can contain the fuel at the same time or different molten salts are provided for the fuel and the cooling or the coolant is used to cool solid fuels. Instead of a molten salt, other coolants (such as liquid metals) can be used, but pure breeding cycles can also be used.

Spectrum (moderator)

In principle, an MSR can be operated with both thermal and fast neutron spectrum. If a thermal spectrum is planned, neutron moderation takes place in the core area, for which graphite is to be used as a moderator.

In addition to graphite, several other moderator materials have been proposed, such as metal hydrides, beryllium, but also water or heavy water, each of which has its own challenges. All moderator materials must be provided with cladding materials to separate the moderator materials from the molten salt. Thermal reactors have safety advantages such as a very slow thermal reaction, a longer neutron lifetime and the practical absence of criticality problems outside the reactor core due to the low accumulation of fissile material in the fuel salt (Greenspan 2021).

In thermal systems, the reactor vessel is also irradiated with a significantly lower neutron flux compared to fast reactor systems.

In MSR with a fast neutron spectrum (MSFR), there is no moderator in the core area. Fast MSR have a strongly negative reactivity coefficient due to the density and Doppler effect, which distinguishes them from fast reactors such as SFR and LFR with solid fuel. The density effect is particularly relevant for reactors in which the molten salt also serves as a coolant.

A thermal spectrum also allows a simpler scheme for actinide burnup without additional reprocessing when using low-enriched uranium or thorium. However, fast systems with reprocessing are more suitable for transmutation due to the neutron balance. The neutron balance is also favoured by the absence of structural materials in the homogeneous liquid reactor core, as well as the flexibility to change the fuel composition without changing the core configuration (Greenspan 2021).

Temperature

MSR have a high operating temperature that could allow other industrial applications in addition to the generation of electricity (e.g. hydrogen production). The melting temperature of typical salts discussed for MSR is around 500 °C, the boiling point in the range of 1400 °C, operating temperatures should be in the range of 600-700 °C (GIF 2002; Riley et al. 2019). The upper limit is approx. 800 °C due to the temperature compatibility of the structural materials discussed today (GIF 2002).

Pressure

Since the coolant (molten salt) does not boil at the operating temperatures in the MSR, the primary cooling circuit of an MSR is not under high pressure, as is the case with water-cooled reactors.

Construction materials

The use of a hot molten salt places special demands on the structural materials due to corrosion, especially for fast systems with a fast neutron spectrum. The structural materials are usually special nickel-molybdenum steel alloys with a high nickel and low chromium content of only 6-7%.

4.4.2 Overview of reactor concepts in the technology line

MSR differ in terms of the fissile materials used, the molten salt used, the neutron spectrum and the design. Concepts with solid and liquid fuels as well as reactor concepts with thermal and fast neutrons have been developed.

Table 4-7 provides an overview of several reactor concepts.

Table 4-7: Thermal and fast MSR: Examples of reactor concepts

Fuel / moderator	Spectrum	Fissile material	Molten salt	Example reactor concept
Solid / graphite	thermal	U	Fluoride	AHTR, HERMES, KP-FHR, Mk1 PB-FHR, SmAHTR, TMSR-SF
Solid / -	fast	TRU	Chloride	SSR-W
Liquid / graphite	thermal	U	Fluoride	IMSR-400, ThorCon, MSR-FUJI, MSTW, LFTR, TAP – MSR, TMSR-LF
Liquid / graphite	thermal	Th-U233	Fluoride	LFTR, TMSR
Liquid / -	fast	U-Pu	Chloride	MCFR, MC-SFR
Liquid / -	fast	U-Pu	Fluoride	U-Pu FMSR,
Liquid / -	fast	TRU	Chloride	DFR
Liquid / -	fast	TRU	Fluoride	MOSART
Liquid / -	fast	Th-U233	Fluoride	IMSBR, MSFR, MOSART

Source: (WNA 2021b; IAEA 2023d; GIF 2021a). The table contains some of the reactor concepts categorised as SMR, see for more detail (Oeko-Institut e.V.; WIP; PhB 2021)

Due to the wide variety of reactor concepts, two reactor concepts are selected as representatives for a more detailed presentation of the development paths of MSR:

- **Thermal MSR:** The development of thermal MSR is most advanced due to the avoidance of material stresses by fast neutrons and the use of fluoride salts. Graphite, which is directly chemically compatible with fluoride salt, is usually used as a moderator. The LFTR is discussed as a specific reactor concept in Chapter 5.5.

  Thermal plant concepts are much less efficient than fast plant concepts when breeding new fissile material. However, the advantage of thermal concepts is that, due to the low energy density in the reactor core itself, components for reactivity control can be accommodated and shielding or moderation is possible to protect components in order to avoid neutron embrittlement. The majority of MSR currently under development have a thermal neutron spectrum and are graphite-moderated. MSR with a thermal spectrum are the most advanced in terms of development status.

- **Fast MSR:** Fast neutron spectrum MSR are better suited for both breeding fresh fuel and transmutation of transuranic elements. Fast MSR could also utilise molten chloride salts. The MCFR is discussed as a specific reactor concept in Chapter 5.6.

  Fast MSR have a better neutron balance for the burnup of actinides than thermal reactors. They share this property with all fast reactor models when transuranic elements (TRU) are used in the fuel or when plutonium is bred from uranium-238 (U-Pu). However, MSR also have the option of using Th-U-233 as fuel, which means that far fewer actinides are produced than when using U-Pu fuel. And compared to thermal MSR with Th-U-233, fewer actinides are formed due to the neutron spectrum in fast reactors and the lower neutron capture in U-233. However, the hard spectrum leads to greater radiation damage in all MSR, so most plant concepts do not provide for an internal reactor vessel structure to regulate reactivity. Monitoring electronics must be heavily shielded.

4.4.3 Historical developments

Historical overviews of MSR developments can be found in (LeBlanc 2010; EPFL 2018; Oeko-Institut e.V. 2017; Greenspan 2021).

The concept of MSR was originally investigated in the U.S. (Aircraft Nuclear Propulsion Program, ANP) at the end of the 1940s due to their high power density and the associated potentially compact design for aircraft propulsion. The first experimental plant was the Aircraft Reactor Experiment (ARE) in 1954 (AEC 1972). This was a thermal reactor with a thermal output of 2.5 MW, beryllium oxide as a moderator and a molten fluoride salt. It was operated for a total of approx. 100 hours at 860 °C (IRSN 2015). Further research was carried out at the Oak Ridge National Laboratory (ORNL) in the U.S. from 1957.

ORNL investigated large graphite-moderated MSR concepts with a U-Th fuel cycle, both for transmutation and for breeding, during the 1960s. Several concepts were proposed: first, a reactor concept with two separate molten salts – one molten salt with fissile fuel and a second, separate molten salt with fertile material (uranium or thorium) to breed new fissile material (AEC 1972). The bred uranium-233 is transferred to the fuel salt by means of fluorination. Fluorination is carried out by bubbling fluorine gas through the fertile molten salt to convert the UF_4 produced into volatile UF_6. The gaseous UF_6 is then converted back into solid LiF-UF_4 pellets and added to the molten fuel salt. The same process is still used today in the MSR process scheme (Greenspan 2021).

The main advantage of an MSR with two molten salts is the reduced initial fissile material inventory and the simplification of the purification processes. Since there is no thorium in the fuel liquid, it is much easier to separate protactinium or soluble lanthanide elements, which would otherwise reduce the breeding of fissile material due to neutron absorption in the thermal spectrum (Greenspan 2021).

The disadvantage of the MSR with two molten salts, however, is the necessary barrier between the core and the fertile zone in a more complex reactor vessel and the additional problems this causes due to material damage caused by radiation and corrosion.

The “Molten Salt Reactor Experiment” (MSRE) was also carried out from 1960 onwards, due to the simpler design with a single molten salt with dissolved fissile material in it. The reactor had a thermal output of 8 MW at 700 °C and was operated for around 18,000 hours between 1965 and 1969. Uranium-235 was initially used as the fissile material, but from 1968 uranium-233 was also used, and finally plutonium (Greenspan 2021). The MSRE did not use any fertile material and therefore did not breed any new fissile material. The molten salt was processed in non-continuous (batch) operation and therefore largely independently of the actual reactor operation (AEC 1972).

Since the shutdown in 1969, the reactor building has remained in an unchanged state;the molten salt is still in the reactor in solidified form , as the matter of disposal not yet been clarified (IRSN 2015).

ORNL continued the development of MSR in a small programme until the early 1980s and propagated another design, the so-called denatured MSR, to counter proliferation concerns. In this design, reprocessing was dispensed with and only krypton and xenon gas was removed from the fuel; the reactor was planned with LEU fuel with an enrichment of 19.75%. The service life was limited to 30 years due to the graphite service life, which is why the concept was called the “30-Year Once-Through Design” (Greenspan 2021).

A summary of the development status achieved with the MSRE by the U.S. Atomic Energy Commission can be found in (AEC 1972). Accordingly, the MSR concept was in an early viability phase at the time. By the time of the report, around USD 150 million had been invested in the

development of civil and military concepts based on MSR technologies since the 1940s. The necessary further development costs for an MSR system were estimated at around USD 2 billion at the time.

The possibility of breeding fissile uranium-233 from thorium in thermal reactor concepts was cited in (AEC 1972) as a major advantage of the MSR concept In addition, the particular advantages of an MSR at that time were considered to be that

- the production, transport and reprocessing of solid fuels could be avoided by using a liquid fuel in combination with on-site reprocessing,
- the estimated costs of the fuel cycle were low, although significant uncertainties were seen in the actual costs of on-site reprocessing,
- the reactor could possibly have relevant safety advantages over other reactor concepts, although no conclusive assessments can be made in this regard due to the early development phase,
- the reactor could be operated with high efficiency in power generation due to high temperatures.

On the other hand, (AEC 1972) also identified significant unanswered questions in connection with this reactor concept:

- Continuous on-site reprocessing of the molten salt: The necessary separation of fission products and protactinium-233, the outgassing of gaseous fission products and the control of salt chemistry to prevent corrosion and deposits in the cooling circuits had not yet been developed.
- Suitable structural materials for the reactor and for on-site reprocessing of the molten salt: The nickel-based steel alloy Hastelloy-N was used as the structural material, which behaved satisfactorily during the four years of experimentation at the MSRE. However, examinations of the metal after the end of the experiment revealed evidence of corrosion problems under conditions of long-term contact between the metal surface and the fission products remaining in the molten salt. Furthermore, the Hatelloy-N used until then was not sufficiently resistant to long-term neutron irradiation. According to (Greenspan 2021), however, later work showed that the corrosion problems could be caused and controlled by the fission product tellurium and that neutron embrittlement could also be controlled by modified alloys.
- Development of a suitable salt composition: The salt used should not have a melting point that is too high so the molten salt does not solidify during operation or in the event of incidents, particularly in the heat exchangers, resulting in a loss of cooling. Furthermore, the lithium contained in the salt might have to consist of enriched lithium-7 in order to reduce the neutron capture in the naturally occurring lithium-6 isotope. Finally, the properties of the irradiated salt in interaction with the fission products it contains were largely open.
- Development of a suitable moderator graphite: The graphite used until then becomes structurally unstable under prolonged neutron irradiation. Furthermore, the graphite would have to be suitably sealed to prevent the diffusion of gaseous fission products and tritium.
- Methods for tritium control: When fluoride salts are used, especially lithium-based ones, neutron reactions produce larger amounts of the radioactive hydrogen isotope tritium. The structural materials used in the MSRE were unsuitable for the sufficient retention of tritium. (Greenspan 2021) points out that after much development, a coolant ($NaF\text{-}NaBF_4$) was later found to capture and remove the tritium and isolate it in a gas purification system.

- Reactor systems and auxiliary equipment: The components used, such as salt circulation pumps, valves and heat exchangers, as well as control and control rods, would need to be scaled up from the experimental boundary conditions in the MSRE to a commercial reactor system (taking into account the need for maintenance requirements). This represents a significant technological development step. The concept of the dump tank (into which the molten salt is drained in the event of reactor malfunctions or for maintenance of the primary cooling circuit) should also be adapted for a high-power system.
- Maintenance technologies: Unlike in reactor concepts with solid fuel, since the radioactive elements are not enclosed in fuel elements but dissolved directly in the coolant, the entire primary cooling circuit is highly radioactively contaminated. Therefore, suitable procedures for the automatic and remote inspection and repair of components in the primary cooling circuit are required. Difficulties in maintaining components of the primary cooling circuit could greatly reduce the availability of a reactor system and thus jeopardise its commercial usability.
- Calculation programs and standards: The calculation programs used for verification would have to be validated for the boundary conditions of an MSR; furthermore, the applied rules and guidelines for the approval of materials and components would have to be adapted to the specific requirements of MSR.
- Safety aspects: An MSR has fundamental safety advantages over other reactor concepts. These include the low operating pressure and the large distance to the boiling point of the coolant (molten salt) as well as the good solubility of important fission products such as iodine and strontium in the molten salt. Due to the on-site reprocessing, the excess reactivity and the total radioactive inventory in the reactor cooling circuit are also low. However, the fact that the entire primary cooling circuit and the downstream systems such as salt storage are heavily radioactively contaminated is considered a disadvantage. The continuous removal of decay heat from the dump tanks must also be ensured with very high reliability under all operating conditions. The extent to which this system would actually achieve safety-related advantages over other reactor concepts could therefore not be assessed at the time.

There was also a need for further research into possible interactions between the neutron-physical and thermo-hydraulic properties of the reactor. For example, (AEC 1972) points out that in the initial phase of operation of the MSRE with uranium-233 there were unexpected and unexplained fluctuations in performance. One possible explanation was seen in the interaction between the flow of helium gas through the reactor core, which has to be added to the molten salt to remove the gaseous fission products.

MSR research and development was also carried out in China, France and above all in Russia (Greenspan 2021). According to (WNA 2021b), the concept for a lead-cooled fast MSR with a plutonium fuel in a molten chloride salt was investigated in Great Britain in 1968-1973. In Switzerland, research projects relating to MSR were carried out between 1973 and 1980 (Křepel 2017). This work focussed on molten salt reactors with a fast neutron spectrum, in which a uranium-plutonium mixture was to be used as fuel in a molten chloride salt. The particular advantages of this reactor concept were identified as the continuous separation of volatile fission products and thus – in combination with low maximum temperatures under accident conditions – the significantly reduced source terms in the event of severe accidents compared to light-water reactors. The disadvantages identified were that the molten salt reactor technology is not yet sufficiently developed, the corrosion of structural materials is a particular problem and this type of reactor has a particularly high inventory of plutonium (Eidgenössisches Institut für Reaktorforschung 1980).

4.4.4 Current developments

A number of current overviews of MSR systems under development are available (Oeko-Institut e.V. 2017; WNA 2021b; Dolan 2017; EPFL 2018; GRS 2020; IAEA 2023d).

Most reactor concepts with liquid fuel currently under development use lithium and/or sodium fluorides as molten salt. The majority of designs rely on a thermal neutron spectrum with graphite as a moderator. Transatomic plans to use zirconium hydride as a moderator for the TAP-MSR.

(Dolan 2017) in particular presents some systems in more detail – some of them by the developers themselves: ThorCon, IMSR, Thorium Molten Salt Reactor – TMSR, Stable Salt Fast Reactor – SSFR, Transatomic Power, Copenhagen Waste Burner, MSTW, Dual Fluid Reactor – DFR (Dai 2017; LeBlanc and Rodenburg 2017; Jorgensen 2017; Kloosterman 2017; Scott 2017; Robertson et al. 2017; Pedersen 2017; Schønfeldt and Klinkby 2017; Huke et al. 2017) . In addition, more exotic concepts such as fusion hybrid reactors (Furukawa et al. 2017) or accelerator-driven systems (Velikhov 2017) with molten salt utilisation are also discussed. Other global activities are also presented in (GIF 2021a; 2020a; 2018a; Dolan 2017; Greenspan 2021; WNA 2021b).

In (GIF 2002), MSR were still primarily discussed as thermal reactors with the aim of reducing actinide stocks. According to (GIF 2014), however, since 2005 work has been carried out on MSR concepts with a fast neutron spectrum (MSFR) in addition to the FHR.

In addition to government research programmes, a large number of smaller development companies have been established in recent years. (EPD 2015; WNA 2021b). Most of these concepts also have the essential characteristics of small, modular reactors. According to (Oeko-Institut e.V.; WIP; PhB 2021), there are currently 18 plant concepts among the SMR under discussion worldwide that can be categorised as molten salt reactors.

Thermal MSR with solid fuel using fluoride salt

In the U.S., research into MSR has been revitalised by ORNL. The development of the Advanced High Temperature Reactor (AHTR), also known as the Fluoride High Temperature Reactor (FHR), is planned. The large reactor with a thermal output of 2400-3600 MW is to use TRISO fuel in a graphite matrix, similar to the Gas Turbine Modular Helium Reactor (GT-MHR) with prismatic fuel (see Chapter 5.10.6). Flibe is planned as the coolant, with an operating temperature of 700 °C. A small version of the reactor is the small modular SmAHTR with a thermal output of 125 MW, which is intended to be transportable and have a refuelling interval of two and a half to four years. The primary circuit is to be cooled with Flibe, while the secondary circuit will use Flinak (WNA 2021b). The AHTR is one of the two reference concepts of the GIF and is similar to the VHTR concepts. However, the use of molten salt as a coolant allows significantly higher power densities and promises thermal outputs of up to 4 GW without the need for active safety systems (WNA 2021b).

Another development strand of a molten salt cooled pebble-bed reactor is the Mark 1 Pebble Bed FHR (Mk1 PB-FHR), a pre-conceptual design developed in 2014 by a consortium of ORNL, Westinghouse and the University of California, Berkley. The reactor consists of modules with a thermal output of 236 MW and is intended to generate a basic electrical output of 100 MW with a peak output of 240 MW through the simultaneous burnup of additional gas. The fuel has a much smaller diameter with 3 cm spheres than the typical 6 cm in other HTR concepts (WNA 2021b).

The KP-FHR is being conceived by Kairos Power in the U.S. and is expected to have a thermal output of 320 MW and operate at an operating temperature of 650 °C using Flibe. Kairos Power is working closely with ORNL. The reactor is intended to use uranium fuel with 20% enrichment in TRISO particles and the fuel is to be supplied during operation. The secondary circuit is to be operated with sodium nitrate, as used in solar thermal plants. Passive shutdown and heat removal systems are planned. The prototype is the Hermes Reduced-Scale Test Reactor with a thermal output of 50 MW and no electricity production. The reactor was selected by the DoE in December 2020 for a USD 629 million programme over seven years. The DoE’s share is USD 303 million from the Advanced Reactor Demonstration Program, with the remainder being borne by the developer. In May 2021, the Tennessee Valley Authority pledged support for the development, operation and licensing of the Hermes project (WNA 2021b).

China is pursuing a dual programme for the development of MSR, with a focus on solid fuel concepts (WNA 2021b). The fuel consists of separate U and Th fuels in TRISO particles in prismatic fuel elements or in blocks or in spheres. The programme to develop the Thorium Molten Salt Reactor (TMSR) was initiated by the Chinese Academy of Sciences and has been carried out at the Shanghai Institute for Applied Physics since 2011. The original plan was to build a pilot plant with a thermal output of 10 MW. The TMSR-SF1 was to be operated with 17% enriched TRISO fuel in 6 cm graphite fuel spheres, similar to the fuel for the HTR-PM (see Chapter 5.8). Cooling takes place with Flibe at an operating temperature of 630 °C. However, the project was not continued and was replaced by the construction of a simulator (TMSR-SF0). The simulator has an electric heater to reach up to 650 °C and a Flinak primary and secondary circuit. Development is to continue with a pebble-bed demonstration reactor (TMSR-SF2) with a thermal capacity of 100 MW with an open fuel cycle starting in 2025 (WNA 2021b).

Thermal MSR with liquid fuel using fluoride salt

In the U.S., work on thermal MSR concepts with liquid fuel is being funded for the first time in 40 years (DOE 2017b). The American Nuclear Society (ANS), in cooperation with the Institute of Electrical and Electronics Engineers (IEEE) and the American Society of Mechanical Engineers (ASME), is currently developing new standards for MSR concepts in preparation for the future licensing of MSR (Holcomb 2017b). The NRC is also currently preparing to have sufficient expertise and an applicable set of rules for future applications for the approval of new reactor concepts (NRC 2016).

A graphite-moderated concept, the IMSR-400 (integral molten salt reactor), is being developed by Terrestrial Energy from Canada. The reactor was conceived in three sizes: 80 MW, 300 MW and 600 MW of thermal power, but the current concept is for 440 MW. In this concept, the fuel is dissolved in a molten fluoride salt at an operating temperature of 600-700 °C. Initially, no lithium or beryllium is planned in the molten salt, but it is possible to use flibe at a later stage. The coolant in the secondary circuit is ZrF_4-KF. The integral molten salt reactor is so called because it is integrated into a compact, sealed and replaceable core reactor unit, the so-called IMSR core unit (WNA 2021b; Dolan 2017). The entire core unit module can be removed for replacement after an operating period of seven years and replaced completely (Greenspan 2021). The concept is currently undergoing a preliminary review for approval in the U.S. and Canada. At the end of 2018, the company completed the first phase of a three-stage review process for subsequent approval by the Canadian regulator (WNA 2021b). In October 2020, the Canadian government pledged financial support of 20 million Canadian dollars for Terrestrial Energy and its IMSR with a planned electrical capacity of 192 MW (GIF 2021a, p. 9).

The company Martingale is designing the graphite-moderated ThorCon MSR. ThorCon is a simple scale-up of the US MSRE at Oak Ridge National Laboratory. BeF_2-NaF at an operating temperature of 700 °C is to be used as the fuel salt. Lithium is to be avoided for cost reasons. Sodium-beryllium is also to be used as the coolant in the secondary circuit. ThorCon only uses commercially available components and materials. The molten salt with dissolved thorium and uranium fluorides (enriched to 19.7%) is also removed and replaced in an eight-year cycle,(Dolan 2017) without reprocessing or separation of fission products during operation. ThorCon is divided into two modules with an electrical output of 250 MW and a supercritical 500 MW turbo-generator. Each module has two sealed reactor vessels (“sealed cans”). Every four years, a vessel that has cooled down during this time is removed and replaced with a new vessel (Greenspan 2021). A ThorCon pilot plant would be similar to mini Fuji.

ThorCon has been working with the Indonesian government and in 2019 the Indonesian Ministry of Energy successfully completed a study on the safety, economic efficiency and grid impact of the prototype with 500 MW of electrical power. Phase 1 is the construction and testing with gradual commissioning, which ends with a type approval for future power plants. Phase 2 is the factory production of ThorCon power plants (Greenspan 2021).

In the Japanese MSR programme FUJI, a graphite-moderated reactor with a thermal output of 450 MW is being conceived with a ThF_4-UF_4 fuel salt and Flibe as coolant at an operating temperature of 700 °C. Higher electrical reactor outputs of up to 1000 MW or SMR mini variants with 10 MW (mini Fuji) are also planned. There are also designs for the use of plutonium and TRU fuel. The reactor is being conceived by the Russian-Japanese-U.S. consortium International Thorium Molten Salt Forum (ITMSF) and is based on ORNL’s MSRE concept. Thorium Tech Solutions Inc (TTS) is planning to commercialise the Fuji concept in cooperation with the Halden test reactor in Norway (WNA 2021b).

The startup Transatomic Power discontinued its operation and development of the TAP-MSR in 2018 due to admitted errors in the plant concept (GRS 2020). The company made false claims about the achievable burnup of actinides (WNA 2021b).

The company Seaborg Technologies from Denmark was founded in 2015 and is developing a graphite-moderated pilot reactor with a thermal capacity of 40 MW and the goal of a commercial module with 250 MW thermal capacity. The Molten Salt Thermal Wasteburner (MSTW) or Compact Molten Salt Reactor (CMSR) is planned to use lithium-7 as fuel salt, with uranium as fuel at the beginning and later the use of plutonium and actinides from spent fuel from LWR. The fission gases are intended to be separated during operation. Flinak at a temperature of 700 °C is planned as a secondary coolant. Seaborg Technologies received funding from the Danish government in 2017, Denmark’s first investment in nuclear technologies since 1985. At the end of 2020, the American Bureau of Shipping confirmed the viability in principle of using the reactor on a barge. Seaborg wanted to build the first complete prototype on a barge in 2025 (WNA 2021b). According to the company’s website, the timeline has since shifted to 2028. The company also states on its website that it will receive funding from an EU programme for innovative companies, the EIC Accelerator, in June 2022.[161]

As part of its dual programme for solid and liquid fuel, China is also developing a closed Th-U fuel cycle with electrometallurgical pyrochemical reprocessing. The aim of the development is initially a pilot plant with a thermal capacity of 2 MW (TMSR-LF1), followed by an experimental reactor (TMSR-

[161] https://www.seaborg.com/press-release-eic-grant

LF2) by 2025 and then a demonstration reactor (TMSR-LF3) with a thermal capacity of 100 MW with complete reprocessing by around 2035. A 1 GW demonstration reactor is to follow later. The TMSR-LF1 uses enriched uranium (UF_4) with up to 20% enrichment and around 50 kg of thorium; flibe with 99.95% enriched Li-7 is to be used as molten salt. The project will start in batch operation, i.e. the fuel will be completely replaced every 5-8 years for reprocessing and separation of the fission gases and minor actinides. During operation, the supply of fresh fuel and the separation of gaseous fission products will be limited. Continuous process control is planned for later operation, as is the continuous transfer from an initial share of 20% thorium in power generation to 80% later on.

The TMSR-LF development line is about 10 years behind the development of the TMSR-SF development line with TRISO fuel. The development is being carried out by the TMSR Centre Shanghai Institute for Applied Physics (SINAP) of the Chinese Academy for Sciences. With this programme, China is striving for technological leadership in the development of thorium MSR reactors. The DoE is collaborating with the China Academy of Sciences in the programme with a start-up budget of USD 350 million. An expenditure of USD 3.3 billion is planned (WNA 2021b). The site for the 2 MW TMSR-LF1 test reactor was selected in 2017 and is located near Wuwei (Greenspan 2021). Construction began in 2018 and is scheduled for completion in 2021; a start-up licence was granted in August 2022.[162]

Fast MSR with liquid fuel using fluoride salt

The U-Pu Fast Molten Salt Reactor (U-Pu FMSR) is being developed in Russia (Holcomb 2017a). The work is based on investigations by the Russian Kurchatov Institute into an actinide burner (MOlten Salt Actinide Recycler and Transmuter, MOSART) (IRSN 2019). Among other things, a breeding variant with thorium is also being investigated (GRS 2020).

Complete preliminary designs for MOSART are not yet available. A single-fluid configuration for the MOSART with a thermal power of 2400 MW has a homogeneous cylindrical core with a fluoride-based salt mixture. It is possible to design a critical homogeneous core fuelled only with TRU trifluorides from UOX or MOX LWR fuel. A special Ni-Mo alloy of the type HN80MTY with a low Cr concentration alloyed with 1% Al was proposed as the structural material for the MOSART fluoride salt container. The composition of the alloy is optimised for corrosion resistance (both in a low-oxygen gas atmosphere and in molten salt fluorides), and the irradiation resistance and mechanical properties at high temperatures have also been optimised (Greenspan 2021).

Parallel to the European EVOL project (“Evaluation and Viability of Liquid Fuel Fast Reactor System”, 2010-2013), work on fast MSR reactors was also carried out in Russia by Rosatom in the MARS project (Minor Actinide Recycling in molten Salt) (GIF 2014).

Since plutonium and minor actinides must be removed from the fuel solvent during reprocessing, the MOSART concept includes a system for removing TRUs from the fuel salt and reintroducing them into the purified salt. This plutonium recirculation loop has the advantage that americium and curium are also returned to the reactor fuel (low separation coefficients within the TRU group). According to (Greenspan 2021), since the higher actinides would always accompany the plutonium, this process would not produce element-pure fissile material that would be attractive for diversion.

[162] https://www.world-nuclear-news.org/Articles/Chinese-molten-salt-reactor-cleared-for-start-up

According to GIF, work has mainly been carried out on MSR concepts with a fast neutron spectrum since 2005, such as the molten salt fast reactor (MSFR) (GIF 2014).[163] The concept of the MSFR was practically only taken up again in the context of the GIF around 2000, see (GIF 2002). The European Union, France, Switzerland, Australia, Canada, the U.S. and Russia are involved in the development and China, Japan and Korea are active observers (GIF 2021a).[164] A “System Agreement” going beyond this, with the development of a specific reactor concept, as exists for other reactor concepts of the GIF, does not exist for the MSFR, but was prepared in 2019 (GIF 2021a). The individual countries participating in the GIF are responsible for the actual implementation of the research work.

The MSFR is to have a thermal output of 3000 MW (IRSN 2015). In addition to molten salt with fuel, the MSFR concept originally envisages another molten salt with thorium fluoride to breed new fissile material. The operating temperature should be 750 °C. Either uranium-233 directly or a mix of enriched uranium, plutonium and minor actinides from spent fuel elements can be used as fuel. The separation of fission products during operation and pyrochemical reprocessing is part of the reactor concept (Greenspan 2021). Since 2008, a thorium-based fuel has been planned as a reference for the MSFR (GIF 2014). In France, work on the MSFR is being carried out by the Centre national de la recherche scientifique (CNRS) in cooperation with Russia and other European partners,(IRSN 2015; IAEA 2016b) such as the Swiss PSI (GIF 2021a, p. 17).

According to (Greenspan 2021), the work on MOSART and MSFR is aimed at using physical and chemical methods, together with materials research, to design a fast spectrum MSR concept with an effective system configuration for the reactor core, for the reprocessing unit and for waste conditioning. This serves to increase confidence that MSR fulfil the GIF objectives for sustainability (Th breeder), non-proliferation (integrated fuel cycle, repeated reprocessing of actinides) and waste treatment (actinide transmutation). According to (Greenspan 2021), no hurdles that are insurmountable in principle have been identified so far, but almost all technologies still need to be tested and a safety approach or concept for this type of reactor still needs to be developed to analyse the potential benefits and test the viability of these innovative MSR concepts with a fast neutron spectrum.

Fast MSR with liquid fuel using chloride salt

The company Elysium in the U.S. is developing a fast MSR with molten chloride salt (MCSFR). The electrical output of the fast reactor is expected to be 50 MW for a small module or 200 MW for a large module. Up to six modules (1200 MW) are located in a secondary salt tank that provides passive cooling through a reactor pool auxiliary cooling system configuration. The modular components can be transported by land. The fuel contains plutonium for startup and there is flexibility to use other fissile and fertile fuels. The design does not include control rods. During normal operation, a portion of the molten salt fuel drains into a dump tank that serves as a continuous drain. The fuel salt is continuously pumped from the dump tank back into the reactor vessel and the salt-salt heat exchangers. Noble gases are removed during operation via systems in the dump tank. A prototype reactor with a thermal output of 10 MW is planned in the same size as a 125 MW demonstration reactor to enable a gradual increase in output as soon as safety data is available (ORNL 2021).

[163] More recently, however, a second AHTR reference reactor has been included by the GIF due to the dynamics.

[164] https://www.gen-4.org/gif/jcms/c_9343/system-arrangements-mou

The Stable Salt Reactor – Waste Burner (SSR-W) reactor concept from Moltex Energy is further advanced in planning. The reactor concept, conceived as an SMR with an electrical output of 300 MW, will also use reprocessed fuel from LWR. Although the fuel is also dissolved in a molten salt, it is enclosed in fuel rods and thus remains separate from another molten salt circuit intended for cooling the reactor. The fuel salt is an actinide chloride/potassium chloride eutectic in a liquid state at atmospheric pressure. In the SSR-W design, the liquid salt fuel mixture is contained in fuel elements that are very similar to those used in conventional liquid-metal-cooled fast reactor technology. The fuel elements are immersed in a pool of fuel-free liquid salt coolant. Each fuel element contains 271 tubes filled with fuel salt. The tubes are fitted with dipping bells at the top to control the release of fission gases. The fuel in the SSR-W consists of potassium chloride salt, plutonium and mixed lanthanide/actinide trichlorides (ORNL 2021).

The fuel for the first reactors of the SSR-W concept will come from converted spent oxide fuel from LWR. The cooling salt is circulated through the reactor core by three pumps, each of which is connected to its own heat removal circuit. In the event of a reactor shutdown and the failure of all active heat removal systems in the SSR-W, the decay heat is removed from the core into air cooling channels around the perimeter of the vessel, which are in constant operation. The heat transfer is sufficient for the passive removal of decay heat at higher temperatures under accident conditions. During power operation, the SSR-W is self-controlling due to the negative reactivity feedback of the temperature coefficient and the ability to continuously transport heat from the fuel tubes. The ability to shut down is ensured by the provision of shutdown rod bundles with boron-containing rods in some positions in the core (ORNL 2021).

Moltex has signed an agreement with New Brunswick Power (NB Power) for the construction of a first unit at Point Lepreau in Canada. In March 2021, the Canadian government announced that it would support the construction with an investment of C$50.5 million (WNA 2021b). Advanced Reactor Concepts has entered into an agreement with New Brunswick Power and Moltex Energy to jointly develop SMR concepts in Canada.[165] Moltex Energy has signed a co-operation agreement with the Canadian National Laboratory. Together with the University of New Brunswick (UNB), a facility for converting spent CANDU fuel into a salt compound (Oxide Nuclear WAste Reduction Demonstration, ONWARD) will be developed and tested.[166] The SSR-W is currently undergoing preliminary testing for approval by the Canadian regulator. The U.S. Jacobs Engineering Group is to operate a test facility for Moltex Energy to experimentally test the heat transfer from the fuel rods into the molten salt used for cooling and thus validate CFD calculations for the SSR-W.[167] Due to the lack of a moderator, the design of the SSR-W is relatively simple and compact. However, the company has also announced the development of a thermal graphite-cooled system, which is larger and more expensive, with a molten fluoride salt (WNA 2021b).

[165] https://www.world-nuclear-news.org/Articles/Canadian-collaborations-move-SMR-plans-forwards, last accessed 15.01.2023.

[166] https://www.world-nuclear-news.org/Articles/CNL-Moltex-collaborate-on-SMR-fuel-development, last accessed 15.01.2023.

[167] https://www.world-nuclear-news.org/Articles/Jacobs-to-assist-Moltex-in-development-of-SSR, last accessed 15.01.2023.

In Germany, a plant concept for a dual-fluid molten salt reactor (dual-fluid reactor - DFR) (Huke et al. 2015; He 2016; Wang 2017) was presented, which could also be used for transmutation (Institut für Festkörper-Kernphysik 2019). It is a lead-cooled reactor with a molten chlorine salt with uranium-plutonium as fuel, with the possibility of coupling the reactor directly with pyrochemical processes for reprocessing. The design is to be operated at an average temperature of 1000 °C. The company moved to Canada in 2021 (Dual Fluid 2021).

Since molten chloride salts have the disadvantage of low thermal conductivity of the salt, the salt in the DFR would have to be pumped quickly through the fuel cycle to create turbulence and thus improve heat transfer to the fuel cladding and intrasalt heat distribution. However, this in turn would make fission product separation more difficult. To remedy this situation, an actinoid metal fuel was therefore investigated in more recent DFR concepts. Uranium/chromium, uranium/manganese and thorium/iron could be considered as eutectics (DFR 2020). There has been no research and development into these eutectics as nuclear fuel to date. As a lead-cooled reactor with liquid metal fuel, the dual-fluid concept would then no longer be an MSR, but an LFR.

Further research programmes[168]

In the 2014 roadmap, the GIF states that the development of MSR requires an international commitment of several years and several billion euros, which has not yet taken place in this form. Only limited funds have been spent on the development of MSR to date (GIF 2014). GIF 2002 originally assumed that the viability phase for MSR would last until around 2013 (i.e. around 10 years) and the subsequent development phase until 2020 (i.e. a further 7 years). This would be followed by another demonstration phase of unspecified duration (GIF 2002). According to the updated planning from 2014, the GIF now estimates that the viability phase could start by around 2025 and the subsequent development phase between 2025 and 2030 (GIF 2014).

In France, the SWATH (Salt at Wall :Thermal excHanges) plant has gone into operation to investigate molten salt heat transfer and phase transitions. Tests on fission product separation were also carried out in the FFFER (Forced Fluoride Flow Experimental Research) test facility. Concepts for “freeze plugs” are also being developed at the FFFER facility (Chisholm et al. 2020). A consortium of mainly French institutions (CEA, CNRS, Orano) in cooperation with the JRC Karlsruhe oriented their research programme towards defining initial design studies for molten salt reactors. Concepts for the post-breeding of fuel and for plutonium burnup (Pu burner) or transmutation were considered. The studies investigated the reactor system (neutronics, materials, components) and the connected fuel cycle (molten salt, corrosion) using chemical and multiphysical simulation and modelling tools, among other things.

As part of the European Research Framework Programme (FP), the project “Evaluation and Viability of Liquid Fuel Fast Reactor System” (EVOL, 2010-2013) was carried out in FP7 (Deutscher Bundestag 2017) and the project SAMOFAR (SAfety features of the MOlten salt FAst Reactor) was completed in 2019 (GRS 2020). The EURATOM project SAMOSAFER (Safety Assessment for Fluid-fuel Energy Reactors), which is coordinated by TU Delft, will continue to develop new tools and models for analysing transients and severe accidents until 2023. The aim is to ensure that MSR will meet all expected regulatory requirements in 30 years (GIF 2021a). Experimental studies on the basic thermodynamic properties of fluoride salts were prepared at the Joint Research Centre in Karlsruhe and corresponding molten salts were synthesised and prepared for irradiation experiments. A new, improved density meter was also designed and tested in 2020 to improve the

[168] This chapter is largely based on (Oeko-Institut e.V. 2023).

density measurements of molten salts (GIF 2021a). The device can be used for molten chloride and molten fluoride salts. The database for molten salts (Joint Research Centre Molten Salt Database – JRCMSD) was also expanded to include other binary systems and relevant chloride systems.

A circuit for FLIBE was put into operation at the Řež research centre in the Czech Republic as part of a cooperation between the U.S. and the Czech Republic to support the development of a fluoride-cooled very-high-temperature reactor (GIF 2021d). According to (GIF 2021d), the “liquid salt test loop” (LSTL) at ORNL in the U.S. will also support the development. Specifically, the companies Terrapower, Kairos Power and Terrestrial Energy U.S. are mentioned according to (GIF 2021d). A complex of experimental facilities is also being built in China to support future research and development of MSR. There have also been SALIENT (SALt Irradiation ExperimeNT) irradiation experiments of small salt samples with Li, Th, U, Pu/F and alloys for structural materials at the Petten research reactor in the Netherlands (GIF 2021d). In Australia, the corrosion behaviour of steels and welded joints in particular has been investigated in recent years (GIF 2021a).

Switzerland is conducting various national research projects relating to MSR, in which the Swiss reactor operators are also participating financially (Pautz 2017). Significant work is being carried out at the Paul Scherrer Institute in the form of scientific qualification theses (MSc, PhD, PostDoc) (Křepel 2017).

4.4.5 Technical development status

With the exception of the large U.S. MSRE and the ARE, hardly any experience has been gained with MSR to date. Thermal reactor concepts are the most advanced and are currently being actively pursued in start-up companies. There is little practical experience to date with fast reactor concepts.

Since the 1940s, many molten salt reactor concepts have been proposed worldwide using different fuels (U, Pu, Th, etc.) and salt compositions (fluorides, chlorides, etc.). These projects have tried to find the optimal solution for the composition of the fuel salt that simultaneously fulfils the following requirements: neutronic properties (neutron moderation, breeding ratio, fission product inventory, etc.), operating temperature of the reactor core (molten salt temperature, radiation stability, transport properties), solubility of actinides and fission products in the molten salt to ensure a homogeneous core composition, compatibility of materials and control of salt chemistry and, last but not least, processing and low waste requirements. (Greenspan 2021)

According to (GIF 2014; 2021d), essential research work still needs to be carried out: the development of a basic concept, research into physico-chemical properties of relevant molten salts, the development of salt and material combinations (characterisation and qualification), on the degassing of the molten salt, on coupled neutron-physical and thermo-hydraulic models of the reactor, on the safety assessment of relevant accident sequences, the interaction of molten salt with air or water under severe accident conditions, the demonstration of safety characteristics on a laboratory scale and beyond, the development of technology for the continuous reprocessing of molten salt during reactor operation and further research and development into the disposal of utilised molten salt.

(GIF 2021d) sees the need for the next few years up to 2028 primarily in the following major challenges:

- the identification, characterisation and qualification of successful salt and material combinations for MSR,
- the development of integrated models (physics and fuel chemistry) for reactor behaviour and safety assessment, which map the corresponding physics and fuel chemistry and are required for the evaluation of reactor performance over all time scales for the licences,
- the establishment of an MSR infrastructure and economics with cost-effective and practical systems for the production, processing, transport and storage of radioactive salts,
- the development of a framework for MSR authorisation and safeguards.

According to (GIF 2021d), this would primarily require inactive test stands, i.e. test stands operated without radioactive material, for testing molten salts or test stands for testing full-scale reactor components, as well as the availability of an active demonstration facility with relevant neutron production for investigating molten salt chemistry and for measuring corrosion, for testing fission product removal and also maintenance techniques. According to (GIF 2021d), progress in the programmes for the construction of MSR demonstrators (IMSR, MCFR, MOSART, TMSR-LF1) is necessary in order to build up this infrastructure.

However, according to (GIF 2021d), despite the increase in private initiatives, recent progress has been limited due to a lack of public investment and the development challenges have only been partially addressed.

The requirements for controlling the combination of radiation and corrosion in the utilisation of molten salts are complex. One advantage of thermal MSR with solid fuels is the decoupling of both phenomena through the ability to expose structural materials only to thermal fluxes, but other advantages of liquid fuel are lost. For molten salt systems, corrosion is a key viability issue in the development phase. Fundamental analyses of the process, experimental characterisation of properties and processes, modelling and lifetime prediction methods and finally validation experiments are required for credible design, safety assessment and approval of reactor concepts,(Calderoni and Cabet 2012) and appropriate monitoring measures and the necessary models and safety requirements must be developed.

In the MSRE experiment, the corrosion of the high nickel content structural material Hastelloy N was well controlled (Calderoni and Cabet 2012). However, other problems occurred with the material, including embrittlement during irradiation and intergranular cracking (Calderoni and Cabet 2012). Due to the corrosion-inhibiting properties of nickel, the steels used in the reactor were alloyed with high proportions of nickel. However, nuclear reactions of nickel with neutrons produce helium, which has an embrittling effect on many metals (GRS 2020). In order to increase the resistance of nickel-based alloys to embrittlement, alloying elements can be added to form finely distributed carbides within the grains of the material (Greenspan 2021).

The material fatigue effects shorten the operating and service life of the reactor and may even require the periodic replacement of particularly stressed reactor components. This increases the volume of radioactive material to be disposed of (GRS 2020). Terrestrial even proposes replacing the entire reactor core every seven years and reprocessing the spent fuel (IAEA 2016f). The service life of the core module of the CMSR should also be limited to around ten years to prevent corrosion and radiation damage (GRS 2020).

The actual corrosion mechanism, the dissolution of alloys by oxides in salt, is well understood thanks to the work at ORNL and the Kurchatov Institute, and there is a lot of literature on the corrosion mechanisms of alloys by molten fluoride salts (Greenspan 2021). How and where these mechanisms occur at hot spots, and then form deposits at cold spots such as in the heat exchanger, must then be experimentally confirmed in scaled experiments for specific reactor concepts.

Due to a lack of experience, the chemistry of chloride-based fuels is less well understood than that of fluoride-based fuels, and little research has been carried out into which materials are compatible with the fission product-containing chlorides (Hombourger et al. 2019; Institut für Festkörper-Kernphysik 2019). The feasibility of enriching chlorine-37 sufficiently to avoid the formation of chlorine-36 and improve the neutron balance has yet to be demonstrated .(Hombourger et al. 2019)

To achieve a high burnup, it is necessary to tolerate high fission product concentrations in the salt. In particular, the lanthanide elements soluble in the salt must be analysed for their influence on the melting point of the molten salt so that no solidification and accumulations can occur during operation (Hombourger et al. 2019).

To date, there is little experience of partitioning and chemical processing of the molten salt beyond the laboratory scale. Tests are being carried out to recover lanthanides in chloride and/or fluoride salts and databases on separation efficiency are being compiled. The acquisition of fundamental data is also required specifically for the separation of actinides and rare earths. The extraction of rare earths is necessary due to the low solubility of these trifluoride elements and their neutron capture, which worsens the neutron balance. Beyond the laboratory scale, the engineering issues need to be investigated and the system needs to be brought from laboratory scale to industrial application (Greenspan 2021). There are currently no valid physical models for simulating the use of partitioning in an operating reactor. In the case of reprocessing, thorium reprocessing processes have so far been little researched and developed. In (Oeko-Institut e.V. 2021), a TRL of 3-4 is specified for the development status for partitioning molten salts, see Chapter 2.4.

New integrated simulation tools are needed to model the physical and chemical changes in the molten salt and its transport throughout the system, especially to map the influence of delayed neutrons from fission products and their precursors when they are removed from the molten salt in different ways (Holcomb 2017a).

Hastelloy N as a material for high-temperature power reactors is currently not part of the ASME simulation tools. A material qualification programme would therefore have to be carried out in order to use Hastelloy N as a material with a safety function. The maximum permissible stress of Hastelloy N decreases rapidly from 600 °C and becomes too low for practical use from a temperature of 700 °C onwards. Hastelloy N has a significantly lower high-temperature strength compared to the typical materials used in the construction of very-high-temperature reactors (Greenspan 2021).

New approaches to the development of structural materials were presented in (ANL 2018; ORNL 2018b). The Kurchatov Institute has also investigated successor materials for Hastelloy N (HN80MTY), which have better strength at high temperatures and good corrosion properties against fluoride salt.

The GIF (GIF 2014) assumes that further development steps will be necessary after the viability phase up to around 2025, with large-scale experiments on handling molten salt and integral demonstration plants without active nuclear fission and finally the construction of demonstration reactors for the licensing procedure. However, the demonstration reactors would still be without continuous reprocessing of the fuel (GIF 2014). If we very optimistically assume a period of only 10 years for each of these development steps (for conception, licensing, construction, operation and experimental evaluation of the respective plants), the first commercial prototype of an MSR cannot be expected before 2050-2060.

Conclusion: Technical development status

The development of MSR reactor technology is more advanced for thermal reactor concepts than for fast reactor concepts. The development of MSR is still at a very early stadium of “applied research”.

In fuel development, MSR concepts with TRISO fuel benefit from fuel development for HTR. For all concepts with fissile material dissolved in a molten salt, the state of development is also in the area of “applied research”. In some cases, there is still a lack of fundamental data on the essential chemical and physical properties of the fuels.

There is also still a great need for research and development with regard to the safety functions of a mature MSR concept, and key properties have yet to be demonstrated. The status of development can be categorised as “applied research”.

With regard to Instrumentation and Control (I&C) and operational requirements, there are discussions on the need for research; the status is also categorised as “applied research”.

There are many unanswered questions regarding the safety demonstration principles; the procedures are currently in the area of “applied research”, and in some cases also in “development”.

Against this background, the authors of this report estimate that the development status of MSR is currently in the “applied research” phase.

4.4.6 Safety

The following presentation was taken from (Oeko-Institut e.V. 2017) and supplemented.

In terms of safety, MSR have a high coefficient of thermal expansion, which ensures a large negative feedback of reactivity due to thermal expansion during heating. When the fuel is liquid, it expands when heated, which reduces reactivity and improves the self-regulation and thus the safety of the reactor. This makes it possible to self-stabilise the reactor output in the event of malfunctions (loss of flow, loss of cooling, etc.). In principle, it may even be possible to dispense with the excess reactivity required to start up the reactor. This can be achieved by preheating the fuel salt in the subcritical drain tanks to the working temperature and then filling the primary circuit with the fuel (Greenspan 2021). The low concentration of actinides in the fuel salt and the large distance to the

saturation temperature ensure that the heat capacity of the primary circuit is considerable and that the transient processes take place without strong temperature changes (Greenspan 2021).

The very strong and instantaneous temperature feedback effect can even mean that control rods become superfluous and control of the reactor can only take place via the heat exchangers. With increased cooling, the power of the reactor increases in order to maintain the critical temperature. If cooling is stopped, the reactor’s power output drops to zero (with the exception of residual heat) without a large increase in temperature. The reactor can also be started by successively filling it with fuel until criticality is reached (Greenspan 2021).

The biggest safety challenge with many MSR is that the fuel salt and thus the fuel is liquid. Reactivity is carried out of the zone where criticality is reached into areas where heat exchange with a secondary coolant takes place through heat exchangers. In this second zone, delayed neutrons and residual power are released in contrast to classic reactor concepts with solid fuels. The proportion of delayed neutrons is thus reduced due to the movement of the fuel and the delayed neutrons enter into less important parts of the reactor (Greenspan 2021). The heat transfer from the fuel salt to the intermediate coolant takes place outside the neutron field, and the interface between fuel and coolant has no influence on the neutronic processes in the core (Greenspan 2021). In the event of possible undercooling of the molten salt, there is a possibility that fissile salts will precipitate, which then have higher fissile material concentrations than the remaining liquid flowing back into the core, so that a power peak can occur. According to (ORNL 2021), this is probably only a short-term effect with adequate negative feedback mechanisms, yet it remains a challenge.

This requires precise knowledge of the coupled neutron-physical, thermo-hydraulic and physico-chemical system as well as the temporal changes in the fissile material content. The fuel must also be very homogeneously dissolved within the molten salt. Otherwise, power peaks and thus overheated areas can occur when the molten salt passes through the reactor. However, the fundamental possibility of achieving a homogeneous distribution of the fissile material is also an advantage of MSR compared to reactor concepts with solid fuels. Degassing can also contribute to fluctuations in the reactivity of the reactor. Monitoring and controlling the distribution of the fuel in the molten salt is therefore an important safety function, the implementation of which has not yet been clarified (IRSN 2015).

With MSR, there is the possibility of continuous fission product removal by physical (helium atomisation for noble gas removal) and/or pyrochemical processes. The fuel salt can be processed in operation or in batches to remove the soluble fission products (lanthanides) and actinides. The actinides are then returned to the fuel cycle. Noble gas removal also prevents xenon poisoning in thermal MSR (Greenspan 2021).

To maintain or improve the criticality of the molten salt, the continuous removal of fission products can also be carried out through a series of filters. The purified molten salt is then fed back into the reactor. Volatile fission products such as noble gases are removed, which must be collected in vessels and stored separately. Other insoluble fission products can also be filtered out. Gaseous fission products and also elements of precious metals and rare earths have isotopes that strongly absorb neutrons and thus limit the possible burnup of the fuel depending on the fissile material concentration (Betzler et al. 2017). However, fission products with an influence on reactivity control are also outgassed, especially precursor nuclides for delayed neutron release are then no longer available.

With regard to residual heat removal, one advantage of MSR is that the primary cooling circuit has low operating pressures, which has a positive effect on the load on components and the effects of leakages (loss of coolant). Based on the operating temperatures, there is also a large distance to the boiling temperatures of the molten salt, so that in the event of temperature excursions or leaks in the cooling circuit, a rapid build-up of pressure or major releases of radioactive substances from the molten salt are not to be expected. Due to the strong thermal expansion of the salts, it is also possible to enable passive heat removal via natural convection in the primary cooling circuit if the system is designed accordingly.

On the other hand, it must be taken into account that residual heat removal must be possible from all areas that can come into contact with the molten salt during operation and in the event of incidents (dump tanks, on-site reprocessing, etc.). This places fundamentally different requirements on the methods and verifications for residual heat removal.

Emergency dump tanks are a central safety element of MSR. In the event of reactor malfunctions, the molten salt should be drained into these tanks, where it should be cooled by suitable systems. Due to the geometric arrangement, the subcriticality of the molten salt should be guaranteed in these dump tanks. If the heat removal from the reactor system is disrupted, the melting of a plug (freeze plug) should passively, i.e. without external switching operations or external energy supply, ensure that the molten salt is drained into the dump tanks. In some concepts, large water tanks are used to ensure passive cooling of the dump tanks. In order to discharge the reactor for operational purposes (such as inspections or repairs), the molten salt can also be discharged in a targeted manner via valves.

The concept of dump tanks was also provided for in historical MSR concepts, although it was not emphasised separately. The need to demonstrate the sufficient reliability of residual-heat removal from such dump tanks under all accident conditions to be considered is already pointed out in (AEC 1972), especially for reactors with large power. (IRSN 2015) notes that the viability and reliability of a purely passive triggering of the drainage of the molten salt in such dump tanks by melting a plug has not yet been demonstrated. In particular, the plug is typically not located in the area of the highest temperatures in the system, so that the question of a timelytriggering before reaching temperature limits in the reactor still needs to be answered. Such a safety function is not available for MSR systems with solid fuels. It is also not clear to what extent a purely passive heat removal of the decay power from the dump tanks can be demonstrated with the necessary high reliability and under all boundary conditions to be considered (such as external effects like earthquakes or aircraft crashes) for the systems discussed in the context of GIF with 3000 MW of thermal power.

Furthermore, other event sequences are conceivable for MSR than for today’s nuclear power plants. For example, if solubility limits are exceeded or if there are significant temperature fluctuations (for example in the heat exchanger between the primary and secondary cooling circuits), precipitation or solidification of fission products from the molten salt can occur. This can clog heat exchangers, which can lead to other requirements for the control of incident sequences (Holcomb 2015; IRSN 2015). The salt chemistry is also complex and the molten salt contains elements with a low critical mass such as plutonium and minor actinides. If these products are deposited in the dump tanks or in the heat exchangers, a critical arrangement must not occur (Greenspan 2021).

Due to the high melting points of the molten salts, there is the problem of reliably preventing the molten salts from solidifying after the plant has been shut down in the event of an accident. This may require options for heating up the melt. Their functionality must also be ensured in the event of a failure of the offsite power supply or in the event of a station blackout (Flanagan 2015).

In the case of continuous separation of fission products, relevant amounts of residual heat can also be generated in areas outside the actual reactor system, in particular in the on-site reprocessing plant but also in storage areas for the separated fission products. Many fission products have the volatile xenon and krypton as precursor nuclides in their decay chain, so that around 40% of the fission products, including large quantities of cesium, strontium and iodine, leave the core (Holcomb 2017a). Sufficient coolability must also be demonstrated for these under all assumed event sequences, comparable to today’s requirements for the cooling of fuel elements in fuel pools. As no specific designs are yet available for on-site reprocessing in MSR, systematic studies to ensure heat removal in these facilities are also not available.

With regard to the confinement of radioactive materials, MSR differ significantly from today’s nuclear reactors, as a large proportion of the radioactive inventory is not concentrated in closed fuel elements, but is transported dissolved in the molten salt in the cooling circuit. Furthermore, there is a coupling with the system areas for on-site reprocessing of the molten salt. Here, too, the confinement of radioactive materials must be ensured at all times.

In principle, it is assumed that many fission products have good solubility in the molten salt and therefore a low release from the molten salt in the event of a malfunction is expected. However, the solubility limits of the fission products in molten salt are not yet known with sufficient accuracy. As a result, there are uncertainties regarding the possible release of fission products from the molten salt in the event of incidents and accidents (source term) (Holcomb 2015). Despite the filtering of volatile and insoluble fission products, fission products remain dissolved in the molten salt.

Various barriers are currently defined for MSR to retain the radioactive substances. Typically, there is a first barrier, a so-called “fuel envelope”, which encloses the reactor area and the dump tanks. The second barrier discussed is a containment that encloses the entire primary cooling circuit, the dump tanks and heat exchangers and the chemical reprocessing. The reactor building is to act as the third barrier. However, the exact function of the respective barriers, their design and the possible failure mechanisms (internal events, external influences) have not yet been determined.

Another major advantage of MSR concepts is that a significant proportion of the fission products produced can be removed from the reactor during operation. This means that the inventory contained in the reactor, which can be released in the event of incidents and accidents, is significantly lower than in reactors with solid fuels. From today’s perspective, it is unclear whether the radioactive inventory in such a system can be kept so low that the intervention guidelines for disaster control measures outside the plant are not exceeded in all incidents and accidents to be considered.

The most critical accident scenario for all MSR is the release of large quantities of molten salt into the environment. Many of the phenomena identified for this accident scenario also apply to other accidents, such as the release of the cover gas from the primary circuit or transients that cause primary systems to fail due to factors such as excessive temperatures or pressures or due to energy excursions (ORNL 2021). During the release of the cover gas, all gaseous or vapour-transported fission products and aerosols can in principle be released in the event of an unforeseen rupture or leakage of the primary circuit and a rupture of the containment, if no further mitigative measures can be taken. During operation, the fission products accumulate in the space above the liquid fuel, with a higher proportion of fission gases than in the plenum of LWR fuel rods. The consequences of such an accident sequences must therefore be prevented and the probability of such an accident must be virtually excluded (“outside of a credible accident space”) (ORNL 2021).

Further problems of a release of molten salt arise from its property to solidify and form a crust. Thermally insulating snow-like deposits can form on surfaces inside the containment, which hinder heat transfer. Mists of salts can form inside the containment, which hinder heat transport through thermal radiation. Molten salt interactions with components and coolants can lead to pressure build-up or even ignite materials and generate fog, thereby hindering heat transport (ORNL 2021).

So far, the possibilities for modelling the release, transport and non-equilibrium conditions are very limited and no integrated experiment on large releases of a molten salt has yet been carried out under the corresponding conditions for model validation. This would be necessary to determine the initial conditions, such as the location of the radionuclides at the beginning of an accident. No existing toolset for reactor modeling includes significant MSR phenomena such as the inert gas and its localisation, precipitated radionuclides or filtered particles, all of which have significant safety implications (ORNL 2021).

The radioactive substances removed from the molten salt must first be stored on site and then suitably conditioned. A systematic safety assessment of possible incidents and accidents in on-site reprocessing is not available due to the lack of a concrete design.

Due to the production of tritium from the main components of the molten salt (Li, Be), the radioactive source term of tritium is high, so that measures to retain and separate tritium are already required under normal operating conditions (Wu et al. 2015). Sufficient confinement of the volatile tritium must be demonstrated for specific reactor concepts, especially under accident and incident conditions. Tritium is primarily formed by the interaction of neutrons with lithium or beryllium, which are often used in the primary coolant. Significantly more tritium is produced in MSR than in LWR. The capture of tritium is technically very complex due to its volatility. Due to their large contact surface and thin-walled structures, heat exchangers are the primary path through which tritium can escape from the primary circuit. Tritium capture in the secondary cooling circuit ($NaF\text{-}NaBF_4$) has been proposed and experimentally demonstrated in the MSRE (Greenspan 2021).

Finally, in addition to radioactive substances, the presence of toxic substances (such as beryllium as a component of molten salt or hydrogen fluoride in chemical reprocessing) is to be expected in MSR.

As a result of the catastrophic accident at the Fukushima nuclear power plant, the importance of the interaction of different plant components on a plant site, especially under the conditions of severe incidents and accidents, has been increasingly taken into account in the safety considerations of nuclear plants. The reprocessing of molten salt is an integral part of MSR concepts. Due to the coupling of the reactor system with an on-site reprocessing plant, the safety-related effects of incidents and accidents in one part of the plant on the other must also be taken into account (Holcomb 2015). As the actual on-site reprocessing plant has not yet been technically developed, no concrete analyses are available to date with regard to possible interactions between this plant and the reactor system.

In today’s safety philosophy, the avoidance of core damage or a core meltdown has taken centre stage. In contrast, the fuel in MSR is already in a molten state. Furthermore, for today’s LWR, an intact reactor pressure vessel is a prerequisite for the reactor core to remain coolable. Proof of the integrity of the reactor pressure vessel therefore plays an important role in the safety case of today’s nuclear power plants. In contrast, the possibility of draining the molten salt from the actual cooling circuit plays an important role in the safety concept for MSR concepts. Therefore, new rules and guidelines for the approval of such concepts must be developed for MSR (Holcomb 2015).

Detailed mathematical analyses of the incident and accident sequences to be considered must then be carried out for the specific verification of the safety level of a new reactor concept. With regard to the verification of MSR, it can currently be stated that many basic neutron-physical and physico-chemical data on molten salts (cross-sections, heat transport properties, solubility limits of the fission products) are not yet known with the accuracy required for a nuclear licensing procedure (Holcomb 2015).

The molten salt is heavily radioactively contaminated during operation – despite continuous reprocessing. This means that all areas that may come into contact with the molten salt during operation (reactor, cooling circuits, heat exchangers, dump tanks) will be subject to high radiation protection requirements in terms of inspection, maintenance and servicing (Flanagan 2015).

The availability of suitable structural materials will also be central to the question of the safe operation of future MSR. These must be able to withstand the high operating temperatures and high neutron fluxes in the reactor as well as the corrosive properties of the molten salt in combination with the dissolved fuel and fission products (Holcomb 2015). Such structural materials are not yet available, and the development and qualification of such materials is associated with high uncertainties and a high time expenditure. It is also necessary to qualify active components for operation in an MSR, such as heat exchangers, pumps and valves, as well as instruments for monitoring the molten salt. Heat exchangers in particular are likely to pose significant challenges here, as they represent a cold spot in the system and ensure a high heat flow. Both properties lead to corrosion and the deposition of corrosion products (Greenspan 2021). In addition, the heat exchangers must also form a barrier to possible tritium diffusion in order to prevent tritium release into the system or the environment (especially for salts with lithium and beryllium). The heat exchangers must also allow inspections and maintenance.

(Greenspan 2021) summarises the following specific challenges for the evaluation of the safety of molten salt reactors:

- The compatibility of salt with other reactor materials under high temperatures and radiation exposure
- High melting points
- Significant amounts of fuel are located outside the reactor core in the heat exchanger, numerous tanks, pumps and, depending on the concept, in a reprocessing plant. Fuel must also be continuously removed and supplied
- Delayed neutrons are distributed through the mobile fuel in the system
- Salt vapour deposits
- Potentially larger volumes of highly radioactive components such as filters and other replaceable components
- The continuous change in salt composition
- The fuel performs essential cooling functions
- Tritium production in MSR with molten lithium and molten beryllium salts
- The presence of bubbles of fission product gases transported in the core

- Noble gas fission products segregate from the fuel into the cover gas. Precious metal fission products can be deposited on surfaces; the fuel salt contains all other fission products. Plants for the treatment of gaseous and soluble fission products or actinides require a specific safety assessment
- The composition of the fuel salt is complex and some products may reach the solubility limit at cold spots or during selective extraction during treatment to cool the salt. Should this occur with fissile elements of low critical mass, this could lead to a criticality accident
- The special operating conditions of MSR have yet to be investigated

Conclusion: Safety

It can be seen from the above that, at present, there is no clear advantage or disadvantage of MSR compared to LWR in terms of reactivity control, cooling and residual heat removal and the confinement of radioactive materials. Some advantages over today’s LWR, such as the very large negative feedback of reactivity with temperature, must be set against new challenges, such as the continuous inflow and outflow of reactivity from the reactor core.

As many details of the various MSR concepts have not yet been defined at the current stage of the development process, many open questions remain with regard to analysing safety, especially with regard to the event spectrum, verification and normal operation. The advantages or disadvantages compared to LWR can therefore not be conclusively answered at this stage.

4.4.7 Fuel supply and waste disposal

The following presentation was taken from (Oeko-Institut e.V. 2023) and supplemented. The focus is on molten salt reactors in which the fuel is dissolved in the molten salt, not on concepts that use molten salt for cooling but solid fuels.

Like all reactors, MSR also generate different waste streams. An overview of the different waste streams is provided in (Riley et al. 2019; Pacific Northwest Laboratory 2018). Waste gas streams (halide and noble gases), salt waste streams, separated salt streams, metal waste streams, graphite and carbon waste streams, decommissioning and decontamination waste streams and operational waste streams are the main categories of waste that can be expected from the operation of MSR (Carlson et al. 2021; Riley et al. 2019; Pacific Northwest Laboratory 2018). Many of these differ from the waste streams from LWR due to their different chemical properties, but also quantitatively (NASEM 2023b).

In MSR with liquid fuels, the radioactive substances produced are not predominantly enclosed in the solid fuel (in the form of fuel elements), but are dissolved in the liquid fuel or are present in the form of particles. Depending on the operating conditions of the fuel (temperature, pressure, chemical composition, contact materials such as pipes or cavities), the resulting elements and molecules remain dissolved in the fuel, are precipitated as particles or are separated from the fuel liquid as gas, mist or aerosol. This also applies to the decay products of the resulting fission products and actinides. In each case, the different waste streams must be suitably conditioned for disposal.

4.4.7.1 Fuel

The spent fuel liquid is highly radioactive due to the unseparated fission products, actinides and salt activation. It can either be disposed of directly or the salt can be treated and split into different waste streams. Some reactor concepts also provide for the recovery of enriched lithium and/or chlorine (^{7}Li and ^{37}Cl).

The planned continuous separation of fission products in MSR with a fast neutron spectrum and with liquid fuels changes the nuclide composition of the fission products compared to other fast reactors, as the fission products are exposed to a significantly lower neutron flux before being removed from the reactor.

The disadvantage of direct disposal of the salt is its solubility when water enters a repository. (NASEM 2023b) assume that direct disposal is only possible in salt; other host rocks are not suitable due to the possibility of water ingress and the high solubility of salt waste. But even with disposal in salt as a host rock, for example for spent fluoride salt fuel, it can be assumed that interaction with water would lead to the formation of uranium phases from the hydrogenation of crystalline UF_4 that are orders of magnitude more soluble, or even corrosive hydrofluoric acid (Tracy et al. 2016; NASEM 2023b). During the storage of waste from the Molten Salt Reactor Experiment (MSRE) in the U.S. (fluoride salt), for example, it has been shown that fluorine compounds and volatile uranium compounds can form via radiolysis due to actinide residues contained in the solid or frozen fuel salt in the storage tank, and this in turn could lead to a criticality problem at enrichments above 8% (NASEM 2023b). However, the concentration of volatile uranium can be monitored by checking the off-gas.

According to (Krall and Macfarlane 2018), initial speculative studies in the U.S. have also shown that direct disposal is not an option for fluoride-based uranium-plutonium fuels. This is also justified by the chemical properties of the uranium tetrafluoride contained in the fuel, which could lead to the formation of highly corrosive hydrofluoric acid on contact with water. According to (Krall and Macfarlane 2018), spent uranium tetrafluoride fuel needs to be reprocessed in order to separate the fluorine and stabilise the uranium, fission products and actinides for disposal.

(Carlson et al. 2021) In a recent assessment of MSR waste and, in particular, of the waste forms available for disposal, the German Federal Environment Agency states that metal and salt waste are the least developed waste forms, with a need for development, also with regard to existing final disposal concepts in the U.S. Suitable technologies also still need to be developed for the final disposal of molten salt (Holcomb 2015). Several reports are available from the U.S. experience with MSRE on issues relating to the disposal of fluoride salt waste (NRC 1999).

For chloride-based molten salts, an enrichment of chlorine-37 is necessary for use in MSR, which has a natural abundance of about 75.77%. Chlorine-35 (24.23%) has a high effective cross-section for neutron absorption and reduces the neutron balance. The resulting chlorine-36 has an extremely long half-life of 300,000 years and leads to problems with final disposal (Riley et al. 2019; IRSN 2015).

It is better to immobilise the salt in a waste form that is chemically more resistant to mobilisation of the radioisotopes it contains than to dispose of it directly. The waste form must retain completely different elements and molecules of halogens, alkali metals, alkaline earth metals, rare earths and actinides in the salt. (Riley et al. 2019) assumes that there will be different waste forms to immobilise chloride or fluoride salts. The common waste forms for highly radioactive salts therefore include

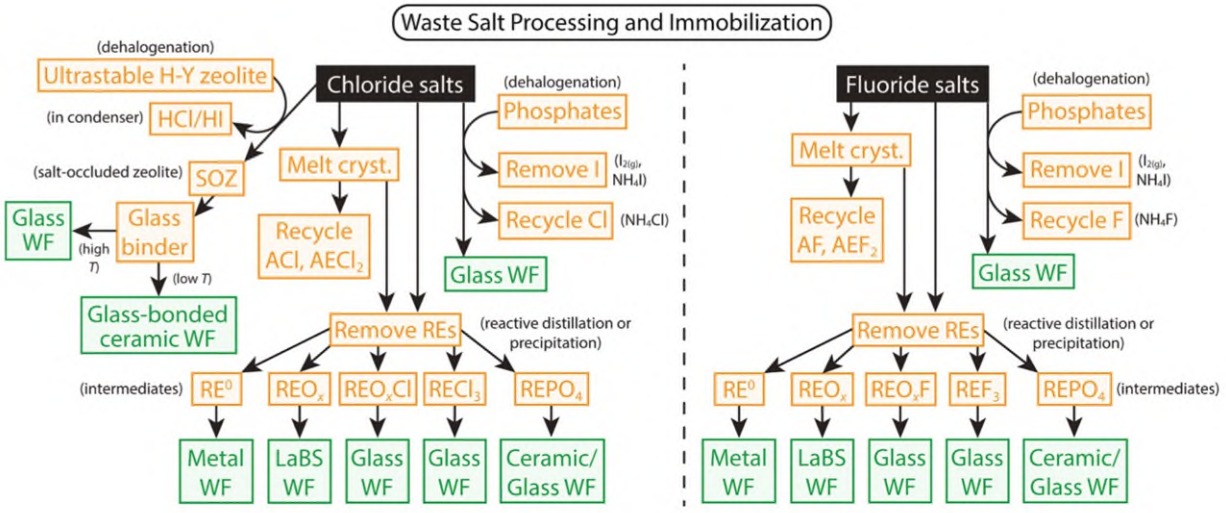
minerals, glasses, ceramics, glass-ceramics, glass-bonded ceramics (also known as glass composite waste) and ceramic-metal composites (cermet) as well as metal halides (halmet) (Riley et al. 2019).

Among the mineral waste forms, titanate (synric), sodalite and apatite were analysed in particular. Soldalite has already been demonstrated in the Experimental Breeder Reactor (EBR) for the treatment of pyroprocessing waste; the end product is a glass-bonded sodalite ceramic (Riley et al. 2019).

Among the possible glasses, there are only a few that are suitable for immobilising salts, as the solubility of traditional silicate glasses for halogens is very low and the volatility of the halogens at the melting temperatures of glasses is very high. Other glasses are therefore required that are not based on silicate, such as tellurite glasses, which were investigated for chloride salts. As a result, some chloride salts were trapped well, but others could not be bound well in the glass, and tellurite glasses are also expensive (Riley et al. 2019).

In addition to stabilising the halides, treatment of the fuel salt before disposal can also consist of separating the waste stream. This would allow valuable raw materials such as enriched ^{7}Li or ^{37}Cl to be recovered, and the separation of actinides or certain fission products would also be possible. Many reactor concepts provide for chemical reprocessing of the fuel in the plant concept, either for reprocessing in batches or during operation. Possible processes include reductive extraction, e.g. of lanthanides, oxidative precipitation (chlorine salts), distillation, melt crystallisation, dehalogenation, phosphorylation, ion exchange and a system for the oxidation and dissolution of glass material (Riley 2020; Riley et al. 2019). If the waste is separated, no additional salts are required to dissolve the fuel. Glasses, glass-ceramic composites and metal alloys can be used as waste packages (see Figure 4-5).

Figure 4-5: Summary of different separation technologies with possible process paths and waste forms for the disposal of MSR salt waste



Source: (Riley et al. 2019)

WF: waste form; RE: rare earth; A: alkali; AE: alkaline earth; LaBS: lanthanide borosilicate

The immobilisation or processing of the salts has a significant influence on the amount of waste. (NASEM 2023b) According to the information provided by the waste producers, the treatment of salt reprocessing waste from the pyroprocessing of EBR-II resulted in a 30-fold increase in the total mass of waste. Reference is also made to the generation of additional waste streams in the area of low and intermediate-level radioactive waste.

The disposal of high-level radioactive salts is being actively researched in the laboratory. Most research has been carried out on fluoride salts. For both chloride and fluoride salts, however, major gaps remain in the assessment of waste package functionality and separation processes. In order to predict the long-term behaviour of the waste forms in a disposal environment, a great deal of data on the chemical composition of the waste would be required, which is not yet available (Carlson et al. 2021). In any case, clarification of the disposal issues would be an integral part of the development of MSR:

> *“Although commercialisation of MSR and electrorefiners are likely decades away, developing a strategy to manage their waste streams before implementation will be integral to their successful deployment” (Carlson et al. 2021).*

Examples include the waste from the MSRE, which was shut down in 1969. For the waste, a DoE report - made public through a Freedom of Information Act release following a request by (Krall et al. 2022) - considers cancelling the disposal of the high-level waste in a repository and containing the waste in-situ, i.e. in the reactor pit (see Appendix Section 3 of (Krall et al. 2022)). The main reasons cited are the cost and technical complexity of removing the salts from the current tanks. Uranium has already been separated from the salt due to the formation of uranium fluoride in order to prevent recriticality and the formation of volatile fluorides. However, there are still problems with radiolysis and the formation of corrosive gases.

Further presentations of the state of research and development on the disposal of waste from MSR were presented by (Pacific Northwest Laboratory 2020; 2018; Riley et al. 2019; Riley 2020).

4.4.7.2 Off-gas treatment

The management of exhaust streams in molten salt reactors is a key aspect of their operation. Unlike in solid fuel reactors, the gaseous fission and activation products are outgassed in liquid fuels. The outgassing of neutron absorbers such as Xe-135 is a desirable effect for improving the neutron economy in the reactor. However, the off-gas in MSR consists of a number of gases, including noble gases (such as the neutron absorbers xenon and krypton), halogens (such as fluorine and chlorine), chemically reactive gases (such as iodine, chlorine, fluorine) and tritium (Riley et al. 2019). These gases are produced by fission reactions, radioactive decay, radiolysis and other chemical reactions within the reactor. However, all types of radioactive noble gases such as krypton and xenon isotopes must then also be treated, including the resulting daughter nuclides (e.g. caesium, barium, rubidium, strontium, lanthanides, zirconium). Tritium is particularly difficult to handle due to its high mobility and the difficulty of confinement. For the capture and handling of iodine, see (Riley et al. 2016; NEA 2022). A number of salts can also be produced in the form of mists or aerosols (Riley et al. 2019). The off-gas treatment systems in MSR are integrated into the reactor operation and are conceived to handle high temperature gases and in some cases operate continuously to consistently capture gases such as xenon. The off-gas system includes, for example, tanks, pipes and filters in which the radioactive substances are collected and decay. The system components must be maintained during

operation and also disposed of separately, depending on the radionuclides that have been captured in the system.

Various methods should be used to capture and treat the off-gas, see Table 4-8 for an overview of waste forms (Riley et al. 2019). Particles and aerosols are captured with high efficiency particulate air filters (HEPA) and fused NaOH systems and immobilised in ceramic or glass. These systems effectively bind tiny particles and aerosols that could carry radioactive materials. For reactive gases such as iodine, chlorine and fluorine, scrubbing methods will be used in which the off-gas is passed through a solution (such as aqueous hydroxide) or solid sorbents that chemically bind and remove the reactive gases from the gas stream (Riley et al. 2019; McFarlane et al. 2020).

For the capture of noble gases, techniques such as cryogenic distillation, where gases are separated based on their different boiling points, and adsorption with materials such as activated carbon or silver zeolite (AgZ) are used (Riley et al. 2019; McFarlane et al. 2020; NEA 2022). The short-lived gases are stored so that they can decay.

One problem with lithium-based molten salts is the production of radioactive and highly volatile tritium by neutron capture in lithium-6, which is contained in about 7.4% of natural lithium. High enrichment to ≥ 99.99% of lithium-7 is required to limit tritium production, as the separation of tritium and subsequent storage is technically very challenging (Forsberg et al. 2017). The extent to which this can be implemented for economic reasons is currently unclear. To exclude relevant releases of tritium from the plant into the environment, structural materials with sufficient retention capacity and technologies for separating tritium from the molten salt would have to be developed. Furthermore, methods for the long-term storage of tritium would have to be developed. For isotopes such as tritium (with a half-life of approx. 12 years) but also krypton-85 (with a half-life of approx. 10 years), the use of cryogenic storage over periods of approx. one century is also being discussed (IRSN 2015). Separation techniques such as cold traps, in which tritium is condensed and collected, and catalytic recombiners, which convert tritium gas into tritiated water, are being discussed (Riley et al. 2019; McFarlane et al. 2020).

Off-gas management in MSR is challenging and complex due to the high temperature environment, the corrosive nature of molten salt and the mixture of gases produced. Due to the high inventory of short-lived radionuclides, the off-gas system of an MSR differs from typical off-gas systems of an LWR (Riley et al. 2019). A possible off-gas system for a commercial MSR, for which most of the components are commercially available, is presented in (Riley et al. 2019). After capture and processing, the radioactive off-gases and particles are prepared for safe storage or disposal. This includes converting them into stable forms, such as solid waste, which can be safely stored or disposed of in specialised facilities.

Redox control is required in all MSR for a number of reasons, including to reduce tritium release. The fission process is oxidising, which changes the redox potential of the salt without redox control (Riley et al. 2019; Guo et al. 2018). This can also corrode other materials if the environment becomes too oxidising. Which fission products are reduced to metals and which remain as chloride or fluoride salts depends on the redox potential. As metals are very poorly soluble in salts, they are deposited on the surfaces of the reactor if they are not removed (Riley et al. 2019). Strong redox changes can cause certain actinides or fission products to precipitate out of the salt solution, changing their chemical structure and increasing the risk of an accident (Riley et al. 2019).

The redox potential can be regulated by adding salt redox buffers, reducing agents and gas sparging, which also removes other impurities from the salt (Riley et al. 2019). In conventional fluoride salt MSR, for example, reducing chemicals are added to change the redox chemistry over time and regulate the ratio of U^{3+} to U^{4+} (Riley et al. 2019).

According to the control of the redox potential, radioactive elements can therefore also be deposited on components and structural material and would then have to be stored and disposed of appropriately according to the deposited radioisotopes.

Table 4-8: Summary of the volatile components of off-gas waste

Waste type	Capture method	Immobilisation method
Particles	Molten NaOH	Ceramic or glass as waste form (WF)
	HEPA filter (≥300 nm diameter)	Size reduction and decontamination
Aerosols	Molten NaOH	Ceramic or glass WF
	Particle traps, charcoal beds	Solids
	HEPA filters	Size reduction and decontamination
Reactive gases	Molten NaOH	Ceramic or glass WF
Tritium (e.g. $^3H(g)$, ^3H_2O)	Separation of gases with low volume or water with high volume (e.g. desiccant, molecular sieve, carbon)	Cement with low water content in vessels
Residual halogens (e.g. F, Cl, I)	Silver mordenite (AgZ)	As AgI in composite WF
	Ag0-functionalised silica aerogel	As AgI in composite WF
	Ag+/0-loaded aluminosilicate aerogels	As AgI in composite WF
	Sulfidaerogel (chalcogel)	In chalcogenide glass
	Molten NaOH	Ceramic or glass WF
N_2, O_2	Activated carbon, magnetic separation, cryogenic	Release
Noble gases (e.g. Kr, Xe)	Cryogenic distillation	Storage as compressed gas
	Solid sorbents (AgZ, HZ, activated carbon)	Storage as compressed gas
	Organometallic frameworks	Storage as compressed gas
	Sulfidaerogel (chalcogel)	Storage as compressed gas

Source: (Riley et al. 2019)

Summary of volatile waste gas components that must or could be captured and/or immobilised. Aerosols: Full salt composition from splashes or condensation of volatile daughters. N_2, O_2. The isotopes produced from a fluoride salt, such as oxygen-16, -17 and -18 for oxygen and nitrogen-15 for nitrogen, are all stable and could be released.

4.4.7.3 Other waste streams

Graphite is used as a moderator in thermal MSR Concepts oras a reflector in MSR with a liquid fuel concept or as a barrier layer and moderator matrix in MSR with TRISO fuel (Riley et al. 2019). Among other things, the long-lived and biologically active isotope carbon-14 is formed. Suitable disposal routes must also be established for the graphite. Even today, there are already large quantities of nuclear graphite from the operation of gas-cooled, graphite-moderated reactors that still need to be fed into a suitable disposal solution (IAEA 2010c). For information on the disposal of graphite waste,

see Chapter 4.6.6. Depending on the reactor concept, the graphite components in the reactor may also have to be replaced relatively frequently.

The limiting factor for the service life of the graphite components is the damage caused by neutron irradiation. In some reactor concepts, the graphite moderator is one of the reactor components that severely limits the potential service life of the reactor. In some SMR concepts, it even has to be replaced every 7 years. The frequency of replacement depends on several factors, e.g. the level of thermal neutron flux in the reactor and the type of salt. During MSR operation, graphite (from moderator and reflector) and fuel matrix carbon will be in close contact with salts, resulting in limited salt penetration into the pores as well as the embedding of fission products (e.g. Sr, Te, Ba, Mo, Ru), decay products of Xe and Kr (e.g. Rb, Cs, Sr, Ba, Ln, Zr) and small amounts of U (Riley et al. 2019).

After decommissioning, metallic surfaces of an MSR will be covered with a layer of salt and fission products, in particular insoluble precious metals (Mo, Pd, Rh, Ru, Tc, Nb, Sb, Ag), which are preferentially deposited on surfaces within the active zone (Riley et al. 2019). These materials therefore require subsequent interim storage or decontamination for dismantling (radiation protection), with further waste streams and potentially high costs. This applies in particular to structural materials made of steel, which are exposed to high corrosion, as is typical for MSR, and therefore have to be replaced more frequently depending on the reactor concept (e.g. every seven years for Terrestrial Energy’s IMSR) (see also Chapter 4.4.5).

Components close to the core that are exposed to high neutron fluxes also contain activation products. (Krall et al. 2022) Therefore, it is assumed that decontamination for recycling of the material will not be possible and, for example, an MSR such as the IMSR will generate up to 1.0 m^3/MWth-a of steel and nickel alloys as long-lived low and intermediate-level radioactive waste. The waste contains long-lived radioisotopes with half-lives of more than several thousand years (e.g. ^{59}Ni, carbon-14, niobium-94, technetium-99, zirconium-93, ^{93}Mo and chlorine-36) in concentrations that categorise it as low or intermediate-level radioactive waste (Krall et al. 2022).[169] As a distinguishing feature, (Krall et al. 2022) specify a neutron fluence greater than approx. 10^{21} neutrons/cm^2 , i.e. a neutron flux greater than 10^{12} neutrons/cm^2/s, with a lifetime of 60 years and a capacity factor of 70%, above which the structural material is so strongly activated that it must be categorised as intermediate-level radioactive waste (greater than category C waste).

In fast MSR, reflectors and shielding materials made of steel or steel-clad components made of beryllium, magnesium, lead or boron carbide are added.

The figures in (Krall et al. 2022) refer to small facilities (SMR). According to (Krall et al. 2022), MSR generate significantly more waste per amount of electricity generated in the category of low-level and intermediate-level radioactive waste than LWR. According to (Krall et al. 2022), the figure is likely to be lower for larger plants in the gigawatt class, for example in a design such as the LFTR (see Chapter 5.5) in which the neutron fluxes at the edge of the reactor are significantly lower, exposing the steel of the reactor vessel to lower fluxes. MSR also have higher operating

[169] (Krall et al. 2022) categorise the waste as “waste greater than category C”. In the U.S., an average activity of 300-2500 Ci/ft^3 is defined for such waste (waste greater than category C) and the need for deep geological disposal is prescribed for such waste. The activity corresponds to $3.9*10^{14}$ Bq/m^3 to $3.2*10^{15}$ Bq/m^3 and thus in Germany this would be high-level waste (activity > 10^{14} Bq/m^3). Category C waste is waste <7 Ci/ft^3. Greater than Category C waste is waste >$9.1*10^{12}$ Bq/m^3 and thus in Germany intermediate-level radioactive waste up to an activity < 10^{14} Bq/m^3.

temperatures compared to LWR, which means higher efficiency and fewer fission products per energy generated.

For low- and intermediate-level radioactive structural materials with long-lived radioisotopes, (Krall et al. 2022) assume that, in contrast to low- and intermediate-level radioactive structural materials with short-lived radionuclides, these require more complicated disposal procedures, as both the operational radiation exposure and the dose on geological time scales must be taken into account.

Conclusion: Disposal

An MSR produces significantly different waste streams compared to an LWR. The conditioning of the waste must do justice to the different waste streams and a number of simple, yet complex technologies must be used, ranging from the separation of the waste streams to conditioning and emplacement. Due to the higher operating temperatures, MSR have a higher efficiency (approx. 45-48% at 550-600 °C) than LWR and therefore produce approx. 25-30% less fission products per electrical energy generated.

Unlike LWR, in which the majority of the radioactivity remains in the fuel and the fuel elements, in MSR, significantly larger quantities of radioactivity are treated in completely different process streams relative to the energy produced. In MSR, volatile fission products escape from the liquid fuel either as mists or aerosols, which have to be filtered and converted into a stable chemical form for disposal. The liquid fuel is also chemically treated to remove fission products. Many reactor concepts also provide for the separation of actinides. All of these processes generate further secondary waste.

Technologies for treating the various wastes are currently at different stages of research and development. From today’s perspective, however, it is unclear to what extent the waste products from the chemical processing of the molten salt or the spent molten salt itself could be disposed of together with the current high-level waste or to what extent separate repository sites with a suitable host rock would have to be sought and developed for this purpose.

Due to the high solubility of the fuel salt, it can be assumed that the fuel salt can only be disposed of directly in salt as a host rock, if at all. A better solution would be to immobilise the salt in a waste form that is chemically more resistant to the mobilisation of the radioisotopes it contains. It may also make sense to process the fuel salt and separate the waste stream. However, there are still too many gaps in the assessment of the performance of waste packages and separation processes for both chloride and fluoride salts to predict the long-term behaviour of the waste forms in a repository environment.

In principle, nuclear fission in an MSR produces similar quantities of radioactive fission products relative to the heat released as in the LWR nuclear power plants in use today. However, due to the lack of more fully elaborated fuel concepts up to now, it is not yet possible to make more detailed estimates here.

As many details of the various MSR concepts and the associated fuel cycles still remain undefined at this stage of the development process, many questions concerning the analysis of the waste inventories ultimately to be disposed of and their composition remain unanswered.

At present, there are no clear advantages or disadvantages of MSR compared to LWR.

4.4.8 Proliferation risks

The following account was adopted from (Oeko-Institut e.V. 2017) and further supplemented.

In MSR concepts, the use of reprocessing in the reactor concept is generally based on the assumption that separated fissile material can be accessed.

MSR can be operated both with high conversion, i.e. the breeding of new fissile material, and as a “burner” to reduce actinide inventories. As a thermal reactor, MSR are also discussed in their function as actinide “burners” with reprocessing of the molten salt. Due to this “burner” function, continuous reprocessing is not required; instead, the molten salt can also be reprocessed discontinuously “offline”. Concepts of MSR as fast breeder reactors, e.g. MSFR, typically provide for the newly bred fissile material to be separated in an on-site reprocessing plant and fed into the molten salt fuel. In some concepts, the fissile material is also regularly separated from the “breeder” molten salt and stored for later use in the reactor or as “start-up” fissile material for further MSR reactors.

Starting up a thorium-based MSR requires a fissile material quantity corresponding to the equilibrium inventory of a reactor. This fissile material must either be obtained through the reprocessing of spent LWR fuel with the associated proliferation risks or, alternatively, a surplus of fissile material would have to be bred in existing MSR, separated and made available in separated form for use in a new MSR. In thorium fuel cycles, the proliferation risks posed by uranium-233 are also influenced by the uranium-232 content of such uranium-233.

This is a prime example of a fundamental problem in the design of future reactor concepts, which should offer advantages over today’s nuclear energy utilisation with respect to all the evaluation criteria considered. On the one hand (e.g. in the case of MSFR), developers are aiming to optimise the criterion of “resource efficiency” through the conception of a breeding blanket that enables a high utilisation of thorium by converting it into uranium-233. However, this leads to specific proliferation risks arising from the need to handle large quantities of high-purity, separated uranium-233. On the other hand, in order to achieve advantages with regard to the criterion of “proliferation resistance”, or at least to avoid higher proliferation risks, the proposal has been made to dispense with the breeding blanket in MSFR. From today’s perspective, it remains to be seen which criterion will ultimately lead to the success of a specific design. What is clear, however, is that no design is currently being pursued that offers advantages in terms of both resource efficiency and proliferation resistance.

The extent to which fissile materials are available in a form that is not directly nuclear weapons-grade thanks to suitable process control, or how high any remaining barrier to accessing directly weapons-grade material is, essentially depends on the ultimate selection of reactor design and fuel concept. In any case, this will also raise questions about proliferation resistance and the possibilities of fissile material control for MSR. According to (GIF 2011), an advantage with regard to fissile material control is that only the amount of fissile material required for the short-term operation of the reactor needs to be present in the system, as the molten salt fuel is circulated once in the primary cooling circuit within a few seconds. However, this also requires the continuous control of the fissile material content in the molten salt, which requires at least a limited supply of external fissile material.

According to (GIF 2011), the fissile material in the inventory of a running MSFR can be monitored, for example, by measuring the fissile material content in the molten salt during operation or by monitoring the operating temperature of the reactor (which correlates with the fissile material content in the molten salt due to neutron-physical feedback properties). There is no information in (GIF 2011)

on how the molten salt in the dump tanks is to be monitored (in the event of necessary maintenance work on the primary cooling circuit). Inventorying is complicated by the constantly changing composition of the fissile material content, which is due to transmutation, chemical processing, the continuous supply of fissile material and possible deposits of precious metals in the primary circuit.

With regard to on-site reprocessing, (GIF 2011) points out that the actual reprocessing steps involved in the MSFR concept discussed there have not yet been specified enough to allow a detailed analysis of the proliferation risks However, the fact that the uranium-232 content produced and separated at the same time as the uranium-233 makes it more difficult to handle and transport separated uranium due to the high radiation is cited as a fundamental advantage of such on-site reprocessing plants. Furthermore, according to this source, the process control in on-site reprocessing would require strong shielding and automated processing due to the high radioactivity of the molten salt, which also places high passive barriers to the diversion of fissile material. Therefore, they conclude, suitable IAEA safeguards can rule out undetected diversion with a high degree of safety.

(Holcomb 2017b) notes that MSR concepts, depending on the actual design, may be vulnerable or resistant to a diversion of nuclear weapons-grade materials. The approaches to fissile material control (safeguards) change fundamentally in every case, as the fissile material is no longer present in individual, clearly verifiable fuel elements, but in a detached state. The fact that the high radiation of molten salt makes it difficult to realise structural modifications to the plant with the aim of diverting fissile material and that concealed diversion of fissile material would not be possible due to the low excess reactivity during operation are cited as advantages.

(Flanagan 2015) also states that, in MSR, the fissile material is distributed throughout the entire cooling circuit (pipework, heat exchangers, drain pools, etc.), whereas in solid fuels, the fissile material to be monitored is concentrated in the fuel elements, so that fundamentally different safeguards concepts are required. Work on the implementation of safeguards concepts for MSR has been initiated by the U.S. National Nuclear Security Administration of the DOE (Holcomb 2017b). The primary aim of this work is to support the IAEA in the development of safeguards technologies and strategies. This work should be incorporated into the design of MSR concepts as early as possible, as later adaptation typically entails higher costs and restrictions in plant availability (Flanagan 2015).

With regard to the properties of the physical protection of an MSFR, (GIF 2011) concludes that it is not yet possible to make any statements about this, as a sufficiently advanced design is not yet available.

Large quantities of tritium will continue to be produced in MSR. Tritium is an isotope required for advanced nuclear weapon designs. In boosted nuclear weapons, tritium is used to increase the explosive power or to miniaturise nuclear weapon designs.

Lithium in the form of lithium deuteride also plays an important role in advanced nuclear weapons. Neutron capture in lithium-6 results in the production of tritium, which contributes significantly to the release of energy in thermonuclear weapons through a fusion reaction with deuterium. Technologies for the enrichment of lithium-6 therefore also exhibit a dual-use character. Since the use of enriched lithium-7 for MSR is currently under discussion, the fate of the lithium-6 separated in the process and the monitoring of enrichment technologies with regard to the proliferation risks of MSR must also be taken into account. Patents have been filed in China, for example, for new enrichment processes for lithium-7 in connection with the development of MSR concepts (Xu 2015). This problem also remains unaddressed in (GIF 2011).

Conclusion: Proliferation risks

The proliferation resistance of MSR depends to a high degree on the concept under consideration and the associated fuel cycle. Material inventorying is more difficult in MSR concepts using liquid fuel. Reprocessing increases potential proliferation risks and requires greater effort for fissile material control and monitoring measures. Some MSR concepts employing thorium as fuel require the use of separated fissile material in order to produce the fuel to start the reactor. However, the proliferation risks associated with using fissile material in MSR represent neither an advantage nor a clear disadvantage compared to technologies currently used in the LWR fuel cycle, such as uranium enrichment or plutonium reprocessing.

4.4.9 Costs

To date, information on the costs of MSR has been sporadic. A number of reactor concepts eliminate the costs of transporting and manufacturing new fuel elements, though some require large components of the reactor system to be replaced at periodic intervals.

The estimated costs for AHTR are less than USD 1000 per kW of installed capacity (WNA 2021b).

For the IMSR-400, it has been indicated that the electricity price for the largest plant concept could be competitive with gas (WNA 2021b).

The manufacturer ThorCon reports electricity costs of 3 to 5 cents per kWh, depending on the size of the plant (WNA 2021b).

Conclusion: Costs

No reliable statements can be made at present with regard to needed investments, the necessary construction times, operating costs, service life and capacity utilisation.

The risks for investors are high, there is no comparable experience available to date.

4.5 Supercritical water-cooled reactors (SCWR)

The basic idea of a supercritical water-cooled reactor (SCWR) is to achieve a higher steam temperature (along with a higher steam pressure) of the live steam compared to today’s light and heavy-water reactors, thereby increasing the efficiency of the reactor system (GIF 2021a; Schulenberg 2020; Huang et al. 2021; Schulenberg and Leung 2016).

The idea goes back to improvements in modern coal-fired power stations, in which the steam temperature has been raised to the supercritical water range in recent decades, so that they now have an efficiency in the region of 45%. To achieve this, the water in coal-fired power stations is first heated, and the water in the critical state is then heated further in superheaters to temperatures of up to approx. 600 °C. This was made possible, above all, by improved materials, which enable the operation of conventional power plants at high temperatures (Schulenberg 2020).

In contrast, the efficiency of today’s light- and heavy-water reactors remains in the region of 35%, as the pressure and temperature of the live steam are around 7 MPa and 290 °C, respectively, while the temperature rise of the coolant as it passes through the reactor is a mere 10 °C (IAEA 2023d).

4.5.1 System description

In an SCWR, the coolant in the reactor should be heated to temperatures above the “critical point of water”, i.e. a temperature of 374 °C and a pressure of 22.1 MPa. Above 374 °C, the surface tension in water approaches zero. As a result, no more water droplets remain in the steam when the water evaporates. Moreover, it is no longer necessary to separate steam and water droplets as required in the steam generators of pressurised water reactors or in the steam space of boiling water reactors (Schulenberg 2020).

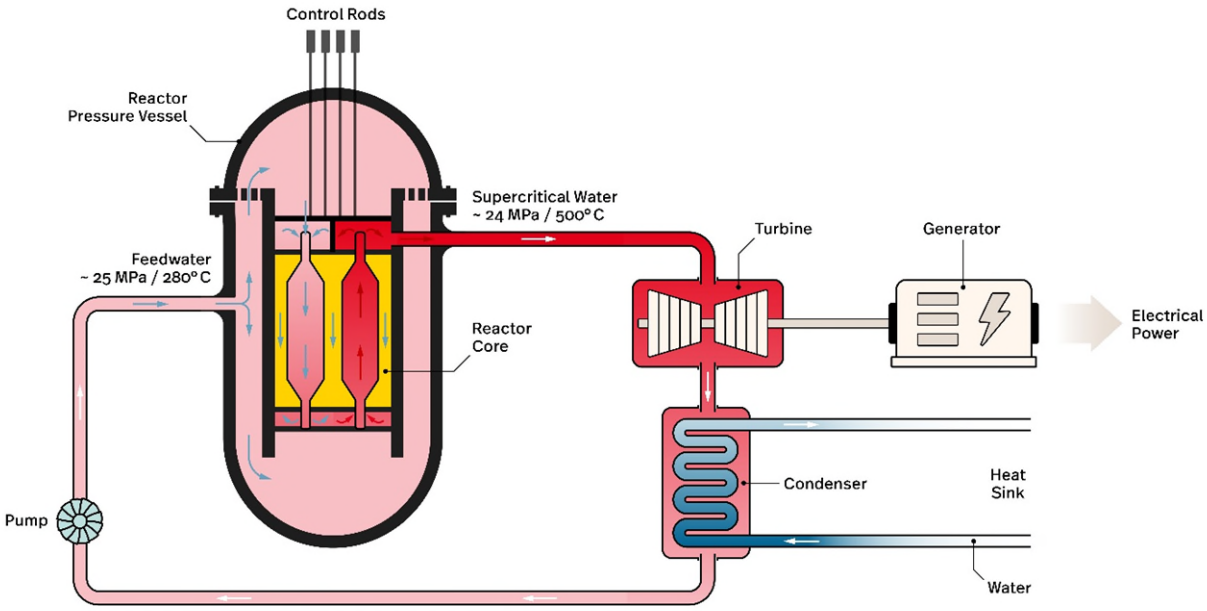
Within the SCWR technology line, the distinction can be made between reactor concepts with pressure vessels corresponding to light-water reactors currently in use and reactor concepts with pressure tubes based on today’s CANDU reactors. Among these reactor concepts, those with a fast neutron spectrum are occasionally discussed in addition to those with a thermal neutron spectrum (GIF 2021a). In this context, (Schulenberg and Leung 2016) have found that, at coolant temperatures above 390 °C, the coolant density becomes so low that a thermal neutron spectrum no longer arises in the reactor core. For this reason, thermal reactor concepts require targeted additional moderation in the form of water gaps in fuel elements in the case of pressure vessel concepts or heavy water in the case of pressure tube concepts.

The following discussion focuses on concepts with pressure vessels, as light-water reactors greatly outnumber pressure tube reactors among the nuclear reactors in use today. The IAEA’s ARIS database also only lists reactor concepts with pressure vessels for the SCWR technology line (IAEA 2023d). Chapter 4.5.3, in particular, discusses important differences within the technology line.

Reactor system

The design of an SCWR (see Figure 4-6) is very similar to designs employed by current pressurised and boiling water reactors (PWR and BWR). As it passes through the reactor core, the coolant is heated at a pressure of approx. 25 MPa to temperatures in the vicinity of 500 °C, i.e. the supercritical region of water. The supercritical water is therefore produced as it passes through the reactor core in a way similar to live steam in modern BWR. However, as the supercritical water does not contain any water droplets, SCWR do not require a water-steam separator and circulation pumps in the reactor pressure vessel, as is the case with BWR. This allows a more compact design of the reactor pressure vessel. At the same time, no steam generators or pressurisers are required, as is the case with current PWR. An SCWR thus combines features of both BWR and PWR.

Figure 4-6: Conceptual diagram of a supercritical water-cooled reactor (SCWR)



Source: Own illustration

The supercritical water is fed to a high-pressure turbine. In various further stages, the steam emerging from the high-pressure turbine can be superheated again and fed to upstream turbine stages. The turbines drive the generator to produce electricity. The expanded steam is condensed in a condenser, while the remaining heat is transfered to the main heat sink. From the condenser, the coolant is fed back into the reactor pressure vessel by the feed water pump. The coolant is typically preheated to approx. 280 °C by means of intermediate superheaters (not shown in the figure).

The reactor pressure vessel must be designed for pressures in the range of 24-25 MPa, i.e. it has a higher wall thickness compared to current pressurised and boiling water reactors. Furthermore, due to its material properties, the steel of the reactor pressure vessel should not be exposed to high steam temperatures in the region of 500 °C. Therefore, the supercritical water produced must not come into direct contact with the wall of the reactor pressure vessel, which requires appropriate internals for flow routing.

As it passes through the reactor core, the coolant is heated from approx. 280 °C to approx. 500 °C. In addition to the density of the coolant, other important thermodynamic variables (such as viscosity or heat capacity and conductivity) also change. For this reason, some reactor concepts require the coolant to be routed through the reactor core multiple times in order to ensure uniform moderation in the reactor core (in thermal reactor concepts) along with the uniform cooling of the fuel elements. Figure 4-6 shows a flow routing in which the coolant first flows from above through the area between the fuel elements and through a section of the fuel elements themselves. The preheated coolant is then collected and mixed below the reactor core and flows back up through another section of the fuel elements, where it is heated to the target temperature. The coolant is then mixed again in an upper collector before it is fed out of the RPV. In other concepts, the coolant may be routed through the reactor core even more frequently; for example, see the description of the HP-LWR in Chapter 4.5.3.

The reactor power is controlled by control rods, which, in most reactor concepts, such as the PWR, are introduced into the reactor pressure vessel from above. Since the coolant evaporates as it passes through the reactor core, boric acid cannot be added to the coolant to compensate for long-term reactivity changes.

Fuel

The fuels for the reactor concepts included in (IAEA 2023d) corresponds to those used for today’s light and heavy water reactors, with uranium dioxide fuels taking first place. However, due to the changed operating conditions in the reactor core (higher temperatures and pressures, corrosion behaviour), zirconium cannot be used as a fuel cladding material, in lieu of which stainless steels are designated for use. For the burnup-related compensation of reactivity losses, the use of gadolinium is planned due to its capacity as a burnup-capable neutron absorber, i.e. an absorber of neutrons that is burnt up during reactor operation itself, as a means to compensate for burnup-related reactivity changes, comparable to current BWR.

Thorium fuels with a proportion of plutonium as the initial fissile material are also being discussed for the Canadian pressure tube reactor concept, as are uranium-plutonium mixed oxide fuels for concepts with a fast neutron spectrum (Huang et al. 2021).

Fuel cycle

Since the structural materials in the reactor core (fuel cladding, fuel element boxes, etc.) have slightly higher capture cross-sections for neutrons, a slightly higher initial enrichment is required for the same burnup as in modern light- and heavy-water reactors; for a target burnup of 45 MWd/kg of heavy metal, an initial enrichment of around 6-7% is specified, and around 9% for a target burnup of 60 MWd/kg. The specified cycle length is 10-18 months (IAEA 2023d). Reprocessing of fuels is neither provided for nor necessary at the level of the technology line, but is nevertheless under discussion for some reactor concepts, see Chapter 4.5.7. There are no other relevant differences to modern light- and heavy-water reactors.

Coolant, temperature and pressure

All reactor concepts use light water as the coolant, which is heated to temperatures above the critical point (374 °C and pressure of 22.1 MPa) as it passes through the reactor core. Operating pressures of 24-25 MPa, inlet temperatures of the coolant into the RPV of 280-310 °C and outlet temperatures of the steam of 500-560 °C are discussed in this context (IAEA 2023d). For the Canadian pressure tube reactor, efforts are being made to achieve outlet temperatures as high as 625 °C (Yetisir et al. 2016).

Spectrum (moderator)

The majority of SCWR reactor concepts provide for a thermal neutron spectrum (IAEA 2023d).

In reactor concepts with pressure vessels, the coolant itself is also used for moderation. Due to the very low density of the steam still produced in the reactor core, different concepts provide for different **flow routing**s for routing the coolant through the reactor. The coolant is fed through the reactor core a certain number of times, depending on the concept, in order to ensure sufficient moderation and cooling of the fuel (Schulenberg 2020). For example, Figure 4-6 shows an exemplary case in which the coolant is first routed through a section of the fuel elements from above, after which the already preheated coolant flows again from below through another section of the fuel elements, where it is heated up to the core outlet temperatures. A further partial flow of the coolant is first channelled between the fuel elements, ensuring the desired moderation.

In reactor concepts with pressure tubes, moderation is ensured by embedding the pressure tubes in a tank with heavy water (Yetisir et al. 2016).

In isolated cases, reactor concepts are also discussed that dispense with additional moderation. Despite the use of light water as a coolant, a fast neutron spectrum is nevertheless used due to the very low density in the reactor core (Oka 2010; GIF 2022c).

Construction materials

The enable the use of modern RPV materials (e.g. 508-III steel, 20MnMoNi 5 5) for the reactor pressure vessels, the walls of the reactor pressure vessels must not come into direct contact with the very high steam temperatures. For this purpose, the coolant injected at temperatures below 300 °C fills the entire interior of the RPV, and the resulting superheated steam is separated from the RPV wall by RPV internals. However, due to the high pressure, the reactor pressure vessels require wall thicknesses of around 440-450 mm, which is almost twice as thick as current light-water reactors (EPR: 250 mm, AP1000: 203 mm). Due to the corrosion properties of the supercritical water, stainless steels are used as structural materials for the fuel rod cladding tubes, the fuel element structural materials and other core internals that come into direct contact with steam at high temperatures (IAEA 2023d; Schulenberg 2020).

4.5.2 Historical developments

Early reactor concepts for SCWR were discussed in the 1950s and 1960s (Huang et al. 2021). In 1957, the U.S. nuclear power plant manufacturer Westinghouse developed a design for a reactor moderated with light water and cooled with supercritical water with a planned core outlet temperature of 538 °C and a thermal output of 70 MW. This was followed by a design for a graphite-moderated pressure tube reactor cooled with light water with an electrical output of 1000 MW in 1962 and, in 1966, a design for a pressurised water reactor cooled and moderated with light water with an

electrical output of 800 MW. As early as 1959, General Electric elaborated the concept of a heavy-water moderated, light-water cooled reactor with a thermal output of 300 MW.

According to (Leung 2017), Babcock & Wilcox pursued a concept for a breeder reactor with a fast neutron spectrum and an operating pressure of 25.3 MPa at an operating temperature of 538 °C.

However, no nuclear reactor with supercritical water has been built to date (Schulenberg and Leung 2016). Though a number of reactors producing superheated steam have been built and operated, these did not work in the pressure range of supercritical water. (Leung 2017) refers in this context to the Russian reactors Beloyarsk-1 and -2 (AMB-100 and AMB-200), which were light water-cooled, graphite-moderated reactors that achieved a steam temperature of 510 °C, as well as to the German superheated steam reactor.

The superheated steam reactor (Großwelzheim Nuclear Power Plant) developed and built in Karlstein am Main, Germany can be seen as a precursor to the SCWR concepts of today (Schulenberg 2020; atw - atomwirtschaft atomtechnik (atw) 1969). This reactor was built by AEG between 1965 and 1969 and had a thermal output of 100 MW. At a pressure of 7.33 MPa, live steam was initially to be generated at a temperature of 457 °C with the aim of reaching a steam temperature of 500 °C. A significant difference to the boiling water reactors of the time was that the saturated steam generated during the (first) pass through the reactor core was not decoupled directly from the reactor; instead, the saturated steam was collected behind the steam separators and passed through the fuel elements multiple times in superheated steam pipes. Only this superheated steam was decoupled from the reactor via a superheated steam collector. Trial operation of this reactor began in autumn 1970. However, it was finally shut down again in the spring of 1971, and the concept was not pursued further in Germany. After further use as a test stand, the superheated steam reactor was completely dismantled in the 1990s.

(Schulenberg 2020) points out that the reasons for decommissioning the reactor were not released to the public. Following discussions with engineers involved in the development at the time, he cites a collapse of the superheated steam pipes as a possible cause. While the fuel elements with the superheated steam pipes were previously tested with electrically heated prototypes, strong local differences in the power release in the reactor core may have resulted in significantly higher temperatures in individual superheated steam tubes, which the structural materials used could not withstand.

There were more recent developments as early as 1986 in Russia, where the design for an SCWR reactor with two circuits and a planned electrical output of 500 MW was explored (Huang et al. 2021).

4.5.3 Current developments

Work on SCWR was taken up again in the scope of the Generation IV International Forum. The IAEA ARIS database currently lists three reactor concepts under the SCWR technology line: CSR1000 (China), HP-LWR (Europe) and JSCWR (Japan) (IAEA 2023d). (GIF 2021a) also cites developments in other countries, above all Canada and Russia. According to (Huang et al. 2021), a research programme in the area of SCWR development was also carried out in the U.S. between 1999 and 2006. Moreover, (IAEA 2014) discusses various SCWR designs from China, the EU, Japan, Canada, South Korea, Russia and the U.S.

Overview of the reactor concepts in the technology line

Significant differences within the technology line result from whether the reactor core is enclosed in a pressure vessel and cooled and moderated by light water or whether the fuel elements are enclosed in individual pressure tubes, with light water being used for cooling and heavy water for moderation (Huang et al. 2021).

Concerning concepts with pressure vessels, discussions also include concepts in which the low density of the coolant in the reactor core results in a fast neutron spectrum (see Table 4-9); these concepts are not considered further in detail below due to the fundamentally higher development requirements compared to thermal SCWR concepts.

In addition, (Huang et al. 2021) points out that various developments for SCWR are also being pursued in the form of SMR concepts. The main variants of SCWR concepts currently in development are briefly described below. Special emphasis is placed here on aspects in which the reactor concept deviates from the above general description of an SCWR.

Table 4-9: Key reactor concepts for the SCWR technology line

Reactor concept	Type	Spectrum	Coolant	Moderator	Outlet temperature
CSR1000 (China)	Pressure vessel	Thermal	Light water	Light water	500 °C
HP-LWR (Europe)	Pressure vessel	Thermal	Light water	Light water	500 °C
JSCWR (Japan)	Pressure vessel	Thermal	Light water	Light water	560 °C
Canadian SCWR	Pressure tubes	Thermal	Light water	Heavy water	625 °C
Japan	Pressure vessel	Fast	Light water	N.A.	501 °C
Russia	Pressure vessel	Fast	Light water	N.A.	540 °C

Source: (IAEA 2023d; Huang et al. 2021)

CSR1000

The CSR1000 is currently being developed in China. This reactor concept is presented in greater detail on the basis of an example in Chapter 5.7.

HP-LWR

The concept of the “high performance light-water reactor” (HP-LWR) was developed in the 2000s under the leadership of the Karlsruhe Institute of Technology (KIT) in Germany (KIT 2012; IKET 2011; 2008b; 2008a). According to the developers, one of the main reasons for these research activities was based on the need registered both in Germany and internationally to build up new expertise in view of the generation change imminently faced by the original reactor developers (Schulenberg 2020, p. 45). The latest information on the HP-LWR contained in the IAEA’s ARIS database dates from 2011 (KIT 2011). The following is therefore based on the more recent account in (Schulenberg 2020).

The HP-LWR was conceived as a reactor with an electrical output of 1000 MW. The design essentially corresponds to the basic design of the technology line outlined above.

In the HP-LWR, the feed water enters the RPV at a temperature of 280 °C and a pressure of 25 MPa, fills it completely and also flows through gaps between fuel element boxes and an internal water gap in the fuel elements. It also acts to moderate the neutrons. Heated to approx. 320 °C, the coolant then enters the interior of the fuel elements from below for the first time and flows through the central third of the fuel elements in the reactor core, i.e. those with the highest power. In the process, it is heated up to approx. 400 °C, i.e. a temperature above the critical point. In an upper mixing chamber, the coolant flow is mixed to equalise temperature inhomogeneities, and the flow direction is reversed for the first time. The coolant then flows through a second third of the fuel elements and is heated further in the process. This coolant, which is simply overheated, is collected and mixed again in a lower mixing chamber and then flows through the last third of the fuel elements at the outer edge of the reactor core, reaching its core outlet temperature of approx. 500 °C.

JSCWR

Toshiba is pursuing the concept of the “Japanese Supercritical Water-Cooled Reactor” (JSCWR). The most recent information about this concept in the ARIS database dates back to 2011 (ARIS 2011b).

According to this source, the development of the JSCWR can be traced back to work at the University of Tokyo starting in 1989. The project was funded by the Japanese Ministry of Economy, Trade and Industry (METI) and jointly pursued by Toshiba, Hitachi-GE, the University of Tokyo, Kyushu University, Kyoto University, the Institute of Applied Energy and the Japan Atomic Energy Agency. The JSCWR is intended to generate a thermal output of 3681 MW and an electrical output of 1700 MW. It is primarily intended to be used for base load supply.

The planned fuel is uranium dioxide or MOX fuel comparable to those used in today’s light-water reactors, with an enrichment of 7% at a target burnup of 45 MWd/kg of heavy metal and stainless steel fuel cladding material. Gadolinium is used as a burnable poison. The coolant enters the reactor at a pressure of 25 MPa and a temperature of 290 °C and is heated to a temperature of 560 °C.

As with BWR currently in use, reactivity control is implemented in the form of control elements that enter the RPV from below. The safety concept corresponds to that of today’s advanced light-water reactors.

The JSCWR is to operate in a closed fuel cycle with reprocessing and plutonium utilisation.

Canadian SCWR

In Canada, the development of an SCWR based on the Canadian CANDU reactor concept has been in progress since the 2000s. The following account is based on (Yetisir et al. 2016).

The Canadian SCWR is a reactor with an electrical output of 1200 MW at a steam pressure of 25 MPa and a steam temperature of up to 625 °C. The efficiency of the reactor is 47%. The reactor is expected to reach a minimum service life of 40 years, with up to 75 years considered possible.

In this reactor concept, the fuel elements are located in individual vertically arranged pressure tubes. Light water is used as the coolant. The pressure tubes are located in a moderator tank, which is filled with heavy water for use as a neutron moderator. The intended temperature of the heavy water is approx. 100 °C, which prevents the walls of the pressure tubes from exceeding a temperature of 120-150 °C. The pressure in the moderator tank is 0.35 MPa.

Unlike previous CANDU reactors, the fuel is loaded and unloaded in batch operation. The fuel elements are inserted into the reactor for three cycles before they are removed. For fuel, the concept provides for thorium dioxide containing 15% or 12% by weight (inner or outer fuel rod ring) of plutonium dioxide as fissile material.

International research activities

In the framework of the GIF,

- system integration,
- materials and chemistry, and
- thermal hydraulics and safety

are cited as current areas of development.

(GIF 2021a) cites, in reference to system integration, the joint European, Canadian and Chinese development programme (ECC-SMART) launched in September 2020 for a supercritical water-cooled SMR, as well as two projects funded by the Chinese Ministry of Science and Technology (MOST) for the further development of the CSR1000 concept and the creation of a knowledge base of previous experiments in the field of SCWR development. (Huang et al. 2020) indicates 925 person-months as the total scope of ECC-SMART.

With regard to material development, (GIF 2021a) also underlines activities within the scope of ECC-SMART with the aim of analysing fuel cladding materials, in particular, for their suitability for use in SCWR. Corresponding investigations are also being funded in China and Canada.

With regard to thermal hydraulics and safety, (GIF 2021a) points out work being carried out in the scope of the ECC-SMART programme and by two research institutions based in Hungary. Furthermore, reference is made in (GIF 2021a) to the installation of a test cooling circuit for fuel element qualification for SCWR in the Czech research reactor LVR-15, which is currently in the approval process. In China, two benchmark studies for thermal-hydraulic calculation programs were launched in November 2020. Current development activities in Canada are focussing on design work for an SCWR-SMR.

4.5.4 Technical development status

In terms of their basic design and safety concept, SCWR are very similar to the modern light-water reactors of today. Nevertheless, further research and development must be done in this field, particularly in the areas of fuel cladding and structural materials, safety functions and safety demonstration principles, before an initial prototype reactor can be built.

According to (GIF 2021a), suitable materials for use in the reactor still need to be developed and tested. (KIT 2011) also cites the provision of suitable structural materials for use in the reactor core as an essential development step. In the interest of identifying suitable structural materials, 16 materials were tested in autoclaves for their corrosion resistance up to 650 °C. According to the study, no suitable material could be found for this temperature range, particularly for such thin-walled components as fuel cladding materials, with the materials analysed demonstrating a temperature limit of 550 °C for the fuel cladding. (Schulenberg and Leung 2016) points out that the ferritic-martensitic steels used in modern coal-fired power plants, in particular, do not have sufficient corrosion resistance for thin-walled components like fuel cladding. (Leung 2017) describes various candidates under investigation for suitable fuel cladding materials. (Huang et al. 2021) also notes that important structural materials have been identified whose suitability needs to be verified in the scope of further studies. For temperatures above 600 °C, however, no suitable fuel cladding material has been identified. Further materials are being investigated on an ongoing basis; for example, see (Bai et al. 2018).

(Schulenberg 2020) views the current safety concepts of modern BWR as being fundamentally transferable to SCWR systems. However, he assumes that previous containment concepts for SCWR plants would have to be recalculated and tested. (Schulenberg 2020, p. 51) underlines the need to calculate and test all relevant components of an SCWR in detail before such a concept can be transformed into a functional power plant. With regard to current developments here, (Schulenberg 2020, p. 57) points towards China, but acknowledges that he considers the construction of such a reactor in China unlikely due to the high financial risk it poses for the power plant manufacturer. (Schulenberg 2020, p. 58) therefore concludes:

> *“The designs developed until now should therefore be regarded more as visions or suggestions and not as drawings for the construction of a power plant.”*

A further development requirement identified by (KIT 2011) is the reliable prediction of neutron-physical and thermal-hydraulic phenomena under the conditions of cooling with supercritical water. With regard to coupled neutron-physical and thermal-hydraulic effects in the reactor core, (KIT 2011) asserts that coupling and iterative calculation in neutronics and thermal-hydraulics programs is possible for equilibrium states (see also (IKET 2006; 2009a)). For transient states (accident analyses), however, suitable calculation programs would still have to be developed. Therefore, the source concludes, realistic tests for the selection of structural materials and flow tests on test fuel elements are first required for further development, and integral tests are necessary for certain safety functions. Finally, it is necessary to design, build and test a low-power prototype reactor in order to validate the overall concept. Therefore, (KIT 2011) sees the development of SCWR more as a long-term continuous further development of current light-water reactors.

(IAEA 2014) presents a summary of findings on the thermal-hydraulic behaviour of SCWR, existing calculation programs and plans for the further validation of calculation programs for SCWR. (Hummel and Novog 2016) report on the results of coupled thermal-hydraulic calculations for transients of the Canadian SCWR. They show that transients can lead to strong power increases, which in turn lead to the self-shutdown of the reactor with a certain time delay due to inherent feedback effects. However, they also cite the possible need for rapid shutdown systems, such as those used in today’s CANDU reactors, in order to ensure the safety of such transients.

In the last GIF roadmap (GIF 2014), the development status of SCWR was estimated to be at the end of the first development phase, the “viability phase” of the GIF, see Chapter 2.6.1. According to this roadmap, out-of-pile testing of experimental fuel elements, the selection of fuel cladding materials and the qualification of calculation methods should have been carried out by 2015, a decision on an SCWR prototype reactor should have been made by 2017 and in-pile testing of experimental fuel elements should have been carried out between 2017 and 2022. The “demonstration phase” of the GIF would then be reached in 2025.

A test stand for fuel qualification is planned at the research reactor of the Research Centre Rez (CVR) in the Czech Republic (Huang et al. 2015; Schulenberg et al. 2012; Raqué and Schulenberg 2011). According to (Schulenberg et al. 2012), this was to be built between 2012 and 2014 and initial tests were to be carried out between 2014 and 2018. (Huang et al. 2021) report that this has now been built and is due to be put into operation shortly. Integral test facilities are cited in (Huang et al. 2021) as necessary for further development. Tests on individual fuel rods and fuel bundles of three, four or seven fuel rods as well as benchmark calculations have been carried out (Huang et al. 2015; Rohde et al. 2016; Huang et al. 2020), but not yet for complete fuel elements (Huang et al. 2021). Further findings from international research are summarised in (Huang et al. 2020).

(GIF 2021a) also sees a significant need for further research in the area of thermal-hydraulic data for supercritical water with prototypical application parameters of SCWR. Even if the failure and accident scenarios for SCWR are comparable to those of current LWR, the thermal-hydraulic calculation programs must be re-validated for purposes of verification due to the considerable differences in the thermal-hydraulic properties at high temperatures and pressures.

The development status of the JSCWR is assessed in (ARIS 2011b) as being at the end of the “viability phase” of the GIF. The second phase, the “performance base”, should begin in 2011 and reach completion by 2020. A prototype reactor was to be planned, approved and built in the 2020s. The first commercial JSCWR reactor set to be built in 2030.

Conclusion: Technical development status

With regard to the development of fuel cladding and structural materials, findings are therefore available from laboratory tests and initial integral tests with a small number of fuel rods that apply to the operating conditions of SCWR. This means that the status of material development is still in the area of “applied research” at the border to that of “development”.

As far as the safety functions required for a complete reactor concept are concerned, the measures and equipment of today’s LWR are largely transferable to SCWR concepts. However, the need exists to confirm transferability, e.g. during transients, loss-of-coolant accidents or during start-up and shutdown processes, particularly in the area of core cooling, see also Chapter 4.5.4. The status of the development of required safety functions can thus be assessed as in the “development” area.

Extensive integral test facilities and a demonstration reactor are still required for the demonstration of a final reactor concept. These are also necessary in order to obtain the data required for verification and, on this basis, to transfer or extend existing calculation programs for verification used for LWR to SCWR, particularly with regard to coupled thermal-hydraulic effects, as well as to validate the results obtained with these calculation programs. This means that the methods for the verification of SCWR are still in the area of “applied research” at the border to the area of “development”.

In view of these circumstances, the authors of this report assess the current state of development of SCWR as being in the area of “applied research” at the border to the area of “development”.

4.5.5 Safety

As it flows through the reactor core, the coolant in an SCWR is converted into supercritical water. This significantly reduces the density of the coolant and thus the coolability of the fuel rods. At the same time, the decrease in density with increasing temperature results in higher flow velocities and thus higher pressure losses when the coolant flows through the fuel elements. In the absence of further measures, therefore, the coolant throughput for fuel elements with higher power would be lower, which may result in considerable temperature differences between the fuel elements in the reactor core. To compensate for this effect, a suitable flow routing of the coolant is required in the core (Schulenberg 2020). A higher number of coolant passages through the reactor core leads to greater homogeneity in coolability and in power release in the core, yet it also increases the complexity of the core structure (Schulenberg and Leung 2016).

(Schulenberg and Leung 2016) point out that there are safety-relevant differences compared to the LWR used today, particularly with respect to the start-up of an SCWR. For example, a dryout cannot be avoided during start-up in the subcritical operating temperature and pressure range, for which reason the maximum fuel cladding temperature must be monitored during start-up. If necessary, special start-up systems must be developed to prevent dryout problems. Furthermore, the occurrence of neutron flux oscillations in the reactor core, which are also possible in today’s BWR, must also be prevented in SCWR. However, due to the very strong change in density of the coolant as it flows through the reactor core, this effect is fundamentally more pronounced in SCWR than in modern BWR and must be prevented by implementing suitable technical measures (IKET 2009b). Finally, the occurrence of xenon oscillations cannot be ruled out, which must also be prevented by taking suitable measures, such as partially retracted control elements during power operation (Schulenberg and Leung 2016).

In all the SCWR concepts currently under discussion, the steam generated in the reactor core is fed directly to the turbine. Unlike BWR, in which radioactivity in the coolant largely remains in the aqueous phase in the reactor pressure vessel, the supercritical water results in a higher input of activity into the conventional plant components (turbine, condenser, etc.). This entails higher radiation protection requirements during normal operation and maintenance (Schulenberg and Leung 2016).

(Huang et al. 2021, p. 578; Schulenberg and Leung 2016, p. 190) see safety advantages in the fact that, due to the single-phase coolant used, there is no boiling crisis in SCWR in the event of reactivity accidents or coolant losses with a small leakage cross-section. At the same time, however, (Huang et al. 2021, p. 579) suggests that the higher operating pressure and the lower coolant inventory in the RPV create a greater need for the intervention of technical safety systems in the event of loss-of-coolant accidents.

Highlighting a safety-related difference to current LWR, (KIT 2011) states that, due to the different flow routing in HP-LWR, a coolant circulation must always be ensured for successful core cooling, as covering the core with coolant alone is not sufficient. For this reason, the rapid automatic depressurisation of the reactor pressure vessel is required as a safety function in the event of an accident.

A safety concept comparable to those used in modern boiling water reactors such as the ESBWR or the BWR-1000 is planned for the HP-LWR (Schulenberg 2020). The reactor is enclosed by a prestressed concrete containment. The pipes for the supercritical water and the feed water that pass through the containment can each be shut off with two passive check valves. An emergency shutdown is performed by inserting the control elements. A boron feed system is provided for as a diverse shutdown system. After a shutdown, the resulting live steam is blown out of the reactor core into a condensation chamber in the containment by means of rapid automatic depressurisation. An emergency cooling system pumps cooling water from the condensation chamber back into the reactor. An emergency condenser located in the containment is also included for passive heat removal. In the event of loss-of-coolant accidents, steam released into the containment also enters the condensation chamber through condensation pipes, which limits the pressure build-up in the containment. The other SCWR reactor concepts also feature comparable safety concepts (Huang et al. 2021).

The use of stainless steel as structural and fuel cladding materials reduces the risk of hydrogen formation in the event of accidents due to reactions with the zirconium used as fuel cladding material in LWR (Huang et al. 2021). However, the use of advanced fuel cladding materials to reduce this risk (“accident tolerant fuels” (ATF)) is also being discussed for future LWR.

(Schulenberg and Leung 2016) point out that accident events with multiple failures of safety equipment and extreme external impacts, such as the crash of a military aircraft or a large civil aircraft, must be considered as design extension scenarios for SCWR concepts, as well. It is also necessary to provide for mitigative measures for serious accidents and the exclusion of accident sequences with early containment failure.

SCWR concepts with a fast neutron spectrum pose the additional problem that a loss of coolant can potentially result in increased reactivity in the reactor core, which would have to be excluded by means of suitable design principles (Schulenberg and Leung 2016).

(Schulenberg 2020, p. 51) does not see any relevant differences overall in the safety concept of SCWR compared to today’s boiling water reactors, but insists that the functionality of the safety systems for supercritical water must first be demonstrated (Schulenberg 2020, p. 58).

Conclusion: Safety

Overall, the safety characteristics and the safety concept of SCWR are largely comparable to modern light-water reactors.

Minor differences arise, for example, with regard to the reduced risk of hydrogen explosions in the event of serious accidents, but these are offset by a more complex reactor core geometry, resulting in higher verification requirements with regard to neutron-physical and thermal-hydraulic effects.

At the level of the technology line, no significant advantage or disadvantage can be assumed overall compared to the LWR in operation today.

4.5.6 Fuel supply and waste disposal

In SCWR, the fuels and fuel concepts are comparable to those of modern light- and heavy-water reactors. Due to the higher neutron losses in the reactor core resulting from the necessity to use different structural materials, a higher initial enrichment is required to achieve the same target burnup. However, due to the higher efficiency in the region of approx. 45% compared to values of today’s reactors, which lies in the region of approx. 35%, the amount of fuel required to generate the same amount of electrical energy is expected to be reduced by approx. 25-30%, meaning a correspondingly lower mass of spent fuel per kilowatt hour of electrical energy produced. (KIT 2011) suggests that the activity and heat generation of a spent HP-LWR fuel element are comparable to those of spent fuel in current light-water reactors.

No other significant differences are recognisable from today’s perspective.

Conclusion: Fuel supply and waste disposal

Due to the inherent properties of this technology line, in particular its higher efficiency, the advantage of a 25-30% reduction in the mass of spent fuel per kilowatt hour of electrical energy produced compared to current light-water reactors can be expected.

4.5.7 Proliferation risks

In SCWR, the fuels and fuel concepts are comparable to those of modern light- and heavy-water reactors. Due to the higher neutron losses in the reactor core resulting from the necessity to use different structural materials, a higher initial enrichment is required to achieve the same target burnup. However, at approx. 6-9%, this is still within the low enrichment range (< 20% uranium-235). The higher degree of enrichment required is offset by a slightly lower fuel requirement to produce the same amount of energy. No other significant differences are recognisable from today’s perspective. (Huang et al. 2021) also sees no significant differences for SCWR concepts with regard to proliferation risks if current uranium fuels are used.

The reprocessing of fuels is not intended or required at the technology line level. Reactor concepts with a fast neutron spectrum, however, provide for the generation of new fissile material in a breeder blankets with subsequent reprocessing (Huang et al. 2021). A closed fuel cycle with reprocessing and plutonium utilisation is also envisaged for the JSCWR (ARIS 2011b). (KIT 2011) also points out

that, in addition to the plutonium produced, approx. 2% uranium-235 remains in the spent fuel of an HP-LWR and that a closed fuel cycle could therefore be suitable for this concept. There may therefore be fundamental differences here with regard to individual reactor concepts, and these must be assessed at the level of the specific reactor concept.

(Ibrahim et al. 2022) , for example, explore the attractiveness of fissile materials for the Canadian SCWR based on a rating scale according to (Bathke et al. 2012). Here, they start from the plutonium/thorium fuel cycle included in the concept and investigate variants with the addition of uranium to the Pu/Th fuel or uranium fuels. Their finding is that adding 9% natural uranium to the fresh fuel minimises the quality of the total uranium (especially uranium-238 and uranium-233) remaining in the spent fuel for weapons applications. They rate the Pu/Th fuel cycle as having the lowest proliferation resistance compared to the other fuel cycles analysed.

Conclusion: Proliferation risks

With regard to possible proliferation risks, there are no fundamental advantages or disadvantages at the technology line level compared to today’s light-water reactors. A more detailed discussion of the advantages and disadvantages of various reactor concepts for SCWR can be found in (GIF 2022c). Differences at the reactor concept level are mainly based on the choice of fuel cycle.

4.5.8 Costs

(GIF 2021a) points to improved economics as a major advantage of SCWR development due to the higher efficiency of SCWR reactors and the higher potential to simplify the reactor system compared to current light-water reactors.

(KIT 2011) explores potential cost savings for SCWR in the region of 20% of the investment costs. This is based on estimates of the amount of steel required and the volume of the containment as cost indicators for the construction costs of a reactor. The two main economic advantages of an SCWR are also discussed in (Schulenberg 2020). On the one hand, a simpler reactor design should lead to savings in the initial investment costs. For example, the reactor pressure vessel can be kept to a similarly compact size as in PWR, while at the same time eliminating the need for steam generators and a secondary cooling circuit. Unlike BWR, there is no need for water-steam separation in the reactor pressure vessel. Overall, this should reduce the initial investment compared to those required for the light water reactors in use today. (Schulenberg 2020, p. 58) also estimates the potential savings to be approx. 20% of the investment costs. At the same time, the efficiency of the reactor is increased, as a higher efficiency of the overall plant in the region of 45% can be achieved thanks to the higher working temperatures.

Due to the different reactor geometry and the materials used in the reactor core, a slightly higher enrichment of the fuel is required to generate the same amount of energy. At the same time, however, the energy generated is better converted into electrical energy thanks to the higher efficiency, meaning that no significant differences in fuel requirements can be assumed.

However, depending on the specific reactor concept, the internal structure of the reactor or the fuel elements may be considerably more complex compared to today’s light-water reactors, which may result in higher costs, especially for fuel element production.

No other cost-related differences are recognisable from today’s perspective.

Conclusion: Costs

It can be assumed that the higher efficiency of the reactor system, which is in the region of 45%, compared to current LWR, which is in the region of approx. 35%, as well as the possible reduction in investment costs estimated at approx. 20%, should result in cost advantages over current LWR provided that the unresolved material-related problems can be solved at moderate expense. Cost effects due to the more complex structure of the fuel elements cannot be estimated more precisely at the technology line level, but it can be assumed that these do not outweigh the advantages mentioned. Due to the cost advantages of an SCWR mentioned above, on can therefore expect inherent advantages of the technology line compared to today’s light-water reactors.

4.6 Very-high-temperature reactors (VHTR)

The basic idea of the very-high-temperature reactor (VHTR) is to utilise very high temperatures to achieve correspondingly high efficiencies for electricity production and/or process heat generation. In order to withstand the high temperatures, graphite is used both as a structural material and to enclose the fuel in spheres or rods, which is intended to achieve a very high burnup. The reactor is also said to offer safety advantages compared to light-water reactors. It has a negative reactivity termperature coefficient. This means that the hotter the reactor core gets, the less power is produced. Its high heat capacity results in sluggish behaviour on the part of the reactor core during power changes, with the reactor exhibiting a lower power density in the core. Due to these properties, a purely passive residual heat cooling system is possible in principle. The neutronically inert noble gas helium is used as the coolant. These advantages are often referred to as inherent safety features of VHTR.

VHTR was chosen as the designation for the technology line, which follows the nomenclature of the Generation IV International Forum (GIF). The declared aim of the GIF is to develop the VHTR for very high temperatures (> 950-1000 °C), surpassing the current state of the art. In the literature and as a historical designation, VHTR are often abbreviated to HTR (high-temperature reactor). In the following description, both terms are used synonymously.

An earlier account and evaluation of very-high-temperature reactors was already provided in (Oeko-Institut e.V. 2017). Parts of the account from (Oeko-Institut e.V. 2017) have been adopted here and further supplemented.

4.6.1 System description

The essential characteristic of VHTR is their high operating temperature and the emphasis placed on the fuel as the primary barrier against release.

Reactor system

The fuel in the reactor core maintains the chain reaction for generating energy (see Figure 4-7). In “pebble bed reactors”, the fuel pebbles pass through the reactor core, with parts of the fuel also located outside the reactor core. Outside the core area, however, the fuel is subcritical due to the geometric conditions, so that only the radioactive decay heat (decay power) is released here. In VHTR with prismatic fuel, the fuel elements are in a fixed location in the reactor core during operation. Depending on the specific design, the chain reaction in the reactor can be controlled by control rods.

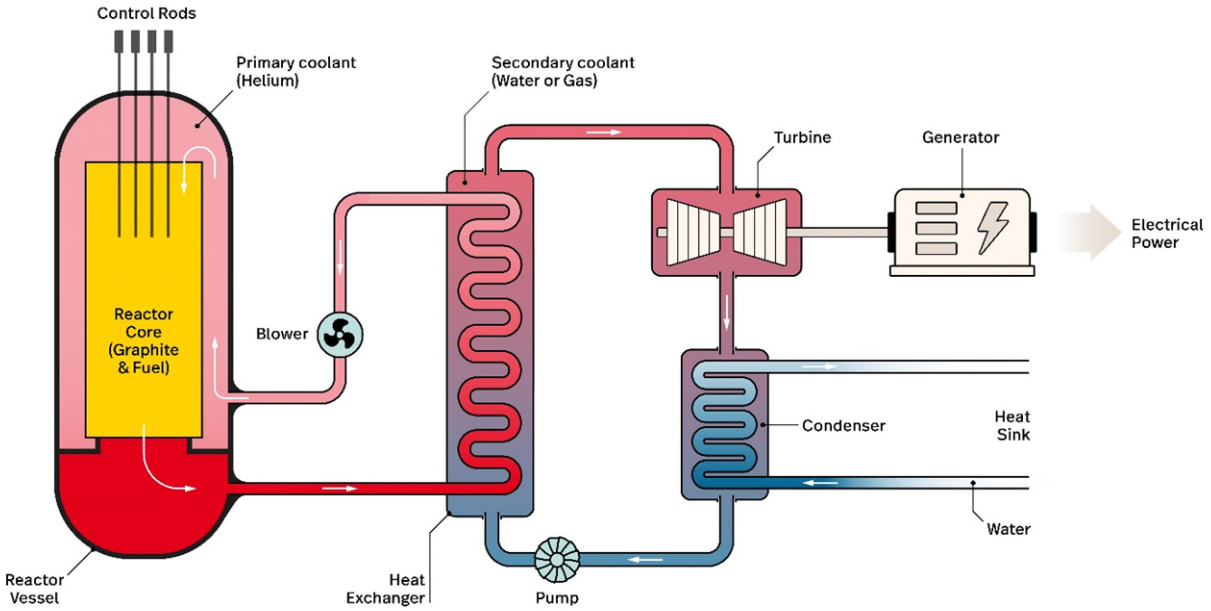
The hot reactor core is cooled by a cooling gas (helium) forced through the reactor core by a blower, which is heated up in the process. The heated gas is conveyed through a heat exchanger in which the energy is transferred to a secondary cooling circuit in order to drive a turbine and thus the generator for the production of electricity. Due to ongoing technical difficulties in manufacturing turbines for helium that can withstand the demanding temperature conditions, only systems with a secondary steam circuit and corresponding heat exchanger have been tested to date.

The remaining waste heat is transferred to an external heat sink, such as a river or the sea. Due to the high operating temperature, some concepts also provide for an application for the industrial production of hydrogen.

VHTR have a lower power density than LWR, falling in the range of approx. 4-10 MW/m^3 (IRSN 2015). In order to increase the external surface area relative to the volume, the reactor vessel has a relatively high length-to-diameter ratio. The maximum possible diameter of the reactor depends on

the resulting radial temperature distribution and the maximum permissible fuel temperature in the reactor core. The length of the reactor core determines the maximum power of the reactor. The high ratio of length to diameter allows the implementation of passive cooling measures of the reactor for residual heat removal, but it also limits the maximum thermal power in the reactor above which active core cooling is required. The pressure losses in the helium flow in the core determine the capacity for the passive cooling of the core and limit the thermal power of pebble bed reactors to < 250 MW and that of VHTR to < 625 MW (GIF 2021a). As a result, many developments in the field of VHTR consist of projects with an electrical output of less than 300 MW, which are often categorised as small modular reactors (SMR). (Oeko-Institut e.V.; WIP; PhB 2021) currently lists 26 SMR concepts that can be categorised as VHTR. Six of these concepts can be further categorised as microreactors.

Figure 4-7: Conceptual diagram of a very-high-temperature reactor (VHTR)



Source: Own illustration

Fuel

VHTR employ a special type of fuel consisting of fuel particles. A fuel particle itself has a diameter of only roughly one millimetre and is covered by multiple layers. The fuel is usually uranium dioxide (UO_2), but uranium oxycarbide (UCO) is also being developed. Thorium/uranium-233 or plutonium can also be used as fissile material. The first layer around the actual fuel particle is a buffer layer of pyrolytic carbon (PyC), which has the dual purpose of absorbing fission gases and compensating for fuel swelling. This layer is followed by a dense layer of PyC and a layer of silicon carbide (SiC), which act as diffusion barriers to trap actinides and fission products. This is followed by a final PyC layer. These particles of fuel and coating layers are called TRISO fuel (TRistructural-ISOtropic) and are surrounded by a graphite coating. In all VHTR concepts, this TRISO fuel represents the decisive component for ensuring the successful containment of radioactivity in the reactor. Due to the

retention function of the TRISO fuel cladding, it is often argued that less importance must be attached to other parts of the safety barrier system, such as the containment.

In pebble bed concepts, the TRISO particles are embedded in graphite spheres (“pebbles”) with the approximate size of a tennis ball, and the bed of the pebbles then forms the reactor core, with the pebbles continuously travelling through the reactor core during power operation. In prismatic reactor concepts, the TRISO particles are embedded in graphite rods measuring a few centimetres, and these are embedded in fuel elements that remain stationary in the reactor during power operation.

The fuel has a demonstrated temperature tolerance of up to 1600 °C before the confinement function fails. More recent TRISO fuel concepts rely on other materials for fuel, such as UCO, and a ZrC layer instead of SiC in the interest of improving the confinement (Helmreich and Hunn 2021).

Fuel cycle

VHTR can be used in open and closed fuel cycles. As a first option, low-enriched uranium (LEU) can be used in the VHTR, but plutonium can also be used as an alternative. The VHTR was recognised very early on as being particularly suitable for a thorium-uranium fuel cycle, as it has good breeding properties (for information on the thorium fuel cycle, see also Chapter 0).

Coolant

The reactor is cooled with helium. Helium is largely inert to interactions with neutrons. Metallic components, such as the reactor’s pressure-tight enclosure, must be kept below the temperature at which creep processes occur and are cooled with the cold cooling gas from the heat exchanger before it enters the reactor core and heats up. In principle, the hot helium could be used directly to drive a gas turbine (Brayton direct cycle) – i.e. a “single-circuit system”. However, the production of such uncooled turbines continues to pose a series of technical difficulties (IRSN 2015). Until such turbines are successfully developed, a secondary cycle is to be used, with the heat utilised with a heat exchanger to generate steam. The hot gas is recirculated by means of gas blowers.

Some VHTR concepts also employ molten salt as a coolant, see Chapter 0.

Spectrum (moderator)

Graphite is used as a moderator and a reflector in the reactor core in order to create a thermal neutron spectrum. As graphite is inferior to water as a moderator, a large amount of moderator must be used in the reactor core compared to an LWR. The result is a low power density in the core.

Temperature

Conventional very-high-temperature reactors (HTR) operate at temperatures of 750 °C to 850 °C, compared to around 300 °C in LWR. The VHTR represents a long-term further development of the conventional HTR and is expected to reach an operating temperature of up to 950 °C, with temperatures of up to 1000 °C also cited in the literature (GIF 2021a). Due to the higher working temperature, the thermal efficiency of the nuclear power plant can be increased to 47% (850 °C) or 50% (950 °C) compared to 34-36% in LWR. The high operating temperatures also allow the heat to be utilised as process heat, for example for the desalination of seawater, coal liquefaction or gasification or hydrogen production.

Pressure

Due to the low density of gases, a higher pressure must be present in the primary cooling circuit. This pressure lies in the range of 1 to 9 MPa.

Construction materials

One of the main construction materials is graphite, which is used as a structural material and also serves as a neutron moderator. To withstand the high temperatures in the reactor core, graphite is a suitable material for temperatures of up to 950 °C. Higher temperatures require other structural materials that have not yet been developed. Graphite is used to embed the TRISO particles in fuel elements and is used as a structural material in the reactor. The entire reactor vessel and all reactor internals that come into contact with the hot cooling gas are made of graphite (reflector, insulation). The reactor pressure vessel is made of steel.

A positive property of graphite is its high heat capacity, which results in a very slow behaviour of the reactor during power fluctuations. A disadvantage of graphite at high temperatures is the risk of graphite fires on contact with oxygen and the formation of flammable water gas, a mixture of carbon monoxide and hydrogen, on contact with water.

4.6.2 Historical developments

The following description of the historical development of VHTR was adopted from (Oeko-Institut e.V. 2017). The description of the PBMR in South Africa has been further supplemented.

The idea of the VHTR dates back to 1946 (ORNL 1947). The first classic HTR test reactor, the “Dragon”, went into operation in Great Britain in 1965 and was operated with prismatic fuel elements until 1976. To date, four electricity-producing reactors have been operated worldwide in Germany and the United States, with further test reactors in the United Kingdom, Japan and China. The operation of the reactors involved a number of incidents and unplanned events, including the ingress of water and oil into the reactor core, fuel failures, helium leaks, etc. A more detailed description of the experiences with the individual test reactors can be found in (Ramana 2016); a summary of significant events is presented in Table 4-10. The following summarises key aspects from (Ramana 2016).

The reactor in Peach Bottom, Pennsylvania was the first commercial very-high-temperature reactor in the U.S. and went critical in 1966. Like the Dragon reactor, the reactor core was operated with prismatic fuel elements, but had a thermal output of 115 MW and an electrical output of 40 MW. The reactor had to be shut down after a mere two months of operation due to problems with the steam generator. Shortly after its recommissioning in 1967, the reactor was shut down once again in 1968 due to increased radioactivity in the helium circuit caused by fuel element failure. After replacement, the reactor was restarted, but 11 more fuel elements had to be replaced later the same year. In 1969, the reactor was restarted with brand new fuel elements, but again there was increased radioactivity in the helium. Nevertheless, the reactor was kept in operation at lower power (lower temperature). The reactor was shut down in October 1969, when another 78 fuel elements had failed. In 1970, the reactor was returned to operation, this time with a modified design of the fuel elements, which operated more satisfactorily. However, due to commercial considerations, it was shut down in 1974, as the fuel elements in the core had to be replaced with new ones. An ingress of oil into the reactor had occurred, which remained undetected by the safety system and was only noticed upon

inspection of the fuel elements. All surfaces of the reactor circuit were also covered with a layer of graphite containing caesium and strontium (Ramana 2016).

The big brother of the Peach Bottom reactor was built in Fort St. Vrain and went into operation in 1974 with a thermal capacity of 842 MW and an electrical capacity of 330 MW. It took five years before the reactor was stable enough to be considered for commercial operation. Helium leaks and the ingress of moisture (which occurred for the first time in 1974) were partly responsible for the lengthy commissioning phase. Other reasons included fluctuations in the core temperature. There were numerous problems over the course of the power station’s operation until 1989: ingresses of water or non-functioning moisture measurement systems, ingresses of air or the failure of the gas measurement system, as well as failure or cracks in graphite tubes and other structural components. The ingress of air had the most serious impact on the safety functions, as it affected the control rods and the backup shutdown system. There was also a failure of six control rods, which could not fully retract into the core. Due to the many incidents, the reactor’s availability was very low (Ramana 2016). Moreover, the reactor was only allowed to run at 70% power due to gas fluctuations in the reactor core and problems with the cooling fan, as well as uncertainties with respect to the guaranteed residual heat removal capacity (Oeko-Institut e.V. 1989b).

The ambitious programme to build 10 high-temperature reactor units in the U.S. was abandoned in the mid-1970s due to poor operating experience – especially with the Fort St. Vrain reactor. Since then, no American operators have indicated a willingness to order a reactor (Ramana 2016). However, in the 1980s, the U.S. Department of Energy was still interested in using an HTR to produce tritium for the nuclear arsenal (Oeko-Institut e.V. 1989b).

The experimental reactor in Jülich, Germany (Arbeitsgemeinschaft Versuchsreaktor (AVR)) went critical in 1966 and was fully operational starting in 1969 with a thermal output of 46 MW and an electrical output of 15 MW. The reactor’s operation until 1988 was beset by numerous problems. The focus was placed on the testing of spherical fuel elements. A number of very different fuel element concepts were tested, over the course of which fuel element failures occurred again and again. Due to faulty temperature measurements, the reactor was operated far above the nominal temperature. Temperature measurements with monitor pebbles indicated that the operating temperature had exceeded 1280 °C in some zones of the reactor core. The prediction of the temperature level and the flow of the pebbles through the reactor proved to be unreliable. In 1978, there was a massive water ingress of 27 t into the reactor, the extent of which remained undetected for several days.[170] The reactor operators initially attempted to heat out the water by operating at a reduced output of 10 MW and to dry out the primary circuit. The leakage caused massive contamination in the reactor building basement. There was also an ingress of oil. The dismantling of the AVR generated high costs due to the contamination, and the entire reactor vessel, in which a number of fuel element pebbles were stuck, had to be temporarily stored in a specially built hall. As of 2012, the partial dismantling incurred costs totalling around EUR 650 million (Aachener Nachrichten 2012). After a decay phase of around 70 years, the reactor vessel must then be cut up by robots and finally disposed of. There were also problems with the AVR due to the contaminated graphite dust in the system. Around 100 TBq of Sr-90 and Cs-137 were found primarily in the contaminated dust after discharge in the reactor vessel; the primary circuit of the AVR is the most heavily Sr-90 contaminated plant in the world, and a further 1.5 TBq of Sr-90 was found in the confinement (Moormann et al. 2018). At the end of the 2000s, one of the developers of the AVR published a comprehensive critique of the AVR based on his experiences (Forschungszentrum

[170] In the same year, there had already been three minor ingresses of water, each involving a few litres of water (Küppers et al. 2014).

Jülich 2008) and, in the following years, derived a comprehensive critique of the pebble bed reactor (see also the account of the HTR-PM provided in Chapter 5.8). A comprehensive account of the history of and events at the AVR can be found in (Küppers et al. 2014).

Construction of the THTR high-temperature thorium reactor with an electrical output of 300 MW in Hamm-Uentrop began in 1971, but it was not put into operation until 1983 and did not generate electricity for the first time until 1987. The THTR does not have any passive safety features and can reach temperatures of well over 1600 °C in the event of a loss-of-coolant accident. Accident scenarios are also possible in the event of water ingress that could potentially result in an explosive release of radioactivity. In addition, there was no pressure containment. The construction was fraught with difficulties right from the start. Shortly before the start of construction, for example, the Krupp company withdrew from the consortium of the companies BBC (Brown, Boveri & Cie), Krupp and Nukem, which was commissioned with the construction, and from the HTR business. In 1988, the reactor was taken out of operation again, as the state refused to assume the financing risks of its continued operation. Due to the many incidents, the reactor’s availability did not reach the targeted 70% in any year, and the reactor was not economically efficient to operate despite a 40% higher guaranteed purchase price compared to LWR electricity. As with the other reactors, graphite dust caused major problems caused in the reactor. A failed attempt to remove the dust from the reactor resulted in a small radioactive release into the environment. The exact cause is still unclear and a subject of controversy. The graphite dust was partly caused by broken fuel element pebbles during the forced introduction of the control rods into the reactor. One of the broken pebbles became stuck in a pipe in 1985 and prevented commercial commissioning, one of the reasons for the reactor’s long commissioning phase. Moreover, the flow of pebbles through the reactor failed to fall in line with predictions, with the pebbles at the center of the reactor core moving much faster, the pebbles at the edge much slower. An inspection of the hot gas pipes between the reactor and the steam generator revealed damage to bolts and other components. There were also massive problems during the construction and commissioning of a helium gas turbine. Helium leaks and contamination occurred, making maintenance impossible. Furthermore, it was impossible to sustain the high temperatures of 950°C with the materials used in the long term. There were also problems with sensors caused by the extreme conditions (temperature, radioactivity). The THTR saw the end of the interest of energy suppliers and the development of the VHTR technology line in Germany, though the technology was sold on to South Africa and China.

Besides the AVR and the THTR, there were a number of other reactor concepts that were planned but not realised. With a thermal output of 200 MW, the HTR module is a simplified design based on the AVR. The main change to the AVR is the separation of the steam generation from the reactor pressure vessel, which is only connected by a hot gas line. In the event of a loss-of-coolant accident, the HTR module can be cooled by passive residual heat removal. The further development of the PBMR in South Africa and the HTR-10 and HTR-PM in China are based on the HTR module (Fütterer et al. 2021; Kugeler and Schulten 1989).

Table 4-10: Previous operating experience with conventional very-high-temperature reactors

Reactor	Power	Start of operation or full power	Decommissioned	Capacity factor	Selected problems
Dragon	20 MW thermal	1965	1976	-	Corrosion of the heat exchangers, helium leakage into the secondary circuit
Peach Bottom	40 MW electrical	1/06/1967	1/11/1974	56.9%	Fuel failure, failure of moisture measurement, graphite dust
AVR	15 MW electrical	19/05/1969	31/12/1988	62%	Fuel failure, water ingress, oil ingress, graphite dust
Fort St. Vrain	330 MW electrical	1/07/1979	29/08/1989	15.2%	Helium leakage, failure of moisture measurement, fuel failure, control rod failure
THTR	300 MW electrical	1/06/1987	29/09/1988	41.3%	Graphite dust, breakage of fuel element pebbles, gas turbine

Source: (Ramana 2016; Englert et al. 2017) and own evaluation

Following a hiatus of roughly a decade, renewed interest in the VHTR only emerged at the end of the 1990s. (Fütterer et al. 2021) cites the main reason for this as a renewed interest in process heat with low carbon emissions.

The German concept for a VHTR was exported to South Africa. For over a decade, South Africa pursued the development of an economically operated, innovative VHTR: the “pebble-bed modular reactor” (PBMR) with a thermal capacity of 200 MW.

According to (Oeko-Institut e.V.; WIP; PhB 2021), the reactor core of the PBMR consists of TRISO particles embedded in graphite pebbles. The graphite pebbles form a loose bed with a height of 11 m in the reactor pressure vessel. Inside the bed is a graphite reflector with a diameter of 2 m. The outer diameter of the bed is 3.7 m. It is surrounded by an outer graphite reflector. The maximum fuel element output is 2.7 kW, the maximum fuel element temperature during operation is 1100 °C. The fuel elements are filled into the reactor from above, exit the reactor core at the bottom and are transported upwards again to be fed into the reactor. They pass through the reactor an average of six times before reaching their target burnup. Cooling takes place via a helium circuit that drives the turbine directly. The cold helium enters the reactor pressure vessel via the outer reflector and flows through the pebble bed from top to bottom.

According to (Oeko-Institut e.V.; WIP; PhB 2021), the key safety features of the PBMR are the exclusion of a core meltdown thanks to passive heat removal in the event of a loss of cooling and the exclusion of water ingress due to the design. Air ingress is limited by design measures. The PBMR has a low number of required safety systems, and the manufacturer states that no event sequences that can be considered possible require evacuations or relocations.

In 1999, PBMR Ltd. was founded as a public-private partnership to develop the PBMR. However, the development was abandoned for economic reasons in 2010 after spending approx. USD 1 billion (Ramana 2016). Around 80% of the investment came from the government, 9% from the South African energy company Eskom, 5% each from Westinghouse and Industrial Development Corporation and 1% from Exelon (atw - Internationale Zeitschrift Für Kernenergie 2010).

It was announced by the Minister of Public Enterprises that the current workforce of PBMR Ltd. was to be drastically reduced to a handful of people who will continue to maintain the expertise. The reasons given were that it had not been possible to attract further investors within the agreed deadline, so further investments totalling ZAR 30 billion (EUR 3.3 billion) were still forthcoming. Furthermore, it was stated the start of construction of the demonstration plant was repeatedly postponed. For new reactors, South Africa would now focus on Generation II and III light-water reactors (atw - Internationale Zeitschrift Für Kernenergie 2010).

In fact, the demonstration model was originally scheduled for completion in 2003 and, in 2010, PBMR Ltd. announced that completion had been postponed until around 2020 (Institute for Security Studies 2010). During the planning phase, the design was modified several times, with the last change being so significant that the environmental impact assessment had to be conducted again. While a plant with a thermal output of 400 MW and 165 MW of electrical output was initially planned (Fütterer et al. 2021), the initial phase only saw 110 MW of electrical output, later becoming 125 MW and then 137 MW (Institute for Security Studies 2010). In the end, the final design only provided for an output of 80 MW (Institute for Security Studies 2010). The design allowed for a very high gas outlet temperature of 900-950 °C.

In terms of costs, the original estimate in 1998 was ZAR 1.1 billion (approx. EUR 180 million $_{1999}$) for the fuel factory and the pilot power plant. In 2003, the South African government approved a prototype for the power company Eskom near Koeberg. The prototype was to go into operation in 2014, and a fleet of 24 PBMR was to follow by 2030. In 2005, the costs were already estimated at ZAR 14.9 billion (approx. EUR 2 billion $_{2005}$) and, in 2009, at ZAR 32 billion (approx. EUR 3 billion $_{2009}$). The costs excluded operating costs, fuel costs, safety costs, final storage, and decommissioning and insurance. Large test facilities were also built in 2007: the “Heat Transfer Test Facility”, a “Helium Test Facility”, a “Pebble Bed Micro Model” and an “Electro-magnetic blower”. Fuel production was also developed, and the first fuel particles were produced in 2008 (Fütterer et al. 2021).

The South African project to build a very-high-temperature reactor was accompanied from the outset by small lobby groups, technically exaggerated hopes prompted by the desire to realise a national nuclear programme of historical import and the associated prestige, and the tense relationship between megaprojects on the one hand and the development needs in South Africa’s young democracy on the other. Sociologist, economist and activist David Figg has provided a detailed account of this background in his article on the history of the pebble bed reactor in South Africa (Institute for Security Studies 2010). He concludes that the pebble bed reactor, plagued as it was by massive cost increases and delays, was unsuitable from the beginning as a solution to South Africa’s energy problems. Despite this financial disaster, the remaining proponents of VHTR technology, such as David Nicholls, Chief Nuclear Officer at Eskom, continue to speak of a window of opportunity (Yelland and Potgieter 2017).

(Fütterer et al. 2021) considers that, similar to other nuclear plants in South Africa at the time, the reactor encountered funding difficulties that forced the business plan to be switched to process heat, after which it nevertheless encountered difficulties in South Africa due to the large coal reserves and lack of carbon emission limitations. The process heat and electricity production were to be used for coal liquefaction and gasification, petrochemical processes, fertiliser production, refineries, oil sands mining, hydrogen production and the desalination of seawater. PBMR Ltd. changed the business orientation and standard design of the PBMR accordingly and entered into cooperative partnerships with Sasol (a South African company) for coal liquefaction, Eskom for electricity, as well as with U.S. and Canadian companies to pursue oil sands extraction, among other things. According to this account, the project was halted in 2010 due to financing difficulties and problems with the licensing authority.

In the case of prismatic VHTR, the “Gas Turbine Modular Helium Reactor” (GT-MHR) with a thermal output of 600 MW was developed by an international group of institutes and companies, including Minatom, General Atomics, Framato and Fuji Electric in Japan. This was based on an earlier design of the “Modular High-Temperature Gas-Cooled Reactor” (MHTGR-350) from General Atomics, which is still being used for code benchmarks in recent times (NEA 2017). The GT-MHR design concept was finalised in 2001, and the project was submitted by General Atomics as part of the “Next Generation Nuclear Plant” project of the DOE (see Chapter 5.9.6.2). However, in the scope of the cooperation between DOE and an industrial alliance, a preference was expressed in favour of AREVA’s ANTARES concept, which was based on the GT-MHR but equipped with a steam circuit. A smaller version, the SC-HTGR with a thermal output of 350 MW, also with a steam circuit, was proposed by AREVA/Framatome (see Chapter 4.6.3.3). This was also the basis for the Japanese GT-HTR300. (Fütterer et al. 2021)

4.6.3 Current developments

The following description of the current development of VHTR has been adopted in part from (Oeko-Institut e.V. 2017) and further supplemented.

Following the discontinuation of the PBMR, development work on the VHTR for the past 10 years or so has been concentrated primarily in the Chinese nuclear programme with the HTR-10 experimental reactor (10 MW thermal output) the development and commissioning of the HTR-PM (see Chapter 5.8).

A smaller experimental reactor, the HTTR (30 MW thermal output), was also operated in Japan, and there are plans for the “Gas Turbine HTR 300” (GTHTR-300). Further plans are in place for the SC-HTGR in the U.S. and the “Prismatic HTR” (Next Generation Nuclear Plant) (see Chapter 5.9), also in the U.S. Terrestrial Energy is planning to develop the Xe-100. The TeResa project is in planning in Poland.

The ARIS database of the IAEA lists five reactor concepts allocated to the VHTR technology line: the GTHTR300, the HTR-PM, the PBMR, the Prismatic HTR and the SC-HTGR (IAEA 2023d).

4.6.3.1 Overview of the reactor concepts in the technology line

An overview of the main current reactor concepts in the VHTR technology line is provided in Table 4-11.

Table 4-11: Key reactor concepts for the VHTR technology line

Reactor concept	Country	Type	Power (thermal)	Coolant
SC-HTGR/Antares	U.S./France	Prismatic	625 MW	Helium
Prismatic HTR	U.S.	Prismatic	350 MW	Helium
Xe-100	U.S.	Pebble bed	200 MW	Helium
HTTR	Japan	Pebble bed	30 MW	Helium
GTHTR-300	Japan	Prismatic	600 MW	Helium
HTR-10	China	Pebble bed	10 MW	Helium
HTR-PM	China	Pebble bed	250 MW	Helium
TeResa	Poland	Prismatic	40 MW	Helium

Source: (IAEA 2023d)

The two main types of VHTR differ in how the TRISO fuel particles are introduced into the reactor core. In prismatic HTR, the TRISO particles are embedded in elongated prismatic graphite rods, while, in pebble bed reactors, the particles are embedded in graphite pebbles with a diameter of approx. 6 cm. The pebbles are continuously fed into the reactor core and travel from the top to the bottom of the reactor vessel over time. Pebbles are either continuously removed from the bottom of the reactor vessel and then fed back into the reactor from above or removed from the reactor once the maximum burnup of the pebble has been reached. Individual fuel pebbles can pass through the reactor multiple times, depending on the previous residence time. In the reactor core, therefore, there are tens of thousands of these pebbles in a single bed. The continuous removal of fuel elements is seen as an advantage, as the reactor does not have to be shut down to change fuel elements. In terms of safety, however, this leads to uncertainties about the exact composition of the reactor core and the exact position of the fuel (Englert et al. 2017).

4.6.3.2 HTR-10

The Nuclear and New Energy Technology Institute (INET) at Tsinghua University in China built the HTR-10 experimental reactor (10 MW thermal output), which went into operation in 2000. The HTR-10 is used to test the properties of the pebble bed, fuel and components and to validate simulation codes. The reactor generates heat in the district heating network of the university campus. The HTR-10 paved the way for the development and approval of the HTR-PM (see Chapter 5.8).

4.6.3.3 SC-HTGR, ANTARES

From 2005 onwards, the programme to develop a VHTR continued in the scope of the DOE’s Next Generation Nuclear Plant (NGNP) project. The NGNP project was cancelled by the DOE in 2011. In the project, the design of the SC-HTGR competed with the PBMR (see Chapter 4.6.2) and the GT-MHR (Gas Turbine Modular Helium Reactor) from General Atomics. The General Atomics concept would later become the Prismatic HTR (see Chapter 5.9).

The reactor[171] subsequently selected by the Idaho National Laboratory (INL) in 2012 is based on the “Steam Cycle High-Temperature Gas-Cooled Reactor” (SC-HTGR) from Framatome (formerly AREVA) (IAEA 2019b) with a thermal output of 625 MW, which in turn is based on the ANTARES reactor concept from AREVA, which was also introduced by AREVA as an SMR concept in 2010 (AREVA 2017).

The SC-HTGR is to contain modules with a thermal output of 625 MW each to be built underground in silos beneath the reactor hall. The exact number of modules is to be based on the requirements of the specific location. The reactor is to reach a core outlet temperature of 750 °C so that it can also provide process heat at 570 °C. The TRISO fuel (UCO) is to be enclosed in small graphite cylinders, as is typical for prismatic HTR. The cylinders are to be embedded in holes located in large hexagonal graphite fuel element blocks with a width of 360 mm and a height of 800 mm. The ring-shaped core contains 102 of these blocks surrounded by graphite reflectors and a central graphite reflector in the centre. The active core is 10 blocks high. Reactivity control is to be ensured primarily thanks to the inherent properties of the strongly negative temperature coefficient together with the slow behaviour of the reactor. In addition, gravity-driven and other reactivity control systems are provided for to enable reactor control and shutdown. The reactor should not require any intervention by the operating personnel or active components for safe operation (“walk away safety”), and no evacuation measures should be necessary in the event of an accident. The fuel element is to be changed by robots via access through the control rod drive openings on the top of the reactor vessel. All pressurised components on the primary side are surrounded by a pressure vessel, including the two steam generators and the hot gas supply line. All walls of the primary system are in contact with recirculated cold gas, which allows the use of normal steel (IAEA 2019b).

The reactor was to be licensed by the U.S. Nuclear Regulatory Commission (NRC) and built in Idaho with substantial participation on the part of industry (WNA 2016b). In August 2008, a strategy for licensing the reactor was jointly issued by the NRC and the DOE, and a series of white papers was published. NGNP pre-licensing interactions between the developer and the NRC began in 2006 and were suspended in 2013 after the DOE decided not to proceed with the project. Disagreements between the DOE and the NGNP industrial alliance over cost sharing under the public-private partnership required by Congress were cited as the reason for the DOE’s decision (for more information, see Chapter 0).[172]

In the U.S., industry has spent over USD 1 billion in development costs over the last decade (2006-2016) for the development of the plant concept and the facilities for utilising the process heat. On the governmental side, the DOE has spent over USD 500 million on fuel development, graphite qualification and materials research. In 2014, further research and development costs for the fuel were estimated at around USD 200 million up to 2023 (Kadak 2016). The total costs for the first power plant were estimated at USD 1.9 billion for the government and USD 3.6 billion for industry.

[171] https://www.world-nuclear-news.org/NN-Areva_modular_reactor_selected_for_NGNP_development-1502124.html

[172] https://www.nrc.gov/reactors/new-reactors/advanced/licensing-activities/ngnp.html

4.6.3.4 Xe-100

In the U.S., the start-up company X-Energy is pursuing plans to develop a gas-cooled pebble bed reactor with an electrical output of 80 MW, similar to the Chinese and South African concept.

According to the description provided in (Oeko-Institut e.V.; WIP; PhB 2021), the reactor core of the PBMR consists of TRISO particles embedded in graphite pebbles. At 425 µm, the UCO particles are slightly smaller than the usual UO_2 TRISO particles with 500 µm. The graphite pebbles form a loose bed in the reactor pressure vessel. The maximum fuel element temperature during operation is 1100 °C. The fuel elements are filled from above and leave the reactor core at the bottom. They pass through the reactor an average of six times before reaching their target burnup. Cooling takes place via a helium circuit, with the heat transferred to the secondary cooling circuit in an external heat exchanger. The cold helium enters the reactor pressure vessel via the outer reflector and flows through the pebble bed from top to bottom.

According to the manufacturer, the properties of the fuel should inherently rule out a core meltdown with releases in all event sequences. Due to continuous fuel element replacement, the excess reactivity remains at the low figure of 2%, which makes it possible to dispense with burnable neutron poisons while maintaining sufficient shutdown margins. Shutdown capability is still guaranteed even in the event of water ingress from the secondary circuit into the reactor. Fuel meltdown, the manufacturer reports, is ruled out up to a temperature of 3920 °C. Even in the event of the most severe accident to be assumed, a loss of coolant circulation and simultaneous loss of pressure (depressurised loss of forced coolant (DLOFC)), no fuel damage should occur. No personnel intervention is required for the safe shutdown of the reactor

In 2020, X-energy LLC was awarded a USD 14.31 million contract by the U.S. DOD to develop a mobile microreactor for military applications.[173] The TRISO-X fuel produced by X-energy is to be irradiated in the MITR test reactor at the Massachusetts Institute of Technology (MIT) in 2020 and then subjected to post-irradiation examinations in the laboratories there. This is supposed to fulfil an important prerequisite for the subsequent approval of TRISO-X.[174] X-energy will produce the fuel for the Japanese high-temperature test reactor (HTTR) together with the Japanese Nuclear Fuel Industries (NFI).[175] In July 2020, X-energy applied for a “vendors’ design review” with the Canadian regulator CNSC.[176] In October 2020, the Canadian company Ontario Power Generation (OPG) pledged financial support for X-energy and its Xe-100 with a planned electrical output of 80 MW (GIF 2021a, p. 9).

In October 2020, the DOE approved funding of USD 80 million for the Xe-100 concept. X-Energy is working with the energy company Energy Northwest as well as with Burns & McDonnell on the development of the Xe-100. The plant will consist of four modules with a total electrical output of 320 MW. The funding is provided in the scope of the “Advanced Reactor Demonstration Plan”, making the Xe-100 the second reactor in this funding programme after the MCFR (see Chapter 5.6).

[173] https://world-nuclear-news.org/Articles/US-Defense-Department-awards-microreactor-contract, last accessed 15/01/2021.
[174] https://www.world-nuclear-news.org/Articles/X-energy-TRISO-X-fuel-to-be-irradiated-at-MIT, last accessed 15/01/2021.
[175] https://www.world-nuclear-news.org/Articles/X-energy-and-NFI-team-up-to-supply-HTGR-fuel, last accessed 15/01/2021.
[176] https://www.world-nuclear-news.org/Articles/X-energy-makes-Canadian-SMR-review-submission, last accessed 15/01/2021.

Further funding totalling USD 3.2 billion can be obtained within the programme over the next 5-7 years. The funds will first be channelled into a planned TRISO fuel fabrication plant, which is to be built for USD 300 million in Oak Ridge, Tennessee, by 2025.[177]

The development of the Xe-100 is currently in an early concept phase. Further development will require the construction of a demonstration power plant, for which investments totalling several hundred million dollars, i.e. several billion, are needed to build a demonstration power plant.

4.6.3.5 HTTR

With a thermal output of 30 MW and prismatic fuel, Japan has been operating the “high-temperature engineering test reactor” (HTTR) since 1999. During this time, it only briefly reached its full output at a gas outlet temperature of 950 °C in 2004. By 2010, the reactor had run through two cycles with 30 days of continuous operation and one cycle with 50 days (Nishihara et al. 2011).

According to the description in (Oeko-Institut e.V.; WIP; PhB 2021, p. 121), the development of this reactor dates back to 1969. Construction took place between 1991 and 1997, and first criticality was achieved in 1998. The HTTR can be viewed as a research reactor for the development of a system with significantly higher power from the GIF development line. The HTTR was decommissioned after the accident in Fukushima, Japan. In June 2020, the operator received approval from the Japanese Nuclear Regulation Authority (NRA) to carry out minor modifications to the reactor in order to implement new safety regulations. The reactor then resumed operation on 30 July 2021.[178]

In the scope of an OECD/NEA research programme, tests on accidents with a loss of forced coolant (LOFC) are to be carried out at the HTTR.

4.6.3.6 GTHTR300

The GTHTR300 concept, which has been under development since the end of the 1990s, represents a further development of Japanese HTTR technology (Sato et al. 2014; IAEA 2011c). The concept thus continues the previous line of development of prismatic VHTR concepts (for more information, see Chapter 4.6.3.3). It consists of a modular plant with a thermal output of up to 600 MW per reactor and is intended for use with a helium gas turbine for the generation of electricity and process heat. As with all VHTR, direct utilisation of the hot gas in the helium turbine would be crucial to ensure economic success. Initial tests for such a turbine have been carried out in a scaled model (30 MW electrical output) (IAEA 2011c).

According to the developers (IAEA 2011c), there is still some development work to be done to achieve the economic targets, including increasing the burnup of the fuel from 90 to 120 MWd/kg of heavy metal and further developing the durability of graphite composite materials and non-destructive inspection methods for components to increase their service life. (IAEA 2020b) states that the developers aim to achieve commercialisation in the 2030s.

[177] https://www.nucnet.org/news/work-begins-on-triso-fabrication-plant-for-x-energy-s-htgr-10-2-2022
[178] https://www.world-nuclear-news.org/Articles/Japanese-gas-cooled-reactor-restarts, last accessed 09/02/2022.

4.6.3.7 FHR AHTR

Besides helium cooling, a further idea is to use molten salt to cool a VHTR (see Chapter 4.4.4, Thermal MSR with solid fuel).

The concept of a “fluoride salt-cooled high-temperature reactor” (FHR) was first proposed in 2003 and combines the properties of two technology lines: the use of TRISO particles in a graphite matrix derived from the VHTR technology line and molten fluoride salt from the MSR technology line. It thereby adopts the properties of a high heat capacity, low chemical reactivity with air and water and a high retention capacity for many fission products (Fütterer et al. 2021).

For the “Advanced High-Temperature Reactor AHTR-100”, which employs prismatic fuel with TRISO particles, there were also considerations to use molten salt for cooling. The AHTR is designed for very high core outlet temperatures of up to 1200 °C (Fütterer et al. 2021).

The “Mark-1 Pebble Bed FHR” (Mk1 PB FHR) is being developed by the University of Berkeley (U.S.) with an electrical output of 100 MW. The core of the fuel pebbles is made of porous graphite to vary the pebble density, and the fuel pebbles are smaller than those typically used in VHTR with a diameter of 6 cm. As a result, they have twice the surface area per volume, allowing for an increase in power density while maintaining nearly the same particle temperatures. The pebbles are lighter than the coolant and therefore float in the salt. They are therefore inserted into the core from below and move slowly upwards through the core. At the top end, they are collected by a defuelling machine and fed back in or sorted out depending on the burnup.

4.6.3.8 TeResa

There is an interest in Poland in investigating the utilisation of reactors that are not intended for direct use for electricity production. In 2017, an initial recommendation was made by a committee of the Polish Ministry of Climate and Environment (MKiŚ) for the implementation of very-high-temperature gas-cooled reactors (HTGR) in Poland to cover the domestic demand for industrial heat at temperatures of up to 700 °C. The GOSPOSTRATEG-HTR project is investigating the concept for a gas-cooled very-high-temperature reactor called TeResa, a research reactor with 40 MW thermal power intended as a technology demonstrator for industrial purposes. (Skrzypek et al. 2022) presents the initial preliminary thermal-hydraulic and neutronic studies of the Polish “National Centre for Nuclear Research” (Skrzypek et al. 2022).

The TeResa multi-purpose and demonstration reactor builds on the preliminary concepts of the GEMINI+ project,[179] which was launched in September 2017 as part of Euratom’s European Horizon 2020 programme to elaborate a reactor concept for combined heat and power generation and the production of process heat, as well as a framework for the approval of the system and a business plan. In doing so, GEMINI+ falls into line with a number of European research programmes on VHTR development over the last 19 years, such as RAPHAEL, PUMA, EURPAIRS, ARCHER and NC2I-R, as well as international collaborations and developments in the GIF, and builds on existing VHTR concepts, such as the GT-MHR, MHTGR and SC-HTGR (Skrzypek et al. 2022).

[179] https://gemini-initiative.com/geminiplus/

4.6.3.9 Other current developments

There are also research efforts underway in South Korea, where a programme was launched in 2006. This programme is currently still in the research and development phase, although construction of a plant was originally expected to begin in 2016 and be completed by 2020 (WNA 2016a). According to (GIF 2021a, p. 14), South Korea completed a research project on various aspects of VHTR development in 2019 and launched a follow-up project in April 2020.

The VHTR System Arrangement of the GIF was signed in November 2006 by Canada, Euratom, France, Japan, South Korea, Switzerland and the United States. China signed the arrangement in October 2008, and South Africa formally joined the GIF Framework Agreement in 2008, but announced in December 2011 that it no longer intends to join the VHTR System Arrangement. Canada withdrew from the System Arrangement at the end of 2012, but is now once again a member and remains active in the GIF hydrogen production project. The newest members of the System Arrangement are Australia (December 2017) and the UK (January 2019). (GIF 2020a)

In 2020, the steering committee for the VHTR technology line updated its work plan for high-level system research and development to support national or international VHTR demonstration projects and the long-term improvement of performance (GIF 2021a).

In December 2021, the UK Department of Energy announced that VHTR had been selected for funding in the “Advanced Modular Reactor Research, Development & Demonstration” programme. The programme is funded by the government with GBP 170 million to accelerate the development of highly flexible nuclear technologies.[180]

4.6.4 Technical development status

The following conclusions can be drawn from the operating experience of VHTR thus far:

- The very high operating temperatures relevant for economic operation repeatedly caused corrosion problems and damage to the pipework and in the steam generator.
- Due to the increased diffusion processes at high temperatures, the underestimation of the actual fuel temperatures (especially in pebble bed reactors) led to fission products diffusing from the fuel matrix into the coolant – even at operating temperatures and increasingly so at higher temperatures – resulting in radioactive contamination in the primary circuit. This was exacerbated by TRISO particle failure and fuel element or pebble failure (breakage). This is particularly problematic, as fission products adhere to the unavoidable graphite dust in the reactor, which in turn settles in inaccessible areas of the reactor, from where they can be remobilised.
- Experience to date has been marked by unreliable operation and low availability.
- Problems were frequently encountered with the use of a gas turbine to utilise the high gas outlet temperatures directly, i.e. without losses of a secondary cooling circuit.

[180] https://world-nuclear-news.org/Articles/X-energy,-Cavendish-team-up-for-UK-HTGR-deployment

Even if these problems faced by VHTR are solved using technological means, such as in the Chinese HTR-PM (see Chapter 5.8), the GIF already pointed out a number of necessary developments in 2014 that would have to be implemented in order to actually achieve a VHTR with suitably high temperatures and not just a conventional HTR (GIF 2014):

- The increase of the permissible core outlet temperature from approx. 800 °C to more than 1000 °C for the entire time of operation.
- A permissible maximum fuel temperature under accident conditions capable of reaching 1800 °C.
- A permissible maximum fuel burn-up of 150-200 MWd/kg of heavy metal.
- The prevention of power peaks and temperature gradients in the core as well as hot gas streams with excessive temperatures relative to the cooling gas.
- The limitation of structural damage caused by water or air ingress.

The achievement of high temperatures, the inherent safety features, high fuel performance in terms of burnup and confinement temperatures, and the coupling of industrial processes in order to utilise process heat while simultaneously generating electricity are therefore the main points of interest of VHTR research and development with the aim of reaching these, in part, conflicting goals. Computer methods and suitable concepts for licensing procedures are also being developed. Prismatic VHTR do not differ significantly from VHTR with pebble beds in many aspects. Moreover, there are synergy effects between the various strands of development.

Fuel production of TRISO with SiC coatings is complex, defect testing is even more complex, but the industrial process can be considered mature overall (Helmreich and Hunn 2021). Also, only minor particle defects are caused during production on an industrial scale.

There are varying statements as to the mechanisms that result in particle failure (mechanical, corrosion). On the one hand, they are well understood, and the fuel can therefore be used under the conditions (temperature, power density, burnup) of modern VHTR designs (Gerczak 2021). More recent irradiation experiments in the U.S. were also successful (EPRI 2019). Compared to previous results, the test range was extended with respect to burnup, power density, neutron fluence, irradiation temperatures and accident temperatures (Reitsma 2021). Research activities in the U.S. investigated the mechanism that, for newer TRISO particle, the fission product palladium can escape through cracks in the inner pyrocarbon due to radiation-induced dimensional changes in the buffer, subsequently corroding the SiC layer (Hunn et al. 2016).

On the other hand, the burn-up dependence of the particle failure is not yet fully understood (Moormann 2011; IRSN 2015). The exact release mechanism for some fission products, such as europium, silver and caesium as a function of radiation-induced defects and corrosion is not yet understood (Moormann et al. 2018). For example, the radiation-induced enhancement of the diffusion of strontium, europium and caesium has only recently been investigated for the first time (Dwaraknath and Was 2016). In parallel, kinetic models have been developed, e.g. for silver and caesium (Ko et al. 2016; Ko et al. 2017). While the diffusion mechanism of silver through the SiC layer is much better understood (Olivier and Neethling 2013) than that of europium and strontium, according to (Dwaraknath and Was 2016), it is difficult overall to separate the effects of thermal diffusion from radiation-induced diffusion enhancement. The diffusion of caesium was considered negligible for high quality fuel. However, according to (Dwaraknath and Was 2016), this has not yet been verified. Recent experiments in the U.S. for UCO-TRISO particles, on the other hand, show

promising results for the retention of fission products and against particle failure, even at temperatures in the region of 1800 °C (EPRI 2019).

The main development efforts for fuel are focussed on achieving high burnups and temperatures. This entails understanding the standard design (UO_2 particles with SiC/PyC coating) and investigating the use of uranium oxycarbide (UCO) as a fuel and ZrC coatings to improve burnup capability and to best contain the fission products. This concerns, in particular, the resistance to core overheating, during which parts of the reactor core can reach temperatures above 1600 °C. This is achieved through an appropriate characterisation of the fuel, post-irradiation examinations, safety tests, the release of fission products, and the chemical and thermomechanical material properties under representative operating and accident conditions. Fuel development also includes the treatment and disposal of spent fuel elements, including the management of spent graphite, and the transmutation of plutonium and minor actinides (MA) to support a closed fuel cycle (GIF 2021a).

In the U.S., work is being carried out on VHTR concepts with a view to developing advanced fuel concepts (TRISO fuels) and new materials for high-temperature applications (DOE 2017b).

The development and qualification of materials, construction specifications and standards, and manufacturing methods are essential to the development of VHTR systems. The key challenges for VHTR structural materials are radiation-induced and/or time-dependent failure and microstructural instability in operating environments (GIF 2021a).

The use of existing materials is specified for core outlet temperatures of up to 950 °C. However, achieving the target of > 1000 °C requires new structural materials (graphite for core structures and the fuel matrix, metals for very high/medium temperatures, as well as ceramics and composite materials) and their qualification for safe operation even under accident conditions and including corrosive processes. This applies in particular to high-temperature heat exchangers and metals in steam generators. At present, interest is still focussed on steam-based process applications at slightly lower core outlet temperatures of 750 °C to 850 °C. A materials handbook has been developed in the scope of the GIF which is used to store and manage VHTR data, facilitate international R&D coordination and support modelling for damage prediction and service life assessment (GIF 2021a).

The temperatures actually demonstrated so far in the utilisation of process heat only allow limited use of the potential range of industrial applications, which has a decisive influence on the economic efficiency of a VHTR. In general, the higher the temperatures achieved, the more efficiently the process heat can be converted and processes such as coal gasification and hydrogen production can be priced.

Two main processes for water splitting were originally considered for hydrogen production: the thermochemical sulphur-iodine cycle and high-temperature steam electrolysis. The evaluation of further cycles has led to a focussed interest in two lower temperature cycles: the thermochemical copper-chlorine hybrid cycle and the sulphur hybrid cycle. The coupling of hydrogen processes with the reactor also requires risk analyses to be carried out in order to cover the possible interactions between nuclear and non-nuclear systems. Thermochemical or hybrid cycles are being investigated with respect to their technical and economic efficiency for pure hydrogen production or combined heat and power generation in order to reduce requirements for operating temperature and to make them compatible with other GIF systems (GIF 2021a).

The use of combined heat and power generation and direct coupling with industrial processes for process heat applications as well as potential transients and external influences must be included in the design considerations for both the nuclear and industrial plant, or innovative decoupling systems must be used (Reitsma 2021).

Computer-aided simulation methods in the areas of thermal-hydraulics, thermomechanics, nuclear physics and chemical transport have been continuously developed, especially in the South African and Chinese programmes. These are required for the evaluation of reactor performance under normal, abnormal and accident conditions as well as for the licensing processes. The validation of codes must be performed by benchmark tests and code-to-code comparisons, from basic phenomena to integrated experiments. According to (GIF 2021a) , computer-aided methods will also facilitate the elimination of unnecessary conservatism in design and improve construction cost estimates (GIF 2021a).

Likewise, essential components for the most important reactor systems (core structures, absorber rods, reactor vessels, pressure vessels, etc.) and for the energy conversion or coupling processes (e.g. steam generators, heat exchangers, pipes, valves, instrumentation) for use at very high temperatures still need to be developed. Some components require advances in on-site manufacturing and construction techniques, including new welding and post-weld heat treatment processes. Some components also need to be tested in specialised large-scale helium test facilities. The results can be used for all strands of development of the VHTR as well as for other technology lines, such as the GFR, see Chapter 4.3 (GIF 2021a).

One of the biggest challenges in SNR development is the granting of licenses, though a number of VHTR have already been approved. However, as pointed out in (Reitsma 2021), the safety and licensing requirements have become considerably stricter over the last 30 years. They also apply specifically to LWR, and it is not easy to apply them to VHTR. On the other hand, the safety philosophy of modern VHTR has changed, making many of the safety systems required in LWR superfluous (Reitsma 2021).

In fact, pebble bed reactors have been licensed under the requirements and licensing concept for LWR. This was already the case in Germany and was recently confirmed again with the HTR-PM in China. In both cases, however, agreement was reached on specific exemptions and additions to the requirements and the concept before the licence application was submitted (Reitsma 2021).

The NRC has also made progress in licensing practice in this area and has made requirements technology-neutral (see e.g. (NRC 2019c), which also contains a special annex on design criteria for modular gas-cooled very-high-temperature reactors). Similar efforts have been made by regulators around the world to ensure an appropriate focus in the licensing of VHTR (Reitsma 2021).

As a result, the main technical difficulties for the safety and commercial utilisation of a VHTR remain unresolved today. Research and development efforts are therefore focussed on improving the existing technology and making it ready for the market.

(GIF 2018b) provides a good summary of the research and development needed for VHTR:

- completion of the capability for fuel testing and qualification (including fabrication, quality assurance, irradiation, safety testing and post-irradiation examinations) in certain countries;
- qualification of graphite, hardening of graphite against air/water ingress, e.g. by SiC infiltration; management of graphite waste;
- coupling technology and related components (e.g. isolation valves, intermediate heat exchangers);
- definition of design codes and standards for new materials and components;
- advanced manufacturing methods (collaboration with the GIF cross-cutting task force);
- cost-reducing R&D and collaboration with industry to optimise VHTR design;
- licensing and site selection: review and validation of computer codes for design and licensing;
- system integration with other energy sources in hybrid energy systems;
- pebble bed modular reactor: demonstration tests (HTR-PM), improved information exchange with multiple start-ups, private investors and new national programmes;
- HTTR: safety demonstration tests and coupling with hydrogen production plant (subject to approval of regulatory authority for restart).

The GIF timetable in 2002 specified the year 2011 as the earliest possible starting time for the “performance” phase (GIF 2002), with the demonstration phase starting in 2014 and a fully developed reactor six years later. The 2014 schedule, however, included the assumption that the “performance” phase had already begun (GIF 2014), although no other reactor was commissioned during this period following the commissioning of the Chinese demonstration reactor HTR-10. Contrary to this, it was assumed that the demonstration phase would begin in 2022 and that a finished, marketable concept could then be available by 2030 at best. The HTR-PM actually went into operation as a demonstrator in 2022. Eight years after this GIF assessment from 2014, it can be assumed that, provided that the HTR-PM is successful, this schedule could be met by 2030. However, the reactor principle with the operating temperatures of the HTR-PM has already been demonstrated with plants from the 1970s and, to ensure its success, the HTR-PM would have to achieve a very high capacity utilisation and operational reliability compared to the plants from the 1970s and 1980s. However, if there are difficulties in operating the HTR-PM, or for reactor concepts with higher temperatures for an improved utilisation of process heat, a marketable reactor cannot be expected until 2035-2040. In view of the as yet considerable need for research and development, a more realistic estimate is 2045.

According to GIF estimates, the corresponding pure costs for R&D in 2002 amounted to around half a billion USD (GIF 2002), which did not include the construction of a research or demonstration reactor. However, in view of the investments made to date in R&D in China and South Africa and based on the current state of development, further R&D costs totalling several billion USD can be safely expected.

Conclusion: Technical development status

Very-high-temperature reactors have been under development for over 60 years. So far, no research and development program has been able to operate a reactor at a high utilization rate over an extended period of time. Most intensive development programmes (Germany, U.S., South Africa) have so far been discontinued in the demonstration power plant phase due to a lack of interest on the part of the operators, which was in turn based on concerns about the plants’ economical and reliable operation. Active research and development programmes are currently underway, primarily in China, the U.S. and Japan.

When assessing the technological readiness level of a VHTR, several goals must be distinguished: achieving core outlet temperatures of 750 °C, up to 950 °C and temperatures above 950 °C.

With regard to temperatures in the region of 750 °C and utilisation for pure electricity generation with a water-steam intermediate cooling circuit, the pebble bed VHTR is currently in the phase between “development” and “operation”. In order to use the reactor outside China under Western licensing standards, further development work would probably need to be carried out, as demonstrated by the development of TRISO manufacturing in the U.S., for example. Further development work would also be required if the pebble bed reactor concept were to be coupled with the utilisation of process heat. The plants also have a modular design with a power limit of approx. 250 MW of thermal output, allowing the use of largely passive safety systems. For concepts of a larger scale, further development work would have to go into safety systems. With the HTR-PM, a demonstration reactor for the pebble bed concept with steam generation is now in operation. However, the reactor would have to achieve high utilisation rate and high operational safety in order to upgrade the demonstration concepts from the 1970s to a marketable system. There is currently no demonstration reactor for prismatic VHTR. With the Japanese experimental reactor, development in this area is most advanced in Japan. However, the development is lagging behind that of the pebble bed reactor.

Moreover, considerable development work still needs to be carried out before a gas turbine can be employed to utilise higher efficiencies; the development status here can be categorised as “development”.

Reactor concepts with temperatures between 750 °C and 950 °C still require a significant amount of development work, particularly with regard to materials and instrumentation. Depending on the reactor concept, the temperature limit of 1600 °C as the maximum fuel temperature may not be sufficient, and further fuel development may be necessary (ZrC coating, UCO particles). This could also open up new applications for process heat or the utilisation of a gas turbine. The degree of maturity here has not yet reached the “operation” phase, and some research work must still be done in the laboratory.

Depending on the area of technology in question, the overall development can therefore be categorised as somewhere between “applied research” and “development”.

For temperatures above 950 °C, new structural materials are required in particular, and the development status can be categorised as “applied research”.

4.6.5 Safety

The following description of the safety of VHTR was essentially adopted from (Oeko-Institut e.V. 2017) and further supplemented.

The VHTR is one of the reactor concepts frequently described as “inherently safe”. In the specific case of VHTR, this inherent safety is often tied to the property of a negative temperature reactifity coefficient. The following should be noted here:

> *“In physical terms, the property at stake is the fuel temperature reactifity coefficient, which is negative for all pressure and boiling water reactors. The fact that it is also negative in the THTR [all HTR] is thus not specific to that reactor. This property is extremely important for control. What this means is that when the fuel cools down, heat production increases and, conversely, when the temperature increases, heat production decreases. The coolant cools the fuel rods, and the power output increases. If there is a sudden rise in temperature for any reason, the output is reduced. If this property did not exist, there would be a constant fear of power excursions.” (Oeko-Institut e.V. 1989b)*

The high heat capacity of the system and the relatively sluggish temperature behaviour of the reactor resulting from this are also cited as a safety feature. In contrast to an LWR, a loss-of-coolant accident only affects the core temperature, as helium is largely inert to interactions with neutrons. A loss-of-coolant accident cannot cause an increase in reactivity with a rapid power excursion, as is possible in all fast reactors. In the case of a VHTR, a criticality accident caused by a loss-of-coolant accident is excluded by the reactor design. In the event of a cooling failure, loss of coolant (loss of pressure) or loss of the heat sink, the reactor cools itself through the natural convection of the cooling gas or through heat conduction and heat radiation.

However, the reactor must be designed to prevent an unintentional extension of the control rods. For pebble bed reactors, the emergency shutdown is achieved by means of control rods or by the falling of neutron-absorbing pebbles caused by gravity, which are also used for long-term shutdown. For the subsequent restart of the reactor, these must be removed from the pebble bed. Due to the high heat capacity of graphite, additional energy in the reactor is first stored in the fuel itself and then in the reactor wall and reflectors before reaching the reactor vessel.

Finally, the low power density of VHTR represents a physical advantage, as it prevents power peaks and thus temperature peaks in the core. The slim design of the reactor vessel also means that the heat can be removed by heat conduction and radiation as well as natural convection.

Based on information provided by the developers, this self-regulating behaviour allows a VHTR to be designed in such a way that the overheating of the fuel can be ruled out in the case of loss-of-coolant accidents (for water or air ingress, see below). The reactor design (size, power density, reactivity behaviour) must therefore ensure that the retention function of the fuel and reactor vessel structures remain protected during design-basis reactor transients.

The provision of fuel in the form of TRISO particles is an essential part of the VHTR safety concept. The safety philosophy behind this is based on the premise that the integrity of the particles must be maintained under all normal operating conditions as well as during design-basis accidents.[181] This

[181] In the words of two members of the “South African Nuclear Regulatory Commission” who were tasked with assessing the pebble bed reactor: “safety design philosophy…is based on the premise that the fuel adequately retains its integrity to contain radioactive fission products for all normal operating and design

means that if the retention capacity of the fuel is maintained even under accident conditions, the radioactive inventory in the fuel matrix of the TRISO particles remains safely enclosed in the graphite pebbles. The result would then be a relatively low source term in the event of incidents and accidents, and any environmental impacts would be severely limited. A VHTR must therefore be designed in such a way that the overheating of the fuel is ruled out. A maximum fuel temperature of 1600 °C is generally assumed as a limit. If this temperature is exceeded, the temperature-dependent diffusion processes result in an increasing failure of the retention function of the particle coating and the surrounding graphite and thus to a release of radioactivity.

Based on this consideration, a safety review would thus have to show that the fuel temperature in a specific reactor concept remains below 1600 °C under accident conditions. The fuel integrity must be demonstrated experimentally under all circumstances with regard to temperature, burnup, power density, etc. One example is a study (Kindt and Haque 1992) that confirms the inherent safety of the reactor concept investigated there with regard to core cooling. In the specific scenarios investigated, which the authors describe as an “extreme combination”, hypothetically only 1536 °C could be reached (Englert et al. 2017; Kindt and Haque 1992).

However, the assumption that there is a threshold of 1600 °C below which no radioactive substances diffuse from the fuel particles would be misleading. This also applies to terms like “leaktight” (IRSN 2015). Even under normal operating conditions, a small proportion of the radioactive fission products diffuse from the fuel particles and through the graphite matrix into the cooling gas. Diffusion is more likely to occur if the coating of the fuel particles is damaged. This cannot be completely prevented during industrial production of the fuel particles. The proportion of defective particles during production is around $3 \cdot 10^{-5}$ (IRSN 2015), which would then be contained in the fuel. Particles can likewise fail during burnup. A rate of around $2 \cdot 10^{-5}$ was determined for the HTR module, for example (IRSN 2015). A number of different mechanisms may bring about particle failure. The silicon carbide coating may be compromised by palladium, gaseous fission products may result in internal gas pressure, irradiation may result in particle shrinkage, temperature gradients in the graphite may result in the migration of fuel particles, and chemical interaction between the graphite of the coating and the fuel particles may occur (Englert et al. 2017). Particle failure may also occur due to the ingress of water or air into the reactor (see below) or due to breakage of or damage to the fuel pebbles, such as in the THTR, where roughly 17,000 pebbles were damaged.

The diffusion rate is strongly temperature-dependent and specific to the individual elements. For example, the diffusion rate of radioactive silver-110 is 15 times higher than that of caesium-137 (Minato et al. 2000). Both are strong gamma emitters, and the release into the cooling gas stream has an impact on operation, the economic feasibility of the reactor (dismantling) and safety (Englert et al. 2017). Moreover, the diffusion processes are not yet fully understood at a fundamental level, e.g. for silver-110 (IRSN 2015; Kim et al. 2015).

While the release remains low at low temperatures, temperature tests have shown that, at 1600 °C, no “cliff edge” effect sets in; instead, metallic fission products, in particular, are released, even at lower temperatures. The justification for the 1600 °C limit in the German fuel development programme was therefore based on the statistical analysis of the onset of particle failure under accident conditions. For this reason, the temperature limit for the HTR module design was set at

basis accident conditions, thereby allowing radiological safety to be assured.”(Clapisson and Mysen 2002) Or, in the description of a panel of the U.S. NRC: “TRISO-coated particle fuel particles are intended to stay intact and effectively retain and contain fission products during normal operation as well as during postulated accidents.” (NRC 2004)

1600 °C, corresponding to a particle failure of $6 \cdot 10^{-5}$ (IRSN 2015). This limit is therefore specific to the moderate conditions of the HTR module for which the fuel was qualified.

According to (Scheuermann et al. 2017), it can be concluded from the irradiation tests available up to that point that higher temperatures than 1600 °C are permissible if the maximum burnup in the fuel element remains strictly below 11% FIMA (fissions per initial metal atom). However, this remains to be proven by new experiments. If the burnup of TRISO particles with uranium oxide is above about 15% FIMA, the temperature must remain strictly limited to below 1600 °C. High-burnup (14% FIMA) UO_2 TRISO fuels have demonstrate particle failure at 1600 °C during the first 300 h. No particle failure occurred with lower burnups (11% FIMA) at 1600 °C. Above 1800 °C, the tests show that caesium release from the particles and the surrounding matrix occurs without delay and that the SiC layer becomes permeable to most fission products. Recent tests for UCO-TRISO particles in experiments based in the U.S., on the other hand, have showed improved fission product retention and less particle failure even at temperatures of 1600-1800 °C (EPRI 2019).

Released, non-gaseous fission products may adhere to graphite dust in the reactor. Graphite dust is unavoidable in a VHTR and is distributed throughout the reactor’s cooling circuit by the hot helium. The dust mainly settles on cold parts of the reactor and on bends and corners. In the event of an accident, the graphite dust may then be remobilised (e.g. leaching during water ingress) and contribute to the source term. The behaviour of graphite dust under accident conditions is not yet well understood.

In summary, it can be said that the 1600 °C temperature limit, but even a lower temperature limit, represents an overestimation of the retention capacity of the fuel. Whether and how much radioactivity is released during an accident depends on the maximum temperatures in the fuel, how long a critical temperature is exceeded, the behaviour of the fuel under corrosive conditions, the irradiation history and the defect rate during fuel production. These parameters are subject to uncertainties that must be taken into account in a safety assessment, especially if the reactor concept does not provide for pressure-tight containment (Englert et al. 2017).

Most VHTR concepts provide for a confinement. This is a vented low-pressure retention space around the reactor pressure vessel. The venting serves to limit possible overpressure in the reactor pressure vessel. During normal operation, a slight negative pressure is maintained in the confinement and, in the event of an accident, the venting is used to limit the pressure in the confinement. This can offer advantages for accident sequences in VHTR. However, the use of a confinement instead of a pressure-tight containment then requires the retention function of the TRISO particles, which must then assume the comparable function of the containment. If this is no longer the case due to faulty TRISO particles or temperatures above 1600 °C, radioactivity may escape to the primary circuit. Graphite dust may also escape, especially in the event of rapid depressurisation of the primary circuit, where deposited graphite dust can be remobilised.

The reactor must be designed to prevent water ingress by limiting the maximum quantity of penetrating water and by designing the reactor for the additional reactivity. When using a steam generator, water ingress is an event that has historically occurred in steam generators with a probability of around 0.001 (large leak) and 0.1 (small leak) per year (Moormann et al. 2018; Bongartz et al. 1988). Water ingress causes reactivity to be added to the reactor, firstly, because the reactor is under-moderated and, secondly, because water is a better moderator than helium. If water enters the core, the reactivity and the power of the reactor increase. This leads to an increase in temperature. Due to the negative temperature reactifity coefficient, the reactivity decreases as the temperature rises, counteracting the input of positive reactifity reactivity caused by the ingress of

water. However, if the water ingress is large enough, the negative temperature reactifity coefficient cannot compensate for the input of positive reactivity quickly enough (Zhang et al. 2005; Moormann et al. 2018). In this case, the temperature of the fuel elements rises, which may result in fuel failure and the release of fission products into the reactor core. Using a gas turbine and dispensing with a steam circuit would be advantageous here, but the possible ingress of oil from the turbine into the reactor, which can also result in an increase in reactivity, must also be taken into account in the safety assessment.

Water also reacts with the hot graphite structures in the reactor core and with the graphite of the fuel pebbles, producing flammable water gas, a mixture of carbon monoxide and hydrogen. The ingress of steam and water gas brings about an increase in pressure in the primary circuit, which in turn may result in the opening of safety valves and the release of radioactive isotopes and explosive gases into the reactor building (Englert et al. 2017). In the case of the AVR in Germany, 30 tonnes of water leaked into the reactor core in 1978, although the reactor remained in operation for a number of days at a lower power level (Moormann 2008; Forschungszentrum Jülich 2008). The combination of the maximum fuel temperature, the operating temperature and the negative temperature feedback (reactivity coefficient) determines the maximum additional reactivity caused by water ingress that can be compensated. For the safety case, an ingress of a certain amount of water is assumed with a certain probability, which can be controlled by the system without exceeding the maximum fuel temperature.

In tests, irradiated UO_2 fuel particles with radiation-induced defects have exhibited a strong burnup dependence for the release of fission product gases like krypton-85 when exposed to steam at 800 °C multiple times. Only 2% of the Kr-85 inventory was released at a burnup of 5% FIMA (fissions per initial metal atom), but 17% at a burnup of 9% FIMA (Scheuermann et al. 2017).

If one or more leaks occur in the primary circuit, air can enter the reactor. The penetrating oxygen oxidises the graphite. Depending on the location of the air ingress, this may affect the graphite internals of the reactor or the graphite coating of the fuel. The temperature in the reactor is also increased by the positive enthalpy of the oxidation process. With a sufficiently large amount of air ingress, the temperature may exceed the maximum fuel temperature and lead to fuel failure, even in reactors with relatively low thermal output (in the region of 200 MW) (Moormann 2011; Englert et al. 2017). According to earlier assumptions, complete particle failure already occurs at 1100 °C, but experiments show that it takes several days at 1100 °C to oxidise the graphite of the pebbles, with the particles remaining undamaged until then. Only prolonged air ingress at temperatures above 1300 °C would also lead to TRISO particle failure. (Scheuermann et al. 2017) Due to the leakage in the primary system in the event of air ingress, radioactivity can also escape there. For the safety case, an ingress of a certain amount of air is assumed with a certain probability, which can be controlled by the system without exceeding the maximum fuel temperature or until countermeasures can be taken to stop the air ingress.

There are still several uncertainties that are relevant for a possible safety case.

- Fuel behaviour: In the case of the pebble bed reactor, the accuracy with which the flow behaviour of the pebbles in the reactor has been modelled has been limited until now. Uncertainties about friction, wear and fracture behaviour under operating conditions make it difficult to predict the flow behaviour of the pebbles and, with that, the structural change of the core over time and the exact prediction of the neutronics, the coolant flow and thus the temperature distribution in the core (Moormann et al. 2018). Temperature peaks in the reactor may occur at points of compaction of the pebbles. Modelling is also difficult because temperature measurement in the reactor core (pebble bed) is only possible to a limited extent, even during reactor operation. The core is constantly changing, and the maximum temperature can only be determined retrospectively, e.g., by mixing in special measuring spheres, which are then analysed after leaving the reactor core. All measuring equipment is exposed to a very corrosive and highly radioactive environment. Corrosion is caused by contamination of the helium with H_2O, CO, CH_4 and H_2 (Graham 1990). This may result in the failure of instruments that are very difficult to replace. In the AVR in Germany, it was only discovered after the fact that, in some places, the maximum temperature in the core was more than 200 °C higher than predicted (Forschungszentrum Jülich 2008). Similar phenomena can occur in prismatic VHTR due to changes in the cooling gap.
- The possible source term is difficult to determine. In the event of water or air ingress, the source term includes fission products that have settled on metallic surfaces in corners or pipe bends in the form of graphite dust. The particles can be washed off or blown off to remobilise them. If a reactor has been in operation for many years, the source term for a release could be much larger than theoretical calculations suggest, especially as the behaviour of graphite dust is not yet fully understood. In the German programme, low source term assumptions led to the abandonment of a pressure- and gas-tight containment. A confinement consisting of the reactor building, a negative pressure system, a building pressure relief system and a filter system was considered sufficient to keep the effects of a release on the environment far below the dose limits of the German Radiation Protection Ordinance. Due to changes in safety standards, it is now questionable whether containment can be dispensed with internationally for plant approvals. On the one hand, as shown by the HTR-PM in China, reactors with a confinement are still being built today, but the HTTR in Japan was fitted with a pressure-tight containment (see also (EPRI 2005; Li et al. 2020)).
- The exact behaviour of the cooling gas in a pebble bed is not yet sufficiently understood. The numerical simulation of the helium gas flow in the reactor is extremely complex, and the fluid mechanics in the pebble bed as well as in the reactor as a whole are difficult to model. As already described in our account of historical operating experience, continuous temperature monitoring is also a highly elaborate process (Shams et al. 2014).

Conclusion: Safety

VHTR are particularly associated with the term “inherent safety”. In view of the low power density in the reactor and the sluggish temperature behaviour, loss-of-coolant accidents are not the main risk, which constitutes an advantage over LWR. Even if active cooling fails, the shutdown reactor can cool itself as long as it does not exceed a certain output level. This, too, is an advantage over LWR. If these passive safety features are to be maintained, the power output of the reactor (or its size) cannot be scaled up in order to optimise the overall costs. The reactors would only be economically attractive if they were combined in a modular design to jointly generate the energy for a turbine and a generator.

A core meltdown like in an LWR is also not possible, as graphite does not sublimate up to 3500 °C. However, the approx. 1 mm large, coated fuel particles (TRISO fuel) encased in the graphite increasingly lose their retention effect from temperatures of 1600 °C onwards, and radioactive elements diffuse from the fuel into the primary circuit. A maximum fuel temperature of 1600 °C is therefore currently regarded as limiting, with some authors also citing 1800 °C as a temperature that could be reached in the future. The maximum fuel temperature also limits the current maximum thermal reactor power to around 250 MW for pebble bed reactors and 650 MW for VHTR with prismatic fuel with no loss of essential passive safety features, as the reactor core heats up too much in some places (above 1600 °C) and massive amounts of radioactivity are released into the primary circuit.

The relevant accident scenarios for a radioactive release, therefore, are not core meltdowns, as in LWR, but the ingress of water and/or air into the reactor core. Above a certain amount, water can trigger a reactivity accident. The graphite of the fuel and reactor structures such as the reactor wall can oxidise and decompose in contact with water or burn in combination with oxygen. A massive release of radioactivity in very-high-temperature reactors is therefore by no means ruled out in principle (i.e. due to inherent properties), and the defense in depth of previous concepts would have to be further strengthened. Ideally, the reactor would be operated without a secondary water-steam circuit with a helium turbine in order to eliminate the possibility of water ingress. To date, however, it has not been possible to build a cost-effective turbine for a helium circuit that can withstand the high operating conditions.

The primary circuit of a very-high-temperature reactor is contaminated with radioactive particles that enter the primary circuit due to fuel failure (pebble breakage or defective TRISO particles) or diffusion processes, even at normal operating temperatures. Radioactive particles also bind to graphite dust, which settles in the primary circuit and can escape in the event of a leak in the primary cooling circuit. It is still difficult to determine the exact source term for an accident in VHTR. This is due, among other things, to uncertainties regarding the position and residence time of the fuel in the reactor (pebble bed), the diffusion of various radioisotopes from the fuel into the reactor, the modelling of the temperature distribution and hot gas flows in the reactor (especially the pebble bed), the accumulation of radioactive graphite dust in the reactor and, finally, the difficult environmental conditions for measuring instruments in the reactor core.

At the technology line level, an overall advantage over current LWR can be assumed if VHTR are used with limited power due to the inherent properties of residual heat removal. The extent to which such advantages can be realised in specific reactor concepts cannot be conclusively assessed at the technology line level.

4.6.6 Fuel supply and waste disposal

The following description of VHTR fuel supply and waste disposal has been adopted in part from (Oeko-Institut e.V. 2017) and further supplemented.

One advantage over LWR is that less heavy metal is required per unit of energy generated due to the high efficiency of the electricity generation. Like all reactors, VHTR also generate different waste streams. In VHTR, the special features compared to LWR are the use of TRISO fuel and the graphite waste streams. VHTR that utilise thorium fuel are also frequently discussed.

4.6.6.1 TRISO fuel

The coated UO_2 fuel particles with layers of pyrolytic graphite (PyC) and silicon or zirconium graphite contain most of the radioactive inventory of fission products, the remaining uranium or thorium, plutonium and minor actinides. The coating itself, therefore, acts as the first barrier against a release in the reactor, but also during disposal (Forschungszentrum Jülich 2015). It also contains porous layers that retain volatile fission products. Among the more recent fuels is uranium oxycarbide (UCO), a fuel with ZrC coatings.

The TRISO particles are embedded in a graphite matrix. The graphite matrix may contain fission products from uranium impurities and from defective fuel particles. It also contains activation products from graphite impurities with elements such as lithium or chlorine and radiocarbon (carbon-14) from the activation of nitrogen-14, carbon-13 and oxygen-17 (Forschungszentrum Jülich 2015).

Typical release mechanisms from the TRISO particles and the surrounding graphite matrix in reactor operation and the current state of research and development for fuel development and particle failure are summarised in Chapter 4.6.4. At very high fuel temperatures, the fuel may also fail, with fission products potentially diffusing out of the fuel particles. Typical radioisotopes primarily include metallic fission products such as strontium-89, strontium-90, silver-110m, iodine-131, caesium-134, caesium-137 (Sun et al. 2020), which were measured, above all, in the AVR due to inadequate early TRISO concepts. Silver-110m, in particular, diffuses at low temperatures and settles in the primary circuit. However, fission products such as barium-140, lanthanum-140, europium-152 and hafnium-181 were also found in the primary circuit of the HTR-10 (Sun et al. 2020). (Sun et al. 2020) state, in summary, that metallic fission products, in particular, have a high diffusion capacity in pyrolytic graphite and are therefore increasingly distributed in the primary circuit. Although there have been considerable improvements in the production of TRISO fuel in recent years, particularly in terms of temperature resistance, the release mechanisms are still not fully understood.

It can be generally assumed that the multi-coating of TRISO particles can also serve as a primary barrier for disposal in a geological repository. Historically, numerous studies have been conducted on the direct disposal of TRISO fuels in a geological repository. The focus of most of these studies is on the properties and functionality of the SiC coating and graphite matrix in a geological repository. HTR fuel elements are considered “well-designed” for direct emplacement in a geological repository due to their multiple barriers within the waste form and matrix (Fachinger et al. 2006; NASEM 2023b).

(NASEM 2023b) come to the conclusion, however, that research and demonstration work is still necessary to determine and prove the effectiveness of the barrier. (NASEM 2023b) also lists a number of recent research projects, mainly from European research institutions. According to this source, the research work of (Fachinger et al. 2006; Grambow 2008; Gerczak et al. 2020; Forschungszentrum Jülich 2015; Katoh et al. 2012; Liu et al. 2020; Johan B Malherbe 2013; Heinz

Nabielek et al. 2010; Peterson and Dunzik-Gougar 2011; van den Akker and Ahn 2013) focuses primarily on the properties and modes of action of the SiC layer and the graphite matrix, which are generally well-suited for direct geological disposal. Appendix G in (NASEM 2023b) provides a more detailed report, the key points of which are summarised below.

The graphite matrix limits the amount of water that can come into contact with the fuel particles (Fachinger et al. 2006; NASEM 2023b). Likely radiolysis is not a significant factor in determining the dissolution rate of the fuel, as the embedding matrix significantly delays the contact of groundwater with the fuel particles in the event of the ingress of water into a repository (Forschungszentrum Jülich 2015). According to (Fachinger et al. 2006; NASEM 2023b), radionuclides are transported in the water-filled porous graphite matrix by means of diffusion processes, with retention times ranging from a few days to a year over distances of up to one centimetre.

According to (NASEM 2023b), the performance of spent TRISO fuel in a geological repository largely depends on the properties and capabilities of SiC or other carbides. The service life of SiC and pyrocarbon layers is estimated to be 10^3 to 10^5 years, depending on the temperature, irradiation dose and corrosion conditions, assuming that no non-uniform corrosion occurs. However, experimental studies are required to confirm this (Fachinger et al. 2006; NASEM 2023b). Based on the assumptions that, after a container failure, water diffuses through the graphite matrix in a few years, that the SiC coating fails after only 10,000 years and that the particles disintegrate in about 10^6 years under reducing conditions and 10^4 years under oxidising conditions, it is concluded that the coatings make spent TRISO fuel elements more resistant to the rapid initial release of radionuclides compared to conventional forms of high-level waste (Grambow 2008; NASEM 2023b; Forschungszentrum Jülich 2015).

According to (NASEM 2023b), further research to better understand the behaviour of TRISO fuel in a geological repository is required to get a full grasp of the effects on coatings irradiated at different temperatures and time periods as well as on radionuclide transport within TRISO particles due to radiation-induced degradation of carbide coatings (Fachinger et al. 2006; Gerczak et al. 2020; van Rooyen et al. 2018). For disposal, a better understanding of the solubility and release of radionuclides, the penetration of water in graphite and the corrosion mechanisms of SiC and other carbide layers under different geological conditions is required (Fachinger et al. 2006; FUKUDA et al. 1982; Grambow 2008; Forschungszentrum Jülich 2015; Johan B Malherbe 2013; ANL 2005; Heinz Nabielek et al. 2010; Peterson and Dunzik-Gougar 2011; van den Akker and Ahn 2013).

Not removing the particles in the TRISO fuel from their graphite prisms or spheres results in a significant amount of waste. The advantage of low power density has a negative impact on disposal. Due to the graphite, the volume of VHTR fuels relative to the energy provided is around ten times greater than that of LWR (Moormann et al. 2018). Another estimate of waste according to (NASEM 2023b) calculates a quantity of 90,000 spent fuel pebbles for a typical pebble bed reactor with a capacity of 200 MWe per year, including around 17 t of carbonaceous waste (NASEM 2023b; Fuks et al. 2020). Converted to a typical SMR, such as the Xe-100 reactor (80 MWe) from X-energy, this would constitute 160 m^3 of fuel elements per year compared to 6.8 m^3 of spent fuel elements per year from a large LWR (NASEM 2023b). Thermal MSR also employ graphite as a moderator that is activated by neutrons and must be disposed of at the end of its operating life. Other concepts for SMR provide for the regular replacement of the moderator. With a service life of seven years, the graphite moderator of Terrestrial Energy (MSR with 180 MWe) will produce eighty tonnes of graphite waste (NASEM 2023b). The ThorCon molten salt reactor (1 GWe) will require 780 t of high quality

graphite and, according to (NASEM 2023b), will produce 100 m^3 of irradiated graphite per GWe each year.

Some VHTR reactor concepts provide for the reprocessing of the TRISO fuel to recover the fissile material or to dispose of the fuel. According to (Del Cul 2002), different processes for aqueous and non-aqueous separation have been developed. After an initial step of crushing and breaking up the TRISO fuel, aqueous separation would be followed by dissolution in nitric acid with the usual process flow of aqueous reprocessing as employed in the PUREX process. In non-aqueous separation, the fuel would be dissolved by carbochlorination. There are also other aqueous and non-aqueous processes(NASEM 2023b), as well as pyrochemical processes. Open research and development issues concern the handling of volatile fission products in the off-gas, increasing recovery efficiency and the scaling of each process stage. Solid residues that remain after nitric acid leaching, such as undissolved precious metals, SiC shell fragments and residues containing SiO_2 and $NaSiO_3$, can result in the formation of silica and impede further separation processes (Del Cul 2002; NASEM 2023b). Development work on the reprocessing of silicon-based fuels is also underway in the context of the disposal of research reactor fuels.

Precautions must also be taken with regard to other fuels with respect to criticality safety, as the use of HALEU fuels with an enrichment of up to 19.75% is also planned for VHTR fuels. In addition, HALEU fuels also have higher heat releases and higher radioactivity due to the high burnup. Moreover, the neutron dose may also be increased in fuels from thermal SNR, such as the VHTR, due to a significantly higher curium-242 and curium-244 content (IAEA 2020f; NASEM 2023b).

When transporting and storing spent HALEU fuels, criticality safety must also be taken into account in order to maintain the subcriticality of the fuel, even in potential accident scenarios.

4.6.6.2 Thorium fuels

The original motivation for utilising VHTR was to use a thorium fuel and to establish a thorium fuel cycle. Most currently planned reactor concepts are based on the use of uranium fuels with approx. 8-12% enrichment in an open fuel cycle. Higher enrichments up to HALEU levels are also being discussed. Discussions concerning a later utilisation of thorium fuels or other fissile materials, such as plutonium, as fuel, are limited to a long-term perspective (for more on thorium, see also Chapter 5.5.3).

The significantly lower build-up of transuranic elements, especially plutonium, the associated advantages with regard to the radiotoxicity of the waste, and a reduction in long-term heat release are frequently seen as significant advantages of a thorium-uranium fuel (IAEA 2005). Although ^{237}Np is produced, it is also separated during the reprocessing of the fuel for the separation of uranium-233, meaning that no plutonium is produced. However, the advantage of eliminating transuranic elements is cancelled out by a certain disadvantage: while less transuranic elements are produced when using thorium, significant quantities of the long-lived ^{231}protactinium and ^{229}U are also produced, which have an effect on radiotoxicity (IAEA 2005).

With regard to the advantages of a lower actinide inventory in a spent uranium-thorium fuel, (NASEM 2023b) also offer a relativising conclusion. Although it is true that actinides dominate in terms of ingestion radiotoxicity and the potential dose to the population, the general assessment that the radioactive hazard is reduced by the use of a uranium-233/^{232}Th fuel cycle compared to a uranium-235/Pu fuel cycle may not necessarily be the case (NASEM 2023b).

The unsuitability of ingestion radiotoxicity as a measure for determining long-term safety in a geological repository is discussed in greater detail in Chapter **8.6.3**. The authors of (NASEM 2023b) also follow this line of argumentation. With regard to thorium fuels, they argue that, while the uranium-233/^{232}Th fuel cycle contains lower quantities of transuranic elements according to (Piet 2013), this does not necessarily mean a lower long-term hazard on average, since focussing on transuranic elements without adequate consideration of the fission products could lead to incorrect conclusions. With regard to the production of long-lived fission products, there are no significant differences between plutonium and thorium-uranium fuels (NEA 2015, S. 100).

However, even with regard to a radiotoxicity comparison, (Piet 2013) states that both uranium-235/Pu and uranium-233/^{232}Th contain essential actinide isotopes. Therefore, from the point of view of an inventory comparison, it cannot be stated that either uranium-233/^{232}Th or uranium-235/Pu is necessarily more harmless than the other fuel cycle from the point of view of long-lived environmental contamination (Piet 2013). See also Chapter **8.6.2** and **Figure 8-18**.

As a further source, (NASEM 2023b) cite the recent study by (Croff and Krahn 2016) on the calculation of the radiotoxicity of a uranium-233/^{232}Th fuel cycle. (Croff and Krahn 2016) state that, under the same framework conditions, e.g. an identical reactor design, fuel design, burnup, specific power, fuel cladding type and composition, and trace element concentration of the fuel matrix, the radiotoxicity of thorium-based fuels and waste disposed of in a repository is broadly similar to that of uranium-based fuels and not “far lower” (Croff and Krahn 2016). Overall, the radiotoxicity of thorium-based fuels containing uranium-233 or plutonium as fissile material is similar to the radiotoxicity of uranium-based fuels containing uranium-235 or Pu for decay times ranging from one year to 20 million years (Croff and Krahn 2016). (NEA 2015) also concludes that, due to some radionuclides, such as uranium-233, ^{231}Pa and ^{232}U, the radiotoxicity of the uranium-233/^{232}Th fuel cycle is more accurately described as comparable to a uranium-235/Pu fuel cycle (see also Chapter **8.6.2**).

(NASEM 2023b) also points out that thorium fuel cycles with plutonium as fissile material were also planned in some concepts. Such a fuel would not have the advantage of low actinide content.

However, the use of thorium fuel could be an advantage due to the lower actinide content in the fuel, especially in scenarios of use in fast reactors like MSR for transmutation.

Furthermore, (NASEM 2023b) state that certain reactor concepts envisage mixing uranium-233 with uranium-238 to increase proliferation resistance during fuel production in order to achieve a lower enrichment of uranium-233. However, such a depletion of uranium-233 would again suffer from the disadvantage that the fuel would contain plutonium and higher transuranic elements. The advantage of higher proliferation resistance must then be weighed against the disadvantage of significantly higher quantities of actinides in the waste stream. For more information on the influence of actinides on disposal in a repository mine, see also Chapter 8.

In addition, the authors emphasise (NASEM 2023b) that an initial core made of HALEU is used in a reactor to be operated with uranium-thorium fuel and that, until all HALEU fuel elements are replaced by uranium-thorium fuel elements, larger quantities of actinides are also produced and require disposal. However, depending on the service life of a reactor, the amount can be much smaller. If thorium-plutonium fuel is used as fissile material in the initial core, larger quantities of actinides are produced accordingly.

If no in-situ breeding is provided for in the reactor, additional waste streams are also generated through the necessary reprocessing. Since the actual fissile material in a uranium-233/^{232}Th fuel cycle must first be bred from thorium, most concepts provide for the reprocessing and separation of the fissile uranium-233. The corresponding reprocessing process, referred to as “THOREX”

(THORium-uranium EXtraction), is a solution-based aqueous process similar to the PUREX process with subsequent chemical separation. However, the process is much less developed and is currently available only in development in laboratories and pilot plants in a few countries (IAEA 2005). The main difficulty lies in the stable compound of thorium oxide, which can only be dissolved by adding hydrofluoric acid to sulphuric acid, which causes the corrosion of process equipment (NASEM 2023b).

4.6.6.3 Graphite waste

Compared to LWR fuel, other materials, in particular graphite, are used in VHTR, and these have an impact on the long-term safety of a repository.

There are several sources of graphite waste in a VHTR. These include, above all, the graphite of the carrier matrix, in which the TRISO fuel particles are enclosed, and the graphite of reactor components (reactor vessels, reflectors, etc.). In addition, graphite from the coating of the TRISO fuel particles is produced during the reprocessing of the TRISO fuel particles, which may contain fission products that have diffused from the fuel into the coating due to its proximity to the fuel (for more information on the reprocessing of VHTR TRISO fuel, see above).

The graphite is contaminated with radioisotopes that originate from impurities in the graphite itself. For example, uranium impurities in the graphite cause fission products, while other impurities result in activation products in the graphite. Other sources of radioisotopes in graphite include defective TRISO particles as well as diffusion mechanisms from intact TRISO particles at high temperatures. The radioisotopes are distributed in the reactor by graphite dust and helium gas.

Among fission products, strontium and barium exhibit a high affinity for graphite and thus do not diffuse through the graphite layers of the fuel particles or the graphite matrix. However, they can be transported in the graphite dust when released from defective particles, for example. Iodine isotopes are extremely volatile, can also form stable compounds with metals and are strongly adsorbed by graphite (Sun et al. 2020).

The activation products generated include tritium from helium, (^{3}He(n,p)^{3}H), but also carbon-14 from carbon-13, nitrogen-14 and oxygen-17, the dominant reaction being the formation from carbon-13. Chlorine-36 is also formed. Other solids in the activation products include ^{51}Cr, ^{54}Mn, ^{59}Fe, ^{57}Co, ^{58}Co, ^{60}Co, ^{75}Se, ^{209}Hg (Sun et al. 2020)and ^{22}Na. Graphite also contains fission products such as strontium-89, strontium-90, silver-110m, iodine-131, caesium-134, caesium-137, as well as barium-140, lanthanum-140, europium-152 and hafnium-181. According to (NASEM 2023b), other reactor types with graphite structures (e.g. Russian-style RBMK reactors) also contain ^{59}Ni, ^{63}Ni, but also other europium isotopes, such as ^{154}Eu and ^{155}Eu, as well as ^{144}Ce and actinides.

Graphite matrix of the fuel

Due to the high volumes of graphite waste, there are considerations to separate the TRISO particles from the graphite matrix prior to conditioning with the ultimate goal of reducing the volume of graphite waste. However, this requires a great deal of effort due to the contamination of the graphite waste and generates further waste streams. A conditioning concept for direct final disposal would therefore be simpler, albeit with significantly higher volumes. Various alternative conditioning methods, including cementing, sand backfilling and salt backfilling, were tested for their behaviour under pressure and temperature.

According to (NASEM 2023b), the TRISO particles can be effectively and almost completely separated from the graphite matrix by methods such as low-temperature (H_2SO_4 + H_2O_2) and high-temperature acid treatments (H2SO4 + HNO3) (Guittonneau et al. 2010; NASEM 2023b). The TRISO particles could then be vitrified in a similar way to the reprocessing waste from plutonium separation or embedded in another waste matrix, such as silicon carbide. The advantage would be the considerable reduction in the volume of the high-level waste (NASEM 2023b). According to (NASEM 2023b), there are two methods for the vitrification of TRISO particles: sintering (at 700 °C) or glass melting (1200-1300 °C). The strong containment properties of the TRISO particles can be preserved to the greatest extent possible by using the lower sintering temperature. Initial tests have shown positive results. At higher temperatures using processes such as liquid sintering or hot pressing (1800-2000 °C), the fuel particles would be damaged and fission products would be released (NASEM 2023b).

Separating the graphite from the fuel alone would make little difference to the volume of the waste; it would only be possible to split the waste streams and develop separate disposal concepts for the two waste streams. According to (Fuks et al. 2020; NASEM 2023b), the graphite should initially be stored temporarily for 5-10 years after an initial separation of the non-radioactive graphite content, after which the radioactive fraction should be stored for a further period of 35-135 years. At the end of the second storage period, the residual waste should then be sent for final disposal as low-level radioactive waste. The EU Carbowaste project (total costs: EUR 12 million; EU contribution: EUR 6 million, 2008-2013) also showed that a number of radionuclides, such as tritium, chlorine-36, ^{60}Co, ^{63}Ni and carbon-14 could be separated from the graphite with high efficiency (NASEM 2023b).

However, the overall size of a repository depends not only on the volume of waste to be stored, but above all on the heat generated by the waste and the resulting distances between the repository containers.

Contaminated components

As a large proportion of the reactor structures in VHTR consist of graphite blocks, these must be dismantled during operation and after decommissioning. According to (Lensa et al. 2020), this poses a challenge due to cracks and weakened structures caused by temperature and irradiation. Parts of these structures may also be contaminated by radioactive graphite dust, activation products or damaged fuel pebbles and must be disposed of accordingly.

Since graphite dust with radioactive particles and broken fuel element pebbles remain in the reactor vessel and in the primary circuit of many historical experimental and test reactors, the dismantling and disposal of these components is also much more difficult than with an LWR. This is particularly problematic for pebble bed reactor concepts. The total production of graphite dust is estimated by (Humrickhouse 2011; NASEM 2023b) at 15 kg/y to 100 kg/y. Reactor components contaminated with deposited radioactive graphite also become low-level, but also intermediate-level radioactive waste. Accidents such as water ingress or graphite fires can also generate graphite dust.

(Sun et al. 2020) provides an overview of the current state of research into the formation, transport and deposition of radioactive graphite dust. (Lensa et al. 2020) summarise the experience gained from the decommissioning and dismantling of HTR worldwide and recommend that calculations on the generation of activation products should already be carried out during the design process of an HTR in order to prevent the generation of excessive quantities of radioactive waste streams and to meet national acceptance criteria for final and interim storage. In addition to the assessment of

(Lensa et al. 2020), it should be noted that the formation of fission products, their transport and deposition behaviour in graphite and other reactor structures and the resulting quantities of intermediate, low-level and high-level waste would also have to be determined. The effects of oil, air or water ingress on these mechanisms and the resulting waste quantities should also be investigated.

Graphite disposal

(NASEM 2023b) describe the following technical means for conditioning the graphite: Irradiated graphite can be pretreated to reduce the radioisotope concentration of the waste. According to (Wickham et al. 2017; NASEM 2023b), possible treatments include oxidation, pressing processes, thermal decontamination and chemical treatments.

In oxidation, the graphite is partially or almost completely gasified to produce CO_2 that contains $^{14}CO_2$ (Worth et al. 2021; NASEM 2023b). After almost complete incineration, only 1-2% residues containing ^{60}Co, caesium-137 and ^{55}Fe remain (Fuks et al. 2020; NASEM 2023b). The carbon-14-doped CO_2 can either be discharged into the environment or captured (e.g. by precipitation with NaOH or amine separation) (IAEA 2004; 1995; NASEM 2023b). Since widespread use of VHTR would probably exceed the limit values if released into the atmosphere (NASEM 2023b), this would have to be followed by an expensive treatment of the graphite incinerator’s exhaust gas in order to sequester the CO_2, which would produce considerable amounts of radioactive lime (Del Cul 2002; NASEM 2023b).

In order to reduce the $^{14}CO_2$ concentration, the CO_2 from the graphite treatment can also be combined with CO_2 from fossil sources. Large voids in the irradiated graphite blocks are limited by pressing processes such as hot vacuum pressing, spark plasma sintering and hot isostatic pressing (Lee et al., 2013). In these processes, the irradiated graphite is crushed, embedded in a matrix (e.g. glass or bentonite) and then pressed to create a type of waste that is impermeable to water and has very low porosity (Yun-zhi Tan et al. 2021; NASEM 2023b). Due to the oxidation of the graphite to produce CO and CO_2, both of which are doped with carbon-14, thermal decontamination of irradiated graphite reduces its radioactivity (NASEM 2023b; Worth et al. 2021). Similar to bulk oxidation, CO_2 can be captured by precipitation at a high pH value (IAEA 2004; NASEM 2023b). One type of chemical treatment is to dissolve irradiated graphite with limited solubility in a high-temperature molten salt (Grebennikova et al. 2021; NASEM 2023b). This treatment can separate fission products, which would reduce the activity of the waste; however, it produces new fission product-contaminated LiCl and KCl waste in the process. According to (Bespala et al. 2019; Sivagami et al. 2019; NASEM 2023b), two other modern treatment approaches are microwave heating and electrochemical decontamination.

(NASEM 2023b) comes to the conclusion that the separation of graphite and TRISO particles is still a relatively new process. While electrochemical processing is an experimental technique, incineration and mechanical separation are somewhat more proven methods for extracting TRISO particles from the graphite matrix (Fuks et al., 2020).

(Fuks et al. 2020) summarise the research and development status of graphite waste:

- There is currently no single, generally accepted method that meets the requirements for the storage of radioactive waste and the disposal of waste gases generated during the processing of graphite.
- The most widespread and cost-effective method is storage over a longer period of time.
- In order for the graphite to be stored as low-level radioactive waste, the graphite must be extracted from the fuel elements.
- The main problem with the separation of fuel and graphite by physical processes is the formation of carbonates containing carbon-14.
- Reprocessing for waste treatment would be possible in connection with the recovery of fuel.
- The release of Wigner energy by graphite in the form of heat is something that requires special attention: if this effect is too strong, the waste itself could ignite.

Due to the fission and activation products it contains, graphite in itself exhibits negligible heat generation. However, irradiation creates defects in the crystal lattice of graphite. These defects store energy (Wigner energy), causing the graphite to expand. The stored energy may be suddenly released in the form of heat, and, if the Wigner energy is too high, the graphite may ignite. This can be prevented by heating the graphite and recombining the defects.

Graphite, and thus all waste, is also flammable, which entails additional requirements for safe transport, storage and disposal.

In order to create a knowledge base for issues related to irradiated graphite waste, the International Atomic Energy Agency has carried out two projects: GRAPA (Graphite Processing Approaches) and IMMONET (a database for data and reports on irradiated graphite). With the CARBOWASTE (Carbonaceous Waste 2008-2013) programme, the European Commission has also expanded the network of experts for evaluating solutions for the characterisation, recovery, treatment, recycling and disposal of irradiated graphite. In Germany, extensive research on graphite disposal has been undertaken in the scope of the CarobDISP project (Kuhne et al. 2015). However, despite the progress that has been made, the issue of carbon-14 release mechanisms under the operating conditions of the Konrad repository has not been conclusively clarified, and further research projects are needed.

Conclusion: Fuel supply and waste disposal

VHTR use TRISO fuel particles. The coated fuel particles with layers of pyrolytic graphite (PyC) and silicon graphite contain the majority of the radioactive inventory. These particles are embedded in a graphite matrix containing fission products from uranium impurities and from defective fuel particles. This multi-coating acts as the first barrier against release in the reactor and also during disposal. It is assumed that the coatings of spent TRISO fuel elements make them more resistant to a rapid initial release of radionuclides than conventional high-level forms of waste from LWR, making them generally suitable for final disposal. However, further research is needed to determine and demonstrate the effectiveness of the barrier in a repository environment.

Due to the graphite matrix, significantly higher volumes of spent fuel elements are produced than with LWR, with correspondingly more space required in a repository. Due to the high volumes of graphite waste, the separation of TRISO particles from the graphite matrix prior to conditioning is also being researched with the ultimate goal of reducing the volume of graphite waste. However, this requires a great deal of effort due to the contamination of the graphite waste and generates further waste streams. The resulting waste stream of medium and low-level radioactive graphite must also be disposed of, and prolonged interim storage will further reduce the activity of the waste. Further treatment of the waste stream, e.g. further separation of graphite from the particle coating as a means to reduce the volume, would be conceivable, but would generate additional waste streams.

The separation and processing of the graphite produces carbonates containing carbon-14. There is currently no generally recognised method for treating the processed waste. A conditioning concept for direct final disposal would therefore be simpler, albeit with significantly higher volumes compared to LWR waste. Storage over extended periods of time is widespread. In addition to the graphite matrix and the particle coating, other graphite waste is produced by structural elements in the reactor core.

Another relevant waste stream is generated by graphite dust. This dust contains fission products bound to dust particles, is distributed in the gaseous coolant in the reactor and settles at trouble spots (e.g. pipe bends). This contaminates reactor components and pipework.

VHTR are often associated with the utilisation of thorium fuel, even though most currently planned reactor concepts do not provide for the use of thorium until a later stage. The resulting lower build-up of transuranic elements, especially plutonium, the associated advantages with regard to the radiotoxicity of the waste, and a reduction in long-term heat release are frequently seen as significant advantages of a thorium-uranium fuel. However, this argument is not valid. The radiotoxicity of thorium-based fuels containing uranium-233 as fissile material is similar to the radiotoxicity of uranium-based fuels. However, the use of thorium fuel could be an advantage due to the lower transuranic element content in the fuel, especially in scenarios of use in fast reactors like MSR for transmutation.

If no in-situ breeding in the reactor is used, additional waste streams are also generated through the necessary reprocessing. Since the actual fissile material in a uranium-233/^{232}Th fuel cycle must first be bred from thorium, most concepts provide for the reprocessing and separation of the fissile uranium-233. In principle, nuclear fission in a VHTR produces similar quantities of radioactive fission products relative to the energy released as in LWR.

Beyond this, there are no significant advantages or disadvantages in the area of fuel supply and waste disposal of VHTR compared to LWR.

4.6.7 Proliferation risks

The following description of the proliferation risks of VHTR was adopted from (Oeko-Institut e.V. 2017) and further supplemented.

Since uranium fuels would have to be enriched in the same way as for light-water reactors, albeit to a uranium-235 content of up to 19% for VHTR (often 8-12%), uranium enrichment, which is sensitive to proliferation, would also be necessary for these fuels. When using thorium fuels, it is also necessary to have a corresponding proportion of fissile material in the fresh fuel (uranium-233, uranium-235 or plutonium). Uranium-233 or plutonium would have to have been separated beforehand, or enriched uranium (uranium-235) would have to be used upon initial start-up of the reactor. For more on the proliferation problem of thorium fuel, see Chapter 5.5.4.

With the use of uranium fuel, VHTR concepts are similar to heavy water moderated reactor systems with respect to the possibility of plutonium diversion. Both systems are less proliferation-resistant than light-water reactors when it comes to plutonium diversion. In addition to the amount of plutonium that can be produced and its weapons-grade quality, this especially applies to the possibility of removing fuel from the reactor during operation, depending on the burnup of the pebble and thus the plutonium content and its isotope vector. In pebble bed reactors, individual pebbles could thus be diverted from the pebble flow. With the help of the measurement signals, pebbles with the optimum plutonium content could be determined, then removed from the extraction section and transported to a reprocessing plant. This would merely require a reprogramming of the extraction equipment. With a mass of approx. 200 g, the individual pebbles are very manageable (Oeko-Institut e.V. 1989b). On the other hand, conventional reprocessing methods cannot be used for plutonium separation.

One advantage of TRISO fuels is that the reprocessing of both fresh and spent fuel is more difficult and has not yet been technically developed. Another slight advantage, depending on the diversion scenario, is that the fuel is embedded in a comparatively large volume of graphite, meaning that larger quantities of fuel would have to be diverted, for example.

VHTR can also be used to produce tritium by inserting pebbles containing lithium into the pebble bed. In the U.S., there were considerations in the 1980s to use a VHTR for this purpose in order to extract tritium for the U.S. nuclear arsenal, but this idea was later abandoned (Oeko-Institut e.V. 1989b). Tritium is used in nuclear weapons as an explosion booster and must be regularly replaced in the arsenals of nuclear weapons states (half-life: approx. 12 years).

Conclusion: Proliferation risks

With regard to proliferation, the use of uranium fuel requires the use of uranium enrichment technology. Due to the continuous exchange of fuel, pebble bed reactors are more susceptible to fuel diversion than LWR. They are also particularly suitable for tritium production.

Overall, it can therefore be assumed that pebble bed reactors have a slight disadvantage in terms of proliferation risks compared to LWR at the technology line level. Differences at the reactor concept level are mainly based on the choice of fuel cycle.

4.6.8 Costs

The economic efficiency of VHTR will essentially depend on the usable core outlet temperature, as other applications for process heat can be utilised with higher core outlet temperatures. Process heat must compete with fossil fuels, their price and the costs incurred for CO_2 emissions. VHTR thus experienced their first major boom during the oil crisis in the 1970s. Major oil companies, such as Gulf and Shell, made a massive financial commitment to VHTR and pushed for their market launch. By 1974, orders and options for VHTR with prismatic fuel elements with a total electrical output of 10 GW had been obtained, which was a breakthrough for the VHTR line considering the circumstances of the time. However, due to various unresolved technical problems, these orders had to be cancelled in 1975, and high penalties had to be paid, as a result of which the oil companies ultimately abandoned VHTR technology altogether.

(GIF 2021a) assume that demonstration projects with low temperatures between 700 °C and a maximum of 950 °C will be pursued in the near future. Typical VHTR from 750 °C to 850 °C can be used to generate electricity with an efficiency of up to 48%, and even higher when combined with process heat. They would then have to be built in the vicinity of large industrial complexes. The transfer of heat to a consumer over a distance of several kilometres can be achieved with steam, gas, molten salt or liquid metals (GIF 2021a).

High-temperature systems like VHTR have always been linked with the idea of using the process heat for the liquefaction or gasification of coal or for the production of hydrogen. For the members of the GIF, hydrogen production, in particular, is a major incentive.

The higher the working temperature, the higher the efficiency for electricity generation and the more efficient the possible coupled uses of the process heat, even enabling certain processes to be used in the first place. For example, utilisation as a desalination plant or for petrochemical processes and oil sand extraction is already possible at temperatures of up to 700 °C, whereas temperatures above 800 °C are required for coal gasification or hydrogen production from methane or with efficient high-temperature electrolysis, and temperatures of up to 1000 °C are required for the thermochemical production of hydrogen, with a threefold increase in efficiency of the processes that can then be used at 1000 °C compared to operation at 750 °C (WNA 2016a). The higher the working temperature of the reactor, the more industrial applications are possible and the more economically attractive the reactor concept becomes.

Even though the current development of the VHTR has achieved significant results, especially with the construction of the HTR-PM, new financing options for a VHTR demonstrator with process heat must be found as soon as possible to ensure that the substantial investments made to date pay off (Fütterer et al. 2021). (Fütterer et al. 2021) sees several initiatives here (see also Chapter 4.6.3), but their success depends on the political framework, social acceptance and economic boundary conditions (gas price, CO_2 price, financing risks).

Companies that are not active in the electricity generation market and would be newcomers to the nuclear sector also need to be won over for the utilisation of industrial process heat.

(GIF 2021a) points out that there are a number of detailed studies available on the costs of VHTR for electricity production and process heat that have been informally shared with the GIF. According to this source, the U.S. results show that the costs would be competitive with the costs of LWR. For process heat, they are competitive with conventional gas-steam combined cycle power plants if the cost of gas is higher than USD 8 per 1 million BTU (EUR 0.027/kWh). The threshold value is lower when CO_2 costs are included. Corresponding gas prices were available on the spot market in the U.S. for a short time in 2022 (also in 2000/2001, 2005/2006, 2008).[182]

An example of a detailed cost estimate of the SC-HTGR (see Chapter 4.6.3.3) can be found in (IAEA 2019b) with costs of USD 3,600/kWth for a module with 165 MWth and steam utilisation for electricity generation.

Conclusion: Costs

The major potential advantage of a VHTR is the possibility of combined heat and power generation and the production of process heat, which could also be used to produce hydrogen, depending on the temperature. This represents an advantage over LWR.

The actual costs could be competitive with LWR. However, as the investment costs for LWR depend heavily on the manufacturer and the country in which construction takes place, while, at the same time, the costs for VHTR depend heavily on the possibility of utilising the process heat, it is ultimately impossible to make a final assessment at the technology line level.

[182] https://www.eia.gov/dnav/ng/hist/rngwhhdM.htm

4.7 Accelerator-driven subcritical reactors (accelerator-driven system, ADS)

The basic idea of accelerator-driven subcritical reactors is to operate a fast reactor with a subcritical reactor core. Since such a reactor cannot maintain a steady-state (i.e., constant over time) neutron population by itself, it requires an “external” neutron source. The reactor system component is subcritical and produces no energy when the external neutron source is switched off. This arrangement is also known as an accelerator-driven system (ADS).

In many concepts, the external neutron source is realised by a proton accelerator whose particle beam collides with a target inside the reactor core and generates high-energy neutrons through a spallation process. The protons penetrate the heavy metal target and cause atomic nuclei of the heavy metal to burst (spallation), resulting in the generation of neutrons in these nuclear reactions. The neutron yield, i.e. the number of neutrons generated per proton, is a key factor in the performance of a spallation neutron source.

Since the 1990s, plans to generate energy with accelerator-driven reactor systems have been gaining new momentum. ADS potentially offer considerable safety advantages due to the subcritical arrangement. The possible use of ADS for the transmutation of radioactive waste is also under discussion. They offer advantages for transmutation due to the fast neutron spectrum and the greater flexibility with regard to the fuel composition compared to critical reactors.

The development of ADS is currently still in the concept and planning phase. In addition to the need to develop a subcritical fast reactor system, the development of a reliable spallation neutron source is required. The accelerator must operate for months without interruption. Accelerators are still expensive and large, although considerable efforts have been made to reduce their size and the construction costs (Biarrotte et al. 2015). Furthermore, there are technical difficulties with the removal of heat from the heavy metal target.

Earlier accounts and evaluation of ADS were already provided in (Oeko-Institut e.V.; ZNF 2015; Englert 2010; IANUS; Oeko-Institut e.V. 1999). Parts of the description there have been adopted here and supplemented.

4.7.1 System description

The essential characteristic of an ADS is the coupling of a spallation neutron source for neutron generation with a subcritical reactor for neutron multiplication and energy generation.

Reactor system

In these hybrid systems, the spallation target is surrounded by fuel elements. The arrangement is similar to that of a fast reactor, except that the spallation neutron source is located in the centre of the reactor core and supplies the neutrons required to operate the subcritical reactor, see Figure 4-8.

The power of the reactor is directly coupled with the power of the accelerator. The arrangement of the fuel elements in the reactor is selected in a manner that the reactor system is subcritical. The reactor functions here in the form of an “energy amplifier”, causing a chain of nuclear fissions through the multiplication of a neutron from the spallation neutron source.

The coupling results in competing priorities. In order to minimise the need for additional neutrons for a specified output of the entire ADS, the reactor core should have the lowest possible subcriticality.

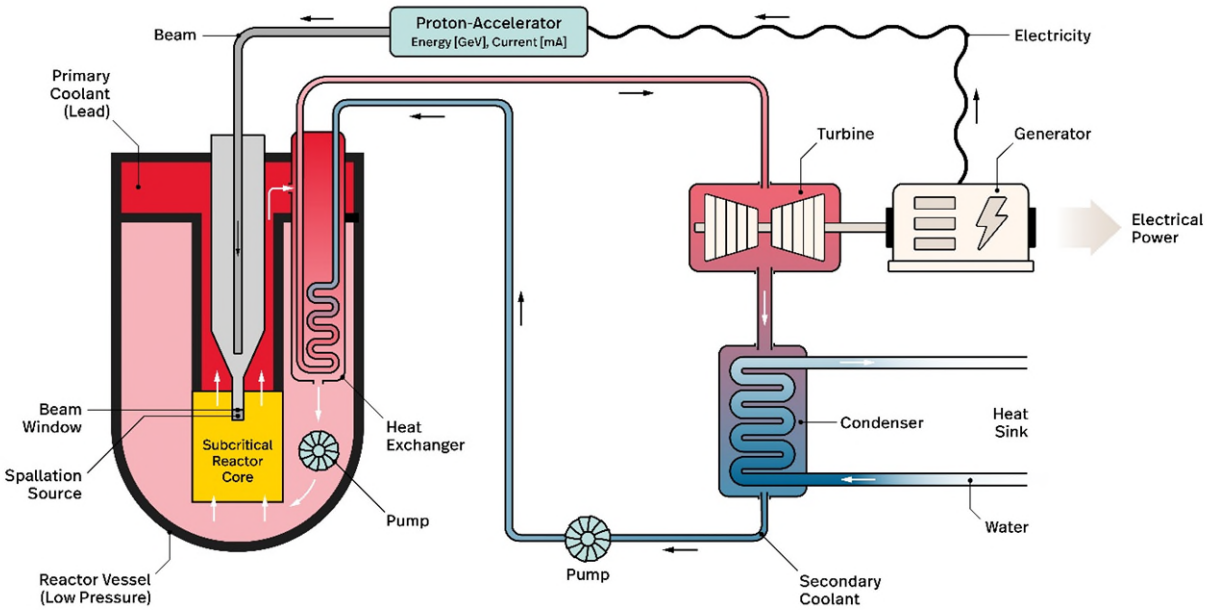
This keeps the power and thus the costs of the accelerator and the spallation neutron source as low as possible.

The closer the system gets to a critical configuration, the more important the reactivity control of the reactor becomes. For safety reasons, the distance to criticality should be kept as large as possible. With sufficient subcriticality, the use of additional reactivity control devices (e.g. control or shutdown rods) can be dispensed with in order to control reactivity changes in the reactor. The reactor is controlled solely by the neutron source; it is dominated by the neutron source, and switching off the accelerator and thus the neutron source should immediately result in a reduction in power. Conversely, an increase in the neutron source leads to an instantaneous increase in reactor power, but not to a power excursion, as in a critical system. There are no final criteria for defining an optimum level of subcriticality (Abderrahim et al. 2021).

The spallation neutron source and the reactor core are located in a reactor vessel. As there is a technical vacuum in the accelerator to prevent obstruction of the particle beam, the vacuum in the accelerator must be separated from the spallation target and reactor core, e.g. by means of a so-called beam window. However, other concepts are also possible.

With the help of a coolant, a pump is used to transport the heat generated in the reactor core and in the spallation target via a heat exchanger into the secondary system, where it drives a turbine and a generator to produce electricity using a water-steam circuit. Part of the electricity generated in the generator is then required for the operation of the accelerator. The residual heat is transferred to a heat sink via a heat exchanger.

Figure 4-8: Conceptual diagram of an accelerator-driven subcritical reactor (ADS)



Source: Own illustration

Fuel

A number of fuels can be used in ADS. Thorium fuels, in particular, were long under discussion, but uranium fuels or MOX fuels are also possible. A special feature of ADS is that fuels with a very high fissile material content of up to 40% minor actinides can be used in the fuel without compromising reactivity control. In uranium-free fuels, plutonium can be used as the remainder of the fissile material, allowing the potential use of a fuel element made of transuranic elements in ADS (Abderrahim et al. 2021).

For further properties of the fuel, see also the technology line for critical lead-cooled reactor systems (Chapter 4.2) and for reactor concepts presented in Chapter 5.3 and Chapter 5.10.

Fuel cycle

ADS are discussed in connection with a large variety of fuel cycles. With a powerful external neutron source, a number of ADS concepts can be conceived as pure thorium reactors without having to use additional fissile material at start-up, as in critical systems. For the special features of thorium fuel cycles, see Chapter 0.

The physical properties of ADS make them particularly suitable for the transmutation of waste (Accelerator Transmutation of Waste). Historically, there were also plans to use the neutron source with or without a subcritical reactor to breed fissile material (plutonium) for civil and military applications and to breed tritium for nuclear weapons (see Chapter 4.7.2).

Spallation neutron source

Due to its subcritical design, the actual fast reactor core is not capable of maintaining a critical chain reaction on its own. For this reason, protons or other particles (e.g. deuteron, He-4) are accelerated to high energy in a high-current accelerator and shot at a heavy metal target. The heavy metal atoms in the target burst, releasing a large number of neutrons. These then induce nuclear fission in the reactor surrounding the target, during which the original neutrons are multiplied again. The power released in the reactor is therefore proportional to the power of the accelerator beam (i.e. the product of the energy of the protons and the proton current).

The most important parameters of a spallation neutron source are the number of accelerated particles over time (accelerator current), their impact velocity (energy) and the number of neutrons produced in the spallation reaction. The more accelerated particles (e.g. protons) collide with the heavy metal target, the more spallation reactions are induced. The higher the energy of the particles, the more neutrons are generated by the disintegration of the atomic nuclei upon impact. Protons are usually used in the accelerator, but other particles are also possible. Typical energies at which the ratio between neutron generation and the energy of the accelerated particles is most favourable lie between several hundred megaelectronvolts (MeV) and two gigaelectronvolts (GeV). In this energy range, the number of neutrons released per spallation reaction is proportional to the proton energy. Depending on the target material, approx. 10-30 neutrons are released per spallation reaction at these energy levels.

Accelerator

Linear accelerators (LINAC) or circular accelerators (cyclotron or synchrotron) are usually used. The dimensions of the accelerator are determined by the energy of the accelerated particles to be achieved. This represents a major cost factor.

The key characteristic of a circular accelerator is that the orbital frequency of a cyclotron is fixed and independent of the particle energy. It can be built much more compactly and has a diameter of approx. 5-20 metres, depending on its performance. This makes cyclotrons inexpensive to build. Disadvantages lie in the limited possibilities they offer in terms of beam focussing and beam extraction from the accelerator, as only a small fraction of the particle beam can be extracted from the accelerator and the achievable accelerator currents are limited (10 mA at 1 GeV). It is also more difficult to control the duration of sudden beam interruptions (Nupecc 2001; Abderrahim et al. 2021).

The advantage of linear accelerators is that, due to the linear arrangement, beam focussing elements can be placed at regular intervals and thus up to two orders of magnitude higher current can be transported than with cyclotrons without suffering beam losses. This means that currents of up to 100 mA and more can, in principle, be achieved using the technology available today (Smith and Schneider 2004). Linear accelerators are better suited for fault tolerance for short beam interruptions. However, linear accelerators are more costly due to their large dimensions and the corresponding radiation protection requirements (van den Eynde et al. 2013). The acceleration distance with linear accelerators can reach several hundred metres.

Most ADS are planned with accelerators with continuous operation, but pulsed operation would also be possible. The accelerator must operate as reliably and efficiently as possible and without frequent failures. A beam failure in the reactor leads to the equivalent of a emergency reactor shutdown, resulting in corresponding thermal stress in the reactor and requiring a lengthy procedure to restart the system. An ADS system must therefore be designed to withstand short beam interruptions of a few seconds.

Spallation target

A limiting factor for the performance of spallation neutron sources is the removal of heat from the heavy metal target. The power of the accelerated particles is largely deposited in the target on impact and must be dissipated through the use of suitable cooling methods. As the design of a spallation neutron source in the research sector and in an accelerator-driven reactor is made to ensure that as many neutrons as possible leave the surface of the target, the target is highly compact. Typical targets have the shape of a narrow cylinder or flat box with a height or diameter of 10-20 cm and a length of up to one metre. As around 2/3 of the total power of the particle beam is deposited as heat within this small volume, cooling represents a major technical challenge. Today’s systems are equipped with liquid metal targets instead of solid tungsten or uranium targets, as the liquid heavy metal simultaneously produces neutrons and can serve as a coolant. Due to their low melting temperatures, suitable target materials include lead, lead-bismuth alloys and mercury, for example. The disadvantage of liquid metal is the high corrosiveness of the coolant.

A spallation neutron source can be designed with or without a beam window. A beam window consists of one or two thin (approx. 1-3 mm) metal plates (e.g. aluminium, Iconel), which are cooled with a liquid or gas. A beam window decouples the high vacuum of the accelerator from the actual target and surrounds the liquid metal. The disadvantage of a beam window is that it is hit first by the high-energy particle beam and is therefore subject to corresponding material damage. However, if the target is also within the high vacuum, no beam window is required. Without a beam window, the particle beam hits the liquid metal directly. However, such an arrangement is technically difficult to implement, as the entire spallation neutron source must be kept under vacuum while impurities are generated by material evaporation and from gaseous fission products from the target material. Nevertheless, this prevents the thermal stresses and radiation-induced material damage that would be caused by the intense particle beam in a beam window.

Coolant

Comparable to critical fast reactors, three coolants are currently being discussed for use in an accelerator-driven reactor: sodium, lead-bismuth and helium. The choice of coolant determines the basic design of the reactor and the resulting safety strategies. High-pressure helium cooling is susceptible to loss-of-coolant accidents, whereas lead-bismuth and sodium cooling systems operate at low system pressures. As a coolant, sodium has the property of being flammable. In contrast, gas cooling is unproblematic in terms of corrosion damage which is relevant for lead-bismuth and sodium cooling systems. Gas-cooled reactors or liquid metal-cooled lead/lead bismuth reactors do not require fuel element channels for flow control and must be operated with a low power density in the reactor.

There is now a broad consensus among ADS developers that the most suitable coolant is lead or a lead-bismuth alloy (Abderrahim et al. 2021).

Spectrum (moderator)

The energy spectrum of the subcritical reactor depends on the purpose of the reactor. However, almost all historical and current systems provide for a fast spectrum.

Temperature and pressure

The pressure and the temperature are similar to the conditions discussed in the technology line for lead-cooled reactors (see Chapter 4.2).

4.7.2 Historical developments

Parts of the following account were adopted from (Englert 2010; IANUS; Oeko-Institut e.V. 1999) and supplemented.

The original idea of using large accelerators and neutrons from the spallation process for the transmutation of radionuclides originated in the 1950s. At that time, however, the intention was not to convert the actinides of the nuclear waste into fission products; on the contrary, fissile material was to be produced with the aid of spallation neutrons.

The possibilities of producing fissile material for the nuclear weapons programme in the U.S. were investigated in the scope of numerous projects, ranging from theoretical studies to demonstration facilities. The MTA (Materials Testing Accelerator) was investigated during the period between 1950 and 1954 at the Lawrence Livermore Laboratory in the U.S. as an alternative to the Hanford and Savannah River reactors for the production of fissile material. In 1952, an electro-nuclear production factory called A-12 was developed, which was designed to accelerate deuterons to 350 MeV with a current of 500 mA. The deuterons would then have been transformed into neutrons by spallation, which in turn would have produced Pu-239. A further, improved concept was developed, which received the name C-50. However, this project was cancelled in 1954 and declassified in 1957 (Magill and Peerani 1999; DOE 1999b).

Several studies on the concept of an accelerator breeder resurfaced in the 1970s. Examples include the “Linear Accelerator Breeder” (Brookhaven National Laboratory 1976) and a concept by the Canadian group that had developed the ING for the production of fissile material (Chalk River Nuclear Laboratories, Atomic Energy of Canada 1978; Atomic Energy of Canada Limited, Chalk

River Nuclear Laboratories 1981). Studies on accelerator breeder concepts were also carried out in Russia in the 1970s (see (DOE 1999b)).

Experiments on the production of plutonium were conducted in the “Fertile-to-Fissile” programme at Los Alamos National Laboratory in 1988 (Gilmore et al. 1988). In Russia, there were also a number of investigations in the 1970s and 1980s into the spallation reaction in uranium and the production of fissile material (Vassil'kov et al. 1991).

France (Lagniel 1998) and the U.S. launched programmes for the production of tritium with spallation neutron sources in the 1990s. The U.S. programme “Accelerator Production of Tritium” (APT) was launched in 1995 and was continued as an alternative until 2002 despite the decision in 1998 to secure tritium production for the weapons programme by irradiating lithium blankets in commercial reactors (Anderson et al. 1999). Among other things, the LEDA (Low Energy Demonstration Accelerator) project was realised (Smith and Schneider 2004).

Since the 1990s, more and more work has been done on concepts for accelerator-driven reactors. One milestone is the idea of the “energy amplifier” developed by Carlo Rubbia (Rubbia 1995; CERN 1995).

The energy amplifier (EA) is a subcritical, fast reactor conceived by a group at CERN led by Nobel Prize winner Carlo Rubbia. The EA utilises various passive safety systems, including, in particular, a passive system for interrupting the accelerator beam and a passive system for heat removal. A liquid lead-bismuth mixture serves as the coolant for the EA, which is intended to circulate by natural convection, i.e. without additional pumps.

The EA uses thorium as a fuel matrix in the form of MOX. The fissile uranium-233 builds up in this matrix during irradiation. During commissioning, additional fissile material is required for the first load. The spent fuel must be reprocessed in order to achieve the low radiotoxicity of the waste envisaged by the EA concept. This reprocessing allows all actinides and a number of long-lived fission products to be separated and fed back into the EA. For the remaining fission products, near-surface storage is provided for in the concept. The EA should be able to supply energy and reduce the radiotoxicity of waste fed to it from other critical reactors. Particularly in countries that have not yet decided on a strategy for the treatment of nuclear waste, the EA should therefore draw interest as a means to transmute waste (“incinerator”) and less as a supplier of energy: the amount of transuranic elements that an EA with a thermal capacity of 1500 MW can burn is expected to amount to roughly 400 kg per year (CERN 1997).

Since the end of the 1990s, there have been numerous research activities worldwide that have explored the possibilities afforded by ADS. A notable example of this is a IAEA benchmark study (ANS 2001) in which many research groups took part.

Parallel to the work on ADS, new concepts for the transmutation of radioactive waste have also been proposed (Accelerator Transmutation of Waste (ATW)), which envisage the treatment of radioactive waste with spallation neutrons for the purpose of converting particularly long-lived isotopes (Arthur et al. 1995; DOE 1999a).

Overall, one can speak of a renewed interest in accelerator-driven systems that has prevailed since the 1990s, as accelerator technology has now reached a level that could potentially fulfil the various use interests. Apart from prototypes, however, only spallation neutron sources for use in research have actually been realised. The use of spallation neutron sources in research to investigate the fundamentals of nuclear physics and the structure and dynamics of matter, as well as the possibilities of future ATW and ADS systems, is also the main purpose of current spallation neutron sources.

Today, there are a number of concept studies underway worldwide on the development of accelerator-driven facilities. A detailed overview of the systems currently under discussion and the state of research in various countries is provided by (IAEA 2015; NEA 2005), for example. However, most of these concepts remain in the project planning phase. Computer-aided simulation programmes have been developed for concept studies and compared with experimental data obtained at the spallation neutron sources that were built for research purposes.

4.7.3 Current developments

Parts of the following account were adopted from (Oeko-Institut e.V.; ZNF 2015; Englert 2010; IANUS; Oeko-Institut e.V. 1999) and supplemented.

Since the 1990s, there has been a renewed interest in accelerator technology for the production of neutrons by spallation, the initially focus of which was on the construction of spallation neutron sources for research purposes (DSF 2009). Large research spallation neutron sources in operation with outputs in the megawatt range include:

- Spallation Neutron Source (SNS) (Oak Ridge National Laboratory – U.S., 1.4 MW)
- MEGAPIE, formerly SINQ (Paul Scherrer Institute – Switzerland, 1.4 MW)

There are also two smaller systems with an output of several 100 kW:

- LANSCE (Los Alamos National Laboratory – U.S., 80-120 kW)
- ISIS (Rutherford Appleton Laboratory – United Kingdom, 200 kW)

China is planning a Chinese ADS project (Chinese Accelerator-Driven System (CADS)), which currently only includes the accelerator, the target and a blanket.

Basic research into accelerator-driven systems is being conducted in Japan, primarily at the J-PARC accelerator. There are also plans to build an experimental system with a thermal output of 60 MW in the future and, later, a commercial system with a thermal output of 800 MW (1 GeV, 20 mA) (IAEA 2015).

In addition, there are intentions to develop accelerator-driven reactors in India (Vandeplassche and Medeiros Romao 2012) and South Korea (IAEA 2015), though these plans have not yet passed the stage of a preliminary study either.

In China, the Accelerator-Driven Advanced Nuclear Energy System (ADANES) has been proposed as a complete energy system for power generation and the transmutation of minor actinides and for fissile material production with an electrical output of 1000 MW.

Despite extensive preliminary studies (DOE 1999a; Bowman 1998), the U.S. has not developed an accelerator-driven system to demonstrate its functional principle or participated in international projects. However, the U.S. has expertise on many technical aspects of an accelerator-driven system, particularly on high-current accelerators and spallation neutron sources (SNS, LANSCE, etc.).

European research efforts are concentrated at the Belgian Nuclear Research Centre (SKC-CEN). This research began with the ADONIS project (1995-1996), which focussed on the production of medical isotopes. From this project, however, the MYRRHA project (Multi-purpose HYbrid Research Reactor for High-tech Applications) emerged in 1997. In the European Commission’s Sixth Framework Programme, the MYRRHA project was continued as part of the EUROTRANS programme (2006-2010), in which the reactor concept EFIT (European Facility for Industrial Transmutation, see below) was also developed as a larger ADS project.

(Kadi 2016) provides an overview of previous and ongoing projects and experiments (see Table 4-12).

Table 4-12: Status of ADS projects and experiments

	Country	Beam characteristics	Spectrum/ Power	Purpose and status
FEAT	CERN	Protons (0.6 to 2.75 GeV) (~10^{10}p/s)	Thermal (≈ 1 W)	Reactor physics of thermal subcritical systems (k≈0.9) With SNS - completed
TARC	CERN	Protons (0.6 to 3.5 GeV) (~10^{10}p/s)	Fast (≈ 1 W)	Lead slowing-down spectrometry and transmutation of long-lived fission products - completed
MUSE	France	DT (~10^{10}n/s)	Fast (< 1 kW)	Reactor physics of fast subcritical systems - completed
YALINA	Belarus	DT (~10^{10}/s)	Fast (< 1 kW)	Reactor physics of thermal and fast subcritical systems - completed
MEGAPIE	Switzerland	Protons (600 Me) + Pb-Bi (1 MW)	-----	Demonstration of a 1 MW target for a short time - completed
TRAD	Italy	Protons (140 MeV) + Ta (40 kW)	Thermal (200 kW)	Demonstration of an ADS with thermal feedback - cancelled
TEF-P	Japan	Protons (600 MeV) + Pb-Bi (10W, ~10^{12}n/s)	Fast (< 1 kW)	Coupling of a fast subcritical system with an SNS in a configuration with MA fuel - reactivated
SAD	Russia	Protons (660 MeV) + Pb-Bi (1 kW)	Fast (20 kW)	Coupling of a fast subcritical system with an SNS - cancelled
EFIT	EU	Protons (800 MeV, 20 mA)	Fast (400 MWth)	Commercial facility - cancelled
TEF-T	Japan	Protons (600 MeV) + Pb-Bi (200 kW)	-----	Facility for demonstration and development of a long-term material database - reactivated
MYRRHA	Belgium	Protons (600 MeV) + Pb-Bi (1.8 MW)	Fast (60 MW)	Experimental ADS - in conceptual phase, not fully funded, 2025?
CADS	China	Protons (0.6 –1.5 GeV)	Fast (100 – > 1000 MW)	Four-phase project: 2011-2032
ADANES	China		1000 MWe	Proposed
U-ADS	Ukraine	Protons (100 MeV)	100 kW	Uranium-based ADS prototype (KIPT) - status ?
ADS	Russia	Protons (250-500 MeV)	1-5 MW	Utilisation of an existing facility in Troitsk - proposed by iThEC

Source: (Kadi 2016) and own research.

4.7.3.1 MYRRHA

For more on MYRRHA, see Chapter 5.10.

4.7.3.2 European Facility for Industrial Transmutation of Minor Actinides (EFIT)

This accelerator-driven system represents an initial conceptual design study for an industrial plant with a thermal capacity of 400 MW cooled with a lead-bismuth alloy (Artioli et al. 2008). The subcriticality of the reactor is characterised by a criticality factor of ke_{eff} = 0.97. This choice was based on previous studies on the XADS and XT-ADS designs (IAEA 2015). The spallation neutron source for EFIT is to be operated with a current of 20 mA and a proton energy of 800 MeV at an output of 16 MW (Biarotte and Müller 2011; Biarrotte et al. 2015; IAEA 2015). EFIT would thus build directly on the experience gained from the MYRRHA demonstration reactor.

EFIT is to be operated with fuel with an inert matrix. The ratio of minor actinides to plutonium in the fuel is to be selected in such a way that the plutonium balance is equalised over the duration of use in the reactor, i.e. plutonium is neither burned nor produced (“isobreeder”). This results in a plutonium content of approx. 45% with 55% minor actinides (IAEA 2015). This ratio causes a relatively low loss of reactivity during burnup. This makes it largely unnecessary to use the spallation neutron source to compensate for reactivity losses during burnup, which requires a lower maximum power of the accelerator. As a result, the costs for the spallation neutron source can be reduced.

The industrial plant should achieve a transmutation rate of 35 kg of minor actinides per TWh and, at a capacity of 80-85%, thereby burn around 100 kg of minor actinides per year (Abderrahim et al. 2021).

The majority of the work on EFIT was completed in the 6th Research Framework Programme of the European Union (2002-2006) in the EUROTRANS project (2005-2010, approx. EUR 43 million, EU contribution: EUR 23 million).

4.7.3.3 ARTMS

In Canada, a consortium led by Deerfield Management Company and Quark Ventures is working on the development of an accelerator-driven plant for the production of the radionuclide Mo-99. In May 2020, ARTMS (Alternative Radioisotope Technologies for Medical Science) Inc. announced that it would support this consortium with a total of 19 million Canadian dollars. The development is being carried out in collaboration with the Canadian particle accelerator centre TRIUMPF (GIF 2021a, p. 10).

4.7.3.4 CADS / CLEAR-I

Two experimental reactors are planned in China, the China LEAd-based Research Reactor (CLEAR-I) and the China initiative Accelerator-Driven System (CADS). The development line also includes the critical system CLEAR-M from the technology line of lead-cooled reactors (Chapter 4.2.3).

CLEAR-I is a lead-bismuth cooled 10 MWth fast experimental reactor to be operated both critically and with a 250 MeV/10 mA accelerator with spallation neutron source. The beam enters from above through a beam window and is directed to the target in the centre of the reactor core. The fuel is uranium oxide with an enrichment of 19.75% in steel fuel cladding tubes, but the reactor is to test different fuel types. The reactor has the typical double-walled configuration of an LFR in a pool design (see also Chapter 4.2 on LFR and Chapter **5.10** on MYRRHA) with 61 fuel rods in a triangular

arrangement. The control rod is retracted from above by weights and a drive spring against buoyancy. Four heat exchangers in double-walled bayonet tubes and a passive air cooling system for residual heat removal transport the heat from the coolant. The coolant circulation is ensured by two vertically installed pumps in the cold section of the pool. Two test facilities were built at the Institute of Nuclear Energy Safety Technology (INEST) in Hefei, China: the CLEAR-S technical validation reactor, which is the same size as CLEAR-I and has the same lead inventory and can be heated to 2.5 MW, and the CLEAR-0 zero-power reactor coupled with the HINEG neutron generator for critical and subcritical operation. A lead-bismuth experimental plant (KYLIN-II) with no nuclear inventory was also built and has already been in operation for 30,000 h. In addition, a virtual CLEAR-V reactor was developed to test and validate the key components. (Alemberti 2021; GIF 2020a)

CiADS is also a fast 10 MWth reactor cooled with lead-bismuth including the power of the accelerator. CiADS is to be used primarily for transmutation research. The facility is intended for research into reliable continuous accelerator operation and the spallation neutron source, as well as for coupling the system components. Software is also to be developed that will allow the conception of a subsequent ADS demonstration facility. The reactor has the typical configuration of an LFR in a pool design (see also Chapter 4.2 on LFR and Chapter 5.10 on MYRRHA) with two coolant pumps and four heat exchangers. The decay heat is transported via natural convection and passive cooling. In order to minimise corrosion, the core inlet and outlet temperatures are 280 °C and 380 °C (see Chapter 4.2.3.10). A molten salt is specified as a secondary coolant. (Alemberti 2021)

4.7.4 Technical development status

Parts of the following account were adopted from (Oeko-Institut e.V.; ZNF 2015; Englert 2010) and supplemented.

Until now, there has been no prototype accelerator-driven system in operation anywhere in the world. The next step in the development of accelerator-driven systems, therefore, is the construction of one or more prototype reactors before an industrial demonstration plant can be built.

Lead-bismuth is currently being pursued as the main coolant. So far, however, there has been little experience worldwide with lead-bismuth cooled reactors. Initial research was carried out in the U.S. in the 1950s and 1960s. The main problem with cooling is corrosion problems at higher temperatures. Experience has also been made with lead as a coolant in reactors for the Russian submarine programme (Mitev 2014). In Russia, the construction of the prototype reactor SVBR-100 is also in planning (Rosatom 2015), as well as the BREST-OD-300 (see Chapter 5.3). For more on the development status of lead-cooled fast reactors, see also Chapter 4.2.

Accelerators are currently being developed mainly for use in fundamental physics. The further development of superconducting accelerator components has driven the technological advances of the last two decades. Whereas, in the 1980s and 1990s, currents of a few 0.1 mA still represented the vanguard of development, today high-current accelerators with a current of a few milliamperes are available for the operation of spallation neutron sources. Currently, the most powerful spallation neutron sources are the SNS at Oak Ridge National Laboratory in the U.S., which is operated with a linear accelerator with a power of 1.4 MW (current 1.4 mA, energy 1 GeV, length 330 m) and the SINQ at the Paul Scherrer Institute in Switzerland, which is operated with a cyclotron with a power of 1.4 MW (2.3 mA, 590 MeV, diameter 9 m). Continuous accelerator currents of approx. 10-20 mA are required for an industrial application in an accelerator-driven system. This has been technically

demonstrated and could be achieved industrially within the next one to two decades with appropriate investment in research and development.

High accelerator availability can be expressed by long average times between two beam interruptions and can be achieved by redundancy, design margins and fault tolerance against beam interruptions. In the case of MYRRHA, a high average time between two beam interruptions of 10 days is to be achieved. This corresponds to three months of operation with “extremely low values” (Biarrotte et al. 2015) of a maximum of 10 beam interruptions lasting more than three seconds. These “extremely strict reliability specifications” (Biarrotte et al. 2015) are necessary because every prolonged beam interruption leads to thermal stress in the beam window of the spallation target, whose material is under extreme stress due to the impact of the particle beam. On the other hand, extended beam interruptions also result in power drops in the reactor and thus to reactor transients with corresponding thermal stresses for the reactor system. If the reactor is shut down unintentionally, a restart with all procedures can take up to 20 hours (Biarrotte et al. 2015). However, according to (Biarrotte et al. 2015), studies from Japan and the U.S. specify less demanding requirements for the reliability of accelerators.

The development of a corresponding interface between the target and the beam tube is also the subject of research. This can be designed as a beam window that is, however, exposed to the significant flux of high-energy protons at high temperatures and in a corrosive environment. It is therefore planned to replace the beam window at regular intervals. This replacement can be avoided with a windowless design. However, this creates the problem of volatile spallation products entering the proton beam tube and contaminating the accelerator. In this design, however, a “cold window” could be installed on the last few metres of the proton beam at a sufficient distance away from the spallation reactions, where the greatest heat is generated. The cold window would then only be irradiated by protons and protect the accelerator against radiation (Abderrahim et al. 2021).

Furthermore, one of the key challenges for powerful accelerators is minimising beam loss. Beam losses result in the activation of the surrounding system structures. In the interest of radiation protection, therefore, the resulting heat output must be reduced to approximately one watt per metre of the wall of a linear accelerator so that the accelerator can still be maintained by personnel. This means that, as the beam power increases, the proportion of beam losses must be kept lower and lower and reduced, for example, to one millionth of the beam power per metre in a megawatt system (Plum 2013).

The high radiation exposure and the high temperatures required for liquid metal expose the structural material of the target to a corresponding degree of material stress. One of the goals of spallation target research and development is to find materials that are suitable for a commercial system with correspondingly high power and its requirements for long service life, low material fatigue and compatibility with the structural materials. For commercial operation, the target will need to be replaced at least once a year; due to the deposited activity, this must be executed using remote handling (IAEA 2015).

Post-irradiation examinations for a lead-bismuth alloy in a spallation target were carried out on the MEGAPIE target at the Paul Scherrer Institute (Wohlmuther et al. 2015). The SNS at Oak Ridge National Laboratory in the U.S. operates with a mercury target.

To date, no material is known that would withstand the conditions of a particle beam from an industrial accelerator-driven facility (beam density of 0.15 mA/cm^2) for one year (NEA Proc. 2015).

Conclusion: Technical development status

Compared to other technology lines, ADS are lagging behind significantly in terms of development. To date, there is no running prototype of such a system anywhere in the world that combines a spallation neutron source with a subcritical reactor. Concrete plans to finance such a system with international partners exist with the MYRRHA project in Belgium, where a first accelerator-driven research and demonstration reactor is to be built.

For an assessment of the development of lead-cooled fast reactors, see Chapter 4.2 and Chapter 5.3.

However, experience has been collected worldwide with spallation neutron sources without coupling with a subcritical reactor. Such sources are used in research to generate neutrons. Moreover, operating experience in the operation of high-current accelerators in the area of research can be transferred to the development of accelerator-driven systems. The functionality of spallation targets in the megawatt range has been confirmed under relevant operating conditions. However, there is still development work to be done to develop neutron sources with an output of 10-20 MW for industrial facilities.

Research and development on spallation neutron sources must therefore be categorised as lying between “applied research” and “development”.

Not all technologies have yet been validated under relevant operating conditions, especially with regard to the coupling of all system components to form an overall system and its safe handling.

Research and development on ADS should therefore be categorised as “applied research”.

4.7.5 Safety

Parts of the following account were adopted from (Oeko-Institut e.V.; ZNF 2015; Englert 2010; IANUS; Oeko-Institut e.V. 1999) and supplemented. Its argumentation is essentially based on (NEA 2002; 2006).

The main safety advantage of accelerator-driven, subcritical systems is the subcritical operation of the reactor. An ADS reacts much more favourably to a sudden reactivity addition than critical systems. As long as the sum of the criticality of the subcritical reactor and the reactivity input does not result in a critical configuration, ADS only react to such reactivity inputs with a fixed increase in power.

Accelerator-driven reactors, therefore, differ from critical fast reactors in their reactivity behaviour. Due to the subcriticality, the margin of additional reactivity that can be introduced into the reactor system is much higher before a significant increase in the power of the system occurs, making the reactivity accidents much easier to control. If the subcriticality of the system is sufficient, reactivity-induced incidents can even be ruled out entirely (Abderrahim et al. 2021).

This behaviour of subcritical reactors can compensate for the disadvantageous effects with respect to the reactivity coefficients and the reduced proportion of delayed neutrons that arise due to a high content of minor actinides in the fuel. In critical reactors such as sodium-cooled fast reactors, the fuel may only have a minor actinide content of 2-5% (Abderrahim et al. 2021). In ADS fuels, this value can be increased to up to 40%.

Typical criticality factors of subcritical reactors include, for example, k_{eff} = 0.95 for MYRRHA and k_{eff} = 0.97 for EFIT. The closer the reactor arrangement is to criticality with k_{eff} = 1, the stronger the neutron multiplication effect and the power amplification of the accelerator-driven system. As a result, fewer external neutrons are required to generate the thermal power and the accelerator has to deliver less power – a direct cost advantage. However, this must be weighed against the safety margins, as the reserves decrease compared to an additional reactivity input into the reactor (Pistner 1999; NEA 2006). Therefore, economic and safety aspects have an impact on the selection; there is no purely physical criterion for determining the optimum subcriticality. (PSI 1996) categorised the criticality factor k_{eff} = 0.98 provided for the energy amplifier as sufficiently subcritical, but also point out need for further analyses of accidents regarding the reactivity of the system. Elsewhere, it is noted that a value of k_{eff} = 0.98 could render it difficult to demonstrate the safety of the system (Assemblée Nationale 1997). This is due, among other things, to the maximum change in k due to protactinium decay (after reactor shutdown) of +0.02 (see (Rubbia, C., et al 1997)). On the one hand, the subcriticality of the Energy Amplifier is cited as the decisive safety advantage of the system; on the other hand, a EURATOM study concludes that the EA is nevertheless no better protected against power excursions than a modern LWR (European Commission, Office for Official Publications of the European Communities 1997).

The possibility exists of a potential reactivity addition through, for example, fuel rod failure and subsequent compression or reduced neutron leakage. Moreover, a low power density discourages a compact design and increases costs (NEA 2002).

Even if the beam power in the accelerator is increased, only a constant power increase will occur. Moreover, the increase in power is not abrupt in either case. With an assumed doubling of the beam power, the power in the reactor increases immediately by 50-60% and, depending on the subcriticality, the reactor only reaches the full double power asymptotically after over a minute. Only when the reactor has just reached criticality (k_{eff} = 1) does the power increase linearly with the available accelerator power. In a supercritical state, the reactor power would also increase exponentially, regardless of the external neutron source (NEA 2006).

There are two fundamental ways to regulate the power of a subcritical accelerator-driven system on short time scales of a few seconds. The first is by the power of the accelerator beam itself, since, in the subcritical state, the total power of the reactor is proportional to the beam power. The second is by using control rods that introduce negative reactivity into the reactor in a similar way to critical reactors. However, the effectiveness of negative reactivity addition through control rods is much lower for subcritical systems than for critical systems. The same negative reactivity addition that immediately reduces the power of a critical reactor to approx. 10% only reduces the power in a subcritical system with k_{eff} = 0.95 to approx. 60% and, with k_{eff}= 0.97, only to approx. 50%. The more subcritical the system, the lower the effectiveness of negative reactivity. This behaviour is particularly significant for transients during which the accelerator cannot be controled or shut down. However, dispensing with the regulation of a reactor through control rods also eliminates another potential cause of incidents.

Important from the point of view of technical safety are reactor transients in which the accelerator beam shuts down as predicted (protected transients) and those in which the shutdown does not occur as predicted (unprotected transients). The latter require special attention, as the dynamic behaviour of accelerator-driven systems in this case deviates significantly from the behaviour of critical reactor systems (NEA 2006).

The main technical safety challenges of accelerator-driven systems are as follows (NEA 2006):

- the dynamic response of the subcritical system to protected or unprotected transients in the event of power change, beam failure, sudden changes in beam power, a sudden increase in reactivity (e.g. due to loss of coolant circulation, reduction in coolant density, blockage of coolant, beam window failure with ingress of coolant) or loss of main heat sink,
- the shutdown characteristics of ADS without control rods from the nominal power to cold state,
- the occurrence of high power spikes in the reactor, especially in highly subcritical systems due to the external neutron source.

The analyses of protected transients with shutdown of the accelerator beam led to the following findings:

- The behaviour of the system during a protected transient is comparable to the situation where the shutdown rods are inserted into a critical reactor. The neutron flux and the reactor power decrease until the power is determined solely by the decay power in the reactor (NEA 2006).
- In an ADS without control rods, the reactor reacts relatively slowly in the event of a shutdown and, depending on the subcriticality, requires several seconds to several tens of seconds until the reactor power drops to a value of less than 10% (at k_{eff} = 0.95 only one second, at k_{eff} = 0.97 already nine seconds). This sluggish behaviour can lead to problems with heat removal in this time, especially during transients with a cooling failure due to failure of the coolant pumps. Cooling by natural convection in conjunction with a high coolant inventory, which serves as a heat sink, has a favourable effect here (NEA 2006).
- Special attention must be paid to temperature reactivity coefficients if a positive reactivity input occurs with a decrease in temperature in the core upon shutting down the neutron source. This significantly extends the time until a drop in reactor power. A positive reactivity input also results when the reactor is switched off from the hot to the cold state during normal operation (NEA 2002). A precise knowledge of all possible positive reactivity inputs into the reactor core is, therefore, an essential prerequisite for the safe operation of an accelerator-driven system without control rods (NEA 2006). This requirement is of particular importance for accelerator-driven reactors that use fuel with an inert matrix. A high positive reactivity addition, which could make such a reactor core prompt critical, is not compensated for by a fail-safe, temperature-induced negative reaction of the uranium-238 (which is not present in fuel with an internal matrix). This could thus result in a power excursion if no internal control and shutdown system is provided for in the system.
- Beam interruptions and variations in the strength of the neutron source are a problem specific to accelerator-driven systems that must be taken into account, especially in the start-up phase of the reactor in systems without control rods, as the neutron flux density of the external neutron source can change rapidly (NEA 2002). With frequent beam interruptions, this can result in excessive thermal stresses in the fuel or in regions above the reactor and in the heat exchangers. The high stability and reliability of the spallation neutron source are essential for safe operation (NEA 2006).

For unprotected transients where the accelerator beam shutdown does not work, the analyses showed the following:

- For the safety-critical case that positive reactivity is introduced into the core during the transient, the sluggish behaviour of the system opens a time window to switch off the neutron source. Accordingly, the likelihood of such transients are assumed to be only very low (NEA 2006). However, they cannot be ruled out.
- Accelerator-driven subcritical systems respond less favourably to negative reactivity input than critical systems. If the power regulation by the accelerator fails, all reactivity control mechanisms, such as control rods or feedback effects, are affected by negative reactivity coefficients. Compared to critical systems, very high negative reactivity additions are necessary in this case for the power of the system to decrease accordingly (NEA 2006).

In summary, depending on the subcriticality, one can distinguish between the two safety-relevant operating modes of accelerator-driven systems: one in the neutron source-dominated range and one in the feedback-dominated range (NEA 2002). In a strongly subcritical system, the neutron flux density and thermal-hydraulics of the system are largely decoupled, and the system reacts very sluggishly. The advantage for adequately subcritical systems, therefore, is that the power control is, in principle, not dependent on delayed neutrons, reactivity feedback or control rods, but is rather determined solely by the external neutron source (source-dominated). However, should the accelerator shutdown fail, it is precisely this sluggishness of the system and the decoupling described above that become a disadvantage, as a result of which the temperature rise in the system can be very high compared to critical systems, with power spikes also possible (NEA 2002; 2006). In less subcritical systems, the coupling of neutron flux density and thermal-hydraulics becomes relevant (feedback-dominated region), and the dynamics more complex. The degree of subcriticality thus becomes an important parameter that determines both the dynamic behaviour and the safety margins (NEA 2006). Due to their dynamic properties, subcritical systems require the continuous monitoring of the subcriticality value if no other control options (e.g. control rods) are provided.

Due to their radioactive inventory, the same safety standards must be applied for accelerator-driven systems as for current power reactors or other technology lines. A comprehensive description cannot be provided here due to the insufficient level of detail in existing concept studies and the lack of operating experience. The following are further safety-relevant properties of accelerator-driven systems identified to date:

- Since the spallation neutron source necessarily penetrates the reactor vessel and the reactor building, there is a penetration of the safety barriers that makes a potential release path possible. This is admittedly comparable to similar penetrations through coolant pipes in existing reactors. However, the accelerator beam with a power of several megawatts may pose a risk to the surrounding structural materials; its power is so high that a deflection of the beam from the target to other structures, such as the beam tube itself or internals behind it, could destroy them if the beam is not interrupted in time (NEA 2002).
- Since the beam tube is under vacuum, reactivity could potentially be introduced into the reactor core if the volume of the beam tube were flooded with coolant and neutron losses were reduced there (NEA 2002).
- As far as ensuring the removal of residual heat after a shutdown of the reactor and the resulting basic safety requirements for the reactor system are concerned, there is no significant difference to a thermal reactor of the same power. The residual heat removal system must typically be able to transport about 0.1-1% of the nominal power to an ultimate heat sink. In lead-bismuth cooled systems, natural convection can be utilised (NEA 2002).

- Spallation neutron sources also produce neutrons with very high energies far above 6 MeV. To prevent the structural materials in the reactor from being exposed to excessive material loads by these high-energy neutrons, a series of buffers are located directly around the spallation target to slow down these neutrons. If these buffers were to be removed, high-energy neutrons would penetrate the core. During the fission processes that would then occur in the core, substantially more neutrons would be released in the fission process due to the high energy of the incoming neutrons, which would significantly increase the power of the reactor (NEA 2002).
- The choice of fuel also has an influence on the safety features. When uranium-free fuels are used, phenomena such as altered melting temperatures, the formation of eutectics, redistribution of actinides within the fuel during irradiation, irradiation damage to the matrix, cladding corrosion, higher fission gas release and gas pressure due to helium formation and changes in residual heat removal must be taken into account (IAEA 2015). Increased helium formation, in particular, can lead to fuel failure and is a potential source of transients caused by gas bubbles in the coolant or its density reduction.

When using ceramic CECER fuels, which have low thermal conductivity, the safety margin until fuel failure is only very small (100-200 K) due to the very high temperatures of up to 1800 K in the fuel. Transients with an increase in temperature in the fuel have a particularly unfavourable effect here. Due to their high thermal conductivity, CEMET fuels have lower temperatures in the fuel (around 1000 K), resulting in much higher safety margins until the fuel melts (Delage et al. 2011).

The melting temperature of the cladding materials in use today (1370 °C) is below the boiling point of the coolant, so that the fuel cladding can melt and be transported out of the core with the coolant in the event of heat removal faults without a reactor shutdown. The result of this is a positive reactivity input due to the absence of the neutron-absorbing effect of the cladding material (Renn 2014).

Conclusion: Safety

Compared to other reactors, an essential feature of accelerator-driven systems is that they are operated in a subcritical condition. This feature makes an accelerator-driven system more robust than critical fast reactors in terms of the reduction of important safety parameters (reactivity coefficients, delayed neutrons), even when minor actinides are used in the fuel.

The decisive advantage of ADS is that the safety margin can be selected by way of the subcriticality. A highly subcritical design of the reactor core increases the safety margin against reactivity accidents, but it simultaneously reduces the energy amplification of the system. This can be compensated for by increasing the power of the spallation neutron source, though this entails correspondingly high costs and requires correspondingly higher accelerator power.

With ADS, not only must reactivity accidents and accidents with a failure of the core cooling be taken into account, but also new scenarios and boundary conditions for incidents and accidents. For example, due to the interaction between the required high-power accelerator and the subcritical reactor core, power excursions based on a sudden power change of the accelerator must be taken into consideration. In addition, due to the subcritical arrangement of the actual reactor core, there is a pronounced drop in power density, which starts from the spallation neutron source and leads to the edge of the reactor core. Finally, the accelerator current must be fed from outside the reactor into the reactor core. It thus penetrates the main barriers that serve to safely contain the radioactive inventory, i.e. the primary cooling circuit and the reactor building.

In terms of safety, accelerator-driven reactors boast significant advantages over LWR, although scenarios for potential serious accidents cannot currently be ruled out for ADS either.

4.7.6 Fuel supply and waste disposal

In terms of supply, ADS do not differ from other technology lines in the use of uranium or MOX fuels on the one hand or thorium on the other.

The special property of ADS is that very high proportions of minor actinides can be used in the fuel, including uranium-free fuels with a minor actinide content of 40% and a plutonium content of 60%. Thorium can also be used as fuel without the need for additional fissile material to be included in the fresh fuel at the start of operation. In terms of fuel selection and the use of fissile materials, an ADS represents the most flexible technology line.

Similarly to MOX fuels, fuel elements with high proportions of transuranic elements have a higher heat output than LWR fuels (Frieß 2017), which can have an impact on final disposal.

Since ADS are particularly suitable for the transmutation of radioactive waste, the actinide inventory of existing radioactive waste could be reduced when the system is used for transmutation, thereby reducing the radiotoxicity of existing waste. For more on the possibilities of ADS, see (ISR 2021; Oeko-Institut e.V. 2023). With uranium-free fuels, it is conceivable to operate an ADS with actinides only, thereby achieving a high transmutation efficiency. (Frieß 2017) presents calculations on the transmutation properties along with the finding that a high transmutation efficiency can only be achieved by a fresh supply of fissile material in the fuel, since the neutron multiplication factor drops sharply after just one transmutation cycle, and thus the transmutation efficiency, as well. In principle, a spallation neutron source could also be operated without the neutron-boosting reactor section as a pure neutron source for the irradiation of long-lived fission products, for example, with corresponding effort and costs. So far, however, only a few isotopes are suitable for such irradiation, which is due both to physical properties and technical problems in the separation of isotopes and the production of targets (ISR 2021; Oeko-Institut e.V. 2023).

The high-energy protons and neutrons in the vicinity of the target lead to significant activation. The activation of the coolant of the spallation target and of structural materials of the spallation neutron source produces additional heat-generating waste, albeit on a small scale.

The special features of the fuel supply and waste disposal of lead-cooled fast reactors also apply, see Chapter 4.2 and Chapter 5.3.

Conclusion: Fuel supply and waste disposal

As many details of ADS concepts and the associated fuel cycles still remain undefined at this stage of the development process, many questions concerning the analysis of the waste inventories ultimately to be disposed of and their composition remain unanswered.

Of all technology lines, ADS are best suited for use in transmutation scenarios due to the possibility of a high minor actinide content in the fuel.

In terms of supply, ADS have advantages over LWR thanks to their high flexibility, especially due to the possible high proportion of transuranic elements. In terms of disposal, ADS fuels have disadvantages over LWR due to their high heat output.

4.7.7 Proliferation risks

In principle, the same considerations apply to ADS operation as to lead-cooled fast reactors, see Chapter 4.2 and Chapter 5.3. Depending on the fuel used, the considerations from Chapter 0, for example, come into play when using MOX fuels or, in the case of thorium, from Chapter 5.5.4. Differences arise when uranium-free fuels are used. With uranium-free fuels, no new fissile material is produced during irradiation in the reactor due to the absence of uranium in the fuel. This means that the isotopic composition of the transuranic elements contained in the fuel changes more than is the case with MOX fuels (DSF 2009). The proliferation relevance of the transuranic elements and especially the plutonium they contain, therefore, decreases over time to a much greater extent than is the case with fast reactors and the use of MOX fuels. However, when transuranic elements are used, corresponding reprocessing technologies are used, which come with corresponding proliferation risks.

Like any strong neutron source, spallation neutron sources with power levels in the megawatt range must also be monitored, as they can be used to breed fissile material. Due to the very hard neutron spectrum, weapons-grade plutonium would be generated in a breeder blanket (Englert 2010; Englert et al. 2006).

Conclusion: Proliferation risks

The proliferation resistance of ADS depends to a high degree on the concept under consideration and the associated fuel cycle. Reprocessing increases potential proliferation risks and requires greater effort for fissile material control and safeguards measures. ADS can do without the use of separated fissile material to produce the fuel for starting the reactor (see Chapter 4.7.1). However, the proliferation risks associated with using fissile material in ADS represent neither an advantage nor a clear disadvantage compared to technologies currently used in the LWR fuel cycle, such as uranium enrichment or plutonium reprocessing.

4.7.8 Costs

A great deal of research and development work is still required for the development of a commercial system. The

> *“operational stability and operability of accelerator-driven, subcritical systems [is] still completely uncharted territory.” (Renn 2014)*

The development costs are correspondingly high.

In a hybrid system consisting of an accelerator and a subcritical reactor, it is not only the level of the accelerator current, but also, and above all, the reliability of the accelerator that has a decisive influence on the safety characteristics of the system and on the capacity utilisation of an industrially operated plant. However, the availability of an accelerator has been (NEA 2000b) and still remains one of the technical challenges faced by future development (Biarrotte et al. 2015).

In any case, an ADS must have similarly high utilisation rates for commercial operation as with today’s LWR. This poses a major challenge for the development of accelerators, as current spallation neutron sources used in research, such as the SNS in Oak Ridge, have a time between two beam interruptions of a few hours at best. The MYRRHA project (see also Chapter 5.10) provides for beam interruptions of up to one minute well over 100 times per day, with possible beam interruption times ranging from one minute to over one hour occuring 10 times per day (Biarrotte et al. 2015).

Additional costs are incurred for the construction of the accelerator and the spallation neutron source and their operation, and also part of the electricity generated is used to operate the facility. An ADS system is therefore more expensive than a comparable LFR.

A special feature of ADS is the possibility of controling the reactor output by modulating the beam intensity and thus the source strength of the spallation neutron source without having to comply with the criticality conditions of critical reactors. This could possibly make an ADS system particularly compatible with other fluctuating energy sources, such as renewable energies, if designed accordingly.

Conclusion: Costs

Due to the possible power regulation without the criticality condition, ADS systems offer an advantage over LWR and the other technology lines when it comes to integration with other fluctuating energy sources.

No reliable statements can be made at present with regard to required investments, the time needed for construction, operating costs, service life and capacity utilisation. However, additional costs are incurred for the construction of the accelerator and the spallation neutron source and their operation, and also part of the electricity generated is used to operate the facility. An ADS is therefore more expensive than a comparable LFR.

The risks for investors are high, there is no comparable experience available to date.

5 Reactor concepts

This chapter discusses the various reactor concepts selected in Chapter **2.3** and examined in more detail as part of a technology line. This makes it necessary to first present the plant concept and its essential properties. Afterwards, a status report on the evaluation criteria of safety, supply and disposal aspects and proliferation risks is collected, allowing the reactor concept to be evaluated on the basis of these criteria. The focus is on specific essential properties of the reactor concept in contrast to the general properties of the associated technology line.

This is followed by a presentation of the technological development status of the specific reactor concept. Finally, with a view to the future implementation of the reactor concepts under consideration, the main analysis focusses on how far the project has come to date and how it will progress in the future, and to what extent statements can be made about the expected costs for the specific reactor concept.

5.1 SFR: BN-800

The BN-800 is a Sodium-cooled Fast Reactor (SFR) with a gross electrical output of 880 MW and a thermal output of 2100 MW. It builds on 60 years of SFR development in Russia. The reactor core consists of a mixed core, which allows the operation of MOX fuel and enriched uranium-dioxide fuel. The main development goal of the BN-800 is to demonstrate a closed fuel cycle consisting of reprocessing, MOX fuel element production and the use of MOX fuel in SFR (Pakhomov 2018)

5.1.1 Description of the plant concept

The BN-800 is a SFR featuring a “pool” design, see also the illustration in **Figure 4-1**. The following description is based on (Pakhomov 2018) and (Schulenberg 2020), unless otherwise stated.

The reactor core has a diameter of 2.56 m and an active height of 0.88 m. The power density in the core is 450 MW/m^3. The core is made up of 565 hexagonal fuel elements. A fuel element has a distance of 96 mm between two parallel sides, with a gap of 4 mm between the fuel elements.

A fuel element is made up of 127 fuel rods with an outer diameter of 6.9 mm. The maximum linear thermal output in the fuel rod is 48.5 kW/m. The fuel is uranium-plutonium mixed oxide fuel (MOX). The core consists of 15.88 t of MOX fuel with a plutonium content in the fresh fuel of 20.5%. The target burn-up is an average burn-up of 67 MWd/kg heavy metal, corresponding to a maximum burn-up of 9.9% of the initial heavy metal. A fuel element remains in the reactor core for 465 days, and a loading cycle is scheduled to take place every 155 days.

The cladding material used is ChS-68 steel, which already finds use in the BN-600. This cladding material covers the cladding strain of around 90 dpa associated with an average burn-up of 66 MWd/kg (Kuznetsov et al. 2018).

The reactor core and the complete primary cooling circuit are housed in a reactor vessel with an internal diameter of 12.9 m, a height of 14 m and a wall thickness of 30 cm. The reactor vessel is made of Cr18Ni9 stainless steel and has an empty weight of 216 t. The primary cooling circuit contains 1000 t of sodium. The heat generated is transferred to a secondary sodium cooling circuit via three heat exchangers located in the reactor vessel. During power operation, the sodium on the primary side is actively circulated by three submersible pumps that are also housed in the reactor

vessel. The residual heat can also be transported from the reactor core to the primary side heat exchangers by natural circulation.

The sodium on the primary side flows through the reactor core from below, at an inlet temperature of 354 °C, and enters the shell side of the heat exchangers below the reactor vessel head, is then cooled in the heat exchangers and exits the lower area of the heat exchangers. From there, it is pumped back into the reactor core from below via the submersible pumps with a coolant mass flow of 8600 kg/s. The core outlet temperature is 547 °C. Between the surface of the primary-side sodium and the reactor vessel head there is a cover gas layer with a volume of 110 m^3 at a pressure of 0.054 MPa.

In the secondary-side cooling circuit, sodium is pumped through the tubes of the primary-side heat exchangers in three loops at a temperature of 309 °C and then flows at a temperature of 505 °C to the external secondary-side steam generators, in which the heat is transferred to a water-steam circuit. Active coolant circulation with a coolant throughput of 2780 kg/s is maintained by an intermediate cooling pump in each circuit. The intermediate cooling circuits also contain a collection and volume equalisation tank for the secondary-side sodium, which has a total mass of 980 t.

The secondary cooling circuit is also connected to an additional section for residual heat removal, in which the secondary-side sodium can be fed to two air-cooled secondary-side heat exchangers via two additional electromagnetic pumps.

In the water-steam circuit, feed water at a temperature of 210 °C is pumped into the secondary steam generators and fresh steam is generated at a temperature of 490 °C. At a pressure of 13.7 MPa, a steam flow of 876 kg/s is maintained in this tertiary cooling circuit. The fresh steam generated is collected and fed to a turbine connected to all three loops. The feed water is pumped back to the three circuits via a common feed water pump.

The steam generators are each made up of ten sectors in which the steam is evaporated and then superheated. In the event of a leak, each section can be separately isolated (IAEA 2021c).

The reactor vessel is completely enclosed by a safety vessel (guard vessel). The safety vessel is in turn protected from external influences by a concrete structure. Located above the reactor vessel are the facilities for loading and unloading fuel elements, which is undertaken through a lock on the closed reactor vessel.

The control and shutdown elements are inserted into the core from above through the reactor vessel head. The performance of the BN-800 can be varied between 17 and 100%.

5.1.2 Safety concept and safety features

A complete presentation and discussion of the safety concept of the BN-800, and the potential for improvements that have already been identified, exceeds the scope of this study; there are extensive analyses for the BN-800, including from the planning for the successor concept to the BN-1200. For a current overview, please refer to (IAEA 2021c).

The general safety advantages of Sodium-cooled Fast Reactors apply to the BN-800 (see Chapter **4.1**). The large coolant inventory in the reactor vessel offers a high heat capacity and, as a result, long grace periods until permissible fuel and coolant temperatures are exceeded in the event of a failure to remove heat from the reactor vessel.

Since the primary cooling circuit is not under high pressure, leaks do not lead to a rapid loss of coolant. Furthermore, avoiding large coolant lines in the design, outside the reactor vessel, as well as the additional safety vessel (guard vessel), prevents a major-loss-of-coolant accident and a subsequent reduction in the coolant level near the fuel.

The safety concept and instrumentation of the BN-800, as well as the maintenance concept, are based on its predecessor project, the BN-600. In particular, further changes to the safety concept were required in order to satisfy the updated Russian licensing requirements in the wake of the Chernobyl accident (IAEA 2021c).

According to (IAEA 2021c), the safety system of the BN-800 includes, among other things, an emergency system for residual heat removal, a reactor protection system, a protection system against loss of coolant, a protection system against overpressure in the reactor and in the secondary cooling circuit, a heat removal system for the fuel elements during loading and unloading processes and in fuel element storage.

According to (Pakhomov 2018), the BN-800 differs from its predecessor in the following points:

- It possesses a passive emergency shutdown system.
- A special cavity above the core was introduced to reduce the effect of the positive sodium void coefficient.
- A core catcher was added to the lower part of the reactor vessel to catch and retard a core meltdown under severe accident conditions.
- A system for removing the decay heat, which transfers the heat to the outside via air heat exchangers and is connected to the secondary circuit, was introduced.

According to (IAEA 2021c) and in comparison to the BN-600,

- the base structure of the reactor vessel and the core support structure were redesigned for improved seismic stability,
- the wall thickness of the reactor vessel and the containment shell was increased from 20 to 30 cm,
- passively triggered shutdown rods were added (which automatically descend into the core if the sodium flow through the reactor core is reduced to less than 50% of the nominal flow rate),
- there is improved separation of caesium from the primary-side sodium,
- as well as improved fire and explosion protection in proximity to the steam generators.

Reactivity control is ensured by control rods, for which, according to (IAEA 2021c), two independent shutdown systems are employed. The BN-800 does not feature an additional, diversely redundant shutdown system.

The BN-800 introduced the concept of an upper sodium plenum (Ohshima and Kubo 2016), see the discussion in Chapter **4.1.5**.

Within the BN-800, the reactor vessel with the integrated core catcher, a gas-tight confinement, an additional enclosure of pipes for the primary auxiliary systems and the ventilation system for the primary sodium system, are used to contain the radioactivity, according to (IAEA 2021c).

The BN-800 is designed to withstand earthquakes of magnitude 7 on the MSK-64 scale (Pakhomov 2018).

The BN-800 is designed to withstand the crash of an aircraft with a mass of 5.7 t at an impact speed of 100 m/s. It is also designed to withstand an explosion pressure wave of 10 kPa for a duration of 1 s (IAEA 2021c).

The core damage frequency of the BN-800 is given in (IAEA 2021c) as approx. $2x10^{-6}$ per year, which is approx. a factor of five better than the BN-600 but still approx. a factor of four greater than the planned BN-1200.

Conclusions on safety

The general safety properties of SFR discussed at the technology-line level apply to the BN-800, in particular the fact that the primary cooling circuit of the BN-800 is a low-pressure system, which limits the effects of leaks, the property of the chemical compatibility of coolant and fuel, and the chemically reactive property of the coolant sodium, when in contact with atmospheric oxygen or water. The opaque, exothermic sodium also makes maintaining the reactor and repairing damage complex. There are no significant differences with regard to the technology-specific assessment.

Reactivity control in the SFR is generally subject to higher requirements than for the LWR. The BN-800 does have two redundant shutdown systems, but no diversely redundant shutdown system, as planned for future SNR as part of GIF, see Chapter **4.1.5**. In terms of reactivity control, this is a disadvantage compared to LWR.

The safety concept of the BN-800 provides for redundant and diverse systems with regard to the required heat removal. There are also options for unlimited heat removal through natural circulation and heat release to the atmosphere. This represents an advantage over current LWR.

With regard to the range of events that need to be taken into consideration for the BN-800, fundamental advantages arise, such as at the technology-line level, from the fact that the cooling circuits are not under high pressure. This is offset by the disadvantage of possible sodium fires in the event of leaks. Overall, this does not result in any relevant advantages or disadvantages for the BN-800 compared to LWR.

Retention of possible releases by means of containment is planned in a similar way to current LWR. This does not result in any relevant advantage or disadvantage compared to LWR.

The core damage frequency reported for the BN-800 is of the order of magnitude of LWR already in operation today. Overall, the BN-800 does not offer any safety advantage over current LWR. In view of the special requirements for reactivity control, it is more likely to be a disadvantage.

5.1.3 Supply and disposal aspects

The Russian strategy for using SFR, especially the BN-1200 as the successor to the BN-800, is based on a closed fuel cycle in conjunction with LWR plants. Spent fuel from LWR plants will be reprocessed and the plutonium obtained in this way will be used in SFR plants in the form of MOX fuel. Spent MOX fuel from SFR plants will also be reprocessed and the plutonium obtained in this way will be used, to some extent, in LWR plants. The production of plutonium with a high fraction of thermally fissile isotopes (plutonium-239 and -241) in SFR plants is necessary in order to maintain a suitable isotopic composition of the plutonium in the fuel cycle for use in LWR (WNA 2021c).

The BN-800 has a breeding rate of 1.04 (Pakhomov 2018). This means that only slightly more fissile material is produced in the BN-800 than was used. Therefore, it is not possible to forego uranium enrichment to supply LWR plants.

LWR are also planned as the main production source of fresh plutonium in Russia for a future SFR with a breeding rate of at least 1.2. The plutonium for the operation of one SFR will be produced in two LWR. The plutonium produced in this SFR is in turn used to produce MOX fuel for LWR. In total, it is expected that 30% of the fuel required to operate the two LWR will consist of MOX fuel, 70% will be classic uranium fuel or uranium fuel from re-enriched uranium stocks from reprocessing (WNA 2021c).

Conclusions on supply and disposal

In principle, the planned coupling of future SFR with classic LWR does not mean that uranium enrichment technology will be dispensed with. In return, this system requires additional facilities for reprocessing, as well as fuel element production for MOX fuels.

Overall, however, the need for fresh uranium to supply the plants could be reduced compared to current LWR. This would represent an advantage over current LWR in terms of supply.

Otherwise, there are no differences when compared to the assessment at the technology-line level.

5.1.4 Proliferation risks

The fuel elements of the BN-800 contain significantly more fissile material (typically about 20% plutonium) than light water reactors. This means that the production and transport of fresh fuels already entails higher proliferation risks than the production and transport of fresh uranium fuels.

The BN-800 is designed for a breeding rate of 1.04. For future SFR in Russia, however, a breeding rate of at least 1.2 is being aimed for. Plutonium bred in this way is particularly suitable for building nuclear weapons (weapons-grade), i.e. it has a particularly high fraction of the isotope plutonium-239. It is planned that the plutonium produced in this way will be used in the BN-800 and its successor plants, as it is intended to establish a closed fuel cycle in which the plutonium produced in the SFR is separated and used to manufacture new fuel elements.

(Oeko-Institut e.V. 2017) points out that fast reactors can in principle also be used to remove plutonium. For this purpose, the plutonium is used in the form of MOX fuel elements in the BN-800 and part of the plutonium is split, or the isotope vector of the plutonium is altered so that it is less suitable for nuclear weapons. The burn-mode is essentially characterised by the fact that the use of breeding elements (blankets) in the reactor shell or in other positions is dispensed with. However, (Kütt et al. 2014) demonstrates that simple reconfiguration of the reactor core to breeding mode, by using breeding elements within the reactor shell, would permit up to 162 kg of weapons-grade plutonium to be bred per year. The radiation barrier in such blankets would also be significantly lower than that of fuel elements for power generation.

Conclusion on proliferation risks

In principle, breeder reactors are well suited to producing large quantities of high-quality plutonium for nuclear weapons applications (see Chapter **2.6.3**). High-quality plutonium is also produced in relevant quantities in the fuel of the BN-800 and its planned successor plants during the breeding phase. This basically gives the plant operator the opportunity to produce material suitable for nuclear weapons by using an alternative loading and unloading strategy.

The infrastructure of the BN-800 and its successor plants, together with the associated reprocessing technology, may also be used for a military programme.

Together with the planned separation of the plutonium for reuse in MOX fuels, this makes the fuel cycle of the BN-800 significantly more relevant to proliferation than LWR with an open fuel cycle.

5.1.5 Technological development status

The objectives of the BN-800 are to demonstrate operation with 100% MOX fuel in the reactor core, to maintain and further develop the know-how on the design, construction and operation of SFR, to advance development of fuels and structural materials for SFR and to test new systems, structures and components, and computer models for SFR (IAEA 2021c; Kostin and Vasil’ev 2007). In addition, Russia is pursuing an extensive research and development programme to further develop the SFR (Kuzina 2021).

Conclusions on technological development status

The operation of the BN-800, the production of uranium or plutonium-containing MOX fuels and the reprocessing of spent LWR fuel have been demonstrated on a large scale and can therefore be classified as being in “deployment”.

For commercialisation and diffusion, further developments in the areas of safety and economic efficiency are required.

5.1.6 Implementation

The BN-800 is the sixth reactor in the Russian BN-development series with the BN-600 being its predecessor and will be followed by the planned MBIR research reactor. The long-term goal of this development is to demonstrate the closed fuel cycle. Currently, the Russian development series can be described by the construction and operation of the following reactors:

- BR-5/BR-10 (1959-2002, Obninsk),
- BOR-60 (1969, Dimitrovgrad),
- BN-350 (1972–1999, Aktau, Kazakhstan),
- BN-600 (1980, Zarechny, Beloyarsk-3),
- BN-800 (2014, Zarechney, Beloyarsk-4),
- MBIR (construction started in 2016, Dimitrovgrad)

The BN-800 can be classified as a demonstration reactor. It will be replaced by a real commercial reactor concept, the BN-1200. A reactor with an output of 1800 MW, the BN-1800, was being pursued as a successor until as recently as 2021 (IAEA 2012b,, 232-237). Construction of a BN-1200 reactor is currently not expected before the mid-2030s. In addition to producing electricity, the BN-800 will also be used to supply district heating (IAEA 2021c).

5.1.6.1 Project timeline

The basic design of the BN-800 was undertaken between 1983 and 1993, using knowledge gained from operating the BN-600, and was already based on new Russian regulatory requirements in response to the Chernobyl accident (IAEA 2021c).

The first design was completed in 1985. After the Chernobyl accident, however, new requirements mandating the need for a negative reactivity coefficient of the reactor power and the coolant temperature were anchored in the Russian regulations in 1989. A corresponding design for the reactor core of the BN-800 was developed by 1992. This was followed by a phase of around five years to demonstrate compliance with the regulatory requirements, both experimentally and theoretically. The government confirmed the intention to build a BN-800 at the South Urals and Beloyarsk sites in 1992, and a building permit for the Beloyarsk site was finally issued in 1998 (Krivitski 2001).

(IAEA 2006a, p. 13) gives the start of construction as 2002, with first criticality then being achieved in 2012. (IAEA 2023g) gives the official start of construction as 2006. It achieved first criticality in 2014, see Table 5-1, and 100% performance for the first time in 2016 (Ohshima and Kubo 2016).

Between 2015 and 2021, the reactor generated 27 TWh of electrical energy with an average operating availability of 65.9%. (IAEA 2023g)

Table 5-1: Key stages in the commissioning of the BN-800

Date	Event
27/06/2014	First criticality
30/07/2015	End of the first critical phase
10/11/2015	Start of power generation
10/12/2015	First connection of the generators to the power grid
10/02/2016	First full-load operation
20/02/2016	Start of the pilot operation phase
09/2016	Completion of the pilot phase and preparation for commissioning of the power plant block
31/10/2016	Operation of the power plant block

Source: Data according to (Pakhomov 2018)

Conclusions on project progress

As with many SFR projects, the development of the SFR in Russia is massively behind schedule compared to the original schedule. The newest plant in operation, the BN-800, was designed in the 1980s and 1990s, but its construction did not officially begin until 2006. The planned successor plant, the BN-1200 (formerly BN-1800), also had ambitious schedules at first, but no specific date for the start of construction has yet been announced. From today's perspective, the future of the BN-800 can therefore be seen as largely open.

5.1.6.2 Costs

In order to reduce the specific construction costs of the BN-800 over the BN-600, the technical and economic performance indicators of the power-generating unit were improved by using a single turbogenerator instead of three, as was the case in the BN-600, and by introducing some other new design solutions (Kostin and Vasil’ev 2007)

In (IAEA 2021c, Fig. 5), the relative specific construction costs of the Russian BN series are compared. According to this, the capital costs of the BN-800 would still be about 65% of the BN-350, while the BN-600 was still about 80% of the BN-350. For the BN-1200, a value of about 50% of the specific construction costs of the BN-350 is expected. This means that the BN-800 is still about 40% higher than the specific construction costs of current Russian LWR (VVER-TOI), while the specific construction costs of the BN-1200 would then only be about 10% higher.

The theoretical, specific construction costs (overnight construction costs) for a future BN-800 double-block plant (which is not planned in Russia), a future BN-1200 plant and a VVER-TOI-LWR are given in (IAEA 2021c, Tab. 2). For the BN-800, these would amount to 3800 USD/kW of electrical power, for the BN-1200 3400 USD/kW and for the VVER-TOI 3100 USD/kW.

In (IAEA 2021c, TAb. 3), the annual operating costs of these plants are estimated at 134 USD/kW for the BN-800, 122 USD/kW for the BN-1200 and 102 USD/kW for the VVER-TOI.

(Pakhomov 2018) gives the planned lifetime of the BN-800 as 40 years.

Assuming a construction period of 6 years, a service life of 60 years and a load factor of 85% for the BN-800, (IAEA 2021c, Fig. 8) determines the LCOE costs at 59 mills/kWh for the BN-800, 46 mills/kWh for the BN-1200 and approx. 44 mills/kWh for the VVER-TOI, at uranium prices below 100 USD/kg.

Conclusions on costs

The investment costs of the BN-800 are higher than those of the LWR, and higher investment costs are also expected for its successor, the BN-1200. At the same time, the availability of the SFR achieved to date, including with the BN-800, is lower than with current LWR. The operating costs are also higher than the LWR.

Even under favourable assumptions that have not yet been realised (short construction time, long service life, high availability over the entire service life), the levelised cost of electricity of the BN-800 would still be around 35% higher than LWR.

Overall, the BN-800 is therefore at a disadvantage compared to LWR in terms of economic efficiency.

5.2 SFR: Travelling Wave Reactor

The US company TerraPower publishes basic statements about its reactor concept on its homepage (TerraPower, LLC 2022a) and in fact sheets (TerraPower, LLC 2020; 2022c).

The descriptions in this subsection are based on (Hejzlar 2021), unless other sources are explicitly cited, as this is the most detailed current source. Additional detailed descriptions of the reactor concept of the Travelling Wave Reactor (TWR) can be found in (Gilleland et al. 2016; Hejzlar et al. 2013). The description for the prototype of the Travelling Wave Reactor (TWR-P) stored on the IAEA ARIS database, however, is not up to date, as it still talks about a planned construction of the TWR-P between 2018 and 2023 (ARIS n.d.). To date, only a few independent studies on the TWR exist, see e.g. (IEER 2013).

The central design constraints for the TWR arise from TerraPower's stated goal of avoiding proliferation risks and efficiently utilising natural uranium reserves. At the same time, it is intended to minimise the amount of radioactive waste generated and increase reactor safety (TerraPower, LLC 2022c).

Based on these objectives, the TWR should be able to use natural uranium or depleted uranium as fuel. At the same time, reprocessing of the spent fuel should be avoided. From this, the developers derive a reactor concept that uses metallic fuels to allow particularly high burn-ups and thus in-situ breeding and burning of plutonium. To enable a high breeding rate, the reactor has a fast neutron spectrum. A liquid metal, in the case of the TWR, sodium, is therefore intended as the coolant.

The developers say that a key difference to previous approaches to sodium-cooled fast breeder reactors such as the BN-800 (see Chapter 5.1) is that there will be no reprocessing of spent fuel elements (TerraPower, LLC 2022a).

The developers expect significant cost reductions through simplified fuel supply and disposal compared to today’s LWR, increased safety, a significant reduction in the amount of highly radioactive waste generated, simplified handling of highly radioactive waste and a high degree of proliferation resistance (TerraPower, LLC 2020).

5.2.1 Description of the plant concept

The TWR is a sodium-cooled fast breeder reactor. The commercial system is to have an electrical output of 1200 MW (Hejzlar 2021), while the TWR-P prototype reactor is specified in (Hejzlar et al. 2013) as having a thermal output of 1475 MW and an electrical output of 600 MW.

The fuel concept of the TWR is based on the so-called travelling wave concept. This process entails the production of additional fissile material during operation in the reactor core by converting uranium from fresh fuel, which itself would not contain enough fissile material to operate the reactor as a critical reactor, into plutonium (breeding process). This is then consumed again through nuclear fission without the need to reprocess spent fuel.

The term travelling wave reactor is based on a concept in which the areas of production and consumption of plutonium in the reactor core shift spatially over time. This basic idea was also investigated for the TWR in the early development phase, but was not pursued further for practical reasons. In contrast, the TWR is currently pursuing an approach in which the areas of production and consumption of plutonium in the reactor core remain (largely) spatially constant and instead the fuel is used in the course of its burn-up by reloading at different positions in the reactor core.

A section of the reactor core is therefore regularly unloaded, fresh fuel is loaded and the fuel elements in the reactor core are moved to positions in the reactor core appropriate to their previous burn-up (Hejzlar et al. 2013).

TerraPower considered various fuel and coolant variants for the TWR concept. A metallic fuel was selected as because it enables a high uranium density in the reactor core and, at the same time, maintains a very hard neutron spectrum, which is a prerequisite for an effective breeding process.

According to (Hejzlar 2021), the fresh fuel includes uranium with an enrichment of up to 10%, low-enriched uranium of less than 5%, natural uranium, depleted uranium or uranium from spent fuel from light water reactors. It is intended that enriched uranium will only be required for the initial reactor core of a new TWR reactor, and further reloads should, if possible, only require fuel from depleted uranium or, depending on the source, natural uranium. When describing a specific initial core composition, however, (Hejzlar 2021, p. 649) also states that initial enrichments of up to 13% are also required. The special feature of this fuel concept is that a reactor core consisting exclusively of fresh fuel (with natural or depleted uranium) cannot sustain a chain reaction. In (Hejzlar et al. 2013), the developers also assumed that additional fissile material for further initial cores of new TWR reactors could be produced in a TWR at the same time, but this is no longer included in (Hejzlar 2021).

The current plans envisage a reactor core with 864 hexagonal fuel elements, with an external distance of the parallel sides of 174 mm. For an initial core, 510 fuel elements are planned to be made from enriched uranium and 354 fuel elements from depleted uranium; further loads will then only use natural or depleted uranium. The reactor core has an active height (i.e. the height of the fuel column in the fuel elements) of 2 m and a radius of 3.25 m. A fuel element consists of 217 fuel rods arranged in a tight, triangular grid. The cladding tube should have a wall thickness of 0.4 mm and an outer diameter of 10.2 mm. The fuel rods are wrapped with a metal wire with a thickness of 0.86 mm to keep them apart. The fuel element is enclosed in a fuel element box with a wall thickness of 4 mm (Hejzlar 2021).

The effective uranium density of the metallic U-10%Zr fuel should be 50-60%, i.e. the fuel is designed with a high porosity to permit free volume for the fission gases produced during burning. However, the internal fuel rod pressure must be further limited to enable the high burn-ups required. The fuel rods are therefore degassed in a controlled manner during operation.

The fuel rod cladding tubes and the fuel element casing are to be made of a particularly radiation-resistant advanced ferritic martensitic steel (TP HT9), which is based on a steel that has already been tested in fast experimental reactors (HT9). Other structural materials in the core are made of stainless steel (316H).

The cycle length is given in (Hejzlar 2021, p. 649) as 550 effective full load days. During loading, 30 fresh fuel elements are loaded into the outer ring of the reactor core. From there, they will initially be transferred further towards the centre of the reactor core over the following cycles, while the fraction of fissile plutonium in the fuel increases through breeding. The fuel elements are then transferred to the centre of the reactor core and in subsequent cycles transferred again towards the periphery of the reactor core. Once the target burn-up has been achieved, the fuel elements are stored for a further five years or so in the edge area of the reactor vessel, such that the loading and unloading of fresh or spent fuel takes place after about 10 years (Hejzlar 2021, p. 650).

For the planned equilibrium core, a target fuel burn-up of 28% is required, i.e. 28% of the total uranium used will have been consumed by fission by the time it is unloaded from the reactor (Hejzlar 2021). The radiation-related cladding strain is given as 520 dpa for a planned service life of a fuel element in the reactor (Hejzlar 2021, p. 650).

The maximum cladding temperature during operation is given as 625 °C (Hejzlar 2021). (Gilleland et al. 2016) a value of 360°C is given for the core inlet temperature and 510°C for the core outlet temperature.

Sodium was chosen as the coolant because it does not moderate the neutrons and has good cooling properties, thus enabling a high power density. At the same time, there is already considerable experience worldwide in dealing with sodium as a coolant (see also Chapter **4.1**), so that the developers assume that such a reactor concept can be implemented early (see Chapter 5.2.5).

The heat is transferred via a primary sodium cooling circuit to a secondary sodium cooling circuit. This serves to prevent a discharge of activity, or a sodium-water reaction in the primary coolant, in the event of leaks in the primary cooling circuit. The intermediate cooling circuit transfers the heat to a tertiary water-steam cooling circuit to produce electricity, and the waste heat is transferred to the environment via water-cooled condensers. The primary cooling circuit is operated as a forced circulation system.

A loop design has been planned for the reactor, in which the reactor core and the complete, pressure-free primary circuit including pumps and heat exchangers are housed in the reactor vessel. The reactor vessel is surrounded by a second, independent protective vessel. This design is comparable to other sodium-cooled fast reactors including the BN-800 (Gilleland et al. 2016).

5.2.2 Safety concept and safety features

The developers themselves expect an increase in safety compared to current LWR (TerraPower, LLC 2020).

The developers point to the general safety advantages of sodium-cooled reactors (see Chapter **4.1**). The large coolant inventory in the reactor vessel offers a high heat capacity and, as a result, long grace periods until permissible fuel and coolant temperatures are exceeded in the event of a failure in the heat removal from the reactor vessel (Hejzlar 2021).

Since the primary cooling circuit is not under high pressure, leaks do not lead to a rapid loss of coolant. Furthermore, avoiding large coolant lines in the design, outside the reactor vessel, as well as an additional protective vessel (guard vessel), prevents accidents featuring a major loss of coolant and leading to a reduction in the coolant level near the fuel. The choice of a metallic fuel avoids chemical reactions when the coolant and fuel come into contact (Hejzlar 2021).

Reactivity control is ensured by control rods, which compensate for the change in reactivity during burn-up in particular. During an equilibrium cycle, the reactivity of the reactor core varies over burn-up between 1.006 and 1.021, whereby the reactivity, unlike in LWR, increases over burn-up due to the breeding of fissile material. Independent and redundant shutdown rods are also provided. Both systems are triggered by the reactor protection system in the event of an accident (Hejzlar 2021).

The developers are aiming for an overall negative temperature coefficient for the reactor core. Due to a typically positive sodium temperature coefficient, an overall negative value must be achieved by additional, sufficiently negative temperature coefficients, for which suitable design boundary conditions must be observed (see Chapter **2.6.2**). Weakly negative values are given in (Hejzlar 2021, p. 650) for the Doppler coefficient of the fuel (-0.092 c/K), as well as the axial (-0.08 c/K) and radial (-0.15 c/K) expansion coefficients.[183] The sodium temperature coefficient is slightly negative at the beginning of a cycle, but due to the change in the isotopic composition of the fuel, it assumes a positive value of 0.27 c/K by the end of the cycle, which must be compensated by the other coefficients.

TerraPower is also developing a passive device to reduce reactivity when the temperature in the reactor core rises (Hejzlar 2021).

The fraction of delayed neutrons, which is important for reactor control, decreases during an equilibrium cycle from 0.0068 to 0.0035 at the end of a cycle.

According to (TerraPower, LLC 2020), it should be possible to ensure the safety of the TWR purely passively without the need for intervention by the operating personnel.

As part of the safety concept of the TWR, it is planned to ensure the removal of residual heat via several pathways through which the residual heat is transferred to the environment. An emergency residual heat removal system uses the primary heat exchangers and the steam generators to remove the residual heat to the atmosphere via a high-pressure system. According to (Hejzlar et al. 2013), in the TWR-P, this system possesses sufficient water supplies for seven days of heat removal, the feed pumps are powered by emergency diesel and the steam is released to the atmosphere. However, the system is not designed as a safety system. A Direct Reactor Auxiliary Cooling System (DRACS) removes the heat from the primary heat exchangers directly to the atmosphere via system-internal heat exchangers over four circuits, two of which are sufficient to remove the residual heat (Gilleland et al. 2016). According to (Hejzlar et al. 2013), DRACS is completely passive and does not need a power supply for activation or operation. The residual heat from the sodium is transferred to molten NaK salt via four heat exchangers. This releases the heat to the atmosphere via four NaK-air heat exchangers, with circulation in each case being purely passive. When operating one DRACS, a coolant temperature of 700 °C can be maintained, or 540 °C when operating two loops.

Heat removal in the event of an incident should be purely passive for an unlimited period of time using air cooling operated in natural circulation (Hejzlar 2021).

Studies into the incidents and accidents under consideration will be performed using the SAS 4A-SASSYS1 calculation code. Compliance with safety requirements could thus be demonstrated for various event sequences that are factors in determining design. In one example of a design-basis accidents featuring a failure of the forced coolant circulation (protected loss of flow), the reactor protection system initially leads to shutdown of the reactor. As a result, the coolant and cladding temperatures rise, whereby the cladding temperatures do not exceed the permissible value of 650 °C for design-basis accidents.

[183] The reactivity of a reactor core is a dimensionless parameter. The reactivity coefficients are given here in units of cents (equivalent to one hundredth of a dollar). In this specification, the difference in reactivity is normalised to the fraction of delayed neutrons, i.e. with a change in reactivity of one dollar, a critical reactor goes critical even without the delayed neutrons, i.e. is immediately critical.

In event sequences classified as exceeding design-basis with a failure of both the control rods and the shutdown rods, significant fuel damage is to be avoided due to the reactivity coefficients of the reactor core.

According to (Gilleland et al. 2016), if the shutdown systems fail, the reactor remains in a stable state with low energy release over the long term.

In the case of a transient triggered by the entry of positive reactivity with a complete failure of the control and shutdown rods (Unprotected Transient Over Power, UTOP), the sodium does not boil within the scope of best-estimate analyses. A reactivity entry of 0.14 USD due to the partial extension of a control rod was assumed for the trigger. The maximum power increases to a factor of 2.3 of the nominal value, the cladding tube temperatures reach 900 °C. In the above-mentioned case of a failure of the forced coolant circulation with superimposed shutdown failure by the reactor protection system, the additional passive reactivity control system is required to limit the effects (Hejzlar 2021, p. 651). This passive reactivity control system is intended to work similarly to a gas expansion module already tested at the FFTF (Gilleland et al. 2016).

Even in the event of an accident exceeding design-base featuring fuel melting, an increase in reactivity would be prevented by the dissolution of molten fuel in the coolant (Hejzlar 2021).

Important radionuclides such as iodine and caesium, which are responsible for the external follow-up dose in severe accidents in LWR, are well retained in the metallic fuel and sodium coolant, so that even in severe accidents the release rate of these radionuclides is significantly reduced compared to LWR. Overall, the developers therefore assume that the emergency planning radii required to date could be significantly reduced compared to LWR. In addition to cost savings, the developers also see this as an opportunity for more flexible site selection for TWR reactors (Hejzlar 2021).

In order to achieve the required high burn-ups in the metallic fuel, the developers plan to release gaseous fission products produced in the fuel rods into the coolant in a controlled manner. Appropriate precautions must be taken to ensure that the release of caesium connected with gas release is as low as possible. Details on the structure and degassing of the fuel rods are considered proprietary information by TerraPower and will not be published (Hejzlar et al. 2013). The caesium released into the coolant is to be removed from the coolant by net-like, glassy carbon traps. This is intended to achieve a coolant activity that is comparable to that of today’s LWR (Hejzlar 2021). Furthermore, the inside of the cladding tube will be coated to limit chemical fuel-cladding tube interaction (Gilleland et al. 2016).

The released fission gas is collected in the cover gas located above the sodium coolant and is separated by a gas cleaning system. In particular, the released krypton-85 will be stored in shielded storage containers (Hejzlar 2021).

To safely contain the radioactive inventory, the entire primary circuit is surrounded by containment, which is intended to prevent the release of radioactivity into the environment and to protect the reactor itself against external influences such as wind, plane crashes or flooding (Gilleland et al. 2016).

For measures for protecting against sodium fires, (Gilleland et al. 2016) refers to techniques that have already been tested in other sodium-cooled reactors. This includes enclosure of the pipes carrying sodium on the primary side, as well as containers, in an atmosphere of inert gas. For the systems carrying sodium on the secondary side, containment systems are also mentioned. A

possible steam generator leak between the secondary sodium circuit and the water-steam circuit is controlled by a leak detection system and rapid steam generator shut-off and pressure relief.

The probabilistic core damage frequency determined in (Gilleland et al. 2016) as 10^{-8} per year for a TWR.

Conclusions on safety

The general safety properties of the TWR discussed at the technology-line level apply to the TWR, in particular the fact that the primary cooling circuit of the TWR is a low-pressure system, which limits the effects of leaks, the property of the chemical compatibility of coolant and fuel, and the chemically reactive property of the coolant sodium, when in contact with atmospheric oxygen or water. This does not result in any significant differences to the technology-specific assessment.

The safety-related proof of the cycle-dependent course of integral safety parameters, such as the fraction of delayed neutrons and the various reactivity coefficients, will be essential for the later construction of a TWR (see Chapter 5.2.2). These varied greatly over the burn-up, such that the permissible safety parameters for all cycle times must be determined sufficiently conservatively.

Reactivity control in the SFR is generally subject to higher requirements than for the LWR. For TWR, this is made even more difficult by the extremely strong change in parameters over burn-up and the locally, highly variegated reactivity contributions from the core.

This is also evident in the events classified as exceeding design basis. In order to achieve the safety objective of avoiding sodium boiling, a passive reactivity control system may be required.

Overall, the higher requirements for reactivity control in the TWR represent a disadvantage compared to current LWR.

The safety concept of the TWR provides for redundant and diverse systems with regard to the required heat removal. There are also options for purely passive, unlimited heat removal through natural circulation and heat release to the atmosphere. Implementing this safety concept in this way, offers an advantage over current LWR.

With regard to possible releases of radioactive materials in the event of incidents and accidents, the developers point to the good retention of significant radionuclides (iodine, caesium) in the coolant. At the same time, however, operational degassing of the fuel rods into the coolant is necessary due to the high burn-ups that are being targeted. To what extent this could result in higher normal operational loads, or potentially higher release pathways in the event of incidents and accidents, is currently unclear. Retention of possible releases of radioactive material using containment comparable to current LWR is planned. Overall, this means that there are no relevant advantages or disadvantages of a TWR compared to current LWR.

With regard to the range of events that need to be taken into consideration for the TWR fundamental advantages arise, such as at the technology-line level, from the fact that the cooling circuits are not under high pressure. This is offset by the disadvantage of possible sodium fires in the event of leaks and the need for active monitoring of the steam generator heating pipes for water entering the sodium intermediate cooling circuit. Overall, this does not result in any relevant advantages or disadvantages for a TWR compared to current LWR.

5.2.3 Supply and disposal aspects

The developers expect simplified fuel supply and disposal compared to today’s LWR, a significant reduction in the amount of highly radioactive waste generated, as well as simplified handling of highly radioactive waste (TerraPower, LLC 2020).

According to (TerraPower, LLC 2022a), the use of natural uranium in the TWR is 30 times more efficient than in conventional LWR. At the same time, the amount of spent fuel will be five times lower.

(Hejzlar 2021) gives the required minimum burn-up for a TWR a value of 30%, i.e. at least 30% of the loaded uranium should be consumed by fission, but elsewhere (Hejzlar 2021) states a target burn-up of 28%.

In contrast, in today’s LWR typically around 4-5% of the initial uranium inventory is converted by fission. The higher consumption of natural uranium compared to a TWR is due to the need to increase the fraction of fissile uranium-235 present in natural uranium through enrichment, so that in addition to the uranium used, additional natural uranium is required for enrichment.

Since operation of the TWR occurs in equilibrium with natural uranium or depleted uranium, the need for uranium enrichment is also largely eliminated. Additional fissile material would only be required for the first core of a TWR. When a TWR is removed from service, a successor plant could be started with the remaining core of a previous plant (Hejzlar 2021).

Due to the high target burn-up and the slightly higher efficiency of a TWR, the energy generated per fuel element would be around a factor of five higher than current LWR, so that the amount of spent fuel generated would be around 80% lower with the same amount of energy provided. This would also reduce the cost of interim storage and repository of spent fuel elements (Hejzlar 2021). Furthermore, due to the smaller waste volume, repository in deep boreholes could become an attractive option instead of geological repositories (Gilleland et al. 2016).

The developers also expect that the storage of spent fuel in the reactor core will reduce the need for intermediate storage and transport of spent fuel elements (TerraPower, LLC 2020).

If depleted uranium is used instead of enriched or natural uranium, the developers also envisage a contribution to the disposal of existing depleted uranium stocks, which would otherwise have to be disposed of as low-level waste (Hejzlar 2021).

Conclusions on supply and disposal

In equilibrium operation, only natural or depleted uranium is intended to supply a TWR. The fact that it is therefore fundamentally possible to forego the enrichment of uranium or the necessary production of uranium-plutonium mixed oxide fuels as with other SFR represents a significant advantage of the TWR compared to the technology line and current LWR. This also means that no additional quantities of depleted uranium are generated.

For a transitional period or for the start of a new TWR, however, fuel with a higher fissile material content in the range of 10% or more is required.

It is also the declared aim of the TWR to be able to forego the reprocessing of spent fuel. This represents an advantage over the SFR technology line, but corresponds to the open fuel cycle that is common today in LWR, so that this does not result in any relevant difference compared to current LWR.

Due to the high target burn-up in the TWR, a smaller volume and a smaller mass of spent fuel elements are generated in relation to the same amount of energy generated. However, these contain at least a comparable amount of fission products to the correspondingly larger amount of spent fuel from LWR. The documents evaluated do not contain any information on the content of transuranic elements in the spent fuel of a TWR, meaning that other parameters such as the radiotoxicity of the spent fuel elements and their heat generation cannot be assessed. There is also no information on the extent to which the highly burned-out metallic fuels can be sent for direct repository or which conditioning steps would be required before geological repository. In this context, it cannot be assumed that a TWR will have any relevant advantages over current LWR in terms of disposal.

The developers also point to the possible contribution to the disposal of global stocks of depleted uranium by using it as fresh fuel in the TWR. In view of the extremely large stocks of depleted uranium available worldwide today, the proportion of quantities that can potentially be used as fuel in TWR represents only an extremely small proportion of these stocks. This does not result in any relevant advantage of the TWR over LWR.

5.2.4 Proliferation risks

The developers expect a high degree of proliferation resistance (TerraPower, LLC 2020).

By not reprocessing, proliferation risks associated with direct access to plutonium stocks will be avoided. At the same time, due to the high target burn-ups, the quality of the plutonium remaining in the spent fuel will be very low for nuclear weapons applications, which is intended to reduce its attractiveness for a potential proliferator. At the same time, fuel elements will only be changed every 10 years approximately, so that possibilities for accessing fuel with a high fraction of high-quality plutonium are reduced (Hejzlar 2021).

It is only necessary to provide the fissile material required for a first core in the form of enriched uranium for the commissioning of an (additional) TWR reactor. This would significantly reduce the need for enrichment capacities (Hejzlar 2021).

According to (Hejzlar 2021), the enrichment level required for the new commissioning of a TWR is less than 10% and thus somewhat higher than for current LWR, but still clearly in the range of low-enriched uranium.

Conclusion on proliferation risks

No enriched uranium is required for the continuous operation of a TWR. The significantly reduced need for enriched uranium compared to current LWR is fundamentally an advantage over current LWR. The fuel required for current LWR made of low-enriched uranium itself, however, only represents a slightly increased proliferation potential compared to natural uranium, and a higher proliferation risk is associated with the uranium enrichment plants. The extent to which the number and capacity of uranium enrichment plants can be reduced overall by using TWR reactors cannot currently be assessed.

The quality of the plutonium remaining in the spent fuel of a TWR is very low for nuclear weapons applications. This represents an advantage over current LWR, but especially also over the technology line.

In principle, breeder reactors are however well suited to producing large quantities of high-quality plutonium for nuclear weapons applications (see Chapter **2.6.3**). It must be assumed that high-quality plutonium is also produced in relevant quantities in the fuel of the TWR during the breeding phase. This basically gives the plant operator the opportunity to produce material suitable for nuclear weapons by using an alternative loading and unloading strategy. To what extent this can be ruled out for a TWR cannot currently be assessed.

The declared aim of the TWR is to be able to dispense with the reprocessing of spent fuel. This represents an advantage over the SFR technology line, but corresponds to the open fuel cycle that is common today in LWR, so that this does not result in any relevant difference compared to current LWR.

5.2.5 Technological development status

In addition to the basic neutron physics design of TWR reactors, (Hejzlar 2021) formulates the main areas of development as being fuel development, thermohydraulic issues and mechanical design limits.

The developers see the possibilities of comprehensive modelling and computational design optimisation before the construction of a prototype reactor as an important advantage over previous, historical lines of development. Neutron physics calculation models, based on 3D diffusion and Monte Carlo programmes, were validated by benchmark calculations for previous fast reactors such as the Fast Flux Test Facility (FFTF), and thermohydraulic and numerical fluid mechanics programmes (CFD codes) are also used. The fuel behaviour is investigated using the finite element calculation programme ALCHEMY, and the ANL’s SASSYS and RELAP-5 3D programmes were used for safety investigations (Gilleland et al. 2016).

A basic neutron physics design is available that achieves equilibrium operation of the TWR with a load exclusively of natural or depleted uranium. The reactor core is constructed in six concentric rings of fuel elements with different burn-up stages.

The achievable fuel burn-up is central to the implementation of the TWR concept.

In order to first breed plutonium from depleted uranium and then use it again to generate energy, the TWR requires the highest possible burn-up. The developers specify a value of 30% as the minimum required target burn-up that makes the use of depleted uranium as fresh fuel possible. This is considerably higher than the highest burn-ups achieved to date with metallic fuel and HT9 cladding tubes, which the developers specify as 20% (Hejzlar et al. 2013).

The achievable burn-up is limited by technical limits. The developers specify the cladding tube strain due to radiation or thermal creep as the key limit.

The cladding tube strain can be influenced by changing the cladding tube wall thickness, but this leads to neutron losses and thus to a higher required minimum burn-up. The developers therefore want to reduce the cladding tube strain by reducing the effective fuel density. A lower effective fuel density means that a larger volume is available for the absorption of fission products, which reduces fuel swelling and the associated load on the cladding tube. The developers are therefore aiming for

an effective fuel density of less than 75%, a value that is lower than the previous metallic fuels for sodium-cooled reactors. In addition, gaseous fission products that are produced will be removed from the fuel rod in order to limit pressure build-up and the associated load on the cladding tube to less than 3% expansion (Hejzlar 2021).

The developers state that the radiation-induced strain on the cladding tube during the intended target burn-up is approximately 500 dpa, but (Hejzlar et al. 2013) estimates this value as 600 dpa. This value is a factor of 2.5 higher than the previous maximum value of approximately 200 dpa, which was achieved with test fuel rods with metallic fuel and a ferritic martensitic steel cladding tube. To achieve this goal, TerraPower is developing an advanced ferritic martensitic steel (TP HT9) with an optimised microstructure. To investigate the irradiation behaviour of this cladding tube material, material samples developed by TerraPower have been bombarded with heavy ions in order to generate radiation exposure comparable to that in reactor use. This material was compared with historical HT9 cladding tube material samples that had been irradiated in the American Fast Flux Test Facility (FFTF). The TP HT9 showed significantly lower swelling behaviour at the same radiation exposure than the historical HT9. However, the heavy ion irradiation does not reflect all the effects of the cladding tube strain during reactor use. These are estimated as part of model calculations. The developers themselves assume that an irradiation programme is necessary to validate the model calculations (Hejzlar 2021). According to (Gilleland et al. 2016), material samples from TerraPower are irradiated in both the Advanced Test Reactor at the INL in the USA and in the BOR-60 at the NIIAR in Russia, and post-irradiation tests are carried out at the INL. A test fuel element for fluid mechanics tests and to demonstrate industrial manufacturability was manufactured by AREVA.

The other structural materials of the fuel element are also exposed to high doses of radiation, which leads to deformation of the fuel element structures during burn-up. To avoid this, reinforced steel structures are typically used in sodium-cooled reactors, which in turn leads to neutron losses. As these cannot be compensated for by increased initial enrichment in the TWR as in other sodium-cooled reactors, TP HT9 is to be used for the structural materials instead. This is intended to limit the deformation of the fuel element boxes to less than 3.3 mm. In addition, the fuel element structures are optimised so that the deformation forces caused by internal pressure differences remain as low as possible (Hejzlar 2021).

As a distance of 3.5 mm is provided between the fuel elements, only a very small residual distance remains with a maximum deformation of 3.3 mm.

In order to implement the gas relief of the fuel rod, cleaning systems for separating caesium from the coolant, as well as systems for separating and storing krypton-85, must be developed in particular.

Another aspect determining design is the need for a core design that is stable in terms of neutron physics. In the case of large SFR, this is largely determined by the sodium temperature coefficient. If the power in the reactor core increases, the density of the sodium decreases. On the one hand, this makes the neutron spectrum harder, which leads to an increase in reactivity, and on the other hand, it can lead to a higher leakage of neutrons from the reactor core. Since a TWR should be designed with as little leakage as possible in order to make the existing neutrons available, as far as possible, for breeding new fissile material, the TWR is expected to have a (strongly) positive sodium temperature coefficient. In order to obtain a negative overall coefficient, taking other temperature coefficients into account, the value of the sodium temperature coefficient must be kept sufficiently small by means of a suitable design. For design parameters that should be used to minimise the total amount of sodium in the core, (Hejzlar 2021) puts forward a fuel rod grid that is as small as possible, a distance between adjacent fuel elements that is as small as possible, and a fuel rod design that does not use sodium filling between the fuel and the cladding tube to improve heat transfer.

The construction and operation of a prototype reactor (TWR-P) on the way to a commercial system is seen by (Hejzlar et al. 2013; Gilleland et al. 2016) as a necessary, intermediate step in order to test developments specific to the TWR, such as the passive shutdown device and the fuel handling of the TWR, and to validate the calculated reactivity coefficients of the overall system and the calculated overall neutronics.

Conclusions on technological development status

With a view to the development of cladding tube and structural materials, specific materials from the TWR are being examined for their suitability for use, including in laboratory tests. However, to date, the operational experience and experimental data are still a long way from covering the parameter space required for a TWR. Irradiation tests and post-irradiation studies are also required for the fuel development itself. This means that the state of material development for the TWR is still squarely in the area of ”applied research”.

Conceptual ideas exist for a number of systems required for operation, such as degassing the fuel rods and the associated necessary coolant cleaning, but these cannot yet be investigated experimentally. Technological solutions also still need to be developed and tested for operational processes such as loading and unloading fuel. In this respect too, the TWR can therefore still be categorised as being “applied research”.

The core design and the exact prediction of safety-related parameters, such as the reactivity coefficient, are central to the safety of the plant. Extensive calculation programmes already exist for this, which are already used as part of the design process. However, the results of these calculations still have to be validated on prototype plants. Verification can therefore be classified as being in “development”.

Overall, the authors of this report estimate the current development status of the TWR to be in the stage of "applied research".

5.2.6 Implementation

5.2.6.1 Planned area of application

The TWR is basically intended for use in power generation. The focus here is on the base load supply of large metropolitan areas (Gilleland et al. 2016). (Gilleland et al. 2016) assumes an installed capacity of 450 GW of TWR reactors in 2100.

However, due to the higher core outlet temperatures (compared to LWR) in the range of 500-540 °C, TerraPower is also examining other possible applications. For this purpose, the TWR will be coupled with an integrated energy system (TWR-IES). For this purpose, the heat generated in the TWR will be initially transferred to heat storage using commercially available molten salts from the field of solar thermal energy. The subsequent use as process heat in industry or for hydrogen production or for variable power generation in load-following operation would then be decoupled from nuclear heat generation. The developers hope that this will continue to lower system costs (Hejzlar 2021).

For the production of hydrogen in a high-temperature electrolyser at 750-900 °C, the developers envisage heating the water vapour using the heat stored at a temperature of 500-540 °C with additional electrical heating. By recovering heat from the electrolyser, the additional energy required for electrical heating remains low, resulting in cost-efficient operation (Hejzlar 2021, p. 654).

5.2.6.2 Project timeline

The idea of reactors with in-situ breeding and burning of plutonium goes back to the late 1950s. Building on these considerations, the company TerraPower was founded in 2006. Early estimates assumed that a prototype reactor could be realised in the early 2020s and a globally available commercial system a few years later (Hejzlar et al. 2013).

In (ARIS n.d.), the prototype plants are still expected to be built between 2018 and 2023, with a commercial plant being commissioned in the late 2020s or early 2030s.

(Gilleland et al. 2016) The planned commissioning date for the first reactor is 2026.

In a fact sheet (TerraPower, LLC 2020), published on the TerraPower website in 2020, the target for the availability of TWR technology was given as the mid-2020s. This would make the TWR concept available much earlier than all other technology lines of the Generation IV Initiative It is pointed out that the company has already built up a supply infrastructure for the most important materials, components and fuel over the last 10 years. The construction of the Versatile Test Reactor (VTR) in the USA is named as a key next milestone. This is intended to provide testing options for fuels, materials and sensors in particular. A first TWR prototype reactor (TWR-P) would serve to demonstrate the reliable operation of such a plant and as a regulatory basis for advanced, commercial reactors.

In an updated fact sheet (TerraPower, LLC 2022a), however, the TWR reactor concept is only viewed as being an important, long-term goal of developments at TerraPower. The focus of TerraPower’s developments has currently shifted to a smaller project under the name “Sodium” that can be categorised within the spectrum of small, modular reactors.

Conclusions on project progress

The basic idea for reactors with in-situ breeding of fissile material is already very old. The TWR concept builds on previous approaches to SFR and aims to develop this further with concepts for in-situ breeding.

However, it has not been possible to keep to the roadmap originally formulated by the developers. While the original schedule assumed the construction of a prototype within 15 years, after 15 years of development the construction of a prototype is still not on the horizon. While the original plan envisaged the construction of a commercial plant after 25 years, the further development of the TWR concept and the construction of a plant are now only seen by the developers as being a long-term goal. Overall, the project's progress to date has been affected by massive delays, even to the point of a shift to other development directions.

5.2.6.3 Costs

Since a sodium cooling circuit is a low-pressure system, the developers expect, together with the low costs for fuel supply and disposal, significant cost savings in the area of investment and operating costs (TerraPower, LLC 2020).

The selected fuel strategy eliminates costs for fuel enrichment and reprocessing. Costs for enriching uranium only arise for the initial core of a TWR. Reprocessing of spent fuel is explicitly not planned. This is intended to avoid costs for the construction and operation of corresponding facilities and for the disposal of remaining waste (Hejzlar 2021).

The developers also argue that interim storage costs are also lower than with currentand repository costs are also lower than with current LWR due to the lower volumes of spent fuel generated per unit energy (Hejzlar 2021).

In (Gilleland et al. 2016), the investment costs (overnight costs) for a TWR are stated as being comparable to current, new LWR. The savings in supply and disposal costs over the course of a 60-year operating life of a TWR are estimated at USD 4 to 5 billion.

The possibility of separating the nuclear heat generation system from the subsequent use of the heat generated via an intermediate heat storage system, based on an intermediate cooling circuit with a molten salt, is identified by (Hejzlar 2021) as a key cost advantage. Accordingly, the high system costs of today’s LWR are largely caused by the complexity of the overall system and the high regulatory requirements. The developers expect significant cost reduction potential through simplification and limitation of the nuclear part of the plant to pure heat production, as well as the spatial and system-technical separation of the application part.

Conclusions on costs

On the one hand, the developers point to possible cost savings in TWR investment costs, but on the other hand they themselves estimate the investment costs to be comparable to current LWR. The available information on the expected service life of a plant is also contradictory. Overall, operating experience with SFR to date indicates a high risk with regard to the expected service life of an SFR, see Chapter **4.1**. From today's perspective, it is not apparent that a TWR could offer significant savings in terms of investment costs compared to current LWR; due to the high technological risk of SFR to date, it is more likely that it represents a disadvantage compared to current LWR.

Even with current LWR, only a few countries are pursuing the strategy of reprocessing spent fuel. This, therefore, does not result in any cost advantages compared to current LWR using direct repository of spent fuel. However, there would be cost advantages over other fast breeder reactor concepts with the concept of reprocessing and plutonium re-use (see, for example, Chapter 5.1). The elimination of costs for uranium enrichment can lead to slight cost reductions in supply.

Small waste volumes can lead to lower interim storage and repository costs. However, the chemical form of the spent fuel, as well as its radiological inventory and its thermal output, also changes. The extent to which this will require special conditioning steps or whether additional costs or time requirements will arise for interim storage is not discussed by the developers.

Overall, fuel supply and disposal in current LWR only constitute a small proportion of the levelised cost of electricity, see also Chapter **2.5**. Therefore, significant cost advantages compared to current LWR cannot be expected on the basis of these differences.

5.3 LFR: BREST-OD-300

The BREST-OD-300 (Быстрый Реактор ЕСТественной безопасности for Fast Reactor with Inherent Safety) is a reactor from the LFR technology line (Chapter **4.2**) and is being built in Russia as a demonstration reactor of a commercial reactor concept with an electrical output of 300 MW and a thermal output of 700 MW for use in a closed fuel cycle. It aims to achieve the same goals as other GIF systems, namely improving safety characteristics, proliferation resistance, economic efficiency and using a closed fuel cycle for better uranium utilisation (Adamov et al. 2021). The reactor will be used to carry out extensive research and development programmes towards an industrial reactor concept.

The first reactor concepts were presented in the early 2000s (Filin et al. 2003; Filin et al. 2001; Khalil et al.). The BREST-OD-300 follows on from the Russian and former Soviet development line of lead-cooled fast reactors and shares most of the essential properties of the technology line. It is the most advanced reactor concept of the LFR series, construction began in summer 2021.[184] A reactor concept with higher output, the BREST-1200, is planned as a follow-up.

The following description is based on (Schulenberg 2020; Alemberti 2021; Adamov et al. 2021; GIF 2021a), unless otherwise stated.

5.3.1 Description of the plant concept

The reactor core and the complete primary cooling circuit, as well as the reflectors and the equipment for changing the fuel elements, are located within the reactor vessel. The reactor vessel is made of reinforced concrete with several caverns and is thermally insulated. The inside of the reinforced concrete is lined with steel. The reactor core, in which spent fuel elements are also stored, is located in a central cavern. The reactor core is surrounded by an inner vessel that separates the hot and cold sections of the reactor. The four circuits of the cooling system are located in four additional caverns, including a coolant pump, two steam generators and two immersion coolers per loop. The design is therefore a mixture of “pool” construction with elements of “loop” construction. The heat is transferred to a secondary water-steam circuit at 505 °C and a pressure of 17 MPa, allowing an efficiency of 42% to be achieved. The feed water is preheated to 340 °C to prevent localised freezing of the lead.

The core inlet and outlet temperatures are 420 °C and 535 °C. The lead flows at a relatively slow speed of 2 m/s from the core through coaxial pipes into the caverns with the steam generators and cools down as it flows through the steam generators. The cooled lead is pumped by the vertical coolant pump to a level that is 1.60 m higher than the level of the hot lead in the reactor. From there, the cold lead flows back through the coaxial pipeline and down the outer wall of the inner vessel that surrounds the core, and from there back into the core. This allows the inner vessel of the core and the steel shell of the central cavern to be cooled while reducing corrosion from hot lead on the vessel wall. The system design is intended to prevent steam bubbles from entering the primary coolant flow in the event of leaks in the steam generators. A pressure reduction system prevents overpressure in the primary system.

Between the central cavern and the four surrounding caverns there is a heating system in which steam, which is initially generated from the residual heat of the reactor, heats the central cavern and the other systems to 400 °C. If the coolant pumps fail, natural convection sets in, initially driven by

[184] https://rosatom-europe.com/de/press-centre/news/19156/

the temperature gradient of the lead until the water levels adjust. In this case, the immersion coolers take over the residual heat removal to a passive system and generate natural convection along the immersion coolers and the outer vessel wall, which are also cooled by cooling systems in the concrete of the vessel wall.

The reactor core, with a diameter of 2.4 m, operates at a power density of 500 MW/m^3. It is located in the central cavern and consists of a wide grid with large spacing to allow large coolant flow for natural convection to remove residual heat. The fuel elements are hexagonal with a wide spacing of 13 mm between the fuel rods, which have a diameter of 10 mm and a length of 3.2 m. The active length is 1.1 m. Blankets for breeding new fissile material are not included; they have been replaced by lead reflectors within the core. However, the fuel rods in the upper and lower sections should also contain uranium-238, in addition to a large fission gas plenum. The cladding tube should be made of ferritic-martensitic chromium steel and exposed to a cladding tube temperature of 650 °C. A cycle time of 300 days between fuel element changes is planned.

The reactor fuel consists of a very thermally conductive uranium-plutonium-nitride fuel (mixed nitride uranium plutonium fuel, MNUP or MNIT). Nitride fuel is much better suited to heat conduction than MOX, permitting higher linear heat generation rate of the fuel rod (approx. 40 to 42 kW/m). Depleted uranium is used, as well as plutonium with a composition that corresponds to the spent fuel of a VVER-LWR after appropriate cooling and reprocessing. Due to its high density, nitride fuel has a positive effect on the in-situ breeding properties in uranium and also on retention properties for fission gases, so that the internal pressure in the fuel rod remains relatively low, for example compared to MOX. During reprocessing, a joint separation of uranium-plutonium and the minor actinides (U-Pu-MA) is then planned, without separating plutonium. Reprocessing takes place directly on the plant site, which avoids the need to transport fresh and spent fuel elements.

In addition to the typical properties of LFR such as a high boiling point, freezing and sealing in the event of certain leaks, high heat capacity and thus longer reaction times for safety systems as well as the neutronic properties and the influence on power distribution within the core, lead as a coolant in the BN-300 also has the property that it is well suited to combining nitride fuels while maintaining a breeding rate above unity, allowing the maintenance of a small excess reactivity and thus preventing a prompt-critical power excursion (Adamov et al. 2021).

5.3.2 Safety concept and safety features

The BREST-OD-300 shares the basic safety properties of the LFR technology line (see Chapter **4.2.5**), such as the inert properties of lead, low pressure and low outlet temperature, an integral primary cooling circuit to avoid loss-of-coolant accidents, avoidance of positive reactivity input through coolant evaporation due to the high boiling point of lead, the possibility of passive residual heat removal through natural convection and low power.

A steam generator heating tube failure is controlled by the spatial separation of the steam generators from the reactor core so that released water vapour does not penetrate the reactor core.

The core is designed so that the excess reactivity is less than the proportion of delayed neutrons and thus prompt supercritical states are avoided, even when the full excess reactivity is supplied, such as when the control rods are fully extended. If the full excess reactivity were supplied, the core would reach a maximum fuel temperature of 1640 °C and a cladding temperature of 1260 °C for a few seconds without the fuel melting or the coolant boiling.

If the power supply fails and the mechanical shutdown systems fail at the same time, maximum cladding temperatures of around 900 °C are reached. If the external power supply fails, two of the four emergency cooling systems are sufficient to remove the residual heat. The probability of a leak with partial loss of coolant with the selected container design was calculated to be no more than $9.7x10^{-10}$ per year.

The developers' safety analyses assume that serious accidents, without external influences, only occur with a probability of $6.5x10^{-9}$ per year. This low probability arises because the developers assume that core destruction with a meltdown, boiling coolant and an interruption of natural convection are practically impossible, and that the release of radioactivity remains within limits even in conservative scenarios. For nuclear power plants with a probability of severe accidents of less than $3.2x10^{-8}$ per year, no evacuation zones or possible relocations to protect the population are to be planned (Adamov et al. 2021).

Conclusions on safety

The BREST-OD-300 shares the basic safety properties of the LFR technology line (see Chapter **4.2.5**) and features additional properties such as combining elements of the “pool” and “loop” design with spatial decoupling of the steam generator from the primary circuit, the prevention of steam entering the primary coolant in the event of a steam generator heating tube leak and low excess reactivity, or low maximum fuel and coolant temperatures even in the event of malfunctions and accidents, which should improve safety when compared to other reactor concepts.

The developers state that serious accidents without external influences are extremely unlikely. It should therefore not be necessary to set up evacuation zones. This would give the BREST-OD-300 an advantage in terms of safety compared to the LWR.

5.3.3 Supply and disposal aspects

The BREST-OD-300 shares basic properties regarding supply and disposal with the LFR technology line when lead is used as a coolant (see Chapter **4.2.6**).

One characteristic of nitride fuel is the formation of nitrogen gas and the formation of radioactive carbon-14 from nitrogen-14 during irradiation. Carbon-14 is problematic when disposed of in high concentrations due to its half-life of 5370 years. To avoid the formation of carbon-14, the recycled fuel should normally be enriched with nitrogen-15, which is associated with associated cost disadvantages. A combination of hydrometallurgical and pyrochemical processes is being researched in Russia for reprocessing of the BREST-OD-300 fuel elements. In these processes, it is planned to prevent carbon-14 from passing into the gas phase as CO_2 and therefore remains in elemental form as a layer on the surface of the melt and can be easily and compactly removed. It is therefore planned to work with non-enriched nitrogen in the BREST-OD-300 fuel (NEA 2018).

In terms of the breeding rate, the core will initially be loaded with reprocessed plutonium from LWR fuel, but it would also be possible to use enriched uranium with an enrichment of less than 15%. The core has a slightly positive breeding rate and an equilibrium state is sought in which new fuel elements are manufactured from the reprocessed plutonium, minor actinides and uranium and only depleted uranium is fed into the fuel cycle. It should therefore only be necessary to dispose of the fission products. Long-term interim storage in special warehouses is planned for these as well as subsequent conditioning in stable waste containers for repository.

Conclusions on supply and disposal

If integrated into the planned fuel cycle, the supply would have the advantage of not requiring fresh fissile material and thus being able to dispense with uranium enrichment; only depleted uranium would be required. This would give the BREST-OD-300 an advantage over an open LWR fuel cycle.

No additional actinides are produced in equilibrium operation, but the amount of fission products would increase. Reprocessing creates additional waste streams and operational radiation emissions and risks. The fission products would have to be disposed of after long-term interim storage. Longer interim storage is not specific to the reactor concept, but could be used in any reactor concept and fuel cycle to reduce the dose and heat output of the waste. The long-term safety analyses for geological repositories are hardly influenced by the actinides, but by long-lived mobile fission products. The BREST-OD-300 would have no significant advantages or disadvantages in terms of disposal compared to current LWR.

5.3.4 Proliferation risks

Uranium enrichment is not necessary. However, the initial core must contain a quantity of separated plutonium or LEU.

All fuel elements contain the same mass of plutonium in equilibrium. Uranium blankets are not planned and the exchange of individual fuel elements for blankets is limited due to the low excess reactivity. However, it is stated that the fuel rods above and below the active zone contain uranium-238, in which almost pure plutonium-239 would be bred.

During reprocessing, a homogeneous material flow of uranium and transuranic elements will be separated meaning that this material mixture would still possess a high radiation barrier. How proliferation-resistant the reprocessing is, depends on the technology used and how easy it would be to switch to pure plutonium separation or to extract an intermediate product with high proportions of plutonium; the plutonium would be comparable to reactor plutonium from LWR.

The reprocessing plant will be built directly on the reactor site. Transportation and the associated proliferation risks would be eliminated.

Conclusion on proliferation risks

The proliferation risks of the BREST-OD-300 differ only slightly from the technology line of the LFR and SFR (see Chapter **0**).

The possibility of breeding weapons-grade plutonium is particularly relevant. Uranium blankets are not planned, however, and the use of uranium blankets is therefore limited for criticality reasons. The use of reprocessing increases possible proliferation risks and requires more effort in fissile material control and monitoring measures. Using pure MOX or transmutation fuels would eliminate the need for uranium enrichment.

Compared to the LWR, the BREST-OD-300 has no significant advantages or disadvantages in terms of proliferation resistance if an open fuel cycle with the use of LEU is assumed.

5.3.5 Technological development status

To date, extensive test programmes have been included in the development of the BREST-OD-300. Testing was performed according to (Adamov et al. 2021; GIF 2021a):

- In the case of a steam generator heating tube leak, it was experimentally proven that a leak would not cause failure and neighbouring tubes would not be damaged.
- Calculations showed that the assumed entry of steam into the core in the event of a steam generator heating tube leak would not cause a positive reactivity effect. The positioning of the steam generators significantly reduced the probability of steam entry. Relevant experiments in the Fast Critical Facility (BFS) are still to be undertaken.
- To test the pumps, a large number of studies were carried out to optimise the flow of the impeller blades using both experiments on scaled models and full-size models, as well as calculations. An automatic control and monitoring system was developed. The bearings were also optimised. Full-scale tests of seals have shown that the leakage of sealing fluid does not exceed the permissible values. The next stage is planned to be a comprehensive check of the functionality of a prototype pump.
- The developers indicate that lead coolant technology is ready for use. The main focus is therefore on the development and testing of specific elements and technological systems. The method of enriching lead with oxygen using oxide granules (PbO) and the method of taking lead samples for analysis have been worked out, measurement methods have been developed and certified. A lead coolant routine has been developed, including cleaning and decontamination, and confirmed by extensive testing. The absence of excessive material corrosion at the regulated oxygen concentration in lead of $1x10^{-6}$ ppm to $4x10^{-6}$ ppm has been demonstrated experimentally. The corrosion resistance of EP302-Sh equipment steel in the range 420-540 °C up to $5.3x10^{4}$ h and EP302M-Sh at 450-550 °C up to $2.8x10^{4}$ h has been demonstrated. The coolant monitoring and control system has been tested and qualified and is ready for installation.
- A large number of experimental studies on the analysis of fission and activation products in the coolant have been conducted to perform safety analyses for operating temperatures and accident conditions with significant increases in coolant temperature to 680 °C. This also made it necessary to clean the lead for the initial coolant inventory while minimising the associated costs.
- Fuel development of nitride fuels has been implemented in pilot production plants and manufacturing improved while industrial production for the BREST-OD-300 is being prepared.
- The burn-up is to be kept low initially, at 6% of the initial heavy metal content, and later increased to 9-10%. To this end, over 1000 fuel elements have already been irradiated in the BN-600 and BOR-60. The test fuel elements were irradiated to a target burn-up of 6%. This corresponds to the target burn-up of the initial load of the BREST-OD-300. In the ETVS-11 experiment, for example, 61 fuel elements similar to the BREST-OD-300 were tested with nitride fuel and EP-823Sh cladding material and a burn-up of 8.3% of the initial heavy metal content was achieved at a cladding strain of 98 dpa. Irradiation to a target burn-up of 8.5% was also successfully carried out, and tests are continuing to a target burn-up of 9.2% of the initial heavy metal content. This achieved a load on the cladding tube material of up to 110 dpa. All essential operating conditions were demonstrated in the tests (GIF 2021a, p. 15).

- Fuel element mock-ups were manufactured under industrial conditions and subjected to a series of material tests, and the loading and unloading of the core was experimentally tested. The first test fuel elements for uranium-plutonium nitride fuels for use in BREST-OD-300 fast breeder reactors were manufactured.[185]

- A large number of tests were carried out on the reactor vessel material – on various concrete samples and on models of the vessel itself. Using a scale model of the vessel bottom, it was confirmed that the maximum permissible temperatures of the building structures are not exceeded. The properties of high-temperature concrete were determined experimentally at temperatures of 400-700 °C and under irradiation. Chemical inertness of the lead coolant with respect to concrete was demonstrated. Using a scale model developed for the central part of the vessel, the properties of the vessel under heat load and the parameters of gas release were determined. Sufficient knowledge has been gained to begin manufacturing the BREST OD-300 reactor vessel.

- Computer codes and experiments have confirmed that the small excess reactivity can be achieved even at the start of operation. The experiments are based on lead test set-ups, as well as a model of the nitride fuel. A full-scale model of the core in the BFS-2 test stand is planned.

- Computational validation of the thermo-hydraulic properties of the core has been carried out. Experimental validation of the hydraulic properties of fuel elements and reflector blocks has been carried out for water and lead and the calculations validated.

- Friction corrosion has been demonstrated in tests on small rod corrosion models conducted in liquid lead for 2500 and 5000 hours, showing cladding wear of less than 0.5 µm. The total thinning of the fuel rod cladding tube was calculated to be 60 µm.

- It is planned to set up a stand for experimental study of vibration characteristics in the steam generators. Lead has a high specific gravity, so it is necessary to analyse the possibility of dependent failure of the steam generator heating tubes when one of them is depressurised.

A factory for the production of uranium-plutonium nitride fuel is under construction at the Siberian Chemical Combine in Seversk.[186] Starting in 2022, planned production of fuel elements for the BREST-OD-300 and BN-1200 should begin here as part of the PRORYV project.

No previous experience has been reported on the development status of nitride fuels with minor actinides, especially with regard to large-scale implementation. No information is given on the development status of reprocessing technologies either.

According to comparable assessments in (Oeko-Institut e.V. 2023) regarding MOX fuel production with minor actinides, development status is considered to be low, as is the case with regard to the necessary reprocessing technologies. In terms of development status, both should therefore be assessed coming under the heading of “applied research”.

[185] https://www.world-nuclear-news.org/Articles/Russia-completes-testing-of-latest-fast-reactor-fu, last accessed 15/01/2021.

[186] https://www.world-nuclear-news.org/Articles/TVEL-unit-launches-CFR-600-fuel-manufacturing-site, last accessed 25/01/2022.

Conclusions on technological development status

Extensive development work has taken place on the BREST-OD-300. In particular, fuel development appears to be well advanced and the developers are confident that they will be able to solve the remaining tasks for scaled-up tests during the construction period of the reactor. Considerable development work is still required for construction of the reprocessing plant and fuel production using minor actinides; both are at the maximum development stage and categorised as “applied research”.

The development stage of the BREST-OD-300 may be estimated as being between “applied research” and “deployment”.

5.3.6 Implementation

The BREST-OD-300 is primarily intended to enable better utilisation of uranium and to use only depleted uranium as fresh fuel after reaching an equilibrium state in the fissile material balance.

5.3.6.1 Project timeline

Initial development work began in 1995 based on experience from lead-bismuth cooled submarine reactors in the Soviet Union. The first feasibility studies appeared in the 2000s (Filin et al. 2003; Filin et al. 2001; Khalil et al.). R&D work has been intensified since 2011 and a preliminary study of the BREST-OD-300 was presented in 2014, which was positively evaluated by a group of experts in 2015. In 2016, construction was planned for the Seversk site, a city located in the Tomsk Oblast. The concept was later revised again and re-evaluated in 2018.

In 2019-2020, the Russian Academy of Sciences also issued a positive assessment, recommending construction and confirming that the design corresponds to the modern state of science and technology. The report recommended starting construction immediately, as some of the problems and the necessary research and development work could be resolved during trial operation. Approval was granted in 2020. Construction began in 2021. Commissioning is planned for 2026. In 2018, plans were presented for a commercial reactor concept, the BR-1200 with a thermal output of 2930 MW and an electrical output of 1260 MW (Adamov et al. 2021). A reprocessing plant is scheduled to start operations in 2028.

Conclusions on project progress

The planning, from the first sketches in the early 2000s to breaking ground in 2021, has progressed more rapidly than with other SNR concepts, especially during the post-2014 phase following an initial detailed preliminary study. However, some development work still needs to be carried out during construction. In view of previous experience in the Soviet submarine programme, the developers are confident that they will be able to solve all of the problems before completion. The reactor will initially be operated with fuel that does not require minor actinides and reprocessing.

5.3.6.2 Costs

No information is currently available on the costs. The costs of development and construction are borne by the Russian state.

Conclusions on costs

Fuel supply and disposal in current LWR only accounts for a small proportion of the levelised cost of electricity. Significant cost advantages over current LWR, e.g. through the use of depleted uranium, cannot be expected on the basis of these differences in the BREST-OD-300 or the LFR technology line as a whole.

The use of reprocessing technology generates additional costs.

Further reliable statements on the cost structure of the BREST-OD-300 are not possible at the current stage of development.

5.4 GFR: GFR reference concept of the GIF

A reference concept for a gas-cooled fast reactor (GFR) is currently being pursued within the framework of the GIF (GIF 2021a). Details of this reference concept are shown, for example, in (GIF 2022b). Accordingly, the reference concept was essentially created up until 2011, after which only minor revisions were made. The developments towards the GFR reference concept are summarized in (van Rooijen 2009), interim developments can be found in (Vasile 2017; Stainsby 2015; Poette et al. 2013; Stainsby et al. 2011).

The main design goals for the GFR reference concept within the framework of GIF are (van Rooijen 2009):

Helium should be used as a cooling gas. The reactor should have a low power density in the fuel of approx. 40 W/g heavy metal and, associated with this, a low power density in the reactor of between 50 and 100 MW/m3 in order to reduce the requirements for coolability. The core design should enable passive heat removal, which requires high-temperature-resistant materials and a high coolant volume in the core.

The aim is to achieve a closed fuel cycle with complete recycling of all actinides. The newly produced amount of fissile material should be sufficient to continue operating the reactor itself. However, a surplus of fuel for other reactors (GFR, LWR or others) should not be produced. A breeding blanket for producing fissile material should also be dispensed with in order to avoid proliferation problems when handling high-purity plutonium. By completely recycling actinides, only fission products and actinides from losses during reprocessing should have to be sent to a repository facility.

Unless explicitly stated otherwise, the following description is based on (GIF 2022b).

5.4.1 Description of the plant concept

The development of the GFR concept is largely based on work by the French Commissariat à l’énergie atomique et aux énergies alternatives (CEA) (Hatala 2021).

The GIF reference concept currently envisages a reactor with 2400 MW of thermal output and a core outlet temperature of 850 °C. The energy is guided out of the reactor via an intermediate cooling circuit with a helium-nitrogen mixture in combination with a subsequent water-steam circuit. On the one hand, this structure is classified by GIF as technologically feasible and at the same time ensures high efficiency of heat conversion (GIF 2021a). The reactor is to have an electrical output of 1150 MW and thus an efficiency of 48% (GIF 2022b).

The high power and the associated larger reactor core lead to lower neutron losses from the core and thus allow a sufficient breeding rate to cover the reactor's own fissile material requirements without having to place too high demands on the fuel (GIF 2022b).

A power density of 100 MW/m^3 is planned for the reactor. This power density is in the upper range of the power densities discussed and represents a compromise with regard to the expected effects on economic efficiency (compact design) and safety (maintaining heat removal under all assumed boundary conditions).

The structure of the GFR reference concept from GIF is similar to the basic structure of a GFR, see Chapter **4.3.1** and **Figure 4-3**.

The fuel is uranium-plutonium carbide, which is enclosed in fuel rods by a ceramic cladding tube. The fuel consists of natural or depleted uranium and a fraction of 15-20% plutonium. In an equilibrium cycle, the fuel still contains about 1% minor actinides. The average burn-up of the fuel is 50.7 MWd/kg(HM), the maximum burn-up is 74.5 MWd/kg(HM).

The reactor core consists of hexagonal fuel elements with 217 fuel rods, which are limited by a fuel element box to guide the flow. The fuel element box is also made of a ceramic material. The reactor core consists of a total of 516 fuel elements. The volume of the reactor core is 23.6 m^3, with about 28% of the core volume being fuel, 27% structural materials and 45% cooling gas.

The structure of the reactor core is intended to contribute to a low pressure loss so that the cooling gas flow is maintained under all operating conditions.

The reactor core has a relatively flat geometry with a core height of 1.65 m and a diameter of 4.268 m in order to minimise pressure losses, which are given as 0.054 MPa for this geometry (GIF 2022b). (Poette et al. 2013) instead, gives a pressure loss across the core of 0.154 MPa.

The cladding tube currently planned is a multi-layered cladding tube, which also shares similarities with developments in the field of accident-tolerant fuels (ATF) for LWR. Two silicon carbide layers are separated by a central metallic liner. The liner is intended in particular to ensure the gas-tightness of the cladding tube. A mesh made of highly porous carbon is provided on the inside of the cladding tube to prevent interactions between the uranium-plutonium carbide pellet and the inner silicon carbide layer. Temperature and irradiation tests were carried out on the individual components of the cladding tube to support the design process (Stainsby 2015). A plenum is provided in the fuel rod to absorb the resulting fission gas, so that even at high burn-ups the internal pressure of the fuel rod remains limited and no degassing of the fuel rods is necessary.

The cladding tubes of the fuel rods must have a specific surface roughness in order to ensure sufficient heat transfer from the cladding tube to the coolant and to prevent pressure losses from the cooling gas flow over the core from becoming excessive (van Rooijen 2009).

The reactor pressure vessel (RPV) of the GFR has a cylindrical shell and is closed off at the top and bottom by a spherical shell. It is divided by an internal structure into an outer area with lower gas temperatures and an inner area with high gas temperatures, so that the outer wall of the RPV is not required to withstand the high core outlet temperatures of the GFR. The helium enters the outer area of the RPV via three cooling circuits at a pressure of 7 MPa and a temperature of 400 °C, where it also helps to cool the entire outer wall of the RPV. It enters the reactor core from below and flows upwards through it. Above the core, the hot gas enters the inner area of the RPV, from where it is guided out of the RPV and into the intercoolers via thermally insulated pipes.

Devices for loading and unloading the fuel elements are housed in the upper area of the RPV. These are loaded and unloaded individually through a lock over the RPV vessel head (Stainsby et al. 2011).

The control elements required for power regulation and shutdown enter the RPV from below.

In three main cooling circuits, the helium releases the heat to an intercooling system via a heat exchanger. The helium circulation in the primary cooling circuit is ensured by active fans. In the intermediate cooling circuits, a helium-nitrogen mixture drives a gas turbine at a pressure of 6.5 MPa and a temperature of 820 °C and releases the remaining energy at a temperature of 565 °C, via an intercooler, to a downstream water-steam circuit. Here, steam is generated at a pressure of 15 MPa and a temperature of 535 °C and used to generate electricity via another turbine.

The intermediate cooling circuits are designed to generate an electrical output of 3x130 MW. In the water-steam circuit, the fresh steam generated is fed to a common turbine, which is designed to generate 730 MW of electrical energy (Poette et al. 2013).

The RPV is also connected to three circuits for residual heat removal, which are designed to ensure the removal of residual heat when the plant is shut down or if the main cooling circuits fail. The circuits for residual heat removal have their own heat exchangers and fans as well as water supplies, which are housed inside the reactor building but outside the protective shell. The heat exchangers are installed at a large height difference with respect to the reactor in order to achieve a coolant flow through natural convection that is sufficient to remove residual heat (van Rooijen 2009). (Poette et al. 2013) states a required height difference of approx. 20 m regarding this.

In addition, several pressure accumulators are connected to the RPV, which are intended to replace the escaping helium with a helium-nitrogen mixture stored under high pressure in the event of a loss-of-coolant accident. Nitrogen can also be used in the pressure accumulators, since the negative neutronic properties of nitrogen are not important after the reactor is shut down and no relevant activation of the nitrogen occurs.

The RPV must be resistant to the helium inlet temperatures of approx. 400 °C and to embrittlement caused by the neutron flux losses from the reactor core. Various materials are being discussed for this purpose, and the RPV steel 316LN is currently being put forward as a reference (Vasile 2017).

The entire primary cooling circuit, including the primary-side heat exchangers and the pressure accumulators, is enclosed in a secondary protective shell, which is intended to ensure that the escaping helium is retained in the event of a loss-of-coolant accident. During normal operation, the secondary protective shell is filled with nitrogen, at a pressure slightly above atmospheric pressure, and must be designed for an overpressure of 0.6 to 1 MPa.

The reactor and the secondary protective shell are protected against external influences by a reactor building made of double-walled reinforced concrete.

5.4.2 Safety concept and safety features

The required high coolant velocities can contribute to vibration of the fuel rods, which can lead to cladding damage (van Rooijen 2009).

The Doppler coefficient of reactivity is approx. -1020 pcm/K at the beginning of a cycle and -876 pcm/K at the end of a cycle. The fraction of delayed neutrons is 360-370 pcm during the cycle. The reactivity effect in the event of a loss of helium coolant is approximately 0.9 USD (GIF 2022b).

Two independent shutdown systems in the form of control elements are provided for reactivity control and shutdown. If sufficient reliability cannot be demonstrated for these, a third shutdown system could be provided. For this purpose, approaches to introducing a liquid absorber at selected fuel element positions in the core have been investigated as diverse systems (Stainsby et al. 2011).

(Hatala 2021) As at the technology-line level, the main challenges for the GFR are, on the one hand, the development of suitable fuels for use under the boundary conditions of a high core exit temperature and sufficiently high power density in the fuel and, on the other hand, ensuring sufficient core cooling for all assumed fault and accident sequences.

In order to be able to maintain a coolant density sufficient for heat removal, even in the event of a loss-of-coolant accident, the protective shell must maintain a helium pressure of 0.6 to 1 MPa. This means that very little active circulation of the coolant should be sufficient to remove heat, and a transition for passive heat removal should be possible within a few hours of shutdown. In an active operating mode with continuous coolant circulation, the emergency cooling system should still be able to ensure removal of residual heat even when pressure is completely equalised (GIF 2022b). In order for the systems for residual heat removal to remove heat through purely passive natural circulation, a pressure of around 10 MPa must be maintained in the system through the protective shell (Stainsby et al. 2011).

The primary-side coolant fan should be mechanically coupled to the gas turbine in the intermediate cooling circuit (Poette et al. 2013).

In the residual heat removal circuits, the energy is transferred to a secondary cooling circuit via a heat exchanger. This transfers the heat to a water reservoir inside the reactor building through natural circulation. There, the heat is transferred to the environment through evaporation (Poette et al. 2013).

Isolation valves are provided in the intermediate cooling circuits to avoid a supercooling transient in the event of a secondary-side coolant loss (Poette et al. 2013).

Protection against external influences is ensured by a reactor building made of double-walled reinforced concrete.

Conclusions on safety

The advantages and disadvantages identified at the technology-line level for the GFR also apply without restriction at the level of the GFR reference concept.

The specific safety concept for the GFR reference concept is still in such an early conceptual phase that a more concrete assessment of the safety of the GFR reference concept is currently not possible.

5.4.3 Supply and disposal aspects

The fuel elements are intended to be used in the reactor for 3-5 years, each nominally for three cycles of 481 full-load days. The spent fuel will be reprocessed. The plutonium contained in it will be separated together with the minor actinides (neptunium, americium, curium). It is intended to use the Global Actinide Extraction Process (GANEX) for this. Alternatively, however, only the plutonium may be separated using the already established PUREX process. This would, in particular, avoid the problem of the high volatility of the americium carbide during carbothermic reduction in the fuel manufacturing process. A value of 0.1% is stated for the process losses during reprocessing (GIF 2022b).

The breeding rate will be one or only slightly higher than one; for proliferation reasons no breeding material will be used in the core, or in any case only a small amount.

In the equilibrium cycle, the fuel then contains a fraction of around 1.1% of minor actinides.

The plutonium content in the reactor core should be less than 10 tonnes per GW of electrical power. This should make it possible to build a larger reactor fleet using plutonium from the reprocessing of LWR fuels within a few decades.

Conclusion Supply and disposal aspects

The advantages and disadvantages identified at the technology-line level for the GFR also apply without restriction at the level of the GFR reference concept.

The specific elements of the fuel cycle (fuel, reprocessing technology) for the GFR reference concept are still in an early conceptual phase, meaning that a more concrete assessment of the supply and disposal aspects of the GFR reference concept is currently not possible.

5.4.4 Proliferation risks

The reactor core of the GFR reference concept should not contain any breeding fuel elements in order to minimise the production of high-purity plutonium. Furthermore, for proliferation reasons, all actinides will be recovered from the fuel during reprocessing and reused together. This is intended to make direct access to separated plutonium more difficult.

The reactor core of the GFR reference reactor contains around 10 t of plutonium per gigawatt of electrical power. In addition, the core contains an average of around 60 kg of neptunium, 400 kg of americium and 120 kg of curium. As a result, each fuel element of a GFR contains more than a significant amount of plutonium, both in the fresh and in the spent state (GIF 2022b).

With this in mind, (GIF 2022b) points out that the main measures contained in the GFR reference concept to reduce proliferation risks are that no enriched uranium is used, the plutonium in the GFR fuel cycle has an unfavourable isotopic composition for weapons applications (either plutonium from the reprocessing of LWR fuel or from spent GFR fuel, in which the fraction of thermally fissile plutonium isotopes is between 50 and 60%) and the fresh fuel elements of the GFR also contain a fraction of minor actinides, which make access to the fissile material difficult due to their radiological properties.

Overall, (GIF 2022b) views the GFR reference concept as having comparable proliferation risks to other fast reactor concepts (SFR, LFR).

Conclusions on proliferation

The advantages and disadvantages identified at the technology-line level for the GFR also apply without restriction at the level of the GFR reference concept.

5.4.5 Technological development status

Key open research and development aspects correspond to the aspects mentioned at the technology-line level.

For the GFR, (Vasile 2017) mentions, among other things, the development of sufficiently robust materials that can withstand the high operating temperatures and the hard neutron spectrum; the control of vibrations in fuel rods due to the high coolant velocities; a sufficiently durable cladding tube that ensures the retention of radioactive substances under all assumed boundary conditions; open questions regarding the design of the systems for residual heat removal and emergency cooling; Questions about the manufacture and quality assurance of the reactor pressure vessel, as well as the availability and industrial processability of high-temperature-resistant structural materials. As a next necessary development step, (Vasile 2017) also points to the construction and operation of an experimental reactor (ALLEGRO). Finally, (Vasile 2017) also sees the need to develop a set of rules for GFR.

Conclusions on technological development status

The development effort required for a GFR reference reactor corresponds in all areas to the open aspects already identified at the technology-line level. The concept itself is still in a very early development phase, as many specific design boundary conditions can only be determined when new findings can be gleaned from the operation of an experimental reactor (ALLEGRO).

Overall, the authors of this report estimate the development status of the GFR reference concept as coming under “applied research”, in line with the development status of the technology line.

5.4.6 Implementation

The GFR reference concept is primarily designed for the provision of electrical energy and the breeding of new fissile material. Providing process heat would, however, be possible in principle.

5.4.6.1 Project timeline

In its technology roadmap (GIF 2002), the GIF originally considered it possible to develop a GFR prototype design by 2019 and to build and commission a prototype reactor by 2025. The end of the GIF “viability” phase was expected to be around 2012. This date was still presented in (GIF 2009). However, in the 2014 update of the technology roadmap (GIF 2014), only the construction and operation of a smaller experimental reactor is expected within the next 10-20 years. The end of the “viability” phase would, therefore, not be reached until around 2022, and the subsequent “performance” phase would not be completed even in 2030. In (GIF 2018b), the completion of the “viability” phase, entry into the “performance” phase and, with a high degree of probability, the start of the “demonstration” phase are expected within the next 10-20 years. (GIF 2022b) points out that the GIF reference concept was only developed slightly further after 2011.

For ALLEGRO, (Vasile 2017) states that the conceptual phase should be completed by 2026 and then a decision should be made on how to continue the project.

Conclusions on project progress

There are currently no concrete timetables for the construction of a GFR reference reactor within the framework of the GIF. The previous assumptions regarding rapid development progress have not proven to be valid. The construction and operation of an experimental reactor (ALLEGRO) is therefore a necessary intermediate step before the potential construction and operation of a GFR reference reactor. Since only in such a reactor can the fuels and materials required for a GFR reference reactor be developed and tested under realistic operating conditions, it is to be expected that at least several decades will be required before planning and construction of a GRF reference reactor can begin.

5.4.6.2 Costs

The GFR reference concept is still in a very early phase of concept development. There are no concrete cost estimates for the construction and operation of such a reactor concept in the literature evaluated. The developers, however, expect cost advantages over current LWR due to the higher efficiency of the plant.

A planned service life of 60 years is specified for the GIF reference concept (GIF 2022b).

Conclusions on costs

The uncertainties identified at the technology-line level also apply to the GFR reference concept.

The developers assume a service life of the plant of 60 years. From today's perspective, this does not seem realistic. To date, no commercial nuclear reactor has achieved a service life of 60 years (IAEA 2022f). A service life of around 60 years or more is often specified for current LWR. At the same time, GFR reactors have significantly higher requirements regarding structural materials, so that from today's perspective, a statement on the service life of a GFR reference reactor seems unreliable.

5.5 MSR: LFTR (molten salt reactors with thermal neutron spectrum)

The Lithium Fluoride Thorium Reactor (LFTR) is a reactor concept from the technology line of Molten Salt Reactors (MSR). The following information on LFTR is based on (EPRI 2015; IAEA 2016e; Sorensen 2016), unless otherwise stated. A concept of the LFTR from the American start-up company Flibe Energy has been in development since 2011. Flibe Energy currently has around 25 employees.

The LFTR uses liquid molten salt as fuel and a graphite-moderated thermal spectrum. The LFTR is based heavily on the MSRE reactor design from the 1960s (see Chapter **4.4.3**), so there is already practical experience to build on. It is listed in the IAEA's ARIS database and was supported by the DoE in 2018 as part of the Advanced Nuclear Technology Development programme. As with other MSR, the high temperatures targeted are intended to ensure high efficiency in power generation, for which a gas turbine is planned in the LFTR.

The LFTR is intended to be operated with thorium fuel in a closed fuel cycle. The LFTR is therefore the only thorium reactor among the selected reactor concepts presented in greater detail in the Chapter 5.[187] At the technology-line level, thorium is intended as fuel in several MSR concepts as well as in VHTR concepts. The main arguments in favour of using thorium as an alternative to uranium are, typically, the large thorium reserves, an improvement in the waste problem and lower proliferation risks (IAEA 2005).

5.5.1 Description of the plant concept

The LFTR from Flibe Energy is a direct successor project to the MSRE experimental reactor and the MSRE reactor concept of the ORNL. The LFTR will be operated with an output of 600 MW (thermal) or 250 MW (electrical). The reactor operates at low operating pressure and high temperatures (500-700 °C).

A general description of an MSR can be found in Chapter **4.4.1**. In the LFTR, the fuel and molten salt blanket are contained in two separate plenums in a structure within the reactor vessel. Hastelloy N is used as the structural material for the reactor vessel, the piping and the heat exchangers. Most of the volume of the reactor vessel is filled with graphite, which serves as a moderator material and also as a structural material. The fuel salt circulates through the reactor in channels in the graphite (IAEA 2016e). The graphite is replaced every four years, as the graphite is exposed to much higher radiation levels than the metallic structures of the reactor (ORNL 2021). The exact arrangement of the graphite tubes for the molten salt flow and how they are connected to the plenums is not yet clear, but must include remote handling for installation, removal and maintenance as a primary design goal (EPRI 2015).

The integrity of the reactor vessel plays an important role in minimising potential accident hazards, as the structures within the reactor define the channels and volumes for the fuel. The reactor vessel should also contain a small heat exchanger to cool the blanket, in which the molten salt circulates in the reactor core by natural convection. In the blanket, the short-lived thorium-233 generates significant amounts of heat in the fuel salt, since the thorium-233 cannot be removed by chemical separation. If the reactor shuts down or chemical processing fails, the heat from the decay of the

[187] The use of thorium in new reactor concepts was evaluated in detail in (Oeko-Institut e.V. 2017). Parts of this evaluation are reproduced here in the LFTR reactor concept.

protactinium would also have to be removed from the reactor vessel. However, with a half-life of 22 minutes, the heat contribution of the thorium-233 ceases relatively quickly (EPRI 2015).

The reactor cell, consisting of the reactor vessel, primary heat exchanger, drain tanks, primary circuit pump and all pipes, must be kept at a temperature by electrical heaters during shutdown so that all liquids remain liquid and do not solidify.

The primary circuit begins and ends at the connections to the reactor vessel and includes the primary pump, primary heat exchanger, gas feed system and drain tanks for the fuel salt and associated cooling systems.

The LFTR reactor concept includes an intermediate cooling circuit with a molten salt that separates the high pressure gas turbine from the primary cooling circuit, with corresponding heat exchangers between the circuits. The intermediate cooling circuit begins and ends at the heat exchanger and includes a pump, the salt side of the heat exchanger to the gas turbine, drain tanks and pressure relief valves. The LFTR is intended to be operated with a supercritical gas turbine using carbon dioxide. Since the coolant salt of the intermediate cooling circuit does not contain any significant radioactivity, no cooling system is required for the drain tanks (EPRI 2015).

Fuel/coolant

LiF-BeF2-UF4 (FLiBeU) is to be used as the fuel salt, LiF-ThF4 (FLiTh) as the blanket salt, and LiF-BeF2 (FLiBe) as the pure coolant salt. The fuel salt leaves the reactor core at 653 °C and at low pressure and flows to the primary circuit pump, where the pressure is raised to 1.1 MPa, and then to the primary heat exchanger, where it is cooled to 500 °C. Part of the fuel salt is diverted to the gas separation plant, where fission product gases such as xenon, krypton, and tritium are separated by helium bubbling. The helium with the fission products is then mixed with a waste stream of the fuel salt (bleed) and stored in a bleed tank, where some of the very short-lived fission products decay. In total, about 0.5 kg of fission products are produced per day. Fuel salt is pumped from the drain tank into the chemical reprocessing and the gas from helium and fission products is fed into the exhaust gas purification system (EPRI 2015).

The LFTR is to be operated with a uranium-thorium fuel, is designed as a breeder and uses separate circuits for the fuel and for breeding new fissile material. Since naturally occurring thorium itself is not fissile, in all concepts for using thorium as fuel, a fissile material must be added to the thorium in a first step, either uranium-235 in increased enrichment or plutonium from the reprocessing of spent reactor fuel.

In a fuel concept based on thorium, the radioactive isotope Th-233 is produced from the naturally occurring thorium isotope Th-232 by neutron capture. This decays (half-life approx. 22 min) via an intermediate step (Pa-233, half-life 27 d) to the highly fissile uranium isotope uranium-233, which is then used to generate energy. The molten thorium salt in the breeding blanket is reprocessed at a rate such that the entire blanket material is processed once every four days. The fuel salt is completely reprocessed every 300 days (EPRI 2015).

The LFTR consumes only a small part of the initial thorium in the breeding blanket. The main function of the chemical reprocessing plant is to separate uranium and protactinium from the breeding salt and to feed the separated uranium-233 into the fuel salt. In addition, gaseous and other fission products are removed from the fuel salt and suitably conditioned. Once the system is in operation and in equilibrium, only small amounts of thorium tetrafluoride need to be added (EPRI 2015).

During the entire chemical reprocessing, almost all actinides remain in the fuel salt and thus within the containment of the reactor until they have been fissioned. Smaller amounts of actinides are removed from the fuel salt and enter the waste stream. This actinide loss could, however, be further reduced (EPRI 2015).

As with most MSR reactor concepts with molten fluoride salt with lithium, the lithium-7 must be enriched for neutronic reasons (see also Chapter **4.4.1**).

5.5.2 Safety concept and safety features

In principle, the LFTR does not differ in terms of safety properties from the representation of thermal MSR with molten fluoride salt (liquid fuel) at the level of the MSR technology line (Chapter **4.4.6**). Some special features of the LFTR from Flibe Energy are presented here again.

Based on the status of the reactor concept from 2015, (EPRI 2015) is undertaking an investigation into the fundamentally important hazards of the LFTR. The following hazards are particularly relevant for the LFTR: Regarding the reactor vessel and the containment, the loss of the molten salt in the breeding blanket, the breakage of one or two graphite tubes, the inflow of contamination, unexpected isotope ratios in the fuel salt and the release of fission gases. When processing fuel salt, the possibility of hydrogen reactions in chemical reprocessing must be taken into account. When reprocessing the blanket salt, insufficient separation of protactinium or uranium can occur. Other safety-relevant systems are discussed in the technology chapter for MSR in Chapter **4.4.6** and are not specific to the LFTR.

The reactivity is controlled by a series of control rods that are guided in the breeding blanket in channels of the graphite elements. They are completely removed during normal operation.

If the blanket salt is lost, the reactivity in the reactor core would be increased due to the strong neutron-absorbing effect. The control rods would then have to retract into their graphite elements and thus compensate for this positive reactivity input by supplying negative reactivity. To do this, it is envisaged that the control rods in the blanket will “float” in the salt so that they lose buoyancy when liquid is lost and automatically and passively retract into the reactor and supply negative reactivity. These control rods can also be used in other operating states for the controlled shutdown of the reactor, for which they would also have to have an active control system. Another series of more conventionally active control rods in the centre of the reactor core with a lower reactivity effectiveness will be used for reactor control. Alternatively, helium could also be used to displace fuel in a chamber within the core of the reactor. More detailed modelling of the reactor control is not yet available (EPRI 2015).

An important safety function of the reactor is part of the primary circuit and is activated if the reactor overheats or the cooling flow is interrupted. In this case, the so-called “freeze plug”, which is actively cooled, is melted and the fuel liquid is passively drained from the primary circuit and the reactor vessel into drain tanks. The drain tanks have an independent cooling system that is passively connected to a heat sink and cools the fuel salt.

If there is a failure in the heat exchanger to the turbine in the intermediate cooling circuit, the pressure relief valves would open and the coolant salt would flow out of the intermediate circuit to prevent damage from overpressure. As a result, the primary circuit is no longer cooled, which would lead to the freeze plug melting and thus the fuel salt being drained into the drain tanks (EPRI 2015).

The safety functions of the reprocessing plant include the handling of highly radioactive materials, their storage and their re-feeding or conversion into stable chemical forms. Cooling systems and drain tanks are provided for each process step. Gaseous substances such as fluorine and hydrogen, which are required for reprocessing, are highly reactive and inventories should be as low as possible to avoid safety risks. The pyrochemical processes in molten salts provided for this purpose have the advantage over reprocessing technologies in aqueous solutions that the possibility of criticality accidents is significantly reduced due to the lower moderating effect of the molten salt compared to water (EPRI 2015).

Different fission products must be handled differently. Many of the fission products form stable chemical compounds with the molten salt and remain there. Gaseous fission products such as xenon, krypton and tritium must be removed from the molten salt and suitably conditioned. A class of fission products such as selenium and tellurium migrate with gaseous hydrogen and hydrogen fluoride and are neutralised in a system with potassium hydroxide. Other fission products are removed from the fuel and are converted into a metallic form in bismuth and later further conditioned. (EPRI 2015) They cite the advantage that no large quantities of the substances would be produced, the handling of which would endanger reactor operation.

Maintenance of the LFTR is one of the main challenges due to the high temperatures and high radiation. The surfaces of the reprocessing plant must be regularly maintained by remote control. How the radiation exposure of structural materials turns out and how often replacement is therefore necessary depends on a number of factors and, according to (EPRI 2015), cannot yet be reliably assessed. It depends on the thickness of the blanket and on the reprocessing plant, since the breeding material is in contact with the reactor vessel. A thinner blanket means that the surfaces of the reactor vessel are exposed to a higher neutron flux and activation products can form. A low reprocessing throughput leads to more protactinium decaying to uranium in the breeding blanket, resulting in more fission and the production of fission products in the breeding blanket. It is therefore not clear what the lifespan of the reactor vessel will be. If the lifespan of the reactor vessel is too short, several spaces for reactor vessels connected to the primary heat exchanger must be provided in the reactor hall. This means that reactor vessels can be taken out of service while another one comes into operation. After a cool-down phase, the irradiated reactor vessels could then be serviced. The graphite in the reactor core is also scheduled to be replaced every four years. The pumps in the primary circuit and the primary heat exchanger will, however, need to be replaced over decades of operation in any case (EPRI 2015).

In principle, a fuel based on thorium has higher melting temperatures and better thermal conductivity properties than a uranium fuel. This is beneficial with regard to the behaviour of the fuel in the event of an accident. The lower build-up of transuranic elements also leads to a lower radioactive inventory, which has a positive effect on possible releases.

Conclusions on safety

The general safety properties of MSR with liquid fuels discussed at the technology-line level apply to the LFTR, as do the same advantages and disadvantages with regard to reactivity control, residual heat removal and the spectrum of events (see Chapter **4.4.6**). With regard to releases in the event of accidents and incidents, the additional use of a reprocessing plant in the LFTR offers an advantage due to the reduction in the radioactive inventory and the separate storage of fission products. However, the segregation of separated fissile material and the complexity of the plant in particular open up further possibility of release and should therefore be considered more of a disadvantage.

This does not result in any significant differences to the assessment in the technology line.

Some of the accident sequences that occur with the LWR cannot occur with the LFTR. However, the LFTR is also associated with specific incident and accident sequences, as well as the possibility of releasing large amounts of radioactivity into the environment.

Overall, there are therefore no significant advantages or disadvantages specific to the LFTR compared to current LWR.

5.5.3 Supply and disposal aspects

Thorium is present in small concentrations in the earth's crust, but its abundance is about three to four times that of uranium (BGR 2016).

(IAEA; NEA 2016) estimates the deposits of thorium that are already known or assumed to exist at 6.2 million tonnes.[188] This value, which is comparable to the uranium deposits, must be viewed against the background that there is currently no economic interest in developing thorium deposits of their own, rather such developments are only carried out in connection with the extraction of other raw materials.

If natural uranium consumption remains at the current level of around 59,200 tonnes per year (NEA 2020b), then the known and suspected reserves of 6.15 million tonnes of uranium will cover global demand for over 100 years. If the quantities that can be mined at a price of up to 260 US dollars per kilogram are taken into account, this will even cover 135 years.[189] According to (NEA 2020b), if progress in the exploitation of uranium resources is estimated, taking into account all conventional reserves (including as yet undiscovered but forecast quantities), then uranium deposits could even last 250 years. Only if nuclear energy were to be expanded drastically, would the resources be used up much sooner or could only be mined at significantly higher costs.

Potentially, much larger quantities of uranium exist; for example, the quantities of uranium dissolved in seawater are estimated at 4 billion tons. So far, however, it has not been technically possible to extract such quantities at economically feasible costs (IAEA; NEA 2016).

Based on these figures, (BGR 2016) also concludes that, from a geological perspective with regard to raw materials, there is sufficient potential to ensure a long-term global supply of uranium.

If one nevertheless assumes that, in the context of very long-term or rapidly increasing future use of nuclear energy, there could be a need to breed additional fissile material, this would in principle be possible with both uranium (by breeding plutonium) and thorium (by breeding uranium-233). This means that even if very long-term use of nuclear energy at a high level is assumed, there would be no need to use thorium as fuel.

An exception, however, are countries like India. India has been pursuing research programmes for some time with the aim of using thorium fuels. The background to India’s interest in thorium is that

[188] More recent reports on uranium resources (NEA; IAEA 2018; NEA 2020b) no longer contain aggregated information on thorium deposits.

[189] In order to actually extract the uranium deposits, the cost must not exceed certain limits. In order to assess how economical and therefore how likely the exploitation of a uranium deposit is, the deposits are divided into cost groups, starting with the category “less than 40 US dollars per kilogram of uranium”. For a long time, extraction costs of up to 130 US dollars per kilogram of uranium were considered the highest cost group. In 2009, an additional cost group of up to 260 US dollars per kilogram of uranium was introduced, as the limit of 130 US dollars for short-term supply contracts was temporarily exceeded in 2007 and 2008, and the costs of extraction have tended to rise.

India has relevant thorium deposits, but practically no relevant uranium deposits. Since India is not a member of the Nuclear Non-Proliferation Treaty, its ability to access international uranium reserves was severely limited for a long time. Despite this high level of interest in large-scale implementation of thorium use compared to other countries, this has not yet been implemented commercially in India.

A key advantage of thorium-uranium fuel is usually the significantly lower build-up of transuranic elements, particularly plutonium, and the associated advantages in terms of waste-related radiotoxicity. Actinide production is lower when thorium is used, but significant amounts of the long-lived protactinium-231 are also produced. Overall, however, the radiotoxicity of thorium-uranium fuel in the spent fuel is significantly less than that of conventional uranium fuel. As discussed in more detail in (Oeko-Institut e.V. 2021), however, this is a misleading argument with regard to the issue of waste, since radiotoxicity is an unsuitable benchmark with regard to disposal in a repository mine.

(NEA 2015, S. 29) also determines this with regard to thorium-uranium fuel. Accordingly, the impact of actinides on the safety proof of a potential repository is small compared to the long-lived, mobile fission products. The main influencing factor with regard to the difficulty of conducting a long-term safety analysis is not so much the type of fuel or reactor system used, but the total energy generated as a parameter for the amount of fission products produced.

With regard to the production of long-lived fission products, (NEA 2015, S. 100) does not observe any relevant differences between a uranium-plutonium and a thorium-uranium fuel.

Conclusions on supply and disposal

Given the uranium resources available today, from the perspective of resource availability there is no need to use thorium as an alternative fuel in new reactor concepts.

In the context of very long-term or rapidly increasing future use of nuclear energy, there may be a need to breed additional fissile material. This would, in principle, be possible with both uranium (by breeding plutonium) and thorium (by breeding uranium-233). This means that even if very long-term use of nuclear energy at a high level is assumed, there would be no need to use thorium as fuel.

When it comes to disposal, the problems discussed in Chapter **4.4.7** apply to fluoride salts as fuel. Due to reprocessing in the LFTR, a number of special waste containers must be developed for repository. The use of thorium instead of uranium as fuel reduces the actinide content in the waste, but does not bring any significant advantages in terms of long-term safety for the final disposal of radioactive waste in geological formations.

Overall, therefore, no clear advantages or disadvantages of the LFTR compared to LWR can be identified in this field either.

5.5.4 Proliferation risks

The uranium isotope uranium-233 is produced as a fissile material in the thorium-uranium fuel. Due to its properties (low critical mass in the range of 15-30 kg, low fraction of spontaneous fission and thus low background of interfering neutrons, long half-life), uranium-233 is a very good material for nuclear weapons (GIF 2011). In addition to the fissile materials uranium-235 and plutonium, as well as natural uranium as breeding material, thorium also falls under the IAEA’s safeguards.

With regard to the proliferation resistance of uranium-233, an important property is that the isotope uranium-232 is always produced in thorium fuel. Uranium-232 decays with a half-life of 69 years,

with various very hard gamma emitters (gamma energy of 2.6 MeV) being formed in the decay chain. This hard gamma radiation is difficult to shield.

Depending on the duration of the irradiation of the fuel, the fraction of uranium-232 in the total uranium produced can range from a few ppm to several thousand ppm. The strong gamma emitters in the decay chain of uranium-232 build up to equilibrium concentration within a few months (after the uranium has been separated). In freshly separated uranium, the radiation from the decay products of uranium-232 is therefore still low. According to (GIF 2011), 1 g of uranium-232 increases the decay rate of the strong gamma emitters by 0.3 GBq per day over the first three months. An equilibrium is eventually reached after a few years, when the decay rate is 270 GBq.

This makes it possible, on the one hand, to detect even small amounts of uranium as part of fissile material monitoring. On the other hand, larger amounts of uranium also result in a relevant dose rate, so that handling and processing of the material is only possible under highly shielded or remote conditions.

(NNL 2010) states that the proliferation risks associated with the use of uranium-233 should be assessed as comparable to the risks associated with handling highly enriched uranium, and at least comparable to those of a uranium-plutonium fuel concept. The physical protection provided by the presence of uranium-232 is generally overestimated; only under very specific conditions can a thorium-uranium fuel concept have a higher proliferation resistance than other fuel concepts. For an in-depth discussion of the influence of this “self-protection” on the proliferation resistance of various conceivable fuel and branching scenarios, see for example (NEA 2015, Kap. 9.2).

Another aspect to be taken into account is that, for any reactor using thorium-based fuels, the addition of a fissile material (either highly enriched uranium or plutonium) is mandatory for initial operation. The extraction and provision of the quantities of fissile material required for this must be included in a complete discussion of the proliferation risks posed by a specific thorium-uranium fuel concept.

Conclusions on proliferation

With regard to proliferation aspects, these would depend very much on the specific technical design of the thorium fuel use and the reprocessing plant.

There are no clear advantages or disadvantages of the LFTR compared to the MSR technology line (see Chapter**0**) with regard to possible thorium use.

In comparison to LWR, enrichment can be dispensed with, but separated fissile material and the reprocessing of the thorium breeding blanket are required to start the reactor.

Overall, therefore, no clear advantages or disadvantages of the LFTR compared to LWR can be identified in this field either.

5.5.5 Technological development status

The authors of (EPRI 2015) still see a need for considerable development effort for the long-term operation of a chemical reprocessing plant, especially with regard to the maintenance of reaction chambers and pipes due to the high radiation and high temperatures.

A detailed assessment of the development status of LFTR with a 10-stage TRL evaluation scheme was presented in (EPRI 2015). The evaluation relates to the essential technical components of the reactor vessel, the primary circuit and the chemical reprocessing plant. Other development areas

essential for the TRL (operational requirements, control technology, safety functions and detection procedures) are given only scant consideration. Similar to what was shown in Chapter 4.5, the authors find it difficult to classify the reactor concept due to its early development stage, as design decisions have not yet been made.

The authors come to the conclusion that some of the components only reached Level 3, i.e. no component-specific laboratory experiment has yet taken place in their TRL scheme. Based on previous experience with the MSRE experiment, these are mainly components of the chemical reprocessing plant, for which they assign TRL 3 and below, but also the emergency shutdown systems. Several components have a TRL of 3-5. In particular, structural parts such as reactor vessels, pipes, heat exchangers and pumps are rated up to TRL 6. In general, the authors also warn against relying too heavily on previous experience from the MSRE and MSRB projects (see Chapter **4.4.3**). The projects and the knowledge are several decades old and the possibility of repeating certain proofs for technical components may be limited. For example, validated software tools may no longer be available, standards may have changed or certain required materials may no longer be manufactured (obsolescence).

Overall, (IAEA; NEA 2016) concludes that, despite previous experience with solid thorium fuels for the LWR and heavy water reactors, these are currently not suitable for commercial use. According to (NNL 2010), the basic suitability of thorium-uranium fuels for use in today’s light or heavy water reactor concepts is basically understood, but would need to be further demonstrated before actual commercial use. This would require further intensive research and development work with a corresponding massive financial commitment over at least 10-15 years. The use of such fuels in future reactor concepts, on the other hand, would require a much longer development phase. The MSR would therefore require, at the very least, a similar level of development effort.

Finally, for the long-term use of thorium fuels, the uranium-233 would have to be recovered from the spent fuel. The required reprocessing technology for thorium fuels is also not yet available worldwide.

Overall, (NEA 2015) determines that the use of thorium as an alternative fuel has received considerable attention in recent years, including in the general media. The reason for this is various thorium interest groups that present thorium as a ready-to-use, problem-free solution to the world’s energy problems. Regardless of the potential use of thorium and any associated benefits, the mostly unbalanced and unscientific portrayal of thorium in the media can only complicate the debate, among decision-makers and the general public, about the actual opportunities and risks associated with the use of thorium.

(NNL 2010) determines that the technology surrounding thorium use is technically innovative, but not yet mature. Due to significant technical and financial risks, without any discernible advantages, thorium is currently not of interest to operators of nuclear power plants.

Conclusions on technological development status

Compared to other concepts in the MSR technology line (see Chapter 5.4), the development of the LFTR still requires a great deal of effort, but there is also historical experience from the MSRE and MSRB of ORNL. The decisive step for the LFTR, in addition to the development of some essential components (e.g. gas turbine, control rod channels, freeze plug), is the development of chemical reprocessing, as this is indispensable when using thorium fuel.

The development of the LFTR can be classified overall as coming under ”applied research”.

5.5.6 Implementation

5.5.6.1 Planned area of application

Similar to other MSR projects, the LFTR should also be usable for high temperatures of up to 700 °C. This increases the efficiency of power generation and opens up the potential for alternative markets such as process heat or heat storage. Different configurations should also be possible. A special feature of the LFTR, in contrast to other MSR concepts that are under active development, is the use of a gas turbine. The waste heat from a gas turbine still has sufficient temperatures to supply a desalination plant.

5.5.6.2 Project timeline

The LFTR concept from the start-up company Flibe has been in development since 2011.

In 2018, Flibe Energy, together with PNNL, was able to raise US$2 million in research funding from the DoE to investigate the properties of nitrogen trifluoride for fluoridation and removal of uranium from molten salt.[190]

The LFTR is listed in the ARIS database (IAEA 2016e).

Conclusions on project progress

Information on project progress is sparse. No reliable statements can be made on the progress of the project.

5.5.6.3 Costs

The economic properties of the reactor should be improved by achieving significantly better efficiencies through the use of high temperatures. The LFTR design is to be coupled with a supercritical CO2 turbine system that works with the Brayton cycle. This should achieve efficiencies of up to 45-50%. The developers state, similar to other systems with performance in the range of the SMR, that modularisation can reduce design and construction costs, based on light water technology in similar fashion to the SMR (EPRI 2015).

According to the developers, it is also possible to scale the LFTR from the size of a non-power producing microreactor up to larger outputs in the gigawatt range, each with an economical and sustainable fuel cycle (ORNL 2021).

In the system description of the LFTR from 2013 (Sorensen 2016), mentions a cost study that Flibe Energy carried out together with its partner Teledyne Brown Engineering under the auspices of the Electric Power Research Institute and Southern Company Services (SCS). This is not publicly available. It is demonstrated in (IAEA 2016e) that the economic performance of the LFTR has not yet been modelled in sufficient detail, but that all indicators suggest that it is highly competitive with other energy sources.

With regard to the use of thorium as a fuel, cost differences in fuel supply and disposal are not of decisive importance for the total electricity generation costs in new reactor concepts. There is also

[190] https://www.energy.gov/ne/articles/us-department-energy-provides-nearly-20-million-domestic-advanced-nuclear-technology

practically no reliable information available to date on the actual costs of a commercial thorium fuel concept.

(NEA 2015) discusses, qualitatively, the cost contributions of the various processing stages in the production of thorium fuel compared to uranium-plutonium mixed oxide fuel (MOX) and comes to the conclusions that

- fuel production is comparable in terms of effort to that of current MOX fuels and thus about three to five times as expensive as that of a pure uranium fuel,
- the reprocessing of the spent fuel to provide the fissile material for the new fuel production is at least as expensive as the current reprocessing of spent uranium fuel and
- no relevant differences should be expected with regard to the costs of final disposal of the remaining highly radioactive waste when comparing the two fuel concepts.

Since the LFTR uses liquid fuel, the statements regarding fuel production cannot be compared directly. However, fuel reprocessing represents an additional cost factor. With regard to disposal, it should be noted that no suitable conditioning of the fuel for subsequent repository has yet been developed and, therefore, no cost statements can be made in this regard.

(NNL 2010) comes to the conclusion that, although small cost advantages compared to other fuel concepts can be achieved through the use of thorium, these cost advantages in fuel supply are not of relevant importance, given the small share of fuel costs in the total levelised cost of electricity.

Conclusions on costs

The supply and disposal of fuel in current LWR only accounts for a small share of the levelised cost of electricity, see also Chapter **2.5**. Significant cost advantages, e.g. through the use of thorium, compared to current LWR cannot therefore be expected on the basis of these differences.

Further reliable statements on the cost structure cannot be made at the current stage of development.

Massive investments amounting to several billion USD are necessary to further develop the LFTR. The willingness of private investors to bear the risks will depend, essentially, on whether the developers succeed in convincing other investors and obtaining further state funding.

5.6 MSR: MCFR (molten salt reactors with fast neutron spectrum)

The Molten Chloride Fast Reactor (MCFR) is a reactor concept from the technology line of molten salt reactors (MSR). The following information on MCFR is based on (Krepel and Kramer 2021), unless otherwise stated. A MCFR concept has been in development by the US company TerraPower since 2015. The TerraPower company has only provided sparse information on its reactor concept. This should be referred to, if the special reactor concept of the TerraPower company is discussed.

The MCFR of the TerraPower company was selected to be a DoE pilot project, as part of the Advanced Reactor Demonstration Programme (ARDP), and an experimental pilot project, the Molten Chloride Reactor Experiment (MCRE), will be built first. The reactor should ultimately offer low costs, high safety, and a reliable, sustainable energy supply.[191]

The advantages of an MCFR can be seen in the fact that the chloride salt enables high solubility of uranium and/or transuranium elements. Like all MSR, it can be implemented as a low-pressure system because the molten salt does not boil at operating temperatures. Structural materials are absent in the reactor core, as no moderation is required and the fuel also serves as a coolant. The high flexibility of the reactor system should make it possible to burn actinides, as well as breed new fissile materials with a range of different fuel compositions, and, depending on the fuel used, it should be possible to avoid enrichment or reprocessing of actinides, thus increasing proliferation resistance.

5.6.1 Description of the plant concept

The MCFR is an unmoderated reactor with a fast neutron spectrum, see Chapter **4.4.2**. The MCFR from TerraPower should have an electrical output of 800 MW. MCFR can, in principle, also be operated with different cooling systems, as homogeneous MCFR with external indirect cooling (concept of the MCFR from TerraPower), but also as heterogeneous MCFR with internal indirect cooling in the reactor core. There are also designs with one or more actinide-containing molten salts.

While the spectrum of fluoride-based MSR is rather more thermal due to the use of moderators or of moderating elements with a low atomic number, such as beryllium, lithium and fluorine, the spectrum of the MCFR is one of the hardest spectra of all fast reactors. The fast neutron spectrum enables in-situ breeding or the use of transuranic fuels (TRU fuel) for transmutation. In the MCFR from TerraPower, the hard spectrum is intended to make the addition of fissile material, after the initial nuclear charge, and the separation of fissile material from the spent fuel, quite unnecessary.

MCFR can be used as breeders or burners in open or closed fuel cycles. Uranium-plutonium, thorium-uranium or a combination with TRU can be used as fuel in the MCFR. With U-Pu fuel, reprocessing may be dispensed with due to the hard spectrum. Such operation without reprocessing was proposed as early as the 1950s and presented again by (Qvist 2021). Since reprocessing during operation is one of the major technical hurdles in MSR concepts, the commercial interest of companies such as TerraPower is probably also due to the fact that it is fundamentally possible to operate the MCFR without reprocessing. The fuel burn-up should reach 183 GWd/MT in once-through operation and up to 334 GWd/MT in twice-through operation, which would correspond to a burn-up of 36%.

[191] https://www.neimagazine.com/news/newsterrapower-and-southern-company-to-demonstrate-molten-salt-reactor-9513910

The basic concept (baseline) of the company TerraPower envisages a U-Pu fuel with the addition of natural uranium or spent uranium during operation (Latkowski 2021). The basic concept is the long-term development of a plant with several 100 MW or GW of electrical power that is physically large enough to carry out in-situ breeding with U-Pu fuel and possesses a sufficiently high breeding rate due to low neutron leakage. The reactors would initially be operated with fresh HALEU fuel with 12% enrichment, the fraction of plutonium in the fission would then be increased over time (Latkowski 2021). Initially, the first commercial reactors (10-100 MWe) will be operated with HALEU as direct successors to a demonstration reactor. An advanced option envisages a “Pu/waste-burning machine” that uses fuel material that has been separated in other MCFR or reprocessed LWR fuel that is chlorinated and separated (Latkowski 2021).

The fraction of fissile materials in the fuel is typically over 10% of the heavy metal in the reactor. The hard neutron spectrum also makes the reactor less sensitive to the build-up of fission products than thermal the MSR, so there is less need to filter out unwanted fission products from the fuel. For reasons of neutron balance, thermal reactors can only be operated with fluoride salts and moderators with low absorption. The TerraPower MCFR will not require reprocessing during operation (online reprocessing) using pyrochemical processes. Lanthanides are not removed from the salt substrate, noble gases and other volatile fission products are flushed out, while precious metals and other insoluble fission products are mechanically filtered (Latkowski 2021).

A molten chloride salt will be used as fuel, since chloride salt enables high solubility of uranium and/or transuranium elements and has a harder neutron spectrum, even if the reactor core is much larger than when using fluoride salt. Several chloride salts are viable here. The most suitable salts are based on sodium, potassium, magnesium or calcium, and the TerraPower patent lists BaCl2, $SrCl_2$, VCl_3, $CrCl_3$, $TiCl_4$ and $ZrCl_4$. (Krepel and Kramer 2021). NaCl-$MgCl_2$ will be used in TerraPower’s MCFR (Chisholm 2021). An important property of chlorine is that it has two stable isotopes. In natural chlorine, Cl-35 occurs at 76% and chlorine-37 at 24%. The neutron capture cross-section of chlorine-35 is, however, significantly higher than that of chlorine-37. In MCFR concepts, chlorine-37 must therefore be enriched to higher purity levels (>75%). In thermal reactors, the neutron capture of chlorine-37 is so high that a Th-U fuel cycle without a continuous supply of fissile material would be out of the question, which also favours a fast spectrum.

In an MCFR with a fast spectrum and the associated low neutron absorption and scattering rates in the salt substrate, the neutrons have a comparatively high mean free path. The homogeneous reactor core is therefore relatively transparent to the neutrons, which leads to high neutron leakage. This necessitates a blanket or reflector in the reactor core, or a high fraction of actinides in the fuel may be an option.

The basic concept of the reactor is a pool with the molten salt fuel. There should be four primary and 4 secondary cooling circuits with tube bundle heat exchangers. The reactor vessel contains 8 parallel circuits, each with its own pump and heat exchanger, which feed the central pool. Further details on possible reactor geometries from TerrPower are not yet publicly available, but (Mausolff et al. 2021) provide a historical overview of some previous MCFR designs and calculate possible reactor geometries and necessary fuel compositions.

5.6.2 Safety concept and safety features

In principle, the MCFR does not differ in terms of safety properties from the representation of fast MSR with molten chloride salt (liquid fuel) at the level of the MSR technology line (Chapter **4.4.6**). Very little concrete information is available on the plant design of an MCFR from TerraPower, for example in (ORNL 2021). After examining historical MCFR designs, the authors in (Mausolff et al. 2021) found that no meaningful safety analysis can be carried out with the available design data. The authors took this as an opportunity to calculate an initial design, which allowed an analysis of the basic safety properties of an MCFR to be studied. To do this, they calculated different geometries for the reactor vessel and the reflector, fuel compositions, power profiles, cooling requirements and reactivity coefficients using simulation codes for the equilibrium case.

In 2016, the MCFR team conducted a preliminary PIRT analysis (Phenomena Identification and Ranking Table), but the relevant document is not publicly available (MCFRG-INNOV-RPT-0002, Rev-0) (ORNL 2021). However, the results have been incorporated into a general study of possible releases from MSR in (ORNL 2021) (see Chapter **4.4.6**).

Reactivity control in TerraPower’s MCFR is achieved by active control rods and feedback through the negative temperature reactivity coefficient.

First calculations on radioactive source terms for a complete release from an MCFR were presented in (Wheeler et al. 2021). Since the TerraPower MCFR is designed without pyrochemical reprocessing (actinides), this does not constitute a possible release path.

Other potential release pathways must also be given due consideration, when separating volatile or metallic fission products. Therefore, the barrier principle of TerraPower's MCFR includes the vessel, the containment and the building of the reactor and the fuel processing plant. This so-called Functional Containment System (FCS) is a new concept that assumes multiple barriers acting together to effectively prevent the physical transport and release of radionuclides into the environment during normal operation, incidents and accident conditions. The FCS is suitable for ensuring that the dose at the boundaries of the plant, as a consequence of a postulated accident, complies with the regulatory limits (Krepel and Kramer 2021).

Conclusions on safety

Some of the accident sequences that occur with the LWR cannot occur with the MCFR. However, the MCFR is also associated with specific incident and accident sequences, as well as the possibility of releasing large amounts of radioactivity into the environment.

The general safety properties of MSR with liquid fuels discussed at the technology-line level apply to the MCFR, as do the same advantages and disadvantages with regard to reactivity control, residual heat removal, releases in the event of incidents and accidents, and the spectrum of events (see Chapter **4.4.6**).

Overall, there are therefore no significant advantages or disadvantages specific to the MCFR compared to current LWR.

5.6.3 Supply and disposal aspects

With regard to the fuel supply, the key aspect to mention is the need for enrichment of chlorine-37. When using a U-Pu fuel, however, the same degree of purity is not necessary for neutronic reasons and, compared to Th-U, only a reduction in the enrichment of chlorine-37 by one order of magnitude is necessary (Forsberg 2021). Chlorine enrichment has not been researched much to date, but in principle it would be possible with the gas centrifuge technology used for uranium enrichment. A monatomic gas such as HCL or methyl chloride (CH_3Cl) would be preferable as a carrier gas, since chlorine gas itself would require a three-component separation (Cl^{35}-Cl^{35}, Cl^{35}-Cl^{37}, Cl^{37}-Cl^{37}). (Forsberg 2006).

In equilibrium operation, only natural or depleted uranium is intended to supply the MCFR. In the MCFR, uranium utilisation of up to 35% should be possible, about 60 times as much as in a LWR. For a transitional period or for the start of a new MCFR, fuel with a higher fissile material content in the range of 10% or more is required. For the start, however, the use of pre-used MCFR fuel (e.g. surplus from another reactor) should also be considered, instead of disposing of the fuel. (Greenspan 2021).

There are a number of disposal options for the fuel from TerraPower’s MCFR. Three variants are presented in (Latkowski 2021). Either direct disposal in a salt matrix without recovery of chlorine-37. Such storage would not be chemically stable and would therefore depend on the geological effectiveness of the enclosure. Another disposal option would be conditioning by oxidising the chlorine to chlorine oxides and thereby separating chlorine-37 and then conditioning metal oxides in SynRoc, a synthetic rock made of titanate minerals. The separation of chlorine from the waste stream could be necessary simply because chlorides as waste forms have a low retention capacity since chlorides dissolve in water.

It would also be possible to recover the enriched chlorine and vitrify the remaining waste, for example in iron phosphate. According to (Latkowski 2021), all disposal options still require considerable research and development effort.

(Latkowski 2021) assume that when the waste is vitrified, either the minor actinides and fission products are separated or, alternatively, the entirety of the liquid waste is vitrified. The feasibility of vitrifying all the waste has not yet been demonstrated, and the waste cannot be reused as fuel.

Overall, (Latkowski 2021) finds that the disposal aspects have not yet been resolved, but that interesting options exist.

Conclusions on supply and disposal

The possible absence of uranium enrichment and the high utilisation of uranium represent an advantage of the MCFR compared to the technology line and current LWR. However, the enrichment of chlorine-37 is required to operate an MCFR.

It is also the declared aim of the MCFR to be able to forego the reprocessing of spent fuel. This represents an advantage over the MSR technology line, but corresponds to the open fuel cycle that is common today in LWR, so that this does not result in any relevant difference compared to current LWR.

For disposal, the waste from the MCFR would have to be suitably conditioned and the disposal options still need to be developed. This represents a disadvantage compared to the direct final disposal of the LWR.

In this context, it cannot be assumed that a MCFR will have any relevant advantages over current LWR in terms of disposal.

5.6.4 Proliferation risks

The proliferation risks of the MCFR differ only slightly from other MSR without reprocessing (see Chapter **4.4.8**). In the TerraPower concept, however, only natural or depleted uranium is fed into the reactor when it is in equilibrium. This increases proliferation resistance. However, an inventory of fissile material in the form of plutonium or HALEU is required to start the MCFR.

(ORNL 2020) offers detailed considerations on the safeguards and monitoring measures based on the MCFR design by TerraPower, which also apply to other liquid reactor systems. According to the report, a number of existing IAEA safeguard technologies are suitable for monitoring, especially for fresh fuel. However, research and development is needed for irradiated fuel. For example, no technologies are known to directly measure the content of fissile material in irradiated fuel salt. Traditional safeguards such as containment and surveillance will probably also have to be modified to cover multiple paths of fissile material flow in the MSR. At the same time, the monitoring technologies must also withstand the extreme radiation conditions. Since there are no isolated fuel elements, an MSR will be treated in a similar way to other systems in which larger quantities of fissile material are present in different process sequences, such as reprocessing plants (bulk vs. item). There is a constant supply and removal of fissile material within the various process loops. At the same time, the fuel composition changes permanently even during equilibrium operation in the reactor. The changing chemical composition of the carrier salt also influences the type and quantity of actinides and the fissile material production. These properties make material balancing difficult (ORNL 2020).

The authors of (ORNL 2020) also come to the conclusion that the high radiation in the reactor hinders the application of monitoring measures, many areas will be inaccessible to inspectors. At the same time, however, this also hinders the possibility of accessing the fissile material and simplifies the application of safeguards.

Another problem, according to (ORNL 2020), is the whereabouts of fissile material in parts of the system. This makes it difficult to determine whether material has been diverted or simply wasn’t accounted for properly (material unaccounted for - MUF). One advantage here, compared to systems such as a large reprocessing plant, is that the total amount of fuel in the system is small and there is no high throughput. Therefore, the overall inventory can be better controlled and uncertainties can be kept small. Good models and computer simulation tools are also required for accurate material balance control and must be developed in parallel.

One advantage of the MCFR is that it is designed without separating fissile material during operation, thus avoiding the proliferation risks of separated fissile material and reprocessing technology. (ORNL 2020) point out, however, that the fraction of fissile material in the fuel is significantly higher than in LWR.

One advantage of the MCFR is that there is only a small amount of excess reactivity in the reactor, so that the diversion of fissile material would be noticeable due to the decrease in reactivity, or the residual reactivity would be insufficient to operate the reactor (Krepel and Kramer 2021).

In (Latkowski 2021), a report on the proliferation risks of a reactor concept in the pre-licensing phase is mentioned (LLNL 2017), the report is not publicly available.

Conclusions on proliferation:

It is the stated aim of the TerraPower company to avoid fuel reprocessing with the MCFR. This represents an advantage over the MSR technology line, but corresponds to the open fuel cycle that is common today in LWR, so that this does not result in any relevant difference compared to current LWR.

During equilibrium operation, only natural or depleted uranium will be fed into the TerraPower company’s MCFR. This constitutes an advantage over other MSR concepts, in which fissile material must be fed continuously. This is also an advantage over the LWR, as no enrichment is necessary. Furthermore, fresh fissile material is required when the reactor is put into operation.

When using safeguards, new approaches and technologies for monitoring fissile material and material balancing must be developed as the safeguards themselves become more complex. This poses a disadvantage compared to the LWR.

Overall, the MCFR has slight advantages over the LWR in terms of proliferation resistance.

5.6.5 Technological development status

Thermal MSR with molten chloride salt reactors were already conceived in the pioneering days of nuclear energy in the late 1950s, but were only pursued theoretically in concept studies due to their poorer breeding properties compared to fluoride molten salt reactors. (Mausolff et al. 2021; Krepel and Kramer 2021) provides a historical overview of the relevant concepts.

A number of recent reports have been written for the further development of the MCFR in collaboration with ORNL and INL, such as a roadmap with irradiation tests, which is mentioned in (Latkowski 2021). There are major gaps in all areas mentioned in Chapter **4.4.5**, which are decisive for development.

With regard to fuel development, historically the data situation for chloride reactions has been poor compared to fluoride reactions (see also Chapter 5.5). It is generally assumed that the melting temperatures for different compositions are similar. Additional studies are needed on corrosion, compatibility with structural materials, redox control, and many other safety and performance characteristics. Regarding the concept of “hard and soft acids and bases”, chlorine is generally softer than fluorine and can therefore cause problems with the stability of chlorine compounds, redox control, and corrosivity (Krepel and Kramer 2021).

The data situation for neutron cross-sections for chlorine-35 and chlorine-37 is also poor. Measurements for the most important cross-section ranges relevant to the MCRE experiment are only now being carried out in collaboration with LANL. The reason for this is that the cross-sections for chlorine-35, for example, differ significantly from currently used evaluated neutron cross-section

libraries such as ENDF/B-VIII.0 (Evaluated Nuclear Data File - ENDF). Accurate cross-section data are crucial for the design and licensing of the MCRE experiment.[192]

Material development also involves producing many components of the system for the first time. However, progress has also been made in recent decades with the MSRE experiment and material research into corrosion-resistant structural materials under hard neutron irradiation. It remains to be determined, however, whether these data, which were mainly obtained for fluoride salt melts, can be transferred to the MCFR.

Experience has yet to be gained with regard to the operational requirements, the control and instrumentation technology, and the safety functions. Experience with the MSRE experiment can also be used here, but the need for an initial small experimental MCRE reactor also shows that these can only be transferred to the MCFR concept to a limited extent.

The detection methods are the most advanced, as the development of simulation codes for development in other technology lines and computer capacities in general enable significant synergies with the development of the MCFR.

Conclusions on technological development status

The development of the MCFR still requires a great deal of effort. The most important next development step for the MCFR is initially the material and fuel development and the measurement of basic physical-chemical properties in the laboratory. At the same time, the technical infrastructure for a system with a molten salt must be developed and the system tested (integrated effects test - IET). The data must be obtained not only from unirradiated materials, but also after irradiation (post irradiation experiments - PIE), which is usually very time-consuming due to the necessary chain of preparation, irradiation and decay. This also requires existing irradiation experiment facilities with the corresponding options.

The development of the MCFR can be classified overall as coming under ”applied research”.

5.6.6 Implementation

5.6.6.1 Planned area of application

Like many MSR concepts, the MCFR should also be usable for high temperatures around 750 °C. This increases the efficiency of power generation and opens up the potential for alternative markets such as process heat or heat storage in the molten salt (TerraPower, LLC 2022b). However, the use of high temperatures also creates new challenges for the molten salt, metallic barriers and reflectors that are exposed to a hard neutron flux (Greenspan 2021).

The electricity supplier Southern Company, which wants to develop the MCFR together with TerraPower, also assumes that the reactor has good properties for both continuous and load-following operation (Chisholm 2021).

[192] https://www.ans.org/news/article-4080/orano-terrapower-get-vouchers-to-study-leu-transport-and-chlorine-chemistry/

5.6.6.2 Project timeline

The timeline that TerraPower is aiming for in 2021 initially envisages separate test benches to investigate different effects (separate effects tests), which will accompany the construction of an integrated test facility (integrated effects test - IET) in 2020-2025 and ultimately lead to the construction of an experimental reactor, the MCRE (Latkowski 2021).

The MCRE is to be built at the Idaho National Laboratory and will become critical in 2025 with either HALEU or uranium-plutonium fuel. Since crucial data for the reactor's chlorine-35 cross-sections are only now being re-measured (see Chapter 5.6.5), it is questionable whether the corresponding schedule can be met. The data is crucial for design and licensing (Latkowski 2021).

The IET was built for USD 20 million and salt production has been successfully scaled up. The experimental data obtained will be used to validate software for calculating the thermal hydraulics for safety proofs for the licensing.

Further development in 2025-2030 will lead to the construction of a demo MCFR reactor, accompanied by a test facility for components. Without scaling up the components, the MCFR will then be developed into a commercial product for the SMR market with 30-300 MW electrical output. Subsequently, from 2030 onwards, a scaled-up “grid-scale” plant with 500-1200 MW electrical output is to be developed as a product (Latkowski 2021).

The time horizon of the individual milestones is constantly shifting. In 2019, construction of the MCFE was still planned for 2022 and the commercial reactor fleet from 2030 (Kramer et al. 2019). In a more recent press release from October 2022, TerraPower and SCS state that the reactor will not be demonstrated until the 2030s.[193]

Conclusions on project progress

The construction of non-radioactive test facilities, such as the IET test bench, was carried out rapidly. It remains to be seen whether construction of the MCFE at the INL will be successfully completed in 2025, as was announced by TerraPower in 2021. Given the development effort still outstanding, the lack of basic data and the need for licensing, this is relatively unlikely. The further development plans with a demo plant in the 2030s should also be viewed with scepticism, even with massive additional investments, given the actual state of development.

5.6.6.3 Costs

TerraPower and Southern Company entered into a partnership in 2015 to develop an MCFR concept as part of a US$45 million DoE grant, the Advanced Reactor Concepts Award.[194] The installation of the IET was part of a further seven-year US$76 million DoE grant to further develop the MCFR system.[195] The MCFR was ultimately selected for further government funding from the DoE's Advanced Reactor Demonstration Programme (ARDP) to build the MCRE in a consortium of Southern Company TerraPower, Idaho National Laboratory, CORE POWER, Orano Federal

[193] https://www.energy.gov/ne/articles/southern-company-services-and-terrapower-build-worlds-largest-chloride-salt-system
[194] https://www.energy.gov/ne/articles/southern-company-services-and-terrapower-build-worlds-largest-chloride-salt-system
[195] https://www.energy.gov/ne/articles/southern-company-services-and-terrapower-build-worlds-largest-chloride-salt-system

Services, the Electric Power Research Institute and 3M Company. The DoE is contributing US$170 million over 5 years, with an 80/20 cost split.

(Latkowski 2021) provide a 2019 updated cost estimate of a large baseline power reactor (nth of a kind – NOAK) with an output of 800 MWe. They calculate the construction costs at US$2.2 billion, US$2800/kWe and the levelised cost of electricity

of US$60/MWh. They see the possibility of cost reductions to <2 billion USD, < USD2500/kWe and < USD50/MWh (Latkowski 2021).

For waste disposal, (Latkowski 2021) states costs of USD1.60/MWh LCOE for 780 MWe over a lifetime of 60 years are given, which are already included in the costs mentioned above. The final disposal costs are expected to be 1 USD/MWh. Decommissioning and dismantling are expected to cost 540 million USD. The costs were calculated according to the Generation IV guidelines (Latkowski 2021).

Cost advantages also arise from the fact that no thick-walled components are required due to the low pressures, e.g. for the reactor vessel, and due to the higher power density, which should enable more compact designs. Uranium utilisation is also significantly increased and, due to the high temperatures, so is the efficiency of electricity generation; marketing process heat will also be possible (Greenspan 2021).

Conclusions on costs

The development of the MCFR to date has been very successful in raising public funds to establish the initial infrastructure for research and development for the reactor. Further cost estimates, especially with regard to construction, operating or decommissioning costs, are currently still speculative without assuming a more specific design, as key features of the cost structure cannot be estimated at this time.

Key advantages of the MCFR in terms of higher burn-ups or the use of TRU fuels, etc. do not represent an advantage over the LWR.

The supply and disposal of fuel in current LWR only accounts for a small share of the levelised cost of electricity, see also Chapter **2.5**. Therefore, significant cost advantages compared to current LWR cannot be expected on the basis of these differences.

Massive investments amounting to several billion USD are necessary to further develop the MCFR. The willingness of private investors to bear the risks will depend essentially on how the MCRE experiment goes and how high further state funding will be.

5.7 SCWR: CSR1000

The Chinese Supercritical Water-Cooled Reactor (CSR1000) is a reactor concept from the SCWR technology line, see Chapter **4.5**. It is being developed under the leadership of the Nuclear Power Institute of China (NPIC). The following description is based on information from (ARIS 2015) and (IAEA 2023d), unless otherwise explicitly stated.

The CSR1000 is said to have significant economic and technical advantages over current light water reactors. On the one hand, these result from the higher efficiency of the plant, and on the other hand from a simplification of the reactor system, since important large components of current pressurised and boiling water reactors can be dispensed with. At the same time, operating the reactor under working conditions above the critical point of water also leads to technical challenges.

5.7.1 Description of the plant concept

The CSR1000 is a reactor cooled and moderated with light water with a pressure vessel and thermal neutron spectrum, see Chapter **4.5.3**.. With a thermal output of 2300 MW, it generates an electrical output of 1000 MW, which corresponds to an efficiency of 43.5%.

The coolant enters the reactor pressure vessel at a pressure of 25 MPa and a temperature of 280 °C and is heated to a temperature of 500 °C as it passes through the reactor. The coolant flow rate through the reactor is 1190 kg/s, which is significantly lower than that of today's light water reactors such as the EPR at 2630 kg/s. This is because of the significantly higher enthalpy increase within the reactor core. The supercritical water produced is fed directly to the turbines via two branches. After passing through a high-pressure, a medium-pressure and three low-pressure turbines, the expanded steam is condensed in the condenser and the resulting condensate is pumped back to the reactor through two loops using feedwater pumps. A turbine that uses a partial flow of the fresh steam is used to drive the feedwater pump.

Uranium dioxide is planned as the fuel, which corresponds to current LWR fuels. With an initial enrichment of 5.6%, a burn-up of 33 MWd/kg of uranium can be achieved, with 6.2% a burn-up of 45 MWd/kg. The cycle duration is then 12 or 18 months. The higher enrichment required compared to current LWR with the same target burn-up is due to the greater neutron losses due to the adapted cladding and structural materials in the core.

The fuel rods have an outer diameter of 9.5 mm. In order to reduce the central fuel temperature and provide a larger volume for the absorption of fission gas, pellets with a central cavity are used. Previous cladding materials would not be sufficiently durable under the operating conditions of the CSR1000. A variant of stainless steel 310S is therefore planned as the cladding material. This is a low-carbon austenitic steel with a high chromium and nickel content. A wire wound twice around the fuel rod serves as a spacer between the fuel rods.

A fuel element consists of four sub-bundles. A sub-bundle consists of 56 fuel rods arranged in a 9x9 grid. The central 25 grid positions remain empty and form a moderator box filled with coolant. The fuel rods are each enclosed on the inside opposite the moderator box and on the outside opposite the other sub-bundles by a fuel element box made of stainless steel 310S. This results in a thermo-hydraulic separation of the coolant flows in the moderator box, in the sub-bundles and between the sub-bundles and between the fuel elements. Cross-shaped control elements, comparable to current BWR control elements, can be used between the four sub-bundles. The four sub-bundles are combined into a fuel element by 4 grids with a height of 30 mm.

An optimised fuel element design for the CSR1000 is reproduced in (Zhu et al. 2021). In particular, the thickness of the moderator box wall and the geometry of the control element have been adjusted.

The reactor core consists of 157 fuel elements and has an active height of 4.2 m and a diameter of 3.38 m. The active height is slightly increased compared to current LWR in order to limit the linear heat generation rate in the core to 15.6 kW/m. The power density in the core is 61.1 kW/kg uranium.

Fresh fuel elements are loaded into the core from the outside and gradually transferred towards the centre of the core in two further cycles. A single fuel element remains in the central position within the reactor core for a fourth cycle.

The central 57 fuel elements form the first fuel element group in terms of coolant flow guidance, the outer 100 fuel elements form the second group. The coolant enters the RPV above the top edge of the core. The largest part of the coolant (94%) fills the upper section of the RPV and from there enters the reactor core through the upper core grid. A portion of 6.5% flows through the moderator boxes of the first fuel element group, a partial flow of 13.7% through the moderator boxes of the second fuel element group, a partial flow of 35.8% serves as a moderator between the fuel elements and sub-bundles of both the first and second fuel element groups and a partial flow of 38% serves to cool the fuel rods in the sub-bundles of the first fuel element group.

A partial flow of 6% of the incoming coolant is directed between the core casing and the RPV wall right to the bottom of the RPV. This partial flow mixes with the coolant that has already passed through the fuel elements once within a mixing chamber below the reactor core. This pre-heated coolant now flows through the reactor core again from below as a coolant for the fuel rods in the sub-bundles of the second fuel element group. The coolant, heated to supercritical temperatures in this way, is collected in a collector above the fuel elements and mixed before it is directed from the RPV to the turbine through thermal sleeves.

This flow guidance ensures that the inner wall of the RPV does not come into direct contact with the supercritical water. This limits the temperature of the RPV inner wall and allows conventional RPV steels to be used. It also ensures that all fuel elements are sufficiently moderated, despite the low density of the supercritical water produced, and that the most even power distribution possible is achieved across the core while, at the same time, providing sufficient cooling

The RPV has an internal diameter 4.855 m and a height of 15.98 m and is structurally comparable to the RPVs of current PWR. The wall thickness is 440 mm and is therefore about twice as strong as that of current LWR. The RPV material is an optimised 508-III steel that is suitable for temperatures up to 350 °C. Internal RPV components ensure the positioning of the fuel elements, the coolant flow and mixing in the areas above and below the reactor core, as well as thermal insulation of the RPV outlet nozzles from the supercritical water.

Reactivity is controlled by control rods, which are inserted into the reactor core from above. To avoid localised power peaks, erbium oxide or gadolinium oxide is used as a combustible neutron poison in the fresh fuel.

5.7.2 Safety concept and safety features

(ARIS 2015) refers, firstly, to the fundamental safety advantages of SCWR reactors, the lack of a phase transition in the coolant, which reduces the risk of a boiling crisis as in PWR, and the use of stainless steel as cladding material, which minimises the formation of hydrogen in the event of serious accidents, see Chapter **4.5.4**.

The safety objective is to significantly reduce the risk to the public, personnel and the environment, which will be achieved by reducing the probability of design basis accidents, events exceeding design-basis, the frequency of core damage and large, early releases. The CSR1000 is to have a lower frequency of core damage and a lower probability of large, early releases than so-called Generation III LWR. In addition, the grace periods for controlling events will be longer. To date, these objectives have only been formulated qualitatively, they are not based on any quantitative specifications, only a target value of 72 hours is specified for the grace period.

To achieve this goal, the focus will be on proven technologies from commercial PWR operations and modern coal-fired power plants cooled with supercritical water. In addition, the defense in depth principle is also being pursued with a combination of inherent properties and active and passive safety systems.

The containment of the CSR1000 is constructed in a similar way to current BWR and is divided into a reactor pit, the steam room and a condensation chamber filled with a coolant supply. Extensive piping systems allow steam to be released from the reactor directly into the condensation chamber. In the event of a loss-of-coolant accident within the containment, the steam is also channelled from the steam room into the condensation chamber through appropriate piping systems, thus limiting the maximum pressure within containment. The cylindrical containment has a diameter of 23 m and a height of 35 m and is designed for pressures up to 0.5 MPa and temperatures up to 145 °C.

In addition to the inherent safety properties of PWR, such as negative void and Doppler coefficients of reactivity, proven safety systems, such as reactivity control and the emergency cooling system, will also be adopted. One fundamental safety requirement is that there is always a sufficient coolant mass flow for core cooling, which must be achieved by feeding in sufficient coolant via the cold feed line and by removing heat sufficiently via the hot line.

If the feed water supply fails, sufficient core feed is initially ensured by a high-pressure feed tank within the containment, similar to the pressure accumulators used in current PWR. The feed pressure is created by a connecting line to the fresh steam line. To ensure simultaneous heat removal, a two-loop automatic pressure relief system is included. The two loops direct the supercritical water from the fresh steam lines and discharge it into a condensation chamber (loop 1) or into the steam chamber of the containment (loop 2). Both loops are secured by a control valve and an insulation valve. After the pressure has been reduced, heat is dissipated via a low-pressure emergency cooling system.

A passive emergency cooling system can remove the decay heat from the reactor at high pressure into the condensation chamber via a heat exchanger. To accomplish this, a closed cooling circuit connects a fresh steam line, via a heat exchanger in the condensation chamber, back to the RPV via a direct feed line. After an isolation valve is opened, this system can passively transfer heat to the condensation chamber.

A passive low-pressure emergency feed system ensures passive feed into the reactor from the condensation chamber after automatic pressure relief or after loss-of-coolant accidents. This ensures core cooling when the active emergency cooling systems are not available.

In the event of a core meltdown, the reactor pit can be flooded with coolant from the condensation chamber through a line that is closed during normal operation. This is intended to cool the RPV from the outside and prevent the corium from being released into the containment.

According to (ARIS 2015), further safety features for exceeding-design-basis events, mitigative emergency protection measures and the design for seismic impacts, have not yet been defined and will be developed in a next design phase. The target value for the seismic design is 0.3 g. Probabilistic safety analyses of levels 1 and 2 also need to be conducted, which is still outstanding.

A number of design faults have already been analysed using programme systems specially developed for the CSR1000. Accordingly, design faults are controlled by the safety systems provided (ARIS 2015). Results of fault analyses for the CSR1000 are summarised in (GIF 2018a). Three-dimensional analyses of control element malfunctions were carried out specifically for the CSR1000. Due to the reactivity control without boric acid in the coolant, transients triggered by control element malfunctions play an important role in the CSR1000 within the reactivity faults. Accordingly, the analysed transients are controlled according to the design without the permissible value for the maximum cladding tube temperature of 850 °C in the core being exceeded (Lianjie et al. 2018).

In addition, studies were carried out into the stability behaviour of the CSR1000 (Shahzad et al. 2018). Accordingly, instability areas need to be avoided, especially when starting up a CSR1000. Additional development of the computer programmes for analysing unstable behaviour in SCWR is therefore necessary; to avoid the occurrence of unstable conditions, automatic real-time monitoring of the operating parameters of an SCWR (pressure, core outlet temperature, power, etc.) should be available.

Conclusions on safety

Overall, the safety properties and the safety concept of the CSR1000, as far as it has been specified so far, are largely comparable to current light water reactors and correspond to the general properties discussed at the technology-line level. There are no significant differences to the assessment at the technology-line level.

Although the developers have formulated qualitative advantages over current LWR as a goal, these have not yet been quantified or proven. In addition, specific statements are still lacking with regard to measures to counter exceeding-design-basis events and mitigative measures.

At the reactor concept level, too, no significant advantage or disadvantage compared to current LWR can be assumed.

5.7.3 Supply and disposal aspects

The aspects identified for the technology line generally apply to the CSR1000: the fuels and fuel concepts are comparable to current light and heavy water reactors. Due to the higher neutron losses in the reactor core, because of the need to use other structural materials, higher initial enrichment is necessary to achieve the same target burn-up. Due to the higher efficiency, however, a reduced fuel requirement is expected to generate the same amount of electrical energy, and thus a correspondingly lower mass of spent fuel per kilowatt hour of electrical energy produced.

A closed fuel cycle is planned for the CSR1000 in order to utilise the fissile materials contained in the fuel as much as possible (ARIS 2015). Other than that, no relevant differences exist compared to the technology line.

Conclusions on supply and disposal

Due to the inherent properties of the technology line, the advantage of reducing the mass of spent fuel per kilowatt hour of electrical energy produced by around 30% can be expected, compared to current light water reactors.

The assumption of a closed fuel cycle can slightly reduce the need for natural uranium or enriched uranium, as well as the amount of plutonium introduced into repository. On the other hand, additional fuel cycle facilities (reprocessing, MOX fuel element production) are required. In this regard, no significant advantage or disadvantage compared to current LWR can be expected at the reactor-concept level.

5.7.4 Proliferation risks

A closed fuel cycle is planned for the CSR1000 in order to utilise the fissile materials contained in the fuel as much as possible.

An analysis of proliferation risks will be carried out in a next design phase (ARIS 2015). Other than that, no relevant differences exist compared to the technology line.

Conclusion on proliferation risks

With regard to possible proliferation risks, the CSR1000 offers no fundamental advantages or disadvantages compared to current light water reactors with regard to the reactor system itself.

The assumption of a closed fuel cycle, however, leads to increased availability of fissile materials that are directly suitable for nuclear weapons, and thus increases the proliferation risks compared to a fuel cycle with direct repository, which represents a disadvantage compared to current light water reactors with an open fuel cycle.

5.7.5 Technological development status

The development of cladding and structural materials that can withstand the high temperatures and pressures in the reactor core is mentioned by (ARIS 2015) as constituting an essential required development step for the CSR1000.

At NPIC, studies are being carried out on the behaviour of various materials under the physical and chemical boundary conditions within a SCWR. Stainless steel 310S is being investigated as a promising candidate for use in the CSR 1000. The future coolant chemistry also plays an important role, since when the coolant transitions from the subcritical to the supercritical state, various impurities remain dissolved in the coolant or can be deposited on internal structures within the core. This can trigger undesirable corrosion processes. Further investigations are required to prove the usability of 310S (Gong et al. 2018). Two variants of 310S stainless steel were irradiated with heavy ions to simulate the possible consequences of radiation when used in the reactor. The experiments showed that even alloying elements with small proportions in the material can have an influence on the material properties (Huang et al. 2020).

The developers also point out that, due to the different operating temperatures, pressures and required loads, the existing control element drives of current LWR must be further developed to meet the requirements of the CSR1000 (ARIS 2015).

The current design, covered in (ARIS 2015), does not yet include any information on measures and facilities for exceeding-design-basis events or for mitigative measures and emergency protection measures. Deterministic safety evidence is only partially available, probabilistic safety analyses have not yet been carried out. Some newer information is now available on this, but the design is not yet complete.

Conclusions on technological development status

The development requirements mentioned at the technology-line level in Chapter **4.5** apply in the same way to the CSR1000.

With a view to the development of cladding tube and structural materials, a specific stainless steel is being tested in laboratory tests for its suitability for use under real operating conditions. The status of material development for the CSR1000 can therefore still be classed as “applied research”, albeit borderline “development”.

With regard to the safety functions required for a complete reactor concept, the developers specifically name, for example, the design of control elements for the specific boundary conditions in the CSR1000.

The design of the reactor concept has not yet been finalised, and only partial evidence of the specific concept is available so far. For the CSR1000, too, the methods for providing evidence for SCWR are still in “applied research” albeit borderline “development”.

Overall, the authors of this report estimate the development status of the CSR1000, in line with the development status of the technology line, to be classed as “applied research”, albeit borderline “development”

5.7.6 Implementation

5.7.6.1 Project timeline

Work in China on an SCWR concept began in 2007 based on a national research programme for fundamental questions of SCWR (Zhang 2016). Basic work on SCWR concepts was already carried out at the Chinese Shanghai Jiao Tong University in 2007. In 2012, the Nuclear Power Institute of China (NPIC) published the CSR1000 concept (Huang et al. 2021).

The work at NPIC was initiated by a national research and development programme with a first phase in 2010-2012, the second phase was completed in 2015. In this framework, the main technical parameters and the design of the reactor core and essential systems and structures were defined. Two test benches were set up at NPIC to investigate the thermo-hydraulic and mechanical properties of SCWR. This allowed investigations at maximum temperatures of up to 650 °C and pressures of up to 25 MPa and a coolant flow rate of 2-5 l/h with a controlled pH value of the coolant (ARIS 2015).

In (ARIS 2015), NPIC states the goal of having a CSR1000 demonstration reactor built by 2027.

According to (GIF 2016a), a new project in Phase II of SCWR development was approved in China, in 2015. The preliminary design of the CSR1000 is intended to be completed in this project. The issues addressed concern the thermo-hydraulic design, safety-related questions of the system design, material-related issues and the development of a test bench for the qualification of fuel elements. An international review of the CSR1000 design is planned for 2018.

Various university working groups have carried out investigations into the safety of the start-up process in the CSR1000 and shown that it complies with the required safety parameters. In addition, various malfunctions were analysed (coolant flow failure, coolant loss malfunctions, reactivity malfunctions) (GIF 2018a; Yuan et al. 2018).

According to (Leung et al. 2018), NPIC proposed designing and building a SCWR prototype reactor, which would demonstrate various SCWR concepts. This prototype reactor should be smaller than a CSR1000, but have sufficiently high performance for the demonstration operation of a turbine operated with supercritical water. This is intended to demonstrate the technical feasibility of concepts such as the CSR1000, the HP-LWR, the JSCWR or the Canadian SCWR. In addition, there are considerations in China to develop a less powerful concept (CSR-150) instead of the CSR1000, which in turn could serve as a demonstration reactor for a CSR1000.

According to (GIF 2020a), work is still ongoing on the preliminary design of the CSR1000. The international design review is now expected to be completed in the phase between 2020 and 2022. According to (Huang et al. 2020), the international CSR1000 design review will take place as part of the research projects started in 2020. The CSR1000 fuel element was optimised within additional works (Zhu et al. 2021).

The Chinese CSR1000 is still in the pre-conceptual design stage; two research projects funded by the Chinese Ministry of Science and Technology (MOST) started in 2020 (GIF 2021a, p. 39). The aim of these projects is to further develop the CSR1000 design. Five Chinese partners are involved: the Nuclear Power Institute of China (NPIC), Shanghai Jiaotong University (SJTU), Xi’an Jiaotong University (XJTU), the China Institute of Atomic Energy (CIAE) and the University of Science and Technology Beijing (USTB).

Conclusions on project progress

The project plans originally envisaged completing the preliminary design of the CSR1000 in Phase II by 2015, conducting an international review in 2018 and building a demonstration reactor by 2027. The design of the CSR1000 has not yet been completed, the international review has not yet taken place and there are no plans for a specific demonstration reactor or actual construction plans.

5.7.6.2 Costs

The reactor is to be designed primarily for a base load supply of electrical energy (ARIS 2015).

With regard to the costs of a CSR1000, the developers refer to the fundamental cost advantages of SCWR compared to current LWR.

Specifically for the CSR1000, the planned service life of the system is given as 60 years, with an operating availability of over 90% being achieved (ARIS 2015, Appendix). There are no further statements on costs specific to the CSR1000.

Conclusions on costs

The inherent advantages of the technology line identified at the technology-line level also apply to the CSR1000.

The developers assume an operating availability of the system of more than 90% for a service life of 60 years. From today's perspective, this does not seem realistic. Firstly, no commercial nuclear reactor has yet achieved a service life of 60 years (IAEA 2022f), and secondly, the average global availability of today's nuclear reactors over the last 20 years has been in a range between 72.2% and 82.6% (IAEA 2022g). However, a service life of around 60 years or more is often specified for current LWR. Although a service life of 60 years or more is often stated for current LWR, SCWR reactors also have significantly higher requirements for the structural materials and, due to high temperatures and pressures, for the cooling circuits as a whole.

5.8 VHTR: HTR-PM

The HTR-PM (high-temperature gas-cooled reactor pebble-bed module) is a high-temperature reactor with a thermal neutron spectrum, see Chapter **4.6**. As a demonstration reactor, the HTR-PM is part of the development of commercial high-temperature reactors in China and is intended to give Chinese science and industry experience with high-temperature reactors. The plant currently comprises two units, each with 250 MW of thermal output, which together supply a turbine with 210 MW of electrical output.

The HTR-PM is conceptually similar to the HTR reactors developed in Germany in the 1960s and 70s (AVR, THTR, see Chapter **4.6.3**). It is a direct continuation of the development line that emerged from the AVR. This is the HTR module with 200 MWth, a simplified design by Siemens/Interatom based on the AVR, but which was not built (later HTR-100). The main change to the AVR is the separation of steam generation from the reactor pressure vessel, which is now only connected by a hot gas line. Development of the PBMR in South Africa was then based on the HTR module (see Chapter **4.6.3**). The predecessor concept of the HTR-PM demonstrator, the HTR-10 experimental reactor, had been planned in China since 1992 and the planning was transferred to the development of the HTR-PM in 2001. The developers also cite influences from General Atomics' MHTGR (see Chapter 5.9) (Zhang et al. 2016).

A commercially viable plant with 600 MWe and 6 modules is planned, the design of which was completed in 2016 (Zhang et al. 2022). The HTR-PM is close to the properties described in the VHTR technology line (see Chapter **4.6**).

The special feature of the HTR-PM is that the design is intended to make fuel temperatures above 1600 °C impossible. The HTR-PM and its successor (HTR-PM600) are to be used primarily for electricity production. Co-production and use of heat is also planned.

5.8.1 Description of the plant concept

The HTR-PM is a pebble bed high-temperature reactor. The reactor core consists of TRISO particles that are embedded in graphite spheres (diameter 60 mm). The graphite spheres form a loose bed in the reactor pressure vessel. There is a graphite reflector inside the bed. The bed is surrounded by an outer graphite reflector. As the HTR-PM consists of two reactor modules that jointly drive a generator, each reactor module contains a reactor pressure vessel with the internal structures of graphite, metal and the pebble bed, a steam generator and a helium blower.

The fuel pebbles are continuously fed into the reactor core from above and move downwards in the reactor vessel over time. Pebbles are continuously removed from the bottom of the reactor vessel to then be fed back into the reactor from above or removed from the reactor after a pebble has achieved maximum burn-up. A discharge machine removes the fuel pebbles individually and separates off broken fuel pebbles. In the HTR-PM, discharge and separation of broken pebbles take place in two separate steps.

Individual fuel pebbles can pass through the reactor several times, depending on their previous duration in the reactor (burn-up). The continuous removal of fuel is considered an advantage, as the reactor does not have to be shut down to change the fuel. From a safety perspective, however, this leads to uncertainties about the exact composition of the reactor core and the exact position of the fuel(Englert et al. 2017).

The steam generator consists of 19 heat transfer tubes, each with a heat transfer capacity of 13 MWth. The tubes are tested in the 10 MWth Helium Engineering Test Facility (engineering test facility — helium technology (ETF-HT), engineering test facility — steam generator (ETF-SG)) at 80% full power. The water flow rate in the heat transfer tubes can be determined from the tests. Other advantages of the HTR-PM heat transfer tubes include the possibility of inspection during operation, mass production and parallel installation, and compatibility with the limited manufacturing experience of Chinese industry (Zhang et al. 2016).

Two designs for the helium blower were developed. The helium blower used is electromagnetically supported and is located at the pressure limit of the reactor's primary circuit to ensure helium sealing and prevent lubricant leakage into the primary circuit. The electromagnetic bearings were sourced from the international market. A second main blower prototype was developed as a backup. The backup blower uses dry gas seals and oil bearings. The motor and oil bearings are located outside the RPV and the blower is located in the vessel (Zhang et al. 2016).

Fuel

The fuel of the HTR-PM, as with all VHTR developments, is based on TRISO particles (see Chapter **4.6.1**) in graphite spheres. A reactor module is filled with 245,000 fuel elements, each sphere has a diameter of 60 mm and contains 7 g of enriched uranium with an enrichment of 8.5% (WNA 2022). The power density in the fuel is only 3.3 MW/m3, about 1/30 of the power density in a LWR (Zhang et al. 2016).

Further details are described in the technology line (see Chapters **4.6.1** and **4.6.6**).

Coolant

Cooling is achieved via a helium circuit. The cooling gas enters the reactor from below at 250 °C and moves up through the channels of the side reflectors to the upper reflector, where it reverses direction and flows downwards through the sphere bed. Bypass flows are fed into the pebble removal tubes to cool the pebbles and into the control rod channels. The cooling gas is heated in the reactor core and, after mixing, exits with an average core exit temperature of 750 °C (WNA 2022). This is far below the temperatures of a future VHTR.

The heat is transferred to the secondary cooling circuit in an external heat exchanger, where it generates superheated steam in the steam generator at 13.25 MPa and 567 °C (Zhang et al. 2016).

5.8.2 Safety concept and safety features

The HTR-PM shares the same basic advantages and disadvantages of all VHTR/HTR: high heat capacity resulting in sluggish behaviour, negative reactivity coefficient due to uranium-238 in the fuel, inert cooling medium, low activity in the primary circuit under normal conditions, inclusion of the fuel in TRISO particles and low power density in the core. The consequences of air or water entering the primary cooling circuit depend on the extent of such leaks and the resulting source term.

One difference to other HTR concepts is that there is only a small excess reactivity due to the continuous fuel flow in the core, which has a positive effect on the safety properties. On the other hand, radioactivity and reactivity are outside the RPV (pebble flow), which must be taken into account in the safety concept, so that the fuel removal and supply, as well as the fuel sorting, must also be taken into account with regard to safety.

Another difference between the HTR-PM and other HTR concepts in the technology line is that the reactor power, and the slim design of the RPV, were chosen so that no temperatures above 1600 °C should occur and the containment of activity in the TRISO fuel should be ensured in all reactivity and loss-of-coolant accidents, even if all active safety systems fail (WNA 2022). To accomplish this, the core outlet temperature of the coolant had to be reduced to 750 °C, which, at the same time, excludes a number of uses of the process heat that are not provided for in the HTR-PM.

Like other HTR concepts, the barrier concept is primarily aimed at the containment of radioactivity in the TRISO fuel particles and the surrounding graphite itself (first barrier). The primary circuit (second barrier) and the steam generator are located in an internal concrete structure/concrete cell (third barrier) that is located in a reactor hall (fourth barrier).

As with most previous HTR concepts (Kugeler and Zhang 2019), the HTR-PM did not feature pressure-tight containment but only confinement with a strong negative pressure. If there were a break in the high-pressure helium cooling system (loss-of-coolant accident), the entire coolant, together with the dust and radioisotopes it contained, would be vented quickly and unfiltered into the atmosphere, via an exhaust stack, as soon as the pressure reached 0.2 atm overpressure (Moormann et al. 2018; Li et al. 2011). This places particular emphasis on the safe containment of the radioactivity in the fuel and on the reliable function of the TRISO particles, even under accident conditions exceeding design-basis (temperature).

Several active safety systems are included to counteract potential water ingress (see also Chapter **0**). Since the core is undermoderated, the addition of water leads to an increase in reactivity, which leads to an increase in temperature and a compensation by the negative temperature coefficient (Doppler broadening). If the density of water vapour in the core exceeds about 0.03 g/cm^3 (without control rods), a temperature excursion can occur, the fuel can overheat and thus fail. If the water vapour density in the core rises above 0.05-0.1 g/cm^3, the additional moderation can lead to a positive temperature coefficient and thus to a heterogeneous power distribution in the core, which can also deceive operating personnel, such that incorrect measures can even lead to an intensification of the accident (Moormann et al. 2018; Lohnert 1992; Ougouag et al. 2004).

Due to the friction of the pebbles and corrosion, graphite dust is created in the reactor, on which fission products can be deposited or absorbed (see also Chapter **0**) and the graphite dust is deposited in the reactor. Understanding of the formation and movement of graphite dust in the reactor and of the fission products absorbed by the dust is still incomplete (Stempniewicz et al. 2018). The radioactive graphite dust can be remobilized in the event of a loss-of-coolant accident (Stempniewicz et al. 2018) and could be released into the atmosphere through the ventilation of the confinement. Previous experience with this in HTR-10 was positive (Moormann et al. 2018; Xie et al. 2017). (Moormann et al. 2018) However, they note that the results are not necessarily transferable to the operating conditions of the HTR-PM. Firstly, because the HTR-PM applies higher loads to the fuel pebbles caused by the larger core. Secondly, because there can be temperature excursions and thirdly because of the penetration of radioactive substances that may not be detected quickly enough.

Originally, 8 control rod drive mechanisms and 22 shutdown systems with small absorber pebbles located in 30 graphite blocks in the rotating side reflectors, were planned for reactivity control. During operation at up to 40% power, the absorber pebbles in the side reflector bores would have been driven out of the reactor core with helium and control rods would have been used to maintain criticality. However, the running main helium blower would have caused a pressure drop in the reactor internals and this would have led to difficulties and uncertainties in the operation of the

shutdown systems, so that the originally planned, absorber pebble shutdown systems could not meet requirements. The developers switched to a system with 24 control rods and only 6 absorber pebble shutdown systems. The reactor can be shut down, started up and operated using the control rods alone. The absorber pebble shutdown systems are now a decoupled shutdown system and during a reactor shutdown the pebbles can now be returned to the storage containers without the main helium blower (Zhang et al. 2016).

(Moormann et al. 2018) addresses a number of safety recommendations in light of the history of HTR:

- Underground storage of spent fuel (see Chapter 5.8.3).
- An extensive commissioning phase with slow temperature ramp-up to quantify temperature excursions and other core parameters.
- Assuming that air ingress will occur and equipping systems with fast-sealing foams and fire containment using halon or similar gases to prevent oxygen-graphite reactions in the reactor and in interim storage facilities. Cutting off the oxygen supply may be the best defence against worst-case accidents.
- Continuous monitoring of silver-110m and caesium-137 levels in the core, which could provide early indications of problematic radionuclide concentrations. The highest activity, and thus the highest safety risk, will be found at the end of the reactor's life in the form of dust accumulations. Regular removal of dust from internal surfaces is recommended.
- Retrofitting the chimney of the ventilated low-pressure containment (confinement) with fast-acting electrostatic filters, or wet scrubbers, to stop the release of most of the radioactive dust into the atmosphere. Other options include the concept of an inflatable balloon made of refractory material or water-sand filters through which the gases can escape but which retain most of the dust.
- Considerations for large-volume waste treatment options to reduce the risk of fire and corrosion. New fuel forms could also be developed to improve traceability, fission product retention and wear and dust development.

Conclusions on safety

Some of the accident sequences that occur with LWR cannot occur in the HTR-PM. However, there are also specific malfunction and accident sequences with the HTR-PM, such as water and/or air ingress, and the possibility of releasing large amounts of radioactivity into the environment.

The HTR-PM should be designed in terms of its geometric dimensions and performance so that temperatures above the critical threshold of 1600 °C cannot occur.

The HTR-PM shares the same advantages and disadvantages of the VHTR technology line in terms of safety properties relating to reactivity control, residual heat removal, releases in the event of malfunctions and accidents, and the spectrum of events (see Chapter **4.4.6**).

At the technology-line level, overall, using power-limited HTR reactor concepts such as the HTR-PM can be assumed to offer a slight advantage in terms of safety compared to current LWR. However, the HTR-PM was built without pressure-tight containment, which in turn reduces this advantage.

5.8.3 Supply and disposal aspects

Uranium enrichment and the production of TRISO fuel particles are required to supply the HTR-PM.

Dry container storage is planned on site for interim storage of the spent fuel elements. Each container, with a wall thickness of just 2 cm, is 1.74 cm in diameter and 4.18 m tall and has a capacity of 40,000 spent fuel elements (Moormann et al. 2018). The containers will be housed in a spent fuel storage building with concrete shielding (Zhang et al. 2016), five containers will be stacked on top of each other in rows of 8 containers. In contrast, in Germany about 2000 AVR fuel pebbles are stored in a container with a wall thickness of 37 cm (Moormann et al. 2018). (Moormann et al. 2018) consider storage in China to be insufficiently safe, a bullet could penetrate the container wall.

According to the developers (Zhang et al. 2016), there is currently no data on metal corrosion near the sea, so forced ventilation is used by air flow in a closed circuit. In the event of a power failure, the residual heat can be dissipated by natural air circulation. The container can also be housed in a standard LWR transport container and transported if necessary. In (Moormann et al. 2018), it is noted that the planned storage does not provide sufficient protection against fire and other threats.

With regard to the disposal of the HTR-PM waste, the total volume due to the graphite is about ten times larger than that of LWR fuels. However, the overall size of a repository depends not only on the volume of waste being stored, but above all on the temperature development of the waste and the resulting distances between the repositories. More storage containers are needed for HTR waste. Without taking a closer look, it is not possible to say in general what distances must be maintained between the containers and whether the larger volume of waste also requires a larger repository area.

Conclusions on supply and disposal

The supply and disposal of HTR fuel has no significant advantages or disadvantages compared to LWR fuel.

5.8.4 Proliferation risks

Pebble bed reactors are more susceptible to fuel diversion than LWR due to the ongoing exchange of fuel. They are also particularly suitable for tritium production to enhance the explosion of nuclear weapons.

Since the HTR-PM uses an open uranium-plutonium fuel cycle, there are no differences to the HTR technology line (see Chapter **0**).

Conclusion on proliferation risks

Overall, the HTR-PM pebble bed reactor is expected to have a slight disadvantage in terms of proliferation risks compared to the LWR.

5.8.5 Technological development status

The VHTR technology line has reached a high level of development due to previous experiments and demonstrator projects.

The Chinese developers worked in close collaboration with German scientists and engineers during the research and development phase (Zhang et al. 2016). However, according to (Zhang et al. 2016),

these collaborations were not sufficient for the construction of the power plant. Firstly, the projects for the HTR module and the MHTGR were ultimately never realised, although there were extensive studies on the reactors. The design drafts and the technologies for manufacturing components were not transferred, with the exception of some software and several technical consulting contracts. Finally, some of the knowledge was lost as the original developers got older and companies closed down. While there were still many suppliers in the 1980s and 1990s, the development of the HTR-PM required the establishment of a new supply chain and a corresponding construction of test benches and experimental facilities, many of which were full-scale experiments in a hot helium environment. The major components were all manufactured in China, such as the helium blower, the steam generator, the fuel exchange system, the storage, control and shutdown systems and the helium cleaning system. Many engineering difficulties arose and had to be solved (Fütterer et al. 2021).

In an interview in 2022 (WNA 2022), Lü Hua Quan, the director of the Nuclear Research Institute of the Huaneng Company, explains which challenges were solved during the construction of the HTR-PM. There were some problems with the nuclear part of the plant, which significantly delayed construction and led to higher construction costs. As an example, he cites problems with the continuous and reliable operation of the fuel element exchange facility. However, he sees fundamental potential to learn from the problems for future plants. Experiences from the HTR-PM projects should also serve to plan the areas for emergency protection for adaptation to future locations. For the future development of HTR, he sees great opportunities for the decarbonisation of sectors that are difficult to address, e.g. in the use of process heat, but some developments such as high-temperature materials, regulatory frameworks, safeguards and disposal certificates for new fuels as well as the economics of the HTR still need to be pushed forward. China has high hopes here, both for its own needs and for export, especially due to the modularity of the HTR of 100 - 1000 MWe. For countries on the new Silk Road in particular, only small to medium-sized plants would be particularly advantageous, as their energy networks are not designed for larger plants or for building plants very close to consumers (WNA 2022).

In 2016, China and Saudi Arabia announced in a joint “Memorandum of Understanding” that they would build an HTR together. At the same time, China declared that it now mastered all key technologies of HTR technology (WNN 2016).

Conclusions on technological development status

With regard to the VHTR technology line, the HTR-PM only has a temperature of 750 °C and is only used for pure electricity generation with a steam cycle.

Due to its commissioning as a demonstrator for pebble bed HTR and due to the historical experience with pebble bed reactors, the HTR-PM is classified as being in “late development” or “early deployment”. Full operational readiness will only be achieved in a commercial follow-up project with the operational experience of the HTR-PM.

With regard to the use of process heat, and with a view to the use of higher temperatures (VHTR) as the goal of the VHTR technology line, the HTR-PM is classified as being in “development”.

5.8.6 Implementation

5.8.6.1 Planned area of application

The long-term goal of developing the HTR-PM into a fully marketable system is to develop large modules with an electrical output of up to 1,000 MWe.

The niche market for high-temperature pebble bed reactors is combined heat and power, but above all the generation of process heat in addition to electricity generation. According to the developers, the HTR-PM also aims to generate electricity and process heat, which can be used for hydrogen production, for example (Zhang et al. 2016). There are no specific plans as of yet.

5.8.6.2 Project timeline

China began developing a high-temperature reactor in the 1980s. Even in the early stages, there was intensive cooperation between companies in the Federal Republic of Germany (KFA, Interatom) and the Institute for Nuclear Energy Technology (INET) at Tsinghua University in Beijing. Construction of the first prototype pebble bed reactor in China began in 1995. The HTR-10 then went critical for the first time in 2000 at a capacity of 10 MWth and a gas outlet temperature of 700 °C. In 2005, after initial planning, China decided to build the HTR-PM, a larger version of the HTR-10.

Chinergy Co., Ltd. was established in 2003 as the architectural and EPC contractor for the nuclear power plant island, and Huaneng Shandong Shidao Bay Nuclear Power Co., Ltd. (HSNPC) was established in 2007 as the owner of the plant. Shang-hai Electric Corporation and Harbin Electric Corporation were contracted to manufacture the main components of the nuclear steam supply system (NSSS).

The HTR-PM was originally planned to have a capacity of 458 MWth in one module. Plans continued in this way until 2006. However, there were problems designing the central reflector. There were two approaches. First, a column of pure graphite spheres in the centre. This approach was rejected by the licensing authority, however, because the developers could not convince the regulator that there was a clear boundary between fuel and pure graphite, as it was possible for fuel spheres to mix with reflector spheres in the zone, or vice versa. In addition, the performance profile would be altered because helium would also flow through the graphite sphere column, and the control rods in the side reflectors in the reactor wall would lose too much of their reactivity effectiveness. The alternative design envisaged a solid column of graphite in the centre of the reactor pressure cylinder. However, this graphite column would have to be replaced during the lifetime of the reactor due to material wear. In addition, three sphere removal devices would now have to be created because the spheres could no longer be removed centrally. This would make the entire cycling of the fuel spheres considerably more complex. There were also difficulties with structural stability. The 458-MWth design was eventually abandoned and, after a cost assessment of the construction costs, the 250-MWth concept with 2 reactor modules was preferred.

In February 2008, construction of the HTR-PM began as one of the flagship projects of the National Science and Technology Programme. The state's goal with the HTR-PM was to develop efficient nuclear technology as an alternative to the LWR, as well as technology for generating nuclear process heat and to participate globally in the development of advanced nuclear technologies.

After the construction permit was granted by the Chinese regulatory authority (Nuclear Safety Administration - NNSA), construction began on December 9, 2012 in Rongcheng in Shandong Province in China. The original construction period was planned to be 5 years and the demonstration power plant was to be connected to the grid in 2017 (Zhang et al. 2016). The work was delayed due to necessary research and development work and the need to establish a supply chain (see Chapter 5.8.5).

In March 2020, work on the primary circuit of the second reactor module was completed.[196] In October and November 2020, cold tests were carried out on the primary circuit components of the two modules to ensure that the welds, connections and piping of the reactor cooling system meet the required quality standards.[197] Hot tests were then started.[198]

First criticality was achieved in Unit 1 in September 2021, followed by Unit 2 in November 2021. The plant was connected to the grid in December 2021. The plant should go into full operation by mid-2022. Construction of up to 18 additional units at the same site is planned. The HTR-PM600 is planned as the successor plant. Here, six modules will supply a turbine with a total electrical output of 650 MWe.[199]

Regarding fuel production, a prototype fuel production plant with a capacity of 100,000 fuel elements per year was built in 2005 to gain initial experience. In 2013, construction of the HTGR fuel production facility began in Baotou, northern China. Commissioning and test production began in 2015. An irradiation test of five HTR-PM fuel balls also took place in October 2012 in the high flux reactor (HFR) in Petten in the Netherlands and ended on December 30, 2014. The post-irradiation tests showed that the fuel achieves the required performance (Zhang et al. 2016).

Conclusions on project progress

The construction of the reactor was delayed by around 5 years, although extensive historical experience from Germany and South Africa could be used. Two main reasons are given for this. Firstly, it was difficult to establish the industrial supply chain for the reactor components in China. On the other hand, an extensive experimental and test infrastructure was also necessary and there was a need to adapt and improve many details of the German and South African design.

[196] https://www.world-nuclear-news.org/Articles/Key-components-of-second-HTR-PM-reactor-connected, last accessed 25/01/2022.
[197] https://www.nuklearforum.ch/de/aktuell/e-bulletin/china-kalttests-beim-zweiten-modul-des-htr-pm-abgeschlossen, last accessed 25/01/2022.
[198] https://www.world-nuclear-news.org/Articles/Hot-functional-testing-of-HTR-PM-reactors-starts, last accessed 25/01/2022.
[199] https://www.world-nuclear-news.org/Articles/Demonstration-HTR-PM-connected-to-grid, last accessed 25/01/2022

5.8.6.3 Costs

The HTR-PM does not use the high temperatures of process heat and is designed purely for electricity production. This means that the economic analysis also excludes the use of process heat for industrial applications, as is otherwise often discussed for VHTR/HTR in the technology line (see Chapter**0**).

Since the HTR-PM is a FOAK reactor, (Zhang et al. 2016) emphasize that the construction of a comprehensive development and test infrastructure of 13 plants was also required in order to provide the appropriate safety evidence. Considerable costs were incurred for this.

Commercial use of the HTR-PM is not expected until the next reactor, the 600 MWe HTR. The concept envisages the use of several reactors in a modular design (multi-module) and thus follows the idea of small modular reactors. A similar concept was already planned in Germany 30-40 years ago as an alternative (HTR module, or HTR-100) in order to reduce costs and make the HTR economically competitive (Reutler and Lohnert 1984; Oeko-Institut e.V. 1989b).

The developers of the HTR-PM want to limit construction costs to just 110-120% of the costs of a light water reactor of the same output with the further development and also assume an increase in the levelised cost of electricity of around 10-20% compared to current (Chinese) LWR. (Zhang et al. 2016). A detailed and early breakdown of the cost structure can be found in (Zhang and Sun 2007). A more recent detailed presentation can be found in (Kugeler and Zhang 2019).

Conclusions on costs

As with many of the HTR demonstration projects, the HTR-PM also experienced significant cost increases during construction due to delays. The forecasts show that the HTR-PM, and its successors, will have slightly higher levelised cost of electricity than current LWR and, at the same time, will forego the commercial use of process heat. This leaves this concept with a slight disadvantage in terms of costs compared to LWR and a disadvantage within the technology line compared to future HTR concepts that have higher temperatures and can use process heat.

5.9 VHTR: Prismatic high-temperature reactors

The Prismatic Modular HTGR from General Atomics is a high-temperature reactor with a thermal neutron spectrum, see Chapter **4.6**. The Prismatic HTR is to be used with 350 MWth of thermal power and a steam cycle, and generate 150 MWe of electrical power.

The Prismatic HTR is based on earlier concepts of a prismatic reactor from the 1960s and 1970s, such as the Fort St. Vrain reactor (see Chapter **4.6.2**) or the later concept of the Modular High Temperature Gas Cooled Reactor (MHTGR) from General Atomics, which had already been pre-licensed in the 1980s, also with 350 MWth.

The earlier reactor concepts of the prismatic reactor were further developed in the U.S. under the Next Generation Nuclear Plant project (NGNP, 2006 - 2013) between several industrial partners and the DoE from earlier preliminary studies (e.g. MHTGR) and the concept of the SC-HTGR from AREVA (see Chapter **4.6.3.3**). The original plan by General Atomics to use a gas turbine (Gas Turbine Modular Helium Reactor - GT-MHR) was not pursued further during the NGNP.

The concept of the Prismatic Modular HTGR ultimately does not differ too much from the HTR-PM and the HTR technology line, apart from the fact that the reactor core uses prismatic fuel (see Chapter **4.6**). The biggest difference is that the fuel is not continuously exchanged, as in the pebble bed reactor, and therefore fuel element changes or repositioning of the fuel are necessary.

There was significant progress, particularly in fuel development, with the development of UCO fuel for TRISO production, but also in terms of licensing efforts. The collaboration between the industry partners and the state institutions ultimately failed in the 2010s, mainly due to differing ideas regarding the required investments as well as how to minimise risk within the licensing process.

5.9.1 Description of the plant concept

The Prismatic HTR is a high-temperature reactor that uses prismatic fuel made of TRISO particles in graphite rods (thumb-sized) that are embedded in fuel elements. As with the HTR-PM, the reactor pressure vessel is separate from the steam generator and connected to it by a connecting vessel. The reactor pressure vessel (inner diameter 6.55 m) is not insulated to allow heat removal in the event of an accident and is as large as the RPV of a large boiling water reactor, with a much lower output (350 MWth instead of about 3 GWth). Steel is used as the structural material for the pressure-retaining components (IAEA 2010a).

The reactor core consists of hexagonal fuel elements and reflector elements made of graphite, plenum elements, starting sources and materials for reactivity control. The active core (outer diameter 3.5 m) consists of hexagonal graphite fuel elements, each containing 210 holes for the compact fuel rods and 102 cooling gas channels. 10 fuel elements are stacked on top of each other. The fuel elements are arranged in a ring shape (3 rings) around a central reflector zone with 19 graphite elements and surrounded by two rings with graphite reflector elements, so that there is a minimum reflector thickness of 1 m around the core (height 7.9 m). The arrangement of the fuel in the core is therefore more heterogeneous than in the pebble bed reactor (IAEA 2010a).

The arrangement was chosen to keep the power density at 5.8 MW/m^3 and to enable passive cooling at a maximum temperature of 1600 °C in the event of a pressure loss or loss of cooling gas. Under normal operating conditions, the maximum fuel temperature is 1250 °C (IAEA 2010a).

The reactivity control is achieved by a combination of rods with burnable neutron poisons in the core, in the form of fuel rods and control rods. Rods with burnable neutron poisons are used to keep the power profile in the core as flat as possible and to control the reactivity over a cycle. The negative temperature coefficient also affects the reactivity control. Six central reflector elements and 24 side reflector elements have channels for control rods that are inserted into the core from above through the upper plenum. The 6 control rods in the central reflector are only used for cold shutdown. The control rods are made of boron in a graphite tube with a metal casing. The inner diameter of the central reflector and the thickness of the fuel element ring were chosen so that the necessary reactivity control is possible with the help of the control rods. If the control rods fail, a shutdown device in the form of boron-containing pellets is activated, which are released into channels in the active core (IAEA 2010a).

There were plans to combine four modules of the Prismatic HTR into one plant, with all safety-relevant systems and components being combined within a special nuclear part of the plant, which is physically and functionally separate from the rest of the plant (turbine, etc.). The reactor modules are located in adjacent but separate concrete-reinforced silos (approx. 9m deep) below the earth's surface (IAEA 2010a).

Fuel

The fuel consists of TRISO particles embedded in a cylindrical graphite matrix (diameter 12.5 mm, length 50 mm). The fuel is uranium oxycarbide (UCO) with an enrichment of 15.5% (LEU). UCO was selected because the fuel can achieve higher burn-ups (IAEA 2010a). The properties of the fuel should allow burn-ups of up to 150-200 MWd/kg heavy metal with a maximum temperature of 1250 °C (see Chapter 5.9.5).

The fuel elements are exchanged using a fuel element exchange machine. The fuel is preferably only exchanged axially and not fully 3D. The cycle length should be 18 months (IAEA 2010a).

Coolant

As with the HTR-PM, the cold cooling gas flows from the helium blower on the outside of the connecting vessel and the inner hot gas pipe to the reactor pressure vessel, then upwards between the wall of the reactor pressure vessel and the reactor core and then flows through the reactor core from top to bottom and is heated to 750 °C. The hot helium gas then flows in the connecting vessel in the inner hot gas pipe to the steam generator and then downwards along the heat transfer pipes in the heat exchanger and is cooled in the process. The gas then flows upwards between the vessel wall of the steam generator to the helium blower. On the secondary side, the feed water enters the steam generator from below and through the heat transfer tubes and exits again as superheated steam
(IAEA 2010a).

5.9.2 Safety concept and safety features

The Prismatic HTR has the same characteristics as described at the technology-line level (Chapter **4.6**): high heat capacity resulting in sluggish behaviour, negative reactivity coefficient due to uranium-238 in the fuel, inert cooling medium, low activity in the primary circuit under normal conditions, inclusion of the fuel in TRISO particles. The consequences of air or water entering the primary cooling circuit depend on the extent of the leaks, especially the source term.

The Prismatic HTR has two active, diverse systems for heat removal. Firstly, the main system for heat removal, consisting of the steam generator and the helium circulation system, which is also used to remove the heat when the reactor is shut down for maintenance or fuel element replacement. However, when the reactor is shut down, the heat can also be dissipated by an independent cooling system, the Shutdown Cooling System (SCS). The SCS consists of a motor-driven heat exchanger below the reactor core inside the reactor vessel. The SCS is water-cooled and is not a safety system for normal operation (IAEA 2010a).

The Prismatic HTR is designed in such a way that in the event of a pressure loss in the primary circuit or a loss of coolant, the core can be cooled passively so that the fuel temperature reaches a maximum of 1600 °C. The Reactor Cavity Cooling System (RCCS) dissipates heat from the uninsulated reactor vessel by thermal radiation and natural convection of air through special cooling panels that surround the reactor. To minimise the release of argon-41 from the reactor pit, the air used for cooling is not mixed with the air in the reactor pit. Due to the continuous natural convection, it should be possible to remove heat in the event of an accident without active components, power sources or intervention by operating personnel. The RCCS also cools the reactor pit during normal operation (IAEA 2010a).

Even in the event of a failure of the RCCS (exceeding-design-basis accident), the Prismatic HTR is designed to continue to maintain cooling through passive heat removal, through heat conduction from the core, through heat radiation from the reactor vessel and through heat conduction into the surrounding walls of the silo.

The accident sequences for air ingress and water ingress are described in the chapter on the HTR-PM (Chapter 5.8) and in the chapter on the technology line (Chapter **4.6**) and differ little. This also applies to the problems of radioactive graphite dust in the event of a rupture of the primary cooling system.

The underground construction of the reactor modules also results in safety advantages, e.g. seismic amplification is lower. Other advantages are a lower need for shielding.

As with most previous HTR concepts (Kugeler and Zhang 2019), the Prismatic HTR did not feature pressure-tight containment but only confinement with a strong negative pressure. To maintain the negative pressure, concrete-reinforced cells connected to the surface form an isolated space designed to only leak one volume per day at an internal pressure of 1 psi (approx. 0.007 mPa) under normal conditions. If the high-pressure helium cooling system were to break (loss-of-coolant accident), the pressure of 1 psi would be exceeded and all of the coolant, together with the dust and radioisotopes it contains, would be rapidly vented, unfiltered, into the atmosphere via the above-ground cells near the ground.

This places particular emphasis on the safe containment of the radioactivity in the fuel and on the reliable function of the TRISO particles, even under accident conditions exceeding design-basis (temperature).

Conclusions on safety

Some of the accident sequences that occur for LWR cannot occur in the Prismatic HTR. However, there are also specific malfunction and accident sequences with the Prismatic HTR, such as water and/or air ingress, and the possibility of releasing large amounts of radioactivity into the environment.

The Prismatic HTR should be designed in terms of its geometric dimensions and performance so that temperatures above the critical threshold of 1600 °C cannot occur.

The Prismatic HTR shares the same advantages and disadvantages of the VHTR technology line in terms of safety properties relating to reactivity control, residual heat removal, releases in the event of malfunctions and accidents, and the spectrum of events (see Chapter **4.4.6**). It has advantages in terms of design against earthquakes and other external influences due to the underground construction.

At the technology-line level, overall, using power-limited HTR reactor concepts such as the Prismatic HTR can be assumed to offer a slight advantage in terms of safety compared to current LWR. However, the Prismatic HTR was built without pressure-tight containment, which in turn reduces this advantage.

5.9.3 Supply and disposal aspects

In terms of supply and disposal concepts, there are few differences between the Prismatic HTR and the technology line (Chapter **4.6.6**) and the HTR-PM (Chapter 5.8.3).

The developers state that the advantages are that initial calculations indicate that the Prismatic HTR achieves very high burn-ups thanks to the possibility of using UCO TRISO fuels and is therefore also suitable for waste incineration.

Conclusions on supply and disposal

The supply and disposal of HTR fuel has no significant advantages or disadvantages compared to LWR fuel.

5.9.4 Proliferation risks

With regard to proliferation risks, there are few differences between the Prismatic HTR and the technology line (Chapter **0**), as the HTR-PM uses an open uranium-plutonium fuel cycle.

In view of the comparatively small amount of fissile material per volume of fuel and the high graphite content, the developers of the Prismatic HTR specify that 50 times more fuel elements would have to be stolen to divert a critical amount of fissile material than with an LWR. However, they do not go into the dimensions of the fuel elements in any more detail; the information probably only refers to the number of elements.

Due to the non-mobile fuel, the Prismatic HTR has advantages in terms of proliferation resistance, as diverting fissile material is more difficult (Chapter 5.8.4).

Conclusion on proliferation risks

Overall, therefore, there are no significant advantages or disadvantages for the Prismatic HTR in terms of proliferation risks compared to the LWR.

5.9.5 Technical development status

The development of prismatic HTR can draw on a number of historical experiences (see Chapter **4.6.2**). When developing prismatic HTR, the aim was to achieve the highest possible temperatures and to use a gas turbine, just as with the pebble bed HTR.

Similar to the HTR-PM, however, the NGNP programme ultimately decided not to develop a gas turbine and instead aimed for a steam cycle. The gas outlet temperature was also limited to 750 °C and the power and dimensions of the reactor were designed accordingly so that the maximum fuel temperature does not exceed 1600 °C with completely passive cooling without coolant. In order to achieve higher gas outlet temperatures, or to be able to use a gas turbine or increase the maximum fuel temperature, development work would have been necessary that would have significantly exceeded the original schedule aimed for in the NGNP programme.

The following presentation therefore focuses on the work required to develop this configuration of a prismatic HTR, the Prismatic HTR in the NGNP programme.

Regarding fuel development, a number of irradiation experiments were carried out on UCO TRISO fuels in the DoE's Advanced Gas Reactor (AGR) programme (see also Chapter **0** and **4.6.4**). The properties of the fuel are said to allow burn-ups of up to 150-200 MWd/kg heavy metal with a maximum temperature of 1250 °C, with 19.4% FIMA (fissions per initial metal atom). In the first AGR-1 experiment with 3,000,000 particles under these conditions, there was no particle failure of the fuel. More recent experiments also showed improved fission product retention and less particle failure even at temperatures of 1600-1800 °C (EPRI 2019). In a second AGR-2 experiment at 1600 °C, except for the fission product Ag-110m, a factor of seven fewer fission products were released in the UCO TRISO (e.g. Kr-85m, Cs-134, Cs-137, Eu-154, Eu-155, Sr-90) than in other previously produced TRISO fuels from the historical US programme or the German HTR programme (Kadak 2016).

Further experiments, AGR 3/4 and AGR 5/6/7, were planned to test the properties of the fuel in the event of intentional fuel failure and to determine the source term for a potential release and corresponding evacuation radii. The irradiation tests were to continue despite the DoE's decision not to pursue the development of a demonstration reactor of the Prismatic HTR (Kadak 2016). The planning of the post-irradiation results is described in (INL 2020). Results are not yet available, but should be able to be used in future HTR development programmes.

Considerable efforts were also made at INL and ORNL, together with European partners, to test different types of graphite. 2000 different types of graphite were characterised as part of the Advanced Graphite Creep test programme and models were created to predict the behaviour of graphite under normal operating conditions and accident conditions. Software codes were developed at the American Society of mechanical Engineers (ASME) and qualification methods at the American Society for Testing and materials (ASTM). The aim of the developments was, on the one hand, to develop acceptable criteria for graphite for use in HTR based on a scientific understanding of the fundamental mechanisms of irradiation, so that they can also be used in the future when HTR are built in the USA (Kadak 2016), and, on the other hand, to demonstrate that the quality of modern graphite produced with today's starting material performs at least as well as historical graphite from past HTR development programmes (IAEA 2010a).

For the Prismatic HTR, it was decided to focus development efforts for structural materials on the temperature range between 750-850 °C, although the goal of chemical hydrogen production remains

a long-term goal. Two materials were investigated. The alloy 617 made of nickel, chromium, cobalt and molybdenum with an addition of aluminium, which has exceptional high-temperature strength and oxidation resistance. And the alloy 800H, an austenitic nickel-chromium alloy with controlled proportions of carbon, aluminium, titanium, silicon and manganese. Significant efforts were made to obtain the data for the alloy 617 and generate it for the ASME code so that the alloy can be used as a pressure barrier at 950 °C. A code qualification for 850 °C and 500,000 hours was also carried out for the alloy 800h. The work continued even after the end of the NGNP programme (Kadak 2016).

Regarding reactor physics, work was also carried out to validate and verify the thermohydraulic and heat conduction codes used. Numerous universities are involved in the numerous aspects of testing individual effects and integrated experiments. At Oregon State University, for example, the High-Temperature Test Facility (HTTF) was built, an integrated test facility designed to test fundamental properties of heat transport even under accident conditions. At Argonne National Laboratory, the Natural Circulation Shutdown Test Facility was built to test the properties of the cavity cooling system of the Prismatic HTR. An overview of the experiments is shown in (Kadak 2016).

The code base for core modelling was also renewed and improved. New codes were developed as part of the multiphysics Object-Oriented Simulation Environment (MOOSE) to achieve 3-D solutions using finite element methods. The models are intended to replace the old diffusion codes such as SCOP. Improvements were achieved through thermal feedback during burn-up calculations and improvements to RELAP-3D for gas reactor analyses. Several benchmarks were carried out within the framework of the OECD and the IAEA, in which the mHTGR-350 served as a reference reactor (Kadak 2016; Lemaire et al. 2017).

The focus in the NGNP programme was primarily on the prismatic reactor design, as the reactor is more complex than the pebble bed reactor. Future challenges will include the analysis of neutron scattering in graphite and elastic scattering in the heavy metal, but also the effect of radiation damage on the thermal properties of graphite and the heat conduction properties at the transition from the core to the reflector. In addition, there are still gaps in the data regarding core behaviour at high burn-up and studies on water and air ingress must be carried out, to name just a few of the requirements that would be needed for licensing in the USA (Kadak 2016).

For future licensing efforts, over 160 publications, concept studies and research reports have been produced in the NGNP programme alone, and numerous discussions with the regulatory authority (NRC) have taken place and white papers have been produced. Overall, however, (Kadak 2016) concludes that little progress has been made on some key discussion points. This concerns containment requirements, determination of the radioactive source term and selection of events for permitting and emergency planning. In collaboration with the NGNP Industry Alliance, the NGNP team identified 2500 individual points that still need to be clarified for licensing, based on the current criteria for LWR. Several reports have been written to address these points, but no decision has been made by the NRC after the NRC review.

Conclusions on technological development status

Regarding the VHTR technology line, the Prismatic HTR only has a temperature of 750 °C and is only used for pure power generation with a steam cycle. Significant progress has been made in development with the new UCO fuel.

Compared to the pebble bed concept, the Prismatic HTR is significantly more complex to develop and there are still unresolved questions about the licensing basis for HTR in the USA. Overall, the Prismatic HTR can be classified as being in “development” due to historical experience with prismatic HTR. Similar to the HTR-PM, however, proof of principle of the reactor concept must first be provided with a demonstrator in order to then be able to achieve use in a commercial follow-up project.

With regard to the use of process heat and with a view to the use of higher temperatures (VHTR) as the goal of the VHTR technology line and the use of a gas turbine, the Prismatic HTR can be classified as being in “development”.

5.9.6 Implementation

5.9.6.1 Planned area of application

As with the HTR-PM (Chapter 5.8) and the VHTR technology line (Chapter **4.6**), a key development goal for the Prismatic HTR is the niche market for combined heat and power, but above all the generation of process heat in addition to electricity generation, e.g. for hydrogen production.

Although some development work on hydrogen production was carried out by the INL and industry during the development of the Prismatic HTR, (Kadak 2016) states that interest has decreased significantly due to the inexpensive and sufficient reserves of gas in the USA.

5.9.6.2 Project timeline

The history of the Prismatic HTR goes back far into the 1980s. With the mHTGR 350, General Atomics submitted a draft safety report in the pre-licensing phase in 1989 and published it through the NRC, then revised and republished in 1995. This forms the basis for further designs based on the prismatic modular reactor concepts such as the GT-MHR or the AREVA SC-HTGR.

The GT-HTR was a 600 MWth reactor concept developed by a group of Russian and US companies, Framatome in France and Fuji Electric in Japan. The design envisaged a gas turbine with a gas outlet temperature of 850 °C. One purpose of the reactor at that time could also have been the burning of military plutonium in the USA, in order to eliminate excess military plutonium from the disarmament between the USA and Russia. The preliminary study of the reactor concept was completed in 2001 and the regulatory process started in 2002, but was not completed (Kadak 2016).

In 2005, the Energy Policy Act was passed in the USA, which required the construction and operation of an HTR by 2021. The law was passed after a multi-year study by US experts on the future of nuclear technology in the USA. As a result of the law, the US Congress decided to build the so-called Next-Generation Nuclear Plant as an HTR to generate process heat and hydrogen. A broad industry alliance was then formed with an interest in the development of an HTR, including General Atomics, AREVA/Framatome and Westinghouse, as well as potential customers such as Dow Chemical and Conoco Philips. During the NGNP programme, up to 33 industrial partners participated in the NGNP industry alliance (Kadak 2016).

Of the three original competing reactor concepts, the prismatic GT-MHR from General Atomics, the ANTARES concept from AREVA with 625 MWth based on the GT-MHR, and the SC-HTGR from Areva/Framatome and a pebble bed concept from Westinghouse, based on the South African PBMR, the SC-HTGR was ultimately selected. The reason for this was that it was not expected that a gas turbine could be successfully developed by 2021 and thus the original gas turbine-driven HTR concept from General Atomics was eliminated. The PBMR was withdrawn because difficulties were feared in licensing the pebble bed concept. Instead, the prismatic HTR concept with a steam cycle at moderate temperatures was selected as the reference concept of the NGNP (Kadak 2016).

In may 2010, General Atomics was commissioned by the DoE to develop a reference concept for a modular helium reactor with steam cycle (steam-cycle modular Helium Reactor - SC-MHR) as a demonstration power plant with 350 MW and an outlet temperature of 725 °C for the NGNP project. (General Atomics 2012). The Prismatic HTR presented here (IAEA 2010a) is therefore an interim stage of an ongoing development process.

Finally, in 2011, the DoE decided, based on a recommendation from the Nuclear Energy Advisory Council, to reduce the entire NGNP project to a pure research project and to abandon all design planning. A key reason was that no agreement could be reached on cost sharing between the industry alliance and government funders. Industry proposed a funding formula for typical industrial investments, i.e. assuming costs during construction, rather than focusing on funding for basic research to justify reactor concepts. Three phases were proposed: in phase 1, research and development would be carried out by the government, in phase 2, with preliminary design and licensing issues, industry would assume 20%, and in phase 3, when the reactor was built, 100% (Kadak 2016). However, due to the lack of results from phase 1 and especially phase 2, with a clear licensing strategy, none of the partners in the industry alliance wanted to assume the risk of building up front. Industry was also concerned about securing long-term government funding, as annual budgets were to be adopted. In total, the industry alliance proposed a cost sharing of US$1.925 billion from the government and US$3.621 billion from industry (Kadak 2016).

Despite the initial hopes and expectations, the development of the prismatic HTR in the USA currently only consists of the completion of the research programme for advanced fuels, graphite and other materials. There are currently no more plans to build a demonstration reactor, as the US Congress originally determined in 2005. The reasons given by (Kadak 2016) are insufficient government funding for research, unrealistic requirements for the temperatures to be achieved, the lack of demand for hydrogen from a hydrogen economy, competition from light-water-cooled SMR, little interest on the part of operators in new technologies, very low gas prices and a difficult licensing process in the USA for reactor concepts that are not based on LWR technology (Kadak 2016).

Conclusions on project progress

The Prismatic HTR is an intermediate stage in the ongoing development of a prismatic HTR reactor concept in the USA. The constraints of a steam cycle and a low exit temperature were chosen in order to be able to meet the original construction schedule by 2021. The plans stagnated and were eventually abandoned, despite extensive historical experience in the USA and a broad industrial consortium. Two main reasons are given for this. Firstly, government donors and industry could not agree on common financing guidelines.

Secondly, risks were still seen in the licensing process in the USA.

5.9.6.3 Costs

The development costs for the Prismatic HTR in the NGNP programme to date amount to more than 1 billion USD from industry up to 2016 and about 500 million USD from supporting research programmes of the DoE, mainly for fuel development, graphite qualification and materials research. The development costs in industry were mainly spent on developing the reactor concept and the systems for generating process heat (Kadak 2016).

As part of the NGNP programme, market studies were also carried out on the various concepts. According to the studies, priority would be given to the markets for natural gas and coal derivatives such as petrochemical products, synthetic fuels, hydrogen production, ammonia and derivatives such as fertilisers, metal production, oil refining and oil extraction from oil sands and oil shale. Typically, an HTR plant would contain several modules, each with around 200 to 600 MWth, with a total plant capacity of 2000 to 2400 MWth (4 to 10 modules), but larger plants, for example for coal liquefaction, are also possible (INL 2010).

In contrast to the LWR for electricity production, it was expected that a whole range of different and specialised reactor concepts would emerge as the HTR technology matured and a number of operators invested in the technology. The reactor concept selected in the NGNP project should therefore demonstrate the widest possible application for potential markets. An exemplary market study came to the conclusion that at a gas price of around 4 USD/MMBtu and assumed CO_2 costs of 50 USD/Mt or around 7 USD/MMBtu, a prismatic HTR could be competitive on the process heat market. The capital costs were assumed to be 650 USD/kWth for the combined cycle power plant and 1700 USD/kWth for the HTR (INL 2010). This is significantly lower than the cost estimates for the comparable SC-HTGR at 3600 USD/kWth (see Chapter **0**). The low capital costs in (INL 2010) are due to assumptions of market penetration of the HTR; in (INL 2010), an example is given with 827 reactor modules of 600 MWth, each in the period between 2020 and 2050.

One conclusion of the NGNP alliance was that there are substantial cost differences between pebble bed concepts and prismatic reactor concepts outside of the achievable performance. However, there is a difference in the modular design of a plant in terms of the capital costs required and the costs for design, development and licensing. The capital costs for an installed capacity of 2400-300 MWth are about 30% lower with a 625 MWth prismatic reactor than with a 250 MWth pebble bed reactor.[200]

Conclusions on costs

The market studies for the Prismatic HTR are primarily aimed at marketing the process heat. The forecasts assume that the Prismatic HTR will only be competitive when gas costs rise. Despite development expenditure of more than USD 1.5 billion, the outstanding development work is not completed. The entire development and construction of the reactor was planned to cost USD 5.5 billion. The use of process heat gives the HTR a fundamental advantage over LWR for this market segment. Compared to future HTR systems, the Prismatic HTR has a disadvantage within the technology line due to its moderate power and temperatures compared to future HTR concepts that can have higher temperatures or greater power.

[200] https://www.world-nuclear-news.org/NN-Areva_modular_reactor_selected_for_NGNP_development-1502124.html

5.10 ADS: MYRRHA

The most advanced planned accelerator-driven system worldwide is the mYRRHA (Multipurpose hYbrid Research Reactor for High-tech Applications) project with a planned 100 MWth. The development of mYRRHA began in 1998 and is mainly supported by the Belgian nuclear research centre SCK-CEN, where the construction of the demonstration and research facility is also planned. Completion is planned for 2026, the first plant concepts date back to the 1990s.

MYRRHA is intended to be a flexible experimental accelerator-driven system that demonstrates the overall ADS concept at low power, in particular the coupling of the three essential ADS components: accelerator, spallation target and subcritical reactor. Based on the experience with the mYRRHA demonstration project, current plans and a successful demonstration envisage scaling up, for example to the fully industrial accelerator-driven system EFIT (European Facility for Industrial Transmutation).

As is typical for materials research reactors, a range of irradiation experiments and applications should also be possible in mYRRHA, from fuel and material development for so-called new reactors and future fusion reactors, to radioisotope production for medical and industrial applications. MYRRHA is also used to research transmutation and thus bundles the research and development efforts in Europe for ADS systems and transmutation (IAEA 2015).

MYRRHA is also intended to gain experience in the operation of a lead-cooled reactor and, in critical operation, serves as a European pilot plant for the development of future lead-cooled systems such as ALFRED (see also Chapter 4.2.3.3).

In contrast to the ADS technology line, mYRRHA will not generate any electricity as a research reactor.

5.10.1 Description of the plant concept

The mYRRHA project has the most sophisticated design to date for an accelerator-driven prototype plant. MYRRHA is intended to consist of a lead-bismuth spallation target that is to be fed by a linear accelerator for protons with an energy of 600 meV at a current of initially 2.4 mA and up to 4 mA, with a total output of 1.44 to 2.4 MW (Biarotte and Müller 2011; Abderrahim et al. 2015).

The beam is fed into the reactor core from above and hits the spallation target through a beam window, where neutrons are generated.

MYRRHA is designed to be operated both in a critical mode, i.e. without a neutron source, with the reactor alone, and in a subcritical mode with a neutron source. MYRRHA is designed as an extremely subcritical system in subcritical operation, with a criticality factor of only keff = 0.95 (Biarrotte et al. 2015). Both modes have specific energy and flux distributions in the core.

As described in the technology line (Chapter **4.7.1**), continuous operation of the accelerator without beam interruptions is essential for the operation of an ADS. High availability of accelerators is achieved through a combination of oversizing and redundancy, as well as fault tolerance of the spallation neutron source. The fault tolerance allows the accelerator to restore the beam after the failure of a single component without causing negative effects on the neutron source and reactor system (Abderrahim et al. 2021).

In the case of mYRRHA, the tolerance for the duration of the beam loss is 3 seconds and no more than 10 failures may occur within a 90-day operating cycle. Shorter beam interruptions are permitted without restriction (Abderrahim et al. 2021; Rimpault et al. 2013). Within this 3-second window, the thermal inertia of the lead-bismuth coolant protects the reactor from thermal transients and the spallation neutron source beam tube from thermomechanical stress. A beam interruption of more than 3 seconds leads to a reactor shutdown with corresponding thermal material loads and a subsequent, longer procedure to gradually increase the performance of the accelerator again. The aim is therefore to achieve an operating time of the accelerator of over 250 hours without beam interruptions.

With regard to the accelerator used, a linear accelerator was chosen, despite the higher construction costs compared to a circular accelerator. The main reason for this is the possibility of using fault tolerance systems in linear accelerators; cyclotrons have no modularity. This also means that a later increase in the beam energy is not a realistic option. A linear accelerator offers a high degree of modularity and has the potential to implement a fault tolerance system so that the beam can be restored within a short time and the beam energy can in principle be increased (Abderrahim et al. 2021). The disadvantage is the length of the accelerator of 400 metres. Further details on the individual technological systems of the accelerator and their development status are available at (Abderrahim et al. 2010).

The reactor part of the ADS plant is lead-cooled with a lead-bismuth alloy in eutectic equilibrium. The total weight of the lead is 7600 t (Abderrahim et al. 2019). All primary cooling systems and the two fuel element exchange systems are located inside the reactor vessel. The fuel element exchange systems are responsible for loading and unloading the fuel elements into the fuel element storage facility. The reactor vessel is located inside a reactor pit, which is provided with a steel casing that serves as secondary containment. In the gap between the steel-lined reactor pit and the wall of the reactor vessel there are vertical cooling pipes to protect the concrete wall and to remove any residual heat if the cooling systems fail.

In the reactor vessel, an upper hot low-pressure zone is separated from a lower, cold high-pressure zone by a partition made of two horizontal plates connected by vertical shells and containing cavities for four fuel element storage areas and several elements that connect the upper and lower zones. The reactor vessel is filled with the coolant made of lead-bismuth and with a cover gas between the surface of the liquid metal and the reactor vessel head, which is protected by a pressure relief system (Schulenberg 2020). Two fuel element handling machines permanently installed in the reactor take care of loading and unloading the fuel elements into the fuel element storage areas, one unloading machine for each side of the core. The upper zone contains the reactor core, through which the liquid lead-bismuth flows from bottom to top, which heats up and flows into the upper plenum. From the upper plenum it is pumped in 2 pipes via a pump per pipe into the lower plenum and releases the heat via two heat exchangers per pipe and cools down in the process. The pump is located behind the heat exchangers to minimise temperatures and thus corrosion of the pump. A maximum pressure of 2.5 bar is planned on the primary side, and cooling with light water with a pressure of 16 bar and a temperature of 200 °C is planned on the secondary side (Sarotto et al. 2013). The fuel element storage is cooled by the flowing lead during operation and by natural convection during shutdown. The four primary heat exchangers transfer the heat to a secondary system (Abderrahim et al. 2021).

The reactor core has a diameter of 1.80 m and a height of 1 m with an active height of 60 cm and consists of 211 positions for fuel elements, control rods, shutdown rods, the spallation window, reflector elements and instrumentation and observation capsules. In critical operation, 78 fuel

elements are planned at the start of operation, in subcritical operation 54, and fuel elements are added during the cycle, so that in equilibrium 108 or 72 fuel elements are used (Abderrahim et al. 2019). 55 of the positions, the multifunctional channels, are accessible from above the reactor vessel head (Abderrahim et al. 2021). Due to the lack of space due to the test and experiment devices and their positions in the core, fuel elements and reflectors are loaded into the core from below and are fixed vertically by buoyancy and radially by a retention system.

The total cycle is to be 540 days before a complete fuel element change takes place. In between, the fuel elements in the reactor are repositioned from the inside to the outside in 6 subcycles of 90 days each. A maintenance phase of 30 days is planned after each subcycle and a longer maintenance phase of 90 days after every 3 subcycles (Sarotto et al. 2013).

A more detailed description of the core and core loading of the mYRRHA FASTEF design is given in (Frieß 2017; Sarotto et al. 2013).

Fuel

A fuel element is 120 cm long and has an active height of 60 cm. The fuel consists of mOX fuel with typical fast reactor plutonium contents between 30-35%. The core is designed so that up to 6 fuel elements can be loaded with minor actinides. Typical compositions correspond to industrial ADS such as EFIT (see Chapter **4.7.3.2**) with 50% plutonium, 46% ameritium, 2% neptunium and 2% curium in an inert fuel matrix, although for practical reasons no curium is to be used in the fuel in mYRRHA (Abderrahim et al. 2021). The omission of curium is probably due to the difficulty of partitioning and subsequent fuel production. Due to their heat development and intensive neutron radiation, curium isotopes lead to problems, particularly in the manufacture of transmutation fuel elements, which have not yet been solved (Poinssot et al. 2012; Modolo et al. 2012; Renn 2014). Omitting curium would mean that americium would have to be separated from the highly active waste stream of the PUREX process and separated from the co-extracted lanthanides. For this option, several potentially suitable processes have been developed within the framework of European R&D projects and successfully tested experimentally in the laboratory (Modolo et al. 2012; Bollesteros et al. 2012; Rostaing et al. 2012). However, large-scale industrial implementation has not yet been demonstrated (Oeko-Institut e.V. 2021).

Coolant

The coolant of the reactor and the spallation neutron source is a lead-bismuth alloy in eutectic phase equilibrium. The temperatures are 270 °C at core entry and 410 °C at core exit (Frieß 2017). To avoid solidification, the temperature must be kept above 200 °C. A maximum pressure of 2.5 bar is planned. The choice of temperatures is based on limiting corrosion and, with the exception of the axial pumps, the pump speed is below 2 m/s to avoid erosive effects.

5.10.2 Safety concept and safety features

According to (Sarotto et al. 2013), the mYRRHA concept includes at least two main and one secondary system for each safety function. The three systems should be completely independent, redundant and diverse.

Buoyancy-controlled control rods are provided for controlling reactivity, which are also used for reactor emergency shutdown and are also used in subcritical accelerator-driven operation or during maintenance phases. The second system for reactor emergency shutdown consists of shutdown rods that are pneumatically introduced into the reactor from above against the buoyancy and then

remain in the reactor thanks to weights. Both systems contain bundles of boron carbide rods for neutron absorption (Sarotto et al. 2013).

Initial accident investigations have also been carried out, such as a leak in the heat exchanger with possible penetration of water vapour into the core, core compaction due to earthquakes, loss of coolant, the ejection of a control rod in critical operation or the breakage of the beam window of the spallation source in ADS operation (Sarotto et al. 2013).

The maximum output is specified as 100 MW, the primary, secondary and tertiary (air cooling) cooling systems are designed to transport an output of 110 MW, with the 10 MW being provided by the residual heat output of the spent fuel elements stored in the reactor vessel. (Sarotto et al. 2013).

Conclusions on safety

The safety properties described for the ADS technology line and their advantages and disadvantages compared to LWR in ADS operation also apply to mYRRHA (see chapter**4.7.5**). The safety properties of lead-cooled critical reactors described in the Chapter**4.2.5** can also be transferred to mYRRHA in critical operation.

Especially in subcritical ADS operation, mYRRHA has significant advantages over a critical LWR and LFR.

5.10.3 Supply and disposal aspects

For mYRRHA, there are no significant differences in the advantages and disadvantages with regard to supply and disposal compared to the ADS technology line.

Conclusions on supply and disposal

As many details of ADS concepts and the associated fuel cycles have not yet been determined at the current stage of the development process, many open questions remain regarding the analysis of the waste inventories to be disposed of and their composition.

Of all technology lines, ADS systems are the most suitable for use in transmutation scenarios due to the potentially high fraction of minor actinides in the fuel.

In terms of supply, ADS systems have advantages over LWR due to their high flexibility, especially due to the potentially high fraction of transuranic elements. In terms of disposal, ADS fuels have disadvantages compared to LWR due to their high thermal output.

5.10.4 Proliferation risks

There is currently no information available regarding the fissile material that is to be used after the construction of the mARRHA plant. Reprocessed plutonium is to be used for mOX production, but it is unclear whether the fuel elements should be reprocessed again or whether already reprocessed material will be used; the same applies to future experimental fuel elements for transmutation.

As mYRRHA is to be used explicitly to research transmutation, this research and development shares the basic proliferation risks of reprocessing technologies and mOX use, especially when separated plutonium is used.

Conclusions on proliferation

The proliferation risks from the spallation neutron source do not differ from the technology line (Chapter**0**). By using reprocessed fissile materials in mOX fuels, the proliferation risks are to be assessed as similar to those in the LWR fuel cycle, assuming reprocessing of plutonium and subsequent mOX use.

5.10.5 Technological development status

Research and development on mYRRHA has been underway for around 20 years and is largely carried out by the Belgian nuclear research centre SCK-CEN. The greatest development challenge is the use of lead-bismuth as a coolant. According to (Abderrahim et al. 2021), several test facilities for the use of lead-bismuth have been built and put into operation to qualify key materials and components in order to investigate metal chemistry, corrosion phenomena, thermohydraulic properties, instrumentation in liquid metals and components in liquid metals. These include the E-SCAPE facility (European SCAle Pool Experiment), a one-to-six-scale hydraulic model of mYRRHA to investigate the thermohydraulic properties of the liquid metal in a complex geometry and to validate computer models of fluid dynamics. Also included is the COMPLOT (COMPonents LOop Testing) facility, in operation since 2014, for isothermal hydraulic experiments on a 1:1 scale to simulate the entire hydraulic path of the mYRRHA core, from the fuel element to the spallation source to the control and shutdown rods and the instrumentation. The HEXACOM (Heat EXchAnger at COMplot) facility with a steam circuit that allows experiments on the secondary circuit of mYRRHA to investigate heat transport and heat exchange from the coolant to the secondary circuit and to obtain experimental data for model development and validation. Finally, RHAPTER (Remote HANDling Proof-of-principle TEst Rig), which has been in operation since 2011 to test the mechanical components immersed in the coolant (Abderrahim et al. 2021).

Furthermore, the chemistry of lead-bismuth is being investigated in the mEXICO (Mass EXchanger In Continuous Operation) experiment to test different control systems for regulating dissolved oxygen and to investigate the lifetime and efficiency of filtration systems for cleaning the lead-bismuth alloy. The HELIOS (HEavy LIquid metal Oxygen conditioning System) system is investigating the conditioning of the lead-bismuth alloy, but also potentials to deal with the possible consequences of a steam leak or air ingress (Abderrahim et al. 2021).

The main infrastructure for materials research is CRAFT (Corrosion Research for Advanced Fast reactor Technology), a facility for investigating long-term corrosion phenomena in lead-bismuth of materials used in mYRRHA under the appropriate conditions such as orbital speed, oxygen content and temperature of the alloy. The LIMETS (LIquid mEtals Test Stands) test stand is used to carry out mechanical tests in lead-bismuth environments on material-liquid metal interactions that affect the mechanical properties of materials (Abderrahim et al. 2021).

The test facilities do not take into account the influence of a fast, hard neutron spectrum on the material properties. This particularly applies to materials that are used directly in the core, such as the fuel cladding tubes. The fuel is to be surrounded by a coated fuel cladding, which is still under development. According to (Sarotto et al. 2013), the full 550 °C fuel cladding temperature can only be achieved with coating materials that are yet to be developed. One option is alloying using pulsed electron beams (Sarotto et al. 2013; Heinzel et al. 2006). The initial plan is to use austenitic 15-15/Ti(SS) steel as the cladding material, but this only allows cladding temperatures of 470 °C and therefore the reactor power must be reduced to 70 MWth when operating with such fuel elements.

But 15-15/Ti(SS) steel, which is used in sodium-cooled reactors, would also need to be further validated for the temperature range of 200-400 °C in order to investigate cladding corrosion and erosion, liquid metal embrittlement and precipitation of oxides (Sarotto et al. 2013).

Conclusions on technological development status

With regard to the technology line of lead-cooled fast reactors (Chapter **4.2.3.10**), mYRRHA is at a similar level of development to other lead-cooled systems and can be classified as being between “applied research” and “development”. The construction of the accelerator with an output of 2.4 MW has already been demonstrated several times using current technology (see Chapter **4.7.4**) and can be classified as being in “deployment”. Sufficient experience is also already available with spallation neutron sources with an output of 2.4 MW (see Chapter **4.7.4**) and the technological status can be classified as being between “development” and “deployment”.

With regard to coupling the systems with a reactor, mYRRHA is the first pilot plant and only a few experiments have been carried out so far. The development can therefore be classified as being between “applied research” and “development”.

5.10.6 Implementation

5.10.6.1 Planned area of application

MYRRHA is planned as a multi-purpose research reactor and should be considered a prototype plant in several respects.

- Further experience will be gained with a lead-bismuth (Pb-Bi) cooled reactor core. The mYRRHA reactor is thus used to gain experience and as a pilot plant for a European lead-cooled LFR.
- Furthermore, the ADS concept will be demonstrated by coupling the three components (accelerator, spallation target and subcritical reactor) at a suitable power in order to gain operating experience for scaling up to industrial application (van den Eynde et al. 2013).
- The fast neutron flux in the reactor, and its design as a multi-purpose reactor, allows various materials, including entire fuel elements, to be irradiated for test purposes and thus experimentally advance the development of suitable fuels and structural materials.
- The reactor serves to bundle European transmutation research.
- The reactor is intended to replace the ageing Belgian research reactor BR-2 and to conduct basic research into nuclear and atomic physics, solid state physics and nuclear medicine. The project is part of the European Strategy Forum for Research Infrastructures and is one of three new research reactors that, alongside the Jules Horowitz reactor in Cadarache (France) and the Pallas reactor in Petten (Netherlands), are intended to form a cornerstone of European material test reactors.

5.10.6.2 Project timeline

As early as the 1990s, the Belgian research centre SCK-CEN in mol was working on an accelerator-driven system in the ADONIS project. Various versions of mYRRHA (1998-2005) were part of successive European collaborative projects in the European Commission’s research framework program. In particular, the 2005 version of the design concept was used as the starting point for the XT-ADS design (2005-2009) (Abderrahim et al. 2011) within the EUROTRANS project (2005-2010) in the 6th Framework Programme. (van den Eynde et al. 2013). XT-ADS was intended as a fast-to-build (target date 2020) and small (50-150 MWth) experimental facility that demonstrates the technical feasibility of transmutation. The mYRRHA-FASTEF design (Abderrahim et al. 2011; Sarotto et al. 2013), which has been pursued since 2009, primarily in the 7thFramework Programme, is based on the predecessor concepts, but has a maximum output of 100 MWth. The facility is intended to permit use in both a critical configuration and as a subcritical, accelerator-driven facility. The FASTEF design is rated for 100 MWth, but this performance will not be achieved in mYRRHA due to a lack of materials that cannot withstand the temperatures at 100 MWth.

In 2013, the developers stated that the plant should be operational by 2024 (Sarotto et al. 2013; Englert et al. 2017). In 2015, the aim was to put mYRRHA into operation in 2026 (De Bruyn et al. 2015). In the 2015 plan, an initial licensing (pre-license) from the regulatory authority was to be obtained in an initial four-year phase, followed by a second licensing phase. As this is a completely new reactor system that is highly complex due to the ADS component, only partial use can be made of previous experience. The project was fully evaluated by the Belgian government at the end of 2014 (Abderrahim et al. 2015).

The latest schedule, which was adopted by SCK-CEN and the Belgian government in 2018, is divided into three stages. In the first phase, up to 2026, a LINAC (Project minerva) with up to 100 meV is to be built and the operational reliability of the LINAC confirmed. A target facility for the production of medical radioisotopes and for basic and applied physics research and materials research is also to be built, including a fusion target station in which materials for fusion reactors are to be tested. The first phase also includes research and development of the LINAC expansion to 600 meV and the subcritical reactor. Finally, the pre-licensing of the reactor is also part of phase 1. The further construction of the accelerator to an energy of 600 meV with a total length of 400 m will then take place by 2033. In parallel, the reactor with the spallation neutron source is to be built in a third phase with completion by 2036 (Abderrahim et al. 2019). Contracts worth EUR 7.6 million for the construction of a first building were signed in 2019.[201]

In 2020, the first RFQ (radio frequency quadrupole) component of the accelerator was connected to the existing low-energy beam path. The next steps at that time included fine-tuning the RFQ, upgrading the ion source amplifier, and developing and building an RF amplifier for the RFQ.[202] In 2022, the first superconducting radio frequency (SRF) cavity was welded, with another 59 of the elements to build the accelerator.[203] In December 2022, the construction permit for the future accelerator facility was granted.[204]

[201] https://world-nuclear-news.org/Articles/Myrhha-project-developers-sign-first-contract (as of 15/1/2023)
[202] https://myrrha.be/about-myrrha/myrrha-reactor (as of 15/1/2023)
[203] https://myrrha.be/news/technological-milestone-myrrha-1st-superconducting-cavity-ready (as of 15/1/2023)
[204] https://myrrha.be/news/update-minervamyrrha-1st-license-fact (as of 15/1/2023)

Conclusions on project progress

MYRRHA would be the first prototype of an ADS worldwide. Although the reactor is of strategic importance for the Belgian nuclear research programme and for experimental reactor research in Europe, the project has been the subject of decades of struggle to finance it. The situation has not improved significantly even after Belgium pledged to finance 40% of the total project costs. The plans for the construction of mYRRHA were finally split into three phases and only the financing of a 100 meV pre-accelerator for research purposes seems to be implemented. The first prototype facility for an ADS with accelerator, spallation neutron source and reactor cannot be expected to be completed before 2036, with corresponding effects on the timeline for the technology line. The delays also have an impact on the further development of lead-cooled critical reactors in Europe (see Chapter **4.2.3.2**), for which mYRRHA should also serve as an experimental reactor.

5.10.6.3 Costs

In September 2015, the Belgian government decided to include the mYRRHA project in the list of Belgian projects for the European Union's new investment programme with a sum of EUR 1.5 billion (SCK CEN 2015). In 2020, the media reported that Belgium was aiming to contribute 40% of the total amount of EUR 1.6 billion, with the remaining amounts to come from a partnership between the European Union, the European Investment Bank and other partners, with 70% of the remaining amount to be borne by EU countries. The participation of other countries, including Germany, has been discussed for some time. The Belgian research centre states that mYRRHA has been evaluated by (Renn 2014) and the report has been sent to the German government. Currently, the discussion about German participation is still ongoing (SCK CEN 2015). Japan also showed interest in principle in contributing 10% of the costs.

In 2018, the Belgian government approved EUR 558 million for the construction costs of the accelerator facility (Project minerva) until 2026 and subsequent operation until 2038.[205] The cost coverage for the subsequent phases of upgrading the accelerator and building the reactor still seems unclear.

Since the project was launched, a number of European and non-European research institutes have worked together and received research and development funding of around EUR 3 million annually. The main financier was the European Commission. In the Horizon 2020 programme, around EUR 11 million will be invested in mYRRHA, of which EUR 9 million will be funded by the European Commission (SCK CEN 2015).

Conclusions on costs

Since mYRRHA is intended to be a prototype for an ADS, and an experimental and research reactor at the same time, no conclusions can be drawn about the technology line of the ADS or a comparison with the LWR. As shown in the project history, financing of the entire project has not been successful for 10-20 years. Reasons for the lack of willingness to invest are not given. So far, there does not seem to be a constellation of government donors who would be willing to take the risk of financing.

[205] https://world-nuclear-news.org/Articles/Belgian-government-approves-funding-for-Myrrha (as of 15/1/2023)

6 Comparative summary of the principal advantages and disadvantages of the technology lines and reactor concepts

For all technology lines under consideration, extensive research and development work has been carried out for several decades, in some cases since the middle of the last century. Depending on the technology line itself, technical test benches for individual phenomena, smaller experimental reactors (for SFR, for example, the US EBR-I and II plants or the Russian BR-10 and Bor-60) and larger demonstration reactors (for the SFR, for example, the French Phénix and Super-Phénix plants or the Russian BN-350 or BN-600 plants) have already been built and operated. However, to date there is no commercially competitive reactor concept in the SNR field.

The most extensive technical experience is available for the SFR and VHTR technology lines. Their technical feasibility in terms of the construction and operation of a demonstration reactor for electricity production has been achieved. However, in order to fully realise the advantages associated with these technology lines compared to current LWR or to exclude possible disadvantages as far as possible, extensive technological developments and, in particular, proof of reliable operation under economic conditions are also required for SFR and VHTR. For the LFR and MSR technology lines, experience with experimental reactors (e.g. M) or from ship propulsion systems for submarines is available or is currently planned, but the technical feasibility in the form of a demonstration reactor is still outstanding. For the GFR, SCWR and ADS technology lines, on the other hand, there is no knowledge from smaller experimental reactors.

For the planning, licensing, construction and operation of such experimental and demonstration reactors, a period of at least one to two decades must be assumed for each reactor project, and probably considerably more based on historical experience. The knowledge gained with these facilities must be evaluated and incorporated into the technical design of a final prototype reactor.

Extensive programmes are necessary, particularly for the development or advancement of fuels and structural materials: basic laboratory and research work, testing of material properties in suitable test benches and the use of materials in research reactors with representative technical boundary conditions (temperatures, pressures, coolants, neutron spectrum, etc.), as well as subsequent post-irradiation examinations. Even with the use of extensive resources, such developments typically require a period of many years to several decades. As the demands on the materials, for example due to corrosion, increase with temperature and simultaneous irradiation, many development programmes also provide for several stages of development, with a phase that places fewer demands on today's materials and later phases at higher temperatures. Higher temperatures are a key point of economic considerations for the efficiency of power generation and the use of process heat.

In some cases, developers of specific reactor concepts plan to forego the intermediate step of building and operating experimental or demonstration reactors due to time constraints. However, this entails high risks, both in terms of the ability to obtain licensing and the actual later functionality and reliability of such reactor concepts. It is evident that developers' schedules are often based on overly optimistic assumptions, developments are delayed by years or even decades, and in many cases development approaches are completely abandoned because the underlying technological difficulties could not be overcome.

The additional time required for the development of SNR is therefore probably in the range of several decades even today. Against this background, it cannot be assumed that such reactor concepts will be used on a relevant scale by the middle of this century. Even the GIF assumes that the introduction of such reactor concepts, in addition to current LWR concepts, is not expected until the second half of this century.

The majority of SNR technology lines envisage reactor concepts with a fast neutron spectrum. In the area of safety, these technology lines generally have higher requirements for reactivity control, since the reactor core is typically not in its most critical configuration Thus – unlike with LWR – malfunctions or accidents that lead to a prompt supercritical state are also conceivable, but such accidents must be practically impossible.

Various alternative coolants discussed for SNR offer a number of favourable properties for heat removal. Helium gas, for example, is chemically and neutronically inert, liquid metals, for example, have good thermal conductivity and a high heat capacity. The primary cooling circuits in systems with liquid metal cooling are also not under high system pressure, which reduces the risk of major loss-of-coolant accidents. On the other hand, water is used widely as a coolant and its properties and the necessary technical systems such as pumps or fittings are therefore extremely well known and developed.

The concepts for containing the radioactive material are very similar in most SNR. Multiple sequential barriers are intended to prevent the release of radioactivity to the outside. The systems essentially differ in this regard due to the accident sequences that can occur in them and the resulting risks for the barriers used to contain the radioactive material. While certain event sequences such as loss-of-coolant accidents may only play a minor role in safety in SNR compared to LWR, other event sequences can determine the achievable level of safety. The chemical reactivity of the coolant (such as sodium in SFR) or the structural materials (such as graphite in VHTR) can play a role, but so can the altered requirements for reactivity control, particularly in fast reactors.

While fundamental safety-related advantages over current LWR are conceivable for individual technology lines, this should not be expected for other technology lines. However, many safety-related issues can ultimately only be analysed against the backdrop of a specific reactor concept, since the level of safety always depends on the specific technical details of a reactor concept. A final safety-related assessment is therefore only possible for a fully defined reactor concept.

The technology lines can achieve greater efficiency in power generation than current LWR due to higher operating temperatures. With regard to supply and disposal aspects, this could reduce the uranium required for energy generation and thus also the amount of highly radioactive waste by probably several tens of percent.

A higher utilisation of the uranium resource is conceivable through the production and further use of plutonium; alternatively, uranium-233 as a fissile material could be obtained by using thorium. Reprocessing of spent fuel and using plutonium, and possibly minor actinides, could also reduce the total inventory of actinides that must be placed in a geological repository. However, the reprocessing and fuel production technologies required for this entail considerable safety and proliferation risks of their own, depending on the specific technical design.

Only in the case of very long-term use of nuclear energy, or extreme expansion of nuclear energy, would such improved utilisation of the known uranium reserves be theoretically necessary. Under current and expected future conditions, such reprocessing and fuel production leads to higher costs

for the required fuels. A significant reduction in the requirements for a geological repository cannot be expected, in any practical sense, by reducing the actinide inventory.

Some of the SNR use different fuels, coolants and moderators than LWR. This affects the quantities and the chemical and radioactive properties of the waste generated. Due to the targeted separation of different waste streams (gaseous fission products, lanthanides, metals) in some technology lines, and the use of other fuel cycle technologies, specific conditioning processes are used or need to be developed for different waste streams.

With a view to the different types of waste streams, the SFR, MSR and VHTR technology lines were examined in more detail, in particular. SFR, LFR and GFR differ mainly in terms of the coolant they use. The waste volumes from spent fuel and structural materials in the LFR and GFR have few special features compared to the SFR. In the GFR, the coolant has no influence on disposal. The special feature of lead-containing coolant in the LFR is the high weight of the coolant. The discussion of the SFR therefore covers the other two technology lines. In the SCWR, the technology line is very similar to the LWR. The ADS is designed as lead-cooled system and is very similar to the LFR. All systems that have improved efficiency due to higher operating temperatures share the property that, in relation to the electrical energy generated, fewer radioactive fission products are produced. With regard to the fraction of fission products in spent fuel elements from SNR, there is no significant difference in terms of disposal compared to the highly radioactive waste from LWR.

In particular, the use of other coolants (sodium) and moderator materials (graphite) leads to larger quantities of medium and high radioactive waste compared to LWR. Waste treatment is made significantly more difficult by the chemical and radiological characteristics of the coolants or moderator materials compared to light water. Many waste treatments provide for the splitting of waste streams because direct disposal is not possible. This results in costs due to complex processes with additional secondary waste. In particular, MSR with liquid fuels require a large number of waste treatment, stabilisation and conditioning processes. Due to the fast neutron spectrum, fast reactor concepts have a higher volume of activated reactor elements and structural components than LWR. Additional waste streams of low and medium-level radioactive waste are also generated by the reprocessing of the fuel, which is planned in many fuel cycle scenarios for the use of SNR.

In various technology lines, plutonium or a mixture of plutonium and minor actinides from the reprocessing of spent fuel is to be used as fissile material instead of enriched uranium. This shifts the proliferation risks from the uranium enrichment plants to the reprocessing and fuel production plants and to the transport of fresh fuel. As long as the use of enrichment technologies cannot be completely dispensed with, additional diversion paths and thus new proliferation risks are incurred.

Only individual reactor concepts, such as the TWR, boast the stated aim of being able to dispense with both the enrichment of uranium, and the reprocessing of spent fuel, by producing and using the fissile material in situ. This could significantly reduce proliferation risks compared to current LWR. However, due to its association with particularly high technological requirements, particularly with regard to the fuels, the TWR is currently only seen by the developers themselves as a long-term development goal.

Current LWR are not competitive with the renewable power generation technologies available today in terms of their levelised cost of electricity. The historical cost development also shows that LWR have increased levelised costs of electricity over time, while in the area of renewable energies, costs have fallen massively, particularly in the last two decades. There are no discernible reasons why this trend should be reversed in the future.

Individual SNR technology lines, such as SCWR or VHTR, could achieve certain economic advantages over current LWR in industrial mass production. The reasons for this are possible savings in the area of investment costs, greater efficiency in electricity generation due to higher operating temperatures and the utilisation of process heat. The extent to which cost advantages can actually be realised in concrete reactor concepts, compared to current LWR, is currently still an open question. In particular, for reactor concepts with high operating temperatures, new materials still need to be developed and their usability proven. Cost estimates are also subject to high levels of uncertainty due to the fact that SNR are still in a very early stage of development. Overall, however, it cannot be assumed that the cost advantages, which can be achieved through SNR, will compensate for the cost disadvantages of current LWR, compared to other power generation technologies, or even turn them into a cost advantage.

General conclusions

The discussion makes it clear that individual technology lines – if designed consistently – could achieve potential advantages over current LWR in individual evaluation criteria (safety, supply and disposal, proliferation, costs). At the same time, however, no advantage over current LWR can be expected for any of the technology lines in all areas; disadvantages compared to current LWR are also possible in some areas.

Different criteria also compete with one another, and improvements in individual areas can lead to disadvantages in other areas. Increasing the safety of a specific reactor concept is often accompanied by additional costs or the need to develop new, improved materials or technologies, and thus increased development effort and time required until it can be implemented. Advances in the area of supply and disposal can lead to new safety risks or an increase in proliferation risks.

Developers of the SNR repeatedly refer to intrinsic properties of technology lines or their reactor concepts in order to justify clear advantages in individual areas compared to current LWR, for example, by excluding event sequences that are of particular importance for safety in current LWR; by the possibility of generating new fissile material during operation of a reactor; or due to higher efficiency of electricity production through high coolant temperatures or the use of process heat.

Such intrinsic properties can indeed lead to advantages over current LWR. However, the differences between SNR and current LWR, which are the cause of such intrinsic properties, typically lead to new technological challenges or disadvantages in other areas. This can lead to other event sequences that are equally relevant for safety, such as the risk of prompt supercritical plant conditions in technology lines with fast neutron spectrum or the risk of graphite fires in VHTR; the fissile material obtained must be capable of being reused, which can lead to proliferation risks, and higher coolant temperatures require new suitable materials that can withstand the increased demands during operation.

Ultimately, only a detailed analysis of a specific reactor concept can provide a final assessment of the extent to which intrinsic properties of the technology line in conjunction with the specific design of a reactor concept will lead to advantages or disadvantages in the individual evaluation criteria.

The expectation, often expressed in public discussions and by developers themselves, that SNR can make a significant contribution to solving today's problems of nuclear technology in the areas of safety, supply and disposal, nuclear non-proliferation and economic efficiency, cannot be considered realistic overall in view of the current state of development of these systems and the actually proven and expected advantages and disadvantages of the individual technology lines.

7 International and selected national regulatory developments affecting the SNR

The aim of the following study is to analyse the regulatory framework for the introduction of so-called “novel” reactor concepts (SNR). For this purpose, selected international and national regulations are analysed for their applicability with regard to the SNR and the existence of regulatory gaps, on the basis of the safety characteristics identified in Chapter **4** and 5 and, in particular, open safety-related issues. Safeguard requirements are also taken into account.

7.1 Introduction

The regulatory requirements for existing and new nuclear power plants are continuously being developed internationally. In particular, after the catastrophic accident at the Fukushima-Daiichi nuclear power plant, important safety principles were examined and revised by national regulators, as well as international organisations. With regard to the SNR, this includes in particular the concept of defence in depth – for example, with regard to reactivity control in fast reactors – and multi layer containment for the radioactive inventories (barrier concept, defence in depth) – for example, with regard to reactor concepts with a liquid fuel. A particular focus is also on questions of the implementation of principles of redundancy, diversity and independence of safety-related facilities. The need to take a comprehensive range of events into account when designing nuclear power plants for safety reasons was also reaffirmed. With regard to the SNR, the relevant event sequences (reactivity excursions, loss-of-coolant accidents, plant-internal events such as fires, chemical reactions of coolant or moderator, etc.) must be identified and safety must be demonstrated. SNR-specific problems can also arise with regard to providing a safety demonstration.

Discussions on regulations take place both within the International Atomic Energy Agency (IAEA) and within the European framework, for example, at the Western European Nuclear Regulators’ Association (WENRA). Basic questions relating to the licensing of the SNR have been discussed in the USA for many decades, but there is also a discussion in other countries relating to new questions surrounding the possible licensing of so-called “novel” reactor concepts.

At the international level, Chapter 7.2 first analyses the existing IAEA regulations with regard to their applicability to new reactor concepts, see Chapter 7.2.1. Furthermore, developments in regulations for application to new reactor concepts that are currently being worked on at the IAEA are presented (IAEA 2023b). A special emphasis is placed on the level of Specific Safety Requirements and Specific Safety Guides in order to examine the existence of specific requirements connected with the SNR at this level. In addition, discussions are being held at the level of international organisations, such as the OECD/NEA.

At the European level, the work of WENRA should be mentioned, see Chapter 7.3. The statement “Safety of new NPP designs” by the Western European Nuclear Regulators’ Association (WENRA) was published in March 2013 and is based on the report of the same name by the “Reactor Harmonisation Working Group” (WENRA 2013). The WENRA/RHWG report describes seven “safety objectives” and formulates statements on selected key safety issues and on lessons to be learned from the accident in Fukushima-Daiichi. The report is intended to serve as a basis for the further development and further harmonisation of WENRA’s safety requirements. The EU's Joint Research Centre has also prepared a paper on the question of the applicability of existing regulations.

Chapter 7.4 also evaluates which national licensing and supervisory authorities are already working on licensing or licensing requirements for the SNR in selected countries.

To this end, the regulatory developments in the USA (Chapter 7.4.1), Canada (Chapter 7.4.2) and the United Kingdom (Chapter 7.4.3) are examined in detail.

Should gaps in the regulations or the need for adjustment have been identified by international institutions or within countries, it will also be examined whether measures such as conducting research projects, further developing computation models or simulation codes or tests to validate existing modelling approaches have been derived from this, and whether and how such measures are planned specifically.

7.2 International regulatory developments

This section discusses international developments within further development of nuclear regulations and harmonisation, with a particular focus on developments related to the SNR.

7.2.1 International Atomic Energy Agency (IAEA)

The safety concept of current nuclear power plants serves to protect people and the environment from the harmful effects of ionising radiation and to minimise the possibility of accidents. The SNR must also meet this goal. The requirements of the nuclear regulations of the International Atomic Energy Agency (IAEA), the “Safety Standards”, are among those used to meet this goal.[206] They were developed as a consensus of the IAEA member states. The experience of the member states with the previous fleet of predominantly land-based, water-cooled reactor concepts was taken into account.

The IAEA also dealt with the issue of SNR regulation early on, see e.g. (IAEA 1980). Starting in the 2000s, it increasingly dealt with general questions about SNR regulation (IAEA; NEA 2002). In doing so, it also produced summary technical reports on specific individual aspects of individual technology lines, see e.g. (IAEA 2020d).

In view of international developments towards the introduction of non-water-cooled reactor concepts, including the SMR, the IAEA has examined the applicability of the existing regulations and also examined the question of whether gaps and applicability issues can be identified in the existing regulations with regard to the new phenomena or technologies in non-water-cooled reactor concepts (IAEA 2022e). The results of this investigation are summarised in the following section. Following this, current changes to the IAEA's regulations will be discussed and two working groups that emerged from the IAEA, the “Nuclear Harmonisation and Standardisation Initiative (NHSI)” and the IAEA's “SMR Regulators' Forum”, will be presented.

[206] The IAEA's nuclear regulations, in the sense of the study carried out here, comprise the three levels of Safety Fundamentals, the (General and Specific) Safety Requirements and the (General and Specific Safety Guides).

7.2.1.1 Applicability of the previous IAEA regulations to the SNR

The report (IAEA 2022e) provides a comprehensive overview of the applicability of the previous regulations with regard to the SNR and SMR.[207] Differences in the supply and disposal of the corresponding plants are also discussed. The results of the report were presented as part of (Vives et al. 2022). The terms used in (IAEA 2022e) are based on (IAEA 2021b). The report covers

- Water-cooled small modular reactors (SMR),
- Sodium-cooled fast reactors (SFR) including SMR-SFR,
- Lead-cooled fast reactors (LFR),
- Very high temperature reactors (VHTR),
- Molten salt reactors (MSR) and
- Transportable nuclear power plants (TNPP).

The GFR, SCWR and ADS are therefore not covered by the report. With regard to molten salt reactors (MSR), the report limits the scope of testing to the “Specific Safety Requirements, Safety of Nuclear Power Plants: Design” (IAEA 2016d), more detailed requirements from other safety standards were therefore not tested. (IAEA 2022e, p. 2).

The key findings with regard to the SNR are summarised below. Aspects that are only relevant for water-cooled SMR or TNPP are not discussed below.

The IAEA's nuclear regulations typically comprise three levels. At the Safety Fundamentals level, basic safety principles are defined. At a second level, the Safety Requirements define requirements with which the basic safety principles can be implemented. At a third level, the Safety Guides, options for fulfilling the Safety Requirements are shown. Within the Safety Requirements and the Safety Guides, a distinction is made between General Safety Requirements (and Guides) and Specific Safety Requirements (and Guides). Since the General Safety Requirements (and Guides) are intended to be applicable to all activities and systems, (IAEA 2022e) assumes that they are fully applicable to the SNR and SMR. An examination is therefore only carried out for Specific Safety Requirements (and Guides) that relate to specific activities and systems.

The report structures the SNR requirements according to the various phases of a plant's life cycle, i.e.

- Site selection,
- Design and construction,
- Commissioning and operation, and
- Decommissioning.

[207] A final version was published as (IAEA 2023a).

In addition to these phases of the life cycle of a nuclear power plant, the report takes into account effects on

- Fuel cycle systems,
- Radiation protection aspects,
- Disposal of radioactive waste,
- Safety management,
- Safety analyses,
- Emergency protection,
- Regulation, and
- Transport safety.

It achieves this by first identifying new properties (deviations) of the SNR and SMR compared to current high-performance, land-based, water-cooled reactor concepts.

The report (IAEA 2022e) identifies deviations and groups them into

- Fundamental deviations (independent of specific technology lines or performance sizes),
- Deviations due to the coolant (for non-water-cooled technology lines),
- Deviations due to the performance size, modularity or multi-block character of systems (typical for SMR concepts) and
- Deviations due to transportability (typical for SMR concepts).

As the investigations carried out here do not give further consideration to SMR concepts that are not part of the SNR, deviations in the last two groups are not discussed further.

For example, with regard to site selection, (IAEA 2022e) notes that open questions have been identified, particularly for the SMR and TNPP. These are not of primary importance for the SNR. For the VHTR, it is pointed out that such plants are also being developed for the direct provision of process heat. In this case, the interactions between the nuclear plant and the process plant must be taken into account.

With regard to the operation of the SNR, the authors of (IAEA 2022e) in (Vives et al. 2022) refer primarily to the intensive use of passive systems and design principles. This raises questions, for example, about the availability of passive systems in plant states with open containment and the verifiability of the function of passive systems under all applicable environmental conditions. Various SNR provide for significantly longer fuel cycles than has previously been the case. With a view to new fields of SNR application, an increased use in the load sequence or, additionally, in plant states with low output or longer shutdowns (for example in plants that would primarily serve to supply district heating) should be expected.

Building on this, (IAEA 2022e) identifies the safety standards assigned to the respective phases mentioned above (see Appendix II in (IAEA 2022e)), and also checks their applicability and completeness on the basis of their deviations from the current reactor concepts identified in the first step. In a further step, the report takes into account possible interactions between the topics of safety, security and safeguards (3S). Specifically for the SNR, the IAEA sees particular challenges in the areas of safety, security and control of fissile material, which influence each other, but also the potential to address these through early and joint consideration in the design, see e.g. (IAEA 2022j). The early and joint consideration of safety, security and control of fissile material, is necessary so as to prevent these factors from impacting each other, for example, as laid out in Requirement 8 of (IAEA 2016d).

In the case of identified gaps and applicability issues in the existing regulations, see the following summary, the report neither assesses the relevance of these gaps nor develops proposals for closing the corresponding gaps.

Regulatory documents with identified need for change

(IAEA 2022e, Appendix II) summarizes the identified need for change in relation to the analysed regulatory documents in a table. A distinction is made between

- No identified applicability problems,
- A small number of identified applicability problems,
- Some identified applicability problems and
- Numerous identified applicability problems (multiple aspects or more than a third of the document affected).

The following Table 7-1 summarises the regulatory documents with numerous applicability problems, Table 7-2 for those with some applicability problems, for regulatory documents with no or only a small number of applicability problems, reference is made to (IAEA 2022e, Appendix II). The following only lists those regulatory documents whose essential limitations are not exclusively due to properties of SMR (such as factory production or multi-module construction) or TNPP (such as safety issues during transport).

Table 7-1: Regulatory documents with numerous applicability problems according to (IAEA 2022e)

Name	Number	Subject	Limitation
Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants	SSG-4	Safety Assessment	Non-WCR; Software and passive system reliability
Design of the Reactor Core for Nuclear Power Plants	SSG-52	Design and Construction	Non-WCR
Design of the Reactor Containment and Associated Systems for Nuclear Power Plants	SSG-53	Design and Construction	Non-WCR; TNPP
Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants	SSG-56	Design and Construction	Non-WCR
Design of Fuel Handling and Storage Systems for Nuclear Power Plants	SSG-63	Design and Construction	Non-WCR; TNPP

Source: (IAEA 2022e, Appendix II), Non-WCR: Non water-cooled reactors (SFR, LFR, VHTR, MSR)

Table 7-2: Regulatory documents with some applicability problems according to (IAEA 2022e)

Name	Number	Subject	Limitation
Safety of Nuclear Power Plants: Design	SSR-2/1 (Rev. 1)	Design and Construction	Non-WCR; TNPP
Safety of Nuclear Power Plants: Commissioning and Operation	SSR-2/2 (Rev. 1)	Commissioning and Operation	Examples are non-WCR; remote operation; multiple modules
Deterministic Safety Analysis for Nuclear Power Plants	SSG-2 (Rev. 1)	Safety Assessment	Non-WCR
Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants	SSG-3	Safety Assessment	Non-WCR; Software and passive system reliability
Storage of Spent Nuclear Fuel	SSG-15 (Rev. 1)	Waste and Spent Fuel Management	Novel fuel types; HALEU fuel
Design of Electrical Power Systems for Nuclear Power Plants	SSG-34	Design and Construction	Passive systems; Multiple modules
Design of Fuel Handling and Storage Systems for Nuclear Power Plants	SSG-39	Design and Construction	Special features
Predisposal Management of Radioactive Waste from Nuclear Fuel Cycle Facilities	SSG-41	Waste and Spent Fuel Management	Novel wastes
Accident Management Programmes for Nuclear Power Plants	SSG-54	Commissioning and Operation	Non-WCR
Design of Auxiliary Systems and Supporting Systems for Nuclear Power Plants	SSG-62	Design and Construction	Non-WCR; Passive systems
Protection Against Internal Hazards in the Design of NPP	SSG-64	Design and Construction	Non-WCR

Source: (IAEA 2022e, Appendix II), Non-WCR: Non water-cooled reactors (SFR, LFR, VHTR, MSR)

As of August 2023, of the rules mentioned in Table 7-1 and Table 7-2 only rules SSR-2/2 (Rev. 1), SSG-3, SSG-4 and SSG-12 are currently undergoing a process of revision, see also Chapter 0.

Conclusion of the evaluation in (IAEA 2022e)

The review of the applicability of the IAEA regulations revealed a number of applicability questions, especially with regard to the design and safety proof of the SNR.

On the one hand, applicability questions can manifest themselves in the fact that previous regulations are not formulated in a technology-neutral manner and can therefore only be directly applied to water-cooled reactor concepts. In these cases, it appears that supplementing the regulatory texts in (IAEA 2022e) with a technology-neutral formulation would, in principle, be appropriate.

In addition, applicability questions were also identified that are due to innovations in the SNR for which corresponding requirements are missing in the existing regulatory texts. While, for some of these innovations, there may already be a certain amount of operating experience from experimental

and prototype reactors in order to formulate corresponding requirements, this is not yet the case for other innovations. In such cases, it is expected that the development of international requirements will take significantly longer.

(IAEA 2022e) summarises the following points as the main results with regard to the SNR:

- Design:

  In the SNR, new types of fuels, new types of coolants and new types of containment concepts, as well as intensive use of passive systems, are being discussed. This gives rise to questions, in particular, regarding the interpretation of previous terms, such as “serious accident”, “beyond-design-basis conditions” and the meaning of the defence in depth concept for safety levels 4 and 5. For example, the IAEA defines a “serious accident” as an event that is accompanied by significant core damage. In molten salt reactors, however, significant quantities of the molten salt fuel are not only present in the core area during normal operation, but also in other structures. For VHTR or MSR concepts, for example, beyond-design-basis conditions do not necessarily have to be accompanied by a core meltdown.

  There are also uncertainties regarding the range of events that need to be assumed and the reaction of plants to the corresponding triggering events. Phenomena, error mechanisms or risks can also exist for the SNR that do not exist for water-cooled reactor concepts.

- Commissioning and operation:

  The increased use of passive systems gives rise to questions regarding proof of the reliable availability of these systems. The function of passive systems is typically based on the presence of large temperature or pressure differences. However, it can be difficult to create these system states for the purpose of commissioning tests or periodic tests in the system. Furthermore, questions arise due to the use of new chemicals. This places special demands on the licensing of prototype systems (FOAK).

- Fuel cycle systems:

  The existing regulations address current fuel production systems. With regard to new fuels and the reprocessing of fuels from the SNR, however, questions arise for which specific regulations would have to be developed. Basic requirements from (IAEA 2017b) are, however, applicable according to (IAEA 2022e).

- Safety analyses:

  Previous terms such as “design-basis accident”, “beyond-design-basis conditionsor “serious accident” may not be directly transferable to the SNR, for example if core meltdowns or serious accidents are no longer to be assumed for these types. There are also questions about how deterministic and probabilistic detection methods should be applied to the SNR.

  A key problem in safety analysis is the limited knowledge of relevant, new phenomena to be considered and a lack of experimental or operational experience in this regard; new risks (e.g. due to other chemicals) have not yet been adequately addressed in the regulations.

There are also no approaches for evaluating new safety functions. This applies, for example, to the evaluation of a functional containment consisting of the properties of TRISO fuels, passive safety properties for heat removal and the protection of the fuel against external influences using the building structures, among other things.

- Radiation protection:

  With regard to radiation protection, questions arise in particular with regard to the possible source terms in the SNR and operational radiation protection during maintenance work and dismantling.

- Disposal of radioactive waste:

  In SNR, new types of spent fuel and other new types of radioactive waste (e.g. activated coolants) will be created, for which specific requirements would have to be developed. In addition, questions may arise with regard to the conditioning of this waste for later final disposal, the interim storage of this waste and facilities for fuel treatment on the plant site (e.g. at MSR).

  In this regard, (IAEA 2022e) notes that SNR developers have so far paid little attention to the aspects of dealing with radioactive waste and spent fuel from the SNR, see also Chapter **4**.

Finally, (IAEA 2022e) points out that new issues arise at the interface between safety, security and safeguards, which should be taken into account at an early stage of the design, using a coordinated approach, in order to avoid the need for more complex measures later. The reasons for this are differences

- In the size of the fuel elements (e.g. TRISO spheres or SFR fuel elements),
- In the fuels used (e.g. in fuels with higher enrichment or a higher fraction of plutonium in the fresh fuel than in current uranium or MOX fuels for light water reactors) and the planned reprocessing of fuels,
- In possible access restrictions during the operation of plants,
- Through continuous fuel changes (e.g. in pebble bed reactor concepts) or online refuelling (e.g. in MSR) and, related to this,
- Through a lack of measurement and monitoring technologies (in VHTR and MSR for balancing the fuels that are introduced into or removed from the plant, as well as their respective fissile material content).

7.2.1.2 Status of revision of IAEA rules

The current structure and status of the IAEA's rule development are described in (IAEA 2023b). Accordingly, as of August 2023, a total of 29 drafts are under development, 24 of which are for the revision of existing standards and five for the creation of new standards. The following Table 7-3 summarises those SSR and SSG that are in the process of being amended or that are to be created and for which there is an applicability to nuclear power plants (NPP) and fuel cycle facilities (FCF). Regulations that refer exclusively to research reactors or facilities are not reproduced. Regulation texts, for which (IAEA 2022e) identifies some or numerous applicability problems, are marked with “yes” in the last column, see Chapter 0.

The review of the applicability of the nuclear regulations to SNR (IAEA 2022e) carried out by the IAEA has initiated various revision processes or expanded their content. (Vives et al. 2022) first of all, refer to a revision of the IAEA Specific Safety Guide SSG-12 “Licensing Process for Nuclear Installations” (IAEA 2010b). The current revision DS539 (IAEA 2022b) aims, among other things,

- to regulate the cooperation of different national regulatory authorities in the licensing of SNR or SMR and
- the procedure for the licensing of a first reactor of an SNR or SMR (FOAK), as well as
- to provide recommendations for reducing regulatory hurdles for SNR or SMR developers in the case of simultaneous licensing procedures in several countries or with regard to the subsequent licensing of a design in another country after an initial design licensing in another country.

The comment phase by the member states is to take place in the third quarter of 2024. The preparation of the revised version is to be completed by the first quarter of 2026 (publication date).

Table 7-3: IAEA rules for the revision process/in preparation as of August 2023

Name	Number	DPP	Step	Need for modification according to (IAEA 2022e)
Specific Safety Requirements				
Safety of Nuclear Power Plants: Commissioning and Operation (2016)	SSR-2/2 (Rev. 1)	DS532	Step 5	Yes
Specific Safety Guides				
Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations (2011)	SSG-18	DS541	Step 5	
Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plant (2004)	NS-G-3.6	DS531	Step 7	
Radiation Protection Aspects of Design for Nuclear Power Plants	NS-G-1.13; NS-G-2.7; NS-G-4.6; WS-G-2.1	DS524	Step 11	
Licensing Process for Nuclear Installations (2010)	SSG-12	DS539	Step 5	Yes
Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants (2010)	SSG-3	DS523	Step 12	Yes
Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants (2010)	SSG-4	DS528	Step 8	Yes
Assessment of the Safety Approach for Design Extension Conditions and Application of the Practical Elimination Concept in the Design of Nuclear Power Plants	New	DS508	Step 12	
Safety Demonstration of Innovative Technology in Power Reactor Designs	New	DS537	Step 12	
Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for Nuclear Power Plants (2002)	NS-G-3.2	DS529	Step 6	
Periodic Safety Review for Nuclear Power Plants (2013)	SSG-25	DS535	Step 5	
Evaluation of Seismic Safety for Existing Nuclear Installations (2009)	NS-G-2.13	DS522	Step 12	
Chemistry Programme for Water Cooled Nuclear Power Plants (2011)	SSG-13	DS525	Step 10	
Safety of Nuclear Fuel Reprocessing Facilities (2017)	SSG-42	DS518a	Step 9	
Nuclear Security Guidance - Safety-Security Interface				
Management of the interfaces between nuclear and radiation safety and nuclear security	New	DS533/ NST067	Step 5	
Security by Design	New	NST071	Step 5	
Evaluation of Physical Protection Systems at Nuclear Facilities	New	NST029	Step 10	
Engineering Safety Aspects of the Protection of Nuclear Power Plants against Sabotage	NSS No. 4	NST063	Step 13	

Source: (IAEA 2023b), the revision processes are divided into 14 formal “steps” (Step 1-14), for a description of the individual steps see (IAEA 2023b, p. 5), DPP: Document Preparation Profile (definition of rule creation or revision process)

Furthermore, (Vives et al. 2022) point to a revision of the IAEA Specific Safety Requirements SSR-2/2 (Rev. 1) “Safety of nuclear power plants - Commissioning and operation” (IAEA 2016c). The revision is intended to record, among other things, the effects of new properties such as

- The need to be able to check the availability of passive (safety) systems,
- New chemical properties or risks (hazards) for commissioning and operation, and
- The need to (re-)define “severe accidents” in plant concepts where it does not make sense to assume the occurrence of a “core melt”.

This should expand the previous revision DS532 of SSR-2/2, in accordance with (IAEA 2022d), to include findings from (IAEA 2022e), where necessary. The comment phase by the member states should take place in the fourth quarter of 2023. The preparation of the revised version is to be completed by the third quarter of 2026 (publication date).

Finally, (Vives et al. 2022) has proposed the preparation of a new Safety Guide with the working title “Safety Demonstration of Innovative Technology in Power Reactor Designs”. A corresponding reversion DS537 has been initiated (IAEA 2022c). Key aspects of this revision include

- Recommendations for dealing with new technologies for which there is currently no operational experience comparable to that of current light water reactors.

This can involve completely new technologies, as well as technologies that have previously only been used in the non-nuclear industry, or technologies that have already been used in nuclear systems but are now to be used in other fields of application or with other application conditions.

The comment phase by the member states is to take place in the third and fourth quarters of 2024. The revised version is to be completed by the second quarter of 2026 (publication date).

The revision of the IAEA Specific Safety Guide SSG-3 “Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants” from 2010 (IAEA 2019a) is based on changes in the associated Specific Safety Requirements and new developments in the member states of the IAEA.

The justification for the revision does not address the development of the SNR. However, it does refer to aspects that are explicitly related to the SNR, in particular the reliability of passive systems.

The revision of the IAEA Specific Safety Guide SSG-4 “Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants” from 2010 (IAEA 2020c) is based on changes in the associated Specific Safety Requirements and new developments in the member states of the IAEA and includes aspects, such as the introduction of Severe Accident Management measures or new findings on the course of severe accidents. The revision does not currently cover any aspects that are explicitly related to the SNR.

In addition to these regulatory processes, which are connected to the findings from (IAEA 2022e), a new Specific Safety Guide “Assessment of the Application of General Requirements for Design of Nuclear Power Plants” is being developed (IAEA 2016a). This is not directly related to the

development of the SNR, but addresses the topics of beyond design-basis plant conditions -basis and the practical exclusion of early or large releases. This guide therefore also addresses topics relevant to the SNR. The comment phase by the member states took place in the third quarter of 2019. The revised version should be completed by the fourth quarter of 2020 (publication date); no new information on its current status is available.

For the other regulatory texts that are currently being revised, the justifications for the revision process generally refer to changes to higher-level regulatory texts and experiences from the catastrophic accident at the Fukushima-Daiichi plant. There are no indications to the need for revisions due to the development of the SNR.

7.2.1.3 Nuclear Harmonization and Standardization Initiative (NHSI) of the IAEA

The IAEA's “Nuclear Harmonization and Standardization Initiative (NHSI)” (initiated in 2022) aims to increase the harmonisation of activities in the nuclear industry and among regulators. The initiative aims to bring together leaders from national governments, regulators, developers, industries, operators, potential users and other international organisations to advance the global development of safe and secure advanced nuclear reactors (IAEA 2023f).

As a result, technical reports will be published and networks of cooperation, particularly between technology providers and developers, will be established to enable further standardisation of industrial approaches to the design, manufacture, construction, commissioning and operation of novel reactors, as well as of associated requirements and criteria. In particular, solutions will be developed for increased regulatory cooperation towards globally harmonised pre-licensing reviews and for multi-national reviews of selected designs. To this end, two work streams will be pursued, the NHSI Regulatory Track and the NHSI Industry Track.

The NHSI Industry Track primarily deals with the topics of “harmonising higher-level user requirements”, “common approaches to codes and standards”, “experimental testing and validation of design and safety analysis computer programmes” and “accelerating the establishment of a nuclear infrastructure for SMR”.

The NHSI Regulatory Track currently includes the working groups “establishing a framework for information exchange”, “developing a multi-national regulatory pre-licensing design review” and “developing approaches to exploit the potential of further regulatory reviews and supporting those regulators who conduct joint reviews”.

In June 2024, the initiative will present the results it has achieved (IAEA 2023e).

7.2.1.4 IAEA SMR Regulators’ Forum

The SMR Regulators’ Forum is a discussion forum of regulatory and licensing authorities from Canada, China, Finland, France, Saudi Arabia, South Africa, South Korea, Russia, the United Kingdom and the United States, with the aim of identifying and improving understanding of future regulatory challenges and decisions that may arise for authorities in connection with SMR projects. This is intended to support efforts to ensure the safety and security of these reactors and to ensure efficient regulatory requirements and procedures, and to help authorities adapt their requirements and practices where necessary. Although the Forum deals explicitly with SMR, the SMR also includes many SNR concepts worldwide, so the Forum’s discussions are summarised here.

The recommendations of Phase 2 of the Forum’s activities (from November 2017 to December 2020) can be summarised as follows (SMR Regulators' Forum 2021d; 2021b; 2021a; 2021c):

- The IAEA should work with the Forum to review the extent to which there are gaps in the existing regulations and potential for improvement. Potentials are identified primarily with regard to the parallel operation and parallel construction of several units/modules and the specific risks associated with this (taking into account shared safety-relevant facilities), as well as risks for facilities that are not used for electricity production and/or are to be located close to other industrial facilities.
- In order to develop a consistent understanding of the application of regulations to SMR, pre-licensing engagements should be established between manufacturers or future operators and the regulatory authorities, as well as an exchange between the authorities.
- In view of the increasing importance of globally operating suppliers for SMR projects, the IAEA should increasingly develop regulatory activities and requirements in this regard.
- The IAEA should develop rules on how an infrastructure can be established to ensure feedback on SMR, covering not only aspects of reactor operation, but also manufacturing, construction and commissioning. The existing International Reporting System (IRS) for operating experience should also be reviewed in this regard.
- The IAEA should provide rules for dealing with the range of passive facilities that may be possible in the future, which should also include expectations regarding the separate justifications to be provided for inherent features of a reactor plant, its passive properties and active elements and their interaction. The safeguarding of the required inherent and passive behaviour will increase the need for integral test activities, taking into account the full range of relevant influences.
- The IAEA should encourage more research efforts to develop methods for quantitatively assessing the reliability of passive systems, as there is currently no generally accepted method for this.
- The IAEA should encourage more work to address the applicability of the single failure criterion to passive properties and inherent characteristics, as it may not be possible to demonstrate the behaviour of these properties and characteristics, reliably, under all transient conditions and for the required time period, especially when there are weak driving forces to ensure the desired function.
- The IAEA should seek to formulate a definition of a “severe accident” in a technology-neutral manner that is also applicable to future advanced reactor technologies.

Overall, the Forum assumes that both future SMR operators and regulatory authorities need time to gain operating experience at the first real plants and to achieve and ensure a high level of confidence in the safety of the reactors. This also applies to potential revisions to the nuclear regulations, as such revisions require a sufficient evidence-based footing, which is largely not yet available. At present, existing safety requirements should not be seen as an obstacle, but as a proven reference standard.

Phase 3 of the Forum's activities (2021 to 2023), for which no reports were available by the end of 2023, will address the following aspects, among other things:

- Interfaces between safety and security
- New approaches to containment/confinement and their impact on the DiD concept
- Regulatory expectations regarding the number of independent physical barriers to contain the radioactive materials
- Consideration of safeguard requirements in reactor design
- Particular characteristics of the long-term procurement of SMR plant components
- Design, taking into account simplified dismantling and handling of radioactive waste

7.2.2 OECD Nuclear Energy Agency (NEA)

Within the framework of the OECD/NEA, the “Working Group on the Regulation of New Reactors (WGRNR)” and the “Working Group on the Safety of Advanced Reactors (WGSAR)”, in particular, deal with new reactor designs.

The WGRNR deals with regulatory issues relating to site selection, licensing and supervision of new commercial nuclear power plants, in particular with inspections during the construction of such plants and the exchange of experience with other working groups of the “Committee on Nuclear Regulatory Activities (CNRA)”, including the “Multinational Design Evaluation Programme (MDEP)”. The CNRA is an international committee, founded in 1989, on safety issues in the area of supervision, licensing and inspection of nuclear plants (NEA 2020c).

The WGSAR aims at providing regulatory perspectives on the regulatory framework and licensing procedures, including required research, through technical reports to help simplify the regulation of “advanced reactors” and harmonise approaches and understanding. “Advanced reactors” are primarily understood as non-light water cooled designs. The WGSAR also exchanges views with other organisations involved in conducting R&D on such reactors, including GIF, IAEA and the EU (NEA 2020d).

(NEA 2021a) sets out the regulatory assessments of safety aspects associated with the calculation and justification of the reactor physics characteristics and criticality assessments of sodium-cooled fast reactors (SFR). It identifies topics where further investigation and potentially research and development are considered necessary. The results presented in the report are based on work by the WGSAR.

The WGSAR reached a common understanding on the following topics:

- The existing regulatory approach of the nations participating in the WG is general enough to be applied to SFR. Many of the existing reactor physics requirements and criteria are sufficiently high-level, with specific details to be considered. This also applies to existing requirements for setting and justifying safety limits for reactor-physics parameters.
- Inherent and passive safety properties should be increasingly applied to the reactor core of SFR (and other “advanced reactors”).
- Determination and consideration of uncertainties in calculation parameters are essential, and all sources of uncertainty must be taken into account. The methods that exist for other reactors in this regard must also be applied to SFR.
- Experimental safeguards must be available for the verification and validation of computer programmes and libraries of cross sections that are to be used for reactor-physics properties of SFR.

The WGSAR also provides the following additional information:

- There is no internationally uniform approach to requirements for calculating reactor-physics parameters.
- The accident scenarios relevant for determining reactivity coefficients are largely comparable. SFR-specific requirements for analysis codes and methods for ensuring criticality safety are not available.

(NEA 2021b) addresses the regulatory assessments of safety aspects that are to be used for fuel qualification in SFR. Topics are identified for which further investigations and potential research and development work are considered necessary. The results are recorded as follows:

- Fuel failure mechanisms have already been developed specifically for older SFR designs, but this still has to be accomplished for new fuel types.
- The proof of fuel rod integrity in normal operation, during transients and under intended operating conditions should be verified experimentally and analytically.
- As part of the fuel qualification programme, tests must be carried out under irradiation that cover at least the operational burn-up and power ranges, as well as the range of expected operational conditions and the leading design-basis accidents.

7.2.3 Conclusions

The international regulations were developed primarily on the basis of knowledge gained from the construction and operation of current water-cooled reactor concepts. On the one hand, they specify basic requirements, but also provide concrete technical details or set requirements with regard to specific technological solutions. Although individual regulatory issues in connection with SNR were discussed at an early stage, for example in the 1980s with regard to SFR, there is no prescriptive set of rules comparable to those for water-cooled reactor concepts. International organisations and their working groups such as the IAEA, the NHSI, the SMR Regulators' Forum or the NEA therefore examine and revise the international regulations with a view to their applicability to SNR, both with regard to the existence of gaps and the applicability of the existing regulations to other types of technology.

SNR differ significantly from current water-cooled reactor concepts, particularly in the use of different coolants (and thus new phenomena, failure mechanisms or risks to be taken into account), increased use of passive safety properties (and associated verification issues), other fuel concepts and possibly other fields of application (in particular high-temperature applications).

Based on such differences, the IAEA has identified numerous application problems of its previous regulations for SNR. This affects many central areas of reactor design and construction, such as the reactor core, containment, cooling systems, fuel handling and storage, and probabilistic and deterministic verification methods.

Due to the different fuel concepts, new questions continue to arise in connection with security and fissile material control (e.g. due to higher initial enrichments, different properties of the fuel elements or a more extensive occurrence of separated fissile material in reprocessing plants).

Questions of applicability can arise from the fact that existing regulations are not formulated in a technology-neutral manner, so that supplementing regulations with a technology-neutral formulation can in principle offer a solution. In addition, applicability questions were also identified that are due to innovations in the SNR for which corresponding requirements are missing in the existing regulatory texts. Due to significantly less operating experience from experimental and prototype reactors compared to water-cooled reactor concepts, it can be assumed that the development of international requirements in such areas will take a considerable amount of time. It requires that such plants have been planned, approved, built and operated, so that the time required for the development of requirements is estimated to be one or more decades rather than a few years.

Some of the identified questions of applicability are now being addressed through revision processes of existing regulations or the development of additional regulations, but gaps still remain.

Overall, it can be assumed that it will take time to gain operating experience at the first real plants and to achieve and ensure a high degree of confidence in the safety of the reactors. This also applies to potential revisions to the nuclear regulations, as such revisions require a sufficient evidence-based footing, which is largely not yet available. This experience will first be available to national authorities and then also be taken up by international organisations, which requires additional time.

7.3 European activities

In Europe, important regulatory developments are taking place at the level of the Western European Nuclear Regulators Association (WENRA). In addition, the European Commission's Joint Research Centre (JRC), for example, has also dealt with regulatory issues related to SNR.

7.3.1 Western European Nuclear Regulators' Association (WENRA)

WENRA is an independent association of nuclear licensing and regulatory authorities from member states of the European Union as well as Switzerland, Ukraine and the United Kingdom. WENRA currently has 19 members (Belgium, Bulgaria, Czech Republic, Finland, France, Germany, Hungary, Italy, Lithuania, the Netherlands, Poland, Romania, Slovakia, Slovenia, Spain, Sweden, Switzerland, Ukraine and the United Kingdom) as well as associated members and observers.

The main objective of WENRA is the further development of harmonised safety requirements for nuclear facilities in the member states. To this end, WENRA has published the “Safety Reference Level” (WENRA 2021b), which the member states have committed themselves to implementing.

Other WENRA publications include coordinated position papers on various nuclear safety issues, such as specific regulatory issues for passive safety systems (WENRA 2018) or the “practical exclusion of event sequences” applied to new nuclear power plant designs (WENRA 2019), including seven safety objectives that new reactor designs should meet (WENRA 2010). These safety objectives should be taken into account in the design, site selection, construction, commissioning and operation of new nuclear power plants and include the following objectives:

- Within the normal operation and abnormal operation (malfunctions) plant states and with a view to preventing the occurrence of accidents, (Safety Objective O1)
  - The frequency of occurrence of malfunctions should be reduced by an improved ability of the plant to remain within the conditions of normal operation, and
  - The potential for the occurrence of accident situations should be reduced by an improved ability of the plant to control malfunctions.
- For accidents without core meltdown, (Safety Objective O2)
  - It should be ensured that such accidents do not lead to any, or only minor, radioactive effects (this means in particular: no need for iodine prophylaxis, for orders to stay indoors or for evacuations).
  - The core meltdown frequency should be reduced as far as reasonably possible, taking into account all assumed causes of accidents and combinations of event sequences, as well as the release of radioactive material.
  - The site selection and the design of the plant should take into account external influences, as well as disruption measures and other influences by third parties.

- With regard to accidents involving core meltdowns (Safety Objective O3),
  - Potential radioactive releases into the environment (taking into account all risks posed by nuclear fuel), including in the long term, are to be reduced to such an extent that
    - Emergency measures outside the plant, which would not have time to be implemented (early releases), or protective measures for the population that would not be limited in terms of space or time (large releases), are practically impossible,
    - For accident sequences that are not practically impossible, design measures are to be provided in such a way that only geographically and temporally restricted protective measures for the population are required (no permanent relocation, no need for evacuations outside the immediate vicinity of the plant, only a limited need to stay in buildings, no long-term restrictions on food consumption) and there is sufficient time for their implementation.
- The independence of the safety levels of the DiD concept (Safety Objective O4)
  - Should be increased with regard to all safety levels, in particular through diverse facilities, in addition to strengthening the individual safety levels themselves, as mentioned in the previous three safety objectives, in order to achieve an overall strengthening of the DiD concept, as far as reasonably achievable.
- Interfaces between safety and security (Safety Objective O5):
  - Safety and security measures should be designed and implemented in an integrated manner, and synergies between these measures should be taken into account.
- Radiation protection and waste management (Safety Objective O6):
  - Appropriate design measures, which include all operating conditions as well as dismantling and decommissioning activities, should lead, as far as reasonably possible, to a reduction of
    - The individual and collective dose of the personnel,
    - The radioactive discharges into the environment and
    - The amount and activity of the radioactive waste.
- Safety-related leadership and management tasks (Safety Objective O7):
  - Effective safety management should be ensured right from the start of the design. This means that the plant operator
    - Establishes effective safety-oriented leadership and management, throughout the new construction project, and provides sufficient technical and financial resources to meet its primary responsibility for safety.
    - Ensures that the staff of all other organisations involved in site selection, design, construction, commissioning, operation and dismantling of new plants are aware of the nuclear safety tasks and the importance of their activities in this regard.

Further information on these safety objectives is contained in (WENRA 2013).

In 2020, WENRA examined the extent to (WENRA 2010) or (WENRA 2013) need to be revised in view of the plans for further developed reactor concepts. The results were as follows (WENRA 2020):

- The safety objectives and the associated further information in (WENRA 2013) are still adequate; the safety objectives are of a higher-level nature and are formulated in a largely technologically neutral manner, which enables flexible application.
- For aspects, which in view of future design developments, may not be adequately addressed in (WENRA 2013), separate position papers would have to be drawn up. This applies, for example, to designs of Small Modular Reactors (SMR).

In 2021, WENRA also looked at the applicability of the safety objectives published in 2013 for new reactors with regard to possible novel safety features of SMR concepts (WENRA 2021a). The results were as follows:

- The safety objectives are applicable to SMR designs, as well as evolutionary improvements of LWR concepts and so-called novel reactor technologies.
- In-depth consideration of the safety objective “O5. Safety and security interfaces” would be recommended, which in its current form does not explicitly include safeguard aspects. An extension of Objective O5 to include the aspect of “safeguards” would therefore be advantageous.

In accordance (WENRA 2021c), the safety objectives must still be regarded as up to date and represent minimum targets to be met for SMR designs.

Independently of this, discussions in WENRA will continue on the further development of overarching safety objectives and safety requirements for new reactor designs. In particular, questions arise in connection with reactors that may be built in densely populated regions and/or in large numbers (mini reactors, for example, to provide process heat).

7.3.2 Review of international agreements by the JRC

The European Commission's Joint Research Centre (JRC) has examined the applicability of internationally binding agreements on nuclear regulatory tasks with regard to SMR (JRC 2022). SMR are understood to be reactors with an electrical output of up to 300 MW per unit. The conventions examined by the JRC include:

- Convention on Nuclear Safety (CNS)
- Joint Convention (JC) on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management
- Early Notification Convention of a Nuclear Accident
- The Assistance Convention in the Case of a Nuclear Accident or Radiological Emergency
- Vienna Convention on Civil Liability for Nuclear Damage
- Protocol to amend the Vienna Convention on Civil Liability for Nuclear Damage
- Convention on Supplementary Compensation for Nuclear Damage (CSC)

- Paris Convention on Third Party Liability in the Field of Nuclear Energy
- Brussels Convention Supplementary to the Paris Convention (BSC)
- Non-proliferation Treaty (NPT)
- Comprehensive Safeguards Agreement (CSA)
- Additional Protocol (AP)
- Übereinkommen über den physischen Schutz von Kernmaterial (Convention on the Physical Protection of Nuclear Material (CPPNM))
- CPPNM Amendment

The result comes (JRC 2022) to the following conclusions:

- The conventions do not address the specifics of the SMR in a consistent manner, although SMR are not explicitly excluded from the scope of the conventions. Some conventions can be applied to all types of SMR, other conventions only to selected types.

  For example, the conventions do not distinguish between different reactor technologies, but between land-based, transportable and ship-based SMR, as well as between civil and military purposes[208].

- The Conventions require adjustments or interpretations to cover all SMR types.
- For some SMR types, the Conventions may not apply if
  - the risks associated with these types are lower than with other reactor types,
  - they contain only small amounts of nuclear material or
  - the effects in the event of an incident or accident are limited.
- It is recommended to carry on further reviews of international agreements (on the use of the seas, space, the environment or other non-proliferation instruments) and of EU legislation with regard to their applicability to SMR.
- With regard to the above-mentioned application exceptions (“low risks”, “low inventory” and “low impact”), (JRC 2022) raises the following questions with regard to the potential implementation of SMR projects:

[208] The “Convention on Nuclear Safety” (as well as the “Vienna Declaration” 2015") only deals with land-based nuclear power plants, not military facilities or SMRs used for research purposes.
The “Joint Convention” does not include military facilities.
The Vienna Convention on Civil Liability, the Protocol of Amendment thereto and the Convention on Supplementary Compensation for Nuclear Damage do not cover air or sea-based transport or military installations. The Convention on Supplementary Compensation for Nuclear Damage would, however, also cover most sea-based SMRs.

- “Low risks”
  - Can it be justified from a safety perspective, for example if there are lower risks, that SMR, including micro-reactors, should not be classified as “nuclear power plants” within the meaning of the “Convention on Nuclear Safety” and the “Vienna Declaration”?
  - Can it be justified from a safety perspective, for example if there are lower risks, that SMR should not be classified as “nuclear facilities”[209] according to the “Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management”?
  - Can SMR release “significant quantities” (in the sense of the “Amendment to the Convention on the Physical Protection of Nuclear Material”) of radiation or radioactive material and to what extent could sabotage endanger people and the environment?
  - The “Protocol Amending the Vienna Convention on Civil Liability for Nuclear Damage” and the “Paris Convention” provide the option for states to exempt facilities from their obligations under the Convention. Can a state claim this for an SMR due to lower risks? To what extent would the risks have to be lower than for conventional nuclear power plants?
- Small inventory of nuclear material
  - According to the Vienna Convention on Civil Liability for Nuclear Damage, small quantities of nuclear material can be exempted from the obligations of the Convention. For which types of SMR with small inventories of nuclear material can this exemption be claimed?
  - According to the Comprehensive Safeguards Agreement, safeguards measures apply to fissile material in quantities greater than one kilogram. Are there types of SMR that contain less than one kilogram?
  - According to the “Additional Protocol to the Comprehensive Safeguards Agreement”, SMR or SMR-related research and development activities that work with nuclear material larger than one kilogram are subject to the Additional Protocol. Are there any types of SMR or research and development activities related to SMR that involve less than one kilogram?
- Limited impact in the event of an incident or accident
  - The “Convention on Early Notification of a Nuclear Accident” also covers SMR to the extent that they can result in cross-border releases. Can all SMR (even if located in remote locations on land or at sea) cause international cross-border releases in the event of an incident or accident, and if so, to what extent?
  - Could an SMR cause damage in a maritime zone other than the one in which it is located in accordance with the “Protocol Amending the Vienna Convention on Civil Liability for Nuclear Damage”? What could be the radius of affected areas in the event of a release of radioactive material in the event of an incident or accident in a floating or submerged SMR?

[209] Nuclear facility: “A ‘nuclear facility’ being a civilian facility and its associated land, buildings and equipment in which radioactive materials are produced, processed, used, handled, stored or disposed of on such a scale that consideration of safety is required.

- Other questions
  - It seems unclear to what extent the conventions referred to cover SMR that are used for transport purposes, operate on the high seas or in space.
  - It seems unclear to what extent the safeguards measures established to date will prove to be sufficiently effective for new, advanced SMR types.

7.3.3 Conclusions

Discussions on regulations in connection with SNR development are also taking place at the European level. For this purpose, WENRA drew up safety objectives for new reactor concepts as early as 2010 and, in a review in 2020, found that these are of a superior higher-level nature and largely technologically neutral, which enables flexible application. Separate position papers would have to be drawn up for future design developments that are not adequately addressed in the previous safety objectives. This particularly applies to the area of the interface between safety and security.

In addition, the JRC has examined international conventions with regard to their applicability specifically to SMR. However, the aspects identified there mainly concern the properties of SMR (low inventory, potentially lower risks, transportability, etc.) and are therefore not directly relevant to SNR developments.

7.4 Selected national regulatory developments

This chapter provides examples of the regulatory developments in connection with SNR in the USA, Canada and the United Kingdom.

7.4.1 USA

A detailed description of the development of SNR in the USA is given in Chapter **3.3.2**, as well as the descriptions of the respective technology lines in Chapter **4**. Another current description of developments in the USA in the area of GIF can be found in (Tsvetkov 2023).

The following describes the activities of the US regulatory authority, the U.S. Nuclear Regulatory Commission (NRC), to further develop the US regulatory framework and to approve SNR.

7.4.1.1 Regulatory requirements for SNR and work of the NRC regulatory authority

Regulatory developments in connection with new reactor concepts (both SMR and SNR) were described in (Oeko-Institut e.V.; WIP; PhB 2021, Kap. 4.1.4 und 4.2.1). A current summary of the latest developments can also be found in (NASEM 2023a, Kap. 7). Detailed information on the licensing of SNR in the USA can be found on the NRC website (NRC 2023aj). An overview of important reports published in connection with the regulation of SNR by the NRC can be found in (NRC 2023ai). The following description is based on these presentations, unless stated otherwise.

In the USA, approvals from the NRC are required for the construction and operation of reactors. Two different application procedures can now be selected (NRC 2020c). Traditionally, licensing was granted in accordance with 10 CFR Part 50 of the NRC. Here, a construction permit (CP) is first applied for. For this purpose, a detailed assessment of the reactor location is carried out. However,

only preliminary safety reports need to be submitted to assess the safety of the specific reactor design. Based on the CP, the applicant can begin construction of the plant and apply for an operating license (OP) at the same time. For this, the applicant must submit final safety reports. With this procedure, the applicant can advance the construction of a reactor while the relevant documents for the operating license are being prepared. However, this also runs the risk that changes to the design of the reactor will have to be made due to the final assessment by the NRC, which means a high risk of additional costs and delays in the completion of the reactor.

Alternatively, an applicant can apply for a combined construction and operating license (COL) in accordance with 10 CFR Part 52. To do this, the applicant must submit complete documents on the safety assessment of the planned reactor for licensing, but this avoids the risk that (new) NRC requirements will lead to changes to plants already under construction. The applicant can also apply for further permits as part of this procedure. On the one hand, independent of a specific reactor concept, he can apply for an early site permit (ESP), which examines the basic suitability of a site for the construction of a reactor. Furthermore, a reactor developer can apply for a design certification (DC) for his reactor concept, which confirms that a specific reactor design complies with the requirements of the NRC. Instead of a DC, a standard design permit (SDC) can also be applied for. This has a less binding effect than the DC, as it only includes confirmation of the requirements by the NRC staff, but not by the NRC Commission or the Atomic Safety and Licensing Board. Both the DC and the SDC can be used to serve as a basis for later issuing a COL at different locations. Finally, a manufacturing license can also be obtained under 10 CFR Part 52, which allows the manufacture of a reactor at a location other than the later reactor location. A COL is then required for the final construction of the reactor at the actual location.

(NASEM 2023a) identifies the key advantages and disadvantages of the procedure according to 10 CFR Part 52. On the one hand, this can reduce the regulatory risk for necessary adjustments during the construction of a reactor. On the other hand, the design of a reactor must be fixed at an early stage so that later changes based on new knowledge or experience from the construction of the reactor can no longer be incorporated into the design. A combined approach can therefore also be chosen by first applying for a construction permit under Part 50 and then applying for a DC under Part 52 for the reactor concept completed by the time the operating license is applied for, instead of an OP, which would only apply to this location. In both procedures, however, the applicant must make considerable investments in preparing the application documents and the licensing process by the NRC.

In 2013, the DoE and the NRC jointly launched an initiative to formulate generic design requirements for novel (non-water-cooled) reactor concepts (DOE 2014; NRC 2023ad). A first result of this work was the submission of a report by the Idaho National Laboratory (INL) on behalf of the DoE, which served as the basis for the subsequent work of the NRC (INL 2014a). The INL report formulated general requirements for SNR as well as exemplary requirements for sodium-cooled fast reactors (SFR) and modular high-temperature reactors (VHTR). The industry also published proposals (EPRI 2016; 2017) based on its strategy for introducing SNR (NEI 2016).

The NRC subsequently developed a strategy for the further development of its regulations (NRC 2016) and a roadmap based on it (NRC 2017). As a first step, the NRC published Regulatory Guide 1.232 in 2018 (NRC 2018d). This provides guidance on how the existing General Design Criteria for Nuclear Power Plants (NRC 2007), which are part of the application under 10 CFR Part 50 but are also used in applications under 10 CFR Part 52, can be transferred to non-water-cooled reactor concepts.

A further development of the basic requirements by the NRC consists in an increasing use of goal-oriented (risk-informed, performance-based) requirements instead of the historically developed and deterministically derived prescriptive requirements (Kadambi et al. 2019). Goal-oriented requirements for the containment of SNR were published in (NRC 2018a; 2018b; 2018c).

In the area of control and instrumentation technology (NRC 2023m) of SNR, the NRC published a review guide in 2021 (NRC 2021b). A first change proposed by the NRC staff concerns the consideration of goal-oriented approaches to determine the required level of diversity and the defence in depth concept for safety-relevant instrumentation technology (NRC 2022b).

The NRC also takes up developments from other organisations. In 2021, the American Society of Mechanical Engineers (ASME) together with the American Nuclear Society (ANS) published a standard for conducting probabilistic risk analyses (PRA) for SNR (ASME; ANS 2021). This is intended to allow the application of PRA methods to SNR of the VHTR, SMR technology lines and liquid metal-cooled reactors (SFR, LFR). The NRC adopted this standard in 2022 as a provisional rule for a trial period of two years (NRC 2022a).

Furthermore, in 2011 ASME added a new Division 5 for use in high-temperature reactors (VHTR) to its ASME “Boiler and Pressure Vessel Code” (B&PV) in Section III (rules for the design of components for nuclear power plants). In 2018, in response to industry requests, ASME requested that the NRC incorporate its 2017 version of the ASME B&PV Code Section III, Division 5 into the NRC regulations. The NRC subsequently commissioned various organisations to review the ASME Code (NRC 2023o). In January 2023, it published a technical review (NRC 2023j) and incorporated the ASME Code with adjustments into its regulations on this basis (NRC 2023a). Since the Code does not contain methods for assessing deterioration that can occur as a result of corrosion, mass transfer phenomena, radiation effects or other material instabilities, the NRC has drafted an internal guideline on this topic and published it for comment (NRC 2023d).

In 2019, the Nuclear Energy Institute (NEI), a trade association of the nuclear industry in the USA, created a guideline (NRC 2023x) for determining initiating events, for the safety classification of systems, structures and components (SSCs) and for evaluating approaches to the tiered safety concept for non-water-cooled reactors as part of the nuclear industry's Licensing Modernization Project (LMP) (NEI 2019). This was adopted by the NRC as (NRC 2020a). As part of the “Technology Inclusive Content of Application Project” (TICAP), the NEI also created a guideline on the structure and scope of the safety report (NEI 2022) and applied to the NRC for its adoption into the regulations. The NRC is developing corresponding regulations as part of its “Advanced Reactor Content of Application Project” (ARCAP) (NRC 2023k).

The application of the procedures according to 10 CFR Part 50 or 10 CFR Part 52 requires the use of the underlying, detailed prescriptive requirements that arose in connection with the introduction and further development of light water-cooled reactors in the USA. The application of these procedures to SNR would therefore also require the development of specific underlying regulations for all SNR technology lines. To avoid this, the U.S. Congress has required the NRC by law to develop a technology-neutral set of regulations for application to SNR by 2027 (NEIMA 2019).

The NRC has committed itself to developing such a set of regulations(NRC 2021a), which is to be published as 10 CFR Part 53 (NRC 2023af). The NRC staff submitted a draft of such a rule to the Commission on 1/3/2023 (NRC 2023h). After licensing by the Commission, a final version of the rule is to be developed by the end of 2024 and published by July 2025 (NRC 2023af).

In the area of emergency planning, the NRC published a new rule for application to SMR and other novel reactor concepts (NRC 2023b) and an associated Regulatory Guide in November 2023 (NRC 2023f). The key aspects of these new rules include (NRC 2023n):

- A new alternative, performance-based emergency response framework, including requirements for demonstrating an effective response during training and exercises for emergency and accident conditions;
- the requirement for a hazard analysis for each facility adjacent to or near an SMR or other novel facility;
- a scalable approach to determining the size of the emergency planning zone for the plume exposure pathway; and
- a requirement that the emergency plan describe the planning for food consumption, including the capabilities and resources available off-site to prevent contaminated food and water from entering the consumption pathway.

The NRC is also revising its Physical Security Requirements regulation and is developing a new rule for SNR (NRC 2023ag). The goal is to publish the rule by the end of September 2025 (NRC 2022h).

Since many SNR use different types of fuel than previous LWR, the NRC has also drawn up requirements for the qualification of such different types of fuels (NRC 2022g). For metallic fuels (NRC 2023s), information is compiled as a basis for the qualification of (U,Zr) fuels in fast reactors in (NRC 2023e). Other fuels investigated by the NRC – partly in cooperation with the Canadian CNSC – include TRISO (NRC 2023v; NRC; CNSC 2023a), molten salts (NRC 2023t; 2022c) and uranium carbide fuels with silicon carbide composite cladding tubes (NRC 2023u).

Safety issues and risks in the production of such fuels are also considered (PNNL 2019; ORNL 2019).

7.4.1.2 NRC licensing and pre-licensing activities

The licensing and pre-licensing activities carried out in the USA as of the end of 2023 are summarised in Table 7-4. Pre-licensing activities include those processes in which a potential applicant clarifies questions related to a future application process with the licensing authority before the actual licensing application is submitted to the authority. For historical licensing activities in the USA related to non-water-cooled reactor concepts, see (Oeko-Institut e.V.; WIP; PhB 2021, pp. 114–115).

Table 7-4: U.S. NRC licensing activities related to non-water-cooled reactor concepts

System	Activity	Applicant
ARC-100	Pre-Application	ARC Clean Technology
Aurora Powerhouse	Combined License Application (eingestellt)	Oklo Power LLC
Energy Multiplier Module (EM^2)	Pre-Application	General Atomics
eVinci	Pre-Application	Westinghouse Electric Company
Fast Modular Reactor	Pre-Application	General Atomics-Electromagnetic Systems
Hermes	Construction License (erteilt)	Kairos Power LLC
Hermes 2	Construction License (beantragt)	Kairos Power LLC
High-Temperature Gas-Cooled Test Reactor	Pre-Application	University of Illinois at Urbana-Champaign; Ultra Safe Nuclear Corporation
IMSR	Pre-Application	Terrestrial Energy USA Inc.
Kairos (KP-FHR)	Pre-Application	Kairos Power LLC
Kaleidos Microreactor	Pre-Application	Radiant Industries, Inc
MCFR	Pre-Application (geplant)	TerraPower, LLC
MSRR	Construction License (beantragt)	Abilene Christian University (ACU)
Natrium	Pre-Application	TerraPower, LLC; GE Hitachi
Xe-100	Pre-Application	X-Energy, LLC

Sources: (NRC 2023ah)

The High-Temperature Gas-Cooled Test Reactor at the University of Illinois at Urbana-Champaign is a test reactor with TRISO fuel, helium cooling and graphite moderation, which is based on the IMSR reactor concept of the Ultra Safe Nuclear Corporation. It is to be built on the university's premises in Urbana-Champaign, Illinois. Since this is a test reactor, the pre-licensing activities related to this reactor are not discussed in detail below, see (NRC 2023ak).

Furthermore, on August 12, 2022, Abilene Christian University (ACU) submitted a construction application for a Molten Salt Research Reactor (MSRR) to the NRC. This is to be built at the university's site in Abilene, Texas. The reactor is to have a thermal output of up to one megawatt and will not be used to generate electricity. It is to use a fuel made from HALEU, which is dissolved in FliBe salt. The containment of the reactor is to meet the requirements of (NRC 2018a; 2018b; 2018c).

Pre-licensing activities for the MCFR from TerraPower should begin in 2019, see (Oeko-Institut e.V.; WIP; PhB 2021, chapter 4.2.1.2), but are still listed by the NRC as being in preparation (NRC 2023ab). The MCFR is one of five reactor concepts for which technical, operational and regulatory challenges for the construction of a demonstration reactor are to be solved within 10-14 years as part of the “Advanced Reactor Demonstration Project” (DOE 2020b).

The findings to date from the other licensing activities are summarized below, as far as information is available on the NRC website.

ARC-100

The ARC-100 is a sodium-cooled fast reactor with 100 MW of electrical power, see (Oeko-Institut e.V.; WIP; PhB 2021, annex 6.2.4.2). The company ARC Clean Technology has initiated various pre-licensing activities for its ARC-100 reactor concept with the NRC (NRC 2023l). These include the topics of

- Fuel qualification,
- Storage of spent fuel in the reactor vessel and
- the regulatory treatment of non-safety systems.

Furthermore, various meetings were held with the NRC. In addition, ARC Clean Technology plans further pre-licensing steps by February 2025 (arc 2023) on the following topics:

- Safety-related shutdown of the main coolant pumps,
- Reactivity changes of the reactor in various event sequences,
- Containment functions,
- Alternative shutdown systems,
- Probabilistic safety analyses,
- Coolant blockages,
- Seismic isolation,
- Operation at 200 MW electrical power and
- The research and development programme.

Final evaluations by the NRC are not yet available (NRC 2023l).

Aurora Powerhouse

On March 11, 2020, Oklo Power LLC applied for a combined construction and operating license (COL) for an Aurora Powerhouse microreactor (NRC 2022e). The fast neutron spectrum micro-reactor is to be operated using a metallic uranium-zirconium fuel at an enrichment of 19.75% (HALEU), with sodium heat pipes for cooling, see (Oeko-Institut e.V.; WIP; PhB 2021, Anh. 6.2.6.1).

The design of the reactor and the safety concept are described in (Oklo 2020a). Due to the small radiological inventory and the exclusion of significant releases in all event sequences considered, Oklo has requested, among other things, an exemption for the need for external emergency protection measures and other exemptions (Oklo 2020b).

The NRC has determined that the documents submitted by Oklo are of such a generic nature that they did not allow the NRC to conduct an assessment of the safety of the reactor concept. It has therefore initiated a two-stage process to process Oklo's application. In a first step, the NRC requested additional information from the applicant, for example on the topics of (NRC 2022f)

- Applicability of regulations,
- Quality assurance,
- Analysis of triggering events and the resulting maximum credible accident (MCA), as well as
- The safety classification of systems, structures and components (SSCs).

In a second step, the NRC would have carried out the actual review of the application documents based on the documents to be submitted by the applicant.

With regard to the applicability of regulations, the NRC has further examined the applications for exemptions submitted by Oklo and has reached a final assessment (NRC 2020b) of which regulations

- Are generally not applicable to non-water-cooled reactor concepts,
- Are generally applicable to non-water-cooled reactor concepts, but due to technical constraints are not applicable to the Oklo concept,
- Are applicable to non-water-cooled reactor concepts including the Oklo concept and
- Are applicable to non-water-cooled reactor concepts, but refer to other regulations that are applicable to non-water-cooled reactor concepts.

On this basis, the NRC considered the possibilities for examining the documents submitted by the applicant in step 2 of the procedure for this point to be given (NRC 2020b). The other open questions regarding quality assurance should be pursued further as part of the discussion on the safety classification of SSC.

In a letter dated January 6, 2022, the NRC terminated the procedure without a final assessment because the applicant had not submitted the documents necessary to complete a licensing process with regard to the analysis of the MCA and the safety classification of SSCs (NRC 2022d).

In its final assessment, the NRC also points out that the applicant had only carried out limited pre-licensing activities with the NRC even before its application for a combined construction and operating license. This included interactions on the topics of (NRC 2022d)

- Reactor core design,
- Risk studies and source terms,
- Basic design criteria,
- Security and emergency planning,
- Radiation protection and automation, and
- Safety concept and external influences.

In particular, the NRC points out that in connection with the pre-licensing activities, Oklo conducted (Oklo 2018) a preliminary study on the application of the new regulation in (NRC 2020a). This Regulatory Guide describes a possible approach to determining initiating events, safety classification

of SSCs and evaluation of approaches to the layered safety concept for non-water-cooled reactors. Despite these activities, Oklo did not refer to this Regulatory Guide in its application for a combined construction and operating license. Oklo also made changes to its reactor concept that limited the usability of the results from the pre-licensing activities (NRC 2022d).

According to the NRC, however, the applicant is free to initiate a new licensing process in the future. As of the end of 2023, pre-licensing activities with regard to the aspects of

- Emergency planning
- Site assessment and seismic design, as well as
- Regulatory requirements

are still being carried out (NRC 2023ae).

Energy Multiplier Module (EM^2)

The EM^2 is a gas-cooled fast reactor with 265 MW of electrical power from General Atomics (GA), see (Oeko-Institut e.V.; WIP; PhB 2021, Anh. 6.2.4.5). The reactor is to use uranium carbide as fuel, which is formed by a cladding tube made of a special silicon carbide composite (SiGA) developed by GA.

GA has submitted a white paper (GA 2021) on accelerated fuel qualification to the NRC, on which the NRC provided feedback to GA in November 2021. No recent information is available on pre-licensing activities for the EM^2(NRC 2023p).

eVinci

The eVinci is a microreactor concept from the Westinghouse Electric Company. It is intended to use TRISO particles or comparable encapsulated fuel particles as the fuel. The reactor is cooled by passive heat pipes with sodium as the working medium, see (Oeko-Institut e.V.; WIP; PhB 2021, Anh. 6.2.6.2). The eVinci is one of five reactor concepts for which technical, operational and regulatory challenges for the construction of a demonstration reactor are to be solved within 10-14 years as part of the “Advanced Reactor Demonstration Project”, see (DOE 2020b) and Chapter **3.3.2**.

Westinghouse has initiated various pre-licensing activities with the NRC (NRC 2023q). Both the planned schedule of activities and the documents submitted by Westinghouse to date have been classified almost entirely as documents containing proprietary information and have therefore not been published or have only been largely redacted, making a meaningful evaluation of these documents impossible.

Fast Modular Reactor

The Fast Modular Reactor (FMR) from General Atomics-Electromagnetic Systems (GA-EMS) is a concept for a helium-cooled fast reactor with 50 MW of electrical power and an operating temperature of 800 °C. Uranium dioxide is planned as the fuel, and a SiC composite material is to be used as the cladding and structural material in the reactor core. The FMR is one of the three reactor concepts for which the design is to be developed as part of the “Advanced Reactor Demonstration Project” so that a demonstration will be possible in the mid-2030s, see (DOE 2020b) and Chapter **3.3.2**.

GA-EMS is pursuing various pre-licensing activities (GA-EMS 2022a; NRC 2023r) with the goal of a demonstration project by 2030 and a commercial launch in the mid-2030s. In doing so, GA-EMS refers, according to (GA-EMS 2022a), to the approach described in NRC Regulatory Guide 1.232 (NRC 2018d), which was supported by (NEI 2018; 2019) and supplemented by (NRC 2020a). In particular, GA-EMS is pursuing a probabilistic approach for the determination of design basis initiating events, to the safety classification of SSCs and the assessment of the adequacy of the selected defence in depth concept.

The technical topics for activities with the NRC are (GA-EMS 2022a)

- Safeguards and security,
- Emergency planning,
- Fuel qualification,
- Earthquakes,
- Flooding,
- Control technology,
- Detection methods,
- Probabilistic safety analyses,
- Quality control and
- The operating concept.

GA-EMS has already submitted white papers to the NRC on various of these topics, as well as a “topical report” on key design criteria (GA-EMS 2022b). GA-EMS's schedule for 2022 called for documents to be submitted to the NRC on the remaining open topics by May 2024. No assessments by the NRC have yet been made (NRC 2023r).

IMSR

Terrestrial Energy USA Inc. started pre-licensing activities for the IMSR concept in 2019 (NRC 2023y).

The design of the reactor and the safety concept are described in (Terrestrial Energy 2020). Accordingly, the IMSR (see (Oeko-Institut e.V.; WIP; PhB 2021, annex 6.2.5.1)) is a molten salt reactor that builds on the developments of the Oak Ridge National Laboratory between the 1950s and 1980s, in particular the Molten Salt Reactor Experiment (MSRE, see (Oeko-Institut e.V.; WIP; PhB 2021, annex 6.2.5.2)). The reactor will have a thermal output of approx. 440 MW and an electrical output of approx. 195 MW. In addition, it is to be able to provide process heat at approx. 600 °C.

The topics covered so far as part of the pre-licensing include

- Reactor design (core unit definition),
- Interfaces to other reactor systems (interfaces between core unit and reactor auxiliary building),
- Essential design principles (principal design criteria),
- Postulated initiating events,
- Safety classification of SSCs (structures, systems, and components classification methodology) and
- Methods for determining the uncertainty of the off-gas source term (uncertainty quantification methodology for calculation of IMSR® off-gas source term).

Substantial parts of the NRC's assessments were classified as documents with proprietary information and were therefore not published or were largely redacted, so that a meaningful evaluation of these documents is not possible.

Kairos: KP-FHR, Hermes and Hermes 2

The Kairos Power company is carrying out pre-licensing activities for its Kairos reactor concept (NRC 2023z). The preliminary review began in November 2018. The Kairos concept is a high-temperature reactor (KP-FHR) cooled with a molten fluoride salt. The fuel is to be in the form of TRISO fuel elements in a pebble bed configuration, see (Oeko-Institut e.V.; WIP; PhB 2021, Kap. 4.2.1.2). The KP-FHR is one of the five reactor concepts for which technical, operational and regulatory challenges for the construction of a demonstration reactor are to be overcome within 10-14 years as part of the “Advanced Reactor Demonstration Project”, see (DOE 2020b) and Chapter **3.3.2**.

The design of the reactor and the safety concept are described in (Kairos Power LLC 2018). According to this, the containment of radioactivity is to be achieved essentially through the combination of the TRISO fuel and the use of a molten salt as a coolant and the associated retention capacity of any radionuclides released from the TRISO fuel elements. It can therefore be presumed that radionuclides will not be released outside the plant site.

Kairos Power submitted a "Regulatory Engagement Plan" for its planned interactions with the NRC in 2020(Kairos Power LLC 2020), a revision 2 was submitted to the NRC on November 14, 2023

(Kairos Power LLC 2023c), but both documents were classified as proprietary and are therefore not public.

The topics covered in the pre-licensing activities include (NRC 2023z)

- Essential design principles (principal design criteria),
- Coolant,
- Scalability of the test program for the KP-FHR,
- Methods for investigating fuel behavior and for fuel quality assurance,
- Quality of metallic structural materials,
- Quality of graphite materials,
- Quality assurance program,
- Methods for source term determination and
- Safeguards.

Furthermore, Kairos Power has submitted an application to build two test reactors (Hermes and Hermes 2) at the Oak Ridge site in Tennessee.

The application for the construction of Hermes was submitted to the NRC on 29/09/2021 (Kairos Power LLC 2023d). This is a reactor with 35 MW thermal output that is not intended to be used for power generation. The safety review of the reactor was completed on the basis of (Kairos Power LLC 2023b) in June 2023 (NRC 2023i). On 12/12/2023, the NRC granted licensing to build this test reactor (NRC 2023c).

An application for Hermes 2 was submitted to the NRC on 14/07/2023 (NRC 2023w). As part of Hermes 2, two test reactors are to be built, each with 35 MW of thermal output, which will have a common water-steam system (Kairos Power LLC 2023a). The NRC plans to submit a final safety assessment by November 2024 and to have carried out a final environmental impact assessment by August 2024.

Kaleidos

The Kaleidos is a microreactor concept from Radiant Industries Inc. The Kaleidos is planned to be a helium-cooled, high-temperature reactor with one megawatt of electrical output. TRISO is planned as the fuel, which is embedded in prismatic graphite blocks. Radiant started pre-licensing activities with the NRC in October 2022 (NRC 2023aa).

Radiant plans to apply for a permit to manufacture its microreactor at the end of the third quarter of 2026 and to deliver a first reactor module at the end of the first quarter of 2028. For this purpose, a “Demonstration of Microreactor Experiment” (DOME) will be carried out in January 2026 as an experimental basis for the submission of permit applications (Radiant 2023).

There were still no statements from the NRC on this reactor concept by the end of 2023.

Sodium

The sodium reactor concept from TerraPower represents a technologically, more simple approach for a sodium-cooled fast reactor with 345 MW electrical power in comparison with the originally pursued Travelling Wave Reactor concept, see Chapter 5.2, and combines design features of the General Electric-Hitachi (GEH) PRISM reactor concept with the TWR concept and is being developed jointly with GEH. Sodium is one of the two reactor concepts for which a demonstration reactor is to be tested, approved and built within 5-7 years as part of the “Advanced Reactor Demonstration Project”, see (DOE 2020b) and Chapter **3.3.2**.

TerraPower is seeking licensing of its reactor concept under 10 CFR Part 50. This process allows TerraPower to apply for a construction permit application (CPA) on the basis of a preliminary safety report and thus begin building a reactor. In contrast, a combined construction and operating permit according to 10 CFR Part 52 would require the submission of a final safety report, which TerraPower, however, intends to develop during construction and submit for the application for the operating license application (OLA). The licensing is to be based on (NRC 2020a) (TerraPower, LLC 2021).

TerraPower started various pre-licensing activities with the NRC in 2021. The topics covered in the pre-licensing activities include, among other things (NRC 2023ac):

- Quality assurance programme,
- Methods for quality assurance of the fuel and the fuel element,
- Test programme,
- Essential design principles (principal design criteria),
- Qualification of the shutdown elements,
- Thermal hydraulic model,
- Emergency protection,
- Hazards from volcanism,
- Methods for determining source terms,
- Methods for determining design events and
- Methods for determining reactor stability.

The submission of a final probabilistic safety analysis is planned for 2027, an application for a building permit was planned for August 2023, and the application for an operating license for March 2024 (NRC 2020a).

The deadline for submitting an application for a building permit was postponed by TerraPower to March 2024 in August 2023, and in October 2023 TerraPower asked the NRC for an assessment of the documents available to date for the building permit submission. In December 2023, the NRC published the procedure for determining its assessment of the current status of the documents for the filing of a building permit by TerraPower (NRC 2023g). The NRC assumes that TerraPower will submit the design documents required for an examination to the NRC by January 10, 2024 and that

the examination can then be carried out within four weeks. As of February 2024, there was no newer information on this on the NRC website.

Xe-100

The Xe-100 is a helium-cooled pebble bed high-temperature reactor from X-Energy LLC, see (Oeko-Institut e.V.; WIP; PhB 2021, annex 6.2.3.5). One module is to have an electrical output of 80 MW and four modules together are to provide a total output of 320 MW. The Xe-100 is one of the two reactor concepts for which a demonstration reactor is to be tested, approved and built within 5-7 years as part of the “Advanced Reactor Demonstration Project”, see (DOE 2020b) and Chapter **3.3.2**.

Pre-licensing activities for this reactor concept were started in September 2018. The topics covered in the pre-licensing activities include, among other things (NRC 2024):

- Quality assurance programme,
- Methods for determining source terms,
- Methods for fuel quality assurance,
- Reactor protection system,
- Methods for verification,
- Quality of the graphite materials,
- Essential design principles (principal design criteria),
- Probabilistic verification methods,
- Seismic design,
- Emergency protection and
- Qualification of calculation codes.

X-Energy has already submitted white papers or “topical reports” to the NRC on various of these topics, on which the NRC has already provided X-Energy with varying degrees of feedback or is still conducting a review of the documents. In addition to the documents already submitted, X-Energy is planning a white paper on the handling of spent fuel (X-Energy LLC 2023b).

As part of the “Long Mott” project, an application for site preparation for a reactor consisting of four Xe-100 modules is to be submitted together with Dow Chemical at the Seadrift site near Victoria, Texas. The associated schedule is classified as proprietary, however, and is therefore not public (X-Energy LLC 2023a).

7.4.1.3 Conclusions

The previous licensing of nuclear power plants in the USA took place within the framework of 10 CFR Part 50 or Part 52. In both procedures, the underlying regulations contain prescriptive requirements that have arisen from the regulatory practice with water-cooled reactors. The NRC is

developing a new set of regulations for the licensing of reactor concepts from the areas of SNR and SMR, which is to be published as 10 CFR Part 53 by 2027. This is intended to offer a technology-neutral approach to the licensing of SNR through increased use of goal-oriented (risk-informed, performance-based) requirements. The NRC has already issued the first regulatory guides and a new rule for the area of emergency planning, and a new rule is being developed in the area of security. The NRC is also dealing with the requirements for the qualification of such fuels for the different types of fuel used in SNR, as well as the safety issues and risks associated with their production.

In addition, the NRC is pursuing extensive pre-licensing activities for various reactor concepts in the area of SNR. In these, the extensive open questions regarding the licensing of SNR are discussed specifically for the respective reactor concepts being pursued. In this context, a combined construction and operating license application for a microreactor (Aurora Powerhouse) was submitted as of January 2024, but the procedure was discontinued by the NRC without a final assessment due to a lack of cooperation from the applicant.

In addition, three construction applications for test reactors (Hermes, Hermes 2 and MSRR) have been submitted to the NRC to date; licensing for the Hermes was granted in December 2023, and the other two procedures are ongoing. As part of the “Advanced Reactor Demonstration Project”, a demonstration reactor is to be tested, approved and built for each of two reactor concepts within 5-7 years starting in 2020. No application for licensing has yet been submitted to the NRC for these two reactor concepts (sodium, Xe-100). Of the four reactor concepts from the SNR area for which technical, operational and regulatory challenges for the construction of a demonstration reactor are to be solved within 10-14 years as part of the “Advanced Reactor Demonstration Project” (KP-FHR, eVinci, BANR, MCFR), three developers have so far started pre-licensing activities with the NRC.

7.4.2 Canada

This section first looks at the development of SNR in the context of the overarching developments in the field of nuclear energy in Canada. This is followed by a description of the activities of the Canadian regulator, the Canadian Nuclear Safety Commission (CNSC), to approve SNR and to further develop the Canadian regulations. Finally, there is an overview of the research agenda pursued by the CNSC in Canada, the research questions identified therein and the knowledge already gained on SNR.

7.4.2.1 SNR development in Canada

Canadian developments for SNR focus on developments in the SMR sector, in which both (light) water-cooled and non-water-cooled technology lines are pursued. A presentation of Canadian developments with a focus on SMR was already created in (Oeko-Institut e.V.; WIP; PhB 2021, Kap. 3.4.3.2). The following presentation is based on this chapter and is supplemented by more recent developments, with a focus on the SMR development from the SNR technology lines. For specific Canadian activities within the framework of the GIF, please refer to the chapters on the respective technology lines.

The Canadian SMR programme is considered by (NCG; NFLA 2019) to be one of the most consistent programmes for the SMR development. Central to this is the “roadmap” for SMR development

published in 2018 by the Canadian Nuclear Association[210] on behalf of the Canadian government (CSMRSC 2018).

In response to the roadmap, an SMR Action Plan (NRCAN 2020) was published in 2020 in which 119 governmental and non-governmental actors committed to measures for the introduction of SMR in Canada. An update of the Action Plan was last published in 2022 (NRCAN 2022).

The Canadian government has allocated a sum of 120.6 million Canadian dollars (CAN$) for the development and introduction of SMR in Canada over a five-year period, beginning in the 2022-2023 budget. Of this, CAN$50.7 million is to support the CNSC in developing capacities and know-how for the licensing of SMR and to pursue harmonisation of regulatory requirements with international partners. A further CAN$69.9 million will enable Natural Resources Canada (NRCan) to conduct research related to radioactive waste from SMR, develop a fuel supply industry, strengthen international cooperation and improve national safety and security policies (Canada Gazette, Part I 2022). In addition, CAN$250 million over the next four years (2023-2027) will enable NRCan to provide support for the introduction of SMR and other infrastructure measures in the energy supply sector. In addition, Canada is investing CAN$1.2 billion to expand the Chalk River Nuclear Laboratories and is providing CAN$100 million for the development of SMR as part of a Strategic Innovation Fund (NRCAN 2022).

The Canadian provinces have also drawn up a joint plan for the introduction of SMR in Canada (Government of Ontario, New Brunswick, Alberta and Sasketchewan 2022). For example, the province of New Brunswick is planning, among many other measures, to introduce SMR to support the process of achieving net greenhouse gas neutrality by 2050. To this end, in addition to an existing CANDU reactor with an electrical output of 600 MW, several new SMR with a total electrical output of 600 MW are to be built at the Point Lepreau site by 2035. These are to be SMR from the SNR area, and the first reactor is to be commissioned as early as 2030 (Province of New Brunswick 2023).

According to (NCG; NFLA 2019, p. 29), four key players in SMR development in Canada stand out in particular, three of which are state-owned:

- New Brunswick Energy Solutions Corporation, a joint venture between the provincial government of New Brunswick and New Brunswick Power (NB Power), which in turn is owned by the provincial government;
- Ontario Power Generation, a company owned by the province of Ontario;
- Bruce Power, a consortium of several private companies that operates eight Candu reactors;
- and Canadian Nuclear Laboratories Ltd. (CNL), which operates the Chalk River site, among others.

In addition, a fourth energy supplier in Canada, SaskPower, is also active within the framework of the SMR Action Plan, with SaskPower focusing on the construction of BWRX-300 SMR (HoC 2023).

[210] The Canadian Nuclear Association is a non-profit organisation founded in 1960 to represent the nuclear industry in Canada and to promote the development and growth of nuclear technologies for “peaceful purposes”.

The CNL sees itself as a global hub for SMR research and technology. The company is a subcontractor of the Canadian state-owned company Atomic Energy of Canada Ltd., which developed the CANDU reactor in the 1950s. It has set itself the goal of building an SMR demonstration reactor at a CNL site before 2030.[211]

In April 2018, CNL called for companies to apply to build an SMR demonstration plant on a campus managed by CNL. A total of four companies applied, three companies with microreactor designs (including U-Battery) and one company with an MSR concept (Terrestrial Energy's Integral Molten Salt Reactor). In December 2020, the CNL signed an agreement with the applicant Global First Power (GFP) for the construction and operation of a reactor at the Chalk River site in Ontario.[212]

This reactor is said to be a prismatic high-temperature reactor (micro modular reactor, MMR) from Ultra Safe Nuclear Corporation (USNC), with 15 MW of thermal and 5 MW of electrical power, see (Oeko-Institut e.V.; WIP; PhB 2021, Kap. 6.2.6.3), which is to be built in cooperation with Ontario Power Generation. In May 2021, the CNSC confirmed that the documents submitted so far by GFP are sufficient to begin the technical review of the application.[213] In June 2023, GFP submitted the first part of a License to Prepare Site (LTPS) to the Canadian regulator CNSC (GFP 2023). In August 2023, GFP announced that USNC had published an update to its reactor design that would permit variable thermal power between 10 and 45 MW.[214]

Terrestrial Energy also managed to obtain government funding in Canada in 2016 for the development of the integrated molten salt reactor (IMSR) concept, see (Oeko-Institut e.V.; WIP; PhB 2021, Kap. 6.2.5.1). In March 2016, the company announced that it had received a grant of CAN$ 5.7 million from the Canadian federal government's SD Tech Fund (SDTC) for sustainable development.[215] In addition, public energy company Ontario Power Generation has announced Terrestrial Energy as one of three SMR power plant developers with whom the company will work as part of its goal of deploying SMR technology. In 2020, the Canadian government invested an additional CAN$20 million (USD 15 million) in federal funds to accelerate the further development of Terrestrial Energy's IMSR. The funding is provided by Canada's Strategic Innovation Fund.[216] Terrestrial Energy is not carrying out the highly specific development of the molten salt fuel alone, but has been collaborating with the US government research facility Argonne National Laboratory on fuel development since 2016. This is funded by the US Department of Energy.[217]

In addition to USNC and Terrestrial Energy, other SMR developers from the SNR sector are active in Canada, including ARC Nuclear Canada, Moltex, Westinghouse and X-Energy, see the next section. The company Moltex also received CAN$ 50 million for the development of its reactor concept and technologies for reprocessing spent CANDU fuel, and the company Westinghouse Electric received CAN$ 27.2 million for the further development of its eVinci microreactor concept (NRCAN 2022).

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[211] https://www.cnl.ca/clean-energy/small-modular-reactors/, as of 02/01/2024. In 2021, CNL still set the goal of building a first demonstration reactor before 2026.
[212] https://www.globalfirstpower.com/, as of 02/01/2024.
[213] https://www.cnsc-ccsn.gc.ca/eng/reactors/power-plants/new-reactor-facilities/, as of 02/01/2024.
[214] https://www.usnc.com/usnc-boosts-mmr-power-flexibility-and-value/, as of 02/01/2024.
[215] https://www.sdtc.ca/en/government-of-canada-announces-support-for-clean-technology/, as of 02/01/2024.
[216] https://www.world-nuclear-news.org/Articles/Canadian-government-invests-in-SMR-technology, as of 02/01/2024.
[217] https://www.neimagazine.com/news/newsfuel-testing-begins-for-terrestrial-energys-molten-salt-reactor-8358784, as of 02/01/2024.

7.4.2.2 Regulatory requirements for SNR and work of the CNSC regulatory authority

An overview of regulatory developments in Canada is available at (Oeko-Institut e.V.; WIP; PhB 2021, chapter 4.2.2). The following presentation is based on that chapter and supplements it with current developments with a specific focus on SNR.

Developments in connection with the further development of the nuclear regulations with regard to SNR in Canada are described by CNSC employees in (Eom et al. 2023). According to this, the Atomic Energy Control Board (AECB) was founded in Canada as early as 1946 as a nuclear regulatory authority based on the model of the US Atomic Energy Commission (AEC). However, unlike the US, Canadian regulation pursued more goal-oriented rather than prescriptive regulation. Goal-oriented regulation formulates overarching target or performance criteria, while prescriptive regulation formulates narrower, technical implementation specifications. As regulation progressed and requirements became more differentiated, however, prescriptive requirements were increasingly included in the Canadian regulations (CNSC 2019b).

(Eom et al. 2023) observe that the respective nuclear requirements, at the time, were not capable of preventing the occurrence of catastrophic accidents (TMI, Chernobyl, Fukushima) and that this is also why the regulatory requirements have been developed continuously. According to (Eom et al. 2023), international cooperation for further development in the field of reactor safety was subsequently carried out through the Multinational Design Evaluation Program (MDEP), which was organised by the NEA starting in 2006. As part of this programme, regulators and their technical support organisations from ten countries were able to exchange their knowledge of new reactor concepts. One goal of this program is to formulate common principles for safety that can then be implemented in national regulations. In addition, work was carried out on the international harmonisation of regulatory requirements in the field of nuclear technology as part of the “CoOperation in Reactor Design Evaluation and Licensing (CORDEL)” programme, which has been run by the World Nuclear Association since 2007.

In addition, (Eom et al. 2023) also refers to developments in Europe, which were initiated by the European Commission and implemented within the framework of ENSREG, and are supported by WENRA and accompanied by the “European Nuclear Installations Safety Standards Initiative” (ENISS). Finally, (Eom et al. 2023) also gives the work of the European operators within the framework of the European Utility Requirements (EUR) is also mentioned as an example of international harmonisation in the area of requirements for the licensing of nuclear power plants, see also Chapter 7.2.

Regulatory work of the CNSC

The CNSC is pursuing the further development of its supervisory capabilities and its regulations in order to enable the licensing of new reactor concepts, in particular SMR concepts. The CNSC started a discussion on the need to further develop the nuclear regulations in 2016 (CNSC 2016). The SMR concepts generally pursued in Canada include not only water-cooled reactor concepts (such as the BWRX-300) but also concepts that can be assigned to the SNR, see below.[218]

The CNSC has also responded to the Canadian SMR Roadmap (CSMRSC 2018). A fifth of the roadmap's “key findings” were that, although Canada's regulatory system is fundamentally well

[218] https://nuclearsafety.gc.ca/eng/reactors/smr/facilities.cfm, as of 31/08/2023, last accessed on 04/01/2024

positioned for the introduction of SMR, some modernisation is considered necessary, see also for details (RRWG 2018).

For this reason, (RRWG 2018) noted that, although there are no fundamental regulatory hurdles to the licensing of SMR in Canada, there are unnecessary requirements in specific areas, such as nuclear liability, staff training, accident management (analysis) and emergency preparedness, as well as safety requirements for Class I nuclear facilities (which include, in addition to nuclear power plants, reprocessing plants and other fuel cycle facilities). (RRWG 2018) formulated a total of seven key recommendations. These include

- A limitation of nuclear liability for SMR operators in the event of serious accidents by introducing upper limits based on a graduated approach,
- The introduction of target-oriented requirements instead of prescriptive requirements in the area of nuclear safety,
- Various changes or limitations to the application of the Canadian “Impact Assessment Act” (Bill C-69) in order to exempt SMR from the need for extensive testing.

Overall, based on these recommendations, the roadmap calls for financial support for the introduction of SMR in Canada and the removal of regulatory hurdles, for example with regard to the conduct of environmental impact studies or with regard to limiting liability amounts in the event of serious accidents with SMR.

In particular, the limitation of nuclear liability and the exemption of SMR projects from the need for extensive environmental impact assessments have also been criticised for contradicting important principles of sustainable development (transparent assessment of possible adverse consequences and the polluter pays principle) (Blaise and Stensil 2020).

In response to the 2018 roadmap, the CNSC has drawn up its own strategy for regulating novel reactor technologies (CNSC 2019b). This is based on the principles of nuclear technology regulation in Canada, which are formulated in (CNSC 2023k).

Based on this strategy, the CNSC has formulated its own measures in the SMR Action Plan (NRCAN 2020). The aim of the CNSC is to formulate a technology-neutral set of rules that is suitable for application to the novel reactor concepts proposed in Canada.

Regulatory requirements for nuclear facilities in Canada are formulated in so-called “Regulatory Documents” (REGDOC) (CNSC 2023l). These contain both mandatory requirements and non-mandatory guidelines or instructions.

The REGDOCs formulate basic requirements for the licensing of nuclear power plants in Canada for site licensing (CNSC 2022c), construction licensing (CNSC 2022a) and operating licensing. (CNSC 2022b).[219]

Additional information specific to SMR approvals is contained in REGDOC-1.1.5, which was newly adopted in 2023 (CNSC 2019a). An SMR, within the meaning of this REGDOC, explicitly includes both water-cooled and non-water-cooled reactor concepts. In addition to these documents, (CNSC 2023a) describes requirements for the reactor design of new reactors in Canada. This REGDOC-

[219] https://nuclearsafety.gc.ca/eng/reactors/power-plants/new-reactor-facilities/index.cfm, as of 21/05/2021, last accessed on 02/01/2024.

2.5.2, however, is explicitly limited to new water-cooled reactors. There are currently no REGDOCs that explicitly refer to non-water-cooled reactor concepts (CNSC 2023l).

As of January 2, 2024, no Canadian regulations are in the drafting process (CNSC 2023c). However, the CNSC is pursuing a long-term plan to update its regulations. During the period from 2023 to 2025, various existing regulations are to be adapted to facilitate the development and licensing of SNR (CNSC 2023e).

In the area of security, the CNSC plans to adapt the Canadian “Nuclear Security Regulation” in order to respond to new forms of threat and to implement international developments in the Canadian regulation. This also aims to reduce existing obstacles to the introduction of new reactor concepts in the SMR sector by replacing existing prescriptive requirements with goal-oriented ones (Canada Gazette, Part I 2022).

Ongoing application procedures

As of the end of 2023, three specific application procedures are underway in Canada. One plant is currently in the licensing phase, a BWRX-300 boiling water reactor at the Darlington site (CNSC 2023g). SaskPower plans to build additional BWRX-300 plants at two potential sites in Saskatechwan, but a decision on this is not expected until 2029 (CNSC 2023j).

In addition, as already mentioned above, Global First Power has submitted an application to prepare the site for the construction of an MMR microreactor (GFP 2023). This is to be built at the Chalk River Laboratories site in Renfrew County, Ontario, about 200 km northwest of Ottawa. The next step in this process requires an environmental impact assessment. The scope required by the authorities for this was determined by a decision of the Canadian regulatory authority Canadian Nuclear Safety Commission (CNSC) on July 16, 2020 (CNSC 2020a), but the corresponding documents have not yet been submitted.

Finally, in June 2023, NB Power, together with ARC Clean Technology Canada, submitted an application for site preparation to the CNSC (CNSC 2023i; NB Power 2023).

Pre-licensing activities

The CNSC offers the option of a Vendor Design Review (VDR), a preliminary review of application documents, to give both the developers and the CNSC the opportunity to gain experience in approving novel concepts.[220] The rules for a VDR are set out in (CNSC 2018).

Within the framework of the VDR, three phases of preliminary review are possible, each of which goes more into the details of the reactor concept. While Phase 1 can be completed in around 12-18 months, Phase 2 is said to take around 24 months. In Phase 3, open aspects from Phase 2 can be dealt with in more detail. The VDR is not a necessary prerequisite for applying for an actual license for a nuclear facility in Canada. At the same time, successful completion of one of the VDR phases does not bind the CNSC with regard to a later actual application for a facility license.

At the end of 2023, according to the CNSC, 10 concepts (with two variants of the MMR concept) will be in different phases of the VDR (CNSC 2023f). The status of the VDR procedures is summarised in Table 7-5. These are reactor concepts from the SFR (ARC-100), MSR (IMSR, SSR), VHTR (MMR,

[220] https://www.world-nuclear-news.org/Articles/Speech-Regulatory-harmonisation-for-SMRs; last accessed 20/01/2021.

Xe-100) technology lines and microreactors (eVinci, U-Battery). Further development of the U-Battery concept was discontinued in 2023 (urenco 2023).

Compared to their status in 2020, three reactor concepts (NuScale from NuScale Power, SEALER from LeadCold and StarCore from StarCore Nuclear) are no longer in the VDR process, see (Oeko-Institut e.V.; WIP; PhB 2021, Tab. 4-3). With regard to SNR, Phase 2 was restarted for ARC-100 from ARC Nuclear Canada and for the MMR from USNC, while Phase 1 for the SSR from Moltex Energy and Phase 2 for the IMSR from Terrestrial Energy were completed. Information on the results of the various VDR phases is available for the completed phases of the SNR concepts ARC-100, IMSR, MMR and SSR in the form of a CNSC summary. These results are presented below.

Table 7-5: Status of Vendor Design Reviews (VDR) in Canada

System	Manufacturer	Performance (MW_e)	Phase	Start of VDR	Status
ARC-100	ARC Nuclear Canada Inc.	100	1	Sept. 2017	Completed
			2	Feb. 2022	Ongoing
BWRX-300	GE-Hitachi Nuclear Energy	300	2*	Jan. 2020	Completed
eVinci	Westinghouse Electric Company, LLC	Up to 5	2*	Jan. 2023	Start announced for 2023
IMSR	Terrestrial Energy Inc.	200	1	April 2016	Completed
			2	Dec. 2018	Completed
MMR-5, MMR-10	Ultra Safe Nuclear Corporation	5-10	1	Dec. 2016	Completed
			2	June 2021	Ongoing
SMR-160	Holtec SMR, LLC.	160	1	July 2018	Completed
SSR	Moltex Energy	300	1 and 2	Dec. 2017	Phase 1 completed
U-Battery	U-Battery Canada Ltd.	4	1	Open	Open
Xe-100	X-Energy, LLC	80	2*	July 2020	Ongoing

Source: (CNSC 2023f)

* While simultaneously working on the tasks of Phase 1

ARC-100

New Brunswick Power (NB Power), the operator of a CANDU reactor at the Point Lepreau site, is planning to build an ARC-100 reactor at the Point Lepreau site in New Brunswick in collaboration with ARC Clean Technology Canada (for the ARC-100, see also the presentation in Chapter 7.4.1.2). The current developments are summarised by (CNSC 2023i).

ARC-100 completed Phase 1 of the VDR in October 2019 after 24 months. The CNSC states the following key findings (CNSC 2019c):

- The developer must further develop its management system to demonstrate that it is able to control its design work, safety analyses and research and development activities in accordance with regulatory requirements.
- The developer must provide further information to show that its research and development work (R&D) is appropriate to the requirements of design development and the conduct of safety analyses.
- Due to the specific new design and safety characteristics of the concept, further information is required on specific aspects. This further information concerns:
  - Sufficient R&D activities to demonstrate the claimed safety characteristics and fuel qualification.
  - Applicability of previous operating data from other fast sodium-cooled reactors to the ARC-100 design.
  - Conformity of the safety classification of systems, structures and components with their safety function or importance.
  - Adequacy of the shutdown systems, the degree of subcriticality achieved and the guaranteed shutdown state, taking into account the independence, separation and diversity of the control and protection systems at all levels (including sensors, control logic and triggering).
  - Adequacy of the containment function and building closure for all triggering events.
  - Verification and validation of the computational models used in safety analyses.
  - Functionality, adequacy and reliability of the inherent and passive systems that are loaded in the safety case.

The CNSC considers these points to be solvable and expects them to be taken into account in further phases of the VDR. Work on Phase 2 of the VDR for the ARC-100 began in February 2022; no further information is available on the CNSC website.

In June 2023, NB Power, together with ARC Clean Technology Canada, submitted an application for site preparation to the CNSC (CNSC 2023i; NB Power 2023).

IMSR

Phase 1 of the VDR for the IMSR, a molten salt reactor from Terrestrial Energy, was completed in November 2017 after 24 months (for the IMSR, see also the description in Chapter 7.4.1.2). The CNSC states the following key findings (CNSC 2017):

- The developer must further develop its management system to demonstrate that it is able to control its design work, safety analyses and research and development activities in accordance with regulatory requirements.
- Various features of the concept are still in an early design phase. Due to the specific new design and safety characteristics of the concept, further information is required on specific aspects. This further information concerns:
  - The adequacy of the predictions of performance and reliability of various systems with regard to their safety significance.
  - The adequacy of the design parameters for safety-relevant systems, structures and components.
  - The importance of a prototype plant for further safety demonstrations.
- Due to new design features, further information is required on specific aspects. This further information concerns:
  - Predicting the behaviour of the reactor core in the event of damage to core components.
  - The developer has classified the failure of the reactor vessel as a severe accident. The question of what constitutes core damage in the IMSR concept needs to be developed further.
  - The control of reactor performance in the IMSR is achieved by controlling heat removal to the primary or ultimate heat sink using negative reactivity feedback in the reactor core. For Phase 2 of the VDR, the developer must demonstrate the validation of predictions for reactor dynamics with regard to ageing mechanisms throughout the entire lifetime of the reactor.
- The appropriate consideration of human factor aspects in the operation and maintenance programmes must be demonstrated.

The CNSC considers these points to be solvable and expects them to be taken into account in further phases of the VDR.

Phase 2 of the VDR for the IMSR-400 began in May 2018 and will be completed after 60 months in April 2023. CNSC notes that the design of the IMSR-400 is not yet complete and further details still need to be developed and evaluated. The results of Phase 2 for the IMSR-400 are summarised as follows (CNSC 2023d):

- Overall, the developer has correctly interpreted and implemented the CNSC's overarching requirements.
- No fundamental obstacles to licensing of an IMSR were identified. However, the IMSR design requires further progress to confirm that all CNSC requirements can be met.

- Additional necessary technical clarifications and findings include:
  - Further details are required on some aspects of the reactor design to confirm the functionality of the in-depth safety concept and safety functions.
  - Sensors and monitoring devices need to be developed for key plant parameters (such as molten salt chemistry), and instrumentation and controls need to be completed.
  - For certain low-power operating conditions, further tests are required to determine the design of the shutdown system and start-up instrumentation.
  - In the area of reactor control and shutdown, further evidence is considered necessary, particularly in connection with the overall negative reactivity temperature coefficient. The coefficient must be demonstrated to be negative for all operating states and boundary conditions.
  - Gaps in knowledge in the following areas must be closed:
    - Operating conditions and possible phase transitions,
    - Chemical changes and their monitoring,
    - Retention of fission products and effects of contamination and
    - Interactions between molten salt and water, unless these can be practically excluded.
  - Further safety evidence is required for the effectiveness of passive safety systems, in particular the internal reactor vessel auxiliary cooling system.
  - Validation and verification of the software tools used for the safety analyses is required.

The CNSC will continue to pursue these points if further VDR steps or specific applications are made for the design of the IMSR. More recent information on this is not available from the CNSC.

MMR

The construction of a VHTR-type Micro Modular Reactor (MMR) is planned by Global First Power at the Chalk River Laboratories site. The concept envisages a reactor part and a decoupled part for the provision of energy (electricity, heat, process heat). The reactor core consists of TRISO particles that are embedded in either a graphite matrix or a SiC matrix. These are embedded in hexagonal graphite blocks as a moderator. The power is transferred, via a helium cooling circuit, to a secondary cooling circuit that uses molten salt. This can store the energy in tanks or transfer it to the energy generation system, see (Oeko-Institut e.V.; WIP; PhB 2021, Kpa. 6.2.6.3). The current developments are summarised by (CNSC 2023h).

MMR completed Phase 1 of the VDR in February 2019 after 24 months. The CNSC states the following key findings (CNSC 2019d):

- The developer must further develop its management system to demonstrate that it is able to control its design work, safety analyses and research and development activities in accordance with regulatory requirements.
- The development of the MMR is being driven forward jointly by various organisations. The developer must ensure that effective cooperation between these organisations is ensured in design, R&D processes and conducting safety demonstrations.
- Due to the specific new design and safety characteristics of the concept, further information is required on specific aspects. This further information concerns:
  - Sufficient R&D activities to demonstrate the claimed safety characteristics and fuel qualification.
  - Applicability of previous operating data from other high-temperature reactors to the MMR design and safety analyses.
  - Consistency of the safety classification of systems, structures and components of structural facilities with their safety function or significance.
  - Appropriateness of the shutdown systems, the degree of subcriticality achieved and the guaranteed shutdown state.
  - Appropriateness of the methods used in the context of probabilistic safety analyses.

The CNSC considers these points to be solvable and expects them to be taken into account in further phases of the VDR. Work on Phase 2 of the VDR for the MMR began in June 2021; no further information is available on the CNSC website.

SSR

The SSR-W300 concept is a fast molten salt reactor with 750 MW thermal and 300 MW electrical power. Uranium, plutonium and other actinides from spent fuel from today's power reactors are used in the form of molten salt in steel fuel rods (Alloy 91). During power operation, the heat is transferred to a secondary cooling circuit via a coolant molten salt in active circulation. The secondary cooling circuit consists of a nitrate molten salt. It includes large storage tanks of this molten salt in order to be able to temporarily store the power from the nuclear circuit and decouple power generation from nuclear generation (Oeko-Institut e.V.; WIP; PhB 2021, Kap. 6.2.5.3).

Moltex Energy, the developer of the SSR-W300, started the VDR process in Phase 1 at the CNSC in November 2017 and completed it in May 2021 after 42 months. CNSC points out that the design of the SSR-W300 has changed during the course of the test and will continue to change. The CNSC states the following key findings (CNSC 2021b):

- Overall, the developer has correctly interpreted and implemented the CNSC's overarching requirements within the scope of Phase 1.
- In many areas, Moltex Energy has chosen a technical approach that differs from the requirements in (CNSC 2023a) due to the specific characteristics of its reactor design. Moltex Energy has stated that it intends to provide appropriate justification for the use of these alternative approaches and understands that it will be expected to demonstrate that these alternative approaches will result in an equivalent or higher level of safety.
- Points that will need to be addressed further in the future include:
  - The developer must further develop its management system to demonstrate that it is able to control its design work, safety analyses and research and development activities in accordance with regulatory requirements.
  - The developer must develop a method for the safety classification of its systems, structures and components (SSC) and explain and demonstrate how it ensures that the SSCs are designed, manufactured, constructed, installed, commissioned, operated, tested, inspected and maintained so that their quality and reliability correspond to the classification.
  - Moltex Energy does not want a second control room. It must be demonstrated that at least an equivalent level of safety can still be achieved.
  - The developer envisages two new types of shutdown devices for the SSR-W300. The reliability of these devices must be demonstrated for all operating conditions and their ability to safely reduce the reactor to low power levels and put it into a guaranteed shutdown state must be demonstrated.
  - The containment concept is still at a conceptual level. Its ability to deal with both design basis accidents and beyond-design basis plant states must be demonstrated.
  - Due to the specific new design and safety characteristics of the concept, further information is required on specific aspects. This further information concerns:
    - The requirements for the interfaces between SSCs of different safety classifications;
    - The adequacy of the research and development activities to substantiate the fuel qualification program, including the role of a first-of-a-kind reactor;
    - Validation and verification of the software tools used for the safety analyses and
    - Functionality, adequacy and reliability of the systems and components of the inherent and passive safety that are specified in the safety case for the reactor.

The CNSC considers these points to be solvable and expects them to be taken into account in further phases of the VDR. More recent information on this is not available from the CNSC.

International cooperation

The CNSC is pursuing a strategy of international cooperation in the area of SNR regulation (CNSC 2019b). It refers to the work within the framework of working groups of the NEA (Working Group on the Regulation of New Reactors, WGRNR; Working Group on the Safety of Advanced Reactors, WGSAR) and the IAEA (SMR Regulators’ Forum), see also Chapter 7.2. It is also striving to work in the IAEA’s Nuclear Harmonization and Standardization Initiative (NHSI) (NRCAN 2022).

In addition, the CNSC pursues a wide range of bilateral and multilateral agreements with other countries, organisations and nuclear regulatory authorities.[221] Memorandums of Understanding (MOU) and other administrative agreements with other regulators are particularly important for developing the regulatory requirements of SNR further.

In order to improve and coordinate their licensing work, Canada and the USA concluded a cooperation agreement on technical cooperation in the regulation of SMR concepts in August 2019 (CNSC; NRC 2019). One result of this cooperation are joint reports (NRC; CNSC 2023b), such as on requirements for the qualification of TRISO fuels for SNR (NRC; CNSC 2023a) or in the context of pre-licensing activities for the IMSR or the Xe-100.

There is also specific cooperation on SNR with the United Kingdom (ONR; CNSC 2022). This includes, among other things, the development of a common approach to the technical review of new reactor concepts, cooperation in the context of pre-licensing procedures, an exchange of information on new reactor concepts and cooperation in the field of research, training and the development of regulatory approaches for new reactor concepts.

The CNSC has also signed an MOU with the nuclear regulatory authority of Poland in 2023 (CNSC; PAA 2023). The authorities thereby declare that they will work together to further strengthen their commitment to the exchange of best practices and experience in the testing of advanced and small modular reactor technologies. This cooperation can be expanded to facilitate a joint technical review of advanced and small modular reactor concepts, including the BWRX-300. The cooperation can include (CNSC; PAA 2023):

- The development of common technical review approaches for advanced and small modular reactor technologies that facilitate the resolution of common technical issues to enable regulatory reviews that comply with each participant's national regulations;
- Cooperation on pre-application activities to ensure that both sides are prepared for the efficient testing of advanced and small modular reactor concepts, including the exchange of independent regulatory review results;
- Collaborating on research, training and developing regulatory approaches to address unique and novel technical considerations to ensure the safety of advanced and small modular reactor technologies.

Other administrative agreements for the harmonisation of the regulations, which were concluded after 2000, exist between Canada and Argentina (2009), Australia (2007), Belgium (2023), China (2016), Estonia (2023), Finland (2010), France (ASN 2019, IRSN 2022), Ghana (2019), India (2015), Israel (2011), Japan (2015), Jordan (2011), Morocco (2019), Netherlands (2022), Romania (2016),

[221] For a complete overview, see https://nuclearsafety.gc.ca/eng/resources/international-cooperation/international-agreements.cfm, as of 19/08/2023, last accessed 04/01/2024.

Sweden (2014), Switzerland (2015), Spain (2020), South Korea (2012), South Africa (2007), Czech Republic (2021), Ukraine (2023) and the United Arab Emirates (2017

7.4.2.3 Identified research needs and previous research results

The CNSC formulated a strategy in (CNSC 2019b) with which it wants to further develop its ability to adequately regulate SNR in Canada. It identifies regulatory challenges due to, among other things, different reactor concepts, new deployment models, new operating concepts, modular construction and new fuel types.

Specifically, (CNSC 2019b) gives examples of differences to current reactor concepts: the use of different fuels and coolants, the increased use of passive systems, a reduction in on-site personnel up to purely remote reactor control, the transportability of reactor systems, the modularity of the systems and the centralised production of large components or systems in factories. The CNSC also points out that for SNR prototypes, an adjustment of safety margins may be necessary to compensate for higher uncertainties due to a lack of experiments or operational experience.

For its further development, the CNSC has established an “Innovation and Research Hub” to coordinate and research new technologies that could impact the regulation of nuclear energy in Canada (CNSC 2023b).

The CNSC runs a research and support programme (CNSC 2023m). In connection with SNR development, various research projects were carried out under this programme. The main questions and results published since 2010 with regard to SNR are summarised in the following.

Already in the 2011-2012 research period, the CNSC had, in the RSP-0273 project, a comparison of Canadian regulatory requirements for the design of nuclear power plants with the requirements of other regulators (USA, Finland, France and the United Kingdom) as well as the WENRA “Reference Levels” and the WENRA "Safety Objectives for New Nuclear Power Plants" carried out. The background to this was that for various reactor concepts discussed in Canada, the respective developers based their design on the respective national regulations of their countries of origin or their reactor concepts had even already been assessed by foreign regulators. For approval of such reactor concepts in Canada, changes to the design of the reactor concepts may thus be necessary, if essential requirements in Canada differ from those of other countries.

The main differences identified concern (CNSC 2012)

- Design measures for severe accident protection,
- Design of the containment,
- How aircraft crashes are handled,
- Dose acceptance criteria and radiological safety objectives,
- Application of the single failure criterion,
- Time available to operators before they must intervene in accidents,
- Extent of the requirements for the electrical systems and
- Inherent reactivity feedback characteristics of the reactor.

In the period 2014-2015 in the RSP-0299 project, the CNSC continued to carry out an evaluation of existing SMR designs with regard to their innovative safety properties and, based on this, an analysis of the Canadian regulations with regard to their applicability to such SMR designs. Six water-cooled SMR designs from non-Canadian manufacturers and one VHTR design (StarCore) from Canadian developer StarCore Nuclear were analyzed (HATCH 2014). The report identifies challenges and problems in the licensing of SMR. On this basis, the report reviews various regulatory documents, including REGDOC-2.5.2 (CNSC 2023a), which was under development at the time, for their applicability to the challenges and problems identified. The report identifies the regulatory handling of inherent and passive safety properties of SMR as a key aspect. It recommends adopting a tiered approach to the safety assessment of SMR. For the licensing of reactor concepts that have already undergone a licensing process outside Canada, the report recommends that the CNSC cooperate with the respective national regulatory authorities in order to enable rapid access to the design principles and the associated regulatory features of these SMR.

In 2011, the American Society of Mechanical Engineers (ASME) added a new Division 5 for use in high-temperature reactors (VHTR) to its “ASME Boiler and Pressure Vessel Code” in Section III (rules for the design of components for nuclear power plants). This code is also relevant for the CNSC. The background to this is the proposal by various SMR developers to use this set of rules for the licensing of their reactor concepts in Canada. Their reactor concepts often work with operating temperatures of up to 750 °C and reach temperatures of over 1000 °C in the event of malfunctions and accidents.

Against this background, the CNSC commissioned a technical seminar in the period 2017-2018, as part of the RSP-688.1 project, to convey requirements, differences in design principles and the technical basis of the code with regard to the structural integrity of materials in high-temperature reactors (Barnes 2018; Burchelle 2018; Jetter 2018; Sham 2018a; 2018b). The subject of the seminar was manufacturing and installation, testing, overpressure protection and quality assurance for various materials in high-temperature reactors.

In the period 2020-2021, the CNSC prepared a report as part of the RSP-715.1 project that addresses the applicability and limitations of the US “ASME Boiler and Pressure Vessel Code” (Section III Division 5) with regard to the design and construction of base materials and the rules for the design of welds for SNR licensing in Canada. Due to the complexity of the ASME Code and its application to safety cases, the CSNC organised a technical seminar for its employees and summarised the results in a report. The report focuses on the justifications and technical contexts for the requirements of the ASME Code, discusses prerequisites and limitations for the application of the code and develops proposals on how identified gaps, affecting the applicability of the code, can be closed (CNSC 2021a).

In the RSP-658.1 project, in cooperation with the U.S. NRC, two two-day technical seminars were held by the Oak Ridge National Laboratory (ORNL), in the period 2018-2019, on design principles and operating experience in the USA with molten salt reactors (MSRE) and sodium-cooled fast reactors (including EBR-II) (ORNL 2018a). The background is that various SMR developers base their reactor concepts on the research and operating experience of the EBR-II or the MSRE. The topics presented by ORNL included the nuclear physics, engineering design and safety analysis for MSR and SFR.

With a view to the licensing of molten salt reactors (MSR), the CNSC has developed an overview of the state of knowledge on the properties of various salt compositions for MSR as part of the RSP-658.2 project in the period from 2019 to 2020. In a two-day workshop organised by the JRC Karlsruhe

on the properties of NaF-KF-UF4/ZRF4 salt, as well as sodium chloride and actinidene trichloride salts, contributions were made on the history of MSR development, experimental facilities for investigating the properties of molten salts, existing uncertainties in the data on the properties of fuel molten salts, the boiling and freezing points of the various salt compositions and the solubility of impurities, the thermal properties of the salts and the monitoring and evaluation of corrosion behaviour. Other topics relevant to the work of the CNSC include the chemical stability of the salt compositions under radiation exposure, the potential for erosion effects on core structural materials such as steels or graphite, and the retention of the molten salts by the structural materials (CNSC 2020b).

As part of the RSP-726.1 project, the CSNC commissioned the Argonne National Laboratory (ANL) to hold a technical seminar on gas-cooled high-temperature reactors (HTGR) for CNSC staff in 2021-2022. The topics addressed included the retention of fission products in the TRISO fuel under normal operating and accident boundary conditions, geometric and structural changes due to the irradiation of graphite, the design of the reactor core and the performance of neutron physics calculations, secured subcritical plant states, the methodology for the selection of initiating events for safety analyses, risk analyses, mechanical design and behaviour of structural materials for high-temperature applications and their failure mechanisms (CNSC 2022d).

7.4.2.4 Conclusions

Canada is pursuing an extensive programme to introduce SMR, both at the national and provincial levels. SMR in Canada include both water-cooled concepts and non-water-cooled concepts (known as ANT in Canada). All SNR concepts specifically pursued in Canada fall within the typical power output range of SMR.

As for the SNR technology lines, SFR, VHTR and MSR concepts are specifically being pursued in Canada.

For this purpose, several pre-licensing procedures (Vendor Design Reviews, VDR) have already been or are being carried out. The time required for this was typically a factor of 1.5-2 to a factor of 3.5 longer than the time required for such VDR procedures planned by the CNSC. Other procedures are still delayed or have been cancelled without results. The results of the completed VDR (Phases 1 and 2) identified extensive open questions that would have to be clarified before an SNR in Canada is approved. Despite such open questions, two applications for site preparations for the construction of SNR (ARC-100 and MMR) have already been submitted in Canada, but no Level 3 VDR and no application for construction or operating permits have yet been submitted.

The Canadian nuclear regulations are more goal-oriented than prescriptive. Nevertheless, the CNSC considers a revision to be necessary for SMR licensing. For this purpose, a number of aspects have been identified in which SMR in general, and SNR in particular, differ from previous reactor concepts and which make a revision of the Canadian regulations necessary. The CNSC has formulated a strategy for the successful regulation of new reactor technologies and is pursuing the goal of formulating a technology-neutral set of regulations. At the same time, regulatory documents were developed and adopted explicitly for water-cooled reactor concepts, but none explicitly for SNR.

The Canadian regulator pursues international cooperation, with close bilateral cooperation with the USA, but also with the UK and, more recently, with Poland.

7.4.3 United Kingdom

With regard to the United Kingdom (UK), the following section first looks at SNR development within the framework of the British government's overarching objectives in the field of power generation. The activities of the British regulator, the Office of Nuclear Regulation (ONR), are presented thereafter. Finally, there is an overview of the research agenda pursued by the ONR in the UK, the research questions identified therein and the knowledge already gained on SNR.

7.4.3.1 SNR development in the UK

The current position of the UK, with regard to SNR development, is described in a policy paper by the Department for Business, Energy and Industrial Strategy (BEIS) in the UK (BEIS 2023b).[222]

Accordingly, in the UK, the term “Advanced Nuclear Technologies” (ANT) refers to two groups of reactor concepts: on the one hand, water-cooled reactor concepts with lower output than current light water reactors (Small Modular Reactors, SMR), and on the other hand, “Advanced Modular Reactors” (AMR), which differ from today’s reactor concepts through alternative coolant or fuel concepts. For both groups, the BEIS defines that these are reactor concepts with lower power than today's light water reactors and that they are designed in such a way that they can be largely manufactured industrially in factories (modularity). For specific UK activities within the framework of the GIF, please refer to the chapters on the respective technology lines.

As early as 2017, the British government had decided to invest up to £460 million in the development of nuclear technologies (fuels, manufacturing processes, reprocessing and advanced reactor concepts) (BEIS 2017). Of this, £180 million was to be invested in the further development of nuclear technologies as part of an “Energy Innovation Programme” by the BEIS (BEIS 2020d).

In the funding period from 2016 to 2021, projects were funded in three phases in the “Nuclear Innovation” programme. In Phase 1, six organisations working on “materials and manufacturing processes”[223], two organisations within reactor concepts[224] and the National Nuclear Laboratories in the area of fuel development and reprocessing received a total of £12.5 million. In Phase 2, £3.7 million was made available for the development of reactor concepts and £4.3 million for fuel development. In a third phase, projects in the field of thermo-hydraulic modelling and the development of digital methods for reactor development were funded (BEIS 2020d).

In another funding programme on “Advanced manufacturing and materials” (AMM), projects were funded with the aim of strengthening the national supply chain, further developing modern manufacturing processes and ensuring core competencies for the design, manufacture, operation and supervision of future reactor concepts. In a first phase from 2017 to 2019, £5 million were made available for this purpose, and in a second phase from 2019 to 2021 a further £20 million (BEIS 2020d).

[222] The BEIS existed until 2023 and was then divided into the Department for Business and Trade (DBT), the Department for Energy Security and Net Zero (DESNZ) and the Department for Science, Innovation and Technology (DSIT), see https://www.gov.uk/government/organisations/department-for-business-energy-and-industrial-strategy, as of 26/10/2023.

[223] Frazer Nash, Amec Foster Wheeler, Brunel University London, Cammell Laird Energy, Nuclear AMRC und University of Sheffield

[224] Frazer Nash und Amec Foster Wheeler

In the first phase, a total of four projects were awarded to Rolls-Royce to develop novel methods for the manufacture of reactor pressure vessels, to Sheffield Forgemasters to produce a flexible electron beam welding cell for processing 200 mm thick steel, to Cammell Laird to transfer modular construction methods from shipbuilding to the nuclear sector and to EDF Energy to review existing computer programmes and nuclear standards for high temperature reactors based on the knowledge available in the UK from the operation of advanced gas-cooled reactors (AGR) (BEIS 2020b).

In the second phase, a total of seven projects were funded, these were carried out by (BEIS 2020b):

- Nuclear Energy Components for the replacement of soluble oil coolants with supercritical carbon dioxide,
- U Battery Developments Ltd for the manufacture and testing of a reactor pressure vessel for its U Battery reactor concept,
- Rolls-Royce for the development of novel methods for signal transmission in control systems,
- Cavendish for novel welding processes,
- Create technologies for the further development of an automated radiography system for weld seam testing,
- Laser Additive Solutions for the development of an additive manufacturing process for complex components supported by artificial intelligence and
- Jacobs for the development of advanced testing methods to demonstrate the quality of components produced using advanced manufacturing processes.

As part of the Energy Innovation Programme, BEIS finally launched an Advanced Modular Reactor (AMR) feasibility and development (F&D) funding programme with funding of £44 million (BEIS 2020c). The aim of this programme was to support feasibility studies for a broad group of alternative reactor concepts that would enable lower levelised costs of electricity, greater flexibility in electricity generation (load-following operation), applications in the field of process heat or hydrogen production or other alternative applications. In Phase 1, a total funding of a maximum of £4 million was made available for this purpose. Eight organisations were each funded with up to £300,000. In Phase 2, a further £40 million was made available for successful Phase 1 projects. In this phase, in addition to a merger project, Westinghouse Electric Company UK received project funding for its concept of a lead-cooled fast reactor and U-Battery Developments Ltd for a gas-cooled high-temperature reactor. A further £5 million was made available to support the British regulatory authority ONR in its monitoring of AMR development (for a more detailed description of the reactor concepts funded, see also (Oeko-Institut e.V.; WIP; PhB 2021, Kap. 4.2.9). Further development of the U-Battery concept was discontinued in 2023 (urenco 2023).

In 2020, the British government formulated its goals for a green industrial revolution in a ten-point programme (HM Government 2020). Point 3 of this also includes the development of ANT. £385 million is to be made available for the development of ANT as part of an “Advanced Nuclear Fund” (ANF). Up to £215 million of this is to be used to develop a national SMR design. A further £170 million is to be used for AMR development. In this context, AMR is understood to mean reactor concepts that operate at temperatures of 800 °C or above (VHTR) and which are to be used for the effective production of hydrogen and synthetic fuels. The funding provided through the ANF is also intended to stimulate up to £300 million in private investment. With a view to the development of

AMR, the aim is to commence operation of a demonstration reactor in the early 2030s. A further £40 million is to be made available for the further development of the regulatory system and the development of a national supply chain (HM Government 2020, pp. 12–13).

The British government has published an “Energy White Paper” to implement the ten-point programme (BEIS 2020a). One of the key goals is to make a decision on building another (large) nuclear power plant (in addition to Hinkley Point-C) within the legislative period. It is pointed out that the nuclear industry has committed itself to reducing the costs of new nuclear power plants by 30% compared to current levels, see (BEIS 2018).

In addition, the government confirms in this Energy White Paper that it will provide the funding for ANT mentioned in the ten-point programme. In this context, (BEIS 2020a, p. 50) notes that a 2014 study (NNL 2014) estimated the global market potential for SMR and AMR at up to £400 billion in 2035. Actually, (NNL 2014) determines the market potential for SMR and AMR in 2035 is 65-85 GW, which would correspond to an investment volume of £250-400 billion. However, it should also be noted that (NNL 2014, p. 19) determines this market potential on the basis that a potential of 20-21.5 GW from SMR and AMR has already been identified by 2025.[225] In contrast, another analysis commissioned by BEIS found that the potential for the introduction of AMR by 2050 is low (Vivid Economics 2019b, p. 44) and that there is very high uncertainty about the costs of as yet undeveloped nuclear systems such as SMR and AMR (Vivid Economics 2019b, p. 54). As key areas of innovation for nuclear energy, especially for AMR, (Vivid Economics 2019a) identifies digitalisation, modularisation, simplification of design, reduction of plant downtimes through new fuels and materials with longer operating times, and increased flexibility in use (high-temperature provision).

In its “Net Zero Strategy” (HM Government 2021), the government reaffirms its planned funding strategy and announces the introduction of a "Future Nuclear Enabling Fund" of £120 million for 2022. This is intended to help overcome potential barriers to the introduction of ANT (HM Government 2021, p. 103). According to (BEIS 2023b), this fund, as of October 2023, has not yet been implemented.

The spectrum of possible AMR was already limited to the VHTR technology line in the ten-point programme (HM Government 2020). To review this focus, a public survey was conducted by BEIS (BEIS 2021a) in 2021 with the aim of limiting the “Advanced Modular Reactor (AMR) Research, Development & Demonstration Program” funded under the ANF to the research and development of VHTR concepts. The reason given was the aim of being able to contribute to the production of hydrogen, the provision of process heat and the efficient production of electricity by providing high-temperature heat. Furthermore, VHTR reactor concepts have been accredited, based on, i.a. (NIRO 2021), a high Technology Readiness Level (TRL) and potential synergies with the previous development of graphite-moderated, gas-cooled reactors in the UK. The TRL and the need for further development for the VHTR, SFR, LFR and MSR technology lines were also examined in (NNL 2020). The results, as provided in (NNL 2020), were a TRL of 7 for VHTR with core exit temperatures of up to 750 °C, a TRL of 5 for VHTR with core exit temperatures of up to 950 °C, a TRL of 7 for SFR, a TRL of 5 for LFR, a TRL of 4 for thorium-based MSR and a TRL of 2 for MSR with fast neutron spectrum. As a result of the survey conducted by BEIS, BEIS concludes that no relevant additional

[225] According to (IAEA 2023g), as of the beginning of November 2023, 2.38 GW of power plants are under construction worldwide that can be assigned to SMRs or SNRs. Two SFRs in Russia with a total capacity of 1.38 GW and two SNRs with a total capacity of 0.064 GW are in operation. In total, this corresponds to less than 20% of the installed capacity in the SMRs and AMRs areas expected by NNL for 2025.

information was provided to change this decision. Therefore, the funding program for AMR was geared towards the development of a demonstration reactor for VHTR by the early 2030s (BEIS 2021b).

In a Phase A, two technology fields were advertised for the period 2022-2023 as part of the AMR funding programme, one for reactor studies, the other for fuel studies. In the area of reactor studies, four projects were funded with up to £500,000. These were projects by EDF Energy Nuclear Generation Limited, the National Nuclear Laboratory, U-Battery Developments Ltd and the Ultra Safe Nuclear Corporation. In the area of fuel studies, two projects in the area of the development of coated fuel particles were funded. These were projects by the National Nuclear Laboratory and Springfields Fuels Limited (BEIS 2023c).

In a Phase B, three further projects will be funded until March 2025 (BEIS 2023d). In a first reactor project, funded with £15 million, the National Nuclear Laboratory is developing the concept for a UKJ-HTR together with the Japanese Atomic Energy Agency. In a second reactor project, funded with £22.5 million, the Ultra Safe Nuclear Corporation (USNC) UK is developing an advanced design of its Micro Modular Reactor (MMR). This design is intended to enable a core outlet temperature of 750 °C and to be used to produce hydrogen and jet fuel. Collaboration with USNC projects in the USA and Canada is planned. In a third project, the National Nuclear Laboratory, in collaboration with the Japanese Atomic Energy Agency, is receiving £16 million to advance the production of coated fuel particles in the UK.

In addition to these projects, the Department for Energy Security and Net Zero (DESNZ), one of the successors to BEIS, is funding a project to retain know-how in the field of AMR with £4 million (BEIS 2023a).

An overview of the entire research and development landscape in the UK is available at (NIRO 2022).

7.4.3.2 Regulatory requirements for SNR and work of the ONR regulatory authority

As part of the BEIS funding, up to £7 million was also invested from 2017 onwards to strengthen the capacity of the regulators (Office for Nuclear Regulation, ONR and Environment Agency, EA) to support and assess advanced nuclear technologies (BEIS 2017). From 2020, the national regulators will also pursue a focused work programme alongside the work under the AMR funding programme. To this end, they have received funding of £830,000 in Phase A and £4.3 million in Phase B. This will enable the authorities, among other things, to review their existing regulatory frameworks with a view to the development and licensing of AMR and to work with the organisations carrying out projects under the AMR (BEIS 2023a).

The ONR has used these funds to expand its own capacity to regulate ANT. To do this, it has identified knowledge gaps and developed a work program to build up the necessary technical expertise in the field of ANT. The ONR also wants to ensure that its processes and guidelines for the licensing of new reactors are also suitable for the licensing of ANT. To this end, it has already revised its “Generic Design Assessments” process (ONR 2023a).

With a view to the introduction of new reactors in the UK, the nuclear regulatory authority Office for Nuclear Regulation (ONR) distinguishes four different fields of work.[226] In a first field of work, the ONR carries out so-called “Generic Design Assessments” (GDA), which are also of relevance for ANT. Two other fields of work include the licensing and construction of new reactors. In these work areas, as of October 2023, ONR names ongoing activities exclusively in relation to the construction of the Hinkley Point C plant. In a fourth work field, ONR deals with regulatory issues relating to ANT.

Regarding the licensing of nuclear facilities in the UK, (ONR 2021) summarises the key regulatory framework. It does not address specific issues relating to ANT.

As a first step before a new reactor is built in the UK, a reactor concept can go through the Generic Design Assessment (GDA) process (ONR 2020b). The process is used to determine whether the reactor design complies with the safety requirements for construction, operation and dismantling (ONR 2019a). The GDA process was revised by the ONR in 2019/2020 in order to also be able to accept possible applications from developers in the ANT field and to incorporate the experience gained from the ONR's collaboration with various developers as part of the BEIS' Advanced Modular Reactor (AMR) feasibility and development programme (ONR 2023a). As a result, new guidelines for applicants (ONR 2019a) and new technical guidelines (ONR 2019b) were developed. According to (BEIS 2023b), the renewed Generic Design Assessment (GDA) process was opened by the ONR in May 2021. As of October 2023, only light water reactor SMR concepts (Rolls Royce and Holtec International) are undergoing the GDA process. Also, only light water reactor concepts have gone through a GDA process so far (ONR 2023c).

As part of the GDA process, the applicant must submit a “safety case” and support it with appropriate documentation that demonstrates compliance with the requirements of the “Safety Assessment Principles For Nuclear Facilities” (SAP) (ONR 2020a). These SAP are supported by associated “Technical Assessment Guides” (TAG) (ONR 2023e). Requirements with regard to security are formulated in the “Security Assessment Principles” (SyAP) (ONR 2022b). A review of the basic guidelines (ONR 2020a; 2022b; 2023e) with regard to their applicability to ANT is planned (ONR 2023a). The first research reports on this are available; see the next section.

According to (ONR 2019a, p. 27), the applicant must also take international regulations, in particular those of the IAEA and WENRA, into account. The ONR notes that the SAP and SyAP were benchmarked against the international regulations in order to ensure basic compliance with international requirements. The “Utility Requirements” (EUR 2012), on the other hand, are not used by ONR as a binding and sufficient set of rules.

As part of the GDA process, ONR can also carry out or commission research to answer open questions that are not clarified by the applicant. The ONR also aims to use information already available from regulatory processes abroad and to develop new information together with foreign supervisory authorities and international organisations (ONR 2019a, p. 37).

According to (ONR 2023e), of 109 TAG as of October 3, 2023, only the two TAG “The single failure criterion” (NS-TAST-GD-011) and “Diversity, redundancy, segregation and layout of mechanical plant” (NS-TAST-GD-036) are under review.

[226] In addition to the nuclear regulatory authority ONR, other authorities such as the EA are involved in the licensing process for new reactors, which will not be discussed further here as they are not concerned with nuclear safety issues.

As part of its work in connection with ANT, the ONR refers to international cooperation, among others within the framework of the IAEA SMR Regulators' Forum, see Chapter 7.2.1.4, and the OECD/NEA Working Group on the Safety of Advanced Reactor Concepts, see Chapter 7.2.2, as well as to general cooperation in the context of regulatory development at the IAEA and WENRA levels and bilateral work with, among others, the Canadian regulatory authority CNSC, see Chapter 7.4.2, and other organisations, such as the Japanese Atomic Energy Agency or the French IRSN (ONR 2023a).

The ONR currently has “Information Exchange Arrangements” (IEAs) with 14 organisations in 13 countries (ONR 2023b). These include Belgium (Federal Agency for Nuclear Control, FANC), China (National Nuclear Safety Administration, NNSA), Finland (Säteilyturvakeskus, STUK), France (Autorité de sûreté nucléaire, ASN; Institut de Radioprotection et de Sûreté Nucléaire, IRSN), India (Atomic Energy Regulatory Board of India, AERB), Ireland (Environmental Protection Agency, EPA), Japan (Nuclear Regulatory Authority, NRA), Canada (CNSC), Poland (Panstwowa Agencja Atomistyki, PAA), Sweden (Strålsäkerhetsmyndigheten, SSM), South Africa (National Nuclear Regulator, NNR), USA (U.S. NRC) and the United Arab Emirates (Federal Authority of Nuclear Regulation of the United Arab Emirates, FARN). These IEAs typically also cover issues related to the licensing of novel reactor concepts or SMR and AMR.

There is specific cooperation on AMR with Canada (ONR; CNSC 2022). This includes, among other things, the development of a common approach to the technical review of new reactor concepts, cooperation in the context of pre-licensing procedures, an exchange of information on new reactor concepts and cooperation in the field of research, training and the development of regulatory approaches for new reactor concepts. In addition, there is fundamental cooperation with the French regulatory authority ASN. (ONR 2022a, pp. 7–8).

7.4.3.3 Identified research needs and previous research results

In connection with its regulatory tasks, the ONR also carries out its own research or commissions it (ONR 2023d). To this end, it published a research agenda in 2019 (ONR 2022a). This sets out the research needs that the ONR believes are necessary for the period 2020-2040. An overview table identifies the overarching research topics (“areas of interest”) (ONR 2022a, pp. 11–13). With regard to the area of ”new reactors including innovation”, the ONR states that only minimal research has been carried out in this field to date, as further clarification of the BEIS’s position on future reactor concepts (SMR and AMR) is first required. After a further determination by the BEIS, however, ONR expects considerable research needs here in the future (ONR 2022a, p. 13).

In an Appendix A, research questions are formulated for five-year periods from 2020-2040. For new reactors including innovation, the ONR expects a research need for the entire period from 2020-2040, but only specifies this within the framework of specific “cross cutting specialism areas”. With regard to AMR, these are:

- Chemistry and chemical engineering:
  - Chemistry of AMR (2020-2030)
  - Production processes of fuels for ANT such as TRISO, SFR fuel and others as well as issues related to HALEU fuels (2020-2030)

- Construction and external influences:
  - Use of seismic insulation for ANT (2020-2025)
  - High-temperature vessel structures made of concrete (2025-2030)
  - Novel materials/nanotechnology (2030-2035)
- Electronic control and instrumentation:
  - Various issues, including: in connection with the use of artificial intelligence (2020-2025)
- Failure analysis, failure studies:
  - Thermohydraulic and physical modelling at SNR including validation (2020-2030)
  - Safety claims and modelling of passive systems (2020-2025)
  - Increased use of AI techniques in modelling safety cases (2020-2025)
- Failure analysis, reactor core:
  - Additive manufacturing technologies for the production of nuclear fuel (2030-2035)
- Failure analysis, probabilistic safety analyses:
  - Modelling of passive safety systems in PPE (2020-2030)
  - PPE for AMR (including new or novel accident sequences or impacts) (2020-2030)
- Human and organisational capabilities:
  - Human factors in the development, justification and application of artificial intelligence and machine learning in nuclear engineering (2025-2035)
- Mechanical engineering:
  - Applications of carbide and ceramic materials and their integrity, strength and durability for nuclear applications (2035-2040)
  - Impact of modular construction on quality assurance in construction (2020-2030)
- Indoor impacts and site safety:
  - Characterisation of hazards associated with novel concepts and materials in ANT, (2020-2040), e.g.
    - consideration of fire initiation mechanisms, explosion hazards, hazard combinations, etc.
    - Models including new and tailored approaches specifically for ANT

- Structural integrity:
  - Impact of new manufacturing processes on structural integrity, e.g. additive manufacturing, hot isostatic pressing, electron beam welding, cladding (2020-2030)
  - Mechanisms of high temperature degradation for AMR (2020-2030)

Results of its research projects are published by ONR (ONR 2023d). So far, only a few projects can be found there with a clear connection to AMR.

In one project, international regulatory approaches for new technologies and new materials were evaluated (RSD; wood n.d.b). The nuclear industry as well as other high-risk technology areas were considered. The aim was to derive recommendations for the development of regulatory requirements in the field of nuclear technology in the UK. For this purpose, specific formulations for requirements were developed as part of a Technical Assessment Guide (TAG) of the ONR. In addition, the applicability of the Safety Assessment Principles (SAP), the Technical Inspection Guides (TIG), the TAG and selected IAEA guidelines was examined and the results summarised in appendices to the study.

In (RSD; wood n.d.a), the effect of temperature changes on the criticality in moderated fissile arrangements is investigated. Simple, infinitely extended critical systems with different moderators, different fissile materials, different enrichments of the fissile material and different moderator to fuel ratios at different temperatures were calculated. MONK was used as the primary calculation programme, and the results were compared with other calculation programs (MCNP, SCALE-KENO). The report states that the criticality of a system generally decreases with increasing temperature, but for strongly moderated systems with Pu-239 as the fissile material, the criticality increases with increasing temperature.

In (ONR 2018), the state of knowledge on iron-based structural materials in the reactor core of SFR is evaluated. The use of austenitic steels, HT9, T91 and oxide dispersion hardened steels (ODS) as cladding material or structural material in the reactor core under the operating conditions of SFR (erosive conditions, high temperatures and high neutron load) is analysed and previous worldwide operating experience is compiled.

In (CRA; ATLAS; GRS 2023), the modelling of software in the context of probabilistic safety analyses and the need for related regulatory requirements are analysed. To do this, different industries (in addition to nuclear technology, the areas of space, air traffic, medicine and the military) were considered with the aim of deriving recommendations for a revision or addition to the SAP or TAG. As a result, additions to the "Technical Assessment Guide 30: Probabilistic Safety Analysis“ (ONR 2019c) were derived.

In (Adelard LLP 2021), the UK's nuclear regulations are analysed to determine whether they formulate suitable requirements for the use of artificial intelligence and machine learning (AI/ML) in a nuclear context or where adjustments or additions may be required The study concludes that the SAP are fundamentally robust with regard to an assessment of the use of AI/ML, but that the underlying TAG require additional interpretations for application and, if necessary, further justifications for meaningful application must be developed.

7.4.3.4 Conclusions

In the UK, ANT development has been extensively supported with government funds since around 2016. In the UK's definition of the term, ANT includes AMR as well as SMR, with AMR in turn being understood to mean reactor concepts with alternative coolants or fuels. In the UK, however, AMR can also be understood as small, i.e. less powerful plants in the range of less than 300 MW electrical output and as modular plants.

At the beginning of funding in the UK, various AMR technology lines were funded, most recently an LFR concept and a micro-reactor concept from Urenco. Development by Urenco was discontinued in 2023. Since around 2020, funding in the UK has been limited to concepts in the VHTR technology line, so that the originally funded concepts, including that of an LFR, will no longer be funded.

In addition to research projects to develop reactor concepts, questions about various cross-cutting aspects (fuels, reprocessing processes, manufacturing technologies, testing techniques, calculation codes and models) will also be addressed within the framework of research funding. Finally, the national regulators will also be provided with their own funding to build up their own expertise in the field of ANT.

In this context, the nuclear regulatory authority ONR is pursuing its own work and research program to strengthen its competencies in the field of AMR and to revise processes and guidelines for the licensing of new reactors. In a first step, the “Generic Design Assessment” (GDA) process was revised and new guidelines published.

To date, however, no AMR concept has begun the GDA process in the UK, nor have there been any applications for the construction or operation of an AMR. The ONR is planning to review basic guidelines with regard to their applicability to ANT. The first research reports on this are available, but the ONR still sees considerable future research needs in the area of ANT.

With regard to its work on AMR, the ONR also refers to international cooperation; in particular with Canada there is explicit cooperation in the area of regulatory development and the licensing of AMR. Finally, ONR also has its own research program, which also addresses issues related to AMR.

7.5 Conclusion on national and international regulatory developments

The interim conclusions of the subsections are presented here once again in a summarised and slightly abbreviated form.

The national and international nuclear regulations were developed primarily on the basis of the knowledge gained from the construction and operation of today's water-cooled reactor concepts. On the one hand, they specify basic requirements (goal-oriented regulations), but they also specify specific technical specifications or set requirements with reference to specific technological solutions (prescriptive regulations). There is no prescriptive regulation for SNR, comparable to that for water-cooled reactor concepts, either nationally or internationally. As a result, regulators worldwide are reviewing and revising national and international regulations with a view to their applicability to SNR, both with regard to the existence of gaps and the applicability of existing regulations to other types of technologies.

SNR differ significantly from current water-cooled reactor concepts, particularly in the use of different coolants (and thus new phenomena, failure mechanisms or risks to be taken into account), increased

use of passive safety properties (and associated verification issues), other fuel concepts and possibly other fields of application (in particular high-temperature applications).

Based on such differences, the IAEA has identified numerous application problems of its previous regulations for SNR. This affects many central areas of reactor design and construction, such as the reactor core, containment, cooling systems, fuel handling and storage, and probabilistic and deterministic verification methods.

Due to the different fuel concepts, new questions continue to arise in connection with security and fissile material control (e.g. due to higher initial enrichments, different properties of the fuel elements or a more extensive occurrence of separated fissile material in reprocessing plants).

Questions of applicability of the regulations can arise from the fact that previous formulations are not technology-neutral, so that supplementing or adapting the regulations in a technology-neutral formulation can in principle offer a solution. In addition, questions of applicability were also identified arising from innovations in SNR for which corresponding requirements are missing in the existing regulatory texts. Due to significantly less operating experience from experimental and prototype reactors compared to water-cooled reactor concepts, it can be assumed that the development of requirements in such areas will take a considerable amount of time. For this, such plants must be planned, approved, built and operated, so that the time required for this is estimated to be one or more decades rather than a few years.

In the USA, nuclear power plants have so far been approved under 10 CFR Part 50 or Part 52. In both procedures, the underlying regulations contain prescriptive requirements. The US NRC is developing a new set of rules for the licensing of SNR, which is to be published by 2027. This is intended to provide a technology-neutral approach through increased use of goal-oriented (risk-informed, performance-based) requirements. The NRC has already issued the first regulatory guides and a new rule for the area of emergency planning, and a new rule is being developed in the area of security. For the different types of fuel used in SNR, the NRC is also dealing with the requirements for the qualification of such fuels, as well as the safety issues and risks associated with their production.

In addition, the NRC is pursuing extensive pre-licensing activities for various reactor concepts relating to SNR. In this context, a combined construction and operating license application for a microreactor (Aurora Powerhouse) was submitted as of January 2024, but the procedure was discontinued by the NRC without a final assessment due to a lack of cooperation from the applicant.

Furthermore, three test reactor construction applications have been submitted to the NRC to date, with one permit (for Kairos Power's Hermes) granted in December 2023. As part of the “Advanced Reactor Demonstration Project”, a demonstration reactor is to be tested, approved and built for each of two reactor concepts within 5-7 years starting in 2020. No application for licensing has yet been submitted to the NRC for these two reactor concepts (sodium, Xe-100). Of the four reactor concepts for which challenges for the construction of a demonstration reactor are to be solved within 10-14 years as part of the “Advanced Reactor Demonstration Project”, three developers have so far started pre-licensing activities with the NRC.

Canada is pursuing an extensive programme to introduce SMR, both at the national and provincial levels. SMR in Canada include both water-cooled concepts and SNR concepts (known as ANT in Canada). All SNR concepts specifically pursued in Canada fall within the performance range of SMR.

As for the SNR technology lines, SFR, VHTR and MSR concepts are specifically being pursued in Canada.

For this purpose, several pre-licensing procedures (Vendor Design Reviews, VDR) have already been or are being carried out. The time required for this was typically a factor of 1.5-2 to a factor of 3.5 longer than the time required for such VDR procedures planned by the Canadian regulatory authority (CNSC). Other procedures are still delayed or have been cancelled without results. The results of the completed VDR (Phases 1 and 2) identified extensive open questions that would have to be clarified before an SNR in Canada is approved. Despite such open questions, two applications for site preparation for the construction of SNR have already been submitted in Canada (ARC-100 and MMR), but no Level 3 VDR and no application for construction or operating permits have yet been submitted.

The Canadian nuclear regulations are more goal-oriented than prescriptive. Nevertheless, the CNSC considers a revision to be necessary. For this purpose, a number of aspects have been identified in which SNR differ from previous reactor concepts. The CNSC has formulated a strategy for the successful regulation of new reactor technologies and is pursuing the goal of formulating a technology-neutral set of regulations. At the same time, regulatory documents were developed and adopted explicitly for water-cooled reactor concepts, but none explicitly for SNR.

In the UK, the development of “Advanced Nuclear Technologies” (ANT) has been extensively supported with state funds since around 2016. In the UK's definition, ANT includes not only SMR but also AMR, which are reactor concepts with alternative coolants or fuels. In the UK, however, AMR can also be understood as small, i.e. less powerful plants in the range of less than 300 MW electrical output and as modular plants.

Initially, various AMR technology lines were promoted in the UK. Since around 2020, funding in the UK has been limited to concepts in the VHTR technology line, so that the originally funded concepts, including that of an LFR, will no longer be funded.

In addition to research projects to develop reactor concepts in the UK, questions about various cross-cutting aspects (fuels, reprocessing processes, manufacturing technologies, testing techniques, calculation codes and models) will also be addressed within the framework of research funding. Finally, the national regulators will also be provided with their own funding to build up their own expertise in the field of ANT.

In this context, the nuclear regulatory authority ONR is pursuing its own work and research program to strengthen its competencies in the field of AMR and to revise processes and guidelines for the licensing of new reactors. In a first step, the “Generic Design Assessment” (GDA) process was revised and new guidelines published.

To date, however, no AMR concept has begun the GDA process in the UK, nor have there been any applications for the construction or operation of an AMR. The ONR is planning to review basic guidelines with regard to their applicability to ANT. The first research reports on this are available, but the ONR still sees considerable future research needs in the area of ANT.

The national regulators in the USA, Canada and the UK refer to international cooperation, in particular between the USA and Canada and between Canada and the UK there is explicit cooperation in the area of regulatory development and the licensing of AMR.

Several of the applicability issues identified are now being addressed at national and international level through revision processes of existing regulations or the development of additional regulations, but gaps still remain. Overall, it can be assumed that the regulatory authorities and international organisations will need time to gain operating experience with individual real plants and to achieve and ensure a high level of confidence in the safety of the reactors. This also applies to potential revisions to the nuclear regulations, as such revisions require a sufficient evidence-based footing, which is largely not yet available.

By moving from a prescriptive to a more goal-oriented (technology-neutral) set of regulations, regulations can be created that are generally applicable to SNR. However, this means that SNR developers have to prove that their respective systems meet the target-oriented requirements without being able to rely on established procedures. At the same time, the licensing authority has to independently check the system-specific evidence submitted by the developers, which in turn requires the development of appropriate expertise and a clear knowledge of the relevant safety issues of the respective technology lines. This creates the risk that the corresponding licensing procedures will take a longer period of time.

8 Partitioning and transmutation

The SNR being discussed today provide for a large number of possible fuel concepts and fuel cycles. Many of the fuel cycles associated internationally with SNR still consider the use of uranium as fuel. There are also thorium fuel cycles in which the fissile material uranium-233 is bred from the fertile thorium. For many decades, the separation and use of the fissile material plutonium produced in LWR for reuse in the reactor in the form of uranium-plutonium mixed oxide fuel (MOX) has been carried out. Historically, fast reactors were also planned to breed new plutonium as a fissile material. Many of the reactor concepts in the technology lines therefore provide for the possibility of using plutonium in the fuel. Most of the fuels and fuel cycles associated with SNR therefore also provide for the use of reprocessing technologies in one form or another to separate transuranic elements (TRU, plutonium and minor actinides) from the fuel and reuse them again in the reactor as fissile material or to use transuranic elements to transmute them. Some reactors such as the MSR provide for the possibility of changing the fuel composition continuously, even during operation.

Generic statements can be made about some of the advantages and disadvantages of partitioning and transmutation (P&T) that are valid for most or all fuel cycles of SNR with P&T. However, many statements require specification of the respective reactor design and the fuel cycle in order to be able to evaluate possible advantages and disadvantages. In the following, therefore, only examples of SNR and corresponding reactor concepts are used, for which sufficient literature is available for this type of evaluation.

This chapter relates primarily to the impact of transmutation on disposal. For other advantages and disadvantages of P&T, please refer to (Oeko-Institut e.V. 2023; 2017; Renn 2014; Oeko-Institut/ZNF 2015).

As the results of the analysis have not changed significantly, older representations of P&T are still somewhat relevant. Therefore, the following descriptions in the subchapters are based on the descriptions from (Oeko-Institut e.V. 2023; Oeko-Institut e.V.; ZNF 2015), some of which have been adopted and supplemented by considerations focussing on SNR and their fuel cycles. An appropriate commentary was placed at the beginning of the respective subchapter.

8.1 Basics of transmutation

The following description was adopted and adapted from (Oeko-Institut e.V. 2023; Oeko-Institut e.V.; ZNF 2015).

During use in a LWR, several percent of the initial inventory of uranium is converted into other elements by fission and neutron capture. After irradiation, the spent fuel elements contain a large number of radioactive nuclides in the form of fission products and transuranic elements (plutonium and minor actinides such as americium and curium). Typically, depending on the time spent in the reactor, spent fuel contains around 1% plutonium, around 0.1-0.2% minor actinides and 4-6% fission products. Instead of repacking (conditioning) the spent fuel elements directly and transferring them to a repository after the required interim storage period, in a P&T system the spent fuel elements are treated in another nuclear facility using chemical and physical processes.

According to its proponents, P&T essentially promises the possibility of reducing the risk of disposal of highly radioactive nuclear waste, or even completely eliminating the need for geological repository, or significantly reducing the long isolation periods required. This would solve one of the main problems of nuclear energy production. The following explanation serves to verify this promise.

Transmutation involves specifically separating long-lived radioactive substances from nuclear waste (partitioning, P) and converting them into shorter-lived or stable isotopes by neutron irradiation in nuclear reactors (transmutation, T). Fast reactors (SFR, LFR, GFR, MSR) are a core component of waste treatment strategies with P&T, even though there are P&T concepts with thermal reactors (VHTR, MSR). In fast reactors, transuranic elements that have been previously separated from the highly radioactive waste can be split using the hard neutron spectrum. In addition, subcritical systems can also be used (ADS).

Uranium-235, plutonium, uranium-233 and minor actinides (neptunium, americium, curium) in a uranium-238 or thorium fuel matrix on a metallic, oxide or nitride basis, can be used as fissile material for transmutation plants. The thorium fuel matrix has the advantage that only a few transuranic elements are re-bred. It also permits the development of uranium-free inert fuels (inert matrix fuels, IMF). The fuels for molten salt reactors are liquid and based on chlorine and fluoride salts, whereby the solubility of transuranic elements in fluoride salts is limited. There are a large number of possible combinations of fissile materials and fuel compositions.

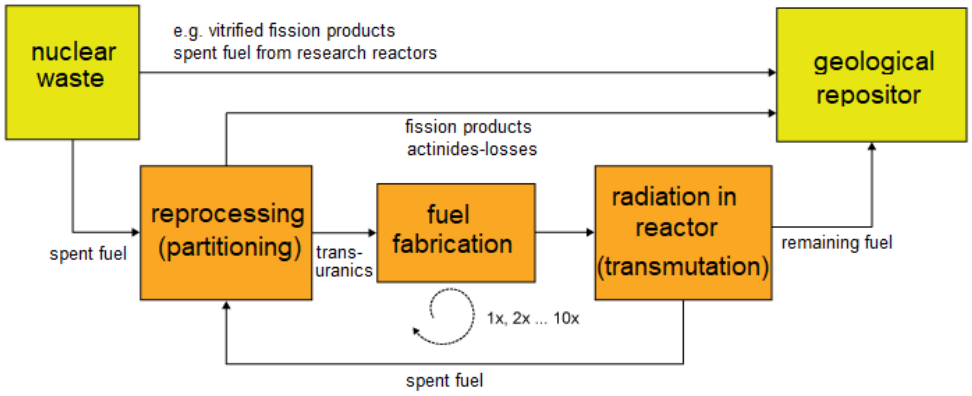
The implementation of a P&T strategy essentially involves the following steps (see

Figure 8-1):

1. Partitioning to achieve chemical and physical separation of elements in spent fuel.
2. Fuel production, or adding the separated elements to a fresh fuel.
3. Transmutation using neutrons by using the fresh fuel in a transmutation reactor in order to convert long-lived fission products into shorter-lived or stable isotopes.
4. Recycling, i.e. repetition of steps 1-3. This then leads to scenarios for the use of P&T in a fuel cycle system.
5. Final disposal. After interim storage and conditioning, remaining waste is transferred to the repository facility for disposal.

As can be seen from this schematic diagram, a large number of combinations of the technologies to be used for the P&T process are possible. In principle, strategies differ depending on whether the P&T technology is embedded in the context of a long-term strategy for the use of nuclear energy (closed fuel cycle with reuse of fissile materials such as plutonium), or whether the main goal is to reduce stocks of plutonium and minor actinides. A further distinction is whether a P&T programme is carried out purely nationally or in cooperation with other countries. For such transmutation scenarios, see Chapter 8.5.2).

Figure 8-1 Schematic diagram of transmutation



Source: Öko Institut e.V.

The individual steps and the technologies employed are described in more detail below. Each step in the process generates further process streams of radioactive waste due to the chemical and physical conversion of the radioactive materials, but also through operational, process and decommissioning waste from the required facilities.

8.2 Partitioning

The following description was adopted and supplemented from (Oeko-Institut e.V. 2023; Oeko-Institut e.V.; ZNF 2015).

In the case of fuels in fuel elements (SFR, LFR, GFR, VHTR, SCWR, ADS), spent fuel elements would be transferred to a reprocessing plant (RP). After use in the reactor, the spent fuel must be stored before reprocessing to allow the radioactivity it contains, and the associated heat generation, to decrease. The fuel is only reprocessed after this *interim storage period*[227]. Typically, the longer the fuel was irradiated in the reactor, the longer the interim storage period required, typically several years.

For liquid fuels (MSR), separation can take place continuously during operation, otherwise the fuel is exchanged in batches. This can significantly reduce the cycle time (Biss and Thomauske 2015).

In the RP, the fuel is chemically dissolved and the radioactive substances it contains are separated into several product streams in various chemical process steps. Each element requires a specific chemical process for separation, but there are also processes for the joint separation of chemically similar elements.

In the case of partitioning processes, most of the technical concepts currently being discussed nationally and internationally are based on the separation of plutonium and other minor actinides (Np, Am, Cm), which are characterised by high radiotoxicity and sometimes very long half-lives. The process steps for separating minor actinides are divided into homogeneous partitioning, i.e. joint separation of all transuranic elements (Pu and MA) or all minor actinides (Pu and MA separately), and heterogeneous partitioning. In heterogeneous partitioning, the individual actinides are separated from one another. In addition, there are concepts that also provide for the separation of radioactive

[227] Italicised terms are important parameters in the analysis of transmutation scenarios in Chapter 8.5.2.

fission products with long half-lives in order to strive for the most complete reduction possible. However, these currently only play a marginal role in the international discussion of P&T strategies (see Chapter 8.5.6).

For all separation processes, the achievable separation factor or separation efficiency is crucial to prevent residual amounts of the element being separated from remaining in the waste stream as separation losses. This is particularly crucial if the purpose of P&T is to reduce the requirements for radioactive waste disposal. The spent fuel must be reprocessed once in each cycle. The achievable separation factor determines which fraction of the remaining TRU inventory passes into the waste stream and is no longer available for a further transmutation step. The higher the separation factor, the lower the total losses that ultimately have to be sent for repository. Typically, separation factors of 99.9% must be achieved.

The number of possible processes is large; an overview of all chemical processes can be found in (Nash et al. 2010). Chemical separation processes for transuranic elements are often divided into wet chemical partitioning processes, using organic solvents, and pyrometallurgical separation processes (pyroprocessing) at high temperatures, e.g. using molten salts.

The individual process steps first consist of dismantling the fuel element and cutting the cladding tubes and then dissolving the fuel, which contains the isotopes produced in the reactor, e.g. in boiling nitric acid or in a molten salt. Industrial techniques for treating radioactive waste are used here, i.e. mechanical dismantling, cleaning, pulverisation, cleaning of exhaust gases and conversion into suitable chemical compounds for conditioning. The majority of the radioactive inventory remains in the exhaust air stream and the solvents, which must be conditioned by converting into a solid phase. Solvent extraction and ion exchange processes produce significant amounts of aqueous waste, as the extraction agents are not particularly radiation-tolerant and must be disposed of after multiple uses (IAEA 2019c).

Wet chemical processes such as PUREX (Plutonium URanium EXtraction) for plutonium separation are now fully developed and are used on an industrial scale. The process has its origins in the separation of plutonium for military use and has subsequently become established in commercially used reprocessing technology. After the fuel has been dissolved, the uranium and plutonium are separated from one another in two re-extraction steps by adding suitable liquid reducing agents. In order to separate the transuranic elements neptunium, curium and americium from the remaining liquid highly radioactive waste, further process steps are necessary, as is the separation of long-lived fission products; the processes are still under development. The non-extracted substances represent the remaining heat-generating waste, which is vitrified after concentration and storage in (glass) matrices (moulds) suitable for repository.

Plutonium is much easier to separate than americium and curium due to its stable oxidation states. Additional separation schemes are required for the extraction of minor actinides and the subsequent co-extraction and purification of lanthanide elements from the highly radioactive waste stream. The lanthanides must be separated in order to obtain americium and curium for subsequent transmutation. Due to the chemically complex behaviour of the minor actinides, the small chemical differences between the lanthanides and americium/curium and their mass ratio of around 1000:1, the separation factors are limited and a large number of stages are required for quantitative extraction. The lanthanides must be separated because these elements contain neutron absorbers and the lanthanides emit high levels of radiation, which makes subsequent fuel element production more difficult (Poinssot et al. 2012).

(NEA 2018) states that new hydrometallurgical processes for complexation and extraction have been developed over the past 20 years that could enable separation factors of up to 99.9% for americium and curium. No further trials have been carried out, however. Although there has been significant progress in this area over the past few decades, there is still no consensus on which system should be developed industrially (NEA 2018).

Although the fraction of curium in spent fuel is small, curium (especially curium-244) is a major neutron emitter and contributes significantly to the heat generated in the waste. Since the separation of curium and americium is chemically difficult, consideration has been given to foregoing the separation of curium (Poinssot et al. 2012; Modolo et al. 2012; Renn 2014). However, this makes subsequent fuel element production considerably more difficult. It would therefore make handling americium much easier, if the curium were separated and not used in transmutation. However, the curium would then be returned to the waste stream and disposed of without transmutation, thus limiting the achievable reduction in the requirements for repository.

Alternatively, pyrochemical separation processes have been tested on a laboratory scale for many decades (e.g. electrolysis, distillation, reduction, liquid metal, etc.). Molten salts (chloride or fluoride salts) or liquid metals (e.g. tungsten, bismuth, aluminium) can be used.

So far, two processes in particular have been developed to the point of a pilot project. The most advanced is pyroprocessing in the form of electrolysis in a medium of molten salt (molten salt electro-refining), which was developed primarily in the USA. On the other hand, the processing of oxide fuels (“oxide electrowinning”) is being developed primarily in Russia (NEA 2018). However, only the pyro-electrometallurgical process allows the separation of minor actinides, due to the difficulty of separating lanthanides from minor actinides in oxidising separation processes.

In the “oxide electrowinning process”, oxide fuels are dissolved in molten salts of sodium chloride-potassium chloride (NaCl-KCl) or sodium chloride-caesium chloride (NaCl-CsCl) after chlorination and plutonium oxide (PuO_2) is separated in a targeted manner. Attainable separation factors of 99.5-99.9% are reported (IAEA 2011d).

The process of molten salt electro-refining was developed for the Experimental Breeder Reactor II (EBR-II) at Argonne National Laboratory. The fast breeder reactor was operated from 1964 to 1969 in combination with the reprocessing of part of the spent fuel (INL 2007); at that time, around 7% of the heavy metal content remained in the molten salt (IAEA 2011b). In addition to metallic fuels, nitride and carbide fuels can also be treated with this process (IAEA 2011b). According to (NEA 2018), over 99.9% of actinides can now be recovered.

The corrosive environment at high temperatures is one of the challenges in developing pyroprocessing beyond laboratory scale (Romero 2007; IAEA 2011b). Pyrochemical processes could also be used as a supplement and in conjunction with the hydrometallurgical processes employed in industry. One advantage of pyrometallurgical separation processes is the potentially higher radiological stability of the chemicals used (IAEA 2011b). (NEA 2018) However, states that the amount of waste from pyrochemical and aqueous processes is comparable.

The NEA presented a comprehensive report (NEA 2018) on the status of chemical separation processes. The proceedings of its workshops “Information Exchange on Actinide and Fission Product Partitioning and Transmutation” complement this write-up (NEA Proc. 2001; 2003; 2005; 2007; 2010; 2012; 2013; 2015; 2017). An overview of the processes and the state of research was presented in (Oeko-Institut e.V. 2023; ISR 2021).

With regard to reprocessing, fuels from fast reactors have some special features that distinguish them from typical LWR fuels. On the one hand, as with MOX fuel for LWR, the specific activity and the decay heat are significantly higher due to the increased burn-up and the higher actinide content. The increased formation of platinum metals from the fission of plutonium can lead to insoluble residues; in the case of fuels that are only stored for a short time, the formation of volatile ruthenium is particularly significant. In addition, some of the fuels (carbidic, nitridic and metallic) have pyrophoric properties (IAEA 2011b).

Despite the very different fuel concepts for the SNR technology lines, the process steps for the pyrochemical and aqueous processing methods of the various SNR are conceptually similar, but they differ in the processes for separating and dissolving the fuel elements (head end) and different amounts of certain radioisotopes accruing with different properties (IAEA 2019c). Each fuel type requires its own plant layout and specific operation, which affects the chemical properties of the waste stream and the final form of the waste products. The waste from future decommissioning and maintenance is also influenced by the plant design (IAEA 2019c).

Regarding solid waste, the disposal of the compacted fuel cladding is of particular importance. Although they do not generate heat, they have a very high overall activity due to the adhering fuels and the induced activity. Ceramics, glasses and crystallines can be used as a matrix to immobilise liquid waste from reprocessing. Glasses are used primarily for immobilisation in aqueous partitioning processes. Waste that contains salts, such as those generated during pyroprocessing and MSR, is particularly difficult. Many of the waste forms examined so far can only absorb small amounts of salt waste. Ceramics are better suited for salt waste, but, like crystalline forms of waste, are still under development (IAEA 2019c).

In addition, operational and decommissioning waste from the partitioning plant is generated. One problem is the creation of secondary waste in the form of solid waste containing alpha emitters through decontamination (IAEA 2019c). The waste from fuel cycle plants also contains high concentrations of metals such as those found in fuel cladding, cladding and other hardware, such as Zr, Cr, Ni, Fe and Al, as well as their nitrates, depending on the process; alkali metal fluorides or chlorides; and metals or compounds contaminated with uranium (IAEA 2007c; 2007b; 2007d).

8.2.1 Conclusions on partitioning

Wet chemical processes such as PUREX (Plutonium URanium EXtraction) for plutonium separation are now fully developed and are used on a large scale. The separation of minor actinides (MA) is still under development. The separation of actinides from lanthanides (neutron absorbers) and the splitting of actinides, especially the separation of americium and curium, is chemically difficult and only limited separation efficiencies have been achieved so far. Without the separation of curium, the subsequent fuel element production is technically complex due to the high heat generation and neutron emission; if curium is separated, it would have to be irradiated in extra targets or would end up in the waste stream.

Pyrochemical processes are still under development. Pyro-electrometallurgical processes are best suited to the separation of MA. The separation efficiency for actinides can reach up to 99.9%.

The reprocessing of spent fuel elements from fast reactors, especially with multi-recycling, is more difficult due to the increased heat production and higher activity. Insoluble residues can occur. Some of the SNR fuels (carbidic, nitridic and metallic) have pyrophoric properties.

The amount of waste generated is specific to the technology lines and fuel cycles used. The metallic structural materials of the fuel elements for fast reactors in particular have a high level of activation. The separated stream of highly radioactive waste is immobilised in glasses, ceramics or crystalline materials. In addition, there is waste from the operation and dismantling of the partitioning plant.

8.3 Fuel production

The following description was adopted and adapted from (Oeko-Institut e.V. 2023; Oeko-Institut e.V.; ZNF 2015).

After separation, fresh fuel elements are made from the separated transuranic elements (SFR, LFR, GFR, VHTR, SCWR, ADS), or these are added to the liquid fuel (MSR). One possibility is to embed transuranic elements in a uranium or thorium matrix. This is already undertaken on a large scale today when separated plutonium is used in uranium/plutonium mixed oxide fuels (MOX), which can be used in conventional light water reactors. For P&T for disposal, the embedding of the minor actinides would be added. For transmutation, in most concepts it is necessary to first separate the elements to be transmuted from the highly radioactive waste in order to then produce new fuel elements from them or, in the case of the separation of fission products, to produce suitable targets for irradiation.

The separated TRU must be processed into new fuel and this fuel must be used for the next cycle in the reactor; the time required for this is the *processing time*. For the sum of the interim storage and processing time, a time requirement of at least 3-5 years can be assumed for solid fuels (Oeko-Institut e.V.; ZNF 2015; Renn 2014). MSR can be an exception here, since many reactor concepts provide for the integration of pyrochemical reprocessing, and then the fuel is reprocessed during operation.

In addition to oxidic fuel types, other fuels are conceivable for use in SNR. These include carbide ((U,Pu)C), nitride ((U,Pu)N), metallic and ceramic-metallic (CERMET) fuels. The various fuel types differ in important reactor physics parameters, for example: the density of fissile material in nitride fuels is up to 40% higher than in oxidic fuels due to the higher density of the fuel matrix. The fuels are also being developed to achieve higher burn-ups than is possible with MOX. However, they are at a lower level of development than MOX fuels. The properties of the fuels (oxidic, nitride, carbide, metallic) with regard to MA transmutation are also roughly the same (Salvatores and Palmiotti 2011). For the advantages and disadvantages of non-oxidic fuels when it comes to disposal, see Chapter **0**.

The production of fuels with higher content of minor actinides is particularly challenging. On the one hand, the content of strong gamma emitters in the fresh fuel is significantly increased. Americium-241 is formed from the radioactive decay of plutonium-241 and decays itself into neptunium-237. The relatively high volatility of americium leads to stricter requirements on process control in order to minimise losses and contamination of the production plant (Delage et al. 2015). The fraction of americium-241 in existing plants for commercial MOX production must therefore be limited. Due to its strong neutron radiation and heat generation, the use of curium would also make fuel production significantly more difficult (Salvatores et al. 2015; Modolo et al. 2012; Poinssot et al. 2012). Alpha radiation is also a major constraint, particularly for radiation protection due to the formation of dust particles (IRSN 2019).

So-called “Inert Matrix Fuels” (IMF) do not contain uranium, which avoids the re-production of plutonium. This fuel is therefore particularly suitable for a P&T strategy to treat a stock of waste, as the efficiency of transmutation is increased compared to MOX fuels. So far, only laboratory-scale research and initial irradiation tests have been carried out for IMFs. IMFs are also being developed for use in accelerator-driven systems.

Thorium fuels (see also Chapter **4.6.6.2**) could also be conceivable for use in P&T systems, as no new TRUs are bred from thorium, but these also have disadvantages due to their high long-term radiotoxicity (see Chapter 8.6.4). Thorium fuels are also still under development.

For fission product transmutation, however, it is possible to produce special targets that are exposed to neutron irradiation (see Chapter 8.5.6).

8.3.1 Conclusions on fuel production

MOX production with plutonium has been demonstrated on a large scale. The production of fuels with minor actinides has not yet taken place on an industrial scale and is technically challenging due to the increased activity, especially when using curium. The volatility of americium also places special demands on the process control and can limit the fraction of americium in the fuel. Other fuels such as carbide, nitride and metallic fuels are being developed, but the fuels hardly differ at all in terms of their transmutation properties. Uranium-free fuels are also being developed and would be particularly suitable for transmutation.

8.4 Recycling and factors influencing P&T scenarios

The following description was adopted and adapted from (Oeko-Institut e.V.; ZNF 2015).

Since only a small fraction of the transuranic elements originally used are fissioned during reactor operation, the steps of separating the fuel, producing the fuel and using it in the reactor must be repeated many times, depending on the to be achieved reduction. This leads to various scenarios for P&T use (see Chapter 8.5.2). A transmutation scenario can be spoken of when individual reactor concepts and fuel cycles are used together over a longer period of time.

First of all, a distinction must be made between scenarios for the permanent use of nuclear energy and scenarios seeking the reduction of an *initial inventory* of radioactive waste (phase-out scenarios).

The most important parameters for assessing a phase-out P&T scenario are the achievable reduction in the initial inventory and the time and cost required to achieve this. However, these parameters are interdependent. The following applies: the less of the initial inventory remains (target reduction see below), the greater the effort required. In scenarios of permanent nuclear energy use, the minimisation of waste generated (waste generation) is in equilibrium, or in the case of long-term dynamic scenarios with changing reactor technologies, the waste generated over the entire implementation time (see below) and the entire fuel cycle, and the costs required for this, are all decisive factors.

In a transmutation step, the fuel remains in the reactor for a number of years, the *irradiation time*. This is typically 3-5 years. A certain percentage of the initially introduced TRU inventory is converted by fission. Typically, the longer the fuel can remain in the reactor, the greater the fraction of the initial inventory in the fuel that is converted by fission (*transmutation fraction*, burn-up). The irradiation time

can be limited by the fact that the remaining fraction of TRU as fissile material is no longer sufficient for the operation of the reactor, or that the fuel itself can no longer be used.[228]

The sum of the necessary periods of time from the irradiation time in the reactor, the interim storage time for decay and the processing time during reprocessing and the production of fresh fuel results in the *cycle time*. It is the total time required to irradiate existing TRU material once in the reactor, allow the fuel to decay, separate the remaining TRU portion from the fuel and produce new reactor fuel from it.

In P&T scenarios for reducing a fixed initial amount of waste, a *target reduction* can be defined based on the initial inventory, which is to be achieved by a P&T scenario. In scenarios of the permanent use of nuclear energy, it makes more sense to use *waste generation* per electrical (or thermal) energy generated as a benchmark (typically in kg/TWh). The target reduction or respectively the waste generation affect the extent to which the requirements for geological disposal may be influenced by P&T.

A key factor in evaluating a phase-out P&T scenario (see Chapter 8.5.2) is the *implementation time*, i.e. the time from the start of a P&T campaign to the achievement of the desired target reduction. Since the cycle time must be run through once for each transmutation step in order to transmute the respective transmutation share (see Chapter 8.5), these two factors, together with the desired target reduction, essentially determine the required implementation time.

Of the total TRU inventory available at a given time, only a certain fraction can be in use in the transmutation reactors; this fraction represents the *reactor inventory*. The remaining TRU fraction is contained in spent fuel or is currently being processed into new fuel. The ratio of the respective reactor inventory to the existing TRU inventory is determined by the ratio of the irradiation time to the total cycle time.

If a uranium-free fuel (IMF or thorium) is used, the transmutation share of an initial amount of TRU also corresponds to the amount of TRU converted by fission, which in turn is proportional to the fission energy released. In phase-out scenarios, since this energy is generated within the cycle length, this also determines the *thermal output* of the required transmutation reactors. If a constant transmutation share is assumed, the existing amount of TRU and thus also the amount of TRU fissioned per year and the resulting thermal output decrease exponentially over the conversion period. This also means that the number of reactors required at any given time would decrease in line with the thermal output produced. While the required number of reactors is typically calculated as a time average of the total energy to be generated, ideally a relatively large number of reactors are actually required at the start of a P&T scenario, which decreases over time. Otherwise, the implementation time would increase. In continuous use scenarios, the handling of fissile material quantities is more flexible and larger quantities of fissile material can be stored and stockpiled over longer periods of time. In some long-term dynamic scenarios with an end to nuclear energy use, a phase-out scenario is carried out at the end to minimise residual amounts of TRU.

In reactors with uranium in the fuel, TRU is bred and the transmutation fraction is lower; this is taken into account by a *conversion factor* ≠ 0. This means that with the same transmutation fraction and the same cycle length, the required installed thermal output is greater. Therefore, either a larger number of reactors is required or the individual reactor must have a higher thermal output.

[228] For example, the build-up of gaseous fission products and the associated fuel swelling can be critical factors in limiting burn-up.

As a reactor once built is basically available for a certain *running time*, the existing reactor fleet would have to be replaced by new reactors after the period of time. At this point, sufficient number of reactors would have to be built to be able to produce the thermal output required at that time. In phase-out scenarios, the shortest possible implementation time can be achieved by using the entire inventory in reactors at the beginning of the scenario, but a proportion of the reactors would then have to be shut down after just one cycle time. If one were to realistically try to reduce the number of reactors required and the associated construction costs, the implementation time would conversely be increased

8.4.1 Conclusions on factors influencing P&T

Since only a small proportion of the transuranic elements originally used are fissioned, the steps of separating the fuel, producing the fuel and using it in the reactor must be repeated multiple times, depending on the desired reduction in TRU. This leads to different scenarios for the use of P&T. A distinction is made between scenarios of the permanent use of nuclear energy and scenarios in which the reduction of an initial inventory of radioactive waste is sought (phase-out scenarios). When considering scenarios, the initial inventory, the achievable target reduction (phase-out) or the minimisation of waste generation (continuous use), the cycle time and the implementation time, the reactor inventory, the thermal output of the reactors used, their conversion factor and their running time are of relevance.

8.5 Transmutation

The following description was adopted and adapted from (Oeko-Institut e.V. 2023; Oeko-Institut e.V.; ZNF 2015).

After a brief introduction to the topic of transmutation, Chapter 8.5.1 will first deal with plutonium recycling. As the reduction of minor actinium (MA) is also necessary for effective transmutation, the challenges of transmutation of MA are discussed in Chapter 8.5.2. In Chapter 8.5.3, possible advantages and disadvantages of the technology lines for transmutation are discussed. As most reactor concepts are used in fuel cycle or transmutation scenarios, phase-out scenarios are first discussed in Chapter 0 and then continuous scenarios of nuclear energy use with simultaneous minimisation of the amount of transuranic waste are discussed in Chapter 8.5.5. The individual technology lines and individual reactor concepts are discussed separately. Finally, a short excursus on fission product transmutation is undertaken.

Chapter 8.5 focuses on the reduction of transuranic elements (TRU) in waste, the achievable target reduction in transmutation scenarios and the amount of waste. However, a lower amount of TRU in the waste stream does not directly equate to an advantage for disposal in a geological repository. The advantages and disadvantages for disposal will be discussed later in Chapter 8.6, Effects on disposal. The key point there will be that the effects of P&T treatment on disposal are limited. In any case, a repository for high-level radioactive waste and one for medium-level radioactive waste will still be needed. In addition, TRU reduction plays no role in a long-term safety analysis of a geological repository in most host rocks; the decisive are the long-lived mobile fission products. P&T therefore only has an impact on very unlikely events (e.g. human intrusion, volcanic eruption) leading to an unrealistic scenario in the repository. Irrespective of this, however, it is clear that the goal of P&T treatment will be missed from the outset, if no significant reduction in the amount of TRU can be achieved in a transmutation reactor concept or transmutation scenario.

The fresh fuel elements are used in transmutation reactors, where they are irradiated, in order to fission the contained transuranic elements. Fast reactors (SFR, LFR, GFR, MSR) but also ADS are particularly suitable as transmutation reactors. The faster velocity distribution of the neutrons in the reactor core compared to the commercial light water reactors commonly used today is favourable for the fission of transuranic elements. There are also transmutation concepts for thermal reactors (MSR). Light water reactors are rather unsuitable for transmutation, since in a thermal spectrum the cross-section ratio of neutron capture to fission reaction is significantly higher than in a fast spectrum. The smaller the ratio, the more fissions occur per capture reaction (NEA 2006).

The reactor concept itself influences disposal due to the materials and operating resources used, e.g. through neutron activation, surface contamination of the reactor internals, but also through the influence of the available contamination methods and the creation of secondary waste. The technology lines and reactor concepts also differ with regard to the complex chemical processes for fuel production and reprocessing.

Only a few concepts for fast reactors are planned without a fuel cycle featuring reprocessing, such as fast reactor concepts such as the Travelling Wave Reactor (see Chapter 5.2), or the SFR “Sodium” from TerraPower in the USA. The use of uranium fuels (HALEU) with an enrichment of up to 19.75% is planned for these reactors, and direct disposal of the fuel elements is planned as the disposal path. Most concepts for fast reactors at least envisage the use of plutonium in the fuel, and in the long term new plutonium is to be bred in the reactor or TRU waste is to be burned.

Many studies on the transmutation properties of SNR are not based on the individual consideration of the transmutation properties of a single reactor concept, but are integrated into scenarios in which several reactor concepts or technology lines are usually part of the plant fleet. The scenarios differ fundamentally in their considerations of the phase-out of nuclear energy, i.e. the reduction of an existing waste inventory for disposal (phase-out), and concepts of the permanent use of nuclear energy (continuous) while simultaneously minimising waste streams (waste generation). The scenarios of the permanent use of nuclear energy are then further divided into scenarios in which fresh uranium-235 continues to be fed into the fuel cycle and fuel cycles without the addition of fresh fissile material.

In some scenario analyses, the isotope-specific composition of the transuranium quantity over time is not considered. This is particularly useful when parametric sensitivity studies are carried out on the influence of various assumptions on a P&T fuel cycle, for example to investigate the influence of the separation efficiency, the cycle length, the transmutation fraction, the target reduction, etc., without carrying out complex fuel cycle simulations. In a real transmutation system, however, the isotope composition would change significantly over time. In particular, the isotopes that are less fissile in fast reactors would be enriched in relation to the isotopes that are more easily fissile. Whether such a P&T scenario would even be possible without, for example, the supply of fissile materials from outside or a diversified power plant fleet must be demonstrated by more detailed time-dependent burn-up calculations and material flow simulations.

The actual goal of transmutation is to reduce the initially (phase-out) loaded transuranic fraction (TRU) or to reduce the amount of waste (continuously) through fission in the reactor or in a reactor fleet. The difference between the amount of TRU initially loaded into a reactor and the amount of TRU discharged represents the transmuted fraction during reactor use. In relation to the original amount of TRU, one speaks of the *transmutation fraction* or the transmutation efficiency. The higher the transmutation fraction that can be achieved in a transmutation cycle (see Chapter 8.5.1), the fewer steps are necessary to achieve the desired target reduction of an initial inventory in a transmutation scenario. Achieveable values in the range of 10-20% are discussed in this respect.

The transmutation rate, i.e. the transmuted mass per energy generated, is often also specified to compare different systems (kgTRU/kWh, but also gTRU/MWd_{th} or gTRU/MWd_{el}). The effort required to reduce waste generation in scenarios of continuous use of nuclear energy is also minimised by achieving the highest possible transmutation fraction. The waste generation is given as the total mass of high-level waste per energy generated (t/GWy_{el} or t/GWy_{th}), also as the waste generation of transuranic elements (tTRU/GWy). A major disadvantage of uranium-containing fuel is that the transmutation rate is significantly lower, since new TRU are continuously produced from the uranium contained (conversion factor ≠ 0). The *conversion factor* is defined as the number of TRU atoms produced from uranium per TRU atom destroyed. This can only be an advantage in scenarios of long-term nuclear energy use in a resource scarcity situation, when the cost of uranium is higher than the cost of reprocessing to recover fissile material. If an inert fuel (IMF) or thorium is used instead, no TRU (IMF) or very little TRU (thorium) is re-bred (conversion factor = 0). To achieve the same transmutation rate, a smaller amount of TRU must therefore be fissioned.

8.5.1 Plutonium recycling

Plutonium recycling is still carried out on a large industrial scale in France, even though MOX fuel is significantly more expensive than uranium fuel. Spent fuel from LWR is reprocessed and the separated plutonium is used to produce MOX for use in LWR. For the status of plutonium use worldwide, see e.g. (Kuperman 2019). The simplest fuel cycle for SNR would therefore be the one-time use of U-Pu-MOX fuels in a fast reactor.

The theoretical maximum Pu reduction would be achieved if all fissions in a reactor were carried out using Pu isotopes only. One gram of fissile material generates around 1 MWd of thermal energy. In a uranium-free fuel, a fission rate of 1 gPu/MWd thermal energy could therefore theoretically be achieved (conversion factor CR = 0). In ADS with uranium-free TRU fuels, a maximum of 1 gTRU/MWd could theoretically be burned, but in practice the rates are more likely to be around 0.75 gTRU/MWd, similar to liquid metal-cooled fast reactors (SFR, LFR, GFR) (Salvatores and Palmiotti 2011). Depending on the burn-up (MWd/kgHM), a different percentage of the initial inventory of plutonium would then be fissioned in the reactor due to the service life of the fuel element, the energy generated in the reactor and the initial inventory in the reactor. In fuels with a uranium matrix, additional plutonium or TRU would be bred from the uranium and the transmutation fraction (percentage of Pu fissioned in one cycle) would decrease accordingly (CR > 0, typically about CR = 0.5-0.7).

Since plutonium is not only fissioned but also converted into other transuranic elements, transmutation rates, i.e. the mass of transmuted plutonium per energy generated, can also be significantly higher than the fission rate of 1 g/MWd (41.67 kg/TWh). Typically, transmutation rates are given in kg/TWh.

Since LWR serve as a reference, the use of LWR for transmutation will first be briefly discussed. For systems with a thermal neutron spectrum such as LWR with MOX fuel, (Lensa et al. 2007) provides a plutonium transmutation rate of -57 kg/TWh_{el}, with 18.6 kg/TWh_{el} of minor actinides being produced. However, thermal reactors are generally less suitable for transmutation than fast reactors (FR) due to the higher ratio of neutron capture reactions to fission reactions. LWR, on the other hand, are used in many transmutation scenarios with continuous use of nuclear energy to use the plutonium produced in LWR. The plutonium is then separated and, in most scenarios, used in reactors of a different technology line in the form of MOX fuel (e.g. FR, ADS). So far, only mono-recycling of LWR fuels has been carried out commercially, which limits the plutonium reduction.

Alternatively, multi-recycling, as originally envisaged in a plutonium economy or envisaged in P&T scenarios, could be developed industrially. Reprocessing of MOX fuels with relatively low plutonium contents (LWR-MOX fuels) has been researched and developed in Russia, Germany, Japan and France. German MOX fuel elements have also been reprocessed, among others. In 1992, 2.1 tHM with a burn-up of 34 GWd/tHM were initially reprocessed from Grafenrheinfeld after a cool-down of 3.5 years, and later 4.6 tHM from Obrigheim, Neckarwestheim and Unterweser with a burn-up of 33-41 GWd/tHM after a cool-down of 5 years. In the first campaign, a high plutonium dissolution of 99.77% was achieved, with about 0.2% of the plutonium remaining as residue. Post-treatment of the residues was able to extract less of the remaining plutonium than had previously been achieved in the laboratory experiments. In the second campaign, only 0.01% of the plutonium from the fuel remained as residue (IAEA 2003). With multiple recycling, the proportion of fissile plutonium-239 also decreases and may need to be compensated for by higher concentrations or by adding other fissile materials, e.g. by enriching the uranium matrix (uranium-235). Multiple recycling is currently totally unattractive from an economic perspective and has not yet been fully developed on a large industrial scale.

According to (Grambow 2021), multiple recycling does allow plutonium inventories to be stabilised and radioactive waste with a low plutonium content (99.8% separation efficiency) to be produced, but this requires total inventories of plutonium in the fuel cycle that are about a factor of 10 higher than today's inventories. However, the total reduction in radiotoxicity is only a factor of 3 over a period of 100 years. Again, according to (Grambow 2021), this is very little compared to the radiological risks posed by the above-ground inventories during this period.

8.5.1.1 Conclusions on plutonium recycling

The separation and one-time use of plutonium in MOX fuel for LWR is now carried out on a large scale. The amount of plutonium in the fuel can be significantly reduced, but this creates large amounts of minor actinides that remain in the spent fuel. Recycling or multi-recycling of the spent MOX fuel is not currently carried out on a large scale, as handling is challenging due to the heat and activity and is totally unattractive economically. Multi-recycling requires higher inventories of plutonium in the fuel cycle (by a factor of about 10), with corresponding radiological risks, while the impact on disposal is limited.

8.5.2 Recycling of minor actinides

In the previous subsection, transmutation was investigated without the use of minor actinides (MA). The investigation is then expanded to include the P&T treatment of minor actinides. First, the basic use of MA is briefly presented. This is followed by a brief look at MA transmutation in the LWR technology line as a reference in order to discuss the problem of creating higher actinides, especially curium.

If the minor actinides are not treated in a P&T treatment, the effect of P&T on disposal remains very limited. Many reactor concepts and scenarios therefore envisage the simultaneous use of plutonium and minor actinides in the fuel. There are also scenarios in which transmutation reactors are designed primarily to treat TRU or the minor actinides, such as the configurations of the sodium-cooled fast reactor concept ASTRID as a Pu or MA burner (see Chapter 8.5.3). As shown in Chapter 8.2, either homogeneous or heterogeneous partitioning processes can be used with minor actinides. With homogeneous partitioning processes, the MA composition with regard to the fraction of actinide elements in the fuel cannot be varied during fuel production. In addition, liquid metal-cooled fast

reactors in particular, can be specifically optimised, for example, for burning plutonium and minor actinides, but also for breeding new fissile material by dividing the core into several zones (homogeneous vs. heterogeneous core). For example, MOX fuels with plutonium can be used in the centre of the reactor core and MOX fuels with Pu and MA in the periphery. Therefore, more precise estimates of transmutation rates and transmutation proportions or the waste generation depend on the reactor configuration and the entire fuel cycle or transmutation scenario. Typical burn-ups for transmutation are 140 MWd/kgHM.

MOX fuels with minor actinides (MA) could also be used in thermal reactors (LWR). Such MA-MOX-containing fuels can currently only be produced with a limited MA content (see Chapter 8.3). (Salvatores and Palmiotti 2011) indicates possible transmutation rates for a closed fuel cycle with multiple recycling. The fresh fuel contains approximately 10% TRU. The transmutation rates of an MOX fuel enriched with 4.9% uranium-235 are -56 kg/TWh_{el} for plutonium, producing 9.89 kg/TWh_{el} americium, 1.38 kg/TWh_{el} neptunium and 5.87 kg/TWh_{el} curium. For a fuel cycle with MA in the fuel (7.7% Pu and 2.3% MA with 0.3% Np, 0.6% Am, and 1.4% Cm), the enrichment of the uranium matrix would have to be increased to 6.45% uranium-235 due to the lower Pu content and the neutron absorbers under the actinides. The result is -21 kg/TWh_{el} for plutonium, while -1.3 kg/TWh_{el} for americium and -1.6 kg/TWh_{el} for neptunium are consumed and 0.2 kg/TWh_{el} for curium is produced. Therefore, if MA were used then plutonium transmutation would be reduced, however, instead of MA being built up, they are in fact burned. Within a fuel cycle scenario, the MA fraction could thus be stabilised and the TRU consumption would be 0.2 g/MWd.

This brief description of MA use in LWR shows, firstly, the property of many fuel cycle scenarios with multiple recycling that curium, in fact, accumulates in the fuel cycle. A fuel with a high curium content would be very difficult to manufacture and use. Secondly, it shows the relative flexibility of compensating for changes in the MA fraction in the fuel by supplying fresh fissile material (Pu or here uranium-235). Of course, the safety boundary conditions for the use of MA must be guaranteed, especially due to the lower proportion of delayed neutrons at higher MA fractions with the associated effects on reactivity control.

Curium is a neutron emitter (NEA 2006). Even without the use of curium, the deterioration of the plutonium isotope vector due to the accumulation of even-numbered plutonium isotopes in the fuel during multiple recycling leads to neutron emission rates that are up to an order of magnitude higher than the LWR MOX fuels, which have a limit of about $1*10^5$ n/ (s kg). In fuel cycles in which curium accumulates, the neutron emission is about two orders of magnitude higher than in fuel cycles without curium recycling, and additional plutonium-242 is produced by the curium decay. Substantial shielding would be required for the production and transport of fuels with a high curium content. The very high emission rates in fuel cycles in which curium is fully recycled can also lead to problems with criticality during transport and storage. Due to thermal radiation, the heat rates of fuels containing curium are also about a factor of 2 higher than LWR-MOX at 2 W/kg, so that appropriate cooling of the fuel elements must also be provided for handling and transport and active cooling methods may also have to be used. Some fuel cycles also provide for separation and interim storage of curium for around 66 years (NEA 2006). Handling curium is therefore one of the major challenges in P&T treatment. The property of curium build-up will occur again and again in other SNR and fuel cycles below.

In addition, the formation of higher transuranic elements than curium should be mentioned, which arise particularly with frequent, multi-recycling. The californium isotopes, californium-250 and californium-252, have a high spontaneous fission rate and therefore contribute significantly to the

neutron background, thereby increasing the problems in the handling of fuels with multi-recycled MA (NEA 2002).

The effects of MA on the reactivity coefficients vary depending on the reactor size and type of coolant. In general, however, the introduction of Am and Np into the reactor core can have undesirable effects on some reactivity coefficients. The Doppler reactivity coefficient becomes less negative and, in the case of SFR, the void reactivity coefficient becomes more positive. In addition, the fraction of delayed neutrons becomes smaller (NEA 2006). A transmutation of the minor actinides therefore fundamentally worsens the neutron physical reactivity coefficients (Renn 2014). This means that in the case of a homogeneous admixture in all fuel elements used, the fraction of minor actinides should not exceed a few percent (NEA 2012).

This undesirable influence of the minor actinides on reactivity coefficients can be reduced, if they are only used in fuel elements at outer positions of the reactor core, where the neutron flux is kept low, so that their impact on the safety properties is also limited. This also requires that the plutonium content of these fuel elements is greatly reduced. Such a heterogeneous admixture leads to a core configuration similar to that of the “classic” fast breeder reactors with an inner fission zone and an outer radial and/or axial jacket, in whose fuel elements proportions of up to 40% minor actinides should then be possible (Salvatores et al. 2015).

8.5.2.1 Conclusions on minor actinides

For the purpose of P&T treatment for disposal, many concepts envisage the simultaneous use of plutonium and minor actinides in the fuel. There are fuel cycles with reactors that are specifically optimised for burning plutonium and minor actinides, but also for breeding new fissile material (homogeneous vs. heterogeneous core). More precise estimates of transmutation rates therefore depend on the reactor configuration and the entire fuel cycle. Thermal reactors are only suitable for transmutation to a limited extent due to the limited fraction of TRU in the fuel. Plutonium is depleted however, MA are created. The higher the proportion of MA in the fuel, the more fissile material needs to be added to the fuel, for example in the form of uranium-235. Curium typically accumulates in the fuel. The build-up of curium makes it difficult to handle the fuel during partitioning, production and transport due to the heat production and neutron radiation.

The fraction of MA in the fuel is limited to a few percent by the safety properties of the reactor concept, if the MA is mixed homogeneously into the fuel; in heterogeneous cores, the MA fraction in the outer area of the reactor core can be increased to up to 40%. Typical burn-ups for transmutation are 140 MWd/kgHM.

8.5.3 Fast reactors (SFR, LFR, GFR) and ADS

The following compares the transmutation properties, especially the transmutation rates, of the different technology lines of liquid metal and gas-cooled fast reactor systems (FR). In particular, the ASTRID reactor concept and the differences between ADS and FR are discussed.

Regarding SFR, there are recent studies on transmutation rates for the now-aborted French SFR project ASTRID, in which homogeneous MOX fuels with a maximum burn-up of 137 MWd/kgHM were to be used. According to (Gabrielli et al. 2015), in a fast reactor such as the ASTRID design, plutonium transmutation could be -13.2 kg/TWh in a configuration designed for Pu burning, with a homogeneous fraction of 25% plutonium in the fuel. This should correspond to a conversion rate of 0.68, i.e. 32% of the initial plutonium would be transmuted in the reactor at the end of 5 reactor cycles

(see below). In this concept, a small fraction of minor actinides would also be used (MA/Pu ratio 1/20). Overall, the MA transmutation rate would be 0 kg/TWh, i.e. no new MA would be added. However, the isotope vector shifts within the MA. While americium is transmuted at -0.9 kg/TWh, 0.8 kg/TWh of curium (and 0.1 kg/TWh of neptunium) are created simultaneously. If, however, the same reactor concept is calculated in a configuration for burning minor actinides (MA/Pu ratio 1/2), with a proportion of approx. 25% Pu and 11% MA in the fuel, -15.4 kg/TWh MA could be burned (-16.3 kg/TWh Am, -0.3 kg/TWh Np), but 1.2 kg/TWh curium would still be produced. The plutonium reduction in this concept would only be -3.7 kg/TWh.

The calculations for ASTRID also showed that the reactor power had to be reduced from 1600 MWth to 1200 MWth in order to achieve more favourable reactivity coefficients in terms of safety, without having to reduce the proportion of minor actinides. In addition, the internal breeding blanket was removed and the lower axial breeding blanket reduced to 2 cm (Gabrielli et al. 2015). This made it possible to achieve a slightly negative void reactivity coefficient for the fuel composition under consideration, taking into account the sodium plenum above the core, although the values in the lower core area remained strongly positive (Gabrielli et al. 2015). Coupled dynamic neutron physics/thermohydraulic analyses are therefore required in order to be able to conclusively assess the accident behaviour of such a reactor core.

(Gabrielli et al. 2015) give a conversion rate of 0.68 for the reactor loadings of ASTRID, in a Pu burning configuration, and 0.9 for plutonium and 0.55 for MA, in an MA burning configuration. At the same time, (Vezzoni et al. 2015) give a reduction of 700 kg Pu with 4800 kg Pu initial inventory, which would correspond to a reduction of only 15%. In an MA burning configuration, the burning of 230 kg Pu with 4400 kg initial inventory and 720 kg MA, with an initial inventory of 2170 kg, corresponds to only 5% Pu reduction and 33% for MA. A reactor concept such as ASTRID could therefore be used both as a Pu burner and as an MA burner in a transmutation scenario.

Transmutation rates for the future LFR BREST-1200 are given in (Ponomarev et al. 2019). The fuel is U-Pu-MA fuel with depleted uranium, plutonium and 3% MA content in the fuel. The reactor is configured as a breeder so that 51 kg/GWy plutonium is bred with a high 17%-fraction of 238Pu in the plutonium. For the minor actinides, -32.2 kg/GWy neptunium, -7.0 kg/GWy americium and -3.3 kg/GWy curium are consumed in the reactor.

(NEA 2002) compares the different MA transmutation properties of LFR, SFR and GFR and different fuels (metallic fuel, oxide fuel and nitride fuel). A commercial SFR, as well as an LFR similar to BREST-300 and a GFR similar to ETGBR all have a normalised transmutation rate of about 7.5-7.7% per year. With a power-normalised initial inventory for the SFR of 695 kgMA/GWth (LFR 1077 kgMA/GWth, GFR 657 kgMA/GWth), the reactors would transmute -53 kgMA/GWth/y for SFR, -83 kgMA/GWth/y for LFR and -49 kgMA/GWth/y for GFR.

The result of this analysis is that, with regard to the differences between the technology lines, it can be determined (Salvatores and Palmiotti 2011) that the transmutation properties between the technology lines of fast reactors (FR)[229] are very similar. The choice of fuel also has little influence (oxide, nitride, carbide fuel) as long as a uranium matrix is used (Salvatores and Palmiotti 2011). The use of a uranium matrix leads to plutonium and other MA being bred from the uranium and the transmutation efficiency being not optimal.

[229] The MSR also has fast systems, which are dealt with separately below (Chapter 8.5.4.3 and 0).

This limitation can be circumvented by using uranium-free fuels, since no new MA are created in uranium-free fuel, while there is also a shift of the element or isotope vector of the MA used in the fuel. For critical fast reactors configured to burn actinides in uranium-containing MOX fuels, the possible TRU content in the fuel is less than 50% and in the case of homogeneous recycling of the minor actinides, typically less than 10%. Typical burn-ups are expected to be around 140 GWd/tHM in the future, but this is presently not achieved. The possibility of using uranium-free fuels with a TRU content of almost 100% in ADS is therefore a great advantage and allows transmutation to be optimised for the purpose of actinide transmutation (NEA 2002).

Accelerator-driven subcritical systems (ADS) are also best suited to overcome the poor safety properties of fuels with high TRU contents in the fuel. ADS have the advantage that their safety properties do not deteriorate in the same way as critical fast reactors (FR) when MA is used, since ADS are subcritical and it is therefore possible to handle high MA contents in the fuel.

The use of an ADS with uranium-free fuel therefore combines several positive properties of a transmutation system. ADS can be considered to be very good MA burners. As a result, comparatively fewer ADS can achieve the same transmutation rates as critical fast reactors (with the same plant power output) and/or, in a transmutation scenario with long-term nuclear energy use, a parallel small fuel cycle for burning MA can be set up, whereby the majority of the plant fleet for power generation would operate without separated MA (often called a “double strata” strategy, based on two parallel fuel cycles). Possible transmutation rates of ADS are listed in the following Table 8-2 under the “B2 ADS” scenario.

For the transmutation properties of fast MSR systems, see Chapter 0.

8.5.3.1 Conclusions on fast reactors

The transmutation properties of the technology lines of fast reactors (FR) are very similar. Likewise, the fuel selection has little influence on the transmutation properties. More precise estimates of transmutation rates depend on the reactor configuration and the entire fuel cycle. FR can be designed either for Pu production or burning but also for MA burning.

The use of an ADS with uranium-free fuel combines the advantage that no new MA is created with the advantage that ADS are subcritical and the safety properties are not very sensitive to the use of MA. ADS therefore have advantages over FR systems, especially in phase-out transmutation scenarios. In scenarios of continuous nuclear energy use, ADS could form their own small fuel cycle (strata) for the burning of MA and the rest of the fuel cycle would be operated without MA separation (double strata). ADS have similar properties to FR in terms of the achievable TRU reductions and can be configured as Pu or MA burners.

8.5.4 Transmutation scenarios - phase-out

The following discusses various results on the use of SFR, ADS and MSR in transmutation scenarios of a phase-out. The studies discussed come to different results regarding transmutation efficiency.

First, the two studies (Vezzoni et al. 2015) and (ISR 2021) on SFR are presented in a phase-out scenario. Then, the results of the parameter studies from (Oeko-Institut e.V.; ZNF 2015) for an EFIT ADS are presented, as well as the transmutation scenario for ADS according to (ISR 2021). Finally, a phase-out scenario is presented based on the calculations from (ISR 2021) and supplemented with results from (Merk et al. 2014).

At the end, results on the generation of low- and medium-level radioactive waste are presented, which is additionally generated by the operation and disposal of the reprocessing and transmutation plants.

8.5.4.1 SFR

In (Vezzoni et al. 2015), a fuel cycle scenario of an ASTRID deployment is calculated (see Chapter 8.5.3 on the transmutation properties of the ASTRID reactor concept), with an initial inventory of 137 t Pu and 38 t MA with an average implementation time of 150 years for a reactor fleet of ASTRID plants, whereby five reactors would have to be configured as Pu burners and two as burners for MA, which would be available during the period of operation.

The scenario is very similar to the phase-out scenarios presented in (Renn 2014) of a nuclear phase-out with a specific initial inventory. The German TRU inventory quantities were assumed, without further fissile material production. Each reactor is filled with 19 t of fuel in the core, of which around 5 t Pu comes from LWR. Reprocessing of the MOX fuel elements only begins once the LWR supply of Pu (or MA in the case of the MA burner configuration) has been used up, in order to obtain new fissile material. In an operation with 5 load cycles until the core is completely reloaded (5 batch operation), approximately 3.8 t of fuel containing 1 t of plutonium would then be used each year in a Pu burner reactor. The plant utilisation is 100%, the cycle length is 10 years, with a 5-year cool-down time. In the end, the authors arrive at an amount of residual waste of 1.8 to 2.1 t TRU (for 6-7 assumed reactors), with a partitioning efficiency of 99.9% for all TRU, which means a target reduction to 1-2%. Approximately 360-420 t of fission products would be produced, although no breakdown by isotopes relevant for disposal is given. The fleet would have an effective conversion rate of 0.62-0.65. The ASTRID cores are deemed that flexible that they can handle all TRU and MA concentrations in the fuel without any significant loss of safety properties. As no service life was specified, the total number of reactors required during P&T treatment is unclear, but due to the small number of reactors at the start of P&T treatment, this scenario is not optimised for the fastest possible implementation, as only a small part of the transmutation inventory can be used in the reactor. The corresponding implementation time is correspondingly long. The remaining 1-2% TRU and the fission products would still have to be disposed of in a final repository. In addition, for this kind of a transmutation treatment there would be the operational and decommissioning waste from the RP and reactors.

In (ISR 2021), a phase-out scenario is also calculated using the transmutation rates from (Gabrielli et al. 2015). The scenario includes a sufficient number of reactors for the respective TRU inventory at any time and an initial inventory of 128.6 t Pu, which drops to 16.8 t within 305 years. The americium inventory drops from 14.3 t to 6.95 t and neptunium from 6.1 t to 5.4 t, while 6.6 t curium

are produced. The authors do not consider curium decay, so the real curium inventory would be significantly smaller due to curium decay, especially to plutonium. In total, 36 t TRU (target reduction) would remain from the initial 150 t TRU, so that the overall reduction would be only 80%. The authors attribute this to the assumption in their calculation that an element-specific calculation was carried out, and the respective different transmutation rates were taken into account, meaning that fewer transmutation cycles were possible than in (Renn 2014) or (Oeko-Institut e.V.; ZNF 2015). The plant utilisation was assumed to be only 83% for a more realistic estimate, and the plant fleet was dynamically adjusted in each case. The burn-up is 140 MWd/kgHM. This produces 13.5 t of technetium-99, 3.6 t of iodine-129, and 9.1 t of caesium-135, the isotopes essential for disposal. In addition, 2300 m^3 of high-level radioactive waste and 84,400 m^3 of low- and medium-level radioactive waste are generated as secondary waste. In addition, there is also waste from decommissioning of the 23 reactors required with a specified service life of 42 years (158,500 m^3) and from the operation and decommissioning of the reprocessing plants and the fuel element production. In total, the authors estimate a quantity of 316,000 m^3 of low and medium-level radioactive waste that will be generated during the operation of the scenario.

8.5.4.2 ADS

To investigate the transmutation properties of ADS in a phase-out scenario, the results of the parameter studies from (Oeko-Institut e.V.; ZNF 2015) for an EFIT ADS are presented, as well as the transmutation scenario according to (ISR 2021).

The calculation in (Oeko-Institut e.V.; ZNF 2015) is based on information in (Renn 2014) and varies parameters of the P&T scenario in order to make corresponding sensitivities visible. The starting point is a phase-out scenario with an initial inventory of 140 t TRU. In (Renn 2014) other phase-out scenarios are also calculated using fast reactors, but they are not presented here as this has already been explained in detail above with the information in (Gabrielli et al. 2015) and the starting points of the scenarios are similar.

In (Renn 2014), in addition to a national scenario, a second regional scenario is discussed in which a P&T application within a European integrated system is assumed. In this regional strategy, the plutonium from German waste is transferred to the European integrated system as a recyclable material and is not considered further. Accordingly, only the minor actinides remain as substances to be transmuted. Such a strategy initially requires the transfer of plutonium disposal to European partners. This scenario also assumes that around half of the transmutation plants completed across Europe will give priority to burning the German stocks of minor actinides, a total of 20 t. The implementation time that could be achieved in this way would be 35 years (Renn 2014). The treatment of radioactive waste from other countries would thus be delayed accordingly, and the risks associated with a longer implementation time would be shifted abroad. Corresponding cross-border scenarios are not presented further here and reference is made to (Renn 2014).

Several scenarios of ADS deployment are compared in the study (Oeko-Institut e.V.; ZNF 2015). Column S in Table 8-1 details a scenario (standard scenario S) that would result from the use of accelerator-driven systems of the EFIT type with a thermal output of 400 MW and the use of uranium-free fuel (IMF) with a conversion factor of 0. Depending on the fuel matrix used and the composition of the transuranium content, an EFIT plant with a thermal output of 400 MW contains a transuranium inventory of around 4.5 t, which remains in the reactor for around 3 years. Alternatively, an SFR such as the ASTRID reactor concept can be used with 1200 MW thermal output (variant 8, Table 8-1), 5 years of irradiation time and 5 t TRU in the reactor. According to (Renn 2014), the transmutation

rates for EFIT are 45 kg/TWh$_{th}$ and for ASTRID 13 kg/TWh$_{th}$ with a corresponding reduction of the TRU share of -11% and -13% per cycle. For ASTRID, the conversion factor is 0.7 because the fuel contains uranium, so the installed reactor power must be increased by a factor of 3.33 compared to the use of uranium-free fuel. When converting 126 t of transuranic elements (14 t of residual inventory), a total of about 345 GWy of thermal energy is generated, assuming that the fission of 1 g of transuranic elements results in about 1 MWd of energy.

Since the fission of transuranic elements produces a roughly equal mass of fission products, the use of EFIT reactors produces 126 t of fission products from the fission of 90% of the initial transuranic inventory, or 135 t if the remaining transuranic inventory were reduced to 5 t. When using ASTRID reactors, the assumed conversion factor of 0.7 results in a power output that is 3.33 times higher and thus a correspondingly higher fission product output of 420 t or 450 t, respectively.

If a target reduction of waste to 10% of the initial inventory (residual quantity) is assumed for a single transmutation cycle of irradiation, storage, separation and fuel production when using EFIT reactors (remaining inventory 14 t, column S in Table 8-1), this requires approximately 25 cycles (6 years cycle time) with a transmutation fraction of 10%. The number of cycles required increases if the target reduction is 5 t (variation 1 in Table 8-1) and the total implementation time of P&T treatment also increases to 242 years, as does the loss of TRU during partitioning. If a separation efficiency of 99.9% is assumed for all transuranic elements, this results in a residual amount of 0.84% of the initial transuranic element inventory in the waste stream, at the end of the entire P&T treatment, due to the frequent recycling (Oeko-Institut/ZNF 2015). If the cycle time increases from 6 years to 8 years or 10 years, the implementation time also increases from 148 years to 199 or 249 years (variations 4-6 in Table 8-1).

Particularly, the interim storage time required before reprocessing can also assume significantly higher values. Even if the required interim storage time is increased to 8 years and the resulting cycle time is 13 years, the implementation time increases to 325 years (variation 7 in Table 8-1). This shows that the cycle length is a critical factor for the feasibility of a P&T scenario. If the sum of the interim storage and processing times is equal to the irradiation time, 50% of the inventory could be in the reactors at any given time (variations S and 5 in Table 8-1). If, however, the sum of the interim storage and processing times is only half as long as the irradiation time, two thirds of the inventory currently in the reactor could be in the reactor and vice versa (variations 4 and 5 in Table 8-1).

If the transmutation fraction can be increased from 10% according to column S to 15% or 20% (variations 2 and 3 in Table 8-1), the required implementation time is reduced from 148 years to 97 or 72 years respectively. Conversely, a lower achievable transmutation fraction would lead to correspondingly longer implementation times.

Separation factors of 99.9% are aimed for in partitioning. If, on the other hand, only a separation factor of 99.5% can be achieved, the losses during reprocessing would increase by a factor of 5 (variation 9 in Table 8-1). Since the losses from reprocessing would already be greater than 5 t with this separation factor, a target reduction to 5 t would no longer be possible. Only with very high separation factors high target reductions are achievable at all.

The achievable target reduction is limited by the fact that in each cycle step a proportion of the existing TRU is transferred to the waste stream. In practice, at the end of a P&T scenario, a proportion of TRU will still remain in the fuel of the last reactor in operation. In order to achieve the same target reduction with a lower separation factor, the reactor inventory remaining at the end of

the scenario must therefore be reduced accordingly. This also increases the required implementation time (also variation 9 in Table 8-1).

If different reactor operating times of 30 or 60 years are assumed, this would result in a correspondingly larger or smaller number of reactors to be built in total (variations 10 and 11 in Table 8-1).

The operation of the individual plants required for P&T results in operational waste, most of which could probably be categorised as non-heat-generating.

An exception could be the decontamination and rinsing water that also accrues or has accrued during the reprocessing of fuel elements in France and Great Britain. The proportion of this type of waste that is temporarily and permanently stored in Germany is accounted for as heat-generating waste, see e.g.(BfS 2015).

Table 8-1: Transmutation (ADS, SFR) with variation of key parameters and impact on target achievement based on an initial inventory of 140 t TRU

Parameter \ variation	S	1	2	3	4	5	6	7	8	9	10	11
Irradiation time (a)	3				5	3	5	5	5			
Intermediate storage + processing time (a)	3				3	5	5	8	5			
Cycle length (a)	6				8	8	10	13	10			
Reactor inventory (% current TRU inventory)	50				63	37	50	38	50			
Transmutation fraction per cycle (%)	10		15	20								
Conversion factor	1								0.7			
Thermal power at t=0 (GW)	6.4		9.6	12.8	4.8	4.8	3.8	3.0	12.8			
Reactor power (MW)	400								1200			
Number of reactors at t=0	16		24	32	12	12	10	8	11			
Target reduction (t)	14	5										
Implementation time (a)	148	242	97	72	199	199	249	325	249	270		
Separation factor (%)	99.9									99.5		
Total separation losses (t)	1.2	1.3	0.8	0.6						6.4		
Reactor lifetime (a)	40										30	60
Total number of reactors	33	37	37	41	30	30	30	30	34	34	39	25
Total thermal energy produced (TWh)	3,100	3,300							10,200	3,200		

Source: Own calculations (Oeko-Institut e.V.; ZNF 2015)

The study (ISR 2021) also looks at the use of ADS in a phase-out transmutation scenario and calculates the target reduction of the TRU inventory to be achieved. A key difference to (Oeko-Institut e.V.; ZNF 2015) is that the TRU reduction is considered element by element, taking into account the transmutation rates of a specific reactor concept, whereas (Oeko-Institut e.V.; ZNF 2015) assumes the same transmutation rate for all TRU elements.

A scenario is calculated in (ISR 2021) with the use of ADS with transmutation rates from (X.-N. Chen et al. 2011). There is a sufficient number of reactors for the respective TRU inventory at any time, with an initial inventory of 128.6 t Pu, which drops to 127.1 t within 88 years. Therefore, due to the low transmutation rate of -5.71 kg/TWh, hardly any plutonium is burned in ADS despite the high initial inventory of 3055 kg Pu in the core. The americium inventory falls from 14.34 t to 4.47 t at a transmutation rate of -46.8 kg/TWh and the neptunium inventory falls from 6.1 t to 5.85 t at -0.86 kg/TWh, while the curium inventory rises from 0.65 t to 2.64 t at 9.7 kg/TWh. The authors do not consider curium decay, especially to plutonium, so the real inventory would be significantly smaller. The initial inventory of minor actinides in the reactor is 3610 kg MA. Overall, at the end of the scenario, i.e. after 88 years, 140 t TRU would remain of the initial 150 t TRU, so that the overall reduction would be only 7%. The authors attribute this to the design of the reactor for MA burning. For a more realistic assessment, the plant utilisation was assumed to be only 74%. The following quantities of isotopes essential for disposal are produced: 9.4 t technetium-99, 2.3 t iodine-129, and 5.8 t caesium-135.

The short duration of the scenario, despite sufficient supplies of fissile material, is largely due to the special assumption of all scenarios in (ISR 2021) that the scenario calculations assume “that the proportion of the individual elements (plutonium, americium, curium, neptunium) in the fuel produced should remain roughly the same over the course of the respective scenario” (ISR 2021). If a transuranic element such as americium is no longer available in the fuel, transmutation is halted, even though there is still sufficient other fissile material (here Pu) in the inventory of the scenario. The results of (ISR 2021) therefore represent a minimum scenario for target reduction to be achieved.

It should be noted that improvements in ADS burning in the scenarios in (ISR 2021) could probably be achieved by adjusting the MA/Pu ratio through different core configurations in various ADS, through different reactor concepts in the power plant fleet and/or by adding fresh fissile material.

The results of using FR or ADS differ significantly in the study (ISR 2021) with regard to the target reduction to be achieved. (Oeko-Institut e.V.; ZNF 2015) come to the conclusion that when using uranium-free fuels, ADS is better suited in achieving a certain target reduction, since less reactor power is required and no further uranium waste is generated through the use of MOX.

8.5.4.3 MSR

One advantage of MSR with a fast neutron spectrum is, in principle, the flexibility of the liquid reactor concept and that this can in principle achieve a very high reduction in TRU. In addition, the flexibility for different fuels also has advantages for transmutation scenarios in which the TRU content in the fuel and the composition of the TRU change over time, since no fresh fissile material is re-bred. Reactivity could be added by flexible choice of fissile material composition. Another advantage of MSR would be that little or, in the case of reprocessing during reactor operation, no cool-down or intermediate storage time is provided for the pyrochemical reprocessing of the fuel, and the duration of the scenario is thus significantly shortened due to the shortened cycle time.

In (ISR 2021), a phase-out scenario is calculated using MSR systems with transmutation rates taken from (Belonogov et al. 2020). For the implementation time, only 2 reactors of the Russian MSR concept MOSART are assumed with an output of 2.4 GWth and lithium beryllium fluoride salts as coolant and fuel. The initial inventory of 128.6 t Pu is reduced to 42.28 t within 55 years. The transmutation rates from (Ponomarev et al. 2019) are used for the calculations with -291.3 kg/GWy of plutonium with a reactor inventory of 3550 kg Pu in one reactor core. The americium inventory falls from 14.34 t to 12.89 t at a transmutation rate of -4.9 kg/TWh (americium-241 -7.2 kg/TWh; americium-243 2.3 kg/TWh) and neptunium from 6.1 t to 0.54 t at -18.8 kg/TWh, while at 9.5 kg/TWh the curium inventory rises from 0.65 t to 3.46 t, although the authors do not consider curium decay. The real inventory would be significantly smaller due to curium decay, especially to plutonium. The initial inventory of minor actinides in the reactor is 395.1 kg MA (223.5 kg Am, 161 kg Np, 10.6 kg Cm). The reactor is therefore designed for plutonium burning. In total, 59 t TRU would remain after the end of the scenario, out of the initial 150 t TRU, so that the target reduction would only be at 39%. For a realistic assessment, the plant utilisation was assumed to be 82%. The following quantities of isotopes essential for disposal are produced: 11.2 t technetium-99, 2.8 t iodine-129 and 9.4 t caesium-135. The short duration of the scenario, despite sufficient supplies of fissile material, is largely due to the scenario's special assumption that the composition of the fuel was kept constant in the scenario calculations, as was the case with the ADS scenarios of (ISR 2021). If a transuranic element such as neptunium, in this example, is no longer available in the fuel, the transmutation is halted, although there is still sufficient fissile material (here Pu and Am) in the scenario's inventory.

The authors of (ISR 2021) assume a very high volume of secondary waste for reprocessing using pyrochemical processes. In the scenario, 10,204 tHM are reprocessed, producing 7,350 m^3 of high-level radioactive waste and 22,960 m^3 of low- and medium-level radioactive waste.

Another hypothetical phase-out scenario is calculated by (Merk et al. 2014). A molten fluoride salt reactor with thorium, with uranium-238 or without fertile material is considered. No safety considerations are made for the use of an MSR with TRU as fuel. The best transmutation rate is achieved with thorium as the fuel matrix. The MSR is to be operated for 50 years with TRU from LWR fuel with 60 MWd/kgHM burn-up as fissile material. The total transmutation efficiency for TRU is 82% for the thorium fuel, 60% for uranium-238 and 71% for a fuel without fertile material. The problem of the last reactor core at the end of a P&T treatment is also analysed. The last reactor inventory will be mixed with previously bred uranium-233 again in order to achieve sufficient reactivity and transmute further TRU. Overall, this should remove another 90% of the plutonium in the TRU of the last core and achieve a total transmutation efficiency of 97% in 108 years. The short time requirement compared to ADS and FR transmutation scenarios is mainly due to the fact that fuel production is not necessary. However, very high actinide inventories in the fuel are necessary, which have not yet been achieved to date. A completely new salt compound that is suitable for high actinide concentrations would first have to be found. Apart from the many questions regarding the technical feasibility of such a system, according to the calculations of (Merk et al. 2014), such an MSR with thorium as a fertile material would be particularly well suited to transmutation and consitute the best system (time and transmutation efficiency) among the SNR for the mass reduction of TRU in a phase-out scenario.

8.5.4.4 Low and medium-level radioactive waste generation

In all transmutation scenarios, the amounts of low and medium-level radioactive waste also increase. Because ADS are very well suited for use in a transmutation scenario due to their properties, a description of the amount of low and medium-level radioactive waste arising in a phase-out scenario using ADS reactors will be presented here.

For the operational waste, only relatively rough estimates of the expected volumes are possible. The following estimates result from the factors given in the scenario from (Oeko-Institut/ZNF 2015; Renn 2014) and those from (Renn 2014):

- When reprocessing the fuel elements that will be available up to 2022 with a heavy metal mass of 9,865 t, a factor of 2 m³ of operational waste per tonne of heavy metal will result in waste generation of around 19,700 m³.
- The amount of operational waste during fuel element production depends on the number of irradiation processes and the mass of heavy metal used, which differs for IMF and MOX fuels. (Renn 2014) assumes a waste generation of 0-1.2 m³/t heavy metal per cycle. For the standard scenario used in this study with nine P&T cycles and a factor of 1 m³/t heavy metal, 630 m³ of operational waste would be generated when using IMF and 3,150 m³ for MOX fuel.
- When reprocessing the transmutation fuels, depending on the separation technology, 2 m³/t heavy metal (for hydrometallurgical processes) or 2.25-2.9 m³/t heavy metal operational waste (for pyrometallurgical processes) are generated according to (Renn 2014). For IMF, the estimated volume of operational waste would therefore be between 1,260 and 1,575 m³ and for MOX fuels between 6,300 and 7,875 m³.
- For reactor operation, (Renn 2014) applies a factor of 8-15 m^3/TWh_{el}. For the standard scenario used here with a thermal energy of 3,100 TWh, an efficiency of 33% would result in a volume of 8,000-15,000 m³ of operational waste.

The total amount of operational waste with the four waste streams mentioned is 29,590 to 45,725 m³.

After the end of the P&T phase, further non-heat-generating waste will be generated when the plants are decommissioned. The volumes depend largely on the number of plants assumed to accomplish the P&T scenario.

In (Renn 2014), the volume of decommissioning waste is estimated at 36,000 to 49,000 m³. However, these figures are based on the unrealistic assumption that the entire transmutation output is provided by six reactors. If the service life of the reactors is taken into account, as well as the decreasing stock of actinides to be transmuted and an adjusted demand for reactors, a more realistic assumption is 30 reactors. This results in a considerably larger volume of waste from the decommissioning of the transmutation reactors. With a waste generation of 3,000 m³ per reactor, based on experience from decommissioning of the Stade nuclear power plant (Renn 2014), the decommissioning of the 30 transmutation reactors alone will generate a total of 90,000 m³ of decommissioning waste.

If we assume, according to (Renn 2014), a further 30,000 m³ of decommissioning waste for the partitioning plants and 1,000 m³ for fuel element production, the volume of decommissioning waste adds up to a total of 121,000 m³.

In total, if P&T is applied in a phase-out scenario for the German TRU inventory, around 150,000 to 170,000 m³ of low and medium-level radioactive operational and decommissioning waste would have to be disposed of, as well as around 2,000–3,000 m³ of separated uranium. This corresponds to another repository for low and medium-level radioactive waste.

8.5.4.5 Conclusions on phase-out scenarios

In phase-out scenarios with a continuously operated, consistent reactor fleet consisting of SFR burners (Pu burners and MA burners), an initial inventory of TRU could be reduced by a factor of around 80 under ideal conditions (separation efficiency, transmutation share). This would result in around two and a half times the initial inventory of fission products. Assuming limitations on the flexibility of SFR, e.g. in the form of a homogeneous, consistent element or isotope vector in the fuel, could drastically reduce the possible reduction. Reactor concepts that are specifically designed for TRU transmutation should, however, be able to deal with a relatively variable isotope composition of the TRU, although sufficient fresh fissile material would then have to be supplied, e.g. in the form of plutonium or enriched uranium (uranium-235). However, there will also be limitations on the possible admixture of elements (e.g. curium) due to safety properties.

Variation calculations for reducing the waste inventory in phase-out scenarios show that the required implementation time and the target reduction react very sensitively to the required cycle time, the separation efficiency and the transmutation share possible per cycle; this applies to all systems used.

In principle, an advantage of MSR with a fast neutron spectrum is the flexibility of the liquid salt reactor concept, the ability to achieve a very high reduction in actinides and the shortening of the cycle length through the possibility of reprocessing during operation. A typical value for the transuranic reduction of an initial inventory in an MSR is a reduction by a factor of 10 in 50 years. A significant limiting factor for the use of transuranic elements or plutonium is the limited solubility of the actinides in reactor concepts based on fluoride salts.

8.5.5 Transmutation scenarios – continuous use of nuclear energy

8.5.5.1 Liquid metal cooled fast reactors and ADS

The results of three major studies are presented below, which compare different scenarios of continuous use of nuclear energy with regard to their impact on waste generation. The transmutation and fuel cycle scenarios consider different technology lines (LWR, SFR, LFR, GFR, MSR, ADS)[230] and different fuel cycles (U-Pu, U-Pu-MA, U-Th, etc.). Many of the scenarios also envisage the combination of different technology lines and fuel cycles in one scenario. The studies compare the total waste generation per energy generated as a possible comparison criterion with regard to disposal.

First, a brief comparison of 40 different fuel cycle scenarios for the continuous use of nuclear energy in equilibrium is made based on the study from (Stauff et al. 2015). Then, the key results of the study from (NEA 2002) are discussed. Both studies are examples of the often used presentation of the influence of transmutation scenarios or fuel cycle scenarios on waste generation (here, mass per energy generated). Then, the results from the study from (Lensa et al. 2007) are presented in more

[230] Liquid metal cooled and gas cooled reactors are summarised below as FR due to their similarity. MSRs are treated separately, although a number of MSR concepts are considered fast reactors.

detail, as this study also takes into account the necessary conditioning of waste and the containers and provides information on the amount of low and medium-level radioactive waste. The results are also the starting point for long-term safety considerations in different host rocks, which are discussed in Chapter 8.6.3.

The study from (Stauff et al. 2015) is a comprehensive comparison of the disposal properties of 40 different fuel cycles presented by the Department of Energy (DoE) of the United States.[231] For the LWR with uranium fuel as a reference scenario, the authors specify a quantity of 21.92 t/GWy_{el} of high-level radioactive waste. The information provided here is only intended as a rough guide between the large variety of possible fuel cycle scenarios. Some scenarios (scenario 3 with heavy water reactor (HWR) and scenario 7 ADS with natural uranium) produce even more waste. Many scenarios using SNR reduce the high-level radioactive waste according to these calculations, e.g. when using VHTR by a factor of around 2 (scenario 2 VHTR with LEU with 9.22 t/GWy_{el}) or when using SFR by a factor of 4 (scenario 4 with 3.99 t/GWy_{el}). Most scenarios with limited reprocessing reduce the waste generation by a factor of around 5-10 with a few exceptions. For example, using MSR with uranium-233/Th (scenario 10) only achieves a reduction to 10.84 t/GWy_{el} (factor 2), while scenario 16 using Pu from a PWR burned in an ADS achieves a much higher reduction (1.52 t/GWy_{el}, factor 14). All other closed fuel cycle scenarios achieve waste generation between 1.25 t/GWy_{el} and 1.79 t/GWy_{el}, with two exceptions. Scenario 20 of TRU use in HWR (2.61 t/GWy_{el}) and scenario 27 of uranium-233/Th use in SFR with enriched uranium feed for initiation (2.25 t/GWy_{el}), both of which have higher waste generation than the rest of the closed fuel cycle scenarios.

The study (Stauff et al. 2015) comes to the following overarching results: For almost all scenarios examined, waste generation is reduced by a factor of between 2 and a maximum of 37. Scenarios involving the separation and use of minor actinides in the fuel lead to a further reduction in waste generation compared to pure uranium/plutonium scenarios without the use of minor actinides. In the most promising scenario of continuous recycling in fast reactors, however, the use of minor actinides does not lead to any improvements. It also makes little difference in TRU recycling whether all minor actinide elements are recycled individually in different fuels, as a single group in a fuel, or in combination with plutonium-containing fuels. Thorium-based fuel cycles have a low minor actinide content in the irradiated fuel and would not significantly benefit from the separation and transmutation of minor actinides In addition, uranium fuels in fast reactors have a higher transmutation rate than thorium fuels in both thermal and fast spectra.

[231] *Once Through Cycles:* 1. PWR LEU, 2. VHTR LEU, 3. HWR NU, 4. SFR TRU/U, 5. High-conversion VHTR LEU + Th, 6. U-233/Th thermal fusion-fission hybrid (FFH), 7. ADS + NU, 8. U-233/Th in fast-spectrum FFH. *With Limited Recycling*: 9. TRU/U in SFR, 10. MSR + U-233/Th, 11. SFR U-233/Th with LEU, 12. Pu/U from HWR in PWR, 13. Pu/U from PWR in PWR burner, 14. U and Pu Bred in SFR in PWR, 15. Pu/U from PWR in SFR burner, 16. Pu from PWR in ADS burner,17. Pu from PWR in an PWR burner with thorium, 18. U-233/Th from PWR in PWR burner. *Continuous Recycling*: 19. Pu/U HWR, 20. TRU/U HWR, 21. Pu/U PWR, 22. TRU/U PWR, 23. Pu/U SFR, 24. TRU/U in SFR, 25. U-233/Th in PWR with LEU support, 26. U-233/Th in MSR, 27. U-233/Th in SFR with LEU support, 28. U-233/Th in SFR, 29. Pu/U from SFR for PWR, 30. TRU/U from SFR for PWR, 31. Pu/U from PWR in SFR burner, 32. TRU/U from PWR in SFR burner, 33. Pu/U from ADS for PWR, 34. TRU/U from ADS for PWR, 35. Pu/U from PWR in ADS burner with IMF, 36. Pu/U in PWR and MA in ADS burner using IMF, 37. TRU/U from PWR in SFR and produce U-233/Th for advanced PWR, 38. U-233/Th from SFR in PWR, 39. U-233 from PWRs and TRU in ADS, 40. U-233/Th from ADS for PWR.

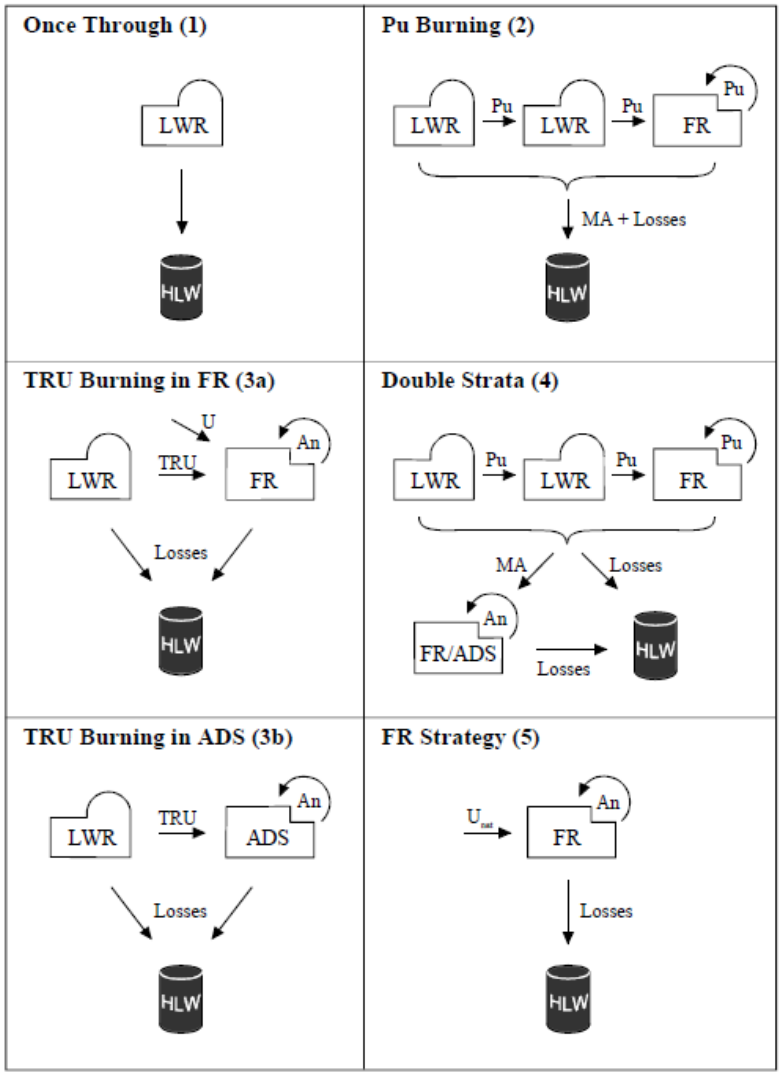
The high reduction factors for waste generation in the scenarios of (Stauff et al. 2015) are based primarily on the separation of uranium from the waste stream, which is then no longer accounted for as high-level radioactive waste. This effect of separating uranium is repeatedly shown in the studies on transmutation scenarios, whereby a nominally very high reduction in waste generation, i.e. in relation to the waste generation in terms of waste mass per energy, can be achieved. However, the effects of the transmutation treatment on the waste generation are distorted by reusing the uranium in multi-recycling and the effect of the actual TRU transmutation on the waste generation if the uranium is excluded from the balance. The reprocessed uranium must also be disposed of as medium-level radioactive waste.

In (NEA 2002), analyses of fuel cycles with ADS and FR were carried out. They compare several combinations of reactor concepts and fuels, whereby these scenarios are also prototypes for other studies. In addition to the reference scenario of LWR use (scenario 1), the following were investigated (see Figure 8-2):

- scenario 2): A fuel cycle for plutonium burning. The plutonium is first extracted and separated in LWR and used in the form of MOX fuel in LWR and FR. Strictly speaking, the scenario is not a transmutation scenario, as no MA treatment takes place. (LWR+UOX, LWR+MOX, FR+MOX)
- scenario 3a): In this, TRU is also separated (TRU burning). The TRU is produced in LWR, separated and then used in MOX in FR (conversion rate 0.5) to burn it. (LWR+U, FR+U+Pu+MA)
- scenario 3b): A scenario without plutonium reuse in fast reactors was also investigated. Instead, the TRU obtained from LWR fuel is used in ADS to transmute the TRU (TRU burning in ADS). The ADS in this scenario make up 25% of the reactor fleet. The plutonium share of TRU in fresh ADS fuel is about 80%. (LWR+U, ADS+U+Pu+MA)
- scenario 4): In this scenario, all systems are used: the use of LWR in which pure uranium fuel is used. Use of plutonium in LWR with MOX (U+Pu) and in fast reactors (FR+U+Pu). The MA are then separated from the MOX fuels and used in Pu+MA fuel for MA burning in a small parallel fuel cycle with ADS (Double Strata). The ADS only make up 3-5% of the reactor fleet. (LWR+U, LWR+U+Pu, FR+U+Pu, ADS+U+Pu+MA)
- scenario 5) Finally, a pure strategy of a closed fuel cycle for all actinides with fast reactors (FR strategy). The only strategy without the use of LWR (FR+U+Pu+MA)

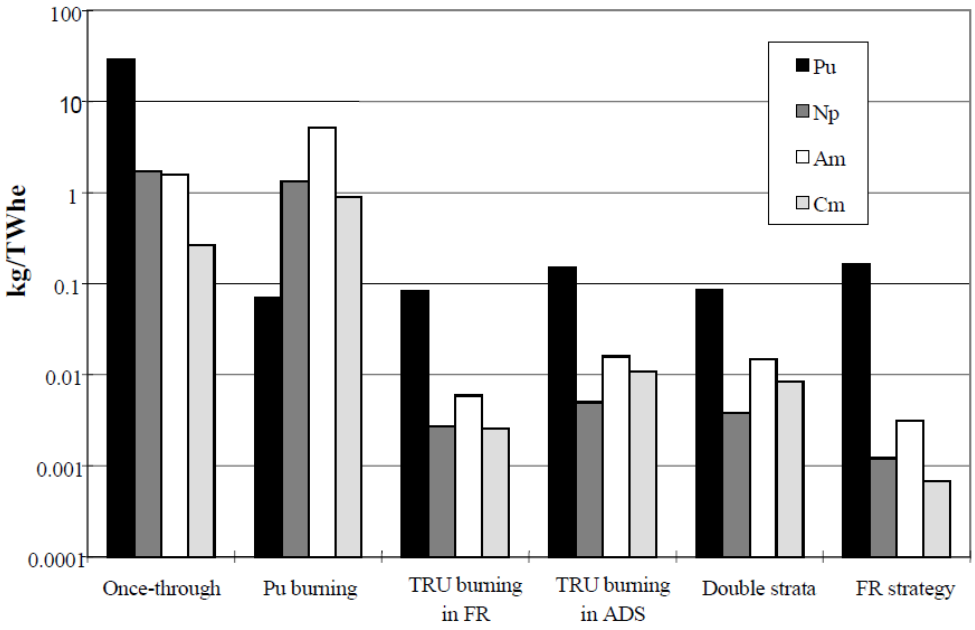
For the reactor and fuel parameters, see (NEA 2002). Except in the pure FR scenario, LWR are used as plutonium suppliers in all scenarios.

Figure 8-2: TRU production of various transmutation scenarios in (NEA 2002)



Source: (NEA 2002) Shown are various idealised transmutation scenarios with FR, ADS and LWR in equilibrium with continuous use of nuclear energy: Plutonium burning (LWR+UOX, LWR+MOX, FR+MOX), double strata (LWR+U, LWR+U+Pu, FR+U+Pu, ADS+U+Pu+MA), TRU in FR burner (LWR+U, FR+U+Pu+MA), TRU in ADS burner (LWR+U, ADS+U+Pu+MA), and a pure strategy with fast reactors (FR+U+Pu+MA).

Figure 8-3: Waste generation from TRU production in various transmutation scenarios in (NEA 2002)



Source: (NEA 2002) Waste generation from the various scenarios in kg/TWh$_{el}$ for the TRU elements plutonium, neptunium, americium and curium on a logarithmic scale.

The plutonium burning scenario (scenario 2) and the double-strata scenario (scenario 4) use MOX fuel in FR, all other scenarios are based on uranium-free fuels in the SNR and a uniform burn-up of 140 GWd/kgHM. All separation efficiencies for TRU were assumed to be 99.9%. These scenarios can therefore be assumed to be idealised scenarios and as maximum scenarios of a P&T strategy with continuous use of nuclear energy given the idealised boundary conditions. In these equilibrium scenarios with idealised boundary conditions, the authors arrive at a waste generation rate for various actinides as shown in

Figure 8-3. The plutonium burning scenario has the lowest waste generation for plutonium, but at the same time a high americium and curium production. In this comparison, the pure FR strategy would be best suited to waste prevention, while at the same time the plutonium isotope vector in this strategy would have a very high proportion of plutonium-239 at many points in the fuel cycle, since plutonium is bred in fast reactors, and would therefore not be proliferation-resistant.

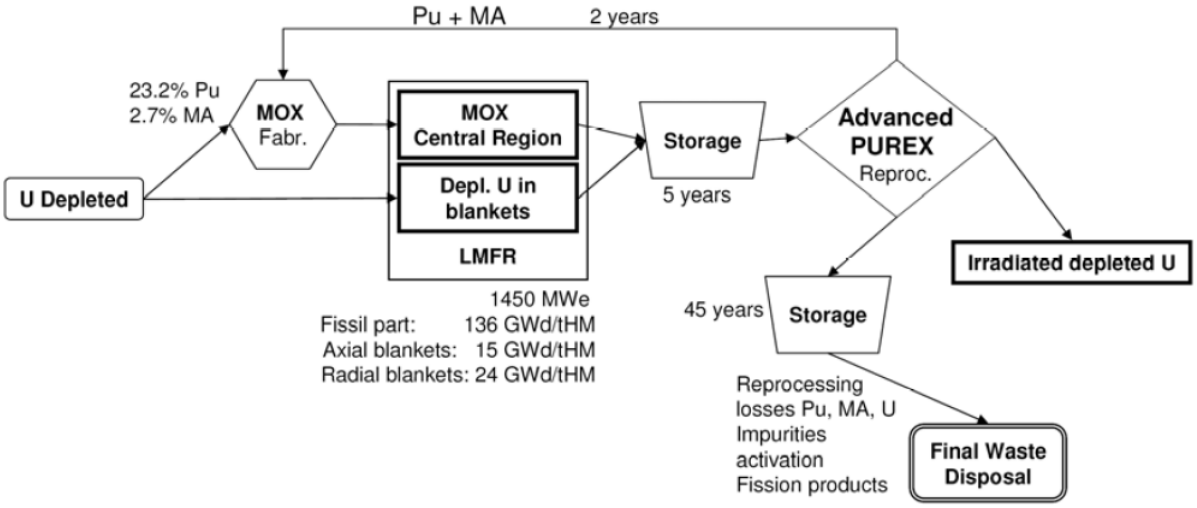
For transmutation scenarios of continuous use in equilibrium, (NEA 2006) also found that the results of fast reactors and ADS hardly differ from fuel cycles with LWR and systems for TRU burning. The choice of coolant also has little influence on the achievable transmutation rates.

The study by (Lensa et al. 2007) is particularly detailed when comparing transmutation scenarios for ADS and FR. This study also calculates and compares several fuel cycles with an LWR with UOX. The fuel cycles are very similar to those described above by (NEA 2006; 2002). The authors also develop variants with, for example, MOX multi-recycling, the use of uranium-free fuels in LWR or FR, the use of thorium or heterogeneous partitioning of the MA (only americium, Am + Cm etc.), but also of short-lived fission products such as Cs and Sr. A variant was also investigated in which an oxide fuel was used for ADS instead of nitride fuel.

In addition to LWR-use as a reference scenario (scenario A1), these are:

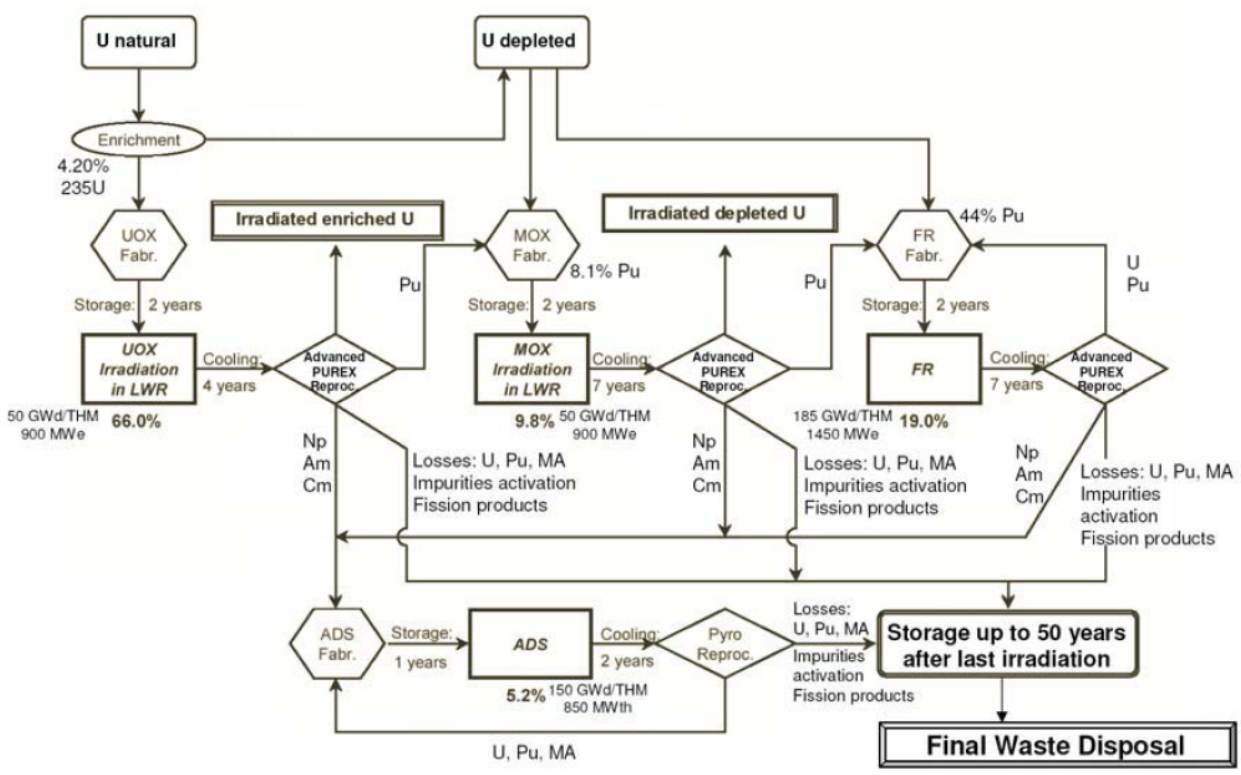
- Scenario A2: the use of 50% LWR with UOX and 50% MOX with 8.5% Pu without multi-recycling (mono-recycling). A number of sub-scenarios concern the multi-recycling of Pu (2x, 3x, infinite), the use of IMF, or homogeneous/heterogeneous thorium fuels and heterogeneous MA recycling. The sub-scenarios are not presented further here.
- Scenario A3: the use of fast reactors with Pu separation. MA are not used.
- Scenario B1: the long-term use of fast reactors (SFR) with TRU. The scenario is very similar to A3, the fuel contains MA. Variants of this scenario serve to examine the separation of strontium and caesium with regard to disposal (see Figure 8-4).
- Scenario B2: a mixed strategy with LWR with uranium fuel and MOX as in scenario A2. In addition, there is another fuel cycle with ADS with uranium-free fuels for burning the remaining plutonium and the MA (double strata).
- Scenario B3: a strategy in which FR is added to B2 (see Figure 8-5).

Figure 8-4: Scenario B1: Use of FR with TRU separation according to (Lensa et al. 2007)



Source: (Lensa et al. 2007) *

Figure 8-5: Scenario B3: Mixed strategy with LWR, FR and ADS according to (Lensa et al. 2007)



Source: (Lensa et al. 2007) *

Table 8-2: Transmutation rates for different scenarios in equilibrium (kg/TWh$_{el}$)

	A1	A2 LWR UOX	A2 LWR MOX	A3 FR U-Pu core	B1 FR U-TRU core	A3-B1* axial blankets	A3-B1* radial blankets	B2 LWR UOX	B2 LWR MOX	B2 ADS
Pu	29.3	29.3	-57.2	-26.5	-22.8	363.7	272.0	29.3	-57.2	-96.5
Am	1.7	1.7	14.7	4.7	-2.5	0.5	0.8	1.7	14.7	-19.0
Np	1.9	1.9	0.5	0.3	-1.8	1.0	0.8	1.9	0.5	-12.5
Cm	0.2	0.2	3.4	0.5	0.0	0.0	0.0	0.2	3.4	-1.3
MA	3.8	3.8	18.6	5.4	-4.4	1.5	1.6	3.8	18.6	-32.8

	Total A1	Total A2	Total A3	Total B1	Total B2
Pu	29.3	20.6	1.0	4.4	1.5
Am	1.7	3.0	4.4	-2.3	-0.3
Np	1.9	1.7	0.3	-1.6	-0.5
Cm	0.2	0.5	0.4	0.0	0.3
MA	3.8	5.2	5.1	-3.9	-0.6

Source: (Lensa et al. 2007) * The values were normalised with the energy production in the blankets, about 4% of the total energy in the reactor. A1: LWR, A2: LWR with MOX, A3: Fast reactors with Pu-MOX, B1: Fast reactors and MOX with TRU, B2: with a combination of ADS and LWR with UO2, Pu- or MOX with TRU.

All scenarios in (Lensa et al. 2007) demonstrate net plutonium production. This is due to the use of LWR and in scenarios A3 and B1 to the configuration of the fast system, which operates in breeding mode. However, Pu production is significantly reduced in scenarios A3, B1, and B2 compared to the reference scenario A1 or the mono-recycling of Pu (scenario A2). Curium is also built up in most scenarios, with high production rates in the fuel cycle, with the corresponding disadvantages due to the additional heat and neutron emissions, especially during fuel handling and transport (see Chapter 8.5.2). The overall americium and neptunium balance is negative. For the fuel cycle variants, see (Lensa et al. 2007). The discussion of the influence of the fuel cycles on long-term safety using a host rock as an example is undertaken in Chapter 8.6.3.

Therefore, the results of the scenarios from (Lensa et al. 2007) differ only slightly from the results from (NEA 2002).

8.5.5.2 High and medium-level radioactive waste

The special feature of the results in (Lensa et al. 2007) is that dedicated waste containers are also assumed for the generated waste. The corresponding volume of the waste streams can be calculated using the capacity of the waste containers for the different waste streams. Finally, the long-term safety of the containers in different host rocks is examined (see Chapter 8.6.3). This results in a chain from the formation of the waste in transmutation scenarios to the dose at the surface at disposal in a geological repository.

Table 8-3: HLW occurrence of the scenarios in (Lensa et al. 2007)

Scenario	A1	A2	A3	B1	B2

Fuel Type	LWR UOX	LWR-MOX	LWR-UOX (reproc.)	FR-MOX (reproc.)	FR-MOX (reproc.)	LWR-UOX (reproc.)	LWR-MOX (reproc.)	ADS TRU-nitride (reproc.)
Waste forms per tHM	2.18	2.21	1.12	2.09	1.98	1.08	1.07	2.61
tHM / TWh	2.46	0.25	2.21	1.15	1.15	1.91	0.24	0.12
Waste forms per TWh	5.35 SFA	0.54 SFA	2.48 UC-V	2.40 UC-V	2.27 UC-V	2.07 UC-V	0.26 UC-V	0.33 UC-V
Waste forms per HLW package	4	1	1	1	1	1	1	1
Waste packages per TWh	1.34	0.54	2.48	2.40	2.27	2.07	0.26	0.33
Package volume (m^3)	2.89	1.51	0.53	0.53	0.53	0.53	0.53	0.53
HLWs volume [m^3/TWh]	3.87	0.82	1.32	1.27	1.21	1.10	0.14	0.17
HLWs TOTAL vol. [m^3/TWh]	3.87	2.14		1.27	1.21	1.41		

Source: (Lensa et al. 2007) A1: LWR, A2: LWR with MOX, A3: Fast reactors with Pu-MOX, B1: Fast reactors and MOX with TRU, B2: with a combination of ADS and LWR with UO2, Pu- or MOX with TRU. “Waste forms” are types of waste, firstly SFA: Spent Fuel Assembly, i.e. spent fuel elements, and secondly UC-V Universal Canister, with vitrified waste from reprocessing.

For the total waste generated in the scenarios, (Lensa et al. 2007) provide the mass flows and resulting volumes for high-level radioactive waste (HLW) (Table 8-3) and for medium-level radioactive long-lived waste (ILW) (Table 8-4).

The volumes of the HLW waste in the scenarios are roughly similar and amount to about 1/3 of the volume of LWR waste, although a large part of the uranium is no longer part of the waste stream of high-level radioactive waste. The reprocessed uranium in fuel cycles A3, B1 and B2 is treated as a recyclable material and is not included in the figures for HLW and ILW. If uranium were defined as waste, this waste stream would also be added as ILW (see below).

More ILW is generated in scenarios A3 and B1 than in scenarios A2 and B2. This is due to the large proportion of structural materials in MOX fuel elements in fast reactors, which is about an order of magnitude higher (3 t/tHM) than in LWR fuel (316 kg/tU). Most of the ILW waste in scenario B2 comes from the reprocessing of LWR UOX fuels. But ADS also generates larger quantities of ILW due to the large proportion of zirconium in zirconium nitride fuel and the accumulation of precious metals during pyroprocessing.

Overall, significantly higher volumes of waste are generated in all scenarios compared to scenario A1 when HLW and ILW are added together. However, the waste streams can in principle be disposed of separately. Scenarios B1 and B2 cause less HLW, but significantly more ILW than would be generated as waste volume in A1. This does not include the quantities of reprocessed uranium, which also have to be disposed of as ILW (see below).

Table 8-4: Waste from scenarios for long-lived intermediate-level radioactive waste

Scenario	A1	A2		A3	B1	B2		
Fuel Type	**LWR UOX**	**LWR-MOX**	**LWR-UOX (reproc.)**	**FR-MOX (reproc.)**	**FR-MOX (reproc.)**	**LWR-UOX (reproc.)**	**LWR-MOX (reproc.)**	**ADS-TRU nitride (reproc.)**
Waste forms per tHM	-	-	1.00	4.10	4.10	1.00	1.01	6.22
tHM / TWh	2.46	0.25	2.21	1.15	1.15	1.91	0.24	0.12
Waste forms per TWh	-	-	2.21	4.71	4.71	1.91	0.24	0.78
Waste forms per long-lived ILW package	-	-	4	4	4	4	4	4
Waste packages per TWh	-	-	0.55	1.18	1.18	0.48	0.06	0.19
Package volume (m^3)	-	-	4.5	4.5	4.5	4.5	4.5	4.5
Long-lived ILWs vol. (m^3/TWh)	-	-	2.49	5.31	5.31	2.15	0.27	0.87
Long-lived ILWs TOTAL vol. (m^3/TWh)	-	2.49		5.31	5.31	3.33		

Source: (Lensa et al. 2007) A1: LWR, A2: LWR with MOX, A3: Fast reactors with Pu-MOX, B1: Fast reactors and MOX with TRU, B2: with a combination of ADS and LWR with UO2, Pu- or MOX with TRU. The waste form here are “universal canisters”, i.e. waste containers with compacted low and medium radioactive waste, e.g. structural materials from the fuel elements.

Long-lived mobile fission and activation products (LLFAP) are particularly relevant for the long-term safety of geological repositories. In Table 8-5 the occurrence of relevant isotopes is considered and compared for the individual scenarios (Lensa et al. 2007):

- In scenarios A3 and B1, due to the higher efficiency (40%) in generating electrical energy in fast reactors, a total of around 10% less LLFAP is generated than in scenarios A2 and B2.
- Compared to LWR, the formation of LLFAP in the HLW is partly lower because these are not immobilised during reprocessing but escape, e.g. iodine-129.
  - Of the volatile isotopes, carbon-14 is mainly created by nitrogen contamination in the fuel elements in scenarios A3 and B1. To minimise the creation of carbon-14, it was assumed in (Lensa et al. 2007) that the fuel was enriched to 97% nitrogen-15. 90% of the carbon-14 is released into the environment during reprocessing and only 10% remains in the HLW waste.
  - The volatile isotope chlorine-36 is created by chlorine contamination in the fuel matrix and is assumed to be only 1% in the HLW stream, 99% is released into the environment, so the values for scenarios A2, A3, B1 and B2 are lower than for A1.
  - Iodine-129, which is also very volatile, is one of the most relevant fission products for long-term safety. In the scenarios with reprocessing, 98% is released and only a small part is in the waste stream. In these scenarios, release into the environment would have to be prevented by suitable

filtration and capture, as well as immobilisation of the fission products, in order to add them back to the waste stream. These would then also play a prominent role in the long-term safety assessment if they are not released into the environment beforehand (see Chapter 8.6.2).

- Strontium-90 and caesium-135 production is lower in A3 and B1. This is due, on the one hand, to the efficiency of FR over LWR, and on the other hand to the fission primarily of plutonium-239, since the fission yield for strontium-90 is lower than for uranium-235.
- The radioisotope niobium-94 is created from the structural materials of the fuel element in stainless steel. Scenarios A3 and B1 therefore have more niobium-94 due to the higher proportion of steel per mass of fuel.
- For ADS fuel, a uranium-free fuel with zirconium as the fuel matrix was assumed, which is also generated as waste in the ILW.

Scenario A2, mono-recycling, only achieves a 37% reduction in TRU waste generated, this is mainly plutonium, with more MA being produced than in the LWR scenario. Scenario A3 also has no MA recycling, but the amount of plutonium waste is reduced significantly, by a factor of 50. However, there are also MA to account for. The total amount of TRU is reduced by more than a factor of 100 in fuel cycles B1 and B2. This has corresponding effects on radiotoxicity (see Chapter 8.6.4), but hardly any impact on long-term safety (see Chapter 8.6.3).

In fuel cycles A3, B1 and B2 it was assumed that the reprocessed uranium would be treated as a valuable material. If the uranium were to be generated as waste, another 2.07 t/TWh$_{el}$ (A2), 0.87 t/TWh$_{el}$ (A3), 0.86 t/TWh$_{el}$ (B1) and 2.0 t/TWh$_{el}$ (B2) of reprocessed uranium would be added as long-lived medium-level radioactive waste.

(Lensa et al. 2007) also point out that the scenarios are equilibrium scenarios and that a fleet of LWR power plants is needed to produce the fissile material for the MOX of the fast reactors.

Table 8-5: Masses of fission and activation products of long-term safety relevance in solid waste (g/TWh$_{el}$) in the scenarios according to (Lensa et al. 2007)

	A1 LWR	A2 LWR with MOX		A3 FR-MOX		B1 FR--MOX		B2 LWR-ADS-TRU-MOX	
	HLW	HLW	ILW	HLW	ILW	HLW	ILW	HLW	ILW
Fission products	125,810	10.6110	195	89,723	189	89,936	189	103,868	191,622*
C-14	0.3	0.03	0.01	0.2	2.3	0.2	2.3	3.1	0.1
Cl-36	6.0	0.3	0.5	0.002	0.0004	0.002	0.0004	0.04	0.6
Se-79	17	17	0.03	16	0.03	16	0.03	17	0.1
Sr-90	571	542	1	249	0.5	250	0.5	513	1.2
Nb-94	18	2.5	15	0	79	0	78.6	0.7	21
Tc-99	2931	2920	6.5	2298	209	2317	208	2969	10.9
Pd-107	837	937	1.5	1207	2.4	1217	2.4	828	296
Sn-126	80	85	0.1	108	0.2	109	0.2	93	0.3
I-129	634	82	5.7	6.7	6.6	6.7	6.7	6.9	9.3
Cs-135	1633	1750	2.9	4270	8.5	4270	8.5	2336	6.5

	A1 LWR	A2 LWR with MOX		A3 FR-MOX		B1 FR--MOX		B2 LWR-ADS-TRU-MOX	
Cs-137	1405	1405	2.5	1137	2.3	1133	2.3	1455	3.8
U	2302048	218730	414	880	176	865	173	2004	535
Np	2021	1890	0.4	547	0.2	2.5	0.5	6.0	1.3
Pu	25430	13688	4.6	485	31	162	32	107	30
Am	4417	4527	0.8	4102	1.8	13	2.6	25	6.9
Cm	52	149	0.01	107	0.02	1.3	0.3	4.9	0.9
TRU	31,920	20,254	5.7	5241	33	179	36	142	39
TOTAL An	2,333,968	238,984	420	6122	209	1044	209	2146	573
Reprocesssed uranium			2,069,080		874,243		861,351		1,996,130

Source: (Lensa et al. 2007) A1: LWR, A2: LWR with MOX, A3: Fast reactors with Pu-MOX, B1: Fast reactors and MOX with TRU, B2: with a combination of ADS and LWR with UO2, Pu- or MOX with TRU. *Zr contained in the ADS fuel matrix.

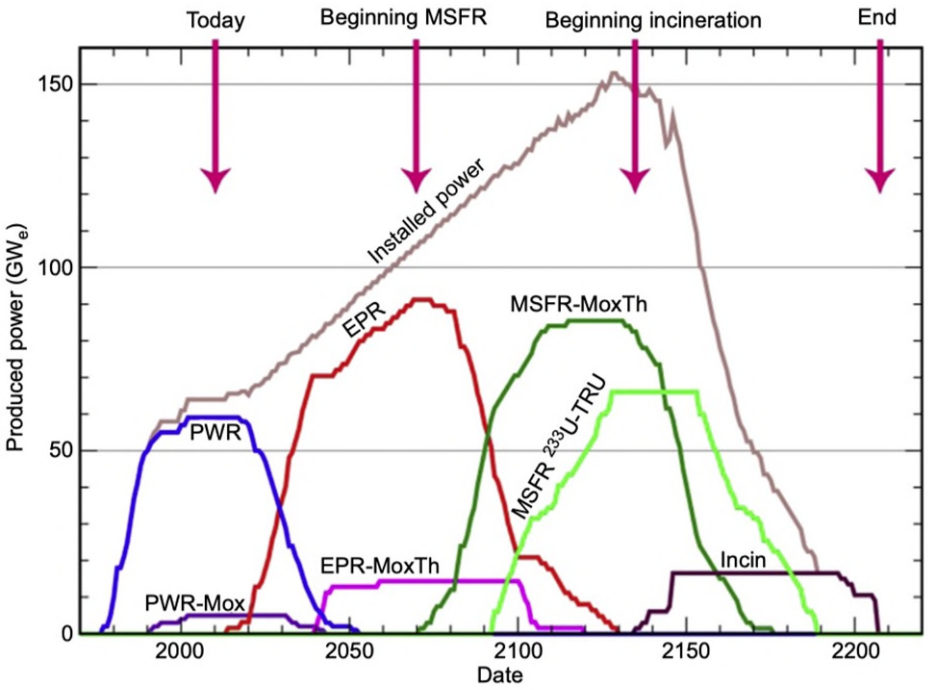
8.5.5.3 MSR

The earlier studies comparing transmutation scenarios (before 2010) did not include MSR, which is why MSR transmutation scenarios for continuous nuclear energy use are treated separately here.

For the MSFR, first calculations on the physics of the reactor (Fiorina et al. 2013) and on scenarios for MSFR use are now available, which are summarised below (Brovchenko et al. 2019; Allibert et al. 2016; Fiorina et al. 2013). Herein, MSFR are integrated in a fuel cycle transition scenario in France, which will be converted after the transition from the current PWR fleet to LWR of the EPR type and will finally be supplemented with MSFR reactors with a U-Th-MOX fuel from 2070 (see Figure 8-6). The EPR fleet is already producing uranium-233 as fissile material for the MSFR fleet for the first reactor loading. Uranium-233 will later be bred in the MSFR. TRU burning will then begin in 2090 in a dedicated fleet of MSFR and will finally be further reduced in a small fleet of burner reactors after the end of MSFR operations. This is a scenario of a mixture of continuous operation and phase-out operation until around the year 2210, with several transitions between reactor generations. The final quantities of the scenario are 2350 t of reprocessed uranium, 5100 t of irradiated thorium, 5 t of Pu in 450 t of irradiated UOX fuel and 0.76 t of Pu in 12.4 t of spent MOX. In addition, there are the quantities of separated MA with 612 t and the inventory from the last burner reactors.

At the beginning, the MSFR must be loaded with existing fissile materials, which determine the reaction behaviour at the start of operation. The long-term behaviour is then determined by the choice of fuel cycle, whether classic uranium-plutonium fissile material (U-Pu), thorium and uranium-233 (Th-U) or the addition of transuranic elements (TRU) or combinations thereof are used. If thorium is started to be used in the fuel cycle, there will initially be no uranium-233, i.e. the reactor would initially be loaded with other fissile material, e.g. enriched uranium or plutonium from PUREX reprocessing. The MSFR would then initially be operated as a Pu burner. It would be particularly relevant to use the MSFR to burn the entire TRU vector produced by a LWR fleet, thereby starting a new Th cycle. One problem here is the maximum concentration of trivalent actinides (mainly Pu, Am, Cm) close to, or above, the solubility limit at the nominal temperature in the carrier salt. It may therefore be necessary to add uranium-235 as a fissile material at the beginning. The MSFR also includes a breeding blanket made of LiF and ThF_4 (Fiorina et al. 2013).

Figure 8-6: Scenario of France's transition to MSFR use



Source: (Allibert et al. 2016) presents the scenario of MSFR use, which is also discussed in (Fiorina et al. 2013; Brovchenko et al. 2019). LWR (here PWR and the French EPR) are used in the course of the scenario. Some of the reactors are operated with uranium-plutonium MOX fuel and from 2040 with thorium-plutonium MOX. MSR, which initially use plutonium as a fissile material and then later with the previously bred uranium-233 and TRU, will only be built from 2070 onwards (MSFR). At the end of the scenario, dedicated MA burners (incinerators) are used to minimise the amount of waste.

The development of an MSFR over time until the equilibrium value of the produced and burned fissile materials and actinides has been reached, is shown in Figure 8-7. Since the reactor would have to be rebuilt several times in the targeted implementation time of 200 years, one can speak of a scenario similar to the phase-out scenarios in Chapter 0. The calculations for the MSFR form the basis and part of the scenarios according to (Brovchenko et al. 2019; Allibert et al. 2016; Fiorina et al. 2013). In addition to a scenario for loading with Th-uranium-233, a hypothetical scenario is shown in Figure 8-7, in which the reactor was initially loaded with a Th-TRU inventory and then new uranium-233 is gradually bred from the thorium as a fissile material.[232] The underlying assumptions are presented in more detail in (Brovchenko et al. 2019). Relevant for the disposal of waste are the required transmutation times. The thorium and uranium concentrations quickly reach an equilibrium state. Plutonium reaches the equilibrium state after about 100 years, americium after 160 years and curium after more than 200 years. Equilibrium is reached when the fission of the actinides is balanced with the formation of the actinides from the thorium. The proportion of curium initially reaches a maximum 26 years after commissioning (Brovchenko et al. 2019). It is assumed that there will be an initial TRU inventory of around 12 t. A reduction of the initial TRU inventory to 10% is achieved for Pu after just 40-45 years, for Am after around 55-60 years, while Cm is only reduced to 10% of the initial inventory after over 130 years. Np is re-bred from thorium.

[232] The reactor would have to be replaced several times during this time. Typical reactor operating times are around 40 years.

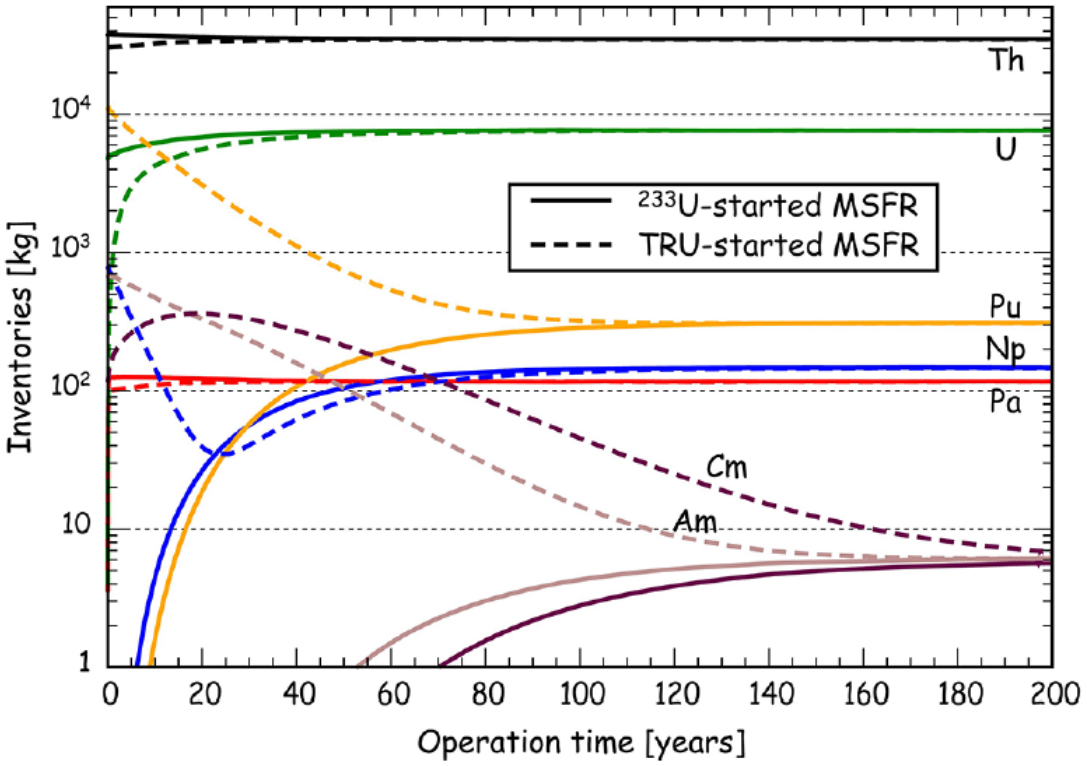
The fuel consists of uranium-233, thorium and 20% TRU from LWR waste, which is slightly less than the solubility of trivalent actinides in the MSFR allows. This means that a reduction of 11.5 t TRU in 50 years is possible with one reactor.

Figure 8-8 shows the development of the total transuranium inventory in the reactor for different simulation environments. After 50 years, a reduction of the transuranium inventory in the molten salt by a factor of 10 has been achieved (Brovchenko et al. 2019).

In the comparison of 40 scenarios in the study (Stauff et al. 2015), which was presented in more detail in Chapter 8.5.5.1, the use of MFR with uranium-233/Th with limited reprocessing (scenario 10) only reduces waste generation to 10.84 t/GWy_{el} (LWR 21.92 t/GWy_{el}), while continuous reprocessing could achieve a reduction in waste generation to 1.25 t/GWy_{el} and thus a similar waste generation to most of the other closed fuel cycles studied, which are between 1.30 t/GWy_{el} and 1.79 t/GWy_{el}, with the MSR having the lowest generation of all scenarios in (Stauff et al. 2015).

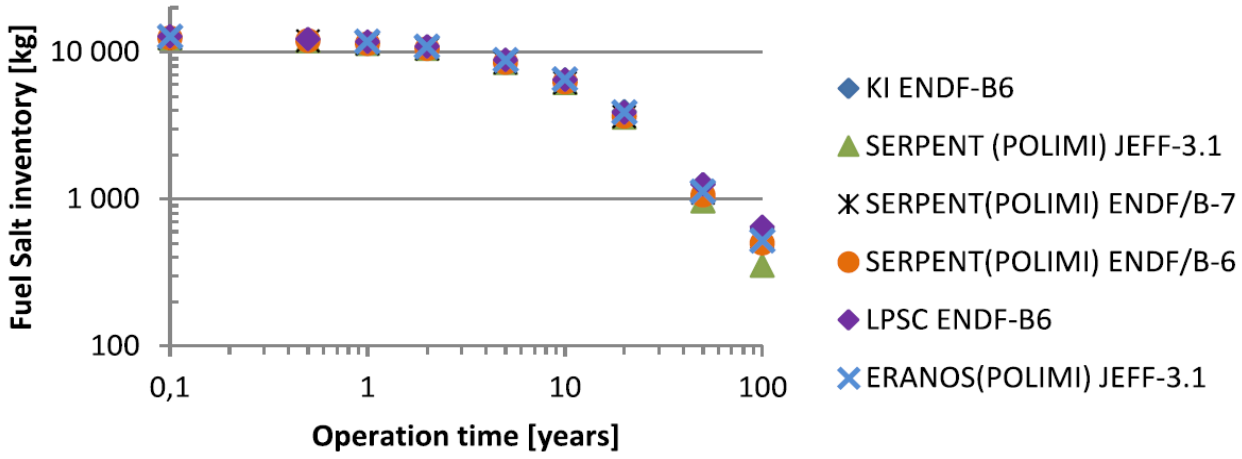
Generic parameter studies with neutron physical transport simulations for MSR are presented in (EPFL 2018). Different fuel cycles for an infinite fuel lattice and the influence of different moderators on reducing the fuel volume and inventory were investigated. The results show that a significant limiting factor for the use of transuranic elements or plutonium is the limited solubility of actinides in reactor concepts based on fluoride salts. Three fluoride-based core designs with thorium-uranium fuel were investigated: two historical graphite-moderated core designs and a modern one with a fast neutron spectrum. In all three reactors, the reactivity was too low to reach criticality due to insufficient fission nuclide inventories. This shows that in fluorine-based concepts, either fissile materials must be added or re-breeding must be carried out.

Figure 8-7: Time-development to equilibrium of the inventory of heavy elements for the MSFR started with uranium-233 (solid lines) and for the MSFR started with TRU (dashed lines).



Source: (Brovchenko et al. 2019). Development of the TRU inventory in an MSFR. The reactor must be rebuilt several times during the 200-year implementation period (scenario). The fuel consists of uranium-233, thorium and 20% TRU from LWR waste. In addition to a scenario for loading the MSFR with Th-uranium-233, a hypothetical scenario is shown in which the reactor is initially loaded with Th-TRU inventory and then new uranium-233 is gradually bred from the thorium as fissile material.

Figure 8-8: Transuranium elements in the molten salt of an MSFR during reactor operation



Source: (Brovchenko et al. 2019). Overall development of the initial TRU inventory of an MSFR.

The use of weapons-grade plutonium (>93% plutonium-239) would have been possible in reactors with a thermal spectrum. The deployment scenario would then have been limited to the disposal of plutonium from decommissioned nuclear weapons. Based on the calculations of fluoride-based concepts, chloride-based core designs were proposed to address the problem of the low solubility of actinides. The limitations of solubility were no longer present in the chloride-based core designs, but MSFR with chloride salt requires considerably larger reactor cores and fuel inventories (EPFL 2018).

8.5.5.4 Conclusions on the continuous use of nuclear energy

When comparing a large number of possible scenarios for continuous nuclear energy use with open fuel cycles and fuel cycles with limited and continuous reprocessing, the following trend emerges (Stauff et al. 2015). In open cycles with SNR, the amount of highly radioactive waste can be reduced by a factor of about 2 (VHTR) to 4 (SFR) compared to the current use of uranium fuels (UOX) in LWR. In scenarios with limited reprocessing with Pu or uranium-233/Th fuels, the amount of waste is reduced by a factor of 5-10. Other scenarios with closed fuel cycles achieve reductions of over a factor of 20 to a maximum of 37. The heterogeneous and homogeneous partitioning of the MA makes little difference in the amount of waste. In the case of thorium fuels, recycling the few MA produced from thorium has little benefit and uranium fuels are better suited for burning MA (Stauff et al. 2015). The reductions are largely due to the waste stream of fertile material (uranium, thorium) removed from the HLW balance in the scenarios.

Overall, in scenarios based on continuous use of nuclear energy, the volume of high-level radioactive waste from reprocessing, as well as spent fuel elements, is similar between the individual scenarios and amounts to about 1/3 of the LWR waste. In contrast, the amount of low- and medium-level radioactive waste is significantly higher. Scenarios with FR in particular have a higher waste inventory than ADS due to the high amount of structural materials. But ADS also produces larger amounts of ILW due to the accumulation of precious metals during pyroprocessing and in zirconium nitride fuels due to the large proportion of zirconium. In addition, waste from reprocessed uranium (or thorium) is also produced.

Systems with increased efficiency due to high operating temperatures have advantages in fission product production in relation to the energy generated.

MSR have a very high amount of secondary waste due to the use of molten salt. According to (Stauff et al. 2015), a typical, maximum value for transuranic reduction is a factor of 18, when using uranium-233/Th fuel in scenarios based on continuous use of nuclear energy. A significant limiting factor for the use of transuranic elements or plutonium is the limited solubility of the actinides in reactor concepts based on fluoride salts.

8.5.6 Fission product transmutation

The following description was adopted and adapted from (Oeko-Institut e.V.; ZNF 2015).

In principle, it is possible to separate and transmute long-lived fission products. The long-lived fission products that are relevant for the long-term safety analysis include selenium-70, zirconium-93, technetium-99, palladium-107, iodine-129, and caesium-135. This is because their high mobility in the geosphere means that they are associated with the release of radioactivity from geological repositories. Reducing the amount of these isotopes is theoretically possible, but requires the development of advanced partitioning methods. However, the corresponding methods are not very

well developed. Overall, the transmutation of fission products is not pursued much today and the state of development is only briefly discussed in this report. For a current article with an overview of relevant literature, see (Wakabayashi et al. 2019; CHIBA et al. 2017).

There are two major problems with the transmutation of fission products. On the one hand, transmutation is not efficient due to the low cross-sections for neutron capture compared to transuranium transmutation (Brenk 2015), this applies to selenium-70 and tin-126, for example. On the other hand, isotope separation would also be necessary for certain elements in order to avoid producing new long-lived isotopes by irradiating stable isotopes and thus further minimising transmutation efficiency (CHIBA et al. 2017).

Furthermore, there are difficulties in producing the corresponding targets. It is still assumed in (NEA 2002) that, in practice, the need for isotope separation and the difficult preparation of targets are problematic for fission product transmutation and that, at the time of writing, the number of candidate nuclides is limited to technetium-99 and possibly iodine-129 (NEA 2002). Only the feasibility of technetium-99 has been proven.

In addition, when transmuting fission products, a proportion of the neutrons in a reactor must be used for this purpose, or dedicated reactors must be used for fission product transmutation. (Salvatores et al. 2015) quote a fraction of 8% to 15% for the required neutron surplus, which the authors say is unrealistic.

Difficulties are also pointed out by (IRSN 2019). The known iodine-129 molecules are thermally and chemically unstable under irradiation and safety concerns exist due to corrosion. In addition, the transmutation rates are low. For technetium-99, the transmutation rates are also low, but so are the separation efficiencies. In view of these difficulties, only a few groups worldwide are pursuing the transmutation of technetium. A transmutation of caesium-135 would also be difficult to implement, as it would have to be separated from the stable caesium-133, which is very complex (caesium-135 forms around 10% of the total caesium in LWR fuel) in order not to re-breed caesium-135. Technetium-99 and iodine-129 would also have to be enriched in order to separate them from other isotopes of the same element that strongly absorb neutrons. With technetium-99, platinum also poses a problem, i.e. technetium is very similar to platinum and separating platinum and technetium is chemically complex. The technetium settles and separation is limited to 80%. In addition, the transmutation rates are low, so that long periods of time, e.g. 15 years in the reactor, are necessary to transmute 50% (ISR 2021). Only very few studies are pursuing the transmutation of caesium-135 (IRSN 2019). In addition, the separation would have to take place in the presence of the highly radioactive caesium-137 isotope. Caesium-135 is also unsuitable for irradiation in targets (ISR 2021). Iodine occurs in various chemical forms as a pure molecule, soluble caesium iodide (CsI), zirconium iodide in solid (ZrI_{4-n}) or volatile form (ZrI_4) (ISR 2021). The separation efficiencies in the PUREX process are 95-98% (ISR 2021).

There is little research into transmutation for the other long-lived fission products that are important for long-term safety. The only way to improve the long-term safety of a repository would be to partition them and then embed them in a particularly stable matrix (CHIBA et al. 2017).

Partitioning is an option for separating strontium-90 and caesium-137, the heat-generating fission products that are relevant in the short term (see Chapter 8.6.2), in order to store the two fission products for the long term, independently from the other waste. Transmutation is not necessary because the natural decay rate would be just as high as the reaction rate in a reactor. A number of

processes are available for separation, such as extraction chromatography for direct separation from the highly radioactive waste stream (TRL 6-8) (NEA 2018).

Many of the fission products, such as those in the German inventory, are also already immobilised in glass. For the disposal option of P&T treatment, it would be necessary to reprocess the vitrified waste canisters in order to extract minor actinides or long-lived fission products and then transmute them. Plutonium and uranium have already been removed. Such a step towards partitioning possible fission products would only make sense, if the P&T processes for the LWR and MOX fuel elements were sufficiently advanced that treating the inventories of minor actinides and long-lived fission products in the moulds would make a difference, in terms of disposal, in case of P&T treatment of all waste and, above all, the fuel elements. To do this, the technical problems relating to the P&T treatment of minor actinides and long-lived fission products would have to be solved.

The vitrified waste was explicitly conditioned for disposal. For this reason, hardly any analyses of further reprocessing of pre-vitrified waste exist, even though the step is chemically possible. Relevant considerations have so far been made, for example, in analyses of the proliferation resistance provided by immobilisation of fissionable materials. The U.S. National Academy of Sciences (NAS 1994), in an investigation into possible ways of disposing of separated military plutonium, came to the conclusion that if the plutonium is immobilised (with or without other highly radioactive waste) in a glass matrix, the technical difficulty of re-extracting it is as high as the reprocessing of LWR fuel elements. Purely chemical separation is no problem for a country like the USA or Russia. However, knowledge of the corresponding chemical process chain is far less widespread than knowledge of reprocessing oxide fuels. However, this is not a fundamental obstacle for other countries with the corresponding expertise in plutonium chemistry to recover the fissile material (NAS 1994).

(ISR 2021) point out that the corresponding processes, according to (Brenk 2015), are complex. According to (ISR 2021), implementation is even difficult in France, particularly on a large industrial scale, and involves high costs and R&D expenditure and, as reported by a study into two possible processes by the French CEA (Commisseriat à l´énergie atomique), the entire project is considered unrealistic (CEA 2008; ISR 2021).

Overall, the transmutation of fission products is not being pursued much these days (Wakabayashi et al. 2019; CHIBA et al. 2017; Salvatores et al. 2015) and there is little evidence that it will be a realistic option in the future (SNL 2019). The number of candidate nuclides is limited to technetium-99 and possibly iodine-129 (NEA 2002).

8.5.6.1 Conclusions on fission product transmutation

Long-lived mobile fission products are important for the long-term safety of a geological repository. The transmutation of actinides therefore has little influence on long-term safety; instead, fission product transmutation, in particular of technetium-99, iodine-129, caesium-135 and carbon-14, would be necessary. Due to the technical difficulties posed by partitioning and target production for irradiation, as well as the isotope separation that is sometimes necessary, fission product transmutation is technically limited to the nuclides technetium-99 and possibly iodine-129. In countries that have undertaken reprocessing, parts of the fission product waste have already been immobilised in glass. Re-partitioning the vitrified waste is considered very challenging. Overall, the transmutation of fission products is rarely pursued today.

8.6 Effects on disposal

The following description was adopted and supplemented from (Oeko-Institut e.V. 2023; Oeko-Institut e.V.; ZNF 2015).

The various effects of P&T on disposal are discussed below. To this end, the effects of P&T treatment on the remaining waste streams are discussed (Chapter 8.6.1). The effects on the volume and heat generation of the waste, which are essential for the design size of a repository, are then discussed in (Chapter 8.6.2). This is followed by the discussion of the effects on the long-term safety of a repository (Chapter 8.6.3). The chapter concludes with a consideration of the radiotoxicity of the waste after P&T treatment. Radiotoxicity comparisons of the ingestion and inhalation toxicity of radioactive waste can be used as a proxy parameter for very unusual scenarios of failure of the repository environment or human intrusion. Despite this limitation, radiotoxicity is used in many P&T studies to compare transmutation scenarios, since actinides contribute significantly to radiotoxicity and actinide reduction by P&T also greatly reduces radiotoxicity. In contrast, actinides play no role in long-term safety in most repository environments due to their immobility.

The end result of this discussion is that P&T has little effect on the requirements for long-term storage of highly radioactive waste and that a repository is indispensable. The most comprehensive American report on P&T by the National Research Council of the National Academy of Sciences (NRC 1996), which is also cited in (NEA 2002), came to the same conclusion in 1996 that it is highly unlikely, if not absolutely impossible, that a repository facility for highly radioactive waste can be dispensed with. (Lensa et al. 2007) also concluded that a geological repository facility is unavoidable, if a P&T strategy is implemented. More recent studies also show the same result that SNR fuel cycles are unlikely to make a contribution to eliminating the long-term problem of nuclear waste disposal (NEA 2011b; Salvatores and Palmiotti 2011; INL 2014b; SNL 2019; NASEM 2023b).

8.6.1 Residual waste streams

In the geological repositories planned to date, transuranic elements (TRU) and long- and short-lived fission products, as well as the uranium from the fuel matrix of LWR and mono-recycled MOX fuels, are to be disposed of together. The requirements for the repository are therefore determined by the properties of these waste groups. The idea behind P&T is to separate and convert the various material groups and thus reduce disposal requirements.

Transuranic elements: For transuranic elements, a large part of the waste inventory is to be converted into fission products by fission. But even if separation technologies were available for all types of waste and corresponding technologies for transmutation were available for all relevant transuranic elements (plutonium, neptunium, americium, curium), residual quantities would still remain in the waste streams:

- Since only a certain percentage of the respective inventory is transmuted per P&T cycle, a residual quantity of transuranic elements always remains in the highly radioactive waste, for which a repository is still required.
- Transuranic elements that remain in the highly radioactive waste also contribute to residual waste, since the efficiency of the separation processes is not 100%. The higher the separation efficiency, the less waste remains. In order to achieve a reduction factor of the TRU by a factor of 100, a separation efficiency of 99.9% is necessary with the burn-ups that can be achieved today (NEA 2002).

Fission products: During partitioning, the fission products that are present in stocks of spent fuel elements are separated, and more are added through the transmutation of the transuranic elements. In scenarios of long-term nuclear energy use, there are few differences between individual fuel cycles in terms of the amount and type of fission products created per energy generated. For scenarios for treating a fixed amount of waste for disposal, the use of transmutation technology creates an additional inventory of fission products, as actinides are split and fission products are created. The fission product waste stream contains relevant proportions of very long-lived mobile fission products, which continue to determine the requirements for disposal, in particular long-term safety. Transmutation of these long-lived fission products by neutron irradiation of targets made from the isotopes is not currently being pursued worldwide and is considered technically difficult (NEA 2002), see also Chapter 8.5.6.

Residual waste streams: The types and compositions of waste that remain at the end of P&T treatment of waste depend heavily on the specifics of the technologies used and the process schemes. In practice, it would be likely that

- Certain types of waste are not treated via transmutation (e.g. in Germany, vitrified waste canisters, special forms such as THTR/AVR waste or other fuels from research reactors),
- After transmutation has finished, residual amounts of transuranic elements remain in spent transmutation fuel elements,
- Residual amounts of transuranic elements or long-lived mobile fission products remain in the waste stream due to too low separation efficiency and
- When the remaining waste is disposed of in a repository, the long-lived mobile fission products continue to dominate the long-term safety assessment.

A combination of disposal options would then be necessary for the highly radioactive short- and long-lived, as well as low- and intermediate-level, radioactive waste remaining after P&T treatment from reprocessing, fuel element fabrication and reactor use.

8.6.1.1 Conclusion on residual waste streams

In no scenario involving the use of alternative fuel cycles with SNR and P&T treatment of waste can a repository for high-level radioactive waste be dispensed with, as residual amounts of transuranic elements, and long-lived fission and activation products, remain in the waste stream. In addition, there are significantly higher amounts of low- and medium-level radioactive waste from the operation and dismantling of the partitioning plants.

8.6.2 Heat and volume

The size of a repository, or the space required for storage, and thus the costs, are largely determined by the total volume of waste and its thermal output. The thermal output requires a necessary minimum distance between the waste containers so that the maximum permissible temperature in the final repository is not exceeded. Waste volume and the thermal output of the waste are interdependent (SNL 2019).

In the first decades, heat production in spent fuel is dominated by fission products. Long-term interim storage of waste would significantly reduce requirements with respect to the heat input of residual

waste. The total heat output of a waste inventory is generated by waste of different ages. In the case of the German inventories available in 2022, after 30 years the fission products will still contribute around 50% to the heat output and after 100 years only 20% (Schwenk-Ferrero 2013). A single fresh typical LWR fuel element would have to be stored for around 60-70 years for fission products to only contribute 50% and after 100 years the fission product share in heat generation would still be around 30% (NRC 1996). After 300 years, this figure is less than 1% and the fission products then no longer have any relevant heat output. The remaining heat of the waste is generated by the actinides.

The isotopes caesium-137 and strontium-90 dominate the majority of heat production from fission products during this period. Pure partitioning of strontium and/or caesium and separate storage, conditioning and repository can reduce the heat input in a repository in claystone and crystalline rock by a factor of 3 to 6, depending on the SNR fuel cycle (Lensa et al. 2007). The required length of the underground storage galleries could also be significantly reduced (Lensa et al. 2007). However, the caesium would still have to be disposed of in a geologically suitable manner, as it also contains the very long-lived caesium-135. The reduction in heat output could also be achieved by simply waiting and long-term interim storage of the waste without P&T treatment. The operation and decommissioning of a partitioning plant results in additional streams of low and medium-level radioactive waste. An improvement in the immobilisation of waste in a repository environment could be achieved by choosing a suitable immobilisation matrix.

(Lensa et al. 2007) discusses the effects on heat production for different fuel cycles. The fuel cycles are presented in detail in Chapter 8.5.5.1. Variations are also calculated for several of the scenarios. The base scenario B1 with a fast reactor with MOX and TRU fuel is investigated to determine the effect of the separation of Sr, or Sr and Cs, on the function of a geological repository.

Two variants of the base scenario B1 were considered by (Lensa et al. 2007) with regard to the impact of the separation of Cs and Sr on the Belgian repository concept for disposal in claystone: either Cs and Sr were separated (B1-a), reducing the content of Sr and Cs in the residual waste to 1% from reprocessing losses, or only Sr was separated from the HLW waste before vitrification (B1-b). The separation of Cs and Sr has a significant impact on the initial heat development of the remaining HLW and thus on the design of the repository mine. The separation of Sr alone leads to a reduction in the required tunnel length by a factor of 1.5 in the base scenario B1.[233] The separation of Cs and Sr means that the thermal output of the vitrified HLW in the base scenario B1 is so low that no cooling times for the residual waste are required. The low heat load is then similar to repository concepts that were developed for the disposal of long-lived medium-level radioactive waste. In such concepts, several waste containers are accommodated in one tunnel section (Lensa et al. 2007).

According to (Lensa et al. 2007), when estimating the required tunnel length for the variant of base scenario B1 with separation of Cs (B1-a), however, it should not be forgotten that the separated Cs must also be conditioned and disposed of in a geological repository, after an assumed decay time of 100 years, since it contains the long-lived isotope caesium-135 (half-life 2.3 million years). The remaining waste can also be stored directly in a repository with a heat load design as for long-lived medium-level radioactive waste. The required tunnel length of the repository, according to (Lensa et al. 2007), is reduced by a factor of 4 compared to scenario B1, without separation of fission products, and by a factor of 13 compared to scenario A1 (current LWR use). In its base scenarios, (Lensa et al. 2007) accounts for a cooling time of 50 years for all spent fuel and HLW.

[233] Compared to the base scenario A1 without reprocessing, this would correspond to a reduction factor of 5.

(Lensa et al. 2007) also points out that the reduction in thermal output may allow for a more compact repository. The distances between the waste have a significant influence on the migration of poorly soluble waste and thus also on the calculated dose from such radionuclides; more compact storage reduces the dose in the long-term safety analysis. The separation of strontium does not make a significant difference to the long-term safety assessment. The separation of strontium and caesium (B1-a), however, reduces the maximum dose by a factor of 9 for the Belgian repository concept in claystone (Lensa et al. 2007).

(Lensa et al. 2007) also conclude that for repositories where the total doses are mainly due to radionuclides with limited solubility (such as the Belgian repository in claystone for HLW from the B1 baseline scenario), the separation of Sr and/or Cs can lead to a reduction in the total doses. However, if the total dose is mainly determined by easily soluble radionuclides (such as the Spanish repository in granite for HLW from the B1 baseline scenario), the separation of Sr and/or Cs has no significant effect on the total dose (Lensa et al. 2007).

Separation of Sr and Cs from LWR waste during reprocessing is also considered in (Nakayama, et al. 2005). Sr and Cs are also separated as part of the separation of transuranic elements (Np, Am, Pu, Cm), uranium and technetium and other metals of the palladium series, which are not discussed further here. 1 Mg SM from LWR fuel burned at 45 GWd/t produces 0.022 m^3 of calcinated Sr-Cs waste. The rest of the waste (excluding U, TRU, Tc) can be vitrified in 0.039 m^3/Mg due to the low thermal output, with a share of 30 wt% of the total weight. Typically, the vitrified unpartitioned waste in Japan contains only 12 wt% at 0.017 m^3/Mg. The vitrified waste with separation of Sr and Cs would therefore have only a third (0.022 m^3/Mg + 0.039 m^3/Mg) of the volume compared to the Japanese vitrified waste from reprocessing with Sr and Cs. (Nakayama, et al. 2005) also provide estimates of the possible secondary process waste that would have to be disposed of and calculate required storage areas in granite rock for different interim storage periods, which are not shown here because they are the same as those of (Lensa et al. 2007). It is also pointed out that, due to the high proportion of long-lived caesium-135, the separated Sr-Cs waste would also have to be disposed of in a geological repository. (Nakayama, et al. 2005) also point out the possibility of long-term interim storage for the separated Sr and Cs waste (Nakayama, et al. 2005).

Since transmutation processes do not treat fission products, the influence of transmutation on heat production is limited to the actinide inventory. In order to fully exploit the lower heat production arising from reduced actinide content through transmutation, the waste would have to be temporarily stored until the short-lived fission products no longer contribute significantly to heat production. The main heat output of the actinides is initially generated by the actinides curium-244 and americium-241(Lensa et al. 2007). The total heat output has dropped to about 10% to 20% of the initial output after 300 years. After 1000 years, heat production is dominated by the actinides americium-241, plutonium-241 and plutonium-239.

In a strategy of dedicated P&T treatment of a waste inventory, fresh fission products would be created through transmutation, which would then dominate heat production and thus the space requirements of a repository for the first 100 years.

A solely volume-based reduction of highly radioactive waste can be achieved by simply separating uranium. However, this increases the thermal output per unit volume of the waste and therefore hardly affects the thermal output and the required distances between the waste containers needed in order to comply with the temperature restrictions of the repository concept. The repercussions on the space requirements for the repository are small. The operation and decommissioning of a

partitioning plant for separating uranium in turn results in additional streams of low and medium-level radioactive waste.

The studies from (Stauff et al. 2015), of up to 40 different scenarios (see Chapter 8.5.5), demonstrate that the main advantages of fuel cycles with reprocessing depend on how the reprocessed uranium and/or thorium must be disposed of. If disposal can be carried out with significantly lower requirements, the total volume of waste for geological repository is reduced by a factor of 10; if uranium/thorium has to be disposed of as well, there is no advantage in reprocessing spent fuel before disposal (Stauff et al. 2015).

(NASEM 2023b; SNL 2019) conclude that alternative fuel cycles, without centuries-long interim storage (aging) of the fission products, have no significant impact on heat management, and that the underground volumes required depend less on the volume of waste generated for disposal than on the heat generated by the waste.

The fuels used in the scenarios[234] from (NEA 2002) (see detailed Chapters 8.5.5.1 and Figure 8-2) feature an unusually high heat rate and neutron source strength, meaning that considerable development work still needs to be done in terms of production, but also operation and achieving high burn-ups and reprocessing. This particularly affects fuels with a high MA content and which can probably only be reprocessed using pyrochemical processes, but they still have 10 to 20 times the amount of heat as normal fuel from fast reactors. This requires new pyrochemical processes with highly corrosive reagents, which still need to be developed, as well as the necessary safety technology. None of these processes have been developed industrially to date, and the corresponding separation efficiencies have yet to be demonstrated.

8.6.2.1 Conclusions on heat and volume

In the first 100-200 years, fission products are particularly relevant for the heat production of highly radioactive waste. Since this determines the required distances between the waste containers and thus the required storage area in a geological repository, the size of a geological repository depends largely on heat production. Separating strontium (and caesium) and storing it separately would significantly reduce the space required, as would storing the waste without partitioning for these periods. This must be weighed against the risks of long-term interim storage of spent fuel elements or partitioning waste and the additional low and medium-level radioactive waste streams generated for partitioning. Removing the actinides would only have a significant impact for periods beyond, with the total heat production from actinides only amounting to 10-20% of the initial values of the total waste after 300 years. Alternative fuel cycles with SNR therefore have only a very limited impact on the space requirements. Since the storage distances and thus the total volume of the repository are determined by the heat input, a volume reduction, e.g. by separating uranium, changes little in the required design.

The reduction in volume and quantity of highly radioactive waste through the separation of uranium is shown in many studies as an advantage of P&T. This sometimes results in very high reduction factors for highly radioactive waste. Every time the uranium is reused as a fuel matrix, the amount of waste per unit of energy generated could then be reduced (halved). However, this is not due to waste being avoided, however, but because significantly more energy is generated without

[234] Plutonium burning (LWR+UOX, LWR+MOX, FR+MOX), double strata (LWR+U, LWR+U+Pu, FR+U+Pu, ADS+U+Pu+MA), TRU in FR burner (LWR+U, FR+U+Pu+MA), TRU in ADS burner (LWR+U, ADS+U+Pu+MA), and a pure strategy with fast reactors (FR+U+Pu+MA).

introducing new uranium into the waste stream. Only in scenarios of continuous use of nuclear energy and permanent recycling of uranium could the waste generated per unit of energy generated be significantly reduced. The reprocessed uranium ultimately results in a separate waste stream as medium-level radioactive waste, which also has to be disposed of in a repository.

8.6.3 Long-term safety

The standard methodology for assessing the radiological risks to humans from underground repositories for spent fuel elements is a long-term safety analysis. This takes into account the conceivable release pathways for radioactive substances from a repository. The long-term safety analysis considers, among other things: geological and man-made barriers, dissolution rates of waste and containers, radionuclide retention in the subsurface, (non-)existence of groundwater pathways into the biosphere, radionuclide mobility in the biosphere and the accumulation of radionuclides in food chains.

Actinides are extremely immobile under liquid-saturated and chemically reducing conditions in the enclosing geological layers. Even when the effects of glaciation on the repository are taken into account and different host rocks such as granite (SKB 2011; Posiva 2012), clay (Marivoet et al. 2008; NAGRA 2002)and salt (Buhmann 1999) are assumed, actinides do not contribute to possible future human exposure in most long-term safety analyses (see, for example, the summary evaluations of various safety analyses in (Swift and Nutt 2012; GRS 2014; SNL 2019)). The only exception is the American Yucca Mountain Tuff with its limited geological barriers: In contrast to the other host rocks, water penetrates into the repository. Only in the case of flood-water penetration into the rock, can the potential dose contributions to humans from plutonium and neptunium be comparable to those from the long-lived mobile fission products (OCRWM 2008).

For example, according to (NEA 2002), (ORNL 1977; Croff and Blomeke 1980) have already come to the conclusion that a P&T strategy for treating highly radioactive waste is not sensible, as the risks are mainly associated with the long-lived fission products, such as technetium-99 and iodine-129, and not with the actinides (see also (NEA 2007a)). Current long-term safety analyses also lead to the conclusion that hypothetical radiation exposure of humans is mainly determined by long-lived fission products. Typical radionuclides, which are mentioned depending on the host rock, are e.g. carbon-14, chlorine-36, selenium-79, zirconium-93, technetium-99, palladium-107, tin-126, iodine-129 and caesium-135 and others (see e.g. (SKB 2011; NAGRA 2002; OCRWM 2008)). In their consideration of the effects of alternative fuel cycles with SNR on different host rocks, (SNL 2019) also come to the conclusion that these have little impact on estimates of long-term repository performance, since mobile fission products dominate the long-term dose estimates in most geological environments.

Such radionuclides are comparatively mobile in the geosphere and biosphere due to their physical and chemical properties and have a half-life of more than a few thousand years.

P&T mainly deals with plutonium and other transuranic elements. Therefore, the transmutation of transuranic elements alone cannot reduce the required isolation periods in the repository. The reduction in long-term risks that can be achieved with P&T treatment of transuranic elements is less than expected ((IAEA 1997) cited in (NEA 2002)). In order to make an effective contribution to the long-term safety of a repository, P&T processes would therefore have to take mobile, long-lived fission products into account. The transmutation of fission products is, however, even more difficult

to achieve technically (see Chapter 8.5.6). Some long-lived fission products that cannot be treated currently by any form of P&T, contribute significantly to the long-term risks.

In addition, the expected doses from the fission and activation products in a carefully selected repository are, according to calculations, well below the permissible limits. (GRS 2014) therefore assume that there will be no positive contribution to the long-term expected dose from the fluid-borne transport of radionuclides through P&T applications of actinides.

In contrast to fluid-borne release, a positive effect from the application of P&T is conceivable for release via the gas path, according to (GRS 2014). This primarily affects the radioactive carbon-14, which is mainly formed in the structural material or occurs in some SNR reactor concepts as a moderator and structural material, and possibly to a lesser extent other nuclides such as iodine-129 and selenium-70. These can occur in gaseous form and, under certain circumstances, be released prematurely from a repository as a so-called “instant release fraction”. For detailed information on the instant release fraction during repository in rock salt, see (GRS 2013). This effect occurs primarily with spent fuel elements and could therefore be reduced by prior conditioning processes, e.g. as part of P&T. Such an effect could in principle also be achieved by integrating corresponding conditioning steps as part of other disposal strategies, but this has not yet been provided for in these concepts.

With regard to P&T, however, it should be noted that carbon-14 and iodine-129 are the highly volatile nuclides that are largely released when the fuel rods are dismantled during reprocessing of the fuel elements. With today's common technologies, the substances are also released into the environment, which should more than compensate for a potential dose reduction in the instant release fraction through P&T. In order to actually reduce the release of volatile nuclides, they would have to be retained in filters with a high degree of efficiency when partitioning the fuel elements, and then fixed in a (as yet non-existent) matrix, in such a way that a significant delay in release during repository could be achieved.

With regard to a mine repository, various publications consider scenarios of unintentional human intrusion into a repository facility and the possible doses for an employee on site. (Marivoet et al. 2008) compares potential doses for an employee who inadvertently drills into a repository area when taking a core sample. In the disposal concept with P&T, for example, the dose for the employee 10,000 years after storage would be almost two orders of magnitude lower than the dose for direct disposal of spent fuel elements. There would therefore be a positive effect for the employees on site.[235] With regard to the risk to the population, for example, (GRS 2014) assume that the danger would be noticed and the borehole would be closed, as required. There would be no risk to the population in this case. If drilling into a repository were not noticed, and the borehole not closed, there could also be consequences for the population. What effects could arise in such a case, and how an inventory modified by P&T could affect this, would also have to be considered in detailed analyses of consequences. Relevant influencing factors could be, for example, changes in the chemical environment and altered dispersion paths that lead to changed release rates of radionuclides (see also (Renn 2014)).

Overall, it can be seen that in scenarios with a fixed initial inventory of radioactive waste, the total inventories of the radionuclides that determine the dose in transport calculations for long-term safety

[235] However, this must be compared with the dose exposure of employees in the nuclear facilities (reprocessing plants and reactors) that are required for P&T.

are not reduced by P&T, but are increased by the fission products created during transmutation, see for example (GRS 2014, p. 45).

The studies in (Lensa et al. 2007) provide an example of the influence of different fuel cycles on long-term safety. (Lensa et al. 2007) discusses the effects on long-term safety for different fuel cycles, as introduced in Chapter 8.5.5.1, for different host rocks.

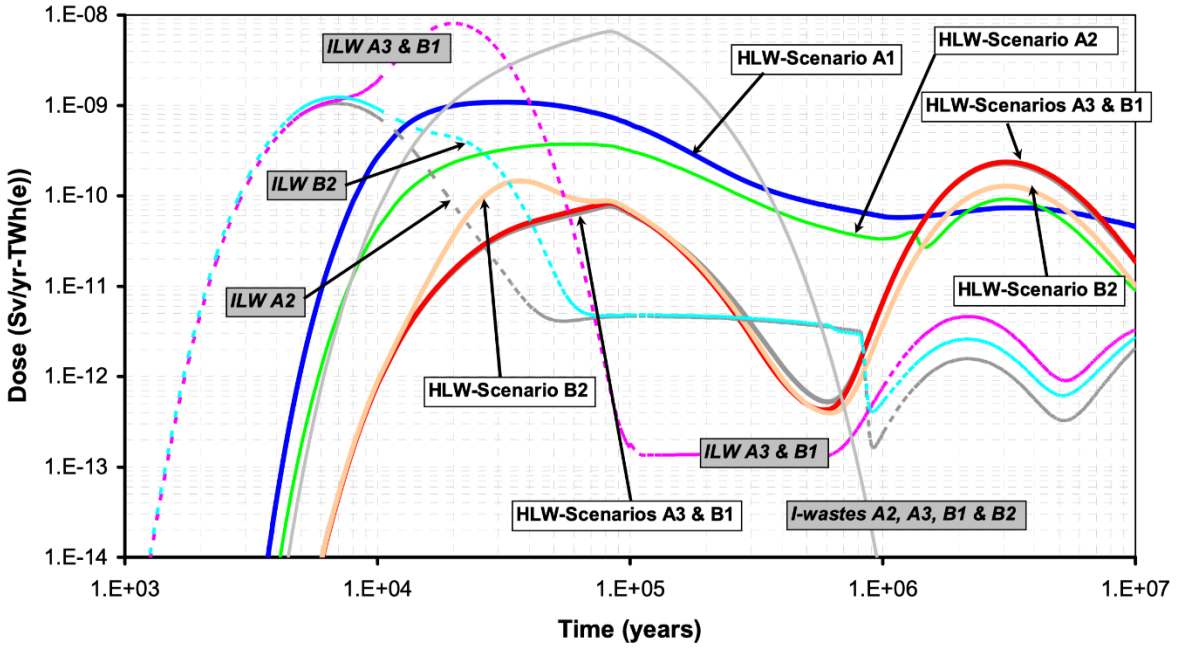
As an example, the study by (Lensa et al. 2007) of the Spanish repository concept in granite rock is presented here for comparison of the fuel cycle scenarios considered on long-term safety (see Figure 8-8). The dose rates per year and per TWh of energy produced are shown for the baseline scenarios for both high-level radioactive waste and medium-level radioactive waste, as well as iodine waste generated during reprocessing. The corresponding inventories are given in Table 8-3, Table 8-4 and Table 8-5 in Chapter 8.5.5.1.

These calculations assume that 98% of the iodine-129 in the spent fuel is immobilised in a vitrified matrix during reprocessing (Lensa et al. 2007). In the reprocessing practised to date, a significant proportion of the iodine-129 is instead released when the fuel dissolves and is then released into the environment (Renn 2014). If long-lived, medium-level radioactive and immobilised iodine waste from reprocessing is transported to the repository and accounted for in the model, the maximum annual doses of all scenarios with reprocessing (A2, A3, B1, B2) are quite similar to each other, for ILW similar to scenario A1, the storage of the LWR fuels without P&T treatment, and, if iodine waste is taken into account, at least six times higher than the peak dose of scenario A1. To reduce the dose from iodine waste, much better immobilisation properties of the glass matrix would be required with confinement times of about 1 million years, significantly longer than the 72,000 years used in the calculations. The authors conclude that for all scenarios that do not only take into account the HLW waste, the long-term radiological consequences for scenarios with reprocessing are equal to or greater than scenario A1 (Lensa et al. 2007).

8.6.3.1 Conclusions on long-term safety

The standard methodology for assessing the radiological risks to humans from underground repositories for spent fuel elements is a long-term safety analysis. This takes into account the release pathways of the nuclides. Actinides are extremely immobile in most repository rocks, except tuff rock, and are not released. A reduction of transuranic elements, therefore, makes no contribution to the long-term safety analysis of a repository. Instead, the main contribution to dose estimates in long-term safety analysis comes from long-lived, mobile fission products such as iodine-129, technetium-99 and activation products such as carbon-14. In transmutation scenarios with a fixed initial inventory of transuranic elements (phase-out), the amount of dose-determining radionuclides would increase due to the resulting fission products. But even in scenarios of longer-term use of nuclear energy, fuel cycles with reprocessing have no advantage in long-term safety considerations over the current use of LWR if all waste streams, including low- and medium-level radioactive waste, are taken into account.

Figure 8-9: Dose for HLW, long-lived medium-level radioactive waste and iodine waste for the five fuel cycles considered for the Spanish repository concept in granite in (Lensa et al. 2007)



Source: (Lensa et al. 2007)

The fuel cycles are the use of 50% LWR with UOX (A1) and 50% MOX with 8.5% Pu without multi-recycling (A2), the use of fast reactors with Pu separation (A3), the use of fast reactors with TRU (B1) and a mixed strategy with LWR with Pu-MOX and ADS with uranium-free fuels with TRU (double strata, B2). The dose rates per year and per TWh of energy produced are shown for the baseline scenarios for both high-level radioactive waste (HLW) and medium-level radioactive waste (ILW), as well as iodine waste generated during reprocessing which, in this case, is not released at the RP but is also stored.

8.6.4 Radiotoxicity

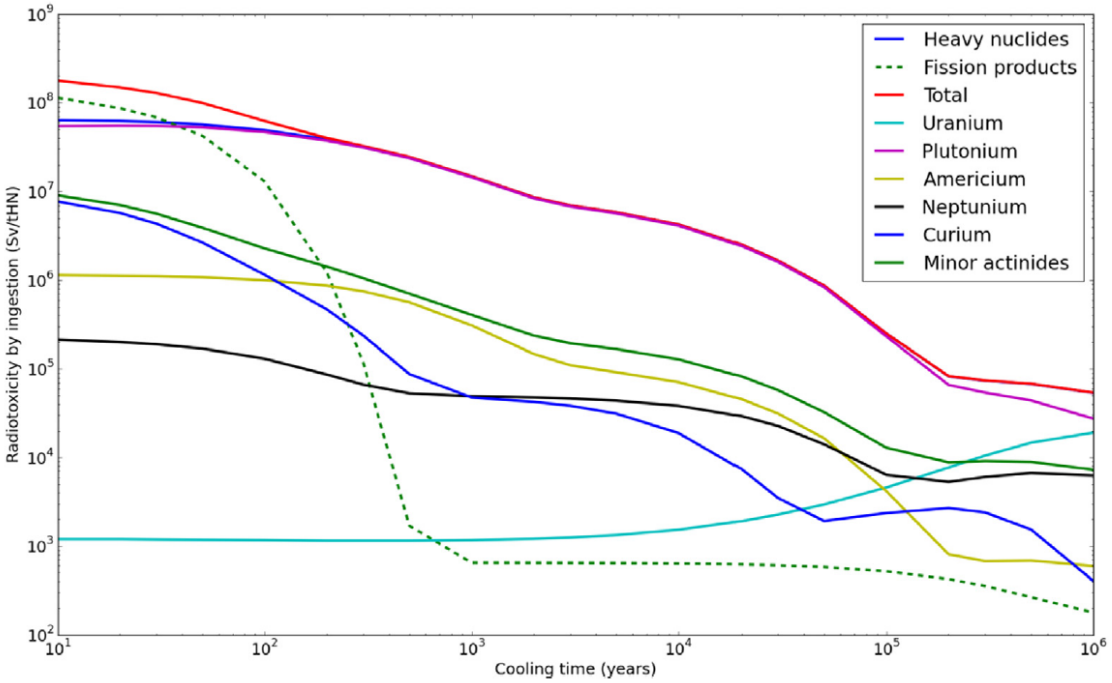
In order to contribute to reducing the long-term radiological risk, P&T must take into account those radionuclides that have the greatest radiological significance for geological repository (GRS 2014; Renn 2014). In the P&T literature in particular, the radiotoxicity index often plays a role as a measure for identifying these isotopes. First, it is shown that radiotoxicity is a criterion for assessing the safety of disposal in a mine repository that is only suitable for a few scenarios of the development of a repository. Since the radiotoxicity index is frequently used in studies on P&T, despite its limited significance, some comparisons and statements on P&T scenarios are nevertheless presented. For example, regarding the effects of separation efficiency on transmutation scenarios. But the effects of the transmutation scenarios from Chapter 8.5.5.1 of (NEA 2002) and (Lensa et al. 2007) are also presented to complete the overall picture. Results for MSR from Chapter 0 regarding the effects of thorium fuels on radiotoxicity are also added, as this is often misrepresented.

Radiotoxicity is a measure of the radiological effects of radioactive substances, which are damaging to health when they enter the human body directly, e.g. via eating, drinking or breathing. The resulting hypothetical radiation exposure is often compared with a hypothetical radiation exposure from a selected amount of uranium ore.

The reduction of transuranic elements in high-level radioactive waste through P&T significantly reduces the radiotoxicity, as defined in this way, because actinides have a high radiotoxicity that only gradually decreases over very long periods of time. A complete transmutation of all transuranic elements could theoretically reduce the radiotoxicity of the residual radioactive waste, i.e. the fission products, to such an extent that the further radioactive decay of the fission products would reduce radiotoxicity through consumption, after a few hundred years, to below the level of natural uranium or medium-level radioactive waste (not shown) (Figure 8-10 and Figure 8-11). If, when enriching uranium ore for LWR to 4.2%, it is assumed that an initial quantity of 7.83 t of natural uranium is needed to produce one tonne of uranium fuel, the radiotoxicity of the initial quantity of natural uranium per tonne of spent fuel would be $2.54*10^5$ Sv/tHM (Magill et al. 2003). However, the comparison with natural uranium can only serve as a rough guide, since pure natural uranium would also have to be disposed of as radioactive waste. (Schmidt et al. 2013) suggests the clearance limit of 0.01 mSv as a comparative standard for evaluating P&T treatment.

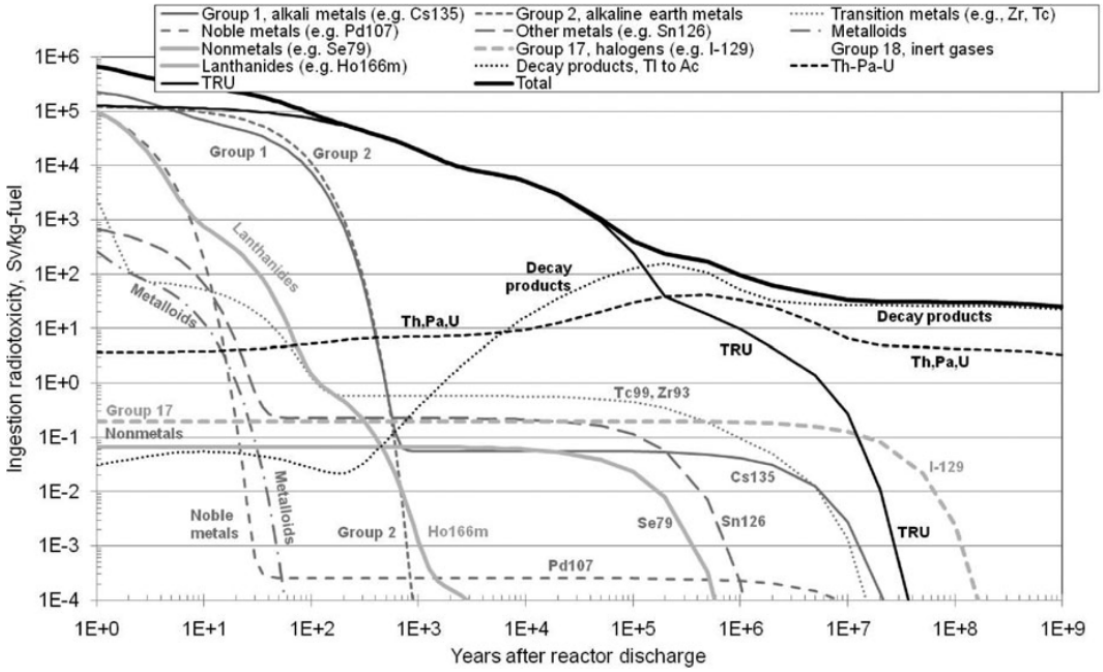
The calculation of radiotoxicity also significantly depends on which reference values are included in the calculation. In Figure 8-10, only the initial amount of actinides is taken as the reference value, without taking the decay products of the actinides into account. Of the actinides, plutonium has the greatest contribution to radiotoxicity and a plutonium transmutation would therefore result in the greatest contribution (Kooyman et al. 2018). Figure 8-11 shows the individual contributions of the elements to radiotoxicity more completely, including the decay products of the nuclides present in the fuel, although the actinide contributions are not resolved element-for-element (Piet 2013). In addition to the actinides, it is mainly the fission products selenium-79, zirconium-93, niobium-93m, technetium-99, palladium-107, tin-126, iodine-129, and caesium-135 that make the greatest contributions to radiotoxicity. Technetium-99 and iodine-129, in particular, have the highest radiotoxicity, even in the long term (Piet 2013).

Figure 8-10 Radiotoxicity by incorporation from ICRP 119 (ICRP, 2012) for the MA vector of an irradiated UOX fuel at 33 MWd/tHMHM



Source: (Kooyman et al. 2018) according to (Bussac and Reuss 1985). The ingestion radiotoxicity is shown without taking decay products into account.

Figure 8-11: Radiotoxicity during ingestion of spent fuel, once-through LWR UOX at 51 MWd/kgHM; no activation products



Source: (Piet 2013). Diagram of the different contributions of the individual element groups to radiotoxicity. Decay products are also taken into account.

This use of the radiotoxicity of stored waste for the radiological characterisation of a geological repository, ignores all processes that take place between the geological location of the waste and the exposure of humans (here through consumption). The isolation potential of the host rock and the behaviour of the radionuclides in the geological subsurface, and in the biosphere, up to the point that they are absorbed into the human body, are not taken into account in radiotoxicity. Radiotoxicity therefore does not correlate with criteria that are crucial for the long-term safety of a repository.

The estimation of radiation doses for humans based on radiotoxicity assumes extremely unfavourable conditions for the food pathway, or incorporation, and ignores a realistic development of the repository. The associated scenarios are extremely unlikely, e.g. direct incorporation of the waste when accidentally entering the repository after its closure or when released from the repository after geological events such as volcanism, earthquakes, etc. If a realistic development of the repository is assumed, the mobility of the elements in the geosphere must be taken into account in order to identify possible advantages of P&T for geological disposal, compared to the disposal of untreated high-level radioactive waste. This substantially modifies the simplistic picture of radiotoxicity.

Regarding the removal of minor actinides from high-level radioactive waste, (Lensa et al. 2007) conclude that the reduction of radiotoxicity has almost no effect on long-term radiological consequences, since minor actinides are almost insoluble and migrate only extremely slowly.

The result that radiotoxicity, as an indicator of P&T treatment of radioactive waste for disposal in geological repository, only reflects very unlikely events has already been established frequently, e.g. in (NEA Proc. 1999; Lensa et al. 2007; Salvatores and Palmiotti 2011; Schmidt et al. 2013; Oeko-Institut/ZNF 2015; ESK 2015; IRSN 2019; SNL 2019; ISR 2021; NASEM 2023b; Oeko-Institut e.V. 2023; Merk et al. 2023) and many others. Alternative fuel cycles with SNR, which use the radiotoxicity index to optimise the disposal of radioactive waste, and thus aim to reduce transuranic elements, therefore, have a completely different optimisation goal (Merk et al. 2023) than in the design of geological repositories that optimises for long-term safety, namely the long-term safe containment of radionuclides through minimisation radionuclide mobility and thus the dose for humans. In fact, these kinds of radiotoxicity comparisons not only prevent any statement to be made about the long-term safety relevance of the radionuclides during repository, they also tend to identify the wrong radionuclides as safety-relevant and therefore P&T-relevant.

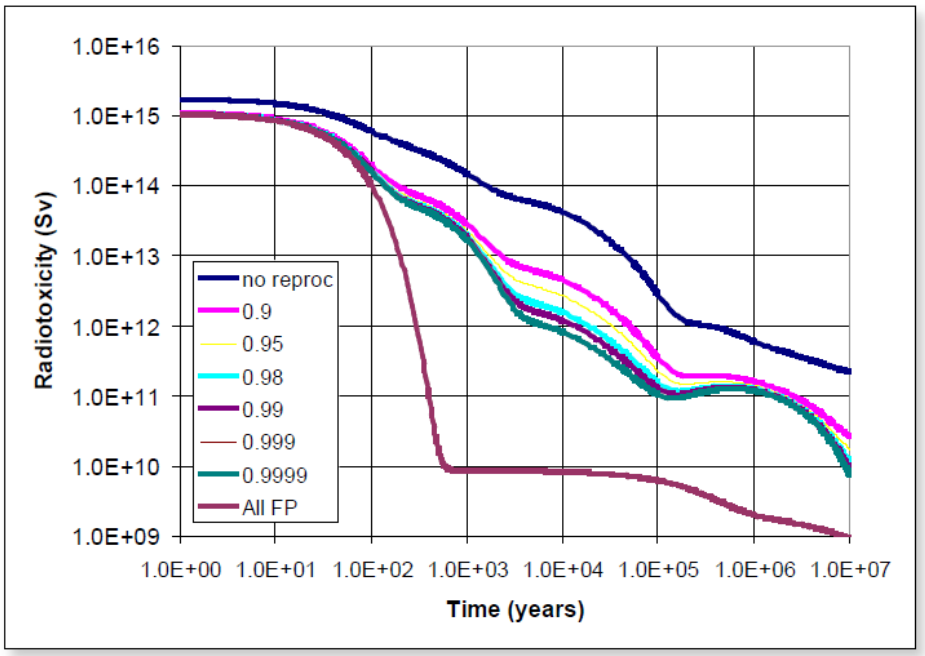
The benefit of radiotoxicity reduction for unlikely events in a geological repository, must be weighed against the risks posed by the radiotoxicity of the radionuclide inventory while it is not yet in a repository. Since the inventory is located at various points in the fuel cycle, it is exposed to the risks of release from accidents and disasters during the P&T treatment period.

Yet, the radiotoxicity index continues to play a major role in the existing literature on the use of P&T for radioactive waste disposal. As has been extensively demonstrated with regard to the safety of a geological repository, the results of this literature have only limited significance for a repository with regard to very unlikely events. Nevertheless, some of the results of the scenarios introduced in Chapter 8.5.2 are discussed here with regard to radiotoxicity, as they allow a comparison of P&T scenarios with one another.

The effectiveness of radiotoxicity reduction depends largely on the separation efficiency of the partitioning step and which actinides can be separated during partitioning, then processed into fuel and finally successfully fissioned. The effect of different separation efficiencies in the separation of plutonium and uranium is shown in Figure 8-12. The description of simultaneous separation of uranium is misleading, since uranium isotopes make a significant contribution to radiotoxicity in the

period between 100,000 and 1 million years (see e.g. Figure 8-11). Figure 8-13 shows the influence of the separation of other actinide elements. The separation of plutonium reduces the long-term radiotoxicity by a factor of 10, even with a low separation factor. According to (Magill et al. 2003), extensive separation of plutonium would mean that the time until the radiotoxicity of this plutonium-free waste falls to the radiotoxicity level of a quantity of natural uranium would be reached after about 10,000 years, instead of after about 130,000 years. Additional separation of minor actinides would shorten this time even further. With a very high separation efficiency >99%[236] and without the separation of the minor actinides, a reduction in radiotoxicity by a factor of about 100 is possible.

Figure 8-12: Radiotoxicity during the separation of plutonium and uranium for different separation factors for LWR UOX fuel with 40 GWd/tHM

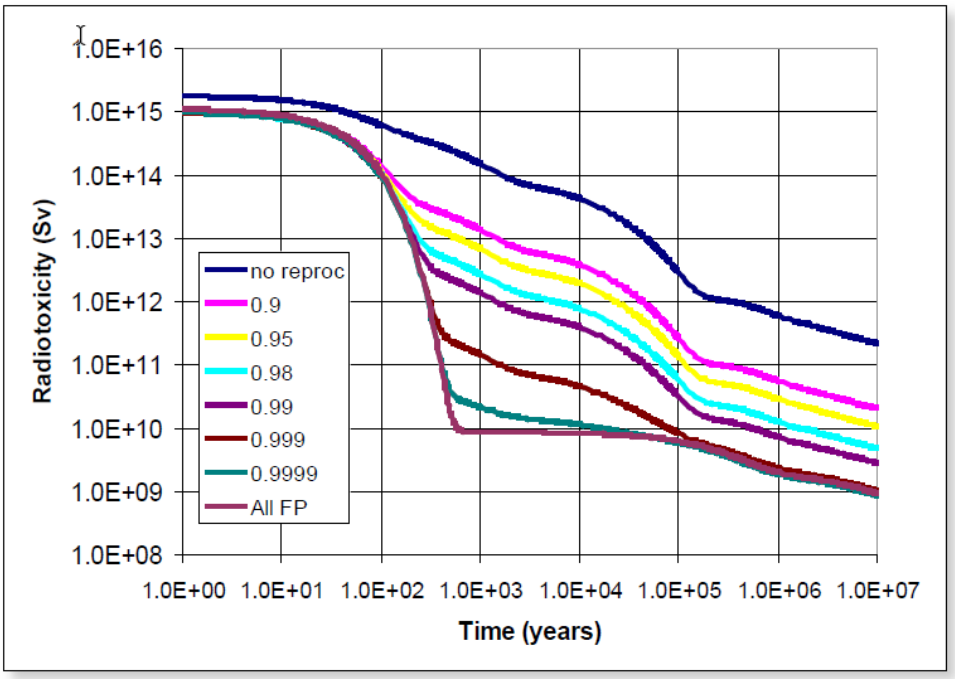


Source: (NEA 2018) 4% uranium-235 – 40 GWd/tHM

It should be noted that the separation factor is assumed for each transmutation cycle and only a limited amount of actinides are converted in one transmutation cycle. The separation losses accumulate over many transmutation cycles and, at the end of the transmutation treatment of nuclear waste, ultimately result in an effective separation efficiency P_{eff} that is significantly lower than the separation efficiency P of a single partitioning step. Even under ideal conditions of infinitely possible transmutation steps (no residual amount of actinides), the effective separation efficiency $P_{eff} = P*T / (1 - (1 - T) P)$, with a transmutation fraction T per transmutation cycle. In numbers, for example, with a transmutation fraction of T = 20%, even with a separation efficiency P = 99.5%, the effective separation efficiency is only 97.5% and drops to 95% at T = 10%. With an inventory of 100 t TRU, even under these idealised assumptions, tons of TRU would still have to be disposed of after P&T treatment. In transmutation scenarios, reductions in actinides by more than a factor of 100 are therefore often sought. In addition, there are residual quantities if, for example, further treatment is stopped after 10 transmutation cycles (see Chapter 8.6.1).

[236] A separation efficiency of >99% for all minor actinides is also discussed in (NEA 2018) as a prerequisite for influencing disposal.

Figure 8-13: Radiotoxicity during the separation of plutonium, uranium and minor actinides for various separation factors for LWR UOX fuel with 40 GWd/tHM

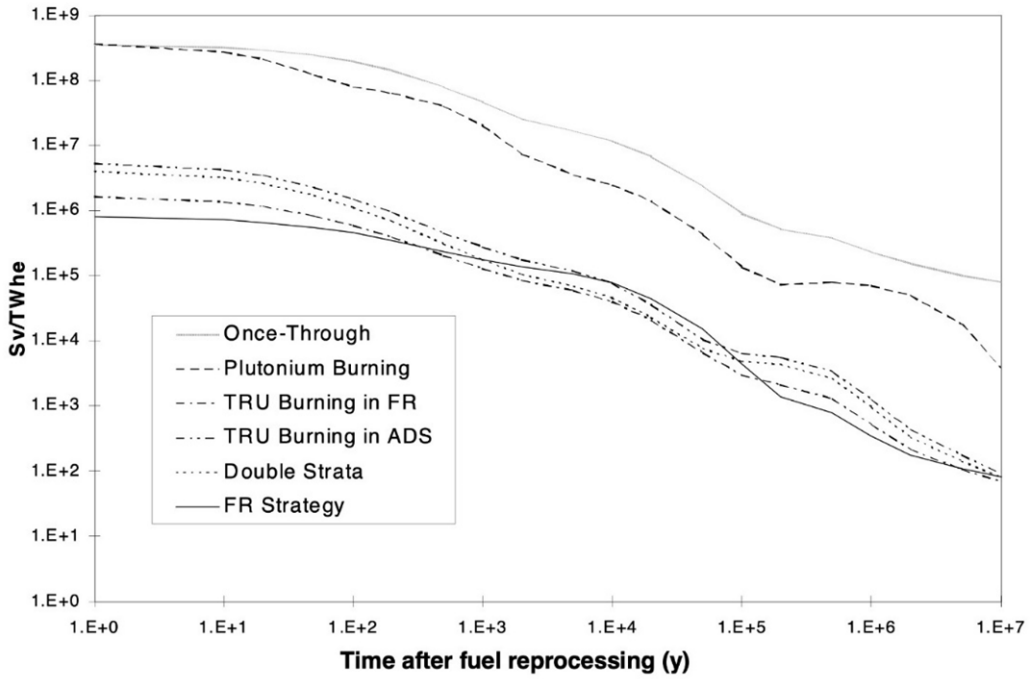


Source: (NEA 2018) 4% uranium-235 – 40 GWd/tHM

The radiotoxicity reduction would differ little over the fuel cycles[237] according to (NEA 2002), which were introduced in Chapter 8.5.5.1 and Figure 8-2, except in the Pu-burning strategy (reduction by a factor of 10); the lower the TRU content in the waste, the lower the radiotoxicity (see Figure 8-14). With regard to the inventory present in the fuel cycle, the double strata strategy and the Pu-burning strategy have the disadvantage that more than half of the radioactive TRU inventory is located outside the reactors, although the double strata strategy also has the highest inventory in the fuel cycle. The TRU-burning-in-FR strategy has the smallest TRU inventory with the advantage of flexible reaction to possible changes in the energy system.

[237] Plutonium burning (LWR+UOX, LWR+MOX, FR+MOX), double strata (LWR+U, LWR+U+Pu, FR+U+Pu, ADS+U+Pu+MA), TRU in FR burner (LWR+U, FR+U+Pu+MA), TRU in ADS burner (LWR+U, ADS+U+Pu+MA), and a pure strategy with fast reactors (FR+U+Pu+MA).

Figure 8-14: Radiotoxicity development of the waste per energy generated at 140 GWd/tHM burn-up for some transmutation scenarios of continuous use of nuclear energy according to (NEA 2002)



Source: (NEA 2002).

Scenario 1 (once-through): Reference scenario with LWR. Scenario 2 (plutonium burning): The plutonium is first extracted and separated in LWR and used in the form of MOX fuel in LWR and FR. Scenario 3a (TRU burning FR): TRU is also separated. The TRU is produced in LWR, separated and then used in MOX in FR. Scenario 3b (TRU burning in ADS): Instead of in FR, the TRU extracted from LWR fuel is used in ADS to transmute the TRU. The ADS make up 25% of the reactor fleet. Scenario 4 (double strata): Scenario with LWR with uranium fuel, plutonium use in LWR with MOX (U+Pu) and in fast reactors (FR+U+Pu). The MA are then separated from the MOX fuels and used in Pu+MA fuel for MA burning in a small parallel fuel cycle with ADS. Scenario 5 (FR strategy). Strategy of a closed fuel cycle for all actinides with fast reactors. The only strategy without using LWR.

(NEA 2002) therefore come to the conclusion that the potential of FR and ADS for reducing TRU, and the resulting reduction in radiotoxicity, are very similar (within a factor of 2), and could achieve an overall reduction in radiotoxicity of more than a factor of 100. The reduction in radiotoxicity depends primarily on the underlying assumptions regarding the achievable burn-up and the losses (99.9%) during reprocessing and fuel element production, especially of the minor actinides. In addition, uranium-free fuels are used in ADS. Heterogeneous transmutation systems in which americium and curium are separated and irradiated in targets are only half as effective in reducing radiotoxicity as homogeneous closed fuel cycles. The sole reduction of plutonium with multi-recycling can only reduce radiotoxicity by a factor of 5 in the long term.

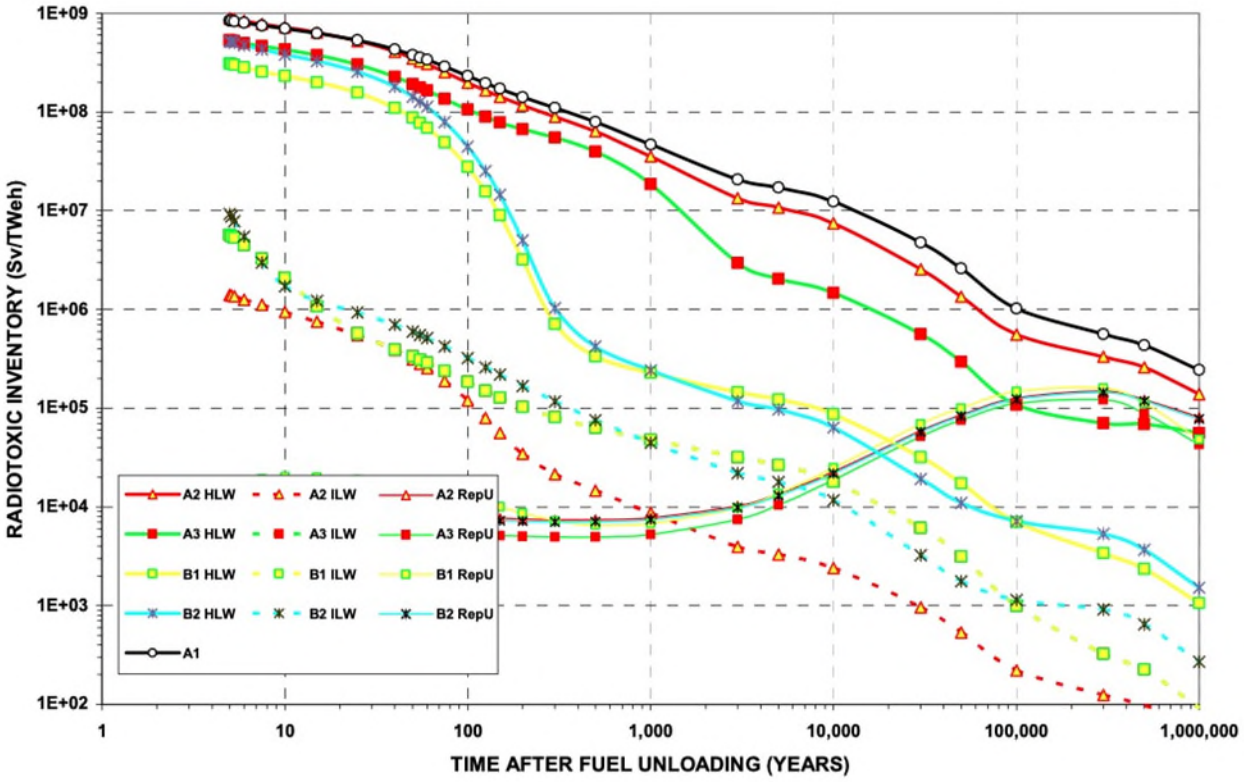
Regarding radiotoxicity, a similar result is also reached in (Lensa et al. 2007) with regard to the HLW waste. The scenarios from (Lensa et al. 2007) were introduced in detail in Chapter 8.5.5.1. Pure Pu use reduces the radiotoxicity of the waste by a factor of about 5, high reductions can only be achieved in scenarios B1 and B2.[238] If the reprocessed uranium produced is treated as waste, the uranium

[238] The fuel cycles are the use of 50% LWR with UOX (A1) and 50% MOX with 8.5% Pu without multi-recycling (A2), the use of fast reactors with Pu separation (A3), the use of fast reactors with TRU (B1) and a mixed strategy with LWR with Pu-MOX and ADS with uranium-free fuels with TRU (double strata, B2), or a strategy in which FR is also included (B3).

waste would contain a similarly high level of radiotoxicity as the HLW, in the long term, from 100,000 years onwards (see Figure 8-15). The usefulness of radiotoxicity reduction would thus be drastically reduced, or limited to the period up to 100,000 years. (Lensa et al. 2007) also discuss the heat development of the waste, but this is not treated here (for heat development, see Chapter 8.6.2).

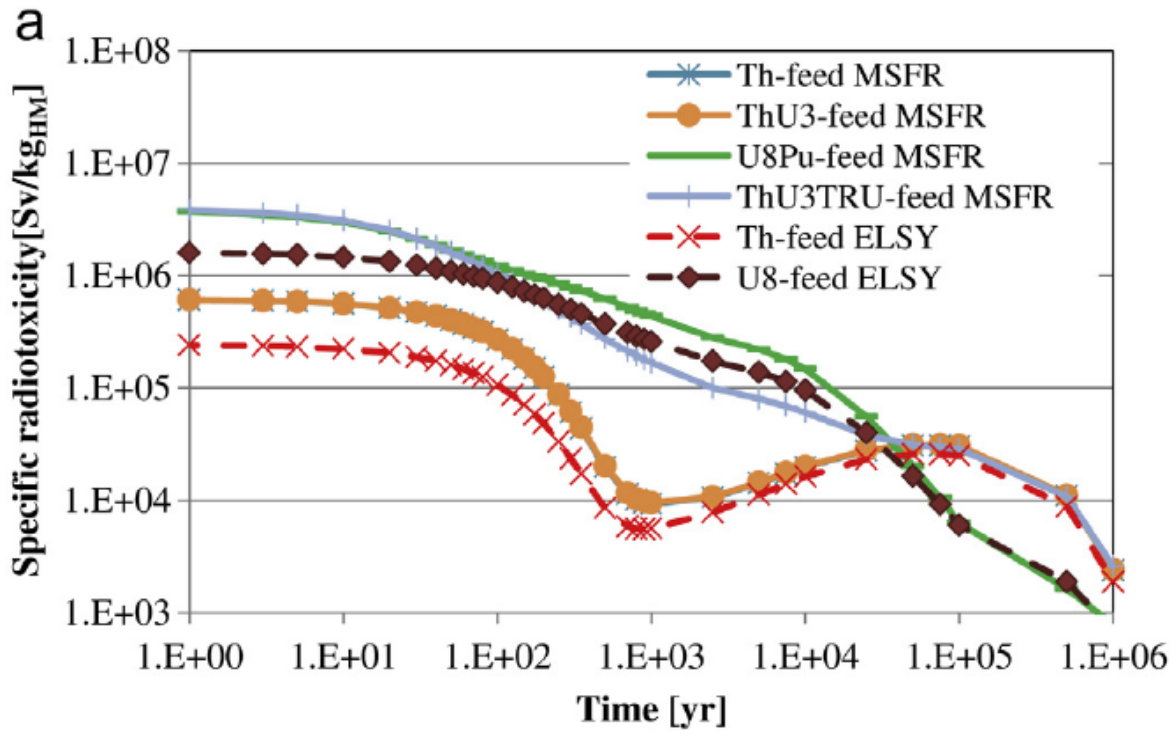
Additional overviews of the radiotoxicity reduction of various fuel cycles with ADS and other fast reactors have also been presented, for example, in (Renn 2014; Salvatores 2012; NEA 2006).

Figure 8-15: Development of radiotoxicity per energy generated for high-level and medium-level radioactive waste, and reprocessed uranium, in the scenarios according to (Lensa et al. 2007)



Source: (Lensa et al. 2007). The radiotoxicity of high-level radioactive waste (HLW), medium-level radioactive waste (ILW) and reprocessed uranium (RepU) is shown. The radiotoxicity of reprocessed uranium is approximately the same as the HLW waste in scenario A3 after 100,000 years. The fuel cycles are the use of 50% LWR with UOX (A1) and 50% MOX with 8.5% Pu without multi-recycling (A2), the use of fast reactors with Pu separation (A3), the use of fast reactors with TRU (B1) and a mixed strategy with LWR with Pu-MOX and ADS with uranium-free fuels with TRU (double strata, B2), or a strategy in which FR is also included (B3).

Figure 8-16: Specific radiotoxicity per fuel mass for different fuel cycles



Source: (Fiorina et al. 2013)

The figure shows the specific radiotoxicity for various fuel cycles calculated with reprocessing losses of 0.1% that remain in the final waste stream. Only the scenarios labeled MSFR are relevant for the discussion here (ELSY are scenarios for a lead-cooled fast reactor). Four scenarios were considered. In the Th-feed scenario, after an initial loading with fissile materials, only thorium is used as fuel, and bred uranium-233 is used for other reactors. In the ThU3-feed MSFR scenario, all uranium-233 is used again in the same reactor (practically overlapping). The last scenario, U8Pu-feed, is the use of uranium-plutonium fuel.

In addition to ADS, MSR are considered particularly suitable for transmutation due to their high potential flexibility with regard to the fuel used. Thorium fuel cycles also have the advantage that, compared to the use of uranium in the fuel, very few new TRU are bred. This means that an MSR using a thorium fuel cycle can be considered one of the most suitable systems for P&T treatment.

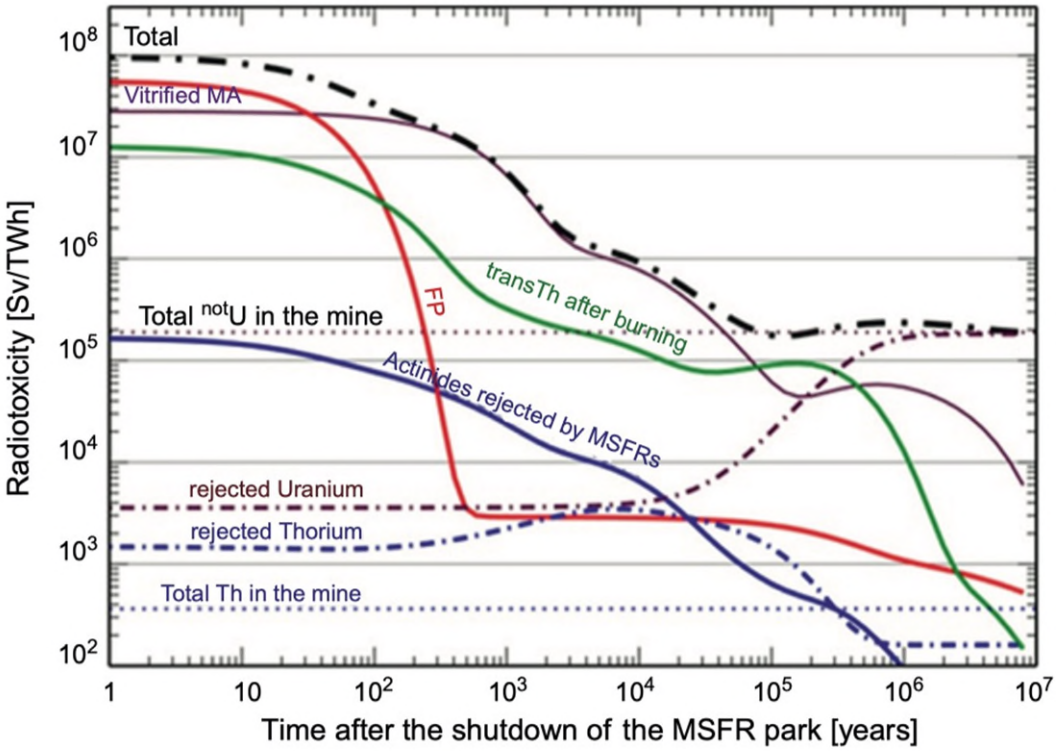
Calculations on the radiotoxicity of various MSR fuel cycles for the MSFR are presented in (Fiorina et al. 2013) (see e.g. Figure 8-16). In the THU3TRU-feed scenario, the MSFR is used for waste incineration. The fuel consists of uranium-233, thorium and 20% TRU from LWR waste, which is slightly less than the solubility of trivalent actinides in the MSFR allows. This means that a reduction of 11.5 t TRU in 50 years is possible with one reactor. Compared to other cycles, the presence of TRU leads to increased short- to medium-term radiotoxicity. In the long term, the daughter nuclides of uranium-233 and uranium-234 lead to an increase in radiotoxicity in thorium cycles compared to the use of uranium in fuel (from approx. 10^5 years). This reduces the decrease in long-term radiotoxicity due to actinide reduction (Fiorina et al. 2013).

With regard to heat development, a first temperature peak in the repository and a second peak after the increase in decay heat in the range of 10^3-10^6 years are calculated for the Th-based options (Fiorina et al. 2013), which determine the repository design. The increase is caused by the slow build-up of thorium-229 and thorium-230 with long half-lives. According to (Fiorina et al. 2013), both peaks are expected to be lower than the initial single peak in the uranium-based counterparts.

However, with high reprocessing rates, the heat production would be comparable to a uranium cycle (Fiorina et al. 2013).

For the scenario of MSFR use, Figure 8-17 shows the development of radiotoxicity for a transition scenario using the reactor in France from 2070 (for the fuel cycle scenario, see also Chapter 0). The diagram includes all waste from the entire fuel cycle system of LWR (PWR, EPR), FR, and MSR with a uranium and a thorium cycle over a period of around 200 years. Since in the fuel cycle the MA from the LWR reprocessing (PWR and EPR), which make up a significant part of the reactor fleet, and from MOX are vitrified over a long period of time, the contribution of MA to radiotoxicity is dominant over long periods of time. But the contribution of irradiated thorium waste for periods of up to 10,000 years is also not insignificant (see below). In these scenarios, the total radiotoxicity reaches the level of natural uranium after about 100,000 years. A significant reduction in radiotoxicity for disposal is not achieved in this temporal transition scenario.

Figure 8-17: Time-development of the individual contributions to radioactivity per energy generated in an MSFR scenario.



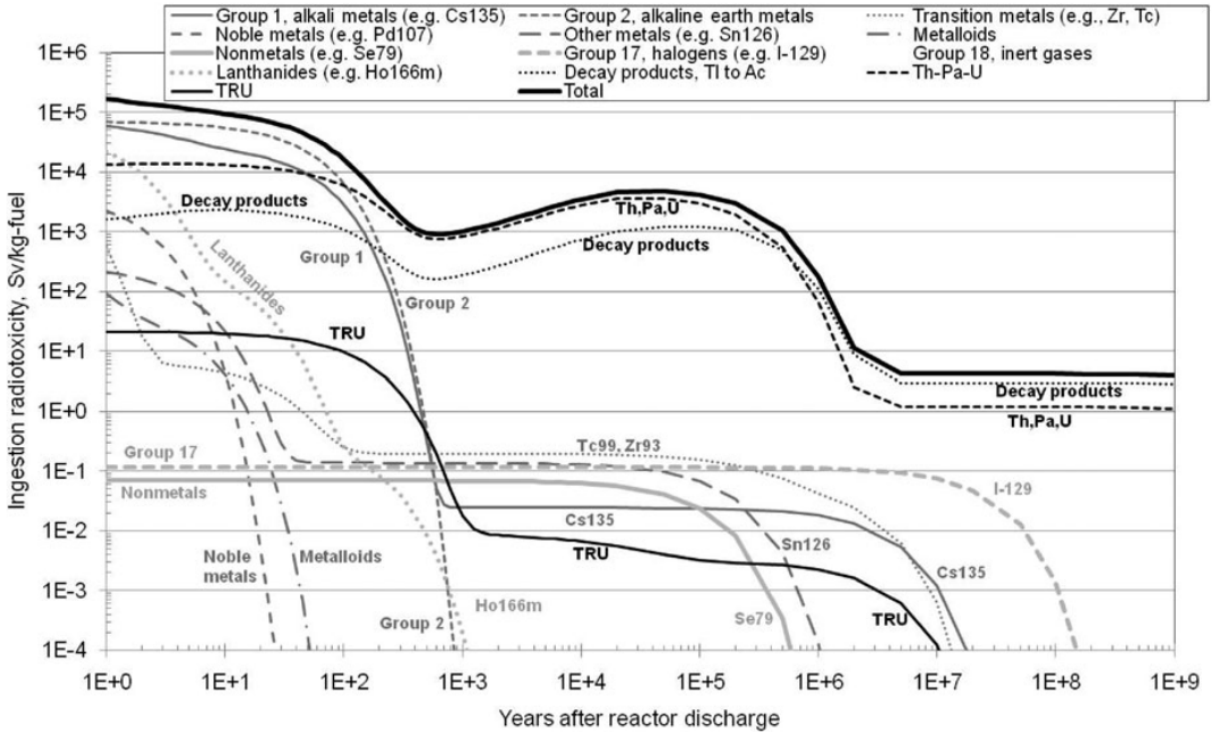
Source: (Allibert et al. 2016; Allibert et al. 2023)

Development of radiotoxicity for a transition scenario using the reactor in France from 2070 (for the fuel cycle scenario, see also Chapter 0). Visible are the different contributions of the fission products (FP), the virtrified minor actinides from LWR that were not reused, transthorium waste from the use of thorium fuels, and uranium waste.

In (Piet 2013), the usefulness of P&T with regard to thorium cycles is also severely limited in terms of radiotoxicity. In addition to the actinides, the calculations mainly consider the fission products selenium-79, zirconium-93, niobium-93m, technetium-99, palladium-107, tin-126, iodine-129, and caesium-135. Technetium-99 and iodine-129, in particular, have the highest overall radiotoxicity. In the period between 1000 and 1 million years, however, the resulting thorium and transthorium

radioisotopes have a similarly high level of radiotoxicity (Figure 8-18) as the U-Pu fuel cycle (Figure 8-11). In (Piet 2013), the authors therefore conclude that the implicit assumption of low TRU inventories in thorium fuel cycles reducing the long-term challenges of disposal is incorrect.

Figure 8-18: Radiotoxicity per fuel mass from spent fuel ingestion, once-through of pure Th/uranium-233 fuel cycle at 18 MWd/kgHM in LWR; no activation products.



Source: (Piet 2013). Diagram of the different contributions of the individual element groups to radiotoxicity. Decay products are also taken into account.

A further comparison of the development of radiotoxicity in 40 different fuel cycle scenarios[239] was carried out in a project of the Department of Energy and documented in several comprehensive reports. The results were summarised in (Stauff et al. 2015). For radiotoxicity after 100,000 years, the maximum difference in radiotoxicity was a factor of 226 between scenario 7 (ADS with natural uranium) and 32. (TRU/U from PWR in SFR burner). In relation to scenario 1 (LWR with UOX), this would result in a maximum radiotoxicity reduction of about a factor of 30 for scenario 32 after 100,000 years. Thorium scenarios in particular have a relatively high radiotoxicity, mainly due to thorium-229, a decay product of uranium-233, but thorium has an overall radiotoxicity that is two orders of magnitude higher than uranium. Likewise, fuel cycles with a high proportion of plutonium in the waste have increased radiotoxicity compared to scenarios in which plutonium is reduced. Scenarios with uranium-233 or Pu in the waste therefore have the highest radiotoxicity (Stauff et al. 2015).

[239] See footnote .231

8.6.4.1 Conclusions on radiotoxicity

Radiotoxicity is a measure of the radiological effects of radioactive substances, which are damaging to health when they enter the human body directly, e.g. via eating, drinking or breathing. The radiotoxicity measure simplifies risk assessment, but ignores all processes that take place between the geological location of the waste and human exposure, as well as the normal development of a repository. It is only suitable for determining risk in the case of very unlikely scenarios. The benefit of radiotoxicity reduction for unlikely events in a geological repository, must be weighed against the risks posed by the radiotoxicity of the radionuclide inventory while it is not yet in a repository.

The reduction of transuranic elements in high-level radioactive waste through P&T significantly reduces the thus defined radiotoxicity, because actinides have a high radiotoxicity that only gradually reduces over very long periods of time. The effectiveness of the radiotoxicity reduction depends largely on the separation efficiency of the partitioning step, i.e. how much residue remains in the waste. Plutonium makes the greatest contribution to radiotoxicity and a plutonium transmutation therefore makes the greatest contribution to radiotoxicity reduction. But only a transmutation of the minor actinides such as americium, curium and neptunium would reduce radiotoxicity by more than the desired factor of 100. Different fuel cycles and SNR differ only slightly from one another in terms of radiotoxicity reduction (factor 2). The thorium fuel cycle represents an exception. Although very few new actinides are created in the thorium cycle and the TRU inventories are low, the long-term radiotoxicity is comparatively high.

8.7 Conclusions on partitioning and transmutation

According to its proponents, partitioning and transmutation (P&T) essentially promises the possibility of reducing the requirements for disposal of highly radioactive nuclear waste. The use of P&T is intended to reduce the amount of waste or the amount of waste generated, the space required in a repository, as well as the radiological risks and storage times required. Transmutation involves the targeted separation of transuranic elements from nuclear waste (partitioning). New fuel must be produced from the transuranic elements. They are then to be converted into shorter-lived or stable isotopes by neutron irradiation in nuclear reactors (transmutation), primarily through fission. Since only a small proportion of the transuranic elements are fissioned during single-use in the reactor, multiple recycling is necessary. Fast reactors (SFR, LFR, GFR, MSR) are a core component of waste treatment strategies with P&T, even though there are P&T concepts with thermal reactors (VHTR, MSR). In addition, accelerator-driven subcritical systems can also be used (ADS).

Wet-chemical and pyrochemical processes are used for partitioning. The separation efficiency must be higher than 99.9% in order to limit the residual amounts of transuranic elements in the waste stream. Plutonium separation processes such as PUREX (Plutonium URanium EXtraction) are now mature and are used on a large scale. The separation of minor actinides (MA) is still under development. The separation of actinides from lanthanides (fission products that act as neutron absorbers) and the division of actinide elements, especially the separation of americium and curium, is chemically difficult and has not yet achieved the required high separation efficiencies outside the laboratory. Pyrochemical processes are still under development. Pyro-electrometallurgical processes are probably the most suitable of the pyrochemical processes for separating MA. The separation efficiency for actinides can reach up to 99.9%.

The production of plutonium fuels such as uranium-plutonium mixed oxide fuel (MOX) has been demonstrated on a large scale, but is not commercially competitive with uranium fuels. The production of fuels with minor actinides has not yet taken place on an industrial scale and is technically even more challenging. The volatility of americium places special demands on the process control and can limit the proportion of americium in the fuel. Due to the high heat development and neutron emission, curium leads to a high cost in fuel production and the transport of fresh fuel. If curium were to be separated, it would have to be irradiated in extra targets or would end up in the waste stream together with the fission products and be disposed of.

The burn-up in MOX limits the reactor service life. Typical burn-ups for transmutation should be 140 MWd/kgHM, but are not yet achieved today. Other fuels such as carbide, nitride and metallic fuels are being developed, i.a. in order to achieve higher burn-ups than MOX. Uranium-free fuels are also being developed and would be particularly suitable for transmutation, as no new production of plutonium and other TRU is taking place.

The one-time use of plutonium in MOX for LWR is currently being carried out on a large scale in a few countries. The amount of plutonium in the fuel can be significantly reduced by irradiation in the reactor, but this produces large amounts of minor actinides that remain in the spent fuel. Repeated recycling or multi-recycling of spent MOX is not currently carried out on a large scale, as handling is challenging due to the heat generated and activity of such MOX fuels and is completely unattractive economically.

The transmutation properties of the technology lines of fast reactors (FR) are very similar. The choice of fuel (with the same achievable burn-up) also has little influence on the transmutation properties of FR. FR can be designed for Pu production or burning but also for MA burning. The fraction of MA in the fuel is limited to a few percent by the safety properties of the reactor concept, if the MA is mixed homogeneously into the fuel; in heterogeneous reactor cores, the MA fraction in the outer area can be increased to up to 40%.

The use of an ADS has the advantage that it is subcritical and the safety properties are therefore not very sensitive to the use of plutonium or MA. ADS could therefore also use uranium-free fuels in which no new MA is created. ADS therefore have advantages over FR systems, especially in phase-out scenarios. In scenarios of continuous nuclear energy use, ADS could form their own small fuel cycle for the burning of MA (Strata) and the rest of the fuel cycle (LWR and/or FR) would be operated without MA separation (Double Strata). ADS have similar properties to FR in terms of the achievable TRU reductions and can be configured as Pu or MA burners.

Since only a small fraction of the originally used transuranic elements is fissioned, the steps of reprocessing, fuel production and use in the reactor must be repeated many times, depending on the required reduction. This leads to different scenarios for the use of P&T. A distinction is made between scenarios of continuous use of nuclear energy and scenarios which aim for the reduction of an initial inventory of radioactive waste (phase-out scenarios). When considering scenarios, the initial inventory of radioactive waste, the minimisation of waste generation (continuous use) or the achievable target reduction (phase-out), i.e. the residual waste, and the implementation time, as well as the required number of reactors, are of importance. Typical scenarios have implementation times of 100 years or more, residual waste continues to be generated and at least several reactors (phase-out) must be built and rebuilt. The required implementation times and the target reduction (phase-out) are very sensitive to the required cycle time, the separation efficiency and the possible transmutation fraction per cycle for all systems used.

In phase-out scenarios with a continuously operated, consistent reactor fleet consisting of SFR burners (Pu burners and MA burners), an initial inventory of TRU could be reduced by a factor of around 80 under ideal conditions (separation efficiency, transmutation share). This would result in around two and a half times the initial inventory of fission products. Assuming limitations on the flexibility of SFR, e.g. in the form of a homogeneous, consistent element or isotope vector in the fuel, could drastically reduce the possible reduction. Reactor concepts that are specifically designed for TRU transmutation should, however, be able to deal with a relatively variable isotope composition of the TRU, although sufficient fresh fissile material would then have to be supplied, e.g. in the form of plutonium or enriched uranium.

For the purpose of P&T treatment for disposal, many scenarios with SNR envisage the simultaneous use of plutonium and MA in the fuel. There are fuel cycles with reactors that are specifically optimised for burning plutonium and MA, but also for breeding new fissile material (homogeneous vs. heterogeneous core). More precise estimates of transmutation rates therefore depend on the reactor configuration and the entire fuel cycle.

Thermal reactors are only suitable for transmutation to a limited extent due to the limited fraction of TRU in the fuel. Plutonium is reduced, but MA is built up. The higher the proportion of MA in the fuel, the more fissile material must be in the fuel, for example in the form of uranium-235. However, curium typically accumulates in the fuel.

In closed fuel cycles, the volume (mass) of high-level radioactive waste can be significantly reduced. However, this is mainly due to the separation of uranium and the allocation of the reprocessed uranium either to another waste stream, e.g. medium-level radioactive waste, or to its consideration as a valuable material. MA recycling leads to a further small reduction in the volume of waste compared to uranium/plutonium fuel cycles. In fast reactors, thorium fuels have a lower transmutation rate than uranium fuels. All fuel cycles with reprocessing produce additional low- and medium-level radioactive waste.

If the packaging and conditioning of the waste is taken into account, in scenarios of continuous use of nuclear energy, the volume of high-level radioactive waste from reprocessing and spent fuel elements is similar among individual closed-fuel-cycle scenarios and amounts to at least about 1/3 of the LWR waste. A large proportion of the uranium is no longer part of the waste stream of highly radioactive waste, but is accounted for as medium or low-level radioactive waste. The amount of low and medium-level radioactive waste is significantly higher. Scenarios with FR, in particular, have a higher waste inventory than ADS due to the high amount of activated structural materials. But even with ADS, further waste can arise due to the large proportion of zirconium in zirconium nitride fuels and the accumulation of precious metals during pyroprocessing. In addition, waste from reprocessed uranium (or thorium) is also produced.

In principle, an advantage of MSR with a fast neutron spectrum is the flexibility of the liquid reactor concept through which a very high reduction in actinides and the shortening of the cycle length can be achieved through the possibility of reprocessing during operation. MSR have a very high amount of secondary waste due to the use of molten salt. A significant limiting factor for the use of transuranic elements or plutonium is the limited solubility of the actinides in reactor concepts based on fluoride salts.

With regard to the effects of P&T waste treatment, P&T is intended to reduce the quantities and volume or heat of the waste, leading to a decrease in size of the required repository facilities. Sometimes, claims are even made that a repository facility may not be necessary. P&T is also intended to have a positive effect on the risks of disposal in a geological repository facility.

In no scenario involving the use of alternative fuel cycles with SNR and P&T treatment of waste can a repository for high-level radioactive waste be dispensed with, as residual amounts of transuranic elements, and long-lived fission and activation products, remain in the waste stream. In addition, there are significantly higher amounts of low- and medium-level radioactive waste from the operation and dismantling of the partitioning plants.

The heat production of highly radioactive waste determines the required distances between the waste containers and thus the required storage space in a geological repository and thus its size. In the first 100-200 years, fission products are particularly prominent in heat production. Separating strontium (and caesium), and storing them separately, would significantly reduce the required storage space, as would storing the waste without partitioning for these periods before storage in a geological repository. This must be weighed against the risks of long-term interim storage of spent fuel elements or partitioning waste and the additional low and medium-level radioactive waste streams generated for partitioning. Transmuting the actinides would only have a significant impact for periods beyond this, with the total heat production from actinides only amounting to 10-20% of the initial values of the total waste after 300 years. Alternative fuel cycles with SNR therefore have only a very limited influence on the heat production of high-level radioactive waste and thus the size of a repository.

The reduction in volume and quantity of highly radioactive waste through the separation of uranium is shown in many studies as an advantage of P&T. This sometimes results in very high reduction factors for highly radioactive waste. Since the required storage area of a repository is determined by the heat input, a volume reduction, e.g. by separating uranium, changes little in the required size of a repository.

With regard to possible radiation risks when disposing of radioactive waste in a geological repository, two main criteria are used: the radiotoxicity of the waste and the long-term safety of the geological repository.

Radiotoxicity is only suitable for representing the risks for very unlikely events in the development of a repository with a release of the radioactive inventory and subsequent incorporation of radionuclides. With regard to these very unlikely events of release and incorporation of radionuclides, P&T treatment using SNR significantly reduces the long-term radiotoxicity of high-level radioactive waste, depending on the separation efficiency of the partitioning technology used and whether minor actinides are separated. This analysis does not take into account the planned development of a repository, as well as the release through mobilisation of radioisotopes and the resulting dose for humans. It therefore makes no statement about the long-term safety relevance of the radionuclides in their repository and rather identifies the wrong radionuclides in connection with P&T treatment. The radiotoxicity of thorium-based fuels that contain uranium-233 as fissile material is defined by the build-up of heavier elements and their decay products and is in result similar to the radiotoxicity of uranium-based fuels.

The standard methodology for assessing the radiological risks to humans from geological repositories for spent fuel is a long-term safety analysis. This takes into account the release pathways of the nuclides. Long-lived mobile fission products are important for the long-term safety of a geological repository. Actinides are extremely immobile in most repository rocks, except tuff rock, and are not released. Instead, the main contribution to dose estimates in long-term safety analysis comes from long-lived, mobile fission products such as iodine-129, technetium-99 and activation products such as carbon-14. Therefore, reduction of transuranic elements makes no contribution to the long-term safety analysis of a repository; instead, fission product transmutation would be necessary, especially of technetium-99, iodine-129, caesium-135 and carbon-14 for example.

In scenarios of longer-term use of nuclear energy, fuel cycles with reprocessing therefore have no advantage in long-term safety considerations over the current use of LWR if all waste streams, including low- and medium-level radioactive waste, are taken into account. In transmutation scenarios with a fixed initial inventory of transuranic elements (phase-out), the amount of dose-determining radionuclides would increase due to the additionally created fission products.

Due to the technical difficulties posed by partitioning and target production for irradiation, as well as the partly necessary isotope separation, fission product transmutation would be technically limited to the nuclides technetium-99 and possibly iodine-129. In countries that have undertaken reprocessing, parts of the fission product waste have already been immobilised in glass. Re-partitioning the vitrified waste is considered very challenging. Overall, the transmutation of fission products is rarely pursued today.

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