

Thomas Schulenberg

The fourth generation of nuclear reactors

Fundamentals, Types,
and Benefits Explained

 Springer

The fourth generation of nuclear reactors

Thomas Schulenberg

The fourth generation of nuclear reactors

Fundamentals, Types, and Benefits
Explained



Thomas Schulenberg
Institute for Thermal Energy Technology and Safety
Karlsruhe Institute of Technology
Eggenstein-Leopoldshafen, Baden-Württemberg, Germany

ISBN 978-3-662-64918-3 ISBN 978-3-662-64919-0 (eBook)
<https://doi.org/10.1007/978-3-662-64919-0>

© The Editor(s) (if applicable) and The Author(s), under exclusive licence to Springer-Verlag GmbH, DE, part of Springer Nature 2022

This book is a translation of the original German edition „Die vierte Generation der Kernreaktoren“ by Schulenberg, Thomas, published by Springer-Verlag GmbH, DE in 2020. The translation was done with the help of artificial intelligence (machine translation by the service DeepL.com). A subsequent human revision was done primarily in terms of content, so that the book will read stylistically differently from a conventional translation. Springer Nature works continuously to further the development of tools for the production of books and on the related technologies to support the authors.

This work is subject to copyright. All rights are solely and exclusively licensed by the Publisher, whether the whole or part of the material is concerned, specifically the rights of translation, reprinting, reuse of illustrations, recitation, broadcasting, reproduction on microfilms or in any other physical way, and transmission or information storage and retrieval, electronic adaptation, computer software, or by similar or dissimilar methodology now known or hereafter developed.

The use of general descriptive names, registered names, trademarks, service marks, etc. in this publication does not imply, even in the absence of a specific statement, that such names are exempt from the relevant protective laws and regulations and therefore free for general use.

The publisher, the authors and the editors are safe to assume that the advice and information in this book are believed to be true and accurate at the date of publication. Neither the publisher nor the authors or the editors give a warranty, expressed or implied, with respect to the material contained herein or for any errors or omissions that may have been made. The publisher remains neutral with regard to jurisdictional claims in published maps and institutional affiliations.

This Springer imprint is published by the registered company Springer-Verlag GmbH, DE part of Springer Nature.

The registered company address is: Heidelberger Platz 3, 14197 Berlin, Germany

Contents

1	Introduction	1
2	Fundamentals of Nuclear Technology	7
3	Innovative Water-Cooled Reactors	37
4	High-Temperature Reactors	55
5	The Nuclear Fuel Cycle	73
6	Sodium-Cooled Fast Reactors	93
7	Lead-Cooled Fast Reactors	115
8	Gas-Cooled Fast Reactors	135
9	Molten Salt Reactors	147
10	Significance for the Economy and Society	167
	Appendix A: Glossary	173
	References	179

Abbreviations

AEG	Allgemeine Elektricitäts-Gesellschaft (General Electricity Company)
AMU	Atomic Mass Unit
ARE	Aircraft Reactor Experiment
AVR	Arbeitsgemeinschaft Versuchsreaktor (Experimental Reactor Working Group)
BREST	Быстрый Реактор ЕСТественной безопасности (Fast Reactor With Inherent Safety)
CANDU	CANada Deuterium Uranium
CEFR	China Experimental Fast Reactor
CRIEPI	Central Research Institute of Electric Power Industry
DIAMEX	DIAMide EXtraction
DOE	Department of Energy
EPR	European Pressurized Water Reactor
ESBWR	Economic Simplified Boiling Water Reactor
FBTR	Fast Breeder Test Reactor
FWT	Feedwater Tank
GTHTR	Gas Turbine High-Temperature Reactor
GTMHR	Gas Turbine Modular Helium Reactor
HDR	Heissdampfreaktor (Superheated Steam Reactor)
HP	High Pressure
HPLWR	High Performance Light Water Reactor
HTR	High Temperature Reactor
HTR-PM	High-Temperature Pebble Bed Modular Reactor
HTTR	High-Temperature Test Reactor
IAEA	International Atomic Energy Agency
IP	Intermediate Pressure
IPPE	Institute for Physics and Power Engineering
JSFR	Japanese Sodium-Cooled Fast Reactor

KAERI	Korea Atomic Energy Research Institute
KNK	Compact Sodium-Cooled Nuclear Reactor Facility Karlsruhe
KWU	KraftWerk Union
LP	Low Pressure
MOSART	MOLTEN Salt ACTINIDE Recycler and Transmuter
MOX	Mixed Oxide of UO ₂ and PuO ₂
MSFR	Molten Salt Fast Reactor
MSRE	Molten Salt Reactor Experiment
MYRRHA	Multi-Purpose HYbrid Research Reactor for High-Tech Applications
NEA	Nuclear Energy Agency
NITI	Научно-Исследовательский Технологический Институт им. А. П. Александрова (Nautschno-Issledovatelskij Technologitscheskiji Institut im. A. P. Aleksandrova, A. P. Aleksandrov Scientific Research Technological Institute)
PBMR	Pebble Bed Modular Reactor
PFBR	Prototype Fast Breeder Reactor
PUREX	Plutonium Uranium Extraction
RBMK	Реактор Большой Мощности Канальный (Reaktor Bolschoj Moschnosti Kanalnyj, High-Power Reactor With Channels)
SANEX	Selective ActiNide EXtraction
SCK-CEN	StudyCentrum voor Kernenergie – Centre d’Étude de l’énergie Nucléaire
SNR	Fast Sodium-Cooled Reactor
SVBR	Свинцово-Висмутовый Быстрый Реактор (Svintsovo-Vismutovyj Bystryi Reaktor, Lead-Bismuth Fast Reactor)
SWR	Siedewasserreaktor (Boiling Water Reactor)
TBP	Tributylphosphate
THTR	Thorium High Temperature Reactor
TMSR	Thorium Molten Salt Reactor
TRISO	TRistructural-ISOrropic
UK	United Kingdom
UNSCEAR	United Nations Scientific Committee on the Effect of Atomic Radiation
USA	United States of America
VVER	Водо-Водяной Энергетический Реактор (Vodo-Vodjanoj Energetiskij Reaktor, Water-Water Energy Reactor)

Formula Symbol

A	Activity [Bq = 1/s]
β	Proportion of delayed neutrons in relation to prompt neutrons [-]
E_f	Energy per fission [eV, J]
Φ	Neutron flux [1/cm ² s]
F	Dose coefficient [Sv/Bq]
k	Boltzmann constant, $8.617 \cdot 10^{-5}$ eV/K
λ	Decay constant [1/s]
N	Nuclide density [1/cm ³]
q	Power density [W/cm ³]
R	Reaction rate [1/cm ³ s]
ρ	Reactivity [-]
σ_a	Cross section for absorption [barn, cm ²]
σ_f	Cross-section for nuclear fission [barn, cm ²]
T	Absolute temperature [K]
$t_{1/2}$	Half-life [s]

List of Figures

Fig. 2.1	Section of the nuclide chart; white: stable, yellow: β^+ emitter, green: β^- emitter	9
Fig. 2.2	Spectrum of the cross section σ_a of neutron absorption in $^{238}_{92}\text{U}$	14
Fig. 2.3	Cross sections of nuclear fission σ_f of $(^{235}_{92}\text{U}$ black) and $(^{238}_{92}\text{U}$ grey)	15
Fig. 2.4	Schematic of the chain reaction of nuclear fission	16
Fig. 2.5	Sketch of the teaching reactor SUR 100	17
Fig. 2.6	Schematic diagram of an RMBK reactor	21
Fig. 2.7	Schematic of uranium enrichment with gas centrifuges; left: Principle of a single gas centrifuge; right: Section of an enrichment cascade with gas centrifuges	23
Fig. 2.8	Principle sketch of a pressurized water reactor	25
Fig. 2.9	High pressure system (primary system) of a pressurized water reactor	27
Fig. 2.10	Schematic of the secondary system of a pressurized water reactor	30
Fig. 2.11	Schematic diagram of a boiling water reactor	32
Fig. 3.1	Change in material properties of supercritical water with tempera- ture at a pressure of 250 bar	39
Fig. 3.2	Schematic of a supercritical steam process for an innovative water-cooled reactor. FWT: feedwater tank; HP, IP, LD: high- pressure, intermediate-pressure, low-pressure turbines	40
Fig. 3.3	Sketch of the high performance light water reactor	42
Fig. 3.4	Fuel assemblies of a high performance light water reactor	42
Fig. 3.5	Sketch of the containment and the emergency cooling concept of an HPLWR	44
Fig. 3.6	Design principle of the superheated steam reactor (HDR) and one of its fuel assemblies	47
Fig. 3.7	Section through a fuel assembly of the HDR	48

Fig. 3.8	Section through a pressure tube of a CANDU reactor with supercritical pressure	51
Fig. 3.9	Containment of a third-generation pressurized water reactor and an innovative water-cooled reactor in size comparison	52
Fig. 4.1	Sketch of the ceramic fuel for a high-temperature reactor with graphite blocks	56
Fig. 4.2	Section through the HTTR experimental reactor in Japan	57
Fig. 4.3	Principle of a modular pebble bed reactor with steam generator	60
Fig. 4.4	Schematic of the high-temperature reactor GTHTR300 with gas turbine and recuperator	62
Fig. 4.5	Schematic of a high-temperature reactor with integrated gas turbine	63
Fig. 4.6	Schematic of the ANTARES high-temperature reactor with gas turbine and tertiary steam cycle	65
Fig. 4.7	Simplified representation of the sulphur-iodine process	68
Fig. 5.1	Uranium and transuranics in the nuclide chart. White arrows: Formation of transuranics from U-238 by neutron absorption and β^- -decay; red arrows: Formation of Pu-238 from U-235; black arrow: formation of Pu-238 by α -decay of Cm-242	74
Fig. 5.2	Build-up of plutonium isotopes in the fuel during the burn-up of uranium with 5% enrichment	76
Fig. 5.3	Radiotoxicity of the spent fuel of a pressurised water reactor in the repository	78
Fig. 5.4	Principle of separation of uranium and plutonium in mixer-settlers	80
Fig. 5.5	Partitioning of transuranium elements from the spent fuel	82
Fig. 5.6	Regeneration factor of nuclear fission of some typical fissile nuclides	85
Fig. 5.7	Fission probability of some transuranium elements in a fast and in a thermal reactor	86
Fig. 5.8	Uranium and thorium in the nuclide chart. White arrows: Generation of fissile U-233 from natural thorium; red arrows: Build-up of heavier nuclides by neutron capture and decay; black arrows: Generation of U-232 and its decay	88
Fig. 6.1	Fission cross section of Pu-239 and example of a fast neutron spectrum of a sodium-cooled reactor	98
Fig. 6.2	Comparison of the fuel rod arrangement in a thermal and a fast reactor	99
Fig. 6.3	Principle of a sodium-cooled reactor in loop design	102
Fig. 6.4	Basic structure of the core of the JSFR-1500	103
Fig. 6.5	Principle of a sodium-cooled reactor in pool design	105
Fig. 6.6	Heterogeneous arrangement of fissile MOX fuel (grey) and fertile U-238 (black) in the core	107
Fig. 6.7	Section through the lower half of reactor 4S	109
Fig. 7.1	Solubility of iron and chromium in liquid lead or lead-bismuth [34]	118

Fig. 7.2	Permissible window for dissolved oxygen concentration in lead-bismuth [33]	119
Fig. 7.3	Principle of the Russian SVBR 75/100 with lead-bismuth cooling	122
Fig. 7.4	Principle of the lead-cooled reactor BREST-1200	125
Fig. 7.5	Principle of the lead-cooled reactor BREST-1200 in plan view	126
Fig. 7.6	Combination of a linear accelerator with a lead-cooled reactor to form an accelerator-driven system	129
Fig. 7.7	Sketch of the MYRRHA reactor in horizontal cross-section	131
Fig. 8.1	Fuel assemblies of the ALLEGRO fast helium-cooled reactor; left: MOX fuel assembly with stainless steel cladding tubes, right: ceramic fuel assembly	137
Fig. 8.2	Design of a ceramic fuel assembly with silicon carbide sandwich plates	139
Fig. 8.3	Structure of the core of the ALLEGRO reactor	139
Fig. 8.4	Schematic of a helium-cooled fast reactor	142
Fig. 8.5	Schematic of the main cooling circuit of a steam-cooled fast reactor with external injection evaporator	143
Fig. 9.1	Principle of the molten salt reactor Fireball	149
Fig. 9.2	Schematic of the experimental reactor MSRE	151
Fig. 9.3	Sketch of the primary system in the MOSART reactor	156
Fig. 9.4	Schematic diagram of the Molten Salt Fast Reactor (MSFR)	159
Fig. 9.5	Schematic of the pyroprocess	160
Fig. 9.6	Cross-section through the planned reactor TMSR-SF1	163

List of Tables

Table 5.1	Fuel composition per kg fuel of a pressurized water reactor after discharge	77
Table 6.1	History of sodium-cooled fast reactors worldwide	95
Table 6.2	Comparison of the physical properties of sodium and water	97
Table 7.1	Thermo-physical properties of lead and lead-bismuth [33]	116
Table 7.2	Nuclear physical properties of lead and bismuth	117
Table 8.1	Comparison of the physical properties of helium and sodium	136
Table 8.2	Physical properties of some alternative fuels [46]	138
Table 9.1	Examples of the physical properties of fluoride mixtures for the molten salt reactor	148
Table 9.2	Examples of fissile material composition in the MOSART reactor	157



1

Introduction

The history of nuclear energy in the USA and Europe began during the Second World War, or rather shortly after it ended. Already towards the end of the 1960s, the first nuclear power plants with a capacity of several hundred megawatts were built in Europe and the USA, which we now call the first generation of nuclear power plants. These were both pressurized water reactors and boiling water reactors, of which only a few are still in operation today. They were followed by nuclear power plants of about 1000 MW capacity or more, which were standardized, i.e. could be built in series; in Germany, for example, the Kraftwerk Union (KWU) convoy plants, a pressurized water reactor type with more than 1200 MW electrical capacity. We refer to these nuclear power plants as the second generation. They started operation towards the end of the 1970s and in the 1980s, in the USA as well as in Europe.

The turbulent development of nuclear power plants in the 1970s was not without consequences for safety. When, in 1979, the core of the Three Mile Island pressurised water reactor in Harrisburg, USA, overheated due to a lack of cooling water and began to melt, safety analyses began to simulate possible accidents and systematically analyse the consequences for the plant. This enabled to provide for appropriate safety measures during the construction phase. Nuclear power plants that went into operation in Germany in the 1980s were already required to have a “probabilistic safety analysis”. This is a calculation in which a probability of failure is determined for each safety-relevant component of the power plant in order to determine the probability of occurrence of a severe accident. The risk is the mathematical product of the probability of occurrence and the consequential damage. If the risk was too

high, the safety system had to be improved. Unfortunately, this safety analysis was not requested for the older power plants that were already in operation in 1980. Nor for the boiling water reactors of the power plant Fukushima I, which went into operation in 1971. They had already been approved.

The damage in Harrisburg had no consequences for the surrounding population. The reactor pressure vessel remained intact during the accident and continued to protect the surrounding area from radioactive radiation. The explosion of a pressure-tube reactor in Chernobyl in 1986, however, showed that a severe accident can also have a very different outcome. This is a completely different type of reactor than the pressurised water reactors and boiling water reactors of the western world, and such an accident could be ruled out in the case of pressurised water reactors for design reasons alone. However, the accident raised awareness that more must be done to protect the population. On one hand, this resulted in improved disaster control, because no one in the vicinity of the Chernobyl reactor would have been exposed to any significant radiation if everyone had been informed immediately and appropriate measures had been taken. On the other hand, this accident led to a further improvement of pressurized and boiling water reactors, which we now call the third generation. The European Pressurized Water Reactor “EPR”, which was jointly developed in the 1990s by Siemens in Germany and Framatome in France, was designed in such a way that even if the reactor were to explode, no one in the surrounding population would have to be evacuated. This is achieved with a lot of concrete and a sophisticated cooling system. At any time, even if everything went wrong, this reactor must be able to be brought back under control. Such EPR reactors can be found today in Finland and France, in China and soon in the United Kingdom (UK).

So what is the fourth generation? I like to compare reactor generations to automobile manufacturing. In this analogy, the second generation is a mid-size car. A mature technology, inexpensively produced and sold a lot. The third generation is more like a full-size luxury car. Not only is it nicer and more comfortable, but it has many additional safety features to protect the occupants and other road users. If driven decently, i.e. not compensating for the increased safety by driving faster, the full-size vehicle should cause fewer fatal accidents. However, this makes the full-size luxury car considerably more expensive, and since I can't afford it, I'd rather stick with the mid-size care and accept the higher risk.

Please do not say now that this is not comparable, that the risk of a nuclear power plant is significantly higher than that of cars. The opposite is the case: every year, we kill more than a million people worldwide with cars. In Germany alone, around ten people a day. And we are proud that there are

only so few of them. As far as I know, no one has ever died as a result of a nuclear reactor accident in Germany.

In this analogy, the fourth generation of nuclear power plants corresponds to racing cars. Such a Formula 1 racing car is neither safer nor cheaper than a normal car. It is neither more fuel efficient, nor more environmentally friendly, nor quieter. Quite the opposite. It's not really good for anything, because it's not street legal. And you would hardly go on holiday in it. It doesn't even have a trunk. So why develop and build such a racing car? This question is difficult to answer and the answer is hardly convincing, but I'll give it a try anyway:

- Because racing cars fascinate us. A job in the racing department of a car manufacturer is probably the dream of every young engineer in this industry, where crazy ideas are allowed and no one immediately asks "what does that cost ...?". Sure, turning a crazy idea into a reliable component of a race car will require quite a bit of good sense, and the question of costs will have to be answered as well. But the job still remains visionary.
- Because, as an engineer, you sometimes have to reach for the stars to achieve more. An engineer should never stop questioning existing boundaries, even if his older colleagues will smile about it, because the applicable rules and standards don't even allow that.

In nuclear technology, the fourth generation is analogously a collective term for concepts that are exciting, but which the market does not even need today. Nor is it certain that the market will ever need these concepts. The fourth generation has actually been around since the beginning of the development of nuclear power plants. There just wasn't that term for it yet. So it doesn't refer to a specific time period, but rather to a vision that drives research. It was revitalized at the turn of the millennium by an initiative of the Department of Energy (DOE) in the United States and has excited many young engineers ever since.

The Generation IV initiative has been international from the outset. In 2001, Argentina, Brazil, France, Japan, Canada, South Africa, South Korea, the United Kingdom and the United States joined forces to form the Generation IV International Forum, with the common goal of advancing research in this field. Together you achieve more and you can share the costs. However, the many international agreements involved do not always make collaboration easy. In the following years, Switzerland, Euratom representing the European Union, the People's Republic of China and Russia joined the international alliance. The technical objectives were intentionally formulated

in very general terms—something that any country could sign without hesitation. Nevertheless, we should take a brief look at them:

Sustainability

This term is used a lot and, unfortunately, it is already a bit hackneyed. By “sustainability”, this International Forum means, on one hand, a much better use of the raw material uranium. The spent fuel from nuclear reactors can indeed be recycled. Not just once, but in some reactor designs almost indefinitely, as we will see later. This secures the fuel supply for millennia. True, uranium is very cheap and abundant today. But today’s uranium deposits will nevertheless last for “only” a few centuries. Reason enough to think about recycling.

Secondly, sustainability means minimising waste. Unfortunately, not everything in the spent fuel is so valuable, and we would be reluctant to recycle it. Many radioactive fission products decay to stable, i.e. harmless, substances within a few hundred years. But there remain some very long-lived radionuclides that we could fission in certain fourth-generation reactor designs, that is, convert them into short-lived radioactive materials. This would significantly alleviate the long-term disposal problem.

Safety and Reliability

No, they are not the same thing. Safety and reliability are actually rather competing goals. A car that won’t start if there’s some minor problem is probably very safe. But you still wouldn’t buy it, because you couldn’t use it reliably. A car without an airbag, on the other hand, might be very reliable to drive, but you wouldn’t be safe in it in the event of an accident. For the fourth generation, that means we’d like to have nuclear power plants that are as reliable as the second generation, but at least as safe as the third generation. That is not easy. A prototype is usually unreliable at first, until we’ve tested it for many years and found and fixed all the design flaws. Therefore, for a reliable prototype, we use only components that have already been tested for many years, if possible. However, this limits the choice of innovative systems considerably.

Economic Competitiveness

This goal interests me primarily as an electricity customer. Of course, I want to get the kilowatt hour as cheaply as possible. For nuclear power plants, this means first of all the lowest possible manufacturing costs. This is achieved, for example, by minimising the number of components. What is no longer needed is left out. This will not cost anything then and cannot fail. On the other hand, the construction time, fuel costs and disposal costs are also included in the electricity costs. Another approach is to generate as much power as possible with the power plant. This reduces the investment and fixed costs per kilowatt. In the case of fourth generation, development costs also play a major role. If the power plant is too innovative, we will need many years of trial operation until it runs reliably, and that will cost not only time but also a lot of money, since we have invested capital but have hardly any revenue.

Although large, powerful power plants are cheaper than small ones—in terms of kilowatt hours—emerging countries in particular have a problem with this. Who will lend me so much money? Can I even sell the kilowatt hours generated? Who bears the financial risk if the construction is delayed? In these cases, small power plants are sometimes more economical.

Proliferation Resistance

Proliferation is the misuse of nuclear fuel to make nuclear weapons. During the Cold War, some nuclear reactors were built in such a way that weapons plutonium could casually be diverted from them. The goal here is just the opposite. Similarly as alcohol, which is denatured as spirits so that it won't be drunk, nuclear reactors can be designed such that they don't produce nuclear weapons-grade plutonium, but instead produce a plutonium composition that would be unlikely to be used to build a nuclear bomb. We call a fuel proliferation resistant if it cannot be used directly for the production of nuclear weapons, but only at great expense. However, such a fuel does not offer absolute protection against proliferation. For this reason, the International Atomic Energy Agency (IAEA) is monitoring also stocks of nuclear fuel from which nuclear weapons can only be produced indirectly, even stocks of natural uranium, but not uranium ore. However, proliferation-resistant nuclear fuel needs to be checked less frequently, say annually or quarterly, than nuclear fuel that could be used directly to produce nuclear weapons.

As a first action of this forum, the many reactor concepts known until then were divided into six categories, which I would like to report on in this book. I consider this classification to be the most important step of all, because only this enabled international cooperation. A steering committee was soon formed for each of the six concepts, which worked out a roadmap [1]. It outlines what would have to be done in principle in order to build such a reactor jointly one day. However, it should be mentioned that the Forum never had its own budget to be able to commission the necessary research work. All contributions from international research groups were voluntary. The groups needed funding from their own countries, from industry or, in Europe, from Euratom. But the funding agencies were even less coordinable, and joint research management required a great deal of patience and persuasion. It is not surprising, therefore, that the Roadmap timetable was difficult to meet. The revised 2014 version of the roadmap [2] should similarly not be mistaken for a strict development plan. It does, however, give newcomers to this technology an initial indication of where it makes sense to get involved.

Before we go deeper into the technologies of the individual reactor concepts, I should explain some basics of nuclear technology. If you know that already, feel free to skip ahead to Chap. 3.



2

Fundamentals of Nuclear Technology

We begin with an excursion into nuclear physics. An atomic nucleus consists of protons and neutrons. The protons are electrically positively charged, with the same charge as an electron, but with a positive sign. The neutrons are electrically neutral. Protons and neutrons together are called nucleons. They are held together by the nuclear forces, which are much stronger than the electrostatic forces, but have a much shorter range. This is why the nucleons sit tightly packed together in the nucleus and do not fly apart, even though the protons repel each other electrically. The number of electrons in the shell around the nucleus is equal to the number of protons in the nucleus, unless the atom is ionized. Thus, an atom that is not ionized is electrically neutral. The electrons are responsible for the chemical reactions, and so is the number of protons. Therefore, a chemical element has always the same number of protons; this number is also called the atomic number in the periodic table. Atoms of the same element, however, can have a different number of neutrons. Chemistry does not notice this, and we have to use a mass spectrometer to detect a difference. The atomic weight is then approximately the number of nucleons times the average weight of a proton or neutron; we therefore refer to this number as the mass number and the average weight of a nucleon as the atomic mass unit (*AMU*).

The multitude of different atoms, taking into account the number of neutrons, are called nuclides. There are many more nuclides than chemical elements, whose periodic table easily fits on an A4 sheet of paper. For a nuclide chart, on the other hand, you would rather need a poster. All nuclides of an element, i.e. those with the same number of protons but different numbers of

neutrons, are called isotopes of the element. Many elements that we find in nature have several stable isotopes. For example, iron, with atomic number 26, comes in four different stable versions, with mass numbers 54 and 56–58. To accurately designate the nuclide, we write the atomic number at the bottom left of the chemical symbol and the mass number at the top left. So, for example, $^{54}_{26}\text{Fe}$ or $^{56}_{26}\text{Fe}$. We leave the right side to the chemists. Shortened, we can also write Fe-54 or Fe-56, because the atomic number is the same for all isotopes of an element and thus already determined by the element symbol.

Nuclides and Radioactive Decay

If you want to understand this book, you need a nuclide chart where you can look up every now and then how the nuclides behave physically. I still have one of those posters hanging on the wall in my office, but rather as a memento than anything else, because modern nuclide charts are more likely to be available as interactive software or online. The amount of information we can report about a nuclide doesn't fit on a poster anymore either.

A quite professional nuclide chart (and I don't want to recommend anything cheaper) is the *Isotope Browser* of the International Atomic Energy Agency IAEA. I would like to explain nuclear physics to you using this nuclide chart. So please download this app to your smartphone, for example from the Google Play Store. The app is free and does not contain advertising.

A nuclide chart shows all nuclides that have ever been observed. The horizontal axis shows the number of neutrons in the nuclide and the vertical axis the number of protons. A small section of the nuclide chart is shown in Fig. 2.1. The isotopes of an element each form a row of the nuclide chart. In the case of stable nuclides, the *abundance* is given, i.e. the percentage of this isotope occurring in nature. In the case of unstable nuclides, the *half-life* is given instead, i.e. the time that elapses until half of the remaining nuclide specimens have decayed.

After you have installed the app, you can first tap on *Chart*. Then the nuclide chart appears as a colorful band that stretches from the bottom left to the top right. For lighter nuclides, the number of neutrons is about the same as the number of protons, because nuclear forces bind both types of nucleons equally well. However, as the mass number increases, we see an increasing excess of neutrons, because the nuclear forces have a short range. Now the weaker repulsive electrostatic forces between the positive charges become increasingly noticeable, so that a neutron can bind to the nucleus more easily than a proton. The coloured band therefore bends a little to the right.

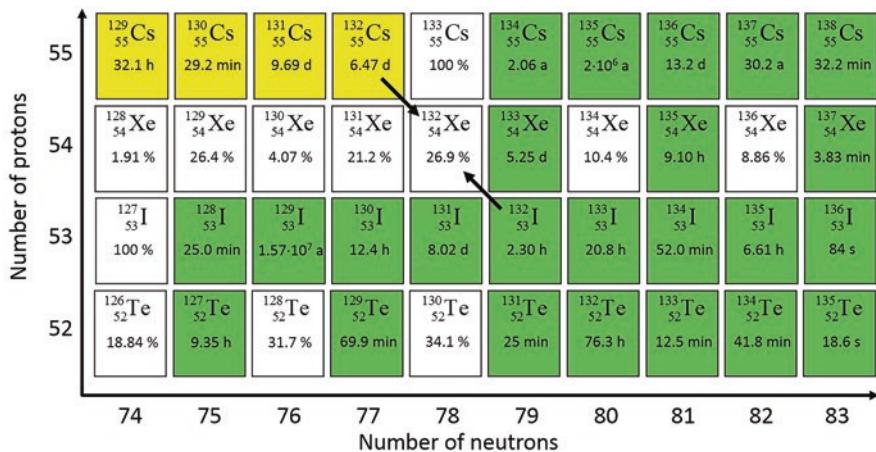


Fig. 2.1 Section of the nuclide chart; white: stable, yellow: β^+ emitter, green: β^- emitter

If you now zoom into the chart, you will see the stable nuclides in white and the unstable, i.e. radioactive, ones in colour. Think of a neutron as a combination of a proton and an electron; this is then electrically neutral. If a nuclide has too many neutrons, a neutron simply throws out a negatively charged electron and becomes a proton. We then speak of a β^- -decay. The Isotope Browser marks such nuclides in green. In the nuclide chart, we move then to the top left, e.g. from $^{132}_{53}\text{I}$ to $^{132}_{54}\text{Xe}$.

If you tap on a nuclide, you get all further information about the nuclide, e.g. how often it decays and in what way, and what radiation is emitted in the process. Similar to the electrons of the shell, the nucleons sit in certain places with very specific energy levels, the quantum states. If a neutron becomes a proton, the nucleus must first shake down to a stable state. During each of these changes in its quantum state, a γ -quantum is emitted, called a photon, just as the electron shell emits a much lower-energy quantum of light when the electrons fall down to other quantum states. The β^- -decay is therefore often accompanied by γ -radiation. The β^- -particles, or electrons, usually fly only a few mm before being picked up by a neighbouring atom. The γ -rays, on the other hand, can travel a few m. We therefore need heavy metals, e.g. lead plates, to protect us from γ -radiation.

Sometimes, if a nuclide has far too many neutrons, it throws out an entire neutron. In the Isotope Browser, such decays are denoted by $\beta^- n$, e.g. in the case of iodine $^{137}_{53}\text{I}$ or tellurium $^{136}_{52}\text{Te}$. Then an electron and a neutron are emitted simultaneously. These nuclides have only a short half-life of a few

seconds, but their neutrons have a great importance for control of a reactor, as we will see later.

The resulting daughter nuclide is often unstable itself. It continues to decay until a stable nuclide is created. The Isotope Browser shows us this *decay chain* in the nuclide chart when we tap on it. Likewise, we can also find out the parents from which this nuclide may have originated.

The decay energy is expressed in keV or MeV. An electron volt (eV) is the energy which an electron has after being accelerated by a potential difference of 1 volt. Since we know the elementary charge and thus the charge of an electron, namely $1.602 \cdot 10^{-19}$ C, we can convert this energy unit into joules: $1 \text{ eV} = 1.602 \cdot 10^{-19} \text{ J}$. This is very little energy, but we are only talking about a single electron.

If a nuclide has too many protons, it has two options: Either it catches an electron from the shell and thus a proton becomes a neutron; the Isotope Browser labels this as *electron capture* (ec). Or, the proton throws out a positive anti-electron, that is, a positron with a positive elementary charge, and thereby becomes a neutron; the Isotope Browser labels this as β^+ -decay. Both of these are labelled in yellow. We get then to the bottom right on the nuclide chart, e.g. from $^{132}_{55}\text{Cs}$ to $^{132}_{54}\text{Xe}$. The positron doesn't get very far. If it hits an electron, only two γ -quanta remain. On the basis of these two γ -quanta, in contrast to the X-ray quanta of the electrons during electron capture, the physicist can clearly distinguish between the two types of decay.

Above lead, that is, above an atomic number of 82, all nuclides are unstable. Even the bismuth isotope $^{209}_{83}\text{Bi}$, which we had for a long time considered as stable, is unstable on close examination. However, with a half-life of $2 \cdot 10^{19}$ years, and that is practically as good as stable. A common decay mode of these heavy nuclides is α -decay. In the Isotope Browser, these nuclides are marked in pink. In this case, the nucleus throws out an entire helium-4 nucleus, i.e. a particularly stable unit consisting of two neutrons and two protons. In the nuclide chart, we move then two steps to the left and two steps downwards. The α -particles have a range of a few μm only. Usually, they catch immediately two electrons from neighbouring atoms, becoming helium atoms. Therefore, in the long run, the material around the α -emitters forms helium bubbles, which we can observe in the electron microscope.

Among the heavy nuclides are two peculiarities: uranium and thorium. All uranium isotopes are unstable, but the isotopes $^{238}_{92}\text{U}$ and $^{235}_{92}\text{U}$ decay so slowly that we still find them in nature. That's why the nuclide chart shows both, an abundance and a half-life. If we assume that these two isotopes were equally present when our solar system was formed, we can infer the age of our solar system from the ratio of the abundance of these isotopes: less of the

faster decaying isotopes remained today. The thorium isotope $^{232}_{90}\text{Th}$ produced in the process is also still quite stable, with a half-life of $1.4 \cdot 10^{10}$ years, and is therefore found rather frequently.

When we walk through the Black Forest (Germany) with a Geiger counter, we find sometimes pitchblende, a uranium-bearing rock, and in it traces of $^{234}_{92}\text{U}$, radium ($^{226}_{88}\text{Ra}$) and sometimes radon ($^{222}_{86}\text{Rn}$). These are nuclides along the decay chain of $^{238}_{92}\text{U}$. If you tap for $^{238}_{92}\text{U}$ on the decay chain in the Isotope Browser, you can follow how it finally becomes the stable lead isotope $^{206}_{82}\text{Pb}$ over millions of years. Consequently, radium and radon are constantly replicated, so they don't seem to decay.

Sometimes there are two isotopes with the same mass number and the same atomic number in the nuclide chart, e.g. the bismuth isotopes $^{210}_{83}\text{Bi}$ and $^{210m}_{83}\text{Bi}$. The small "m" after the mass number indicates a metastable state. The nucleons of both isotopes sit then on different quantum states and the isotopes decay accordingly quite differently.

The Isotope Browser gives us also the binding energy per nucleon of a nuclide (*Binding energy/A*). For uranium, this binding energy is lower than for the lighter nuclides, and stable nuclides have the highest binding energy per nucleon. From the difference, we can calculate, in principle, how much energy is released when we fission uranium, provided we know the fission products. But that depends entirely on how a neutron happens to hit the uranium nucleus. On average, we get 193 MeV per fission plus the energy of some neutrinos, but they escape quickly. Their energy is technically unusable. Converted into energy units, which are more descriptive for us, 1 g of $^{235}_{92}\text{U}$ produces an energy of about 24 MWh, or 1 megawatt-day (MWD), when we fission it. That is as much as 2600 l of petrol if we burn it in the engine.

The average yield of a certain fission product, which is typically produced during nuclear fission, in relation to all fission products produced, is given to us by the Isotope Browser as the *Fission Yield (FY)* for each nuclide. For example, we find a high fission yield of more than 5% for nuclides with a mass number of about 140, i.e. in the vicinity of iodine or caesium, or with a mass number of about 90, i.e. in the vicinity of strontium or yttrium. So two different sized fragments are usually produced, a smaller one and a larger fragment. On the other hand, two fragments of the same size, each with a mass number of about 118, are rather rare. Because uranium had a considerable neutron excess before fission, the fission products are always nuclides with a high neutron excess. For these smaller nuclides, however, this excess is quite unstable, and so the fission products decay rapidly by β^- -decay or even with emission of neutrons to more stable nuclides. Only when a nearly stable

nuclide has been reached, the decay proceeds more slowly until finally a stable nuclide is produced.

Energy is released in every decay process. If we convert the electron volts given by the nuclide chart into joules by a factor of $1.602 \cdot 10^{-19}$ and multiply them by the number of decays per second, we get a heating power. Compared to the heating power of nuclear fission, this is only about 1–2%, but this heating cannot be switched off. If, for example, a reactor had a heat output of 3000 MW during operation, the fission products will consequently still have an output of 30–60 MW after shutdown. That's enough to melt down the whole reactor core if we don't cool it. What decays faster produces more heat, but is also gone faster. Therefore, the residual heat from the fission products decays over time. After a year, it is only about 0.2% in the pressurized water reactor.

Now that we have spent some time browsing the nuclide chart, we can draw the following conclusions:

- Fission products are radioactive; mostly β^- and γ -emitters, sometimes also neutron emitters. We have to shield them to protect ourselves from them. Uranium is comparatively harmless because it decays extremely slowly, so that it hardly radiates at all. Don't be afraid to touch a fresh fuel assembly with enriched uranium, if you ever get the chance. This also makes it easier to manufacture fuel assemblies.
- The radioactive fission products produce heat that cannot be switched off. Therefore, every reactor needs an emergency cooling system in case that the power supply fails. The various fourth-generation reactor designs have quite individual solutions for this. When we talk about the safety of a reactor, we usually mean how reliable this emergency cooling system is. But there are also quite different safety requirements, as we will see in the next chapter.

Nuclear Reactions of Neutrons

Neutrons can fission uranium. So they play an important role in the design of a reactor. But first let's take a look at what else neutrons do in a reactor.

The nuclear reactions of neutrons can be easily measured. To do this, we first need a neutron emitter. In Otto Hahn's time, a mixture of radium sulphate and beryllium sulphate was used for this purpose. The stable $^{9}_{4}\text{Be}$ absorbs the α -particles of $^{226}_{88}\text{Ra}$ and forms ordinary carbon $^{12}_{6}\text{C}$, emitting a neutron. Today, we get neutron beams more effectively if we go to a research

reactor with a nuclide sample and get a laboratory place reserved for us at one of the beam tubes that run out of the research reactor into the laboratory hall.

The reactor can provide us with neutrons with a broad energy spectrum. They are produced by nuclear fission and initially have an energy of about 1–5 MeV. They are called the prompt neutrons of nuclear fission. As they pass through a water bath, they are gradually slowed down by collisions with hydrogen atoms. This is called “moderation” and the water accordingly a moderator. At an energy of about 25 MeV, which is almost 100 million times less, they just fly around like gas atoms. Temperature alone then keeps them moving, and we can easily calculate their average energy according to the kinetic theory of gases as $(3/2) \cdot kT$, where k is Boltzmann’s constant and T is the absolute temperature (in Kelvin). Such slow neutrons, whose energy depends only on the temperature, are called thermal neutrons. In order to slow down the neutron energy to near zero, we would therefore have to moderate them in a bath of liquid helium with almost 0 K.

The experiment on the beam tube is similar to an optical experiment where we hold a glass plate into a beam of light and then use a prism to separate the light into its spectrum. We find then some black lines between the bright colours of the spectrum. These are the absorption lines of the electrons of our glass plate. When the energy of the incident photon is just enough to raise an electron to a higher quantum state, the photon is absorbed, leaving a local minimum in the light spectrum. From the position of the black lines in the spectrum, we can tell which chemical elements our glass is made of.

The neutron detector, which is located at the beam tube behind the nuclide sample, measures a similar effect. As an example, we see in Fig. 2.2 the spectrum of neutron absorption of a sample from $^{238}_{92}\text{U}$. On the x -axis the neutron energy is plotted logarithmically, otherwise we would not be able to see anything in this huge energy spectrum. On the y -axis the cross section of the neutron absorption is plotted, also logarithmically. If we multiply this cross section σ_a by the neutron flux Φ , i.e. by the number of neutrons passing through a surface in a time interval, and by the number of uranium nuclei $N_{\text{U-}238}$ in a sample volume, we get the number of reactions R , in this case the absorptions per second in this sample:

$$R = \sigma_a \Phi N_{\text{U-}238}.$$

The cross-section of a reaction is usually given in units of barn; that is 10^{-24} cm^2 . The cross-section of each reaction is a characteristic material property of the respective nuclide, which can be looked up in databases. It is

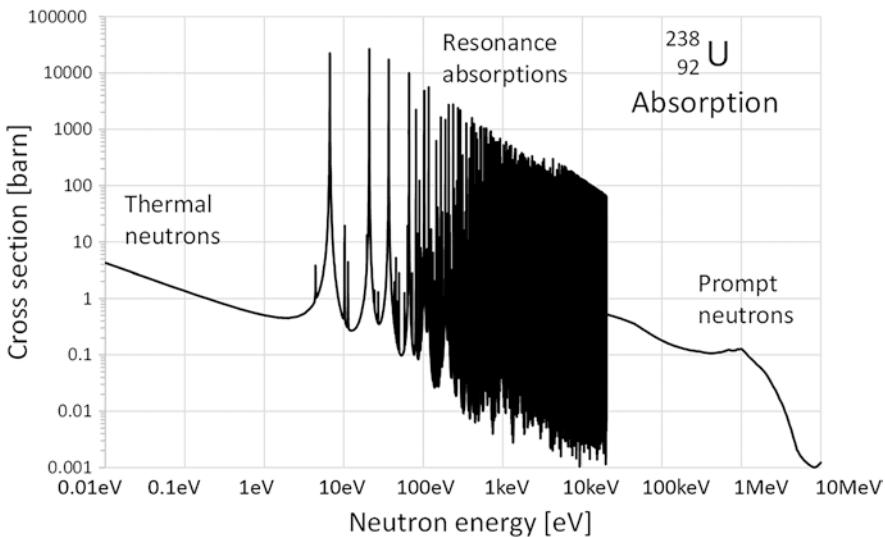


Fig. 2.2 Spectrum of the cross section σ_a of neutron absorption in $^{238}_{92}\text{U}$

independent of the experimental setup. You can find this data, for example, in the table of nuclides of KAERI [3].

The many maxima in the energy range from 5 eV to 20 keV are the black lines in the neutron spectrum. We call them resonance absorptions of the uranium. Only when the energy of the neutron is just sufficient to lift the neutron in the nucleus to a certain energy level is it absorbed. After that, the nucleons in the nucleus shake themselves back into place until a stable arrangement is reached, associated with corresponding γ -emissions. The database therefore calls the process an (n, γ) -reaction.

After absorbing a neutron in $^{238}_{92}\text{U}$, the uranium isotope has become $^{239}_{92}\text{U}$. However, the new, heavier nuclide is quite unstable, and it decays with a half-life of 23.5 min by a β^- -decay into the neptunium isotope $^{239}_{93}\text{Np}$. This nuclide is also still quite unstable. With a half-life of 2.4 days, it decays to $^{239}_{94}\text{Pu}$ (plutonium), which, with a half-life of 24,000 years, can almost be considered as stable.

The lighter nuclide $^{235}_{92}\text{U}$ absorbs neutrons even better. However, we see a big difference between the two isotopes $^{238}_{92}\text{U}$ and $^{235}_{92}\text{U}$ in the reaction “fission” (Fig. 2.3). The fission cross section of $^{238}_{92}\text{U}$ is almost negligible. Nuclear fission produces two fission products and two to three prompt neutrons with an energy greater than 1 MeV. Uranium-235 is fissioned about 500 times better by thermal neutrons than by prompt neutrons, as we can easily see in

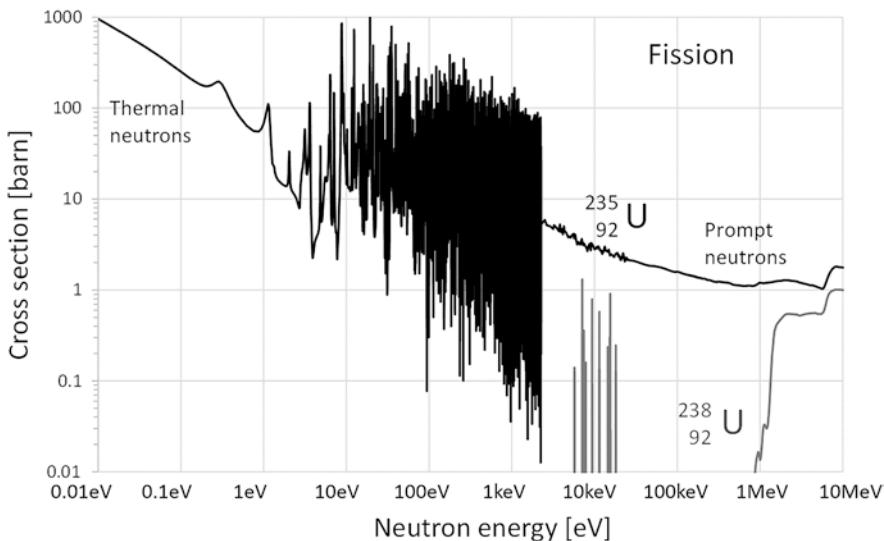


Fig. 2.3 Cross sections of nuclear fission σ_f of ($^{235}_{92}\text{U}$ black) and ($^{238}_{92}\text{U}$ grey)

Fig. 2.3. Therefore, in order to use the neutrons most effectively for nuclear fission, we moderate the prompt neutrons of nuclear fission until they become thermal neutrons. A reactor that uses this principle is called a thermal reactor. We can calculate its average power density q , i.e. the thermal power per volume element, in a simplified way from the thermal neutron flux Φ , the average density of $^{235}_{92}\text{U}$ -atoms, $N_{\text{U-}235}$, and the cross section σ_f of nuclear fission of thermal neutrons as

$$q = \sigma_f \Phi N_{\text{U-}235} E_f.$$

E_f is the energy per fission, i.e. the 193 MeV, which we already mentioned in the last chapter. The power density is therefore directly proportional to the neutron flux.

The chain reaction of neutrons in a thermal reactor is shown schematically in Fig. 2.4. It proceeds as follows: When a thermal neutron hits a uranium-235 core, two fission products and two to three prompt neutrons are produced. Some of these prompt neutrons escape; we refer to this as the neutron leakage. The remaining neutrons are decelerated in the moderator until they are thermal, and then they fission again. So if we change the neutron leakage or the absorption, we can influence whether the number of neutrons keeps increasing in the process, decreases, or stays constant on average. In physics, we

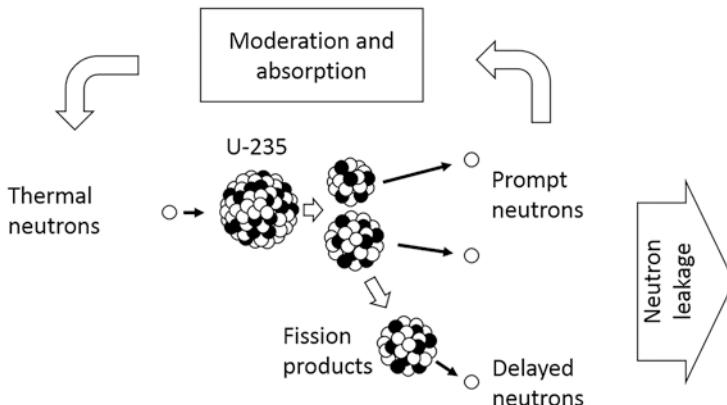


Fig. 2.4 Schematic of the chain reaction of nuclear fission

express this by the effective multiplication factor k_{eff} . However, the cycle lasts only about 10–100 μs , which is much too fast to be able to control the neutron flux reasonably.

Some of the fission products emit neutrons as well during their decay. With an energy of some 100 keV, these neutrons are almost as fast as the prompt neutrons. However, it takes a few seconds for these neutrons to be emitted, depending on the half-life of these fission products. We therefore call them delayed neutrons. Relative to the prompt neutrons, they are much rarer, accounting for less than 1%, but they are extremely valuable for controlling the neutron density. We call this fraction β . We then control the reactor in such a way that the number of neutrons keeps increasing only with the help of the delayed neutrons. The delayed neutrons can escape or be absorbed in the same way as the prompt neutrons, and when we change the neutron leakage or absorption, we consequently change the neutron flux very gradually.

If we adjust the neutron absorption so that the neutron flux remains constant in time, the reactor is said to be critical. If the neutron flux can only increase with the help of the delayed neutrons, it is called delayed supercritical, and if the prompt neutrons alone cause the neutron flux to increase, the reactor is called prompt supercritical. Of course, we'd better not do that. That would be too fast. The multiplication factor should therefore not be much greater than one.

In the literature on reactor physics, you will often find the term “reactivity”, usually referred to as ρ . This refers to the relative deviation of the multiplication factor k_{eff} from the number 1:

$$\rho = \frac{k_{\text{eff}} - 1}{k_{\text{eff}}} .$$

If $\rho = 0$, the reactor is critical. If $\rho > 0$, the reactor is supercritical, and for $\rho > \beta$, the reactor is prompt supercritical. So now, we have also a measure of when the reactor is getting prompt supercritical.

In Stuttgart, there is a small teaching reactor of the type SUR 100, which works exactly according to this principle. I have often operated it with some students and was always amazed at how easy it is to produce a self-sustaining chain reaction of nuclear fission, compared to a self-sustaining nuclear fusion, for which one needs this gigantic tokamak reactor, which is being built in Cadarache in southern France since quite some time. I have sketched the principle of this teaching reactor in Fig. 2.5: Two polyethylene blocks containing finely dispersed enriched uranium must first be moved together to

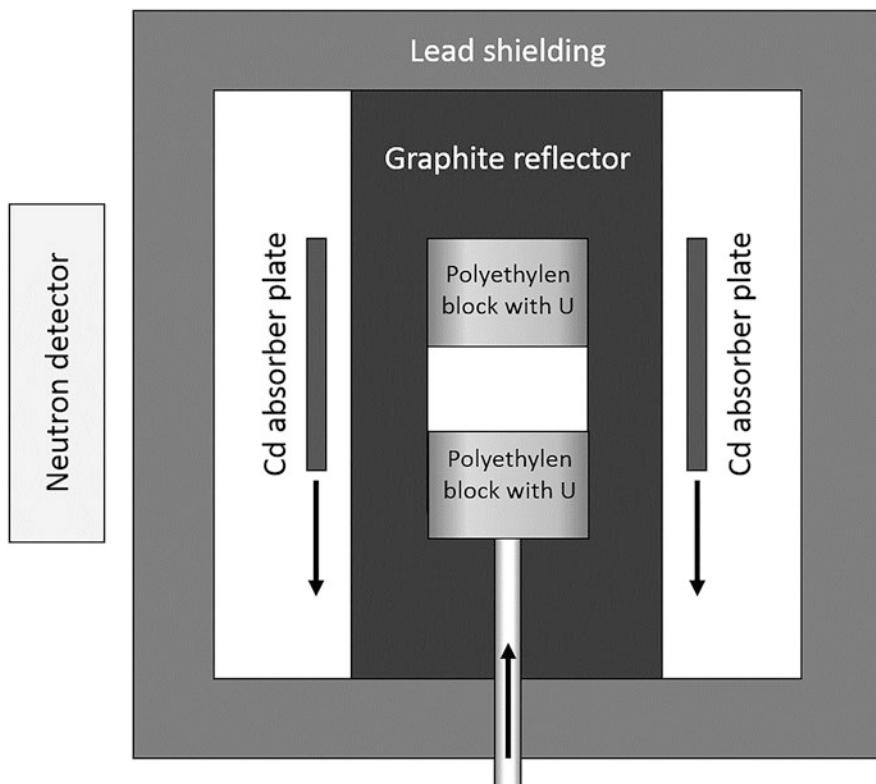


Fig. 2.5 Sketch of the teaching reactor SUR 100

reduce the neutron leakage. A small neutron source consisting of an α -emitter and beryllium is driven underneath the blocks to start the chain reaction. Next to the blocks are cadmium plates as neutron absorbers, which can be moved out until the reactor becomes delayed supercritical. Now, the neutron density increases slowly and we measure an increasing neutron flux in a neutron detector located next to the reactor core. The moderation here is done by the hydrogen atoms in the polyethylene and by the carbon atoms in the graphite reflector. The power of the teaching reactor is less than 100 mW, so we don't need cooling or emergency cooling. This kind of reactor is also called a "zero power reactor", but strictly speaking, of course, the power cannot be "zero". If the neutron flux in the detector, and thus the power, gets too high, the bottom block falls down with a loud "plock" and the students can restart the reactor again. This was then a reactor scram, as we call the fast shutdown.

It is not so easy to run this teaching reactor exactly critical. Depending on the position of the cadmium plates, the neutron flux either increases or decreases slowly, and one hardly manages to keep the neutron flux constant over a longer period of time. This is quite different in a power reactor. As soon as the temperature increases with increasing power or the density of the moderator decreases for other reasons, there are physical effects that control the neutron flux by themselves.

The first, important effect comes from the absorption of neutrons in uranium. The resonance absorption, that is, the absorption in the many peaks in the absorption spectrum in Fig. 2.2, increases as the uranium becomes warmer. In general, the probability of absorption is highest when the neutron brings exactly the energy needed to occupy a particular energy level. When the atom is warm, it vibrates in the crystal lattice. If it moves then towards the neutron, the relative velocity between the neutron and the atom is higher than if the atom moves away. Then the kinetic energy of the neutron is also higher. Therefore, the chance that the incident neutron will bring just the right energy with it becomes higher the warmer the atom is and the more it oscillates. The resonance peaks become wider then. This effect is comparable to a moving sound source. If you can only pick up sound at a very specific frequency, the faster the sound source moves back and forth, the more likely it is that the sound source is just at the right frequency. If the sound source is coming towards you, the frequency is higher, otherwise it is lower. By analogy with acoustics, this effect is therefore also called the Doppler effect. In the reactor, this effect means that neutron absorption increases as the uranium gets warmer. So if the neutron flux increases and consequently the power of the

more frequent nuclear fissions increases, this effect controls the neutron flux back again all by itself.

The second effect is due to the temperature of the moderator. As a general rule, the density of the moderator decreases when it becomes warmer. Then the collisions between the neutrons and the light atoms of the moderator become less frequent. As a result, the moderation becomes worse. So fewer thermal neutrons are produced and the neutron leakage increases instead. We call this effect the density effect. This effect also controls the neutron flux back on its own when the power of the reactor increases, because the moderator becomes warmer as a result.

The third effect is the void effect. Void is the term used to describe an empty space. This refers, for example, to the volume of the steam bubbles of a water-cooled reactor. When the cooling water boils, which is also the moderator in this type of reactor, part of the moderator is missing because there are comparatively few water molecules in the steam bubbles. As a result, moderation deteriorates, fewer thermal neutrons are produced, and neutron leakage in turn increases. The void effect therefore controls the neutron flux in a similar way as the density effect. It decreases the neutron flux when the power of the reactor increases, provided that the coolant boils.

Imagine we are running a water cooled reactor at a constant power. Now we take out a neutron absorber a little bit. This will increase the chain reaction of nuclear fission. The reactor is now delayed supercritical. Due to the increasing neutron flux, there are now more nuclear fissions, thus the power increases and the temperature increases in the uranium as well as in the water. The Doppler effect, the density effect and possibly the void effect, if the water is boiling, control however the neutron flux so far back again, until a new condition with constant power is reached. This time with a slightly higher power, with warmer water and with warmer uranium. So steady state operation is a stable state, and the reactor finds to that state all by itself. Unlike a zero power reactor, we don't have to control it at all. The question is justified why we denote a steady-state operation as "critical". This can only be understood historically.

The more we take out the neutron absorber, the higher the reactor power and the hotter the reactor. So we can adjust the power with the neutron absorbers. That's why we call the absorber rods control rods.

These three effects are not only practical for the operation of the reactor. They are also enormously important for its safety. As a reactor operator, you learn that you must never run a reactor prompt supercritical, and the control rod drives don't even allow that. If, for some silly mistake, it happens anyway, these three effects help to avoid a disaster. A reactor in which one of these

three effects is positive, such that the neutron flux increases when the power increases, should better not be allowed for safety reasons.

The reactor accident in Chernobyl can serve as a bad example here. In order to explain this accident sequence to you in more detail, I must first tell you about a fourth effect: the xenon effect.

The xenon isotope $^{135}_{54}\text{Xe}$ is produced by β^- -decay of the iodine isotope $^{135}_{53}\text{I}$, a fairly common fission product with a half-life of 6.6 h. You will find these two nuclides on the right in the section of the nuclide chart in Fig. 2.1. Xe-135 is an extremely good neutron absorber. With an absorption cross-section of 2,650,000 barn, it absorbs thermal neutrons about a million times better than U-238, so that even small amounts of this nuclide can powerfully alter the neutron balance. Once it has absorbed a neutron, it becomes the isotope $^{136}_{54}\text{Xe}$, which absorbs hardly any neutrons. On the other hand, $^{135}_{54}\text{Xe}$ can also decay. With a half-life of 9.1 h, it becomes the caesium isotope $^{135}_{55}\text{Cs}$. If we operate the reactor, an equilibrium is reached after a few hours between the production of Xe-135 by decay of I-135, by the consumption by neutron absorption and by the decay of Xe-135. However, if we shut down the reactor, one of these three processes ceases: neutron absorption, because there are no more neutrons to be absorbed. Consequently, the concentration of Xe-135 then increases as I-135 continues to decay forming Xe-135, but Xe-135 itself decays relatively slowly. About 10 h after shutdown, Xe-135 has reached its maximum concentration. It takes about 1.5 days for the Xe-135 concentration to return to pre-shutdown levels. During this time, the reactor should not be restarted. The reactor operators say then “we are stuck in the xenon trap”. If you still want to restart the reactor, you would have to move all the absorber rods out to become supercritical, because Xe-135 catches most of the neutrons. But then something surprising happens: as soon as the neutron flux increases again, Xe-135 is also consumed again. As a result, the Xe-135 concentration falls, fewer neutrons are absorbed, and the neutron flux continues to rise. So this is an effect that accelerates itself, sometimes faster than one can retract the absorber rods.

The RBMK-type reactor that exploded at Chernobyl on April 26, 1986, is a pressure-tube reactor [4], outlined in Fig. 2.6. Vertical pressure tubes containing the uranium fuel are located in a large graphite block that serves as a moderator. Water evaporates inside the pressure tubes, cooling the fuel assemblies. Steam and water from the pressure tubes are first separated in steam separators and the steam is then fed to a turbine that drives a generator. So the cooling and moderation functions are separated in this reactor: graphite for moderation and boiling water for cooling. At low power, when the water is just starting to boil, the liquid water absorbs some of the thermal neutrons

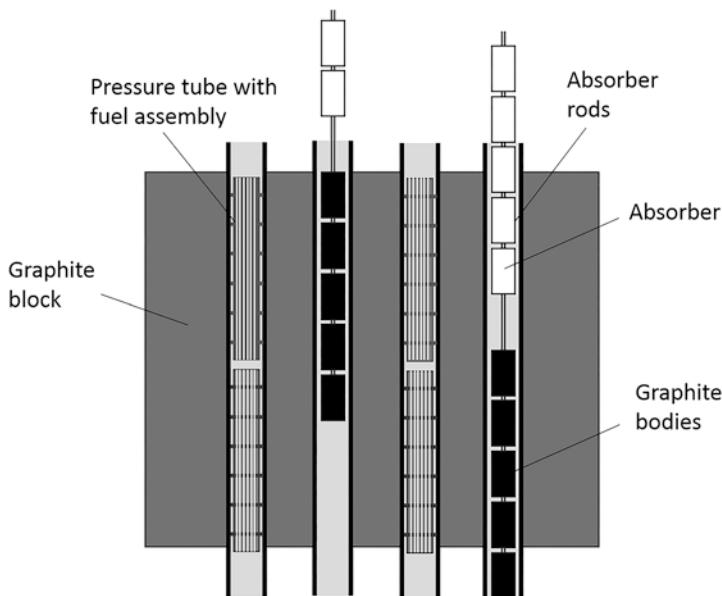


Fig. 2.6 Schematic diagram of an RMBK reactor

that have been slowed down in the graphite. Moderation by this water, on the other hand, is small and the graphite block prevents an increase in neutron leakage when steam bubbles form. This absorption becomes weaker as more steam is in the pressure tubes. This gives the reactor a positive void effect at low power: the more steam is produced, the higher the thermal neutron flux in the uranium becomes. But this goes straight in the wrong direction, unlike, for example, the pressurized water reactor, which has a negative void effect.

On the day of the accident, after a few hours at full power, the reactor operators reduced the reactor first to half power, and then finally to about 22% of power, just for testing. This caused a lot of Xe-135 to form, which increased neutron absorption, and the power continued to reduce on its own down to 1%. It would have made sense to shut down the reactor now. However, it was started up again and all control rods had to be pulled out. Due to the renewed neutron flux, the Xe-135 concentration fell again, which caused the neutron flux to increase even further. This was now compounded by the positive void effect, which also increased the neutron flux as the power increased. As a result, the reactor became prompt supercritical even before the shutdown rods could be retracted. Graphite bodies at the tip of these shutdown rods even improved moderation and further increased power before the absorbers took effect. The pressure tubes then exploded at about 100 times

the reactor's rated power. The graphite block ignited and the fire drove the released fission products to great heights. The further consequences are known. The UNSCEAR committee [5] carefully followed up all personal injuries for 20 years and reported on them in a thick report. In total, fewer people died than in a plane crash, far fewer than had been feared at the time, but the damage was still enormous. As a result, many people lost confidence in nuclear technology.

The xenon effect was certainly known to the reactor operators. So ignoring it was a stupid operational mistake. They should not have restarted the reactor that day. But the positive void effect is rather a design flaw in this reactor that we must not allow in fourth generation reactors. The reactor design must also allow prompt supercritical conditions without a catastrophic consequence.

Nuclear Fuel Fabrication

By nuclear fuel we mean enriched uranium in conventional reactors. I don't think I need to mention that uranium does not "burn" in the reactor, but it is fissioned. Chemically, the uranium in the fuel assembly is in the form of uranium dioxide, UO_2 . This is a ceramic material with a melting temperature of about 2800 °C. It can therefore be heated to a very high temperature in order to transfer the heat to the surrounding cooling water.

We find uranium in almost every country in the world. There are even almost unlimited uranium deposits dissolved in seawater. However, the concentration is usually so low that it is expensive to extract this uranium. The "Red Book" of the NEA and IAEA [6], which regularly reports on the occurrence, extraction and demand of uranium, therefore distinguishes uranium deposits according to cost groups. In 2015, the world's developed and secured uranium resources up to 130 USD/kg_{uranium} amounted to about 3.5 million tons. If we add the uranium resources known beyond that, these resources are enough to secure today's uranium consumption for more than 135 years. It therefore makes little sense to speculate on this.

Uranium ores used to be produced in mines, sometimes also in open-cast mines. The ore was finely ground and the uranium was then dissolved out with sulphuric acid or nitric acid. Nowadays, however, the uranium is more likely to be dissolved out directly on site. This is done by pumping carbonates or acids into boreholes, which leach the uranium directly out of the rock there. It is then precipitated with ion-exchange resins. After precipitation with ammonia, a brown or black powder is produced, which is sold under the name *Yellow Cake*, because it originally looked mostly yellow. It contains

uranium oxide, predominantly as U_3O_8 . At the beginning of 2019, the price of yellow cake was around USD 80/kg.

$\text{U}-235$, which is fissile, which means that it can be fissioned with thermal neutrons, is present in it at only 0.72%. Although there are some reactors, so-called pressure tube reactors, which can be operated with this low concentration, it is more economical today to enrich the uranium-235 to a higher concentration. Chemically, $\text{U}-235$ and $\text{U}-238$ are identical, and so the only way to separate the two isotopes is to distinguish the individual atoms based on their mass. This would require uranium as a gas, but uranium vaporizes only at extremely high temperatures. Instead, uranium hexafluoride, UF_6 , is produced. UF_6 vaporizes at ambient pressure at 56.4°C , so it is already gaseous at reasonable temperatures. At lower temperatures, UF_6 is solid, so that it can be easily stored. Each UF_6 gas molecule contains exactly one uranium atom. Fluorine occurs in nature with only one stable isotope, so we do not run the risk of enriching only fluorine with heavier fluorine isotopes when separating the gas molecules.

By far the most common method of enrichment today is the gas centrifuge. Figure 2.7 shows a simple sketch of this process. UF_6 gas is fed into the centre of tall, vertical cylinders. The cylinders rotate as fast as possible without breaking apart, and the molecules with the heavier isotope $\text{U}-238$ separate by

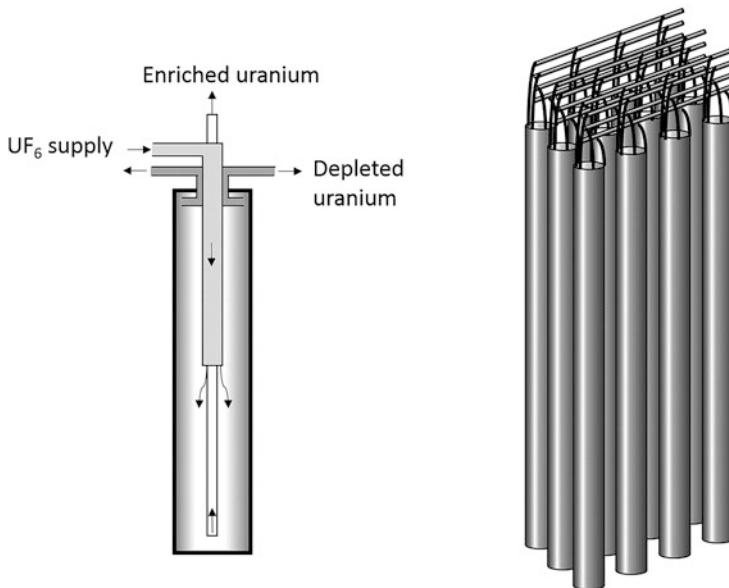


Fig. 2.7 Schematic of uranium enrichment with gas centrifuges; left: Principle of a single gas centrifuge; right: Section of an enrichment cascade with gas centrifuges

centrifugal force mainly on the cylinder wall, where they are extracted, while those with the lighter isotope U-235 are more likely to be found in the centre of the cylinder at the bottom. The separation factor of a single stage is about 1.4, which means that the gas extracted in the centre contains 40% more U-235 than the gas extracted at the rim. For example, to obtain an enrichment with U-235 of 3.5%, about ten of these separation stages have to be connected in series. This is called an enrichment cascade. The enriched mass flow is passed to the next downstream stage for further enrichment, while the depleted mass flow is returned to the upstream stage for co-processing. Thus, at the inlet of the cascade at the cylinder wall, we get a depleted uranium with a U-235 concentration of about 0.2%, and at the exit of the cascade, in the centre of the cylinder, we get an enriched uranium with 3.5% U-235. The mass flow going through each centrifuge is quite small; therefore, many of these centrifuges have to be connected in parallel.

A simple mass balance shows that we need an input mass of 6.3 kg of natural uranium to produce 1 kg of uranium with an enrichment of 3.5%, producing 5.3 kg of depleted uranium with a residual concentration of U-235 of 0.2%. So the enrichment process produces quite a bit of depleted uranium that we could use as well for power production. But we will come to that later.

In the next step, UF_6 must be converted into uranium dioxide UO_2 . To do this, UF_6 is dissolved in water, forming hydrofluoric acid, and neutralized with ammonia. The resulting UO_3 is then reduced with hydrogen to UO_2 . Pure UO_2 is a black powder that is pressed into pellets, which are then sintered at 1700 °C to become more compact and solid. Finally, the pellets are ground to the exact diameter required so that they fit precisely into the cladding tubes of the fuel assembly.

The final UO_2 fuel today costs about €1 per MWh of heat that can be generated from it. That is 10 times cheaper than coal and 25 times cheaper than natural gas. Enrichment accounts for only about $\frac{1}{3}$ of the cost. In addition, there are the manufacturing costs for the fuel assembly, for disposal and some fixed costs that are independent of enrichment, in total of the same order of magnitude. Therefore, it is understandable to try to get a little more U-235 into the fuel assembly. Today's enrichment plants for conventional nuclear fuel are designed for an enrichment up to 5% and the new nuclear power plants can also use this fuel.

Conventional Water-Cooled Reactors

Actually, I didn't want to talk much about conventional reactors in this book. However, I should perhaps report a few basic things, otherwise it will not be clear afterwards what is different about the fourth generation.

Pressurized Water Reactor

By far the most frequently built reactor type is the pressurized water reactor. With this reactor, water is heated by just 30 °C at approx. 155 bar, from approx. 290 °C to approx. 320 °C. At this pressure, the water would boil at 345 °C. Therefore, at the exit from the reactor core, it is still too cold for boiling. Figure 2.8 shows a sectional view of such a reactor. The reactor pressure vessel is about 12–13 m high and about 5 m in diameter. The fuel assemblies

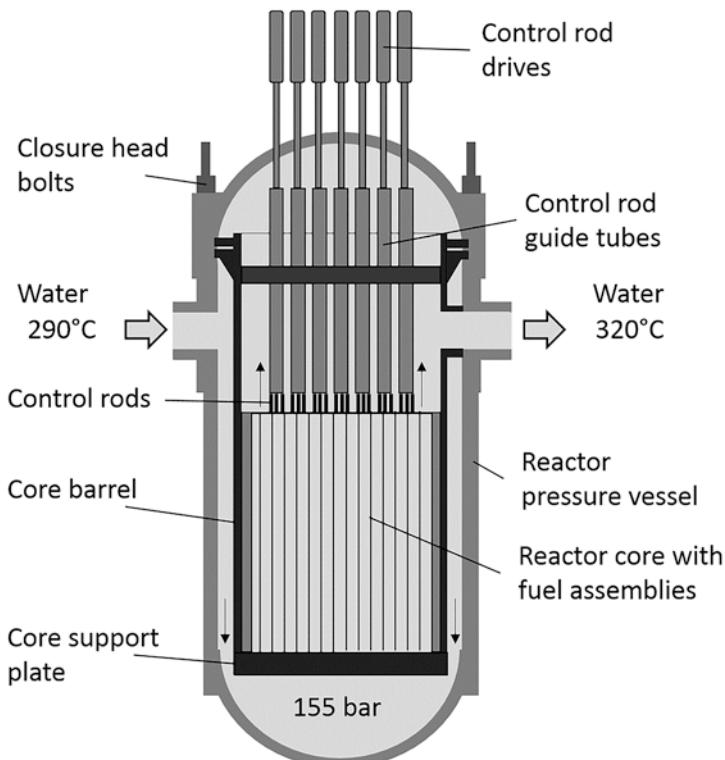


Fig. 2.8 Principle sketch of a pressurized water reactor

stand on a thick core support plate in the lower half of the reactor. It forms the bottom of the core barrel, which is suspended in the pressure vessel. The core support plate is highly permeable to water. The slightly colder water that enters the pressure vessel through the inlet flanges, on the left in our picture, flows first around the core barrel, then through the core support plate, and from there upward through the fuel assemblies of the core. Above the core, the heated water is then collected and discharged to a steam generator via the outlet flanges. About every third fuel assembly has some control rods inside that we can use to control neutron absorption. They are connected to drive rods that run out the top of the reactor closure head. A magnetic control rod drive above the pressure vessel can move these control rods in or out incrementally. When the power is switched off, the control rods simply fall down under their own weight and the reactor shuts down.

A fuel assembly of the European Pressurised Water Reactor “EPR” consists of 265 fuel rods each with a diameter of 9.5 mm, arranged in a square with a pitch of 12.5 mm and with 24 control rods in every third fuel assembly. The cladding tube is made of Zircaloy, an alloy primarily of zirconium with small amounts of other metals. Zircaloy looks similar to stainless steel and is also similarly corrosion resistant. Unlike stainless steel, however, zirconium absorbs hardly any neutrons and is therefore downright transparent to neutrons. The height of the fuel column made of the UO_2 pellets in a fuel rod is 4.2 m. The fuel rod can generate up to 40 kW of heat per meter. For comparison: a powerful immersion heater has only about 2–3 kW of power. At this power, the pellet is white-hot inside, like a lightbulb, and the water boils at the cladding tube surface, even if it is still subcooled at a distance of a few mm from the cladding tube. The average power density q in a pressurized water reactor reaches up to 100 W/cm³ or 100 MW/m³ expressed in larger units.

The heated high-pressure water is cooled back down to 290 °C in steam generators. These are large vessels with tube bundles located next to the reactor, through which the hot water flows. Figure 2.9 shows that these steam generators are even much larger than the reactor. On the outside of the pipes, inside the steam generators, is water at a lower pressure of about 70 bar, which boils at 286 °C already. This is how we generate the steam that we need to drive the steam turbines. The steam generator has many droplet separators and a steam dryer in the upper half for this purpose. To pump the hot high-pressure water back into the reactor pressure vessel, there are large coolant pumps in the return line.

Finally, this closed high pressure system needs a pressurizer. This is a surge tank like the one I have at home for my heating system, only much larger and adjustable. The pressurizer is filled half with water and half with steam. This

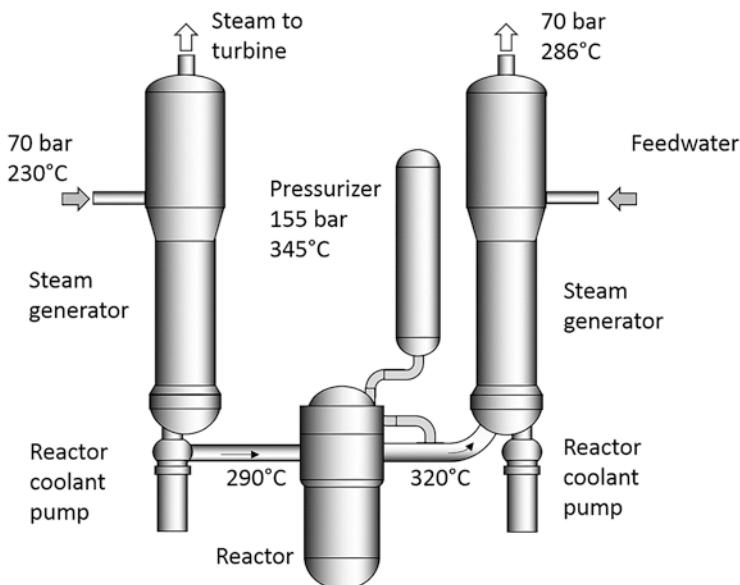


Fig. 2.9 High pressure system (primary system) of a pressurized water reactor

water must be heated up to 345 °C, otherwise there is no steam. If this water is made a little hotter, the steam pressure increases and with it the pressure in the reactor pressure vessel. If you let it cool down or perhaps even spray colder water into it, the pressure drops.

Once a year, the pressure vessel is opened in order to shuffle the fuel assemblies and to fill in new ones. To do this, the reactor is first allowed to cool down. Then the space above the reactor in the power plant has to be completely flooded with water, because the old fuel assemblies still produce residual heat. They have to be cooled permanently, and their γ -radiation would be too dangerous for people on the loading machine. The water, however, shields the radiation quite well. Then, first, the control rods are disconnected from their drive rods. This is done by a small latch inside the drive rod. Then the closure head bolts are opened, the closure head is lifted off with a crane. Finally, the control rod guide tubes are taken out as a whole package. All under water. Then one-third of the fuel assemblies are replaced step by step with new ones. The remaining ones are moved to a new position in the core. The spent fuel is transported under water to the neighbouring fuel assembly storage pool, where it has to cool down for a few more years.

The fresh fuel assemblies have initially more U-235 than would be needed for a self-sustaining chain reaction. This, however, is consumed over the course of the year. So we have to put more neutron absorbers into the reactor at the

beginning, because we need a smaller neutron flux then to generate the same power. Therefore, we have to capture the excess neutrons away. That's why the cooling water contains boric acid. Boron, more specifically $^{10}_5\text{B}$, is a good neutron absorber. As more and more U-235 is fissioned over the year, and we need a higher neutron flux for nuclear fission, the boric acid is simply replaced more and more by clean water. We use the control rods only to fine-tune the reactor power and to shut it down. Consequently, some water is permanently being taken out of the closed high-pressure system, which is cleaned and then pumped back again with the right boric acid concentration.

If the power supply fails during operation, the control rods fall into the reactor core and it is shut down. However, the residual heat must still be removed. For this reason, the steam generators are arranged somewhat higher than the reactor. The hot water rises then on its own to the steam generators and runs back after it has been cooled down. This is called natural circulation. You don't need an emergency power supply for that. On the low-pressure side, the water continues to boil on the outside of the pipes of the steam generator, and all that needs to be done now is to fill up water there. There is an emergency feedwater system for this, which can also be operated with emergency power.

Generally, the safety system of a reactor is designed in such a way that no radioactive radiation nor radioactive substances can escape to the outside. To shield γ -radiation, for example, we need lead plates. However, a thick concrete wall or a thick layer of water helps as well. To shield neutrons, light atomic nuclei, preferably hydrogen, help such that the neutrons lose as much energy as possible when colliding with these atomic nuclei. Hydrogen is found in high density in water, but also in concrete, which is mixed with water. That's why the reactor is surrounded by a lot of concrete. The β -rays travel only a few mm, the α -rays even much less.

Radioactive substances are, primarily, the fission products that result from the nuclear fission of uranium. They are locked up like prisoners in a high-security prison: the fuel is surrounded by three barriers, each of which must be hermetically sealed so that there is always another barrier if the first or second barrier should fail. The first barrier is the cladding tube of the fuel. It is closed by welding after it has been filled with fuel pellets. However, some fission products are gaseous, and so the pressure in the cladding tube continues to rise during operation. To prevent the cladding tube from bursting, there is an empty space above the pellets, the so-called fission gas plenum, which can collect the gases. Then the pressure rises more slowly. However, it is not allowed to drill a small hole into the cladding tube to simply release

these gases into the surrounding cooling water. That would violate the first barrier.

The second barrier is the primary system, which is the high pressure system as shown in Fig. 2.9. It must again be hermetically sealed. If everything is closed by welding, this is not really a problem. However, there are basically two places where moving parts must penetrate the primary system: the control rod drives on the reactor closure head, Fig. 2.8, and the drive shafts of the coolant pumps. The control rods that run out of the drives at the top are therefore still encased in a gas-tight protective tube, and the drive shafts of the coolant pumps either have a sophisticated sealing system or the drive motor itself even runs inside the cooling water, i.e. is inside the barrier.

The entire primary system is located in a closed concrete bunker called a containment. It is lined on the inside with sheets welded together to form a gas-tight seal, because concrete alone is not sufficiently gas-tight. This is the third barrier. Only the steam pipes from the steam generators to the turbine and the feedwater pipes to the steam generators run in and out of it. They each have a large valve inside the containment and a valve outside. Both valves can also close by themselves, without a power supply, if the steam line or feedwater line is damaged. This ensures that even in the event of an accident, the containment remains hermetically sealed. A leak in these lines inside the containment, on the other hand, would mean that the reactor would lose cooling water. The lost water is then collected in a basin at the bottom of the containment. From there, it must be pumped back into the reactor using emergency cooling pumps, which can also be driven by emergency power.

The special feature of the third generation of pressurised water reactors is that this containment must remain intact even if the reactor is completely destroyed in some severe accident. Even in the worst case imaginable, the aim is not to create a situation that would require the surrounding population to be evacuated.

The water-steam cycle, which finally produces electricity from the generated steam, is called the secondary system. I have sketched the most important parts of this secondary system schematically in Fig. 2.10. The steam temperature at the outlet from the steam generator corresponds to the boiling temperature at this pressure. The steam is not overheated. There would not be a sufficiently hot heat source in our power plant for that. This is also called saturated steam. More than 96% of this steam goes to a high pressure turbine (HP turbine). A fast-acting stop valve at the inlet can shut off the steam supply in less than a second, if necessary. This will be required, for example, if the generator suddenly had to be disconnected from the grid. The steam is then vented into the condenser via the bypass valves instead. The second valve

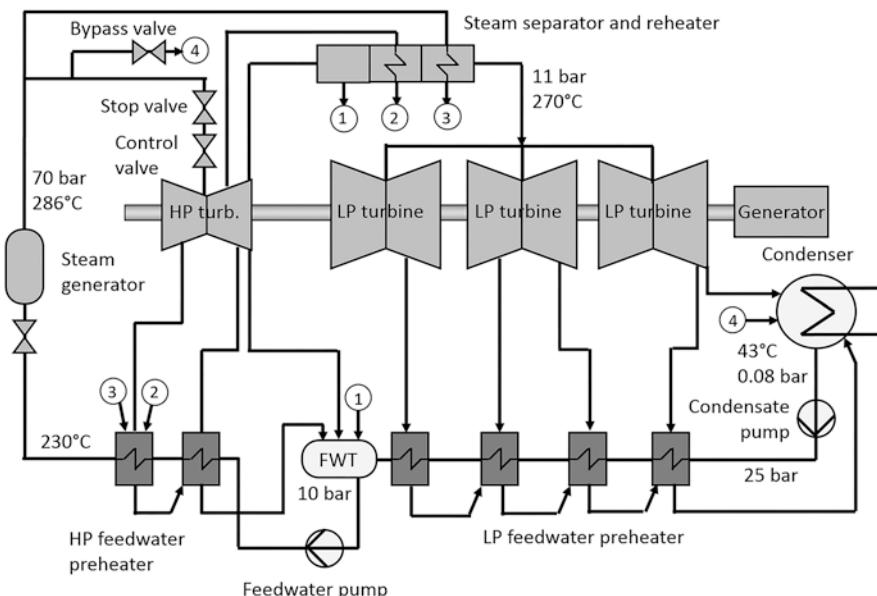


Fig. 2.10 Schematic of the secondary system of a pressurized water reactor

behind it is a control valve. It opens and closes only slowly, thereby controlling the pressure in the steam generator. A small portion of the steam produced is extracted before or inside the high-pressure turbine and fed to a reheater. This allows the steam coming out of the high-pressure turbine to be dried and superheated avoiding too many droplets in the low-pressure turbine. They would otherwise damage the turbine blades.

The large low-pressure turbines (LP turbines) turn the generator together with the high-pressure turbine. For the usual output of more than 1300 MW, this turbine shaft rotates at half the grid frequency, i.e. in Europe at 25 Hz (1500 revolutions per min). The generator has two pairs of poles in its rotor for this purpose; it generates then three-phase current at 50 Hz again despite half the speed. When the steam finally comes out of the low-pressure turbine, it has a pressure of less than 100 mbar, depending on the temperature of the cooling water inside the pipes of the condenser. The steam condenses on these condenser tubes and the condensate is collected at the bottom, in the hotwell of the condenser. In principle, we could feed it back to the steam generator from there. However, we can achieve a higher efficiency of this steam cycle by extracting some steam from the high-pressure and low-pressure turbines and condense it inside feedwater preheaters. Before we do this, however, we have to bring the condensate back up to pressure, otherwise it would boil again as

soon as it was heated. This is done by the condensate pumps at the lowest point of the power plant. From now on, the condensate is called feedwater.

In four heating stages, the feedwater is first preheated to about 150 °C with low-pressure steam. It then enters a large feedwater tank (FWT) where it is “boiled” with steam from the high-pressure turbine to remove the dissolved gases. They would otherwise oxidize the feedwater pipes. Consequently, the feedwater tank contains water and steam, usually at about 10 bar. This corresponds to a boiling temperature of 180 °C. The feedwater tank is located in the turbine building on one of the upper floors. The feedwater pumps, which are located down in the basement, take water from it and bring it to a pressure high enough to fill the steam generator. Two more high-pressure feedwater preheaters follow, heated by steam from the high-pressure turbine and by condensate from the reheater. Before entering the steam generator, the feedwater line still needs a valve to control the water level in the steam generator.

Boiling Water Reactor

The second most common reactor type is the boiling water reactor. Instead of producing hot water first, the boiling water reactor evaporates the water for the steam turbine directly by nuclear fission inside the reactor core. Figure 2.11 shows that the boiling water reactor is something like combining a reactor with a steam generator. The pressure vessel is about twice as high as that of a pressurized water reactor. In the reactor core, water evaporates at about 70 bar, or 286 °C. Coolant pumps, which in this case were integrated directly into the pressure vessel, pump water into the reactor core from below. Steam is leaving above the core, but with it also a lot of droplets, because the fuel assemblies should always be wetted and should not run dry. Steam and droplets are first collected in the dome above the core and from there fed into the steam separators. The steam gets a fast swirl there, which flings the droplets against the tube wall. There they are peeled off and run back into the water pool while the steam flows upwards. At the outlet from the steam separators is another steam dryer. These are zigzag shaped sheets, and at each turn of the zigzag, more droplets are thrown out. The evaporated water is replaced by feedwater, and the water supply keeps the level in the reactor at a constant level.

The fuel assembly of a boiling water reactor is smaller than that of a pressurized water reactor. It usually has only 72 fuel rods and an internal water box or tubes through which water flows. It is surrounded by a fuel assembly box, also made of Zircaloy, which creates a water gap between the fuel assemblies. This ensures that there is enough moderator everywhere, even with

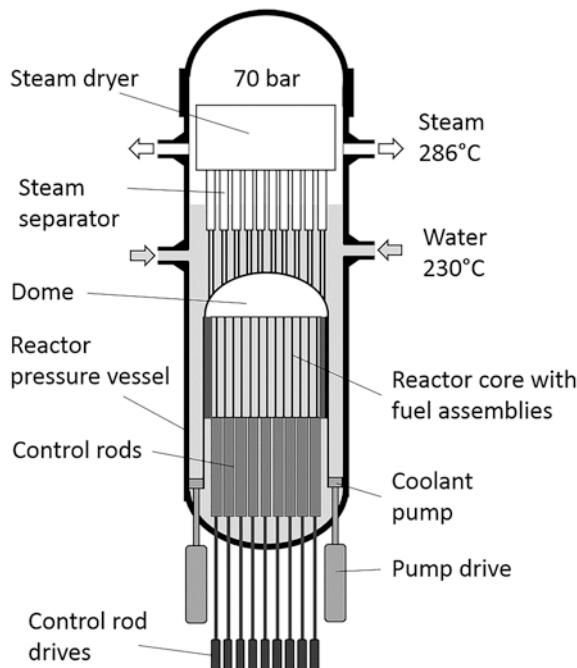


Fig. 2.11 Schematic diagram of a boiling water reactor

higher levels of steam in the upper half of the reactor core. The steam alone could no longer slow down the neutrons sufficiently. A fuel rod of a boiling water reactor can reach a similar high power as the one in a pressurized water reactor. Only in the upper part of the reactor core, we have to ensure that the fuel rods do dry out. They would otherwise overheat. However, the additional water gaps and water boxes require additional space. The average power density of this core is therefore only half that of a pressurised water reactor, about 50 MW/m^3 .

In the boiling water reactor, the control rods are inserted from below. Above the reactor core, there would be too little space between the steam separators. To allow the control rods to “fall” upwards without power supply, they have a hydraulic emergency drive which is constantly kept under pressure during operation. When the power is switched off, this hydraulic system shoots the control rods upwards. There they lock in place and can only be extracted again when the power is restored.

In this reactor, too, there is initially too much U-235 in the core after loading. But you can't add boric acid to the cooling water to absorb more neutrons. Boric acid evaporates at a higher temperature than water, and if we

boiled the water during operation, the boric acid would concentrate. The power distribution of the reactor could then hardly be reasonably controlled. The more boric acid we get, the less power is produced. Therefore, boric acid can only be used for emergency shutdown. Instead, the boiling water reactor has some fuel rods with gadolinium oxide, Gd_2O_3 . Gadolinium exists in nature in seven different isotopes. Two of them, namely ^{155}Gd and ^{157}Gd , are strong neutron absorbers. Once they have absorbed a neutron, they become the next heaviest isotope, which absorbs hardly any neutrons then. During operation, the gadolinium absorber therefore becomes weaker and weaker, but chemically it remains gadolinium. In this way, the neutron flux can increase as U-235 is consumed. Such a neutron absorber, which is consumed when U-235 burns up, is called a burnable neutron absorber.

As with the pressurized water reactor, this reactor does not require coolant pumps to remove the residual heat in an emergency case. The steam in the core and in the steam separators above it is much lighter than the water in the annulus around the reactor core, and this hydrostatic pressure difference is sufficient to generate sufficient flow through the core. However, an emergency feedwater pump, which can also run on emergency power, is required to replenish evaporated water.

The water-steam cycle of a boiling water reactor is actually the same as that of a pressurized water reactor, i.e. same components, same pressure and same temperatures. I could call Fig. 2.10 a schematic of the boiling water reactor as well, if I simply renamed the “steam generator” to “reactor”. We notice one difference, however, when we enter the turbine building. In case of the boiling water reactor, we have to wear the same protective clothing as when we enter the reactor building, i.e. overalls and a dosimeter in our pockets. The steam that is fed to the turbines comes here directly from the reactor. There, some oxygen atoms of the water molecules absorbed a neutron and emitted a proton. This is how the radioactive nitrogen ^{16}N was created from ^{16}O . With a half-life of 7.1 s, it quickly decays back to ^{16}O , but during this process it emits high-energy γ -rays of more than 6 MeV, which we can measure when standing next to the turbine. This is not a big problem for the power plant personnel, because, while the power plant is running, no one is needed in the turbine building anyway. And when we shut down the power plant to inspect or repair something, the problem has disappeared after a few minutes, because N-16 has already decayed again.

However, we can easily see that the barrier concept is violated in the boiling water reactor during normal operation. The first barrier, the cladding tube, has the same function as in the pressurised water reactor. It is hermetically sealed. However, the second barrier, the primary system, penetrates the

containment around the reactor because the steam generated inside the reactor goes directly to the turbine, that is, into the secondary system. So the reactor and the secondary system must form the second barrier together. Therefore, in the event of a leak in the turbine building, the steam lines and the feedwater lines that penetrate the containment are closed. This can even be done passively, i.e. without power supply. Only then does the containment form the third barrier again, like with the pressurized water reactor.

Such a boiling water reactor is undoubtedly larger and more expensive than a pressurised water reactor. However, all other components of the primary system in Fig. 2.9 are omitted. If the costs of all these components are added together, a power plant with a boiling water reactor is cheaper. Like pressurised water reactors, boiling water reactors have been built since the 1960s, mainly in the USA, Europe and Japan. China currently has a major nuclear power plant expansion program. Since 2010, China has commissioned more than 40 new reactors. However, these are all pressurized water reactors.

One could report a lot more about conventional water-cooled reactors, but I think this is enough to be able to assess the reactor concepts of the fourth generation. If you would like to look up one or the other question about these thermal reactors, I can recommend e.g. the book of Lamarsh and Baratta [7].

Glossary

Atomic number	Number of protons in the atomic nucleus
Binding energy	Energy that is theoretically released when an atomic nucleus is composed of free nucleons
Boltzmann constant	$k = 8.617 \cdot 10^{-5} \text{ eV/K}$
Containment	An enclosed space that prevents the escape of radioactive substances in case of failure of the primary system, e.g. a prestressed concrete containment with an inner steel liner
Cross section	Probability that neutrons cause a certain nuclear reaction, scaled with the neutron flux and the density of the irradiated atomic nuclei
Decay chain	Series of unstable nuclides created by successive decays until a stable nuclide is reached
Delayed neutrons	Emitted by fission products (approx. 200–600 keV)
Density effect	Influence of the moderator temperature on the moderation of neutrons
Depleted uranium	Uranium with a concentration of U-235 of less than 0.72% (N_2)

Doppler effect	Influence of fuel temperature on neutron absorption
Electron	Elementary particle of the atomic shell
Enriched uranium	Uranium with a concentration of U-235 exceeding 0.72% (N_+)
Enrichment cascade	Connection of several separation stages in series to increase enrichment
eV	Electron volt; $1 \text{ eV} = 1.6019 \cdot 10^{-19} \text{ J}$, energy unit
Feedwater	Water supplied to the evaporator
Fission gas plenum	An empty space in the closed cladding tube which can collect gaseous fission products
Fission yield	Probability that a nuclide is produced by nuclear fission
Half-life	Time that elapses until half of an unstable nuclide has decayed
Reheater	Superheater of the intermediate pressure steam, i.e. after the steam has left the high pressure turbine and before it enters the intermediate pressure or low pressure turbine
Isotope	Nuclides with the same number of protons
Mass number	Number of nucleons in the atomic nucleus
Moderation	Deceleration of neutrons by elastic collisions with light atomic nuclei
Moderator	Material that decelerates neutrons by elastic collisions (e.g. water or graphite)
Multiplication factor	Relative change in the number of neutrons in a reactor per fission
Neutron flux	Number of neutrons hitting a sample per unit area (cm^2) and second
Neutron	Electrically neutral subatomic particle
Nucleon	Subatomic particle (proton or neutron) of a nucleus
Nuclide	Class of atoms characterized by their number of protons, their number of neutrons, and their nuclear energy state
Photon	Electromagnetic radiation quantum that is produced during electron or nucleon transitions to another quantum state
Pressurizer	A vessel filled partly with water and partly with steam, both at boiling temperature, to control the pressure by temperature

Proliferation	Misuse of nuclear fuel for the production of nuclear weapons
Proliferation-resistant	The property of a fuel or process that it cannot be used directly, but only with great effort, to produce nuclear weapons
Prompt neutrons	Fast neutrons produced during nuclear fission (approx. 1–10 MeV)
Proton	Positively charged subatomic particle
Quantum state	Energy state of a nucleon or electron in quantum theory
Reactivity	Relative deviation of the multiplication factor from the number 1
Resonance absorption	Absorption of a neutron in an atomic nucleus, when the neutron energy is just sufficient to lift the neutron to a certain quantum state
Saturated steam	Steam at boiling temperature
Separation factor	Ratio of enriched to depleted U-235 concentration of a separation step $[N_+(1 - N_-)]/[N_-(1 - N_+)]$
Steam dryer	Zigzag-shaped plates which separate further droplets from the steam by centrifugal force
Steam separator	Vertical tubes with a swirl generator at the inlet, separating droplets by centrifugal force
Superheated Steam	Steam with a temperature higher than the boiling temperature
Thermal neutrons	Neutrons which, like gas atoms, are kept in motion only by temperature (approx. 25 meV at room temperature)
Thermal reactor	Nuclear reactor in which the prompt neutrons are slowed down to thermal neutrons to fission more effectively
Void effect	Influence of the vapour bubbles of the coolant on the neutron moderation
Xenon effect	Influence of the concentration of Xe-135 on neutron absorption



3

Innovative Water-Cooled Reactors

A conventional water-cooled reactor, whether a pressurised water reactor or a boiling water reactor, has a fairly modest efficiency of about 35%. If we can get very cold cooling water for the condenser, for example the Baltic Sea water in Finland, the efficiency is a little higher, about 36%, but that is still very low compared to a modern coal-fired power plant. The water-steam cycle of a conventional water-cooled reactor is basically still the same as in the 1960s. In the intervening years, however, coal-fired power plants have evolved considerably. As better and better boiler steels became available, higher and higher steam temperatures could be realized. The water is first evaporated in the tube wall of the boiler and then gradually superheated to more than 600 °C. As the steam temperature increased, the steam pressure was also allowed to rise more and more, and the steam turbine thus generated more power. On one hand, this lowers the specific costs of a coal-fired power plant, i.e. the ratio of manufacturing costs to power plant output. On the other hand, the heat demand of the coal firing system does not increase as much as the turbine output increases. This increases the efficiency of the power plant. A modern coal-fired power plant today has an efficiency of about 46%. How about applying this technology to a nuclear power plant?

Let's take a closer look at the coal-fired power plant. In principle, a boiling water reactor corresponds to a drum boiler. Feedwater, which has been pre-heated to almost the evaporation temperature, is initially only partially evaporated in the boiler tubes around the flames and the water-steam mixture rises into a steam separator. In a coal-fired boiler, this is called the "drum". The liquid water from the drum is recirculated by a pump back into the evaporator and the separated steam goes into a superheater. The boiling water reactor

has a steam separator as well, but such a superheater does not exist. The superheated steam from the coal-fired boiler is fed to the high-pressure turbine. The higher the steam temperature at the turbine inlet, the higher the steam pressure must be, because the steam turbine produces more power only then. Otherwise, just hotter steam would come out of the low-pressure turbine.

Supercritical Water

The higher the pressure, the hotter the water is when it evaporates and the lower its surface tension. At 374 °C, the surface tension finally becomes zero. The boiling pressure at this temperature is 220 bar. In thermodynamics, we call this point the critical point, and the corresponding pressure the critical pressure. If we bring water to a higher pressure than 220 bar before evaporation, we can produce steam without bubbles or drops, because there is no surface tension then. The density of water just keeps decreasing then as it heats up, and above about 380 °C we call it steam.

Figure 3.1 shows how also the other material properties of supercritical water change with heat-up at a pressure of 250 bar. Viscosity and thermal conductivity decrease continuously with heat-up, as does the density. At a temperature of 384 °C, this change occurs very rapidly. At the same temperature, the specific heat capacity has a pronounced maximum. This means that the temperature of the water changes only slightly when heated in this range, but the material properties change strongly. This effect can also be called pseudo-boiling. To the left of this maximum, the supercritical water is more comparable with a liquid and to the right of it more with steam.

When you run a coal-fired boiler with supercritical water, there is no longer a drum, because you can't separate supercritical steam from supercritical water; there are no more droplets. Instead, the water is completely vaporized and superheated without any of it being fed back into the evaporator.

Since the mid-1990s, coal-fired power plants have been using a supercritical steam process. It has not only advantages for efficiency, but also for power output and thus for the electricity costs. A power plant operator must use part of the revenue to pay its investment costs. The higher the pressure and temperature of the steam, the higher the turbine output and the more electricity can be sold every day. Higher pressure requires thicker pipe walls, but these extra costs are comparatively low.

In a water-cooled reactor, supercritical steam has another advantage: the steam separators and the steam dryer in the steam generator of the pressurized

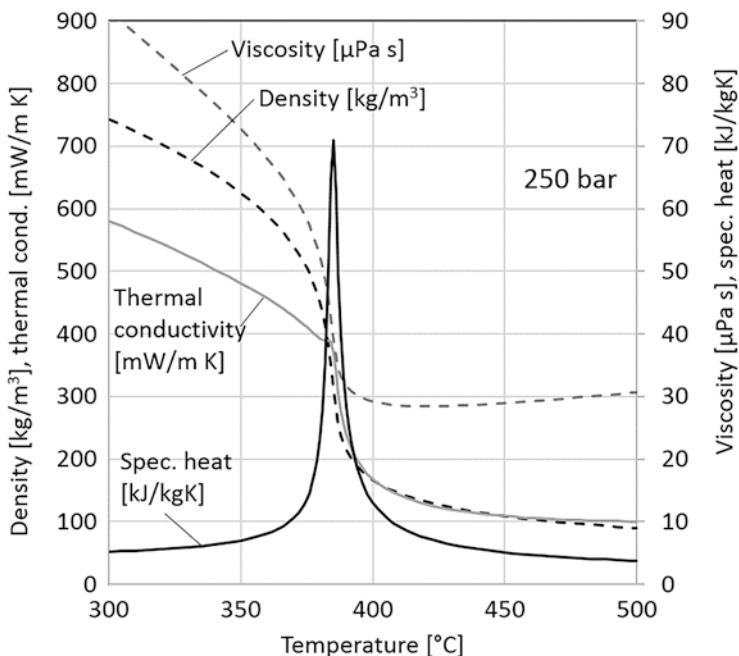


Fig. 3.1 Change in material properties of supercritical water with temperature at a pressure of 250 bar

water reactor or in the reactor pressure vessel of the boiling water reactor can be omitted. There are no droplets to be separated anymore. This makes the boiling water reactor as compact as a pressurized water reactor, but at the reactor outlet we get already superheated steam, which can be fed directly to the high-pressure turbine.

I have sketched such a supercritical steam cycle of an innovative water-cooled reactor in a very simplified form in Fig. 3.2. By far the largest part of the supercritical steam at 500 °C is fed directly to a high-pressure turbine (HP). A control valve at the inlet to this turbine keeps the steam pressure at 240 bar. A small part of the steam is diverted beforehand. It is needed to reheat the intermediate-pressure steam leaving the high-pressure turbine to about 440 °C before it is fed to the intermediate-pressure turbine (IP). Without further reheating, the low-pressure steam exiting the intermediate-pressure turbine enters then the low-pressure turbine (LP). Condensation and feedwater preheating are the same as in a conventional water-cooled reactor, except that the feedwater pumps have to generate more than 250 bar of pressure now and the steam, which is extracted from the turbines, has to produce a feedwater temperature of 280 °C. In total, seven stages of feedwater

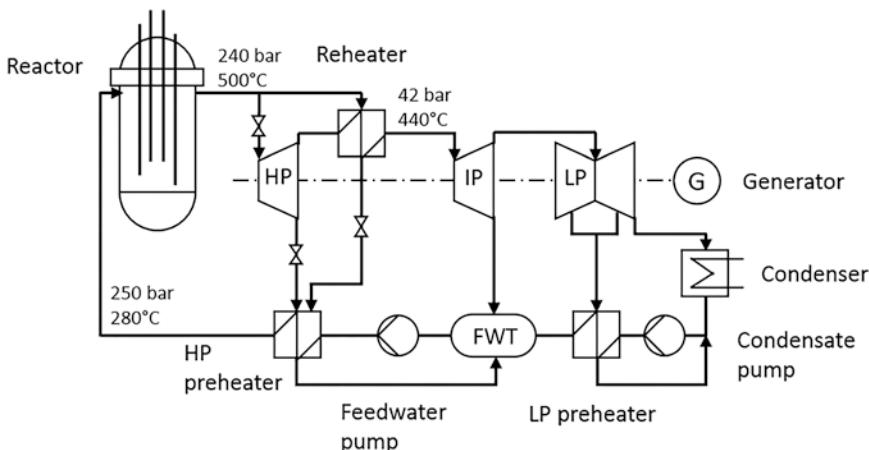


Fig. 3.2 Schematic of a supercritical steam process for an innovative water-cooled reactor. FWT: feedwater tank; HP, IP, LD: high-pressure, intermediate-pressure, low-pressure turbines

preheating are needed, of which only two are shown here. The feedwater temperature is controlled by valves in the steam lines of the turbine extractions.

We can see another cost advantage in Fig. 3.2: The coolant pumps, which had to be installed inside the pressure vessel from below in the case of the boiling water reactor, are also unnecessary, because there is no water to be recirculated if droplets are not separated anymore. Such a supercritical water-cooled reactor would have an efficiency of more than 43%. So it makes about 20% more electricity from the same fuel. This is still less than a coal-fired power plant with a supercritical steam process would achieve, because the steam is still 100 °C colder at the turbine inlet, but it is already a significant improvement over today's pressurized water reactors.

Reactor Concepts with Supercritical Water

At the beginning of 2000, a group of students and young engineers in Europe had worked out such a reactor concept. It was published under the name “High Performance Light Water Reactor” (HPLWR) [8]. Similar concepts were also developed in Japan at this time under the name “Super Light Water Reactor” [9], also there mainly by undergraduate and graduate students under the guidance of some experienced engineers. The background of the support programs by industry and by Euratom was mainly that a whole generation of nuclear engineers retired and had to be replaced by new specialists. But you

can't learn power plant engineering just by reading books and listening to lectures. And so we decided to design a new nuclear power plant together and work through the calculations. The students were enthusiastic, also about the international cooperation, and designed far more details than we had ever planned. Most of them quickly found a well-paid job in the nuclear industry afterwards.

At this point, however, let us take a more technical look at the concept. The components of the steam cycle in Fig. 3.2 are mainly the same as those used in a modern coal-fired power plant. The electrical output of the HPLWR was therefore assumed to be 1000 MW, the same as that of coal-fired power plants built during that time. The turbines there run at a speed of 50 Hz. They had been cost-optimized over decades and the HPLWR concept already got a significant cost advantage from this synergy. The only new feature was the reheater, which differs significantly from those in a coal-fired power plant. The reheating is done there with flue gas of the boiler instead. However, the reheater is not a particularly sophisticated component that I would have to report on.

The reactor itself, on the other hand, is interesting. At first glance, it looks like a pressurized water reactor, as sketched in Fig. 3.3. The reactor pressure vessel is about 14 m high, but has considerably thicker walls than a pressurised water reactor. It must withstand a pressure of 250 bar. The control rods are also inserted from above and the control rod drives are the same as in a pressurised water reactor. The feedwater at 280 °C flows around the entire inside of the pressure vessel, so that it can be built from the same steel as a pressurized water reactor.

The reactor core, however, is clearly different. The fuel assemblies, outlined in Fig. 3.4, are similar to those of a boiling water reactor. They each have 40 fuel rods with UO₂ pellets with up to 7% enrichment in U-235. The fuel is inside cladding tubes with 8 mm in diameter. The fuel rods in the corners of the fuel assembly have some pellets with gadolinium oxide to absorb some neutrons first, as long as the fuel is fresh. The fuel assembly is wrapped by an assembly box, creating a water gap of 9 mm width between two boxes. Another, smaller water box is installed inside the fuel assembly. This water, which has hardly been heated beforehand and consequently still has a high density, serves as a moderator to slow down the neutrons, because, at the high exit temperature of 500 °C, the generated steam would have too little density to be able to moderate effectively.

Both, the cladding tubes and the water boxes, are made of stainless steel, i.e. not Zircaloy. Some corrosion tests with Zircaloy had quickly shown that Zircaloy is unsuitable for supercritical water, although it would undoubtedly

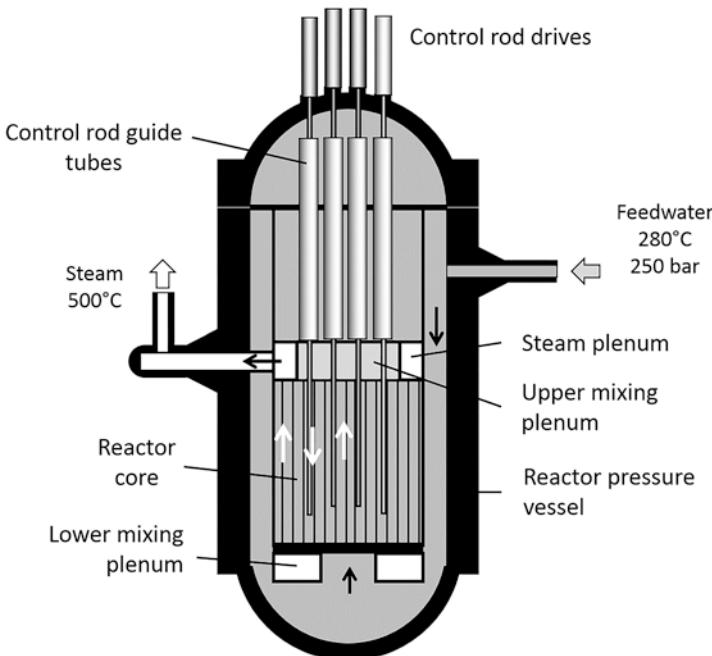


Fig. 3.3 Sketch of the high performance light water reactor

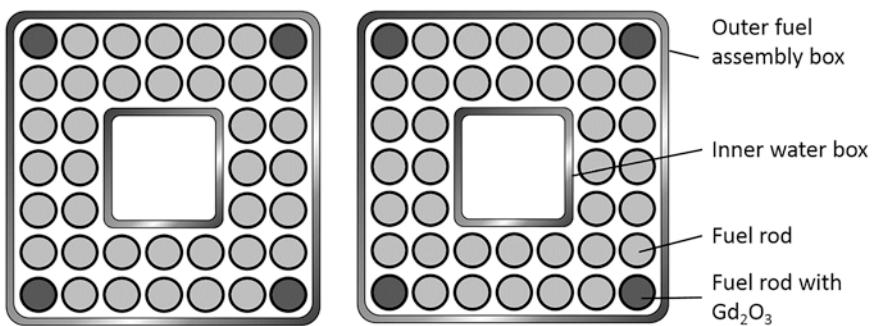


Fig. 3.4 Fuel assemblies of a high performance light water reactor

be the better material for the neutrons. Stainless steel unfortunately absorbs significantly more neutrons. For this reason, an enrichment of 7% had to be chosen, more than for a pressurized water reactor. A wire is used as a spacer between the fuel rods, which is running spirally around each cladding tube and which is fixed at the top and bottom. The swirling flow that this creates ensures that the cooling water flowing through the fuel assembly is well mixed. This avoids hot spots inside the fuel assembly, because the hottest fuel rod in

the reactor core will limit the average exit temperature of the generated steam. The aim of the design was to keep this hottest cladding tube colder than 630 °C. Otherwise, it would be too hot for the stainless steels available today. However, the analyses showed that this is not so easy to achieve at a mean exit temperature of 500 °C.

Since the coolant flow is enclosed in the fuel assembly box, the generated steam cannot mix with that of a neighbouring fuel assembly. However, not all fuel assemblies have exactly the same power. Some are located at the rim of the reactor core, where more neutrons escape through leakage. There the neutron flux is lower. In other fuel assemblies, a control rod has been inserted more or less deeply into the inner water box to control the power. There, too, the neutron flux is correspondingly lower. Finally, there are fresh and somewhat older fuel assemblies in the reactor core. They have different concentrations of U-235, so that they produce different power even with the same neutron flux.

Unfortunately, it is physically the not case that a fuel assembly with higher power also receives a higher mass flow of cooling water, but quite the opposite. The hotter the steam produced, the lower its density and consequently the higher its velocity. However, the higher steam velocity produces a higher pressure loss, and the cooling water then prefers to go through the colder fuel assemblies, which would not need it at all. It is therefore very important with this reactor concept that the cooling water or the steam of the individual fuel assemblies is mixed properly again and again in between.

This is the reason for the unusual flow routing of the cooling water through the reactor core, which is outlined with white arrows in Fig. 3.3. The feedwater, which enters the reactor at 280 °C, first flows along the entire pressure vessel, through the gaps between the fuel assembly boxes and through the inner water boxes. It serves as a moderator for the neutrons. When it enters the reactor core from below for cooling, it is already almost 320 °C warm, but still has a high density of more than 700 kg/m³. The openings in the core support plate only allow this cooling water to enter one-third of the fuel assemblies, those in the central part of the reactor core. When this water exits at the top, it is already around 400 °C hot, and with a density of 167 kg/m³ already rather vaporous. Accordingly, we could call the first heating stage an “evaporator”, but with supercritical water, the transition from water to steam is continuous. Now the steam is first mixed properly in the upper mixing chamber. Afterwards, it goes down through the fuel assemblies around the evaporator and heats up further to around 470 °C. This is the first superheater. Now the steam is mixed again uniformly in an annular lower mixing chamber, and the final heating up to 500 °C is only carried out very carefully, i.e. with little power inside the fuel assemblies located at the rim of the reactor core, because

we don't want to overheat anything. The annular steam plenum around the upper mixing chamber finally guides the generated steam through the outlet flanges into the steam lines and from there to the turbine.

The average power density of this certainly innovative reactor core concept is about 50 MW/m^3 , which is comparable to a boiling water reactor. The reactor has to be opened once a year to replace one-third of the fuel assemblies and to shuffle the other fuel assemblies, as in a conventional pressurised water reactor.

The generated steam is fed from the reactor directly to the high-pressure turbine, similar to a boiling water reactor. Therefore, it was obvious to design the containment and the emergency cooling system similar to the latest boiling water reactors. The principle of the containment of an HPLWR is explained in Fig. 3.5. It is a prestressed concrete construction with an inner diameter of 20 m and an inner height of 23.5 m. This is extremely compact compared to a containment of a pressurized or boiling water reactor. The feedwater pipes lead into the containment from the outside through two check valves. In an emergency, boric acid can be injected into these lines to shut down the reactor if the control rods should jam. However, no boric acid

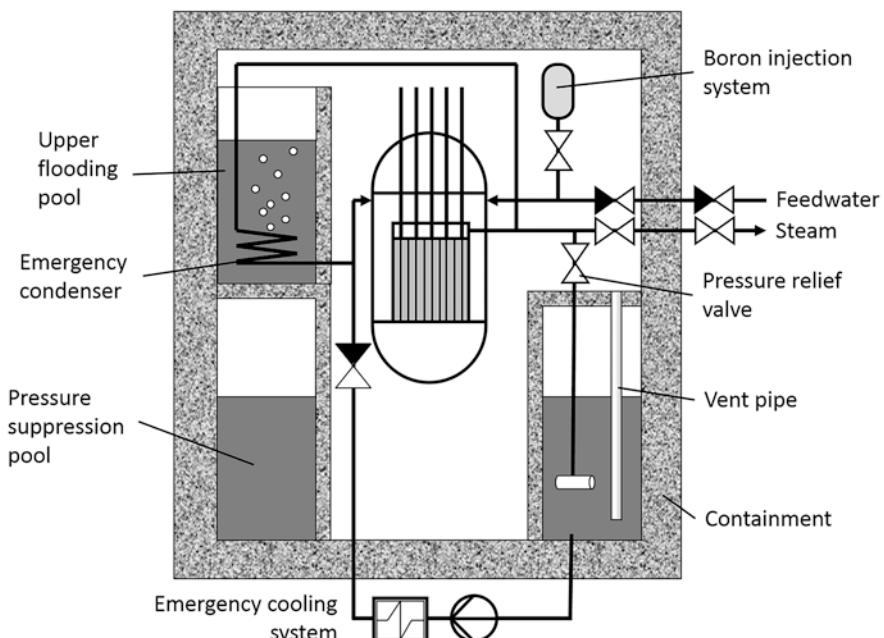


Fig. 3.5 Sketch of the containment and the emergency cooling concept of an HPLWR

is added during normal operation. The superheated steam produced goes out of the containment through two isolation valves per line, both of which also close passively if a leak should occur outside, as for the boiling water reactor. When these valves close, first the control rods fall into the reactor. Now, however, the steam generated by the residual heat has to go somewhere else. It is then blown off via pressure relief valves into the annular pressure suppression pool in the containment. From there, an emergency cooling system can pump the water back into the reactor. Now, the water-steam cycle is closed again then inside the containment via the emergency cooling system.

If these pumps fail for any reason, the residual heat can also be removed via an emergency condenser installed in an upper flooding pool of the containment. To do this, the pressure in the reactor must first be relieved via the pressure relief valve. At subcritical pressure, steam is generated and this steam rises through the steam lines into the emergency condenser, condenses there and flows back into the reactor.

In the event that a steam line inside the containment should burst, the check valves of the feedwater lines and the steam isolation valves also close immediately and limit the damage to the inside of the containment. The reactor shuts down and the pressure in the reactor goes down all by itself because the reactor has now a leak. However, the pressure in the containment rises in return, and the containment quickly fills with steam. Vent pipes are provided for this case. These are pipes that lead from the containment into the closed pressure suppression pool. They are simply open at both ends, and the increased pressure in the containment now blows steam into the cold pool, where it condenses in the cold water of the pressure suppression pool. This keeps the pressure rise in the containment limited and is the only reason why we are allowed to build the containment so compact. The emergency cooling system pumps the lost water from the pressure suppression pool back into the reactor.

The containment concept with its safety systems is basically the same as that of the third generation boiling water reactors. We find something like this in the American ESBWR [10] and the European SWR-1000 [11], both advanced boiling water reactors for which a great deal has already been developed and tested, but which have never been built yet. For example, the containment concept of the ESBWR has been tested at the PANDA facility of the Paul-Scherrer-Institute in Switzerland and the containment concept of the SWR-1000 has been tested at the INKA plant in Karlstein am Main in Germany. So it's not all that new. In combination with an HPLWR, however, the situation is somewhat different, because its reactor core has a significantly higher pressure loss than a boiling water reactor. So the containment concept

would certainly have to be recalculated and tested before building something like this.

The aim of this innovative reactor concept was to improve the economic efficiency of conventional pressurized and boiling water reactors. I think this can be clearly seen, as all components have become smaller and more compact and some large components have even been eliminated entirely. Moreover, the efficiency has increased. However, I cannot claim that this concept would be safer than a boiling water reactor of the latest design. After all, the safety concept was “only” derived from third generation boiling water reactors. Of the four criteria I mentioned in the introduction for assessing fourth-generation concepts, only the third criterion could be said to be a clear improvement.

A reactor with supercritical pressure has never been built before. There would also be a lot to do before such a concept will become a functioning power plant. Not only do you have to analyse all the components in detail beforehand, but you also have to test all the relevant components to ensure that they work. The effort involved should not be underestimated, because almost everything is relevant in terms of safety, both in the reactor and inside the containment.

The Superheated Steam Reactor

There were times when companies in Germany were more courageous and built an innovative reactor, even if only a few components were tested beforehand. A reactor concept with superheated steam dates back to this time. Although it operated at a significantly lower pressure, it was also intended to generate superheated steam at 500 °C. A prototype of this “superheated steam reactor” (Heissdampfreaktor, HDR), as it was called at the time, was actually built between 1965 and 1969. It was located in Großwelzheim, now Karlstein am Main, and it was far ahead of its time. Unfortunately, it never worked properly, but let me first explain the concept.

The HDR was basically a boiling water reactor with superheater, as outlined in Fig. 3.6. Unfortunately, there is little documentation on this concept, but the journal “Atomwirtschaft – Atomtechnik” reported something about it in November 1969 [12]. The prototype had a thermal output of 100 MW, of which 75 MW was produced in the evaporator and 25 MW in the superheater. The feedwater pressure at the reactor inlet was 90 bar. The HDR was to produce superheated steam at 457 °C at a pressure of 73.3 bar, initially still less than the design target of 500 °C.

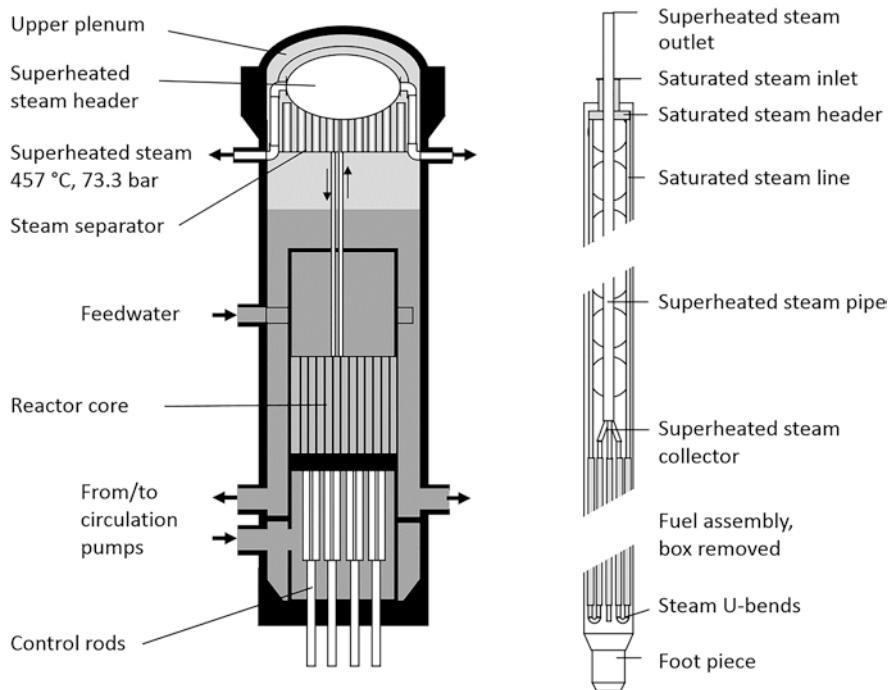


Fig. 3.6 Design principle of the superheated steam reactor (HDR) and one of its fuel assemblies

The reactor core in the lower half of the reactor pressure vessel consisted of 52 fuel assemblies with 24 fuel rods each. 21 cross-shaped control rods were inserted into the reactor core from below, as is still the case today with boiling water reactors. The circulation pumps, which pumped the water to be evaporated through the reactor core, were located next to the reactor. They were therefore not yet integrated into the reactor, as is the case today, but this was also the state of the art in other boiling water reactors at the time. The feedwater was supplied halfway up the reactor. It kept the level inside the reactor at about $\frac{3}{4}$ of the reactor height. In the upper part of the pressure vessel were the steam separators, which separated the droplets of generated steam by centrifugal force and returned them to the feedwater pool.

Unlike boiling water reactors, however, the saturated steam generated was sent through the reactor core again to superheat it. For this purpose, saturated steam pipes ran from the steam separators back to the fuel assemblies, and superheated steam pipes finally brought the superheated steam into a superheated steam header in the upper plenum of the reactor.

The fuel assembly design would still be considered highly innovative today. Figure 3.6, right, shows a longitudinal section and Fig. 3.7 a cross-section through this fuel assembly. It consisted of 24 fuel rods each with an outer diameter of 26.5 mm and a gadolinium rod in the middle as an additional neutron absorber for fresh fuel assemblies. The fuel rods stood in a fuel assembly box made of Zircaloy. Unlike today's fuel rods, however, there was a second cladding tube made of Inconel 625 inside each fuel rod. This is a high-temperature nickel-based alloy that is still used today for high-temperature heat exchangers. The outer cladding tube was stainless steel. The enriched uranium was filled as powder together with some annular UO_2 pellets into the space between both cladding tubes and was compressed there.

The water supplied to the reactor core by the two circulation pumps initially evaporated outside the fuel rods and water and steam rose inside the fuel assembly. Six of these fuel rods were each connected together to form a superheater unit, marked by dashed lines in Fig. 3.7. After steam separation, the saturated steam was then first fed to inner tubes 1 and 3. After a U-bend in the fuel assembly foot piece, this superheated steam then rose up again through the inner tubes of fuel rods 2 and 4. The two superheated steam lines were then combined and the steam was given a second superheat down through fuel rod 5 and up again through fuel rod 6. A superheated steam collector at the top of the fuel rods combined then the superheated steam lines of the four superheater units and fed the superheated steam jointly through superheated steam pipes to the large steam header in the upper plenum.

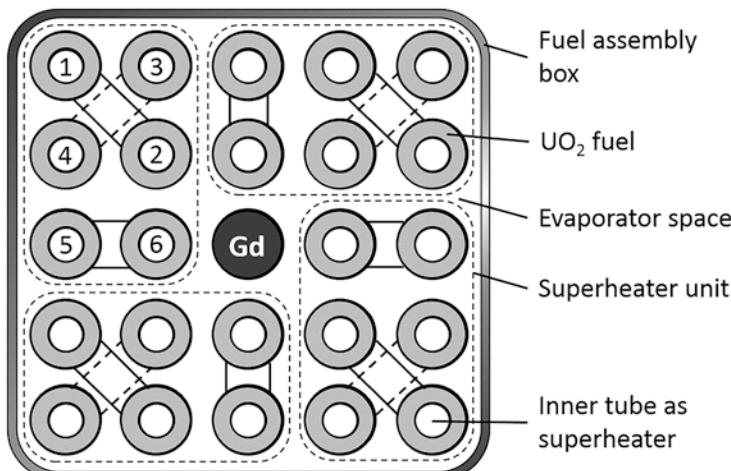


Fig. 3.7 Section through a fuel assembly of the HDR

So here, too, the steam was already sent several times through the reactor core, but without a mixing chamber in between. This was perhaps the decisive mistake, because the reactor did not survive the first trial operation in autumn 1970 for long. It was finally shut down early in spring 1971. The concept of the superheated steam reactor has not been pursued further since then. The former power plant division of AEG, which built this reactor, merged with the corresponding division of Siemens in 1969 to form Kraftwerk Union (KWU) and from then on built only boiling water reactors without superheaters. The HDR served as a test stand until the 1980s and was finally demolished in the 1990s.

The reasons for the early failure of this reactor were not published unfortunately. Who likes to report publicly on their mistakes? Though it would have been helpful for future generations of scientists and engineers to learn from these mistakes. From discussions with some older engineers who could still remember this reactor, I learned that some of the hot steam pipes collapsed. That sounds plausible, at least. A single fuel assembly had been tested quite successfully before, being electrically heated to simulate the heat from nuclear fission. It is conceivable that it was underestimated how much the power of the individual fuel assemblies in the reactor core could differ. If the steam in some fuel elements had become hotter than planned, these fuel elements would have received less water because their pressure loss increased, making the steam even hotter. However, if the superheated steam pipes become too hot, they can no longer withstand the external pressure of the feedwater and indent. Then the superheated steam at this constriction gets even a greater pressure loss and the fuel assembly consequently gets even less water, until finally the superheated steam tube completely overheats and collapses. A proper mixing of the superheated steam of all fuel assemblies before each superheating step could certainly have limited the consequences, but this would have led to a completely different reactor design.

Pressure-Tube Reactors

There is another concept of a reactor with supercritical water that has been worked out in Canada in recent years. The principle is a pressure-tube reactor in which supercritical water is superheated to more than 500 °C at about 250 bar inside individual pressure tubes. The pressure tubes are surrounded by heavy water at atmospheric pressure.

Heavy water differs from ordinary light water in that the hydrogen atoms have a deuterium nucleus, that is, one proton and one neutron each.

Deuterium, ^2H , is found everywhere in nature. It is not radioactive and occurs with a frequency of 0.0115% of all hydrogen atoms, as the nuclide chart tells us. Since deuterium is twice as heavy as light water, we need more collisions to moderate the neutrons, but in return the cross section for the absorption of thermal neutrons is only 0.5 mbarn, compared to 332.6 mbarn of a light hydrogen nucleus. The neutron absorption of a reactor moderated with heavy water is thus considerably lower than when moderated with light water.

In the 1960s, these considerations led to the concept of the CANDU reactor, which can even be operated with natural uranium because it allows a higher thermal neutron flux, and which was built quite frequently, especially in Canada. In the original CANDU reactor, cooling water is heated at about 100 bar inside pressure tubes. The pressure tubes run horizontally there and are continuously supplied with fresh fuel during operation, because the little bit of U-235 in the natural uranium is consumed quickly. In the early years of nuclear technology, this type of reactor had the advantage of not needing large forging presses to produce the reactor pressure vessel, and not needing an enrichment plant for the uranium; both key technologies that not every country had. Meanwhile, however, this reactor is specifically more expensive than a pressurized water reactor.

It was obvious to apply the supercritical steam cycle also to this reactor principle in order to make it more economical. The horizontal pressure tubes were replaced by vertical pressure tubes and the natural uranium was replaced by enriched uranium, so that the continuous charging with fresh fuel could be omitted. The principle of a pressure tube of such a CANDU reactor with supercritical pressure is sketched in Fig. 3.8.

The thick pressure tube separates the supercritical, light water inside from the heavy moderator water at normal pressure outside. In order not to heat up the moderator water unnecessarily, thermal insulation is required inside the pressure tube, which at the same time keeps the pressure tube cold, giving it more strength. The feedwater supply to the fuel rods is inside the pressure tube; the flow there is downward. The pressure tube is closed at the bottom; the feedwater is thus directed into the surrounding annular fuel assembly, where it flows upwards and is heated. Further details of the reactor concept are described by Yetisir et al. [13].

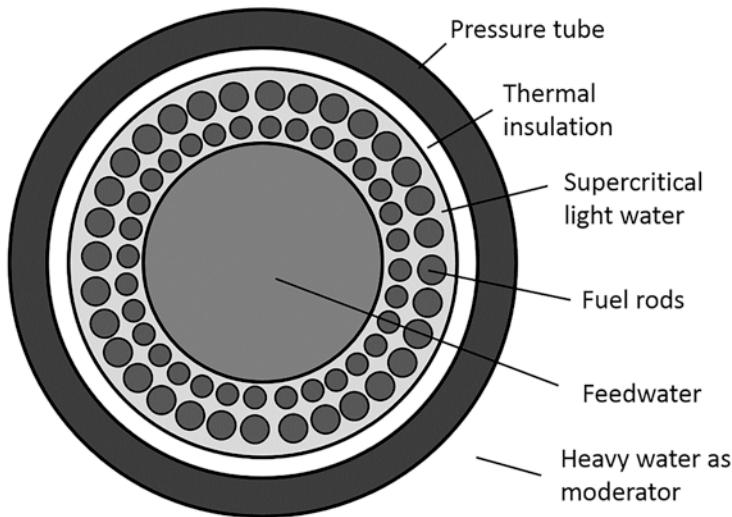


Fig. 3.8 Section through a pressure tube of a CANDU reactor with supercritical pressure

Further Developments

However, this pressure-tube concept had not convinced the engineers in China, so that work is only being done there today on a reactor with a pressure vessel for supercritical water, similar to the one outlined in Fig. 3.3. The main reason for this development work is again to train young engineers using exciting and challenging topics in reactor technology. In China, several new nuclear power plants are currently coming on line every year, and the demand for young, dedicated nuclear engineers is consequently huge. That alone is reason enough to study these topics at universities. Whether such a reactor will ever be built there is still uncertain. But I think it is rather unlikely. The risk of building a power plant for several billion euros that doesn't run afterwards is generally too high for a power plant manufacturer. It could quickly ruin him financially. This is why coal-fired power plants were not improved in leaps in the past and the steam temperature did not jump from 300 °C to 600 °C, but the development proceeded continuously over many years. With each new power plant that was built, the concept was improved a little, the temperatures and pressures were raised a little in each case, components were simplified or modernised and the power plant output increased. This is even more important for nuclear power plants with their even significantly higher investment costs and quality requirements.

Conclusion

Water-cooled reactors with supercritical steam parameters are an attempt to transfer the knowledge gained over the past 30 years in the development of coal-fired power plants to nuclear power plants.

Economic Competitiveness

The result would be a fairly compact design: some large components such as steam generators, steam separators and coolant pumps could be completely eliminated and the specific investment costs could thus be reduced by about 20%. Maybe even more. As a result of the high steam temperature, the efficiency of the power plant increases by about 20%. The current designs therefore appear to be more economical than the third-generation water-cooled reactors. If we draw the containment of the AP 1000, a third generation pressurised water reactor, and the containment of the HPLWR side by side on the same scale, as I have attempted to do in Fig. 3.9, it becomes clear why the innovative water-cooled reactors should be cheaper. However, there is still a

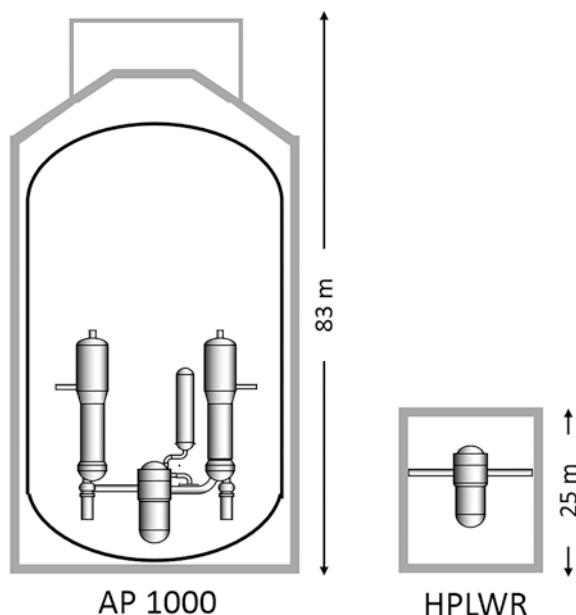


Fig. 3.9 Containment of a third-generation pressurized water reactor and an innovative water-cooled reactor in size comparison

long way to go, even if the components of the secondary system could be largely adopted from coal-fired power plants. The designs so far are therefore to be understood more as visions or suggestions and not as drawings for the construction of a power plant.

Safety

The safety systems of these reactor concepts do not look much different from those of a third-generation boiling water reactor. Therefore, I find no reason why such an innovative water-cooled reactor should be any safer. On the contrary, it would have to be shown first that these safety systems work just as well for a reactor with supercritical water.

Reliability

The high reliability of existing nuclear power plants can certainly not yet be expected for a new reactor of this type. I think that applies to all power plants that are being built for the first time. That's why we should rather improve a power plant incrementally than in one big leap. On a purely design basis, I could give no reason why the innovative water-cooled reactors should be more reliable than conventional ones.

Sustainability

The innovative water-cooled reactors are initially just as unsustainable as conventional water-cooled reactors. They consume uranium and produce spent fuel, the possible further use of which we will have to consider in more detail in later chapters.

Proliferation Resistance

A pressurized or boiling water reactor does not produce fuel from which nuclear weapons could be made directly. The uranium in the fresh fuel assemblies would only be nuclear weapons grade if enriched to 20% or more. That high enrichment is never needed. Therefore, the risk of nuclear fuel proliferation is very low. Innovative water-cooled reactors are no different.

Glossary

Pressure suppression pool	A space in the containment that is partially filled with cold water. If steam is generated in the containment, it can condense there so that the pressure increase in the containment remains limited
Pseudo-boiling	The continuous transition of liquid water into steam without bubbles or droplets; only occurs with supercritical water
Specific heat capacity	Ratio between the heat supplied and the temperature increase caused by it
Supercritical water	Water or steam at a pressure greater than the critical pressure (220.64 bar)
Superheated steam	Steam with a temperature higher than the boiling temperature
Thermal conductivity	Ratio between heat flow and temperature gradient of a substance



4

High-Temperature Reactors

A pressurized water reactor generates saturated steam in its steam generators at <300 °C. The cladding tubes of the fuel assembly are made of Zircaloy, which does not melt below about 1850 °C, but it corrodes in water already at much lower temperatures, already above 450 °C. Therefore, reactors that generate supercritical steam have cladding tubes made of stainless steel, which does not corrode below 650 °C; they allow steam temperatures of about 500 °C. If we want to go much higher with temperature, only ceramics will help.

Ceramic Fuel Elements

The fuel UO_2 is already a ceramic itself. It has a melting temperature of about 2800 °C and may also be heated to almost this temperature. A cladding tube made of silicon carbide (SiC) would indeed tolerate up to 1600 °C. However, it is very brittle and the fuel rod could easily break during installation. For this reason, a completely different type of fuel element was developed as early as the end of the 1950s, outlined in Fig. 4.1. Small spherical particles of approx. 0.5 mm diameter are formed from the fuel UO_2 . These are coated with graphite and SiC so that they are enclosed gas-tight. This composite layer then forms the required first barrier against the release of radioactive substances. The innermost layer is a porous graphite, which can absorb the gaseous fission products in its pores. It forms the fission gas plenum. Around this are three dense layers, the first one of pyrographite, a graphite with high thermal

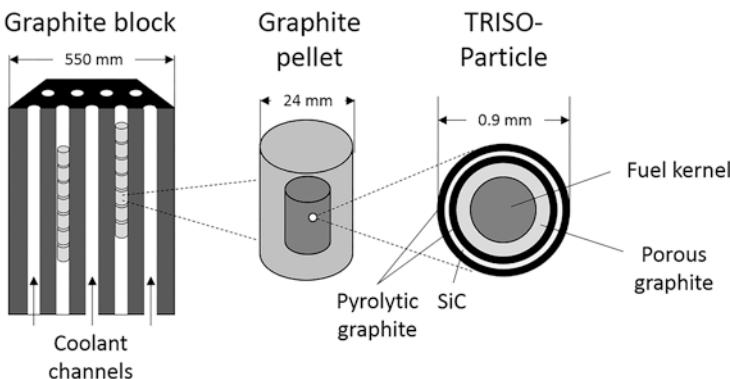


Fig. 4.1 Sketch of the ceramic fuel for a high-temperature reactor with graphite blocks

conductivity, on top of this a layer of SiC with high mechanical strength and finally another dense layer of pyrographite. The finished particle has a diameter of about 0.9 mm and is called TRISO particle (*TRistructural-ISOtropic*). It can be used up to surface temperatures of about 1600 °C before it fails. Thousands of these tiny spheres are then embedded in a graphite body that is either cylindrical having about the thickness of a thumb, or spherical with a diameter of 6 cm, about the size of a billiard ball. In this form, the fuel is easier to handle than the small TRISO particles.

At operating temperatures of 800 °C or more, water is neither suitable as a coolant nor as a moderator. CO₂ or, better, helium, which is chemically inert so that neither oxidation nor corrosion need be expected, can be used as a cooling medium. As a moderator, however, helium has too little density. Graphite is more suitable here. Indeed the carbon atom has a mass 12 times greater than that of a hydrogen atom, and we would need about 6 times as many collisions to decelerate a neutron to thermal energy. To absorb thermal neutrons, however, carbon, ₆¹²C, has a cross section of only 3.5 mbarn, compared to 333 mbarn for the hydrogen atom, as a look at the nuclide chart tells us. This means that only a few neutrons are lost during moderation. Carbon is therefore, despite its high atomic weight, a quite good moderator.

History

Today, there are two lines of development of these helium-cooled and graphite-moderated reactors. The American line of development is based on cylindrical fuel pieces filled into hexagonal graphite blocks, sketched on the left in

Fig. 4.1. The oldest reactor of this design was Peach Bottom, a 40 MW prototype that started operation in 1967, followed by Fort St. Vrain, which started operation in 1976 and produced already 330 MW of electrical power [14]. The helium there had an average temperature of 775 °C at the core exit. It produced superheated steam of about 540 °C in steam generators underneath the reactor core, which was supplied to a steam turbine train consisting of high-pressure, intermediate-pressure and low-pressure turbines. Interestingly, the reactor had a pressure vessel made of prestressed concrete, but this did not prove to be very effective in the following decades.

A smaller reactor, also with graphite blocks but with higher core exit temperatures of 950 °C, was operated in Japan since 1998: the HTTR. It produced 30 MW of thermal power, but no electricity, and is therefore purely an experimental reactor [15]. The reactor core is 2.3 m in diameter and 2.9 m high. It consists of graphite blocks with fuel pellets, drawn in white in Fig. 4.2, with a U-235 enrichment of 3–10% (6% on average). The fuel pellets are housed by cladding tubes made of graphite, so that an entire fuel column can be loaded or unloaded at a time. The other hexagonal graphite blocks (dark grey) have some holes for the control rods. The core has an average power density of 2.5 MW/m³. The reactor core is surrounded by a graphite reflector and stands in a pressure vessel, which is designed for a helium gas pressure of 40 bar.

The graphite blocks are loaded with a surplus of fuel at the beginning, since the fuel is also consumed here during operation. This type of reactor therefore needs burnable neutron absorbers like a boiling water reactor, e.g. gadolinium, to compensate for the excess neutrons at the beginning. Control rods are

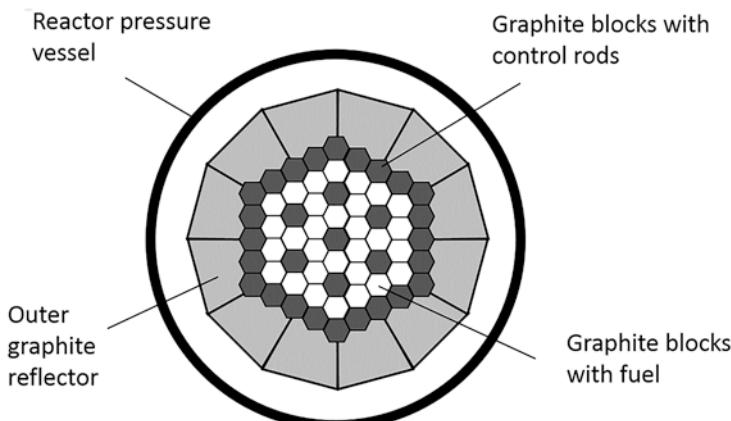


Fig. 4.2 Section through the HTTR experimental reactor in Japan

made of boron carbide, B_4C , running from the top into holes in some graphite blocks to control the power. Furthermore, all graphite blocks have a lot of cooling holes through which helium flows and heats up.

Changing the fuel in such a reactor is more complex than in water-cooled reactors, where the spent fuel elements are simply transported under water to the neighbouring fuel assembly storage pool. Unlike in the helium-cooled reactor, they are always well shielded and cooled. In contrast, the pebble bed reactor, which was developed in Germany at about the same time as the American development, had an almost ingenious concept for fuel element handling. This type of reactor uses graphite spheres of 6 cm diameter in which thousands of TRISO particles are embedded. These spheres are conveyed by a pneumatic tube to the reactor lid where they are continuously replenished. They are rolling then to a position inside the pebble bed that can only be predicted approximately. At the bottom of the reactor, the spheres are removed again via a funnel. A beam detector can then easily determine how many fission products have formed, and thus how much uranium has already been fissioned. If there is still enough fissile uranium, the spheres are simply returned to the reactor lid. Only after they have passed through the reactor many times, they need to be sorted out as spent. All this runs fully automatically.

In this way, the pebble bed reactor can also be constantly loaded with just as much fuel as it needs to be critical. So it doesn't need any surplus fuel and therefore no neutron absorbers to compensate. That makes it very safe. If you simply turn off the helium cooling in this reactor, the spheres heat up and the Doppler effect of the fuel causes the neutron absorption to increase to the point where the reactor becomes subcritical again, so it shuts itself down. Now it has only to be ensured that it does not get hotter than 1600 °C locally, so that the TRISO particles are not damaged. This is quite possible for reactors with a thermal output of a few hundred MW. The spheres then dissipate the residual heat to the outside of the reactor by heat conduction, and the peak core temperature of the reactor actually stabilizes at less than 1600 °C.

One disadvantage of the pebble bed reactor is that we cannot predict exactly how the fuel is distributed inside the core. The spheres distribute themselves largely randomly and so does the local power density. On the other hand, the average power density in such a reactor core is extremely low. In the AVR, the first pebble bed reactor, which went into operation in Jülich (Germany) in 1967, it was only 2.6 MW/m³. By way of comparison, a pressurised water reactor has an average power density of 100 MW/m³, i.e. almost 40 times that much. In such a pebble-bed reactor, it is therefore not so important where exactly which pebble is located. If sufficient safety margin is maintained, it

will not become locally too hot, even if the arrangement happens to be unfavourable. The AVR experimental reactor had a thermal output of just 46 MW and generated superheated steam in a steam generator above the reactor core to produce 13 MW of electricity. After some optimizations, the AVR already reached an average core exit temperature of 950 °C. However, the maximum temperature peak in the core could not be measured during operation, which still leads to discussions today as to whether this operation was not a little too hot.

The further development of the AVR was the pebble bed reactor THTR-300, which was built in Hamm-Uentrop in Germany and started operation in 1986 [16]. It was intended to generate 300 MW of electrical power. Inspired by the Fort St. Vrain reactor in the USA, the THTR-300 had also a reactor pressure vessel made of prestressed concrete. The reactor core had a rather large diameter of 5.6 m. Thus, it became necessary for some control rods to enter the pebble bed to control the power. The control rods had a dent at the tip that allowed the control rod to catch a sphere and to drive it through the pebble bed. Unfortunately, this didn't always work as planned and the control rods would occasionally crush a sphere. Inside the prestressed concrete vessel, steam generators stood next to the reactor core, driving a steam turbine train like the one at Fort St. Vrain.

In 1986, the reactor at Chernobyl exploded, as reported. This was a completely different type of reactor with water-cooled pressure tubes, but also moderated with graphite. As a result, the government in Germany quickly lost confidence in innovative reactors, and the utilities had already decided for pressurized and boiling water reactors anyway, which were already well developed at the time. Thus, after only a few years of test operation, the THTR-300 was finally abandoned in 1990.

Newer Concepts

The pebble bed reactor was subsequently further developed in South Africa and in China. In China, a small reactor with 10 MW thermal power and 750 °C core outlet temperature was built first, the HTR-10. It is located at Tsinghua University and went into operation in 2000 [17]. The aim was first to learn about this new technology in China, which included the production of TRISO particles, before building a large reactor. A power plant with two large HTR-PM type reactors of 250 MW thermal power each is currently under construction in Shandong Province, China. It is expected to generate 210 MW of electricity upon completion [18].

The principle of this HTR-PM is a modular pebble bed reactor with a steam generator, as outlined in Fig. 4.3. One module consists of two pressure vessels for a helium pressure of 70 bar, one of which contains the reactor core with its pebble bed and the other one a steam generator and a helium blower. The blower pushes helium at 250 °C from the top through the pebble bed, where it heats up to 750 °C on its way down. The 420,000 fuel spheres have a diameter of 60 mm and are filled with TRISO particles containing the fuel UO₂ with 8.6% enrichment. After the bad experiences with the THTR-300, the control rods do not run into the pebble bed, but outside into the reflector.

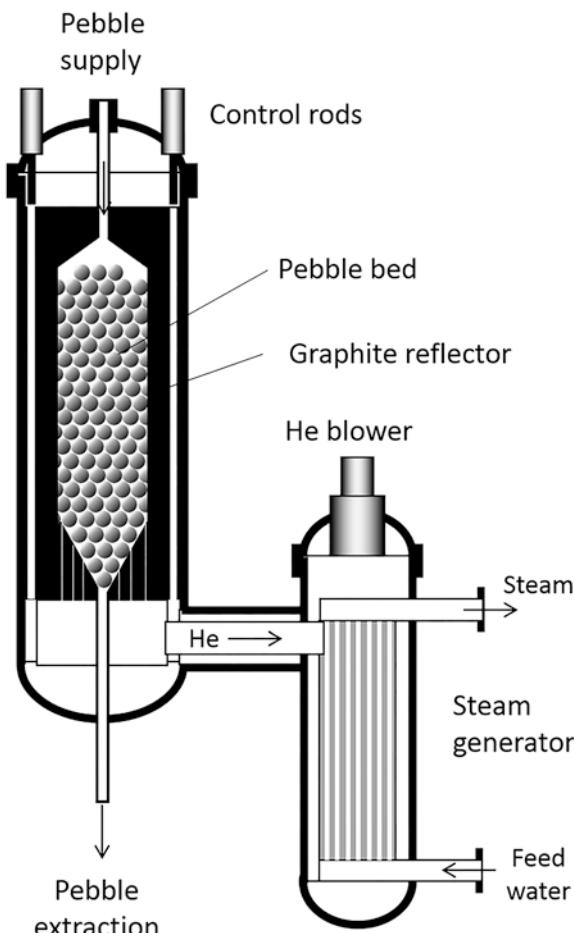


Fig. 4.3 Principle of a modular pebble bed reactor with steam generator

A second, independent shutdown system are small absorber spheres, which can fall into vertical holes in the reflector in case of emergency.

The pebble bed of the HTR-PM in China has a diameter of 3 m and a height of 11 m. This corresponds to a power density of 3.3 MW/m^3 in the core, i.e. only slightly more than in the AVR. At this diameter and low power density, the reactor can dissipate its residual heat by conduction within the spherical fill through the surrounding graphite reflector to the vessel wall. If the fan is switched off, the spheres heat up to $<1600^\circ\text{C}$ and the reactor shuts itself down by the Doppler effect. The TRISO particles survive this without damage. Therefore, the reactor does not require an emergency cooling system, but only external cooling.

The hot helium is coaxially supplied inside the helium return pipe to the steam generator. It generates superheated steam at 567°C at 132.5 bar. The helium temperature of 750°C at the core outlet is well above the temperature of superheated steam, and thus sufficiently high for this steam generator. There is therefore no need for the reactor to be operated at a higher temperature. The steam generator is deliberately positioned lower than the reactor, which is different from the AVR. If the steam generator should ever leak, no water can drip into the reactor core during standstill.

As in all pebble bed reactors, fuel spheres are constantly refilled into the reactor from above and removed again at the bottom. They pass through the reactor several times before being sorted out into an intermediate storage facility that is cooled with air.

The idea of such a modular reactor is to simply combine several of these modules to a large power plant. As long as the thermal power of 250 MW is not exceeded in each of these modules, residual heat removal can be removed via the vessel wall and a reactor emergency cooling system becomes unnecessary. In this way, a large power plant should not be much more expensive than a pressurized water reactor, despite the low power density of the core.

Such a high-temperature reactor with steam generator is admittedly recognised as a fourth-generation reactor; after all, it is also the only high-temperature reactor currently being built. However, this principle does not take advantage of the much higher core exit temperatures of 950°C that have already been run at the AVR and the HTTR in Japan. The goal of the International Forum was therefore rather a “Very High Temperature Reactor”, i.e. technologies that also make use of these higher temperatures and even make them necessary. Ideally, the heated helium is fed directly to a gas turbine, which drives a compressor and a generator.

The principle is explained in Fig. 4.4. The temperatures given there are from a study of the Japanese GTHTTR300 [19]. The hot helium at 950°C is

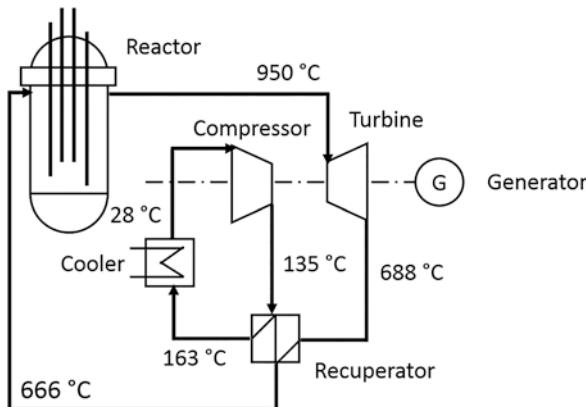


Fig. 4.4 Schematic of the high-temperature reactor GTHTR300 with gas turbine and recuperator

fed to a gas turbine, where it is expanded from 62.3 bar to 33.2 bar and cooled to 688 °C. The remaining heat is used to preheat the helium before it enters the reactor. Heat recovery is also referred to as recuperation. When it exits the recuperator, the helium has only 163 °C. It is cooled then to 28 °C by a cooler before entering the compressor. The compressor increases the gas pressure from 32.1 bar back up to 64.2 bar and pumps the helium through the recuperator at 666 °C back into the reactor. Such a power plant could achieve an efficiency of more than 50%. Compared to the 42% of the HTR-PM with steam generator, this is a significant improvement.

After the end of research on the pebble bed reactor in Germany, it was the company Eskom in South Africa that took on this principle. The PBMR with pebble bed reactor and gas turbine was to generate an electrical output of 165 MW [20]. The concept was already very far developed and there was also a site for the prototype. The International Forum soon expected the first plant to be built. There was therefore great disappointment when Eskom abandoned development in 2010 because a construction was no longer financially viable. It is a long way from the design to the construction of the first plant, which is often underestimated.

There are other interesting studies of high-temperature reactors with graphite blocks driving a gas turbine, according to the scheme in Fig. 4.4, or even with intercooling of the helium after the first compressor stages. I would like to mention in particular the design of the GT-MHR by General Atomics in the USA [21]. This design integrates the gas turbine with compressor, the generator and all heat exchangers into a second pressure vessel, which is located next to the reactor pressure vessel. The arrangement is sketched in

Fig. 4.5. The reactor should produce a thermal output of about 600 MW and the generator should provide about 300 MW of electricity to the grid. The reactor pressure vessel is 23.7 m high and 8.2 m in diameter. By comparison, a pressurized water reactor with 1300 MW of electrical power is only about half that height. All the rest of the power plant in the vessel next to it, on the other hand, is quite compact. At this power size, the reactor requires an active residual heat removal system installed below the core. The integrated design of all components in the second pressure vessel eliminates the need for a shaft sealing system between the turbine and generator; the system is hermetically sealed by the pressure vessels. Oil-lubricated bearings are no longer advisable in this case; the design therefore includes magnetic bearings, as well as a roller bearing in case the magnetic bearing should fail. The coolers look compact in the sketch, but it will still be a challenge to actually build them that compact. A simple tube and shell heat exchanger would be way too large. This power plant has never been built and there are no plans to build it, but as a vision, the design is quite spectacular.

In any case, there is still a long way to go. Let's run through a few potential incidents in order to better understand the development effort. The direct

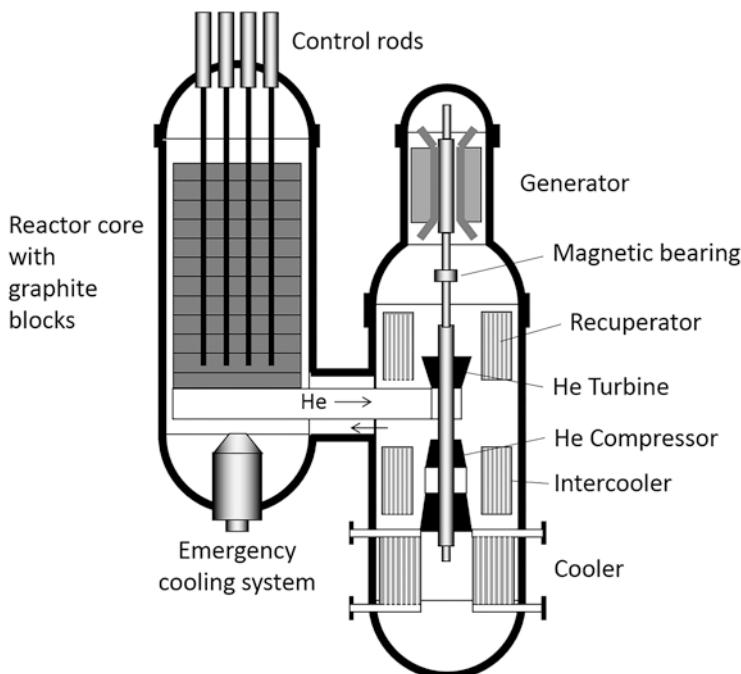


Fig. 4.5 Schematic of a high-temperature reactor with integrated gas turbine

coupling of a turbine with a reactor core is nothing new. In boiling water reactors, the steam from the reactor is also fed directly to a turbine.

Let's assume that we have a short circuit in the power grid at the GT-MHR, for example because a power pole has fallen down in a storm or because an excavator has damaged a cable. If this happens near the power plant, the generator breaker must open within 50 ms, otherwise the generator will overheat. In the case of the boiling water reactor, the turbine valve closes then in less than a second and the bypass valve of the turbine blows the steam into the condenser. This prevents the turbine from accelerating to overspeed and breaking apart. The GT-MHR's turbine rotor, however, has no turbine valve at all. It is built more like a gas turbine or like an aircraft engine. There instead, the fuel valve closes in less than a second and the turbine speed drops in case of a short circuit. However, we can't do that with the GT-MHR. The reactor core has far too much stored heat to shut down in less than a second. Consequently, the generator must not be shut down in the event of a short circuit, but an emergency consumer must be provided, such as huge resistors in the cooling tower, which quickly take over the generator's power then. Only then, the generator can slow down the speed of the turbine rotor.

Or let's assume that a compressor blade or a turbine blade breaks off. Unfortunately, this is not at all uncommon with gas turbines. Since it is not possible to shut down the turbine of a GT-MHR so quickly, this event would quickly produce a high level of consequential damage, perhaps even a total loss of the turbine rotor. The steam turbine of a boiling water reactor can quickly be separated then from the reactor. It is located in the turbine building, and the steam pipes to it have fast acting valves where they pass through the containment. So the reactor stays protected, shielded and cooled and we can replace the broken turbine later. The GT-MHR, however, does not have such containment valves.

I don't want to paint the devil on the wall now or even make you afraid of this technology. Such problems can certainly be solved by a proper design. But you don't get a reliable power plant by design and calculation alone. This requires a lot of testing, initially with a heat source that is more harmless than a reactor core. These tests will cost many years of development time and many billions of euros. Even an engine for an aircraft has to be tested for years before it is allowed to be used, and even if the manufacturer had already decades of experience with similar engines.

The steam generator of the HTR-PM is an almost primitive technology compared to a turbine rotor. Such steam generators have been built very often. If the steam turbine of this power plant should have a problem, it does not

concern the steam generator. If necessary, it could blow its steam into the condenser or even into the air. The steam is not radioactive.

A compromise between these two technologies is the high-temperature reactor ANTARES [22], which was developed by Framatome in France. There are no plans to build it today, but I would still like to briefly introduce the concept.

A high-temperature reactor with graphite blocks with a thermal output of 600 MW heats helium to 850 °C at a pressure of 55 bar. Similar to the HTR-PM, this helium is cooled to 355 °C in a heat exchanger next to it, but this time it does not do so by generating steam; instead, it heats a gas mixture of 80% nitrogen and 20% helium from 300 °C to 800 °C at a similar pressure. The reactor and its heat exchanger are inside a containment that can be isolated with fast-closing valves in case of emergency. The schematic is shown in Fig. 4.6.

The hot gas mixture drives a gas turbine and the remaining heat at the turbine outlet is used to generate steam. Such a combination of a gas turbine with a heat recovery steam generator is common in natural gas-fired power plants today. The steam cycle is shown here only in a very simplified form. It requires a condenser, condensate pumps and feedwater pumps and usually has several steam turbines and a reheat, which are not shown here. The gas mixture is recompressed after it leaves the steam generator. This heats it up to about 300 °C before it is returned to the heat exchanger inside the containment.

In this design, the generator produces a net output of about 300 MW. The power plant thus has an efficiency of about 50%. Such a concept separates the turbine part from the nuclear primary system similar to a pressurized water

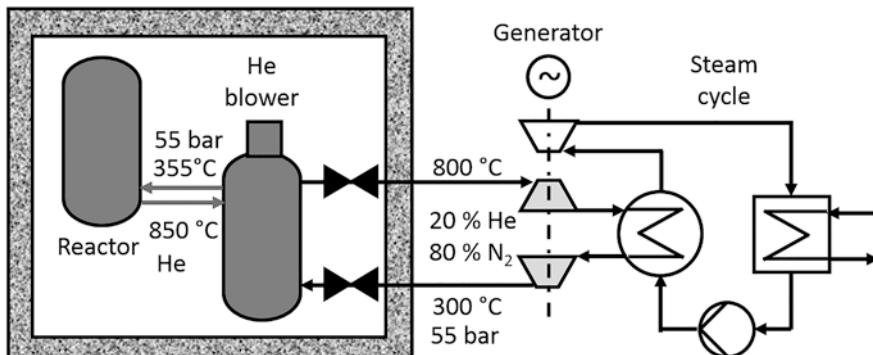


Fig. 4.6 Schematic of the ANTARES high-temperature reactor with gas turbine and tertiary steam cycle

reactor. The water-steam cycle is a conventional technology that is already available today. The nitrogen-helium gas turbine, on the other hand, would still have to be developed and tested.

Inherent Safety

So what is the advantage of a high-temperature reactor? The developers of the pebble bed reactor liked to call it “inherently safe”. By this they meant: the reactor does not need an emergency cooling system to remove the residual heat. If you turn off the helium cooling, the fission chain reaction will physically stop by itself. The reactor gets very hot then, but the graphite spheres survive it, and the residual heat is dissipated by heat conduction to the outside of the pressure vessel. A core meltdown is therefore physically impossible. Of course, the pressure vessel has to be cooled then from the outside, otherwise it won't work. Some designers would therefore have liked to put the reactor outside, without a containment, so that the ambient air could keep the reactor cold without a fan. Alternatively, the building in which the reactor is located could be well ventilated.

In the meantime, the research centre in Jülich has begun to dismantle the old AVR experimental reactor. Since the radiation of the spheres had already reduced far enough, they could examine them more closely. Unfortunately, some surprises emerged that call this concept into question again [23]. The spheres had a high concentration of radioactive strontium, $^{90}_{38}Sr$, and caesium, $^{137}_{55}Cs$ and $^{134}_{55}Cs$ at the surface. These are metallic fission products with longer half-lives that can be detected for several years after the shutdown. Apparently, these fission products had diffused out through the ceramic layers of the TRISO particles during reactor operation, while helium and iodine remained safely confined. In general, diffusion increases exponentially with temperature. So the strontium and caesium diffusion must have happened when the reactor was operating with a core exit temperature of 950 °C. For a short time it was even 990 °C by mistake. The spheres had survived this operation mechanically well and the operators were proud that the reactor allowed such high exit temperatures. How hot it was at the hottest point of the pebble bed could not be measured. Since the spheres constantly change their position, it is not possible to place a thermocouple or a neutron detector between them. However, the amount of fissile uranium in each sphere is slightly different depending on how many times it has passed through the core, and so it is quite possible that random spots of higher power may form. Furthermore, the spaces between the spheres through which helium flows can be different

depending on the arrangement of the spheres, so it is also impossible to predict the local temperature of the spheres accurately. This is a general disadvantage of the pebble bed reactor compared to the high temperature reactor with graphite blocks.

As the spheres slowly move downwards, they rub against each other and some graphite dust is produced. The radioactive strontium and caesium are therefore also contained in these dust particles and are carried away with the helium flow. If the reactor had leaked, these dust particles would have escaped into the environment. It follows that the pebble bed reactor would have needed a gas-tight containment, i.e. a third barrier. The first barrier, the TRISO particle, was obviously not as reliably tight as previously thought. However, a gas-tight containment must be able to withstand the pressure that occurs when helium escapes. During normal operation, it must also be kept at negative pressure so that air from outside enters rather than gas escapes from inside, it needs a filtering system to clean the extracted gas, and it needs cooling so that the reactor does not heat up the containment. All of this cannot be done physically alone, but ultimately requires an active or passive safety system. This is feasible, but that makes the formerly simple system quite expensive. It seems more meaningful to me to develop the TRISO particles further so that they will contain the fission products better in the future.

On the other hand, a gas-tight containment could also be filled with an inert gas. This would also eliminate two further problems which, although unlikely, should nevertheless be taken into account in the careful design of the power plant:

- If the steam generator tubes should leak, steam enters the reactor. At the hot surface of the spheres, it reacts with graphite to form carbon monoxide, CO, and hydrogen, H₂, and the pressure rises. The pressure relief valve must therefore vent gas, but in an inert gas it cannot do any further damage.
- In the certainly very unlikely event that the reactor gets a large break, air could enter the reactor and ignite the graphite. If the reactor is located inside a containment with inert gas, this case can also be easily ruled out.

In contrast to the pebble bed reactor, a high-temperature reactor with graphite blocks can precisely be calculated. One can definitely say where which fuel is located and how large the cooling channels are. Furthermore, the blocks do not move and no graphite dust is produced. This type of reactor would therefore be much more suitable for high temperatures of 950 °C. However, according to current knowledge, a containment would probably still be built around it.

Further Developments

For what do you actually need such high temperatures? At 950 °C, a high-temperature reactor with a gas turbine can achieve a power plant efficiency of more than 50%, while the HTR-PM in China with a core exit temperature of 750 °C reaches only 42%. With higher efficiency, the reactor consumes less uranium per kWh of electricity generated. As long as uranium is cheap, however, there is hardly any financial advantage to be gained from this. This is offset by the higher waste volume. If the HTR-PM has a power density 30 times smaller than that of a pressurized water reactor, it will generate about 10 times as much waste, which we will have to fill then into Castor casks and put into an interim storage. Not 30 times as much, because the HTR-PM has about twice as much fissile U-235 in its fuel as the pressurized water reactor, and it has a higher efficiency, which is just 35% for the pressurized water reactor. So the spheres generate significantly more kilowatt-hours of electricity. Nevertheless: the high temperatures can hardly be justified with the better efficiency alone.

When it comes to the high-temperature reactor, however, we should not only think about electricity generation. The high core exit temperatures also enable chemical processes that are hardly possible at the low temperatures of a pressurised water reactor. The production of hydrogen using the sulphur-iodine process is often mentioned as an application, Fig. 4.7. Hydrogen iodide, HI, and sulphuric acid, H_2SO_4 , are produced from iodine, sulphur dioxide and water in an exothermic reaction, i.e. without the addition of energy. This is the well-known Bunsen reaction. Hydrogen iodide and iodine

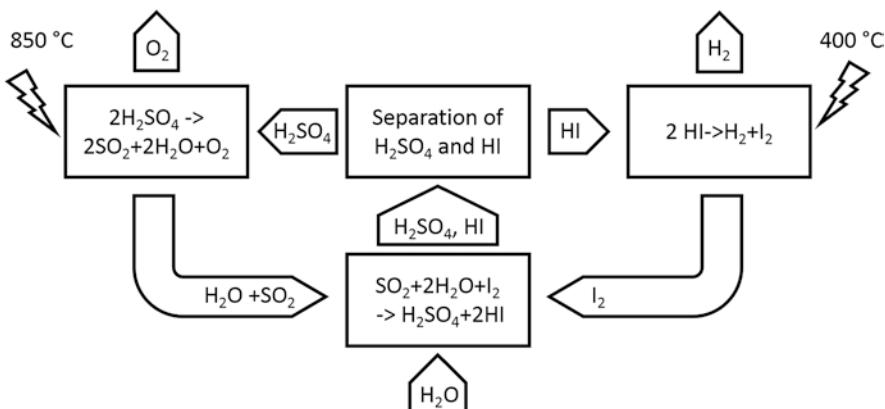


Fig. 4.7 Simplified representation of the sulphur-iodine process

on the one hand and sulfuric acid on the other hand are immiscible and can be easily separated. At about 400 °C, the hydrogen iodide splits again into iodine and hydrogen and at about 850 °C the sulphuric acid splits into sulphur dioxide, oxygen and water vapour. Nuclear process heat is thus used to obtain the reactants I₂ and SO₂ of the Bunsen reaction again, while hydrogen and oxygen can be extracted as end products.

Hydrogen could gain strategic importance in the future for transport as a CO₂-free fuel. Or one could use hydrogen and oxygen to produce steel from iron ore without oxidizing coal, i.e. without producing CO₂. Alternatively, however, hydrogen could also be produced from electricity by electrolysis. Then it doesn't matter where the electricity comes from, a nuclear power plant, a pressurized water reactor, or a high-temperature reactor, or even from renewables. Ultimately, it's just a matter of cost.

How economical is such a high-temperature reactor? The manufacturers of the HTR-PM in China have calculated that six of these modules combined to form a power plant, which would then generate more than 600 MW of electrical power, should only be about 10–20% more expensive than a pressurised water reactor with this output. However, the new pressurized water reactors that are now being manufactured in China with more than 1200 MW are still somewhat cheaper. So this high temperature reactor will certainly not be cheaper than a pressurized water reactor.

For the high-temperature reactor GTHTR300 with gas turbine and 950 °C core outlet temperature, a Japanese study comes to a much better result [24]: The capital costs of a power plant consisting of four modules GTHTR300 with 274 MW electrical output each should be about 25% cheaper than those of a pressurized water reactor with 1180 MW, i.e. with about the same total power output. The fuel costs were to be comparable. However, the cost of developing and testing this new technology was not included. The bottom line is that this is not a convincing cost saving either, and so to date no investor has been found for this project.

The Fort St. Vrain reactor and the THTR-300 had another peculiarity: the fuel was a mixture of thorium and highly enriched uranium. Now this is neither economical nor proliferation resistant, because highly enriched uranium is directly nuclear weapons grade when chemically separated from thorium. Thorium itself, on the other hand, is not a fissile material and therefore contributes only indirectly to the production of nuclear weapons. The fuel was merely a first attempt towards a sustainable fuel supply. I will go into more detail in the next chapter. The reactors considered in the Generation IV International Forum, on the other hand, use only a conventional fuel of slightly enriched UO₂. It can be enriched somewhat higher than in the

pressurized water reactor because the TRISO particles allow higher burnup than Zircaloy fuel rods. Then the proliferation risk remains low, comparable to a pressurised water reactor.

Conclusion

High temperature thermal reactors are cooled with helium and moderated with graphite. There are two lines of development: one with graphite blocks and a second one with graphite spheres. Compared to pressurized water reactors, the reactor core in both concepts has only a low power density. This is initially a significant cost disadvantage. On the other hand, the coolant helium can become much hotter than water, which results in a significantly higher efficiency than in pressurized water reactors. A cost advantage of the high-temperature reactor could arise if the containment and emergency cooling systems could be omitted.

Safety

Usually, the high-temperature reactor has no emergency cooling system and the concept of this reactor relies on the fact that it also does not need emergency cooling. If the cooling fails, for example because the fan fails or because the reactor has a leak, the temperature can only rise to a maximum of 1600 °C for physical reasons and the TRISO particles, the first barrier of the protection system, remain gas-tight. This has earned this type of reactor the rating “inherently safe”. Unfortunately, the new findings from Jülich call this inherent safety into question again: even at significantly lower temperatures, such as 1250 °C, metallic fission products diffuse through the closed shell of the TRISO particles.

At the low core exit temperatures of the HTR-PM, which is currently being built in China, the simple safety concept still seems to work. However, this was not the aim of the fourth-generation Very High Temperature Reactor. In the case of reactor concepts with core exit temperatures of 950 °C, the entire safety concept must be rethought. In reactor cores with graphite blocks, the maximum temperature in the core can be predicted far more accurately than in a pebble bed reactor, in which the fuel distribution and the cooling channels are more random. Presumably, one will therefore be able to achieve higher core exit temperatures with a reactor with graphite blocks than with a pebble bed reactor.

Economic Competitiveness

Only the high-temperature reactor with gas turbine and with high core exit temperatures has the potential to become more economical than a pressurized water reactor. But that is still a very ambitious goal. Such a reactor has never been built. If, on the other hand, only steam is generated with the hot helium, a similarly expensive steam cycle to that of a pressurised water reactor will have to be built and there will no longer be any cost advantage over the pressurised water reactor.

Sustainability

The new concepts of the high-temperature reactor use uranium fuel, UO_2 , with enrichment up to 9%. There are no plans at present to recycle the fuel after use. The higher efficiency of the high temperature reactor saves some uranium compared to the pressurized water reactor, but the much larger volume of waste increases the cost of disposal in return. This is not sustainable. The option of using thorium to build a closed fuel cycle for this type of reactor will be discussed in more detail in the following chapter.

Proliferation Resistance

As long as the high-temperature reactor uses only slightly enriched uranium, the risk of nuclear fuel proliferation is as low as in the pressurized water reactor.

Glossary

Burnable neutron absorber	An absorber material (e.g. gadolinium) which loses its absorption property through neutron absorption
Diffusion	Substance transport of atoms or molecules by concentration gradients
Pebble bed reactor	A reactor having a core consisting of a bed of graphite spheres
Pyrographite	A special crystal structure of graphite with strongly anisotropic material properties
TRISO particle	A spherical fuel element coated with three layers of pyrographite and SiC



5

The Nuclear Fuel Cycle

The generation four reactor designs we have discussed so far have a thermal neutron spectrum. They are hotter at the core exit than third-generation reactors, which results in higher efficiency and hopefully lower specific costs for the power plant. On the other hand, the other fourth-generation reactor concepts, which I want to introduce to you next, have a fast neutron spectrum. However, to understand the background of these concepts, we first need to look at the nuclear fuel cycle.

What happens to the nuclear fuel after it has been spent? Immediately after being discharged from the reactor core, the fuel still has such a high decay heat so that it has to be cooled for at least a few years. But can it be recycled afterwards? We also recycle paper, glass, cans, plastic bottles and much more. Why not uranium? It can be done indeed, and the basic technologies are more than 50 years old now. But let us first take a look at the composition of spent fuel. Then it will be easier to discuss what options we have for recycling. For that we need the nuclide chart again.

Suppose we use uranium with 5% enrichment in a pressurized water reactor. This is the maximum enrichment used today for new pressurized water reactors. Then the fresh fuel assembly has 50 g of U-235 and 950 g of U-238 per kg of uranium. The thermal neutrons in the reactor core can fission U-235 and we get fission products: usually a larger one with a mass number of about 125–155 and a smaller one with a mass number of about 80–110. The nuclide chart, such as the Isotope Browser, gives the probability that a nuclide is produced by fissioning with thermal neutrons. It is called *cumulative thermal neutron fission yield*, abbreviated also as *cumulative thermal n FY*. The fission

product decays along a *decay chain*, which we can also find in the nuclide chart, to a stable isotope. This decay releases a lot of energy, which we call residual heat. Half a minute after the reactor is shut down, that's still 3% of the reactor's power, after 5 min it's still 2%, and after 3 h it's only 1%. So most of the fission products decay quite quickly. On the other hand, after 30 years of cooling in the spent fuel pool and in the interim storage facility, there is still 0.1% of the original reactor power left.

U-235 is called a fissile material. The heavier isotope U-238 is also fissionable, but it cannot be fissioned with thermal neutrons. Instead, it likes to absorb a thermal neutrons (*thermal neutron capture*). We call this a fertile material. Let us take a look at the top part of the nuclide chart to understand what it becomes then. I have cut out a small section of it, greatly simplified, in Fig. 5.1. Analogous to the Isotope Browser, I have labelled β^- -decays in green, β^+ -decays in yellow, and α -decays in pink. When a neutron is absorbed, we move one position to the right in the nuclide chart.

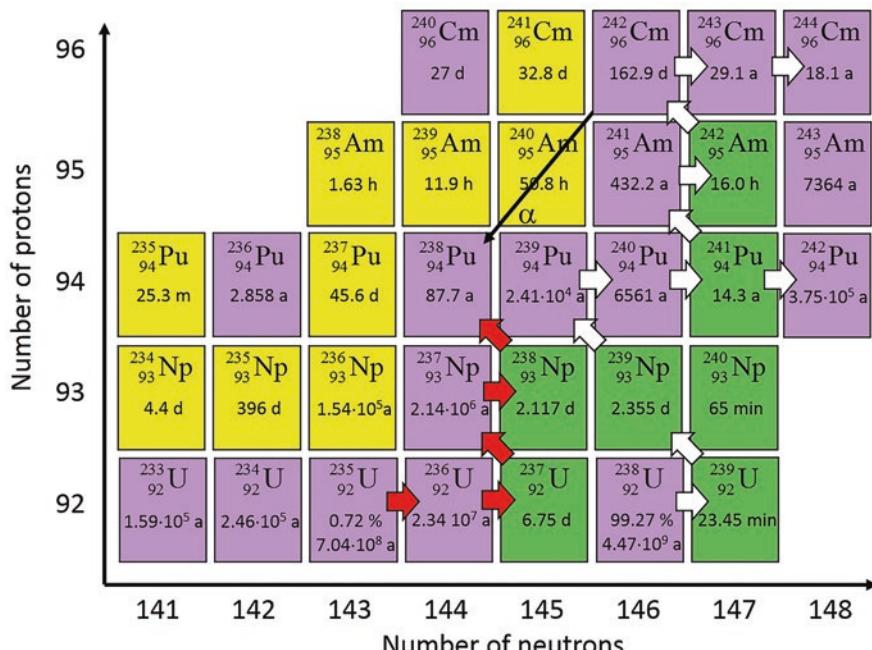


Fig. 5.1 Uranium and transuramics in the nuclide chart. White arrows: Formation of transuramics from U-238 by neutron absorption and β^- -decay; red arrows: Formation of Pu-238 from U-235; black arrow: formation of Pu-238 by α -decay of Cm-242

Formation of Plutonium in the Reactor

The generated U-239, correctly named $^{239}_{92}\text{U}$, decays very soon, namely with a half-life of 23.45 min to neptunium, $^{239}_{93}\text{Np}$, which in turn decays with a half-life of 2.356 days to the plutonium isotope $^{239}_{94}\text{Pu}$. With these β^- -decays, we move to the top left of the nuclide chart. Since all this happens very quickly, larger amounts of $^{239}_{92}\text{U}$ and $^{239}_{93}\text{Np}$ cannot build up. There is also hardly any time for these nuclides to absorb a neutron in turn. The plutonium isotope $^{239}_{94}\text{Pu}$, on the other hand, has a fairly long half-life of 24,110 years before it becomes the uranium isotope $^{235}_{92}\text{U}$ by an α -decay. So more and more Pu-239 is produced during reactor operation, which is good because Pu-239 is also a fissile material. The cross section for fission by a thermal neutron is even 752 barn. By comparison, U-235 has only a cross section of 566 barn. Neutron absorption thus turns U-238, which cannot be fissioned with thermal neutrons, into a fissile nuclide. At the end of burnup, when most of the U-235 has already been spent, the generated Pu-239 actually contributes as much to reactor power as U-235.

We usually express the burnup as thermal energy produced from 1 kg of uranium. Since fission of 1 g of U-235 yields a thermal energy of about 1 megawatt-day (MWd), we can therefore achieve a thermal energy of 50 MWd by fissioning 1 kg of uranium enriched to 5%. So the maximum burnup possible would be 50 MWd/kg_{uranium}. In fact, we get a slightly higher burnup because Pu-239 contributes towards the end of the operating time. On the other hand, U-235 cannot be completely fissioned and a small residual amount, <0.5%, remains in the fuel at the end.

The produced Pu-239 lives very long and can therefore absorb a neutron as well, forming $^{240}_{94}\text{Pu}$. Similar to U-238, we cannot fission this nuclide with thermal neutrons. Pu-240 has a fairly long half-life of 6561 years. Therefore, an inventory of Pu-240 builds up as well with the burnup. Neutron absorption in Pu-240, on the other hand, produces a fissile isotope again: Pu-241. It has a cross section of 1010 barn when fissioned with thermal neutrons, so it is even better than Pu-239. The build-up of all these Pu isotopes with burnup is shown in Fig. 5.1 with white arrows and as a sum in Fig. 5.2. The proportion of the isotope Pu-239 reaches a maximum of just over 5 g/kg_{uranium} at a burn-up of about 40 MWd/kg_{uranium}. Afterwards, as much Pu-239 is produced from U-238 as is destroyed again by nuclear fission and neutron absorption. The proportion of Pu-240, on the other hand, continues to increase as is hardly fissioned.

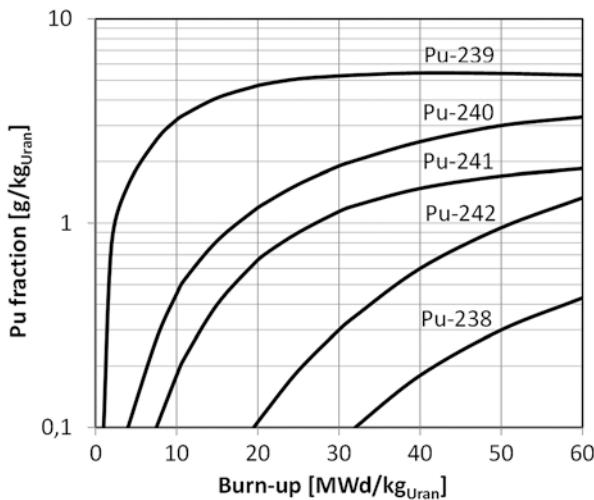


Fig. 5.2 Build-up of plutonium isotopes in the fuel during the burn-up of uranium with 5% enrichment

The isotope Pu-241 has a comparatively short half-life of 14.3 years. It is a β^- -emitter and decays to americium, ^{241}Am , which is a fertile material again. In general, we can say that only those heavy nuclides are fissile that have an odd number of neutrons. ^{241}Am however, has $241 - 95 = 146$ neutrons, an even number.

But the isotope Pu-241 can also absorb a neutron. Despite its β^- -decay, it lives long enough to do so. Then we get Pu-242. This isotope has an even number of neutrons and is therefore not fissionable with thermal neutrons, i.e. it is not fissile. Therefore, we see in Fig. 5.2 that the fraction of Pu-242 continues to increase as well.

The lighter isotope Pu-238 is formed quite differently. First, we get it by neutron absorption from U-235. When a neutron is absorbed in ^{235}U , it becomes ^{236}U , which is pretty much stable with a half-life of 23.4 million years, and from that, by further neutron absorption, we get ^{237}U . This isotope has only a short half-life of 6.75 days, during which it becomes ^{237}Np via a β^- -decay. With a half-life of 2.144 million years, this neptunium isotope is also quite stable. However, it can in turn absorb a neutron, and the resulting ^{238}Np decays then rapidly to ^{238}Pu . This process is indicated by red arrows in Fig. 5.1. A second formation process of ^{238}Pu runs via neutron absorption of ^{241}Am and β^- -decay of the resulting ^{242}Am to the curium isotope ^{242}Cm , which then becomes ^{238}Pu by an α -decay.

In the course of the burn-up, therefore, five different plutonium isotopes are produced, depending on the initial enrichment and burn-up. We also get americium and curium isotopes, which are highly radioactive, and some neptunium isotopes. What will happen if we simply pack the fuel assemblies then into a suitable cask and store them?

Fuel Composition at Direct Disposal

Ten years after shutdown, 1 kg of spent fuel from a typical pressurised water reactor, discharged at a burn-up of 33 MWd/kg_{uranium}, has an inventory as compiled in Table 5.1.

In order to assess how dangerous this inventory will be for us humans, we must convert it to its radiotoxicity. Dividing the mass inventory by the respective atomic weight and multiplying by the Avogadro constant, we first obtain the number N of atoms in a kg of uranium. Then the decay law is used to calculate the activity A , i.e. how many decays the respective substance undergoes per second:

$$A = \lambda N$$

The decay constant λ is calculated from the half-life $t_{1/2}$ as

$$\lambda = \frac{\ln 2}{t_{1/2}}$$

Table 5.1 Fuel composition per kg fuel of a pressurized water reactor after discharge

	Uranium	955.4 g
Transuramics	Neptunium	0.5 g
	Plutonium	8.5 g
	Americium	0.6 g
	Curium	0.02 g
Long-lived fission products	Iodine I-129	0.2 g
	Technetium Tc-99	0.8 g
	Zirconium Zr-93	0.7 g
	Caesium Cs-135	0.3 g
Shorter-lived fission products	Caesium Cs-137	1.0 g
	Strontium Sr-90	0.7 g
Stable nuclides	Lanthanides	10.1 g
	Other elements	21.8 g

The numerator is the natural logarithm of the number 2, i.e. approximately 0.693. The activity is given in becquerel (Bq), i.e. decays per second. Finally, we multiply A by the dose coefficient F to calculate the radiotoxicity. The dose coefficient is different for each nuclide [25]. It further depends on how we incorporated the nuclide and which body organ was damaged. It is expressed in sievert per becquerel (Sv/Bq).

A typical accident scenario for the assessment of a repository could consider the case that traces of the nuclides stored there get into the drinking water. If one consequently assumes an incorporation of these radionuclides by ingestion, one obtains the radiotoxicity of the discharged fuel as a function of time after discharge, as shown in Fig. 5.3. The double logarithmic plot unfortunately distorts the curves considerably, but there is little else I can do to show these large time periods and the large differences in radiotoxicity. Since the unit $\text{Sv}/\text{kg}_{\text{Uranium}}$ does not tell us much and we do not yet know what quantities we are talking about, I have added the radiotoxicity of natural uranium to this diagram. We find dissolved uranium in mineral water and seawater, for example, but only in concentrations of a few $\mu\text{g}/\text{litre}$. Consequently, we also find these concentrations in many plants, in meat and in fish. For us this is harmless.

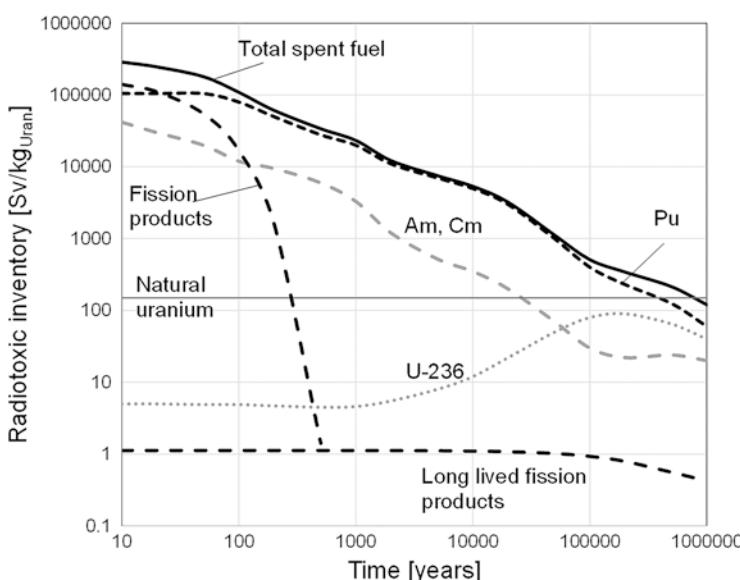


Fig. 5.3 Radiotoxicity of the spent fuel of a pressurised water reactor in the repository

At the beginning, 10 years after the reactor has been shut down, the fission products still form the main part of the radiotoxicity. After a few centuries, however, the short-lived fission products have decayed and only the long-lived ones remain. However, their large half-lives mean a very small decay constant, and the long-lived fission products therefore produce little activity. The sum of all these long-lived fission products is about a hundred times smaller than the radiotoxicity of natural uranium. We can therefore neglect this contribution.

The radiotoxicity of the spent fuel in the repository comes then exclusively from transuranics, predominantly from the isotopes of plutonium. Americium is essentially Am-241, which is formed from β^- -decay of Pu-241. It thus builds up first during final disposal and then decays again with a half-life of 432 years to Np-237. However, this neptunium has a half-life of more than 2 million years, and consequently only a very small decay constant. It hardly produces any activity and is therefore not shown in Fig. 5.3. Curium, after 10 years, is essentially just Cm-244. It has a half-life of 18 years, so that more than 99.9% of it has become Pu-240 after two centuries.

The radiotoxicity of all transuranics, in the long run, also comes from the nuclides along their decay chains, which you can view in the Isotope Browser. Finally, after millions of years, it all becomes lead. The activity of the nuclides from the decay chains has been included in the activity of the transuranics in Fig. 5.3 to simplify the figure.

U-236 is produced from neutron absorption in U-235 during reactor operation, but it is also the decay product of Pu-240, so that we see an increase in U-236 after a few millennia in Fig. 5.3. However, with its long half-life, U-236 produces little radiotoxicity and thus does not cause a serious disposal problem.

From this consideration we can already derive two good reasons to recycle plutonium and not to put it into a repository:

- (a) We stretch the fuel reserves of uranium by indirectly fissioning U-238, of which there is much more than U-235.
- (b) We would relieve the repository of hazardous materials that decay extremely slowly.

The recycling of plutonium is also known as reprocessing. I would like to explain now briefly how this works.

Separation of Uranium and Transuramics

A mature technical process for separating plutonium and uranium from spent fuel is the PUREX process [26]. A large PUREX plant is located, for example, at La Hague in Normandy. There, the fuel assemblies are first chopped into small pieces and dissolved in hot nitric acid. The Zircaloy pieces of the cladding tubes and the spacers of the fuel assemblies are retained and disposed as low-level waste. The separation of uranium and plutonium from the nitric acid is performed with tributyl phosphate (TBP), which can form a chemical complex with uranium and plutonium and thus chemically binds these substances. The TBP is diluted with kerosene to form a water-insoluble organic phase. The nitrates are dissolved in the aqueous phase and are present there as U^{4+} and Pu^{4+} ions.

To separate U and Pu, the TBP-kerosene mixture on the one hand and the nitric acid with the dissolved fuel on the other hand are placed in a *mixer-settler*, as shown in Fig. 5.4. Both phases are first stirred in the mixer-settler so that many bubbles are formed and thus a large water-kerosene interface is formed, over which U and Pu are extracted. The mixture is then allowed to settle and the lighter organic phase settles on top. Repeating the procedure

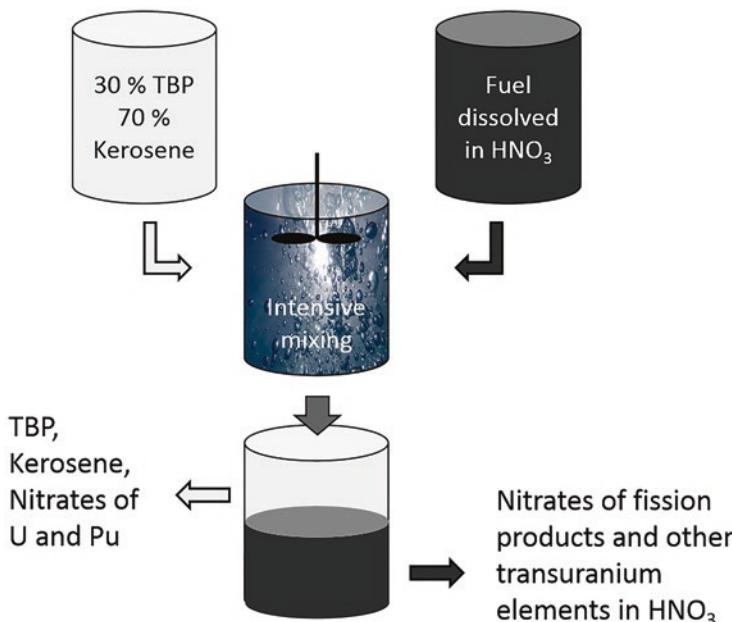


Fig. 5.4 Principle of separation of uranium and plutonium in mixer-settlers

several times, U and Pu can be removed with the organic phase, and the fission products, together with Np, Am and Cm, remain in the aqueous phase.

In a further step, Pu can be separated from U in an analogous manner. U and Pu can only be bound by TBP if they are tetravalent ions. To separate U and Pu, the plutonium ion must therefore be chemically reduced to trivalent plutonium. It can no longer be taken up then by TBP and remains in the aqueous phase. In the final step, the uranium is re-extracted from the organic phase and leaves the process in a nitric acid solution. U and Pu can then be denitrated and evaporated in separate product streams before being oxidized to UO_2 and PuO_2 .

The whole process runs remotely controlled as a continuous process behind a thick concrete shield. Care must be taken to ensure that there is no critical arrangement of the fissile materials, which are well moderated in the aqueous solution. Otherwise a fission chain reaction will start. This has happened indeed in the past. The containers must therefore not be too large. Furthermore, the concentration of the fissile materials in the respective solution must permanently be measured and monitored. Such nuclear material management and safeguard system is also necessary to check if fissile material has been diverted. Unfortunately, these many analyses generate also some radioactive waste. Here the PUREX process has still some potential for improvement and there are currently several research projects on how to reduce this waste considerably. The technical approach to this is microfluidics, with the help of which a chemical analysis can already be carried out on a single droplet.

In principle, the PUREX process could also be used to separate neptunium, but there has been no need for this so far. There is actually no repository problem with neptunium because the neptunium isotopes either decay quickly or have very low activity. However, since the transuranic elements americium and curium would radiate much longer in the repository than the fission products, additional separation processes have been developed in recent years to separate these two elements as well [27]. I have sketched the process in a simplified form in Fig. 5.5. First, the DIAMEX process separates the fission products from Am and Cm (which belong chemically to the group of minor actinides). Unfortunately, this process separates as well the elements from the lanthanide (Ln) group, which behave very similarly chemically. Therefore, another separation process is necessary, the SANEX process, which finally separates Am and Cm from the lanthanides. The extraction agents are again complex organic compounds, similar to TBP. However, the equipment for this exists only on a laboratory scale so far.

In the meantime, the first fuel pellets with americium have already been produced and tested in a reactor. Curium, on the other hand, has a very high

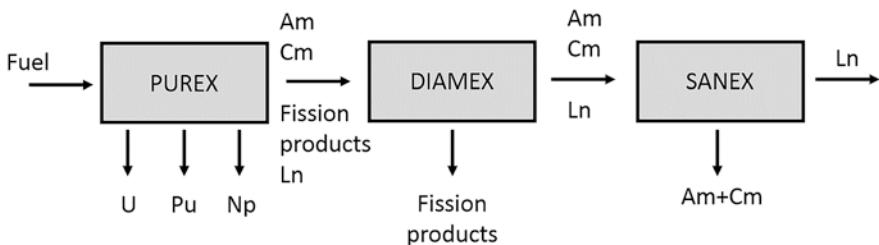


Fig. 5.5 Partitioning of transuranium elements from the spent fuel

activity and occasionally spontaneously fissions by itself. It is questionable whether one really wants to recycle such a fuel or whether it would not be better to wait until the curium has decayed by itself.

All waste that is not recycled as fuel is vitrified. To do this, the liquid waste stream is first concentrated and then converted into oxides. If you put these oxides together with glass into a melting furnace, the oxides combine with the glass and can be cast off as a solid block of glass. You simply pour this glass into stainless steel containers and let it solidify. This “glass mould” is leach-proof, even if the glass should form cracks, can be welded into a thick-walled repository container and taken to the repository. Ideally, only fission products would enter the repository. Then the activity of the waste will be gone after a few hundred years. It should surely be possible to design a cask for a nuclear waste repository for these glass moulds in such a way that it remains leak-proof for 500 years, regardless of which repository it is placed in.

Recycling in Thermal Reactors

Until the 1990s, a common disposal concept for German pressurised and boiling water reactors was to transport the spent fuel to La Hague for reprocessing. There, uranium and plutonium were separated and the remaining waste, i.e. fission products and the minor actinides Np, Am and Cm, were vitrified. The vitrified waste was returned to Germany, where it is stored now in interim storage facilities awaiting final disposal. From the separated plutonium and from depleted uranium, new fuel assemblies were produced, so-called MOX fuel assemblies, which were reused in some German nuclear power plants. In so far, recycling of plutonium in thermal reactors has already 40 years history in Germany.

MOX fuel assemblies cannot be produced quite as easily as enriched uranium fuel assemblies. They radiate from their beginning and produce heat.

The five plutonium isotopes are α - or β^- -emitters, partly accompanied by weak γ -rays, which can be easily shielded even by thin walls. We find the decay energies of each nuclide in the Isotope Browser ordered as *Decay Radiations*. For all plutonium isotopes, γ -decays are also given, but only with a low intensity of much less than 1%. So they are very rare. However, small dust particles containing plutonium are dangerous, especially if they get into our lungs. The fuel must therefore be assembled in glove boxes, today largely automated. Such a production facility is located in Marcoule, France, for example.

The heat produced by these MOX fuel assemblies comes essentially from Pu-238. The kinetic energy of the emitted α -particles is given by the Isotope Browser as $Q\alpha = 5593.27$ keV. Converting this energy into joules and multiplying this number by the activity of 1 g of Pu-238 gives a specific power of 0.568 W/g.

The finished MOX fuel assemblies are then transported to the nuclear power plant and stored there with the spent fuel assemblies in designated storage pools until they are used. Even though these MOX fuel assemblies contain the same fraction of fissile material as the original uranium fuel assemblies, the reactor core must be redesigned for this fuel, because the plutonium isotopes have different cross-sections for fission and absorption and produce slightly more neutrons during fission than U-235. It is certainly a considerable success for the development of recycling concepts for plutonium that all this has worked out without any problems to date.

In order to obtain the highest possible fraction of fissile plutonium, the spent fuel is processed as soon as possible. Already after about 5 years of cooling in the spent fuel pool, it was transported away to La Hague, otherwise too much of the fissile Pu-241 would have decayed. In this way, there was also no need for dry interim storage of the spent fuel.

However, the recycled MOX fuel can never completely replace the fresh uranium fuel. Let us consider a new reactor with fuel assemblies with U-235 of 5% enrichment. They thus contain 50 g U-235 per kg uranium. As we can easily see in Fig. 5.2, the reactor produces only 5.4 g Pu-239 and 1.7 g Pu-241 from this, of which perhaps only 1 g remains by the time it is reused in the reactor. Thus, only about 12–13% U-235 can be saved. Conventional uranium fuel still makes up the larger proportion of fuel in the reactor core.

Without plutonium, the vitrified waste contains only 10% of the radiotoxicity of directly disposed fuel assemblies as soon as the fission products have decayed, as we can see in Fig. 5.3. This looks good at first and would mean a significant reduction in the burden on the repository. However, the thermal

reactor can only fission Pu-239 and Pu-241, and the fractions of Pu-238, Pu-240 and Pu-242 keep increasing during reuse. It may be possible to recycle the MOX fuel a second time, but at some point we will be left with a quantity of even-numbered plutonium isotopes that will ultimately have to go into the final repository if we cannot find another use for them. Under the bottom line, only about half the radiotoxicity of direct disposal is avoided in the long term. Recycling in thermal reactors is therefore not yet a truly sustainable concept.

Sustainable Concepts

The goals of the Generation IV programme go well beyond this. Ideally, in the fourth generation, no new fuel should be needed and nothing but fission products should end up in the repository. This can be achieved with fast reactors, i.e. reactors that fission with a fast neutron spectrum, but not with those with a thermal spectrum. I would like to explain this briefly below.

Let us consider again the neutron balance of nuclear fission. In order to keep the chain reaction of nuclear fission stationary, we need at least one neutron per fission, which is left over from the prompt neutrons after losses due to leakage and absorption and is available for the next fission reaction (see Fig. 2.4). If we now require, in addition, that a new fissile nucleus is produced for each fissioned nucleus, we need two neutrons per fission, one for the next fission, as before, and another one to be absorbed in the fertile nuclide U-238. After decay of the resulting U-239 and then of Np-239, it finally becomes a fissile Pu-239 nuclide. Even such a reactor will certainly have some neutron leakage to the outside of the reactor core and some neutron absorption elsewhere, i.e. not in the U-238 and not in the fissile material. So depending on the design of the reactor, we need 2.5–3 prompt neutrons after nuclear fission for each neutron that was absorbed by the fissile nuclide causing nuclear fission. This ratio is also called the regeneration factor of nuclear fission of this nuclide.

The regeneration factor depends on the fissile nuclide used and on the kinetic energy of the neutron that fissions this nuclide. Figure 5.6 shows this regeneration factor for three typical fissile nuclides, namely U-235, Pu-239 and U-233. The prompt neutrons have an energy of 1–10 MeV. Even if we do not use a moderator, they are still slowed down somewhat by collisions with heavy atomic nuclei. Therefore, a reactor without a moderator has a broad maximum in the neutron spectrum, ranging from 50 keV to 5 MeV. This is called a fast neutron spectrum. On the other hand, if we decelerate the

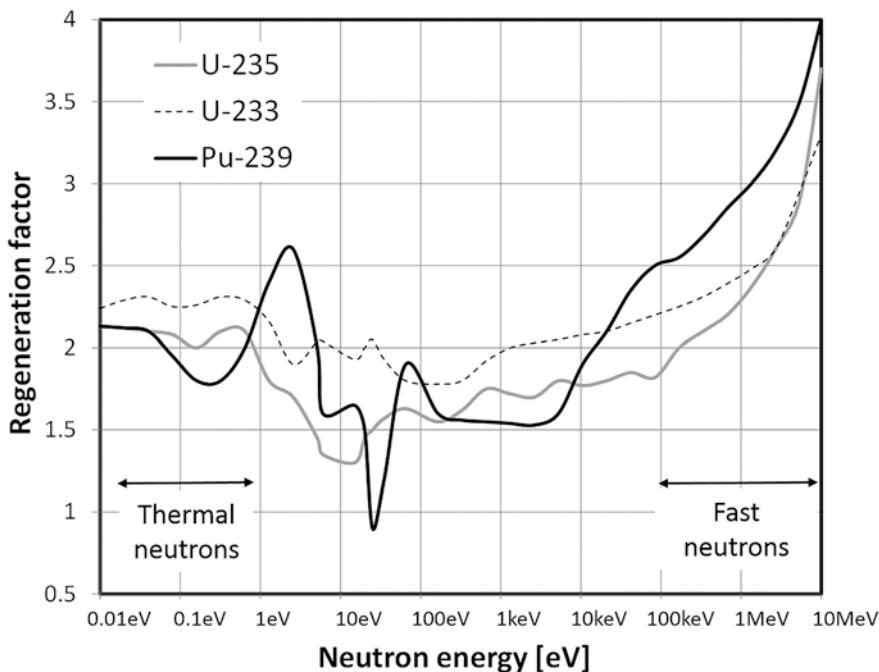


Fig. 5.6 Regeneration factor of nuclear fission of some typical fissile nuclides

neutrons to thermal energy, the thermal neutron spectrum has a maximum between 0.01 eV and 0.2 eV. We see in Fig. 5.6 that we achieve a regeneration factor >2.5 only with Pu-239 as the fissile material, and only with a fast neutron spectrum. Above 2 MeV, U-233 and U-235 would also have a high regeneration factor, but the fast neutron spectrum decreases there already noticeably. On the other hand, since exactly this Pu-239 is formed by absorption in U-238, this results in a sustainable concept. For this, we need an initial inventory of Pu-239, which is constantly replenished. In the end, we only consume the abundant U-238, which is mainly available as depleted uranium from uranium enrichment, but also as uranium from spent fuel, together about a hundred times more than the amount of uranium that has been fissioned so far. That's enough for millennia, assuming naively that we will consume a similar amount of electricity annually in that time as we have done in the last 50 years. If neutron losses are carefully minimized, even more Pu-239 can be formed than is fissioned. This is called a fast breeder reactor.

The other side of sustainability concerns waste. The problem with the thermal reactor was that it can only fission a few fissile transuranics. This looks much better with a fast neutron spectrum. Figure 5.7 shows a comparison of

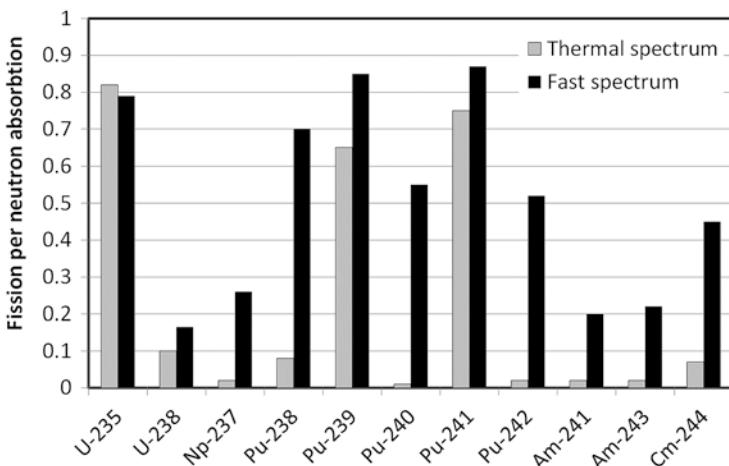


Fig. 5.7 Fission probability of some transuranium elements in a fast and in a thermal reactor

the fission probability for each absorbed thermal or fast neutron. The fissile materials Pu-239 and Pu-241 have a high fission probability in the fast spectrum again. For the other transuramics, only every second or even every fifth fast neutron causes fission, but you still get these nuclides reliably destroyed with them. With a thermal spectrum, on the other hand, their fissioning would be negligible. A fast reactor is therefore a veritable waste incinerator for transuranium elements. You can recycle the waste as often needed.

In principle, it is also possible to produce americium fuel rods. Some fourth-generation reactor concepts envisage this already. However, Am-241 is a powerful γ -ray emitter. Therefore, such a fuel rod can only be produced in a hot cell. A hot cell is a shielded room into which we can only look through a thick lead glass window. To the right and left of it are so-called remote manipulators. These are arms with pulleys and ropes and a pincer handle with which a pair of tongs inside the hot cell can be operated. That is not so easy. When I was allowed to try it once, I felt like a baby trying to build a tower out of building blocks. With a little practice, you can already perform a little more complicated hand movements, but compared to a glove box, the effort is already considerable.

With curium fuel, the effort would be even greater, because curium can spontaneously fission and thus also produce neutron radiation. While the reactor physicists had already started to design reactor concepts with curium, the colleagues from the hot cells told me that they cannot allow such a fuel in their hot cells because of the risk of a fission chain reaction. Since then, it seems easier to me to wait until Cm-244 has decayed.

Proliferation Resistance

Actually, we would not need such a variety of transuranics for this sustainable concept. It would be sufficient if we were to remove the fertile fuel assemblies with U-238 after a low burn-up of about 3 MWd per kg of uranium and send them for reprocessing. Then we would have almost pure fissile Pu-239, as Fig. 5.2 shows. This would save a lot of waste. If all humans were peaceful and decent, this would work. Unfortunately, however, not all people are peaceful and decent. They have often used Pu-239 in the past to build nuclear weapons. It is therefore the stated goal of the Generation IV International Forum not to develop reactors that directly produce weapons-grade plutonium.

Technically, it is Pu-238 that makes it difficult or even impossible to use plutonium for weapons production. This isotope generates heat, which makes the plutonium much more difficult to handle, transport and store. We speak then of physical self-protection. This is comparable to alcohol that is denatured with other chemicals to make spirit so that it is not drunk. It is not completely impossible to get pure alcohol out of it, but it is so inconvenient that nobody makes this effort. In the case of the plutonium isotopes, an enrichment process like with U-235 is hardly possible, because the fissile isotopes differ only by a single mass number from the fertile ones, and nobody would contaminate his enrichment plant with the radiating plutonium isotopes. He could neither maintain nor repair it afterwards.

In contrast to the fast breeder reactors of earlier years, the fast reactor concepts of the fourth generation should therefore have no more fertile assemblies if possible, i.e. no fuel assemblies made of pure U-238. Then no operator or, better said, no government would get the stupid idea of producing nuclear weapons from the plutonium produced. The U-238 is then only present in the fuel as a mixture with all five plutonium isotopes and the newly produced Pu-239 has physical self-protection from the outset.

Thorium Fuel

Figure 5.6 also indicates another sustainable concept that is even possible with thermal reactors. The fissile material U-233 does not exist in nature, but we can produce it by irradiating the fertile thorium Th-232 with neutrons. Thorium is a fairly common chemical element. There is about three to four times more of it than uranium. The only thorium isotope we find in nature is $^{232}_{90}Th$. Unfortunately, it cannot be fissioned with thermal neutrons. Instead,

it absorbs neutrons in a nuclear reactor. Then it becomes $^{233}_{90}Th$, which decays to protactinium $^{91}_{90}Pa$ with a short half-life of 22.3 min. By further β^- -decay, with a half-life of 27 days, it becomes $^{233}_{92}U$. We see this process in the section of the nuclide chart shown in Fig. 5.8. The generated U-233 is almost stable. It has a half-life of 159,200 years and can be fissioned with thermal neutrons.

If we look at the regeneration factor of U-233 in Fig. 5.6, we read a value of 2.2–2.3 for thermal neutrons. Now this is not that much more than 2, but if we are very sparing with the neutrons of the reactor, just as much U-233 could be replicated as is fissioned. An ideal reactor for this concept would be the high-temperature reactor of Chap. 4. The moderator graphite has low neutron absorption. Further structural material which could absorb neutrons, e.g. steel, does not exist in this reactor. Thanks to the low power density, the reactor core is very large. It therefore has a small surface area compared to its volume and consequently a small neutron leakage. The surrounding graphite reflector returns much of the lost neutrons without absorbing them. That is why the Fort St. Vrain reactor in USA and the THTR-300 reactor had a thorium fuel. But the initial thorium could not be mixed with U-233, because

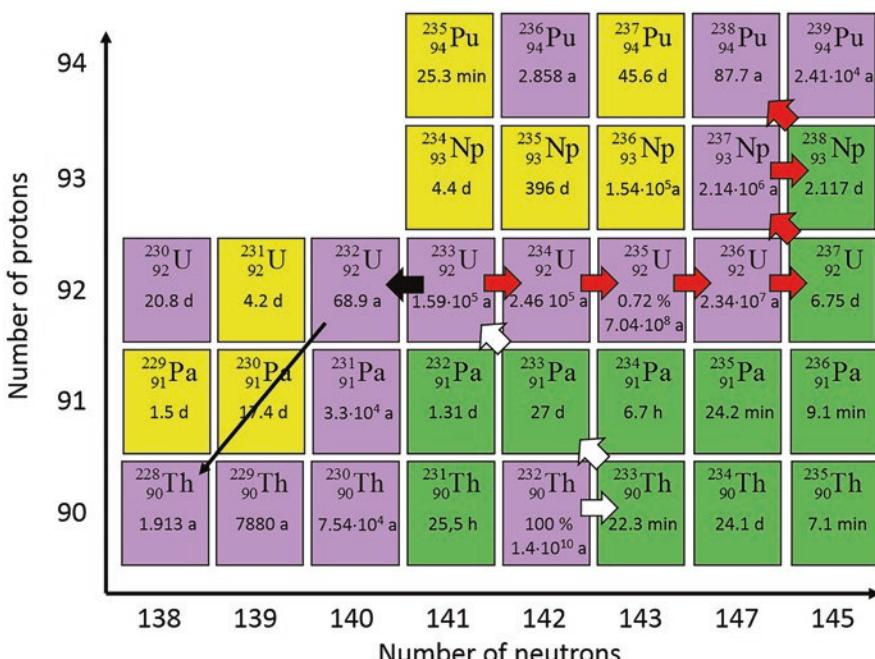


Fig. 5.8 Uranium and thorium in the nuclide chart. White arrows: Generation of fissile U-233 from natural thorium; red arrows: Build-up of heavier nuclides by neutron capture and decay; black arrows: Generation of U-233 and its decay

that was not available for purchase. So they used highly enriched U-235 instead, which of course was not proliferation resistant, because highly enriched U-235 is weapons grade. It was not economical either, because it would have been cheaper to fission it as low-enriched uranium in a pressurized water reactor. So it was just a physical experiment.

Thorium fuel studies and experiments were conducted in India for many years. India has large thorium deposits; however, large uranium deposits were not known there at the time. Thorium fuel also appears repeatedly in other fourth-generation reactor designs. I would like to briefly summarize the advantages and problems associated with it here so that you can better assess the new thorium concepts.

In addition to the advantage that thorium is abundant, the fuel produced from it has another advantage: almost no transuranium isotopes are formed when it is irradiated. When U-233 absorbs a neutron, it becomes U-234, which becomes U-235 by another neutron absorption, and eventually becomes U-236 then. These are all α -emitters with a very long half-life, i.e. low activity. They do not produce a nuclide with a higher atomic number, but decay to nuclides with a smaller atomic number, which are quite radioactive but decay rapidly. Only U-237 is a β^- -emitter, which decays to Np-237, which after further neutron absorption then becomes Pu-238. The fission products of U-233 are quite similar to those of U-235, but there will be no problem with americium and curium in the repository with thorium fuel.

However, there is a problem with U-232. This uranium isotope is created from U-233 when that captures a high-energy neutron. When the neutron energy exceeds about 5.5 MeV, the neutron knocks two neutrons right out of the nucleus. Few prompt neutrons have that much energy, and the concentration of U-232 is consequently very low. U-232 has a half-life of 68.9 years; unfortunately, that's short enough to cause a high activity and at the same time too long to get rid of it quickly. It decays along a long but fairly rapid decay chain to the stable lead isotope $^{208}_{82} Pb$. In the Isotope Browser, we see that $^{232}_{92} U$ initially decays to $^{228}_{90} Th$, which continues to decay to $^{224}_{88} Ra$ with a half-life of 1.9 years. Then everything happens very quickly; the remaining nuclides along the decay chain have half-lives of only seconds to minutes. The last nuclide before stable lead is reached is the thallium isotope $^{208}_{81} Tl$. When it decays, it emits γ -rays with an energy of 2.6 MeV. It is thus one of the hardest γ -emitters in the nuclide chart. Consequently, the generated U-233 inevitably emits γ -rays at quite a high energy.

This has the following consequences: On one hand, a glove box is no longer sufficient to fabricate a fuel assembly with U-233, as it was for MOX fuel with plutonium. The hard γ -radiation requires that the fuel be assembled in hot

cells, behind concrete walls, thick lead glass, and with remote handling. Unfortunately, this makes it much more expensive than a MOX fuel assembly. But on the other hand, this γ -radiation means physical self-protection against proliferation. No one would make nuclear weapons with such a nasty fissile material.

Once this problem was recognized in India, interest in thorium fuel quickly waned. In the meantime, larger uranium deposits are also known there, so that India is no longer dependent on thorium. Nevertheless, thorium fuel could be an interesting alternative for future reactor concepts if the fuel were particularly easy to be produced. We will come back to this in the chapter about molten salt reactors.

Conclusion

A sustainable nuclear fuel cycle can only be achieved with fast reactors, i.e. reactors with a fast neutron spectrum. Only these reactors can produce more fuel than they consume, and only these reactors can almost completely fission the plutonium that is produced. The separation of plutonium from spent fuel and its reuse in new fuel assemblies has already been tested on an industrial scale for 50 years.

Complete recycling of plutonium reduces the radiotoxicity of waste in the repository to about one-tenth. It could be reduced considerably further if americium and curium were also recycled. Their separation and recycling is possible, but has so far only been tested on a laboratory scale. There is still a lot to be done before these processes can be implemented on an industrial scale.

A sustainable nuclear fuel cycle with thermal reactors would in principle be possible if thorium instead of uranium were used as the initial fuel. Then U-233 is produced instead of plutonium and no americium and curium is built up. However, this process failed in the past due to the high γ -radiation of the decay products of the uranium produced.

Glossary

Activity	Decays of a nuclide per second (also: radioactivity)
Burn-up	Thermal energy gained per kg of uranium (or heavy metal in general) by fission
Decay constant	The constant in the exponential law of radioactive decay

Dose coefficient	Conversion factor from activity to radiotoxicity of a nuclide
Fertile material	A material which can absorb neutrons forming fissile material, but which cannot be fissioned with thermal neutrons
Fissile material	A material which can be fissioned with thermal neutrons
Fissionable material	A material which can be fissioned either with thermal or with fast neutrons
Glove box	A workspace that is hermetically sealed gas-tight
Hot cells	A heavily shielded room with remote handling and lead glass windows for handling highly radioactive materials
Lanthanides	Chemical elements with an atomic number of 57–71
Minors actinides	Neptunium and chemical elements with an atomic number >94 (plutonium)
MOX fuel	A mixture of UO_2 fuel and PuO_2 fuel
Radiotoxicity	A measure of how harmful a nuclide is to health, e.g. when ingested through food or drinking water
Regeneration factor	Number of neutrons emitted per absorbed neutron during nuclear fission
Transuranics	Chemical elements with an atomic number >92 (uranium)



6

Sodium-Cooled Fast Reactors

After this excursion into radiochemistry, we can look in the following chapters at how to build the fast reactors needed for a truly sustainable concept.

History

Fast reactors cooled with sodium have been around for a long time, just like water-cooled reactors. As early as 1956, the construction of a small experimental reactor, the BR-10, with a thermal output of 8 MW began in the then USSR. At the same time, the larger Fermi reactor was built in the USA, which went into operation for the first time in 1963, but did not produce the full output of 61 MW of electricity until 1970, because it overheated locally in the meantime due to a coolant blockage. Shortly thereafter followed the EBR-II in the USA, which already produced the full capacity of 20 MW electricity in 1965, and in the USSR the BOR-60 reactor with 12 MW electrical capacity. The latter one is still in operation today, but is due to be shut down shortly. So initially, there was a real competition between the USSR and the USA to develop this type of reactor. The first small reactors were mainly needed to produce fast neutrons, which could be used then to measure cross sections at high neutron energies and to test materials. France also had such an experimental reactor in Cadarache, called RAPSODIE, which went into operation in 1967. It had a thermal output of 40 MW, but did not yet generate any electricity.

In the 1950s and 1960s, development programmes were probably more or less integrated into military research. It is therefore not surprising that the first fast reactors were only built in countries that possessed nuclear weapons. It was not until the Nuclear Non-Proliferation Treaty, first signed in 1968, that a strict separation of the civilian use of nuclear energy from the military use was enforced, thus enabling international cooperation in science and research in the development of the sodium-cooled fast reactor. Germany and Japan signed the Nuclear Non-Proliferation Treaty in 1969 and 1970, respectively, and were henceforth able to participate in the development as well.

Experimental reactors in Germany and Japan therefore followed somewhat later. The KNK reactor was built in Karlsruhe, Germany, and was initially designed and operated as a water-cooled thermal reactor KNK I in the mid-1960s. From 1974, the conversion to sodium cooling followed as KNK II. In 1977, the KNK II reactor went “critical” for the first time with a fast neutron spectrum, i.e. it had a self-sustaining chain reaction with fast neutrons at zero power. In 1979, it produced full power of 20 MW of electricity for the first time. The Joyo reactor in Japan was first critical in 1979. This was also a purely experimental reactor that did not generate electricity. The reactor is still in operation today.

From then on, it was not long before the first large power reactors went into operation. Particular mention should be made of the Phénix reactor in France, which already produced 255 MW of electrical power in 1974, the BN-600 in Russia with 600 MW of electrical power and still in operation today, or the PFR in Scotland, which achieved an electrical output of 250 MW in 1977. Further development was then perhaps a little too fast. The Super-Phénix in France, with an electrical output of 1242 MW the largest sodium-cooled reactor ever built, and the MONJU reactor in Japan with an electrical output of 280 MW had major technical problems which delayed operation. In fact, the MONJU reactor has only generated electricity for about an hour since it was commissioned, so it has virtually never run properly. In Germany, it was rather political disagreements in the assessment of the safety concept and plutonium production: the SNR 300 in Kalkar was built, but then never went into operation.

India has not signed the Nuclear Non-Proliferation Treaty to this day. The country was thus excluded from international cooperation, but nevertheless managed to develop and build a small sodium-cooled reactor: the FBTR with 40 MW thermal power and 13 MW electrical power was first critical in 1985 and reached full power in 1996. A larger reactor, the PFBR with 500 MW of electrical power has been under construction since 2003, but has not yet been operated.

Table 6.1 History of sodium-cooled fast reactors worldwide

Reactor	Country	Electrical power (MW)	Thermal power (MW)	Start of construction	First criticality	First full load	Shutdown
Fermi	USA	61	200	1956	1963	1970	1975
BR-10	Russia	0	8	1956	1958	1959	2003
DFR	UK	11	60		1959	1962	1977
EWC II	USA	20	62.5	1958	1965	1965	1998
Rapsodie	France	0	40	1962	1967	1967	1983
BOR-60	Russia	12	60	1965	1969	1970	2020?
BN-350	Kazakhstan	130	750	1964	1972	1973	1999
Phénix	France	255	563	1968	1973	1974	2009
PFR	UK	250	650	1966	1974	1977	1994
KNK II	Germany	20	58	1974	1977	1979	1991
Joyo	Japan	0	140	1970	1979	2003	In operation
FFT	USA	0	400	1970	1980	1980	1996
BN-600	Russia	600	1470	1967	1980	1981	In operation
FBTR	India	13	40	1972	1985	1996	In operation
Super Phénix	France	1242	2990	1976	1985	1986	1998
MONJU	Japan	280	714	1985	1994	1995	2017
CEFR	China	20	65	2000	2010	2014	In operation
BN-800	Russia	880	2100	2006	2014	2016	In operation
PFBR	India	500	1250	2003	2022?		

The list of sodium-cooled fast reactors in Table 6.1 is thus already impressively long. Further technical information on these reactors and some sketches of their construction can be found in a technical report of the IAEA [28]. The report was prepared in international collaboration and is quite extensive, but fortunately freely accessible.

No other type of reactor that we classify now as fourth generation has been built so many times. Some of these reactors were in operation for more than 30 years. However, they were always experimental reactors or prototypes. They were never commercialized subsequently. Let's look at this technology in a little more depth to understand this.

Sodium

For a fast neutron spectrum, you need a coolant with heavy atoms that do not slow down the generated prompt neutrons too much. Sodium, with its mass number of 23, is not really heavy yet, but it is a fantastic coolant: It has a very high thermal conductivity, five times as high as stainless steel and more than ten times as high as water, so it can still cool fuel assemblies with high power.

Furthermore, it melts at approx. 98 °C, so that it is already liquid at moderate temperatures, but it does not boil below 883 °C under ambient pressure. The reactors mentioned were operated without overpressure and had core exit temperatures of about 500–550 °C, i.e. still far below the boiling temperature, but already so hot that the stainless steel cladding tubes of the fuel assemblies reached their strength limit. Sodium has a density like water, a higher viscosity, but only about a quarter of the specific heat capacity of water. A high density and a high specific heat capacity improve heat transfer and reduce the temperature increase during heating. A high viscosity, on the other hand, worsens heat transfer. In the case of sodium, however, the positive effect of higher heat conduction predominates. A particular advantage of sodium is that it has excellent chemical compatibility with stainless steel. Even after 30 years of reactor operation, the EBR-II looked as good as new from the inside. The sodium did not cause any corrosion on the pipes and internal structures.

But sodium reacts all the more violently with water. If you put a piece of cold sodium on a water bath, it hisses around on the surface: sodium hydroxide and hydrogen are formed, which produces the hissing. Contact of sodium with water must therefore be carefully avoided in the power plant. Even more problematic is that hot sodium burns when exposed to air. So if a pipe with liquid sodium has even a small leak, sodium drips out and immediately starts to burn in the air. You can see then a burning drop falling down. Now, of course, you can't extinguish it with water. An easy way to extinguish the fire is to put a metal plate underneath the pipes. When the burning drop hits the metal plate, its heat is dissipated and the flame goes out immediately. So you extinguish with metal, so to speak. For this reason, experimental facilities with liquid sodium are always placed on a stainless steel tray. Furthermore, a sump tank is needed in the basement, which is connected to the plant via thick pipes. In the event of a leak, the sodium is quickly drained into the tank and the plant is filled with inert gas instead. The fire then goes out after a few minutes.

There is another important difference to water: water is transparent, sodium is opaque. In a pressurized water reactor, you can look down through the water pool into the reactor when you change the fuel assemblies. It's easy to see if anything is damaged, and it's relatively easy to repair any damage. In a sodium-cooled reactor, on the other hand, you can't see anything. This can be quite a hindrance in the event of damage. An overview of the properties of sodium and water is given in Table 6.2.

Table 6.2 Comparison of the physical properties of sodium and water

	Sodium	Water	
Melting temperature at 1 bar	97.8 °C	0 °C	
Boiling temperature at 1 bar	883 °C	100 °C	
Absorption cross section of thermal neutrons	430 mbarn	332 mbarn (H)	
<i>Important material data for core entry and exit conditions:</i>			
Temperature (°C)	400	550	290
Pressure (bar)	1	1	155
Density (kg/m ³)	856	820	746
Viscosity (μPa s)	277	220	92
Thermal conductivity (W/m K)	70	62	0.58
Specific heat capacity (kJ/kg K)	1.28	1.26	5.24

Nuclear Fission with Fast Neutrons

In addition to the different coolant, a fast reactor also needs a completely different core design. Let us first look at the fission cross section of Pu-239, Fig. 6.1, which replaces U-235 as the fissile material. At first glance, the cross sections look quite similar to those of U-235 in Fig. 2.3, but now we cannot use the highly effective cross section of thermal neutrons of 755 barn. Instead, the cross section on the right-hand side of the spectrum, above about 10 keV, is only 1.5–2 barn; it is thus about 500 times smaller. First, this means that more fissile material is needed to obtain a critical arrangement. The typical plutonium content of large reactors is 15–20% plutonium in the UO₂ fuel. In small reactors, as much as 30%. This is significantly more than the maximum 5% enrichment with U-235 in the pressurized water reactor. However, we do not have to provide a higher fissile material content at the beginning of a burn-up, which would have to be compensated for initially with suitable neutron absorbers. The plutonium is permanently reproduced during the burn-up and the plutonium content remains almost constant.

In Fig. 6.1, I have also drawn a typical neutron spectrum of a sodium-cooled reactor with a grey curve. The neutron flux can be read on the right axis. Above 1 MeV, only the prompt neutrons from the fission reaction are responsible for the spectrum. The smaller neutron energies, on the other hand, were produced by collisions with uranium, plutonium, sodium or with the cladding material. By the way, the minimum at about 3 keV comes from the collisions of neutrons with sodium. Sodium not only behaves like a hard billiard ball, but it can also excite a nuclear resonance during the collisions. This is called resonance scattering, in which the neutron transfers significantly more energy.

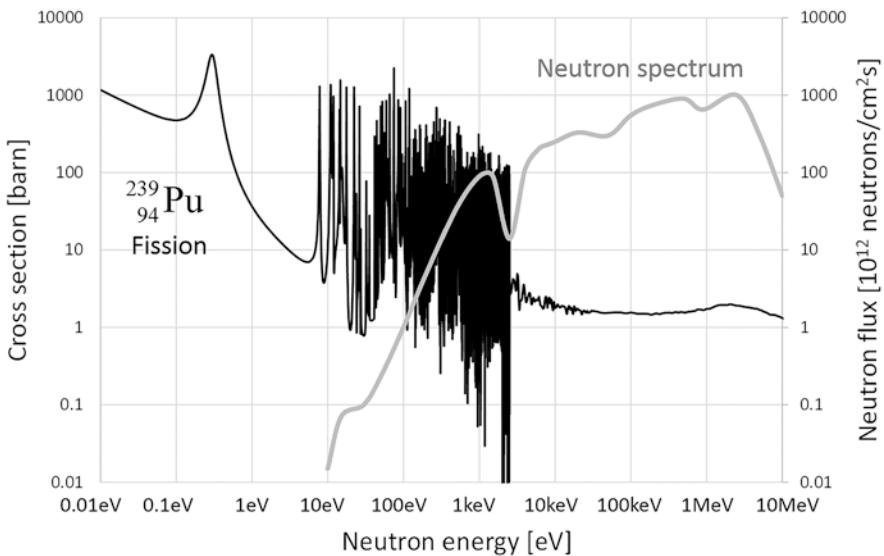


Fig. 6.1 Fission cross section of Pu-239 and example of a fast neutron spectrum of a sodium-cooled reactor

Below 3 keV, the cross section of nuclear fission of Pu-239 has many resonance peaks. The resonance absorptions of U-238, Fig. 2.2, extend even up to 21 keV. A great many neutrons are absorbed here, predominantly in U-238, and the neutron spectrum therefore decreases rapidly towards lower energies. Practically no neutron escapes from this resonance range and consequently we do not get any thermal neutrons with < 10 eV. This was intended to be the case. In the resonance absorption region, the neutron spectrum still has about 10% of the neutron flux of the fast neutrons at the maximum. This is very important, because these resonance peaks absorb more neutrons with increasing temperature of the absorber. This is the Doppler effect, which was already helpful in the thermal reactor to control the neutron flux. So when the fuel gets hotter, it absorbs more neutrons due to this effect and thus controls the power of the reactor back again by itself. Compared to a thermal reactor, however, only a smaller part of this resonance range is used. Consequently, the fast reactor has a weaker Doppler effect than the thermal reactor.

If one integrates over the fast neutron spectrum, one gets a typical neutron flux of about $3 \cdot 10^{15}$ neutrons/cm²s. This is about a hundred times more than the neutron flux of a pressurized water reactor. Such a high neutron flux is necessary to get a similar power density of nuclear fission as in a thermal reactor, because the probability of nuclear fission of the fast neutrons is 500 times

smaller. This is quite a challenge for the structural materials, i.e. for the cladding tubes and the reactor vessel. The neutrons strike flaws inside the metal and cause the steel to become brittle. As a result, the cladding tubes have only a limited service life. In principle, one could achieve unlimited burnup by constantly producing fissile plutonium. However, the service life of the cladding tube would then still require a fuel assembly change after a burn-up of about 100 MWd per kg of heavy metal (uranium or plutonium).

The high neutron flux also limits the lifetime of the reactor vessel. It must therefore be shielded from the reactor core. On the other hand, the reactor vessel does not have to withstand high internal pressure, as is the case with the pressurised water reactor. Since sodium does not boil below 883 °C, the reactor can easily be operated at ambient pressure. Then, the reactor vessel has only to withstand the hydrostatic pressure of the sodium, i.e. a few bar at the vessel bottom.

Let us compare a section of a typical fuel assembly of a sodium-cooled fast reactor with that of a pressurized water reactor, Fig. 6.2. In the pressurized water reactor, the fuel rods are usually arranged in a square array. Only the Russian VVER reactor makes an exception here. The distance between the fuel rods is chosen in such a way that enough water fits between them for moderation. If the fuel rods are too close together, moderator is missing and the reactor has too few thermal neutrons. If the fuel rods are too far apart, the water absorbs too many neutrons. In both cases, the reactor produces less power or even shuts down. So the pitch of the fuel rods is close to the optimal moderator-to-fuel ratio. More precisely, it is always designed to be a little under-moderated, a little too close. Then some moderator is missing and the neutron flux decreases even further if the water starts boiling. The steam displaces the water then and even more moderator is missing; we move further away from the optimum. This gives us then the desired negative void effect.

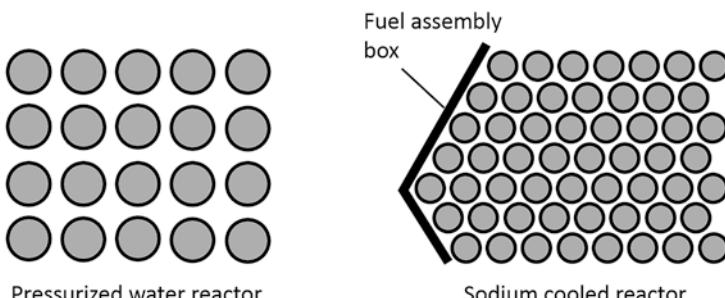


Fig. 6.2 Comparison of the fuel rod arrangement in a thermal and a fast reactor

In the fast reactor, we don't need a moderator, and so the fuel rods can stand much closer together. In this way, they absorb fewer neutrons and more neutrons remain to produce new plutonium. The fuel rods are usually arranged in a hexagonal array for this. This creates a denser packing. The fuel in most fast reactor designs is also MOX fuel, which is a mixture of UO_2 and PuO_2 . The maximum power we can produce with a single fuel rod is then the same for both reactor types, about 40 kW per m of rod length. At a higher power, the UO_2 would melt in the centreline of the pellet. The smaller the diameter of the cladding tube, the smaller its surface area over which this heat is transferred to the coolant, and thus the higher the heat flux in Watts per m^2 surface area. In the case of water, there is an upper limit for the heat flux at which the water is boiling so violently that the cladding tube burns out. Sodium, thanks to its good thermal conductivity, has no such upper limit, and so we can make the diameter of the fuel rods of a sodium-cooled reactor much smaller. As a result, the sodium-cooled reactor has a power density that is more than twice that of the pressurized water reactor.

In normal operation, the sodium does not boil. It enters the core from below at approx. 400 °C and exits at the top at approx. 550 °C on average. So we still have a high safety margin compared to the boiling temperature. However, if cooling channels become blocked, it could still boil. Then the sodium vapour displaces the liquid sodium, fewer neutrons are absorbed, and the neutron flux increases. That's a positive void effect, and that's bad. If the reactor got hot enough somewhere to boil the sodium, it would get then even hotter because of this effect. Again, the Doppler effect will control the power back, but the fuel rod may have already burned out by then.

There is another effect for the physical control of the chain reaction. As already discussed in Chap. 2, the chain reaction is caused by the prompt neutrons, which come directly from nuclear fission, and by the delayed neutrons, which come some seconds later from the decaying fission products. If the reactor power is to increase, the reactor is always run supercritically only to such an extent that only the sum of the prompt and delayed neutrons causes the chain reaction to increase. This is then called delayed supercritical. Otherwise the increase of the neutron flux and thus of the power would be too fast. This applies in principle to both fast and thermal reactors.

Fast reactors, however, emit more neutrons with each fission, as Fig. 5.6 shows. That was the whole point of the fast spectrum. The fission products of plutonium, on the other hand, are quite similar to those of uranium. They produce as many, or rather as few, delayed neutrons after each fission. So the ratio of delayed neutrons to prompt neutrons is less favourable in a fast reactor. If we pull the control rods out a bit to ramp up the reactor power, we have

less margin there for delayed supercritical operation. In normal operation, that certainly doesn't make a problem. You can adjust the control rods accordingly. However, if the power goes up on its own, for example because the sodium is boiling, we have less safety margin this way until the reactor gets prompt supercritical.

One can see from this discussion that a fast reactor is in many ways more sophisticated than a pressurized or boiling water reactor. The fast reactor has a much higher neutron flux, a weaker Doppler effect, a positive void effect, and a smaller fraction of delayed neutrons. In addition, sodium can burn when it leaks. It is obvious why a power plant operator has little interest in sodium-cooled reactors, especially since enriched uranium is currently cheaper than recycled MOX fuel and a sodium-cooled reactor is more expensive than a pressurized water reactor. On the other hand, the history of sodium-cooled reactors shows that these problems were manageable in the past. There has never been any serious accident like in water-cooled reactors. So this is not utopian. Nevertheless, the problems described above gave reason to improve the concept even further in the fourth generation.

Reactor Concepts

The first sodium-cooled reactors were based in principle on the design of a pressurized water reactor: reactor, heat exchanger and coolant pumps were assembled with pipes to form a primary system. Today, this is called a loop design. The principle is explained in Fig. 6.3. A pressurizer, as in the pressurized water reactor, is not needed. The sodium is in the reactor at ambient pressure. The reactor is not completely filled, but has a protective argon gas plenum at the top that could accommodate the thermal expansion of the sodium without increasing the pressure. Unlike the pressurized water reactor, however, the heat exchanger is not a steam generator. In all reactor designs using sodium, secondary systems using sodium are always provided instead, still today. The sodium of the primary system transfers its heat always to at least one secondary sodium system, which in turn supplies heat to a steam generator in the adjacent room. If contact between sodium and steam should ever occur in this steam generator, the sodium can be drained from the affected secondary system into a sump tank and the remaining secondary system or any other emergency cooling system can continue to cool the reactor. The damaged steam generator can therefore be completely disconnected from the reactor in an emergency.

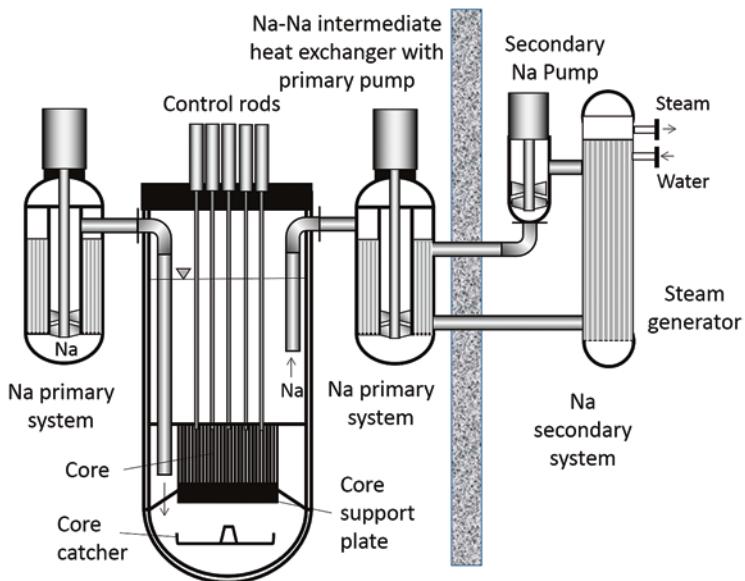


Fig. 6.3 Principle of a sodium-cooled reactor in loop design

The reactors Fermi, DFR, Rapsodie, BOR-60, KNK-II, BN-350, Joyo and even MONJU were such loop reactors. We still find these concepts in the fourth generation of nuclear reactors. The concept of Japanese JSFR-1500, which is the basis of Fig. 6.3, is a reactor with 3570 MW thermal power and 1500 MW gross electrical power [29]. A double-walled reactor vessel with 10.7 m inner diameter heats sodium from 395 °C up to 550 °C. Two intermediate heat exchangers, each integrating the coolant pump of the primary system, heat the sodium of the secondary system to 520 °C, which finally produces superheated steam at 497 °C at 192 bar.

The goal of this design is a power plant with sodium-cooled reactors that could compete economically with pressurized water reactors. The secondary sodium system, however, causes additional costs compared to the pressurized water reactor. Therefore, the intermediate heat exchangers were designed to be very compact to keep these additional costs low. And this is also the reason for the rather high power of the reactor, which is of the order of recent pressurized water reactors. The intermediate heat exchangers and their pumps are still relatively easily accessible in this design. In this way, they can still be easily inspected and repaired if this should become necessary. This reduces costs as well.

The structure of the reactor core is shown schematically in Fig. 6.4. Each of the 255 fuel rods is surrounded by a hexagonal fuel assembly box about 20 cm

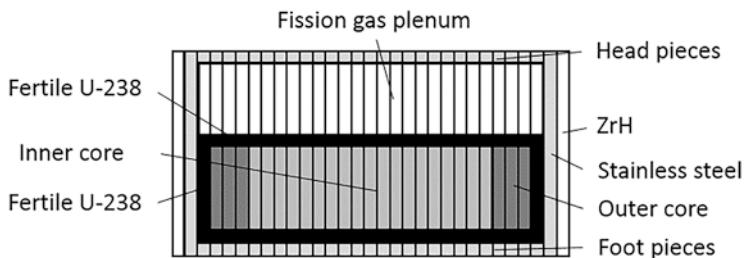


Fig. 6.4 Basic structure of the core of the JSFR-1500

wide. They each have a foot piece, with which they stand in the thick core support plate, and a head piece, by which they can be pulled out. The fuel assembly box allows the coolant mass flow to be adjusted for each fuel assembly by means of an orifice in the foot piece. MOX fuel, i.e. UO_2 and PuO_2 , is only foreseen in the lower part of the fuel rods of the inner and outer core. In the outer core, the plutonium content of 13% is somewhat higher than in the inner core, where it is only 11.5%, so that the temperature distribution of the sodium at the core exit is somewhat more uniform. About every tenth fuel assembly contains control rods made of boron carbide for controlling the reactor power or for shutdown.

The lowest 20 cm of the fuel column in the fuel rod and the top 20 cm consist of depleted uranium, i.e. about 99.9% of U-238 as UO_2 . On one hand they form a neutron reflector and on the other hand they absorb neutrons in their many resonance peaks. Thus they serve as fertile fuel to produce further plutonium and the neutron leakage is well used. The same task is performed by the blanket assemblies of U-238, which are arranged around the outer core, indicated in black in Fig. 6.4. In older reactors, this zone was thicker in order to produce as much plutonium as possible.

A total of 13.6 t of plutonium is required to fuel this reactor. Of this, 7.56 t was assumed to be Pu-239. For comparison, a pressurized water reactor with 1300 MW of electrical power leaves only about 1 t of plutonium in the spent fuel when it is discharged. The active reactor core has a height of only 1.4 m. It achieves an average power density of 390 MW/m³, almost four times that of a pressurised water reactor. The fuel assemblies are shuffled every 800 days and remain in the core for a total of 3200 full load days. Thus they achieve an average burn-up of 150 MWd per kg of heavy metal. They have produced then approx. 10% more fuel than they have consumed. If the fertile fuel on the periphery of the outer core is replaced by stainless steel elements, 3% more fuel is produced than was consumed. This variant without fertile fuel

assemblies would be in line with the fourth generation goals of higher proliferation resistance.

The high burn-up produces a lot of Pu-238, which makes the fuel unusable for nuclear weapons after discharge. But it also produces a lot of gaseous fission products. For this reason, all fuel rods have a large fission gas plenum in the upper half, i.e. an empty space in which these gases can be collected without getting an unacceptable pressure inside the cladding tube. The fertile U-238 assemblies on the rim of the core are still surrounded by a ring of reflector assemblies. They look like fuel assemblies from outside, but their fuel rods are just solid stainless steel rods. Around that reflector is another ring of “fuel assemblies” in which the cladding tubes are filled with zirconium hydride, ZrH. The hydrogen in ZrH is a moderator for the neutrons. It slows the neutrons down to thermal energy so that they have only a range of a few cm. This protects the reactor vessel from high neutron irradiation. However, the thermal neutrons must not be allowed to return to the core under any circumstances. There, with the high fuel inventory, they would generate a very high fission power, because the fission cross-section is much larger with thermal neutrons. The stainless steel elements on the inside are therefore needed to absorb these thermal neutrons, i.e. to protect the core from thermal neutrons.

For refuelling, the spent fuel assemblies are first pulled into a sodium can, in which they are taken then to a cooled sodium container outside the containment. The lid on the reactor remains always closed, except a small cut-out for the fuel assembly, which can be opened. Thus, it cannot be opened completely, as in the case of a pressurised water reactor. After a few years of cooling in the sodium pool, the fuel assembly can completely be freed of sodium. Then it can then be stored and transported more easily. To do this, the sodium is drained off and the fuel assembly is cleaned with an inert gas to which steam is added in doses. In this way, the sodium reacts to form caustic soda, which can be neutralised with acid to form salt water.

If electrical power is not available for the pumps despite the emergency power supply, submersed dip coolers inside the reactor can also transfer the residual heat to an air cooler located higher up without any power supply. The tubes of the dip cooler are also filled with sodium, which circulates solely by gravity. Furthermore, the residual heat can be released, also by natural circulation, to the secondary system, which has also such an emergency cooling system. This emergency cooling can be maintained unlimitedly without power. This makes the reactor even safer than a second generation pressurized water reactor in this respect. If everything should go wrong and the reactor core should melt down, the reactor vessel must be prevented from failing in any case. That is why there is a sodium-cooled core catcher underneath the

core, but still inside the reactor vessel, which prevents the core melt from coming into contact with the bottom of the reactor.

In Japan, the design of the JSFR-1500 has already been worked out in great detail. However, so far there is no concrete plan to build this power plant.

The loop design may be more economical, but if the primary sodium system leaks and sodium drips out, this design has a problem: you can't just drain the primary system. The sodium is constantly needed there to cool the reactor core, even if you shut it down. So early on, there was an alternative design that would safely protect the primary system from sodium leakage. One simply places the entire primary system inside a large, double-walled sodium pool, outlined in Fig. 6.5. This is called a pool design. The EBR-II was already designed in this way, as were the Phénix, the Super-Phénix and the BN-600. This design is now also preferred for current reactors, such as the BN-800 in Russia, the PFBR in India and the CEFR in China.

In the BN-800, a sodium-cooled reactor with a thermal output of 2100 MW and an electrical output of 870 MW, three submersible pumps and three heat exchangers are arranged around the core inside the sodium pool. The pool has an internal diameter of 12.9 m. The hot sodium comes out of the top of the

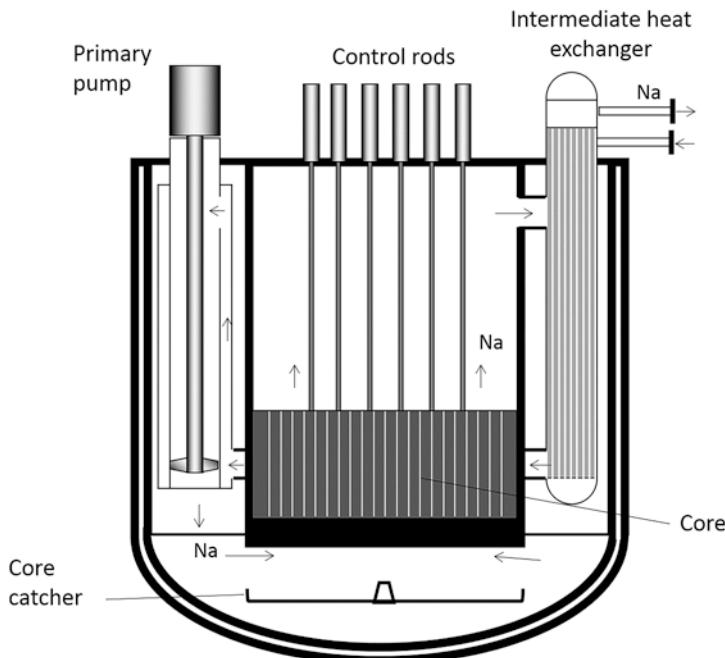


Fig. 6.5 Principle of a sodium-cooled reactor in pool design

reactor core at a temperature of about 550 °C, and from there it first goes to the three heat exchangers, where it flows down the outside of the heat exchanger tubes. It is then directed in the circumferential direction of the pool to the adjacent primary pumps, one of which is outlined in Fig. 6.5 on the opposite side of the reactor. These pumps then convey the sodium back to the reactor core.

The sodium of the secondary system flows on the inside of the heat exchanger tubes. It leads to three external secondary loops with steam generators, a sodium collection tank and a coolant pump. The residual heat can also be removed from the reactor core to the internal intermediate heat exchangers by natural convection, i.e. without the primary pumps running. The secondary sodium system has an emergency cooling system that can remove the residual heat to two cooling towers even without electrical power.

The Super-Phénix in France had even four such secondary loops, each with two intermediate heat exchangers. There, the eight intermediate heat exchangers and four primary pumps were arranged inside a sodium pool with an internal diameter of 21 m. Should there be a sodium leak in any secondary sodium loop, it could be emptied into a drain tank in a matter of minutes. In the case of the Super-Phénix, this amounted to about 400 m³ of sodium. The other three secondary loops took over then the residual heat removal. In October 1984, this actually happened once with the Super-Phénix [30]. The leak could be repaired on the same day and the secondary loop was filled with sodium again the following day.

In a pool design, dip coolers can be inserted as well into the sodium pool from above, which removes the residual heat to an intermediate sodium system, which in turn is connected to an air cooler. Since the air cooler is much higher than the reactor, the sodium rises by itself to the air cooler and then falls by itself back down to the dip coolers. Thus, the residual heat can be removed indefinitely even without any power supply. The Indian PFBR, for example, has such dip coolers.

The JSFR-1500, the BN-600, the Super-Phénix, and the PFBR all have a positive void effect if the sodium starts boiling. One can certainly debate whether this is safety relevant or not. You do have to assume that the sodium is boiling violently in large parts of the core without reactor scram before a severe accident would occur. That could happen only in case of a gross operational failure. In 60 years of reactor history, this has never created a problem. However, since the Chernobyl reactor went out of control due to a gross mistake by the operating crew and eventually exploded due to a positive void effect, this problem cannot be explained away. We would rather have a reactor that can forgive even gross mistakes.

Therefore, some reactor designs have been worked out in the past years which shall avoid a positive void effect. One can more or less compensate the void effect of sodium by another effect: When sodium is boiling, not only does its neutron absorption decrease, but so does its poor moderation of neutrons. The neutrons then become faster and fly further away on average. As a result, the neutron leakage increases. A reactor with a high neutron leakage has therefore in total a decreasing neutron flux with increasing vapour fraction of sodium, i.e. a negative void effect. In the past, this effect was hardly used, because the neutron leakage was minimised in order to optimize the plutonium production. However, some reactor concepts of the fourth generation have some interesting approaches to this problem, as I would like to show in the following.

A higher neutron leakage is achieved in a large reactor core, e.g. by a heterogeneous arrangement of fertile and fissile assemblies. Figure 6.6 shows two examples of such a design. The fertile assemblies with U-238 are then located not only on the outside of the core, but also inside, either as an annular arrangement of fertile assemblies (Fig. 6.6, bottom) or as horizontal layers of U-238 pellets in the fuel rod (Fig. 6.6, top). When sodium is boiling, more neutrons escape into these fertile zones and the neutron flux in the fissile

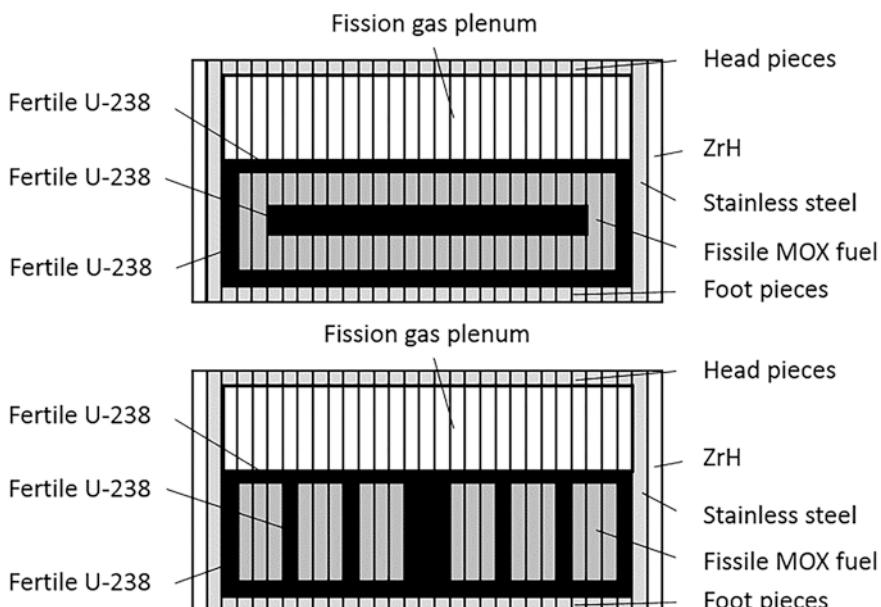


Fig. 6.6 Heterogeneous arrangement of fissile MOX fuel (grey) and fertile U-238 (black) in the core

zones decreases. On the other hand, the constantly higher leakage is well used by producing new fuel. However, this solution has a disadvantage: the fertile zones act like absorber rods in a reactor. To get it nevertheless critical, one needs a higher fraction of fissile material in the MOX fuel, depending on the design rather 20–25% instead of 11.5–13% as in the JSFR-1500.

But this creates a new safety risk: the new pressurized water reactors being built must be designed such that even in the unlikely event of a core meltdown, the neighbourhood does not have to be evacuated. Of course, this applies equally to sodium-cooled reactors. For this purpose, they have a core catcher below the reactor core, which is intended to collect and cool the melt. In the case of an annular, heterogeneous arrangement of fissile and fertile assemblies, however, only the fissile assemblies would melt initially. The fertile assemblies hardly get hot. If the melt accumulates then on the core catcher, the fertile fuel rods are missing there as absorbers. It's like pulling the control rods out of a reactor. It might become critical. In the event of a meltdown, it could even cause an uncontrolled criticality. That could be dangerous. So the concept of heterogeneous core design is not really convincing yet.

A completely different approach to a sodium-cooled reactor with negative void effect is the reactor concept 4S. It was developed by Toshiba Corp. in cooperation with the research institute CRIEPI [31]. The name 4S stands for “Super Safe, Small and Simple”. The goal is a small reactor with 30 MW of thermal power that can deliver just 10 MW of electrical power to the grid. It could be used in very remote areas, such as Alaska or northern Canada. The reactor should be able to operate for 30 years without having to be opened. It is therefore fully enclosed. There are no plans to replace fuel assemblies inside the power plant. Consequently, there is no need for a fuel assembly storage or a loading machine in the power plant. Instead, the reactor is to be delivered to the site together with its fuel and returned to the manufacturer after 30 years. This does not only make this reactor more resistant to proliferation; it also requires significantly fewer personnel on site, which is an important cost advantage for such a small reactor.

Figure 6.7 shows a cross-section through the lower half of the reactor. The reactor core is formed by 18 hexagonal fuel assemblies, whose fuel rods are loaded with fuel over a length of 2.5 m in the lower part and which have a 2.7 m long fission gas plenum in the upper part. Inside the core, there is an absorber rod with which the reactor can be shut down. Otherwise, the power is controlled by the reflector, whose six circular segments can be raised hydraulically or lowered suddenly, like the control rods of a conventional reactor. Above the reflector are six empty barrels per reflector segment, which displace the sodium and thus allow further neutron leakage from the reactor core,

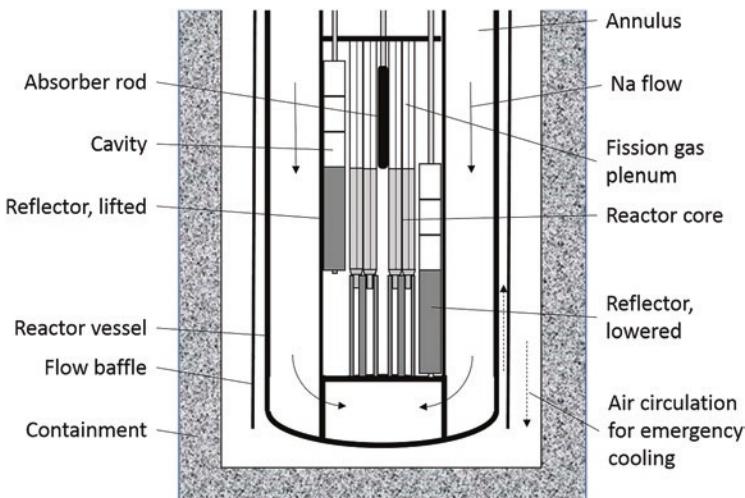


Fig. 6.7 Section through the lower half of reactor 4S

depending on the position of the reflector. The prototype of this reactor is to receive enriched uranium as fuel, so no plutonium yet. The enrichment should be less than 20% to keep the fuel proliferation resistant. This prototype produces then less fuel than it consumes, but the void effect is clearly negative.

An annular shell-and-tube heat exchanger is located in the upper part of the wide annular space. The hot sodium of the primary system flows down through the vertical heat exchanger tubes, cooling down from 510 °C to 355 °C, while the sodium of the secondary system heats from 310 °C to 485 °C on the outside of the tubes. Below that are electromagnetic pumps that drive the sodium from the primary cooling loop down through the annulus back into the reactor core. So again, this is a pool design. Electromagnetic pumps don't have an impeller, they drive the flow purely by alternating electromagnetic fields. This is not as effective as a pump impeller, but it makes these pumps virtually maintenance free. In remote areas, this is more important than high efficiency. The reactor vessel has an internal diameter of 3.5 m and a height of 19 m, including the hydraulic drives on the lid. So for a reactor that generates just 10 MW of electrical power, it's not really small.

In this reactor, the secondary sodium system consists only of a single loop in order to save costs. Should a leak occur there, so that the sodium has to be drained into a tank, the reactor is shut down and the residual heat is released via the outside of the reactor vessel to an air circulation system. The hot air passes up there inside a flow baffle and is released to the atmosphere. For small reactors, this is quite sufficient. The secondary sodium system produces superheated steam at 453 °C for the tertiary steam cycle. This sodium is also driven

by an electromagnetic pump and can be cooled by an air cooler in an emergency, without a power supply.

The development of this reactor was already very advanced in 2007. Many of the components had already been tested by then and there was already a site in Alaska, in Galena on the Yukon River, where this power plant was to be built. Nevertheless, a construction decision was never made until today. The expected safety and reliability of this small power plant were convincing and the inhabitants in Galena were quite open-minded for such an innovative power plant. However, economics is always a problem with such small reactors. It is quite difficult to compete against an oil tank and some diesel generators. Then in 2013, there was a momentous flooding of the Yukon River, where almost the entire town had to be evacuated. No one was talking about an innovative nuclear power plant in Galena anymore since then.

But back to the technology: How did Toshiba manage to design a reactor core that does not need to be recharged for 30 years? The active part of the reactor core of the 4S has a volume of about 1.8 m^3 and produces just 30 MW of heat. This corresponds to an average power density of about 17 MW/m^3 . Take a JSFR-1500, which has an average power density of 390 MW/m^3 and which needs to be refuelled every 2 years. If you run it at only 5% of its rated power, which is a power density of 19.5 MW/m^3 , it will take 40 years before it needs to be reloaded. The fuel will then have received the same burnup and the cladding the same neutron dose as the JSFR-1500 after 2 years. So the long periods for reloading are actually not that difficult to achieve.

Having understood the trick, we could also design easily a sealed pressurized water reactor for Alaska that wouldn't need to be opened for 30 years. We simply lower the power density of the core by a factor of 30. In fact, there have been numerous designs of small, safe pressurized water reactors for remote areas in recent years, but none of them could be sold yet, because it doesn't make that small reactor more economical than a large reactor. Typically, we tend to go the other way instead in reactor development. The more power we can achieve with a reactor, the lower its specific costs, i.e. the euros or dollars per kW installed power. On the other hand, all the other components of the power plant can be built correspondingly smaller, i.e. the pumps, heat exchangers, steam generators, turbines and the generator. So the size of the reactor core is not so important anymore. The best overview is given by a report of the IAEA on the so-called "Small Modular Reactors" [32]. There you will find 26 concepts of water-cooled reactors, which can be installed on land or mobile on boats or pontoons, in the range of 6–300 MW electrical power. Since I do not include pressurized and boiling water reactors, regardless of their size, among the fourth-generation reactors here, I will not discuss them further.

Conclusion

Sodium-cooled reactors have already a 60-year history, during which everything from small prototypes to large power reactors has been built and operated. However, these reactors are certainly more sophisticated than pressurised and boiling water reactors. Let us discuss them here in brief on the basis of the objectives of the fourth generation.

Sustainability

The closed fuel cycle and the sustainable use of uranium are obviously state of the art thanks to this technology. In the past, sodium-cooled reactors have actually produced more fuel than they have consumed. This reactor does not leave plutonium for a repository; it can recycle all plutonium isotopes. The reactor core contains already americium in small quantities, namely from the decay of Pu-241. Utilization of larger quantities is envisaged in many concepts, but has not yet been tested on a large scale. Today, there is not yet that much americium anyway.

Safety

In some safety respect, a sodium-cooled reactor is even better than a second-generation pressurized water reactor: In the event of a complete loss of power, i.e. a failure of off-site and emergency power, a sodium-cooled reactor can remove its residual heat to the ambient air for an unlimited period of time. Water-cooled reactors generally cannot do this. However, from a neutron physics point of view, the sodium-cooled reactor has smaller safety margins: compared to the pressurized water reactor, it has a smaller fraction of delayed neutrons, a weaker Doppler effect and usually even a positive void effect of the coolant. However, these weaknesses will only become important in case of severe accidents. They do not pose a problem in the case of minor accidents.

Reliability

The sodium-cooled reactor has never shown any corrosion problem. On one hand, this makes it very reliable. On the other hand, a sodium leak always leads to a fire, for which the power plant must be prepared in order to be able to quickly drain the affected part of the plant. Furthermore, the

sodium-cooled reactor has a neutron flux 100 times greater than that of a pressurized water reactor, which causes the materials in the vicinity of the reactor core to age correspondingly faster. In the event of damage inside the reactor vessel, inspection and repair under sodium is unfortunately considerably more difficult than under water.

Economic Competitiveness

The sodium coolant requires always an intermediate sodium system in order to be able to separate the steam generator from the reactor in the event of a coolant leak, which increases costs. The handling of the spent fuel assemblies, which subsequently have to be cooled with sodium for some time, is also considerably more complex than under water. Despite their long operating history, power plants with sodium-cooled reactors are therefore still more expensive than water-cooled ones. In addition, recycled plutonium fuel is more expensive than fresh uranium fuel. Recycling is therefore not (yet) economically viable. But we know this from other recycling strategies as well.

Proliferation Resistance

It is possible to build a sodium-cooled reactor in such a way that weapons-grade plutonium is never produced. It has a high burn-up then and does not have any fertile fuel assemblies. This requires compromises with regard to the reusability of the plutonium for electricity generation, but there are fewer concerns about the export of such reactors.

Research still sees some potential for improvement of sodium-cooled reactors: new core concepts with a negative void effect are currently being pursued, but are not yet quite convincing. Further studies are trying to further increase the economic efficiency and to improve inspection and repair under sodium. Furthermore, there are now increasingly better methods of predicting the course of severe accidents with destroyed reactor cores, in order to be able to assess the extent to which the smaller safety margins of neutron physics could become a problem.

Glossary

Breeding	Production of fissile plutonium by neutron absorption in U-238
Fertile zone	Part of the reactor core containing fertile fuel
Fertile assemblies	Fuel assemblies which initially contain only U-238 in order to produce fissile plutonium from it
Loop reactor	A reactor with steam generators and pumps outside the reactor vessel
Pool reactor	A reactor with steam generators and pumps inside the reactor vessel
Resonance scattering	An elastic collision of a neutron with an atomic nucleus in which the latter is excited to a higher energy state



7

Lead-Cooled Fast Reactors

The two most serious problems of the sodium-cooled fast reactor can be solved simply by using a different coolant. Liquid lead is a good choice here. It does not react violently with water or air. It simply solidifies when it escapes through a leak, and at most oxidizes a little on the surface. Furthermore, the boiling temperature of lead is so high that the cladding tubes melt rather than the lead starts boiling. Therefore, a positive void effect is unlikely to do any damage. Furthermore, with a mass number of 206–208, the lead atom is much heavier than a sodium atom. In elastic collisions with lead, the neutron therefore loses much less energy than in collisions with sodium. The neutron spectrum thus becomes a bit faster and can produce more plutonium. Unfortunately, on the other hand, lead causes a few other problems that we need to take a closer look at first.

Lead and Lead-Bismuth

Let us first consider the thermo-physical properties of lead, see Table 7.1. Similar to sodium-cooled reactors, lead-cooled reactors can also be operated at atmospheric pressure. A typical temperature range of a lead-cooled reactor is about 400 °C at the core inlet to 550 °C at the core outlet. Locally, 650 °C can also occur in the core. Liquid lead is known to have a very high density of more than 10,000 kg/m³; as a result, the liquid has a ten times higher hydrostatic pressure deep inside the reactor. The viscosity of lead is low, although higher than that of water. The thermal conductivity is comparable to that of

Table 7.1 Thermo-physical properties of lead and lead-bismuth [33]

	Lead	Lead-bismuth	
Composition	100% Pb	55.5% Pb, 44.5% Bi	
Melting temperature at 1 bar	327 °C	124 °C	
Boiling temperature at 1 bar	1750 °C	1670 °C	
<i>Important material data for core entry and exit conditions:</i>			
Temperature (°C)	400	550	300
Pressure (bar)	1	1	1
Density (kg/m ³)	10,536	10,384	10,337
Viscosity (mPa s)	2.2	1.7	1.8
Thermal conductivity (W/m K)	16.6	18.3	12.1
Specific heat capacity (kJ/kg K)	0.147	0.144	0.146
			0.143

steel, so not as good as sodium, but still much better than water. Unfortunately, the specific heat capacity is about nine times worse than that of sodium. However, at the same flow rate, only the product of density and specific heat capacity enters into the heat balances, and that is still higher for lead than for sodium.

With its high melting temperature, pure lead unfortunately has the disadvantage that the reactor must always be kept at a minimum of 400 °C, otherwise it might freeze. However, one can easily lower the melting point to 124 °C, only slightly higher than that of sodium, by using a eutectic alloy of 55.5% lead and 44.5% bismuth. We see in Table 7.1 that this hardly changes the thermo-physical properties. However, bismuth is more than ten times as expensive as lead and has also some nuclear physical disadvantages, as we will see below.

Lead is found in nature in the form of three stable isotopes, namely $^{206}_{82}\text{Pb}$, $^{207}_{82}\text{Pb}$ and $^{208}_{82}\text{Pb}$. The lighter $^{204}_{82}\text{Pb}$ is also practically stable; it has a half-life of more than 10^{17} years. I have compiled the cross sections for the absorption of thermal neutrons in Table 7.2. Though the absorption cross section of $^{207}_{82}\text{Pb}$ is higher than the one of sodium, lead absorbs less neutrons than sodium because this isotope is found with an abundance of 22.1% only. If these isotopes are absorbing a neutron, we get another stable lead isotope, except for $^{208}_{82}\text{Pb}$ which, after decay of the radioactive $^{209}_{82}\text{Pb}$ with a half-life of 3.23 h, becomes the stable bismuth isotope $^{209}_{83}\text{Bi}$.

If, on the other hand, this $^{209}_{83}\text{Bi}$ absorbs a neutron again, we get $^{210}_{83}\text{Bi}$, which decays to the polonium isotope $^{210}_{84}\text{Po}$ with a half-life of about 5 days. But this polonium is a problem. It is a powerful α -emitter that decays to the stable $^{206}_{82}\text{Pb}$ with a half-life of about 138 days. Not only does it contaminate our lead-bismuth, but it also produces a lot of heat as it decays. Its α -particles have an energy of 5407.5 keV, as the *Isotope Browser* tells us, producing a heat

Table 7.2 Nuclear physical properties of lead and bismuth

Isotope	$^{204}_{82} Pb$	$^{206}_{82} Pb$	$^{207}_{82} Pb$	$^{208}_{82} Pb$	$^{209}_{83} Bi$
Abundance (%)	1.4	24.1	22.1	52.4	100
Half-life (years)	1.4 · 10^{17}	Stable	Stable	Stable	2 · 10^{19}
Cross-section of thermal neutron absorption (mbarn)	703	26.6	622	0.23	32.4

of 144 W/g, about 250 times more than the α -emitter Pu-238. In a large reactor filled with lead-bismuth, a few kg of polonium can build up over the years, producing a similar heat as the residual heat of the fission products. But this does not create a long term problem for a waste repository; it decays too quickly for that. After 4 years, only 0.1% of it is left, and 4 years later another 0.1% of it; so after 8 years there is practically nothing left. If, on the other hand, a reactor is operated with pure lead, the neutron absorption in $^{208}_{82} Pb$ will produce some $^{209}_{83} Bi$ as well, which decays to $^{210}_{84} Po$, but not nearly as much.

Lead-bismuth has another physical property as a coolant that could potentially become a problem: It expands slowly as solid lead-bismuth when it has previously solidified rapidly. Initially, it shrinks after solidification, like other metals. A hexagonal dense sphere packing of both elements is formed. But then, the lead and bismuth atoms begin to arrange themselves into a new crystal lattice, called the γ -phase, which takes up more space. About 100 min after the sudden solidification, the volume of the solid, cold lead-bismuth is already as large as that of the liquid again, and then it starts growing further and possibly bursting the container. We know the problem from ice. A reactor that has been filled with lead-bismuth should not be allowed to freeze, or it may have cracks afterwards. If the lead-bismuth cools down slowly, on the other hand, this problem does not occur: The solid phase has almost the same density as the liquid then. This problem does not occur with pure lead either; it shrinks during solidification. But a lead-cooled reactor should, of course, not be allowed to freeze anyway.

Chemically, both lead and lead-bismuth are harmless when they come into contact with water or air. At high temperatures, however, both liquids attack the steel, and this is the actual problem with lead-cooled reactors. Iron and nickel are dissolved in the liquid metal. The problem increases exponentially as the temperature increases. In lead-bismuth, corrosion problems below 450 °C are usually acceptable. In pure lead, comparable problems do not occur below 500 °C, as can be seen from Fig. 7.1. For a lead-cooled reactor, however, much higher temperatures are required. A solubility of 1 ppm means:

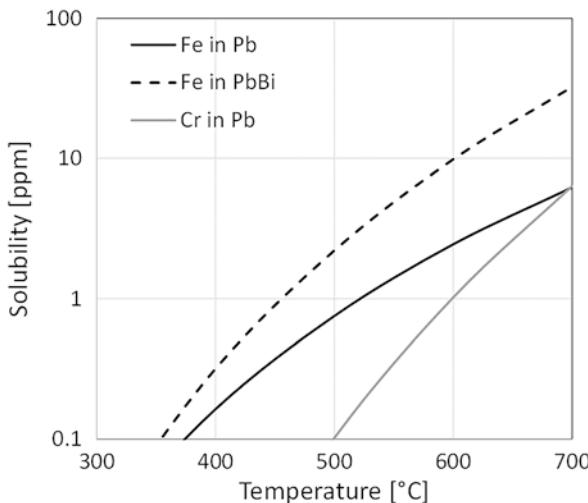


Fig. 7.1 Solubility of iron and chromium in liquid lead or lead-bismuth [34]

in one cubic meter of liquid metal, which is about 10 t, about 10 g of iron will dissolve. Unfortunately, this does not happen evenly distributed over all surfaces, but it starts at the hottest spots. And then even 10 g can be a lot.

Nickel is particularly soluble in lead or lead-bismuth. Its solubility is far above the values shown in Fig. 7.1. Therefore, stainless steel or even nickel-based alloys are not a good solution, although these steels are known to us to be particularly corrosion resistant in water. Instead, ferritic steels can be used. They consist mainly of iron with a few percent chromium.

Corrosion problems can be avoided by applying a protective coating to the cladding tubes of the fuel rods and to other internal structures of a lead-cooled reactor. It consists of iron oxide, Fe_3O_4 , which forms by itself on the surface if some oxygen is dissolved in the liquid metal. But if you dissolve too much oxygen in the liquid metal, lead oxide, PbO , forms and precipitates as crystals in the liquid, clogging the cooling channels. This leaves only a small window for the oxygen concentration and for the temperature, to avoid both problems. Figure 7.2 shows how small this window is for lead-bismuth. In order not to risk either corrosion of the cladding tubes nor clogging of the fuel assemblies by lead oxide, considering a temperature range of 400–550 °C with a small safety margin, the oxygen concentration may only vary between 0.001 ppm and 0.01 ppm. This is an extremely small concentration. Considerably more oxygen would be soluble in lead-bismuth. At 0.01 ppm oxygen, however, lead oxide is already formed below about 250 °C, i.e. well above the melting temperature of lead-bismuth, as we can see in Fig. 7.2.

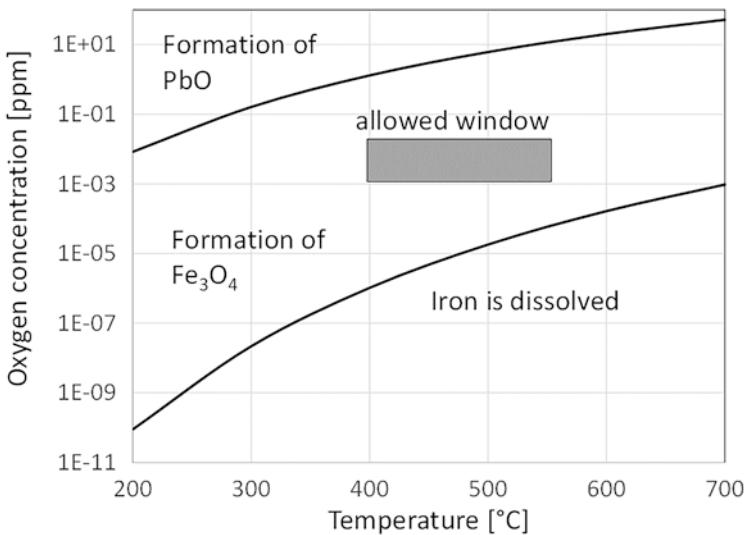


Fig. 7.2 Permissible window for dissolved oxygen concentration in lead-bismuth [33]

Consequently, to maintain such a small concentration window, we need very precise oxygen measurement and control. In principle, oxygen measurement in liquid lead or lead-bismuth is carried out by measuring the electrochemical potential, i.e. the voltage between an electrode immersed in the lead and a second electrode in a reference liquid metal. Usually, this reference liquid metal is filled into a ceramic tube, which thus forms the electrolyte, and which is immersed in the liquid lead. Now the voltage between the inside and outside of this tube can be measured, which yields the oxygen concentration then.

The easiest way to control the oxygen concentration is via the inert gas upon the surface of the liquid metal. As with sodium-cooled reactors, the reactor must not be completely filled with liquid metal, but a protective gas layer must remain on top so that the liquid metal can expand when heated. If some hydrogen is added to this inert gas, usually argon, it pulls oxygen out of the liquid metal. If, on the other hand, a little water vapour is added, the oxygen content in the liquid metal increases. However, the liquid metal must then remain in constant motion and must flush all parts of the reactor uniformly. Otherwise, too much oxygen will remain in niches that are not flushed and lead oxide crystals will form there. 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, so it's better to start with lower temperatures first avoiding hot spots in the reactor core which might cause damage. Again, this is more in favour of lead-bismuth, despite its higher cost and its polonium problem. In contrast to sodium-cooled reactors, however, experience to date is rather limited.

History

Lead-bismuth was actually used to cool submarine reactors in the Soviet Union as early as the 1960s. The reactor was much smaller than a light water reactor and the submarine could thus be built more compact and run faster. Unfortunately, I do not know the details of these reactors. They are still subject to military secrecy to this day. But a field report about these reactors was published by the IAEA [35].

The first prototype, the K-27 submarine, went into service in 1963. The liquid metal cooled reactor, however, was not a success. It failed as early as 1968. The oxygen in the lead-bismuth had either not been properly removed after the last overhaul or lubricating oil from the pumps had entered the lead-bismuth. As a result, an oxide plug got stuck inside the reactor core and the core overheated locally. This caused the Doppler effect to control the power back, and when the seamen tried to increase the reactor power again by pulling the control rods, some of the fuel rods melted. The damage could not be repaired at sea and the reactor eventually failed. During this accident, nine sailors died from an increased dose of radiation. The K-27 was finally sunk at sea in 1982.

Subsequently, the Alfa class (also known as Project 705 or Lira) was built in the 1970s. These submarines had again a lead-bismuth-cooled reactor. They were in service until 1990. Although this allowed the submarines to run faster than other submarines, they were rarely used because the reactors had to be kept permanently warm with superheated steam when they were shut off to prevent them from freezing. An external steam supply, however, was only available in the home port. When even this steam supply broke down in the early 1980s, the reactors had to keep running constantly to stay warm. However, these reactors were not designed for this; inspections and overhauls were virtually impossible. Continuous operation resulted in numerous faults and eventually caused a steam leak via the steam generator into the lead-bismuth, which as a result even released steam inside the submarine that was contaminated with polonium. Finally, the reactor had to be abandoned at sea.

The sailors took refuge in shelters and fortunately survived the accident. Since this failure, submarine reactors worldwide are only cooled with water.

There were also two onshore experimental reactors in Russia cooled with lead-bismuth, one at the IPPE research centre in Obninsk and one at the NITI research centre in Sosnovy Bor. Details of these reactors have never been made public. They are also subject to military secrecy. However, as a result, there are about 80 reactor-years of experience with lead-bismuth-cooled reactors in Russia to date.

So, unfortunately, the history of lead-bismuth-cooled reactors has not been crowned with much success, unlike that of sodium-cooled reactors, but there is certainly some lesson to be learned from these mistakes. Reactors cooled with pure lead have never been built before. Even experimental facilities to develop this technology are rare to date and have only been operated at lower temperatures, if at all. So there is still a lot to develop. For temperatures of 650 °C or more, there is even a lack of suitable materials for a test section. Nevertheless, many concepts have already been worked out for liquid lead or lead-bismuth within the framework of the Generation IV programme, about which I would like to report in the following.

SVBR-75/100

The Russian experimental reactor SVBR 75/100 builds on the long experience with Russian submarine reactors [36]. It is a pool-type reactor, sketched in Fig. 7.3, which is filled with 193 t of lead-bismuth. It was designed in two versions, as a lower temperature prototype to produce a net power of 75 MW, and as a later commercial reactor with 100 MW of electrical power.

The reactor core consists of 55 hexagonal fuel assemblies with a width of 225 mm and a height of 1845 mm. The fuel rods with a diameter of 12 mm and a length of 1638 mm are filled with MOX fuel only in the lower part over a height of 1000 mm. The space above inside the cladding tubes serves as a fission gas plenum. In the centre of each fuel assembly, a tube for the control rod made of boron carbide is provided. The shutdown rods must drop down into a dry tube, otherwise they would float in lead-bismuth. The guide tubes of the shutdown rods are therefore open at the upper end, outside the reactor, and closed at the lower end. An interesting feature of this reactor are melting rods on some of the shutdown rods. When the lead-bismuth becomes hotter than 700 °C, they melt and the shutdown rods fall down by gravity.

The fuel rods have a larger pitch than in the sodium-cooled reactor. With the sodium coolant, the fuel rods have to be as close as possible so that the

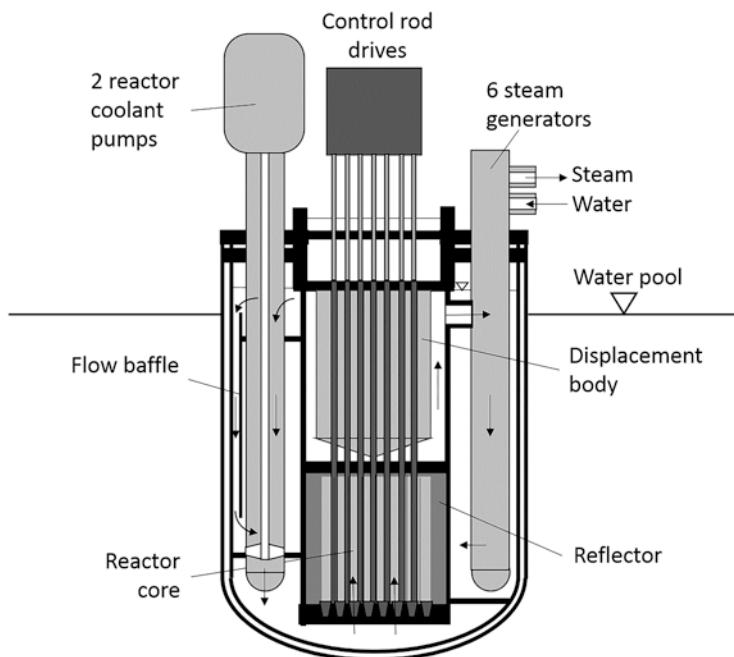


Fig. 7.3 Principle of the Russian SVBR 75/100 with lead-bismuth cooling

sodium does not moderate too much. That's not a problem with this reactor. When a neutron collides with a heavy nucleus of lead or bismuth, it's about like a billiard ball hitting a bowling ball: it bounces back with undiminished speed, so it's hardly "moderated" at all. However, a slightly larger distance between the fuel rods allows a smaller flow velocity of the liquid metal. Experience to date has shown that corrosion problems only arise above 2 m/s. The flow velocity in the reactor core of the SSVBR 75/100 is therefore limited to this velocity.

The reactor core has an average power density of 140 MW/m³ and achieves an average burn-up of 71.5 MWd/kg. The fuel assemblies cannot be discharged individually, but the core must be removed as a whole after 6–8 years, or 2200 full load days, and transported to the manufacturer. It would not be easy to replace individual fuel assemblies anyway: the fuel assemblies float up in lead-bismuth! Instead, the core is pulled as a whole into a transport container, together with the lead-bismuth, which is allowed to solidify in it after some time. Furthermore, there are also no fertile fuel assemblies. This makes this concept more resistant to proliferation. After unloading, the MOX fuel will have produced on average 4% more plutonium than was consumed. Alternatively, the manufacturer offers also to operate the reactor with 16% enriched uranium. In this case, however, it consumes 13% more fuel than it

produces because the U-235 releases fewer neutrons during fission. The longer operating time and the relatively low power density, compared to other fast reactors, follow a similar strategy to the 4S reactor. However, operation for more than 8 years does not seem advisable to me in view of the corrosion problems associated with lead-bismuth cooling.

The reactor core heats the lead-bismuth from 320 °C at the inlet to an average outlet temperature of 482 °C. For the prototype with an electrical output of 75 MW, the coolant temperatures are reduced, heating the lead-bismuth up from 286 °C to 435 °C only. The cladding tubes of the fuel rods must not become hotter than 600 °C. They are made of a ferritic-martensitic steel with 10–12% chromium. Unlike the sodium-cooled reactor, they are not made of stainless steel. Since the hottest point can never be predicted so precisely, it is more advisable to start with these lower temperatures first and only raise the temperature slowly after a few years of operating experience.

In contrast to the sodium-cooled reactor, a lead-cooled reactor does not require a lead intermediate system. The steam generators are therefore located directly inside the reactor pool. In the SVBR 75/100, there are a total of six steam generator modules whose saturated steam is collected and separated in two large, horizontal drums above the reactor. The steam rises there by itself and the steam generators do not need a recirculation pump. These six smaller steam generator modules fit better inside the annular space between the reactor core and the vessel than two large ones. There is no superheater foreseen for the generated steam. In the commercial version with 100 MW, the saturated steam has a pressure of 95 bar and a temperature of 307 °C. This results in a net efficiency of 34.6%, similar to the low efficiency of today's pressurized water reactors. One can see from these figures that this reactor is designed quite conservatively. After the difficult experience with the submarine reactors, this seems very reasonable to me.

The two coolant pumps in the reactor pool together deliver 11,760 kg/s at a pressure of 5.5 bar, after the lead bismuth has previously cooled down in the steam generators. The pumps were deliberately placed in the cold leg because the speed at the pump impeller must be higher than 2 m/s to generate this pressure. However, at the lower temperature, the risk of flow-induced corrosion is lower.

The lead-bismuth enters the steam generator at the top and exits at the bottom. Should a leak occur in the steam generator and steam enter the liquid metal, a box around the steam generators prevents steam bubbles from entering the reactor core. The steam must first rise to the elevated pump inlet, where it can be released to the inert gas above the lead bismuth. This is important because the reactor core has a positive void effect, unless it is operated with enriched uranium only, and the steam bubbles would increase power.

The inert gas system has a steam condenser to allow the steam to be removed from it. Should there suddenly be a large break in the steam generator, as happened in the submarine, the inert gas system has also a rupture disk that blows off the excess steam into the large pool of water around the reactor, where it condenses. This steam must not be blown off into the open air, because it is contaminated with polonium.

However, this flow routing is unfavourable for natural circulation. When the colder liquid metal exits the steam generators at the bottom, it has no reason to rise to the pump inlet on its own. So if both pumps should fail, perhaps because neither offsite power nor emergency power is available, the colder liquid will merely collect at the bottom like in a siphon. For this reason, part of the liquid metal is directed around the pump along the vessel wall. This is ensured by a flow baffle that extends around the entire circumference of the vessel wall on the inside. This part is directly fed to the pump inlet further down. If both pumps should fail, this partial flow cools down at the tank wall and thus drives the natural circulation. Therefore, the outer water pool has an important role in emergency cooling.

The reactor vessel is double-walled to prevent lead-bismuth from freezing at the wall. The gap has a leak detection system leaving enough time to discharge the reactor in case of corrosion of the inner vessel wall. The reactor can also be preheated with steam via the gap between the two vessels before it is filled with lead bismuth. The vessel has an inner diameter of 4.13 m and a height of 7 m.

Another striking feature of this reactor are the steel displacement bodies, especially the large displacement body above the reactor core. They minimize the necessary mass of lead and bismuth. As these bodies heat up from neutron and gamma radiation, they have numerous longitudinal holes through which some lead-bismuth rises and cools these displacement bodies. The large spaces between the steam generators and the pumps are also largely filled with such displacement bodies. There is therefore only 18 m³ of lead-bismuth inside the reactor vessel, which is only about a quarter of the interior space.

The whole concept of this reactor is already quite mature and elaborated. I could well imagine that this reactor will be built one day. However, there is no concrete construction decision so far. The company AKME Engineering, which is currently marketing this reactor, is still looking for further investors at home and abroad. Whether it will actually be economically competitive one day, as the manufacturer claims, remains to be seen. Such small reactors always have a problem competing with large reactors. Compared to other reactors of this size, however, the SVBR 75/100 appears very compact and cost-effective.

BREST-1200

Reactors cooled with lead-bismuth are regarded in the Generation IV International Forum rather as an intermediate step, as long as the materials for the lead-cooled reactor have not yet been sufficiently improved. However, the design of the Russian reactor BREST-1200, which shall be cooled with pure lead, may give us a first impression of what such a future power reactor might look like [37]. It is currently being developed in two versions in Russia, a smaller version with about 300 MW of electrical power, the BREST-OD-300, which could serve as a prototype, and a large version with 1200 MW as an economic replacement for pressurized water reactors sometime in the future. The basic principles of both versions are similar.

At first glance, the reactor looks like a combination of a loop design and a pool design. The reactor vessel consists again of an inner vessel and an outer vessel, sketched in Fig. 7.4, but the inner vessel looks more like a loop reactor. The central vessel section contains only the reactor core with its neutron reflectors and required shielding. The steam generators and pumps, on the other hand, are located in four peripheral sections arranged around the central vessel. This can be seen a little better in the plan view, Fig. 7.5. Each of these

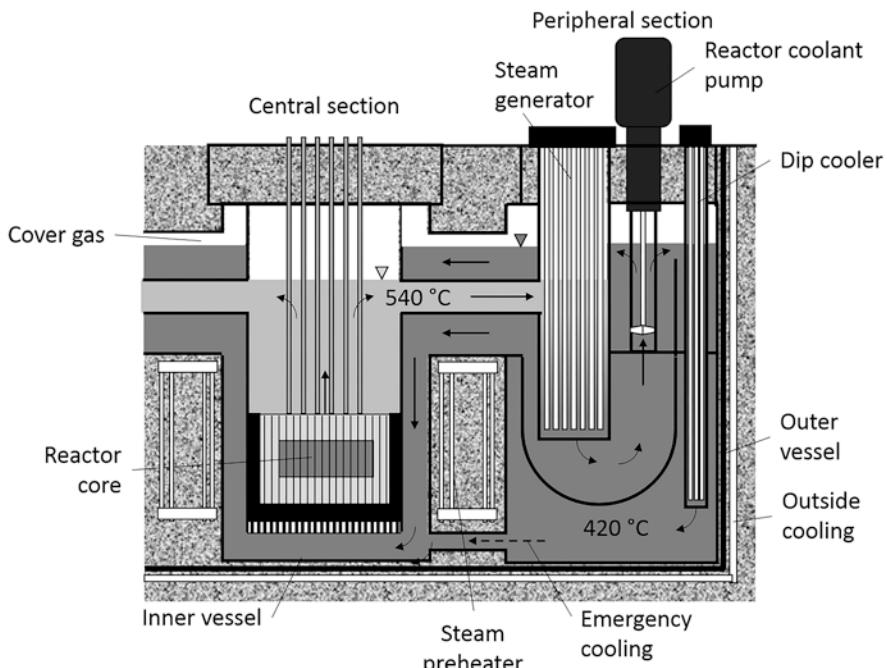


Fig. 7.4 Principle of the lead-cooled reactor BREST-1200

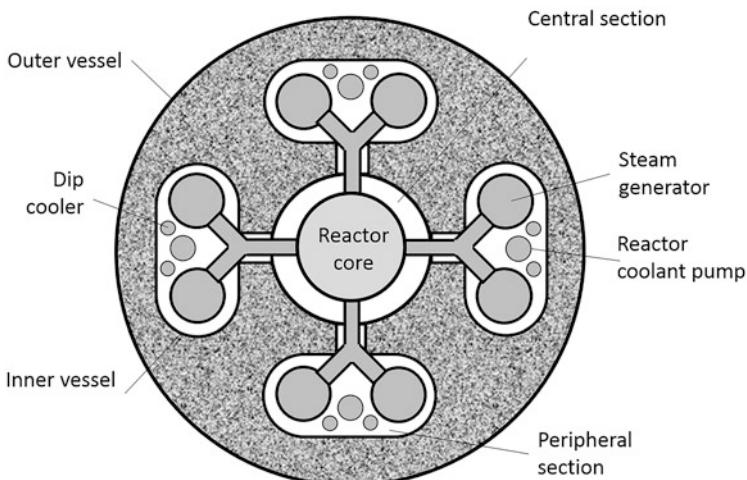


Fig. 7.5 Principle of the lead-cooled reactor BREST-1200 in plan view

peripheral sections has two steam generators, a coolant pump and two emergency coolers. The hot lead leaves the reactor core at a temperature of 540 °C and is fed through a coaxial pipe to the steam generators, where it is cooled down to 420 °C. The coolant pump provides a pump head then of 1.60 m above the liquid level in the reactor core. The four peripheral sections are then, on their cold side, something like elevated tanks for the liquid lead. From there, the lead runs through the annular cross-section of the coaxial pipe back to the central reactor core. In this way, the inner vessel wall is only exposed to a lead temperature of 420 °C, at which corrosion problems are not to be expected.

The whole arrangement is placed inside a common, large outer vessel set in concrete. Now this is rather a pool construction. The space between these vessels is used to preheat the whole system with steam. This can be achieved by lots of vertical boiler tubes in it, where superheated steam slowly preheats the reactor and all its external components to more than 400 °C. They can also keep the reactor warm in case of shut-down for a longer time period. In the short term, the residual heat of the reactor takes over this task. Heat must then rather be dissipated. This can initially be done by the steam generators with the aid of the coolant pumps. If there is no electricity to drive these pumps, it takes a while for the level in the peripheral vessels to equalise with the level in the central vessel, which initially maintains a forced flow through the reactor. Then, the two dip coolers in each of the peripheral vessels can transfer the residual heat to a passive emergency cooling system, similar to the one already

used in the sodium-cooled reactor. The hot lead rises from the reactor core to these dip coolers by itself and also runs from the dip coolers back to the central vessel by itself.

The outer vessel must also be cooled. This is done by outer cooling coils inside the concrete near the vessel wall. They keep the concrete colder than the melting temperature of lead, and would therefore freeze the lead on the vessel wall if the inner vessel were to leak.

This quite innovative design has the advantage that a leakage from the steam generator cannot transport steam bubbles into the reactor core. Otherwise, the steam could trigger an uncontrolled power increase due to the positive void effect. The steam pressure in this reactor must be set very high. Since lead melts at 327 °C, the feed water for the evaporator must be pre-heated to 340 °C to avoid a risk of lead freezing locally in the evaporator. At 340 °C, however, we are not far from the critical temperature of water. At such a high temperature, the feed water must already be supercritical to remain liquid. Therefore, the BREST-1200 reactor has a supercritical steam process with a live steam pressure of 245 bar and a live steam temperature of 520 °C at the inlet to the high-pressure turbine.

The reactor core of the BREST-OD-300 reactor consists of hexagonal fuel assemblies [38]. Unlike the sodium-cooled reactor, they do not need a fuel assembly box and their cross-section is similar to that of the fuel assemblies of the Russian VVER pressurized water reactor. The fuel rods have a similar pitch of 13 mm like in pressurized water reactors. They are about 9–10 mm in diameter. The speed at which the liquid lead flows through these wide fuel assemblies is then less than 2 m/s, which is helpful for the removal of residual heat by natural circulation. The active part of the fuel rod is only 1.1 m high. It is supplemented at the top and bottom by a layer of U-238 and by a large fission gas plenum. The whole fuel rod is then 3.2 m long. These ratios are similar to those of sodium-cooled reactors. The core diameter of the BREST-OD-300 is given as 2.4 m. The larger reactor BREST-1200 has an active core of 4.75 m in diameter. Fertile fuel assemblies are not to be installed, in accordance with the goals for higher proliferation resistance. The BREST-OD-300 is expected to achieve an average core power density of more than 500 MW/m³.

A ferritic-martensitic chromium steel is taken as cladding tube material, which will still be a challenge for the envisaged maximum cladding temperature of 650 °C. A uranium-plutonium nitride, (U-Pu)N, is currently favoured as the fuel. This fuel has occasionally been used for sodium-cooled reactors before. It has about ten times the thermal conductivity of MOX fuel. As a result, the fuel inside the fuel rod does not get as hot. The maximum

permissible rod power can then chosen to be somewhat higher. In the BREST reactor, however, it is only about 40–42 kW/m, similar to MOX fuel. Furthermore, the density is about 40% higher. This, together with a higher effective cross-section, means that the neutrons are more likely to be absorbed in the fuel and thus less likely to be absorbed in the coolant. This means that the reactor produces more new fuel such that fertile fuel assemblies can be omitted.

A disadvantage of this fuel is that the nitrogen in this fuel should not be the ordinary ^{14}N , because this nitrogen isotope likes to absorb a neutron emitting a proton, such that ^{14}C is formed. With its half-life of 5730 years, we usually take this isotope for age determination, but in high concentration it is more of a repository problem. Therefore, the nitrogen in (U-PU)N contains highly enriched ^{15}N instead. This is stable and present in nature, but quite rare and therefore expensive to enrich.

There are quite a few other designs for pool reactors cooled with lead or lead-bismuth, but they have not yet been worked out in such detail. They have emerged in the last few years in Japan, Korea, the USA and Europe. I think it's getting a bit tedious to introduce these concepts all individually. Instead, I would like to draw attention to a completely different concept, which is unlikely to become economically competitive as a power plant, but is nevertheless technically highly interesting.

MYRRHA

Fast reactors are not only needed to generate electricity. In research, they are still needed today as neutron sources, e.g. to measure cross sections, to irradiate materials or to produce radionuclides, e.g. as medical isotopes. In particular, current research on fusion reactors, where materials will be exposed to a very high neutron flux with high energy, has required many irradiation experiments. High-energy protons can be produced with a particle accelerator, either a linear accelerator or a cyclotron, where the protons are accelerated in circular orbits. If the protons are fired at a target, e.g. a metal plate, neutrons are knocked out of the nuclei, which can be used as a neutron source. The particle accelerator SINQ at the Paul Scherrer Institute in Switzerland, for example, generates a neutron flux of $10^{14} \text{ n/cm}^2\text{s}$ in this way. If the protons are accelerated to an energy of a few hundred MeV, they can literally smash heavy atomic nuclei. This is called a spallation. It produces different fission products with each hit, as well as about 10–20 neutrons, depending on the energy of the incident proton. But it makes the target very hot. You either build it out

of tungsten, which has an extremely high melting point, and cool it intensively with helium. Or you can use lead or lead-bismuth, which melts quickly but can be easily cooled down again as a liquid metal in a heat exchanger. Such a liquid metal target with lead bismuth has already been used successfully in the SINQ facility in Switzerland [39].

An even higher neutron yield is obtained if the neutrons produced also generate a few nuclear fissions. The nuclear fuel then serves as an amplifier of the generated neutron radiation. For this purpose, such a target is installed in the middle of a reactor core, preferably of a fast reactor, if a fast neutron spectrum is required for the irradiation experiments. For this purpose, the proton beam is injected into the reactor from above by means of deflecting magnets, sketched in Fig. 7.6. The beam tube ends in the centre of the core with a spherical calotte, around which liquid metal flows from the outside. If a heavy liquid metal is used, such as lead-bismuth, the reactor coolant is also the liquid target material.

The proportions in Fig. 7.6 are by no means exaggerated. On the contrary, only the rear part of the particle accelerator is shown. To the left of the picture, there is an acceleration section which is just as long. When I first got to see a drawing of this facility, the reactor seemed like a small beam dump. The particle accelerator was 450 m long, but the reactor was only 7 m high. It was immediately clear to me: this is no way to build a power plant. But this is not about a power plant at all.

In this operation, the reactor is not critical, but only an amplifier of the neutrons produced. This is comparable to a microphone and an amplifier, where the speakers either emit more sound when you speak louder into the microphone, or by turning up the amplifier more. In this reactor, we can increase the neutron flux by amplifying the proton beam or by pulling the control rods out a little further. If we pull the control rods too far, the reactor

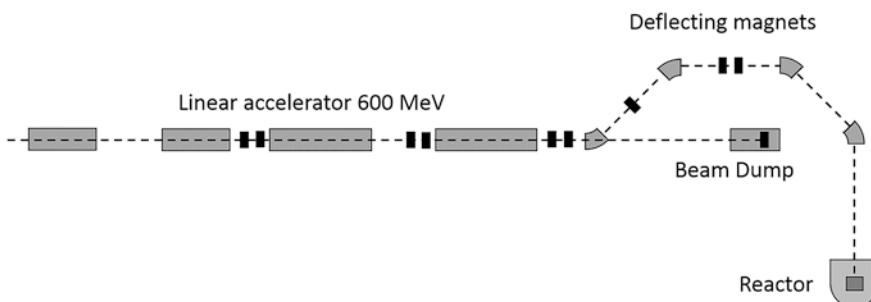


Fig. 7.6 Combination of a linear accelerator with a lead-cooled reactor to form an accelerator-driven system

becomes supercritical. This can be compared to the microphone amplifier, which emits a piercing noise due to the feedback of the sound waves between the microphone and the loudspeaker if you turn it up too high. The volume then increases uncontrollably. In the reactor, on the other hand, this condition can be controlled within a certain range thanks to the delayed neutrons, as long as it does not become promptly supercritical; the neutron flux then controls itself via the Doppler effect, the density effect or the void effect. But even if we have hardly any delayed neutrons, too little Doppler effect, or a void effect in the wrong direction, it is still possible to operate such a reactor under control by simply remaining subcritical, i.e., using it only as an amplifier. Such a reactor is called an *Accelerator Driven System (ADS)*.

The Belgian research centre SCK-CEN decided a few years ago to replace the old research reactor BR2 by such an accelerator-driven system [40]. The reactor of this plant, called MYRRHA, is cooled with lead-bismuth, which is heated from 270 °C to 410 °C in the reactor core. The thermal power of 100 MW is only delivered to a condenser. It is not intended to generate electricity there. Rather, as a research reactor, it is intended to provide the largest possible neutron flux of more than 10^{15} n/cm²s with a neutron energy of more than 0.75 MeV.

The reactor core is constructed from hexagonal fuel assemblies with a width across flats of 87 mm. The reactor core has a diameter of 1.80 m and a height of 1 m, of which only 600 mm are loaded with fuel. The fuel rods have an outer diameter of 6.55 mm and are 1200 mm long in total. Of the 151 core positions, 37 are available for measuring positions, control rods or absorbers, instrumentation or other experiments. In the centre of the reactor, the beam tube of the proton accelerator is inserted into the reactor instead of a fuel assembly. It thus generates a spallation neutron source in the middle of the reactor core. Channels with material samples can be inserted into the measuring positions from above in order to irradiate them, but also samples with americium or curium in order to fission it experimentally. The additional neutrons produced there are not a problem for the control of the reactor in subcritical operation. This would make MYRRHA an ideal facility to test such nuclides before they are fissioned in larger quantities in a critical reactor. On the other hand, MYRRHA should also be designed in such a way that the reactor can be operated critically, i.e. without a particle accelerator.

The many access points to the measuring positions above the reactor would have been a hindrance to conventional refuelling from above. MYRRHA therefore opted for an unconventional alternative: The fuel assemblies would be changed from below. In the heavy liquid metal, they are lighter anyway and float upwards. The loading machine therefore first pulls the fresh fuel

elements underneath the reactor core and then allows them to float upwards into the core support plate at the respective core position. Actually, they would stop there by themselves due to the buoyancy. However, they are still secured by a locking device. The loading machine had to be specially developed for this purpose, as well as an ultrasonic inspection of the reactor core from below in order to be able to check the core. After burn-up, the spent fuel assemblies are first placed in a fuel assembly storage area next to the core, i.e. inside the reactor pool, where they can cool down for a while before being taken out again by the loading machine.

The design of the reactor has changed several times over the years [41]. The latest design of which I am aware is sketched in Fig. 7.7. The cylindrical pool reactor sits in an outer cylindrical vessel set in concrete, which serves as protection against leakage. A cooling screen of vertical tubes in the gap between the two vessels keeps the surrounding concrete sufficiently cold. It can also dissipate residual heat should the normal cooling system fail. Surrounding the core are four steam generators and two pumps in the cold leg, which force the lead-bismuth through the core. Furthermore, the two loading machines with an airlock for the fuel assemblies are located in this annular space. They can load the core from two sides. The inert gas above the surface of the liquid metal has a pressure relief system with a rupture disc that limits the pressure there to a maximum of 6 bar.

The secondary system consists of two independent loops, each of which can remove the full heat. For this purpose, the steam generators first discharge the

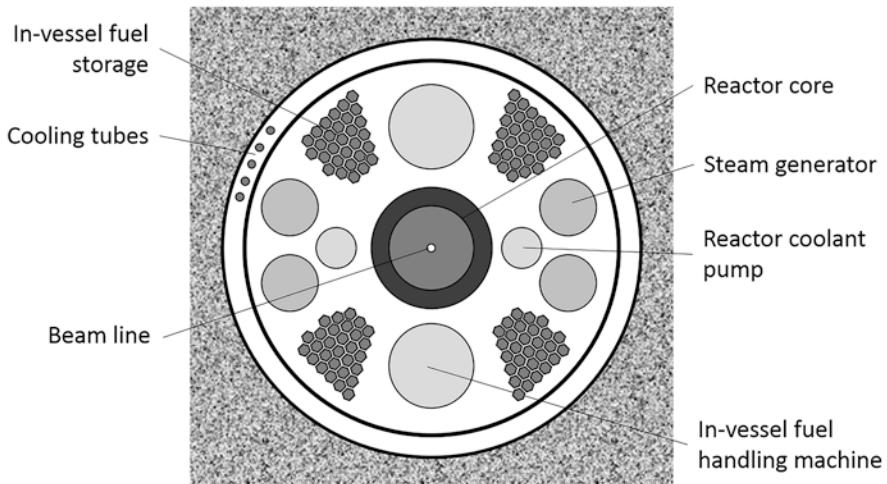


Fig. 7.7 Sketch of the MYRRHA reactor in horizontal cross-section

steam by buoyancy alone to a drum in which the steam is separated from the liquid. The saturated steam is then fed directly to an air condenser via a throttle. The secondary circuit is kept at a pressure of 15.5 bar in order to stay above an evaporation temperature of 200 °C so that the lead-bismuth does not freeze on the evaporator tubes. Furthermore, the residual heat can also be dissipated via the cooling screen, which cools the vessel from the outside.

In the meantime, the Belgian government has approved the funds for the construction of this facility. Construction will start first with the linear accelerator. This gives the developers of the reactor a little more time for the detailed reactor design. In case of success, the MYRRHA reactor would be the first reactor outside of Russia to be cooled with lead bismuth. The reactor should be ready for operation from 2036.

Conclusion

The lead-cooled reactor is obviously a promising alternative to the sodium-cooled reactor. Concepts that have already been worked out in great detail and may soon be realized use lead-bismuth as a coolant and are operated at lower temperatures, but otherwise do not differ fundamentally from lead-cooled reactors. However, let me briefly compare sodium- and lead-cooled reactors on the basis of the goals of the fourth generation.

Sustainability

A reactor cooled with lead or lead-bismuth can produce plutonium just as sustainably and recycle transuranics just as well as a sodium-cooled reactor. Polonium Po-210, which is produced from bismuth during irradiation, is a safety issue for reactor operation but not a repository issue. It decays too quickly for that.

Safety

Since lead and lead-bismuth behave largely inert in water and air, these coolants have a clear safety advantage over sodium. Moreover, the very high boiling point of these heavy liquid metals gives them a safety advantage over sodium: the positive void effect, which can hardly be avoided in sodium, can hardly cause any damage even under very hot operating conditions. In the

case of lead-bismuth, on the other hand, the highly radiative polonium Po-210 poses a new safety problem, as experience with Russian submarines has shown. Pure lead as a coolant would have this problem only slightly.

As was already the case with sodium-cooled reactors, residual heat removal is also easily achievable with passive systems. It can be removed by natural circulation without the need for an electrically driven system.

Reliability

Even though the sodium-cooled reactors were prepared for a sodium fire in terms of safety, there were always outages in operation of these reactors because of fires. This problem does not exist with lead. Otherwise, however, the sodium-cooled reactor is a mature technology from which we can expect a high level of reliability. Chemically, sodium behaves completely inert inside the reactor, even over many years of operation. Corrosion in lead or lead-bismuth, on the other hand, will remain a challenge for the time being. For the lead-cooled reactor in particular, there is no operating experience yet.

Economic Competitiveness

Lead or lead-bismuth do not need an intermediate system between the liquid metal cooled primary system and the steam cycle. This undoubtedly saves costs compared to the sodium-cooled reactor. So a lead-bismuth cooled reactor is likely to be more economical than a sodium cooled reactor. However, if the steam generator is integrated directly into the reactor vessel, care must be taken to prevent steam bubbles from entering the reactor core. Otherwise, a serious safety problem might arise.

Proliferation Resistance

In accordance with the goals of the fourth generation, strict attention was paid to the new, lead-cooled concepts to ensure that no fertile fuel assemblies were used. But they still produce more plutonium than they consume. Consequently, the risk of proliferation is lower. Actually, however, sodium-cooled reactors could be designed in this way as well. The differences in core design are not due to the coolant, in my opinion.

Glossary

Electrolyte	A solid or liquid ion conductor
Eutectic alloy	A mixture of two metals in such a way as to give the lowest melting point
Ferritic steel	A steel with the crystal structure of iron, consisting predominantly of iron but may contain up to about 18% chromium to improve corrosion resistance
Ferritic-martensitic steel	A ferritic steel with a higher carbon content that can be hardened, transforming its microstructure to martensite
Spallation	Shattering of an atomic nucleus by bombardment with high-energy neutrons, producing multiple fragments and multiple neutrons
Target	The “target” of a particle beam, i.e. the object that is to be bombarded



8

Gas-Cooled Fast Reactors

There are other options for cooling a fast reactor. Just as you can cool a thermal reactor with gas, it would also be conceivable to cool a fast reactor in this way. Gases that could be used for this purpose are, for example, helium, CO₂ or water vapour. I would like to briefly present a few examples in this chapter.

Helium

For thermal reactors, helium ended up being the best gaseous coolant. Helium does not react with air or water like sodium; nor is helium corrosive like lead or lead-bismuth. Helium is simply a noble gas and does not undergo any chemical reactions. The density of helium is so low that it does not moderate neutrons, at least not appreciably. The neutron absorption of helium is practically zero. Helium doesn't boil, so it doesn't produce a void effect, just a small, albeit positive, density effect of the coolant: as helium density decreases, reactor power increases. However, when a helium-cooled reactor gets hotter, it is essentially the Doppler effect of the fuel that controls the power back. Since helium neither moderates nor absorbs, a helium-cooled reactor has a fairly good yield of fast neutrons. Its neutron spectrum is harder, which means that it has higher neutron energies than a sodium-cooled reactor. This enables to produce a lot of new fuel during burn-up. Helium is not activated by neutron capture, so it does not produce problematic nuclides. Helium is also transparent, unlike liquid metals, so that you can look inside the reactor, at least with

Table 8.1 Comparison of the physical properties of helium and sodium

	Helium		Sodium
Cross section of thermal neutron absorption	0		430 mbarn
<i>Important material data for core entry and exit conditions:</i>			
Temperature (°C)	500		500
Pressure (bar)	1	70	1
Density (kg/m ³)	0.062	4.13	835
Viscosity (μPa s)	38.6	38.6	236
Thermal conductivity (W/m K)	0.301	0.304	64.2
Specific heat capacity (kJ/kg K)	5.195	5.195	1.258

cameras. From a chemical and neutronics point of view, helium would be the ideal coolant for a fast reactor.

I have compared the physical properties of helium with those of sodium in Table 8.1. Not only the considerably lower density of helium is striking, especially at atmospheric pressure, but also its low thermal conductivity. It is still better than that of many other gases, but compared to sodium, helium is more of a thermal insulator. This results in the general difficulty with this type of reactor: the poor heat transfer of helium. On the one hand, a helium-cooled reactor must be operated at higher pressure. At 70 bar, the density is already almost 70 times as high as at atmospheric pressure. However, the heat transfer from the fuel rod to helium would still be more than 500 times worse than that in sodium at the same flow rate and temperature. The reactor core must therefore be cooled at much higher flow rates and the power of the fuel rods must be reduced in order to be able to maintain acceptable cladding temperatures.

Helium-Cooled Fast Reactor Concepts

A helium-cooled fast reactor has never yet been built. However, there have already been numerous approaches to this [42]. In Germany, quite detailed studies of a “gas breeder”, as a fall-back option in case the sodium-cooled fast reactor should not be successful, were already made in the 1960s [43]. The design included a reactor integrated into a pre-stressed concrete vessel together with its steam generators. Alternatively, a reactor in a pressurized vessel was proposed, with the hot helium driving a gas turbine. However, the studies were discontinued in favour of the sodium-cooled reactor after 1971. Similar approaches were also underway in the U.S. at this time, led by General Atomics. In the 1970s, studies continued in the United Kingdom (UK), where early experience with CO₂-cooled thermal reactors was relied upon.

Finally, in the 1990s, studies were carried out in Japan, but never led to the construction of a prototype.

After 2000, within the framework of the Generation IV programme, a team of French scientists decided to work again on a new concept for such a reactor. This essentially resulted in two reactor designs, namely a large reactor with 2400 MW thermal power [44], in order to be able to assess the economic viability of the concept, and a small prototype with 75 MW thermal power, since no one would build such a large reactor without having tested first a small prototype. The prototype was given the beautiful name ALLEGRO [45]. Today, an international team of European scientists continues this work.

In the design of the reactor core, unfortunately, we cannot take over much from the helium-cooled thermal reactor. For a fast neutron spectrum, we must omit all graphite in the reactor core. There have been repeated attempts to use the proven TRISO particles, but the pyrographite therein could not withstand the high neutron flux for long. For a first design of the reactor core, we can rather assume the fuel assemblies of a sodium-cooled reactor, i.e. MOX fuel in a stainless steel cladding tube. The fuel rods of the Phénix reactor had a diameter of 6.55 mm and were wrapped with a wire to keep them at distance and to mix the coolant well. The ALLEGRO fuel rods would then be 1.30 m long and tightly packed hexagonally, as sketched in Fig. 8.1 on the left. Moderate heating of the helium in the core from 260 °C to 516 °C, as recently envisaged, would certainly be sufficient for a prototype reactor. The ALLEGRO reactor should be operated at a pressure of 70 bar.

However, with such fuel rods, the maximum permissible cladding temperature would be limited to approx. 650 °C. At higher temperatures, stainless

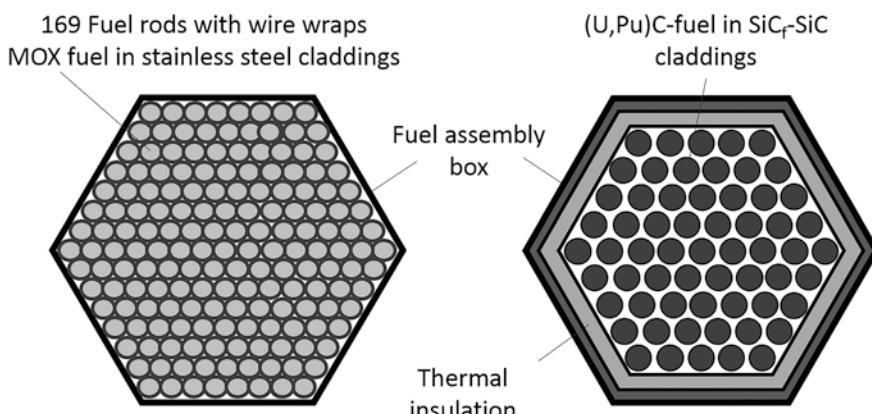


Fig. 8.1 Fuel assemblies of the ALLEGRO fast helium-cooled reactor; left: MOX fuel assembly with stainless steel cladding tubes, right: ceramic fuel assembly

steel begins to creep noticeably and the cladding tube could either collapse under the external pressure or inflate under the internal pressure. Alternatively, the cladding tubes could be made of fiber-reinforced silicon carbide ($\text{SiC}_\text{f}-\text{SiC}$). To prevent them from breaking off easily, these tubes have a larger diameter, as shown in Fig. 8.1 on the right. The fuel inside the cladding tubes should then be uranium and plutonium carbide ($\text{U},\text{Pu}\text{C}$). Unlike MOX fuel, ($\text{U},\text{Pu}\text{C}$) has a higher density and much higher thermal conductivity, so that higher power could be obtained with such fuel rods. Table 8.2 shows some physical properties of these fuels in comparison.

What is probably more important in this application is that a carbide fuel is chemically more compatible with the silicon carbide cladding tube. However, this fuel, like the ceramic cladding tubes, is still little tested today. The development strategy for the ALLEGRO reactor therefore envisages first building a reactor core from known MOX fuel assemblies and then testing individual ceramic fuel assemblies in it at a later stage. Both fuel assemblies therefore shall have the same width across flats of 110 mm. The ceramic fuel assembly should be able to withstand higher temperatures. For testing, these fuel assemblies would have to be throttled somewhat at the inlet so that less helium passes through. However, the box walls of the ceramic fuel element must be thermally insulated then on the inside, otherwise the neighbouring metallic fuel element box will overheat.

Alternative fuel assembly designs with silicon carbide plates, between which ($\text{U}-\text{Pu}\text{C}$) fuel pellets are to be mounted, have also been published [44]. Figure 8.2 shows such a design. The fuel pellets with a diameter of just over 11 mm and a thickness of 6.5 mm are stuck in a honeycomb structure, each formed by two plates of fibre-reinforced $\text{SiC}_\text{f}-\text{SiC}$. These plates must be lined on the inside with a thin layer of tungsten to prevent fission products from diffusing through. Each pair of plates with internal fuel pellets forms a sandwich structure with a width of 8.4 mm. This is inserted into a ceramic fuel assembly box, as sketched in Fig. 8.2 on the right. A 4 mm gap is left between the sandwich plates, through which the helium flows upwards and cools the plates. This design already allows some heat transfer by conduction across the flow and thus temperature equalization of hot spots. However, such an

Table 8.2 Physical properties of some alternative fuels [46]

	UO_2	MOX	UN	UC
Thermal conductivity (W/m K) at 1000 °C	2.79	2.60	22.9	28.8
Thermal conductivity (W/m K) at 2000 °C	2.12	2.33	28.7	64.6
Density (kg/m ³) at 20 °C and 5% porosity	9664	9770	13,619	12,970
Melting point (°C)	2850	2750	2850	2365

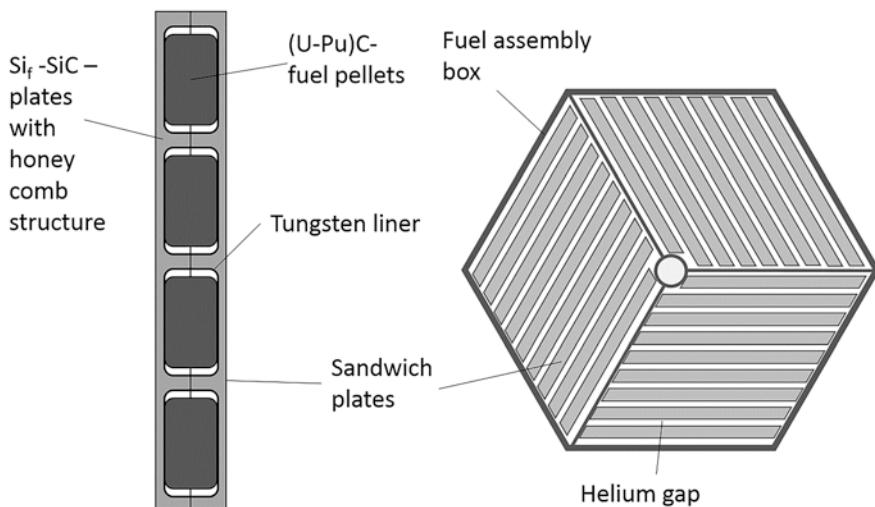


Fig. 8.2 Design of a ceramic fuel assembly with silicon carbide sandwich plates

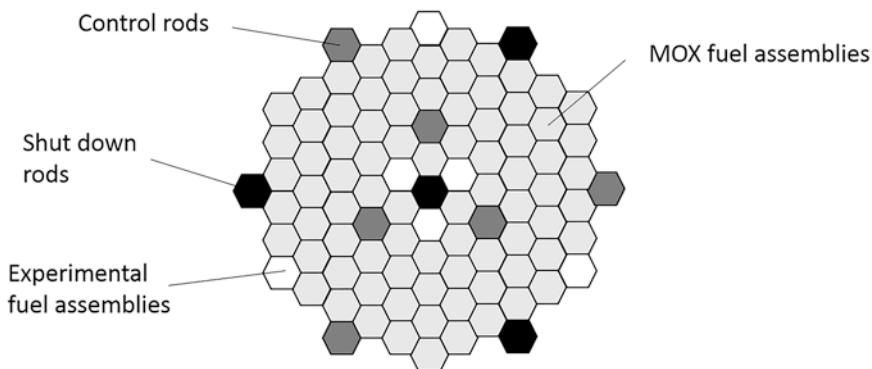


Fig. 8.3 Structure of the core of the ALLEGRO reactor

innovative fuel element is completely new and still requires many preliminary tests before it can be used.

The core of the ALLEGRO reactor consists basically of 81 MOX fuel assemblies, as outlined in Fig. 8.3. In addition, there are six core positions which can be equipped with ceramic experimental fuel assemblies, as well as ten core positions for the control and shutdown rods. The hard neutron spectrum of the helium-cooled fast reactor and the lack of absorption in helium require good shielding of the core against the reactor pressure vessel. Therefore, the reactor core is surrounded by four additional rows of reflector elements made of stainless steel and around them by three rows of elements made of

boron carbide, B_4C . The carbon in the boron carbide slows down the fast neutrons to thermal energy so that the boron isotope $^{10}_4B$ can absorb them. The stainless steel reflector prevents these thermal neutrons from getting back into the active core. In addition, the MOX fuel rods of the active core have a 30 cm thick layer of reflector material at the top and bottom and a 50 cm thick layer of absorber material below and above them, respectively. On the other hand, the core has no fertile fuel assemblies. The hard neutron spectrum makes it possible to retain 97% of the plutonium used in the ALLEGRO reactor at the end of burn-up. Thus, no further plutonium is produced.

The active part of the ALLEGRO core has a volume of about 0.75 m^3 . The MOX fuel would have to contain about 30% plutonium in this small core to make the reactor critical. At 75 MW thermal power, the core has then a power density of about 100 MW/m^3 . Today, however, a power density significantly lower than this is also being considered.

In the ALLEGRO reactor, the hot helium is supplied to two heat exchangers located next to the reactor. They are simply cooled with water, which then releases its heat into the environment. Electricity production is not yet planned for this prototype. A helium blower with 426 kW power generates a helium overpressure of approx. 1 bar (e.g. from 69 to 70 bar), which pumps the helium through the core with a mass flow of 28,000 kg/s. The mean flow velocity in the core is approx. 80 m/s.

In the larger power reactor, three heat exchangers are provided instead, which could be used to generate steam for a steam cycle, for example. However, the pressure on the secondary side is then higher than in the primary system, so that we must check a potential accident in which steam enters the primary system. Similarly, water could leak into the reactor pressure vessel during shutdown. An alternative secondary circuit would then be another gas system consisting of a nitrogen—noble gas mixture, which has already been proposed for the ANTARES high-temperature thermal reactor. This gas mixture would then drive a gas turbine in a closed circuit, the exhaust gas of which would supply heat to a steam cycle via a heat recovery steam generator.

More problematic is the removal of residual heat if the helium blower should fail. This is not a problem with liquid metal cooled reactors. They can simply transfer the residual heat to an emergency cooler by natural convection of the coolant inside the reactor without the reactor core overheating unacceptably. A pressurized water reactor can also transfer its residual heat from the reactor core to the steam generators by natural convection alone. However, in a helium-cooled reactor, the emergency cooler needs a fan. If both the off-site power and the emergency power fail, the fan of the emergency cooler could also be driven by a steam turbine, if necessary.

Even more difficult is an accident in which the reactor has a leak. If a liquid metal-cooled reactor has a leak at the top, only the inert gas escapes and only until the pressure in the containment has risen to the pressure of the inert gas. The reactor is otherwise pressureless. A leak in the lower part of the reactor vessel must be avoided, otherwise coolant will be lost. This is why these vessels are double-walled. If a leak were detected in the gap between the two vessels, there would be sufficient time to discharge the reactor. In a pressurised water reactor, this accident scenario is more problematic. The cooling water is also forced by the steam pressure through a leak that sits above the reactor. In the case of the reactor at Three Mile Island, the water had escaped through an open valve in the pressurizer, which is at the highest point of the primary system. For this case, there are pumps for emergency injection of water and a large water reservoir in the containment from which the water is taken for emergency injection. Furthermore, some pressure accumulators are required that can feed water into the primary system while the pumps are being started.

In the case of the helium-cooled reactor, however, an appropriate emergency supply of helium is even more difficult. Huge pressurized accumulators are needed to hold sufficient helium. A total of six large accumulators are provided for this purpose, three of which are filled with helium and three with nitrogen, and which feed into the primary system in the event of a coolant leak. Nitrogen has a density seven times greater than helium and a larger absorption cross section for thermal neutrons of 80 mbarn.

The gas must not be allowed to escape into the environment in the event of such a loss-of-coolant incident. Therefore, a guard vessel is provided around the primary system and its emergency coolers. Depending on the size of the guard vessel, it must be able to withstand a pressure of 5–10 bar, because neither helium nor nitrogen can be condensed. On the other hand, a higher pressure is also helpful for heat transfer in the reactor core.

The limited possibilities for residual heat removal complicate as well the normal fuel change. The reactor may not simply be opened for this purpose. Instead, the fuel assemblies must be brought individually through a lock in the reactor lid into the interior of the pressure vessel, where an internal fuel assembly loading machine sets the fuel assemblies in position. Figure 8.4 shows how the large power reactor might look like. In order to provide sufficient space for the fuel assembly loading machine, the control rods must be driven into the reactor from below.

The second or even third cooling system for normal operation, the three emergency coolers with fans for the failure of the actual cooling system, the six large pressurized accumulators and the guard vessel that encloses all this are not required for a thermal high-temperature reactor. At this point, at the

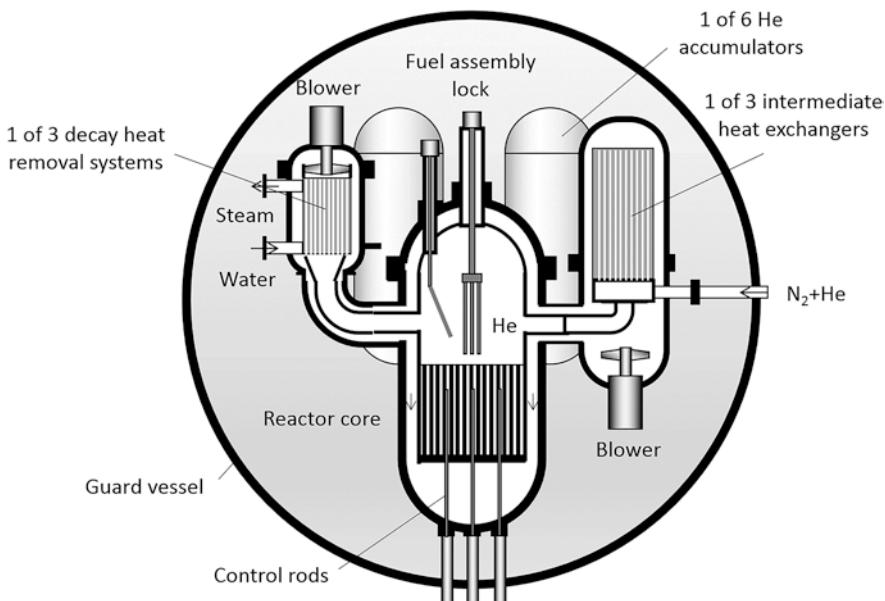


Fig. 8.4 Schematic of a helium-cooled fast reactor

latest, it becomes apparent that the high-temperature thermal reactor was actually an ingenious invention. Its low power density on one hand, only $3\text{--}4 \text{ MW/m}^3$, and its large inventory of graphite on the other hand make an emergency cooling system unnecessary. If necessary, the graphite can simply dissipate the residual heat by conduction to the outside of the reactor without causing the core to overheat unacceptably. Such a simple system would undoubtedly be desirable as well for the fast helium-cooled reactor, but up to now, no one has had a bright idea.

Steam Cooled Fast Reactors

In the case of pressurised or boiling water reactors, the emergency cooling concept is not as simple as in liquid metal-cooled reactors, but their safety systems have been proven for many years. For example, it is easy to convert a boiling water reactor into a fast reactor by omitting all water channels in the reactor core, arranging the fuel rods as closely as possible and increasing the plutonium content until the reactor still becomes critical. There have also been proposals for reactors with supercritical water [9]. In general, however, all these concepts have the same safety problem: there is a large amount of

water and a large amount of fissile material inside the reactor pressure vessel. Only because the fuel rods are close together and the reactor core is strongly under-moderated, a thermal neutron spectrum does not occur. If the same amount of fissile material and water were arranged differently in the pressure vessel, so that the volume of water in the reactor core increased, a thermal spectrum with correspondingly high cross sections for nuclear fission would result and the reactor core would become prompt supercritical with this high fuel inventory.

Certainly, during normal operation of the reactor, such an event does not occur. However, in the course of a severe accident, for example, some of the fuel assemblies could melt and thus make room for more water in the core. Or, during refuelling, several fuel assemblies could be pulled out by mistake and the water content increased too much until the fast reactor becomes critical as a thermal reactor. In a thermal reactor, such an accident cannot occur, not only because there is enough absorber material in the core, but also because both a significant increase and a decrease in the moderator fraction makes the reactor subcritical. It is designed close to the optimum moderator-to-fuel ratio.

An early design of a “steam breeder” was intended to circumvent this problem [47]. A fast reactor, which was to be cooled only by steam, was designed for this purpose so that the steam was generated in an external injection evaporator. The process schematic is shown in simplified form in Fig. 8.5. The core of this reactor was designed of hexagonal fuel assemblies with tightly packed fuel rods, as was also common for sodium-cooled fast reactors at that time. It had an inner fissile zone with a lower plutonium content of about 13% and

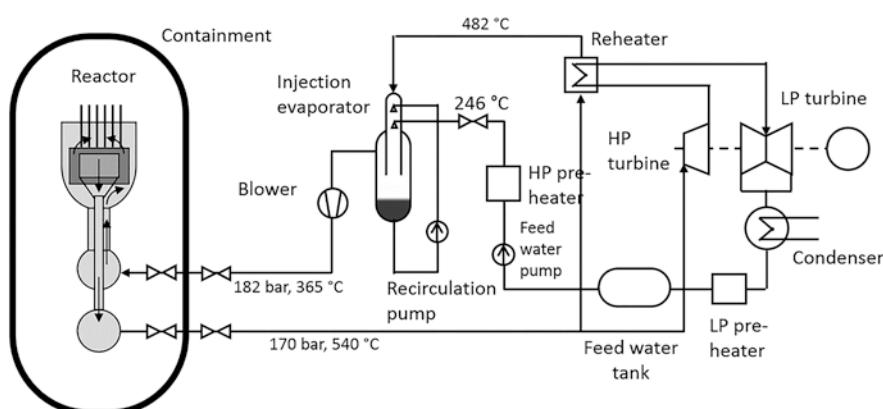


Fig. 8.5 Schematic of the main cooling circuit of a steam-cooled fast reactor with external injection evaporator

an outer fissile zone with a higher plutonium content of about 16%. In addition, it was surrounded radially and at the top and bottom by fertile zones with U-238. For cooling the reactor core, saturated steam at about 182 bar enters first from below into the outer fertile zone and into the surrounding neutron reflector, where it superheats somewhat as it rises. Then it flows downwards through the fissile zone, where it is finally superheated to 540 °C. Part of this steam (about 60%) is fed directly to the high-pressure turbine. The remainder first cools down to 482 °C in the reheater between the high-pressure and low-pressure steam turbines and then flows on to the injection evaporator. There, preheated feed water from the steam cycle at 246 °C is injected, which cools the steam to saturated steam temperature and simultaneously generates further steam. The steam generated in this way at 165 bar has still to be brought back to a pressure of 182 bar by a blower before it can be fed back into the reactor.

For refuelling, this reactor should be flooded with water so that the spent fuel can be transported under water to the adjacent fuel pool, as in a pressurised water reactor. The power plant concept also includes two auxiliary cooling circuits within the containment, each of which can remove residual heat without the need for the main cooling circuit shown. They also operate according to the principle of the injection evaporator. On the one hand, the steam generated there forms the emergency cooling system of the reactor; on the other hand, this steam is also used during normal operation of the power plant to generate secondary steam via another steam generator. It drives the blower, the feedwater pumps and a small steam turbine, which covers the house load of the power plant. An auxiliary steam boiler is also required for start-up.

This concept thus consistently avoids large quantities of water in the reactor pressure vessel and in the surrounding components. On the other hand, however, a large water inventory is available from the feedwater tank of the main cooling circuit, with which steam could be regenerated if a leak were to occur.

The striking features of this concept are the reactor's steam supply from below and the downward steam flow in the core. This was deliberately chosen. Should the reactor overheat to such an extent in an accident that the fuel assemblies melt, the steam expels the destroyed fuel assemblies downwards. In the narrow steam tube below the reactor, they are well cooled by the rising steam on the one hand. On the other hand, this prevents them from forming a critical accumulation of core melt. The concept was designed in such a way that the reactor could be flooded with water in an emergency or even in the event of a core meltdown. In the cold, fully flooded state, the core was found

to be subcritical in the calculations. However, when the water boiled, the core became prompt supercritical, unless the shutdown rods had been inserted beforehand [48]. The concept was thus practically dead for the time being.

This reactor was never built, primarily because the sodium-cooled fast reactor turned out to be simpler and safer. In the Generation IV program, such proposals were not taken up again either. I still like to report on these early concepts, even if they were not successful, because I got the impression that mistakes would be repeated if we did not understand them. Incidentally, the power plant was already designed and calculated in astonishing detail, considering how simple computer technology was at the time.

Conclusion

For gas-cooled fast reactors, mainly helium is considered as a coolant today, similar as for the thermal high-temperature reactor. However, the moderator graphite is omitted. The reactor core has thus a much higher power density than that of the thermal high-temperature reactor.

Although helium cooling is chemically and neutronically very promising for fast reactors, it creates problems with the removal of residual heat in the event of an accident, which would be comparatively easy to solve with liquid-metal-cooled fast reactors. Steam cooling, on the other hand, carries the risk of a prompt supercritical reactor if the reactor core were to get too much water. That is even worse. Nevertheless, as in the previous chapters, let us briefly discuss of how the concepts for helium-cooled fast reactors would have to be evaluated according to the criteria of the Generation IV programme.

Sustainability

A fast reactor cooled with helium has a harder neutron spectrum than a liquid metal cooled fast reactor and, if properly designed, could produce even more plutonium. However, the reactor designs of recent years do not make use of this. The aim there was rather to keep the plutonium inventory approximately the same.

Safety

Chemically, helium is inert, so that no problems with corrosion or even violent reactions with air or water are to be expected. Physically, there cannot be a positive void effect, because the reactor is only filled with gas during operation; this cannot boil. When the reactor gets hotter, the Doppler effect always reliably controls the reactor power back. A safety problem, however, is the removal of residual heat if the cooling fan should fail or even if there should be a leak in the primary system. A passive safety system for this case is difficult to design, since the flow in the core always requires a high velocity.

Reliability

A gas-cooled fast reactor has never been built before. Many experiments would therefore be required before a reliable system could be developed from the designs available today. In particular, there is a lack of experience with ceramic fuel assemblies. Unfortunately, the many years of experience with helium-cooled thermal reactors do not help either.

Economic Competitiveness

The safety problem of residual heat removal in the event of an accident may be solvable, as recent publications on the subject have already indicated. But all this does not look cheap. However, it is still too early to be able to derive a serious economic analysis from the current power plant designs.

Proliferation Resistance

Even with the new concepts for helium-cooled reactors, care is taken to ensure that no fertile fuel assemblies are used. The risk of proliferation is then similarly low as with lead-cooled fast reactors.

Glossary

Carbide fuel	A fuel made of uranium and plutonium carbide, which has a higher density and thermal conductivity than MOX fuel
Fissile zone	Area of the reactor core that contains fissile material, in contrast to the fertile zone
Injection evaporator	A device for generating steam by injecting water into superheated steam



9

Molten Salt Reactors

The sixth reactor type in the Generation IV programme is something completely different. The fuel uranium or plutonium is not present here as a solid pellet or particle, but in liquid form. Uranium, thorium, but also transuranium elements form salts with fluorine that melt even at moderate temperatures. Mixed with other fluorides, such a molten salt can then be both fuel and coolant. For research, this type of reactor is currently the most exciting option. So far, it is mainly physicists and chemists who have done the preliminary work, and there is still a lot of design work to be done before it becomes a real power plant. But I'd be happy to explain how far we've come so far.

Fluorides

All nuclear fuels, i.e. uranium and plutonium, but also thorium, neptunium, americium and curium form fluorides (ThF_4 , UF_4 , NpF_4 , PuF_3 , AmF_3 , CmF_3). As pure fluorides, they have a melting point of slightly more than 1000 °C, but as salts they are miscible with other fluorides and can thus have a much lower melting point of about 500 °C. Fluorine has only a very small cross section for the absorption of thermal neutrons, about 9.5 mbarn. Suitable mixing partners are first sodium fluoride, NaF , which produces a low melting point, and zirconium tetrafluoride, ZrF_4 , because zirconium, in particular, has also a low absorption cross section. Better than sodium fluoride, however, with its high absorption cross section of 517 mbarn, are lithium fluoride, LiF , and beryllium fluoride, BeF_2 , since ^7Li and ^8Be have absorption cross

sections of only 45 mbarn and 8.5 barn, respectively. This gives a better neutron balance. Lithium has two stable isotopes: a smaller part, only 7.5%, is 6_3Li , which however absorbs neutrons forming tritium. The rest, 92.5%, is 7_3Li . To avoid the neutron absorption and the tritium problem, highly enriched 7_3Li is used, even if it is more expensive. Beryllium has the additional advantage of producing more neutrons when hit by α -particles from the transuranics. This can serve also as a neutron source to start the reactor. Beryllium, on the other hand, has the disadvantage that it is highly toxic, but compared to the radiation of fission products in the molten salt, this is rather a minor problem.

The trivalent fluorides PuF_3 , AmF_3 and CmF_3 are unfortunately less soluble in this salt mixture, so that other salt mixtures are currently being investigated. The ideal salt has therefore not yet been found.

The fluorides mentioned hardly react in air and are also poorly soluble in water. In contrast to metallic sodium, they are therefore easy to handle. However, they are corrosive at high temperatures. In the past, reactor vessels were made of Hastelloy N™. This is a high-temperature nickel-based alloy with 7% chromium and 16% molybdenum, which still had sufficient corrosion resistance at temperatures up to 870 °C. However, traces of water can significantly increase corrosion.

For comparison with water or liquid metals, I have listed a few physical properties of such salt mixtures in Table 9.1. They have a better thermal conductivity than water, but liquid metals are still much better. The viscosity is five times higher than that of liquid lead. But since we don't have to pump the molten salt through a narrow fuel assembly, this doesn't matter much. The

Table 9.1 Examples of the physical properties of fluoride mixtures for the molten salt reactor

Reactor	MSRE	MSFR
Composition (mol%)	65% LiF, 29.1% BeF ₂ , 5% ZrF ₄ , 0.9% UF ₄	77.5% LiF, 22.5% ThF ₄
<i>Important material data for core entry and exit conditions:</i>		
Melting temperature (°C) at 1 bar	434	565
Temperature (°C)	650	700
Pressure (bar)	1	1
Density (kg/m ³)	2270	4100
Viscosity (mPa s)	8.1	10.1
Thermal conductivity (W/m K)	1.4	1.0
Specific heat capacity (kJ/kg K)	1.968	1.594

heat capacity, on the other hand, is even better than that of sodium, so that these molten salts are actually good fluids for heat transfer.

History

A molten salt reactor existed already in the 1950s. While people were already starting to equip ships and submarines with nuclear propulsion systems, the idea of powering also an aircraft with a nuclear reactor came up. The application at that time was of course also military, and so unfortunately nothing has been published. Today we find only a few sources that report something about it [49]. The reactor was basically a sphere, as sketched in Fig. 9.1. It was filled with a salt mixture of 50 mol% NaF, 46 mol% ZrF₄ and 4 mol% UF₄. The uranium was highly enriched U-235. In principle, this reactor would also work with low enriched uranium, but as it was, the reactor was very compact and thus not too heavy for an aircraft. The sphere was about 1.4 m in diameter. Inside the sphere and around it were beryllium bodies as moderator, cooled with sodium. So it was a thermal reactor. In the middle of the sphere,

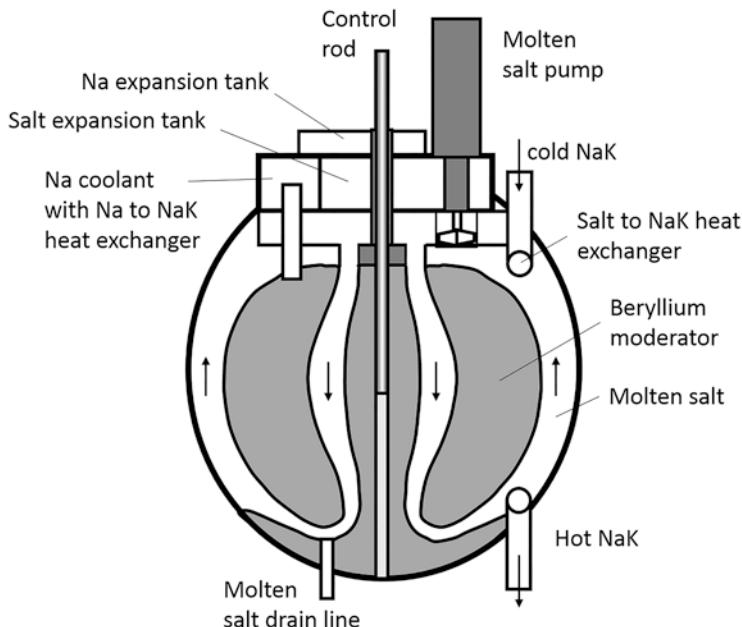


Fig. 9.1 Principle of the molten salt reactor Fireball

a control rod could be retracted. Officially this reactor was called “*Aircraft Reactor Test*”, but it became better known as “Fireball”.

The molten salt was driven by a circulating pump that was inserted from the top of the reactor. Nuclear heat was generated primarily in the central annulus, where the neutron flux was highest. The outer annulus of the reactor contained a heat exchanger that was used to heat liquid NaK. Only the two headers of the heat exchanger are shown in the figure. The eutectic alloy NaK of sodium and potassium is already liquid at room temperature. It was supposed to transport the nuclear heat with a temperature of 871 °C to heat exchangers, which were placed inside the aircraft engines instead of the combustion chamber. The reactor was to reach a thermal output of 60 MW. For emergency shutdown, the salt of this reactor could also simply be drained into sump tanks.

A small prototype of the reactor, or rather a pre-experiment (Aircraft Reactor Experiment, ARE), with a thermal output of 2.5 MW was actually built and is said to have functioned reliably. After all, it has been operated at about 870 °C for more than 100 h. Fortunately, the fission products produced fluorides as well, which were readily soluble, or found in the inert gas above the molten salt. So they did not settle anywhere. However, no aircraft ever flew with the Fireball. The main problem was the shielding of γ -rays. Thick concrete walls or lead shielding were much too heavy for an airplane. However, some basic principles of the Fireball served later as a basis for the design of another molten salt reactor.

The goal of an aircraft reactor was abandoned in 1960. Instead, Oak Ridge National Laboratory (USA) developed a molten salt thermal reactor in the 1960s, which was operated at the Aircraft Reactor Experiment facility, first with enriched U-235 and later with U-233 from thorium [50]. A sustainable thorium fuel cycle is possible with a thermal reactor, as reported in Chap. 5, but otherwise fails because of the high γ -radiation that U-232 brings as a by-product of U-233. It is therefore almost impossible to make a fuel assembly from U-233. However, this is not a general problem with the molten salt reactor: it does not have any fuel assembly.

This Molten Salt Reactor Experiment (MSRE) was a reactor with a thermal power of 8 MW. I have listed the salt composition used and its physical properties in Table 9.1. The reactor core was 1.40 m in diameter and 1.62 m in height. Thus, the power density was similar to the low power density of the high-temperature thermal reactor. Vertical square graphite rods with $5 \times 5 \text{ cm}^2$ cross section served as moderator. All metal parts were made of Hastelloy N™. Control rods could be inserted into the reactor from the top. They were

separated from the molten salt by protective tubes. The molten salt was heated from 632 °C to 654 °C in the core.

The primary circuit of the reactor was of loop design, as indicated in Fig. 9.2. The circulating pump and a heat exchanger were external to the reactor and transferred the generated heat to a secondary circuit, which was operated with a molten salt of 66% LiF and 34% BeF₂. This finally released the generated heat to the ambient air via an air cooler. In the basement below the reactor were two drain tanks, each of which could be cooled by a steam generator. This was the emergency cooling system. The steam generator could even freeze the molten salt. The steam was condensed in a cooler and returned to the steam generators. Furthermore, there was a storage tank in the basement through which fresh molten salt could be filled into the reactor, and a flush tank to remove molten salt. Each of these four tanks had to be much smaller than the reactor, otherwise the contents could have become critical and started to generate heat by themselves.

The entire primary circuit was shielded as in a hot cell. All components could only be reached from above via long arms. It was therefore important to design the plant to be as maintenance-free as possible. Heating wires were

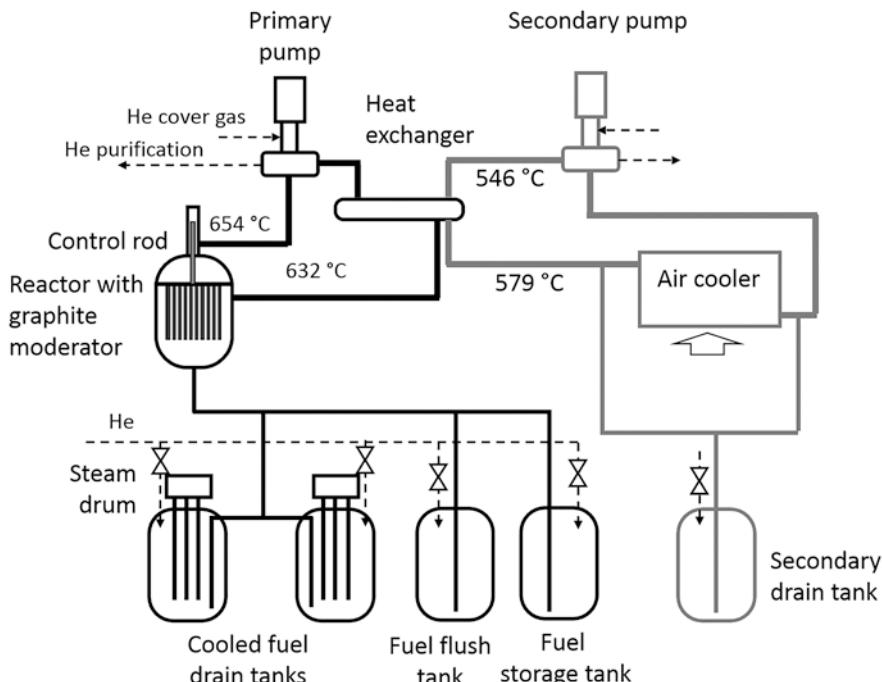


Fig. 9.2 Schematic of the experimental reactor MSRE

wrapped around each of the pipes and tanks to preheat the system before filling and to keep pipes warm in case of stagnant flow. Instead of valves, the salt in the piping could simply be allowed to freeze and thaw out again via the heating wires, i.e. to open them. The drain valve underneath the reactor was simply a flattened piece of pipe that was kept cold with compressed air from the outside. If the compressed air purge failed, or if the reactor overheated, this “valve” opened by itself. So this was the emergency shutdown system. From the tanks in the basement, the molten salt could be pumped back into the reactor by bringing the helium protection gas in these tanks to overpressure.

As with liquid metal-cooled reactors, the drive shafts of the two pumps protruded from the molten salt. Consequently, the pump housing had a helium cushion at the top, which served as an expansion vessel and as a fission gas plenum. There, the pump shaft had to be sealed against the release of fission gas. For this purpose, helium was added to the shaft seal as a sealing gas with some overpressure, which then escaped into the fission gas plenum. At the same time, helium was drawn off there and cleaned via activated carbon filters. In this way, the fission gases xenon and krypton were permanently removed. But also some noble and semi-noble metals such as palladium (Pd), ruthenium (Ru), silver (Ag), molybdenum (Mo), niobium (Nb) and antimony (Sb), which do not form fluorides, could partly be extracted there as colloids. During prolonged operation, they could otherwise settle on the surface of the heat exchanger. To further enhance the fission gas extraction, the pump shaft produced a droplet spray that increased the surface area.

The reactor was in operation for 4 years, from 1965 to 1969. For testing, it was operated first with depleted U-238. Then U-235 was added, which was gradually replaced by U-233 during operation. This came from another reactor; the molten salt reactor itself was never operated with thorium. However, the experiment showed that the high activity of U-233 was not a problem. To extract uranium from the molten salt, fluorine bubbles were allowed to rise in the melt in a separate tank. The uranium is then found as UF_6 in the gas phase. Similar to uranium enrichment, the UF_6 was oxidized then to UO_2 .

The cited field report by Haubenreich and Engel [50] gives a rather positive impression of this reactor. Nevertheless, the experiments were not continued in the 1970s. Worldwide, too, there was little interest in this type of reactor until the end of the last century. Only with the Generation IV initiative in the early 2000s did it become clear that a molten salt reactor with a fast neutron spectrum could actually fulfil the Generation IV criteria better than any other fast reactor.

Fast Molten Salt Reactors

You can get a fast molten salt reactor from a thermal molten salt reactor by simply omitting all graphite and increasing the fissile material content instead. The reactor core is then practically empty inside except for a few absorber rods. These can be retracted into protective tubes. Designing a reactor in this way is quite different from designing a conventional reactor with fuel assemblies. Whereas in the case of solid fuel assemblies it is necessary to ensure that the fuel burns up evenly everywhere, the concentration distribution of liquid fuel mixes by itself. No cladding tube can become brittle or even break due to the neutron radiation. On the other hand, however, care must also be taken in a liquid reactor core to ensure that the temperature is distributed as evenly as possible, because it has a significant influence on the solubility of the salts. Furthermore, the surrounding reflector must be protected from overheating.

Let us first consider neutron physics. The neutron spectrum looks similar to that of a sodium-cooled reactor, but has a somewhat lower energy on average, as the salt causes some absorption and moderation of neutrons. However, unlike the sodium-cooled reactor, a warmer or even boiling molten salt does not only decrease the absorber density, but also the fuel density. This has a much greater effect than the change in density of the absorber or the moderator. The power of the reactor therefore decreases. The molten salt reactor is thus the only fast reactor with a negative void effect. In addition, there is the Doppler effect, which is somewhat stronger than in the sodium-cooled reactor.

Delayed neutrons also exist in the molten salt. However, during the time from the nuclear fission reaction to the decay of the fission products emitting neutrons, the molten salt has moved on. Thus, the delayed neutrons are emitted at a different place than the prompt neutrons. Consequently, one must couple a flow analysis with a neutronic analyses for the reactor design [51]. However, the overall effects on the dynamic behaviour of the reactor remain the same. If the reactor is critical at a certain mean temperature and one changes the heat removal, it returns to this mean temperature by itself due to these reactivity feedback effects. If it gets hotter, the fuel density and consequently the power decreases, and if it gets colder, the power increases. The average temperature can therefore be changed by adding or removing more fissile material or by introducing absorber rods. Thus, if one drains the melt, the reactor is shut down in the same way as if one retracts the absorber rods.

This stable behaviour makes power control of the reactor very simple. In a power plant, one would usually use the secondary circuit to generate steam that drives a steam turbine. If one opens the turbine valve at the inlet of the

steam turbine, the power of the steam turbine increases and at the same time the pressure in the evaporator decreases. This first makes the secondary circuit colder and then the molten salt in the reactor as well. Then, the reactor produces more power until it reaches the set temperature again. Consequently, control rods are actually not needed to control the power output.

The fast molten salt reactor initially requires a fairly large inventory of plutonium to reach criticality. But since it continually replenishes its fuel, it could be designed to replenish as much fissile material as it consumes. Then it could run indefinitely. However, more and more fission products are formed during operation, which absorb neutrons. Some of these fission products are removed via the inert gas and separated in activated carbon filters. On the other hand, however, it is also necessary to constantly remove some melt in order to remove the remaining fission products from it. This amount is replaced by fresh, clean molten salt. The molten salt reactor should therefore be combined with a salt purification process that separates and disposes of the fission products. I will come to this later. The fresh molten salt needs some additional fresh uranium U-238, but the composition of the transuranics can remain unchanged during salt purification. Even americium and curium, with their high radiation, should not cause any problems; the MSRE, after all, has also been operated with the problematic fuel U-233. Since no fuel assemblies need to be made, the molten salt fast reactor can probably recycle all the long-lived waste from the spent fuel assemblies, leaving only fission products that will decay in a few centuries. This sounds fantastic, and it has also motivated researchers to take a closer look at this type of reactor.

Instead of refeeding fresh U-238, one could also refeed the thorium isotope Th-232. Not because a shortage of U-238 is to be expected. The U-238 in the spent fuel assemblies and the depleted uranium from enrichment would be sufficient for several thousand years of electricity supply. But because hardly any transuranium elements are formed when thorium fuel is irradiated. In the long run, americium and curium would be avoided altogether. But that is already thinking very far into the future.

The general safety philosophy of the molten salt fast reactor differs from other fast reactors in one important respect: the fuel is not surrounded by a cladding tube, so there is no such first barrier. Instead, the entire primary system is the first barrier. Consequently, the primary system of reactor, circulating pumps and heat exchangers is heavily contaminated with highly radioactive fission products and transuranics. In a sodium-cooled reactor, for comparison, the sodium is hardly contaminated unless cladding damage occurs. Lead or especially lead-bismuth, on the other hand, contain the radioactive polonium, which also requires careful shielding. Consequently, the

primary system of the molten salt reactor must be shielded as in a hot cell and one cannot repair anything on it by hand. However, experience with the MSRE shows how this problem can be solved: The hot cell forms the gas-tight second barrier. Around it, a containment is needed as a third barrier, which, however, can normally be entered.

The emergency cooling concept in the event of a power failure is very simple here. The molten salt simply flows through the large drain line into several drain tanks, which must be placed far enough apart from each other so that they cannot result in a critical arrangement. The pipes to them, however, must be kept constantly warm for this purpose, or the melt will freeze at the walls before it reaches the drain tank. These drain tanks are then passively cooled, either by steam generators with condensers, or by other passive systems which do not require any power supply or active control. Also, in the event of a leak, perhaps because the reactor or a pipe is corroded, the molten salt is simply drained. The concept of the MSRE was then simply to allow the remaining salt in the primary system to freeze before opening the appropriate flanges of the primary system and remotely changing the affected component. However, such a case never occurred with the MSRE.

In the following, I would like to show two examples of how such a fast molten salt reactor could look like in future. However, there are still no detailed power plant concepts as with the liquid metal cooled reactors.

The MOSART Concept

The *Molten Salt Actinide Recycler and Transmuter* (MOSART) [52], currently being designed in Russia, aims to fission the transuramics of spent fuel from Russian pressurised water reactors without producing new fuel. Figure 9.3 shows that the reactor core consists only of a large empty space. The molten salt is supplied to the core from below at a temperature of 650 °C via nozzles in a distribution plate, which distribute the flow evenly over the cross-section of the reactor core. This avoids hot spots inside the core. The nozzles generate vertical jets in the core, which mix the melt well. On the other hand, the pressure drop across the distribution plate allows the lateral graphite reflector to be cooled as well. Control rods or shutdown rods do not yet appear in the designs to date. Although the reactor is in principle controllable without control rods, I assume, however, that it may not be operated without shutdown rods. The space above the reactor gives enough room to accommodate any control rod drives. The melt leaves the top of the reactor at a temperature of 750 °C and is distributed to the surrounding four pumps and heat exchangers.

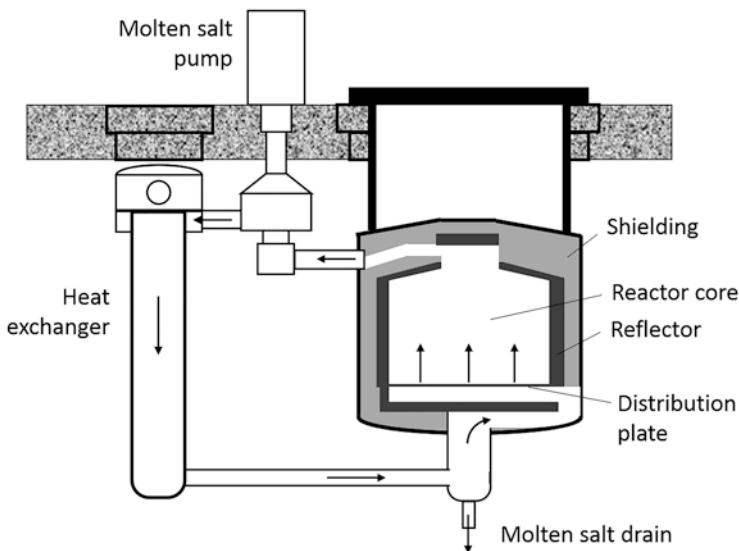


Fig. 9.3 Sketch of the primary system in the MOSART reactor

The reactor core has a diameter of 3.4 m and a height of 3.6 m. The molten salt consists of LiF and BeF₂ and the trifluorides of Np, Pu, Am and Cm. The temperature window of 600 °C to 750 °C results on the one hand from the minimum temperature for the solubility of the salts, so that no salts, in particular no transuramics, are precipitated. On the other hand, the steel for the components of the primary circuit limits the salt temperature to 750 °C. The reactor is to be operated with a thermal power of 2400 MW in order to generate about 1000 MW of electricity from it. This corresponds to an average power density in the core of 74 MW/m³, less than the power density of a pressurized water reactor. The neutron flux is 10¹⁵ neutrons/cm²s, which is comparable to other fast reactors. The dissolved fission products are to be removed over a period of 1–3 years and the salt needs to be replaced accordingly by clean molten salt. The noble gases can be continuously removed via the helium inert gas and the poorly soluble noble metals and semi-noble metals via a filter.

The primary system is filled with about 50 m³ of molten salt, of which about 32 m³ is inside the reactor, the rest outside in the pumps, heat exchangers and connecting piping. Some initial design calculations of the core have been performed with different compositions of transuramics, two scenarios of which are listed in Table 9.2 [53]. In this design, the melt deliberately does not contain uranium and thorium so as not to create a new fissile material. Scenario 1 corresponds to a composition of pressurized water reactor fuel

Table 9.2 Examples of fissile material composition in the MOSART reactor

Transuranic	Scenario 1	Scenario 2
Total	3.5 t	4.5 t
Neptunium	6.42%	6.51%
Plutonium	89.78%	85.65%
Americium	2.88%	7.25%
Curium	0.92%	0.58%

assemblies 1 year after discharge from the reactor and Scenario 2 corresponds to an interim storage of spent fuel assemblies of 10 years. In scenario 2, more transuranics have to be used because the fissile Pu-241 has already partially decayed to the fertile Am-241. For the production of the initial molten salt, the extended PUREX process could be used, for example, in which americium and curium are separated by the DIAMEX and SANEX processes (see Chap. 5). Of these transuranics, the reactor consumes about 500 kg of plutonium and 250 kg of other transuranics per year.

However, a reactor without uranium and thorium has only a very small Doppler effect. The resonance absorptions extend up to 21 keV for U-238, only up to 6 keV for Pu-240, up to 1 keV for Pu-242 and finally only up to 0.16 keV for Am-241. This means that, without U-238, the fast neutron spectrum has only a small fraction of neutrons in the resonance absorption region, which is responsible for the Doppler effect. Therefore, the graphite reflector surrounding the core brings some neutrons to lower energy. However, a liquid metal cooled reactor, which is fully dependent on the Doppler effect because its power increases as the liquid metal gets hotter, could not be operated with this fuel at all. In contrast, the molten salt reactor's power decreases as the molten salt gets hotter, and the MOSART reactor uses this effect to compensate for the small Doppler effect.

The secondary system, and in particular the safety systems of this reactor, would still have to be designed. The concept is based on the experience with the MSRE, so it also provides for cooled drain tanks into which the molten salt can be released in an emergency. Furthermore, facilities for the production and continuous purification of the salt from dissolved fission products are required, but this development is still in an early stage.

The Molten Salt Fast Reactor (MSFR)

When designing a molten salt reactor, one has the choice of destroying or preserving plutonium. So you first have to decide whether to regard plutonium as waste or as a raw material. If a country's goal is to generate its electricity with nuclear energy in the long term, it will tend to produce more plutonium, at least as much as it consumes or even more if necessary, in order to be able to use the abundant U-238 for electricity generation.

The *Molten Salt Fast Reactor* (MSFR) concept seeks a compromise between these two options. If the U-238 is replaced by Th-232, but the transuranium elements from the spent fuel are used as the fissile material, the inventory of plutonium and americium can be reduced by a factor of about 30 in the long term, over a period of more than 100 years [54]. Instead, U-233 is produced from Th-232, which the molten salt reactor can readily utilize, even though it causes high γ -activity, as the MSRE had already shown. Also in this case, plutonium, americium and curium would first have to be separated with the extended PUREX process in order to be able to give the thorium tetrafluoride an initial inventory of transuramics for fission.

The structural design of the MSFR is similarly simple to that of the MOSART reactor. For a reactor with a thermal power of 3000 MW, the design envisages a liquid reactor core of 2.255 m diameter and 2.255 m height, operating at an average temperature of 700 °C with a heat-up of about 100 °C. The molten salt consists of 77.5% LiF and 22.5% thorium or transuranics, thus avoiding the use of beryllium. The melting point of this mixture is thus about 130 °C higher. The most important material data are listed in Table 9.1. Figure 9.4 shows a sketch of this reactor. The 16 pumps and heat exchangers arranged around the reactor core are to be made of nickel-based alloys, as are the reflectors surrounding the core. The compact arrangement is more reminiscent of a pool design. Half of the salt inventory of 18 m³ is inside the core, the rest in the surrounding components. For one circulation through the core, the pumps and the heat exchangers, the molten salt needs just under 4 s.

The toroidal displacement body and the conical bottom and lid of the reactor resulted as a consequence of the initial flow field calculations. Since this reactor design does not provide a distribution plate that could distribute the molten salt over the cross-section, large flow recirculation areas were calculated at the lateral edges. However, the low flow velocities there produced high salt temperatures. Therefore, the contour was improved such that it does not

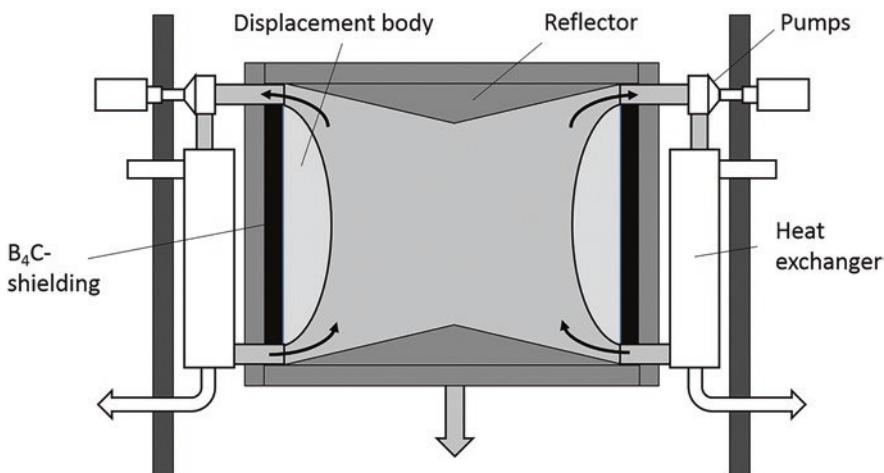


Fig. 9.4 Schematic diagram of the Molten Salt Fast Reactor (MSFR)

show any more flow recirculation areas and the molten salt heats up evenly from the bottom to the top [55].

On the other hand, this displacement body can now also be used as a breeding zone. For this purpose, it is purged with a separate liquid salt circuit with pure Th-232, from which one can continuously separate U-233 formed there. The experiments at MSRE have shown how this works with fluorine gas, which is bubbled through this molten salt.

An open point with both concepts is the salt purification. The noble gases can easily be separated via the inert gas and the noble metals can probably be separated with a filter. However, the dissolved fission products must be removed chemically. In principle, this can be done electrochemically directly from the molten salt. A quite similar process with chlorides has already been developed for the separation of uranium and plutonium from the fission products, which I would like to report on below.

The Pyroprocess

The pyroprocess is an electrochemical separation process in molten salts that separates uranium, transuramics and fission products from spent uranium and MOX fuel, e.g. from pressurized or boiling water reactors [56]. It was originally developed at Argonne National Lab in the USA as a proliferation resistant alternative to the PUREX process, which remained in the military domain in the USA after signing the Nuclear Non-Proliferation Treaty and

thus unfortunately could not be used to produce MOX fuel for power plants. In the meantime, the pyroprocess has also been further developed in Japan and South Korea, while it is hardly known in Europe. The principle is explained in Fig. 9.5.

In the first step, the oxide fuel is removed mechanically from the cladding tubes and, if necessary, further crushed. The second step is an electrochemical reduction. This requires a hot molten salt, about 650 °C, a current source with a few volts DC and three electrodes: a cathode (+), an anode (-) and a reference electrode against which the potential at the cathode and anode can be set. Then the fuel is given into a metallic basket and placed inside the molten salt. During electrochemical reduction, it consists of lithium chloride, LiCl, with a small amount (about 1%) of lithium oxide, Li₂O. The basket with its contents is the cathode of the electrochemical separation process. There, the metal oxides receive the necessary electrons to split into metals, which remain in the basket, and oxygen ions, O²⁻, which migrate through the melt to the platinum anode. The oxygen ions each donate their two electrons back there, so that the oxygen rises as O₂ gas at the anode, while the electrons migrate back to the cathode via a wire. The metals in the basket consist of uranium and transuramics, but also of metallic fission products. The remaining fission products remain in the melt.

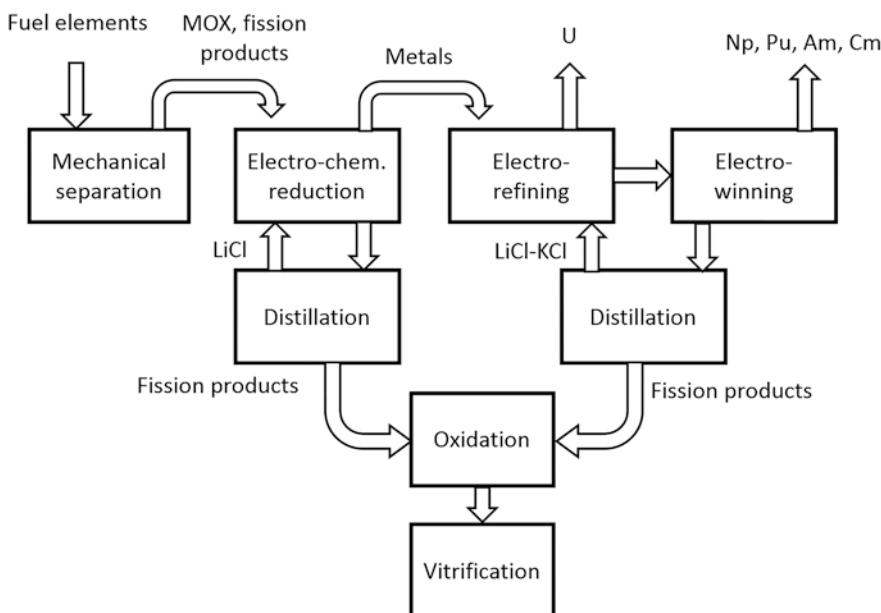


Fig. 9.5 Schematic of the pyroprocess

In the third step, electrorefining, these metals are added to the anode basket of another molten salt, consisting of a eutectic mixture of lithium chloride, LiCl, and potassium chloride, KCl. Electrons are removed from the metals there. As a result, they become positive metal ions, i.e. they dissolve in the molten salt. If the potential is set correctly, the trivalent uranium ions, U^{3+} , settle at the cathode, where they receive the missing electrons to become metallic uranium again. This is also called galvanic deposition. After some time, long crystals of metallic uranium grow at the cathode. The other metals remain in the molten salt.

The fourth step is electrowinning of transuranics. This is another galvanic separation of metals from the remaining molten salt. However, a cathode of liquid cadmium is rather used there, in which the transuranium elements are found then again. A further separation of plutonium, americium and curium is no longer possible. The remaining fission products remain in the salt.

The respective salt is finally recovered by distillation, thus remaining in the cycle. The residue of the distillation is the fission products, which must now be oxidized before they can be vitrified.

Unlike in the PUREX process, the transuranium elements plutonium, neptunium, americium and curium therefore remain inseparable. This has two advantages:

- The vitrified moulds that are filled at the end during vitrification contain only fission products and no transuranics. This reduces the time during which this waste radiates in a repository to less than 500 years.
- The plutonium obtained is definitely unsuitable for nuclear weapons.

On the other hand, this plutonium was hardly suitable for the production of fuel assemblies because it was too radioactive. This would not be a problem for the molten salt reactor.

The first two steps are only required for oxide fuel. They would be needed to process the spent fuel assemblies from the interim storage facility for the molten salt reactor. The third and fourth steps would be required to regularly clean the salt of fission products. It would, of course, make more sense to carry out the electrorefining and electrowinning process already with fluorides, so that it would not be necessary to convert the materials into chlorides first. There have already been initial attempts to do this, but development is still in its infancy.

To develop the pyroprocess, the fission products have so far been replaced by stable isotopes and the transuranics limited to plutonium, so that the whole process could be set up in a large glove box. This did not include

vitrification yet, but this could presumably be carried out in a similar way to the PUREX process. Hot cells would be required for true fission products as well as for americium and curium. But for this to happen, the process must first work reliably enough to be automated.

The pyroprocess is significantly smaller and thus presumably less expensive than the PUREX process. The concepts of the molten salt reactor therefore assume that this plant will be built on the same site as the reactor and its power plant. But we are still a long way from that today. Several more years of research will be required before it is worthwhile to construct the entire plant.

TMSR Development Plans

I think it has become clear that such a molten salt reactor cannot be developed at a desk. It requires a great deal of laboratory work and preliminary testing before a prototype can be built. The Shanghai Institute of Applied Physics (SINAP) has set itself the goal of investing heavily in an appropriate research infrastructure, which undergraduate and graduate students in particular will also be able to use. In cooperation with the Oak Ridge National Laboratory, a research laboratory is currently being built, which will eventually result in a *Thorium Molten Salt Reactor* (TMSR) for thorium fuel [57]. The new laboratory at Jiading has already designed hot cells as well as glove boxes, a molten salt circuit to test the necessary pumps, and initial equipment to develop the pyroprocess such as fluoride fabrication, uranium electrorefining, and salt distillation to purify the fluorides. There are centrifuges for the enrichment of Li-7, which have already reached concentrations of more than 99.99%, a production plant for lithium beryllium fluoride and for lithium sodium potassium fluoride, facilities for the production and processing of Hastelloy N, and materials laboratories for corrosion tests.

In the first step, one would like to use the molten salt only for cooling a reactor. Since graphite has proven to be well compatible with fluoride melts in the MSRE on the one hand, and China can produce graphite spheres with TRISO particles for the high-temperature reactor HTR-PM itself on the other hand, a pebble bed reactor is considered, which is to be cooled with lithium beryllium fluoride. Figure 9.6 shows a cross-section of the proposed reactor TMSR-SF1 [58]. An regular arrangement of spheres in an octagonal prism of 139 cm width and height is cooled vertically upwards by a binary molten salt of 66.7% ^7LiF and 33.3% BeF_2 (mol%). The graphite spheres have a diameter of 6 cm, in which TRISO particles are located in an inner volume of 5 cm diameter. Thus, they are the same spheres as in the HTR-PM

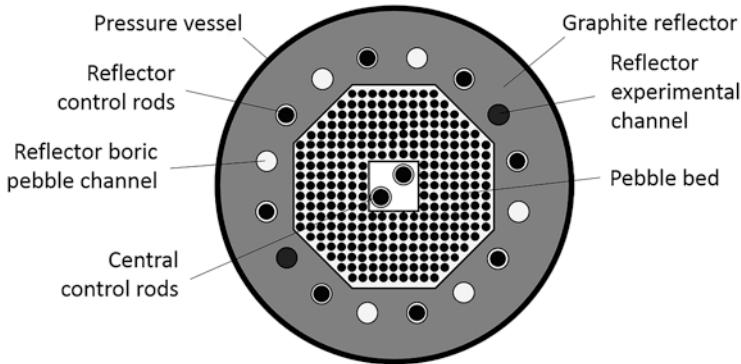


Fig. 9.6 Cross-section through the planned reactor TMSR-SF1

(see Chap. 4). In a first step, a thermal power of 2 MW is to be generated. The molten salt only heats up from 600 °C to 620 °C in this process. In the second step, the heat output is then to be increased to 20 MW, for which purpose the mass flow must be doubled and the heat-up increased to 700 °C. The secondary system shall be operated with a molten salt of lithium-sodium-potassium fluoride.

Consequently, the molten salt in this reactor contains neither fuel nor fission products, so it does not need to be purified. This is less challenging than the molten salt reactors described above, but it is a first step towards becoming familiar with molten salt.

There are already plans to build a real molten salt reactor, the TMSR-LF1. Like the MSRE in the USA, however, it will first be a thermal reactor that is moderated with graphite rods in the same way as the MSRE. By then, however, the pyroprocess for salt purification should be working. These are very ambitious plans and we'll see what comes out. However, they show that China is very serious about this molten salt reactor.

Conclusion

The molten salt reactor is a relatively old concept. It was built early on, but fell into oblivion again when water-cooled thermal reactors had their economic breakthrough. Somehow, nobody could apply this technology anymore and the development was hardly continued. During the Generation IV program, however, it became clear that the molten salt reactor with a fast neutron spectrum had a few fundamental advantages that no other fast reactor type could

offer. I would therefore like to compare this fast reactor in particular with other fast reactors on the basis of the criteria of the Generation IV programme.

Sustainability

In principle, the fast molten salt reactor can not only recycle plutonium sustainably, but it can also destroy americium and curium more easily and safely than any other fast reactor. For this purpose, these elements do not even have to be separated individually, but it is sufficient if they are recycled together with plutonium and neptunium via the pyroprocess. There is also no need to produce a fuel pellet or a fuel assembly from them, which would otherwise be quite an effort given the high activity of these isotopes.

Furthermore, the molten salt reactor is the only reactor type that could provide a sustainable fuel supply of thorium. The high γ -activity that inevitably arises with the U-233 produced is not a problem for this reactor. While some other reactors can produce the fissile material U-233 from thorium as well, they have not yet succeeded in subsequently fabricating a fuel element from it.

Safety

The fast molten salt reactor is the only fast reactor with a negative void effect, i.e. in which a decreasing density of the coolant causes the reactor power to decrease consistently. This makes it very safe, at least in this respect, and even allows the use of fuels that would no longer be controllable in other fast reactors.

In the event of a loss of power or reactor leakage, the molten salt can simply be drained off. The drain tanks can simply be cooled passively. The safety concept is thus much simpler than with any other fast reactor.

In this reactor, however, the cladding tubes are missing as the first barrier against the escape of radioactive substances. Instead, the entire primary system must be considered as the first barrier. In the event of a leak, not only coolant but also fuel and fission products escape. Therefore, the primary system must be housed inside a gas-tight hot cell. This forms then the second barrier, but not yet the containment, which is also required as an enveloping third barrier.

Reliability

The operating experience with the MSRE does indeed suggest that the molten salt reactor is quite reliable. However, it has to be said that a power plant with salt purification, which has only been envisaged so far, has not even been developed yet. It would certainly be premature to talk about its reliability at this stage.

Economic Competitiveness

The same applies to the economic viability. So far, the concepts still look quite simple and one can hardly imagine what is supposed to be expensive about it, but one can only talk about the economic competitiveness once the power plant has been designed so far.

Proliferation Resistance

In combination with the pyroprocess, no fuel will occur in the nuclear fuel cycle of the molten salt fast reactors that could be used to produce nuclear weapons. No other fast reactor has such good proliferation resistance.

That all sounds very positive. The only decisive disadvantage is that there is still a lot to do. As long as the actual problems are not yet known, everything looks quite positive for the time being. However, China has taken up this challenge. While other countries are still hesitating about whether they want to get involved with this concept, China has already made a start.

Glossary

Electrochemical separation	Process for separating metals from melts or solutions by applying an electric field
Electrorefining	Galvanic process to remove certain metals from an alloy
Electrowinning	Galvanic deposition of metals from a molten salt onto an electrode
Nickel-base alloy	A metal alloy that contains mainly nickel and hardly any iron and is highly resistant to corrosion



10

Significance for the Economy and Society

So what is the fourth generation? Just technologies that had already been developed in the past but that never prevailed economically? From a rational point of view, this can certainly be said. Generation IV nuclear reactors are still more the domain of research. There was never any market demand for them. In the market, power plant operators must be customer oriented. When I sign an electricity supply contract as a customer, I take a quick look at my account and choose the cheapest tariff. An industrial customer will care as well that the electricity is delivered reliably, because the consequences of a power outage can sometimes cost him more than the electricity itself. Therefore, the power supplier can only build what produces electricity reliably and cheaply. If he produces electricity in a country that is not afraid of nuclear energy, he chooses a pressurized water reactor. Such a reactor has already been built several hundred times, is offered by several manufacturers in competition and there are decades of experience with it, not only among the manufacturers and operators, but also among the sub-suppliers, experts and authorities. So far, none of the fourth generation reactors could ever compete with the pressurized water reactor in terms of cost and reliability. And I don't expect that to change in the coming years.

One can rather say: the fourth generation of nuclear reactors is a visionary goal with a real background. Much of what has been developed here has already been built in a similar form. So it's not all castles in the air. Some of it has already worked just fine.

Research thrives on such visionary goals. A visionary goal is virtually the prerequisite for getting research funded. If a development leads to a profitable

product in the foreseeable future, it can be paid for by industry, as was the case with the pressurised water reactor. At that time, in the 1960s, it did not need research funding. Instead, the nuclear research centres in Germany preferred to work on high-temperature reactors and fast reactors. That was more visionary. Unfortunately, the consequence is often that research results do not lead to an economic product. If I were to list today which research results later became a market success, even outside nuclear technology, I would stutter. There are not many.

After all, it is a long way from a research result to a commercial product. It is often underestimated when a scientist shows beautiful, vivid pictures of what a new nuclear power plant could look like later. All the components in it do not only have to be designed in every detail so that they can be built. They also have to be tested. Let's take a fuel assembly of an innovative reactor as an example. We first have to test the materials, i.e. the cladding tube material, the fuel pellet and the materials of all the other components of the fuel assembly, over the entire operating life and over all the loads that these materials will later have to endure. This inevitably takes years. Then a manufacturing process for the cladding tubes and the fuel pellets must be qualified, monitored and controlled in such a way that no rejects will be produced later. Only then can these materials be released. In parallel, the design of the fuel assembly must be checked, because it might have design errors. So first a similar fuel assembly is built with an electric heater and cooled in a test rig simulating precisely the reactor conditions. Building a test facility like this for a new coolant is a challenge in itself, because we need several MW of heating and cooling power. So these facilities are not small. Then the fuel assembly has to be tested under accidental conditions, i.e. it has to be overheated in a special test facility so that it oxidizes or melts, because we have avoid the risk of catastrophic consequences.. Then the fuel assembly must be tested with nuclear heating. To do this, a new fuel assembly from a pressurised water reactor, which has so far survived all these tests, can be placed inside an existing reactor and allowed to run for a few years. It can be checked annually then whether the predictions were correct. With a new coolant, this is more difficult. Sometimes you need a new test reactor to create this specific coolant environment. These costly tests are no different with other components such as control rods and their drives, pumps, valves and much more. In total, several thousand engineers are involved in this work for years. In order to keep the development effort and its costs within limits, manufacturers are therefore forced to change as little as possible, and so it is not surprising why only a few new ideas can be realized later.

Not only in research, but also as students, often already as pupils, we follow our visionary goals. This is important because it motivates us to look at things more closely, to question them and to develop our own approaches to solutions. They can sometimes captivate and inspire us for weeks (or nights) on end. A master thesis or doctoral dissertation is more than just a job, a task we do to get a certificate. It is also a great opportunity to get one step closer to our own visionary goal. I didn't get to know the harsh world of manufacturing and market economy until I graduated. Once I understood how to navigate that world, it began to interest me and my own visionary goals took a back seat. However, the basics and methods I had worked out by then for my visionary goals could help me amazingly well.

I therefore believe that it is even the most important task of research to inspire budding scientists and engineers to pursue visionary goals. It is not so much the research result that industry needs later on, but it is these people, the brains, who have the courage to start new things that no one has dared to do before and who have learned the necessary basics. The subsequent innovations of the industry will be based on them. The patents that the research institute had applied for in this context, on the other hand, are rarely used.

But our society needs visionary goals as well, and so do politics. If we were to decide today whether the old, dismantled fuel assemblies should be taken to a final repository, we could only do so on the basis of visionary goals. From a purely economic point of view, it would be better to leave them in an interim storage. That is cheaper. The investment of a final repository cannot be amortized in the foreseeable future. But we have to face the question of whether our descendants will be grateful to us for the gigantic energy resource we have left them with this spent fuel, or whether they will condemn us for the dangerous waste we have left them. There is a 100 times more energy in depleted uranium than we have hitherto used as nuclear energy. But to use that energy, we need plutonium. We don't find that in nature. It's only found in these spent fuel assemblies.

The question is not easy to decide today. But there is also no reason to decide it quickly. Plutonium will be just as easy to recycle in 200 years as it is today. Better, in fact, because the fission products and curium will have largely decayed by then, making recycling much easier. The biggest radiation problems will have disappeared on their own by then. In 200 years, the technology to recycle this fuel and generate electricity from it will be as old as James Watt's steam engine is today. I can't imagine that our descendants will be overwhelmed by it. On the contrary, when I look at the topics the younger generation is working on today, I get the impression that they will do a lot better than us.

The expansion of renewable energies was also based on a visionary objective. In purely market-based terms, we would certainly never have decided to do this. However, visionary goals are eminently arguable. They are superior to every norm and every law, and sometimes even to the laws of nature. It was not my intention to get involved in that dispute with this book. Instead, as a physicist and engineer, I wanted to explain to you factually and in simple terms what lies behind the vision of sustainable nuclear energy. In the end, you will have to judge for yourself.

As a result of such discussions, Germany's commitment to fourth-generation nuclear reactors is negligible today. In other countries, on the other hand, not only are studies being carried out in the Generation IV programme, but the one or other prototype reactor is even being built.

We currently see the greatest commitment in this area in China, even though historically this country had contributed little to the development of such generation IV reactors. In China, a new sodium-cooled reactor went into operation in 2011, the CEFR with 65 MW of thermal power. The reactor is of pool design and runs on MOX fuel. Currently, there are also engineering designs for a large sodium-cooled reactor with 2100 MW of thermal power, but I have not heard of any concrete plans to build it.

A small high-temperature reactor with graphite spheres, the HTR-10, first went into operation at Tsinghua University in 2000 and reached its planned thermal output of 10 MW in 2003. It is considered a prototype for the larger pebble-bed reactor, HTR-PM, with 250 MW of thermal power, which produces steam for a conventional steam cycle that will ultimately generate 105 MW of electrical power. Two of these reactors are expected to drive a common 210 MW steam turbine. This power plant is currently under construction in Shida Bay, China.

Reactors cooled with supercritical water or lead-bismuth are now only being designed in China. Numerous basic experiments are under way, but the construction of a prototype reactor has not yet begun. In contrast, research work is more concrete in the field of molten salt reactors, for the development of which a large research laboratory has been set up at the Shanghai Institute of Applied Physics. A graphite-moderated pebble bed reactor, which is to be cooled with a molten salt, is regarded as a first milestone towards the development of a molten salt reactor.

In Russia, a new sodium-cooled reactor, the BN-800 with 2100 MW of thermal power, came on line in 2014. The BN-800 is a pool-type reactor with three secondary sodium systems feeding a steam cycle that generates 890 MW of electrical power. Also very noteworthy are the engineering designs of lead-cooled reactors in Russia, which are already very mature, such as the SVBR-100

with 100 MW of electrical power, which is to be operated with lead bismuth, and the BREST-OD-300, which should achieve an electrical power of 300 MW with pure lead. The first SVBR-100 is to be built at the Dimitrovgrad research centre in Russia.

The MYRRHA research reactor, which is also cooled with lead bismuth, is to be built in Mol in Belgium in the coming years. This pool reactor can be operated both with a spallation neutron source of a particle accelerator and as a critical reactor. The funds for the construction of the particle accelerator have now been approved as the first stage of expansion. Further research activities in Europe only led to a large number of innovative reactor concepts, which are not built according to the current status, such as the sodium-cooled reactor ASTRID in France, the high-temperature reactor ANTARES, the lead-cooled reactor ELSY, the gas-cooled reactor ALLEGRO or the reactor with supercritical water HPLWR.

The HTTR high-temperature reactor with 30 MW thermal power, which went into operation in Japan in 2001, has a reactor core made of graphite blocks. Today, it serves as a research reactor for the irradiation of materials and novel fuels. However, construction of the sodium-cooled reactor 4S is no longer planned for the time being after the Fukushima accident. The design had already been worked out in some detail by then, and there was also a planned site in Galena in Alaska, but had to be abandoned in the meantime.

Extensive development work on sodium- and lead-cooled reactors can also be found in Korea, as well as process development for the production of TRISO particles for high-temperature reactors. However, there are currently no plans to build a research reactor using these fourth-generation technologies. Spectacular fourth-generation reactor designs are also repeatedly reported in the USA. The company Terrapower demonstrates that private companies are also involved. However, the construction of a prototype has not yet begun in this century either.

Finally, I should mention the development work in India, even though this country is not a member of the Generation IV International Forum. The PFBR sodium-cooled pool reactor with an electrical output of 500 MW is currently being commissioned there. Other development work there includes, for example, high-temperature reactors and molten salt reactors.

If you would like to delve a little deeper into this material, I can recommend the Generation IV Handbook [59]. This handbook is also still written in a way that is understandable for a layman and contains numerous concrete examples of fourth generation reactors.

Glossary

Activity	Decays of a nuclide per second (also: radioactivity)
Atomic number	Number of protons in the atomic nucleus
Binding energy	Energy that is theoretically released when an atomic nucleus is composed of free nucleons $k = 8.617 \cdot 10^{-5} \text{ eV/K}$
Boltzmann constant	
Breeding	Production of fissile plutonium by neutron absorption in U-238
Burnable neutron absorber	An absorber material (e.g. gadolinium) which loses its absorption property through neutron absorption
Burn-up	Thermal energy gained per kg of uranium (or heavy metal in general) by fission
Carbide fuel	A fuel made of uranium and plutonium carbide, which has a higher density and thermal conductivity than MOX fuel
Containment	An enclosed space that prevents the escape of radioactive substances in case of failure of the primary system, e.g. a prestressed concrete containment with an inner steel liner
Cross section	Probability that neutrons cause a certain nuclear reaction, scaled with the neutron flux and the density of the irradiated atomic nuclei

Decay chain	Series of unstable nuclides created by successive decays until a stable nuclide is reached
Decay constant	The constant in the exponential law of radioactive decay
Delayed neutrons	Neutrons emitted by fission products (approx. 200–600 keV)
Density effect	Influence of the moderator temperature on the moderation of neutrons
Depleted uranium	Uranium with a concentration of U-235 of less than 0.72% (N^-)
Diffusion	Substance transport of atoms or molecules by concentration gradients
Doppler effect	Influence of fuel temperature on neutron absorption
Dose coefficient	Conversion factor from activity to radiotoxicity of a nuclide
Electrochemical separation	Process for separating metals from melts or solutions by applying an electric field
Electrolyte	A solid or liquid ion conductor
Electron	Elementary particle of the atomic shell
Electrorefining	Galvanic process to remove certain metals from an alloy
Electrowinning	Galvanic deposition of metals from a molten salt onto an electrode
Enriched uranium	Uranium with a concentration of U-235 exceeding 0.72% (N^+)
Enrichment cascade	Connection of several separation stages in series to increase enrichment
Eutectic alloy	A mixture of two metals in such a way as to give the lowest melting point
eV	Electron volt; $1 \text{ eV} = 1.6019 \cdot 10^{-19} \text{ J}$, energy unit
Feedwater	Water supplied to the evaporator
Ferritic steel	A steel with the crystal structure of iron, consisting predominantly of iron but may contain up to about 18% chromium to improve corrosion resistance
Ferritic-martensitic steel	A ferritic steel with a higher carbon content that can be hardened, transforming its microstructure to martensite

Fertile assemblies	Fuel assemblies which initially contain only U-238 in order to produce fissile plutonium from it
Fertile material	A material which can absorb neutrons forming fissile material, but which cannot be fissioned with thermal neutrons
Fertile zone	Part of the reactor core containing fertile fuel
Fissile material	A material which can be fissioned with thermal neutrons
Fissile zone	Area of the reactor core that contains fissile material, in contrast to the fertile zone
Fission gas plenum	An empty space in the closed cladding tube which can collect gaseous fission products
Fission yield	Probability that a nuclide is produced by nuclear fission
Fissionable material	A material which can be fissioned either with thermal or with fast neutrons
Glove box	A workspace that is hermetically sealed gas-tight
Half-life	Time that elapses until half of an unstable nuclide has decayed
Hot cells	A heavily shielded room with remote handling and lead glass windows for handling highly radioactive materials
Injection evaporator	A device for generating steam by injecting water into superheated steam
Isotope	Nuclides with the same number of protons
Lanthanides	Chemical elements with an atomic number of 57–71
Loop reactor	A reactor with steam generators and pumps outside the reactor vessel
Mass number	Number of nucleons in the atomic nucleus
Minors actinides	Neptunium and chemical elements with an atomic number > 94 (plutonium)
Moderation	Deceleration of neutrons by elastic collisions with light atomic nuclei
Moderator	Material that decelerates neutrons by elastic collisions (e.g. water or graphite)
MOX fuel	A mixture of UO ₂ fuel and PuO ₂ fuel
Multiplication factor	Relative change in the number of neutrons in a reactor per fission

Neutron	Electrically neutral subatomic particle
Neutron flux	Number of neutrons hitting a sample per unit area (cm^2) and second
Nickel-base alloy	A metal alloy that contains mainly nickel and hardly any iron and is highly resistant to corrosion
Nucleon	Subatomic particle (proton or neutron) of a nucleus
Nuclide	Class of atoms characterized by their number of protons, their number of neutrons, and their nuclear energy state
Pebble bed reactor	A reactor having a core consisting of a bed of graphite spheres
Photon	Electromagnetic radiation quantum that is produced during electron or nucleon transitions to another quantum state
Pool reactor	A reactor with steam generators and pumps inside the reactor vessel
Pressure suppression pool	A space in the containment that is partially filled with cold water. If steam is generated in the containment, it can condense there so that the pressure increase in the containment remains limited
Pressurizer	A vessel filled partly with water and partly with steam, both at boiling temperature, to control the pressure by temperature
Proliferation	Misuse of nuclear fuel for the production of nuclear weapons
Proliferation-resistant	The property of a fuel or process that it cannot be used directly, but only with great effort, to produce nuclear weapons
Prompt neutrons	Fast neutrons produced during nuclear fission (approx. 1–10 MeV)
Proton	Positively charged subatomic particle
Pseudo-boiling	The continuous transition of liquid water into steam without bubbles or droplets; only occurs with supercritical water
Pyrographite	A special crystal structure of graphite with strongly anisotropic material properties
Quantum state	Energy state of a nucleon or electron in quantum theory

Radiotoxicity	A measure of how harmful a nuclide is to health, e.g. when ingested through food or drinking water
Reactivity	Relative deviation of the multiplication factor from the number 1
Regeneration factor	Number of neutrons emitted per absorbed neutron during nuclear fission
Reheater	Superheater of the intermediate pressure steam, i.e. after the steam has left the high pressure turbine and before it enters the intermediate pressure or low pressure turbine
Resonance absorption	Absorption of a neutron in an atomic nucleus, when the neutron energy is just sufficient to lift the neutron to a certain quantum state
Resonance scattering	An elastic collision of a neutron with an atomic nucleus in which the latter is excited to a higher energy state
Saturated steam	Steam at boiling temperature
Separation factor	Ratio of enriched to depleted U-235 concentration of a separation step $[N+(1 - N-)]/[N-(1 - N+)]$
Spallation	Shattering of an atomic nucleus by bombardment with high-energy neutrons, producing multiple fragments and multiple neutrons
Specific heat capacity	Ratio between the heat supplied and the temperature increase caused by it
Steam dryer	Zigzag-shaped plates which separate further droplets from the steam by centrifugal force
Steam separator	Vertical tubes with a swirl generator at the inlet, separating droplets by centrifugal force
Supercritical water	Water or steam at a pressure greater than the critical pressure (220.64 bar)
Superheated steam	Steam with a temperature higher than the boiling temperature
Superheated steam	Steam with a temperature higher than the boiling temperature
Target	The “target” of a particle beam, i.e. the object that is to be bombarded
Thermal conductivity	Ratio between heat flow and temperature gradient of a substance

Thermal neutrons	Neutrons which, like gas atoms, are kept in motion only by temperature (approx. 25 meV at room temperature)
Thermal reactor	Nuclear reactor in which the prompt neutrons are slowed down to thermal neutrons to fission more effectively
Transuranics	Chemical elements with an atomic number >92 (uranium)
TRISO particle	A spherical fuel element coated with three layers of pyrographite and SiC
Void effect	Influence of the vapour bubbles of the coolant on the neutron moderation
Xenon effect	Influence of the concentration of Xe-135 on neutron absorption

References

1. A Technology Roadmap for Generation IV Nuclear Energy Systems, issued by the U. S. DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, Dec. 2002. https://www.gen-4.org/gif/jcms/c_40473/a-technology-roadmap-for-generation-iv-nuclear-energy-systems
2. Technology Roadmap Update for Generation IV Nuclear Energy Systems, Issued by the OECD Nuclear Energy Agency for the Generation IV International Forum, Jan 2014. <https://www.gen-4.org/gif/upload/docs/application/pdf/2014-03/gif-tru2014.pdf>
3. Nuclear Data Center at KAERI, Table of Nuclides. <http://atom.kaeri.re.kr/>
4. Malko MV (2002) The Chernobyl reactor: design features and reasons for accident. Semantic Scholar, Corpus ID 1490526
5. United Nations Scientific Committee on the Effect of Atomic Radiation. The Chernobyl accident, UNCEAR's assessment of the radiation effects. <http://www.unscear.org/unscear/en/chernobyl.html>
6. Uranium (2016) Resources, production and demand, a joint report by the nuclear energy agency and the international atomic energy agency. <https://www.oecd-nea.org/ndd/pubs/2016/7301-uranium-2016.pdf>
7. Lamarsh JR, Baratta AJ (2001) Introduction to nuclear engineering. Prentice Hall, Upper Saddle River, ISBN: 0-201-82498-1
8. Schulenberg T, Starflinger J (2012) High performance light water reactor, design and analyses. KIT Scientific Publishing, Karlsruhe. ISBN: 978-3-86644-817-9
9. Oka Y, Koshizuka S, Ishiwatari Y, Yamaji A (2010) Super light water reactors and super fast reactors. Springer, Boston. ISBN: 978-1-4419-6034-4

10. GE Hitachi Nuclear Energy. The ESBWR General Description. https://nucleargepower.com/content/dam/gepower-nuclear/global/en_US/documents/ESBWR_General%20Description%20Book.pdf
11. Stosic ZV, Brettschuh W, Stoll U (2008) Boiling water reactor with innovative safety concept: the generation III + SWR-1000. Nucl Eng Des 238(8):1863–1901
12. HDR – Heißdampfreaktor Großwelzheim, Atomwirtschaft – Atomtechnik, XIV(11): S 526–556, Nov. 1969
13. Yetisir M, Gaudet M, Pencer J, McDonald M, Rhodes D, Hamilton H, Leung L (2016) Canadian supercritical water-cooled reactor core concept and safety features. CNL Nucl Rev 5(2):189–202
14. Habush HL, Harris AM (1968) 330 MW(e) Fort St. Vrain high temperature gas-cooled reactor. Nucl Eng Des 7(4):312–321
15. Shiozawa S, Fujikawa S, Iyoku T, Kunitomi K, Tachibana Y (2004) Overview of HTTR design features. Nucl Eng Des 233(1–3):11–21
16. Bäumer R, Kaliniwski I, Röhler E, Schöning J, Wachholz W (1990) Construction and operating experience with the 300-MW THTR nuclear power plant. Nucl Eng Des 121(2):155–166
17. Wu Z, Lin D, Zhong D (2002) The design features of the HTR-10. Nucl Eng Des 218:25–32
18. Zhang Z, Dong Y, Li F, Zhang ZM, Wang H, Huang X, Li H, Liu B, Wu X, Wang H, Diao X, Zhang H, Wang J (2016) The Shandong Shidao Bay 200 MW_e high-temperature gas-cooled reactor pebble-bed module (HTR-PM) demonstration power plant: an engineering and technical innovation. Engineering 2:112–118
19. Yan X, Kunitomi K, Nakada K, Shiozawa S (2003) GTHTTR300 design and development. Nucl Eng Des 222:247–262
20. Koster A, Matzner HD, Nichols DR (2003) PBMR design for the future. Nucl Eng Des 222:231–245
21. Baxi B, Perez E, Shenoy A (2006) Evolution of the power conversion unit design of the GT-MHR. In: Proceedings of ICAPP ‘06. Reno, NV, June 4–8, 2006, Paper 6213
22. Copsey B, Gauthier JC, Lecomte M (2005) The framatome ANP high temperature reactor concept. In: Proceedings of ICAPP 05. Seoul, Korea, May 15–19, 2005
23. Moormann R (2008) A safety re-evaluation of the AVR pebble bed reactor operation and its consequences for future HTR concepts, Berichte des Forschungszentrums Jülich, Jül-4275
24. Takei M, Kosugiyama S, Mouri T, Katanishi S, Kunitomi K (2006) Economical evaluation on gas turbine high temperature reactor GTHTTR300. Trans AESJ 5(2):109–117
25. ICRP Publication 72 (1996) Age-dependent doses to the members of the public from intake of radionuclides part 5, compilation of ingestion and inhalation coefficients. SAGE Publications Ltd

26. Simpson MF, Law JD (2013) Reprocessing of nuclear fuel. In: Tsoulfaidis N (ed) Nuclear energy. Springer, pp 153–173. ISBN: 978-1-4614-5716-9
27. International Atomic Energy Agency (2008) Spent fuel reprocessing options, IAEA-TECDOC-1587. https://www-pub.iaea.org/MTCD/Publications/PDF/TE_1587_web.pdf
28. International Atomic Energy Agency (2006) Fast reactor data base, 2006 update, IAEA-TECDOC-1531. <https://www.iaea.org/publications/7581/fast-reactor-database-2006-update>
29. Katoh A, Hayafune H, Kotake S (2014) Design features and cost reduction potential of JSFR. Nucl Eng Des 280:586–597
30. Sauvage J-F (2004) Phénix, 30 years of history: the heart of a reactor. CEA Valrhô, France
31. Tsuboi Y, Kasuga S, Sakashita Y, Matsumiya H, Hasegawa K, Ueda N, Grenci T, Handa N (2009) development of the 4s and related technologies (1), plant system overview and current status. In: Proceedings of ICAPP '09, Tokyo, Japan, May 10–14, 2009, Paper 9214
32. International Atomic Energy Agency (2018) Advances in small modular reactor technology developments, a supplement to: IAEA advanced reactors information system (ARIS). https://aris.iaea.org/Publications/SMR-Book_2018.pdf
33. Nuclear Energy Agency (2007) Handbook on Lead-bismuth eutectic alloy and lead properties, materials compatibility, thermal-hydraulics and technologies, 2007 Edition. <https://www.oecd-nea.org/science/reports/2007/nea6195-handbook.html>
34. International Atomic Energy Agency (2002) Comparative assessment of thermo-physical and thermohydraulic characteristics of lead, lead-bismuth and sodium coolants for fast reactors. IAEA-TECDOC-1289
35. Zrodnikov AV, Chitaykin VI, Gromov BF, Grigoryv OG, Dedoul AV, Toshinsky GI. Use of Russian technology of ship reactors with lead-bismuth in nuclear power. https://inis.iaea.org/collection/NCLCollectionStore_Public/31/058/31058476.pdf
36. Zrodnikov V, Toshinsky G, Komlev OG, Stepanov VS, Klimov NN (2011) SVBR-100 module-type fast reactor of the IV generation for regional power industry. J Nucl Mater 415(3):237–244
37. Glazov G, Leonov VN, Orlov VV, Sila-Novitskii AG, Smirnov VS, Filn AI, Tsikunov VS (2007) BREST reactor and plant-site nuclear fuel cycle. At Energ 103(1):501–508
38. Dragunov YG, Lemekhov VV, Moiseyev AV, Smirnov VS, Yarmolenko OA, Vasyukhno VP, Cherenpnin YS, Afremov DA, Lemekhov YV (2017) BREST-OD-300 reactor facility, development stages and justification, IAEA-CN-245-539. In: International conference on fast reactors and related fuel cycles, Yekaterinburg, Russia, June 26–29, 2017

39. Latge C et al (2016) MEGAPIE: the world's first high-power liquid metal spallation neutron source. In: Revol JP, Bourquin M, Kadi Y, Lillestol E, de Mestral JC, Samec K (eds) Thorium energy for the world. Springer, Cham
40. Engelen J, Ait Abderrahim H, Baeten P, De Bryn D, Leysen P (2015) MYRRHA: preliminary front-end engineering design. *Int J Hydrogen Energ* 40:15137–15147
41. Fernandez R, De Bruyn D, Baeten P, Ait Abderrahim H (2017) The evolution of the primary system design of the MYRRHA facility, IAEA-CN-245-358. In: International conference on fast reactors and related fuel cycles, Yekaterinburg, Russia, June 26–29, 2017
42. van Rooijen WFG (2009) Gas-cooled fast reactor: a historical overview and future outlook. *Sci Tech Nucl Installations* 965757(2009):1–11
43. Dalle-Donne M et al (1971) Gasbrüter-Memorandum. Kernforschungszentrum Karlsruhe, KFK-1375
44. Malo JY et al (2008) Gas cooled fast reactor 2400 MW_{th}, end of the preliminary viability phase. In: Proceedings of the ICAPP '08, Anaheim, CA, USA, June 8–12, 2008, Paper 8175
45. Poette C, Brun-Magaud V, Morin F, Pignatell J-F, Stainsby R, Mikityuk K (2009) ALLEGRO: the European gas fast reactor demonstrator. In: Proceedings of the 17th International Conference on Nuclear Engineering (ICON17), Brussels, Belgium, July 12–16, 2009
46. International Atomic energy Agency (2008) Thermophysical properties of materials for nuclear engineering: a tutorial and collection of data. ISBN 978–92–0–106508–7, IAEA
47. Müller A et al (1966) Referenzstudie für den 1000 MWe dampfgekühlten schnellen Brutreaktor (D1). Kernforschungszentrum Karlsruhe, KFK-392
48. Kiehaber E (1967) Reactivity coefficients of steam-cooled fast breeders. Kernforschungszentrum Karlsruhe, KFK-0629
49. <https://energyfromthorium.com/2006/04/22/a-brief-history-of-the-liquid-fluoride-reactor/>
50. Haubenreich PN, Engel JR (1970) Experience with the molten-salt reactor experiment. *Nucl Appl Tech* 8:118–136
51. Aufiero M et al (2014) Calculating the effective delayed neutron fraction in the molten salt fast reactor: analytical, deterministic and Monte Carlo approaches. *Ann Nucl Energ* 65:78–90
52. Maschek W et al (2008) Report on intermediate results of the IAEA CRP on Studies of advanced reactor technology options for effective incineration of radioactive waste. *Energy Convers Manag* 49(7):1810–1819
53. Ignatiev VV, Feynberg OS, Khaperskaya AV, Kryukov OV (2019) Molten salt reactor for the management of minor actinides from VVER 1000/12000 used fuel. In: IAEA International conference on the management of spent fuel from nuclear power reactors, CN-272, Vienna, June 24–28, 2019

54. Heuer D, Merle-Lucotte E, Allibert M, Brovchenko M, Ghetta V, Rubiolo P (2014) Towards the thorium fuel cycle with molten salt fast reactors. *Ann Nucl Energ* 64:421–429
55. Aufiero M, Cammi A, Geoffroy O, Losa M, Luzzi L, Ricotta ME, Rouch H (2014) Development of an openFOAM model for the molten salt fast reactor transient analysis. *Chem Eng Sci* 111:390–401
56. Simpson MF (2018) Fundamentals of spent fuel pyroprocessing. In: Zhang J (ed) Nuclear fuel, reprocessing and waste management. World Scientific, Boston
57. Hongjie Xu, Status and perspective of TMSR in China. https://www.gen-4.org/gif/upload/docs/application/pdf/2017-05/03_hongjie_xu_china.pdf
58. Xiao Y, Hu L-W, Forsberg C, Qiu S, Su G (2013) Licensing considerations of a fluoride salt cooled high temperature test reactor. In: Proceedings of the 2013 International Conference on Nuclear Engineering ICONE21, Chengdu, China, July 29–Aug 2, 2013
59. Pioro L (Ed.) (2016) Handbook of generation IV nuclear reactors. Woodhead Publishing Series in Energy: Number 103, Elsevier