OPERATIONAL MATURITY OF GEN IV NUCLEAR REACTOR DESINGS

Lappeenranta–Lahti University of Technology LUT

Master’s programme in Nuclear Engineering

2025

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ABSTRACT

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LUT School of Energy Systems

Energy Technology

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Operational Maturity of GEN IV Nuclear Reactor Designs

Master's thesis

2025

98 pages, 19 figures, 20 tables and 5 appendices

Examiners: Professor, D.Sc. (Tech.) Juhani Hyvärinen and D.Sc. (Tech.) Juhani Vihavainen

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Keywords: GEN IV, Nuclear power, Operability, Operating Experience, VHTR, SFR, LFR, MSR, SCWR, GFR

As multiple new nuclear projects are being commenced globally, interest in the fourth generation of nuclear reactors is growing. These innovative designs bring major conceptual changes to the existing practices and promise benefits such as increased efficiency, improved suitability for industrial cogeneration applications and in some instances, the possibility of net fissile material generation through fuel breeding. The GEN IV reactors, however, are currently not perceived as commercially mature, facing multiple design challenges that are not yet demonstrated as solved. These reactor concepts have existed for a long time, but the notable challenges compared to light-water reactors, such as with fuels and structural materials historically set them as less preferred and developed options.

This thesis conducted for Fortum's Nuclear R&D focuses on the operability-related issues of the GEN IV reactors. The main phenomena and the technological readiness to their solutions are assessed using the phenomena identification and ranking table (PIRT) method. This method is used to identify the phenomena that have the most significant impact to plant operability and show the least technological maturity. This information can be used as a guide to assess individual reactor designs, or specific vendor design choices from the viewpoint of plant operability. In addition, an overview of the current state and timeline of GEN IV reactor development is presented through an assessment of current vendors and ongoing projects.

TIIVISTELMÄ

Lappeenrannan–Lahden teknillinen yliopisto LUT

LUTin energiajärjestelmien tiedekunta

Energiatekniikka

Pyry-Pekka Koistinen

Neljännen sukupolven ydinreaktoreiden käyttötoiminnan kypsyys

Energiatekniikan diplomityö

2025

98 sivua, 19 kuvaa, 20 taulukkoa ja 5 liitettä

Tarkastajat: Professori, TkT Juhani Hyvärinen ja TkT Juhani Vihavainen

Ohjaaja: DI Konsta Värri

Avainsanat: GEN IV, Ydinvoima, Käytettävyys, Käyttökokemus, VHTR, SFR, LFR, MSR, SCWR, GFR

Useita uusia ydinvoimaprojekteja on alkamassa ympäri maailmaa. Näiden ohella myös kiinnostus neljännen sukupolven ydinreaktoreita kohtaan on kasvamassa. Tämänkaltaiset innovatiiviset reaktorimallit tuovat merkittäviä muutoksia perinteisiin laitostyyppeihin nähden ja lupaavat etuja, kuten korkeampia prosessihyötysuhteita, parempaa soveltuvuutta teollisuuden yhteistuotantoon ja joissain tapauksissa mahdollisuuden nettopositiiviseen fissiilin materiaalin tuotantoon hyötämällä. Neljännen sukupolven ydinreaktorit eivät ole kuitenkaan vielä riittävän kypsiä kaupalliseen käyttöön. Ennen tätä on ratkaistava lukuisia suunnitteluhaasteita, joiden ratkaisuja ei ole vielä pystytty osoittamaan toimiviksi. GEN IV reaktorimallit ovat olleet olemassa jo kauan aikaa, mutta ne eivät ole toistaiseksi saaneet kevytvesireaktorien kaltaista kaupallista asemaa.

Tämä diplomityö on tehty toimeksiantona Fortumin Nuclear R&D:lle ja se keskittyy GEN IV-laitosten käytettävyyden ongelmiin. Tärkeimpiä käytettävyyteen liittyviä ilmiöitä ja niiden ratkaisuja arvioitiin phenomena identification and ranking table (PIRT)-menetelmällä. Tätä käytettiin tunnistamaan ja luokittelemaan ne ilmiöt, jotka ovat laitosten käytettävyyden kannalta tärkeimmät ja tällä hetkellä vähiten ymmärretyt. Tätä tietoa voidaan käyttää esimerkiksi yksittäisten reaktorien tai laitostoimittajien arviointiin. Lisäksi yleiskatsaus GEN IV laitosten kehityksen tilasta ja aikataulusta tehtiin tällä hetkellä meneillä olevien projektien ja nykyisten laitostoimittajien arvioinnilla.

ACKNOWLEDGEMENTS

I would like to thank my colleagues from Fortum, especially my supervisor Konsta Värri and my manager Eero Vesaoja, for their support and helpful comments on this thesis.

I would also like to express my gratitude to the nuclear engineering department of LUT University. My master’s studies there were truly inspiring and exciting. Warm thanks to my examiners, Juhani Hyvärinen and Juhani Vihavainen.

Finally, I would like to thank the many people I have had the pleasure of working with, studying with, and spending time with at the university and during the many summer internships over the years. Special thanks to my family and to Eeva for continuous daily support.

Loviisa, November 2025

Pyry-Pekka Koistinen

SYMBOLS AND ABBREVIATIONS

Roman characters

EG	net energy generated	Wh
EPL	unplanned extensions to outages loss	Wh
FEL	forced energy loss	Wh
FLR	forced loss rate	%
LF	load factor	%
OF	operation factor	%
P	power	W
p	pressure	bar, Pa
PEL	planned energy loss	W
T	temperature	K, °C
t	time	s
UA7	unplanned automatic scrams per 7000 hours critical	
UCF	unit capability factor	%
UCL	unplanned capability loss factor	%
UEL	unplanned energy loss	Wh

Abbreviations

AGR	Advanced Gas-Cooled Reactor
AOO	Anticipated Operational Occurrence
ARIS	Advanced Reactor Information System
ASME	American Society of Mechanical Engineers
CANDU	Canadian Deuterium-Uranium Reactor

COT	Core Outlet Temperature
DOE	U.S. Department of Energy
ETA	Event Tree Analysis
FMEA	Failure Modes and Effects Analysis
FOAK	First of a Kind
FTA	Fault Tree Analysis
GFR	Gas-Cooled Fast Reactor
GIF	Generation IV International Forum
HALEU	High-Assay Low-Enriched Uranium
HTR	High-Temperature Reactor
IAEA	International Atomic Energy Agency
IEEE	Institute of Electrical and Electronics Engineers
IHX	Intermediate Heat Exchanger
LBE	Lead-Bismuth Eutectic
LFR	Lead-Cooled Fast Reactor
LOCA	Loss-Of-Coolant Accident
LWR	Light-Water Reactor
MOX	Mixed Oxide
MSR	Molten Salt Reactor
NOAK	N:th of a Kind
NRC	US. Nuclear Regulatory Commission
OLC	Operating Limits and Conditions
PHA	Process Hazard Analysis
PIRT	Phenomena Identification and Ranking Table

PRA	Probabilistic Risk Assessment
PRIS	Power Reactor Information System
PWR	Pressurized Water Reactor
SCWR	Supercritical Water Reactor
SFR	Sodium-Cooled Fast Reactor
SMR	Small Modular Reactor
STUK	Finnish Radiation and Nuclear Safety Authority
TRISO	Tri-Structural Isotropic Fuel
TRL	Technological Readiness Level
VHTR	Very-High Temperature Reactor
WANO	World Association of Nuclear Operators
WNA	World Nuclear Association

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Appendix 5. Identified major initiating events and corresponding safety systems.

DECLARATIONS

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The originality of this thesis has been reviewed with the Turnitin similarity checking service.

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The author, Pyry-Pekka Koistinen, takes full responsibility for the content of this thesis and has reviewed and edited the content generated by the possible use of AI tools.

1 Introduction

The nuclear industry is currently facing a period of globally high acceptance and favour. The benefit of stable and greenhouse gas-free production have led to ambitious goals towards developing the industry. This includes new nuclear projects and setting the right conditions for long-term operation of existing plants. In the 2023 United Nations Climate Change Conference (COP28) a goal of tripling global nuclear capacity by 2050 was set, and by 2025, a total of 31 countries and 140 nuclear industry companies have pledged to support this. (WNA, 2025)

Currently, most of the new nuclear projects in the world are focused on the current reactor designs that are licenced, constructed and operated. Additionally, major interest is shown in evolutionary designs needing only confirmatory testing or engineering. These are sometimes classified as GEN III and GEN III+ nuclear reactors, that in general resemble the previous generation light-water reactors (LWRs). In the background, there is a growing interest in the GEN IV reactors, or so-called innovative designs, bringing major conceptual changes to the existing practice.

The GEN IV designs promise notable benefits, such as increased thermal efficiencies and improved suitability for industrial cogeneration applications. In some instances, the net-positive generation of fissile material through fuel breeding is also possible. These reactor concepts have existed for a long time; most being developed simultaneously with the first-generation light-water reactors. Compared to the "simpler" reactors, the notable challenges, such as with fuels and structural materials, historically set them unfavoured.

Many historical predecessors of the GEN IV reactors, operated as experimental and demonstrator units, suffered from poor operating performance. While the comparison of these with the currently operating and highly performing reactors is not completely equitable, the demonstration of safe and reliable operation is needed before any utility would consider employing them. This master’s thesis is produced for the Finnish energy company Fortum's Nuclear R&D. The aim of the thesis is to find out what are the main operability-related issues to GEN IV reactors and what is the technological readiness to their solutions. In addition, an overview of the current state of development with ongoing projects and their timelines is conducted. This thesis is conducted as a preliminary study to understand the current state, and possibilities of future development of GEN IV reactors.

With the focus on novel reactor concepts, first a brief overview of the six GEN IV reactor concepts and their technical specifications is presented. After this, an overview of reactor vendors and the currently ongoing projects varying states of development are discussed. The vendor-estimated development and deployment timelines are presented along with financial estimations based on past projects. The main nuclear power performance indicators with their global trends is also introduced.

1.1 Methods and aims

The method selected for assessing the designs in this thesis is an applied phenomena identification and ranking table (PIRT). The method is used to identify the most important operability-related phenomena of the selected designs and then ranking the phenomena based on their defined importance and knowledge levels. The base of knowledge for the method comes from two main sources. These are the historical operational experience data from the current and past reactor operators and the IAEA database information on historical production values and performance indicators. This study is conducted on the four most developed GEN IV designs: the very-high temperature reactor, the sodium-cooled fast reactor, the lead-cooled fast reactor and the molten salt reactor.

The PIRT studies completed individually for the four reactor designs allow comparison between them. The amount of identified issues, along with their severity and the state of knowledge can be compared to see which designs are closer to deployment and which need further development. The resulting tables for each reactor can be used to assess individual reactor designs, or vendor design choices. Secondarily, the results of the study can be used as an input for further GEN IV reliability or safety issues.

As new research is continuously being conducted on the issues with the GEN IV reactors, the assessment of operability issues done in this thesis can be later be updated if necessary. Issues identified from the historical predecessors might be less important, or completely avoided in contemporary designs. Inversely, new issues might also be identified as new experimental reactors begin operating.

2 Generation IV reactors

When assessing the evolution of nuclear reactors, the designs are typically classified into "generations." These are the generations I, II, III, III+ and IV. These four generations of nuclear reactor designs are displayed in a timeline in Figure 1.

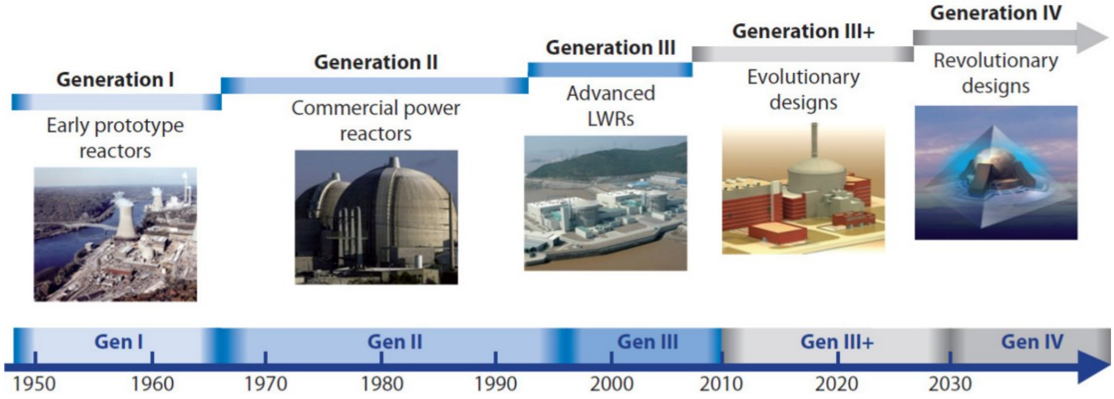

Figure 1. Four generations of reactor designs (GIF, 2018).

The first generation (1950-1970) consisted of the first early power and prototype reactors, such as the reactors of Shippingport and Dresden-1. Currently, no more GEN I reactors are operating, and the majority have undergone decommissioning. The second generation (1970-1990) is the matured commercial, economical and reliable generation of nuclear reactors, which makes up the majority of all currently operating reactors. GEN II is dominated by light-water reactors, namely the pressurized water reactor and the boiling water reactor. The typical design lifetime of GEN II reactors was 40 years; this has been or is being exceeded in many cases with technical and economic-driven long-term operations. (Goldberg, 2011)

The third generation (1990-2000) reactors improve on the second generation by implementing newer technologies on the areas of fuel design, thermodynamic efficiency, standardization, modularization and nuclear safety. Safety systems and their design are the main difference between GEN III and GEN III+ as well. GEN III+ (2000-2030), or "evolutionary" reactors, aim to further improve the safety and operability. Some of the latest designs utilize passive safety systems, which do not depend on electricity, but are driven by gravity and natural convection. In addition, most of the more technologically ready small modular reactors (SMRs) belong to this category rather than in the GEN IV. (Goldberg, 2011)

The fourth generation (2030-), or the “innovative” designs, offer a next potential development step in nuclear reactor design. GEN IV explores multiple vastly different reactor designs compared to the current LWRs. They claim to further improve on the areas of efficiency, safety and economic feasibility while introducing the sustainability and efficiency of material use to the scope. (Goldberg, 2011)

2.1 Definitions and goals of the GEN IV development

The current development of the fourth generation of nuclear reactors is coordinated by the generation IV international forum (GIF), founded by the U.S. department of energy in 2001. Currently, GIF consists of 13 countries* as well as EURATOM. As the international forum was formed, some general goals were selected for the research and development. These goals are sustainability, safety and reliability, economics, and proliferation resistance & physical protection. The goals for the reactor designs were set on the basis of developing nuclear energy forward and possibly achieving benefits that had not existed in the past. (GIF, 2014)

The first goal set by GIF, sustainability, was set on the claim that utilizing GEN IV technology would improve the efficient use of natural resources and reduce the amount of produced waste. This is set on the assumption that some of the reactor designs would be able to either breed fissile material or consume generated waste, both requiring that a closed fuel cycle could be adopted. The claimed increase in efficiency is also linked to this goal. The more advanced coolants and fuels would allow higher core coolant outlet temperatures and better power conversion efficiencies. (Dunlap, 2025)

The fourth generation aims to reach the level and exceed the degree of safety and reliability of the third generation. This is aimed to be achieved through the possible utilization of passive safety systems, coupled together with the intrinsic design safety features of the reactor designs. The newer designs, however, have their own specific challenges compared with LWRs. The focus of reactor design is planned to be coordinated so that the core damage frequency would be minimal. The reduction of risks such as operator error and the reliance on off-site emergency response are acknowledged as well. (Dunlap, 2025)

* Argentina, Australia, Brazil, Canada, China, France, Japan, Korea, Russia, South Africa, Switzerland, the United Kingdom and the United States.

Russia is not included in the new GIF framework agreement effective on 1st March 2025.

Economic competitiveness and financial risk management are a critical factor in the commercialization of new nuclear reactors. GIF claim that there is a potential to develop a system that has a clear advantage in the plant life cycle and production costs over other energy sources. This would be achievable through higher process efficiencies, utilization of modular designs, and if applicable, the high efficiency of the fuel cycle. In addition, the financial risks associated with new nuclear builds need to be reasonable, and preferably on the same level as with other energy projects. (GIF, 2014)

The goal of proliferation resistance is introduced in the GEN IV goals, since a heightened risk of dual use has been associated with the historical predecessors of the selected reactors compared to LWRs. Dual-use in commercial GEN IV designs is intended to be made difficult or impossible through fuel-and reactor design choices. The risk of proliferation is also a matter to be considered with the closed fuel cycle. Closely related to this, improved physical protection against theft or acts of terrorism need to be considered. (Dunlap, 2025)

2.2 GEN IV reactor technology overview

In the early 2000s, GIF selected six general reactor technologies to be further developed among the member countries and parties. The decision was made on the basis that these reactors were the most likely to meet the set design goals. The selected reactors are the very high-temperature reactor (VHTR), the supercritical water reactor (SCWR), the sodium-cooled fast reactor (SFR), the molten salt reactor (MSR), the lead-cooled fast reactor (LFR) and the gas-cooled fast reactor (GFR). The main technical specifications these six GEN IV reactor designs are tabulated in Appendix 1. The operational values vary based on the reactor size and specific design but are in the given range. (GIF, 2018)

As a common factor, the increased thermodynamic efficiencies and high core outlet temperatures can be seen as a clear advantage in all the listed reactor designs. The fast reactors generally have a higher power density than generic LWRs. This means that while the reactor volume stays the same, the thermal power and residual heat generation increase. This is both an economic benefit and a safety concern. The scalability of the reactors is also quite good, both large and small units have been considered in design. In all six of these designs, the nuclear fuels are more advanced and different from the current LWR fuels. Prior to deployment, many of these will require more thorough development.

2.2.1 VHTR

The very-high temperature reactor is the development step from the high-temperature reactor (HTR) and further, the GEN II advanced gas-cooled reactors (AGR). VHTR is a gas-cooled, thermal reactor with a high core outlet temperature in the range of 700-850 °C, with the target design temperatures reaching 900-1000 °C. VHTR uses a graphite moderator and helium coolant at moderate pressure of 40-100 bars. Helium is a chemically and neutronically inert gas, which is beneficial for the reactor operation and in the case of a coolant leak. The main driver for this technology is the high core outlet temperature, allowing vastly better thermodynamic efficiency compared to LWRs, as well as heightened possibilities for use in cogeneration for industrial purposes. (Locatelli, 2013)

The two main design lines of the VHTR are the prismatic core (column) type and the pebble bed type. Both utilize the same tri-structural isotropic (TRISO) fuel but differ in the way that the fuel is packaged. The prismatic core consists of stacked graphite elements, which hold the fuel and coolant channels. The fuel rods inside the fuel channels consist of TRISO-particle filled fuel pellets. In the pebble bed design, the fuel particles are packed in spherical graphite pebbles. The pebble bed core consists of a pile of thousands of fuel pebbles, through which the coolant flows. (Yan, 2023)

Both designs have a pressure vessel with a graphite reflector surrounding the core. Control rods are driven to the core from the top of the vessel. The main differences are the related to the power density. Both reactors have a low power density, pebble bed having lower. This correlates to the maximum core outlet temperatures and the maximum feasible size of the reactor. The maximum power density is also limited by the aim to utilize passive residual heat removal. This favours a tall and thin core. The maximum feasible sizes are about 700 MW_{th} for the prismatic and 300 MW_{th} for the pebble-bed reactor. (Locatelli, 2013) Figure 2 shows the VHTR designs, the prismatic high-temperature rest reactor HTTR and the pebble-bed reactor HTR-10.

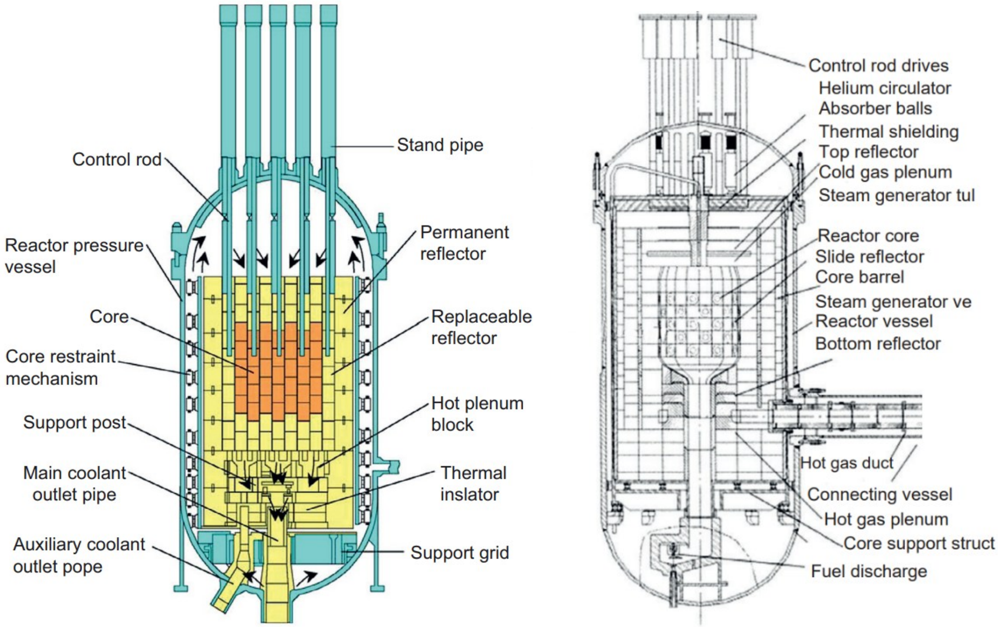

Figure 2. The HTTR (left) and the HTR-10 (right) (Yan, 2023, adapted).

The power conversion process in the VHTR can be either a Rankine or a Brayton process. The traditional Rankine process works with a steam generator transferring heat from the primary helium circuit to the secondary water circuit and further the steam turbine. More advanced designs aim to simplify the process by utilizing only helium in both circuits. This kind of process would allow a more compact and efficient helium gas-turbine process to be used, but is yet to be demonstrated. (Locatelli, 2013)

Since the VHTR is designed to operate at temperatures reaching 1000°C, the conventional nuclear fuels such as zirconium-cladded UO_2 are not suitable. Nuclear fuel must maintain its integrity and keep the fission products contained in normal, as well as accident conditions. This has led to a development of a fuel that has high capabilities for heat transfer, as well as the ability withstand high temperatures with minimal release of fission products. Instead of using a cladding, the TRISO fuel is held together by ceramic elements. TRISO fuel consists of a spherical UO_2 core particle surrounded by four ceramic layers. These are the loose pyrolytic carbon layer, the inner dense pyrolysis carbon layer, the silicon carbide layer and the outer dense pyrolytic carbon layer. (Liu, 2025) Figure 3 shows an overview of a TRISO particle.

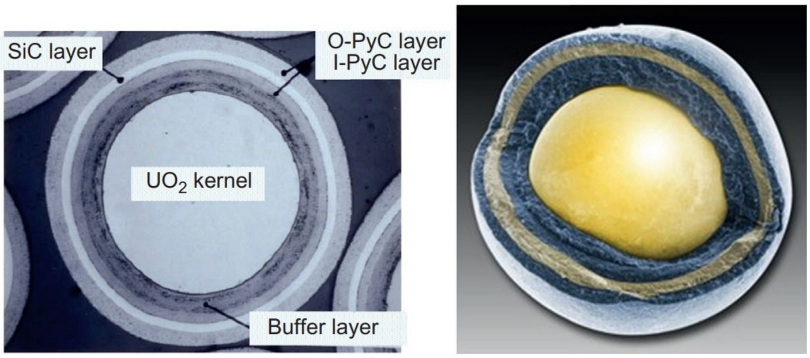

Figure 3. TRISO fuel particle (Yan, 2023).

The TRISO fuel also is the main driver of inherent safety in the VHTR. The fuel is demonstrated to withstand temperatures up to 1600 °C, which would allow the removal of residual heat through passive safety systems. The TRISO fuel is also proliferation resistant. The physical composition of the fuel particles makes the illicit use extremely difficult. In addition, the designed high burnup of power reactor TRISO fuel makes the isotopic composition of the fuel less desirable for dual use. (Yan, 2023)

2.2.2 SCWR

The supercritical water-cooled reactor (SCWR) is considered to be the evolutionary step from the current boiling water reactors. The plant specifications are quite similar, with the exception in the utilization of supercritical water as coolant. In thermodynamics, supercriticality is a condition of a fluid where separate gaseous and liquid phases do not exist anymore. The supercriticality is reached at the thermodynamic critical point. For water this is 374 °C and 22.1 MPa, dictating the operating pressure of the SCWR. The typical design pressures of the SCWR are around 25 MPa, core inlet temperatures being around 280 °C and outlet temperatures at 500-600 °C. (Wu et al, 2022)

The SCWR can be designed to function with fast, thermal or a mixed neutron spectrum. This flexibility is caused by the neutronic properties of supercritical water. As water above the transition temperature becomes a supercritical fluid, its density drops significantly. This reduces the neuron moderation so much that the reactor can stay in fast spectrum. SWRCs with a thermal spectrum use different means of neutron moderation. While the fast spectrum would bring the benefit of fuel breeding, safety concerns are a challenge. One major problem is the LOCA accident, where a negative void reactivity coefficient would have to be ensured, especially since the SCWR has an overall low water inventory in the core. (Marsault, 2004)

The two main design lines of the SCWR are the pressure vessel and pressure tube type. The pressure vessel type SCWR has a traditional pressure vessel, whereas the pressure tube type consists of modular pressurised channels, similar to those in a CANDU reactor. The thermal and mixed spectrum reactors are typically designed to function so that the coolant passes the core twice to even the axial power. In the mixed spectrum reactor, the first and second passes are divided into the fast and thermal regions of the core. (Wu et al, 2022)

The power conversion process of the SWCR is the supercritical Rankine cycle, which is in use in various conventional power plants. This is a clearest advantage of the SCWR, since the technology is proven. The process efficiency is around 45 %, which is around 10 percentage points higher than with the GEN III LWRs. In addition, the need of traditional BWR process components, such as the steam separator, steam dyer and the recirculation system are removed. (Locatelli, 2013)

The use of supercritical water in a nuclear reactor is yet to be demonstrated, the main concerns being the harsh conditions and corrosion on reactor materials. Given the high operating pressure of the reactor, the fuel cladding needs to be sufficiently thick, which creates a challenge regarding the neutron absorption. The traditional LWR cladding, zirconium, cannot be used in high temperatures. Other alternative materials such as aluminium are not strong enough, so the most common material is stainless steel. (Liu, 2025)

2.2.3 SFR

From the six GEN IV reactors, the sodium-cooled fast reactor is the most widely tested and developed. Over 20 sodium-cooled fast reactors have been built for various purposes and with various operational performance. This corresponds to hundreds of years of reactor operation, which yield valuable information on their operability. The SFR operates with a fast neutron spectrum and uses liquid sodium as coolant. The usage of fast neutron spectrum brings the ability to use the reactor to breed fertile material into fissile, but as in any breeder reactors the full benefit of breeding requires adapting the closed fuel cycle. The SFR can also be designed for “burning” of high-level waste, such as plutonium and minor actinides. (Locatelli, 2013)

The two main design lines of the SFR are the pool type- and the loop type reactor. The pool type reactor consists of a single reactor vessel, which houses the entire primary coolant system. This includes the reactor core, control rods, intermediate sodium heat exchanger and the primary coolant pumps. In the loop type reactor, the intermediate heat exchanger is located outside the reactor vessel and connected to it via piping. In both designs, the intermediate heat exchanger links the reactor coolant system to the secondary sodium system, which then in turn transfers heat to the power conversion system. The loop-type design has the benefit of a smaller reactor vessel; this eases maintenance activities and reduces overall plant capital costs. The main benefit of the pool-type layout is the increased safety considering the thermal inertia of the sodium volume inside the reactor vessel and the lower number of primary sodium system penetrations. The two reactor designs are displayed in Figure 4 below. (Ohshima, 2023)

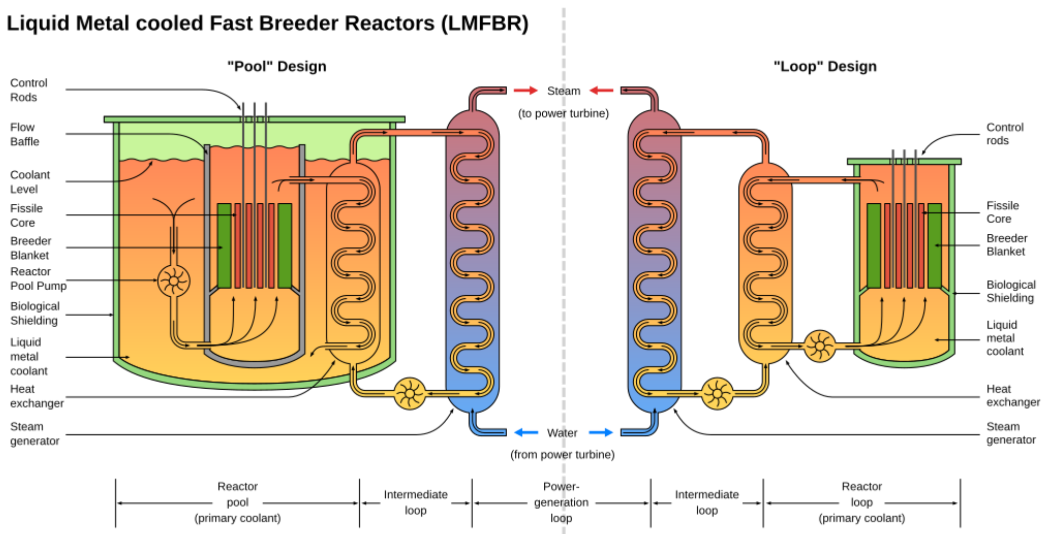

Figure 4. Two designs of the sodium-cooled fast reactor (Dunlap, 2025).

As reactor coolant, sodium brings many differences to the ordinary light water in terms of reactor design. Sodium is a metal with a melting temperature of 97.8 °C and a boiling temperature of 882.9 °C, which allows a large range of operation with atmospheric pressure. The ability to operate without pressurising the reactor system greatly decreases the risk of a loss-of-coolant accident. In case of a leak, the coolant is typically designed to be confined to an outer safety vessel. This also limits the maximum volume of coolant lost. Sodium has a high thermal conductivity (72.2 W/mK) and low neutron absorption- and scattering cross-sections, which makes it a suitable coolant for fast reactor. The downsides are the challenging chemical properties. In high enough temperatures, liquid sodium spontaneously combusts when in contact with air. Sodium is also highly reactive with water, producing hydrogen and heat. The risks related to the chemical reactivity of sodium are well known, and they have caused numerous operational incidents in history. Sodium has only one naturally occurring isotope, ^{23}Na. Neutron absorption of sodium leads to it becoming beta-active ^{24}Na with a half-life of 15 hours, which then in turn decays into stable ^{24}Mg. Radiation protection challenges related to sodium are low, since-it has no long-living by-products. (Ohshima, 2023)

The power generation process of the SFR is dictated by the boiling point of sodium. The typical values of sodium core outlet temperatures are in the range of 500-550 °C to ensure a high-enough safety margin to boiling. Formation of sodium bubbles in the coolant must be avoided, since the void reactivity coefficient of sodium is positive due to the small, but present neutron absorption. Rankine process operating with superheated steam at these temperatures has an efficiency of approximately 40 %. The power conversion process serves as a heat sink during normal operation in the SFR, but accident- and residual heat removal in newer SFRs are designed to function with passive systems. These include convective cooling by air and natural circulation. The high thermal conductivity of sodium is a benefit in this. (Ohshima, 2023)

The SFR fuel has to meet multiple design goals. These are achieving high fuel burnup (150-200 MWd/kgU), high irradiation resistance and the possibility of including minor actinides for burning. SFR cores typically consist of the fissile core region and the breeding/conversion region. The fissile region consists of high-enriched fuel, to supply high neutron production, while the breeding region consists of fertile materials to be turned into fissile. Some newer reactor designs tend to not favour separate breeding zones (or blankets) due to issues with proliferation prevention. These are called mixed core designs. The standard fast reactor fuels are oxide fuels, such as in LWRs, but carbide (U,Pu)C, nitride (U,Pu)N and metal fuels (U-Pu-Zr) are in development for their improved neutronic and heat transfer properties. The selection of cladding is dictated by the neutron absorption cross-section and the compatibility with sodium. Traditional cladding material, zirconium, is not suitable for fast reactors since its neutron absorption cross-section is too high. Other materials such as aluminium, chromium, vanadium and nickel have either insufficient material strength or incompatibility with sodium. Historically, it has been determined that best materials suitable for SFR fuel cladding are austenitic steels. (Liu, 2025)

2.2.4 MSR

The molten salt reactors are either fast or thermal reactors cooled by liquid molten salts. MSRs can be categorized to three main designs. These are the circulating fuel design (homogenous reactor), static fuel design and the salt-cooled high-temperature reactor. The most pursued type, the circulating fuel design, functions with molten salt serving as both fuel and coolant. The fissile and fertile fuel is contained in a molten salt and circulated through a primary circuit, becoming critical at the primary vessel where a solid moderator is present. The second type, static fuel design, resembles the traditional solid-fuel reactor. Here the fuel salt is contained in sealed tubes, and the cooling is provided with separate cooling salt. The third design utilizes molten salt only as a coolant, with fissile material confined in a solid fuel. (Dunlap, 2025)

The fuel salts that could be used with MSRs are fissile and fertile actinide fluorides (UF_4,PuF_3,ThF_4). The only natural fluorine isotope ^{19}F has a low absorption cross-section, which is fitting for the fuel. The fluoride salts, however, have high melting temperatures of above 1000 °C, which makes the reactor design difficult. To lower the melting temperature, the fuel salts can be mixed with fluorides such as zirconium (ZrF_4), beryllium (BeF_4) or lithium (LiF_4) with a fluoride to fuel ratio of around ¾. This lowers the melting temperature to around 500 °C. This also lowers the corrosiveness of the fuel and reduces risks such as fuel solidification in the primary circuit. For the salt-cooled high-temperature reactor, one pursued possibility is to use TRISO-fuel, such as in the VHTR. (Allibert, 2023)

Thermal MSRs consist of a core typically fitted with a graphite moderator. Fuel salt travels through channels in the core to an intermediate heat exchanger, where heat is transferred to the secondary salt circuit. The secondary circuit then transfers heat to the power conversion process via a steam generator. The core outlet temperatures of around 600-700 °C allow high thermodynamic efficiencies of around 45-50 %. The fast MSR functions in a similar way, except it has no moderator and can have the intermediate heat exchanger located inside the reactor vessel, similarly to a pool-type SFR. The fast neutron spectrum also allows the reactor to be operated as a breeder. This configuration is quite interesting, since the newly bred fuel could possibly be reprocessed and recirculated while the reactor is running. (Dunlap, 2025)

The MSR system has some distinct features compared to an LWR. These include emergency core dump tanks and a fuel chemical processing facility. In the case of loss of cooling, the fuel obviously cannot melt. The corrosiveness of fuel does, however, increase as the salt temperature increases, which can lead to dangerous structural material failures. If necessary, the fuel salt can be removed from the reactor into separate dump tanks. The fuel salt circuit of the MSR is connected to a fuel chemical processing system, where the composition of the fuel can be altered. The fissile material content can be adjusted, which allows online refuelling, and continuous removal of fission products. This, however, requires a facility for safely storing the accumulated solid and gaseous fission products. Another positive side is the ability to operate the reactor with minimal excess reactivity. (Dunlap, 2025)

2.2.5 LFR

The lead-cooled fast reactor is a lead or lead-bismuth eutectic cooled reactor operating on the fast neutron spectrum. The LFR shares many traits with the sodium-cooled fast reactor, both being fast, liquid-metal cooled reactors typically operated close to atmospheric pressures. As required for a fast reactor coolant, lead has a low neutron absorption cross section. Lead also has a higher atomic weight than sodium, so the elastic neutron scattering does not slow the neutrons down as much as sodium does, hardening the neutron spectrum. In terms of heat transfer, lead is an effective coolant with good thermal conductivity (17 W/mK), albeit less than that of sodium. The melting temperature of pure lead is 327.5 °C, and the boiling temperature is 1749 °C. Typical LFR operating temperatures are around 500 °C, the corrosion of structural materials in higher temperatures setting the limit. This operational temperature can achieve thermodynamic efficiency of around 40-45 %, similar to that of an SFR. (Dunlap, 2025)

The reactor is typically designed in a similar way as a pool-type SFR. The reactor core, main coolant pumps and the steam generator are all housed in a reactor vessel. The loop-type design is more challenging since the flow rate of lead has to be limited. This is due to corrosion and pressure loss challenges. The low flow rate also requires large diameter piping (Smith, 2010)

The lead coolant has some safety benefits compared to sodium. The most obvious is the lack of dangerous reactivity with air or water, which eliminates the need of an intermediate circuit to avoid primary coolant to water interaction. Contact between coolant and air should be avoided, however, since this leads to formation of solid lead oxides, which can build up and clog the reactor cooling system. Another benefit is the high boiling point. Lead coolant has a positive void reactivity coefficient such as sodium, but the margin from operating temperature to boiling is much higher. In practice, the positive void reactivity could be considered a problem only in the case of a loss-of-coolant accident, which is also much less of a concern than in LWRs, since the operating pressure is equal or close to atmospheric. (Smith, 2010)

The coolant used in the LFR can be either pure lead or lead-bismuth eutectic (LBE). One concern in the LFR operation is the possible solidification of coolant during shutdown or transient conditions. In these situations, the lead is kept in liquid form via large electric heaters. The high melting point of lead can be lowered from 327.5 °C to 123.5 °C by making the coolant a lead-bismuth alloy. The main downsides of LBE are the increased cost, corrosivity and radiation hazards. After a neutron capture, ^{209}Bi becomes beta-active ^{210}Bi which decays into ^{210}Po with a half-life of five days. ^{210}Po then alpha decays into ^{206}Pb with a half-life of 138.4 days. The long half-life causes polonium to come into an equilibrium in the reactor system. Polonium is a serious radiation and toxicity hazard in the case of a coolant leak and in primary circuit maintenance activities; it also considerably extends the time of residual heat generation. (Dunlap, 2025)

The LFR is designed to use either MOX or uranium-plutonium nitride fuel. The use of nitride fuels over oxide and metallic fuels is favoured due to their better compatibility with lead, high density, radiation swelling resistance and thermal conductivity, making them suitable for the conditions of the LFR. The nitride fuels, however, are still in the development phase. Suitable cladding materials include ferritic/martensitic steels and austenitic steels. (Liu, 2025)

2.2.6 GFR

The gas-cooled fast reactor is a reactor operating on the fast neutron spectrum and cooled by a gaseous coolant. The GFR shares similarities with the VHTR, both aim for high core outlet temperatures resulting in good thermodynamic efficiency and extensive production possibilities. The GFR aims to bring the possibility of fuel breeding to the operation of gas-cooled reactors, which has not been done before. Consequently, the GFR is the least developed of the six GEN IV designs, with no functioning reactors built in history. (Dunlap, 2025)

The GFR concept system consists of a reactor vessel, which houses the reactor core consisting of hexagonal elements of ceramic-cladded mixed-carbide fuel. The core outlet temperatures are in the range of 600-900 °C. Compared to liquid metal cooled fast reactors, the gas coolant could bring some benefits. The total neutron cross-section of typical gaseous coolants is smaller than those of liquid coolants. This allows a harder neutron spectrum, which improves the fuel breeding ratio. The possible gas coolants are helium, carbon dioxide and dinitrogen dioxide. The use of helium has already been proven in VHTRs, helium is inert and has a low neutron absorption cross-section. The thermal VHTR uses a graphite moderator, which brings thermal inertia into the reactor system and allows an easier design of the passive residual heat system. (Dunlap, 2025)

Due to the low density and low heat capacity, the GFR coolant has to be pressurized to 80-100 bars to reach adequate heat transfer in the core. In the case of reactor depressurization and coolant dissipation, efficient safety systems are needed to provide effective residual heat removal. While this is relatively easy to implement in the VHTR, the lack of thermal inertia is an issue with the GFR. Another safety drawback is the low, but still positive temperature coefficient of reactivity, caused by the hard neutron flux. (Dunlap, 2025)

3 Current development

In the recent years, the development of GEN IV reactors in the western world has shifted from national laboratories and research facilities towards privately owned companies. These include both traditional large nuclear energy companies as well as small start-ups. The most mature designs and ongoing construction projects are currently located in China, Russia and India, controlled by the public sector. The major challenge for the western companies is to acquire and utilize the knowledge generated during the operation of the historical experimental and demonstration reactors, as well as to develop a technologically and economically feasible system. The major GEN IV reactor developers, their design timelines, and a preliminary financial estimation are presented in this chapter.

3.1 Design development

IAEA publication: “Terms for Describing Advanced Nuclear Power Plants” has set guidelines for classifying new reactor designs as well as their development status. Three terms typically used to describe new reactor designs are “advanced”, ”current” “evolutionary” and “innovative.” A solid consensus on using these terms is unclear, and their cross-use is frequent, especially in marketing materials. The “advanced” design is defined as a reactor with significant improvement over its predecessors and can be understood as an umbrella term for all GEN III and GEN IV reactor designs. The “current designs” are advanced reactors already in operation, under construction or licenced. Current LWR designs such as the Westinghouse AP-1000 and EDF EPR could be classified as current and advanced designs. (IAEA, 2023)

The term “evolutionary” designs refer to advanced designs that aim for improvement with proven and technologically ready design choices. Compared to the current designs, the evolutionary designs are not ready for immediate deployment but require further testing and development. Going even further, the “innovative” designs bring major conceptual changes to the existing practice. They typically require extensive R&D, as well as construction of technological demonstration or prototype units prior to commercialization. Generalizing, the LWR SMRs could be classified as “evolutionary” and the six GEN IV designs as “innovative” reactors. The relationship between the terms is displayed in Figure 5 below. (IAEA, 2023)

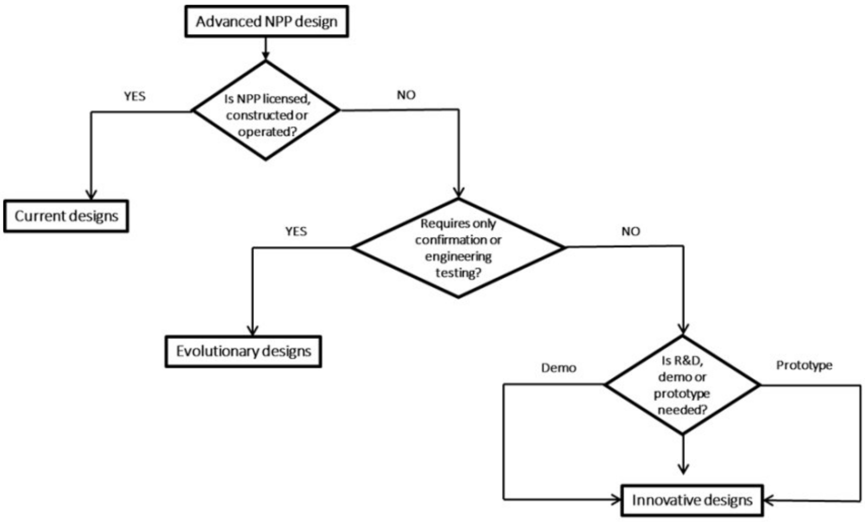

Figure 5. Relationship between terms related to NPP design categories (IAEA, 2023).

For reactors in the design stage, IAEA uses a model of four design phases to describe the states of development. These are the concept description, conceptual design, basic design and detailed design.

The design development begins with the concept description phase, where the basic idea and goal are described, and development and testing needs are identified. The second phase is conceptual design, where the design of key components and systems are extended and a preliminary analysis is done on relevant accident scenarios. The third phase is the basic design, where a complete system description is done, safety analyses are completed and technical specifications for major components are made. The fourth and final phase, detailed design, marks a nearly complete design. Here the construction schedule is ready, along with manufacturing, procurement and commissioning specifications. The final safety analysis report is also finished. (IAEA, 2023)

The end of detailed design, however, does not mean that the reactor is necessarily ready for deployment. As stated before, innovative designs might require construction of engineering test units, and partial or full-scale demonstration reactors prior moving to commercial deployment. In a MIT study: “The Future of Nuclear Energy in a Carbon-Constrained World” (Petti et al, 2018) the development of innovative nuclear technologies towards commercialization is described with four steps. The first step is the research and development, which consists of the four development steps described by the IAEA. After this come the engineering demonstration, performance demonstration and commercial demonstration steps.

The engineering demonstration is necessary if the reactor technology has never been built before and a proof-of-concept is needed. Scaling up to full or near-full size, comes the performance demonstration, where the purpose is to gain operating experience and to validate the system. Finally, the commercial demonstration is a full-scale unit, which corresponds with the following commercial offerings of the technology. For the more mature innovative designs, all these steps are not necessary, and the overall proceeding towards commercialization is faster. For example, for the more developed VHTR, one possible route to commercialization for a reactor developer could be to build one reduced scale performance demonstrator, which is scaled up for the first commercial unit. (Petti et al, 2018)

3.2 Ongoing projects

The IAEA keeps a record of currently ongoing projects of evolutionary and innovative nuclear reactors in the "Advanced Reactor Information System" (ARIS). Reactors presented in the ARIS system, along with a few non-power reactor additions is included in a table in Appendix 2. The table lists currently ongoing GEN IV projects, along with their general specifications and developer-estimated first deployment dates. The state of development is divided into five steps: the first three come from the IAEA guide for describing the development of advanced nuclear power plants, and the fourth and fifth show the state of possible demonstration unit construction. These states of development are presented in Table 1 below.

Table 1. Stages of development and identified projects in each stage.

Stage of development	Description	Identified projects
1. Conceptual design	Key components, layout drawings, single line diagrams are available	31
2. Basic design	Brief descriptions of key components and systems are provided	5
3. Detailed design	System descriptions, safety analyses and licensing documents are available. Itemized cost estimate and master schedule for the construction phase have been prepared.	10
4. Engineering Demonstration	Construction or planning of a nuclear demonstrator/prototype plant. Gain of operating experience.	9
5. Commercial demonstration	Construction of a first of a kind, full-scale plant	2

As an overview, notable currently ongoing GEN IV projects are displayed in Figure 6 below, a larger figure with all ongoing projects is included in Appendix 3. The x-axis displays the developer estimate for the year of first prototype or commercial unit deployment, and the y-axis displays the state of development as is described in Table 1. As clarification, the Russian, Chinese and Indian governmental projects have been marked with an asterisk.

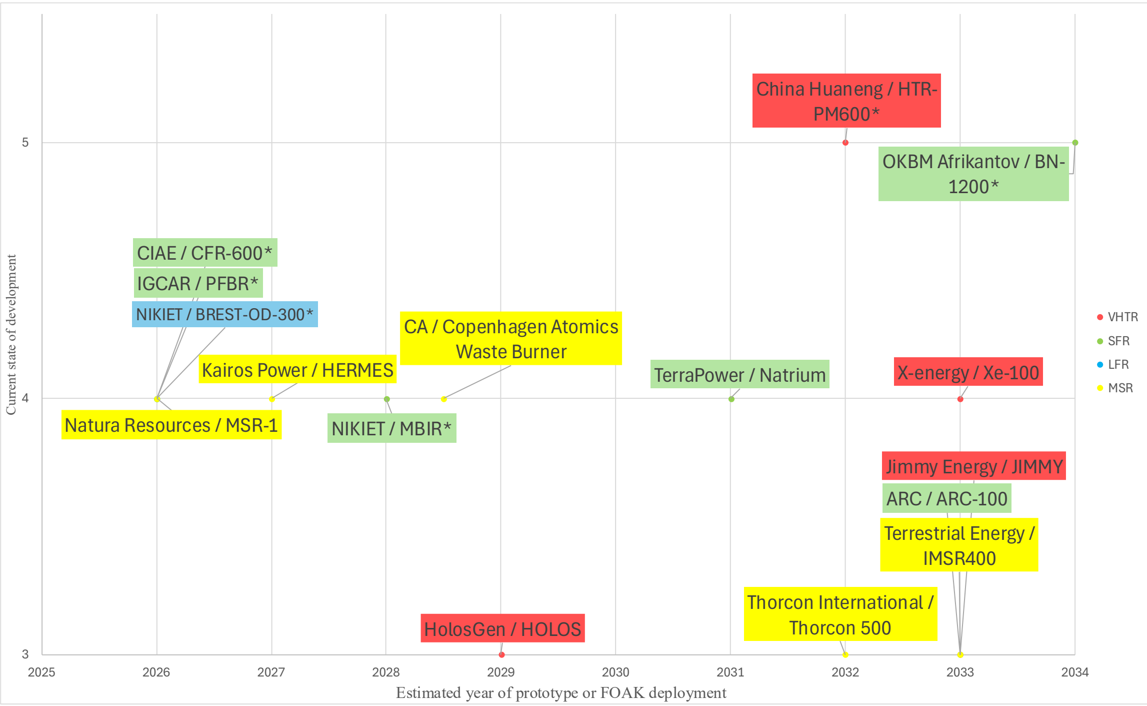

Figure 6. Currently ongoing GEN IV projects, their state of development and estimated year of deployment according to the vendor.

The accuracy of the estimated year of deployment is extremely low, since the data is supplied by the reactor developers and is part of their marketing. Majority of the designs are still in the initial stages of development, so it is understandably difficult to estimate when the design is ready to be licensed and demonstrated. The general accuracy of the estimated year of development increases as the stage of development increases and further, as a demonstration unit is designed and planned. This would mean that the data of reactors currently in detailed design or with planned demonstrator units is more accurate. In the next few years, the Russian demonstrators LFR BREST-OD-300 and SFR MBIR are planned to enter operation along with a Chinese double-unit SFR CFR-600 and the Indian SFR PFBR.

In the western countries, multiple demonstration reactors are planned for the coming years. The molten salt reactors Natura Resources MSR-1 and Kairos power KP-FHR demo HERMES have been granted construction permits by the NRC. These two are small-scale non-power test facilities (1 MW and 35 MW), which might explain the fast granting of the permits. (NRC, 2025) Other demonstrators planned are the SFR TerraPower Natrium, VHTR X-energy Xe-100, and the MSR Copenhagen Atomics waste burner. TerraPower and X-energy have applied for construction permits and plan to operate the reactors in the early 2030s.

At the moment, there are two projects in the most advanced stage of development: the HTR-PM600 and the BN-1200. Both these reactors have “demonstrator” units currently in operation (HTR-PM, BN-800). The options of wide commercialization of these reactors are uncertain. The HTR-PM600 might be sold outside the domestic Chinese market, but the local Russian and Chinese applications are the main targets for these reactors. One particular interest is in the cogeneration applications of the HTR-PM600, these types of reactors could be used in replacing Chinese fossil heat generation in the steel industry, district heating, and water desalination. (Zhang, 2022) The applications in Russia are most likely focused on traditional electricity generation. In addition, Russia has a long history of nuclear cogeneration, with both BN-600 and BN-800 reactors producing district heating alongside electricity. (IAEA, 2021)

The time it takes to reach commercial maturity is dependent on the success of constructing and operating the demonstrator units. If using the HTR-PM as an example, the detailed design was completed and construction began in 2012, and full power was reached in 2022. If the commercial demonstrator HTR-PM600 is finished in 2032, this would mean a 20-year long development from the end of detailed design to ready commercial deployment. It is also noteworthy that long prior to the design of the HTR-PM, the experimental HTR-10 begun construction in 1995, and went into operation in 2000. Estimations of the general dates of GEN IV technology commercial deployment have been made multiple times in recent history. After GIF was founded in 2000, the first estimations stated that all six systems could be deployed before the year 2030. (DOE, 2002) The development was not this fast, however, and was slowed down by events such as the Fukushima disaster in 2011 and the overall lack of funding and interest in more novel designs. Based on the estimates of current demonstrator reactor deployment and initial deployment estimates presented in “The future of nuclear energy in a carbon-constrained world” (Petti et al. 2018, 88), Figure 7 presents the currently possible GEN IV deployment dates.

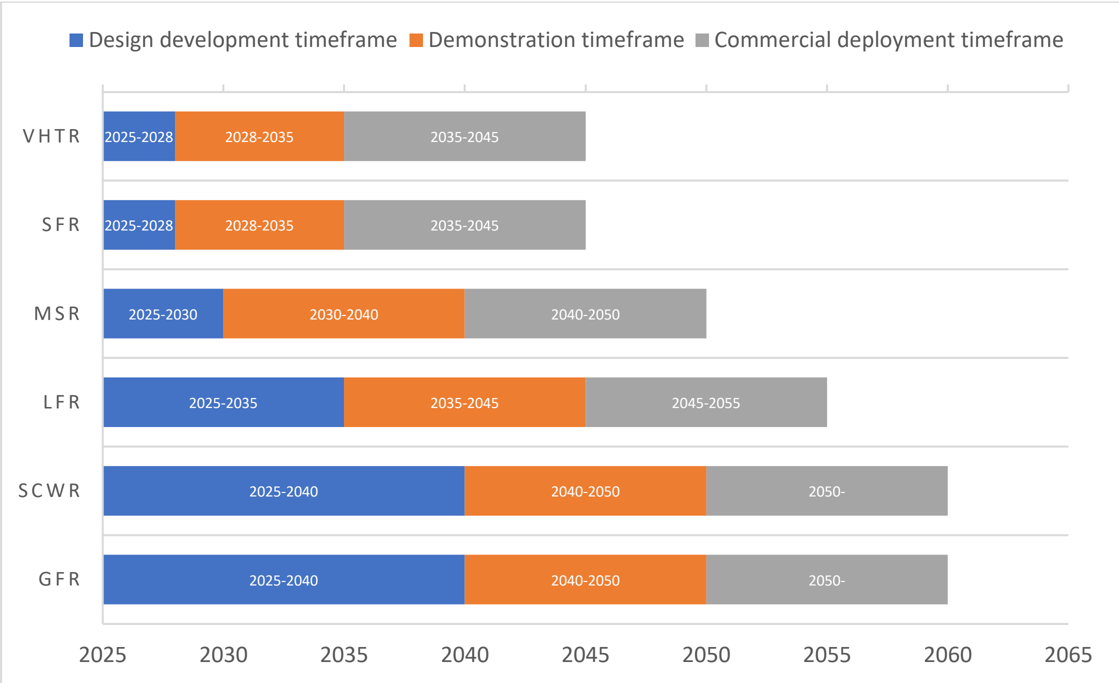

Figure 7. Estimated GEN IV technology deployment timeframes.

The VHTR and SFR are the most developed among the six technologies. They are also the closest to possible commercial deployment. The development and interest towards the MSR and LFR in coming years dictates their rate of development. They could possibly become available in the 2040s or 2050s. The SCWR and GFR are currently the least developed, and only a few major projects currently exist. Their feasibility might be known better later, as they advance towards higher states of development.

As there are large uncertainties and only little available and relevant data on the SCWR and GFR, they are excluded from the rest of this thesis.

3.3 Financial estimations of GEN IV reactor projects

Estimating the financial aspects of innovative reactors is extremely challenging due to the lack of detail in design. Majority of available information on currently developed GEN IV reactor capital- and operating costs comes from different kinds of marketing material, with little factual value. Often, the early estimates of capital- and O&M costs for nuclear are overly optimistic and get more realistic as the design is developed further. Acquiring accurate cost estimations of “generic” designs of the GEN IV reactors would require a large and independent study based on detailed plant designs. A limited amount of open-source information is available on the historical prototypes, but also on the recently commissioned demonstration reactors (namely the BN-800 and HTR-PM). These values for Russian and Chinese plants are not directly indicative to what western newbuild reactors would cost. This is affected by several factors, such as differences in costs of labour, materials and the licencing process. As a result, the overnight costs in western countries might easily be twice as high.

An IAEA document “Limited Scope Sustainability Assessment of Planned Nuclear Energy Systems Based on BN-1200 Fast Reactors” (IAEA, 2021) has listed estimations on overnight capital costs of Russian BN-type sodium-cooled fast reactors. The listed specific capital costs for a BN-800/1200 reactor are reported to be in the range of 3800-4500 $\$_{2025}$/kW(e). A value of 3700 $\$_{2025}$/kW(e) is given as a reference for a VVER-TOI plant. This would imply that Russian SFRs could have roughly 5-10 % higher capital costs than corresponding PWRs. As a comparison, the final public estimations for the cancelled Finnish Fennovoima AES-2006 VVER were approximately 6000 €$_{2021}$/kW(e). (World Nuclear News, 2021)

Overnight costs for the Chinese HTR-PM have been presented in “Economic feasibility of flexible energy productions by small modular reactors from the perspective of integrated planning” (Nian, 2020). Here the value for capital cost is 4000 $\$_{2025}$/kW(e), assuming the plant is used only for electricity generation.

Financial estimations for the LFR and MSR are currently extremely early and uncertain. A possible range of a NOAK MSR overnight cost is between 6000-7000 $\$_{2025}$/kW(e) (Petti et al, 2018,70). For the LFR, an indicative value for the overnight capital cost is 5100 $\$_{2025}$/kW(e), which is the predicted cost of the Russian BREST-OD-300, planned for commissioning in 2026. (Steigerwald, 2023)

4 Operability and performance

The performance of a nuclear power plant can be measured with multiple different indicators since the definition itself is quite broad. The world association of nuclear operators (WANO) has set of outcomes for a well-performing power plant in the manual "WANO Performance Objectives and Criteria" (2019). The key outcomes determined by WANO include sustained event-free operations; avoidance of long, unplanned shutdowns and well-managed and understood safety, design and operating margins. The performance objectives are met with operational quality and plant availability, which are inevitably linked with nuclear safety.

Multiple historical predecessors of the GEN IV reactors have suffered from low operating performance. While some of this can be explained as "teething issues," or FOAK equipment problems, issues from the general reactor design also cause them. The definitions of nuclear performance indicators, and the interconnection between nuclear safety and availability are presented in this chapter.

4.1 Operating performance indicators

The most general meter of operation is the power plant availability. In the IEEE standard "IEEE Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Systems and Other Nuclear Facilities" (2016), availability is defined as: "The probability that an item or system will be operational on demand." Availability can be measured with multiple quantitative methods, often based on yearly net energy output. In addition to this, the nuclear energy sector has various specific performance indicators.

The simplest power plant performance indicator is the load factor (or capacity factor). The load factor is the ratio of the net energy generated to the reference energy generation in a set time period:

$$LF(\%) = \frac{EG \cdot 100(\%)}{REG} \tag{1}$$

where LF is the load factor, EG is the net energy generated [MWh], and the REG is the reference energy generation [MWh]. The net energy generated is the electrical or heat energy supplied to the grid or the user. The reference energy generation is the amount of energy that could be produced if the unit operated the given time period on the reference unit power. (IAEA, 2005)

The load factor is used to tell the actualized unit utilization; this ultimately is what generates revenue. The load factor cannot be used to distinguish planned, unplanned, or station-independent production losses. Another performance indicator is the operation factor, which is the ratio of the hours the unit was connected to the grid to the total hours in the given period:

$$OF(\%) = \frac{t \cdot 100\%}{T} \tag{2}$$

where *OF* is the operation factor, *t* is the number of hours online [h] and *T* is the reference time period [h]. Compared to the load factor, the operation factor is not lowered by the changes of power output during plant startup or shutdown, or the operation during flexible operation and variable load. (IAEA, 2005)

Other performance indicators are linked with the planned and unplanned energy losses. One is the unit capability factor:

$$UCF(\%) = \frac{(REG - PEL - UEL) \cdot 100\%}{REG} \tag{3}$$

where *UCF* is the unit capability factor, *PEL* is the planned energy loss [MWh] and *UEL* is the unplanned energy loss [MWh]. Planned energy losses mean energy losses caused by planned outages and testing activities. Unplanned energy losses are caused by unplanned outages, outage extensions or repairs. *UEL* does not consider issues such as grid losses, lack of demand, environmental limitations, labour strikes, or fuel coastdown. A high UCF value indicates a good level of maintenance activities and outage planning. (IAEA, 2005)

Performance can also be measured via looking at failures. If considering only the faults during the operating period, we can use the operating period forced loss rate:

$$FLR(\%) = \frac{FEL \cdot 100\%}{REG - (PEL + EPL)} \tag{4}$$

where *FLR* is the forced loss rate, *FEL* is the operating period forced energy loss [MWh] and *EPL* is the unplanned extensions to outages loss [MWh]. The *FLR* does not take planned outages or their extensions into account but is affected by unplanned outages or load regulations. A high *FLR* value indicates common equipment failures, faults due to human factors or possible safety deviation-related losses. (IAEA, 2005)

If taking all unplanned energy losses to account, we get the unplanned capability loss factor:

$$UCL(\%) = \frac{UEL \cdot 100\%}{REG} \tag{5}$$

Where *UCL* is the unplanned capability loss factor and *UEL* is the total unplanned energy loss [MWh]. This factor is affected by all unplanned power losses and is a good meter of the level of maintenance and outage planning.

One important nuclear-specific performance indicator is the unplanned automatic scrams per 7000 hours critical:

$$UA7 = \frac{(number\ of\ automatic\ scrams\ while\ critical) \cdot 7000}{total\ number\ of\ hours\ critical} \tag{6}$$

Similarly, *US7* indicator shows the total scrams per 7000 hours critical, also counting manual scrams. The hours critical is normalized to 7000 since this is roughly the yearly average time critical per unit. A high amount of unplanned reactor scrams is undesirable due to loss of production, and the thermohydraulic transients they cause. Automatic scrams are initiated by the reactor protection system, which for example, is triggered when determined safety margins are exceeded or when safety-critical systems become unavailable, leading to a breach of operational limits and conditions. A low UA7 value indicates safe and reliable operation, and a good level of maintenance. (IAEA, 2005)

4.2 Factors affecting power plant availability

As the most important performance indicators show, the reliability of a power plant can be measured with available and unavailable production capacity. The world nuclear association (WNA) has analysed in a report “Optimized Capacity: Global Trends and Issues” (2015), various causes of energy losses in the global reactor fleet. During the report observation period (2008-2012), the average load factor of all operating power reactors was 78.54 % and 94.43 % for the best-performing 10 %. These values have stayed relatively similar since the year 2000, the world average being 81.5 % in 2023. (WNA, 2023) When dividing the production losses to planned and unplanned unavailability, it is seen that the majority of losses are planned. Figure 8 shows that in both global and best-performing plants the planned energy losses are dominant, followed by a lesser fraction of unplanned and external losses. Both the fractions of planned and unplanned losses are noticeably smaller in the best-performing plants, but the share of unplanned losses is more notable in the case of all reactors. (WNA, 2015)

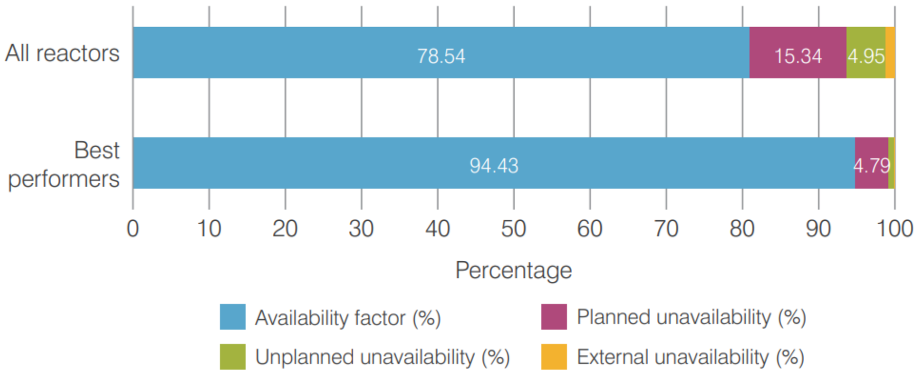

Figure 8. Availability of reactor units in 2008-2012 (WNA, 2015).

WNA states that there is no overall detectable age-related decay in the power plant load factors. Related to this, the early operations after a reactor has entered commercial operation typically show lower performance. This is caused by hidden flaws detected only after operations have begun, as well as the "learning curve," or improvement of methods in the plant operations. Figure 9 shows data from the early operations of reactors commissioned in recent years (2015-2025) from IAEA power reactor information system (IAEA, 2025b). The average load factor typically drops after the first year of operation, indicating either a longer first outage, or a failure-caused unplanned unavailability. Most of the recently commissioned plants have reached desired load factors in the first five years of operation.

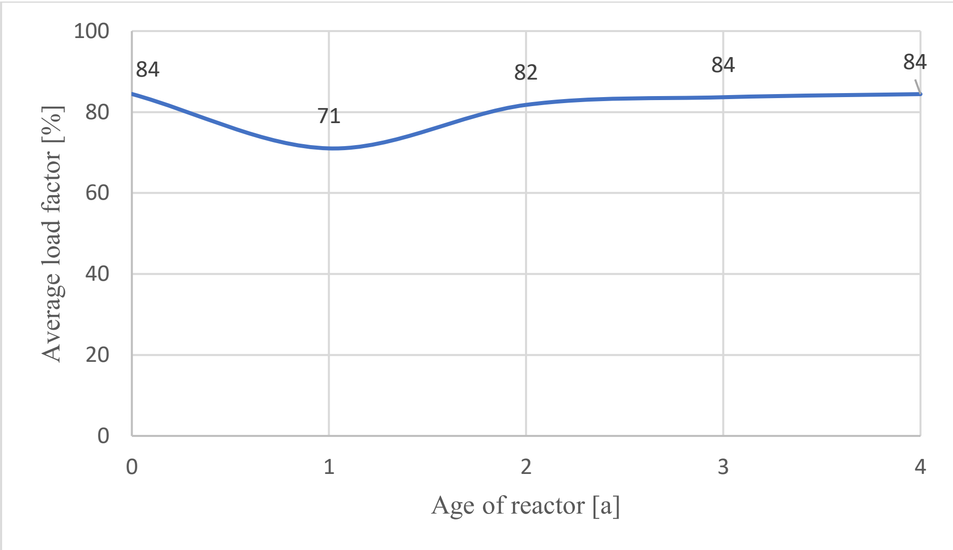

Figure 9. Average load factors in early operations from reactors commissioned in 2015-2025.

The planned unavailability of nuclear plants mainly consists of scheduled maintenance- and refuelling outages. The extent of work, possibly including large modernizations or backfitting extends this loss. Usually, the most time-critical work in nuclear power plant outages is refuelling. The minimum number of days that a refuelling outage requires is dictated by the length of cooldown after reactor shutdown. In addition, the amount of mechanical work needed before the fuel can be accessed affects this. The duration of planned outages can be optimised with various means, such as performing as much maintenance activities as possible while online and utilizing precise and high-quality outage planning. (Miller, 2011)

The unplanned energy losses are caused by reasons not under plant control. These include extensions of planned outages, automatic and manual scrams and deferred and immediate controlled shutdowns. The most common causes for these are plant equipment problems and faults. Less common causes are human factor-related problems, regulatory requirements, and fires. (WNA, 2015)

Figure 10 shows the unplanned energy losses by system for all and the best-performing plants. Generally, most of equipment faults are in the power-conversion side of the nuclear power plant process. The turbine, electrical power supply systems and the main generator systems cause the most losses on average. It is noted that the safety systems cause relatively little unplanned losses, being among the lowest in the case of the best performers. This can be explained with the rigorous availability requirements these systems have, due to the need to operate reliably during any adverse reactor events. (WNA, 2015)

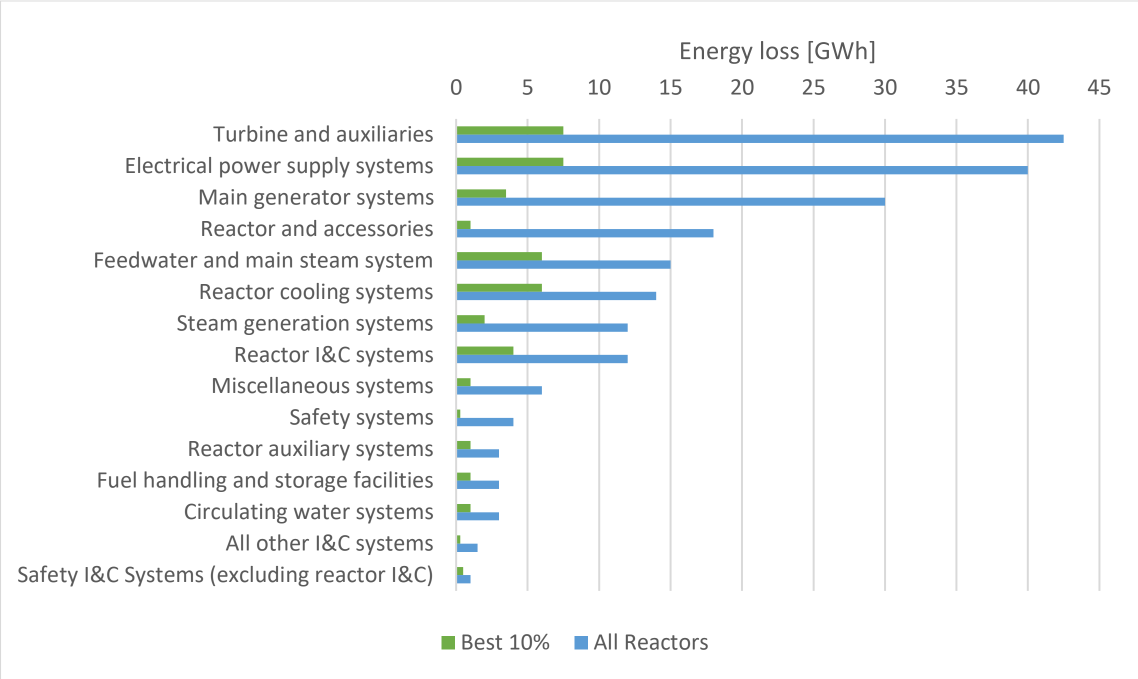

Figure 10. Unplanned energy loss by system (2008-2012), reproduced from (WNA, 2015).

In can be summarized that a well-operated and maintained nuclear power plant has a good control over all production losses. The planned maintenance and refuelling outages should be optimized and well-prepared for. The unplanned production losses can be mitigated with good maintenance and equipment reliability programs, as well as reducing the human factor by improving organizational and human performance. In a more uncommon case, the unavailability might be caused by entirely external factors, such as market and financial conditions, fuel availability, public relations and extreme environmental conditions.

4.3 Nuclear safety and performance

While some component failures may cause plant unavailability directly and immediately, failures or abnormal operation in safety systems or safety-related systems may cause unavailability indirectly through the plant's operating limits and conditions (OLCs). Breaching these conditions requires the plant to be brought to an allowed state, such as a reduced power level or reactor shutdown. Safe operation is thus required for good performance.

The IAEA safety guide: “Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants” (IAEA, 2022) states that: “The plant shall be operated within the operational limits and conditions to prevent … anticipated operational occurrences or accident conditions, and to mitigate the consequences of such events if they do occur.” The OLCs include safety limits, limits for normal operation, and action statements for deviations from normal operation.

The safe operation of a nuclear power plant is maintained by the limits and conditions for normal operation. The safety limits, safety system settings and operational limits are set for different process parameters. Typical parameters include the neutron flux and flux distribution, power oscillation, temperature of cladding and coolant, coolant flow rate, feedwater temperature and flow rate, containment pressures, levels of radioactivity in primary and secondary circuits, and coolant leakage monitoring. The OLCs also describe which systems and components are required to be operable at a given plant state. In general, safety limits are intended to assure the fulfilment of the fundamental safety functions: control of reactivity, removal of heat from the fuel and confinement of radioactive materials. (IAEA, 2022)

The Finnish radiation and nuclear safety authority (STUK) has set requirements for OLCs in the YVL A.6 regulatory guide (STUK, 2018a). The guide states that OLCs should be prepared so that the plant is operated in accordance with the design bases specified in the safety analysis report, and the requirements and limits set in the OLCs should be verified by means such as probabilistic risk assessment (PRA). If the plant is detected to be at a state not consistent with the OLCs, the remedial actions shall be taken immediately in order to reestablish compliance. The regulatory guide also gives some room for management of deviations. If deviations can be justified through a safety analysis and review, an exemption may be granted. This could give enough time to fix the detected faults, and to avoid plant unavailability. The same kind of a “grace period” for equipment repair time can be set in OLC conditions.

For the GEN IV reactors, the more exotic coolants and fuels bring differences to the current LWR safety regulations, including the OLCs. As the initial setting of these is done in the licencing phase, good communication with the regulator is necessary to determine right limits without the risk of an unnecessary performance loss.

5 Operational challenges and historical operating experience

While the GEN IV reactor designs bring many benefits regarding efficiency and fuel use, there is a great number of known technological risks related to them. These include challenges related to reactor materials, maintainability, safe operation, fuel, and the fuel cycle. The causes and effects of these risks are analysed in this chapter.

Most of the historical predecessors of the modern GEN IV designs were engineering and performance demonstration reactors. Their purpose was to prove the proposed technologies and gain knowledge about their operation. As expected, these reactors utilized many innovative components and solutions, which sometimes proved challenging. Majority of the problems had a root cause in the reactor materials, either structural, fuel or coolant-wise. The extensive experience the operation of these reactors provided is nonetheless valuable, when addressing the similar modern designs.

5.1 VHTR

The identified principal challenges from literature and available historical operational experience data of the VHTR are displayed in Table 2 below.

Table 2. Identified VHTR principal challenges.

Issue	Cause	Effect
Helium contamination	Coolant impurities	Oxidation, Carburization / Decarburization of steels
Fuel element failures in pebble substrate or prismatic block	High variations of temperature, anisotropic thermal expansion of graphite	Fuel element wear, possible fuel leak or failure
Graphite dust	Wear of graphite components, such as fuel pebbles	Dust accumulation in primary circuit surfaces. Radiation protection challenge
Control rod shroud material integrity	High peak temperatures at reactor core	Need of heat-resistant protective control rod shrouds
Spatial oscillations of tall and thin cores	Core shape due to low power density and radial heat transfer requirement	Xenon-oscillations
Need of qualified heat-resistant alloys	Desired core outlet temperatures of 800-1000 °C	High temperatures accelerate corrosion and may cause mechanical overloads
Fuel qualification at high temperatures	Lack of demonstration	TRISO qualification and testing needed at accident-conditions (beyond 1600 °C)
Effectiveness of the passive heat removal system	Lack of demonstration	Difficulty in licencing
Air ingress due to a primary coolant system leak	Structural material failure / failure in pressure boundary	Air ingress leads to corrosion, creep and fatigue of structural materials
Water ingress due to a primary coolant system leak	Steam generator tube rupture	Water induces reactivity, increased corrosion, loss of primary heat sink due to need of draining the SG

5.1.1 Principal challenges

From the design and operation of historical high-temperature reactors, multiple principal challenges have been identified. The most notable challenge is the high temperatures the reactors operate on. The inlet temperatures range between 400-500 °C and core outlet temperatures between 750-1000 °C. The areas most affected are the intermediate heat exchangers (or steam generators) and the interconnecting hot gas duct. Currently, the maximum feasible core outlet temperature for longer-term operation is 750 °C, dictated mainly by the heat exchanger materials. The range of core outlet temperature is also used as classification between the VHTR and HTR. Typically, reactors operating at 700-900 °C are classified as HTRs and reactors operating over 900 °C are classified as VHTRs. (Zinkle et al, 2019)

The VHTR reactor pressure vessel typically needs to be cooled. For example, the Chinese HTR-PM uses a solution where the fissile core is surrounded by an inner graphite reflector and an outer brick layer held in a metallic core barrel inside the pressure vessel. The inner barrel and the pressure vessel are cooled by helium redirected from the heat exchanger cold leg and by an external reactor cavity coolant system. When the pressure vessel is cooled and kept at temperatures below ca. 370 °C, the use of standard PWR pressure vessel steels such as SA508 or SA533 is possible. If cooling is not possible, alloy steels with good creep resistance, such as 9Cr-1Mo, or ASME grade 91 steel are needed. Grade 91 steel is currently widely used in heat exchanger tubing at fossil plant applications. Melting and forging a large VHTR pressure vessel from these steels, however, is currently not possible. (Wright, 2019)

The other metallic components in the reactor coolant system are the hot gas duct and the intermediate heat exchanger. Also depending on the selected material, the heat resistance of the control rod protective shrouds need to be considered. Here the maximum permittable temperature follow the maximum core outlet temperature. For the contemporary maximum temperatures below 800 °C, Fe-Ni-Cr alloy 800H can be used. Alloy 800H has been used in the German pebble bed reactors, as well as the US FSV plant, and is ASME code qualified. Operation in higher temperatures of 750-1000 °C would require the use of nickel-based alloys, such as Inconel 617, Haynes 230 or Hastelloy X. These materials would require more thorough research and testing to be qualified for use. The main challenge with nickel-based alloys is the high volume of chromium, which causes corrosion problems. (Zinkle et al, 2019)

High temperatures make the reactor structural materials susceptible to degradation, which is caused by metal creep, fatigue and their interaction. In addition, corrosion by helium coolant impurities degrade metals. The VHTR coolant will inevitably have a non-negligible number of impurities. These include O_2, N_2 and H_2O leaking inside from seals and welds, and CO, CO_2, H_2 and CH_4 coming mainly from coolant interaction with hot graphite. Accumulation of these gases creates a possible fire hazard. Depending on temperature, impurity concentration and alloy composition, these impurities can react with metallic surfaces with oxidization, decarburization and carburization. These processes degrade the alloys, and cause losses in their load-bearing capabilities. Therefore, the functionality of the helium coolant purification system and leak tightness of the primary system is critical to the VHTR operability. (Zinkle et al, 2019)

Another problem with the primary system leak tightness is the possibility of water ingress from the steam generator. In case of a tube rupture, the water entering the primary circuit could have a possibility of increasing reactivity due to the increased neutron moderation. The large negative reactivity feedback from temperature would quickly halt the power increase. The first response to such incident would be an automatic scram combined with the draining of the steam generator. The residual heat would be designed to be removed through the reactor cavity cooling system, and the TRISO fuel should maintain its integrity. A large enough ingress of water could, in the worst case, increase the primary circuit pressure so that a pressure relief valve would have to open and drain the primary circuit from the helium coolant. The accident could lead to a serious radiological emission, since the relieved helium coolant containing various radioactive activation products would be difficult or impossible to be completely contained. (Chen, 2018)

In addition to metals, properties of the graphite moderator in the VHTR need to be considered. Operation at higher temperatures would require research and development of a new graphite type. Irradiation of graphite leads to changes in properties such as dimensions, dynamic Young's modulus, coefficient of thermal expansion and thermal conductivity. The effect of irradiation changes depending on temperature. The determination of these values relies heavily on irradiation testing, which is expensive. In addition, the purity requirement of nuclear-grade graphite is high, since any such impurities could lead to unwanted activation in the core, and possibly induce corrosive materials into the system. The number of manufactures of nuclear-grade graphite has fallen since the 1980s, and the availability of suitable graphite might become a problem. (Zinkle et al, 2019)

The use of graphite also brings a problem with spreading of graphite dust in the primary circuit. Graphite dust in high-temperature reactors is mainly generated via the wear of graphite components in the core, especially the fuel pebbles in a pebble-bed reactor. Due to the fuel circulation, the mechanical interaction between the fuel pebbles causes dust to form. The formed dust can accumulate on the primary system surfaces and affect their functions. For example, accumulated dust on the intermediate heat exchanger tubes might eventually cause a loss in heat transfer efficiency. The spread of this dust among the primary circuit via helium creates radiation protection problems in regular operation and maintenance activities, but also adds to the source term in the case of a loss-of-pressure accident. (Luo, 2017)

The design of TRISO particles aims to minimize the fission product leakage during both normal operation and accident situations. Fission products can, however, be released to the coolant and graphite dust via defective particles or diffusion at high temperatures. During the operation of the AVR pebble bed reactor in Germany in 1967-1988, a release of metallic fission products Sr-90, Cs-137 and Ag-110 from the fuel was observed. The release of these fission products was caused by both defective fuel and diffusion due to higher than anticipated operating temperatures. It is estimated that the fission product release with modern fuels in the AVR would have been lower, but still present. (Moormann, 2008)

The contemporary SiC-coated TRISO fuels have reached full technological maturity with routine commercial operations. The amount and availability of TRISO-manufacturers, however, could be an issue. Development of TRISO fuels is focused on improving the fuel safety in accident conditions. The current fuels can withstand temperatures up to 1600 °C with minimal leakage, but planned operation at higher temperatures would require qualification at higher temperatures. Research of TRISO fuels is focused on the particle inner fuel materials, such as uranium carbides, nitrides and silicides. The coatings under research are zirconium and niobium carbides. The main challenges with these types of new fuels are the increased susceptibility to oxidization and more difficult manufacturing. (Liu, 2025)

5.1.2 Historical operating experience

Historically, eight high-temperature experimental and demonstrator reactors have been constructed and operated. Both the prismatic block type (Dragon, Peach Bottom, FSV, HTTR) and the pebble bed type (AVR, THTR-300, HTR-10, HTR-PM) have been in operation. These reactors are presented in Table 3 below. The older, already shut-down reactors operated with varying operational performance, ranging from good to sub-optimal. The first constructed reactors, Dragon and Peach Bottom, reached relatively good availability in commercial operation with operation factors ranging from 60-80 %. The main goal of these facilities was to test the TRISO and TRISO-like fuel in operation. The largest US demonstrator, FSV, did not perform well economically. The reactor suffered from multiple primary coolant system leaks, which introduced water to the primary system. The damages were serious, but they did not prevent the reactor form resuming operation after maintenance. However, the repeated and long overhaul programs caused the plant to be shut down due to economic unviability. (Dunlap, 2025)

The German pebble-bed reactors operated with relatively average performance, averaging 50-60 % operation factors. The larger prototype, THTR-300, experienced fuel-related problems. Multiple fuel pebble failures were observed, and the fuel feeding system was jammed on one occasion. Multiple problems regarding graphite dust were also recorded. (Dunlap, 2025)

Table 3. Current and historical high-temperature reactors (Dunlap, 2025), (IAEA, 2025b).

Reactor	P [MW_{th}]	Type	Country	Operating period	Core outlet temperature [°C]	Load factor [%]	Operation factor [%]
Dragon	21.5	Experimental	UK	1965-1976	720	60.7	60.9
Peach Bottom	115	Experimental	USA	1967-1974	725	56.9	72.15
AVR	46	Prototype	Germany	1967-1988	950	62	65.9
FSV	842	Demonstration	USA	1979-1989	775	15.2	31.1
THTR-300	750	Prototype	Germany	1985-1989	750	41.3	56
HTTR	30	Experimental	Japan	1998-	950	-	-
HTR-10	10	Prototype	China	2000-	700	-	-
HTR-PM	2x250	Demonstration	China	2021-	750	27	72

The currently operating HTRs are located in Japan and China. The Japanese HTTR has been the longest-operating high-temperature reactor, used for contemporary studies regarding fuels and materials. The Chinese pebble bed reactors, HTR-10 and its scaled-up version HTR-PM have operated successfully as well. The HTR-PM represents the current state-of the art, sometimes being classified the first operational GEN IV reactor. The current plans are to scale up and improve on the design for the upcoming HTR-PM-600 reactors. (Dunlap, 2025)

According to IAEA PRIS data, the first year of commercial operation of the HTR-PM showed varying performance values. The operation factor was 72.05 %, the load factor was 20.66 %, the planned unavailability factor was 66.13 % and the forced loss factor was 8.25 %. This would indicate either very long planned outages, load-regulation, or possibly continued commissioning tests. Explanations on the operating experience for this year have not been made public, so the reasons behind the unavailability are unclear. (IAEA, 2025b)

5.2 SFR

The identified principal challenges of the SFR are displayed in Table 4 below.

Table 4. Identified SFR principal challenges.

Issue	Cause	Effect
Refuelling and fuel handling	In-sodium fuel identification and transport	Outage length, possible equipment failures difficult to repair
Maintenance of primary system equipment	Decontamination of sodium, primary systems difficult to access	Difficulties with maintenance
Remote operations inside the reactor vessel	Need of system leak tightness	Slow maintenance actions, reliability risks
Sodium leaks	Bad seals, failure of structural materials	Fires, possible spread of radioactive contamination
Cover gas purity	Primary system leaks, noble gas buildup	Corrosivity, sodium impurity
Sodium impurity	Intrusion of oxygen, water and hydrogen	Corrosivity, forming of oxide plugs
Strength of fuel cladding	Temperature, swelling, irradiation and high burnup	Strength deterioration, possible ruptures
Water intrusion to sodium systems	Steam generator tube ruptures	Fires, hydrogen formation
Air intrusion to sodium systems	Bad seals, failure of structural materials	Oxygen buildup in sodium, fires

5.2.1 Principal challenges

Compared to the other GEN IV reactor designs, the SFR has relatively low material-specific requirements for the primary system components. The sodium coolant is not inherently corrosive, and the low operating pressures allow the use of thinner-walled vessels. The main concerns regarding materials come from the elevated temperatures and desired high fuel burnup. The core outlet temperatures in SFRs are in the range of 500-550 °C, which corresponds to maximum fuel cladding temperature of 650-700 °C. The fuel burnup is expected to reach 150-200 MWd/kgU on average. High burnup levels lead to problems related to fuel swelling, as gaseous fission products cause an increase in the fuel pin pressure. The same effect is present in LWRs, but the possible presence of mixed oxide fuel increases the amount of fission gas release and pressure buildup. These, combined with the expected high neutron doses inflicted set the requirements for the fuel and cladding material. (Zinkle et al, 2019)

The cladding and fuel shroud (duct) should have a low neutron absorption cross-section, low metal swelling properties, high ductility and good creep strength. In the past, austenitic stainless steel has been used as cladding material, but experience has shown problems with radiation-induced void swelling. Possible austenitic steels suitable for the SFR are titanium and niobium stabilized austenitic steels, such as 316 and 304 stainless steels. One alternative is to use ferritic steels, such as HT9, but they tend to have problems related to radiation embrittlement. (Liu, 2025)

The corrosion effects that the coolant has are caused by the dissolution of metal alloying elements from structural materials and sodium impurities such as oxygen. The most notable elements dissolving to coolant are Cr, Mn, Ni and Si. As the coolant dissolves the outer layer of structural materials, a ferritic layer is left on the steel surface. The dissolved ions react with the oxygen present in the sodium coolant, forming corrosion products such as chromium oxide ($NaCrO_2$). The formed corrosion products are transported through the system, depositing on the colder parts as their solubility is a function of temperature. The deposition of corrosion products causes a reduction in load-bearing capability, transport of radioactive materials and possibly form blockages. As the corrosion rate of steels is mainly caused by the oxygen concentration, oxygen control is the main task of the SFR sodium purification system. (Zinkle et al, 2019)

Small amounts of oxygen mixed in the sodium coolant form sodium oxides. These oxides typically have lower saturation temperatures than pure sodium, so different temperature-based filters can be used in purification. The SFR coolant purification system typically consists of plugging indicators, removable coarse filters and cold traps. The plugging indicators are used for online measurement of oxygen content. They function by cooling the sodium until the sodium oxides start crystallizing and decreasing the meter flow rate. The cold traps function in a similar way. A portion of the main coolant flow is redirected to a cooler and then a filter, where impurities are caught and deposited. In addition to oxygen and metallic impurities, the SFR needs detection systems for cover gas and hydrogen contamination. Cover gases, such as argon or helium are used in the upper parts of the reactor vessel. The gas seals the primary system and prevents the intrusion of air into the reactor coolant. The gas can be contaminated by possible leaks from outside of the reactor, and in time by gases released from the reactor. When the number of impurities in the cover gas exceeds acceptable values, it needs to be replaced. The detection of hydrogen is important in the early detection of a water leak into the sodium system. The reaction between water and sodium produces hydrogen, so precise detection is needed to eliminate the leak as quickly as possible. (IAEA, 2013)

The sodium reactions are primarily of interest in the secondary sodium circuit. The possibility and risk associated with these are also the reason of the introduction of the secondary sodium circuit in the design, as the aim is to prevent the release of radioactive material from the primary circuit. The water-sodium reaction can occur if a steam generator tube ruptures. The reaction releases notable amounts of heat and hydrogen. If the leak is large enough, this causes a pressure shock to the secondary system. Consequences may be failures of the secondary system components and possibly the intermediate heat exchanger tubes, in case the pressure damages them. The risk of an immediate large tube break is less than that of a smaller, slowly propagating one, highlighting the importance of detection. The secondary circuit is also the most likely place of a sodium leak outside the system, since it contains the largest total area of sodium piping. A large sodium leak leads into combustion when in contact with air. This can cause mechanical failures due to heat and overpressurization. Mitigation measures against these leaks include early detection and loop draining, as well as proper sizing and design of containment structures able to maintain integrity even if a leak occurs. (Ruggieri, 2017)

Operation-wise, the most notable challenge comes from the maintenance and fuel-transfer actions inside the primary system. Understandably, the SFR cannot be refuelled or maintained such as an LWR, where the pressure vessel can be opened. While filled with sodium, all actions inside the pressure vessel need to be conducted remotely. This requires special inspection and handling devices designed to operate in sodium. If any in-vessel components, such as the fuel, sodium pumps of intermediate heat exchangers need to be moved outside the vessel, special in-sodium transfer equipment is needed. In addition, the components need to be cleaned from sodium before any maintenance actions can be done, or before the spent fuel can be stored outside sodium. This causes a variety of problems, most notably the length of a refuelling outage. (IAEA, 2013)

5.2.2 Historical operating experience

Several sodium-cooled fast reactors have been built in the past. Most of these were experimental- and demonstration reactors, but a few commercial-scale reactors intended for power generation have been constructed as well. From these, the most recent larger-sized units are selected for assessment. A comprehensive table of currently operating and historical SFRs is included in Appendix 2.

The largest sodium-cooled fast reactor in the United States was the 400 MW_{th} fast flux test facility (FFTF), which operated from 1974 to 1994. FFTF had no capability to generate electricity but was used for material testing and irradiation purposes. The operating experience of this facility was overall positive. The annual operation factors (by reactor availability) were in the range of 60-80 %. Some noteworthy incidents have been recorded in literature. During the operation, a total of 12 fuel ruptures occurred resulting in sodium contamination, 11 of these happened in experimental fuel assemblies. The ruptures caused radiation protection challenges related to the fission product caesium, which deposited on the cool cover-gas filled areas of the reactor. Other noted incident was a problem related to the fuel transfer system, which included transfer ports in the reactor vessel head. One of these ports malfunctioned due to a tilted and stuck nozzle. These transfer ports were surrounded by steel-cased uranium shield rings. One of these casings had ruptured, and the leaked uranium oxidized and swelled. This swelling caused the tilt and jamming. The leaking shield was removed, and the issue didn't occur again. (Swaim, 1991)

The first larger French SFR demonstration reactor was the 563 MW_{th} Phenix, which operated from 1973 to 2009. During its active power production period of 1974-1990, the average load factor was 59.6 %. The loss of availability was caused by several operational incidents, caused by sodium leaks, water-sodium reactions, intermediate heat exchanger failures and fuel cladding failures. The longest production losses were in 1976-1977, caused by a leakage in the intermediate heat exchanger and in 1982-1983, when a total of four sodium-water reactions occurred in the steam generators. (IAEA, 2013)

The largest SFR built to date was the 3000 MW_{th} Superphenix, which operated from 1985 to 1997. The Superphenix was purposed to be a demonstrator of commercial SFR viability, but it faced challenges in its operation and public acceptance. During the operation period, Superphenix was in regular operation for 53 months, 25 months in maintenance and 54 months with no authorization to operate. If excluding the administration-caused unavailability, this resulted in an average operation factor of 23 %. The overall durations of unavailability by system were: 1035 days for the reactor, auxiliary systems and the steam generators, 191 days for the power conversion systems and 54 days for the instrumentation and control. The long unavailability by unplanned maintenance was caused by three major operational events. (Guidez, 2017)

The first event happened in 1987, a sodium leak occurred in the fuel transfer storage tank, causing a 10-month outage. This tank was used analogously to LWR fuel pools, where spent fuel is stored after being removed from the core. The leakage happened in a weld of the storage tank cooling circuit, leaking sodium to the surrounding argon-filled safety vessel. The material used there was ferritic steel (15D3), which had experienced embrittlement by impurity hydrogen and was later evaluated as unfitting for sodium circuits. The second event was an 8-month outage caused by oxygen impurities in the sodium. The impurities were caused by a leak in the reactor argon cover gas system, which at that time, had insufficient purity monitoring. The resulting coolant purification took multiple campaigns to complete, challenge being the location of the filters inside the reactor vessel and the accumulated large number of impurities in coolant. The third event was a 7-month outage caused by an argon cover gas leak in the intermediate heat exchanger. The leak occurred on the argon seal located between the reactor vessel hot and cold sodium pools. The cause of the leak was again determined to be a defective weld but caused by a single manufacturing issue that was not observed elsewhere. The repair, however, had to be conducted remotely, which required long planning and execution. (Guidez, 2017)

The latest experience from large SFR operation come from the Russian BN-600 and BN-800 reactors. Both reactors are currently in operation, BN-600 started production in 1980 and BN-800 in 2016. After the early operations, the average operation factors have stabilized to the range of 60-80%, mainly affected by the reasonably long planned outages twice a year. The BN-600 has a well-documented history of operational experience. In the early ears of 1980-1994, the reactor experienced 12 water-sodium reactions in the steam generators and 27 direct sodium leaks, all but one of them in the secondary sodium circuit. The leaks caused fires and temporary unavailability but were able to be supressed without major damage occurring. (IAEA, 2013) Figure 11 below shows the IAEA operational experience database data for the causes of unplanned outages by system for the BN-600 and BN-800 during their entire lifetime.

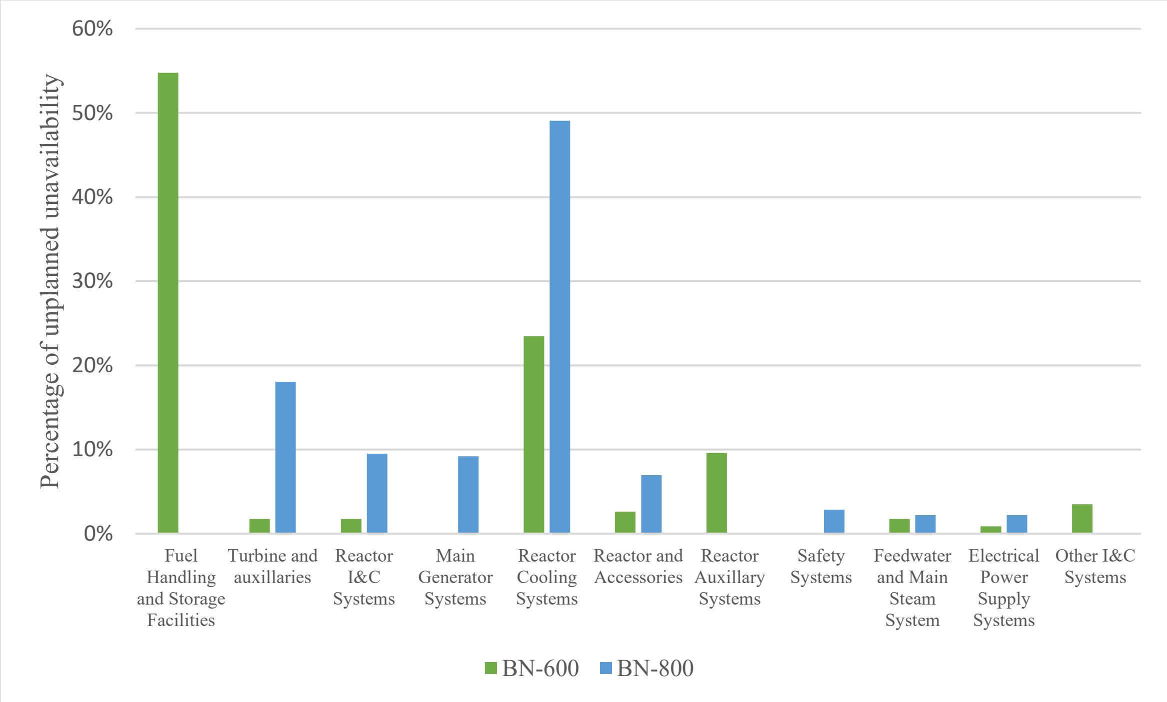

Figure 11. BN-600/800 Lifetime average unplanned unavailability by system (reproduced from IAEA, 2024a).

The main causes of unplanned outages for both these reactors come from the reactor island of the power plant. The fuel handling and reactor cooling systems have caused the most problems. This refers to the in-sodium fuel transfer equipment, intermediate heat exchangers and the secondary sodium circuit. The experience from these reactors shares similarities with the Phenix and Superphenix, except for long outages that have been avoided in Russia. This can be explained by the long operational history of the BN-600, providing not only valuable feedback for operations and maintenance, but also the experimental nature of these facilities. It is quite likely that the power production strategy of these reactors is not conducted in a similar fashion as the rest of the Russian reactor fleet.

The operational experience data for planned outages shows that for an outage for inspection, maintenance or repair combined with refuelling, the BN-600 has lost 1337 hours on average per reactor year during its lifetime. If this is assumed to belong to the annual outages, the planned outage days per year is 56. For BN-800, the amount is 76 days. (IAEA, 2024a)

5.2.3 Estimation of SFR performance values

An IAEA document "Fast Reactor Database 2006 Update" (IAEA, 2006) has listed design values for lengths of routine shutdowns for experimental, demonstration and commercial SFRs. This value can be used to estimate how long an average SFR outage would last, and then what could be an average load factor. The values from the database data, along with calculated effective full-power days (EFPD) and load factors are presented in Table 5 below.

Table 5. Estimated SFR operation factors from historical designs (Adapted from IAEA, 2006).

Reactor	Mean length of routine outage [days]	Mean length of run [days]	Planned shutdown days / year	EFPD	Load factor [%]
Commercial					
Superphenix	120	640	68	297	81 %
BN-800	17	140	44	321	88 %
Demonstrator					
Phenix	7	90	28	337	92 %
PFBR	22	240	33	332	91 %
Monju	30	148	74	291	80 %
PFR	21	90	85	280	77 %
BN-350	10	105	35	330	90 %
BN-600	15	160	34	331	91 %
Experimental					
Rapsodie	10	80	46	319	88 %
FBTR	7	60	43	322	88 %
JOYO	16	60	97	268	73 %
BOR-60	45	100	164	201	55 %
EBR-2	7	49	52	313	86 %
BR-10	12	100	44	321	88 %
CEFR	14	73	70	295	81 %
Average	24	142	61	304	83 %

Based on the values of designed outage lengths and mean run lengths, the average value of planned shutdown days per year is 61, which corresponds to 304 effective full-power days and a load factor of 83 %. With an improbable assumption there would be no production losses due to failures, load regulations or else, the SFR load factor would be similar to the world power reactor average (81.5 %), but considerably lower than with the top performing 10 % (94.4 %). (WNA, 2015)

5.3 LFR

The identified principal challenges of the LFR are displayed in Table 6 below.

Table 6. Identified LFR principal challenges.

Issue	Cause	Effect
Corrosion of structural materials	Physical properties of lead	Loss of structural material strength, corrosion product formation
Coolant slagging by lead oxides	Too high oxygen content in coolant	Flow blockages, possible fuel overheating
Steam generator integrity	Wear, corrosion	Water ingress, secondary circuit contamination
Lead solidification	Failure of coolant heating system	Material stress, possible fuel damage
Remote operations	Lead opacity, system leak tightness	Difficult and slow maintenance and refuelling
Coolant impurities	Primary system leakages	Corrosion, oxide formation, loss of material strength
Personnel protection	Lead toxicity	Difficulties in maintenance activities
Radiation protection	Production of polonium, if LBE coolant is used	Increased doses, difficult maintenance activities

5.3.1 Principal challenges

Compared to sodium, lead is notably more corrosive. When in contact with structural materials, lead causes effects such as material dissolving, embrittlement and thermal transport of mass. A difference between lead and sodium is that stable protective oxides can form on LFR steel structures in right circumstances. As the protective layer is formed, there is a balance between dissolution of alloys by lead and formation of protective elements that prevents dissolution. The coolant chemical control system of the LFR must then keep a balance between having a high enough oxygen concentration to prevent dissolution but low enough to prevent lead oxide formation, or "slagging." The formed lead oxides form solid masses than can deposit on any primary system surfaces and cause coolant blockages. The corrosiveness of lead sets the limit to maximum core outlet temperatures. The optimal temperature range for protective oxide formation is 300-470 °C, temperatures between 470-550 °C show a transition from oxide formation to dissolution, while temperatures above 550 °C show high dissolution. Operation at lower temperatures can be reached with current material technology, but higher temperatures would require more advanced materials or resistant coatings to current steels. This is needed as no protective oxide layers would form in high temperatures. (Zinkle et al, 2019) An overview of the protective oxidization related to core outlet temperatures is presented in Figure 12 below.

Effective corrosion protection	Transition zone	Additional protection needed
Compact stable oxide barrier on ferrite/martensite and austenite	Oxide formation on ferrite/martensite	Unstable metal oxide layer
	Mixed corrosion mechanism: oxi-dation / dissolu-tion on austenite	Stable FeAl alloy coating

400°C 500°C 550°C 600°C

Figure 12. Protective action via controlled dissolved oxygen at increasing temperature (Smith, 2010).

Typical structural steel for the LFR reactor vessel is austenitic steel, such as AISI 316L. More special materials, such as aluminized steel, are needed for components that are expected to undergo high erosion. One example of this are the primary coolant pump impellers. The fuel cladding experiences similar problems as other reactor structures. Typical design materials for cladding are ferritic/martensitic steels with 9-12 % Cr content, the historical Russian LBE-submarine designs used a 11Cr steel EP823. The operating temperature window for these steels is 350-550 °C, lower limit set by irradiation hardening and embrittlement, and upper by rapid dissolving by coolant. The LFR fuel cladding needs more thorough testing and development, since the technological readiness for corrosion resistant and high-burnup tolerable fuel is quite low. (Zinkle et al, 2019)

Since lead is chemically inert with water, the majority of LFR designs do not use an intermediate circuit between the reactor and the steam generator such as in the SFR. The event of steam generator tube rupture still poses some risks. Water or steam ingress to the core might induce positive reactivity and primary system overpressurization. If the steam generator fails to be isolated after such an incident, lead from the primary system might leak to the secondary systems of the power plant. This would be a challenging regarding contamination and cleanup, similarly to a PWR primary-secondary leak. (Alemberti, 2010)

The LFR needs reliable systems for coolant heating during reactor shutdown. This is especially necessary if LBE coolant is not used, since the freezing point of pure lead is high.

Coolant solidification leads to formation of blockages and disruption of decay heat removal. A complete coolant solidification can cause excessive material stress and failures, possibly rendering the reactor inoperable for a long or permanent shutdown. From the view of accident management, however, complete (and isolated) lead solidification might even be beneficial. (Alemberti, 2010)

From the operating perspective, the most notable challenges of the LFR come from the need of utilizing remote operations inside the primary circuit. The challenges are very similar as in the SFR, the need of system leak tightness and coolant opacity causing them. A benefit compared to sodium is the lesser need of cleaning and decontamination of equipment, as lead is inert with air. Additional challenges come mainly from the radiation and personnel protection, as lead coolant is extremely toxic. This, coupled with the presence of polonium if LBE coolant is used, brings a notable challenge to all maintenance activities not performed remotely. The shutdown temperature of the reactor is also an issue with the LFR. Compared to sodium, which has a melting point of 97.8 °C, lead and LBE have melting points of 327.5 °C and 123.5 °C, respectively. Since coolant solidification must be avoided, all maintenance actions must be done in coolant temperatures above the melting points. This causes issues with equipment used in the primary circuit maintenance.

5.3.2 Historical operating experience

Despite the similarities with the SFR, only a few lead-cooled fast reactors have been built in the past. In the 1950s, the LFR was selected by the Soviet Union as a possible submarine propulsion reactor. The high-power density allowed building small and lightweight reactors needed for submarines faster than the LWR counterparts. All LFRs built to date are Russian military-purpose land prototype and submarine reactors, and very little is known about their design or operating experience. All these reactors used LBE-coolant, so technically no reactor operating with pure lead has been operated. A list of these historically operated reactors is presented in Table 7 below. The first LFR was the 70MW$_{th}$ prototype reactor 27/VT, which was put to operation in 1959. This reactor was a land prototype for the coming submarine K-27, which experienced a partial meltdown accident in 1968. The second generation of these submarines used 155 MW$_{th}$ reactors. Seven submarines were constructed along with one land prototype. All these reactors experienced similar issues related to polonium contamination and coolant slagging. (IAEA, 2013)

Table 7. Historical lead-cooled fast reactors.

Reactor	P [MW_{th}]	Type	Country	Operating period	Note
27/VT	70	Experimental	Russia	1959-?	Land prototype
K-27	2x73	Submarine	Russia	1963-1968	Partial meltdown
KM-1	155	Experimental	Russia	1978-?	Land prototype
BM-40A	155	Submarine	Russia	1971-1996	4 produced
OK-550	155	Submarine	Russia	1971-1996	3 produced

The land prototype 27/VT experienced multiple incidents, these included lead oxide slagging in the core and coolant leakages. The operation of the reactor showed that the depressurization of the reactor during refuelling outages caused lead oxide formation due to air ingress to the core. This, in addition with leaking oil seals from primary circuit pumps caused a deposition of slags in the core, which caused flow blockages and local temperature increases in fuel elements. No fuel damage occurred, but the incidents required extensive repairs. The primary coolant leakage was caused by auxiliary pipeline corrosion due to moisture ingression from defective welds. This caused a leakage of 20 tonnes of coolant to the reactor compartment, which caused a long and difficult outage due to cleanup and radiation protection measures due to polonium. (Troyanov, 2023)

These faults with the land prototype carried on to the first submarine K-27. These reactors had no cover-gas system and no coolant purification systems, which were introduced only for the next generation of protypes. Similarly to the prototype facility, other of the two parallel K-27 reactors became slagged with lead oxides during operation. Due to operator error following the slagging, the reactor partially melted down and resulted in death of 9 crew members and the discontinuation of this submarine. (Troyanov, 2023)

The next land prototype KM-1 introduced a coolant purification system, which functioned by hydrogen regeneration. Use of this system eliminated the problems related to slagging, but experience showed that total removal of oxygen from the coolant was also harmful. This showed the need of balancing between problems with slagging and corrosion. The KM-1 reactor and the succeeding seven submarines operated with varying success, similar issues persisted with coolant leaks, causing difficulties with radiation work. The overall difficulties and expenses compared to LWR submarines caused an early termination of the program. The Russian program resulted in operation history of 15 cores with about 80 reactor-years of operating time. (Troyanov, 2023)

5.4 MSR

The identified principal challenges of the MSR are displayed in Table 8 below.

Table 8. Identified MSR principal challenges.

Issue	Cause	Effect
Corrosion of structural materials	Corrosion by salt and fission products	Loss of structural material strength
Remote operations	Radiation due to circulating fuel	All inspections and maintenance where fuel circulates must be remotely conducted
Online fuel processing	Difficult separation of fissile materials and fission products	System availability, reactivity and radioactivity hazards
Heat exchanger integrity	Material failures	Water intrusion, secondary circuit contamination
Control of salt composition	Faults in redox control or salt processing systems	Corrosion of structural materials
In-core instrumentation	Corrosion by salt / heavy irradiation in fast MSR	Availability of in-core measurements
Embrittlement	Irradiation	Nickel alloy degradation at high temperatures
Tritium control	Formation of large amounts of tritium	Permeation, possible spread to secondary circuit
Waste management	Solid and gaseous fission products and irradiated structures	No disposal methods developed yet
Graphite wear	Irradiation	Dimensional changes and degradation, possible short lifetime of moderator

5.4.1 Principal challenges

The corrosivity of salts, and its effect on the reactor structural materials is a major concern in the MSR development. Most typically selected carrier salt, fluoride salt, dissolves structural steel alloying components. When steel is exposed to a fluoride salt, the alloying chromium will dissolve at the fastest rate, with iron and nickel being more resistant. For this reason, MSR designs favour steels with low chromium and high nickel content, such as Hastelloy N used in the MSRE, or steels that have been coated with a dissolution-resistant material. The reaction between fluorine and chromium should be avoided, so a chemical control method such as reduction-oxidation (redox) control needs to be used. Different redox control methods include gas phase additions, metallic additions and salt chemistry altering. Redox control works by controlling the fluorine potential through introducing materials that fluorine preferentially reacts with over chromium. These include hydrogen gas, zirconium and beryllium. The downside of this kind of corrosion control is the deposition of the redox materials to structural material surfaces, and possible neutronics-affecting properties that the materials have. The dissolution of metal is also affected by factors such as moisture ingress from outside the system, or certain fission products such as tellurium. The effect of these set requirements for the salt control and purification system. (Zinkle et al, 2019)

Another problem is the management of produced tritium. Tritium is formed from neutron absorption reactions of lithium and beryllium used in the primary salt. Tritium permeates structural materials, which is of concern in the thin-walled primary heat exchangers. This could be one possible radiation release path, as tritium would eventually reach the power conversion circuit and possibly the environment. In addition to tritium, another problem is the lifetime of the graphite used in thermal MSRs. The combined effect of irradiation and salt penetration to graphite pores cause losses in structural strength. In some design cases, the estimated replacement interval for the moderator graphite is as low as 4 years. (Serp, 2014)

One specific issue that is yet to be solved is the handling of fission product waste. The majority of the radioactive nuclides are not stored in sealed fuel elements, but are spread across the plant. This is an issue both for safety and security. The MSR needs a system or a facility for storing the gaseous and solid fission products. The gaseous fission products are typically designed to be removed online via an off-gas system. The solid fission products are either designed to be separated from the salt or disposed along with it. Both these would require separate storage and handling systems and similar material requirements as in the primary circuit would apply here too. (IAEA, 2024b)

IAEA technical report “Status of molten salt reactor technology” (2024), lists the following topics among the most important challenges in MSR development: supply chain, regulation, fuel salt waste disposal, safeguards and security, and operations and maintenance. The challenges related to supply are linked to the availability of manufacturing MSR-specific components, mainly caused by material requirements. The fuel supply challenges are limited by the designed use of the reactor. MSR initial cores might need higher enriched fuel, which has limited availability. After that, the possible use of thorium or transuranic fuel brings supply challenges as well. The regulatory challenges are related to the unfamiliarity with this reactor type. Licensing an MSR would require thorough safety evaluation, but suitable tools to do so might not be available. In addition, the MSR lacks industry standards for design safety, fuel transportation, salt composition and fissile material accounting. The MSR is currently lacking a disposal path for fuel salt wastes. The composition of the wastes, depending on the reactor operation and possible fission product separation during operation is also uncertain. Challenges regarding safeguards and security come from the non-traditional composition of fuel and the fuel reprocessing on-site, again related to fissile material accounting. (IAEA, 2024b)

The main challenges regarding operations and maintenance are caused by the need to develop completely remote maintenance and decontamination methods due to high radioactivity of the primary salt circuit. If the reactor is designed to operate with a circulating fuel salt, all systems containing the primary salt are going to be dangerously radioactive. This includes the reactor, primary coolant pumps and heat exchangers, fuel processing facility and the waste storage. In addition to irradiation resistance, the maintenance equipment needs to be temperature resistant as well. The high melting point of coolant (400-500 °C) brings similar issues in maintenance actions as in the LFR. (IAEA, 2024b)

5.4.2 Historical operating experience

The development of the molten salt reactor begun in the USA in late 1940s with the aircraft propulsion program. The program resulted in the construction of the aircraft reactor experiment (ARE), which became operational in 1954. ARE was a 3 MW_{th} reactor using liquid fluoride salt fuel and sodium coolant, separated by corrosion-resistant Inconel tubes. The reactor used a solid beryllium oxide moderator. ARE operated only for four days, serving as a physics demonstrator for the technology. After ARE, the aircraft propulsion program was terminated, but research on molten salt technology continued with the molten salt reactor experiment (MSRE). MSRE was a 7.4 MW_{th} reactor, which operated from 1965 to 1969. MSRE used fluoride salt fuel and a solid graphite moderator, with the fuel salt operating also as the coolant. The development of the molten salt reactor continued after the discontinuation of the MSRE, but no more functioning reactors were built. In more recent times, interest in the MSR technology has grown in China. The Chinese R&D program focused on the thorium fuel cycle has resulted in the construction of the Chinese thorium molten salt reactor (TMSR), which was put to operation in 2023. Information on the early operations of this reactor are extremely limited. A table of historically operated MSRs is displayed in Table 9 below. (Dunlap, 2025)

Table 9. Historical molten salt reactors. (Dunlap, 2025)

Reactor	P [MW_{th}]	Type	Country	Operating period
ARE	2.5	Experimental	USA	1954
MSRE	7.4	Experimental	USA	1965-1969
TMSR-LF1	2	Experimental	China	2023 -

The MSRE has a well-documented history of operating experience, but as a small test reactor, not all of it is directly applicable to industrial-sized plants. MSRE experienced a few issues during its run. The reactor vessel became progressively embrittled due to neutron irradiation, and the structural steel Hastelloy N experienced fission product caused surface cracking. These kinds of phenomena would not be tolerated for a large power reactor with a long lifetime. Less notable problems included fuel pump oil leakages and freeze valve cracks. The MSRE reactor system was housed in a sealed containment and all maintenance activities there were planned to be done remotely. In case of failures, the individual components were planned to be removed and replaced completely, but no such maintenance had to be conducted during the operating years. Curiously, the recorded dose rate levels in the reactor containment were hundreds of sieverts per hour during the operation, and tens during shutdown. This would mean that no maintenance would have been able to be done in-person. (Holcomb, 2017)

6 Operational risk assessment methods

The design of a nuclear power plant should result in a system that is reliable both in terms of nuclear safety and plant operability. In the traditional nuclear safety design, it is often required that a comprehensive probabilistic risk assessment (PRA) is conducted prior to plant licencing and kept up to date during the operation. The PRA focuses on accident scenarios, concretely the level 1 PRA is defined as “determining the sequences leading to fuel damage and estimating their probabilities”. (STUK, 2019b) Using similar reliability engineering calculations, methods such as reliability and maintainability analysis are more motivated by the financial aspects of system failures. The goal of these methods is to reduce the plant downtime through eliminating possible causes of unplanned unavailability.

The article “Solving the issue of reliability data for FOAK equipment in an innovative nuclear energy system” (Chen et al, 2018) addresses these issues from the perspective of the Chinese HTR-PM demonstrator. The article states that (innovative) first-of-of-kind equipment is unlikely to have sufficient data on reliability statistics, being either nonexistent or unavailable. One possible solution would be to test the equipment with a lengthy reliability experiment, but this is often not possible in a limited timeframe. This creates challenges both for the formulation of the PRA and the reliability and maintainability analysis.

As the state of reactor development advances from conceptual design to detailed design and licencing, the methods that can be used for addressing safety and reliability issues shift from mainly qualitative to quantitative means. There are several methods designed for safety and reliability assessment suitable for different states of reactor design and maturity. An overview of different industry-standard safety assessment methodologies has been provided in “Development of a technology-inclusive methodology to analyze the environmental, safety, and health risks associated with advanced nuclear reactor designs as demonstrated on the molten salt reactor experiment” (Chisholm, 2020). These are presented in Figure 13 below. As the design maturity develops, the tools that can be used shift from more simple process hazard analysis (PHA) methods towards a fully quantitative PRA. The progression of these methods is presented by arrows, many of which indicate an iterative approach.

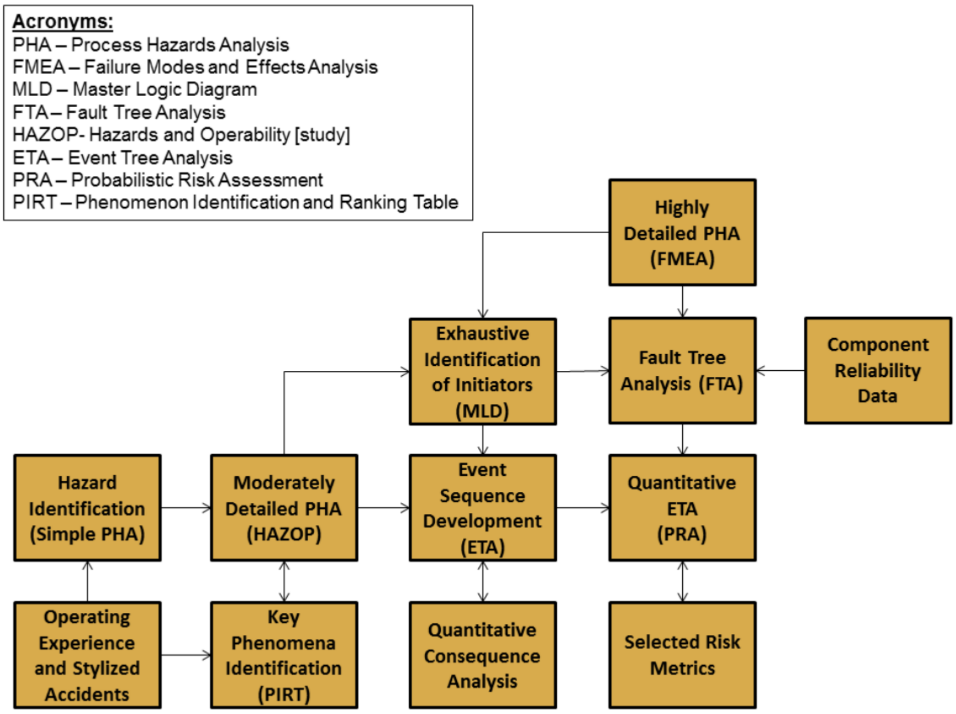

Figure 13. Overview of industry-standard safety assessment methodologies (Chisholm, 2020).

The different safety assessment methods in this overview can be seen as “building blocks” with the goal of building a full PRA. Simpler PHA methods that are suitable for reactors in earlier design stages include methods such as the “what-if” analysis. Going further, studies such as the hazard and operability study (HAZOP) and the phenomena identification and ranking table (PIRT) method can be used. These two are used for understanding the accident phenomena and hazard scenarios and can provide initial information for an event tree analysis (ETA). If addressing a more mature and detailed system with available component reliability data, methods such as a system-level failure modes and effects analysis (FMEA) can be formed and further used in a quantitative fault tree analysis (FTA) and the PRA. (Chisholm, 2020)

6.1 Phenomena identification and ranking table method

The method selected for use in the preliminary operational risk assessment in this thesis is an applied phenomena identification and ranking table. PIRT is systematic method used for gathering information of a specific issue and ranking the information based on their importance and the level of knowledge. Originally, PIRT was designed to be a decision-making tool in nuclear safety design that systematically identified and ranked physical phenomena dominating the response of a reactor system in a postulated accident scenario. Since then, PIRT-studies have been extended to other contexts, such as prioritization of research needs and evaluation of concepts through a single key parameter, or figure-of-merit (FOM). The FOM is a measurable value or a criterion that is used for determining the importance of the selected phenomena. (NEA, 2018)

A PIRT is typically performed with the help of a panel of experts, similarly to a HAZOP study. The PIRT conducted in this thesis is not formulated with an expert panel, but the method accuracy is improved by the extension of the knowledge and uncertainty parameters. Reason for excluding the panel is the scope and schedule of this thesis, along with an understandably low amount of available experts with knowledge regarding the selected issues.

6.2 Method procedure

A standard procedure for the PIRT-study adopted by the US NRC is typically divided to nine steps. These are presented in Figure 14 below.

- **Step 1**—define the issue that is driving the need for a PIRT;
- **Step 2**—define the specific objectives for the PIRT;
- **Step 3**—define the hardware and the scenario for the PIRT;
- **Step 4**—define the evaluation criterion;
- **Step 5**—identify, compile, and review the current knowledge base;
- **Step 6**—identify plausible phenomena, that is, PIRT elements;
- **Step 7**—develop importance ranking for phenomena;
- **Step 8**—assess knowledge level (KL) for phenomena; and
- **Step 9**—document PIRT results.

Figure 14. The Nine-Step PIRT Process (Ball, 2008).

First, it is defined why the PIRT-study is conducted and how it will be used. After this the exact scope of the study is formulated, a specific reactor design is chosen along with a scenario, a figure of merit is selected as an evaluation criterion, necessary data is collected, all possible phenomena are identified, importance and knowledge levels are assessed and finally results are documented and analysed.

6.3 Definition of issues and objectives

This PIRT study is conducted as a preliminary operational risk assessment for the selected four GEN IV reactor types. The results can be used as a guide to assess individual reactor designs, or specific vendor design choices. A secondary goal for the study is to function as a possible input for further research on GEN IV reliability or safety issues.

The objectives of this PIRT are:

1. Identification of the most important operability-related phenomena of the selected reactor designs
2. Ranking the phenomena based on their importance and knowledge level
3. Formulating preliminary risk assessment matrices to be used in further studies

6.4 Hardware and scenario definitions

The four GEN IV reactors selected for this study are the VHTR, SFR, LFR and MSR. These four reactors are assessed at a general level, without going into assessment of (contemporary) individual reactor designs. This allows the scope of the study to be manageable, with wider application possibilities of the results.

The basic scenario for the study is an event causing plant unavailability. This includes unplanned unavailability from system failures, anticipated operational occurrences (AOOs), and limiting conditions of operation. The number of phenomena related to these scenarios is high at any reactor design, so a limiting requirement is exclusiveness to the individual design.

6.5 Evaluation criterion

With the focus on plant operability and reliability, the evaluation criterion must be a power plant performance indicator. The chosen figure-of-merit is the plant load factor. The phenomena that most affect this through long downtimes are the most important. Since the load factor is affected by the total electricity generation, smaller events such as power reductions affect this as well. A secondary evaluation criterion is the system maintainability after a failure. While low maintainability directly affects the load factor, it may have other effects such as large maintenance costs, reduced equipment lifespan, and safety issues.

6.6 Definition of importance and uncertainty ranking

When a typical PIRT-study is conducted, the panel of experts is asked to rank the identified phenomena. This is usually done in a simple manner, due to the scope and nature of the method. The importance and knowledge are typically ranked by “levels” ranging from low to high. When the size of the panel is large, a typical three-level ranking of “low-medium-high” is used. (GIF, 2011) Since a wide panel of experts is not used in this study, the importance and knowledge ranking are expanded to achieve more accurate results.

The importance ranking was expanded with methods from the failure mode and effects analysis (FMEA). FMEA is a reliability engineering tool used to quantitatively estimate risks at a system level. FMEA identifies system failure modes by looking at possible sub-system failures and then determines their effects and severities. This study is typically done in the system design phase to possibly eliminate failures in the process before the product reaches the customer. (Stamatis, 2019)

A typical FMEA-study ranks failure modes with the parameter’s severity, occurrence, and detection, and then combines these to an index called the risk priority number. In this study, the risk priority number is used as the importance level in the PIRT ranking. The Risk priority number is calculated as follows:

$$RPN = S \cdot O \cdot D \qquad [7]$$

Where RPN, is the risk priority number, S is the severity rating, O is the occurrence rating and D is the detection rating. A scale of 1-5 is used for all these ratings. (Stamatis, 2019)

For determining the severity rating, defining meters are needed for how serious the phenomenon (or failure mode) is regarding to the plant operability. Following the selected figure-of-merit, the meter for severity is the load factor. If the failure mode causes a major effect in operation or a long unavailability of production, a high severity rating is assigned. The definition of the severity rating is displayed in Table 10 below.

Table 10. Severity definition.

Rank	Definition	Meter
Rank 5	Major effect in operation, or causes long unavailability of production	> 5% Decrease in load factor
Rank 4	Moderate effect in operation, or causes unavailability of production	> 2.5% Decrease in load factor
Rank 3	Minor effect in operation, or causes loss of efficiency	> 1% Decrease in load factor
Rank 2	No effect in operation, or causes short loss of efficiency	> 0.5% Decrease in load factor
Rank 1	Negligible effect on operation	No effect load factor

For the occurrence parameter, the determining factor is how often the failure occurs in the design lifetime. For the selected phenomena, a high rank is assigned if there are multiple cases of occurrence from historical experience or the failure can be expected to happen. The definition of occurrence rating is displayed in Table 11 below.

Table 11. Occurrence definition.

Rank	Definition	Meter
Rank 5	Phenomenon has certain or inevitable occurrence	Multiple cases from historical experience
Rank 4	Phenomenon has frequent occurrence	Some cases from historical experience
Rank 3	Phenomenon has occasional occurrence	A few cases from historical experience
Rank 2	Phenomenon has minor occurrence	Individual cases from historical experience
Rank 1	Phenomenon has remote occurrence	Probability acknowledged

The detection parameter is determined by the possibility that a specific cause of failure is detected before a failure occurs. In general, failures that are difficult to be prevented through inspections or condition monitoring are assigned a higher rating. The definition of detection rating is displayed in Table 12 below.

Table 12. Detection definition.

Rank	Definition	Meter
Rank 5	Very remote chance to detect a potential cause of failure	Approximate 0-20% chance of detection
Rank 4	Low chance to detect a potential cause of failure	Approximate 20-40% chance of detection
Rank 3	Moderate chance to detect a potential cause of failure	Approximate 40-60% chance of detection
Rank 2	High chance to detect a potential cause of failure	Approximate 60-80% chance of detection
Rank 1	Potential cause of failure is almost certainly detected	Approximate 80-100% chance of detection

The knowledge ranking is done in this study inversely as uncertainty. The two parameters chosen to determine this are the technological readiness index (TR) and data availability (DA). This kind of division of the knowledge ranking is quite common. For example, in safety design, the selected parameters might be the availability of data, and the availability of models. The definitions of technological readiness and data availability rating are presented in Table 13 and Table 14 below. (NEA, 2018)

Table 13. Technological readiness index definition.

Rank	Definition	Meter
Rank 5	Very limited knowledge, uncertainty cannot be characterized	TRL 1-2, Concept described, functional principles observed
Rank 4	Partially known, large uncertainty	TRL 3-4, Technology confirmed in laboratory tests
Rank 3	Known, moderate uncertainty	TRL 5-6, Technology demonstrated under relevant operating conditions
Rank 2	Known, small uncertainty	TRL 7-8, Prototype demonstrated under relevant operating conditions
Rank 1	Fully known, small uncertainty	TRL 9, Technology proven under relevant operating conditions

Table 14. Data availability definition.

Rank	Definition	Meter
Rank 5	No data availability, phenomenon poorly understood	No research available
Rank 4	Limited availability of data, large uncertainty about phenomenon	Only limited amount of research available
Rank 3	Partially available, moderate uncertainty about phenomenon	Moderate amount of research available
Rank 2	Adequate amount of data available, phenomenon known	Adequate amount of research available
Rank 1	Sufficient and good quality data available, phenomenon fully understood	Good amount of available research

The calculation of final importance and uncertainty levels is done simply by multiplication of the corresponding factors. The outcome is normalized to the scale of 1-5 according to equations 8 and 9 below.

$$IL = \sqrt[3]{S \cdot O \cdot D} = \sqrt[3]{RPN} \quad [8]$$

$$UL = \sqrt[2]{TR \cdot DA} \quad [9]$$

Where IL is the importance level and UL is the uncertainty level. This kind of simple determination quite easily distinguishes dominating phenomena. As a downside, a single parameter with a high value might dominate the factor.

6.7 Anticipated results

After the PIRT-study is completed, a prioritized list of operability-affecting phenomena is provided. The results are graphically presented in an outcome chart, functioning similarly as a risk assessment matrix. For distinguishing the outcome phenomena, the matrix can be divided into three areas: the area of high uncertainty, the area of high importance and the combined area of high uncertainty and high importance. These are presented in Figure 15 below. The general visualization of the method is adapted from “Phenomena Identification and Ranking Table. ” (NEA, 2018).

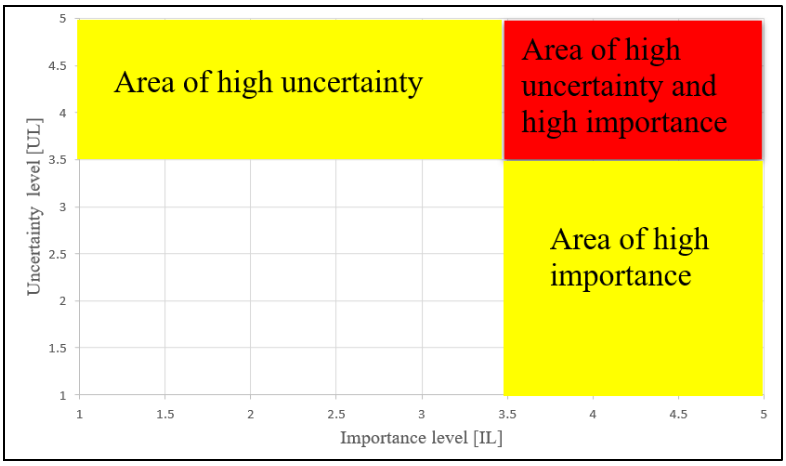

Figure 15. PIRT outcome matrix.

Phenomena that are located in the area of high uncertainty and high importance are considered the most crucial. These phenomena need extensive R&D effort to reduce the level of uncertainty. This typically means developing solutions that also reduce the importance level. Multiple phenomena identified in this zone indicate that the general state of the reactor design is not yet suitable or mature enough for reliable and cost-efficient operation. The overall definitions of the importance level and uncertainty levels are presented in Table 15 below.

Table 15. Overall definitions of the importance and uncertainty levels.

Rank	Importance level (IL):	Uncertainty level (UL):
5	Phenomenon has controlling impact on operability	Phenomenon not understood, large uncertainty
4-5	Phenomenon has high impact on operability	Phenomenon partially known, large uncertainty
3-4	Phenomenon has moderate impact on operability	Phenomenon known, moderate uncertainty
2-3	Phenomenon has low impact on operability	Phenomenon known, low uncertainty
1-2	Phenomenon has insignificant impact on operability	Phenomenon fully understood, low uncertainty

7 Phenomena identification and ranking results

The basis of knowledge for this PIRT study is the generic reactor design information, historical operational experience data and IAEA database information on historical production values and performance indicators.

The phenomena identified are significant events, such as process or system failures, maintenance actions or postulated initiating events. For all the four reactors, the phenomena are categorized in six categories: 1. Fuel and core, 2. Maintenance activities, 3. Structural materials, 4. Heat transfer and coolant properties, 5. Postulated initiating events, 6. Power conversion.

Majority of the identified operability-related phenomena are presented and explained in the chapter 5 tables of principal challenges (Table 2, Table 4, Table 6 and Table 8). In addition to these, a collection of phenomena identified to reactor-specific anticipated operational occurrences, i.e. postulated initiating events and their main corresponding safety systems are presented in Appendix 5. Several sources of information were used to collect data on operational experience. Along with those referenced in chapter 5, the most notable references are provided in Table 16 below.

Table 16. Most notable references for operational experience data.

Reactor	Reference
VHTR	Beck et al. (2011) Copinger et al. (2003) Zhang (2024)
SFR	IAEA (2002) IAEA (2013) Pakhomov (2018)
LFR	Allen (2007) IAEA (2000) Troyanov et al. (2023)
MSR	Holcomb et al. (2017) Holcomb et al. (2018) IAEA (2023)

7.1 VHTR results

The resulting PIRT for the VHTR is presented in Table 17 and Figure 16.

Table 17. Phenomena identification and ranking table for the VHTR.

Phenomena	#	Failure mode / Scenario	Importance ranking				State of knowledge		
			S	O	D	IL	TR	DA	UL
1. Fuel and core									
TRISO fuel integrity	1.1	Large fuel failure / Fission product release	5	1	3	2.47	2	2	2.00
TRISO fuel supply	1.2	Unavailability of fresh fuel	5	1	2	2.15	3	2	2.45
TRISO fuels for COT above 800°C	1.3	Fuel integrity if coolant is lost	5	1	3	2.47	5	3	3.87
Graphite dust	1.4	Maintenance difficulty, loss of efficiency	3	5	4	3.91	4	3	3.46
Spatial oscillations	1.5	Power reduction	2	3	1	1.82	1	2	1.41
Online refuelling (in pebble-bed cores)	1.6	Failure of refuelling mechanism	4	3	4	3.63	2	3	2.45
2. Maintenance									
Maintenance actions in depressurized shutdown	2.1	Air or water ingress	5	4	3	3.91	2	3	2.45
Refuelling actions through vessel head (in prismatic cores)	2.2	Failure of refuelling mechanism	4	3	4	3.63	3	3	3.00
3. Structural materials									
Pressure vessel materials for COT below 800°C	3.1	Structural material failure	5	2	2	2.71	2	3	2.45
Heat exchanger materials for COT below 800°C	3.2	Structural material failure	4	3	3	3.30	2	3	2.45
Pressure vessel materials for COT above 800°C	3.3	Structural material failure	5	3	3	3.56	5	3	3.87
Heat exchanger materials for COT above 800°C	3.4	Structural material failure	5	3	3	3.56	5	3	3.87
Control rod shroud materials	3.5	Failure to operate / need of premature replacement	3	2	2	2.29	2	2	2.00
Graphite wear	3.6	Need of premature replacement	5	3	3	3.56	2	3	2.45
4. Heat transfer / Coolant properties									
Functioning of coolant purification system	4.1	Corrosivity caused by impurities	3	3	3	3.00	2	3	2.45
Effectiveness of the residual heat removal system	4.2	Failure to remove residual heat	4	1	2	2.00	3	2	2.45
Moisture removal from coolant and fresh fuel	4.3	Corrosion	3	5	3	3.56	4	3	3.46
5. Postulated initiating events									
Water ingress due to steam generator tube rupture	5.1	Corrosion, extensive maintenance needed	5	3	4	3.91	3	3	3.00
Large air ingress to primary system	5.2	Corrosion, rapid oxidation of graphite	5	3	2	3.11	2	2	2.00
Small air ingress to primary system	5.3	Corrosion, oxidation of graphite	2	4	4	3.17	2	2	2.00
Large primary coolant leakage	5.4	Large loss of coolant	5	1	2	2.15	2	3	2.45
Small primary coolant leakage	5.5	Small loss of coolant	2	4	4	3.17	2	3	2.45
Emergency depressurization of primary system	5.6	Total loss of coolant, radioactive release	5	1	2	2.15	3	2	2.45
Gas circulator failure	5.7	Loss of coolant flow	2	4	2	2.52	2	2	2.00
6. Power conversion									
Rankine cycle	6.1	Failure in power conversion process	2	4	2	2.52	1	1	1.00
Brayton cycle	6.2	Failure in power conversion process	2	4	2	2.52	5	2	3.16
Use of reactor in high-temperature industrial applications	6.3	Failure in heat supply process	2	4	2	2.52	3	2	2.45

7.2 SFR results

The resulting PIRT for the SFR is presented in Table 18 and Figure 17.

Table 18. Phenomena identification and ranking table for the SFR.

Phenomena	#	Failure mode / Scenario	Importance ranking				State of knowledge		
			S	O	D	IL	TR	DA	UL
1. Fuel and core									
Fuel handling	1.1	Failure resulting in fuel damage	5	2	3	3.11	3	3	3.00
Fuel handling mechanism	1.2	Failure of handling machinery	4	5	3	3.91	3	3	3.00
Fuel identification	1.3	Failure of fuel identification	5	3	2	3.11	2	3	2.45
Decontamination and storage of spent fuel	1.4	Fires, remote operation failures	2	2	2	2.00	2	3	2.45
Positive void reactivity effect	1.5	Reactivity incident if the coolant boils	5	1	1	1.71	3	2	2.45
Supply of oxide fuels	1.6	Unavailability of fresh fuel	5	1	2	2.15	2	2	2.00
Supply of non-oxide fuels	1.7	Unavailability of fresh fuel	5	1	2	2.15	4	3	3.46
2. Maintenance									
Remote operations inside reactor vessel	2.1	Failure of remotely operated machinery	5	4	3	3.91	3	4	3.46
Inspections and maintenance of in-sodium equipment	2.2	Fires, failure of remotely operated machinery	4	4	3	3.63	3	4	3.46
3. Structural materials									
Irradiation and temperature-resistant core materials	3.1	Structural material failure	5	3	2	3.11	3	3	3.00
Corrosion by coolant impurities	3.2	Structural material failure	5	4	3	3.91	3	3	3.00
4. Heat transfer / Coolant properties									
Functioning of coolant purification system	4.1	Failure resulting in coolant impurities	4	3	3	3.30	3	4	3.46
Cover gas purity	4.2	Failure needing cover gas replacement	3	5	2	3.11	3	3	3.00
Effectiveness of the residual heat removal system	4.3	Failure to remove residual heat	5	1	1	1.71	3	2	2.45
Loss of a secondary circuit	4.4	Failure in secondary coolant circuit	3	4	3	3.30	2	3	2.45
5. Postulated initiating events									
Air ingress to sodium systems	5.1	Formation of sodium oxides	4	4	3	3.63	2	4	2.83
Water ingress from steam generator tube rupture	5.2	Water-sodium reactions, hydrogen formation, and fires	5	3	3	3.56	3	2	2.45
Large primary sodium leak	5.3	Large leakage causing fires and radioactive contamination	5	2	2	2.71	4	3	3.46
Small primary sodium leak	5.4	Leakage causing minor fires and radioactive contamination	3	3	4	3.30	2	2	2.00
Large secondary sodium leak	5.5	Large leakage causing fires and loss of a cooling loop	5	3	2	3.11	4	3	3.46
Small secondary sodium leak	5.6	Leakage causing minor fires and loss of a cooling loop	3	5	4	3.91	2	2	2.00
Cover gas leak	5.7	Total loss of cover gas resulting in large air ingress	5	3	3	3.56	3	4	3.46
Coolant solidification	5.8	Material stress and loss of cooling	5	1	1	1.71	2	3	2.45
Failure of a primary coolant pump	5.9	Loss of forced circulation	2	3	2	2.29	2	2	2.00
Failure of control rod insertion	5.10	Partial or full loss of reactivity control	2	3	2	2.29	2	2	2.00
6. Power conversion									
Rankine cycle	6.1	Failure in power conversion process	2	4	2	2.52	1	1	1.00

7.3 LFR results

The resulting PIRT for the LFR is presented in Table 19 and Figure 18.

Table 19. Phenomena identification and ranking table for the LFR.

Phenomena	#	Failure mode / Scenario	Importance ranking				State of knowledge		
			S	O	D	IL	TR	DA	UL
1. Fuel and core									
Fuel handling	1.1	Failure resulting in fuel damage	5	2	3	3.11	4	4	4.00
Fuel handling mechanism	1.2	Failure of handling machinery	4	5	3	3.91	4	4	4.00
Fuel identification	1.3	Failure of fuel identification	5	3	2	3.11	3	3	3.00
Fuel integrity	1.4	Corrosion of fuel cladding	4	3	4	3.63	4	4	4.00
Fuel availability	1.5	Unavailability of fresh fuel	5	1	2	2.15	4	3	3.46
2. Maintenance									
Remote operations inside reactor vessel	2.1	Failure of remotely operated machinery	4	4	3	3.63	4	5	4.47
Polonium hazard (If LBE-coolant is used)	2.2	Failure of polonium hazard prevention	3	4	2	2.88	3	3	3.00
Maintenance actions in temperatures above 330°C	2.3	Failure of maintenance equipment	4	4	2	3.17	4	5	4.47
3. Structural materials									
Structural materials for COT below 500°C	3.1	Corrosion by coolant, structural material failure	5	4	3	3.91	4	3	3.46
Structural materials for COT above 500°C	3.2	Corrosion by coolant, structural material failure	5	4	3	3.91	5	4	4.47
4. Heat transfer / Coolant properties									
Functioning of coolant purification system	4.1	Coolant oxidization resulting in corrosion and oxide formation	4	5	4	4.31	5	3	3.87
Effectiveness of the residual heat removal system	4.2	Failure to remove residual heat	5	1	2	2.15	3	3	3.00
5. Postulated initiating events									
Water ingress due to steam generator tube rupture	5.1	Primary system overpressure, secondary side contamination	5	3	2	3.11	4	3	3.46
Coolant slagging due to lead oxides	5.2	Coolant flow blockages	4	3	3	3.30	5	3	3.87
Air ingress to reactor	5.3	Minor coolant oxidization	3	4	3	3.30	4	3	3.46
Cover gas leakage	5.4	Total loss of cover gas	4	2	2	2.52	4	4	4.00
Coolant leakage	5.5	Lead leakage causing contamination and need of repair	4	4	3	3.63	4	4	4.00
Failure of a primary coolant pump	5.6	Loss of forced circulation	2	3	2	2.29	2	3	2.45
Failure of control rod insertion	5.7	Partial or full loss of reactivity control	2	3	2	2.29	2	3	2.45
Coolant solidification	5.8	Material stress leading to fuel ruptures	5	2	2	2.71	4	2	2.83
6. Power conversion									
Rankine cycle	6.1	Failure in power conversion process	2	4	2	2.52	1	1	1.00

7.4 MSR results

The resulting PIRT for the MSR is presented in Table 20 and Figure 19. Phenomena marked with an asterisk are not present in a salt-cooled HTR.

Table 20. Phenomena identification and ranking table for the MSR.

Phenomena	#	Failure mode / Scenario	Importance ranking				State of knowledge		
			S	O	D	IL	TR	DA	UL
1. Fuel and core									
In-core instrumentation	1.1	Unavailability of measurements	4	3	4	3.63	4	4	4.00
Online refuelling and salt processing*	1.2	Failure of online fuel processing	5	3	3	3.56	5	3	3.87
Online removal of fission products*	1.3	Failure of online fuel processing	4	4	3	3.63	5	3	3.87
Fuel availability	1.4	Unavailability of fresh fuel	5	1	2	2.15	4	3	3.46
Waste management	1.5	Failure of waste salt handling systems	3	3	3	3.00	4	4	4.00
2. Maintenance									
Remote operations inside reactor compartment*	2.1	Failure of remotely operated machinery	5	4	3	3.91	4	4	4.00
Failure of contaminated equipment	2.2	Failure resulting in remotely conducted maintenance	5	5	3	4.22	5	4	4.47
Personnel protection	2.3	Radiation and tritium protection	2	4	2	2.52	2	2	2.00
Maintenance actions in temperatures above 430°C	2.4	Failure of maintenance equipment	4	4	2	3.17	4	5	4.47
3. Structural materials									
Corrosion resistant structural materials	3.1	Corrosion by salt, structural material failure	5	3	3	3.56	5	3	3.87
Irradiation resistant structural materials	3.2	Failure of reactor or fuel processing structural materials	5	3	2	3.11	4	3	3.46
Graphite wear	3.3	Irradiation and salt penetration caused wear requiring replacement	4	5	3	3.91	5	3	3.87
Tritium permeation	3.4	Tritium permeation through structural materials	3	5	2	3.11	3	3	3.00
4. Heat transfer / Coolant properties									
Control of salt chemistry	4.1	Failure resulting in deposits primary circuit	4	3	3	3.30	4	3	3.46
Effectiveness of the residual heat removal system	4.2	Failure to remove residual heat	5	1	2	2.15	3	3	3.00
5. Postulated initiating events									
Water ingress due to heat exchanger failure	5.1	Pressure increase, secondary circuit contamination	5	2	2	2.71	3	4	3.46
Air ingress to primary system	5.2	Formation of impurities, oxidation of graphite	4	3	3	3.30	3	4	3.46
Fuel salt leakage*	5.3	Leakage of fuel salt to reactor containment	5	4	3	3.91	4	4	4.00
Coolant salt leakage	5.4	Leakage of coolant salt to reactor containment	2	4	3	2.88	4	4	4.00
Failure of the off-gas system*	5.5	Gaseous fission product buildup or release	4	3	3	3.30	4	3	3.46
Salt pump failure	5.6	Loss of forced circulation	2	4	2	2.52	2	4	2.83
Inadvertent freeze (drain) valve opening*	5.7	Transient, inadvertent shutdown	2	3	2	2.29	2	3	2.45
Core flow blockage	5.8	Flow blockages resulting in elevated temperatures	4	2	3	2.88	3	4	3.46
Coolant solidification	5.9	Structural material stress	4	2	2	2.52	3	4	3.46
6. Power conversion									
Rankine cycle	6.1	Failure in power conversion process	2	4	2	2.52	1	1	1.00

7.5 Analysis and discussion

A total of ninety-nine operability-related phenomena were identified across the four reactor designs. Of these, thirty-one were ranked as phenomena with high-priority and forty-six with high uncertainty. Twenty-two were ranked both. These phenomena that are poorly understood and have a serious impact on plant operability are the most important to address.

The results from the VHTR show an average value of 2.93 for the phenomena importance level and 2.59 for the uncertainty level. This indicates that the general relevance level for the phenomena is moderate. The VHTR ranking results from Table 17 are presented in Figure 16 below.

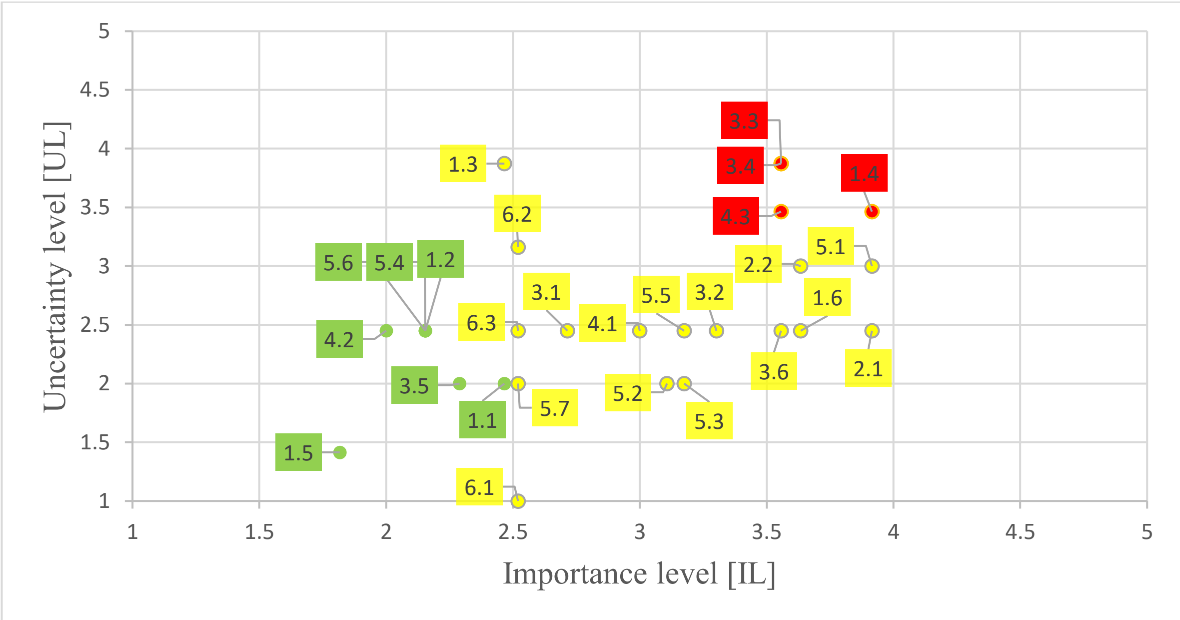

Figure 16. VHTR Ranking results.

The figure shows the ranked prenomena with color coding representing their relevance. It can be seen that the majority of the phenomena have both an average level of importance and understanding. The distinct phenomena with both high importance and uncertainty levels (marked with red) are: graphite dust, pressure vessel and heat exchanger materials for core outlet temperatures above 800 °C, and moisture removal from coolant and fresh fuel.

The phenomenon of graphite dust accumulation is present in both prismatic and pebble bed cores. Graphite dust is formed by the wear of graphite components. In pebble bed cores, the dust is formed by mechanical interaction of fuel pebbles with surrounding pebbles, online refuelling system and the graphite reflector. Graphite dust is also formed by the thermal expansion of graphite components, which again leads to mechanical interaction and wear. More uncommonly, failure for fuel elements also releases graphite dust. The severity ranking of graphite dust is moderate, as is most likely doesn’t cause direct unavailability, but it can potentially affect the plant efficiency and the length of maintenance actions. The issues with graphite dust have occurred in multiple HTR reactors in the past, and the possibility and effectiveness of dust filtration is uncertain.

The phenomenon with material availability for core outlet temperatures above 800 °C will most likely not be solved in the near future. The metal creep, fatigue and their interaction along with coolant impurity-caused corrosion problems set requirements for the primary system structural materials. Contemporary means of vessel cooling with currently ASME qualified materials, such as Alloy 800H do not allow operations above 800 °C. Possible materials used in fossil plant applications are often not suitable for forging large pressure vessels, and more experimental heat exchanger materials are not yet qualified and often face corrosion problems. The low technological readiness of these materials, along with severe and expected failures set the high ranking.

The phenomenon of moisture in the VHTR reactors is a problem due to corrosivity. Excluding the low-possibility water ingress from a steam generator tube rupture, sources of a small moisture ingress to the reactor coolant can be leaking welds, leaking seals seals or fuel pebbles during online refuelling. This moisture ingress has a degrading effect on the reactor structural materials. Even in smaller amounts, the moisture will eventually wear down the primary system components and structural materials, causing an early need of replacement or repair. In a more extreme case, the reactor would have to be shut down for moisture removal if the water concentration in the coolant would be dangerously high. The phenomenon of moisture in the VHTR coolant is inevitable, so functioning of the coolant purification system is extremely important. The severity of the phenomenon is also important, possibly affecting the load factor through unplanned maintenance or coolant purification. The technology for coolant purification and moisture removal from fuel pebbles has been demonstrated, but its effectiveness and the overall prevention of moisture ingress is uncertain.

For the SFR, an average value of 2.95 was obtained for the phenomena importance level and 2.70 for the uncertainty level. This indicates that the general relevance level for the phenomena is moderate. The SFR ranking results from Table 18 in are presented in Figure 17 below.

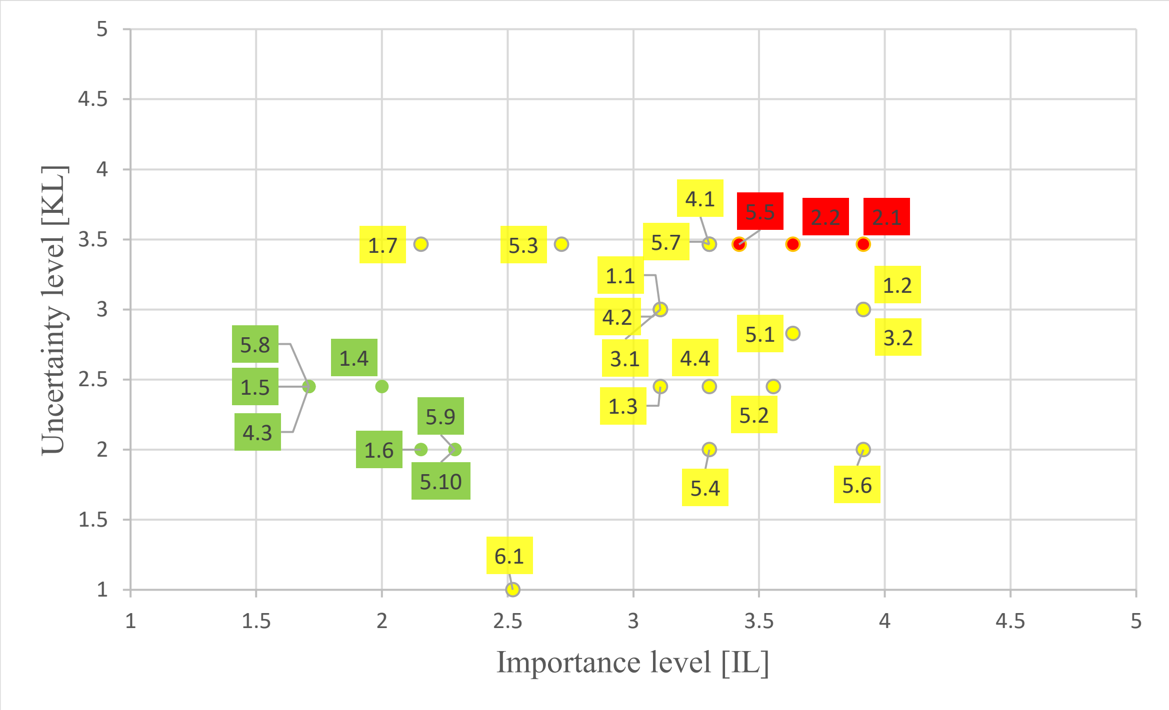

Figure 17. SFR Ranking results.

The phenomena with both high importance and uncertainty levels are: remote operations inside the reactor vessel, inspections and maintenance of in-sodium equipment and large secondary sodium leaks.

In SFRs, conduction of remote operations inside the reactor vessel is a major operability risk. Due to the need of system leak tightness, the SFR must be refuelled without opening the reactor vessel. In addition to refuelling, all maintenance actions on primary sodium components must be done the similar way. Compared to an LWR, the SFR needs a more complex reactor refuelling and maintenance system often consisting of a rotating plug vessel head, in-sodium component detector, in-sodium fuel manipulator, fuel transfering machine and a sodium-filled fuel storage tank. All of these systems are required to be completely leak tight, as any air ingress to the reactor as well as sodium leaks outside the system need to be avoided. The reliability requirements for these handling components need to be high, since repairing or replacing in-sodium equipment is difficult and time-consuming. This is due to the need of system leak tightness and the need of cleaning any residual sodium before maintenance. Many of these remotely controlled systems are designed to be operated only during a refuelling outage, so their failure would most likely cause a delay to the outage schedule. With the current and historically operated SFRs, the refuelling and maintenance actions are notably slower than with LWRs. This is caused primarily by the complex operations inside the reactor vessel. The reliability of the in-sodium component handling equipment is critical from the view of plant operability. Multiple failures of these systems have been recorded in history and they are also the main cause of low maximum performance factors. The issues and general principles of the in-sodium handling equipment are, however, well known and documented.

From the SFR-specific incident scenarios, sodium leaks from the secondary sodium circuit are the most probable to occur and to cause interruptions to operations. The other major events, such as water-sodium reactions in steam generators and primary sodium leaks are much less likely to occur, although they might be more severe. The secondary sodium circuit of the SFR is large and not as isolated as the primary circuit, which is typically sealed with a cover gas and confined inside a separate safety vessel. A large sodium leak from the secondary circuit might be difficult to contain, and it has potential for serious structural damage. Multiple incidents like these have occurred in the past, causing notable downtime. However, technological readiness for mitigation and detection systems for sodium leaks have been demonstrated, and there is plenty of data available on the issue.

For the LFR, an average value of 3.12 was obtained for the phenomena importance level and 3.46 for the uncertainty level. This indicates that the general relevance level for the phenomena is moderately high, with most of the identified phenomena in the area of high uncertainty. The LFR ranking results from Table 19 are presented in Figure 18 below.

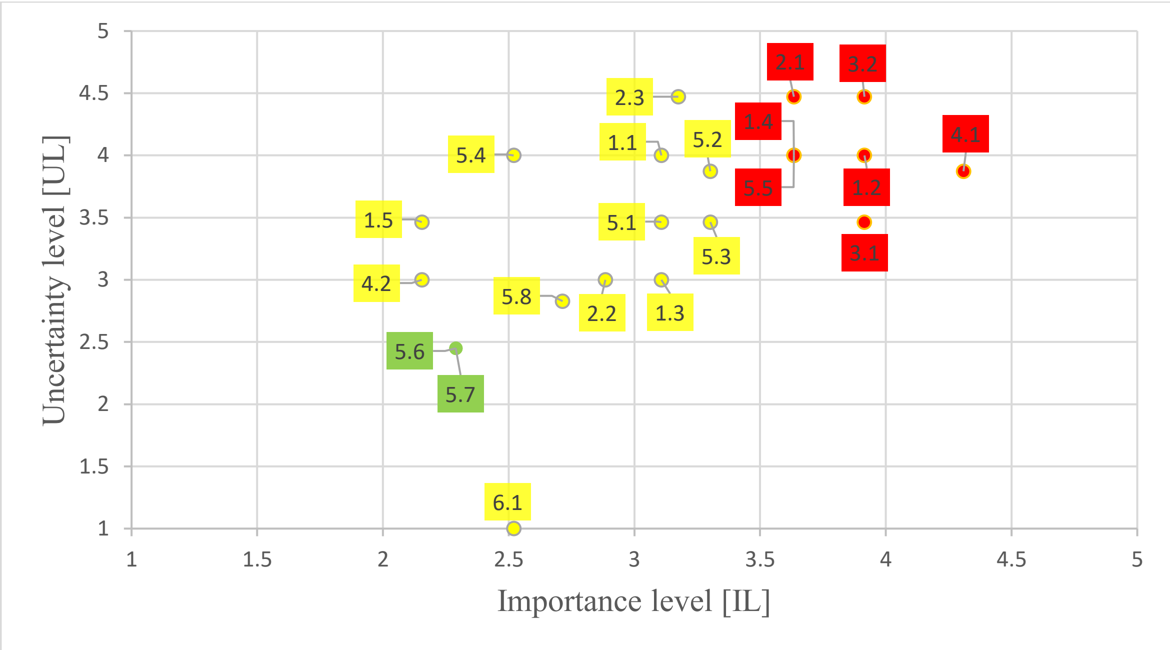

Figure 18. LFR Ranking results.

The phenomena with both high importance and uncertainty levels are: fuel handling mechanism and remote operations inside reactor vessel, fuel integrity, structural materials (above and below 500 °C), functioning of the coolant purification system and coolant leakages.

The LFR shares many similarities with the SFR, and the issues related to fuel handling and remote operations inside the reactor vessel are among them. The LFR requires complex and leak-tight equipment for fuel handling and in-vessel maintenance actions. Faults in these systems are expected and their effect in plant operations are severe. A minor benefit in using lead coolant is the inertness with air, but air ingress to the reactor should be avoided in addition to spreading of toxic and radioactive contamination via the coolant. This is even more important if LBE-coolant is used, due to the produced polonium. Compared to the SFR, the high shutdown temperatures due to avoidance of coolant solidification will bring problems as well. All maintenance tools on the primary circuit components, as well as the refuelling machinery need to withstand high temperatures. The high severity and expected failures in the primary circuit maintenance activities set the high ranking. The historical operation of the LBE-submarines, albeit not directly applicable, supports this.

The phenomena of fuel and structural material integrity in LFRs are important. Since lead coolant is notably corrosive, requirements for structural materials and fuel cladding are strict.

Typical materials for LFR cladding are ferritic/martensitic steels and austenitic steels. Structural materials for reactor vessels are austenitic steels. Lead coolant corrodes structural materials through dissolution, but in a suitable temperature range of 400-550°C it is possible that a protective oxide layer forms on the steel surface, working as a corrosion inhibitor. This requires that a suitable oxygen content in the coolant is maintained and kept at levels that do not cause a formation of blockages. In addition to this kind of chemical control, the corrosion can be inhibited by using different coatings such as aluminium on the steels. The functioning of the chemical control system to inhibit corrosion has been demonstrated in the historically operated LBE-submarines, but very little data is available on the subject. The protective coatings are not yet demonstrated and require more thorough testing and development prior to deployment. These, combined with the high severity and expected occurrence of structural material or fuel cladding failures sets the high ranking.

For the MSR, an average value of 3.11 was obtained for the phenomena importance level and 3.46 for the uncertainty level. This indicates that the general relevance level for the phenomena is moderately high. The MSR ranking results from Table 20 are presented in Figure 19 below.

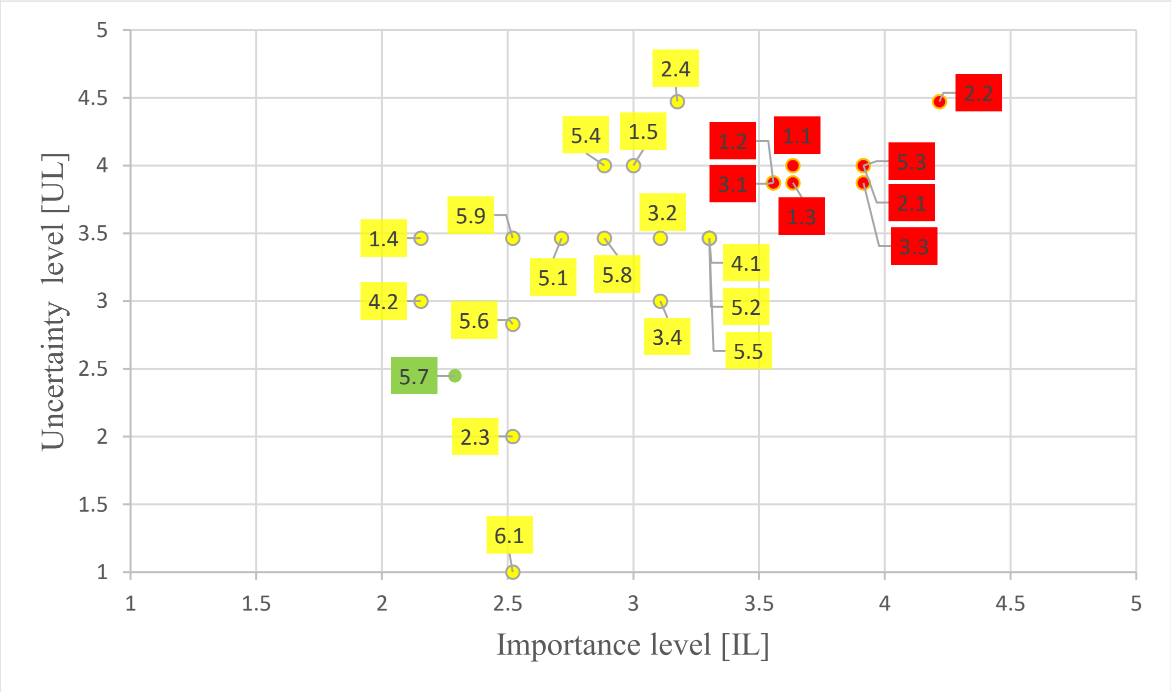

Figure 19. MSR Ranking results.

The phenomena with both high importance and uncertainty levels are: online refuelling and salt processing, online removal of fission products, availability of in-core instrumentation, remote operations inside reactor compartment, failure of contaminated equipment, corrosion resistant structural materials, graphite wear and fuel salt leakages.

The phenomena related to fuel salt processing are of large importance and uncertainty in the liquid-fuelled MSRs. The MSRE experiment demonstrated online refuelling and the removal of uranium from the molten salt through a fluorination process and the removal of gaseous fission products through an off-gas system. Other solid fission products remained in the fuel salt. The need of controlling the salt chemistry and fission product concentration in the fuel salt come from the corrosivity. Salt impurities such as oxygen and moisture along with fission products such as tellurium corrode structural materials. Some rare earth fission products have low solubility and a high neutron absorption cross-section, disturbing the reactivity balance. Failures in the salt processing system are highly severe to reactor operability, as refuelling or salt corrosivity control can cause long unavailability. The chemical control system has been tested and demonstrated, but not in a full-scale reactor. Processes for removal of fission products, such as helium sparging and pyrochemical separation are less technologically mature, with issues such as actinide and lanthanide separation.

The corrosion structural materials by fluoride salts is a major concern. The tested nickel-based alloys, such as Hastelloy-N suffer from irradiation damage and salt penetration. Preventing structural material failures is planned to be conducted by the salt chemical control systems, and possibly by coating the alloys with dissolution-resistant materials. In addition to the historically tested materials, new alloys are being tested and developed for better irradiation corrosion resistance. Overall, the material issues with the MSRs are severe, and the knowledge regarding them are quite limited to the historical MSRE operations and later laboratory tests. Testing and development in relevant operating scale and conditions are needed before wider application.

The most important operations and maintenance phenomena with the MSRs are caused by the difficult environment around the primary salt circuits. The high shutdown temperatures and major radiation dose rates create a need for utilizing remote maintenance technologies. The effect of radiation is most important in circulating fuel designs, but its effect in static fuel reactors and salt-cooled HTRs is notable as well. When irradiated equipment in the MSR primary circuit need to be repaired or replaced, they most likely need to be remotely removed and decontaminated before such maintenance can be conducted. While similar kind of remote radioactive component transfers and decontamination are done in LWRs, the lack of demonstration and experience, along with notably higher amount of individual irradiated components make this a challenge. Remote maintenance and component replacements were not necessary during the MSRE reactor operation, and its decommissioning is still under planning, so no relevant operating experience yet exists on the subject. Similar issues are faced when dealing with possible salt spillages and leaks. The expected failures and overall low maintainability of irradiated components and remote handling systems make these phenomena extremely important.

7.5.1 Comparison of results

The PIRT results show varying levels of importance and uncertainty among the four reactors for the identified operability-related phenomena. The average values of importance for the VHTR and SFR are moderate, indicating that the phenomena are predicted have some impact on plant operations. The average importance level of the LFR and MSR is noticeably higher. The issues identified for these reactors are more severe, indicating major possible effects to production.

For the levels of uncertainty, the VHTR was ranked the lowest close to the SFR. This is explained by the overall satisfactory level of historical operational experience and good overall system technological readiness levels. Both these reactors have been tested and are currently being operated under relevant operating conditions. However, solutions for issues regarding advanced materials and maintenance activities are still not ready.

The LFR and MSR have high rankings of uncertainty. The phenomena are known with moderate to large uncertainty and the technological readiness to their solutions are not yet mature. The issues regarding plant operability need further research and development, again primarily focused on the reactor materials. Multiple operations and maintenance challenges still exist, and acceptable solutions to them need to be developed prior to full-scale application.

7.6 Applicability of results and possibilities of further research

The goal of the PIRT study was the identification, ranking and comparison of the most important operability-related phenomena of the selected reactor designs. This was completed along with formulating graphical presentations of the solutions. The results indicate that the VHTR and SFR are more mature and ready for operations and the LFR and MSR are much less. This was expected and is in-line with the estimations of technology development timeframes presented in Figure 7. The most important challenges of reactor design were well known, but specific focus on operability issues and their relation to each other was not available.

After completing the study, some shortcomings were observed with the method. First, the lack of an expert panel created issues. Typical PIRT panel consists of people from various positions and backgrounds who know the assessed systems and issues in detail. The study conducted here was expanded with expansion of the assessment criteria, notably with the FMEA method, to bring more depth to the results. The amount and adequacy of gathered data on historical operational experience was good, but defining accurate rankings for the severity and the technological readiness of the solution proved difficult on some occasions due to the lack of available information on detailed design. The set definitions of the evaluation criterion and importance and uncertainty ranks was clear, but the evaluation is still done subjectively, this is why a panel is typically needed.

Continuing research on the operability issues would most likely need to be focused on a single reactor type, design, and vendor. Then, depending on the state of development and available data, a suitable method for the analysis could be chosen. If a PRA model of the plant would be available, it could possibly be utilized as an input for a probabilistic availability assessment. An issue here is the scope of the study. PRAs focus on items important to safety, not operability.

The technological readiness to solutions of the phenomena identified in this thesis could potentially be evaluated through information presented in the design preliminary safety analysis reports. In a less in-depth scale, vendor design information on outage activities and their duration could be used to calculate values of planned unavailability. This kind of design evaluation would most likely require a formal contract to access the detailed design values and information. If these evaluations would indicate that solutions to the major known problems would be available and ready, further collaboration between a utility and the developer could be considered.

Other key areas that could benefit from further research include analyses of the economical aspects and the fuel cycle. The reference values for the historical costs of these plants covered in this thesis do not accurately represent what they would be in western countries in a modern context. Building an accurate model of GEN IV overnight- and operating costs would require that completed models of the selected plant would be available. In other words, the detailed design would have to be completed.

The question on the fuel supply and cycle is of major importance in all GEN IV designs. Most of these designs would require the use of high-assay low-enriched uranium (HALEU), which currently has very few suppliers. The SFR and the LFR have a benefit of being able to use oxide fuels, but the TRISO fuels in the VHTR and the fuels salts in the MSR bring major fuel supply issues.

The question on the fuel cycle is perhaps of a lesser importance. Many GEN IV reactor designs are designed to be operated as breeder reactors. If the benefit of generating more fissile material than is consumed is to be achieved, fuel reprocessing is needed. This brings questions on how, where, and how expensively this would be conducted. Further studies on the applicability of this would be then needed.

8 Conclusion

This thesis studied the operational issues of GEN IV reactors. The aim of the thesis and the conducted PIRT studies was to identify the main operability-related issues of the reactors and to see which issues from their historical predecessors still remain relevant. Secondarily, the study was to serve as a preliminary risk assessment that could potentially be used as an input in further research and design assessment.

The main findings of the PIRT study were specific issues identified from the historical operating experience of these reactors. Almost all of the identified phenomena were linked to the challenging conditions caused by the high temperatures and exotic coolants. All of these reactors have issues with structural materials, caused either by corrosion, irradiation embrittlement or temperature-dependent creep and fatigue. The corrosion issues are caused by either coolant impurities or the coolant itself. Preventing this requires that a functioning coolant purification system is designed, or that protective coatings are applied to the structural materials. The coolant purification systems have been demonstrated to some degree but they, along with metal coatings, are not yet technologically mature. Likewise, suitable materials for extremely high temperatures are either not yet qualified or they are not fitting for forging large pressure vessels.

Other identified major issues are related to the operations and maintenance of these reactors. The coolant properties can create difficulties due to a requirement of total system leak tightness or difficult radiation conditions. This results in difficulties with maintenance, as remote operations need to be utilized in actions such as refuelling, and primary system component maintenance. The need to utilize remote operations creates a reliability risk, since both the remote handling equipment and the remotely maintained systems may cause unavailability. In addition, the prevention of coolant leaks is an issue in all of these reactors. Coolant leaks are a risk to both safety and operability as in light-water reactors, but due to the nature of coolants used these designs, additional risks such as fires and spread of toxic contamination are introduced.

The study deduced that reactors having the longest historical operating experience, the SFR and VHTR, are also the most technologically mature from the viewpoint of operability. Their operation has been demonstrated under relevant conditions, but uncertainties exist that need to be solved before large commercialization can happen. The LFR and the MSR are less technologically mature, still facing principal challenges to operability that have not been solved. However, research on these issues is being continuously conducted, and both nuclear and non-nuclear experimental and demonstration facilities are planned to assist in research on these questions.

The PIRT study done in this thesis showed some limitations. Typically, this kind of a study is conducted on a large panel of experts, but due to the nature of the thesis, this was not done. In addition, the reactor designs were assessed on a concept level based on historical operating experience, so the results do not represent any individual reactor. This is also beneficial, since the application possibilities of the results are wider.

All the six GEN IV designs are currently being pursued by multiple different developers. These include traditional national laboratories and research facilities, as well as large nuclear energy companies and small start-ups. Recently in western countries, the development of GEN IV reactors has shifted to privately owned companies. At the moment, there are multiple privately and publicly backed projects that are aiming to construct their own demonstrator units. The success of these projects is of course dependent on economic factors, and the operation of the first units should show if and how they have solved the issues presented in this thesis.

The results of this thesis indicate that the GEN IV reactors are in varying states of technological maturity, but none are yet ready for commercialization. Some designs have less uncertainty about the issues, but even they would need to demonstrate their operation before a utility might consider them. Possible identified topics of future research are on the areas of economic modelling, in-depth availability analysis and fuel cycle research. From the viewpoint of a utility, these analyses would be most beneficial if conducted on a single reactor in an advanced state of development, such as in the end of detailed design. The major operability issues identified in this thesis could then be assessed accurately.

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Appendix 1. GEN IV General technical specifications.

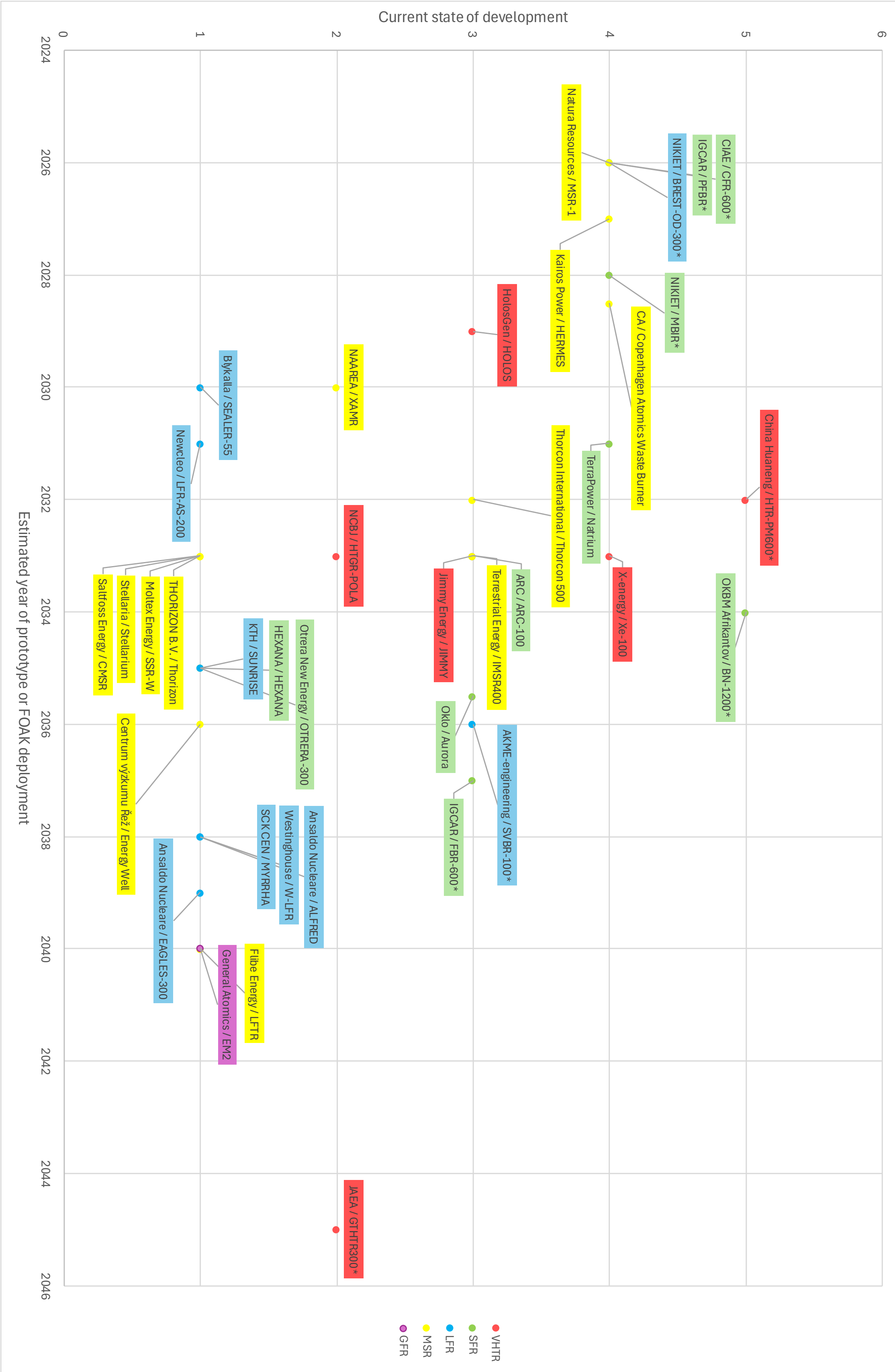
Reactor	VHTR	SCWR	SFR	MSR	LFR	GFR
Spectrum	Thermal	Thermal / Fast	Fast	Thermal / Fast	Fast	Fast
Moderator	Graphite	Light / Heavy water	-	Graphite / None	-	-
Coolant	Helium	Light water	Sodium	Molten salt $LiF_4/BeF_4/ZrF_4$	Lead / LBE	Helium/ CO_2
Fuel type	TRISO UO_2/MOX	Steel-cladded UO_2/MOX	Steel-cladded UO_2/MOX	Fuel salt UF_4,PuF_4 Or TRISO	Steel-cladded UN/MOX	UPuC
Fuel enrichment [%]	8-20	5-7	15-30	20	15-30	20-30
Thermal power [MW]	100-1000	2000-4000	100-3000	100-3000	100-3000	600-3000
Power density [W/cm^3]	2-8	100-300	300-500	20-300	100-300	100
Core outlet temperature [°C]	700-1000	500-600	530-550	600-700	480-540	600-900
Operating pressure [Bar]	40-100	250	1	1	1	50-100
Thermodynamic efficiency [%]	45-55	45	45	45-50	40-45	45-50
Power conversion process	Rankine / Brayton	Supercritical Rankine	Rankine	Rankine	Rankine	Brayton

Appendix 2. Currently ongoing GEN IV projects (Adapted from ARIS, 2025).

Company	Type	Reactor	Country	Purpose	Power [MW_{th}]	State of development
China Huaneng	**VHTR**	HTR-PM600	China	Commercial	6x250	Commercial Demonstration
X-energy		Xe-100	USA	Commercial	4x80	Detailed Design
Jimmy Energy		JIMMY	France	Commercial	20	Detailed Design
HolosGen		HOLOS	USA	Commercial	22	Detailed Design
Terra Innovatum		SOLO	USA	Commercial	4	Basic Design
NCBJ		HTGR-POLA	Poland	Demonstration	30 demo, 180 full-scale	Basic Design
JAEA		GTHTR300	Japan	Demonstration	600	Basic Design
BRIN		PeLUIt-40	Indonesia	Commercial	100	Basic Design
STL Nuclear		HTMR100	South Africa	Commercial	100	Conceptual Design
Framatome		SC-HTGR	USA / France	Commercial	625	Conceptual Design
JAEA		HTR50S	Japan	Demonstration	50	Conceptual Design
OKBM Afrikantov		MHR-T	Russia	Commercial	30-1000	Conceptual Design
OKBM Afrikantov		MHR-100	Russia	Commercial	215	Conceptual Design
OKBM Afrikantov	**SFR**	BN-1200	Russia	Commercial	2800	Commercial Demonstration
China Institute of Atomic Energy		CFR-600	China	Demonstration	1500	Engineering Demonstration
IGCAR		PFBR	India	Demonstration	1250	Engineering Demonstration
NIKIET		MBIR	Russia	Experimental	150	Engineering Demonstration
TerraPower		Natrium	USA	Commercial	840	Engineering Demonstration
Oklo		Aurora	USA	Commercial	75	Detailed Design
ARC Clean Technology		ARC-100	USA	Commercial	286	Detailed Design
Toshiba		4S	Japan	Commercial	30, 135	Detailed Design
IGCAR		FBR 1&2	India	Commercial	1250	Detailed Design
GE-Hitachi		PRISM	USA	Commercial	840	Detailed Design
HEXANA		HEXANA	France	Prototype	400	Conceptual Design
Otrera New Energy		OTRERA300	France	Prototype	300	Conceptual Design
JAEA		JSFR	Japan	Commercial	3530	Conceptual Design
NIKIET	**LFR**	BREST-OD-300	Russia	Demonstration	700	Engineering Demonstration
AKME-engineering		SVBR-100	Russia	Prototype	280	Detailed Design
Newcleo		LFR-AS-200	UK	Prototype	480	Conceptual Design
Westinghouse		W-LFR	USA	Prototype	950	Conceptual Design
Blykalla		SEALER-55	Sweden	Commercial	140	Conceptual Design
Ansaldo Nucleare		ALFRED	EU	Demonstration	300	Conceptual Design
Ansaldo Nucleare		EAGLES-300	EU	Demonstration	1500	Conceptual Design
KTH		SUNRISE	Sweden	Experimental	0.5	Conceptual Design
SCK CEN		MYRRHA	Belgium	Experimental / Accelerator	100	Conceptual Design
Natura Resources	**MSR**	MSR-1	USA	Experimental	1	Engineering Demonstration

Company	Type	Reactor	Country	Purpose	Power [MW_{th}]	State of development
Kairos Power		HERMES	USA	Demonstration	35 demo, 320 full-scale	Engineering Demonstration
Copenhagen Atomics		Copenhagen Atomics Waste Burner	Denmark	Commercial	100	Engineering Demonstration
Terrestrial Energy		IMSR400	USA	Commercial	442	Detailed Design
Thorcon International		Thorcon 500	USA	Commercial	557	Detailed Design
NAAREA		XAMR	France	Commercial	80	Basic Design
Moltex Energy		SSR-W	Canada	Prototype	750	Conceptual Design
Saltfoss Energy (Seaborg)		CMSR	Denmark	Commercial	250	Conceptual Design
Thorizon		Thorizon	Netherlands	Prototype	250	Conceptual Design
Stellaria		Stellarium	France	Prototype	250	Conceptual Design
Centrum výzkumu Řež s.r.o.		Energy Well	Czech Republic	Commercial	20	Conceptual Design
Flibe Energy		LFTR	USA	Commercial	600	Conceptual Design
CNRS		MSFR	France	Demonstration	3000	Conceptual Design
MoltexFLEX		FLEX Reactor	UK	Prototype	60	Conceptual Design
International Thorium Molten-Salt Forum (ITMSF)		FUJI	Japan	Commercial	450	Conceptual Design
General Atomics	**GFR**	EM2	USA	Commercial	500	Conceptual Design
General Atomics		FMR	USA	Commercial	100	Conceptual Design
CREIPI		Kamado FBR	Japan	Demonstration	3000	Conceptual Design
Euratom		ALLEGRO	EU	Demonstration	75	Conceptual Design
NPIC	**SCWR**	CSR1000	China	Demonstration	2300	Conceptual Design
KIT		HP-LWR	EU	Demonstration	2300	Conceptual Design
Toshiba		JSCWR	Japan	Demonstration	3681	Conceptual Design

Appendix 3. Figure of currently ongoing GEN IV projects with vendor-estimated deployment dates.



Appendix 4. Historical sodium-cooled fast reactors (Adapted from PRIS, 2025).

Reactor	**P [MW_{th}]**	**Type**	**Country**	**Operating period**	**Operation factor [%]**	**Load factor [%]**
EBR-I	1.4	Experimental	USA	1951-1963	-	-
EBR-II	62.5	Experimental	USA	1963-1994	-	-
Fermi 1	200	Experimental	USA	1963-1975	-	-
SEFOR	20	Experimental	USA	1969-1972	-	-
Fast flux test facility	400	Experimental	USA	1980-1993	-	-
Dounreay FR	65	Experimental	UK	1959-1977	34.3	33.8
Prototype FR	250	Demonstration	UK	1974-1994	38.2	26.9
Rapsodie	40	Experimental	France	1967-1983	-	-
Phenix	563	Demonstration	France	1973-2009	41.2	40.5
Superphenix	3000	Commercial	France	1985-1998	14.4	7.9
KNK 2	58	Experimental	Germany	1972-1991	26.8	17.1
FBTR	40	Experimental	India	1985-	-	-
Monju	714	Demonstration	Japan	1994-1996, 2010.	-	-
Joyo	140	Experimental	Japan	1978-2007, 2021-	-	-
BN-350	750	Demonstration	Kazakhstan	1972-1999	51.6	43.4
BR 1/2	1	Experimental	Russia	1955-1956	-	-
BR-5	5	Research	Russia	1958-1971	-	-
BOR-60	60	Research	Russia	1969-	-	-
BR 10 Obninsk	8	Research	Russia	1973-2002	-	-
BN-600	1470	Commercial	Russia	1980-	78.4	76
BN-800	2100	Commercial	Russia	2014-	71.1	67.2
CEFR	65	Experimental	China	2010-	-	-

Appendix 5. Identified major initiating events and corresponding safety systems.

Initiating event	Corresponding safety system
VHTR	
Gas circulator failure (loss of forced cooling)	Residual heat removal system (through cavity cooling or SG)
Primary leak in boundary (air ingress)	Coolant leak detection system
Primary IHX interface failure (water ingress)	Secondary circuit draining system
Overpressurization of primary system	Pressure relief valves (primary circuit)
Primary relief valve accidental opening	Pressure relief valves (primary circuit)
Catastrophic failure of coolant piping	Reactor cavity cooling system (ECCS)
Failure of control rod insertion	Secondary reactivity control system (absorber pebbles), emergency shutdown control rods
SFR	
Loss of primary coolant flow	Residual heat removal system (secondary sodium to air / water)
Loss of secondary flow	Emergency residual heat removal system (in case of all secondary sodium circuits fail)
Sodium leakage from reactor vessel	Sodium leak detection system, fire protection system
Sodium leakage from secondary circuit	Secondary circuit emergency drain system, fire protection system
Steam generator tube rupture (water ingress)	Steam generator drain system
Coolant solidification	Coolant heating system
Primary coolant contamination / blockage formation	Cover gas system, coolant purification system
Failure of control rod insertion	Emergency shutdown control rods
LFR	
Loss of primary coolant flow	Residual heat removal system
Steam generator tube rupture (water ingress)	Steam generator drain system, emergency residual heat removal system (through vessel cooling)
Primary coolant contamination / blockage formation	Cover gas system, coolant purification system
Air ingress to primary coolant	Cover gas system, coolant purification system
Coolant solidification	Coolant heating system
Lead leakage from primary system	Coolant leakage containment system
Failure of control rod insertion	Emergency shutdown control rods
MSR	
Loss of primary coolant flow	Residual heat removal system
Primary coolant contamination (air ingress) / blockage formation	Salt processing system, off-gas system
Failure / leakage of the off-gas system	Waste gas containment system
Water ingress due to heat exchanger failure	Secondary circuit draining system
Inadvertent freeze valve opening	Freeze valves (fuel salt drain system)
Excessive graphite radiation damage (if graphite moderated)	Graphite condition monitoring system
Fuel salt leakage	Leaked salt catch-pans, fuel salt drain system
Waste salt leakage	Waste salt storage system
Coolant solidification	Coolant heating system
Failure of control rod insertion	Fuel salt drain system (with freeze valves), emergency shutdown control rods