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Literature Review of Emerging SMR Technologies

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Executive summary

The purpose of this report is to identify systems that are termed Emerging Small Modular Reactor Technologies and determine barriers to their deployment. Emerging Small Modular Reactor Technologies refers to Small Modular Reactor (SMR) systems that are receiving commercial interest but are unlikely to be ready for commercial deployment before 2030. SMRs are defined as reactor systems that operate below 300 MWe and are designed to take full advantage of modularisation. Therefore most of the fabrication is carried out in a factory setting, thereby reducing construction times and exploiting fully the costs reductions associated with gaining experience when producing many units.

The following reactor systems have small modular variants and were identified as Emerging SMR Technologies: Sodium-cooled Fast Reactors (SFRs); Lead-cooled Fast Reactors (LFRs); Molten Salt Reactors (MSRs); and High Temperature Gas Reactors (HTGRs). None of these reactor systems have been deployed commercially but all of these systems have at least been operational as small power experimental units.

Many of the Emerging Small Modular Reactor Technologies are fast reactor systems. Fast Reactors are characterised by the large proportion of high energy neutrons residing within the reactor core. Operating with high energy neutrons allows the reactor to extract far more energy from the fuel than is achievable with current reactors. Therefore, considerable benefits arise regarding fuel utilisation and reductions in high level waste volumes. However, Fast Reactors are inherently more difficult to design and operate than their thermal reactor counterparts, in part because thermal reactors exhibit very favourable reactivity feedback characteristics.

Beyond the general benefits attributable to Fast Reactors, SFRs offer considerable safety improvements with respect to managing the heat produced from nuclear fuel under accident conditions. Sodium's reasonably high boiling point at atmospheric pressure and excellent heat transfer properties allow SFRs to remove decay heat without the need for active (externally powered) systems nor sacrificing high power densities. This is a considerable benefit over current commercial reactor systems. However, this benefit needs to be balanced against sodium's high chemical reactivity with water and air. In addition, the optical opaqueness and chemical reactivity of liquid sodium makes In-service Inspection and Repair (ISI&R) difficult.

LFRs are far less mature than SFRs. The coolants employed in LFRs do not exhibit high chemical reactivity with water and air, which is a significant advantage that lead-based coolants have over liquid sodium. However, LFRs face major deployment barriers with respect to the material challenges posed by lead-based coolants. The coolants employed in LFRs are very corrosive and relatively low coolant velocities can result in excessive erosion of reactor structural materials. With current materials it is necessary to accurately control oxygen concentrations in LFRs which is yet to be proven at a commercial scale under prototypic LFR conditions. Furthermore, the high density and high melting point of lead-based coolants result in even more challenging ISI&R environments than those facing SFRs.

MSRs are unique with respect to the other emerging technologies as they can readily operate with either fast or thermal spectra depending on the choice of materials employed. The liquid fuel nature of MSRs, and the fact that in most MSRs the liquid fuel also functions as the coolant, departs radically from existing solid fuel experience. The use of a liquid fuel that also functions as the coolant allows for extensive online reprocessing to significantly improve fuel efficiency. Molten salt coolants exhibit similar benefits and drawbacks to lead-based coolants; both coolants exhibit high boiling points and are less reactive with water and air than sodium but their high melting points and corrosiveness raises difficulties.

The difficulties associated with deploying MSRs depends on the extent of online reprocessing and how far designers aim to replicate historical experience that focused on low power density, thermal spectrum reactors. Furthermore, most experience with molten salt coolants is based on molten fluoride salts. Hence, MSRs operating with graphite moderators and with reprocessing limited to elements that naturally separate from the liquid fuel are likely to be ready for deployment sooner compared with fast spectrum MSRs with extensive online reprocessing. However, both MSR types are far less mature than SFRs and HTGRs.

HTGRs are able to draw upon the considerable historic and ongoing research programmes. Compared with the other emerging technologies their designs are significantly less complex, with a heavy reliance on a well-tested fuel form to confer safety and performance benefits. Furthermore, reactivity control in thermal reactors is easier to achieve under a variety of scenarios than with fast spectrum systems. The material challenges with HTGRs are heavily dependent on coolant temperatures and are known to be of limited concern if the helium coolant is less than around 800°C. A considerable disadvantage with HTGRs is the difficulty in reprocessing the fuel form; however, the counterargument is that this increases proliferation resistance and fuel utilisation is currently of no concern when uranium is widely available. Yet spent fuel volumes will become considerable in the event HTGRs are widely deployed.

The lack of experience licensing HTGRs and the known issues surrounding graphite oxidation and the production of graphite dust embedded with radionuclides, will likely delay HTGR deployment. In addition, the absence of commercial suppliers of large quantities of HTGR fuel and potential issues surrounding procuring fuel enriched beyond 5 wt.%, are sufficient to warrant it unlikely that HTGRs will be deployed commercially before 2030.

There is no reason to believe that commercial deployment of HTGRs could not be achieved by around 2035 and HTGRs are very likely to be commercially deployable before the other emerging technologies. With respect to the other small modular systems, SFRs are closer to commercial deployment than MSR and LFRs. Low power density thermal spectrum MSRs, with limited online reprocessing, and LFRs, are at comparable levels of maturity. Fast spectrum MSRs with extensive online reprocessing and operating at high power densities are the least mature and likely to take the longest time to deploy.

1. Introduction

The purpose of this report is to review so-called Emerging Small Modular Reactor Technologies. Emerging Small Modular Reactor Technologies refers to Small Modular Reactor (SMR) systems that are receiving commercial interest but are unlikely to be ready for generating electricity, space heating or heat to drive industrial processes for a customer on a commercial basis before 2030.

SMRs are defined as reactor systems that operate below 300 MWe and are designed to take advantage of modularisation. Therefore most of the fabrication is carried out in a factory setting, thereby reducing construction times and exploiting fully the costs reductions associated with gaining experience when producing many units.

Given the above definition, and the information in subsequent chapters, the following reactor systems belong to the Emerging SMR Technology category:

- Sodium-cooled Fast Reactors;
- Lead-cooled Fast Reactors;
- Molten Salt Reactors; and
- High Temperature Gas Reactors.

There are varying degrees of experience with all of these reactor systems, with historical attempts, and ongoing research, to deploy near-commercial Sodium-cooled Fast Reactors and High Temperature Gas Reactors in both small and large sizes. On the other hand there have never been any near-commercial Lead-cooled Fast Reactors or Molten Salt Reactors deployed. Nevertheless, both LFRs and MSR systems have been deployed as low power military propulsion systems and small scale research units, respectively.

Historically, the development of new reactor systems initially focused on building a low powered (less than ~100 MWth) experimental system and, once sufficient experience had been gained, a near-commercial demonstration system would be constructed. These near-commercial systems typically had much higher powers (~1000 MWth). The reason behind increasing powers was that the cost of demanding more power from systems was relatively small and hence operating at high power levels improved their economic performance. However, sometimes the greater power demands had detrimental impacts on the operational performance of reactor systems. Hence, there is an interest in focusing on small reactors with comparable power outputs to early experimental systems. It is hoped that benefits associated with improved operational performance and/or exploiting fully the costs reductions from gaining experience when producing many units can compensate for the loss of benefits associated with operating systems at high powers.

In order for a reactor design to successfully pass through a licensing regime it must demonstrate that the various systems (such as fuels, pumps and structural materials) will operate within their design limits. Therefore, most Emerging SMR Technologies aim to utilise components and systems that have previously been demonstrated to operate successfully under conditions similar or identical to those within their reactor design. By attempting to rely on technologies that have prior pedigree, this should allow reactor designers to minimise the amount of expensive experimental programmes that must be performed to determine the design limits. Furthermore, such a pragmatic approach considerably reduces the risks associated with attempting to fully commercialise a technology with limited experience. New technologies may turn out, after further testing, to be incapable of operating in the required manner, for example, the structural materials envisaged for deployment may rapidly degrade in the presence of an irradiation field.

This report reviews the technologies underpinning the reactor systems of interest. Therefore, given that many of the technologies reactor vendors are aiming to deploy are meant to be relatively mature, much of the review is focused on historical technology. Hence, there are very few discussions on the plethora of novel technologies (such as advanced materials and surface coatings) that may allow for significant improvement in reactor performance but necessitate lengthy test programmes to establish their performance benefits.

Many of the Emerging Small Modular Reactor Technologies are fast reactor systems. Fast Reactors are characterised by the large proportion of high energy neutrons residing within the reactor core. Operating with high energy neutrons allows the reactor to extract far more energy from the fuel than is achievable with current reactors. Therefore, considerable benefits arise regarding fuel utilisation¹ and reductions in high level waste volumes.

Fast Reactors ensure neutrons have high energies by minimising their interaction with other materials within the reactor core. This is achieved by reducing the quantity of materials within the core that readily absorbs energy from neutrons and by keeping the reactor core compact. Hence Fast Reactors typically operate at very high power densities. The large proportion of highly energetic neutrons present in the core and high power densities makes the material challenges in Fast Reactors very significant. Furthermore, Fast Reactors have inherent characteristics that makes reactivity control more difficult in comparison with current reactor systems, such as Pressurised Water Reactors (PWRs). Hence, the drawbacks associated with fast reactor technologies are that they are inherently more difficult to design and operate.

Current reactors operate with low energy neutrons referred to as thermal neutrons. These thermal reactors, which include PWRs and HTGRs, extract only a small fraction (<1%) of the available energy from the fuel. However, their design and operation are generally simpler than Fast Reactors, in part because thermal reactors exhibit very favourable reactivity feedback characteristics.

¹ Note that fuel utilisation only becomes important in the event that uranium availability becomes an issue.

2. Sodium-cooled Fast Reactor

Sodium-cooled Fast Reactors (SFRs) utilise liquid sodium as the coolant, with typical inlet temperatures around 400°C and outlet temperatures around 550°C [1]. The use of sodium as a coolant has the following favourable characteristics:

- It exhibits excellent thermal properties, in particular, a very high thermal conductivity and a high heat capacity;
- It has a reasonably high boiling point (880°C) at one atmosphere of pressure; and
- Unlike many other potential coolant media it offers good chemical compatibility with conventional structural materials.

The latter point is important as many coolants, such as water, molten salts and lead-based alloys, are very corrosive at temperatures around 500°C; thereby leading to degradation of key structural components. The corrosiveness of molten salts and lead-based alloys either results in considerable uncertainties regarding material performance during long residence times (>5 years); necessitates further lengthy R&D into suitable material candidates and/or requires large components within the primary circuit to be routinely replaced, which is economically detrimental.

The prime disadvantages related to sodium as a coolant are:

- Its optical opaqueness makes it difficult to visually monitor the core which results in some complications regarding fuel handling operations and significant difficulties relating to In-Service Inspection and Repair (ISI&R) in comparison with transparent coolant media (such as water and some molten salts);
- It undergoes an exothermic chemical reaction with water and air;
- The relatively low mass of the sodium nucleus which appreciably slows down neutrons relative to heavier metals, such as lead, adversely impacts neutron economy; and
- The irradiation of sodium (which naturally consists solely of Na-23) results in the production of the problematic isotopes Na-24 and Na-22 which have half-lives of 15 hours and 2.6 years, respectively. The relatively short half-life of Na-24 and the penetrating gamma rays produced during its decay are part of the reason behind necessitating an intermediate cooling loop, which significantly increases plant costs. The quantities of Na-22 produced are relatively small which permits work to be performed on the primary circuit after only a few days after shutdown.

Many coolant types were initially screened during the early development of fast reactors. Sodium became the preferred coolant medium for fast reactors due to its inherent favourable thermal and material compatibility properties. It was believed that the disadvantages associated with sodium could be engineered around; for instance the risk associated chemical with incompatibility of sodium with water and air can be considerably reduced using a combination of inert gases, containment, detection, isolation and countermeasures in the event sodium comes into contact with water or air.

There have been numerous leaks of sodium from SFRs (with a number of cases highlighted in Section 2.2.3). However to the author's knowledge there have been no personnel injuries associated with sodium leaks even though many of the leaks have caused considerable interruption to plant operation thereby significantly reducing plant capacity factors.

An important differentiator between SFR designs is the layout of the primary system which are either of pool or loop type configurations (see Figure 1). In the pool type configuration the entire primary circuit (i.e., the reactor, primary pumps and intermediate heat exchangers) is located in a large sodium pool reactor tank. In the loop type configuration the primary pumps and intermediate heat exchangers are located outside the reactor vessel via interconnecting piping. Neither primary circuit implementation is superior to the other and both pool and loop designs have their proponents, with Japan investing heavily in the development of loop type SFR systems and the US, UK, Russia and France preferring pool type SFR designs. Table 1 lists the relative advantages and disadvantages of the pool and loop type SFR implementations.

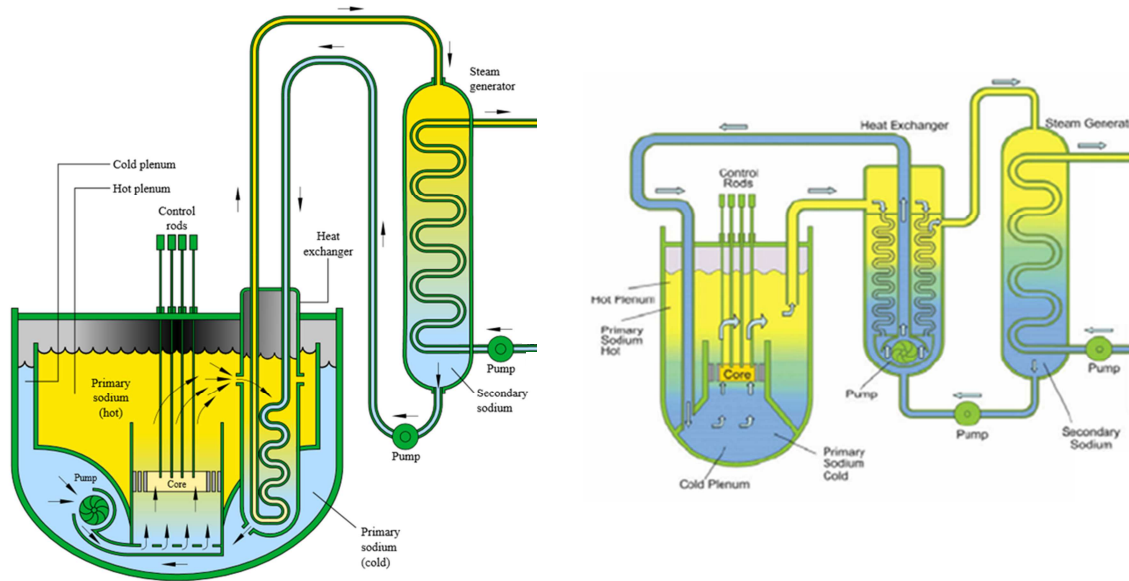


Figure 1: Diagram showing the difference between pool and loop type Sodium-cooled Fast Reactor systems.

Table 1: Advantages (+) and disadvantages (-) of pool and loop type sodium-cooled fast reactor designs [5].

	Pool type	Loop type
Economics	- Large vessel size, which increases capital costs	+ Small vessel size, which decreases capital costs
	+ Smaller reactor building	- Larger reactor building
Safety	+ Slower and milder transients	- Faster transients due to less thermal inertia
	+ Reduced probability of breach in primary circuit resulting in fuel damage	- Greater possibility of sodium leakage
Operation and Inspection	- More difficult to perform in-service inspection and repair	+ Easier in-service inspection and repair
	- Greater shielding requirement as heat exchangers containing sodium neighbour active core	+ Overall less shielding; however, external loops carrying activated sodium and therefore require shielding

Table 2 details the small modular Sodium-cooled Fast Reactors referred to in this report. (Note that in Table 2, transients refer to undesirable changes in reactor parameters (e.g. coolant temperature) that can occur, for instance, from an increase or decrease in the electrical load on the turbine generator.) Whilst PRISM is above the 300 MWe threshold for SMRs it is marketed as a Small Modular Reactor and has therefore been included. ASTRID is not an SMR but is a modern design with considerable design information in the public domain along with detailed reports summarising challenges the designers are trying to overcome. Therefore, ASTRID is referenced heavily in this report and is included for comparison.

Table 2: List of Small Modular Sodium-cooled Fast Reactors that are currently being progressed and have design information in the public domain [2, 3, 4, 24, 117, 118].

Reactor Name	Thermal/Electrical Output	Fuel Form	Primary system layout	Outlet Temperature
PRISM	840 MWth/311 MWe	U-Pu-Zr alloy in HT9 clad	Pool	499°C
ARC-100	260 MWth/100 MWe	U-Zr alloy in HT9 clad	Pool	510°C
Toshiba 4S	30 MWth/ 10 MWe – 135 MWth/ 30 MWe	U-Zr alloy in HT9 clad	Pool	510°C
Travelling Wave Reactor	737.5 MWth/ 300 MWe	U-Zr alloy in HT9 clad	Pool	500°C
ASTRID	1500 MWth/ 600 MWe	UO ₂ in AIM1 clad	Pool	550°C

2.1. Materials and Chemistry

2.1.1. Fuel and Cladding Materials

Metallic (U-Pu-Zr and U-Zr) and oxide fuels (U,Pu)O₂ have undergone considerable research, with metallic fuels being favoured in the USA and oxide fuel experience focused in Europe and Russia. Whilst the level of experience with oxide fuel is significantly larger than metallic fuel experience (see Appendix A1.2.), both fuel types have been shown to perform well under SFR conditions and acceptable conversion ratios (the net production of fissile material) can be achieved with either fuel type [13]. Furthermore, both fuel types have been shown capable of achieving very high burnups (where burnup refers to the amount of energy generated per unit mass of fuel)² but, again, there is significantly more data available surrounding the performance of high burnup oxide fuel relative to U-Pu-Zr and U-Zr fuels [14, 18].

Table 3: Comparison of the properties of fast reactor fuels where the Pu metal fraction = 0.2. *500°C for U-Pu-Zr [6].

	(U,Pu)O ₂	(U,Pu)C	(U,Pu)N	U-Pu-Zr
Density (g/cm ³)	11.0	13.6	14.3	15.6
Heavy metal density (g/cm ³)	9.7	12.9	13.5	14
Melting temperature				
Liquidus (°C)	2775	2480	2780	1160
Solidus (°C)	2740	2325	2720	
Thermal conductivity at 1000°C* (W m ⁻¹ K ⁻¹)	2.9	19.6	19.8	35

² Note that generally speaking all reactors try to maximise the energy generated from the fuel that resides within the core (i.e. they target high burnup). However, there are often constraints such as ability of materials to withstand high burnups that limit the highest achievable burnup.

Thermal expansion from 20°C to 1000°C* (10 ⁻⁶ /°C)	12.6	12.4	10.0	16.5
Compatibility with water	Good	Poor	Average	Bad
Compatibility with sodium	Poor	Good	Good	Good
Dissolution in nitric acid (HNO ₃)	Yes	No	Yes	No

A comparison of key physical characteristics of fast reactor fuels, including the more experimental nitride and carbide fuel forms, are shown in Table 3. Oxide fuel exhibits a significantly lower heavy metal density than U-Pu-Zr and oxygen is also weakly moderating, which adversely impacts the neutron economy. The main drawback associated with these two characteristics is a lower conversion ratio than is achievable with U-Pu-Zr fuel; however, given the current abundance of uranium, the importance of superior conversion ratios is arguably limited in the near-term.

Oxide fuel undergoes a chemical reaction with sodium but experience has shown that this behaviour is not overly detrimental (see Appendix A1.1.). The high melting point and poor thermal conductivity of oxide fuel and the low melting point but high thermal conductivity of metallic fuel results in similar fractional margins to melt for these two fuel types ($T_{max}/T_{melt} \approx 0.8$ for both fuel types).

Besides oxide fuel being used extensively in LWRs and therefore considerable experience in its behaviour and manufacturability, the low swelling rate of oxide fuel was another reason behind the early adoption of oxide fuel in fast reactors [7]. The low swelling rate attributable to oxide fuels helps reduce the stress applied to the cladding material; high swelling rates are a major reason behind clad failure at high burnups.

Metallic fuel has historically suffered from large amounts of swelling under irradiation. However, strategies to overcome swelling have been successfully developed. Firstly, fabricating the fuel with relatively high porosity helps reduce net swelling under irradiation. Secondly, taking advantage of the good chemical compatibility between U-Pu-Zr and sodium, which allows for a relatively large space to exist between the fuel slug and clad if the gap is filled with liquid sodium (an excellent heat transfer medium), also considerably helps in reducing stress applied to the cladding material [7]. Hence, very high burnups have been achieved in metallic fuel that are comparable to those achieved in oxide fuel (~200 Gigawatt-days per tonne of Heavy Metal (Gwd/tHM), where the term heavy metal refers to elements with atomic numbers greater than actinium) [17].

Research in metallic fuels for sodium fast reactors initially included unalloyed uranium and various uranium alloys (e.g. U-Mo, U-Zr, U-Cr and Pu-Al) [91]. EBR-II was started up and operated with an alloy denoted U-5 wt.% Fs, where Fs stands for fission, a mixture of metals: 2.46M, 1.96Ru, 0.28Rh, 0.19Pd, 0.1Zr, and 0.01Nb (in wt.%). The bulk of the metal fuel database for sodium fast reactors is dominated by U-5Fs. Although the properties of U-5Fs were sufficient for EBR-II operation, it was found that its swelling behaviour was unfavourable and the addition of plutonium also resulted in chemical interactions with cladding materials which reduced fuel melting temperatures [7, 17, 91].

Due to the limitations associated with U-5Fs, alternative U alloys were investigated in the U.S. U-Zr and U-Pu-Zr received significant interest, with around 13 000 U-Zr rods and around 700 U-Pu-Zr rods being irradiated [91, 92]. A large proportion of the irradiated U-Pu-Zr rods were around one third the length of conventional fast reactor rods and many were clad in D9 rather than HT9 [92].

U-Pu-Zr and U-Zr undergo fuel constituent redistribution whilst the fuel is operating within the reactor, with Zr migrating towards the fuel centre and U migrating towards the clad surface. Plutonium does not show significant redistribution [17]. Zr redistribution is more pronounced in U-Pu-Zr than in U-Zr fuel [91]. Redistribution has significant effects on the material properties, for instance Zr depleted regions exhibit lower solidus temperatures than the original U-Pu-Zr fuel [91]; however, under normal operating condition the fuel temperatures should not drop below the solidus temperature of the Zr depleted region in U-Pu-Zr fuel [91].

During irradiation, cladding constituents are known to diffuse into the U-Zr and U-Pu-Zr fuels which can result in a mixture forming between the clad and fuel (referred to as a eutectic) with a lower

melting point than either the clad or original fuel material [93]. Diffusion of cladding constituents into the fuel is more pronounced in U-Pu-Zr than U-Zr fuel [91]. The eutectic formed between U-Pu-Zr fuel and the clad has been shown to undergo melting at temperatures as low as 675°C, such temperatures can arise under certain transient conditions in sodium fast reactors [91]. Hence there is interest in incorporating a Zr liner within the cladding material for U-Pu-Zr to limit the formation of eutectics between the clad and fuel [94]. However, irradiation tests on metallic fuel that incorporates a Zr liner are relatively limited, with peak burnups up to around 30 GWd/tHM being performed [95].

Significant work has been undertaken to determine the behaviour of oxide fuel under transient conditions, for instance, the flow disturbance and overpower transient tests performed in the Scarabee reactor [96]. These tests simulated transients caused by a blockage accident and/or increases in rod power. The results from these tests demonstrated that whilst melting of the oxide fuel did take place, in most cases the fuel rods remained intact. Only under the most extreme scenarios, such as reductions in coolant flow rate by 91% or rod powers increasing by around a factor of 4, did fuel rod failure occur.

It must be stated that whilst metallic and oxide fuels have achieved very high burnups (~200 GWd/tHM), the database at such high burnups for both fuel types is limited [15, 16]. Most data on routinely achievable burnups under SFR conditions have been around 100 GWd/tHM. Whilst the licensing process may be prolonged for metallic fuels relative to oxide fuels (because of the smaller performance database relative to oxide fuel) there should be few reasons limiting licensing of SFR fuel up to around 100 GWd/tHM for both fuel types. However, this will likely not be the case if whole core burnup within the SFR design approaches 200 GWd/tHM for either fuel types [15, 16]. There is generally a desire for SFR fuel to target burnups ~200 GWd/tHM in order to improve economic performance by minimising fuel costs and outages associated with lower burnup core designs.

To gain sufficient confidence in the ability of fuel rods to achieve burnups ~200 GWd/tHM requires fuel performance data that includes information on changes in fuel dimension and temperature as a function of time. In addition, post-irradiation measurement on the chemical composition of the fuel, clad damage, fuel microstructure and porosity would also be required. Furthermore, there will inevitably be some scatter in the data due to different experimental procedures, stochastic defects in manufactured fuel rods and varying irradiation environments (neutron and gamma fluxes along with coolant temperature conditions); hence the need for detailed results from a large number of fuel rods with burnups ~200 GWd/tHM. Finally, the behaviour of fuel rods at burnups of ~200 GWd/tHM must be exposed to a variety of transient conditions in order to ensure performance is satisfactory under normal and off-normal conditions.

The focus of this report is on the current status of the most mature reactor system technologies. With respect to cladding materials there are a large number of potential cladding choices ranging from materials that have been manufactured into fuel rods and performed well up to high burnups (such as HT9), to materials that have not achieved either of these criteria. Only the relatively mature cladding concepts are discussed in this report.

Two material types have been heavily investigated for use in SFRs: austenitic steel and Ferritic-Martensitic (FM) steel. These two types of steel are differentiated by their crystal structure and exhibit different properties.

Austenitic steels tend to exhibit good creep strength at high temperatures (~600°C) and resistance to corrosion but have the disadvantages of low thermal conductivity, poor swelling behaviour under irradiation and are susceptible to embrittlement (loss of ductility) [7, 8]. FM steels on the other hand tend to exhibit greater resistance to embrittlement and irradiation induced swelling and they also have higher thermal conductivities. However, FM steels generally show low creep strength and limited corrosion resistance [7, 8].

High energy neutron irradiation, which is typical in fast reactors, results in atoms being displaced from their original location within materials thereby creating defects such as vacancies in the material's crystal structure. Hence, the extent of irradiation damage in materials is measured in terms of displacements per atom (dpa).

Vacancies can agglomerate leading to so-called void swelling. Void swelling is a major life limiting phenomenon and is responsible for the irradiation induced swelling witnessed in cladding materials in fast reactors at high neutron doses [7].

Ideally cladding materials would be capable of routinely achieving doses in the range 150 – 200 displacements per atom (dpa), whilst being exposed to coolant temperatures up to around 700°C, in order to improve the economic performance of the reactor system [7, 19]. Moreover, cladding materials should reduce the concentration of highly neutron absorbing elements (such as nickel) to minimise the parasitic loss of neutrons.

Early SFRs employed austenitic steel type 316. However, the extent of irradiation induced swelling became excessive at doses around 50 dpa, which limited steel 316's use in fuels to relatively low burnups [7]. Therefore, interest in FM steels as a cladding material gathered pace. However, in some reactors utilising oxide fuel (where fuel and cladding temperatures tend to be high) the improved high temperature strength of austenitic steels was sufficient to warrant further development of austenitic steels. For instance, the ASTRID reactor plans to utilise austenitic steel AIM1 (Austenitic Improved Material 1) cladding material which has been demonstrated to exhibit good swelling resistance up to around 100 dpa [24]. To achieve higher irradiation swelling resistance, work is under way to develop an improved variant of AIM1 called AIM2, which is envisaged to exhibit swelling resistance up to 120 dpa [24].

FM steels that have received considerable attention include HT9 (9Cr-1.8W-0.5Mo) and T91 (Mod 9Cr-1Mo). HT9 is a so-called first generation FM steel and is a popular choice for cladding material in the SFRs listed in Table 2. HT9 has successfully operated at irradiations up to 155 dpa but only at relatively low temperatures (440°C) [15]. Peak clad temperatures in SFRs can reach approach 700°C under certain circumstances. There is significant uncertainty on the performance of HT9 cladding at high doses (> 100 dpa) and high temperatures (~600°C) [15]. Furthermore, there may be issues surrounding the availability of large quantities of HT9 material from nuclear qualified manufacturers which may postpone the deployment of SFRs aiming to employ this cladding material [15].

T91 shows improved high temperature performance relative to HT9 but the extent of irradiation performance data is far less substantial than HT9 [15, 19]. The limited irradiation performance data probably explains why many SFR designs that are aiming to achieve commercial deployment over relatively short time frames have decided against employing T91 as a cladding material.

2.1.2. Structural Materials

Austenitic steels are favoured for use as structural materials in ASTRID. Austenitic steels contain large quantities of chromium (>13 wt.%), which forms a passive layer of chromium oxide (Cr_2O_3). As such, their general oxidation and corrosion behaviour is good. However, stress has been shown to aggravate corrosion resistance, as observed in type 316 steel, where chromium carbides particles form at the grain boundaries. This can give rise to stress corrosion cracking (SCC). Lower carbon steels, for example 316L, have shown significant improvements in SCC behaviour but the reduced carbon content also results in decreased strength. The addition of nitrogen to 316L, resulting in 316L(N), is known to offset this loss of strength whilst still maintaining good SCC behaviour [20]. 316L(N) is a candidate for many of the reactor components (pumps, heat exchangers, vessel surfaces and the Auxiliary Cooling System (ACS)) in French and Indian SFRs [21].

Austenitic steels have been successfully deployed in a number of SFR systems [21]. Furthermore, such steels have a large R&D database; they are resistant to liquid metal degradation and exhibit favourable high temperature properties.

2.1.3. Coolant Chemistry

The primary aim during SFR operation is to minimise the oxygen levels within the primary sodium loop in order to limit the extent of corrosion [7]. Typical oxygen limits are < 5 ppm with France operating a more stringent oxygen limit (< 3 ppm). At oxygen concentrations less than 3 ppm French SFRs have not suffered from chemistry related issues [18].

2.2. Components and Instrumentation

2.2.1. Pump Technologies

Two pump types have been considered for use in Sodium-cooled Fast Reactors: mechanical and Electro-Magnetic (EM) pumps. Mechanically driven pumps have been used extensively throughout the world in SFRs and have in general exhibited good reliability performance [22]. Mechanical pumps are also around 50% more efficient than their EM counterparts. In addition, their more compact design relative to EM pumps makes mechanical pumps better suited for locations where space is limited, such as inside the primary vessel.

EM pumps have only been used in small reactors and have had limited use in the primary loops of SFRs [1]. Given their large size relative to mechanical pumps there is a preference to locate EM pumps outside of the primary vessel and hence they have been used in secondary circuits. In principle, EM pumps offer superior reliability since they contain no seals or bearings that are subject to failure or erosion, but have low inertia that may need to be enhanced artificially with an engineered system to ensure adequate cooling in the event of power loss (for example, PRISM uses EM pumps in the primary circuit coupled to pony motors that drive a flywheel that provides the required inertia). The inherent reliability of EM pumps is intended to offset their higher upfront costs and decreased efficiency.

2.2.2. In-service Inspection and Repair

A major challenge that must be overcome to ensure successful, economically competitive, deployment of Sodium-cooled Fast Reactors is the need to achieve Capacity Factors (CFs) comparable to modern Light Water Reactors (i.e. > 90%). The Russian BN-600 SFR has achieved a CF of ~75%, which is the highest capacity factor achieved for any SFR, but is significantly lower than those achieved by modern LWRs [5]. The BN-600 system has recorded a number of sodium leaks during its operation which have ultimately limited its ability to achieve high capacity factors [5]. Hence it is important that the reliability and availability of SFR systems are significantly improved.

A dominant contributing factor behind LWRs' high CFs is the use of effective In-service Inspection and Repair (ISI&R). The need to develop similarly effective ISI&R methods for SFRs is currently the subject of a large R&D programme and is considered a major challenge [9]. A considerable complication with ISI&R in liquid metal-cooled fast reactors is the opaqueness of the coolant medium prohibiting visual inspection. There are a number of ways to try and compensate for this property including [9]:

- Simplify reactor design so that the number of structures and components requiring inspection is reduced;
- Development of measurement techniques for continuous monitoring during reactor operation and during periodic reactor shutdown; and
- Utilising remotely controlled devices (robots) that are capable of performing ISI&R.

All of these strategies are being developed for improving ISI&R for the ASTRID R&D programme [9]. In the areas of monitoring: core power; sodium flow rates; sodium temperatures; fuel failure; leak

detection and coolant quality, there is significant experience from previous and current SFR programmes and these areas can be considered relatively mature [1, 10].

Considerable historical effort has been undertaken to develop ultrasonic viewing techniques for use in liquid sodium-cooled fast reactors [106]. Ultrasonic viewing techniques in SFRs are capable of achieving resolutions of ~1mm and have proved successful in inspecting core components, such as the fuel subassemblies in the UK's Prototype Fast Reactor [107]. However, there are a number of limitations that surround ultrasonic viewing techniques. Firstly, ultrasonic viewing systems are known to perform relatively well at temperatures up to around 260°C in liquid sodium. However, within the primary circuit, such temperatures are only achievable under shutdown conditions. This implies that preference is given to performing inspection when the reactor is not operating and this can affect a system's capacity factor.

Secondly, historical data indicates that fatigue cracks of less than approximately 25% of the material's wall thickness were not detectable in immersed sodium by ultrasonic viewing techniques [108]. This is problematic as through-wall cracks can significantly degrade a component's mechanical properties. Further work is required to overcome this shortcoming using ultrasonic inspection techniques.

Besides ultrasound technologies, eddy current based methods have been used to support fast reactor programmes [108]. Although eddy current based methods are generally very good at detecting small defects in materials such as surface-breaking cracks [PNNL-16253], much of the experience for nuclear applications has been gathered under LWR conditions.

A challenge for utilising eddy current inspection techniques in SFRs is the relatively high electrical conductivity of the liquid sodium coolant. The high electrical conductivity not only attenuates the signal between the probe and workpiece but also acts as conductive path between crack faces [109]. Furthermore, the probe must be guided to the location where inspection is to be performed within an opaque liquid medium. The use of ultrasound methods for imaging and careful consideration during the design process of the reactor regarding how to accommodate inspection devices would considerably help in the process of guiding sensors.

Reliably detecting defects and performing maintenance whilst minimising disruption to the operating plant is relatively immature, especially in comparison with ISI&R in LWRs. For non-removable components that are immersed in sodium coolant, the reference ASTRID tool for performing repairs requiring welding operations is the use of a system capable of maintaining a leak-tight cavity, void of sodium, around the components. This system, called a sodium-tight diving bell, is currently at the early stages of development [11, 12]

2.2.3. Steam Generators

For SFRs employing a steam power conversion system (see Section 2.3), for which there is considerably more experience than the alternative gas power conversion option, there is a need to ensure reliable operation of steam generators. A number of SFRs have suffered poor operational histories directly due to inadequate steam generator or secondary circuit pipework performance (see Appendix A1.3.).

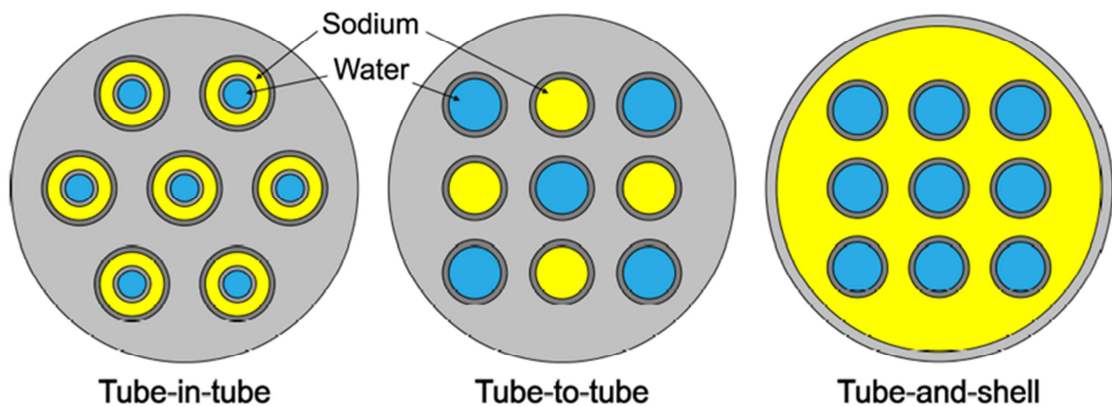


Figure 2: Cross-section for different steam generator designs employed in SFRs.

Early designs of steam generators used tube-in-tube and tube-to-tube configurations (see Figure 2). These were reliable and robust, making sodium-water interaction less likely. However, these were costly due to their low thermal efficiency, large size and complicated construction. Nearly all current designs are tube-and-shell designs employing forced vertical circulated steam generators with a straight or helical coil design [97, 98]. Nevertheless, there is scope to overcome some of the cost disadvantages associated with these robust yet costly designs using advanced manufacturing methods such as near net shape forming.

For any steam generator design it would be advantageous to use advanced manufacturing methods that avoids welds in critical areas, in addition to utilising welding methods that minimise the negative impacts welding can have on material performance (such as adverse chemical and micro-structural changes). Furthermore, many early steam generator designs suffered poor performance by choosing materials that were susceptible to certain failure mechanisms [110] (PFR, for example, utilised low alloy steels that were susceptible to stress corrosion cracking failure).

Unless the failure probability associated with selected steam generator design can be demonstrated to be very small, it would be sensible to ensure that chosen steam generator design readily permits replacement or major refurbishment during planned outages. Moreover, the steam generator design must allow for routine inspection and repair to detect the onset of failure and allow for the replacement of steam generator elements before failure results in unplanned outages.

It is also vital that reliable, fast acting leak detection systems are employed and that the detection system works in conjunction with isolation and sodium dumping systems [99]. Furthermore, there should be a means to reduce pressure and manage the hydrogen produced during sodium-water interactions. By utilising fast acting leak detection/isolation systems, this would avoid relatively small component failures cascading into major component failures resulting in extended unplanned outages.

The challenges associated with steam generators in sodium fast reactors are well documented. Additionally, solutions exist to overcome the problems that were encountered with early steam generator designs, without overly sacrificing economic performance in order to achieve a robust and reliable steam generator design. Nevertheless, it appears that significant work is still required to demonstrate these design solutions would work in a commercial SFR.

2.3. Power Conversion System

Two Power Conversion Systems (PCS) options exist for use in SFRs:

- Steam PCS (Rankine cycle)
- Gas PCS (Brayton cycle)

Most experience world-wide has been with the Rankine steam cycle but a significant disadvantage of this steam PCS is the potential for sodium-water interactions [23]. Hence, whilst the Rankine steam cycle is relatively mature there is still significant international development attempting to reduce the probability of sodium-water interactions.

He, N₂ and Supercritical-CO₂ (S-CO₂) have been considered for use in use SFR gas power conversion systems. Whilst He is inert and has a relatively high thermal conductivity its low specific heat capacity relative to N₂ necessitates high flow rates which are detrimental to plant thermal efficiency. The higher density of supercritical-CO₂ relative to the other gases has the advantage of minimising the compression work requirement but S-CO₂'s exothermic reaction with sodium is disadvantageous. For a gas PCS the French favour the N₂-based systems even though N₂ has poorer thermal properties than steam [23].

The preferred class of heat exchangers for use in N₂ based PCS are compact heat exchangers, these exhibit large heat transfer surface areas and relatively small volumes [23]. However, currently there are no nuclear qualified manufacturers of compact heat exchangers and in-service inspection appears to be an outstanding issue.

A significant drawback of implementing the gas power conversion systems in SFRs is that they are considerably less mature than the Rankine steam cycle. The use of a gas power conversion system based on helium or N₂ (which removes the risk of sodium-water reaction) does not make it possible to avoid the use of the sodium secondary system. This is because it is necessary to keep a barrier in order to stop gas entering the core and affecting reactivity, in addition to excessively raising the pressure of the primary circuit [24].

2.4. SFR Conclusion

Sodium-cooled Fast Reactors (SFRs) have benefited greatly from the research and development that has been ongoing worldwide for several decades. The performance of the fuel and cladding material for SFRs are relatively mature and as long as burnups are limited to around 100 GWd/tHM then fuel failure is highly unlikely. There are some issues surrounding the formation of eutectics and fuel constituent redistribution in metallic (U-Pu-Zr and U-Zr) fuels. These eutectics can result in localised melting within the fuel, and the relatively limited irradiation tests performed on prototypic U-Pu-Zr fuel with HT9 clad (full length fuel rods, operating under appropriate temperature and irradiation fields) are likely to prolong the licensing of an SFR that relies on U-Pu-Zr fuel relative to (U,Pu)O₂ fuel.

It should be noted that SFRs typically target burnups around 150 GWd/tHM in order to reduce fuel costs and outage times, which implies that reactor designs aiming for burnups around 100 GWd/tHM or less will likely exhibit relatively low capacity factors and high fuel costs.

The exothermic reaction of sodium with air and water is a significant drawback associated with sodium. However, experience gathered from numerous experimental SFR programmes has shown that the chemically reactive nature of sodium is not prohibitive. Moreover, the excellent thermal properties of sodium results in considerable advantages with respect to improving heat transfer within the core and aiding decay heat removal; both of which improve the safety performance of the reactor by reducing the likelihood of high temperatures leading to failure of structural materials and fuel cladding.

Whilst a substantial amount of experience has been gathered operating Sodium-cooled Fast Reactors, these programmes have generally been limited to prototype systems with relatively low powers and low capacity factors (see Appendix A1.2.). The low powers make them highly relevant to the small modular Sodium-cooled Fast Reactor designs discussed in this report. However, the low capacity factors raise concerns around the economic viability of SFRs unless strategies to improve capacity factors are successfully implemented. Besides trying to achieve high burnups to allow for greater fuel residency times and minimise fuel costs, it will likely be necessary to significantly improve in-service inspection and repair techniques before a commercial SFR can be realised. Furthermore, many SFRs have suffered extended outages due to poor steam generator performance (see Appendix A1.3.). Hence, ensuring that the steam generator design is robust, well-tested and the onset of failure can be readily detected are key requirements that will play an important role in determining whether an SFR can achieve successful commercial operation.

The significant development required with respect to in-service inspection and repair is sufficient in itself to postpone commercial deployment. In addition, whichever steam generator design is chosen for a commercial SFR, there will be a need to ensure sufficient experimental work has been performed to demonstrate that the selected design will operate reliably. Finally, the ability of suppliers to readily produce some components, such as cladding tubes based on HT9 steel, also make deployment before 2030 unlikely.

3. Lead-cooled Fast Reactors

The Lead-cooled Fast Reactor (LFR) utilises molten lead (Pb) or Lead Bismuth Eutectic (LBE) as a coolant resulting in a fast neutron spectrum. The use of Pb and LBE coolants potentially allows for considerable enhancements in the safety characteristics relative to light water and sodium-cooled reactor systems. However, no commercial LFRs have been built and, outside of Russia, experience with Pb and LBE coolants is much more limited than with Sodium-cooled Fast Reactors (SFRs). Table 4 details the key physical properties of Na, Pb and LBE liquid metal coolants.

Table 4: Comparison of physical properties of liquid metal coolants. T_m and T_b are the melting and boiling temperatures, respectively. Densities (ρ), specific heat capacities (c_p) and thermal conductivities (k) are given at 427°C [29, 45].

Property	Na	Pb	LBE
ρ (g/cm ³)	0.847	10.48	10.45
T_m (°C)	98	327	125
T_b (°C)	883	1750	1670
c_p (kJ/kg·K)	1.3	0.15	0.15
ρc_p (kJ/m ³ ·K)	1.1×10^3	1.6×10^3	1.6×10^3
k (W/m·K)	70	16	13

The utilisation of Pb and LBE (collectively referred to here as lead-based coolants) offers the following advantages:

- A very high boiling point (~1600°C), thereby allowing for much higher safety margins than associated with other coolants. However, it should be noted that failure of structural materials will occur at a significantly lower temperature than 1600°C;
- Does not exhibit a strongly exothermic chemical reaction with water or air (unlike sodium) resulting in improved safety characteristics; and
- Low neutron absorption and moderation which permits higher coolant to fuel ratios. (The large coolant to fuel ratios is necessary in LFR designs to reduce pumps speeds and limit temperature gradients.)

However, there are technical challenges associated with the use of Pb and LBE, these include:

- Their corrosiveness at temperatures above approximately 500°C; and
- Protective iron oxide films are eroded by coolant flows greater than 2 m/s.

Moreover, Pb's high melting temperature (327°C) results in the potential for freezing. The significantly lower melting temperature of LBE (125°C), was one of the primary drivers behind LBE's use in early nuclear power applications (Russian submarines) [31]. However, LBE has a number of specific drawbacks relative to Pb, these are:

- High activity associated with the production of Po-210, which necessitates robust containment to ensure isolation of the coolant from the environment; and
- The high cost and scarcity of bismuth which raises questions around the deployability of LBE-cooled reactors on a scale beyond a few demonstration systems.

Russia has built up a large amount of experience with LFR systems through their operation of LBE-cooled fast reactors in submarines, which are no longer operational. However, a large proportion of the information surrounding Russia's experience with LFRs has not been fully disclosed. In addition, nuclear submarines tend to spend a large proportion of their time operating at low power and only occasionally at nominal power levels [38]. Moreover, most submarines employ whole core refuelling

which usually takes place once decay heat has dropped to a sufficiently low level that refuelling can be performed without having to be concerned with the added complexity of removing large quantities of heat. Hence, refuelling typically occurs around 12 months after core shutdown [38]. None of these characteristics are favourable economically as they tend to increase fuel costs and significantly reduce plant capacity factors. Therefore, whilst operation of Russia's LFR submarines fleet demonstrates the principle of LFRs using LBE coolant, it does not demonstrate that a commercial system can successfully operate over many decades with a large capacity factor (>~90%). The closed nature of Russia's experience also impedes the ability to perform analyses on the effort required to progress such a system through a modern regulatory regime.

Table 5: List of Small Modular Lead-cooled Fast Reactors that are currently being progressed and have design information in the public domain.

Reactor Name	Thermal/Electrical Output	Fuel Form	Coolant	Outlet Temperature
BREST-300	700 MWth/300 MWe	UN in EP823 clad	Pb	540°C
SEALER	8-26.7 MWth/3-10 MWe	UO ₂ in Fe-10Cr-4Al-Zr clad	Pb	450°C
Hyperion	70 MWth/25 MWe	UN in HT9 clad	LBE	~500°C
ENHS	125 MWth/50 MWe	U-Pu-Zr in HT9 clad	LBE	~500°C
LSPR	150 MWth/53 MWe	UN in HT9 clad	LBE	~525°C
SVBR-100	280 MWth/100 MWe	UO ₂ in EP823 clad	LBE	490°C

Table 5 details the Small Modular Lead-cooled Fast Reactors (SMLFRs) that are currently being progressed and also have design information in the public domain. As materials can suffer excessive degradation in the presence of molten lead coolants, the types of materials that are planned to be employed and the coolant temperatures these materials are exposed to are detailed in Table 5. Fuel cladding materials are most susceptible to degradation as they are exposed to high temperatures, intense neutron and gamma radiation and high mechanical loadings as fuel swell during irradiation. Furthermore, cladding materials can suffer wear induced by flowing coolant. Therefore, given the susceptibility of cladding materials to failure and their importance as a barrier to fission product release, a particular focus is placed on their behaviour under prototypic LFR conditions.

International R&D on LFRs is focused on utilising Pb as a coolant due to the issues surrounding LBE coolant. Interestingly, there is interest in Russia for moving away from LBE coolants, due to the issues discussed above, even though Russia has significant experience operating LBE cooled reactors from their naval programme [31]. However, many of the SMLFR concepts (see Table 2) intend to utilise LBE as the coolant, likely because of the greater wealth of data on material performance in molten LBE compared with molten Pb [32].

The low coolant velocities permitted in LFRs (in order to minimise erosion of material surfaces) necessitates larger flow areas than compared with SFRs in order to achieve the same heat removal rate. Hence, LFRs have larger coolant to fuel fractions than SFRs [29]. The benefit of lattice configurations with large flow areas is to reduce pumping power requirements and enhance passive safety such as the removal of decay heat via natural circulation or the total elimination of pumps during operation and shutdown. In addition, the 45% higher volumetric heat capacity (term ρc_p in Table 1) for lead coolants relative to sodium and the much higher boiling points of Pb and LBE relative to Na result in LFR cores exhibiting large degrees of thermal inertia. Large thermal inertias are favourable as in the event a transient sequence results in a considerable quantity of thermal energy generated within the core, there is a significant time period before operators or safety systems must perform countermeasures.

3.1. Materials and Chemistry

3.1.1. Fuel and Cladding Materials

For Pb and LBE at temperatures above approximately 500°C the corrosive nature of lead-based coolants to commonly employed reactor component materials (Ferritic-Martensitic (FM) steels and austenitic steels) raises concerns over the integrity of these materials during their service lives [34]. Three life limiting phenomena that have received particular attention in the presence of lead-based coolants are:

- Erosion (i.e. the wearing away of metal surfaces due to the forces induced by the flowing coolant);
- Corrosion (e.g. the reaction of iron with oxygen dissolved in the coolant to form magnetite (Fe_3O_4)); and
- Dissolution (i.e. the preferential incorporation of one component of an alloy into the coolant medium).

Erosion can be maintained within acceptable limits if the coolant velocity is limited to 2 m/s. Hence, most LFR designs restrict coolant velocity to this limit. However, for designs that utilise pumps within the primary circuit, coolant velocities inside the pumps can be significantly above this level. Pump technologies are discussed in more detail in Section 3.2.1.

FM steels are a popular choice for use in LFRs - all of the cladding materials listed in Table 5 are FM steels - since austenitic steels typically have high nickel concentrations (8-15 wt%) and nickel exhibits a high solubility in lead-based coolants. Hence, austenitic steels are particularly susceptible to failure via dissolution.

Figure 3 details the dependence of corrosion behaviour of FM steels on coolant temperature. To summarise, relatively thin and stable oxide layers are known to form at temperatures below approximately 500°C in the presence of oxygen concentrations greater than $\sim 10^{-6}$ wt% [32]. In the temperature range 500°C to 550°C, the corrosion behaviour appears to make a transition from oxidation to dissolution and corrosion may be acceptable given adequate oxygen control [34, 35]. For temperatures above 550°C, a very thick and potentially unstable oxide layer is formed. This thick oxide is particularly susceptible to erosion. Hence, for these reasons most LFR designs aim to maintain coolant outlet temperatures below 550°C during normal operation and some limiting outlet temperatures to 500°C (see Table 5).

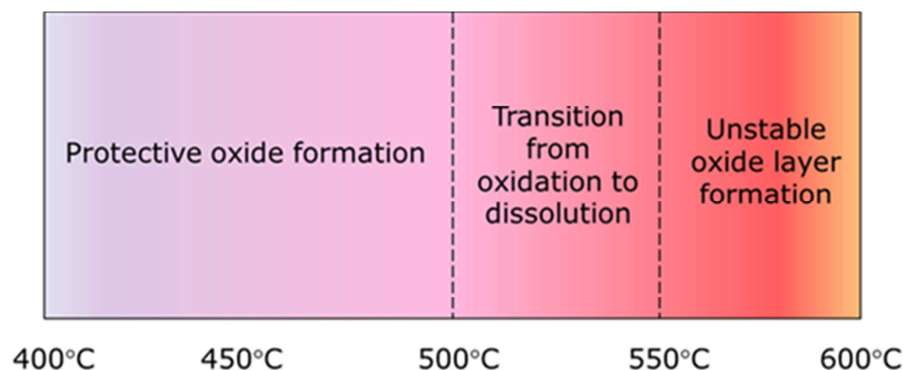


Figure 3: Sensitivity of the corrosion behaviour of FM steels to coolant temperature [34,35].

It should be noted that whilst coolant outlet temperatures may be limited to 550°C the surface of fuel rods can be significantly higher than the coolant outlet temperature during normal operation. This is because the power distribution, and therefore temperature, varies across the reactor core both axially and radially, and as a function of time. Furthermore, the temperature difference between the clad surface and bulk coolant temperature can be high in lead-based coolants ($\sim 50^\circ\text{C}$) [88]. With current

materials, clad surface temperatures above 550°C can lead to excessive material degradation. Nevertheless, coolant outlet temperatures are indicative of maximum clad surface temperatures during normal operating conditions and in the absence of data detailing maximum clad surface temperature the coolant outlet temperature is used as a guide for the maximum clad surface temperatures.

Besides the lower limit on oxygen concentrations of $\sim 10^{-6}$ wt%, there is also an upper limit on oxygen concentration ($\sim 10^{-4}$ wt%) to prevent excessive oxidation taking place and/or the formation of undesirable insoluble oxides (principally lead oxide (PbO)) [32]. PbO formation is problematic as it forms a solid which can block narrow channels and deposit on heat surfaces, reducing the efficiency of heat transfer [36].

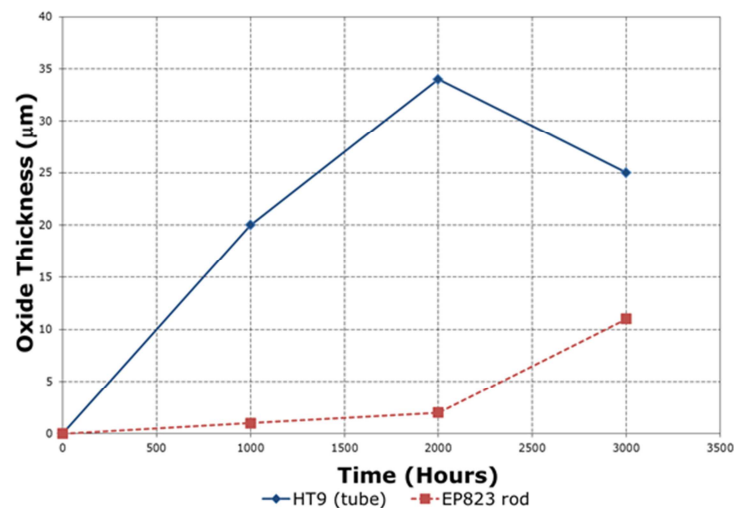


Figure 4: Oxide thickness for candidate Lead-cooled Fast Reactor cladding materials exposed to flowing LBE coolant at temperatures of 550°C with a flow rate of 1.9 m/s and oxygen concentrations of $3 - 5 \times 10^{-6}$ wt% [37].

Many of the SMLFRs shown in Table 5 intend to use EP823 and HT9 as fuel cladding. Figure 4 shows the results from a series of corrosion performance tests performed under representative LFR conditions (appropriate flow rates, temperatures and oxygen concentrations) but not in an irradiation field and for a relatively short time period (125 days). The results show that under these test conditions the Russian alloy EP823 exhibits very good corrosion resistance; however, HT9 suffers relatively high levels of oxidation. Furthermore, HT9 experiences high levels of oxide layer removal after approximately 2000 hours. The reduction in oxide layer raises concerns around the ability to maintain an oxide layer of sufficient thickness that can inhibit further oxidation and/or dissolution of clad materials.

Irradiation tests have shown that EP823 is susceptible to irradiation induced embrittlement [89, 90]. The degraded material performance of EP823 in the presence of an irradiation field highlights the importance of carrying out material tests under prototypic LFR conditions.

Much of the corrosion testing in the open literature has been performed for time periods less than 5000 hours (~ 200 days), implying that unless proprietary data is being employed by SMR vendors then a significant amount of extrapolation is being performed when designs are stating fuel residence times greater than or equal to 1 year. Furthermore, whilst there has been a relatively large number of corrosion experiments performed in LBE coolants, the number of Pb coolant experiments is very limited [32]. The range of experiments for LBE coolants also makes it difficult to build up a database for a specific material exposed to particular set of coolant conditions. Hence, for any individual material there does not appear to be the performance data necessary to determine the likely condition of the clad for a given flow rate, set of chemical conditions and range of temperatures over time periods stretching several years.

The cladding constitutes an important barrier to fission product confinement (with the reactor vessel and containment building constituting the remaining barriers) and hence the cladding material must

not fail during normal operation³. The overall effect of the uncertainties surrounding cladding material performance and its important safety function is that, without reducing uncertainties, it is not possible to conceive how LFR designs could be licensed in the near-term.

Besides corrosion; dissolution; and erosion, mechanical failure mechanisms are also very important. For this reason it is necessary to have in place a comprehensive database on the mechanical properties of the selected structural materials utilised in the reactor design. However, whilst a significant database of materials under SFR conditions has been built up over a number of decades, and will have some applicability to operating conditions within an LFR, the mechanical properties database for materials under LFR conditions is sparse [42]. Properties such as yield stress, ultimate tensile strength, ductility and resistance to fatigue and crack propagation in contact with Pb coolants (covering temperatures typical of normal and transient conditions) must be constructed.

Both uranium oxide (UO₂) and uranium nitride (UN) are popular fuel material choices for lead coolants [18, 34]. Both materials are reportedly compatible with molten lead coolants [39, 41]. Ref. [41] reports that UN was exposed to molten lead at temperatures of 650°C and 800°C for up to 2000 hours and at temperatures of 1200°C and 1300°C for 5 hours. No interactions between the molten lead and fuel from this series of experiments were observed. Constituents of U-Pu-Zr are highly soluble in lead coolants, implying that for these designs it is important in the event of clad failure, coolant clean-up systems are capable of removing fission products and/or rapid isolation of failed fuel channels occurs [39, 40].

The interaction of Pb and LBE with UO₂ appears less well understood than nitride fuels and work is being progressed to study the behaviour of these coolants with UO₂ fuel [25]. However, the available data on oxide fuels under fast reactor conditions is large and the overall behaviour of UO₂ fuel under irradiation is very good [18]. Whilst this data is based on SFR conditions, there are a number of similarities between fuel operating under LFR and SFR environments relating to operating temperatures, cladding materials (such as HT9) and the neutron spectrum. UO₂ also benefits from a simpler manufacturing route for the production of pellets relative to UN and there is no added complication associated with the production of the biologically hazardous nuclide C-14 from N-14 irradiation. Nevertheless, UN offers key advantages with respect to higher fissile loading capability and significantly better thermal conductivity performance; hence its popularity in LFR designs [34].

3.1.2. Coolant Properties and Chemistry Control

3.1.2.1. Oxygen Control

Oxygen concentration must be tightly controlled in LFRs to ensure core materials form protective oxide layers and the formation of solid lead oxide deposits is limited. The thickness of oxide layers and quantities of solid lead oxide deposits are both highly temperature dependent. As the temperature varies along the primary circuit, it will be necessary to accurately monitor the oxygen concentration and perhaps alter the quantity of dissolved oxygen within the primary circuit as temperature changes.

Oxygen sensors must be capable of repeatedly providing accurate measurements of very low oxygen concentration and sufficiently robust to operate in the harsh environment of a molten lead-based primary circuit. Any measurement failure that results in oxygen concentrations drifting out of the necessitated range or mechanical failure resulting in debris entering the primary circuit could have serious consequences on the safe operation of the reactor. Many sensor concepts have shown issues regarding robustness under the harsh environment of molten lead-based coolants [32]. It appears that

³ More precisely licensing regimes require that cladding materials remain intact during all normal operating conditions but allowances are made for a very small number of random statistical fuel failures such as those associated with manufacturing defects. However, in the event of such failures there must be measures in place to ensure that fission products entering the coolant are isolated on a short time scale.

no sensor, where detailed information on their performance under prototypic conditions, has shown sufficient robustness and reliability that it can be considered a mature design [32, 33]. Therefore, accurate control of oxygen within LFRs is considered to currently be a major challenge.

3.1.2.2. Polonium Production

Naturally occurring bismuth consists of a single isotope (Bi-209) which upon exposure to neutrons is converted into the alpha-emitting nuclide Po-210, which has a half-life of 138 days, and decays directly to stable Pb-206. Lead bismuth eutectic constitutes 45% Pb and 55% Bi, therefore the inventory of Po-210 in LBE can be very high [30]. Po-210 is also the principal short-term nuclide of most concern in Pb coolants due to the irradiation of bismuth impurities within the coolant and/or numerous nuclear reactions occurring with certain Pb isotopes; however, the concentrations of Po-210 with Pb coolants are much lower than in LBE coolants (polonium activity is a factor of 10^4 lower) [29, 30]. The main hazard associated with polonium is the production of compounds forming radioactive aerosols [26].

During normal operation, when the primary circuit is sealed, the production of Po-210 is of limited concern. However, in the event of coolant leakage, during maintenance or refuelling intervals, where the vessel head is removed, Po-210 constitutes a radiological hazard. Any licensing regime would necessitate robust methods for ensuring isolation of Po-210. There are effectively three ways to ensure Po-210 is isolated from the environment:

1. The reduction in the concentration of nuclides that are precursors to Po-210;
2. The reactor primary circuit is sealed until the decay of Po-210 has resulted in sufficiently low concentrations of this nuclide and can be shown to be sufficiently robust that the escape of Po-210 is deemed highly unlikely;
3. An effective Po-210 extraction and isolation process has been demonstrated so that in the event of refuelling or coolant escape the quantities of Po-210 that are not isolated are deemed of limited concern.

Item 1) is not feasible for LBE coolants as bismuth only occurs naturally as Bi-209. With Pb coolants, besides ensuring very low impurity concentrations of Bi, it does not appear to be being pursued by any reactor concept, likely due to the difficulties in enriching lead coolants. Item 2) appears to be a process that is being pursued by only the ENHS concept as the fuel rods and LBE coolant are contained within a module that is shipped to the reactor site with no need for access to the active region of the core (Table 5 provides details on the ENHS concept). However, this strategy would likely be challenging as a large database on the performance of core materials would need to be in place to ensure sufficient confidence to operate the reactor without ever needing to perform in-core maintenance procedures or refuelling. Item 3) is an active area of research and various methods are detailed in Ref. [26]; however, all of the polonium extraction methods highlighted suffer from one or more of the following difficulties:

- Poor safety characteristics due to the production of explosive mixtures of H₂ gas containing Po;
- Material performance due to the combination of high temperatures and corrosive environments; and
- Very limited to no experience.

Hence, the issues surrounding polonium production appear to be still outstanding. Moreover, the Russian developed BREST-300 concept has moved away from LBE as the coolant due to concerns surrounding Po production; even though Russia has considerable experience with regards to the difficulties surrounding activation of LBE coolant. One of the reasons cited for the move towards Pb coolants in the BREST-300 reactor is the reduction in polonium production [31].

The SVBR-100 utilises LBE with the proposed isolation being based on robustness of containment and low operating pressure [84], in addition to filtration systems [27]. Whether these basic principles would be sufficient to progress through licensing is uncertain.

3.1.2.3. Coolant Freezing

The high melting point of Pb (327°C) results in major concerns regarding the likelihood of coolant temperatures decreasing below these temperatures and the effects caused by large solid formations of coolant within the primary circuit. It is noteworthy that the Russian submarines utilising LBE as a coolant (with a melting point (125°C) significantly lower than Pb) suffered numerous problems relating to coolant freezing [100]. Of primary concern is induced stresses caused by volumetric changes associated with changes in phase [28]. In addition, if the solidification has caused a blockage in the primary circuit this could impair heat removal within the core and result in clad failure [29]. For this reason, redundant electric heating systems are required to ensure the coolant can be maintained in a liquid state, adding to the complexity of the power plant [29].

3.1.2.4. Water-lead Interactions

Whilst water does not undergo exothermic reactions with lead-based coolants there is the potential for water, from, for example, the steam generator in the event of a leak, to come in contact with hot Pb or LBE and rapidly heat up. The super-heated water vapour will result in an increase in pressure within the reactor vessel which could result in structural failure of core components. Therefore it is necessary that the reactor design precludes lead-water interaction within the reactor vessel or systems are in place that can reduce reactor internal pressure in the event super-heated water vapour is produced.

Some LFR designs intend to incorporate steam generators into the primary vessel, however, the risks associated with water-lead interactions may result in these designs not being granted licences to operate by regulators [46]. Furthermore, to entirely preclude water-lead interactions an intermediate circuit (as employed in SFRs) may be necessary.

3.2. Components and Instrumentation

3.2.1. Pump Technologies

The sensitivity of erosion to Pb coolants creates a major challenge in the design of mechanical pumps, where spinning components (impellers) within pumps that are exposed to the coolant can quickly become severely damaged. Most designs that utilise mechanical pumps state maximum impeller speeds in the range of 10-15 m/s, with maximum coolant velocity within the active region of the core limited to 2 m/s. However, no materials are known to successfully operate under such conditions for extended time periods (~1-10 years) [43, 44]. A number of promising candidates have been identified [43] but to the author's knowledge no demonstration pump systems based on these materials have been exposed to prototypic LFR conditions for extended periods of time.

There does not appear to be any interest in the use of electro-magnetic pumps in LFR systems. Ref. [42] states that this is likely because of the inefficiency of such devices to circulate heavy liquid metals.

Some designs do not use primary circuit pumps and therefore erosion of pump components is of no concern. However, the ability to model such systems where neutronics and thermal-hydraulics becomes tightly coupled and the reactor becomes susceptible to undesirable flow instability phenomena necessitates a significant experimental and theoretical undertaking to rule out such issues⁴.

⁴ Similar concerns face Small Modular-PWRs (SM-PWRs) that intend to rely on natural convection cooling (see Appendix 2 for a discussion on the benefits and challenges relating to natural vs forced convection cooling SM-PWRs).

3.2.2. Fuel Assemblies and Control Rods

The high density of lead results in complications in designing fuel assemblies and systems that can rapidly insert negative reactivity into the reactor core, as the buoyancy forces of the lead-based coolants must be overcome. In the case of fuel assemblies, it is possible to add ballast, in the form of tungsten weights, to counteract the upward lift force of the coolant.

Ballasts can also be used in control rod systems; however, the speed at which negative reactivity insertion can be achieved simply with a gravitationally driven system is relatively low in LFR coolants [46]. Therefore, pneumatically driven control rod systems are required. It would be possible to place control rods beneath the reactor core and rely on the upward lift force to drive control rod insertion, yet again this method of negative reactivity insertion is relatively slow [46].

3.2.3. In-service Inspection and Repair

As in the case with SFRs, the optical opaqueness of lead-based coolants significantly complicates In-service Inspection and Repair (ISI&R). However, the characteristics of lead-based coolants (high density and high melting point) are likely to further complicate ISI&R relative to SFRs for the following reasons [46]:

- The higher melting point of lead-based coolants, in particular molten lead, implies that ISI&R would have to be performed at high temperatures. Therefore ISI&R systems must be capable of surviving such an aggressive environment;
- The high density of lead-based coolants may make it difficult to insert inspection systems due to buoyancy forces; and
- There will be a reliance on ultrasound measurement systems to image core components due to the optical opaqueness of lead-based coolants. However, unlike liquid sodium, the limited density difference between lead-based coolants and steel implies a reduction in the sensitivity of ultrasound inspection methods.

3.3. LFR Conclusions

Lead and lead-bismuth eutectic (LBE) have two very attractive features: they do not undergo a strongly exothermic reaction in the presence of water or air; and its inherent properties confer a large amount of thermal inertia. The latter point is due to lead's high boiling point, coupled with its high volumetric heat capacity, which allows for considerable improvements in safety margins in the event of temperature deviations.

There are considerable challenges that must be overcome before the benefits associated with LFRs can be realised. One major issue is the relatively sparse database on material performance, and whilst a number of materials have been tested, the database consists of a mixture of tests performed under various conditions. The variability in the experiments performed makes it very difficult to accurately determine whether or not a certain material is capable of operating under the conditions of a particular reactor design. In addition, material performance in the presence of neutron and gamma radiation field is lacking. Nevertheless, the relatively sparsely populated material database indicates that the material challenges with lead-based coolants may be limited if: temperatures are kept lower than around 500°C; coolant velocities are low (< 2m/s); and adequate oxygen control is achieved.

A means of ensuring adequate oxygen control still requires significant further research and development. Oxygen sensors that are capable of providing reliable and accurate measurements for prolonged periods of time under the harsh environment of a lead-cooled fast reactor must be demonstrated.

The utilisation of In-service Inspection and Repair (ISI&R) is important for all reactor concepts as a means of ensuring materials are operating within their design limits and any components that are becoming degraded can be repaired or replaced with minimum downtime. As lead-based coolants are

optically opaque this significantly complicates ISI&R but the inherent characteristics of LFR coolants (high density and high corrosiveness) will further complicate ISI&R methods.

The high melting point of lead (327°C) raises concerns over freezing of the coolant during operation and for this reason LBE has historically received significant attention. However, LBE suffers from producing large quantities of polonium-210 when irradiated and the limited availability of bismuth. For these reasons international research on LFRs has become focused on lead as a coolant.

Finally, information in the open literature on the operation performance of Lead-cooled Fast Reactors (LFRs) is very limited. Russia has previously employed LFRs for use in their submarine fleet but has since moved away from LFRs and adopted Pressurised Water Reactors (PWRs) for submarine propulsion, implying that the performance of LFRs were suboptimal relative to PWRs. Russian LFR experience, therefore, is probably best viewed as proving the principle of using lead as a coolant since much work is required to improve the operation and economic performance of LFRs.

4. Fast Spectrum Molten Salt Reactors

Molten Salt Reactors (MSRs) are a class of reactor systems where the fuel is an inorganic liquid containing dissolved fissile material (U-233, U-235 or Pu-239), with the fuel also normally serving as the primary coolant⁵. The fuel is circulated at low pressure between the core (where criticality is achieved) and the heat exchanger.

MSRs are capable of operating with a range of neutron spectra which can typically be divided into fast and thermal. The characteristics of these two MSR types (fast and thermal) results in very different demands on reactivity control, structural materials and the reprocessing system. The neutron spectrum also influences the choice of coolant salt and the fissile inventory. Given the key differences in technology options and system requirements for thermal and fast spectrum MSRs, separate reviews have been performed for these two MSR types.

There are no fast spectrum MSRs in the public domain that are aiming for near-term commercial deployment and can be considered small and modular. Table 6 details two fast spectrum MSRs that are currently being proposed; however, only the French Molten Salt Fast Reactor (MSFR) has enough design information in the public domain for review. Whilst both the Stable Salt Reactor (SSR) and the MSFR are large (> 300 MWe) there is no fundamental reason as to why an SMR variant for these designs cannot exist.

Table 6: Details on fast spectrum Molten Salt Reactors [51, 60].

Reactor Name	Thermal/Electrical Power	Fuel Salt	Outlet Temperature (°C)	Primary Piping	Online reprocessing beyond noble metals and gaseous fission products
SSR	2500 MWth/ Unknown	NaCl-(U,Pu)Cl	Unknown	PE16	No
MSFR	2500 MWth/ 1000 MWe	LiF-ThF ₄	750	Ni-W-Cr alloy	Yes

The Stable Salt Reactor is unique amongst MSR concepts in that the liquid fuel is contained in cylindrical tubes but is cooled by a separate molten fluoride coolant. Therefore the SSR has many similarities to solid fuelled fast reactor systems.

MSR systems have primarily employed fluoride salts because of their chemical stability at high temperatures and the low neutron absorption characteristics associated with fluorine [47]. Chloride salts are also of interest but are significantly less mature than fluoride salts [47, 49]. Chloride salts have the advantages of lower melting points and are able to hold greater concentrations of fissile material. However, chloride salts have: lower boiling points; tend to be more corrosive than fluoride salts; produce the problematic Cl-36 nuclide under irradiation, which is highly mobile under waste repository conditions; and chlorine exhibits a relatively high neutron absorption characteristics in thermal reactor systems [49].

Fluoride salts, as indicated in Table 7, tend to exhibit high volumetric heat capacities (ρc_p) [47]. Therefore, for a given volume of coolant the thermal inertia would be expected to be higher in an MSR system relative to Lead-cooled Fast Reactors (LFRs) and Sodium-cooled Fast Reactors (SFRs). However, the volume of coolant in an MSR is much smaller than in SFR and LFR systems. In the

⁵ Sometimes the definition of a molten salt reactor encompasses all reactor systems that utilise a molten salt as the primary coolant and does not discriminate between systems that utilise solid fuel and those where the fuel is in liquid form. In this report, molten salt reactors refers only to those systems based on liquid fuel as there are a number of distinct benefits and challenges associated with such an implementation.

SVBR-100 LFR system, for example, the thermal power to coolant volume (ie the thermal output divided by the total coolant volume) is ~ 0.5 MWth/m³, for comparison in the MSFR it is 170 MWth/m³ [50, 81].

Table 7: Comparison of the physical properties of the molten salt LiF-ThF₄ with liquid metal coolants. T_m and T_b are the melting and boiling temperatures, respectively. Densities (ρ), specific heat capacities (c_p) and thermal conductivities (k) are given at 427°C for the liquid metals and a 600°C for LiF-ThF₄ [29, 45, 47].

Property	LiF-ThF ₄ (0.78-0.22)	Na	Pb
ρ (g/cm ³)	4.45	0.847	10.48
T _m (°C)	568	98	327
T _b (°C)	Unknown	883	1750
c_p (kJ/kg·K)	1.0	1.3	0.15
ρc_p (kJ/m ³ /K)	4.45×10^3	1.1×10^3	1.6×10^3
k (W/m·K)	$\sim 1.5^*$	70	16

* Value at temperature equal to 750°C

4.1. Materials and Chemistry

4.1.1. Structural Materials

Much of our knowledge on the behaviour of materials exposed to molten salts prototypic of MSR environments was gained during Oak Ridge National Laboratory's (ORNL) Molten Salt Breeder Reactor (MSBR) programme. As part of the MSBR programme, a small molten salt research reactor – the Molten Salt Reactor Experiment (MSRE) – was constructed. Whilst MSRE operated with a thermal spectrum and operated at low power densities (~ 4 MWth/m³) a large amount of material performance data was generated during its operation which will be of relevance to fast spectrum MSRs.

Most alloys that are used in high temperature environments contain large concentrations of chromium to improve corrosion resistance. However, chromium (Cr) is highly soluble in molten fluoride salts. Therefore early MSR designs employed nickel based alloys, which generally exhibit favourable high temperature and corrosion resistant properties, with no Cr addition [IGN2012]. Initial operation of MSRE utilised the Cr free nickel based alloy Hastelloy B, which had additions of molybdenum and iron at 29 wt.% and 5 wt.% respectively [60].

Whilst nickel based alloys with low Cr contents exhibit good corrosion resistance to molten fluoride salts; for components that contained molten fluoride salts but were exposed to air, their air oxidation properties were poor. Hence Cr was added to nickel based alloys at concentrations of around 7 wt.% to improve air oxidation resistance but not so high that corrosion in the molten salt environment became excessive. The resulting alloy – Hastelloy N – showed overall favourable corrosion resistance to both molten fluoride and air environments. However, after prolonged exposure to the active molten salt environment, Hastelloy N was shown to suffer irradiation induced embrittlement and develop small cracks [49]. Work was performed to overcome these limitations of Hastelloy N, resulting in modified Hastelloy-N, which contained additions of 1-2 wt.% niobium. Unfortunately, this material was never tested under prototypic MSR conditions that included exposure to an irradiation field, appropriate chemical conditions and temperature gradients [49].

A major drawback associated with nickel based alloys, especially when exposed to large fast neutron doses, is the transmutation reactions of nickel that lead to the production of helium atoms. Helium atoms can combine with one another resulting in severe intergranular embrittlement. However, some nickel based alloys, in particular PE16, have shown successful operation up to very high burnups in Sodium-cooled Fast Reactors [52].

The favourable historical experience with PE16, and the fact that nickel based alloys tend to exhibit good molten salt corrosion resistance, are likely the reasons for its planned deployment in the SSR system (see Table 6). However, the lack of current experience with PE16 and the fact that PE16 has not been qualified for use in any type of fast reactor implies that its deployment in molten salt reactors is likely to be a prolonged process.

Fast spectrum MSR do not contain graphite moderator in the most active region of the core. Therefore, fast spectrum MSR do not have to be concerned with the degradation in graphite's properties during irradiation, unlike thermal spectrum MSR (see Section 5.1.2). The lack of graphite is a major advantage fast spectrum MSR have over their thermal spectrum counterparts.

4.1.2. Coolant Chemistry

Most knowledge of coolant chemistry control in MSR is based on experience gained during the MSBR programme and in particular operation of MSRE. Whilst MSRE employed LiF-BeF₂ salt, the knowledge gained should be transferrable to other fluoride molten salts [60]. Section 5.1.3 details the coolant chemistry control regime in MSRE, to summarise; successful chemistry control relies on minimising impurities within the molten salt and tight control of the redox potential via careful control of the ratio of UF₃ to UF₄.

The ability to continuously remove chemical elements during power operation, so-called online fuel reprocessing, offers considerable benefits, especially in thermal spectrum systems. Online reprocessing enables the removal of fission products that impede nuclear chain reactions and also permits the segregation of actinide elements. Certain actinides will after the passage of time decay into material that readily undergoes fission. Therefore segregation allows the operator to reintroduce actinide material into the reactor core when it can confer maximum benefit to core reactivity. However, segregation of the actinide elements does raise significant proliferation issues if inappropriately designed.

It is important to note that online reprocessing was very limited in MSRE. MSRE only removed gaseous (Xe and Kr) and noble metal fission products, as these species naturally separate from the molten salt mixture. More elaborate fission product removal will have to test materials capable of withstanding the harsh chemical environment likely to arise in the reprocessing system and accommodate the decay heat produced via unstable fission products [49].

An advantage fast spectrum MSR have over thermal spectrum MSR is the relative insensitivity of fast spectrum reactors to the fission product inventory. This insensitivity means fast MSR require only minimal daily salt reprocessing (~10 litres per day for a 1 GWe system) compared with the much higher volumes required in thermal MSR (~1000 litres per day) [60]. Therefore, if online reprocessing is desired to maximise fuel utilisation then the reprocessing strategy will be far simpler in fast spectrum MSR. Nevertheless, the difficulties associated with online reprocessing beyond gaseous fission product removal and noble metals are likely to greatly complicate the design of the power plant and significantly delay deployment of the MSR concept.

4.2. Components and Instrumentation

4.2.1. Reactor Components

Demonstration that various components can withstand the aggressive coolant environment in MSR will ultimately depend on the performance of materials used to construct components such as pumps and heat exchangers. Candidate materials will have to demonstrate adequate strength, low-irradiation degradation, fabricability and corrosion resistance.

For the SSR concept, demonstration of reactor components will be considerably easier than with other MSR concepts as the coolant will not contain significant quantities of fission products under normal

operation. Therefore, irradiation resistance and the added complications of chemical reactions with the plethora of fission products that are produced during nuclear fission will be far more limited.

4.2.2. Instrumentation

As with other emerging technologies, further work is needed to prove that robust and reliable instrumentation and control systems can be deployed to accurately determine various parameters of the liquid fuel. Unfortunately the work that was undertaken as part of MSRE is likely to have limited applicability as there were limited in-situ instrumentation devices present in the MSRE system [56]. Furthermore, as part of MSRE, few studies were performed regarding the stability of the control system during transients. Therefore it is considered that the development of robust and reliable instrumentation and control systems are likely to provide a major challenge to the development of MSR systems.

4.3. Safety Performance

A key barrier to deployment of both fast and thermal MSR concepts is the limited experience with licensing such reactors and the need to understand the consequences and develop mitigation strategies for various failure mechanisms. For those concepts where the fuel is dissolved into the coolant, many uncertainties arise surrounding the use of a highly active coolant.

A highly active coolant will undoubtedly complicate repair and maintenance procedures. Furthermore, the multi-constituent nature of MSR concepts containing fuel dissolved into the coolant, could result in certain fission products plating onto structural materials and not being removed when the coolant is drained from the primary circuit. These fission products could create a heat source, which, in the event of inadequate cooling after the coolant has been removed from the primary circuit, would eventually degrade structural materials.

The high melting point of molten salts raises issues surrounding the risk of coolant freezing. Coolant freezing could result in blockages causing impaired heat removal in one section of the reactor and lead to structural materials overheating and failing.

A significant advantage of fast spectrum MSRs relative to other fast reactor concepts currently under development is the strong negative thermal feedback inherent in fast spectrum liquid fuel concepts. However, it must be noted that under certain scenarios where the reactor core rapidly cools down, a highly negative thermal feedback can be disadvantageous. This is because the rapid cooldown of the primary circuit will result in core reactivity increasing which must be compensated for via a reactivity control mechanism, such as the addition of control rods.

The safety performance of MSRs is discussed further in Section 5.3 in the context of issues that were identified during the MSBR programme.

4.4. Fast Spectrum MSR Conclusions

Fast spectrum MSRs have advantages over thermal spectrum MSRs with respect to:

- Not suffering from the issues surrounding graphite damage;
- Far less demanding reprocessing requirements when extensive online reprocessing is employed; and
- The ability to achieve a higher conversion ratio with either Th/U or U/Pu fuel cycles.

However, the lack of experience with operational fast spectrum MSRs and the greater demands on metallic structural materials imply that realising these benefits are likely to be more prolonged than with thermal spectrum MSRs. All fast spectrum MSRs, including the Molten Salt Fast Reactor (MSFR) and the Stable Salt Reactor (SSR), suffer from relatively immature structural material concepts. This is because the structural materials that are envisaged for deployment, Ni-W-Cr and PE16 in the case of the MSFR and SSR respectively, have not been tested under prototypic conditions of fast spectrum MSRs.

Two considerable advantages fast spectrum MSR have over Sodium-cooled Fast Reactors and Lead-cooled Fast Reactors are their ability to achieve strong negative thermal feedback characteristics and their use of an optically transparent coolant medium. The former is a favourable characteristic since in the event temperatures increase there will be a corresponding decrease in reactivity and such feedback can be difficult to achieve in other fast reactor concepts. However, a strongly negative thermal feedback characteristic can be unfavourable in the event rapid cooldown occurs within the primary circuit and hence the reactivity control system must be capable of compensating for reactivity increases in the event primary circuit temperatures rapidly decrease.

An optically transparent coolant is very helpful in allowing for visual inspection of the reactor core. However, in a similar manner to lead-based coolants, the chemical aggressiveness of the primary coolant will still make in-situ inspection and repair difficult.

The Stable Salt Reactor (SSR) which employs a molten chloride fuel with a separate molten fluoride salt coolant is able to benefit from many of the advantages associated with molten fluoride coolants but sacrifices the ability to maximise fuel utilisation through online reprocessing. The lack of online reprocessing and maintaining a low activity coolant are very favourable with respect to minimising deployment barriers. However, the use of molten chloride as a fuel salt is a very immature concept. There is considerably more experience with molten fluoride salts for deployment in nuclear reactors than with molten chloride salts. Furthermore, by utilising a liquid fuel concept there is the risk associated with rapid contamination of the primary circuit if the tubes containing the liquid fuel fail.

Developing a satisfactory safety case in addition to the issues surrounding development of appropriate materials are likely to prove considerable barriers to commercial deployment. For these reasons, it is expected that deployment will take longer than both Sodium-cooled Fast Reactors and High Temperature Gas Reactors.

5. Thermal Spectrum Molten Salt Reactors

Molten Salt Reactors (MSRs) are a class of reactor systems where the fuel is an inorganic liquid containing dissolved fissile material (U-233, U-235 or Pu-239), with the fuel also serving as the primary coolant⁶. The fuel is circulated at low pressure between the core (where criticality is achieved) and the heat exchanger.

Much of our knowledge on MSR systems results from the work undertaken at Oak Ridge National Laboratory (ORNL) as part of the thermal spectrum Molten Salt Breeder Reactor (MSBR) programme. The MSBR programme started off with the construction of Molten Salt Reactor Experiment (MSRE). MSRE was an 8 MWth thermal spectrum system that reached criticality in 1965 and was shutdown in 1969, achieving 1.5 effective fuel power years due to its low capacity factor as a research reactor. A great deal of information was gathered during the operation of this low-power prototype MSR and whilst results were encouraging there were a number of outstanding issues which are detailed in the following sections. Many Small Modular MSR system designs are based on the MSRE system. The MSBR programme culminated in the design of a 2250 MWth reactor called MSBR.

Table 8: Details on thermal spectrum Molten Salt Reactors currently under development [51, 53, 54, 101]. *A number of fuel salts are under consideration with NaF-RbF-UF₄ the current reference choice [51].

Reactor Name	Thermal/Electrical Power	Fuel Salt	Outlet Temperature (°C)	Primary Piping	Online reprocessing beyond Noble metals and gaseous fission products
LFTR	2225 MWth/ Unknown	LiF-BeF ₂ -(Th,U)F ₄	Unknown	Hastelloy-N	Yes
ThorCon	8 MWth/3.6 MWe – 557 MWth/250 MWe	NaF-BeF ₂ -(Th,U)F ₄	704	SS316Ti	No
IMSR	80 MWth/32.5 MWe – 600 MWth/291 MWe	NaF-RbF-UF ₄ *	700	Unknown	No

Table 8 gives details on MSRs that are currently under development. Many of these MSRs have stated powers > 300 MWe and therefore are not technically Small Modular Reactors. However, the systems highlighted have at least considered SMR variants of their current open-literature designs.

5.1. Materials and Chemistry

5.1.1. Metallic Structural Materials

The chemical environment within an MSR is very aggressive and therefore during the MSBR programme significant effort was put into developing structural material capable of withstanding the high temperature LiF-BeF₂ salt. This resulted in the development of the nickel-based alloy named Hastelloy-N, which was employed in MSRE.

The general corrosion resistance of Hastelloy-N was very good. However, post-irradiation examination identified poor resistance to radiation induced embrittlement and the development of small cracks on the inside surface of the Hastelloy-N piping employed in the MSRE system [49]. Significant R&D was

⁶ Sometimes the definition of a molten salt reactor encompasses all reactor systems that utilise a molten salt as the primary coolant and does not discriminate between systems that utilise solid fuel and those where the fuel is in liquid form. In this report, molten salt reactors refers only to those systems based on liquid fuel as there are a number of distinct benefits and challenges associated with such an implementation.

undertaken after the shutdown of the MSRE which indicated that modified Hastelloy-N, containing 1-2% niobium, operating under molten salt temperatures <math><650^{\circ}\text{C}</math> is a strong candidate for resistance to crack propagation and radiation induced embrittlement [49]. Unfortunately modified Hastelloy-N has not been tested for prolonged periods of time under prototypic MSR environments and therefore these tests would need to be performed before sufficient confidence can be gained in its performance characteristics.

Data on the material performance of the 300 series of stainless steels appears limited above

5.1.2. Graphite

Graphite is known to be chemically compatible with molten salt and the addition of graphite into the core limits damage to metallic structural materials by lowering the fast neutron dose to structural materials. A key issue facing thermal MSRs that contain large quantities of graphite within the core is the effect of neutron irradiation on graphite properties. Graphite will initially exhibit a volume reduction at the beginning of irradiation but at higher neutron doses graphite begins to swell. Shrinkage of graphite is less of a concern if the core design has accommodated the reduction in volume. However, swelling is more difficult to accommodate as eventually swelling will result in cracks and the development of pores within the graphite medium until total disintegration of the graphite takes place [58].

As part of the MSBR programme it was estimated that at power densities of around

Given the sensitivity on neutron dose and that modern reactor systems aim to achieve core lifetimes of around 60 years it seems unlikely that a thermal MSR could be realised that removes the need to replace graphite within the core. Although, it may be possible to allow portions of the graphite that experience the highest neutron fluxes throughout core life to be more readily replaceable, and therefore greatly reduce the complexity of routinely replacing large portions of the core internals. The difficulties associated with graphite are one of the primary motivations behind developing fast spectrum MSRs.

5.1.3. Coolant Chemistry

The MSRE successfully demonstrated ways to adequately control the chemistry within the primary loop, with a large focus on the careful control of the redox potential within the primary circuit. Furthermore, MSRE undertook relatively limited online reprocessing, which benefited from natural separation mechanisms that inherently occur within the liquid salt such as the insolubility of Xe and Kr fission gases. Some MSR concepts aim to perform more extensive fission product removal in order to remove parasitically absorbing elements from the liquid salt mixture and considerably improve fuel utilisation.

MSRE operated at low power densities which not only limited material damage but limited the effect of neutron irradiation driving chemical changes through transmutation processes within the fuel salt. At high neutron fluxes it is expected that transmutation mechanisms will result in redox changes and the formation of oxide impurities [62].

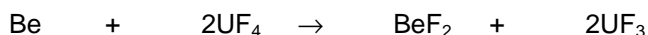
Given that most of our experience with chemistry control within MSR systems resides with the MSRE programme, it is prudent to review the strategies employed in MSRE and their effectiveness. It is worth noting that whilst MSRE focused on the LiF-BeF₂ salt, the basic principles for chemistry control employed in MSRE are applicable to MSR systems relying on other molten fluoride salts compositions [60].

The MSRE programme demonstrated that given relatively simple fission product removal (focusing on the natural separation that takes place with gaseous fission products and noble metals) successful chemistry control hinges on [62, 63]:

1. Maintaining a highly pure salt mixture, in particular ensuring that the presence of reactive oxides, such as H₂O, are kept sufficiently low; and
2. Tight control of the redox potential within the MSR system helps limit the extent of corrosion of structural materials.

It is necessary to limit the presence of reactive oxides; otherwise there is the risk that actinide dioxides (such as UO₂) form and precipitate out of the molten salt solution into solid deposits resulting in blockages within the primary circuit [47]. Any blockages could impair heat removal and therefore cause damage to the structural materials (potentially leading to a loss of highly activate coolant) if countermeasures do not begin quickly enough. The MSRE project showed that a sufficiently pure salt can be maintained to limit the formation of actinide dioxides and that also the addition of ZrF₄ can be added to the salt solution to further limit the formation of actinide dioxides [62]. However, ideally the addition of ZrF₄ is best avoided as it has a slightly parasitic effect on neutrons and also complicates online reprocessing [56].

In MSRE it was found that making the coolant conditions more reducing, via the careful control of the ratio of UF₃ to UF₄, further inhibited corrosion of the Hastelloy-N structural material [47]. The concentration of UF₃ was increased through the addition of beryllium (Be) metal which took advantage of the following chemical reaction [62]:



However, the UF₃ concentration cannot become too high otherwise graphite starts to chemically react with the liquid salt and there is an increasing risk that other materials within the primary circuit may become degraded [47].

The complexity of any MSR system is likely to be proportional to the extent of online reprocessing being performed. There is considerable uncertainty with respect to the performance of more extensive online reprocessing methods as these methods are unable to benefit from historical experience. Moreover, systems containing unstable fission products will need to accommodate the heat produced within these systems once shutdown has occurred. This highlights a significant disadvantage with separating out fission products from the primary circuit as each system will then have to demonstrate it can achieve adequate decay heat removal, thereby increasing the overall complexity of the plant. There are also highly challenging material performance issues to overcome as the fission product removal process becomes more extensive since chemical conditions within the individual reprocessing systems are very aggressive [49, 56].

5.2. Components and Instrumentation

5.2.1. Reactor Components

Demonstration that various components can withstand the aggressive coolant environment in MSR systems will ultimately depend on the performance of materials used to construct components such as pumps and heat exchangers. Candidate materials will have to demonstrate adequate strength, low-irradiation degradation, fabricability and corrosion resistance.

5.2.2. Instrumentation and Control

As with other emerging technologies, further work is needed to prove that robust and reliable instrumentation and control systems can be deployed to accurately determine various parameters of the liquid fuel. Unfortunately the work that was undertaken as part of MSRE is likely to have limited applicability as there were limited in-situ instrumentation devices present in the MSRE system [56]. Furthermore, as part of MSRE, few studies were performed regarding the stability of the control system during transients. Therefore it is considered that the development of robust and reliable instrumentation and control systems are likely to provide a major challenge to the development of MSR systems.

5.3. Safety Performance

Whilst the liquid fuels exhibit a number of very favourable characteristics (high boiling point, optical transparency, low pressure operation and no exothermic reaction with air or water), MSRs are unable to benefit directly from the several thousand reactor years of operation solid fuel systems have achieved. Having the fuel dissolved in the coolant during normal operation results in significant complications in the event overcooling or leakages occur.

Most of our experience with MSR systems is from a single research reactor: MSRE. It is important to remember that MSRE was a prototype and therefore its aim was to prove the fundamental technology principles of MSR systems, which in many cases it did so successfully. However, as a low power experimental system it did not provide sufficient information to thoroughly test the safety performance of systems at prototypic powers, which will be necessary for developing a safety case capable of meeting modern regulatory requirements.

LiF-BeF₂ has a melting point of 455°C, which is relatively low for a molten fluoride salt, but is significantly higher than other coolants for reactor systems (c.f. melting point of sodium is 98°C and even pure lead is significantly lower at around 327°C) [47]. Therefore, the risks attributed to salt solidification cannot be ruled out, especially when transient MSR calculations indicate the formation of cold spots within the reactor system [46]. Cold spot formation followed by salt recrystallisation could impair coolant flow and lead to localised failure of structural materials.

It is sometimes stated [59] that even in the event of piping failure, the risk of radioactive release and/or an uncontrolled criticality event is minimal if an appropriately designed containment tank surrounds the MSR system. The containment tank would contain leaks and allow for drainage into separate subcriticality configured vessels. The implementation of a containment tank is a sensible mitigation approach to pipe failures but overlooks a considerable drawback with MSR systems: the highly active, multi-constituent, high melting temperature, nature of the liquid fuel. In the event liquid fuel is released from MSR piping the containment tank could become highly activated, especially if components of the liquid fuel do not successfully enter subcritical tanks; for instance due to rapid recrystallisation. A highly active containment facility would make repair and maintenance more difficult, which would undoubtedly be necessary in the event of pipe failure, and at the very least would be expected to degrade the capacity factor of the plant if significant time must pass before the dose inside the containment tank is low enough to permit workers to enter the containment structure. It may be necessary to also adopt multi-barrier piping (as is proposed in some Sodium-cooled Fast Reactor (SFR) concepts) and numerous valves to allow for isolation of failed pipes without contaminating the containment structure. However, the highly active nature of the coolant/liquid fuel would still make repairs more difficult than in other reactor systems.

In a similar manner to other systems that operate at high temperatures (such as Lead-cooled Fast Reactors) there is the risk of steam explosions occurring in the event water comes into contact with the hot liquid salt. Any pressure increase must not result in structural damage to the containment structure. There appears limited publicly available research into water-liquid fuel salt interactions [46].

The safety performance of an MSR system relies heavily on the effectiveness of the drainage system. Many MSR concepts rely on so-called 'freeze-plugs' which consist of a block of material that will melt in the event of off-normal events that result in deviations in liquid fuel temperatures. Once the freeze-plug has melted, the inventory of liquid fuel will then drain to tanks that are designed to keep liquid fuel

in a subcritical configuration (so-called 'dump tanks') and allow for decay heat removal, usually via natural convection; thereby removing the need for electrically powered pumps.

A number of issues arise with this safety procedure regarding:

1. The speed and reliability of the freeze-plug melting process;
2. The precise details regarding indefinite decay heat removal; and
3. The robustness of the dump tank design.

Freeze-plugs are likely to occupy a volume of $\sim 100 \text{ cm}^3$ and if the plug consists of a molten salt, which exhibit relatively high specific heat capacities and low thermal conductivities, plug melting will probably take several minutes unless an external heating source is employed [61]. Furthermore, in order to stop the freeze-plug from melting during normal operation, the plug must be actively cooled, for instance by blowing air around the freeze-plug. Actively cooling the freeze-plug is beneficial under some circumstances, as in the event of a loss of power incident the freeze-plug will begin to melt and the liquid salt will eventually drain into the dump tanks without the need for operator intervention. However, by necessitating active cooling, there is the risk that too much cooling could be provided and as Ref. [61] shows, the speed at which freeze-plug melting occurs is highly sensitive to its initial temperature. Furthermore, if drainage of the liquid salt takes several minutes it is necessary to determine the survivability of core components at elevated liquid salt temperatures whilst the drainage process is taking place. In addition, there is always the risk that blockage could have occurred within the MSR system away from the freeze-plug which inhibits the ability to drain salt within the proposed time period [61].

Decay heat produced from the liquid salt must be adequately removed. As the power density of the MSR system rises the difficulty in removing decay heat also becomes larger. MSRE had a very low power density for an MSR and therefore decay heat removal was of limited concern for this research reactor. In the French MSFR, decay heat is removed by a pool containing several thousand cubic metres of water at around 20°C , in which the dump tanks are immersed. The volume of water is sufficient to allow indefinite heat removal [46]. However, there will always be the risk that coolant medium surrounding the subcritical tanks comes into contact with the liquid salt, in the same way it is difficult to eliminate the risk of sodium-water or sodium-air interactions in SFR systems.

There is little publicly available information detailing the interaction between liquid fuel salts and water but it is possible that hydrogen fluoride (HF) could be produced. HF is a highly corrosive and toxic gas and therefore the interaction of liquid salts with water cannot be considered benign.

Given the important safety function of dump tanks regarding fission product confinement, reactivity control and decay heat removal, they will need to demonstrate exceptionally high levels of robustness. Dump tanks will need to ensure that they are capable of containing liquid salts at elevated temperatures in the event that, in the process of draining, the salt temperature has significantly deviated from its nominal operating temperature.

Tritium production is a significant problem for MSR systems (especially those reliant on salts containing lithium and/or beryllium), as tritium is expected to be the only radionuclide that, under normal operating conditions, has the potential for significant release [50]. MSRs will produce significantly more tritium than light water reactors and, at the high temperatures MSRs operate at, tritium will permeate relatively easily through available structural alloys [50]. Heat exchangers are particularly susceptible to tritium release due to their large surface area and thin walls. As part of the MSBR programme, a small test loop was constructed to test the suitability of sodium fluoroborate (NaF-NaBF_4) as a secondary coolant for trapping tritium released through heat exchangers [65]. Sodium fluoroborate was shown to successfully trap around 90% of the tritium inventory; however this still leaves a substantial amount of tritium remaining. The efficiency can be improved further when factoring in that helium bubbling will also remove a fraction of the tritium inventory [60]. Ref. [56] highlights considerable uncertainty regarding the behaviour of sodium fluoroborate with water and LiF-BeF_2 . Concerns relating to enhanced corrosion in the presence of water, contamination of the primary loop and the production of toxic BF_3 if the liquid fuel were to come into contact with sodium fluoroborate have been highlighted [56]. If water is not used as the working fluid then issues regarding enhanced corrosion can be precluded.

Concerns regarding the thermal feedback behaviour of graphite moderated MSR systems have been raised [60]. However, Ref. [64] indicates that positive thermal feedback is avoidable in graphite moderated MSR systems with appropriate core layout.

The precise behaviour of fission products, in particular the noble metals, within the primary circuit is relatively poorly known [47, 56]. It is known that a fraction of the noble metals can be removed via the helium bubbling process as they are insoluble and this process could potentially be enhanced [60]. However, a significant fraction of noble metals have been found to plate out onto structural materials and will not enter the dump tanks in the event the liquid fuel needs to be removed from the primary circuit [47]. The decay heat from these unstable noble fission products may result in failure of structural materials.

5.4. Thermal Spectrum MSR Conclusions

The Molten Salt Reactor Experiment (MSRE) demonstrated that a low power density liquid fuel system is feasible as long as:

1. There is a high confidence in the reliability of structural materials over core life;
2. Online fission product removal is limited to gaseous fission products and noble metals; and
3. The operator is willing to accept downtime associated with large quantities of core graphite removal every ~4 years.

Unfortunately MSRE was unable to demonstrate the reliability of the chosen structural material (Hastelloy-N) over long periods of time and at the end of the programme Hastelloy-N was shown to exhibit embrittlement and cracking. A programme was undertaken to try to overcome these issues, resulting in a promising candidate called modified Hastelloy-N; however, this material was not tested under prototypic conditions (appropriate irradiation fields, coolant chemistry conditions and temperature gradients). Any uncertainty in the reliability of the structural material could result in large quantities of primary piping needing replacing several times during core life, which will likely be difficult due to the highly active nature of the liquid fuel.

For concepts that rely heavily on the work undertaken as part of the MSBR programme, there is considerable uncertainty in the ability to successfully license a reactor design that relies heavily on results produced during the 1960s. It is likely that a low-power demonstration system (~10 MWth) would be required for any MSR system, before being able to construct a system able to successfully meet the demands associated with commercial operation (capacity factors ~90%) and deliver cost competitiveness with other reactor systems. Such a low-power demonstration system would allow for: testing of promising structural materials and state-of-the-art reactor grade graphite; assessing the safety performance of key systems to support modern licensing requirements; and provide input to validate computer models. This approach is supported by IRSN (the French nuclear technical authority) and is also the approach being undertaken in China with the planned construction of a prototype 2 MWth MSR by around 2020 and a 100 MWth MSR demonstration system in 2035 [46, 57]. Any concepts that differ from the MSBR design will very likely suffer a more prolonged development programme, as these designs will not be able to benefit from the particular solutions developed as part of the MSBR programme.

There are however significant advantages associated with developing concepts that depart from the MSBR design, such as avoiding issues related to graphite replacement and the higher conversion ratios fast MSR designs are able to achieve. Table 3 summarises the advantages (+) and disadvantages (-) associated with thermal (graphite moderated) and fast MSR designs.

Table 9: Advantages (+) and disadvantages (-) for thermal (graphite moderated) and fast Molten Salt Reactors.

Thermal (graphite moderated) MSR	Fast MSR
+ Able to take advantage of the significant experience built up as part of the MSBR programme	+ Does not suffer from the issues surrounding high graphite throughput
+ The addition of a moderator limits damage to structural materials	+ Less demanding reprocessing requirements than a thermal MSR if online reprocessing is employed
+ Lower fissile inventories which allows for a fleet of MSRs to be started up with less Pu and/or enriched uranium	+ Ability to achieve high conversion ratio and fast MSRs are less sensitive to FP build-up
- Online reprocessing much more difficult than in fast MSR	+ Strong negative thermal feedback is unique for fast reactor systems currently under development
- Lower conversion ratio in thermal MSRs than fast MSRs and reactor is more sensitive to the build-up of fission products	- High fast neutron dose to structural materials
- High throughput of graphite will likely degrade capacity factor and result in large inventory of radioactive waste	- Very limited worldwide experience with fast MSRs

6. High Temperature Gas Reactors

High Temperature Gas Reactors (HTGRs) make extensive use of ceramic materials by utilising a robust, all-ceramic fuel and graphite as a structural material. Helium is used as the primary coolant due to helium's excellent compatibility with the ceramic structural materials. The high operating temperatures of the ceramic materials and helium's chemical compatibility allow HTGRs to achieve outlet temperatures around 800°C.

HTGRs are unique in their proven ability to achieve outlet temperatures of around 800°C (see Table 10). Molten Salt Reactors and Lead-cooled Fast Reactors aim to achieve comparable outlet temperatures but material performance issues will likely limit their outlet temperatures to approximately 700°C and 500°C, respectively, until advanced materials can be demonstrated for these coolants. Higher temperatures allow reactor systems to achieve improved thermal efficiencies and therefore increase the economic competitiveness of the reactor system. Furthermore, higher temperature can also be employed for driving a variety of industrial processes.

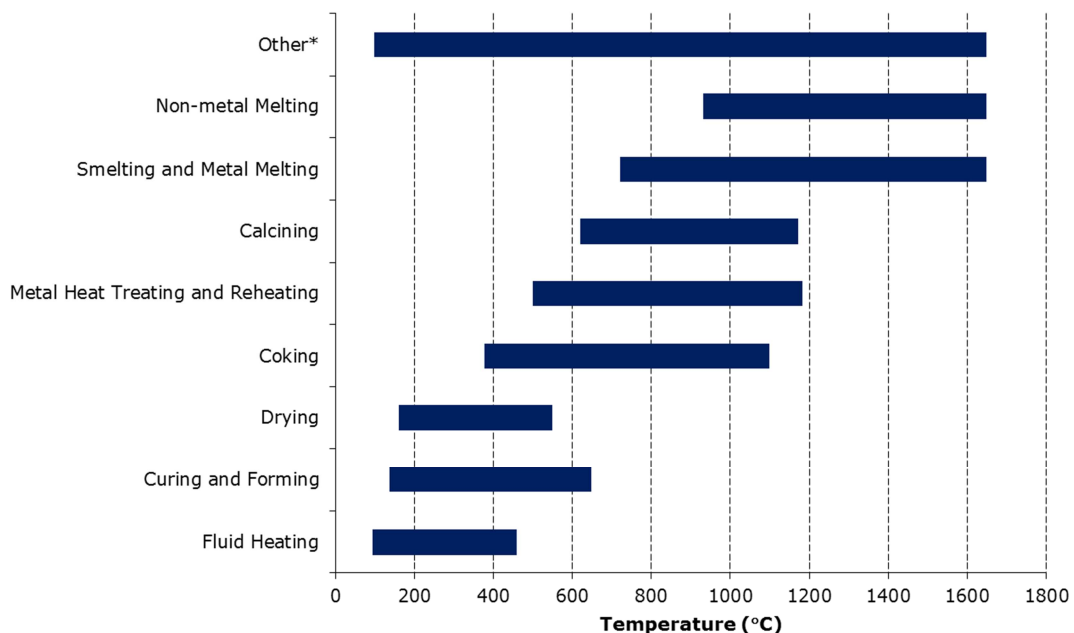


Figure 5: Typical temperature ranges for common industrial processes [66]. *Includes preheating, catalysis, thermal oxidation and incarnation.

Figure 5 shows the typical temperature ranges required for common industrial processes and that there are a number of industrial processes where a reactor capable of achieving outlet temperatures of 800°C could be utilised. However, it should be noted that industrial heating is dominated by processes operating at relatively low temperatures (~500°C or less). Fluid Heating, for instance, makes up 29% of US industrial thermal energy consumption and Drying consumes 22% [67], which can, to a large extent, be provided by SM-PWRs. Hence, any reactor system aiming to provide industrial heat does not necessarily have to achieve temperatures in excess of 800°C.

Figure 5 shows industrial processes that are currently well established but not those that are under development, such as novel hydrogen production methods. There has been interest in utilising HTGRs to provide process heat for hydrogen production via the sulphur-iodine cycle that has been behind the recent efforts in HTGRs. However, to maximise the benefits from the sulphur-iodine process requires temperatures around 1000°C. At such high temperatures the performance of materials in HTGRs

rapidly diminishes and hence why most HTGRs have only been able to achieve sustained operation at 800°C.

It is possible to increase the outlet temperature from a reactor system utilising heat pumps [103]. However, heat pumps for boosting outlet temperatures are yet to be demonstrated with HTGRs. Furthermore, there are a number of challenges with respect to commercialising the sulphur-iodine process due to the combination of high temperatures in a chemically aggressive environment. Nevertheless, there is likely to be a need for low-carbon hydrogen production in order to reduce greenhouse gas emissions in certain industries; as a method to support decarbonising transport systems; and as a means for energy storage [104, 105]. However, in the near-term deployment of HTGRs for industrial process heat is likely to be limited to the processes shown in Figure 5.

A defining feature of HTGRs are their low power densities, which are typically around 5 MW/m³, for comparison Light Water Reactors are usually around 100 MW/m³ and Sodium-cooled Fast Reactors around 400 MW/m³. HTGRs have low power densities due to the inherent characteristics of the materials and coolant employed; namely:

- The poor heat transfer properties of helium, which limits the ability of the coolant to remove thermal energy from the core; and
- Graphite does not effectively slow neutrons in small concentrations and therefore large volumes of graphite are required within the core in order to achieve the desired thermal spectrum.

A drawback of the relatively large size of HTGRs for a given power output and the cost of a relatively complex fuel form does degrade the economic competitiveness of HTGR systems. However, the low power densities and the ability of the core materials to withstand temperatures in excess of 1600°C allow HTGRs to achieve very high safety margins (namely, indefinite decay heat removal even under quite severe accident scenarios such as depressurisation accidents [46]), which may prove useful when aiming to co-locate HTGRs with hazardous industrial facilities in which HTGRs provide process heat.

The fuel elements in HTGRs can either be cylindrical, as used in prismatic cores, or spherical, used in pebble-bed cores. The prismatic cores contain stationary fuel elements during operation which are then removed during refuelling outages, whereas the spherical fuel elements are capable of being continuously removed and added to the core during operation. Section 6.1.1 gives more details on HTGR fuel.

Table 10 details experience with HTGR operation. Table 10 shows that most HTGR operational experience has been gathered from low power (<500 MWth) reactors and HTGRs have typically operated around 800°C. The operational performance from high power HTGRs has been relatively poor (see Appendix 3).

Table 10: High Temperature Gas Reactors that have been built and operated along with key parameters [73, 83].

Reactor Name	Operation Dates	Power	Fuel Form	Outlet Temperature
Dragon	1964-1975	20 MWth	(Th,U)O ₂ and UO ₂ fuel in prismatic form	750°C
Peach Bottom 1	1966-1974	115 MWth/40 MWe	(Th,U)C ₂ fuel in prismatic form	~700°C
AVR	1967-1988	46 MWth/ 15 MWe	(Th,U)O ₂ and UO ₂ fuel in pebble-bed form	750-950°C
Fort St. Vrain	1976-1989	842 MWth/ 330 MWe	(Th,U)C ₂ fuel in prismatic form	~800°C
THTR	1985-1991	750 MWth/ 300 MWe	(Th,U)O ₂ fuel in pebble-bed form	750°C
HTRR	1998-present	30 MWth	UO ₂ fuel in prismatic form	850-950°C
HTR-10	2000-present	10 MWth	UO ₂ fuel in pebble-bed form	700-950°C

Small modular HTGRs that are currently being progressed are shown in Table 2. Table 2 indicates that the operational parameters for proposed small modular HTGRs have been limited to those of HTGRs that have prior pedigree of operating successfully.

Table 11: List of Small Modular High Gas Temperature Reactors that are currently being progressed and have design information in the public domain [78, 79, 80, 102].

Reactor Name	Thermal/Electrical Output	Fuel Form	Power Conversion System	Core Outlet Temperature
HTMR100	100 MWth/35 MWe	(Th,U)O ₂ fuel in pebble-bed form	Steam Cycle	750°C
HTR-PM	2×250 MWth/211 MWe	UO ₂ fuel in pebble-bed form	Steam Cycle	750°C
U-battery	10 MWth/4 MWe	UO ₂ fuel in prismatic form	In-direct Brayton Cycle	800°C
Xe-100	125 MWth/45 MWe	UCO fuel in pebble-bed form	Steam Cycle	Unknown

6.1. Materials and Chemistry

6.1.1. Fuel

6.1.1.1. TRISO Fuel Particles

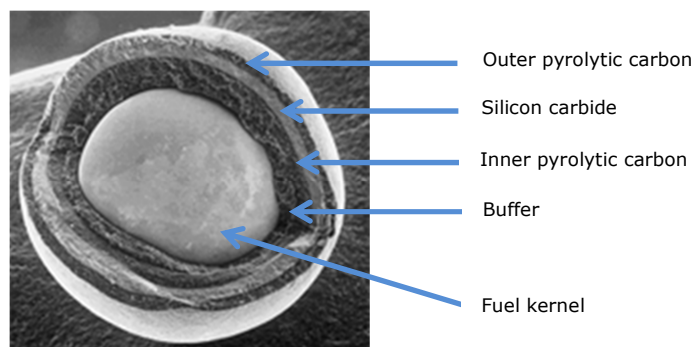


Figure 6: TRISO particle constituents.

High Temperature Gas Reactors gain many of their benefits from utilising an extremely robust all-ceramic fuel which is in particulate form. At the centre of the fuel particle is a sphere, typically around 250 μm in size, of heavy metal (U or Th) oxide or carbide - the so-called fuel kernel. There are four layers consisting of three isotropic materials that surround the fuel kernel (see Figure 6). Hence, the fuel is termed tri-structural isotropic (TRISO) fuel. The tasks of the surrounding layers are as follows [68, 69, 46]:

- The first layer surrounding the fuel kernel is a porous graphite buffer layer that is around 100 μm thick and acts a sacrificial layer to protect the surrounding layers from the energetic fission products. In addition, the porous buffer layer accommodates gaseous fission products thereby limiting pressure build up within the TRISO fuel.
- The second layer consists of dense pyrolytic carbon (approximately 40 μm thick). The purpose of this inner pyrolytic carbon (IPyC) layer is to protect the neighbouring layer of silicon carbide (SiC) and limit the permeation of noble gases and halogens produced during fission.
- The SiC layer (around 35 μm thick) confers a high degree of mechanical and thermal resistance. Moreover, the SiC layer acts as a leaktight seal around the previous layer, and is particularly good at retaining metallic fission products and tritium. The SiC layer only exhibits good mechanical properties when it is under compression.
- Finally, there is a further layer of pyrolytic carbon termed the outer pyrolytic carbon (OPyC) layer, with a thickness of approximately 40 μm . OPyC acts as further barrier to impede the release of fission products and serves as a bonding surface between the TRISO particle and the surrounding host material. Furthermore, the behaviour of the pyrolytic carbon layers during irradiation ensure that the SiC is under compression otherwise the SiC layer is prone to failure.

TRISO fuel particles are then embedded in graphite to form fuel elements. Fuel elements can either be spherical or cylindrical, which are used in pebble-bed and prismatic HTGR designs respectively.

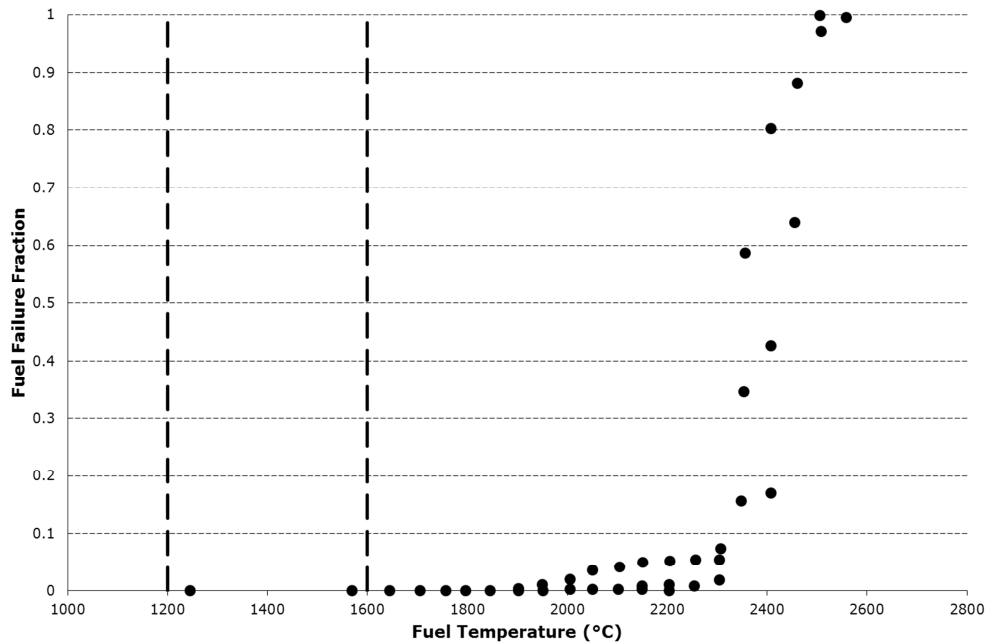


Figure 7: Fuel failure fraction for TRISO particle as a function of temperature. Typical HTGR maximum normal operating (1200°C) and transient (1600°C) temperature limitations are indicated by dashed vertical lines [70, 75].

Figure 7 shows the sensitivity of fuel failure on fuel temperature. Note that the SiC layer begins to rapidly decompose at temperatures above 1900°C, hence the swift transition in particle failure rates above 1900°C [70]. Furthermore, Figure 7 exhibits some spread in failure fractions as fuel failure depends on a number of properties, including:

- The effectiveness of the manufacturing process;
- Fuel burnup; and
- The duration particles spend at different temperatures.

The effectiveness of the manufacturing process is very important as faulty particles are the main reason for fission products entering the primary coolant. Given the large number of particles in a HTGR ($>10^8$), it is not feasible to check the integrity of each particle before loading them into the core. Therefore, it is necessary to sample a proportion of the particles produced during manufacture to ensure that the number of defective particles entering the reactor is very low. German HTGR fuel generally exhibited very low fission product release fractions and is often used for making comparisons when assessing HTGR fuel performance. In this regard, the proportion of defective particles produced towards the end of the German HTGR programme was 1.14×10^{-5} [69].

Faulty particles do not only arise due to inadequate fabrication and quality control but can also occur during reactor operation as TRISO particles become degraded as they are irradiated and/or their temperature increases. UO_2 TRISO fuel has demonstrated low failure rates when operating for prolonged periods of time at temperatures around 1200°C and burnups limited to approximately 100 GWd/tHM [70].

UO_2 TRISO particles are capable of operating for $\sim 10^2$ hours at very high temperatures (1600°C) without suffering pronounced fuel failure rates. German UO_2 TRISO fuel showed particle failure rates of 1.65×10^{-5} during heating tests at temperatures up to 1620°C [69]. Depending on the HTGR design, temperatures around 1600°C, for relatively short periods of time, are conceivable under certain hypothesised accident scenarios, for instance in the event reactor pumps fail and helium begins to escape the primary circuit. It may be possible to ensure low fuel failure rates under certain accident scenarios at temperatures between 1600°C and 1900°C, but the current database on UO_2 TRISO fuel at these temperatures is insufficient to determine with statistical confidence whether UO_2 TRISO can sustain such temperatures [70].

Fuel kernels have typically been UO_2 and have shown adequate performance characteristics under a range of tests as detailed above. However, the main disadvantage of using UO_2 fuel arises from the left over oxygen that was associated with the uranium atom prior to uranium undergoing fission. Most of the residual oxygen will readily combine with a number of fission products but there will be an excess of oxygen which can oxidise the carbon that surrounds the fuel kernel, resulting in the production of carbon monoxide (CO) [69]. The build-up of CO can result in overpressurisation of the TRISO particle, which can lead to severe degradation of the TRISO particle. Carbon monoxide formation becomes particularly important at very high burnups [69].

To overcome the issues surrounding CO formation, the use of uranium oxycarbide (UCO) as the fuel kernel has been investigated. UCO consists of a mixture of uranium oxide (UO_2) and uranium carbide (UC_2), any liberated oxygen results in the following chemical interaction:



which avoids the production of problematic carbon monoxide (CO). Whilst UO_2 as a fuel kernel material has the most experience, there has been significant irradiation of TRISO fuel containing UCO [70, 71]. These irradiation tests indicate that UCO fuel is capable of achieving low failure rates when operating for prolonged periods of time at temperatures around 1200°C with burnups greater than 100 GWd/tHM [70].

Thorium oxide fuels have historically received a lot of attention, as shown in Table 10. However, unlike UO_2 and UCO TRISO fuels there has been no recent irradiation tests performed. Therefore, designs aiming to employ $(\text{Th,U})\text{O}_2$ as a fuel kernel material may struggle through the licensing process if regulators require up-to-date fuel performance data under prototypic HTGR conditions.

An important barrier to the deployment of any commercial HTGR system may be the lack of facilities capable of producing large quantities of TRISO fuel bearing fuel in the desired form (spherical or cylindrical). This is further complicated by the fact that most uranium facilities are licensed to handle enriched uranium up to 5 wt.% and that HTGR fuel typically has enrichments between 5 and 20 wt.%. Whilst fuel enriched to between 5 and 20 wt.% is below the limit (20 wt.%) that differentiates between high and low enriched uranium, there are non-technical barriers (such as current plant safety procedures only allowing fuel enriched up to 5 wt.%) that may significantly postpone deployment time frames for HTGR systems.

6.1.1.2. *Pebble-bed vs Prismatic Cores*

In the pebble-bed reactor, the spherical fuel elements are contained in a large tank surrounded by graphite, which constitutes the reactor core, and the fuel elements at the bottom of the tank are removed during power operation. Fuel elements are checked using a system called a scrap separator for mechanical soundness and then their burnup is determined using gamma spectroscopy [72, 83]. Fuel elements that are mechanically intact and below the burnup threshold are returned to the core. After around 6-10 passes through the core the fuel elements are permanently removed and fresh spherical fuel elements are introduced [72]. As fuel elements travel along random paths as they pass through the core, there is an uncertainty associated with the position of a particular fuel, this uncertainty can result in local hot spots due to fresh fuel pebbles neighbouring one another.

The advantages of having circulating pebbles are [72]:

- Minimises enrichment requirements as the fuel within the reactor only need to generate sufficient reactivity until fresh fuel can be reintroduced;
- Fuel utilisation is improved as not only are enrichment requirements minimised but each fuel element makes multiple passes through the reactor, hence fuel elements achieve similar burnups once discharged;
- Outages are not governed via refuelling requirements but during maintenance requirements, with the turbine generator requiring maintenance every six years. Hence there is an expected increase in capacity factor; and
- By minimising the excess reactivity in the core this lessens the demands on the reactivity control systems, allowing for potentially improved shutdown margins.

However, the disadvantages of pebble-bed cores are associated with the relative complexity of continuously adding and removing pebbles; increased uncertainty in power distributions as the fuel randomly moves through the reactor; and the wear induced on fuel elements compared with stationary fuel elements [73, 83].

Both pebble-bed and prismatic core types are fundamentally sound design choices with historical experience demonstrating their effectiveness. Nevertheless, during the operation of both reactor types there have been instances where unexpectedly large fuel failure rates have arisen [73]. In some cases the high fuel failure rates were due to insufficient knowledge on the limitations of the particular fuel concept employed and/or poor fabrication and quality control methods [73]. In the remaining cases these fuel failures have occurred due to the inadequate performance of another system (such as the mechanical behaviour of control rods) which ultimately degraded the fuel, and poor instrumentation not being able to detect the anomalous reactor behaviour. These situations arose due to poor reactor design and were not fundamental issues with the fuel form.

6.1.1.3. Spent Fuel Waste

A downside with the low power density and encapsulated nature of the HTGR fuel is the relatively large quantities of irradiated spent fuel produced. Furthermore, the robustness of the fuel form makes reprocessing with current technology unfeasible. Hence, the volume of spent fuel is around a factor of 10 greater from a HTGR system than compared with spent LWR fuel. The difficulty in reprocessing does, however, improve the proliferation resistance of HTGR spent fuel relative to fuels associated with other reactor technologies.

6.1.2. Graphite

6.1.2.1. Graphite Behaviour Under Irradiation

A key feature of HTGRs is the reliance on graphite as not only the moderator but also as a major structural component. Graphite acts as neutron reflector, radiation shield, fuel container and flow channels for the coolant [83]. Graphite also acts as a heat sink during reactor trips and transients [74]. Therefore any severe degradation of the graphite material could lead to damage of safety critical reactor components.

During irradiation, many of graphite's physical properties can significantly change. Graphite will initially shrink during irradiation but at higher neutron doses graphite begins to swell. The changes in volume must be factored in to the reactor design to ensure stress limits are obeyed, which otherwise could lead to component failure. At very high irradiation doses, swelling will result in cracks and the development of pores within the graphite medium until total disintegration of the graphite takes place [58].

For any proposed HTGR the type of graphite utilised will need to have sufficient irradiation data gathered that span the temperature ranges the HTGR operates under. Most previous HTGR operation has typically had outlet temperatures of around 800°C (see Table 1) with some reactor concepts achieving temperatures of 950°C at brief points in operation. To further complicate matters, many of the historic HTGR programmes utilised now obsolete graphite grades [74].

6.1.2.2. Graphite Dust

During operation of HTGRs graphite dust can be produced, due primarily to the abrasion of surfaces made of graphite [83]. Any ingress of contaminants into the primary circuit, such as water vapour from

the steam generators or hydrocarbons from leaking components containing oil, can lead to graphite corrosion, further increasing the mass of graphite dust. The ingress of contaminants should be strictly limited and therefore abrasion is the dominant mechanism.

Graphite dust can become highly active during reactor operation by, for instance, low levels of uranium that were present in graphite during manufacture interacting with neutrons or, in the unlikely event fuel particles fail, fission products escape the particles and interact with the graphite [83]. Hence, the tight control of contaminants during manufacture and minimal fuel failure must be ensured.

The production of graphite dust raises two safety concerns [83]:

- a potential source of radionuclide release in the event radioactive graphite dust escapes the primary circuit; and
- a potential explosion hazard if dust particles are sufficiently fine and their concentration is above a certain threshold in the case air enters the primary circuit.

The use of filters can be used to reduce the inventory of activated dust particles in the primary circuit. Furthermore, experiments during the AVR test programme found that dust particles tended to bond tightly to surfaces within the reactor [83]. The combination of these two factors may considerably reduce the fraction of dust particles that could be released in the event the primary circuit is breached. However, further data to back these claims may be necessary to support a reactor's safety case.

Graphite dust can become deposited on the surfaces within heat exchangers and impede the transfer of thermal energy which can reduce the plant operational efficiency. The activated nature of graphite dust would also likely complicate repair and maintenance of components (such as heat exchangers) where the dust has collected [73].

The production of graphite dust is expected to be more pronounced in pebble-bed HTGRs than prismatic designs due to continual movement of pebbles within pebble-bed designs [83]. Hence there are greater incidents of abrasion occurring within pebble-bed HTGRs.

6.1.2.3. Graphite Oxidation

Graphite will oxidise at temperatures above ~600°C in the presence of air, with the oxidation rate increasing as a function of temperature. In addition, the oxidation process is very exothermic which can raise temperatures and therefore increase the oxidation rate [83]. Given that graphite not only acts as a barrier to fission product release since it surrounds the TRISO particles but that graphite is also an important structural material, any graphite degradation mechanism can have serious consequences. Moreover, the layers surrounding fuel kernel will be susceptible to oxidation once the host material (graphite) has become degraded. Graphite oxidation, in particular prolonged oxidation, would only arise in the event of a major fault. The likelihood of a major fault in any reactor that has successfully been licensed should be very low, otherwise it should not have been granted a licence to operate.

To achieve prolonged graphite oxidation requires air to be continuously circulated through the core at high flow rates to provide sufficient oxygen to drive the oxidation process. The air circulating through the core would also have to not induce a net cooling effect as the oxidation process decreases with lower temperatures [46]. A potential scenario that could result in optimal air ingress conditions would be two breaks in the primary circuit that are positioned in such a way that the heated air produced during oxidation leaves the primary circuit and draws air into the core [46]. Such a scenario seems unlikely but nevertheless may have to be taken into consideration during licensing.

The oxidation process described above can also result in the production of flammable gases, such as carbon monoxide, which, in addition to graphite dust, could theoretically result in violent chemical reactions taking place. However, the energy released under such a scenario would be significantly less than the violent steam reactions considered in PWR severe accident analyses [46].

The presence of any impurities in the helium coolant can also result in localised oxidation and the degradation of graphite within the core. Hence it is important to keep helium purities to very low levels in HTGRs.

6.1.3. Coolant Chemistry

Besides the importance of keeping the concentration of impurities within the helium to very low levels to minimise graphite corrosion (see Section 6.1.2), there is also the need to minimise impurities to reduce damage to systems within primary circuit. In particular, ensuring contamination of the primary circuit does not occur due to the failure of systems containing water (such as heat exchangers) or lubricants (such as helium circulators). The ingress of moisture from leaking components has resulted in numerous extended shutdowns with HTGR systems [73]. Hence, it is important that systems that could contaminate the primary circuit are highly robust and inspection routines are in place to detect the onset of failure.

6.2. Reactor Components

6.2.1. Intermediate Heat Exchangers

Intermediate Heat eXchangers (IHXs) are employed when designs necessitate separate fluids for cooling the reactor core and powering the turbines. Some designs allow the helium coolant to directly power the turbine, referred to as direct Brayton cycle, but such an approach risks spreading contaminants throughout the primary circuit. The spread of contaminants can result in failure of core components if the helium coolant is not kept sufficiently pure and/or radionuclides entering the turbine machinery. Radionuclides present in the turbine machinery can dramatically raise worker dose rates, thereby complicating maintenance procedures. However, utilising direct Brayton cycle removes the complications associated with IHXs discussed in this section.

For HTGRs that employ IHXs the high temperatures and long service life of the IHXs places high demands on the materials used in IHXs [82]. Most HTGR research and development conducted during the 1960s and early 1980s focused on steam-cycle power conversion systems [82]. Therefore, for HTGR systems employing steam cycles with shell-and-tube or helical steam generators and with outlet temperatures limited to $\sim 800^{\circ}\text{C}$, for which there is considerable experience, will likely have the fewest barriers to deployment. Alloy 617 (a nickel-based alloy) is the leading candidate for use in IHXs and has been extensively tested at temperatures up to 850°C [82].

The disadvantage with helical and shell-and-tube steam generators is their relatively large size and for systems with high power outputs (>1 GWth) can result in very expensive steam generators; hence the interest in more compact IHX designs. Compact IHX designs are, however, considerably less mature than steam generators and the relatively thin wall thickness of channels results in susceptibility to damage induced by corrosion and high stress concentrations. Furthermore, methods for performing non-destructive inspection have not been sufficiently developed [82].

6.2.2. Reactor Pressure Vessel

Given the long service life of Reactor Pressure Vessels (RPVs) it is important that the temperatures they are exposed to and neutron doses RPVs receive are minimised. It is usual practice to only expose the vessel wall of the RPV to the cooler inlet helium [75]. The low power densities of HTGRs (typically between 2 and 10 MW/m^3 [76]) helps minimise neutron doses to the vessel wall. Nevertheless any proposed RPV structural material will need sufficient material test data to ensure it suitability for prolonged use.

6.2.3. Control Rod Materials

Control rod sleeves are another component that can experience high temperatures and neutron doses. Alloy 800H (consisting of 42% Fe, 33% Ni, 21% Cr and small quantities of Al, Ti and C) is a leading candidate material for control rod sleeves but is only qualified for temperatures up to 760°C [82]. Furthermore, irradiation testing may be required to ensure the survivability of Alloy 800H for prolonged periods in the reactor due to irradiation induced damage [82].

6.3. Safety Performance

As discussed above, the safety performance of HTGRs in terms of retention of radionuclides during normal and transient conditions is highly reliant on the effectiveness of TRISO fuel at retaining fission products. Any scenarios that degrade the TRISO fuel are of concern.

The following events could potentially lead to the failure of the TRISO fuel:

- Chemical interactions between graphite and contaminants (discussed in Section 6.1.2.3);
- Mechanical damage; and
- Any event leading to fuel temperature increases sufficient to degrade the TRISO layers.

There have been a number of instances where poor reactor design has resulted in mechanical damage of TRISO fuel particles. Inadequate control rod design resulted in mechanical damage of the fuel elements in THTR (see Appendix 3). Failure of reactor components can also arise due to highly energetic coolant inducing vibrations within flexible structures within the primary circuit, such as heat exchangers [73]. The vibrating structures can eventually fail due to mechanical wear or fatigue. HTGRs must demonstrate that the design being employed is not susceptible to such failure mechanisms.

Typical processes that can lead to elevated temperatures resulting in TRISO fuel degradation include: the inadvertent addition of material that can increase core reactivity; accidental partial activation of one or more reactivity control systems; poorly characterised coolant flow; and exothermic chemical interactions, such as graphite oxidation.

Pathways for inadvertently adding material to the core resulting in an increase in power production should ideally be eliminated during the design process. For HTGR designs employing a steam cycle there is the possibility that large volumes of water could enter the reactor core and, water being a neutron moderator, could result in an increase in reactor power and ultimately fuel failure. It is possible to design cores such that the amount of water capable of entering the region containing fuel is limited to such an extent that resulting power increases does not result in fuel failure [46]. There are cases, such as misloading of fuel elements during reloading operations that can arise due to poor operational procedures. If the misloading were to lead to power increases and fuel failures, core instrumentation must be capable of detecting the resulting abnormal power distribution and initiate countermeasures (e.g. reactor shutdown).

Historical HTGR operation has showed that partial activation of reactivity control systems, such as control rods, can occur without being detected [73]. Over time the reactivity power profile can drift from its nominal distribution and result in fuel damage. Such scenarios have affected large HTGRs (see Appendix 3) and highlight the importance of robust, reliable and accurate instrumentation and control systems.

A number of HTGRs have suffered from unpredicted and undesirable coolant flow pathways arising during operation, known as bypass flow. Insufficient cooling around reactor components can occur if a large proportion of the coolant is not flowing along desired pathways within the core and creates irregular temperatures resulting in failure of core components. It is therefore essential that HTGR systems ensure bypass flow is minimised and accounted for in reactor design.

The ingress of air at high temperatures cannot only lead to graphite degradation (see Section 6.1.2.3) but, once particles are exposed, also destruction of the outer pyrolytic carbon TRISO layer. The outer pyrolytic layer oxidises over a period of $\sim 10^2$ hours in an air atmosphere at around 1000°C [71]. At higher temperature ($\sim 1400^\circ\text{C}$), in the presence of air, the SiC layer starts to degrade and once severe degradation arises within the SiC layer then a number of hazardous fission products can escape the

fuel particles [71]. However, for such severe degradation to arise then temperatures within the breached primary circuit must remain very high and be sustained over long time periods. Such a scenario may not be possible as there will very likely be a number of mechanisms, for example the inherent reduction in decay heat as time passes, that limit core temperatures. Nevertheless, HTGR designs will need to show that air ingress does not result in excessive release of radionuclides.

6.4. HTGR Conclusions

High Temperature Gas Reactors (HTGRs) benefit from the considerable historic work performed and recent programmes involving operational HTGRs and qualifying TRISO fuel performance. Most experience has been with UO₂ TRISO fuel and HTGRs employing steam raising plants.

UO₂ TRISO fuel has been demonstrated to withstand high temperature operation (~1200°C) and burnups up to 100 GWd/tHM. Furthermore, under transient conditions - lasting around 100 hours, and resulting in fuel temperatures reaching ~1600°C - UO₂ TRISO fuel is capable of effectively retaining all the hazardous fission products. Uranium oxycarbide (UCO) fuel has demonstrated improved irradiation performance relative to UO₂ fuel but its performance under transient conditions appears to require further demonstration to ascertain UCO's performance at very high temperatures. A key part of successful HTGR operation, and indeed any reactor, is ensuring that the capabilities of the fuel are well understood and reactor operation, including under accident conditions, does not surpass these limits.

HTGRs are capable of achieving sustained reactor operation with core outlet temperatures ~800°C. Higher outlet temperatures up to 950°C with current technology may be possible but it is uncertain whether sustained operation at such temperatures is possible. Outlet temperatures ~1000°C are not possible with current technology as rapid degradation of a number of reactor components occurs.

HTGRs come in two variants depending on fuel element type, with pebble-bed cores utilising spherical fuel elements that continuously travel through the reactor core, and prismatic cores utilising stationary cylindrical fuel elements. Pebble-bed cores are able to utilise fuel with lower enrichments; have the potential for greater shutdown margins; and have higher capacity factors due to online refuelling. Prismatic cores on the other hand benefit from reduced fuel element wear (with an expected reduction in graphite dust production); reduced uncertainty in power distributions as fuel does not stochastically move through the reactor; and overall a simpler reactor design which may reduce downtime.

Whilst HTGRs are generally regarded as exhibiting very high safety margins, due to the use of robust TRISO fuel and the use of graphite as a structural material, there are certain incidents that could lead to significant radionuclide release and/or major damage to the reactor structure. The following potential events would need to be factored in to the safety case:

- The release fraction of activated dust in the event the primary circuit is breached;
- Scenarios that could lead to sustained graphite oxidation; and
- The production of explosive mixtures of flammable gases (from graphite oxidation) and/or graphite dust (from abrasion or the ingress of contaminants causing graphite corrosion).

It appears that small HTGRs have a greater likelihood of deployment in the near-term compared to larger reactors due to inherent beneficial reactivity and instrumentation characteristics associated with smaller HTGRs. This is corroborated by the fact that small HTGRs have tended to perform well historically relative to their larger counterparts. However, there are a number of outstanding issues with HTGR technology which, taken together, are likely to result in delaying deployment until after 2030. The issues identified are: uncertainties in the quantity and mobility of graphite dust; overcoming current limitations in enrichment; ensuring there is sufficient material performance data on the grade of graphite selected for HTGR construction; and having in place a commercial supplier of the desired fuel elements. Nevertheless, the maturity of HTGRs and the relative ease of performing in-service inspection and repair in comparison to the other emerging technologies, means that commercial deployment by around 2035 is deemed possible.

7. Conclusions

The Emerging Small Modular Reactor Technologies are:

- Sodium-cooled Fast Reactors (SFRs);
- Lead-cooled Fast Reactors (LFRs);
- Molten Salt Reactors (MSRs); and
- High Temperature Gas Reactors (HTGRs).

None of these systems have demonstrated successful commercial operation. They have not demonstrated the ability to operate with high capacity factors (~90%) over long time periods (>10 years) whilst being cost competitive with other commercial reactor systems such as Pressurised Water Reactors (PWRs) and Boiling Water Reactors (BWRs). However, all offer some benefits (along with some disadvantages) over current commercial reactors.

The fast spectrum emerging technology systems (SFRs, LFRs and some MSR designs) are able to offer major improvements in uranium fuel utilisation and considerable reductions in the burden posed with waste produced from conventional reactor systems. With regards to fuel utilisation, it must be noted that uranium is not currently a scarce resource and therefore fuel utilisation is unlikely to confer significant benefits for some time.

Of the fast reactor systems, SFRs are the most mature, with around 400 years of reactor operation, and considerable historic and ongoing research. Beyond the general benefits attributable to fast reactors, SFRs offer considerable safety improvements with respect to managing the heat produced from nuclear fuel under accident conditions. Sodium's reasonably high boiling point at atmospheric pressure and excellent heat transfer properties allow SFRs to remove decay heat without the need for active (externally powered) systems nor sacrificing high power densities. This is a considerable benefit over current commercial reactor systems. However, this benefit needs to be balanced against sodium's high chemical reactivity with water and air. In addition, the optical opaqueness and chemical reactivity of liquid sodium makes In-service Inspection and Repair (ISI&R) difficult.

LFRs are far less mature than SFRs. However, the coolants employed in LFRs (Lead Bismuth Eutectic (LBE) and molten lead) do not exhibit high chemical reactivity with water and air, which is a significant advantage lead-based coolants have over liquid sodium. Thanks to the properties of LBE and molten lead, LFRs have very high thermal inertia which is beneficial under a variety of accident conditions. However, LFRs face major deployment barriers with respect to the material challenges posed by lead-based coolants. Both LBE and molten lead are very corrosive and relatively low coolant velocities can result in excessive erosion of reactor structural materials. With current materials it is necessary to accurately control oxygen concentrations in LFRs which is yet to be proven at a commercial scale under prototypic LFR conditions. Furthermore, the high density and high melting point of lead-based coolants result in even more challenging ISI&R environments than those facing SFRs.

MSRs are unique with respect to the other emerging technologies as they can readily operate with either fast or thermal spectra depending on the choice of materials employed. The liquid fuel nature of MSRs, and the fact that in most MSRs the liquid fuel also functions as the coolant, departs radically from existing solid fuel experience. The use of a liquid fuel that also functions as the coolant allows for extensive online reprocessing to significantly improve fuel efficiency. Molten salt coolants exhibit similar benefits and drawbacks to lead-based coolants; both coolants exhibit high boiling points and are less reactive with water and air than sodium but their high melting points and corrosiveness raises difficulties. A significant benefit fast spectrum MSRs have over other fast reactor designs is the strong negative thermal feedback achievable in fast spectrum MSRs.

The difficulties associated with deploying MSRs depends on the extent of online reprocessing and how far designers aim to replicate historical experience that focused on low power density, thermal spectrum reactors. Furthermore, most experience with molten salt coolants is based on molten fluoride salts. Hence, MSRs operating with graphite moderators and with reprocessing limited to elements that naturally separate from the liquid fuel are likely to be ready for deployment sooner

compared with fast spectrum MSRs with extensive online reprocessing. However, both MSR types are far less mature than SFRs and HTGRs.

HTGRs are able to draw upon the considerable historic and ongoing research programmes. Compared with the other emerging technologies their designs are significantly less complex, with a heavy reliance on a well-tested fuel form to confer safety and performance benefits. Furthermore, reactivity control in thermal reactors is easier to achieve under a variety of scenarios than with fast spectrum systems. The material challenges with HTGRs are heavily dependent on coolant temperatures and are known to be of limited concern if the helium coolant is less than around 800°C. A considerable disadvantage with HTGRs is the difficulty in reprocessing the fuel form; however, the counterargument is that this increases proliferation resistance and fuel utilisation is currently of no concern when uranium is widely available. Yet spent fuel volumes will become considerable in the event HTGRs are widely deployed.

The lack of experience licensing HTGRs and the known issues surrounding graphite oxidation and the production of graphite dust embedded with radionuclides, will likely delay HTGR deployment. In addition, the absence of commercial suppliers of large quantities of HTGR fuel and potential issues surrounding procuring fuel enriched beyond 5 wt.%, are sufficient to warrant it unlikely that HTGRs will be deployed commercially before 2030. However, the relative maturity of HTGR technology means that deployment by around 2035 is deemed possible.

In summary, small modular SFRs, LFRs, MSRs and HTGRs are unlikely to achieve commercial deployment before 2030. There is no reason to believe that commercial deployment of HTGRs could not be achieved by around 2035 and HTGRs are very likely to be commercially deployable before the other emerging technologies. With respect to the other systems, SFRs are closer to commercial deployment than MSRs and LFRs. Low power density thermal spectrum MSRs, with limited online reprocessing, and LFRs, are at comparable levels of maturity. Fast spectrum MSRs with extensive online reprocessing and operating at high power densities are the least mature and likely to take the longest time to deploy.

8. References

- [1] P. Tsvetkov, A. Waltar and D. Todd, "Reactor Plant Systems" in Fast Spectrum Reactors, Chapter 12, Springer (2012)
- [2] Toshiba Corporation and CRIEPI, "Super-safe, small and simple reactors (4S, Toshiba Design)", IAEA Advanced Reactor System Information (2013)
- [3] B. S. Triplett, E. P. Loewen and B. J. Dooies, "PRISM: A Competitive Small Modular Sodium-cooled Reactor", Nuclear Technology, Vol. 178, pp 186 – 200 (2010)
- [4] D. C. Wade and L. Walters, "ARC-100: A Sustainable, Modular Nuclear Plant for Emerging Markets", International Congress on Advances in Nuclear Power Plants, San Diego, USA (2010)
- [5] H. Zhao, H. Zhang, V.A. Mousseau and P.F. Peterson, "Improving SFR economics through innovations from thermal design and analysis aspects", Nuclear Engineering and Design, Vol. 239, pp 1042-1055 (2009)
- [6] Y. Guérin, "In-reactor Behaviour of Fuel Materials" in The Nuclear Fuel of Pressurized Water Reactors and Fast Reactors design and behaviour, Chapter 3, CEA (1999)
- [7] B. Raj, "Core Materials" in Fast Spectrum Reactors, Chapter 11, Springer (2012)
- [8] R. L. Klueh and A. T. Nelson, "Ferritic/martensitic steels for next-generation reactors", Journal of Nuclear Materials, Vol. 371, pp 37-52 (2007)
- [9] F. Baque et al, "ASTRID In Service Inspection and Repair: review of R&D program and associated results", International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios, Paris (2013)
- [10] F. Baque et al, "Challenges and R&D program for improving inspection of sodium cooled fast reactors and systems", Fast Reactor and Related Fuel Cycles: Challenges and Opportunities, Kyoto (2009)
- [11] F. Jadot et al, "In-Service Inspection and Repair for the ASTRID Project: Main Stakes and Feasible Solutions", International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios, Paris (2013)
- [12] K. Vulliez et al, "R&D Program on sealing issues for in-service inspection and repair tools on ASTRID sodium prototype, International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios, Paris (2013)
- [13] J. Sackett, "General Safety Considerations" in Fast Spectrum Reactors, Chapter 13, Springer (2012)
- [14] IAEA, "Status and Trends of Nuclear Fuels Technology for Sodium Fast Reactors", IAEA Nuclear Energy Series No. NF-T-4.1 (2011)
- [15] M. Denman et al, "Sodium Fast Reactor Safety and Licensing Research Plan – Volume I", Sandia National Laboratories, SAND2012-4260, (2012)
- [16] L. Walters et al, "Sodium Fast Reactor Fuels and Materials: Research Needs", Sandia National Laboratories, SAND2011-6546 (2011)
- [17] T. Ogata, "Metal Fuel" in Comprehensive Nuclear Materials, Chapter 3.01, Elsevier Ltd. (2012)
- [18] Y. Guérin, "Fuel Performance of Fast Spectrum Oxide Fuel" in Comprehensive Nuclear Materials, Chapter 2.21, Elsevier Ltd. (2012)
- [19] J.S. Cheon et al, "Sodium fast reactor evaluation: Core materials", Journal of Nuclear Materials, Vol. 392, pp. 324-330 (2009)
- [20] H. Kumar D, A. Somireddy and K. Gururaj, "A review on critical aspects of 316LN austenitic stainless steel weldability", International Journal of Materials Science and Applications. Vol. 1, No. 1 (2012), pp 1-7

- [21] K. Natesan and M. Li, "Materials Performance In Sodium-Cooled Fast Reactors: Past, Present, And Future", International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios, Paris (2013)
- [22] IAEA, "Liquid Metal Cooled Reactors: Experience in Design and Operation", IAEA-TECDOC-1569, 2008
- [23] G. Laffont, L. Cachon, V. Jourdain and J.M. Faque, "ASTRID power conversion system: assessment on steam and gas options", International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios, Paris (2013)
- [24] CEA, "4th-Generation sodium-cooled fast reactors: The ASTRID technological demonstrator", CEA (2012)
- [25] Teodora 25gan et al, "State of the Art for Fuel-Coolant Interactions Research for LFRs", In-pile testing and instrumentation for development of generation-IV fuels and materials, Halden, Norway (2012)
- [26] L. O. 26 and J. Baret, "Purification of Lead-Bismuth Eutectic in Accelerator Driven Systems", WM2009, Phenix, AZ (2009)
- [27] G. I. Toshinsky et al, "Principles of Providing Inherent Self-Protection and Passive Safety Characteristics of the SVBR-75/100 Type Modular Reactor Installation for Nuclear Power Plants of Different Capacity and Purpose", Global 2007, Boise USA (2007)
- [28] A. Alemberti et al, "Lead-cooled Fast Reactor (LFR) Risk and Safety Assessment White Paper", Gen IV International Forum, (2014)
- [29] K. Tucek, J. Carlsson and H. Wilder, "Comparison of Sodium and Lead-cooled Fast Reactors Regarding Severe Safety and Economical Issues", 13th International Conference on Nuclear Engineering
- [30] IAEA Nuclear Energy Series, "Liquid Metal Coolants for Fast Reactors Cooled by Sodium, Lead, and Lead-Bismuth Eutectic", IAEA (2012).
- [31] E. O. Adamov, "Demonstration of Neutral Safety: BREST-300 Reactor" in Chapter 15 of "White Book of Nuclear Power", NIKIET (2001)
- [32] NEA, "Handbook on Lead-bismuth Eutectic Alloy and Lead Properties, Materials Compatibility, Thermal-hydraulics and Technologies", OECD-NEA, no. 6195 (2007).
- [33] NEA, "Handbook on Lead-bismuth Eutectic Alloy and Lead Properties, Materials Compatibility, Thermal-hydraulics and Technologies – 2015 Edition", OECD-NEA, no. 7268 (2015).
- [34] T. R. Allen and D. C. Crawford, "Lead-cooled Fast Reactor Systems and the Fuels and Material Challenges", Science and Technology of Nuclear Installations (2007).
- [35] C. Smith, "Lead-Cooled Fast Reactor (LFR) Design: Safety, Neutronics, Thermal Hydraulics, Structural Mechanics, Fuel, Core, and Plant Design", Lawrence Livermore National Laboratory, LLNL-BOOK-424323 (2010)
- [36] Gen IV International Forum, "Technology Roadmap Update for Generation IV Nuclear Energy Systems", OECD-NEA (2014)
- [37] N. Li et al, "Corrosion Test of US Steels in Lead-Bismuth Eutectic (LBE) and Kinetic Modeling of Corrosion in LBE Systems", LA-UR-01-5241 (2001)
- [38] IPPE, EDO 'Gidropress' (Russian Federation), "Lead Bismuth Cooled Fast Reactor SVBR-75/100", IAEA, (2007)
- [39] L. Cinotti, "LFR: Fuels", IAEA Education and Training Seminar/Workshop on Fast Reactor Science and Technology, CNEA Bariloche, Argentina (2012)
- [40] E. P. Loewen and A. T. Tokuhira, "Status of Research and Development of the Lead-Alloy-Cooled Fast Reactor", Journal of Nuclear Science and Technology, Vol. 40, pp 614-627

- [41] Rogozkin, B.D. Stepenova, N.V. et al, "Mononitride U-Pu Mixed Fuel and its Electrochemical Reprocessing in Molten Salts", IAEA TECDOC 1348 (2003)
- [42] L. Cinotti, C.F. Smith et al, "Lead-Cooled Fast Reactor (LFR) Design: Safety, Neutronics, Thermal Hydraulics, Structural Mechanics, Fuel, Core, and Plant Design", in "Handbook of Nuclear Engineering", Springer (2010)
- [43] M. Frogheri, A. Alemberti and L. Mansani, "The Lead Fast Reactor: Demonstrator (ALFRED) and ELFR Design", IAEA, International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios, Paris, France (2013)
- [44] A. Weisenburger, A. Gessi et al, "Materials for ALFRED and ELFR – Selection and Challenges", IAEA, International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios, Paris, France (2013)
- [45] V. Sobolev, "Properties of Liquid Metal Coolants" in "Comprehensive Nuclear Materials", Elsevier (2012)
- [46] IRSN, "Review of Generation IV Nuclear Energy Systems", IRSN (2015)
- [47] O. Beneš and R. J. M. Konings, "Molten Salt Reactor Fuel and Coolant" in Comprehensive Nuclear Materials, Chapter 3.13, Elsevier Ltd. (2012)
- [48] D. F. Williams, L. M. Toth and K. T. Clarno, "Assessment of Candidate Molten Salt Coolants for the Advanced High-Temperature Reactor (AHTR)", ORNL/TM-2006/12 (2006)
- [49] V. Ignatiev and A. Surenkov, "Material Performance in Molten Salts" in Comprehensive Nuclear Materials, Chapter 5.10, Elsevier Ltd. (2012)
- [50] J. Serp et al, "The molten salt reactor (MSR) in generation IV: Overview and perspectives", Progress in Nuclear Energy, Vol. 77, pp. 308-316 (2014)
- [51] T. Griffiths, J. Tomlinson and R. O'Sullivan, "MSR Review: Feasibility of Developing a Pilot Scale Molten Salt Reactor in the UK", Energy Process Developments Ltd (2015)
- [52] R. M. Boothby, "Radiation Effects in Nickel-Based Alloys" in Comprehensive Nuclear Materials, Chapter 4.04, Elsevier Ltd (2012)
- [53] ThorCon website: <http://thorconpower.com/design/thorcon-can> Last accessed on 05/10/2015, DE10007/06/48/02
- [54] World Nuclear Association website: <http://www.world-nuclear.org/info/Current-and-Future-Generation/Molten-Salt-Reactors/> Last access on 05/10/2015, DE10007/06/48/01
- [55] E.L. Compare et al, "Fission Product Behaviour in the MSRE", 55, (1975)
- [56] R. C. Robertson, "Conceptual Design Study of a Single-Fluid Molten-Salt Breeder Reactor", 56 (1971)
- [57] H. Xu, "Thorium Energy R&D in China", Thorium Energy Conference, CERN (2013)
- [58] T. D. Burchell, "Radiation Effects in Graphite" in Comprehensive Nuclear Materials, Chapter 4.10, Elsevier Ltd (2012)
- [59] D. LeBlanc, "Molten salt reactors: a new vision for a generation IV concept", Sustainable development through nuclear technology: 29th annual conference of the Canadian Nuclear Society and 32nd CNS/CAN student conference, Toronto, Canada (2008)
- [60] T. Auger et al, "The CNRS Research Program on the Thorium cycle and the Molten Salt Reactors", CNRS (2008)
- [61] M. Richardson, "Development of a Freeze Valve for use in MSRE", 61, (1962)
- [62] L. M. Toth et al, "Molten Fluoride Fuel Salt Chemistry", Space Nuclear Power Conference, Albuquerque, New Mexico, January 9 – 11, 1995
- [63] D. F. Williams and L. M. Toth, "Chemical Considerations for the Selection of the Coolant for the Advanced High-Temperature Reactor", ORNL/GEN4/LTR-05-011 (2005)

- [64] J. 64, C. Fiorina and B. Hombourger, "PSI Studies on Advanced fuel cycle options for Fast/Thermal MSR Utilizing Thorium", Thorium Energy Conference, CERN (2013)
- [65] G. T. Mays, A. N. Smith and J. R. Engel, "Distribution and Behavior of Tritium in the Coolant-Salt Technology Facility", ORNL/TM-5759 (1977)
- [66] A. Thekdi, "Waste heat to power – Economic tradeoff and consideration", 3rd Annual Waste Heat to Power Workshop 2007, Houston, Texas, September, 2007
- [67] R. B. Chapas and J. A. Colwell, "Industrial Technologies Program Research Plan for Energy-805 Intensive Process Industries," prepared by Pacific Northwest National Laboratory for the U.S. 806 DOE (2007)
- [68] K. Sawa, "TRISO Fuel Production" in Comprehensive Nuclear Materials, Chapter 3.06, Elsevier Ltd. (2012)
- [69] K. Verfondern, "TRISO Fuel Performance Modeling and Simulation" in Comprehensive Nuclear Materials, Chapter 3.24, Elsevier Ltd. (2012)
- [70] D. A. Petti et al, "TRISO-Coated Particle Fuel Performance" in Comprehensive Nuclear Materials, Chapter 3.07, Elsevier Ltd. (2012)
- [71] IAEA, "Fuel Performance and fission product behaviour in gas cooled reactors", IAEA-TECDOC-978 (1997)
- [72] A. C. Kadak, "A future for nuclear energy: pebble-bed reactors", International Journal of Critical Infrastructures, Vol. 1 (2005)
- [73] J. M. Beck and L. F. Pincock, "High Temperature Gas-cooled Reactors Lessons Learned Applicable to the Next Generation Nuclear Plant," INL (2011)
- [74] B. J. Marsden, "Nuclear Graphite for High Temperature Reactors", AEA Technology (2000)
- [75] D. Chapin, S. Kiffer and J. Nestell, "The Very High Temperature Reactor: A Technical Summary", MPR Associates Inc. (2004)
- [76] B. K. Mcdowell et al, "High Temperature Gas Reactors: Assessment of Applicable Codes and Standards", Pacific Northwest National Laboratory, 76 (2011)
- [77] General Atomics, "New Power For the Next Century – A Harvest of Technologies", General Atomics.
- [78] <http://www.thorium100.com/docs/Presentation%20for%20website.pdf> last accessed on 22/10/2015, DE10007/06/48/03
- [79] R. Arnold, "CHP and High Temperature Reactors", NC2I Conference, Brussels, September, 2015
- [80] E. Mulder, "X-Energy", NRC-DOE Workshop on Advanced Non-LWRs (2015)
- [81] AKME Engineering, "SVBR-100: New Generation Nuclear Power Plants for Small and Medium-Sized Power Applications", AKME Engineering (2013)
- [82] R. Wright, J. Wright and C. Cabet, "Material Performance in Helium-Cooled Systems" in Comprehensive Nuclear Materials, Chapter 5.11, Elsevier Ltd. (2012)
- [83] AREVA NP Inc., "Pebble Bed Reactor Technology Readiness Study", US DOE (2011)
- [84] A. V. Zrodnikov et al, "A Concept of the Innovative Nuclear Technology Based on Standardized Fast Reactors SVBR-75/100 with Lead-Bismuth Coolant for Modular Nuclear Power Plants of Different Capacity and Purpose", Global 2007, Boise USA (2007)
- [85] A. Vasile, "Sodium Cooled Fast Reactors Fuels", IAEA Seminar on Fast Reactors, Bariloche, Argentina, 1-5 October (2012)
- [86] P. Tsvetkov and A. Waltar, "Fast Reactor Data" in Fast Spectrum Reactors, Appendix A, Springer (2012)
- [87] <http://www.world-nuclear.org/info/Current-and-Future-Generation/Fast-Neutron-Reactors/> last accessed on 19/02/2016, DE10007/06/48/05

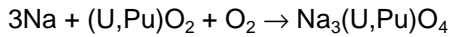
- [88] W. Ma et al, "Experimental and numerical study on lead-bismuth heat transfer in a fuel rod simulator", *Journal of Nuclear Materials*, Vol. 45, pp 415-424 (2011)
- [89] S. A. Maloy et al, "Tensile testing of EP-823 and HT-9 after irradiation in SINQ target-3", *Journal of Nuclear Materials*, Vol. 356, pp 27-49 (2006)
- [90] S. I. Porollo et al, "Microstructure and mechanical properties of ferritic-martensitic steel EP-823 after neutron irradiation to high doses in BOR-60", *Journal of Nuclear Materials*, Vol. 329, pp 14-18 (2004)
- [91] D. C. Crawford et al, "Fuels for sodium-cooled fast reactors: US perspective", *Journal of Nuclear Materials*, 371, pp 202-231 (2007)
- [92] R.D. Leggett and L. C. Walters, "Status of LMR fuel development in the United States of America", *Journal of Nuclear Materials*, 204, pp 23-32 (1993)
- [93] A. B. Cohen et al, "Fuel/cladding compatibility in U-19Pu-10Zr/HT9-clad fuel at elevated temperatures", *Journal of Nuclear Materials*, 204, pp 244-251 (1993)
- [94] B. Lee et al, "Preliminary Burnup Limits Analysis of the Barrier Cladding for KALIMER", *Transactions of the Korean Nuclear Society Autumn Meeting*, Gyeongju, Korea, November (2003)
- [95] C. Lee et al, "Status of SFR Metal Fuel Development", *International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios*, Paris, France, March 4-7 (2013)
- [96] F. Schmitz et al, "Fuel Behavior Aspects of the Interpretation of the Scarabee Fast Reactor Safety Experiments", *Nuclear Engineering and Design*, Vol. 59, pp 275-286 (1980)
- [97] IAEA, "Fast Reactor Database 2006 Update", IAEA-TECDOC-153 (2006)
- [98] Y. Chikazawa et al, "Technology Gap Analysis on Sodium-heated Steam Generators Supporting Advanced Burner Reactor Development", *ANS*, Vol. 164 (2008)
- [99] T. Kim, V. S. Yugay and J. Jeong, "Acoustic leak detection technology for water-steam small leaks and microleaks into sodium to protect an SFR steam generator", *Nuclear Technology*, Vol. 170, pp. 360-369 (2010)
- [100] M. Rawool-Sullivan, P. D. Moskowitz and L. N. Shelenjova, "Technical and Proliferation-Related Aspects of the Dismantlement of Russian Alfa-Class Nuclear Submarines", *The Nonproliferation Review* (2002)
- [101] Terrestrial Energy, "The Integral Molten Salt Reactor", *Molten Salt Reactor Workshop*, Delft University, 17th April 2015
- [102] IAEA, "Status Report 96- High Temperature Gas Cooled Reactor – Pebble-Bed Module (HTR-PM)", IAEA (2011)
- [103] A. Marmier et al, "Nuclear Cogeneration of High Temperature Process Heat and Electricity with Heat Pump Driven Superheating", *Proceedings of ICAPP 2007*, Nice, France, May 13-18, 2007
- [104] Committee on Climate Change, "The Fifth Carbon Budget - The next step towards a low-carbon economy", CCC, November 2015
- [105] Energy Technologies Institute, "Hydrogen - The role of hydrogen storage in a clean responsive power system" ETI (2015)
- [106] J. W. Griffin et al, "Under-Sodium Viewing: A Review of Ultrasonic Imaging Technology for Liquid Metal Fast Reactors", *Pacific Northwest National Laboratory*, PNNL-18292 (2009)
- [107] IAEA, "Status of Liquid Metal Cooled Fast Reactor Technology", IAEA, IAEA-TECDOC-1083 (1999)
- [108] L. J. Bond et al, "Instrumentation, Monitoring and NDE for New Fast Reactors", *Advanced Nuclear Fuel Cycles and Systems*, GLOBAL, September 9-13, 2007

- [109] T. Wu and J. Bowler, "Limitations of Eddy Current Testing in a Fast Reactor Environment", 42nd Annual Review of Progress in Quantitative Nondestructive Evaluation Conference, Minneapolis, Minnesota (2015)
- [110] S.C. Chetal and G. Vaidyanathan, "Evolution of Design of Steam Generator for Sodium Cooled Reactors", IAEA (1997)
- [111] IAEA, "Status of Fast Reactor Research and Technology Development", IAEA TECDOC 1691 (2012)
- [112] <http://www.world-nuclear-news.org/NN-Chinese-fast-reactor-completes-full-power-test-run-1912144.htm> Last accessed on 19/02/2016, DE10007/06/48/06
- [113] <http://mragheb.com/NPRE%20457%20CSE%20462%20Safety%20Analysis%20of%20Nuclear%20Reactor%20Systems/Steam%20Generator%20Leakage%20at%20the%20BN-350%20Desalination%20Plant.pdf> Last accessed on 19/02/2016, DE10007/06/48/07
- [114] T. R. Allen et al, "Corrosion of Zirconium Alloys", Chapter 5.03 of Comprehensive Nuclear Materials, Elsevier Ltd (2012)
- [115] K. Ogata et al, "Separate effects of factors affecting outside-in cracking of high burnup fuel cladding", TopFuel 2008, Seoul (2008)
- [116] F. Garzarolli et al, "Optimization of BWR fuel rod cladding condition for high burnups", ANS international topical meeting on LWR fuel performance, Orlando (2010)
- [117] K. D. Weaver et al, "Extending the Nuclear Fuel Cycle with Traveling-Wave Reactors", Proceedings of Global 2009, Paris, France, September 6-11 (2009)
- [118] P. Hejzlar et al, "Terrapower, LLC Traveling Wave Reactor Development Program Overview", Nuclear Engineering and Technology, Vol. 45, pp 731-744 (2013)

Appendix 1

A1.1. Compatibility of Sodium with (U,Pu)O₂ Fuel

When (U,Pu)O₂ comes into contact with sodium the following reaction takes place [18]:



This reaction occurs at the periphery of the fuel pellet and necessitates that availability of oxygen. As sodium coolant has a very low oxygen concentration (< 5 ppm) the oxygen is supplied from the fuel itself. Once the ready supply of oxygen is consumed the reaction stops. The sodium uranoplutonate (Na₃(U,Pu)O₄) layer is typically ~100 µm. The main effects of this layer are:

- A decrease in fuel conductivity associated with changes in fuel chemical make-up; and
- A swelling associated with the production of uranoplutonate which has a density around half that of (U,Pu)O₂.

The overall effect is an unfavourable reduction in the properties of (U,Pu)O₂ which could lead to further clad breach. However, experience has shown that in most failure cases there was limited fuel material release into the coolant and the fuel-sodium reaction that took place formed a scab over the breach site which would limit further release of fission products as long as the fuel rod operated under nominal power conditions for the remainder of its life [18].

A1.2. List of Sodium-cooled Fast Reactors that Have Achieved Criticality

Table A1: List of sodium-cooled fast reactors that have achieved criticality.

	Reactor Name	Country	Thermal Output	Electrical Output	Fuel	Date Critical	Shutdown
1	EBR-I	USA	1.4 MWth	0.2 MWe	U metal	1951	1963
2	BR-5	Russia	5 MWth	0 MWe	PuO ₂ , UC	1959	1971
3	BR10	Russia	8 MWth	0 MWe	UN	1959	1971
4	DFR	UK	60 MWth	15 MWe	U-Mo	1959	1977
5	EBR-II	USA	62.5 MWth	20 MWe	UFs, U-Zr	1963	1994
6	FERMI	USA	200 MWth	61 MWe	U-10Mo	1963	1972
7	Rapsodie	France	40 MWth	0 MWe	MOX	1966	1982
8	SEFOR	USA	20 MWth	0 MWe	MOX	1969	1972
9	BOR60	Russia	55 MWth	12 MWe	MOX	1969	In operation
10	KNK-II	Germany	58 MWth	21 MWe	MOX	1977	1991
11	BN-350	Kazakhstan	750 MWth	130 MWe	UO ₂	1972	1999
12	Phenix	France	563 MWth	250 MWe	MOX	1973	2009
13	PFR	UK	650 MWth	270 MWe	MOX	1974	1994
14	JOYO	Japan	140 MWth	0 MWe	MOX	1977	Interrupted since 2007
15	FFTF	USA	400 MWth	0 MWe	MOX	1980	1994
16	BN-600	Russia	1470 MWth	600 MWe	UO ₂	1980	In operation
17	FBTR	India	40 MWth	13 MWe	(U,Pu)C	1985	In operation

18	Super-Phenix	France	2990 MWth	1242 MWe	MOX	1985	1996
19	MONJU	Japan	714 MWth	280 MWe	MOX	1994	Interrupted since 1995
20	CEFR	China	65 MWth	20 MWe	UO ₂	2010	In operation
21	BN-800	Russia	2100 MWth	870 MWe	MOX	2014	In operation

A1.3. Issues with Steam Generators and the Secondary Circuit in Sodium-cooled Fast Reactors

Table A2: Performance data for steam generators in sodium-cooled fast reactors.

Reactor	Electrical Output	Issues with steam generators and secondary circuit
DFR (NaK) 1959-1977	15 MWe	The DFR Steam Generators (SG) were of extremely conservative design. Each consisted of separate but parallel serpentine tube coils (austenitic stainless steel) for sodium and feedwater/steam which were connected with each other only by brazed copper laminations [110]. Each water tube was surrounded by 4 sodium carrying tubes. The heat transferred from sodium to water was by conduction through the copper laminations [110]. The primary circuit of the reactor was NaK coolant but the secondary circuit was sodium. No significant problems were encountered.
EBR-II 1963-1994	20 MWe	EBR-II's steam generators never experienced a sodium-to-water or sodium-to-steam leak, which differed from the typical experience with sodium-cooled power plants world-wide. This performance was attributed to the robust duplex tube design of both the superheaters and evaporators [111]. As well as duplex tubes, a duplex tubeplate was used and arranged so no welds had water on one side and sodium on the other. One drawback with EBR-II's steam generator design was that the bonded tubes exhibited variable heat transfer efficiency as the tubes could separate due to thermal expansion [110].
BOR 60 1969-Present	12 MWe	BOR-60 has used a number of SG designs (U-tubes, straight tube and helical tubes). One major SG failure occurred in the testing of the BN-600 design and about 200g of sodium came into contact with 2kg of water before the systems could be shutdown [111].
KNK-II 1977-1991	21 MWe	KNK-II used the serpentine steam generator concept [110]. Only one sodium-water reaction occurred due to a faulty tube-to-tube weld just after the start of operation [111]. There were no hydrogen detectors installed at the time of the leak and failure was successfully detected by pressure changes in the argon cover gas [111].
FBTR (India) 1985-Present	13 MWe	FBTR utilises a serpentine shell and tube steam generator concept akin to the Phenix serpentine SG but with an integrated evaporator-superheater unit [110]. The only problems reported relate to a small number of tube end case failures [110].
CEFR (China) 2010-Present	20 MWe	The design of the reactor is heavily influenced by Russian experience. The reactor went critical in 2010 but only achieved full power in December 2014 [112]. There appears to be no data in the public domain regarding steam generator design or operation performance.
Fermi (USA) 1963-1972	60 MWe	The steam generators in Fermi, which were of serpentine design, were subject to numerous leaks and design faults, even during initial pre-preparation testing [110].
BN-350 1972-1999	130 MWe	The reactor is designed for both electricity generation (130 MWe) and water desalination (150 MWth) [113]. BN-350 uses U-tube superheaters and baronet evaporators [110]. Issues with steam generators initially limited power output to approximately one third of nominal power [113]. Leaks were detected in all the evaporators immediately during commissioning tests. Many tubes had to be plugged. There was one major incident in 1974 with a large failure in an evaporator, which allowed 800 kg of water to enter the sodium side [110]. The steam generator was replaced with a new modular design [110].

Phenix 1973-2009	250	Each secondary loop is connected to a steam generator consisting of 3 stages (evaporator, superheater, and reheater) with 12 modules in each stage. A serpentine tube-in-shell design was used [110]. Phenix had a number of issues with leaks in secondary circuit during early operation [110]. In the last phase of operation the only leaks were in a faulty hydrogen detection pump and the secondary circuit purification system [111].
PFR 1974-1994	270	The PFR design was U-tube steam generator design. The steam generators in PFR had serious and continuing problems with small leaks [110]. Problems appeared in superheaters and reheaters at the start of commissioning. Cracking occurred in poorly heat treated low alloy welds. Tubes were initially plugged and shot-peening the welds were tried to alleviate the problems. The problems continued with operation and leaks also found in the evaporators. In the end it was decided to sleeve all the welds with an explosive weld on the tubeplate and a Ni braze on the tubes. This effectively solved the problem after 1984 [110]. One case of a major fretting failure was found in a superheater in 1987. The unit was automatically shut down and sodium dumped in 10s, but despite this the failure had propagated to 39 other tubes. The superheaters and reheaters were replaced in the late 1980s but the prospect of further replacements was one of the influences in the decision to close the reactor.
BN-600 1980- Present	600	BN-600 has a straight shell-and-tube once-through modular steam generator design with evaporators, superheaters and reheaters [110]. A total of 12 failures have taken place. Six of these were in the first year of operation and no leaks have taken place since 1991. All except one failure were in the superheaters and reheaters, with failure at the rolled and welded tube to tubeplate joint. Repair was done by replacing the modules that had suffered failures. All failures were caught at an early stage and the effect on operation was minimal. The SG was designed to operate for 15 years and the evaporators were replaced in the mid-1990s.
Super Phenix 1985-1996	1242	Super Phenix used helical steam generators. This design did not use tubeplates; each tube penetrated the shell through thermal sleeves [110]. Individual tubes could be plugged from the outside [110]. No significant problems occurred during operation.
MONJU 1994 - Present	280	Monju has not had much operation due to a substantial loss of sodium (3 tonnes) from the failure of a thermowell in the secondary circuit that was not noticed. This occurred in 1995 just 4 months after commissioning. Monju's steam generators are of once-through helical type, which worked well during the limited operating time [110].
BN-800 2014-	800	BN-800 only achieved full power in December 2015 and is still undergoing pre-commercial operation trials.

Appendix 2

A2.1. Benefits and Challenges Related to Small Modular PWRs using Natural Convection Cooling

Some SM-PWR designs intend to eliminate the need for primary circuit pumps and hence rely on the removal of heat from the core via natural convection cooling. The vast majority of experience with regards to licensing and operating Pressurised Water Reactors has been gained from PWR systems that utilise primary circuit pumps (forced convection cooling).

Operating a PWR with primary circuit pumps allows for more efficient removal of heat from the core during normal operation. This allows the core to achieve higher power output than the case where pumps are not utilised. The higher power output contributes significant economic benefits as more electricity can be produced from a given core volume. Also, by operating with forced convection cooling there is a reduced likelihood for flow instabilities to arise that could impact the neutronic behaviour of the reactor core.

Some SM-PWRs eliminate the need for primary circuit pumps by relying on natural convection cooling driven by temperature, and therefore density, differences between the coolant at the top and bottom of the reactor core. The outlet temperatures for SM-PWRs of both the forced and natural convection cooling varieties are typically around 320°C. Inlet temperatures; however, vary, with natural convection cooling systems having lower inlet temperatures.

Elimination of primary circuit pumps simplifies reactor design, thereby potentially reducing the likelihood of faults occurring, and also avoids the need to purchase, install and maintain primary circuit pumps. Therefore, there are some economic advantages associated with removal of primary circuit pumps, which will to some extent offset the economic disadvantages of operating without pumps.

The lower power density of PWR systems that do not use primary circuit pumps is expected to improve the flexibility of the reactor system with respect to power changes. This is because the likelihood of fuel rod failure is greatly increased in situations where the pellet-clad thermo-mechanical equilibrium is perturbed. Hence, large changes in reactor power output over relatively short periods of time can result in fuel failure. Therefore, load following manoeuvres can significantly increase the likelihood of fuel failure if not carefully controlled. Fuel rods that operate at lower powers will exhibit less thermal expansion and greater retention of chemically aggressive fission products than those operating at higher powers. These attributes significantly reduce the likelihood of fuel rod failure during power manoeuvres for cores that operate with a low power density.

SM-PWRs that operate without primary circuit pumps will be capable of varying their power levels; however the speed at which power changes can occur is currently unclear. This is because in the event that the power level is increased in any PWR concept, the amount of thermal energy generated by fuel rods increases, and it is important that there is sufficient coolant flow to ensure fuel rod surface temperatures do not become too high, otherwise failure of the cladding could occur. PWR systems that rely on natural convection cooling will exhibit poorer heat transfer characteristics compared to PWR systems that employ primary circuit pumps. However, the lower power density of natural convection cooled SM-PWRs will limit the extent of rod surface temperature rises for a given flow rate. Hence, it is possible that with sufficiently low power densities natural convection cooled SM-PWRs would be capable of varying their power levels in a manner similar to SM-PWRs that incorporate primary circuit pumps. This would need to be demonstrated.

The low inlet temperatures in natural convection cooled SM-PWRs raise concerns associated with clad embrittlement. Oxidation reactions take place between the zirconium-based cladding materials and the water coolant. A small proportion of the liberated hydrogen from the H₂O coolant is also incorporated into the cladding. The presence of hydrogen in the cladding is a concern once the hydrogen begins to form zirconium hydrides (ZrH_{1.6} or ZrH₂), which happens more readily at low temperatures (<~250°C), since these hydrides are very brittle and hence their presence in the cladding reduces its fracture toughness and increases its susceptibility to cracking [114, 115, 116]. The degraded clad properties associated with clad embrittlement could place limits on the ability of natural convection cooled SM-PWRs to load follow relative to SM-PWRs that employ primary circuit pumps.

There are advantages associated with the removal of primary circuit pumps including design simplification and avoiding the need to purchase, install and maintain primary circuit pumps. Furthermore, the low power density of non-pumped SM-PWRs is beneficial with respect to flexibility. However, these benefits need to be assessed against reduced electrical output, the potential for undesirable flow instabilities, the need for test-programmes to demonstrate the thermal hydraulic behaviour and the potential for clad embrittlement to reduce the flexibility benefits.

Appendix 3

Complications Arising with Large HTGRs

Historically the operating performance of small (<~500 MWth) High Temperature Gas Reactors (HTGRs) has been more encouraging than large HTGRs (>~500 MWth). For instance the AVR (46 MWth) and Peach Bottom 1 (115 MWth) achieved capacity factors of ~60% and ~75% respectively, which given their prototype nature and limited prior experience with HTGR technology, can be considered very good operational performance histories [76, 77]. However, the subsequent large HTGRs, that were meant to build on the experience gathered during AVR and Peach Bottom 1 operation, exhibited poor operational histories.

Fort St. Vrain (FSV), an 842 MWth reactor, that used Peach Bottom 1 as the basis for its design, achieved very low capacity factors (~20%) [77]. The THTR, which was an evolution of the AVR system, also suffered a number of problems during operation resulting in low capacity factors (~40%) [76].

The small sample size for large HTGRs (only two HTGRs operated above 500 MWth) is not sufficient on its own to indicate whether there are insurmountable challenges arising with large HTGRs. However, there are certainly a number of difficulties associated with large HTGRs including:

- Large core geometry necessitates a greater number of control rods than with small reactors, with the neutronic impact of each control rod in a large HTGR only having marginal influence on core reactivity. Therefore, the accidental insertion of a small number of control rods can go undetected if instrumentation is not sufficiently detailed and for very large cores it can be difficult to achieve adequate shutdown margins if a number of control rods fail to engage.
- Large cores are more susceptible to perturbations in power profiles becoming magnified due to differential depletion in fuel. The perturbed power profile can result in hot spots developing and if no countermeasures are initiated then fuel failure can arise.

In principle the above points are able to be resolved with the addition of sufficient number of control rods and robust, reliable and detailed instrumentation. However, both FSV and THTR failed in these respects.

Besides the fundamental issues arising with large HTGRs a number of issues arose in THTR and FSV that were likely a result of poor design choices, in the case of the THTR these were:

- Blockages occurred within the pebble handling system due to a design error resulting in helium flowing in the wrong direction within a compartment of the pebble handling system [83]; and
- In order to overcome the difficulties with low control rod impact within large HTGRs, the THTR design placed control rods in the central fuel region. However, this decision resulted in control rods damaging fuel pebbles. The quantity of damaged fuel pebbles required numerous reactor shutdowns as the design of the pebble handling system necessitated reactor depressurisation when emptying drums containing damaged pebbles [83].

The FSV HTGR experienced a number of problems associated with poor instrumentation, including [73]:

- Not detecting that the secondary shutdown system had partially activated; and
- Divergent power profiles going undetected which resulted in fuel failure.

By far the biggest issue that faced operators of FSV was moisture ingress, primarily associated with using water as a lubricant within the helium circulator system which repeatedly leaked large amounts of water into the reactor during power operation. There was also no means to readily remove large quantities of moisture during operation which ultimately resulted in repeated shutdown and hence the low capacity factor. The ingress of water vapour resulted in numerous issues including: failure of control rods and secondary shutdown system; excessive corrosion of graphite within the core; and positive reactivity insertion [76].

It should also be noted that as the power output of HTGRs increases it becomes more difficult to ensure that passive means to confer high safety margins (e.g. the removal of decay heat via conduction, natural convection and radiation) can be relied on [46].

Overall, whilst large HTGRs have greater difficulties which necessitate robust, reliable and detailed instrumentation most of the setbacks associated with large HTGRs have been poor design choices.