Latest generation
. . . and what's to come

Nuclear power stations generate about 11% of the world's base load electricity but many older nuclear plants are near the end of their service life. What are their likely replacements? This article examines present day reactors and the new Gen IV designs.

First, let's look at the most common current design, the pressurised water reactor (PWR) and then we will describe the six Gen IV designs, all selected by the international Gen IV Forum (GIF) committee:

- Sodium Fast Reactor (SFR),
- Lead Fast Reactor (LFR),
- Gas Fast Reactor (GFR),
- Supercritical Water Reactor (SCWR),
- Very High Temperature Reactor (VHTR) and
- Molten Salt Reactor (MSR).

The pressurised water reactor accounts for 65% of the world's ~450 nuclear power plants (NPPs). This wasn't always the case and in the 1950s many countries developed their own designs.

Thus, the Canadians developed the CANDU heavy water moderated reactor that used natural uranium (99.27% U-238, 0.73% U-235). The UK developed the gas-cooled reactors (eg, Magnox and AGRs) which also used natural uranium and are very safe on account of their low power density (with lots of graphite in the core and not a lot of fuel).

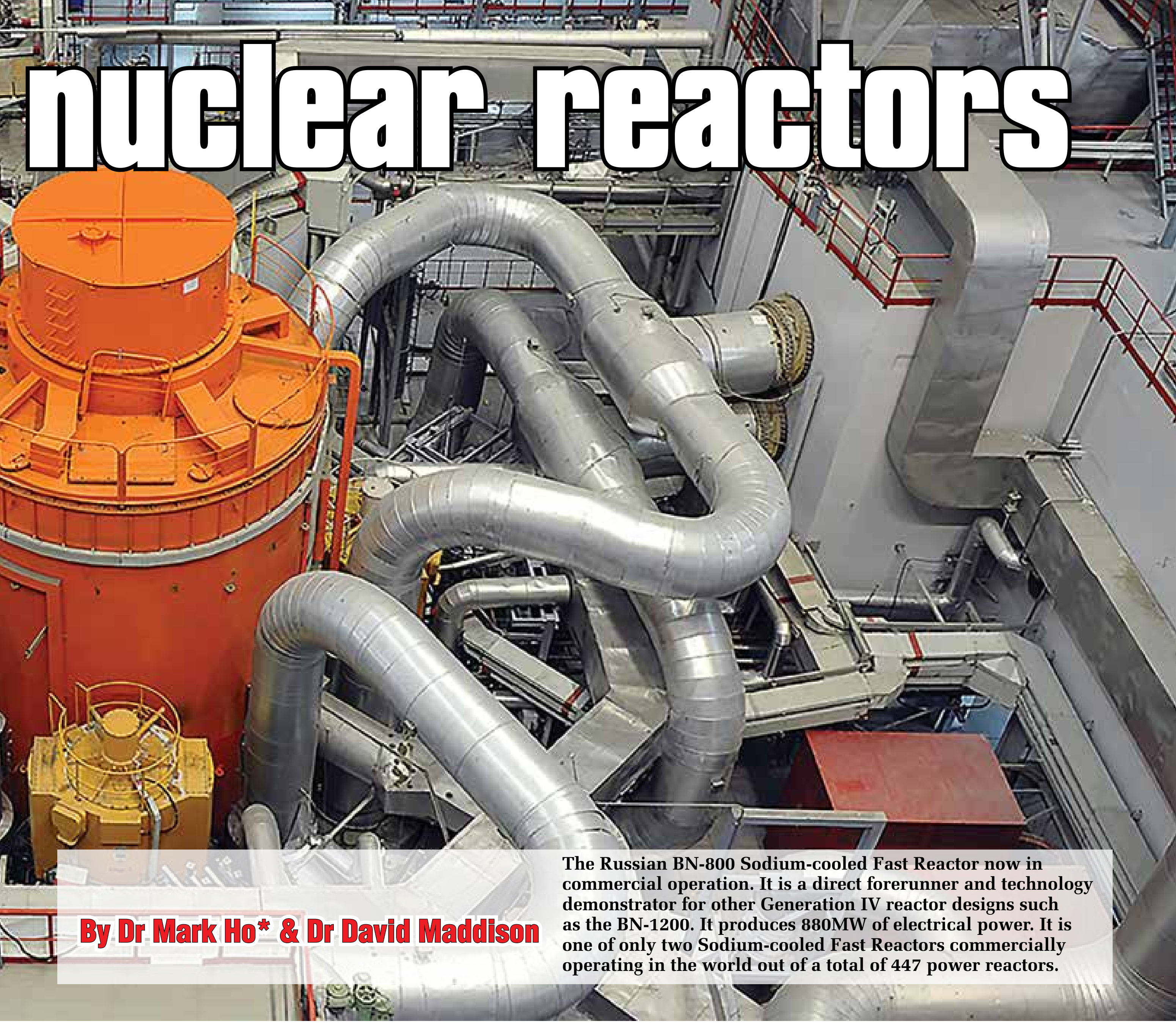
For their part, the Americans developed a compact pressurised water reactor (PWR) that used highly enriched uranium (>20%) to power their naval vessels. From there, they developed land-based PWRs up to 1350MWe (megawatts of electrical power) using low enriched uranium (5%). These have been found to be very economical to operate.

Subsequently, PWRs have been widely deployed in Russia, China, Japan, UK, France and other European Countries, displacing these countries' own designs.

PWRs are very safe on account of their negative thermal reactivity feedback – meaning that the hotter the core gets, the less nuclear reaction takes place in the core. The materials and heat transfer characteristics of PWRs are well known.

Water under pressure is well understood, as are the properties of steel which makes up the reactor pressure vessel (RPV) and the zirconium alloy 'fuel pins' (ie, hollow tubes) that contain the sintered uranium-dioxide fuel pellets.

nuclear reactors



By Dr Mark Ho* & Dr David Maddison

The Russian BN-800 Sodium-cooled Fast Reactor now in commercial operation. It is a direct forerunner and technology demonstrator for other Generation IV reactor designs such as the BN-1200. It produces 880MW of electrical power. It is one of only two Sodium-cooled Fast Reactors commercially operating in the world out of a total of 447 power reactors.

So nuclear regulators have confidence in these designs and PWRs have become the mainstay of the global nuclear fleet.

After some 50 + years of operations, these Generation II PWRs are nearing the end of their service life and are being slowly replaced by Gen III PWRs and BWRs (Boiling Water Reactors which generate steam directly in the reactor core).

Gen III reactors have active and passive safety systems which ensure heat can be removed from the reactor core after shutdown.

Why is this necessary?

In a nuclear reaction, a typical uranium-235 nucleus with 92 protons and 143 neutrons can split after absorbing a neutron, producing two elements of lower mass numbers (fission products), 2-3 neutrons and some energy in the form of gamma radiation.

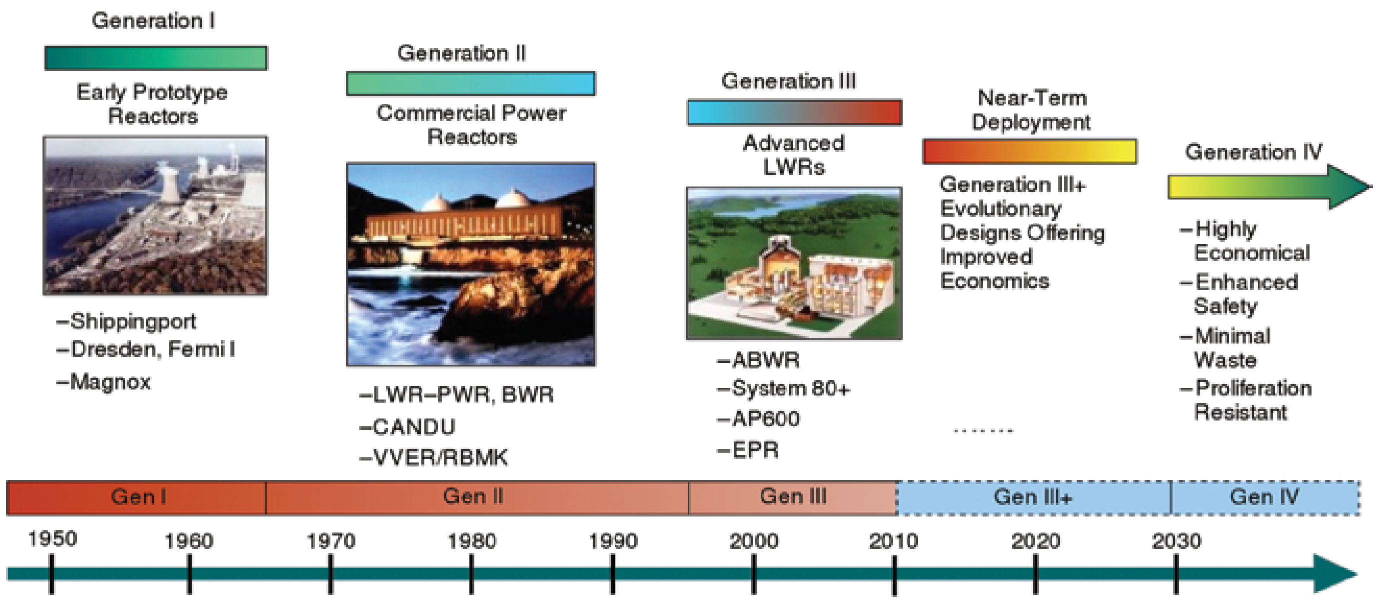
The fission products continue to radioactively decay after shutdown, generating roughly 1.2% of the reactor heat at full power one hour after the control rods are dropped. So for a 3000MW-thermal / 1000MW-electric reactor, the core continues to generate 36MWth (megawatts of thermal output) one hour after shutdown.

This 'decay heat' is removed either by pumps to drive water through the core or as in the case of some Gen III reactors, by natural circulation which does not require pumps or off-site power. New PWRs and BWRs are often built with large water reservoirs that act as a "thermal-sink" for decay heat removal.

By eliminating the need for off-site power, Fukushima-type accidents would be eliminated.

Apart from needing improved safety features, there are other other features which one might have for a nuclear reactor. These include:

(1) to breed nuclear fuel without creating nuclear weapons materials (ie, non-proliferation)
(2) to burn radioactive waste
(3) to burn nuclear fuel more completely
(4) to supply high temperature heat for industrial processes



Timeline showing development of various generations of reactors. Generation IV reactors are intended to be deployable no later than 2030. ***Image credit: US Nuclear Engineering Division***

(5) to operate more economically.

Not surprisingly, these attributes are the expressed goals of the Gen IV forum (GIF) which is a group of 14 nations (now including Australia) working together on the next generation of power reactors.

So let us discuss these desired points.

Fuel breeding and non-proliferation

Currently, PWRs cannot breed enough fuel to be self-sustaining. In fact, readers might be surprised to know PWRs and BWRs do create fuel by exposing the 'fertile' uranium-238 content (95% of the uranium-dioxide) to neutron bombardment. This results in neutron absorption and transmutation into the fuel plutonium-239.

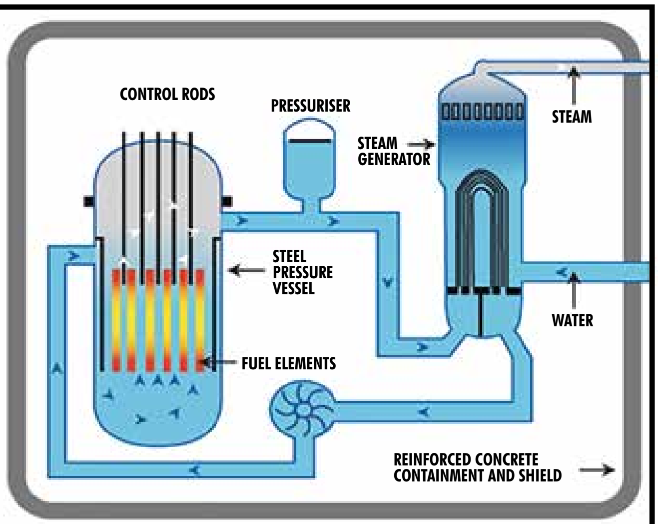
What is more interesting is that about half of the power that comes from a usual 18 month burn-cycle (the duration a fuel bundle is in the core) actually comes from burning plutonium created in the core when exposed to neutrons! Thus the bred plutonium is beneficial as it's essentially 'free power'.

Some people may ask whether "bomb-grade material" is being made in the reactor. The short answer is no, because plutonium 240 is also made along with Pu-239 in the core and the mixture of both makes it unusable as a bomb material.

There is also no easy way to separate Pu-240 from Pu-239 without a dedicated isotopic-separation facility which is difficult to engineer, requires large amounts of power to operate and thus is difficult to hide from satellite surveillance.

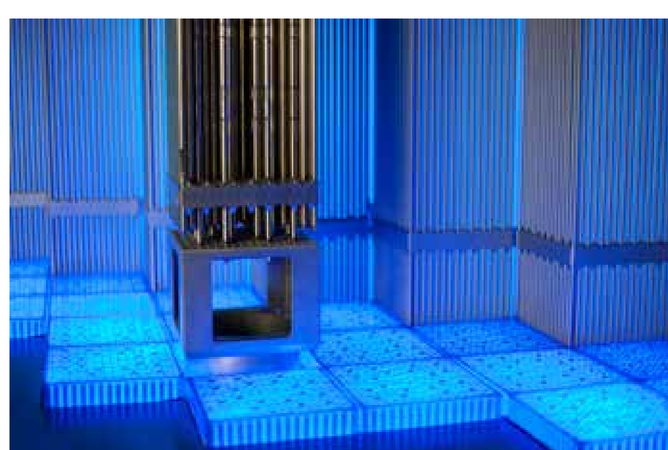
Despite progress made to maximise fuel breeding in PWRs, the maximum PWR conversion ratio (ie, total fuel produced/ total fuel burnt) is about 0.6 or 60%.

A self-sustaining fuel cycle would require a conversion ratio above 1.0. To do so would also require a very different type of reactor, one that operates in the 'hard neutron spectrum'.

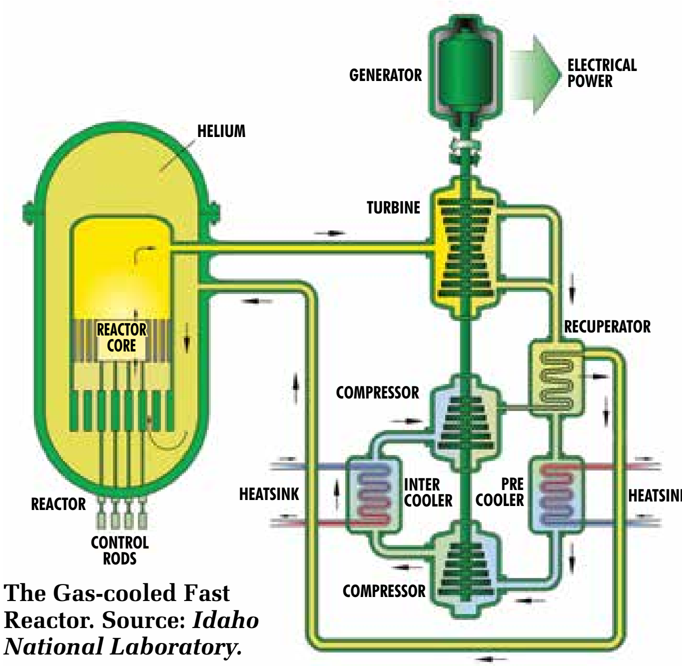


Pressurised Water Reactor

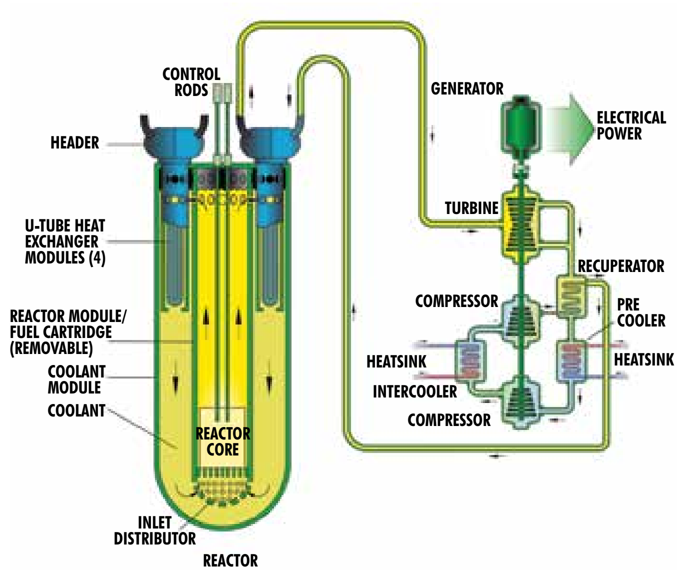
Fuel: uranium dioxide (4 - 5% enriched)
Fuel Cladding: Zircaloy (98% zirconium, 2% tin)
Moderator: light water
Loops : 2 – primary & secondary
Coolant: light water – light water
Core temperature: 300 – 330°
Operating pressure: 150 atm
Rankine (steam) cycle: 33% efficiency



Nuclear fuel inside a reactor.



The Gas-cooled Fast Reactor. Source: *Idaho National Laboratory.*



Lead-cooled Fast reactor. Note the natural convective pathway for cooling. Source: *Idaho National Laboratory.*

PWRs operate in the thermal neutron spectrum, when neutrons are slowed to the speed of gas molecules at room temperature, about 0.25eV (electron volts). Fast neutron reactors operate in the hard neutron spectrum with neutrons zipping around at 5% the speed of light at ~1MeV.

An example of a much-studied fast reactor is the SFR, the Sodium Fast Reactor.

The conversion ratio for the SFR is theoretically limited to 1.3. Since the conversion value is > 1.0, it's called the "breeding ratio".

The probability of neutron capture for all nuclear fuels are two to three orders of magnitude less in the fast spectrum than in the thermal spectrum. Thus a fast neutron reactor requires a lot more fissile material than a 'thermal reactor' like the PWR. Hence, one can see why thermal-neutron reactors have been in wide usage, as they require less fissile material per reactor to achieve criticality.

For a reactor to be stable, the amount of neutrons produced is balanced by an equal amount of neutrons lost. It is known as achieving criticality in the core when the core reactivity is equal to 1. Less than 1 is sub-critical and more than 1 is super-critical

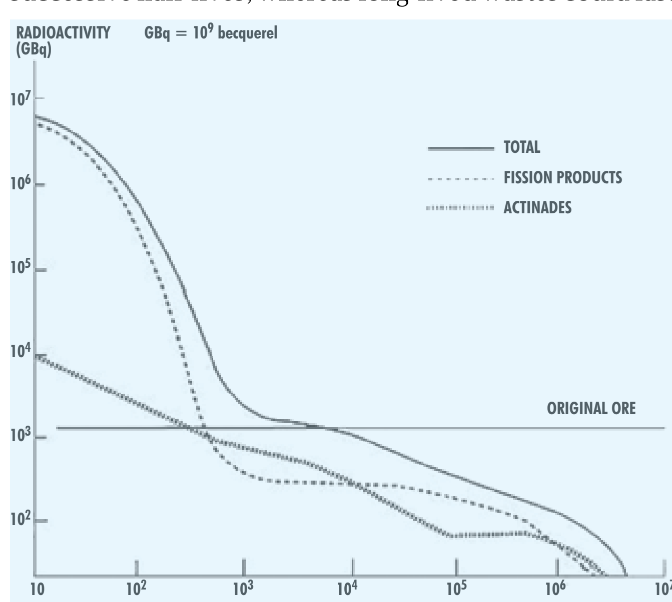
Russian hexagonal PWR fuel bundle.

Burning radioactive waste

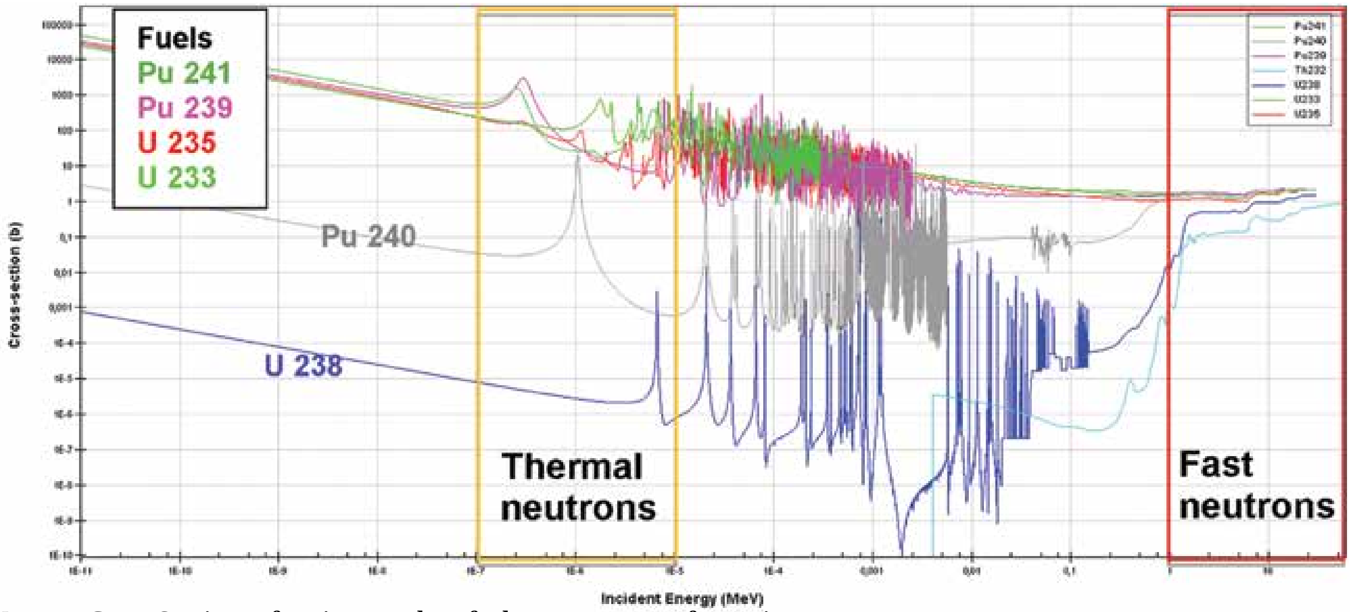
Radioactive waste created in PWRs and BWRs can be loosely separated into two categories: long-lived and short-lived waste. Short-lived waste comprises fission products with a half-life of about 30 years.

Long-lived waste comprises high mass-number elements created from uranium-238 capturing several neutrons and transmutating into elements such as neptunium, plutonium, americium and curium. These trace elements are known as 'minor actinides' as they are actinides created in small quantities.

What is important to note is that short-lived wastes pretty much fully decay after about 300 years or about 10 successive half-lives, whereas long-lived wastes could last



Decay in radioactivity of high-level waste from reprocessing one tonne of spent PWR fuel. The straight line shows the radioactivity of the corresponding amount of uranium ore. Source: *OECD NEA 1996, Radioactive Waste Management in Perspective*.



Neutron Cross Sections of various nuclear fuels over a range of energies.

for 100,000+ years.

But it is the short-lived waste that is the most radioactive as it's decaying at a much faster rate than the long-lived waste. In reality, radioactive waste is not an insurmountable issue as it is possible to engineer containing structures that are very good at shielding radiation and resistant to corrosion.

When spent fuel is reprocessed and the useful uranium and plutonium content is extracted, the remaining fission products are usually immobilised as glass (vitrified) and this is known as high-level waste which is radioactive for 10,000 years.

For unprocessed fuel assemblies held in hardened, shielded casks, the time it takes for the waste to reach a level of radioactivity no more than in uranium ore is about 120,000 years.

Still, there are some who wish for minor actinides to be destroyed and this can be achieved by "burning" them in a fast neutron reactor. In fact, the Russian BN-600 SFR has been burning excess weapons-grade plutonium since 2012 as per their arms-reduction agreement with the USA. Similarly reprocessed actinide waste can be burnt in the form of mixed-oxide (MOX) fuel.

Better burn-up of nuclear fuels

As stated earlier, PWRs and BWRs use uranium dioxide pellet fuels enclosed in thin-walled zircalloy cladding. These long fuel pins are injected with helium gas and sealed to improve heat conduction. Uranium dioxide is a ceramic with a very high melting point (2865°C!) but is relatively low in thermal conductivity at 2.0 – 2.5W/(m.K) between 900 and 2200°C.

In comparison, stainless steel has a conductivity of 15-18W/(m.K) and Zircalloy 21.5W/(m.K). More important to note is the thermal conductivity in uranium dioxide degrades as fission gasses build up, causing cracks to occur.

Naturally, we want thermal conductivity in the fuel to be as high as possible for effective heat transfer, so fuel must be removed from the reactor before the structure of the pellets starts to degrade substantially. Another factor to consider is fission product (FP) build-up which accumulates as the fuel is burnt.

Fission products parasitically absorb neutrons, affecting the core's neutron economy and thus they restrict the fuel's residence time in the core. For these reasons, fuel bundles usually stay in the core for no longer than two years. The maximum burn-up of reactor fuel is measured as the power created divided by the tons of heavy metal 'burnt'.

For uranium dioxide at 5% enrichment, the burn-up tops out at around 60GW-days/ton of heavy metal (where 'heavy metal' (HM) is a mix of uranium, plutonium and minor actinides). Fast neutron reactors which do not suffer as much for the effect of fission-product build up have been shown to achieve a burn up of up to 200GWd/tHM.

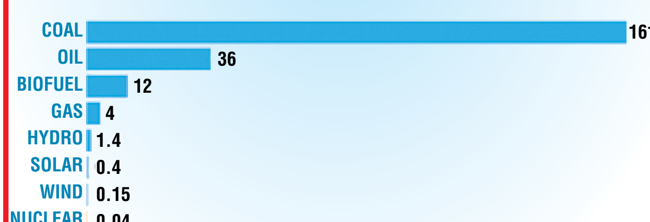
Readers might be surprised to know that PWR-spent fuel

Safety of nuclear power

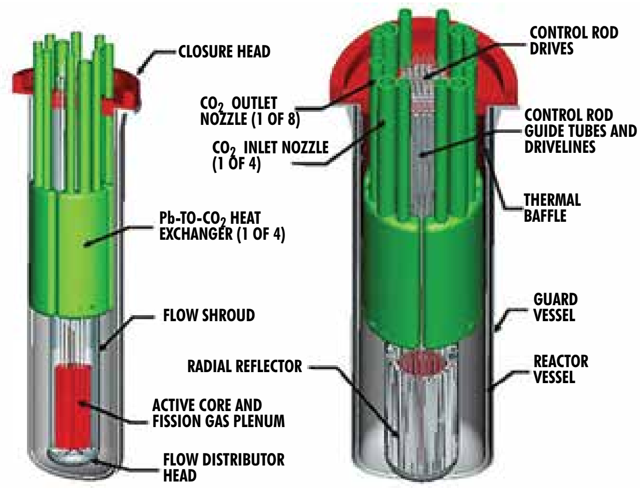
Despite the claims made often in the popular press, nuclear power is **by far** the safest form of energy production, from mining right through to waste disposal.

In three significant nuclear incidents, Three Mile Island, Chernobyl and Fukishima, no one died in the first one, 38 died (four in a helicopter accident) in the second one and nobody died in the last one despite 20,000 people dying in the associated tsunami.

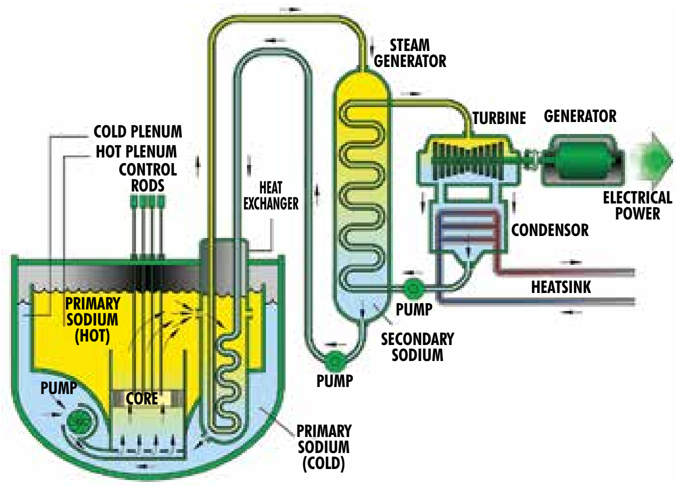
The Chernobyl reactor was a simple and cheap design whose purpose, apart from producing electricity, was to generate as a by-product plutonium for nuclear weapons with no regard to safety. Even so, the area around Chernobyl is now a wildlife paradise with many once-endangered species now thriving.



Deaths per terrawatt-hour of electricity produced



SSTAR reactor concept. It is a compact design that has an electrical output of 20MW and when fuel needs to be changed it is removed as a "cassette" by the reactor supplier and replaced with a fresh one. This design is scalable up to an electrical output of 180MW however development seems to have ceased at the moment. A 100MW version would be around 15 metres high and 3 metres in diameter and weigh 500 tonnes.



The GE Hitachi PRISM (Power Reactor Innovative Small Module) reactor is another type of Sodium-cooled Fast Reactor under development. It is a breeder reactor and closes the fuel cycle. It will be produced as 311MW units that are factory assembled. The UK has analysed some scenarios to burn the country's reprocessed spent-fuel using this reactor which could supply the UK's current electrical demand for the next 500 years.

still contains 95% U-238 which can be reprocessed and reused as Mixed Oxide (MOX) fuel in a PWR or any of the other Gen IV reactors.

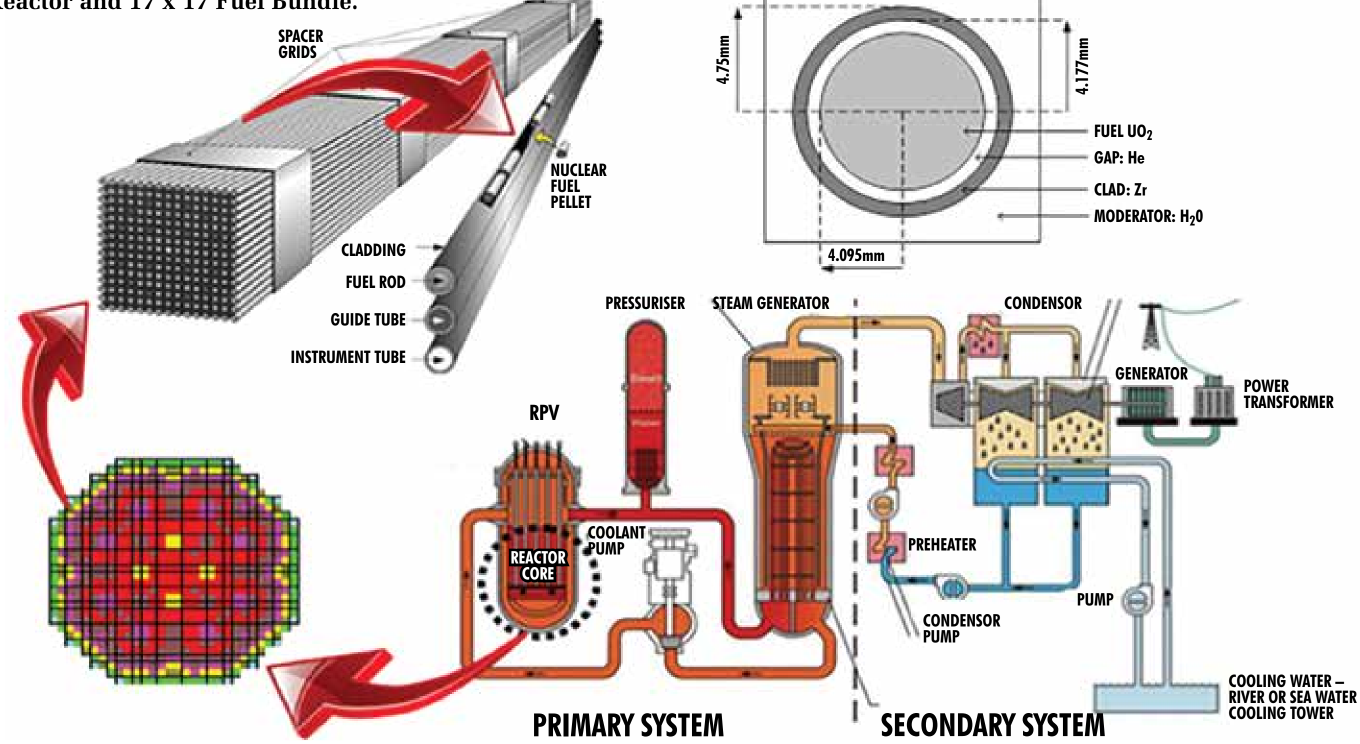
High temperature reactors to supply heat for industrial processes

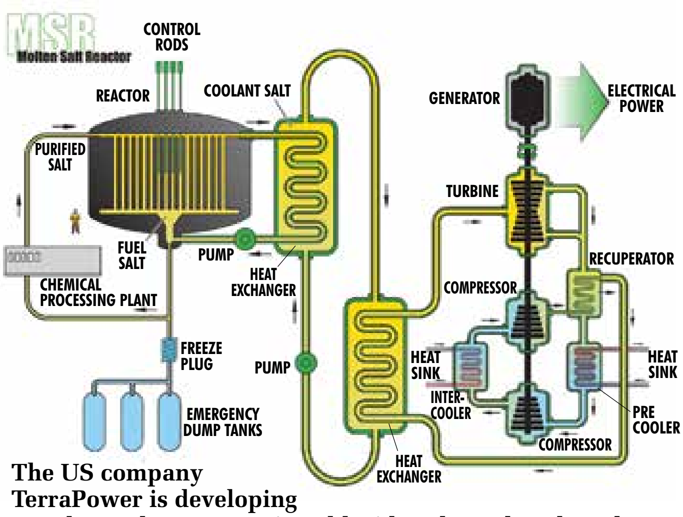
Today's PWRs and BWRs operate at about 300°C which is sufficient to drive a steam turbine at a thermal efficiency of 33% but they are unable to supply the very high temperature heat required for direct-thermal minerals refinement, hydrogen production or synthetic fuel manufacturing.

The limitation for PWRs and BWRs is of water which must remain pressurised to prevent boiling, dry-out and core meltdown. With the exception of the Supercritical Water Reactor, all Gen IV designs circumvent this problem by using more exotic coolants that remain liquid at very high temperatures and without pressurisation.

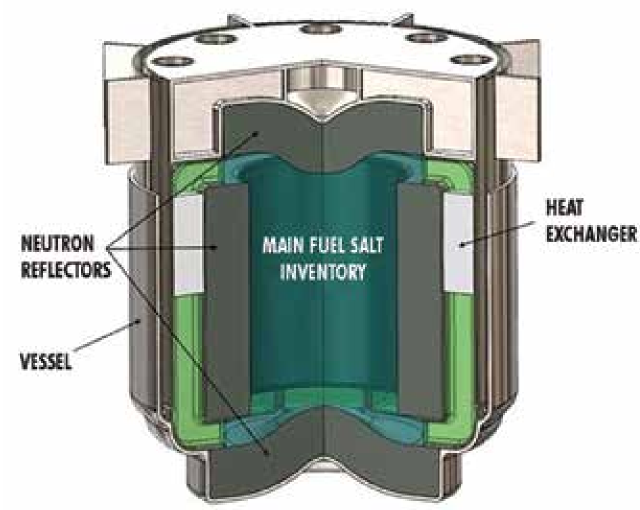
Some of these liquids include sodium (boiling point 892°C), molten salt (bp ~1400°C) and lead (bp 1737°C) which are used in three of the six Gen IV designs. And Very High Temperature Reactors use helium gas instead of a liquid coolant.

Pressurised Water Reactor and 17 x 17 Fuel Bundle.





The US company TerraPower is developing a molten salt reactor using chloride salts rather than the more conventional flouride salts, the Molten Chloride Fast Reactor. It is doing this research alongside its other development project, the Travelling Wave Reactor.



TerraPower's Molten Chloride Fast Reactor.

Economic construction and operation

The Levelised Cost of Electricity (LCOE) is often used to assess the overall cost of a generation system averaged over its lifetime. This takes into account the Capital Cost (build cost), Operating Cost (eg, fuel and maintenance), Grid Connection Cost (eg, grid build-out, stand-by supply) and Financing Cost.

Established nuclear power plants have very low operating costs (as low as 3 US cents/kWh) because the build and financing which currently accounts for 80% of the lifetime costs have usually been paid off.

On the other hand, the LCOE of new nuclear reactors is highly sensitive to the cost of financing (ie, the discount rate usually set at 7%) because nuclear is capital-intensive and much of the investment happens initially during the 5-7 years build phase. Experience in building nuclear reactors also contributes greatly to cost reductions. South Korea has built PWRs continually over the last 30 years and has a LCOE nearly half that of the UK and the United States who are only just restarting their new-build programs.

To counter rising costs, some reactor designers, such as NuScale, are simplifying and miniaturising PWRs in the form of small modular reactors (SMR) that generate 50MWe instead of 1000MWe. (See Silicon Chip, June 2016: "Small Nuclear Reactors" [**siliconchip.com.au/article/9957**]).

The intention is to install then in banks of 12 inside a common pool to provide passive heat removal after shutdown. With a bank of 12 50MWe modules, the plant could produce 600MWe, well suited to replace coal plants, for small grid systems or for remote deployment.

The aims are to reduce the build time to three years, improve costs and quality control by building each reactor in a controlled factory environment (instead of an external environment) and to accumulate experience more quickly by building many reactors on an assembly-line, similar to aircraft manufacturing.

To ensure Gen IV designs remain cost-competitive, it will be important to combine the lessons of continual build, design simplification and modular construction with clever design work that incorporates new materials, fuels and exotic coolants.

Reactor safety

Reactor safety involves four main concerns:

(1) ensuring the reactor has a negative thermal reactivity characteristic so that an increase in core temperature decreases fission activity;

(2) maintaining structural integrity in the fuel, cladding and primary loop containing the coolant that circulates through the core;

(3) avoiding total coolant phase-change (and thus loss of flow) in the core in the event of a reactor power excursion or reactivity spike and

(4) the ability to remove decay heat after shut-down. PWRs have by-and-large demonstrated these characteristics. Only when there is insufficient decay heat removal does the question of boiling, structural integrity and fission product release come into play.

To improve the intrinsic safety of future reactors, three Gen IV designs: the Sodium Fast Reactor, Lead Fast Reactor and Molten Salt Reactor (SFR, LFR, MSR) use unpressurised, high boiling-point liquid coolants that can ensure uninterrupted passive decay heat removal.

Liquid metal coolants such as sodium and lead are also very good conductors of heat, so the task of decay heat removal is easily achieved. The Very High Temperature Reactor and Gas Fast Reactor (VHTR, GFR) circumvent the coolant phase change problem entirely by using helium gas as the coolant.

For high temperature reactors such as SFRs, LFRs, VHTRs and MSRs, passive decay heat removal using air instead of water is achievable because of the large temperature difference between the core and the ambient air temperature.

Now let us take a look at the Gen IV designs, focusing on the sodium fast reactor and molten salt reactor.

Sodium Fast Reactors

The end of the Second World War ushered in the Atomic Age which promised a seemingly inexhaustible energy supply. But there was concern amongst scientists that the world's uranium resources were limited and could be quickly exhausted. Thus, work started on "breeder reactors" which could create more fuel than was burnt.

In the course of testing the neutron cross-section of different materials, it was found that sodium was one of the most neutron-transparent, being six times less neutron absorbing than lead. This made sodium an excellent candidate as a reactor coolant to maximise the reactor core's neutron flux.

More neutrons in the core meant the possibility of using excess neutrons to transmute fertile uranium-238 into plutonium-239 fuel or burning neutron-parasitic actinide-waste. Another feature of sodium is that it is only lightly moderating which means a sodium-cooled reactor could operate in a fast spectrum and directly burn uranium-238, something that thermal-neutron spectrum reactors cannot do.

By calculations, a sodium fast reactor could theoretically attain a breeding ratio of 1.3, meaning that 30% more fuel could be produced than is used. In comparison, a lead fast reactor has a theoretical breeding ratio of 1.0 (making it an "iso-breeder") and a PWR has a conversion ratio of 0.6 (making it a "converter" as noted earlier).

By utilising SFRs, it has been calculated that uranium resources can extend the life of economically recoverable reserves by at least 60 times. Before the Gen IV forum started, there was already much co-operation between the US, Russia, France and the UK on SFRs.

Sharing SFR research in the interest of reactor safety was deemed more important than the possibility of future commercial conflicts of interest. So information on materials neutron cross-section measurements, zero-power critical assembly studies, SRF core layouts optimisation studies and safety analysis research were shared. As a result, the SFR core layouts of most countries ended up being quite similar.

Specific advantages of Generation IV reactors

- Greater fuel efficiency than current Generation III+ reactors with 100 to 300 times more energy output for a given amount of fuel. There will be less useful fuel left over in waste.
- In some reactor designs, existing nuclear waste can be consumed, extending the effective nuclear fuel supply by orders of magnitude. For example, it has been estimated that if the existing nuclear waste of the United States was dug up and used in new reactor designs it could keep the entire US supplied with nuclear electricity for 70 years. This concept also closes the nuclear fuel cycle, meaning the waste is reprocessed as opposed to the "once through" or "open fuel cycle" in which waste is buried rather than reprocessed.
- Waste products that are hazardous for only centuries instead of thousands of years. From current engineering experience we know that structures such as buildings can easily last hundreds of years, even those built with centuries old technology so underground containment structures should pose no problem.
- Many different types of nuclear fuels can be used with different encapsulation methods such as in ceramics or no encapsulation.
- Reactor designs are designed to be intrinsically safe with no external emergency shut down systems or power required in the event of an emergency and (depending on design) low pressure reactor operation. A Fukushima type event where external power failed would not lead to reactor failure.

SFR fuel

SFRs are similar to PWRs in their use of uranium dioxide and plutonium dioxide fuels. In the future uranium nitride, which can carry a higher uranium loading per unit volume and metallic fuels, which have better heat conductivity, could become a possibility.

Plutonium has a larger neutron cross section than uranium for neutrons above 1MeV. Thus, a Fast Neutron Reactors is actually optimised to burn plutonium.

Also, the number of neutrons produced per plutonium-239 fission is 25% more than from uranium-235 and neutrons produced from Pu-239 are more energetic, thus are better at maintaining the fission process. As mentioned earlier, U-238 under neutron bombardment transmutes into Pu-239 and Pu-241 that can be burnt as fuel and some U-238 can be directly burnt by 1MeV neutrons.

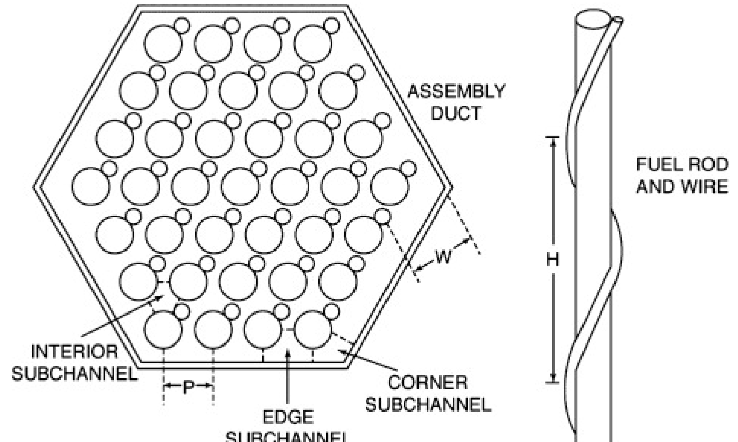
SFR design

A typical SFR fuel bundle is shown opposite. The fuel pins which contain uranium dioxide pellets are packed into a tight hexagonal arrangement to maximise the core's neutron flux. Stainless steel instead of Zircalloy is used for the fuel rods as stainless steel is transparent to fast neutrons, not-corroded by sodium and relatively inexpensive to fabricate.

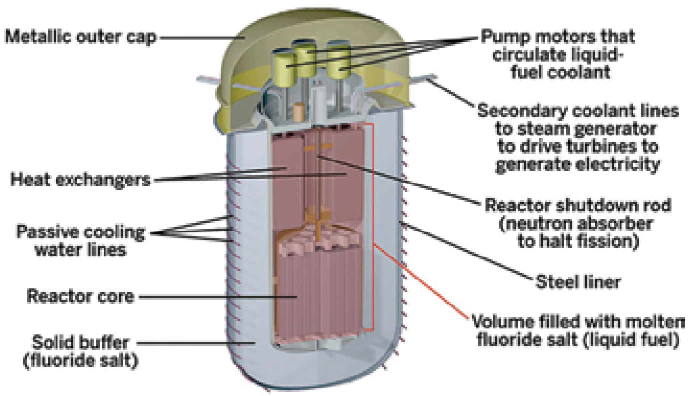
The fuel rod wires that curl around the fuel pin promote flow, mixing and prevent flow dead-spots from forming. Finally the hexagonal fuel bundle is surrounded by a hexagonal shroud to prevent the possibility of large cross flows which would result in fuel bundle vibrations. To maintain a high neutron flux, SFR cores are typically smaller than PWRs (eg, The Dourneay FR 65MWth was the size of a rubbish bin) but because of the smaller neutron cross sections of 1MeV neutrons, the fissile loading of SFRs are typically three times that of PWRs.

A higher core power density necessitates a superior form of coolant which is why liquid metal is used. Passive reactor control is maintained by a strong negative temperature coefficient which for fast reactors is dependent on the Doppler Broadening phenomenon. When nuclear fuel is heated, the resonance energies for capturing neutrons broaden, resulting in neutron absorption instead of fission. (ie, the fuel becomes self-shielding from neutrons).

Since sodium is very reactive to water, most SFRs use an 'integral design' to prevent coolant leakage. In an integral configuration, the core sits in a large pool of liquid sodium with a cover gas – typically argon.



Typical Hexagonal SFR fuel bundle cross section.



Integral Molten Salt Reactor (IMSR).

Having the total primary sodium coolant held inside the thick walled reactor vessel minimises the risk of sodium leakage. For the BN-800 reactor heat removal is accomplished by three independent coolant loops supplying power to a common turbine.

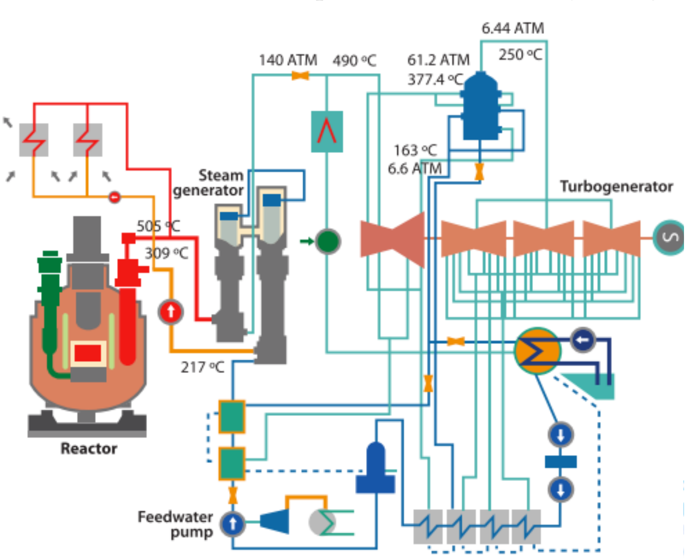
Each loop is comprised of a primary, secondary and tertiary circuit which transfers power to the turbines but also isolates the very radioactive primary sodium coolant from the water-based tertiary coolant. The SFR core operates at a higher temperature than PWRs with an exit temperature of 547°C which allows it to drive a superheated steam cycle at ~40% efficiency.

Future of SFRs

In total, 20 SFRs have operated since the 1950s, accumulating a total of 400+ SFR reactor years of experience. The list of past SFR prototypes includes:

(1) Experimental Breeder Reactors 1 & 2 (USA)
(2) BOR / BN series (Russia)
(3) Phénix and Superphénix (France)
(4) Dounreay FR and PFR (UK)
(5) Monju (Japan) and
(6) CEFR (China).

After a flurry of initial research, most SFR prototypes have permanently shut down after uranium reserves were found to be much more plentiful than initially thought



BN-800 fuel flow diagram. Three consecutive coolant circuits prevent radioactivity from penetrating into the steam generators.

and PWRs & BWRs were optimised to run economically. The exception is in Russia who has operated the BN-600 (600MWe) SFR since the 1980s and have recently commissioned their BN-800 reactor.

There are plans to build an even larger BN-1200 reactor which will further simplify the core design and test new fuels and materials in the quest to close the nuclear fuel cycle (ie, fully consume all radioactive waste generated).

In terms of cost, SFRs are currently more expensive to run than PWRs. It was reported that BN-800 capital costs were 20% more than a Russian VVER-1200 (Russian PWR) and BN-800 operational costs were 15% more than a VVER.

Still, work continues on SFRs in some countries such as France who are planning to build the next generation SFR called "Astrid" and have studied scenarios to replace half of the current PWR fleet with SFRs.

The UK Department of Energy and Climate Change had also studied scenarios of eventually phasing out PWRs with SFRs but has opted to focus on PWRs and BWRs in its new-build program. China, which is currently building most of the world's PWRs, plans to build its own BN-800 reactor with Russian assistance.

In the West, multiple SFR designs are on the drawing board such as the GE Hitachi PRISM reactor and the TerraPower Travelling Wave reactor (TWR). TerraPower recently entered into partnership with China National Nuclear Corporation (CNNC) to further develop the Travelling Wave reactor. The intended purpose of the TWR is to burn spent fuel generated in PWRs using less nuclear fuel and producing less nuclear waste than today's PWRs.

Molten Salt Reactors

Molten salt reactors use fluoride or chloride salts as coolant and can be designed to burn either solid fuels (SF) or liquid fuels (LF).

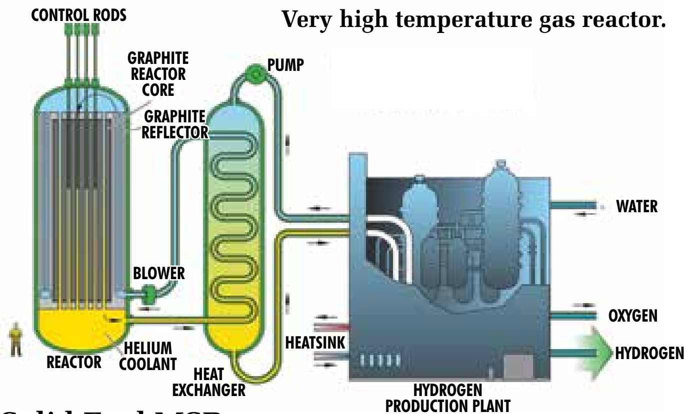
The salt is not dissolved in water; the salt in molten form is the coolant. The choice between a chloride or fluoride salt depends on the desired neutron spectrum. Lithium-beryllium fluoride (FLiBe) works as a thermal spectrum salt on account of the low mass numbers of lithium and beryllium.

Chloride salts paired with heavier elements are much less moderating and good at maintaining a fast neutron-spectrum. All salts have excellent heat transfer characteristics. For example FLiBe salt has the same volumetric heat capacity as water but remains a liquid up to 1400°C without pressurisation.

This is due to the FLiBe salts having a very low vapour pressure (ie, rate of evaporation). Other attractive aspects of the salt include a low neutron absorption cross section, resistance to radiation damage on account of their ionic bonds, being non-reactive to air or water and visually transparent.

MSRs possess a substantial safety margin between the reactor's operational temperature and the salt's much higher boiling point, as boiling could lead to a loss-of-flow accident in the core. Added to this, since pressurisation is not required, the reactor pressure vessel (RPV) can be designed to have a thinner wall compared to the 20cm thickness of a PWR RPV.

Due to the MSR's high core temperature, a Brayton-cycle gas turbine operating at a high thermal efficiency of 45% can be used.



Very high temperature gas reactor.

Solid Fuel MSRs

Current SF-MSR designs are salt-cooled, graphite-moderated reactors that use TRISO (Tri-structural-isotropic) fuel that was developed during earlier research into High Temperature Gas Reactors (HTGRs). TRISO fuel is composed of thousands of 0.5 mm diameter uranium dioxide kernels wrapped in layers of carbon and silicon carbide that trap solid and gaseous fission products without degrading the fuel's thermal conductivity.

A sphere of ten thousand TRISO particles is surrounded by layer of graphite, making a 6cm diameter ball (known as pebble fuel). Alternatively, TRISO fuel can be made into large prismatic blocks of graphite with TRISO particles dispersed on the surfaces that interface with the salt coolant.

TRISO fuel is more accident-tolerant than standard PWR fuel and has been tested to withstand temperatures up to 1800°C without fission product release but the layers of silicon carbide and carbon also make the fuel difficult to reprocess and reuse so this is counter to the goal of closing the fuel cycle.

One may think of the SF-MSR design as being very similar to a HTGR. Both use TRISO pebble fuel and operate in a thermal neutron spectrum but the helium coolant in a HTGR is swapped out for the FLiBe salt. The operation of SF-MSRs is similar to PWRs as both need periodic refuelling but fuel burn-up is enhanced due to TRISO fuel's superior thermal-performance.

One advantage of the SF-MSR is that it is more compact than a HTGR due to the salt's higher volumetric heat capacity. On the other hand, FLiBe coolant is more expensive to manufacture than helium. Currently, the Shanghai Institute of Applied Physics (SINAP), Oak Ridge National Laboratory (ORNL) and Kairos Energy based in California are continuing research on SF-MSR designs.

Liquid Fuel Thermal MSRs

Liquid fuel, molten salt reactors use fuel (233UF4, 235UF4 or 239PuF4) that is directly dissolved into the primary coolant itself. Having the fuel dissolved provides some advantages for thermal-spectrum LF MSRs: 135Xenon – a highly neutron parasitic fission product – can be removed as a gas during operation and refuelling can occur while the reactor is running.

The ability to constantly remove fission products means a much higher rate of burn-up can be achieved (>50%) and also means less decay heat to contend with after the reactor is shut down. The fact that both the fuel and the beryllium moderator are in a liquid form results in them readily expanding at high temperatures, giving the MSR a highly negative reactivity thermal coefficient that prevents a run-away chain reaction.

However, having a fuel in solution also means the primary coolant salt becomes highly radioactive, complicating maintenance procedures and the chemistry of the salt must be monitored closely to minimise corrosion. Another advantage of the liquid fuel molten salt design is that it allows the breeding of 233U from 232Th in the thermal/epithermal neutron spectrum instead of using a fast-spectrum.

Neutron capture by thorium-232 results in beta decay (one of the neutrons in the thorium nucleus expels an electron to become a proton) thus transmutating into rotactinium-233 which further beta decays into uranium-233. The U-233 could then be used as an MSR fuel. The thorium fuel cycle holds promise and studies have shown that a breeding ratio of 1.06 to 1.14 is possible for thermal and epithermal spectrum MSRs.

Despite the potential for breeding fuel, current efforts are focused on simply bringing the LF-MSR to the commercial market – one which satisfies the nuclear regulator's stringent demands for safety. Various LF-MSR start-up companies are approaching the problem from different angles.

Terrestrial Energy's (Canada) "Integral Molten Salt Reactor" (IMSR) uses low enriched (5%) uranium (ie, denatured uranium) dissolved in the salt coolant. The reactor vessel is designed to be swapped out every seven years to address possible issues with salt corrosion.

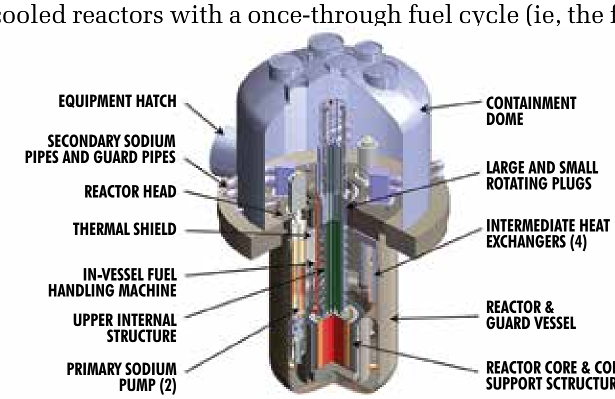
Another company, ThorCon, has a similar design, using a FLiBe salt and graphite moderator but fitted on a ship. Transatomic has a design using lithium-fluoride salt instead of FLiBe and zirconium hydroxide instead of graphite as the moderator with a view to burn radioactive waste.

The Shanghai Institute of Applied Physics is also pursuing a LF-MSR design and has worked with Oak Ridge National Labs and with ANSTO on corrosion resistant materials development. SINAP has secured $3.3 billion USD to build a 10MWth thermal-spectrum LF-MSR prototype by 2020.

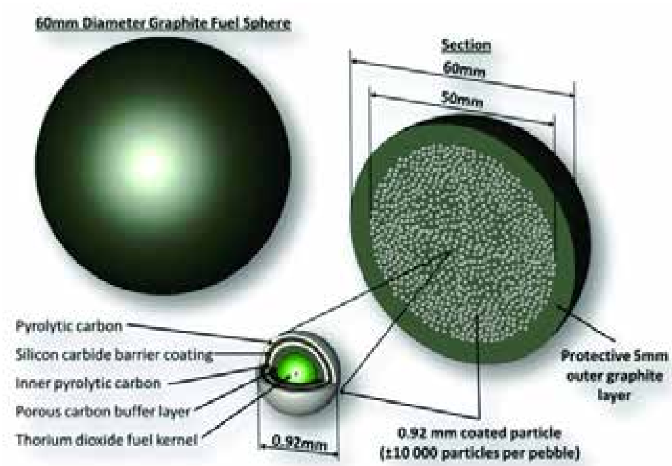
Fast spectrum, chloride-salt designs are being pursued by the European SAMOFAR (Safety Assessment of the Molten Salt Fast Reactor) consortium, Elysium Inc. (USA) and Terrapower's MCFR (Molten Chloride Fast Reactor) which aims to burn the 700,000 tonnes of uranium held in spent fuel from PWR and BWR operations in the USA.

VHTRs, GFRs, SCWRs & LFRs

Very High Temperature Reactors (VHTR), like their predecessor the HTGRs, are graphite-moderated, helium-cooled reactors with a once-through fuel cycle (ie, the fuel



The travelling wave reactor (TerraPower).



Fuel pellets for Terrapower Molten Chloride Fast Reactor

is not reprocessed) using TRISO fuel. VHTRs have a target operational temperature of 900°C whereas HTGRs' core outlet temperature is about 700°C.

The higher temperature of 900°C would enable hydrogen production or the delivery of heat for industrial processes. Difficulties in realising a VHTR design are mainly due to the limitations of material performance as the rate of material corrosion increases linearly with temperature. Thus materials research is continuing to enable the VHTR concept.

The USA, Russia, South Africa, Japan and the UK have all built experimental HTGRs. China is close to completing two HTR-PM (High-Temperature Reactor – Pebble Module) prototypes which will deliver superheated steam to a common turbine generating 210MWe.

Limiting the thermal output of each HTR-PM unit to below 300MWth ensures the maximum fuel temperature limit of 1600°C will not be compromised after reactor shutdown, thus guaranteeing the reactor's inherent safety. It is envisaged new HTR-PM units will replace current coal plants which drive the same superheated steam cycle and so quickly reduce China's pollution problems.

Gas Fast Reactors (GFRs) can be thought of as an extension of VHTR technology but with a higher fissile loading (on account of the fast spectrum) and without the presence of moderating graphite. It is a challenging design as the removal of the graphite severely reduces the core's thermal inertia (ie, the ability of the core material to 'suck up' the decay heat).

Progress on this design has been slow and depends on the outcome of VHTR research.

The Supercritical Water Reactor (SCWR) could be thought of as a Boiling Water Reactor with the primary loop directly driving a steam turbine.

The water coolant is heated beyond 375°C and 22.1MPa in a super-critical state whereby the total liquid inventory behaves like steam and the transitional dynamics of boiling can be avoided.

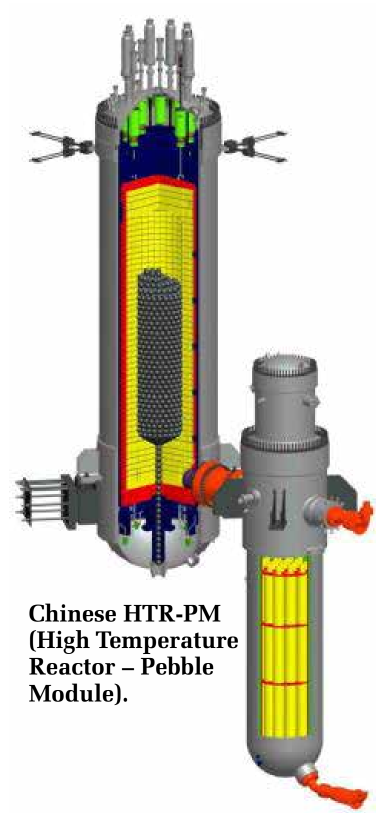
This design is focused mainly on improving the efficiency of the thermal cycle but faces the challenges of increased thermal stress on reactor components, accelerated corrosion rates at elevated temperatures and a reduced water inventory in the primary loop which normally serves as a buffer for sharp changes in reactor power.

(For more on super-critical steam see SILICON CHIP, December 2015 – siliconchip.com.au/article/9634).

The last Gen IV reactor design is the Lead Fast Reactor. Both the US and Russia have studied reactor concepts using a lead coolant but only Russia has fielded the LFR in its naval vessel, most notably in the Alfa-class attack-submarine (see below).

Since fast reactors operate with a compact core to maximise the neutron flux, the lead-bismuth cooled OK-550 fast reactor with an output of 60MWe could fit inside a small cross-sectional hull and propel the submarine up to 41 knots (76km/h!)

These submarines have all been decommissioned but plans for the new BREST-300 LFR was recently granted approval for construction in Seversk, Russia and will serve as the demonstrator unit before the larger BREST-1200 unit is built.



Chinese HTR-PM (High Temperature Reactor – Pebble Module).

This concludes our brief run-down of all the six Gen IV designs. All nuclear power reactors must extract large amounts of power from the small volume of the core. This necessitates both the fuel and the coolant to be in close contact with one another to maximise reactor heat transfer.

Added to this, neutrons must be used sparingly via the careful selection of component material so that excess neutrons can be used to either bred fuel or burn radioactive waste.

In the end, the whole reactor system must be contained by a durable and inexpensive material, resistant to corrosion and radiation damage. *SC*

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The Russian Alfa-class submarine which used a lead-bismuth fast reactor. It could run at up to 41 knots.