PDHonline Course E339 (3 PDH)

Nuclear Power Volume III - The Future of Nuclear Power

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2020

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Nuclear Power Volume III
The Future of Nuclear Power

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Preface

This is the third in a series of three courses about the nuclear power industry.

The series covers the nuclear industry from the physics of nuclear reactions to the types of plants in operation today as well as the potential of the next generation of nuclear power plants that are likely to appear in the first half of the 21st century.

The complete series includes three courses:

1. Volume I – The Nuclear Power Industry
2. Volume II – Nuclear Power Plants
3. Volume III – The Future of Nuclear Power

The first course, *Volume I – The Nuclear Power Industry*, gives a broad overview of the nuclear power industry. This course goes into the details of nuclear reactions and the physics of nuclear power. The prime fuel source, uranium, is covered too.

The second course, *Volume II – Nuclear Power Plants*, reviews the classifications of nuclear power plants and the basic components of a nuclear power plant. The course covers the design and operation of the current generation of nuclear power plants in operation today.

The third course, *Volume III – The Future of Nuclear Power*, gives an overview of the types of plants that are being considered for the next generation of power plants. Some of the designs covered are already operating in experimental stages, some are modifications of current designs, and others are radical new concepts that have not been commercially validated.

It is not necessary to take the courses in sequence. However, for the best comprehensive it is suggested that the courses be taken in the order presented.

Introduction

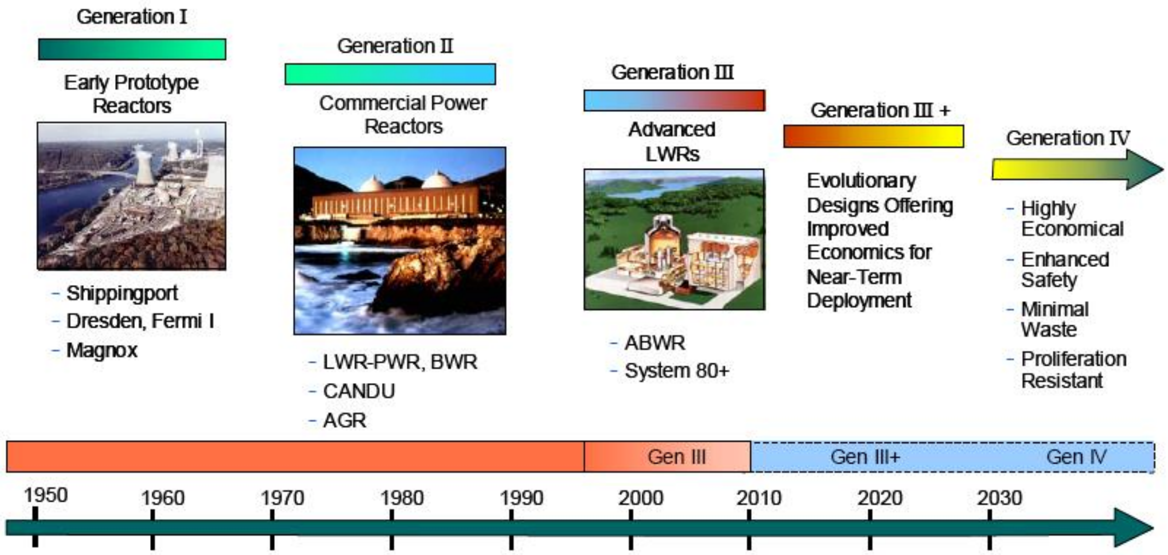
The world's population is expected to expand from about 6 billion people to 10 billion people by the year 2050, all striving for a better quality of life. As the Earth's population grows, so will the demand for energy and the benefits that it brings: improved standards of living, better health and longer life expectancy, improved literacy and opportunity, and many others.

Many of the world's nations, both industrialized and developing, believe that a greater use of nuclear energy will be required if energy security is to be achieved. They are confident that nuclear energy can be used now and in the future to meet their growing demand for energy safely and economically, with certainty of long term supply and without adverse environmental impacts.

Approximately 16% of world electric demand is served from 449 nuclear power plants with a capacity of 391 GW. The United States produces the most nuclear energy, with nuclear power providing 19% of the electricity it consumes, while France produces the highest percentage of its electrical energy from nuclear reactors—72% as of 2017. It is quite possible to utilize nuclear power to provide the vast majority of an entire country's need for electricity.

This course is the third in a series about the use of nuclear energy to generate electricity. Volume I in this series included an overview of the nuclear industry and covered the basics of nuclear physics and uranium as a fuel source and Volume II delved into the specific types of nuclear reactors in use around the world today.

The current generation of commercial nuclear power plants are known as Generation II reactors (the earliest experimental reactors are classified as Generation I reactors.) Generation III plants are, for the most part, improved versions of Generation II plants and are near term ready for commercialization. See the timeline below.



Generation IV nuclear reactors will be revolutionary new designs that the industry hopes will be very economical and efficient. These reactors will likely not appear until at least 2020.

In this course, we will start with a brief overview of the current crop of Gen-III plants that are likely to be seen in commercial operation in the next few years. Subsequent chapters will cover several of the promising Gen-IV reactors. There are currently six Gen-IV reactors that hold significant promise. They are:

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

Let's look at Chapter One for an overview of the Gen-III reactors.

Chapter 1
Generation III Reactors

This chapter covers a few of the evolutionary designs that are basically improvements on existing commercially operating nuclear reactors. These units may be in operation in the early 2020's. The units covered in the chapter include the Westinghouse AP1000, The EPR, and Hyperion Power Module. There are other Generation III reactors that are not included in this discussion.

AP1000

Westinghouse Electric Company's AP1000 reactor design is the first Generation III reactor to receive final design approval from the Nuclear Regulatory Commission (NRC.) It is an evolutionary improvement on the currently operating AP600. It is essentially a more powerful model with roughly the same land use. The AP1000 is a two-loop PWR planned to produce a net 1,154 MW.

Reaction: Fission

Moderator: Water

Coolant: Water

The design is less expensive to build than other Gen-III plants partly because it uses existing technology. The design also decreases the number of components, including pipes, wires, and valves. Standardization and type-licensing should also help reduce the time and cost of construction. Because of its simplified design compared to current generation PWR's, the AP1000 has:

- 50% fewer safety-related valves
- 35% fewer pumps
- 80% less safety related piping
- 85% less control cable
- 45% less building volume

Like existing PWR's, the refueling cycle is 18 - 24 months.

The Nuclear Regulatory Commission approved the final design certification for the AP1000. This means that prospective builders can apply for a Combined Construction and Operating License (COL) before construction starts, whose validity is conditional upon the plant being built as designed.

In this design Westinghouse's Passive Core Cooling System (PCCS) uses less than twenty explosively operated valves which operate within the first 30 minutes of an incident, even if the reactor operators take no action. The electrical system required for initiating the passive systems doesn't rely on external or diesel power and the valves don't rely on hydraulic or compressed air systems. If the active process to turn on the passive system works, the design is intended to passively remove heat for 72 hours, after which the PCS gravity drain water tank must be topped up for as long as cooling is required.

The first commercially operated AP1000 reactors will likely be in China. China has officially adopted the AP1000 as a standard for inland nuclear projects. It plans to have the first units in operation in the early 2020's.

Evolutionary Power Reactor

Reaction: Fission

Moderator: Water

Coolant: Water

The Evolutionary Power Reactor (EPR) is another one of the Gen-III pressurized water reactor (PWR) designs. This reactor design was initially called the European Pressurized Reactor.

Four EPR units are under construction. There is one each in Finland and France and two in China.

The main design objectives of the EPR design are increased safety while providing enhanced economic competitiveness through improvements to previous PWR designs scaled up to an electrical power output of 1,650 MW. The reactor can use 5% enriched uranium oxide fuel, optionally with up to 50% mixed uranium plutonium oxide fuel. The EPR is the evolutionary descendant of the French N4 and KONVOI reactors.

The EPR design has several active and passive protection measures against accidents. There are four independent emergency cooling systems, each capable of cooling down the reactor after shutdown. Leak tight containment is provided around the reactor. And there is an extra container and cooling area if a molten core manages to escape the reactor. In addition, there is a two-layer concrete wall, designed to withstand impact by airplanes and internal overpressure.

It has been stated that the EPR is the only new reactor design under consideration in the United States that "...appears to have the potential to be significantly safer and more secure against attack than today's reactors."

Hyperion Power Module

Reaction: Fission

Moderator: Water

Coolant: Water

Hyperion Power Module (HPM) is a relatively small 25 MW nuclear reactor, which will be modular, inexpensive, inherently safe, and proliferation-resistant.

The Hyperion Power Module is being developed by Gen4 Energy, formerly Hyperion Power Generation company.

The uranium nitride fuel incorporated in the design is generally similar in physical characteristics and neutronics to the standard ceramic uranium oxide fuel that is used at present in modern light water nuclear reactors. However, it has certain beneficial traits - higher thermal conductivity - and thus less retained heat energy - that make it preferable over oxide fuels when used at temperatures greater than the 300C temperatures that are found in light water reactors. By operating at higher temperatures, steam plants can operate at a higher thermal efficiency.

The Hyperion module has sufficient fuel for 3,650 full power days at 25MW, is capable of load following, and is meant to be built in pairs; one module can be at power, while another can be under installation or removal at the same time, ensuring reliable supply of electricity.

The HPM uses natural circulation of the lead-bismuth coolant through the reactor module as a means of primary cooling. Coolant temperatures within the primary loop should be approximately 500C. Powered intermediate heat exchangers, also using lead-bismuth coolant, are located within the reactor and run an intermediate loop going to a third heat exchanger (the steam generator), where heat is transferred to the working fluid, heating it to approximately 480C. Two schemes of power generation exist at this point: either using superheated steam or supercritical carbon dioxide to drive Rankine cycle or gas turbines.

The thermal hydraulics of the lead-bismuth reactor is dictated by the high heat capacity and unique properties of the lead-bismuth eutectic coolant. This coolant has several extremely beneficial properties for a reactor: it is opaque to gamma radiation, but transparent to neutron flux; it melts easily at a low temperature, but does not boil until an extremely high temperature is reached; it does not greatly expand or contract when exposed to heat or cold; it has a high heat capacity; it will naturally circulate through the reactor core without pumps being required - whether during normal operation or as a means of residual decay heat removal; and it will solidify once decay heat from a used reactor has dropped to a low level.

> A ***eutectic system*** is a mixture of chemical compounds or elements that has a single chemical composition that solidifies at a lower temperature than any other composition.

Four mechanisms of control are used in the reactor. There are two types of control rods - rapid shutdown rods, designed to promptly absorb a large quantity of reactivity from the reactor to bring it below the shutdown margin, and fine-grained working control rods, also known as shims, which are used to compensate for the long-term decrease in reactivity that comes from the nuclear fuel being depleted and fission products being formed. There is a secondary shutdown system consisting of neutron-absorbing boron carbide balls that can be launched into the core in the event the shutdown rods are not responsive and rapid shutdown is called for. Fourth, there is the prompt negative temperature coefficient of reactivity, which prevents the reactor from remaining critical if it should enter into an unsafe temperature range. The reactor is designed so that once shut down, it does not require external assistance aside from natural conduction and convection to surrounding natural media to remove residual heat, qualifying it as highly safe.

The reactor weighs 22 tons fully fueled (including coolant), and it can be transported by truck or by rail to its destination. Radiation protection during transport is integral, making it nearly impossible for any transport accident to threaten the release of radiation. As the coolant is composed of lead (a strong absorber of gamma radiation), the reactor is very safe for humans to be in close proximity to while the reactor is transported; further, if the reactor is allowed sufficient time to eliminate decay heat prior to transport, the lead-bismuth coolant will be in solid phase, thus fixing the internals of the reactor in place, causing the reactor to behave as a single piece of metal if subjected to external shock.

Chapter 2
Very-High-Temperature Reactor

The Very High Temperature Reactor (VHTR) is a reactor concept that is helium cooled, uses graphite for moderation, and has a once-through uranium fuel cycle. The core can be built from either prismatic blocks or may as a pebble bed reactor. Outlet temperatures of over 900C enable thermo-chemical hydrogen production via an intermediate heat exchanger, with electricity cogeneration, or direct high-efficiency driving of a gas turbine (Brayton cycle). There is some flexibility in fuels, but no recycling is planned for initially. Modules of 600 MW thermal are planned. The VHTR has potential for high burn-up, completely passive safety, low operation and maintenance costs, and modular construction. See Figure 1 below.

Reaction: Thermal

Moderator: Graphite

Coolant: Helium

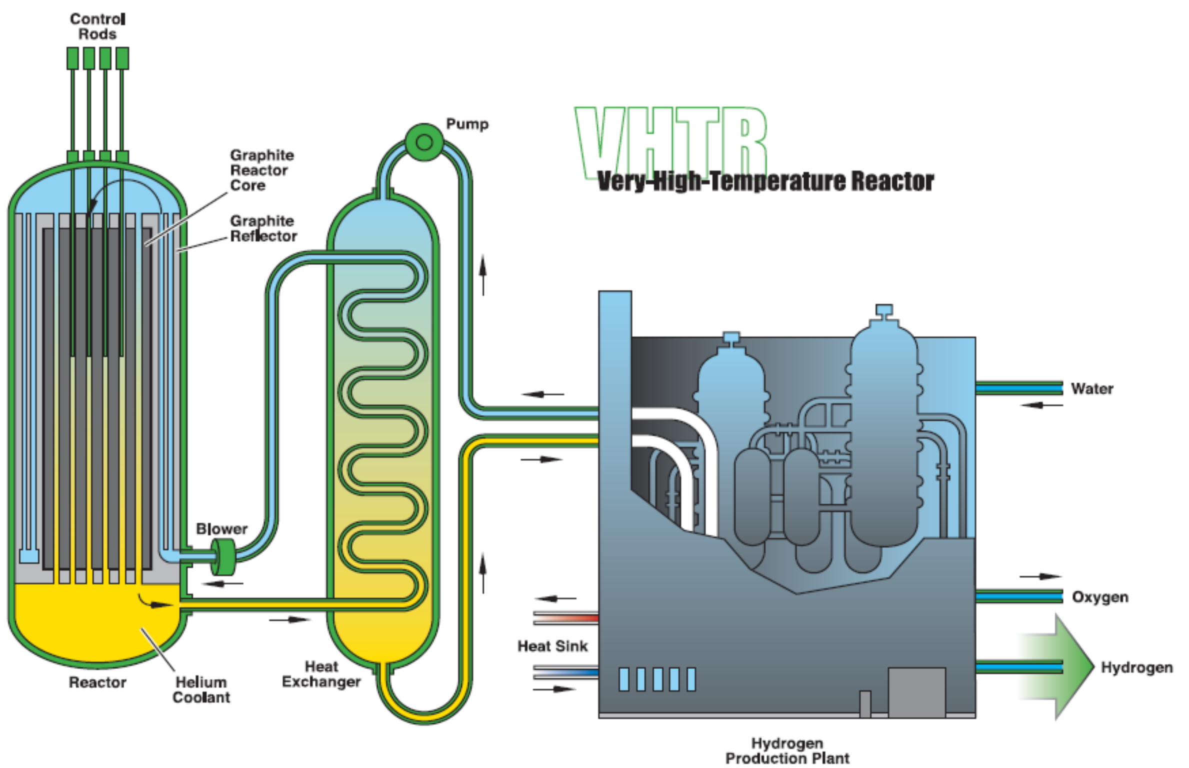


Figure 1

Courtesy: Department of Energy

The earlier version of this design was known as a high temperature gas-cooled reactor or the HTGR, and to an extent, the general type of this reactor is still known by that name; the very high temperature reactor representing a modern and highly evolved version of the original HTGR design.

Some designs refer to a prismatic block core configuration, where hexagonal graphite blocks are stacked to fit in a spherical pressure vessel. Pebble bed designs are also being studied and have been used at lower temperatures than those envisioned for the VHTR.

TRISO (Tri-structural-isotropic) fuel is a type of micro fuel particle. It consists of a fuel kernel composed of UOx in the center, coated with layers of carbon and silicon.

The fuel will most likely be uranium dioxide in a TRISO configuration. A combination of uranium dioxide and uranium carbide may also be used to reduce internal pressure in the TRISO particles caused by the formation of carbon monoxide, due to the oxidization of the porous carbon layer in the particle. The TRISO particles are either dispersed in a pebble for the pebble bed design or molded into compacts/rods that are then inserted into the hexagonal graphite blocks.

A pebble bed helium cooled reactor type is the dominant one being studied; its primary design uses a 600-MW thermal core with a helium outlet temperature of 1,000C. Helium has been used in most high temperature gas reactors (HTGR) to date. Helium is an inert gas, so it will not chemically react with any material. Additionally, exposing helium to neutron radiation does not make it radioactive, unlike most other possible coolants.

The molten salt cooled variant, the LS-VHTR uses a liquid fluoride salt for cooling in a pebble core. The pebble fuel floats in the salt, and thus pebbles are injected into the coolant flow to be carried to the bottom of the pebble bed, and are removed from the top of the bed for recirculation. The LS-VHTR has many attractive features, including: the ability to work at high temperatures low pressure operation, high power density, better electric conversion efficiency than a helium-cooled VHTR operating at similar conditions, passive safety systems, and better retention of fission products in case an accident occurred.

In the prismatic designs, control rods are inserted in holes cut in the graphite blocks that make up the core. The LS-VHTR is controlled by inserting the control rods into the surrounding graphite reflector. Control can also be attained by adding pebbles containing neutron absorbers.

The VHTR can produce hydrogen from only heat and water by using thermo-chemical iodine-sulfur process or from heat, water, and natural gas by applying the steam reformer technology to core outlet temperatures greater than about 900C. The VHTR can generate electricity with high efficiency, over 50% at 900C. Co-generation of heat and power makes the VHTR an attractive heat source for large industrial complexes. The VHTR can be deployed in refineries and petrochemical industries to substitute large amounts of process heat at different temperatures, including hydrogen generation for upgrading heavy and sour crude oil.

For electricity generation, the helium gas turbine system can be directly set in the primary coolant loop, which is called a *direct cycle*. For nuclear heat applications such as process heat for refineries, petro-chemistry, metallurgy, and hydrogen production, the heat application process is generally coupled with the reactor through an intermediate heat exchanger (IHX), which is called an *indirect cycle*.

Research is required to adapt the chemical process and the nuclear heat source to each other with regard to temperatures, power levels, and operational pressures. Qualification of high temperature alloys and coatings for resistance to corrosive gases like hydrogen, carbon monoxide, and methane will be needed.

Performance issues for the VHTR include development of a high-performance helium turbine for efficient generation of electricity. Modularization of the reactor and heat utilization systems is another challenge for commercial deployment of the VHTR.

The increase of the helium core-outlet temperature of the VHTR results in an increase of the fuel temperature and reduced margins in case of core heat up accidents. Fuel particles coated with silicon-carbide are used in HTGRs at fuel temperatures of about 1,200C. Irradiation testing will be needed to demonstrate that TRISO-coated particles can perform acceptably at the high burnup and temperature associated with the VHTR. Following irradiation, high temperature heating tests are needed to determine that there is no degradation in fuel performance under accident heat up conditions up to 1,600C as a result of the more demanding irradiation service conditions. These fuel demonstration activities will require about 5 to 7 years to complete following fabrication of samples. Complete fuel qualification will require an additional 5 to 7 years in which statistically significant production scale fuel is irradiated to confirm the performance of the fuel from the production facility.

Above a fuel temperature of 1,200C, new coating materials such as zirconium-carbide (ZrC) and/or improved coating techniques will need to be considered. Use of ZrC enables an increase in power density and an increase in power size under the same coolant outlet temperature and allows for greater resistance against chemical attack by the fission product palladium. The limited fabrication and performance data on ZrC indicates that although it is more difficult to fabricate, it could allow for substantially increased operating and safety envelopes (possibly approaching 1,800C).

To realize the goal of core outlet temperatures higher than 1,000C, new metallic alloys for reactor pressure vessels have to be developed. At these core-outlet temperatures, the reactor pressure vessel temperature will exceed 450C. LWR pressure vessels were developed for 300C service, and the HTTR vessel for 400C. Hasteloy-XR metallic materials are used for intermediate heat exchanger and high temperature gas ducts in the HTTR at core-outlet temperatures up to about 950C, but further development of super-alloys will be required for the VHTR. The irradiation behavior of these super alloys at the service conditions expected in the VHTR will need to be characterized. Such work is expected to take 8 to 12 years.

The VHTR assumes a once-through fuel cycle. Like LWR spent fuel, VHTR spent fuel could be disposed of in a geologic repository or conditioned for optimum waste disposal. The current HTGR particle fuel coatings form an encapsulation for the spent fuel fission products that is extremely resistant to leaching in a final repository. However, as removed from the reactor, the fuel includes large quantities of graphite, and research is required to define the optimum packaging form of spent VHTR fuels for long-term disposal. Radiation damage will require graphite replacement every 4 to 10 years.

Commercial production of VHTR's is not likely until well after 2020.

Chapter 3
Supercritical-Water-Cooled Reactor

Reaction: Thermal

Moderator: Water

Coolant: Water

The Supercritical water reactor (SCWR) is a reactor concept that uses supercritical water as the working fluid. SCWRs are high-temperature, high-pressure water-cooled reactors that operate above the thermodynamic critical point of water. These systems may have a thermal or fast-neutron spectrum, depending on the core design. SCWRs are basically Light Water Reactors (LWRs) operating at higher pressure and temperatures with a direct, once-through cycle. It will operate on a direct cycle, much like a BWR, but since it uses supercritical water as the working fluid, will have only one phase present. It will operate at much higher temperatures and pressure than both current PWRs and BWRs.

Supercritical water-cooled reactors (SCWRs) are promising advanced nuclear systems because of their high thermal efficiency (i.e., about 45% vs. about 33% efficiency for current light water reactors) and considerable plant simplification.

The SCWR is built upon two proven technologies, LWRs, which are the most commonly deployed power generating reactors in the world, and supercritical fossil fuel fired boilers, a large number of which are also in use around the world.

The SCWR uses water as a neutron moderator. Moderation comes primarily from the high density supercritical water. This high-density water is either introduced from cooling tubes inserted into the core or as a reflector or moderated-part of the core.

> ***Actinides*** are the 14 chemical elements with atomic numbers from 90 to 103, thorium to lawrencium. Actinides are radioactive and release energy upon radioactive decay; Actinides, uranium and plutonium, are used in nuclear reactors.

The neutron spectrum will be only partly moderated, perhaps to the point that the SCWR will technically become a fast neutron reactor. There are three main advantages for having a fast neutron spectrum. First, fast neutron reactors have a higher power density and can therefore generate more power for the same size of reactor. Second, the fast neutrons are able to split the long lived actinides, destroying the most long lived nuclear waste through nuclear transmutation. Third, since fission events induced by fast neutrons produce more neutrons per fission event, it becomes possible to design a breeder reactor, which could extract roughly 100 times the energy from the same quantity of uranium as could a traditional reactor design.

See Figure 2 on the next page for a diagram of a proposed SCWR design.

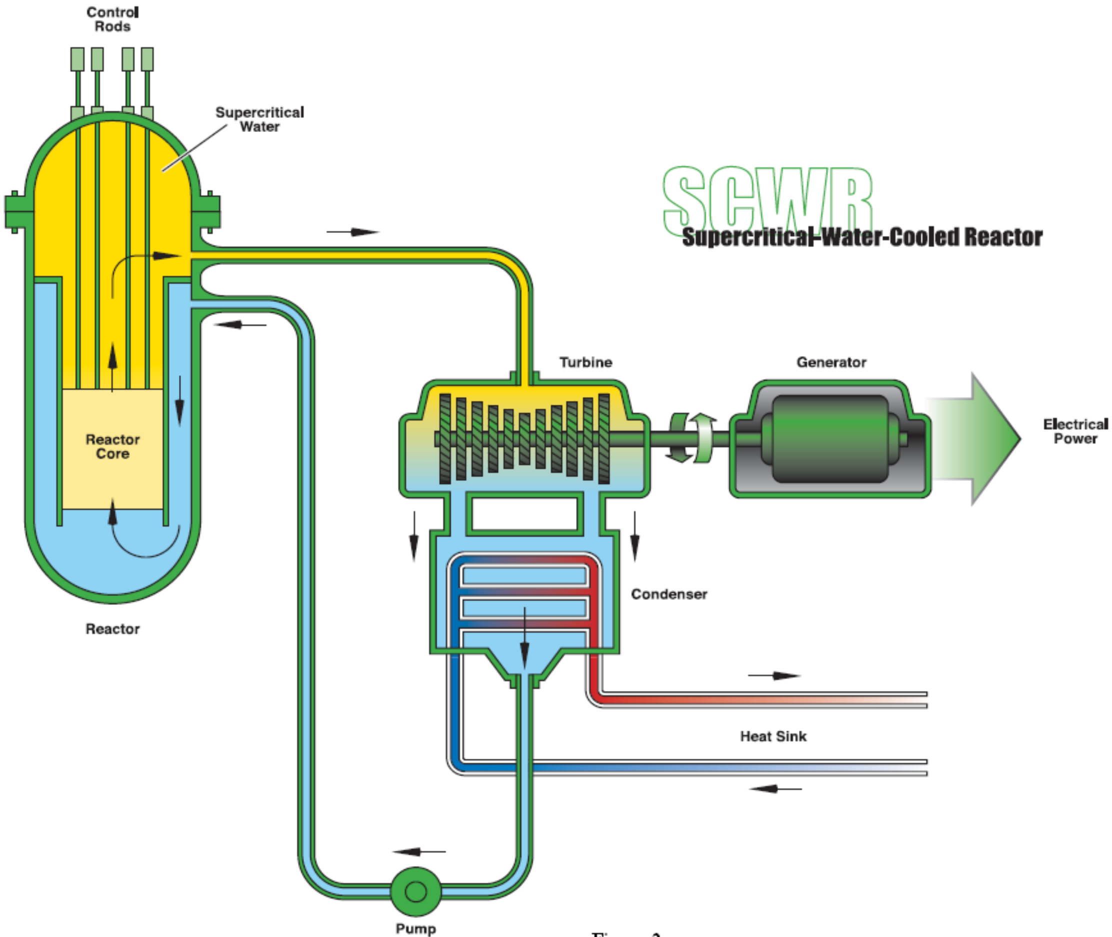


Figure 2

Courtesy: Department of Energy

The proposed fuel will resemble traditional LWR fuel. However, it is likely the SCWR will use channelized fuel assemblies like the BWR in order to reduce the risk of hotspots caused by local variations in core thermal hydraulic properties. Because the SCWR will operate under conditions exceeding current experience with LWRs and LMFBRs, new criteria for core materials (especially for the fuel cladding) must be developed to ensure safe operation to maintain fuel rod integrity during abnormal transients, normal power operation as well as concerns for release of the fission products caused by oxidation corrosion of the cladding. There are four failure modes considered for the fuel rod's integrity during abnormal transient conditions: mechanical failure, buckling collapse, overpressure damage and creep failure. It is expected that hydrogen will need to be injected into the coolant to reduce oxidation corrosion of the sheath.

The coolant will be supercritical water. Operating above the critical pressure ensures the coolant remains single-phase in the core. At a lower pressure it would boil, producing voids (bubbles) with less density and therefore less moderating effect, making the reactor power output hard to predict and control. At extreme pressure, above the critical point, steam and liquid become the same density, and are indistinguishable. The hope is that more of the heat produced from fission can be converted into electricity in reactors cooled and/or moderated with supercritical water. Additionally, the elements handling water's phase change from liquid to gas in conventional light water reactors are not needed. This simplification should reduce construction costs and improve reliability and safety. Current LWRs need recirculation and jet pumps, pressurizers, steam generators, and steam separators and dryers, all or most of which would not be required.

SCWRs will likely have control rods inserted through the top, as is done in PWRs.

Because supercritical water has different chemical properties from liquid water, and because it requires a higher operating temperature, materials developed for more traditional reactors may not be suitable for an SCWR. While there is a lot of experience with supercritical water in fossil fuel plants, the intense neutron radiation present in a reactor means that the material requirements in an SCWR differ from a fossil fuel plant.

The SCWR can also be designed to operate as a fast reactor. The difference between thermal and fast versions is primarily the amount of moderator material in the SCWR core. The fast spectrum reactors use no additional moderator material, while the thermal spectrum reactors need additional moderator material in the core.

Much of the technology base for the SCWR can be found in the existing LWRs and in commercial supercritical-water-cooled fossil-fired power plants. However, there are some relatively immature areas. There have been no prototypes SCWRs built and tested. For the reactor primary system, there has been very little research done on potential SCWR materials or designs.

The supercritical water (SCW) environment is unique and few data exist on the behavior of materials in SCW under irradiation and in the temperature and pressure ranges of interest. At present, no candidate alloy has been confirmed for use as either the cladding or structural material in thermal or fast spectrum SCWRs. Potential candidates include stainless steels, solid solution and precipitation-hardened alloys, ferritic martensitic alloys, and oxide dispersion-strengthened alloys.

The fast SCWR design will result in greater doses to cladding and structural materials than in the thermal design by a factor of five or more. These doses will result in greater demands on the structural materials in terms of the need for irradiation stability and effects of irradiation on embrittlement, creep, and corrosion. The generation of helium by transmutation of nickel is also an important consideration in both the thermal and fast designs because it can lead to swelling and embrittlement at high temperatures. The data obtained during prior fast reactor development will play an important role in this area. To meet these challenges, research is needed for the cladding and structural materials in the SCWRs that focuses on acquiring data and understanding the following key property needs: corrosion and SCC, radiolysis and water chemistry, dimensional and micro-structural stability, and strength and creep resistance.

Here are a few of the advantages and disadvantages of an SCWR.

Advantages

- The higher temperatures and use of a supercritical brayton cycle improves thermodynamic efficiency, estimated to be 45% versus the current 33% of light water reactors.
- The SCWR design is far simpler than current designs, eliminating circulation pumps, pressurizers, steam generators, steam separators and dryers.
- An SCWR will be smaller than current designs, thereby requiring a smaller containment structure.
- Supercritical water can potentially be used as coolant in a fast breeder reactor

Disadvantages

- The SCWR will require extensive material development, and it is still unclear if suitable materials can be found.
- The chemistry of supercritical water under radiation conditions is not well understood.

SCWR nuclear reactors may appear as early as 2025.

Chapter 4
Molten-Salt Reactor

Reaction: Fast

Moderator: None

Coolant: Liquid Metal

A molten salt reactor (MSR) is a type of nuclear fission reactor where the primary coolant is a molten salt mixture, which can run at high temperatures while staying at low vapor pressure for reduced mechanical stress and increased safety, and is less reactive than molten sodium coolant. The nuclear fuel may be solid fuel rods, or dissolved in the coolant itself, which eliminates fuel fabrication, simplifies reactor structure, equalizes burnup, and allows online reprocessing.

In an MSR, the uranium fuel is dissolved in the sodium fluoride salt coolant which circulates through graphite core channels to achieve some moderation. Fission products are removed continuously while plutonium and other actinides can be added along with U-238, without the need for fuel fabrication. Coolant temperature is 700C at very low pressure. A secondary coolant system is used for electricity generation, and thermo-chemical hydrogen production is also feasible.

Compared with solid-fuelled reactors, MSR systems have lower fissile inventories, no radiation damage constraint on fuel burn-up, no spent nuclear fuel, no requirement to fabricate and handle solid fuel, and a homogeneous isotopic composition of fuel in the reactor. These and other characteristics may enable MSRs to have unique capabilities and competitive economics for actinide burning and extending fuel resources.

The attractive features of the MSR fuel cycle include: the high-level waste is comprised of fission products only, hence shorter-lived radioactivity; small inventory of weapons-fissile material; low fuel use; and safety due to passive cooling up to any size.

For the MSR there are two baseline concepts:

- The Molten Salt Fast Neutron Reactor (MSFR)
- The Advanced High-Temperature Reactor (AHTR)

The AHTR has the same graphite core structures as the VHTR and uses molten salt as coolant instead of helium, enabling power densities four to six times greater than HTRs.

See Figure 3 for a diagram of a proposed MSR reactor.

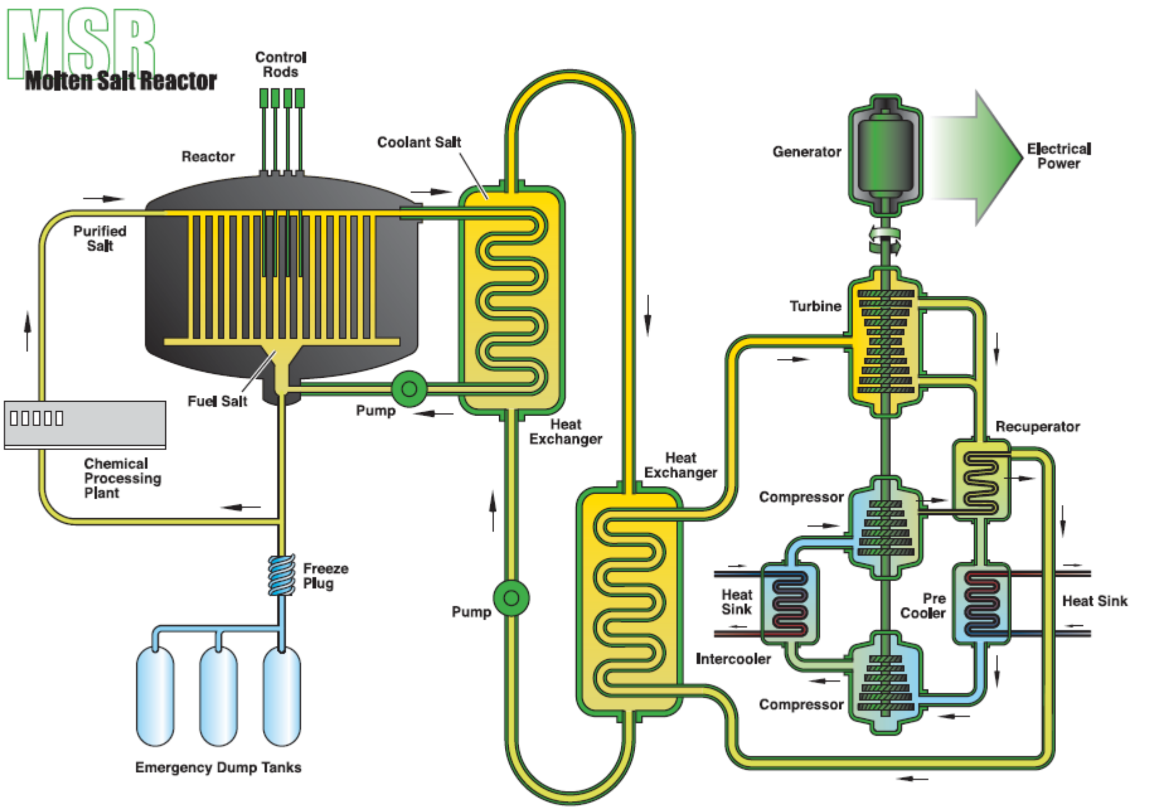


Figure 3

Courtesy: Department of Energy

The MSR produces fission power in a circulating molten salt fuel mixture. MSRs are fueled with uranium or plutonium fluorides dissolved in a mixture of molten fluorides, with Sodium and Zirconium fluorides as the primary option. MSRs have the following unique characteristics,

- MSRs have good neutron economy, opening alternatives for actinide burning and/or high conversion
- Molten fluoride salts have a very low vapor pressure, reducing stresses on the vessel and piping
- Inherent safety is afforded by fail-safe drainage, passive cooling, and a low inventory of volatile fission products in the fuel
- Refueling, processing, and fission product removal can be performed online, potentially yielding high availability
- MSRs allow the addition of actinide feeds of widely varying composition to the homogenous salt solution without the blending and fabrication needed by solid fuel reactors.

The reactor can use U-238 or Th-232 as a fertile fuel dissolved as fluorides in the molten salt. Due to the thermal spectrum of the fluoride MSR, Th-232 achieves the highest conversion factors. All of the MSRs may be started using low-enriched uranium or other fissile materials.

The range of operating temperatures of MSRs ranges from the melting point of eutectic fluorine salts to below the chemical compatibility temperature of nickel-based alloys.

Recent research has focused on the practical advantages of the high-temperature low-pressure primary cooling loop. Many modern designs rely on ceramic fuel dispersed in a graphite matrix, with the molten salt providing low pressure, high temperature cooling. The salts are much more efficient at removing heat from the core, reducing the need for pumping, piping, and reducing the size of the core as these components are reduced in size.

Another advantage of a small core is that it has fewer materials to absorb neutrons. The improved neutron economy makes neutrons available so that the thorium-232 can breed into uranium-233. Thus, the compact core makes the molten salt design particularly suitable for the thorium fuel cycle.

MSR plants should be safe to operate and maintain because molten fluoride salts are mechanically and chemically stable at sea-level pressures at intense heats and radioactivity. Even given an accident, dispersion into the atmosphere is unlikely. The salts do not burn in air or water, and the fluoride salts and radioactive fission products are generally not soluble in water.

There is no high pressure steam in the core, just low-pressure molten salt. This means that the MSR's core cannot have a steam explosion, and does not need the most expensive item in a light water reactor, a high-pressure steam vessel for the core. Instead, there is a vat and low-pressure pipes constructed of thick sheet metal. The metal is an exotic nickel alloy that resists heat and corrosion, but there is much less of it, and the thin metal is less expensive to form and weld.

The thorium breeder reactor uses low-energy thermal neutrons, similarly to light water reactors. It is therefore much safer than fast-neutron breeder reactors that the uranium-to-plutonium fuel cycle requires. The thorium fuel cycle therefore combines safe reactors, a long-term source of abundant fuel, and no need for expensive fuel-enrichment facilities.

The molten-salt-fueled reactor operates much hotter than LWR reactors, from 650C to as hot as 950C. So, very efficient gas turbine generators are possible. The efficiency from high temperatures reduces fuel use, waste emission and the cost of auxiliary equipment (major capital expenses) by 50% or more.

MSRs work in small sizes, as well as large, so a utility could easily build several small reactors (say 100 MW) from operating income, reducing interest expense and business risks.

A concern with MSR’s is that when optimized for breeding, thorium breeder reactors may require on-site reprocessing which might allow diversion of fuels to weapons. In addition fluoride salts naturally produce hydrofluoric acid when in contact with moisture, which may lead to release of hydrofluoric acid fumes during reactor shutdowns, decommissioning, or flooding. A molten salt reactor's fuel can be continuously reprocessed with a small adjacent chemical plant.

The MSR has a number of technical viability issues that need to be resolved. The highest priority issues include molten salt chemistry, solubility of actinides in the fuel, compatibility of irradiated molten salt fuel with structural materials and graphite, and metal clustering in heat exchangers.

The first commercial MSR's are expected to be in operation around 2025.

Chapter 5
Gas-Cooled Fast Reactor

Reaction: Fast

Moderator: None

Coolant: Helium

The Gas-Cooled Fast Reactor (GFR) system features a fast-neutron spectrum and closed fuel cycle for efficient conversion of fertile uranium. The reactor design is a helium-cooled system operating with an outlet temperature of 850C using a direct gas turbine for high thermal efficiency. Several fuel forms are being considered for their potential to operate at very high temperatures and to ensure an excellent retention of fission products: composite ceramic fuel, advanced fuel particles, or ceramic clad elements of actinide compounds. Core configurations are being considered based on pin- or plate-based fuel assemblies or prismatic blocks, which allows for better coolant circulation than traditional fuel assemblies. The reactors are intended for use in nuclear power plants to produce electricity, while at the same time; breeding new nuclear fuel, respectively.

Like other helium-cooled reactors, GFRs will be high-temperature units - 850C. They employ similar reactor technology to the VHTR. The planned design GFR unit is 1,200 MW, with thick steel reactor pressure vessel. For electricity, the helium will directly drive a gas turbine. It will have a self-generating (breeding) core with fast neutron spectrum and no fertile blanket. Robust nitride or carbide fuels will include depleted uranium and any other fissile or fertile materials as ceramic pins, with plutonium content of 15 to 20%. As with the SFR, used fuel will be reprocessed on site and all the actinides recycled repeatedly to minimize production of long-lived radioactive wastes.

The GFR base design is a fast reactor but in other ways similar to a high temperature gas cooled reactor. It differs from the HTGR design in that the core has a higher fissile fuel content as well as a non-fissile, fertile, breeding component, and of course there is no neutron moderator. Due to the higher fissile fuel content, the design has a higher power density than the HTGR.

In a GFR reactor design, the unit operates on fast neutrons, no moderator is needed to slow neutrons down. This means that, apart from nuclear fuel such as uranium, other fuels can be used. The most common is thorium, which absorbs a fast neutron and decays into Uranium-233. This means GFR designs have breeding properties—they can use fuel that is unsuitable in normal reactor designs and breed fuel. Because of these properties, once the initial loading of fuel has been applied into the reactor, the unit can go years without needing fuel. If these reactors are used for breeding, it is economical to remove the fuel and separate the generated fuel for future use.

The gas used can be many different types, including carbon dioxide or helium. It must be composed of elements with low neutron capture cross sections to prevent positive void coefficient and induced radioactivity. The use of gas also removes the possibility of phase transition induced explosions, such as when the water in a water cooled reactor flashes to steam upon overheating or depressurization. The use of gas also allows for higher operating temperatures than are possible with other coolants, increasing thermal efficiency, and allowing other non-mechanical applications of the energy, such as the production of hydrogen fuel.

No true gas-cooled fast reactor design has ever been brought to criticality. The main challenges that have yet to be overcome are in-vessel structural materials, both in-core and out-of-core, that will have to withstand fast-neutron damage and high temperatures, (up to 1,600C). Another problem is the low thermal inertia and poor heat removal capability at low helium pressures, although these issues are shared with thermal reactors which have been constructed.

It is estimated that a prototype system could be placed in operation by 2025.

Performance issues for GFR include:

- Development of materials with superior resistance to fast-neutrons under very-high-temperature conditions
- Development of a high-performance helium turbine for efficient generation of electricity
- Development of efficient coupling technologies for process heat applications and the GFR's high temperature nuclear heat.

Figure 4 on the next page is a diagram of a proposed GFR design.

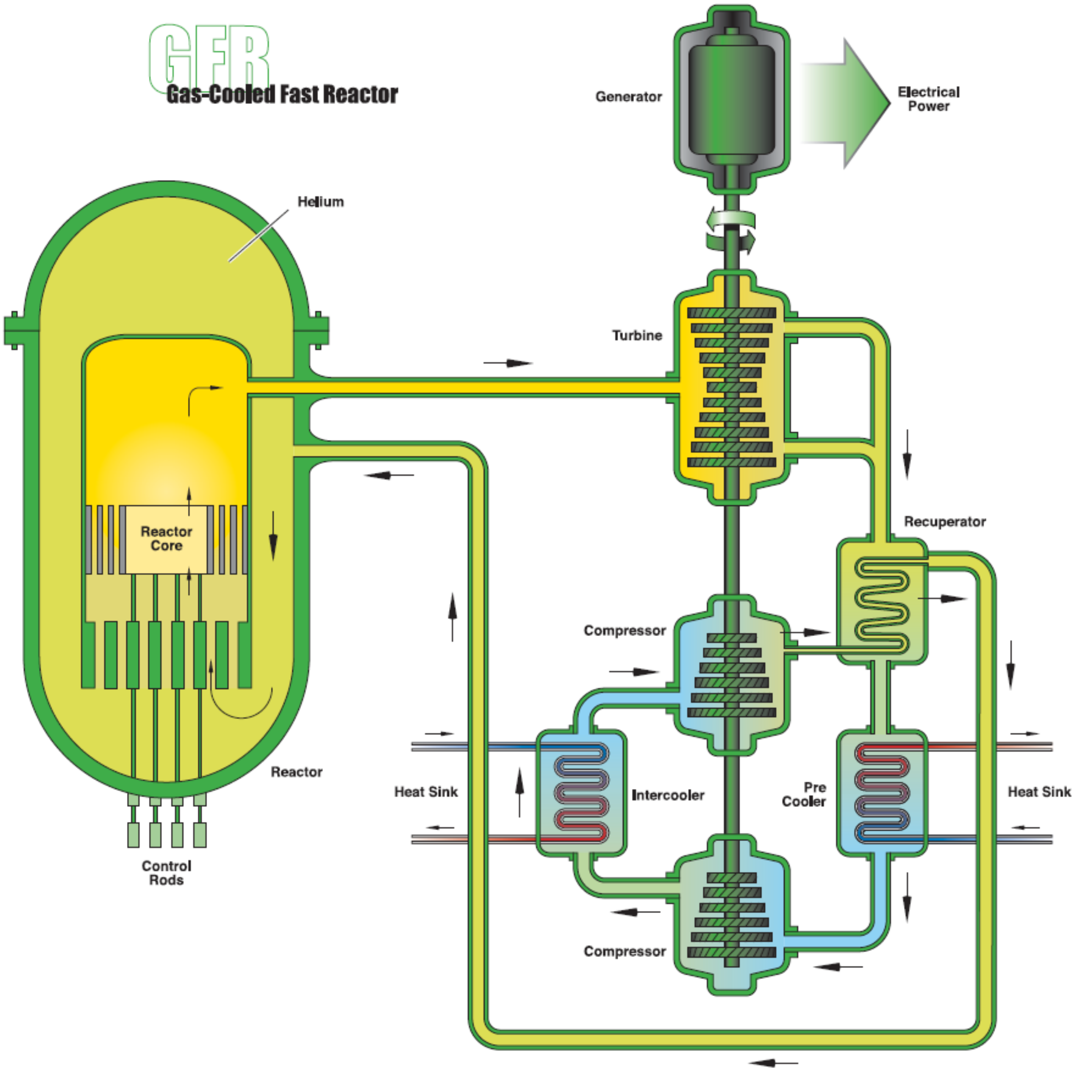


Figure 4 Courtesy: Department of Energy

GFR research is focused on studying potential candidate fuels and evaluating their technical feasibility based on existing information on the structural integrity and radiation resiliency of the coating system and the chemical compatibility among the different materials for the GFR service.

Fuel fabrication techniques must be developed to be compatible with on-site processing for actinide recovery and remote fuel fabrication. Innovative methods such as vapor deposition or impregnation are among the candidate techniques for on-site manufacturing of composite ceramic fuel (e.g., cercer). The main challenges are in-vessel structural materials, both in-core and out-of-core, that will have to withstand fast-neutron damage and high temperatures, up to 1,600C in accident situations. Ceramic materials are therefore the planned option for in-core materials.

The most promising ceramic materials for core structures are carbides such as Silicon-Carbide (SiC.) Inter-metallic compounds like the alloy, Zr_3Si_2, are promising candidates as fast-neutron reflector materials.

For other internal core structures the candidate materials are coated or uncoated ferritic-martensitic steels. The main candidate materials for pressure vessels (reactor, energy conversion system) and cross vessel are martensitic steels.

The innovative GFR design features to be developed must overcome shortcomings of past fast-spectrum gas cooled designs, which were primarily low thermal inertia and poor heat removal capability at low helium pressure. Various passive approaches will be evaluated for the ultimate removal of decay heat in depressurization events. The conditions to ensure a sufficient back pressure and to enhance the reliability of flow initiation are some of the key issues for natural convection, the efficiency of which will have to be evaluated for different fuel types, power densities, and power conversion unit. Dedicated systems, such as semi-passive heavy gas injectors, need to be evaluated and developed. There is also a need to study the creation of conduction paths and various methods to increase fuel thermal inertia and, more generally, core capability to store heat while maintaining fuel temperature at an acceptable level.

The most important issues regarding economic viability of the GFR are associated with the simplified and integrated fuel cycle, and the modularity of the reactor.

Chapter 6
Sodium-Cooled Fast Reactor

The Sodium-cooled fast reactor or SFR is an advanced fast neutron reactor and is closely related to existing LMFBR Reactors. The objective is to produce a fast-spectrum, sodium-cooled reactor and a closed fuel cycle for efficient management of actinides and conversion of fertile uranium-238.

Reaction: Fast

Moderator: None

Coolant: Sodium

The SFR utilizes depleted uranium as the fuel and has a coolant temperature of 550C enabling electricity generation via a secondary sodium circuit, the primary one being at near atmospheric pressure. Three variants are proposed:

- 50-150 MW type with actinides incorporated into metal fuel requiring electrometallurgical processing (pyroprocessing) integrated on site,
- 300-1500 MW pool-type version of this, and
- 600-1500 MW type with conventional MOX fuel and advanced aqueous reprocessing in central facilities elsewhere.

See Figure 5 for an example of a SFR reactor design.

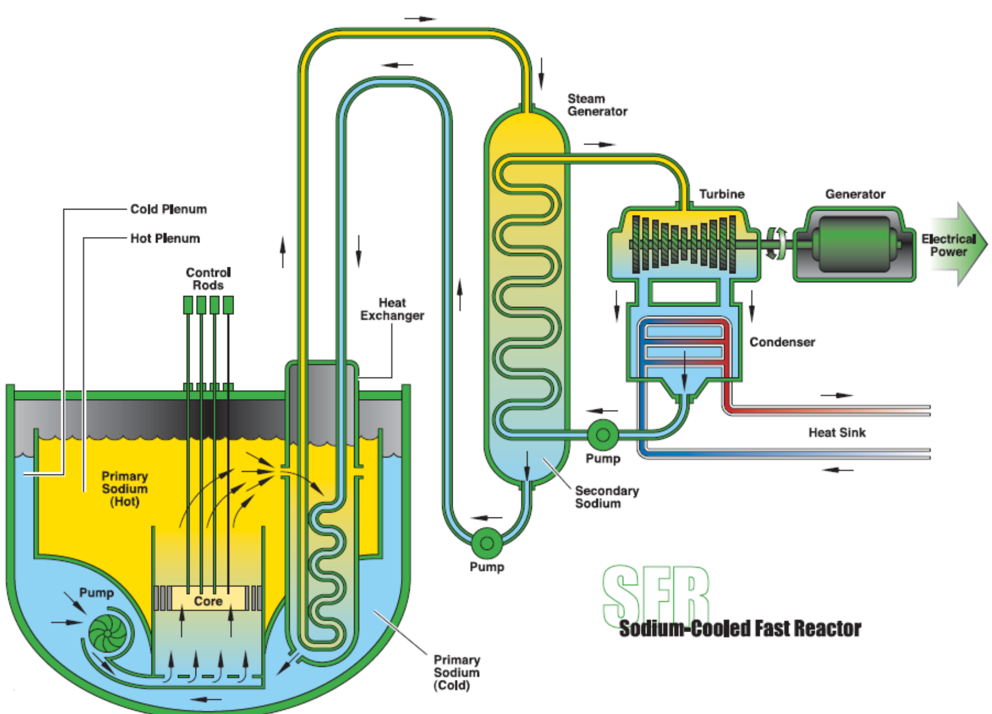


Figure 5 Courtesy: Department of Energy

Two fuel options exist for the SFR: MOX and mixed uranium-plutonium-zirconium metal alloy (metal). The experience with MOX fuel is considerably more extensive than with metal. SFRs require a closed fuel cycle to enable their advantageous actinide management and fuel utilization features.

The primary fuel cycle technologies include an advanced aqueous process, and the *pyroprocess*, which derives from the term, pyro-metallurgical process. Both of these processes will allow recovery and recycling of 99.9% of the actinides, and have inherently low decontamination factor of the product, making it highly radioactive. These fuel cycle technologies must be adaptable to thermal spectrum fuels in addition to serving the needs of the SFR. This is needed because the startup fuel for the fast reactors must come ultimately from spent thermal reactor fuel. This also allows for the waste management advantages of the advanced fuel cycles to be realized and fuel from thermal spectrum plants will need to be processed with the same recovery factors. Thus, the reactor technology and the fuel cycle technology are strongly linked.

The primary coolant system can either be arranged in a pool layout, or in a compact loop layout. For both options, there is a relatively large thermal inertia of the primary coolant. A large margin to coolant boiling is achieved by design, and is an important safety feature of these systems. Another major safety feature is that the primary system operates at essentially atmospheric pressure, pressurized only to the extent needed to move fluid. Sodium reacts chemically with air, and with water, and thus the design must limit the potential for such reactions and their consequences. To improve safety, a secondary sodium system acts as a buffer between the radioactive sodium in the primary system and the steam or water that is contained in the conventional Rankine-cycle power plant. If a sodium-water reaction occurs, it does not involve a radioactive release.

Water is difficult to use as a coolant for a fast reactor because water acts as a neutron moderator that slows the fast neutrons into thermal neutrons. While it may be possible to use supercritical water as a coolant in a fast reactor, this would require a very high pressure. In contrast, sodium atoms are much heavier than both the oxygen and hydrogen atoms found in water, and therefore the neutrons lose less energy in collisions with sodium atoms. Sodium also need not be pressurized since its boiling point is higher than the reactor's operating temperature. A disadvantage of sodium is its chemical reactivity, which requires special precautions to prevent and suppress fires. If sodium comes into contact with water it explodes, and it burns when in contact with air.

The operating temperature must not exceed the melting temperature of the fuel to prevent Fuel-to-cladding chemical interaction (FCCI). FCCI is melting between the fuel and the cladding; uranium, plutonium, and lanthanum inter-diffuse with the iron of the cladding. The alloy that forms has a low melting temperature. FCCI causes the cladding to reduce in strength and could eventually rupture.

The SFR is designed for management of high-level wastes and, in particular, management of plutonium and other actinides. Important safety features of the system include a long thermal response time, a large margin to coolant boiling, a primary system that operates near atmospheric pressure, and intermediate sodium system between the radioactive sodium in the primary system and the water and steam in the power plant. With innovations to reduce capital cost, such as making a modular design, removing a primary loop, integrating the pump and intermediate heat exchanger, or simply find better materials for construction, the SFR can be a viable technology for electricity generation.

The SFR's fast spectrum also makes it possible to use available fissile and fertile materials (including depleted uranium) considerably more efficiently than thermal spectrum reactors with once-through fuel cycles.

Most plants so far have had a core plus blanket configuration, but new designs are likely to have all the neutron action in the core.

Assurance or verification of passive safety is an important performance issue. Some consider the acquisition of irradiation performance data for fuels fabricated with the new fuel cycle technologies to also be a viability issue, rather than a performance issue.

With the advanced aqueous fuel cycle, the key viability issue is the minimal experience with production of ceramic pellets (using remotely operated and maintained equipment) that contain minor actinides and trace amounts of fission products. Further, it is important to demonstrate scale-up of the uranium crystallization step.

A key performance issue for the SFR is cost reduction to competitive levels. The extent of the technology base for SFRs is noted above, yet none of the SFRs constructed to date have been economical to build or operate.

The first SFR's reached commercialization in 2016.

Chapter 7
Lead-Cooled Fast Reactor

Reaction: Fast

Moderator: None

Coolant: Lead

The lead-cooled fast reactor (LFR) is a nuclear power reactor that features a fast neutron spectrum, molten lead or lead-bismuth coolant, and a closed fuel cycle. Options include a range of plant ratings, including a number of 50 to 150 MW units featuring long-life, pre-manufactured cores. Plans include modular arrangements rated at 300 to 400 MW, and a large monolithic plant rated at 1,200 MW. The fuel is metal or nitride-based containing fertile uranium and transuranics. The LFR is cooled by natural convection with a reactor outlet coolant temperature of 550 C, possibly ranging over 800 C with advanced materials. Figure 6 is a diagram of an LFR design.

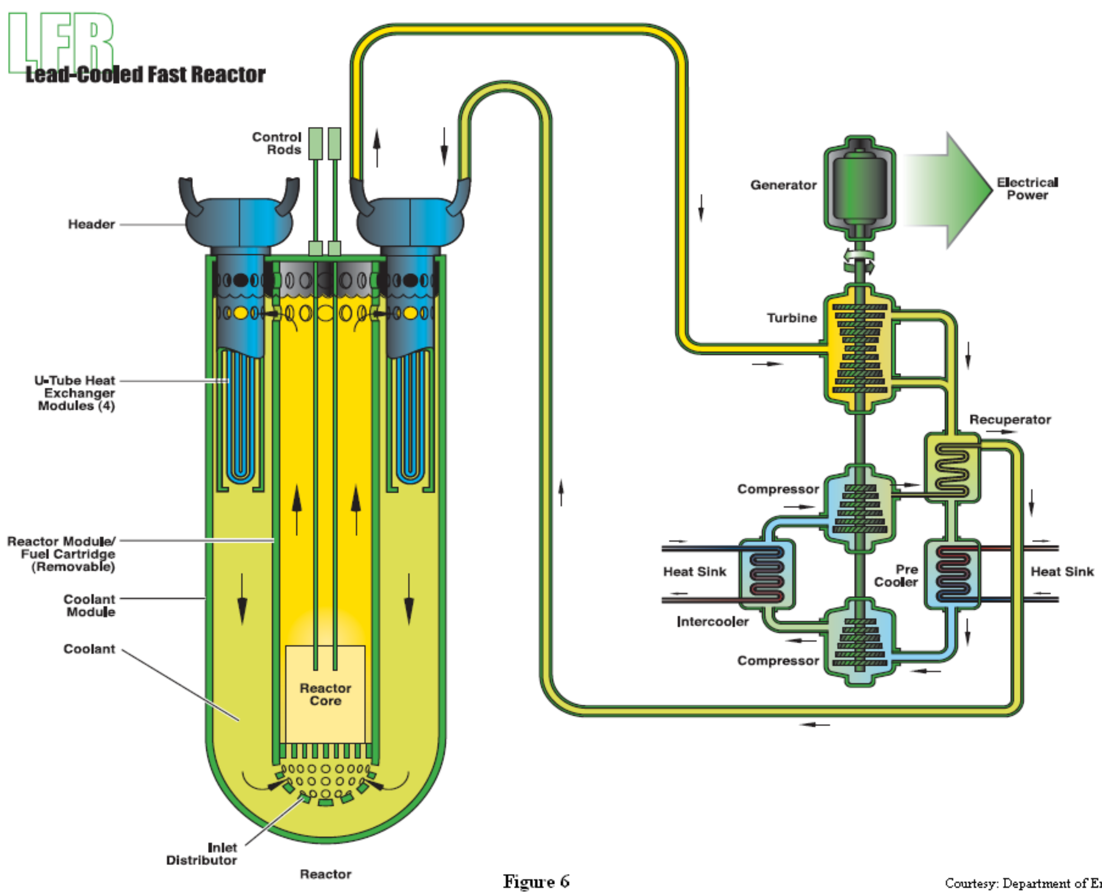


Figure 6

Courtesy: Department of Energy

The LFR battery is a small factory-built turnkey plant operating on a closed fuel cycle with very long refueling interval (15 to 20 years) cassette cores or replaceable reactor modules. Its features are designed to meet market opportunities for electricity production on small grids, and for developing countries that may not wish to deploy fuel cycle infrastructure to support their nuclear energy systems. The modular "battery" system is designed for distributed generation of electricity and other energy products, including hydrogen and potable water.

Russian has a couple of proof of concept LFR's in operation today.

The options in the LFR class may provide a time-phased development path: The nearer-term options focus on electricity production and rely on more easily developed fuel, clad, and coolant combinations and their associated fuel recycle and refabrication technologies. The longer term option seeks to further exploit the inherently safe properties of lead and raise the coolant outlet temperature sufficiently high to enter markets for hydrogen and process heat, possibly as merchant plants. LFR holds the potential for advances compared to state-of-the-art liquid metal fast reactors in the following:

- Innovations in heat transport and energy conversion are a central feature of the LFR options. Innovations in heat transport are afforded by natural circulation,
- The favorable properties of lead coolant and nitride fuel, combined with high temperature structural materials, can extend the reactor coolant outlet temperature into the 750–800ºC range in the long term, which is potentially suitable for hydrogen manufacture and other process heat applications. In this option, the alloying agent is eliminated, and the less corrosive properties of Lead help to enable the use of new high-temperature materials.

The favorable neutronics of Lead and Lead-Bi coolants in the battery option enable low power density, natural circulation-cooled reactors with fissile self sufficient core designs that hold their reactivity over their very long 15- to 20-year refueling interval. For modular and large units more conventional higher power density, forced circulation, and shorter refueling intervals are used, but these units benefit from the improved heat transport and energy conversion technology.

Plants with increased inherent safety and a closed fuel cycle can be achieved in the near- to mid-term. The longer-term option is intended for hydrogen production while still retaining the inherent safety features and controllability advantages of a heat transport circuit with large thermal inertia and a coolant that remains at ambient pressure. The favorable sustainability features of fast spectrum reactors with closed fuel cycles are also retained in all options.

Summary

New nuclear power plants are essential to meeting the future electric demands of the planet. The availability of affordable, clean energy will improve the standard of living and life expectancy of the world's population. The current generation of nuclear power plant designs is morphing into the next generation (Generation III) power plants and research is active on the fourth generation of nuclear power plants. The Generation IV plants, which will likely begin appearing after 2020, are the future of the power industry for clean, affordable energy.

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