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U.S. DEPARTMENT OF ENERGY

Quadrennial Technology Review 2015

High Temperature Reactors

Chapter 4: Technology Assessments

Introduction

High temperature reactor (HTR) systems (i.e., reactors with core outlet temperatures between 700°C and 950°C) offer higher thermodynamic efficiency of converting the heat generated in the reactor to electricity (e.g., ~50% at 950°C or 47% at 850°C) than light water reactors (LWRs); this could greatly improve the economics of reactor systems.[1,2] However, the higher temperature also limits the number of fuel, coolant, and material choices available for the nuclear reactor designer, increasing the emphasis on materials such as high-strength creep-resistant nickel-based metallic alloys, graphite, and ceramics, and motivates R&D on improved materials.

These high outlet temperatures could also supply process heat for a variety of industrial applications. Under the Next Generation Nuclear Plant (NGNP) Project, a variety of process heat applications were assessed, and it was concluded that much of the process heat market could be served by an HTR producing outlet temperatures in the 750-800°C range. The NGNP Technology Roadmap was updated[3] to focus on plants which could be built with materials that are qualified (or undergoing qualification) for operation in this temperature range. The modular HTR (helium or salt-cooled) could thus serve multiple electricity and nonelectrical markets.

Internationally, there are only two high temperature systems under consideration: high temperature gas-cooled reactor systems (HTGR) and molten salt reactors (MSRs). Seven gas-cooled HTRs have been built around the world in England, Germany, the United States, Japan, and China.[4,5] They range in power level from 10 MWt to 842 MWt, are based on both prismatic and pebble bed design, and encompass both demonstration and prototype missions. Currently, there is a two-unit (250MWt each) 500 MWt (total) helium-cooled modular pebble bed power plant under construction in China; five more are planned in coming years. Thus, the technology is reasonably mature, though not yet to a commercial stage in most markets.

Process Heat Applications

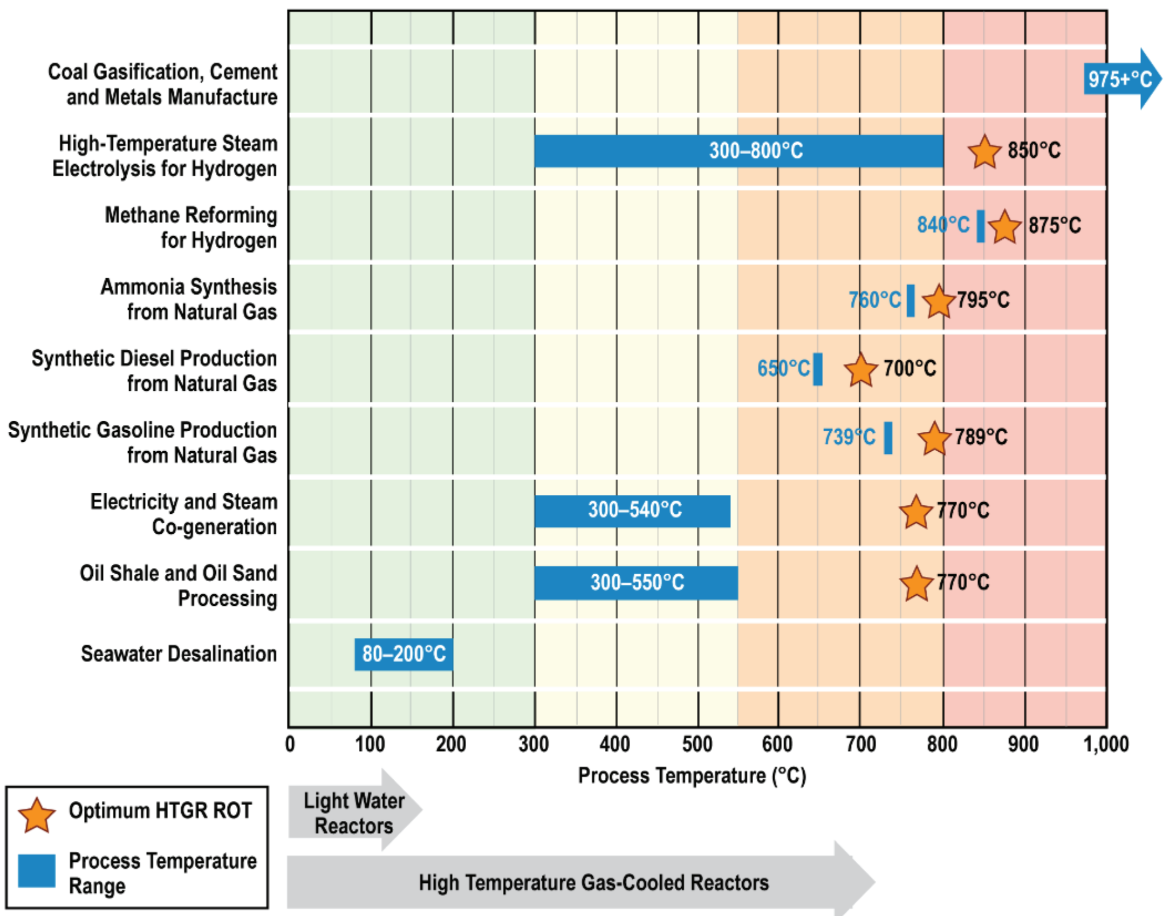
Beyond electricity, the high outlet temperatures of HTRs could drive a number of industrial processes that require heat input in the 650°C to 950°C range, such as petrochemical and fertilizer production, extraction of hydrocarbons from oil sands, the conversion of coal and biomass to high quality liquid fuels, and hydrogen production. For example, hydrogen production is much more economic at the temperatures available from the HTR because more of the energy needed to split water can come directly from the thermal energy of the fluid rather than by converting it first to electricity (with the attendant losses), as is done with LWRs.

Currently, the heat for all of these industrial thermal processes is almost exclusively provided by the burning of fossil fuels and is the source of about 20% of total U.S. energy-related CO_2 emissions.[6] In a study conducted for the NGNP project, a 25% penetration of these four markets by 2030 would prevent 230 to 560 million metric tons of CO_2 emissions using about 400 HTRs with a size of 600-MWt each or roughly 5-10% of the total annual US energy-linked CO_2 emissions.[7]

A study performed for the NGNP Project summarized the range of process heat temperatures suitable for a number of applications and also the optimal temperature for each.[8] The results are illustrated in the following chart.

Figure 4.J.1. Optimum HTGR outlet temperatures and the process temperature range associated with HTGR-integrated industrial processes[9]

Credit: Idaho National Laboratory



As with other small modular reactors (SMRs), the economics of the HTR cannot be proven without an actual demonstration, but its compatibility with nonelectrical applications makes it more attractive for the hybrid energy market. Unlike other nuclear plants with much larger exclusion zones, the industrial facility being driven by the process heat could be located next to the HTR with minimal risk of contamination or dynamic coupling because of the high degree of inherent safety and confinement of the radioactive material. The large heat capacity of the graphite core spreads power transients over tens of hours or even days. The robustness of the TRISO (Tristructural-Isotropic) -coated particle fuel coupled with the low power density of the core allows excess heat to dissipate naturally with no active intervention so that the fuel cannot melt. This eliminates the need for active safety systems or off-site electric power to prevent severe accidents. Furthermore, the potential for radioactive releases is so low that the regulatory requirements for evacuation planning for the public outside the reactor site could be reevaluated. Discussions between the Department of Energy and the Nuclear Regulatory Commission (NRC) commenced under the NGNP Project to agree on new guidelines for emergency planning based upon validated source term models.

Gas-Cooled HTRs

An HTGR is a graphite-moderated, helium-cooled reactor with thermal neutron spectrum. These gas-cooled HTRs are characterized as a source of fission power with smart choices of coolant, fuel, and moderator that are mutually compatible to enhance safety. Furthermore, design of the system is focused on achieving passive safety through the use of robust TRISO-coated particle fuel, large thermal inertia of the graphite moderator, and low power density. Two technology options are currently available in the world: (1) a pebble-bed HTR where TRISO-coated particles are embedded in graphite pebbles that slowly circulate by gravity through the core; and (2) a prismatic HTR where TRISO-coated particles are embedded in graphite cylinders that are placed in hexagonal graphite blocks. Both are at about the same level of technical readiness because they share so many common systems and components. In both designs, chemically inert helium flowing through the core removes the fission heat. The outlet temperature of the helium can be set to match the mission for the reactor. Designs with a 750°C outlet temperature of the helium are used to produce steam via an indirect Rankine cycle. The helium outlet temperature can approach 900°C for higher temperature process heat needs. The heat carried by the helium can be transferred to another fluid through an intermediate heat exchanger for subsequent process heat application and/or electricity. In many process heat applications, delivery of both heat and electricity is required, and the HTR system is flexible enough to alter the mix of electricity and heat to accommodate end-user requirements. This dual mode of operation is unique to the HTR. In even more advanced designs, the hot helium would be transferred directly to a gas turbine via a Brayton cycle or combined Brayton/Rankine power conversion system to obtain a thermodynamic efficiency approaching 50%. The HTR is modular and very scalable with the size of the unit and/or the number of units selected to meet the mission. As many as four to eight large 600 MWt (each) collocated units are envisioned for industrial and conventional electricity markets. Smaller units (10 to 100 MWt each) are under study for off-grid and remote power applications.

To ensure passive decay heat removal, gas-cooled HTRs are limited to 600 MWt each or less. Like all SMRs,[10] this compromises their economic competitiveness in the electricity market because the high capital cost of nuclear plants tends to favor economies of scale for individual units. This disadvantage is partially offset with HTRs because (1) the high outlet temperature boosts plant thermodynamic efficiency and (2) the passive heat removal system avoids the cost and complications of active decay heat removal systems that meet nuclear plant reliability requirements. The system can be deployed in multiple small units, depending on need, and indeed there is a growing interest in using small HTRs (10–600 MWt each) for off-grid and remote power. The smaller thermal output (<600 MWt) is essential to passive safety because the small cores are better able to reject heat passively at this scale.

Prior to the Three Mile Island (TMI) accident, HTRs were large plants with high power densities that competed directly with LWRs in the electricity generation market. Over a dozen large HTRs were on order during the 1970s during the rapid expansion phase of nuclear technology in the United States. All but one of those orders (Fort St. Vrain in Colorado) was cancelled in the post-TMI era. Following TMI, HTR designers in the United States and Germany radically redesigned their concept, with a focus on passive safety. Through a large reduction in power density and the use of a large graphite moderator, designers were able to slow power transients so that they would extend over tens of hours or even days, a unique safety characteristic of the HTR. The robustness of the TRISO-coated particle fuel coupled with the low power density of the core allows excess heat to dissipate naturally so that the fuel never melts or releases large amount of radioactive fission products. This eliminates the need for active safety systems or off-site electric power to remove the heat and prevent severe accidents. Thus, small modular HTRs (300 to 600 MWt each) were born, with the focus on a pebble bed reactor in Germany and a prismatic reactor in the United States. Over the past decade, the focus in the United States has been on process heat, with an outlet temperature of 750°C, to meet near term market needs at minimum risk.[11] However, more recently, off-grid and remote location needs for electricity have prompted interest in very small (~10-40 MWe each) HTRs, where such systems can be competitive with other options for those markets given the high cost of electricity in those cases.[12]

The TRISO fuel form being developed is believed to have two additional advantages beyond safety which preclude the need for reprocessing. First, the fuel has been demonstrated to be capable of much higher burnup, resulting in a more optimal utilization of the initial fissile material. Second, the TRISO fuel form can provide excellent long-term containment for the used fuel fission products as part of a direct disposal process. If reprocessing is desirable, proof-of-principle experiments have been conducted to demonstrate the feasibility of separating the TRISO coating from the fissile kernel. In addition, preliminary research has been done to develop front-end processes that allow the fissile kernels to be recycled via either an aqueous or pyroprocessing route.

The cost of a 600 MWt gas-cooled HTR was estimated based on three independent sets of vendor input during the NGNP project. They are reasonable estimates at this stage of development and experience. Overnight costs for a four-module 600 MWt (each) plant (Nth of a kind) were estimated to be between \$2,350/kWt (750°C Rankine cycle) and \$4180/kWt (950°C Brayton cycle).[13]

The major impediment to deployment of HTRs for process heat is the current low price of natural gas and the high capital cost of nuclear systems (not just HTRs). At \$4/MBtu for natural gas, HTRs are not competitive. Studies done over the past five years for a number of process heat options indicate that HTRs could be economic when or where natural gas is priced between \$6 and \$10/MBtu.[14] Based on detailed technology development roadmaps established for the technology,[15,16] the key technical challenges for these systems include development and qualification of materials that will survive the high temperature environment (especially at outlet temperatures in excess of 750°C for the anticipated 60-year lifetime of these reactors), development of intermediate heat exchanger technology for process heat applications that require hot helium or air (around ~ 900°C) instead of steam, and qualification of the TRISO-coated particle fuel and graphite components at high temperature, radiation dose, and fuel burnup. Understanding and having a predictive modeling and simulation capability to predict the integrated behavior of the reactor system is necessary as part of the safety demonstration for these systems, so that safety margins can be established in worst-case events.

To license an HTR in the United States for applications that use a steam generator, key enabling research and development (R&D) that began over a decade ago and continues today must be completed. This R&D includes the following: (1) completing the qualification of fuel and graphite as part of the Department of Energy Nuclear Energy (DOE-NE) Advanced Reactor Technologies program (the current schedule calls for this research to be completed in late 2021 or early 2022, depending on budgets); (2) completing the American Society of Mechanical Engineers (ASME) qualification of high temperature materials for higher outlet temperature applications (up to 950°C), which should occur in the next three years; and (3) validating reactor design and safety analysis modeling and simulation tools. Experiments underway in the DOE-NE Advanced Reactor

Technologies program will provide data for comparison to modeling predictions over the next three to five years. These modeling and simulation tools are needed to help optimize safety margins and increase confidence in the ability to simulate reactor behavior. Seven to eight years are required to complete this enabling R&D and move the technology to a level of maturity that it could be deployed. With this R&D completed and a reactor design established, an applicant would be able to pursue a license from the NRC. An important aspect of the licensing process will be to continue to pursue the risk-informed technology-neutral framework that was proposed to the NRC as part of the pre-application interactions under the NGNP project.

In the longer term, for process heat applications that require hot gas (helium or air), the development of intermediate heat exchanger technology (e.g., gas-to-gas heat exchangers) is required for these high temperatures, and engineering scale studies and testing on how to integrate the reactor system with the proposed process heat mission are needed (e.g., high temperature valves). For high temperature direct cycle applications, the Brayton cycle technology (e.g., gas turbines) must be demonstrated for these systems. As operational experience is gained, R&D can also focus on improving performance and/or reducing capital and operating costs associated with the nuclear portion of the system.

Molten-salt cooled HTR

Two MSR options are under study today: (1) the fluoride salt-cooled high temperature reactor (FHR) in which solid, graphite-moderated reactor fuel is cooled by the molten salt and (2) a dissolved-fuel MSR in which the fuel forms part of the salt working fluid. FHRs have recently gained interest in the United States because the thermal properties allow operation of the reactor at high temperature, low pressure, and moderate power density while avoiding the technical challenges of a highly radioactive primary coolant. A fast spectrum MSR variant is under study in Europe. A summary technology assessment of each option is provided below.

Fluoride HTR

An FHR is an attractive combination of two technologies: the robust HTR TRISO-coated-particle fuel embedded in a graphite moderator and the good heat transfer afforded by the salt. Like the helium-cooled HTR, FHRs can rely on passive decay heat removal and can be used to drive generators or industrial processes. FHRs, unlike helium-cooled HTRs, are not limited in output power in maintaining passive decay heat removal and consequently can support both large and small plant sizes. Studies,[17] are examining the potential role of FHRs in integrated hybrid energy systems. Unlike the helium-cooled version, FHRs can operate at a low pressure with a higher core power density, which would improve its economics. The limitation to core power density in an FHR does not arise from safety but from available fissile material loading. TRISO particles provide a lower fissile material loading than LWR fuel pellets, which necessitates either frequent or on-line refueling, a lower power density, or an alternate fuel format. Hence, either on-line refueling via a pebble bed or the use of the silicon carbide cladding along with high uranium density pellet fuel is being considered for FHRs.

Major technical challenges still must be overcome and thus the overall technology readiness is low (about a technology readiness level [TRL] of 3). A 25 to 30 year development time is estimated as needed to reach commercial deployment status with a vigorous R&D effort. A prototype or demonstration power plant has never been operated, although a proof-of-concept reactor operated at Oak Ridge National Laboratory (ORNL) in the 1960s and 1970s,[18] and both 10 MW and 100 MW pebble bed versions are being designed in China today. The major remaining technical challenges are corrosion and tritium control. In particular, controlling the chemistry of the salt to minimize its corrosive effects on the structural materials at high temperature is a key challenge. At the high temperatures of the system, tritium transports freely through the cooling system structural alloys and could readily escape the plant if not otherwise controlled. This is an active area of R&D,

but ultimately there will be a need to validate that design solutions function as intended. A leading candidate for the salt, Flibe (a mixture of lithium fluoride and beryllium fluoride with excellent neutronic and thermal properties) is toxic (carcinogenic) and expensive, and the ability to deal with this salt coolant in practical terms during maintenance and operation is also an outstanding issue. Lithium-7 is a substantial constituent of the salt, and a cost-effective isotope separation technique will need to be reindustrialized (the United States ceased production of lithium-7 in 1963)[19] for the salt to be affordable. Solving these challenges will require non-reactor loop testing, then in-pile testing with fuel, graphite, and active chemistry control to demonstrate that corrosion can be controlled. For these reasons, industry interest in the concept has been limited to participation in university projects by a few reactor vendors.

Dissolved Salt

A fluoride salt reactor in which the uranium fuel was dissolved in the salt was built and operated at ORNL in the 1960s. The Molten Salt Reactor Experiment (MSRE) was a 8 MWt reactor that operated from 1965 to 1969.[20] Significant international work on MSRs was performed from the early 1960s to the end of the 1970s with key work on fast spectrum MSRs performed by the United Kingdom Atomic Energy Agency and the Swiss Federal Institute for Reactor Research. Denatured, thermal-spectrum MSRs continued to be studied at ORNL until the early 1980s. The Chinese and European Union have dissolved-fuel MSR designs that feature on-line fuel processing. The liquid fuel does facilitate online chemical processing to remove fission products, inject fresh fuel, and recycle transuranics, all during operation. These same capabilities enable dissolved fuel MSRs to burn spent LWR fuel and destroy plutonium. A fast spectrum dissolved-fuel MSR can potentially burn higher atomic number actinides and increase uranium utilization with a much higher conversion ratio than its thermal spectrum cousin, thus reducing spent fuel toxicity and extending fuel supplies. Both chloride and fluoride salts have been considered for fast spectrum MSRs as have both uranium and thorium based fuel cycles. Proposed thermal spectrum system MSRs employ fluoride salts and most commonly use Flibe as the carrier salt. Thermal spectrum systems nearly always employ graphite as their moderator. The graphite irradiation lifetime makes the moderator the first component requiring replacement.

A wide variety of dissolved fuel MSR designs have been proposed.[21] Designs developed prior to the early 1970s typically featured on-site separation of fissile materials from the remainder of the salt and thus would have had similar proliferation characteristics to the integral fast reactor with its on-site pyroprocessing. More modern designs generally do not separate fissile materials and feature only minimal coolant chemistry adjustment. Avoidance of fissile material separation is not, however, universal, and continued vigilance on the proliferation characteristics of each specific design remains necessary. The liquid fuel of MSRs also affords the potential for proliferation resistance attributes not possible in solid fueled reactors. Once started, MSRs will contain an undesirable isotopic fissile material composition (similar to very high burnup fuel), and any fresh fuel that is added would immediately be dissolved into the mixture, preventing the “short-cycling” that can enable enrichment and diversion that is possible with solid fueled reactors. Moreover, fast-spectrum MSRs are intended to be refueled with natural or depleted uranium and thereby avoid the requirement for uranium enrichment facilities and reduce the overall nuclear power enterprise proliferation vulnerability. Several small private companies are developing thermal-spectrum dissolved fuel MSR concepts. While substantial uncertainty remains in the commercial designs, avoiding on-site separation of fissile materials is a major point of design emphasis for several of the vendors.

The major challenges with this system are as follows:

- Designs that feature on-site separation of fissile materials would eliminate a barrier to diversion of nuclear material.
- Coolant (if Flibe is used) is toxic (carcinogenic) and potentially prohibitively expensive. Lower-cost industrial scale lithium isotopic separation technology needs to be developed.

- Licensing of a "dissolved moving fuel" concept is viewed as difficult in the United States given the precedent of having stationary solid fuel in more conventional reactor concepts and at a minimum would require different review and evaluation tools and methods than are currently available.
- Operations and maintenance would have to be entirely remotely performed due to the high radiation levels.
- Despite the success of the MSRE and molten salt breeder reactor (MSBR) experiments, which operated for a few years each, much work remains to be performed in corrosion control, especially for the FHR concept, when inherent chemistry control through the two valence states of uranium is not available as in the case of the dissolved salt MSRs. High-temperature material creep and creep-fatigue for power plant components are also needed and expected to run for decades. Further, some concepts use a different salt than Flibe, which would require even more corrosion work to prove the efficacy of that salt. In that case, the overall technology readiness of the concept is lower.

The U.S. fusion community has studied Flibe as a blanket coolant for fusion reactors, largely for the same reasons that make it a good candidate for the FHR.[22] The challenges (corrosion, toxicity, tritium permeation, etc.) of using Flibe were recognized and have been studied, but progress has been limited such that its use would require further research and testing.

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Acronyms

ASME	American Society of Mechanical Engineers
DOE NE	Department of Energy Office of Nuclear Energy
EIA	United States Energy Information Administration
Flibe	Lithium fluoride and beryllium fluoride salt mixture
FHR	Fluoride salt-cooled high temperature reactor
HTGR	High temperature gas or gas-cooled reactor
HTR	High temperature reactor
LWR	Light water reactor
MWe	Megawatt-electric
MWt	Megawatt-thermal
MSBR	Molten salt breeder reactor
MSR	Molten salt reactor
MSRE	Molten Salt Reactor Experiment
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
R&D	Research and development
SMR	Small modular reactor
TRL	Technology Readiness Level

TMI	Three Mile Island nuclear power plant
TRU	Transuranic elements
TRISO	Tristructural-isotropic coated particle fuel

Glossary

Actinides	The actinide elements are the chemical elements starting with actinium (atomic number 89) and going through lawrencium (atomic number 103).
Brayton cycle	In the Brayton cycle the working fluid remains a vapor throughout the entire power cycle as opposed to the standard Rankine cycle where the operating fluid is continuously evaporated and condensed.
Colocate	Colocate is placing two or more units or facilities near each other. Examples include multiple units of a reactor, especially small modular reactors, or a reactor, like a high temperature reactor, and an industrial facility that makes use of the process heat from the reactor.
Enrichment	Enrichment is the process by which the amount of the uranium-235 isotope is increased from its natural amount in uranium compared to the uranium-238 isotope.
Fast (spectrum) Neutrons	Neutrons released during fission that have high energy levels and are travelling at very high velocity.
FHR	Fluoride salt-cooled high temperature reactors are a subset of the Generation IV high temperature reactor systems. It uses molten fluoride salts as the coolant instead of helium gas. A solid fuel similar to that used in high temperature gas-cooled reactors is also used for the FHR.
GFR	The gas fast reactor or GFR is a Generation IV advanced reactor design (https://www.gen-4.org/gif/jcms/c_42148/gas-cooled-fast-reactor-gfr). The proposed reactor design operates at high temperatures and uses helium as a coolant. The reactor uses fast or high-energy neutrons and would likely employ a continuous recycle fuel cycle. Because of the high-temperatures generated, the system is proposed for potential support of a wide range of industrial processes requiring large amounts of heat or steam.
Generation IV Reactor	Generation IV reactors are the next generation of reactors that are currently being researched for potential deployment in the future. Reactors operating today are primarily Generation II and III designs. New reactors under construction in the United States are considered Generation III+.

Flibe	Flibe is a molten salt made from a mixture of lithium fluoride and beryllium fluoride. It is under consideration as a coolant for molten salt reactors.
HTGR	High temperature gas or gas-cooled reactor or HTGR is an advanced reactor design that operates at high-temperatures (above 700°C) and uses helium as a coolant (http://www.ngnpalliance.org/index.php/htgr). The fuel is coated compounds of uranium (often uranium dioxide). The reactor is typically proposed for operation using a once-through fuel cycle. Because of the high-temperatures generated, the system is proposed for potential support of a wide range of industrial processes requiring large amounts of heat or steam.
HTR	High temperature reactor or HTR is an advanced reactor design that operates at high-temperatures (above 700 °C) and uses either helium or molten salt as a coolant. The fuel is coated compounds of uranium (often uranium dioxide). The reactor is typically proposed for operation using a once-through fuel cycle. Because of the high-temperatures generated, the system is proposed for potential support of a wide range of industrial processes requiring large amounts of heat or steam.
LWR	Light water reactors are the standard reactor design deployed today. They use normal water (H_2O) as the coolant and neutron moderator to lower the energy of the neutrons to thermal levels. The fuel is typically uranium-dioxide pellets that are placed into cladding of a zirconium alloy. The system can operate with low-enriched uranium. In the United States and a number of other countries, LWRs are operated using a once-through fuel cycle, but some countries also deploy a limited recycle option.
Moderator (neutron)	Material used to lower the energy level of neutrons (from fast to thermal) that are generated from fission. Moderators are materials like natural water, heavy water, or graphite. The energy of the neutron is lowered due to collisions with the moderator atoms.
MSR	Molten salt reactor or MSR is a Generation IV advanced reactor design (https://www.gen-4.org/gif/jcms/c_9359/msr). The MSR is distinguished by its core in which the fuel is dissolved in molten fluoride salt. The salt is both the fuel and coolant. The reactor can be designed to operate with either low or high-energy neutrons. The MSR has been proposed for operation as both a once-through fuel cycle and a continuous recycle fuel cycle.
MSRE	Molten Salt Research Experiment or MSRE was a small prototype (8 MWt) molten salt reactor that was operated at Oak Ridge National Laboratory. It went critical in 1965 and operated through 1969. It demonstrated a number of the technical aspect of MSRs.[23]

Next Generation Nuclear Plant The Energy Policy Act of 2005 tasked the Secretary of Energy to establish the Next Generation Nuclear Plant Project. Under the Act, this project consists of the research, development, design, construction, and operation of a prototype plant that is supported by the Generation IV Nuclear Energy Systems Initiative, and the reactor shall be used to generate electricity and/or produce hydrogen.[24] The focus of this work was therefore on high temperature reactors.

Pebble Bed A pebble bed reactor is a type of high temperature reactor in which the fuel is spherical and the approximate size of tennis balls. The pebbles are made of pyrolytic graphite and contain TRISO fuel particles. The pebble bed design uses hundreds of thousands of tennis ball-sized spherical fuel elements. The pebbles are stacked together in contact with each other like gumballs in a vending machine. The pebbles are added at the top, circulate through the reactor core, and are removed from the bottom. Fuel replacement in a pebble bed design is continuous and allows for online refueling.[25]

Prismatic A prismatic design HTR uses cylindrical fuel elements that are pressed into channels drilled into graphite blocks. These fuel-bearing blocks are stacked in columns in fixed locations in the reactor core. Refueling is accomplished by shutting down the reactor, removing the fuel-bearing blocks, and replacing the oldest ones with new blocks. The cylindrical fuel elements contain TRISO fuel particles.[25]

Rankine Cycle The Rankine cycle is the fundamental operating cycle of most power plants where an operating fluid is continuously evaporated and condensed. The selection of operating fluid depends mainly on the available temperature range.[26]

Reprocessing or recycling Reprocessing or recycling is the chemical treatment of used reactor fuel to separate uranium and plutonium and possibly transuranic elements from the fission products. The recovered uranium, plutonium, and transuranic elements can be recycled to a reactor to be burned. The fission products can be converted to high-level waste for disposal. Example technologies include aqueous-based processes like PUREX and dry processes like electrochemical recycling or pyroprocessing.

Thermal (Spectrum) Neutron A neutron whose energy has been reduced by collisions with moderator materials such that the neutron is in thermal equilibrium with the medium in which it is interacting.

Transuranics Transuranic elements or TRU are artificially made, radioactive elements that have an atomic number higher than uranium in the periodic table of elements such as neptunium, plutonium, americium, and others.

TRISO	Triso (tristructural-isotropic) fuel particles are triple-coated spherical particles of uranium fuel, less than one millimetre in diameter. A uranium center is coated by a layer of carbon, which is then coated by silicon carbide, with an outer shell of carbon. In effect, this gives each tiny particle its own primary containment system. The particles are then fabricated into fuel pellets.[27]

Further Reading

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