Nuclear Energy for the Future

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Abstract – Nuclear breeder reactors hold the promise for limitless energy supplies without the use of fossil fuels or renewables at acceptable costs. After about a 30-year hiatus, the designs for an improved water cooled nuclear reactor are being developed by a consortium of nuclear reactor builders and users under an agreement with the US Department of Energy (DOE) and the Nuclear Regulatory Commission (NRC). The development of high-temperature helium-cooled pebble-bed or prismatic bed reactors may herald the entry of an entirely new, safer, and less costly type of reactor to replace the water-cooled reactors of the past and current types.

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I. A Brief History of Nuclear Reactor Development

In 1930, at a World Power Conference in Berlin, Arthur Eddington presented an address in which he challenged the assembled experts to learn how to use the enormous amounts of energy contained in all atoms according to Albert Einstein's mass(m)-energy(E) relation $E=mc^2$, where c is the velocity of light. The first engineering response to this challenge was produced by Enrico Fermi's group at Chicago in December 1942, in conjunction with the development of a nuclear weapon under the aegis of the World War II Manhattan Project, when Fermi's group operated Chicago Pile 1, the first self-sustaining nuclear reactor. At the present time, nuclear energy is used to generate about 16% of the world's electrical power while coal, natural gas, and oil are used to produce about 65% with the collection of "renewable" sources contributing the balance of 19% (of these, hydroelectric and geothermal energy are the primary sources - about 16% of total). The percentages of nuclear energy used for electricity generation range from about 77% for France to 35% for Japan to 20% for the US to 15% for Canada. Because of the very large total generating capacity in the US, most of the currently operating nuclear reactors are located in the US (104 of about 440).

We refer to the encyclopedia article in Ref. (1) for a good overview of the principles of nuclear engineering.

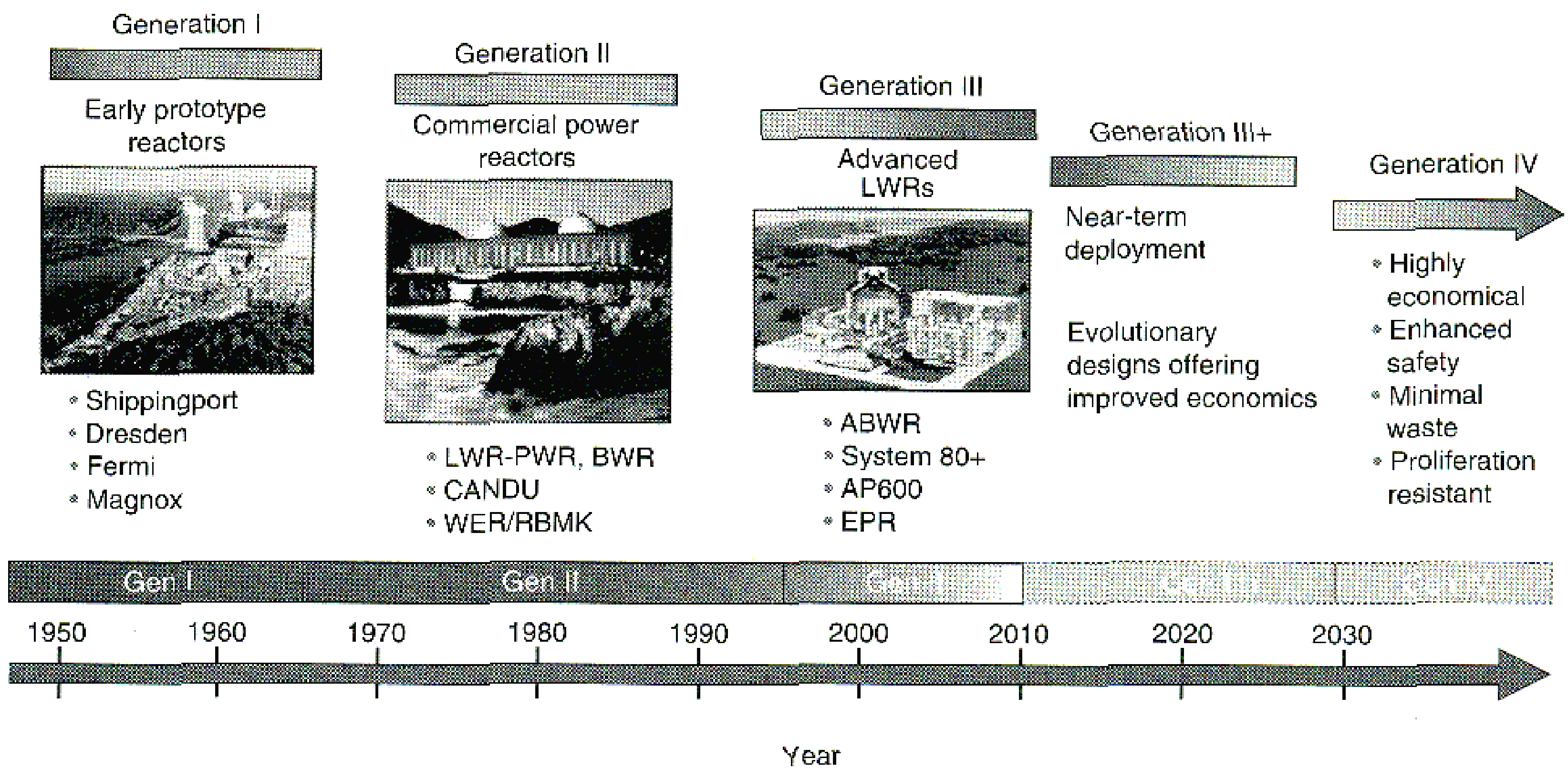


Fig. 1. A subsequent generation of nuclear plants is built on the experience of previous generations. LWRs = light water reactors; PWRs = pressurized water reactors; BWRs = boiling water reactors; CANDU = Canadian deuterium uranium reactor; WER/RBMK = Russian light water/boiling water reactors; ABWR =advanced boiling water reactor; System 80+ = advanced version of pressurized water reactors; AP 600, see the following discussion; EPR = European pressurized water reactor. Reproduced from a figure of the U.S. Department of Energy.

I-1. Passively Safe Systems

It is worth noting how the AP600 system is designed to produce passive safety in case of a LOCA (=loss of coolant accident). The basic idea is simply that the application of gravity cannot fail. The AP600 is the first reactor with this passively safe feature and is certified by the U.S. Nuclear Regulatory Commission (NRC). Tanks elevated with respect to the reactor core are filled with cold water containing a dissolved salt (e.g., sodium borate). If a LOCA occurs while the reactor core pressure remains elevated, the core make-up tanks (CMTs) circulate cold water through the core as the result of negative buoyancy. At somewhat reduced pressures, forced water injection is caused by high-pressure nitrogen. At low pressures, forced water injection is caused from refueling water-storage tanks. The flashing steam condenses on the internal walls before it is recycled back to the water-storage tank and the lower containment compartment. Heat generated in the reactor core drives the convection cooling processes to operate as long as injection cooling is needed. The AP600 has sufficient redundancy to guarantee continued reactor cooling even when some but not all of the systems fail. In the highly unlikely event that all of the passive safety systems for water-cooled reactors fail simultaneously, safety systems such as the containment vessels and activated cooling systems are designed to ensure public safety. The passive safety margins for pebble-bed reactors (PBRs) and prismatic helium cooled reactors should generally ensure operational safety.

I-2. Nuclear Waste Disposal

The nuclear fuel cycle describes the life cycle of nuclear fuel from mining to disposal in a repository. The uranium is separated from its ore to produce "yellowcake" (U_3O_8) which is converted to gaseous UF_6 by chemical conversion. UF_6 is then progressively enriched in its U-235 content in a gaseous diffusion or centrifugal separation process based on the fact that lighter constituents diffuse more rapidly through permeable membranes than heavier constituents. After enrichment to about the 3% level, the gas is processed to produce uranium dioxide fuel pellets. The fuel is "burned" in a nuclear reactor for about 3 years to produce electricity. Fuel rods holding the spent fuel may be stored in water pools for several years to allow decay of the most intensely radioactive fission products. Subsequently, they are removed to dry storage about 300 m underground in a dry, geologically stable medium. Alternatively, the spent fuel rods are reprocessed by separating the uranium and plutonium, which is followed by reprocessing to produce new fuel. Isotopes for specialty medical or industrial uses may also be separated. Long-lived fission products and actinides may be separated and returned to the reactor for transformations to stable or rapidly decaying isotopes (transmutation). Because uranium and plutonium make up 97.1% of the spent fuel while industrially usable isotopes and actinides constitute 2.6%, reprocessing utilizes valuable waste components while reducing the waste to about 1/300 of its initial volume. The result is that the annual per capita high-level waste is about the size of an aspirin tablet. This waste may be encapsulated in molten glass, cooled, and then buried about 300 m deep in a dry geologic medium impermeable to water before being packed in bentonite clay. The radioactive components will decay to the level of natural uranium within about 10,000 years*. The currently preferred burial site is Yucca Mountain in Nevada where the water table is about 550 m deep. At a depth of 300 m, the waste will not be exposed to either rainfall or upwelling groundwater. At the site, a vertical central shaft will be connected to horizontal tunnels holding the waste canisters, which are made of corrosion-resistant titanium or steel. After filling with waste, this structure will be sealed with bentonite clay. Use of the described multiple barriers should retain the high-level wastes safely for the required long periods of time.

* Safe storage continues beyond 10,000 years and perhaps beyond 1 million years as advocated by some concerned environmentalists.

To minimize leaks during shipment, the shipping casks are designed to withstand a 9.1 m fall to a firm surface (i.e., from an overpass), a 1.02 m fall on a 15.2-cm diameter pin (i.e., a bridge abutment), a 30-minute fire at 801.7 °C (i.e., a crash with flammable material), an 8-hour submersion in water (i.e., a roadside fall into a creek). The casks must survive an 84 mile/hr impact with a concrete barrier, a broadside hit from a locomotive at 80 miles/hr, all followed by fire tests without leaking.

I-3. Reprocessing and Proliferation

The plutonium-239 separated from spent nuclear fuel could be diverted to produce nuclear weapons or it could be fissioned in a nuclear reactor to produce electricity. The spent-fuel reprocessing facility could be used to convert weapons-grade plutonium to low-enriched mixed oxide (MOX), which is a nuclear fuel for electricity production, as has been proposed for dismantled nuclear weapons produced in the former Soviet Union. Weapons-grade plutonium has at least 92% of the isotope ^{239}Pu, whereas typical commercial spent fuel has less than 80% of this isotope. US policies have shifted from opposition to research on advanced reprocessing technologies during Democratic presidential terms to support for research on reprocessing with Republican administrations. Unfavorable economics for civilian applications are, in part, to blame for these policy shifts. Commercial reprocessing plants are subject to nuclear material safeguards verification by the IAEA (International Atomic Energy Agency) under the 1970 Treaty on the Nonproliferation of Nuclear Weapons which has been signed by 188 member states (with India, Israel, and Pakistan as non-signators and North Korea recently stating its intention to withdraw).

I-4. Nuclear Fuel Sources (Ref.1, pp. 383-394)

The original fuel for light water reactors is obtained by conversion and enrichment from uranium ore mining and milling. Natural uranium is used for heavy water reactors. Spent nuclear fuel (SNF) is disposed of as radioactive waste.

I-5. A Condensed History of Nuclear Power Agencies (Ref.1, pp. 395-408)

Eisenhower (1953): “Atoms for Peace” initiative.

Atomic Energy Act of 1954.

Price-Andersen Act (Private Insurers offered $60m, Congress added $500m).

Antinuclear Movement (including the idea since 1970 that nuclear pollution must be considered).

The Arab Oil Embargo of 1973 provided clear indication of the need to develop nuclear-powered electricity.

AEC abolished and replaced in the Energy Reorganization Act of 1974 by ERDA and NRC.

II. Fission Reactors (Boiling (BPR) and Pressurized Water (PWR) Reactors)

The BPRs and PWRs of the 21st century will be the successors to the Light Water Reactors built during the last 40 years, following initial development for naval nuclear power beginning with the USS Nautilus in 1954 (submarine applications were pioneered by Admiral Hyman Rickover) and both US and Soviet Union surface ships in 1959. The US Atomic Energy Commission sponsored the development of the 90-MW_e Shippingport PWR (1957) while the first commercial 250-MW_e PWR (Yankee Rowe) was completed in 1960 and operated until 1992. General Electric built the first large BWR (200 MW_e) which operated from 1960 to 1978. The first Soviet PWR (210 MW_e) started operation in 1964 and was soon followed by a 440 MW_e unit. Since 1960 electric utilities have built both types of reactors (65% PWRs and 23% BWRs). Recently, 441 units generated a total of 1,367,000 MW_e throughout the world (2 billion MWh in 2001) with 14 countries generating more than 30% of the electricity used by them and 8 countries currently building these systems, while new designs are being developed in many countries. There are numerous nuclear fuel-assembly designs for both BWRs and PWRs. Both military (submarines, cruisers, aircraft carriers) and civilian (merchant ships, icebreakers) nuclear- powered ships remain in service. Various types of nuclear fuel designs are being used.

III. The NuStart and Dominion Initiatives

After a hiatus in the US of nearly 30 years, construction of the first advanced nuclear reactor has been under consideration by the NuStart Energy Development LLC, a consortium of nine nuclear power companies† (which currently generate 58% of US nuclear electricity), together with the two major US nuclear reactor builders (GE and Westinghouse). Another group is led by Dominion Energy, GE and Bechtel Corp. A cooperative agreement has been signed with NRC to design and construct the new reactor after obtaining operating licenses by 2010.

The current agreement calls for cost-sharing of $520 million between the NRC and the NuStart and Dominion companies for the preliminary design costs. Two designs for specified sites will be submitted by 2008 to NRC for two combined operating licenses.

Following NRC review, two construction permits will be issued, hopefully by 2010, to all of the participants or to as yet undetermined subgroups of these companies.

† The companies involved are Constellation Energy, Duke Energy, the subsidiary of Electricité de France known as EDF International North America, Energy Nuclear, Exelon Generation, Florida Power and Light Company, Progress Energy, Southern Company, and the Tennessee Valley Authority (TVA).

IV. Perspectives on Fast Breeder Reactors (FBRs)

IV-1 Historical Overview

1945: E. Fermi, "The country which first develops a breeder reactor will have a great competitive advantage in atomic energy."

1946-53: Clementine, the first US experimental FBR operated at Los Alamos, had a 2.5 l core fuelled with Pu metal and was cooled by mercury, 25 kW_t at full power.

1951-63: 200 kW_e generated by EBR-I (=experimental fast breeder reactor-I) at the National Reactor Testing Station in Idaho.

1963: The EBR-II (=FBR) program was directed at the Argonne National Laboratory by W. Zinn.

1959: 5 MW_t SBR-5 in Russia.

1959-80: 12 MW_e "demonstration fast reactor" (=DFR) at Dounreay, UK.

1967: 20 to 40 MW_t Rapsodie in France.

Incidents, accidents and safety issues have involved non-nuclear systems components. Development must proceed to passively safe nuclear reactors before large- scale development of breeder reactors becomes politically acceptable.

Plans for early development of breeder reactors have been deferred because of recent discoveries of large reserves of uranium which have extended the time horizon for lower cost conventional nuclear reactors by 50 to more than 100 years.

IV-2. Why FBRs

The current generation of nuclear reactors produces about 400 GW_e and will deplete known nuclear fuel resources in about 100 years.

With FBRs, known fuel sources will last about as long as intelligent life is likely to exist on the planet. The energy content of 1g of naturally occurring uranium is approximately (1MW-day/g of U) x (24 hr/day) x (10^3 kW/MW) x (10^7erg/$2.78\text{x}10^{-7}$ kWh) = $8.6\text{x}10^{17}$ erg/g of U. Therefore, for use in breeder reactors, the ocean uranium content has the energy value ($1.5\text{x}10^{18}$ m^3 of water) x (1 mt of water/m^3 of water) x ($3.3\text{x}10^{-9}$ mt of U/mt of water) x ($8.6\text{x}10^{17}$ erg/g of U) x (10^6 g of U/mt of U) = $4\text{x}10^{33}$ erg. The uranium content of the earth's crust has been estimated as $6.5\text{x}10^{13}$ mt. The energy equivalent for use in a breeder reactor of this resource is then ($6.5\text{x}10^{13}$ mt of U) x (10^6 g of U/mt of U) x ($8.6\text{x}10^{17}$ erg/g of U) = $5.6\text{x}10^{37}$ erg. At the estimated year 2050 per capita energy-consumption rate of $2\text{x}10^{28}$ erg/year, this energy source would last for $2.8\text{x}10^9$ years, which is a large fraction of remaining human life on the planet with needed energy supplies for 10 billion people enjoying the US standards of living of the year 2000 provided the human population remains unchanged and 40% energy conservation can be implemented.

FBRs have been successfully tested, built and used in a number of countries. Nevertheless, with the discovery of new high-grade uranium resources, a world-wide decision to terminate large-scale construction and development was implemented at the end of the nineties. In what will hopefully not turn out to have been a futile effort to rescue technological achievements from oblivion for the benefit of future generations, we produced a summary of design, development and construction status prepared by leading experts on FBRs in 1998 with seminal contributions of leading authorities from the USA, France, Russia, Germany, the UK, Japan, and India. Our conviction that the use of FBRs will be needed for the advancement of human civilizations has not been dimmed by time. J.P. Cretté (seee p. 581 of Ref. 4) summarized the European status in the Abstract of his chapter by noting that early development of the liquid metal fast breeder reactor (LMFBR) was initiated in England and in France in the mid-50s. The demonstration plants PFR in Scotland and Phénix in France were started up in the mid-70s. The 1200 MW_e, commercial-size LMFBR Superphénix was built and operated by a consortium of experts from France, Germany and Italy, and first reached full power in 1986. Besides electricity production and the demonstration of breeding gain, it was scheduled to burn plutonium and other actinides in a fast flux. Future European developments were aimed at designing a 1500 MW_e European Fast Reactor (EFR) which was expected to become cost-competitive with LWRs when the market price of uranium had risen by a factor of 2 to 3. The EFR was not built and the entire worldwide breeder program, with the exception of Indian activities, was terminated around 1998.

V. Coastally-Based Nuclear Reactors - Present Developments in China and Russia

The American Physical Society recently launched a political initiative to augment the populations of scientists and engineers in US graduate schools while enrollment of foreign-born students is rapidly declining because of opportunities and initiatives designed to keep the brightest and ablest in their countries of birth. During a recent extensive visit to China, senior members of the American Chemical Society were impressed by the high educational proficiency and outstanding educational and job opportunities of advanced students in many fields of science and engineering. The educational contents and standards, which will ultimately be reflected in frontier technologies, have indeed changed dramatically since one of us (SSP) served as chair of a study on the impact of foreign born engineering students on US science and engineering. In this evaluation, it was found that, contrary to the then prevalent view that we were educationg our competitors, the vast majority (upwards of 95% for many countries) of the US-trained science and engineering graduates remained or returned to the US and helped to provide the impetus for US supremacy in advanced technologies. This was a bargain system for US industry because we got the ablest graduate students educated to the advanced college levels at the cost of their home countries, provided for their final training, and then reaped the benefits of their superior performance after they became US residents and citizens. Now we must learn to stay ahead with our own students, educated at our expense, and hope that we can keep them here in the face of rapidly growing attractive opportunities abroad.

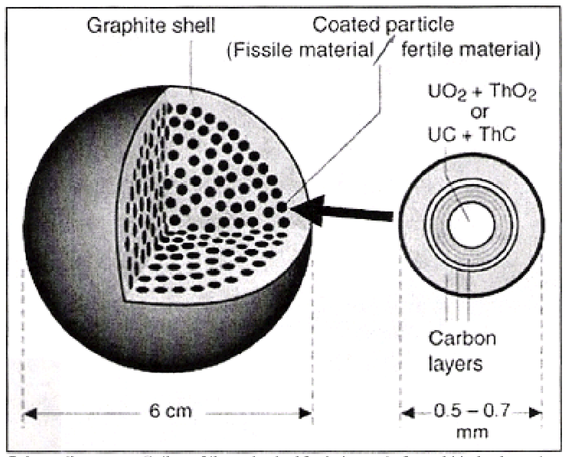
In the energy area, we are beginning to see foreign initiatives that may change the face of our industrial system. China, which contains 16 of the planet's 20 worst cities in terms of health and environment, is pioneering the development and construction of small (ca. 10 MW_e) floating, coastally-based nuclear reactors. A summary of the Chinese pebble-bed reactor program has been given by Fred Singer in his inspirational sermon called TWTW of 05 September 2004. On June 26, 2005, Mosnews.com published an announcement from the Russian Atomic Energy Agency (RosAtom) that a floating nuclear power plant of 70 MWt (which would supply electricity and heat for about 50,000 people) would be constructed at Severodvinsk within 5 years at a cost of $180 million.

VI. Modular Pebble-Bed (MPBR) and Prismatic Block Nuclear Reactors

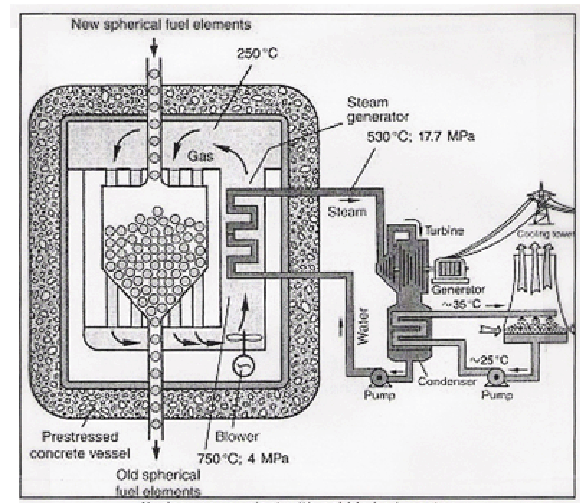
VI-1. Reactor concept and details

The basic idea for this type of reactor was first proposed by Farrington Daniels while he was on leave during the nineteen forties at the Argonne National Laboratory from the University of Wisconsin where he was one of the senior professors of physical chemistry. His proposal was to use beryllium oxide or graphite as moderator materials in nuclear reactors using fuel carriers in the form of round pebbles. This idea was soon referred to as the "Daniels pile" in the reactor community. A reactor with a Daniels pile was not built at the time because of the design preference chosen by Fermi and perhaps in part because one of the proposed moderator materials (beryllium oxide) was judged to be too toxic and costly and too hazardous for the 4% or so segment of the population subject to the fatal consequences of berylliosis resulting from minute human depositions of beryllium oxide. Current efforts to commercialize the new reactors deal exclusively with graphite as chosen moderator material. In spite of the clear priority of Daniels' proposal, the MPBR concept has been attributed by some authoritative sources to Rudolf Schulten who apparently independently proposed this idea during the 1950s in Germany for the purpose of using spheres that served simultaneously to hold the nuclear fuel as moderator material and for containment. Since the melting points of beryllium oxide, α-silicon carbide and carbon in the form of graphite are, respectively, about 2550°C, 2700°C, and 3700°C, any one of these materials is suitable for use in reactors with peak temperatures around 2000°C. The nuclear fuel feature is imparted by doping the solid carrier materials with appropriate amounts of uranium 235, whereas the beryllium and carbon contents determine the desired moderator effect. Containment is achieved by natural packing of the solid spheres which provides for ducting and core design. Reactor cooling is achieved by passing chemically inert helium through the reactor core. Replacement of spherical pebbles by hexagonal prismatic blocks (about 1 m tall and 30 cm long on flat faces) in the General Atomics design (called the GasTurbine-Modular Helium Reactor or GT-MHR and described in Sec. VII) is supposed to accomplish improved thermal conductivity and prevent mishaps caused by fractioned pebble pieces which may clog pebble inlet and outlet lines and thereby require costly system shut-downs and repairs.

A schematic diagram of the first operating and now decommissioned pebble-bed reactor (PBR) is shown in Fig. 2 and is reproduced from a publication by the European Nuclear Society.



Schematic representation of the spherical fuel element of a pebble bed reactor.



Nuclear power plant with pebble bed reactor

Fig. 2. Gas-cooled high-temperature reactor with a core of spherical fuel and moderator (graphite) elements. The decommissioned nuclear power plants (AVR in Jülich and THTR-300 in Uentrop) had pebble-bed reactors. The THTR-300 contained about 600,000 spherical fuel and moderator elements. The fuel elements consist of kernels of U-235 and thorium surrounded by a graphite matrix 6 cm in diameter.

A reactor dubbed the AVR (=Arbeitsgemeinschaft Versuchsreaktor) operated at Jülich (near Aachen in Germany) from 1966 to 1988 when it was decommissioned after line clogging caused by pebble break-up. The AVR was originally designed as a breeder for U233 from Th232 but containment of U233 was so effective that it was cheaper to extract U235 from natural uranium than to extract the bred fuel from the AVR. The coolant employed in the AVR was He which absorbed so few neutrons (i.e., possessed such low nuclear reactivity) that it could be used directly in the turbine generators for power production while exposing local workers to less than about 20% of the radiation levels associated with the operation of conventional light-water reactors.

China has a license to use the German AVR technology. Current R&D at Tsinghua University in Beijing is aimed at the construction of 200-MW_e modules beginning in 2007. A commercial unit using 30 modules by 2020 is expected to produce 6 gigawatts of electrical power. Planning still further ahead, 300 gigawatts of new nuclear power plants may be constructed by 2050 leading to the largest assembly of nuclear power plants anywhere in the world. The Chinese program is closely associated with future efforts to produce hydrogen as fuel for transportation use from water dissociation using nuclear reactor heat.

VI-2. Proliferation Resistance

The efficiency of burn-up of radioactive materials in the MPBR is so great that the resulting waste is a less concentrated supply for weapons-grade material than natural uranium. As the result, widespread use of these reactors will not contribute to weapons proliferation by leaving highly radioactive residues. The pebbles typically pass through the reactor 10 times before examination for residual radioactivity and further passage through the reactor or disposal as harmless waste. The following are requirements to produce 10 kg of weapons-grade Pu239 from MPBRs: destruction of 10 billion coated particles from about 1 million fuel elements followed by decomposition of these fuel elements using as yet unknown methods and reprocessing of the final Pu-U mixtures. Other methods for weapons diversions should be both simpler and cheaper.

VI-3. Current Programs

Eskom in South Africa is the world-wide technology leader on the development of the MPBR. Construction of a 110 MW_e prototype pebble-bed reactor at Koeberg was approved by the government in June 2003. Modules of 165 MW_e will be used in commercial installations, beginning at Pelindaba. The required uranium is to be imported from Russia via Durban. The use of needed numbers of small modules will mitigate siting and safety approval issues for large installations. When He is used as coolant, immediate access of the gas to the generator turbines is allowed because the chemically inert, non-radioactive gas may serve both as primary coolant and as working fluid for electricity generation. The use of non-polluting, chemically inert He as working fluid, in conjunction with pre-approved, passively safe modular nuclear reactors, supports the vision of rapid construction of passively safe, non-polluting, proliferation-proof, low-cost reactor modules for electricity generation anywhere in the world.

A closed-cycle system using He as the working fluid has been tested in South Africa. Whereas acceptable delivery of electricity to the distribution grid requires close synchronization between the generator system and required power delivery, the turbine compressors may be easily decoupled from the electricity-generation system. Power generation using He as coolant does not cause radioactive contamination in the He compressors, turbine, and recuperator. In South Africa, the generated nuclear power will be used in part to provide for peak power loads and to desalinate water. If attempts to obtain financial support are successful, then Eskom's business plan may include the construction of 10 MPBRs for local use and the export of 20 plants per year. There is strong local South African opposition from environmental groups to the Escom construction, utilization and business plans for MPBRs.

Development of MPBRs for transportation applications is being pursued in the Netherlands by Romara B.V. and also by Adams Atomic Engines (AAE) in the US.

VI-4. Inherent Safety Features in the MPBR Designs

With proper design, the laws of physics provide protection against a LOCA (=loss of coolant accident). As the MPBR units get hotter, U-238 atoms absorb more neutrons, which limits the extent of heating by reducing the power output of the reactor without human intervention. Furthermore, reactor core cooling is maintained by the inert circulating He cover. The system simply heats up to a designed temperature limit. This design feature was tested experimentally with the AVR reactor.

MPBRs have nuclear fuel elements consisting of continuously inspected pebbles. These "TRISO" fuel elements contain 4 of the 7 MPBR containment features. In a typical reactor, the pebbles have diameters of about 2 inches, weigh about 200 grams and hold 9 g of uranium. A 120 MW_e unit has 380,000 pebbles. The ceramic materials of which they may be made have melting points above the equilibrium operating temperatures of the reactor and serve as fuel-containment holders and moderators for the nuclear reactors. The outer sphere is a 60 mm diameter hollow pyrolytic graphite structure, serving as moderator. These spheres hold around 15,000 encapsulated moderator spheres with "seeds" consisting of ~0.5 mm kernels of fissionable materials in the form of uranium oxide or carbide. Each fissionable kernel serves as nuclear fuel and is surrounded by layers serving as containment spheres. Next to the fuel, the following structural features are found: first a layer of low-density pyrolytic porous graphite, then a layer of high-density pyrolytic graphite, next a silicon carbide layer and, finally, a pyrolytic graphite outer layer. The coated particles are encased in the graphite of the pebble (see Fig. 2). During operation, the pyrolytic graphite slows neutrons, and serves to maintain structural integrity. Since graphite is potentially combustible, the layer of SiC serves as a fire break. Fissionables are carbides or oxides of U, Th or Pu which cannot be burned.

Safety hazards (e.g. pebble break-up) have been noted and have led to both safety concerns and design changes to prevent accidents from occurring. As might be expected, opposition to nuclear reactors of any type has led to critiques of the absence of a containment structure (which is not really needed), the use of flammable graphite as a moderator, the assumption of maintaining "nearly perfectly" spherical pebbles during operation, and difficulties of fuel handling. All of these obviously manageable features plus pebble break-up, which led to the shut-down of the AVR, are cited as crucial flaws of the MPBRs but amount to little more than contrived rhetoric in view of what is already known.

VI-5. A Swiss Evaluation of Generation IV Nuclear Reactors

Table 1 is reproduced from Ref. 7 which deals with nuclear energy futures for Switzerland and is based on perspectives of 100 experts from the 11 GIF members (=Generation IV International Forum) comprising 10 countries plus the Euratom organization, which have agreed to cooperate in developing long-term nuclear energy supplies. Acceptability for development requires sustainability, low costs (economics) safety and reliability, proliferation resistance, meeting of clean-air standards, efficient fuel utilization, hydrogen and process-heat production, and actinide management. The leading contenders are listed in the first column of Table 1. Common Features of the six selected systems are rapid fast-neutron fuel cycles (with reprocessing), transmutation and recycling of actinides, high coolant temperatures, and ready process-heat utilization. Of the six identified systems, only the HTGR and SFR were identified as possessing likely application potential before about 2040 while opposition to the Superphénix in France and the impact of the Monju accident with the SFR in Japan reduce the field to the VHTR (HTGR), i.e. the type of system being developed at General Atomics. Nuclear fuel availability is estimated at about 120 years with a moderate increase in utilization efficiency. IAEA scenarios for nuclear energy production include capacity growth between 7 and 60% by 2030.

It is important to emphasize that operational lifetimes of nuclear reactors have increased from about 30 years at the time of construction to 50-60 years. For Switzerland, a reasonable development estimate is maintenance of current nuclear energy contributions during the next 25 to 35 years, which implies increases in domestic production above current levels equal to about 16% by 2035 and 47% by 2050.

Table 1. Overview of Generation IV Systems, slightly modified from Ref. 7.

System	Abbre-viation	Neutron spectrum	Coolant	Max. temp-erature, °C	Pressure	Fuel	Fuel cycle	Output, MW_e	Output
Gas-cooled fast reactor	GFR	fast	helium	850	high	U-238, MOX	in-situ closed	288	electricity, hydrogen production
Liquid metal (e.g., Pb) cooled fast reactor	LMFR	fast	Pb-Bi	550-800	low	U-238, MOX	closed, regional	50-150, 300-400, 1200	electricity, hydrogen production
Molten-salt reactor	MSR	epitherm al	fluoride salts	700-800	low	UF_6 in salt	in-situ closed	1000	electricity, hydrogen production
Sodium-cooled fast reactor	SFR	fast	sodium	550	low	U-238, MOX	closed	300-1500	electricity
Supercritical, water-cooled reactor	SCWR	thermal/ fast	water	510-550	very high	UO_2	open(th), closed(f)	1500	electricity
Very high-temperature gas-cooled reactor	VHTR‡	thermal	helium	1000	high	UO_2	open	250	electricity, hydrogen production

‡ Substantially equivalent to the HTGR of GA

VII. The General Atomics Modular Helium Reactor

The Modular Helium Reactor (MHR) is the culmination of almost 50 years of development of the High Temperature Gas-cooled Reactor (HTGR) concept. In both the pebble bed and prismatic block forms, the MHR meets the objectives established by the Generation IV (Gen IV) International Forum to define reactors that would have improved sustainability, economics, safety, waste management and proliferation resistance in order to form the basis for the "next generation" of nuclear power for generation of electricity and production of hydrogen for the world energy economy. As stated in Section VI, the MHR is a likely Gen IV design for deployment in the near term. The Modular Helium Reactor being developed by General Atomics in San Diego, CA uses stationary hexagonal prismatic blocks of graphite to hold the coated particle uranium fuel and, for the production of electricity, will use a closed-cycle gas turbine power-conversion system. Because of the high coolant outlet temperature (850 to 950°C), electricity can be generated at high efficiency (48-52%) or high-temperature process heat can be produced to make hydrogen from water by high-temperature electrolysis or thermochemical water-splitting.

VII-1. Historical perspective

The HTGR concept evolved from early air-cooled and CO_2-cooled reactors. To-date, seven HTGR plants have been built and operated (Table 2). The first was the 20 MWt Dragon test reactor in the UK, followed by the 115 MWt Peach Bottom I in the US and the 49 MWt AVR in Germany. These plants were followed by the 842 MWt Fort St. Vrain (FSV) plant in the US and the 750 MWt THTR plant in Germany. These early plants demonstrated electricity generation from HTGR nuclear heat using the Rankine (steam) cycle with achievable efficiencies of 35-40%. In addition to demonstrating the use of helium coolant (with outlet temperatures as high as 950°C) and graphite moderator, these early plants also demonstrated the utility of coated particle fuel, a fuel form that employs ceramic coatings for containment of fission products at high temperature, which is a key feature of HTGRs.

General Atomics used the HTGR technology from these early plants to design several 2000 – 3000 MWt, HTGR plants and orders were received for 10 of these plants. These orders were canceled, along with the cancellation of orders for a large number of other nuclear power plants, following the oil embargo in the early 1970s and the ensuing energy-conservative measures that temporarily reduced energy demand and the need for new electricity generation capacity.

Recently, two additional HTGR test reactors have been constructed and are successfully operating, namely, the 30-MWt High-Temperature Test Reactor (HTTR) in Japan and the 10 MWt High Temperature Reactor (HTR-10) in China (Table 2), with design outlet temperatures of 950ºC and 900ºC, respectively. These reactors are current research-phase designs.

Table 2. HTGR plants constructed and operated

Reactor	**Dragon**	**Peach Bottom**	**AVR**	**Fort St. Vrain**	**THTR**	**HTTR**	**HTR-10**
Location	UK	USA	Germany	USA	Germany	Japan	China
Power (MWt/MWe)	20/ –	115/40	46/15	842/330	750/300	30/ –	10/ –
Fuel Elements	Cylindrical	Cylindrical	Spherical (Pebble)	Hexagonal	Spherical (Pebble)	Hexagonal	Spherical (Pebble)
He Temp (In/Out°C)	350/750	377/750	270/950	400/775	270/750	395/950	300/900
He Press (Bar)	20	22.5	11	48	40	40	20
Pwr Density (MW/m^3)	14	8.3	2.3	6.3	6	2.5	2
Operation Years	1965-1975	1967-1974	1968-1988	1979-1989	1985-1989	1998 -	1998 -

The modular HTGR concept began in 1984 when the US Congress challenged the industry to investigate the potential for using HTGR technology to develop a “simpler, safer” nuclear power plant design. The goal was to develop a passively safe HTGR plant that was also economically competitive. In the low-power density HTGR cores, it was found that decay heat could be transferred passively by natural means (conduction, convection and thermal radiation) to a steel reactor vessel wall and then thermally radiated (passively) from the vessel wall to surrounding reactor cavity walls for conduction to a naturally circulating cooling system or to the ground itself. To maintain the coated particle fuel temperatures below damage limits during passive decay-heat removal, the core physical size had to be limited to about 200 MWt for a solid cylindrical core geometry. However, a 200 MWt power plant is not projected to be economically competitive. This estimate has led to the development of the annular core geometry to enable larger cores and, therefore, higher reactor powers while retaining full passive safety. The current reference core power is 600 MWt, and is known as the Modular Helium Reactor (MHR).

The latest evolution made for the purpose of economics has been replacement of the Rankine steam cycle power conversion system with a Brayton (gas turbine) cycle power conversion system to boost the thermal conversion efficiency to ~48%. The coupling of the MHR with the gas turbine cycle (GT) forms the GT-MHR, which has full passive safety characteristics and is projected to have more attractive economics than any other generation alternative. A 1,100 MWe power plant would consist of 4 of these 285 MWe modules.

VII-2. GT-MHR Design Description

The GT-MHR (Figure 3) couples a gas-cooled modular helium reactor (MHR), contained in one pressure vessel, with a high efficiency Brayton cycle gas turbine (GT) power conversion system (PCS) contained in an adjacent pressure vessel. The reactor and power conversion vessels are located in a below-grade concrete silo that provides high resistance to sabotage.

Key HTGR technology design characteristics of the MHR are the use of helium coolant, graphite moderator, and refractory coated particle fuel. The helium coolant is inert and remains single phase under all conditions; the graphite moderator has high strength and stability to high temperatures; and ***the refractory coated particle fuel retains fission products up to high temperatures***.

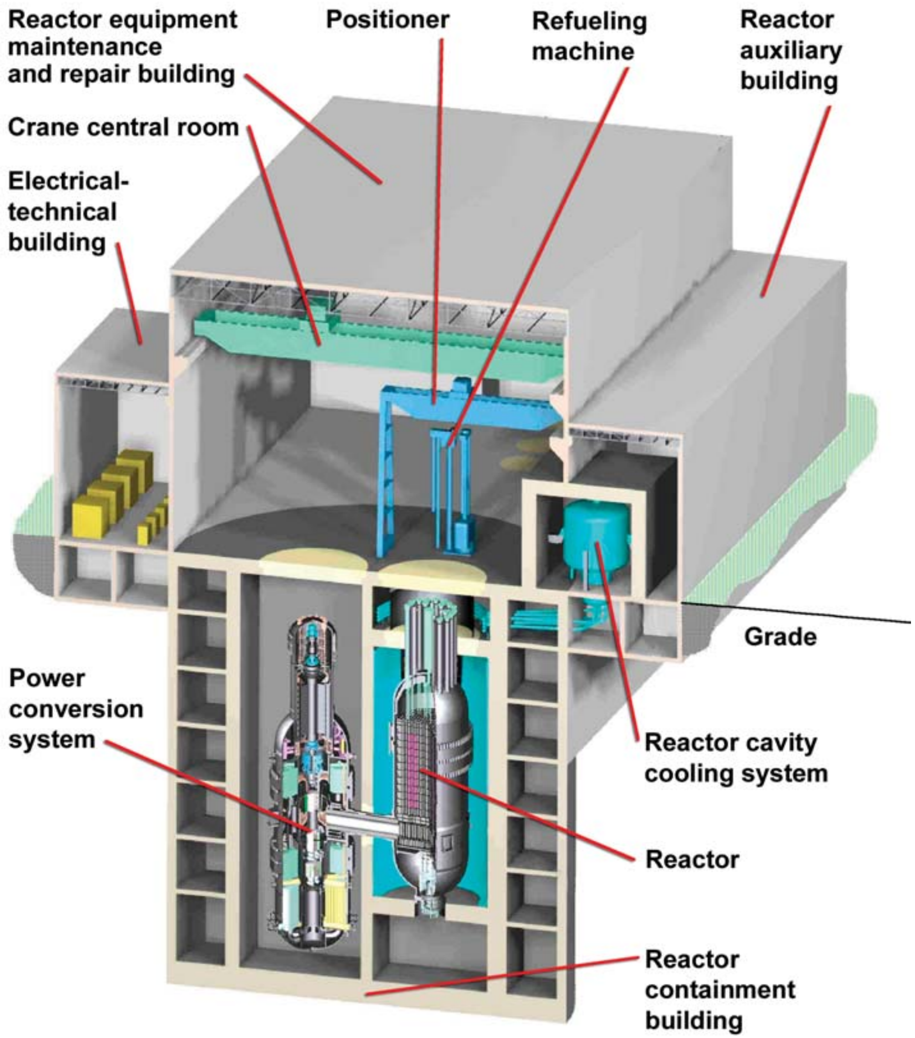


Fig. 3. The GT-MHR module.

Fuel - The MHR refractory coated particle fuel (Fig. 3), identified as TRISO coated particle fuel, consists of a spherical kernel of fissile fuel encapsulated in multiple layers of refractory coatings. The multiple coating layers form a miniature, highly corrosion- resistant pressure vessel and an essentially impermeable barrier to the release of fission products. The diameter of TRISO-coated particles is about 1 mm. The TRISO coated fuel particles are bonded into cylindrical fuel compacts and loaded into hexagonal fuel blocks (Figure 4). The TRISO coatings provide a high-temperature, high-integrity structure for retention of fission products to very high fuel burnups and temperatures approaching 2000°C. Normal operating temperatures do not exceed about 1250°C and worst-case accident temperatures are maintained below 1600°C.

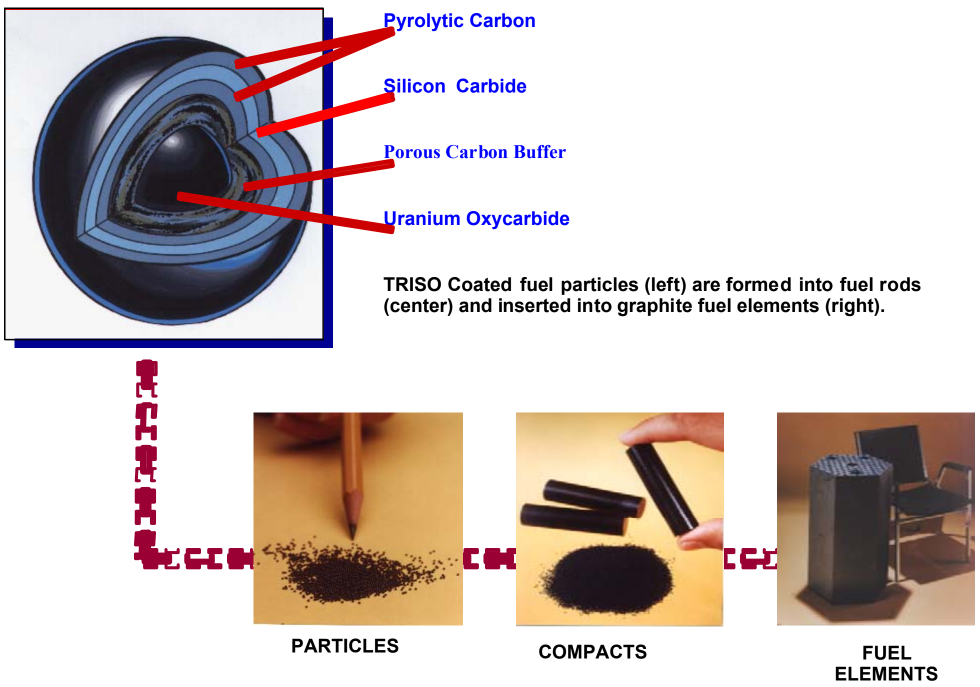


Fig. 4. The GT-MHR TRISO coated particle fuel.

Reactor - The MHR core consists of an array of hexagonal fuel elements. The fuel elements are stacked 10 high in an annular arrangement, enclosed in a steel pressure vessel. Control- rod mechanisms are located in the reactor vessel top head, and a shutdown cooling system for maintenance purposes only is contained in the bottom head. The helium outlet temperature is 850°. The hot outlet helium flows from the reactor core to the gas turbine power-conversion system (PCS) and returns to the reactor 490°C. All of the core components exposed to the heated helium are either graphite or thermally insulated from exposure to the high-temperature helium. Graphite has high strength, is difficult to ignite, readily extinguished, and has dimensional stability to very high temperatures (~2300°C). Because of the accident at Chernobyl in 1986, the role of graphite in reactor safety has received increased attention. However, the consequences of the Chernobyl accident were caused by massive fuel failure and not by graphite oxidation. High-purity, nuclear-grade graphite reacts very slowly with oxygen and would be classified as noncombustible by conventional standards.

Power-Conversion System - The GT-MHR direct Brayton cycle (gas turbine) power conversion system contains a gas turbine, an electric generator, and gas compressors located on a common vertical shaft supported by magnetic bearings. The use of the direct Brayton cycle to produce electricity results in a net plant efficiency of approximately 48%. This is ~50% higher than for current LWR nuclear power plants. Nominal full power operating parameters for the GT-MHR are given in Table 3.

Table 3. GT-MHR nominal full power operating parameters.

Reactor Power, MWt	600
Core Inlet/Outlet Temperatures, °C	491/850
Core Inlet/Outlet Pressures, MPa	7.07/7.02
Helium Mass Flow Rate, kg/s	320
Turbine Inlet/Outlet Temperatures, °C	848/511
Turbine Inlet/Outlet Pressures, MPa	7.01/2.64
Net Electrical Output, MW(e)	286
Net Plant Efficiency, %	48

VII-3 GT-MHR Safety

The GT-MHR safety design objective of complete passivity is achieved through a combination of inherent safety characteristics and design selections that take maximum advantage of the following characteristics:

(a) The helium coolant is in the gas phase, chemically inert, and does not become radioactive. Loss of coolant is not a catastrophic event.
(b) The graphite core has high heat capacity and therefore slow thermal response. It retains structural stability to very high temperatures. The reactor cannot melt down under any circumstances.
(c) The refractory coated particle fuel retains fission products at temperatures much higher than normal operation and postulated accident conditions. A large release of radioactivity is therefore not possible.
(d) The negative temperature coefficient of reactivity causes shut down of the core somewhat above normal operating temperatures. In any accident situation, the reactor shuts itself down with no action required by operators or equipment.
(e) The annular, low-power-density core (6.5 watts/cc) in an uninsulated steel reactor vessel is surrounded by a natural circulation reactor cavity cooling system (RCCS). The operation of this system prevents the reactor vessel from overheating under any circumstances.

The result is a reactor with total passive safety. Under any possible accident scenario, the public is protected. The large heat capacity of the graphite core structure is an important inherent characteristic that significantly contributes to maintaining fuel temperatures below damage limits during loss-of-coolant events. The core graphite heat capacity is sufficiently large that any heat-up or cool-down, takes place very slowly. A substantial time (on the order of days vs. minutes for other reactors) is available to take corrective actions to mitigate abnormal events and to restore the reactor to normal operations. If no corrective actions are taken, the reactor still shuts itself down and safely protects the public.

VII-4. GT-MHR Economic Competitiveness

The GT-MHR is projected to have economic advantages over any other source of base-load generation capacity. The economic competitiveness is a consequence of the use of the direct Brayton cycle power-conversion system and the passive safety design. The direct Brayton cycle provides high thermal conversion efficiency and eliminates extensive power conversion equipment required for the Rankine (steam) power conversion cycle. Reduction in the complexity of the power-conversion equipment reduces both capital and operation and maintenance (O&M) costs. The passive safety design eliminates the need for extensive safety-related equipment that also reduces both capital and O&M costs.

The capital cost for the reference GT-MHR plant containing four modules is projected to be ~1000 $/kWe. The construction period required for the first module of the GT-MHR plant is estimated to be ~3 years. The 20-year levelized busbar generation cost is projected to be 3.1 cents/kWh, including capital, O&M, fuel, waste disposition, and decommissioning.

VII-5. GT-MHR Environmental Benefits

The GT-MHR has significant environmental impact advantages relative to light water reactor plants. The thermal discharge (waste heat) from the GT-MHR is significantly less than that from the PWR plant because of its greater thermal efficiency. The GT-MHR produces less radioactive waste per unit energy produced because of high thermal plant efficiency and high fuel burn-up. Similarly, The GT-MHR produces less total plutonium and Pu^{239} (materials of proliferation concern) per unit of energy produced.

The deep-burn capability and high radionuclide containment integrity of TRISO particles offer potential for improvements in nuclear spent fuel management. A high degree of degradation of plutonium and other long-life fissile actinides can be achieved by the deep-burn capability. The spent fuel particles contain significantly reduced quantities of long-life radionuclides and very degraded fissile materials that can then be placed in a geologic repository with high assurance that the residual products have insufficient interest for intentional retrieval and will not migrate into the biosphere by natural processes before decay renders them benign. The TRISO fuel particle coating system, which provides containment of fission products under reactor operating conditions, also provides an excellent barrier for containment of the radionuclides for storage and geologic disposal of spent fuel. Experimental studies have shown that the corrosion rates of the TRISO coatings are very low under both dry and wet conditions, and indicate that the TRISO coating system should maintain its integrity for a million years or more in a geologically stable repository environment.

VII-6. Hydrogen Production Using the MHR

A significant "Hydrogen Economy" is predicted to limit dependence on petroleum and reduce pollution and greenhouse gas emissions. Hydrogen is an environmentally attractive fuel but contemporary hydrogen production is primarily based on fossil fuels, mostly natural gas. Hydrogen can be produced from nuclear energy by several means. Electricity from nuclear power may be used to separate water into hydrogen and oxygen by electrolysis. The net efficiency is the product of the efficiency of the reactor in producing electricity times the efficiency of the electrolysis cell, which, at the high pressure needed for distribution and utilization, is about 75%. If a GT-MHR with 48% electrical efficiency is used to produce the electricity, the net efficiency of hydrogen production could be about 36%. Electrolysis at high temperature, providing some of the energy directly as heat, promises efficiencies of about 50% at 900°C. Thermochemical water-splitting processes similarly offer the promise of heat-to-hydrogen efficiencies of ~50% at high temperatures. Thermochemical water-splitting is the conversion of water into hydrogen and oxygen by a series of thermally driven chemical reactions that could use nuclear energy as the heat source. Only water and high-temperature process heat are input to the cycle and only hydrogen, oxygen and low-temperature heat are outputs. All of the chemical reagents are regenerated and recycled. There are no effluents. An intermediate helium heat-transfer loop would be used between the MHR coolant loop and the hydrogen production system. Nuclear production of hydrogen using the MHR could be competitive at today's prices for natural gas. As the price of natural gas rises with increasing demand and decreasing reserves, nuclear production of hydrogen would become more and more cost-effective while producing no greenhouse gas emissions.

VII-7. Development Pathway

The path forward for deployment of the GT-MHR technology is necessarily a demonstration project because of a number of heretofore-unproven characteristics embodied in the design: the passive safety design approach, the fuel operating conditions (burnup, fluence, temperature), and the power-conversion system design. These development needs make attempts to obtain project financing by private industry extremely difficult. The potential benefits of the GT-MHR for the generation of electricity (passive safety, good economics, reduced environmental impact, and high proliferation resistance), coupled with the potential for efficient production of hydrogen, provide significant incentives for a government-sponsored demonstration program

VII-8. Conclusions

The GT-MHR offers several advantages. It is meltdown-proof and passively safe. Use of the Brayton cycle helium gas turbine in the GT-MHR provides electric generating capacity at a net plant efficiency of about 48%, a level that can be obtained by no other nuclear reactor technology. The coated particle fuel provides a superior spent fuel waste form for both long-term interim storage and permanent geologic disposal, retaining their integrity in a repository environment for hundreds of thousands of years. Relative to water reactor plants, the GT-MHR thermal discharge is about 40% less and the actinide (Plutonium) production is 50% less per unit electricity produced. The GT-MHR spent fuel has very high proliferation resistance. The GT-MHR levelized busbar generation cost is evaluated to be less than that for competing water reactor and gas-fired combined cycle plants. The high temperature capability of the MHR is ideally suited as an energy source for efficient production of hydrogen for the future Hydrogen Economy. These advantages make the GT-MHR an ideal candidate for a near-term demonstration project that will lead to the ideal “next generation” of nuclear power.

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