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Status of high temperature gas-cooled reactor technology

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Abstract. Over a period spanning more than half-a century, the High-Temperature Gas-cooled Reactor (HTGR) design has evolved from early experimental prototypes with single-coating fuel to more recent modular designs featuring TRISO fuel and a direct-cycle gas turbine design, promising enhanced safety and improved economics. In this paper, the current status of the technology is reviewed, starting with a brief introduction and a descriptive history of the evolving design. This is is followed by an overview of the special fuel and core design aspects, including core physics, thermal-hydraulics, reactivity control and fuelling schemes. An overview of safety performance is also presented, followed by an outline of the various power conversion unit layouts, the Brayton cycle main characteristics and the potential process heat applications of this particular design. A brief overview of HTGR-related IAEA activities as well as current international projects is also presented.

1. Introduction

Gas-cooled reactor design concepts have been evolving since the 1940's and in recent years there have been a surge of global interest in their modular variants due to their promising features of enhanced safety and improved economics. Modular HTGR designs are currently considered one of the leading reactor concepts considered for any future nuclear power plant deployment. There are currently various related projects around the world and the IAEA is following their progress, coordinating research and facilitating information exchange among its Member States. In the next section, we present an outline of the evolving design history and in sections 3-5, the specific fuel design, core design and safety performance aspects are overviewed, respectively. Section 6 includes an outline of the various power conversion unit layouts and the Brayton cycle characteristics and in section 7, potential process heat applications are noted. Finally, the status of current HTGR activities and international HTGR projects are overviewed in section 8.

2. History of gas-cooled reactor development

Early gas-reactors were used for plutonium production and aimed at military applications. They were basically natural uranium-fuelled piles, graphite-moderated and air-cooled. Commercial gas-cooled nuclear power for electricity production started in 1956 with the operation of 4 units at Calder Hall, UK. The design, which came to be known as Magnox, featured carbon-dioxide as pressurized coolant and magnesium alloy cladding. Thermal efficiency was still limited at 20% or so and later designs switched to stainless steel cladding, enriched uranium oxide fuel and higher CO_2 pressures and temperatures in what came to be known as Advanced Gas Reactors (AGRs), in order to raise thermal efficiency.

The move to helium cooling and ceramic coated particle fuel design came with the Dragon reactor proptotype which was in operation at Winfrith in the UK between 1965 and 1976, as an OECD project. Featuring a steel pressure vessel, coated fuel particles of highly-enriched uranium-thorium carbide and a helium outlet temperature of 750 °C, the 20 MWt prototype served as a test bed providing valuable information on fuel, material and component behavior under high-purity helium conditions.

Another successful prototype which also supplied 15 MW of electricity to the grid was the AVR reactor which was operated at Juelich in Germany for 21 years (1967-1988). A gas core outlet temperature of 950 °C was achieved. A steam generator located above the core was the interface between the primary and secondary loops. This particular design consisted of 100,000 coated fuel spheres travelling downwards through the core inside a graphite reflector pot, featuring what came to be termed as the Pebble Bed Modular Reactor (PBMR) concept. During the last few years of its operation, the AVR was used to perform tests related to HTGR performance and safety. In the US, Peach Bottom Unit 1 was the first HTGR demonstration prototype owned and operated by Philadelphia Electric, now Exelon. Rated at 40 MWe, the unit was operated between 1967 and 1974 and early operation was plagued by cracked fuel elements, which prompted a modification in the fuel particle design and the introduction of an additional buffer coating.

In the eighties, two distinctive HTGR design categories have emerged, one being the pebble-bed type and the other the prismatic block type. Design features include a pre-stressed concrete reactor vessel and a more advanced form of the coated fuel particle design, known as TRISO. Design and power was raised to 300 Mwe. The German Thorium High Temperature Reactor (THTR-300) represented the first category, while the US Fort St. Vrain unit represented the second. Licensing and funding problems led to the early closure of the first, while helium circulator bearing problems plagued the operation of the latter [1].

In the wake of the Three-Mile Island and Chernobyl accidents, modular HTGR designs featuring reduced power, low power density and passive safety features have been advocated [2]. The General Atomics (GA) MHTGR design was rated at 350-450 MWt while the German HTR series design was rated somewhat lower at 200-300 MWt.

In recent years, with advances in magnetic bearings, gas turbine and high-temperature component technology, a direct-cycle version of the MHTGR design is being considered for deployment. The potential of reaching 50% or more in thermal efficiency in a Brayton cycle operation mode, together with the passive safety features, promise to make modern HTGRs both competitive and safe.

3. Fuel design aspects

HTGR fuel is in essence a spherical kernel of fissile and fertile coated particles, usually in the form of Uranium or Plutonium oxide or oxy-carbide. The particles, varying in diameter from about 200 to 500 microns, are known in the industry as TRISO particles to signify the three levels of coating (Fig. 1). A low-density sacrificial zone provides a 100-micronbuffer to accommodate fission gas and is surrounded by an inner pyrocarbon coating, a silicon carbide coating and an outer pyrocarbon coating, each about 40 microns in thickness. The three coatings provide a corrosion-resistant pressure vessel and a barrier to fission product release. In the General Atomics design, the TRISO particles are bonded within a graphite matrix to form cylindrical compacts 13 mm in diameter and 51 mm-long. Approximately 3000 of these compacts form a hexagonal graphite fuel element, the same type used in Fort St. Vrain. In the alternative PBMR design, the particles are also inbedded in a graphite matrix but in this case, in the form of spherical pebbles, each 6 cm in diameter, with hundreds of thousands of these filling the core.

The design of TRISO fuel features high termal capacity and high-temperature stability, with a low probability for coating failure below a temperature of 1600 °C. Irradiation tests conducted in material testing reactors and operating gas-cooled reactors to bunups of up to 15.6% fissions per initial metal atom (FIMA) showed minimal probability for particle failure (~ 10^{-4} to 10^{-5}). Post-irradiation heating experiments have also been carried out on TRISO fuel at high temperatures (1600 °C and above) with heating periods as long as 500 hours, and fission gas release was monitored.. While low-burnup fuel showed minimal release up to a temperature of 2200 °C, higher-burnup samples (~8% FIMA) showed a noticeable increase in fission gas release at heating temperatures above 1600 °C, as shown in Fig. 2 [3]. Several mechanisms giving rise to coating failure and radioactivity release include diffusion due to the temperature gradient, fission product interation with the coating material and fission gas pressure effects. With this in mind, and as will be discussed in Section 5, the value of 1600 °C is currently taken as an upper design limit for maximum fuel temperature under normal and abnormal HTGR operation while current research efforts are focusing on the feasibility of further stretching the envelope of irradiation and temperature stability to improve HTGR fuel performance.

HTGR fuel can be Uranium or Plutonium based and can include fertile material such as Thorium, allowing improved fuel utilization and the possibility of burning stockpiles of civilian and military Plutonium. While early designs were inclined towards higher enrichments of 20 % or more, recent trends have moved towards lower enrichment of 8% or less, for improved safety.

4. Core design aspects

The combination of coated fuel particle design, graphite moderator and helium coolant, gives the HTGR design its distinctive features of low power density, high gas temperature and high burnup operation. HTGRs are currently designed with a discharge burnup of up to 100,000 MWD/t, about twice that of light-water reactors, and a core outlet coolant temperature of up to 950 °C, far above what is permissible with light-water cooling.

4.1. Core physics design

The combination of graphite as moderator with its low absorption cross-section and helium as an inert gas coolant with negligible absorption, helps improve HTGR neutron economy. The core is usually annular in shape, which helps flatten the radial power distribution. It features a central graphite refelector column and radial as well as axial outside reflector columns. The inner columns are usually designed as replaceable due to their exposure to high neutron fluence. Two other related core physics features are worth noting for the HTGR design. One is the larger migration area of graphite. While this feature implies an increase in core size, it also allows a fairly low and benign power density , as low as 2-3 MW/m3, which is far below that of light water reactors. Another is the characteristically negative core temperature coefficient which increases in magnitude at lower enrichments and higher burnup. Both of these core physics features are put to good use, as will be explained in 4.2 and 4.3.

Serious efforts have been directed in recent years towards validation of HTGR core calculation methods, by benchmarking them against experimental tests such as those at the Proteus critical facility in Switzerland, the HTTR reactor in Japan and the and HTR-10 reactor in China. The codes used vary from detailed Monte Carlo methods to a combination of cell transport and core diffusion models. Streaming effects and double-heteregeneities at the fuel cell level are some of the challenges encountered in these calculations [4]. While results naturally vary from code to code, it is worth noting that some of the methods used in the HTR-10 benchmark predicted the core critical loading within 1% in terms of number of pebbles needed.

4.2. Core thermal design

Taking advantage of the low power density and high core thermal capacity of the HTGR design, modular low-power designs in the range below 600 MWt have been conceptualised to ensure passive decay heat removal under all normal and abnormal operating conditions. A large height-to-diameter ratio core, an annular core geometry and a steel pressure vessel design also help in this regard. Another related feature of this inherently-safe thermal design is the Reactor Cavity Cooling System (RCCS), which is located in the concrete structure external to the reactor pressure vessel and ensures passive removal of core residual heat in the event of unavailability of normal cooling. There is however a design penalty to pay with this, in the form of a certain percentage of heat loss, in the order of 5%.

Helium is the choice for coolant since it is an inert gas, with no affinity for chemical or nuclear activity and radiactivity transport in the primary circuit is therefore minimal under normal operation. The gaseous nature moreover avoids problems related to phase change and water-metal reactions and therefore improves safety. It also allows the use of a direct Brayton cycle, improving thermal efficiency and economics.

The high pressure gas flow in the core is guided through borings of the side reflector, travelling upwards towards the cool gas plenum, cooling the external reflector regions and the top core support structure before entering the core in the downwards direction (Fig. 3). The gas is heated up to a temperature of 800 to 950 °C by the time it reaches the lower hot gas plenum and exits the core.

4.3. Reactivity control and fuel cycle design

In current modular HTGR designs and in perticular PBMRs, control rods for safety and operational purposes are located in the outside reflector region, in order to limit their high-temperature exposure. This of course has direct influence on their worth and annular small-diameter cores are usually designed with this in mind. The use of low-enriched fuel and online refuelling imply a low-reactivity inventory and improves the safety features of the control system.. Power can be conveniently controlled by varying the helium inventory in the primary loop, affecting core flow and taking advantage of the negative temperature coefficient feature in the range of 25% to 100% power. This is an attractive feature for load-following operation.

While prismatic core designers have sticked with a fixed few-year refuelling interval strategy, the pebble bed designers have been favoring the use of online refuelling. Fuel balls are loaded at the top of the core and discharged at the bottom. In the Once-Through Then Out (OTTO) scheme, the balls transit the core only once and are not recycled, while in the alternative multi-pass scheme, they are recycled a few times through the core, until they reach their target burnup limit. The latter scheme is usually preferrable, due to its power-flattening features [5]. Another important feature in the HTGR design is its potential for Plutonium burning and fertile fuel conversion, thanks to its spectrum features. For example, the Fort Saint Vrain design used Thorium as fertile material, and the current GT-MHR design uses Plutonium as part of its fuel scheme [5]. This of course enhances fuel utilization and fuel cycle economics.

5. Safety performance aspects

Transient events affecting modular HTGR performance can be classified into 2 main parts, reactivity-initiated events and loss of flow events with or without depressurization. A combination of these events have been analyzed by hypothetizing an unscrammed core heatup accident scenario. Within minutes, a rapid decrease in core power is achieved, due to the strongly-negative Doppler effect, followed by slow core heatup. Heat transfer to the RCCS takes place in the form of conduction and radiation, with the latter dominating. Core temperatures show a gradual increase on the scale of hours after the initiation of the event before dropping again, without exceeding the limit of 1600 °C beyond which fuel particle integrity may be compromised [6].

While water-cooled reactor safety issues such as Departure from Nucleate Boiling (DNB) and Pellet Clad Interaction (PCI) are not of concern for HTGRs, there are other safety concerns to address instead such as air ingress which may oxidize and weaken the graphite in the core. However, only in the unlikely event of multiple ruptures in the pressure vessel together with two openings in the concrete surrounding it, can enough oxygen be drawn into the core, leading to massive graphite oxidation and potential fuel particle failure. Even with this unlikely combination of simultaneous events, several days can be allowed to seal the concrete and stop air flow into the core[2].

Water ingress in the core has also been investigated as yet another safety concern, due to its positive reactivity contribution to th core. However, the core physics design and the amount of heavy metal loading limits the effect of water ingress well within safety limits [2]. For all possible accidents, reactor shutdown is ensured by three independent absorber systems and stopping helium circulation in the core. Any one of these procedures is sufficient to stop the fission process.

6. Power conversion unit and Brayton cycle design

While early versions of modular HTGR designs were based on the Rankine steam cycle, current plant designs have adopted the Brayton cycle, mainly to improve thermal efficiency. Values close to 50% are within reach of this design, making it even more competitive vis-a-vis fossil fuel combined-cycle designs. The Power Conversion Unit (PCU) includes all equipment necessary to convert the core thermal energy into mechanical and then electrical energy. Included in the PCU are the gas turbine which is connected to a generator, turbo-compressors for helium pressurization, a pre-cooler and intercooler for compression temperature control and a recuperator which is a regenerative helium-to-helium heat exchanger (Fig. 4). The precooler, intercooler and recuperator are all essential to improve cycle efficiency.

In the direct-cycle design, a cross-duct connects the reactor vessel with the power conversion vessel, while in the alternative indirect-cycle design, an intermediate heat exchanger is provided as an interface between primary and secondary circuits. The latter design is preferred by some designers, since it adds an extra barrier against potential radioactive contamination of the turbo-machinery. This of course is expected to carry some penalty in terms of cycle efficiency. Other variations of the design include single-shaft vs multi-shaft systems for mounting turbines and compressors, horizontal vs vertical turbo-machinery and external vs submerged generators. The HTGR Brayton cycle development has coincided with advances in turbo-machinery components such as large-size gas turbines, efficient compact recuperators and large magnetic bearings. It is expected that with further advances in high-temperature material technology, higher efficiencies, as high as 60% may be achievable in the future.

In the Brayton cycle, High-temperature helium is expanded in the power gas turbine driving the generator and then enters the recuperator where it gives up much of its heat to the helium returning to the core. From the recuperator, the helium enters a precooler, discharging heat to an external heat sink, before entering the first stage of compression. Usually, two stages of compression are used and an intercooler in between the two compressors is provided to take away the resulting heat of compression, thereby improving efficiency. Finally, the compressed helium is preheated in the recuperator before being returned to the core in the case of direct-cycle designs or to the intermediate heat exchanger for indirect-cycle designs.

7. HTGR process heat applications

Besides electricity generation, the potential exists for the HTGR design to provide both high-temperature and low-temperature process heat for various applications. For example, the production of hydrogen using thermo-chemical splitting or steam methane reforming involves an endothermic reaction. High-temperature heat increases the efficiency of the process and can be readily provided by the HTGR[7]. Another potential application is thermal desalination processes which rely on low-temperature heat and which can also benefit from the available waste heat at the precooler level, promising a signficant reduction in the cost of potable water production. HTGR operation in co-generation mode can boost the overall thermal effciency to 80% or more, which is a significant and attractive feature[8].

8. IAEA activities and HTGR projects[9]

8.1. IAEA Activities

IAEA HTGR activities are conducted with input and feedback from Member States, through the Technical Working Group on Gas-Cooled Reactors (TWG-GCR). The main role of the Agency is to coordinate research activities and facilitate information exchange. Several Coordinated Research Projects (CRPs) have been conducted on safety-related physics, after-heat removal under accident conditions, fuel and fission product behavior and heat utilisation systems. Two major CRPs are currently active. The first is focusing on benchmarking core physics and thermal-hydraulic calculation methods against experimental results, with the objective of reducing design uncertainties. Results of a first set of benchmarks have been published, while work continues on a second set. Meanwhile, a second CRP is focusing on high-temperature coated fuel particle technology, with the objective of gaining a better understanding of the fabrication process and performance envelope under normal and accident conditions. Several member states with interest in HTGR technology, are participating in these two IAEA projects.

8.2. PBMR Project

The South African electrical utility ESKOM joined by BNFL and the Industrial Development Corporation (IDC) of South Africa continues its efforts to license and construct a PBMR demonstration plant in Cape Town. The design basically follows that of the German HTR-MODULE and is currently rated at 400 MWt/168 Mwe. the long-term design objectives include a capital cost of ~US$1000/Kwe, a construction period of 24 months and an emergency planning zone of 400 m. Feasibility studies and safety reviews have now been completed and preparations are under way for assuring lead components, licensing procedures and international financing.

8.3. NGNP Project

In the US, the Next Generation Nuclear Plant (NGNP), is a project supported by DOE, as part of its 2010 nuclear power initiative. It entails the design and construction of a high-temperature gas-cooled reactor for co-generation of electricity and hydrogen. The plant, rated at 600 MWt with a 1000 °C core outlet temperature, is expected to start operation around 2016.

8.4. GT-MHR Project

The GT-MHR development program is a venture involving MINATOM of Russia and General Atomics (GA) of the US, among others. The proposed prismatic design, rated at 600 MWt/293 Mwe is meant to be utilised for destroying inventories of weapons plutonium, with a long-term goal of commercial development. While research and development work continues, the realization of the project depends on ongoing bilateral and financing negotiations involving the project stake holders.

8.5. HTTR Project

In 1987, the Japanese Atomic Energy Commission recommended the construction of a High-Temperature Test Reactor (HTTR), with an eye on high-temperature process heat applications. Construction was finished in 1996 and first criticality achieved in 1998. rated at 30 MWt, the test reactor features an annular prismatic fuel design, with an intermediate heat exchanger equipped to supply process heat. The HTTR is currently undergoing safety demonstration tests. Efforts are also planned to design a power plant prototype, the GTHTR-300, with hydrogen production in mind.

8.6. HTR-10 Project

HTR-10 is a 10 MWt PBMR-type high-temperature test reactor operated by the Chinese Institute of Nuclear Energy Technology (INET) in China. Local fuel fabrication has been developed based on German fuel technology and criticality was reached in late 2000. A steam power cycle is to be used in the first phase, followed by a gas turbine cycle at a later phase. Plans also exist for constructing a power plant prototype, the HTR-PM, within this decade.

8.7. European Technology Network

In Europe, an HTGR technology network (HTR-TN) has been established in 2000, supported by the European Commission and several projects have been launched addressing core physics and fuel cycle, fuel irradiation and testing, material and component technology as well as safety and licensing issues. Several industrial, research and educational institutes from several european countries are taking part.

9. Concluding remarks

There is growing global interest in modular HTGRs, due to their attractive features of enhanced safety and economic competitiveness. TRISO fuel provides an effective barrier against radioactive release and the Brayton cycle allows high thermal efficiencies to be achieved. The design is considered by many as one of the leading candidate concepts for future nuclear power deployment.

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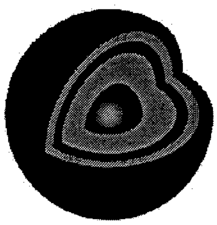

FIG. 1. TRISO coated fuel particle cross-section

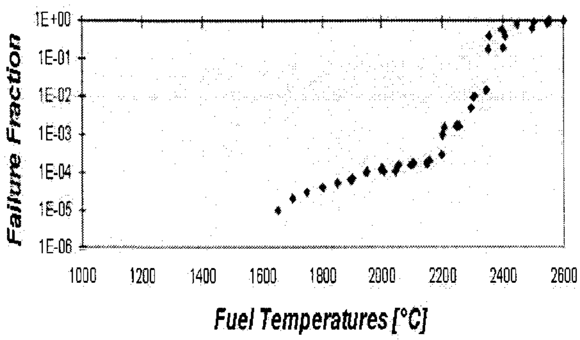

FIG. 2. TRISO fuel performance as function of temperature

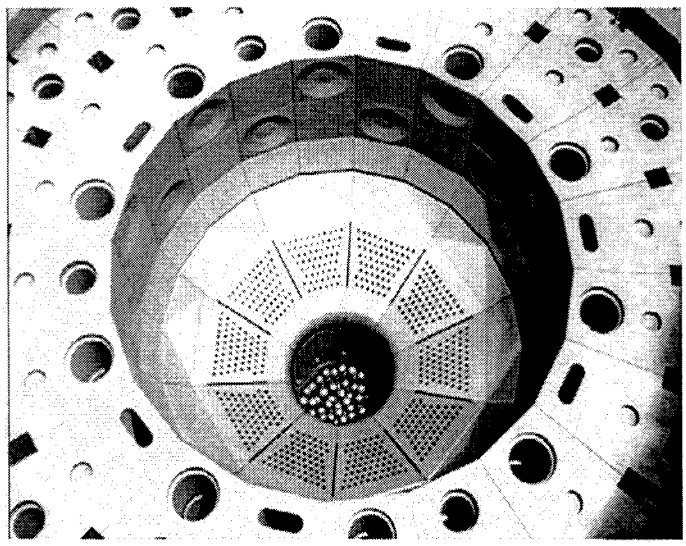

FIG. 3.HTR-10 PBMR-type bottom and side reflector cross-section

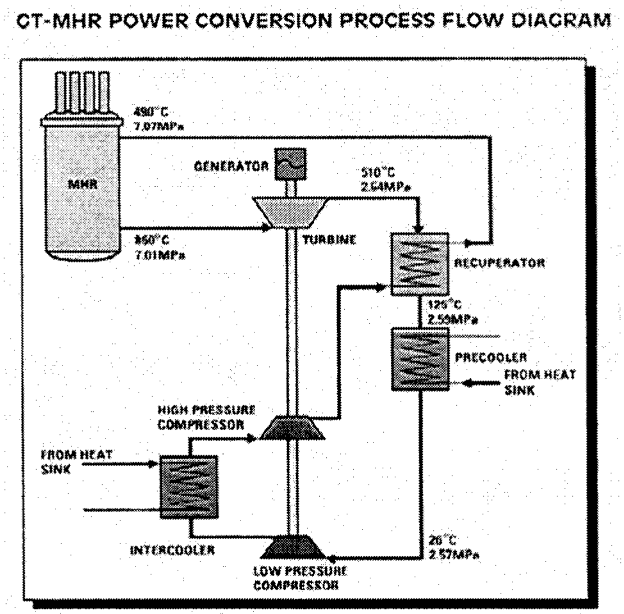

FIG. 4. GT-MHR sketch of power conversion unit layout