Gas-Cooled Reactor, Advanced Heavy Water Reactor

K.S. Rajan
Professor, School of Chemical & Biotechnology
SASTRA University
Table of Contents

1 GAS-COOLED REACTOR (GCR) ............................................................................................................. 3
2 ADVANCED GAS-COOLED REACTOR (AGCR) .................................................................................. 4
3 HIGH-TEMPERATURE GAS REACTOR (HTGR) ................................................................................. 5
4 ADVANCED HEAVY WATER REACTOR (AHWR) ............................................................................. 6
5 FAST REACTORS ............................................................................................................................... 7
6 REFERENCES/ADDITIONAL READING ......................................................................................... 7
In this lecture, we shall discuss about the working of Gas Cooled Reactor (GCR), High Temperature Gas Reactor (HTGR), Advanced Heavy Water Reactor (AHWR) and Fast Reactor.

At the end of this session, the learners will be able to
(i) understand the working of gas-cooled reactor
(ii) distinguish the power cycle of gas-cooled reactor from that of liquid-cooled reactors
(iii) distinguish between gas-cooled reactor and high-temperature gas reactor

1 Gas-Cooled Reactor (GCR)

A schematic diagram of a gas-cooled reactor is shown in Figure 1. This is a type of nuclear reactor that uses a gas as the coolant. Mostly CO$_2$ is used. Graphite blocks are used as moderator, within which channels are made for housing fuel rods. Control rods are inserted into the graphite blocks. Channels are established between the graphite blocks for the flow of coolant. Natural uranium is used as the fuel while cladding is made of a magnesium alloy called magnox. This reactor derives its name from the alloy used for cladding, ‘magnox’. The coolant gas (CO$_2$) is supplied by a gas circulator and enters the core from bottom. Gas flows through the coolant channels between the graphite blocks. As the gas moves up through the core, it gets heated up and leaves the top of the core at high temperature.

![Schematic diagram of gas-cooled reactor](image)

**Fig 1. Schematic diagram of gas-cooled reactor (Redrawn from Ref. [6])**
This high temperature gas exchanges heat with water in a heat exchanger, resulting in the production of steam, which runs the turbine. The spent steam is condensed and returned back to the heat exchanger, while the gas returns to the reactor. The heat exchanger is located outside the pressure vessel and the containment. Magnox reactor finds extensive use in United Kingdom.

2 Advanced Gas-Cooled Reactor (AGCR)

A major difference between the GCR and AGCR is the location of heat exchangers (2 in number) within the reactor pressure vessel and the containment in AGCR. The other differences are the use of stainless steel fuel cladding and the use of enriched fuel (2.5 – 3.5 % U-235). Higher energy per unit mass of the fuel is obtained owing to higher fuel enrichment. Due to the integration of heat exchanger inside the reactor vessel and in a pool of hot gas, higher thermal efficiencies are achieved compared to that in a GCR.
3 High-Temperature Gas Reactor (HTGR)

High-temperature gas reactor uses Helium gas as the coolant, owing to its chemical inertness and better thermodynamic properties. Fine particles of UO$_2$ (~ 0.5 mm) are coated with layers of porous carbon, pyrolytic carbon, silicon carbide and pyrolytic carbon again. Coating with silicon carbide provides protection against melt down up to 1600 °C. The diameter of coated particles is about 0.92 mm. This type of fuel is called TRISO (Tristructural-isotropic) fuel.

If these coated particles are embedded in spherical graphite matrices of 60 mm diameter to form pebbles, and used in the reactor with the voids created by the packing of pebbles making the path for coolant (helium gas) flow, the reactor is called Pebble Bed Reactor (PBR). The moderator (graphite) is integrated with the fuel such that the separation of moderator from fuel never occurs. There are no control rods in these reactors for power control. Increase in temperature of the core causes the fission rate to decrease by a phenomenon called ‘Doppler broadening’. This ensures that the heat generation does not exceed beyond design limits.

The heat extracted by helium may be directly used to operate a high temperature gas turbine or can be used to generate steam by extracting heat from helium in a heat exchanger/steam generator. If the fuel particles are compacted and placed in a graphite block (as in a Gas-Cooled Reactor), the reactor is called prismatic-block-gas-cooled reactor. An advantage of PBR over prismatic block reactor is the facility for continuous replacement of fuel in PBR. South Africa is the pioneer of PBR technology.
There were two commercial HTGR plants: (i) 330 MWe plant at Fort St. Vrain in USA and (ii) 300 MWe (THTR-300) plant at Germany. These were in operation from early 1970 till 1990 and early 1980 till 1990 respectively. The two prototype HTGR in operation, as on 2011, were HTTR (Japan) and HTR-10 (China) with the capacities of 30 MWt and 10 MWt respectively.

4 Advanced Heavy Water Reactor (AHWR)

This reactor is an Indian version of next generation Heavy Water Reactor. Boiling water is the coolant while heavy water is the moderator. The proposed reactor design is of vertical, pressure tube type rated at 920 MWTh and 300 MWe. This reactor is designed to utilize U-233 as the fissile isotope, the initial loading obtained from breeding in fast breeder reactor by the nuclear transmutation of Th-232 to U-233. At the centre of a typical fuel cluster is a displacer rod containing dysprosia in a zirconia matrix along with a water tube for the injection of water from Emergency Core Cooling System (ECCS) on to fuel pins directly. The details of a typical ECCS will be discussed in the later modules with reference to BWR and PWR. Surrounding the central rod are two rows of (Th-232 & U-233) oxide fuel pins (30 in number, 11.2 mm diameter) and a row of (Th-232 and Pu-239) oxide fuel pins (24 in number, 11.2 mm diameter). This configuration ensures a slightly negative reactivity coefficient similar to that of BWR. The pressure tubes are 120 mm in diameter. The number of coolant channels is ~ 452. The pressure in the reactor is about 70 bar, with the calandria being 8000 mm in diameter and 5000 mm in length.

By using a mixture of U-233 and Th-232 in the core, simultaneous breeding (Th-232 to U-233) and burning of U-233 can be achieved with thermal neutrons. The capture cross section of Th-232 for thermal neutrons is high to facilitate transmutation and breeding. Hence this proposed reactor is the first-of-its-kind thermal breeder. This reactor facilitates utilization of Thorium that is found in abundance in India.

Heat released during fission is removed by the boiling of light water under natural circulation, making it as an inherently safe passive feature. This circumvents the need for coolant pumps and hence brings about a reduction in cost of primary coolant system. Use of a large water tank on top of the primary containment called Gravity Driven Water Pool (GDWP) is a unique feature of this design that adds to the safety features of the reactor.

AHWR, being a reactor cooled by natural circulation, power density is lower. Hence lower level of neutron flux is required, which serves to increase fertile to fissile fuel conversion.
The important safety features of AHWR are:

(i) Negative void coefficient of reactivity, implying lower heat generation with increase in void fraction (volume fraction occupied by steam)
(ii) Passive safety systems that do not require operator intervention
(iii) Presence of a large 6000 m$^3$ of water in the form of gravity driven water pool
(iv) Heat removal from core by natural convection during normal operation and shut down

5 Fast Reactors
Fast reactors are the ones that utilize fast neutrons for fission. Modules 12-15 exclusively deal with sodium cooled fast reactors, their design and operation. Hence we shall discuss about this reactor in brief during this lecture.

The reactor is devoid of any light nuclei which act as moderators. The use of fast neutrons necessitates the use of Pu-239 as the main fissile isotopes at high enrichment levels. The reactor core is hexagonal, with fuel pins consisting of mixed oxide (PuO$_2$-UO$_2$). The power cycle is three-coolant, steam cycle. Sodium is used as coolant in two cycles, while water is used as coolant in third cycle. Use of liquid sodium as coolant permits reactor operation at higher temperatures and subsequently steam generation at higher temperatures. Primary sodium is the sodium in contact with the core and hence is radioactive, while secondary sodium is the one in thermal contact with primary sodium on one side of the loop. On the other side of the loop, secondary sodium (non-radioactive) is in thermal contact with water for steam generation. Two sodium loops are used to prevent even accidental contact of radioactive sodium with water. Steam-water cycle is similar to that used in PWR or PHWR.

6 References/Additional Reading