HIGH TEMPERATURE GAS COOLED REACTORS

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Summary

Generally nuclear fuel is less costly than fossil fuels so there is not the same motivation to develop high efficiency thermodynamic cycles for nuclear plants as there is for conventional fossil plants. Nevertheless good thermodynamic efficiency will reduce fuel costs and benefit the economic production of electric power. The high temperature reactor, by supplying heat to the thermodynamic cycle at a high temperature, allows a higher cycle efficiency, compared with other reactors particularly water cooled reactors, to be achieved. This is possible by using a ceramic fuel and a graphite moderator, which can sustain much higher temperatures than metallic alloys, and an inert gas as a coolant. Generation of steam can therefore be at the limiting temperatures and pressures of conventional fossil fuel fired boilers which use hot combustion gases as a heat source.

Although developed in the United Kingdom, Germany and the United States and reaching the level of commercial operation in the latter they were not favored above other more developed reactor systems. In more recent years however interest in high temperature gas cooled reactors has been revived due to their safety features and possibility of modular construction.

The first high temperature gas cooled reactors had fuel particles imbedded in graphite blocks through which helium coolant flowed. The hot helium then passed through a steam generator where its heat was transferred to a conventional steam cycle. An alternative version had the fuel particles imbedded in graphite spheres in a packed bed through which the helium flowed. Since it is advantageous, with regard to heat transport, to pressurize the helium to increase its density, it was considered feasible to use the hot high pressure helium directly in a gas turbine; hence the use of the helium coolant as the fluid in a thermodynamic cycle. Replacing the large steam generators and heavy steam turbines with smaller and lighter gas turbines enabled a modular design to be conceived. Significant components could be built at the manufacturing works and shipped to the plant site for quick erection.

An attractive feature of the high temperature gas cooled reactor is its inherent safety in the event of a power excursion or loss of coolant. The large mass of graphite in the core has sufficient thermal inertia to prevent a rapid rise in temperature. The large heat transfer surface in the core also promotes cooling by natural circulation even under depressurized conditions.

1. Introduction

1.1. Thermodynamic Cycle

In any thermodynamic cycle where heat is used to produce work, the key requirement for high efficiency is that the heat should be received by the working fluid of the cycle at as high a temperature as possible and rejected from the cycle at as low a temperature as possible. The low temperature is governed by ambient conditions while the high temperature is limited by material properties. In a steam cycle the maximum steam temperature, as determined by the characteristics of the steel alloy tubes in which the steam is heated, is around 540°C for most applications. To induce adequate heat flow from the reactor coolant to the working fluid the reactor coolant temperature must be considerably higher, that is, at least 650°C. This is at about the limits of the Advanced Gas Cooled Reactor AGR.

1.2. High Temperature Requirements

With a view to further development of the AGR concept, particularly with regard to increased temperatures, an alternative core configuration and a different reactor coolant were employed. The use of non-metallic fuel cladding and core structure allowed for higher operating temperatures and a greater margin to fuel damage in the event of a power excursion. Furthermore the use of inert helium as the reactor coolant avoided the high temperature corrosion that occurs at an increasing rate with rising temperature when using carbon dioxide as a coolant.

1.3. Historical Background

The High Temperature Gas Cooled Reactor (HTGR) was developed in both the United Kingdom and the United States to the point of commercialization. In the United Kingdom this was one of several possible routes for commercial development at the time but an alternative path was chosen and no further development of the HTGR took place. In the United States one commercial plant was built but it suffered a series of technical problems and was decommissioned prematurely. The concept however was well proven and much technical experience was gained in rectifying various design problems. At the time of decommissioning there was a general lack of demand for new large scale electrical generating capacity so interest in the concept waned. The HTGR however has been proven to be viable as a commercial unit and has certain inherent safety characteristics not found in water cooled reactors.

1.4. The Dragon Reactor

The Dragon Reactor Experiment was an experimental high temperature reactor (HTR) reactor built at the UKAEA Winfrith, England as an international effort to study the irradiation of fuels and fuel elements for future HTRs. Technical data for the Dragon reactor is given in Table 1.

Parameter	Value
Total Core Height	2.54 m
Active Core Height	1.60 m
Total Core Diameter	1.50 m

Active Core Diameter	1.08 m
Number of Fuel Elements	37
Number of Reflector Element	30
Number of Control Rods	24
Fuel Rods per Element	6
Reactor Coolant Pressure	2 MPa
Coolant Inlet Temperature	350°C
Coolant Outlet Temperature	750°C
Helium Mass Flow Rate	9.62 kg/s
Thermal Power Output	21.5 MW

Table 1: Dragon reactor technical data

The fuel, as shown in Figure 1, consisted of highly enriched uranium dioxide kernels 0.8 mm in diameter, having on top of a buffer layer of porous carbon, a triple (TRISO) layer of inner pyrocarbon, intermediate silicone carbide and outer pyrocarbon to give an overall diameter of 1.1 mm. These coated particles were bonded together in a carbonaceous matrix to form a hollow cylinder. Six such cylinders were arranged in circular fashion to form hexagonal fuel elements as shown in Figure 2. The central position was for testing various fuels, such as low enriched uranium, thorium and plutonium, in the same type of matrix. The body of the fuel elements surrounding the fuel cylinders, consisted of graphite to serve as the moderator. Provision was made for the top and bottom portions of each element to be graphite only and hence to serve as a reflector. Similarly graphite blocks were arranged around the outside of the core also to act as a reflector. Figure 3 shows a cross section of the core having 37 fuel elements surrounded by a reflector.

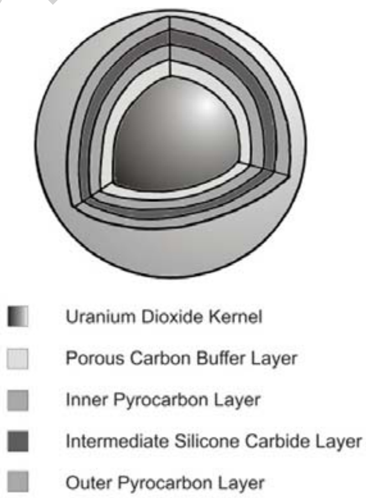

Figure 1: Dragon reactor TRISO fuel particle

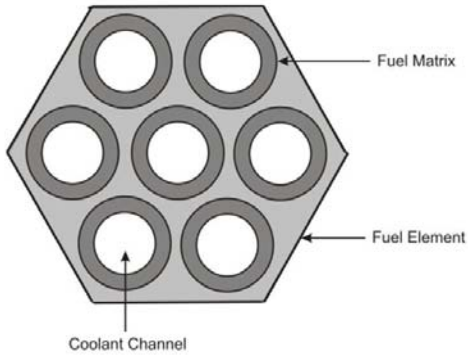

Figure 2: Dragon reactor fuel element cross section

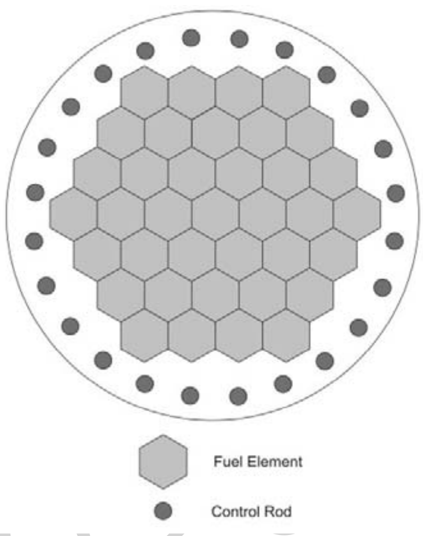

Figure 3: Dragon reactor core arrangement

Control of the reactor was primarily by 24 control rods inserted into spaces in the reflector as shown in Figure 3. By manipulating the control in a non-symmetrical pattern fuel temperatures could be varied as required for testing purposes. Generally, during normal operation, the fuel peak temperatures did not exceed 1250°C with graphite surface temperatures only 200°C lower. Overall the reactor had a strong negative coefficient of reactivity making it inherently stable and self regulating. In fact the reactor could be controlled over a wide range of power by varying the helium flow rate. An increase in the coolant flow rate would tend to cool the fuel, increase the reactivity and hence increase power until the fuel temperature returned quickly to near its original value.

The TRISO coated fuel particles with the impervious silicon carbide intermediate layer were very effective at containing fission products. Maintenance operations during the operating life as well as decommissioning at the end of the service life of the reactor were facilitated by the general very low levels of radioactive contamination prevailing in the reactor vessel. This allowed work with minimal personal protection after removal of the fuel.

1.5. The Peach Bottom Reactor

The Peach Bottom reactor was a prototype high temperature reactor (HTR) developed in the United States to prove the concept of using a high temperature gaseous coolant to produce steam and hence electric power via a superheated steam cycle. It operated successfully for seven years from 1967 to 1974. Technical data for this reactor is given in Table 2. The fuel consisted of highly enriched uranium carbide and thorium carbide particles 0.15 mm to 0.40 mm in diameter and coated with pyrolytic carbon to give an overall diameter of 0.25 mm to 0.50 mm. The porous nature of the pyrocarbon layer allowed fission gases to escape. These were purged by a separate small flow of helium and collected in internal and external traps. The coated particles were bound together by graphite in a matrix and made into grooved fuel compacts 69.85 mm (2.75 in) in diameter and 38.1 mm (1.5 in) in height. These were stacked on a central graphite spine and encased by a graphite sleeve to make a fuel element 3.658 m (12 ft) in length and 88.9 mm (3.5 in) in diameter. The active length of the fuel in the central region was 2.286 m (7.5 ft). The core was made up of 804 fuel elements arranged in a cylindrical manner to give an active core diameter of 2.792 m (9.2 ft). Graphite guide tubes within the core allowed for the insertion of 36 control rods and 19 shutdown rods as required during various operational states.

Parameter	**Value**
Total Core Height	3.658 m
Active Core Height	2.286 m
Active Core Diameter	2.792 m
Number of Fuel Elements	804
Number of Control Rods	36
Number of Shut Down Rods	19
Reactor Coolant Pressure	2 MPa
Reactor Inlet Temperature	334°C
Core Inlet Temperature	346°C
Core Outlet Temperature	749°C
Reactor Outlet Temperature	734°C
Helium Mass Flow Rate	55.4 kg/s
Thermal Power Output	115.3 MW
Core Power Density	8.24 kW/L
Feedwater Inlet Temperature	218°C
Steam Outlet Pressure	10 MPa
Steam Outlet Temperature	538°C
Steam Mass Flow Rate	45.4 kg/s
Generator Gross Output	46 MW
Plant Net Output	40 MW
Plant Overall Efficiency	34.7%

Table 2: Peach Bottom reactor technical data

Helium coolant at a pressure of 2 MPa flowed upwards on the outside of the fuel elements while ¼% of this flow passed through grooves in the fuel compacts to purge fission product gases. The thermal output of the reactor was 115 MW. This heat was transferred to the steam cycle in two external shell and tube forced circulation steam generators each having an economizer, recirculating evaporator, steam drum and superheater. Superheated steam at 538°C (1000°F) and 10 MPa (1450 lbf/in^2 g) produced 46 MW of gross electrical power and 40 MW of output from the station.

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Biographical Sketch

Robin Chaplin obtained a B.Sc. and M.Sc. in mechanical engineering from University of Cape Town in 1965 and 1968 respectively. Between these two periods of study he spent two years gaining experience in the operation and maintenance of coal fired power plants in South Africa. He subsequently spent a further year gaining experience on research and prototype nuclear reactors in South Africa and the United Kingdom and obtained M.Sc. in nuclear engineering from Imperial College of London University in 1971. On returning and taking up a position in the head office of Eskom he spent some twelve years initially in project management and then as head of steam turbine specialists. During this period he was involved with the construction of Ruacana Hydro Power Station in Namibia and Koeberg Nuclear Power Station in South Africa being responsible for the underground mechanical equipment and civil structures and for the mechanical balance-of-plant equipment at the respective plants. Continuing his interests in power plant modeling and simulation he obtained a Ph.D. in mechanical engineering from Queen's University in Canada in 1986 and was subsequently appointed as Chair in Power Plant Engineering at the University of New Brunswick. Here he teaches thermodynamics and fluid mechanics and specialized courses in nuclear and power plant engineering in the Department of Chemical Engineering. An important function is involvement in the plant operator and shift supervisor training programs at Point Lepreau Nuclear Generating Station. This includes the development of material and the teaching of courses in both nuclear and non-nuclear aspects of the program.