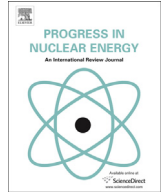




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High temperature reactor technology development in India

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ABSTRACT

High Temperature Reactor technology development programme was initiated in India with an aim to provide high temperature process heat for nuclear hydrogen production by splitting water. As high efficiency hydrogen production needs process heat at temperatures around 1123 K, a challenging technology development goal for the high temperature reactors was set to achieve coolant temperature of 1273 K. Currently development is in progress for a Compact High Temperature Reactor (CHTR), and a 600 MWth Innovative High Temperature Reactor (IHTR). Current design version of CHTR has ^{235}U based TRISO (TRistructural-ISOTropic) coated particle fuel, Beryllium oxide (BeO) as moderator, graphite as reflector, and lead-bismuth eutectic (LBE) as the coolant. The design incorporates many passive safety features for reactor heat removal. Current design version of IHTR is based on pebble bed fuel configuration with molten salt as coolant. For both the reactors, reactor heat is removed passively by natural circulation of the coolant. Technology development for these reactors include development of TRISO coated particle fuel, lead-bismuth eutectic and molten salt coolant technologies, BeO and graphite, oxidation resistant coatings, high creep strength alloys compatible to these coolants, high temperature instrumentation for these coolants, as well as high efficiency hydrogen production and electricity generation technologies.

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1. Introduction

Due to exhausting world reserves of petroleum-based products and the environmental concerns with use of fossil fuel, it has become important to find an alternative energy carrier for transport applications. Hydrogen is considered an attractive alternative. Large numbers of institutions including R & D laboratories, educational institutes as well as many industries in India are involved in carrying out developmental work related to all the aspects of hydrogen energy. This encompasses hydrogen production, storage, utilisation in transport and power generation sector, development of safety codes and standards etc. While options for production of hydrogen from fossil fuel, such as steam methane reforming and coal gasification have been developed to satisfy demand during interim periods, nuclear energy based hydrogen production by splitting water is being developed as a sustainable and environmentally benign option. Indian High Temperature Reactor (HTR) technology development programme is aimed at nuclear hydrogen production by splitting water. Water splitting

hydrogen production processes, either by thermo-chemical or electrolysis routes, need process heat at high temperatures or electricity or both depending upon selection of the specific process. The efficiencies for these processes are usually higher at higher temperatures. Thermo-chemical processes operating in the temperature range of 823–1123 K have been reported to have efficiencies in the range 40–57% depending on the thermo-chemical process selected (Sinha and Banerjee, 2005; Prasad and Dulera, 2009) have described various processes and their energy requirements). Nuclear reactors have a great potential for supplying energy for these hydrogen production processes at required high temperature conditions in a sustained manner. Indian HTR programme is aimed (Sinha, 2013) at development of reactor systems capable of providing process heat at 1273 K.

Currently, technology development program for HTR has been initiated to develop a small power Compact High Temperature Reactor (CHTR), and a 600 MWth Innovative High Temperature Reactor (IHTR). The CHTR would serve as technology demonstrator for technologies related to high temperature reactors. IHTR is aimed at producing large scale hydrogen, although it can as well be used for high efficiency electricity generation.

In the subsequent paragraphs, the paper briefly reviews HTR concepts developed in the world. This is followed by CHTR and

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details of its design, its fuel, in-core and structural materials, reactor physics design, thermal hydraulics design, and passive systems of the reactor. Subsequently the paper describes IHTR and molten salt related thermal hydraulics and material compatibility studies. Finally paper very briefly describes developmental work for hydrogen production.

2. History of high temperature reactors

As it is not the purpose of this paper to review history of HTRs and current developments, a very brief and indicative review of some of the HTRs has been provided to highlight similarities and differences of Indian reactors. (Brey, 2003), has briefly described evolution and future of developments of HTGRs):

2.1. Gas cooled reactors (GCRs)

Gas cooled reactors were the initiator for all High Temperature Gas Cooled Reactors (HTGRs). Commercial experience with gas cooled reactors began in 1956 with generation of electricity from Calder Hall plant in United Kingdom (UK). Subsequently 26 Magnox and 15 Advanced Gas Cooled Reactors (GCRs) were operated by UK. In France, Japan and Spain also development of commercial GCRs took place. HTGRs were developed to improve upon the performance of GCRs.

2.2. Early HTGRs

Development of HTGRs began in 1950s. HTGRs utilise ceramic particle fuel with coatings. Fuel is usually dispersed in graphite matrix. These reactors use graphite as the moderator material. Either prismatic type graphite moderator blocks are used or spherical fuel elements are used. Coolant Helium flows through coolant holes in the blocks or through the interstices in the pebble bed core. HTGRs can operate at very high core outlet temperatures because of all ceramic core.

The early HTGR plant designs were based on steel primary system pressure vessels. Dragon in UK, AVR in Germany and Peach Bottom in USA were the first HTGRs. Some of the early HTGRs have been described in the following paragraphs:

2.2.1. Dragon

It was a 20 MWth reactor built in UK. It first operated in 1965. It was an international project, started in 1959. The primary objective was to demonstrate the HTGR feasibility. The reactor used enriched uranium carbide fuel elements based on coated particle fuel. Helium was used as coolant. The core inlet and outlet temperatures were 623 K and 1023 K respectively. It did not produce any electric power, but served as a platform for development of helium gas cooled reactors and advanced fuel particle coatings.

2.2.2. AVR

In Germany this 15 MWe Arbeitsgemeinschaft Versuchsreaktor (AVR) started operation in 1967. This was a pebble bed type of reactor, used particle fueled graphite spheres of 0.06 m diameter. This had a steel containment vessel. AVR had a coolant outlet temperature of 1223 K. The reactor was operated for about 1,22,000 h till 1988.

2.2.3. Peach Bottom

This 40 MWe HTGR in USA operated from 1967 to 1974. This reactor utilised coated particle fuel. This reactor used two kinds of core. The fuel particles for core-1 had a single layer of anisotropic carbon to prevent hydrolysis of carbide fuel kernels and had several failed fuels. This led to development of BISO (Buffer isotropic

pyrolytic carbon) coatings on the fuel kernels.

2.2.4. Fort St. Vrain

This reactor commissioned in USA used Pre-stressed concrete as the primary system containing vessel. This used a core of hexagonal graphite block fuel elements and reflectors with fuel in the form of TRISO coated particle fuel, once through steam generator modules producing superheated and reheated steam, steam turbine driven helium circulators. This reactor served as a test-bed for demonstrating valuable technologies including the reactor core in the form of hexagonal graphite blocks with TRISO coated fuel particles, reactor internals, steam generators, fuel handling and helium purification systems.

2.2.5. THTR-300

This Thorium High Temperature Reactor 300 MWe plant was sponsored by Germany. This reactor also had its reactor vessel made of pre-stressed concrete. It operated during 1985 to 1988.

Large HTGR steam cycle plant designs for the following reactors were worked out:

2.2.6. HTR-500

HTR-500 was a 500 MWe German design. It made considerable simplifications and optimisations based on the practical experience gained by THTR-300. This plant featured a simple design with the primary system components located within a single cavity Pre-stressed Concrete Reactor Vessel (PCR). This was a pebble bed reactor. This reactor had a capacity of 1390 MWth to produce 550 MWe of electricity.

2.2.7. VG-400

This was 1060 MWth Russian pebble bed reactor design for co-generation i.e. electricity production and supplying heat at core outlet temperature of 1223 K.

Currently following two small power HTGRs are operating.

2.2.8. HTTR

The High Temperature Engineering Test Reactor (HTTR) of the Japan Atomic Energy Research Institute (JAERI) is a graphite-moderated and helium gas cooled reactor with an outlet temperature of 1223 K and a thermal output of 30 MWth. The major objectives of the HTTR was to establish and upgrade the technological basis for advanced HTGRs and to conduct various irradiation tests for innovative high temperature basic researches. This was shut down since Fukushima Accident in 2011. JAEA, however, is under preparation of its operational re-start.

2.2.9. HTR-10

The 10 MWth High Temperature Gas-Cooled Reactor-Test Module is located at the Institute of Nuclear and New Energy Technology (INET) of Tsinghua University in China. It is a pebble bed, helium cooled, graphite moderated modular HTGR. The HTR-10 utilizes spherical fuel elements containing TRISO coated particle fuel. It attained first criticality in December 2000. Safety demonstration tests have been carried out for loss of forced coolant and control rod withdrawal without SCRAM demonstrating its safety aspects. Phase 1 (Steam Turbine Cycle: HTR -10ST) is continuing, and transitional works towards Phase 2 (Gas Turbine Cycle: HTR -10GT) are under way.

Future HTRs include HTR-PM, under construction; and VHTR, under development:

2.2.10. HTR-PM

This program in China is of HTGR plant demonstration and commercialization, and is based on experiences of HTR-10. HTR-PM

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