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Development of fast breeder reactor technology in India

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ABSTRACT

India is pursuing a three stage nuclear power program employing its natural uranium and Thorium reserves. FBRs form the second stage of the program linking the first phase with natural U and third phase with Th fuels. The saga of FBR technology development in India is presented in this paper. The valuable experience in design and operation of Fast Breeder Test Reactor (FBTR) is recounted briefly. The R&D for the techno-economic demonstration of Prototype Fast Breeder Reactor (PFBR) is explained including technology development for addressing manufacturing challenges and engineering full-scale qualification. Finally a brief projection is made on the developments for FBR600 with improved safety and economics as well as metal fuel technology for future experimental and power reactors to follow.

1. Introduction

The indispensability of Fast Breeders for the sustained growth of nuclear energy in India was realized by Dr. Homi Bhabha, even in the 1950s. He recognized that the limited reserves of Uranium in India cannot sustain the nuclear energy programme for long, and gave out the breeder route for the efficient use of Uranium. In the long run, Thorium, which India has in abundance, was slated to play a major role in India's Nuclear Programme (Fig. 1) (Kakodkar, 2008). The Department of Atomic Energy (DAE) has been actively pursuing the well known 'three stage nuclear programme' chalked out by Dr. Bhabha (Venkataraman, 1994).

Even when the major thrust of R&D activities in Bhabha Atomic Research Centre (BARC) was on the first stage of Indian nuclear power program, studies on the second stage had been initiated in parallel. Fast Breeder Reactor (FBR) technology in India had its roots in the several theoretical studies carried out in 1968 on several design options for fast reactors in the Indian context. It was concluded that for FBRs, carbide or metal fuel will be best suited for high breeding in U-Pu fuel cycle. Before launching on a large scale indigenous programme on fast reactors, however, there was an obvious need for a test reactor aimed at providing experience in fast reactor operation and large scale sodium handling. The reactor was also required to serve as a test bed for irradiation of fast reactor fuels and materials. The need for comprehensive R&D on all aspects of fast reactors, up to the closing of the fuel cycle, was also recognized.

Abbreviations: FBR, Fast Breeder Reactor; FBTR, Fast Breeder Test Reactor; PFBR, Prototype Fast Breeder Reactor; DAE, Department of Atomic Energy; IAEA, International Atomic Energy Agency; CEA, French Alternative Energies and Atomic Energy Commission; IGCAR, Indira Gandhi Centre for Atomic Research; LHR, Linear Heat Rating; DHR, Decay Heat Removal; DND, Delayed Neutron Detection; SGDHR, Safety Grade Decay Heat Removal System; OGDHR, Operational Grade Decay Heat Removal System; IFTM, Inclined Fuel Transfer Machine; SADHANA, SAfety Grade Decay Heat removAl loop in Natrium; CDA, Core Disruptive Accident.

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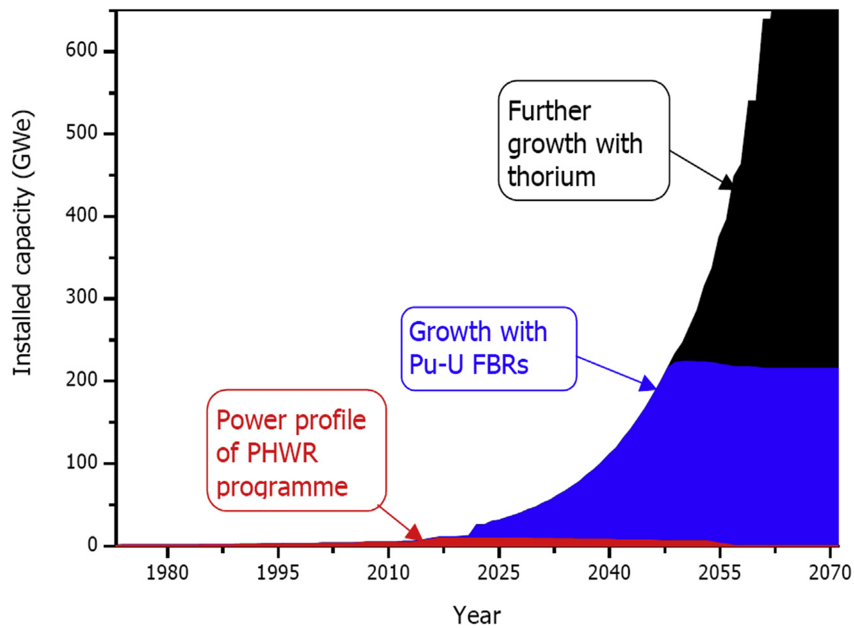

Fig. 1. Nuclear energy scenario of India.

1.1. Indira Gandhi Centre for Atomic Research

An R&D centre, exclusively devoted to the development of Sodium-cooled Fast Reactor (SFR) technology in India was established in 1971 at Kalpakkam, where construction of the first indigenous twin units of 235 MWe Pressurized Heavy Water Reactors was in progress. It was initially christened as 'Reactor Research Centre'. At the heart of the centre was proposed a sodium cooled test reactor, named Fast Breeder Test Reactor (FBTR), which would serve as a test bed for irradiation of fuels and materials and provide experience in large scale sodium handling and reactor operation. An agreement was signed with French Alternative Energies and Atomic Energy Commission (CEA) for transfer of the design of the Rapsodie reactor, training of personnel in Rapsodie and transfer of manufacturing technology of critical components.

Parallel to the construction of Fast Breeder Test Reactor, engineering halls with sodium loops for component testing, hot cells for Post-Irradiation Examination and laboratories with advanced facilities for materials and metals research, safety studies, sodium and fuel chemistry studies, development of instrumentation and R&D on reprocessing were also established. Thus the entire gamut of R&D for the second stage of our programme was included in the mission of the centre. The Reactor Research Centre (RRC) was renamed as Indira Gandhi Centre for Atomic Research (IGCAR) in 1985.

This paper focuses primarily on the reactor design and technology development. Other aspects such as fuel fabrication, reprocessing and waste management are covered in the companion papers (Setty et al., 2017; Natarajan, 2017; Wattal, 2017).

2. Fast Breeder Test Reactor

FBTR is a loop type, sodium cooled fast reactor (Suresh Kumar et al., 2011). Though adapted from Rapsodie, FBTR has several design modifications. The major change was the incorporation of a steam-water circuit in place of the sodium-air heat exchangers deployed in Rapsodie. The steam-water circuit has four once-through steam generator modules of the design used in the French Phenix Reactor. The construction of FBTR was started in 1972, and civil works were completed by 1977. Most of the components were installed in 1984. The reactor was made critical on 18th October 1985.

Fig. 2 shows the schematic flow sheet of the heat transport circuits and the temperatures shown are as per the original design at the rated conditions of operation. Heat generated in the reactor is removed by two primary sodium loops and transferred to the corresponding secondary sodium loops through Intermediate Heat Exchangers. Each secondary sodium loop is provided with two once-through steam generator modules. Steam from the four modules is fed to turbine-generator (TG). A Dump Condenser of 100% capacity is also provided in the steam water circuit for continued reactor operation when TG is not available.

2.1. The carbide fuel

Rapsodie used MOX fuel with 30% PuO_2 and 70% UO_2 (with the latter enriched to 85%) as the driver fuel. This design was originally proposed for the core of FBTR. In view of the non-availability of highly enriched uranium, studies on MOX fuel with 70% PuO_2 & 30% UO_2 (natural uranium) were carried out, which indicated poor performance in terms of the linear heat rating, swelling and compatibility with sodium. Uranium mono-carbide fuels had been tested, on a limited scale, in many fast reactors all over the world. Mixed carbide had, however, not been used in any of the reactors as a driver fuel. Mixed carbide with high Pu content had not been tested in any reactor. Nevertheless, based on out-of-pile studies, it was decided to go in for Pu rich carbide MK-I fuel (70%PuC+30%UC) as the driver fuel for the first core of FBTR. Being a fuel without any irradiation data, the core was made small. The first criticality was achieved with a small core with 23 subassemblies of MK-I carbide fuel as against the originally intended core with 65 MOX fuel subassemblies (Srinivasan et al., 2006). The initial target burn-up was set at 25 GWd/t.

The operating Linear Heat Rating (LHR) of the MK-I fuel was conservatively set at 250 W/cm. The reactor power was limited to 8 MWt for the 23 subassemblies (SA) core. Out-of-pile tests by electrical resistance heating up to 620 W/cm showed no sign of melting, giving the scope for raising the LHR to 320 W/cm. With the

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