ELSEVIER

Fusion Engineering and Design 51–52 (2000) 299–307

Fusion Engineering and Design

www.elsevier.com/locate/fusengdes

Blanket R&D activities in Japan towards fusion power reactors

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Abstract

Research and development on both the solid breeder blanket and the liquid breeder blanket have been conducted in Japan towards the fusion power reactors. The solid breeder blanket, one of the most realistic systems based on the present technical data base, is under development within Japan Atomic Energy Research Institute in collaboration with the Japanese universities. On the other hand, the liquid breeder blanket has been studied mainly within the universities as a future alternative blanket system considering its merits like a simple structure and a high accommodation to radiation damage. The present paper overviews the long-term development program and the current status of R&D on the breeding blankets towards the DEMO reactor. © 2000 Elsevier Science B.V. All rights reserved.

Keywords: Breeding blankets; Japan; Fusion power reactors

1. Introduction

Recent progresses on the fusion plasma researches have made it possible to proceed intensively a development plan for a breeding blanket system aiming at the electric power generation in the fusion reactors. As a most realistic blanket system to produce high heat for the power generation, a solid breeder blanket cooled by high temperature pressurized water has been proposed and applied for the design of DEMO reactor SSTR [1]. It utilizes layered small pebbles of tritium breeder and neutron multiplier installed in a reduced activation ferritic steel box structure. The electric power generation efficiency is expected to be comparable with that of the fission reactors due to its cooling conditions. In order to realize a higher efficiency, an advanced solid breeder blanket cooled by high temperature pressurized helium gas has been proposed for the design of the power reactor DREAM [2]. In the design, SiC/SiC composite is utilized instead of the ferritic steel as a structural material. Presently, our major efforts are concentrated on the R&D on the water-cooled solid breeder blanket for the DEMO

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PII: S0920-3796(00)00295-7

reactor considering the ferritic steel as a most realistic structural material. As an alternative, one also pursues an advanced blanket system based on liquid breeder cooling considering its merits like a simple structure and a high accommodation to radiation damage. Force free helical-like reactor (FFHR) is a design study for DEMO reactor which utilizes Flibe (Li_2BeF_4) as breeding material and coolant conducted by National Institute for Fusion Science (NIFS) and researchers in Japanese universities. Development of the solid breeder blanket is primarily conducted by Japan Atomic Energy Research Institute in collaboration with the Japanese universities, while the liquid breeder blanket has been studied mainly within the universities and NIFS.

The present paper overviews the long-term development program and the current status of R&D on the breeding blankets towards the DEMO reactor.

2. Development program of breeding blanket

Major purposes of the development plan on the solid breeder blanket are to develop fabrication technologies of the blanket module, to establish an engineering database for mock-up design and fabrication, and then to validate the design concepts by in-pile small-scaled mock-up tests and also by out-of-pile prototypical mock-up tests [3]. This program has been carried out so that the DEMO blanket test module to be tested in the International Thermonuclear Experimental Reactor (ITER) will be in time for the ITER operation, according to the following phases (See Fig. 1). In Phase 1 where the authors are presently involved, elemental development on the blanket has been carried out. It covers fabrication technology development of a blanket structure, breeder and multiplier pebbles and tritium permeation barriers, as well as material-oriented engineering tests on the breeder and multiplier pebbles and a variety of engineering tests on packed pebble beds, including safety-related issues. In Phase 2, in-pile irradiation tests and out-of-pile thermo-mechanical performance tests will be carried out on the basis of the elemental development. Within the in-pile tests, small-scale mock-ups, typically 10 cm in diameter and 1 m long will be irradiated in the Japan Material Testing Reactor (JMTR) to examine thermal responses and tritium release characteristics. In the out-of-pile performance tests, prototype blanket mock-ups will be manufactured and their thermo-mechanical performances will be demonstrated. In Phase 3, one plans to construct a DEMO blanket test module based on the elemental development and the in-pile and out-of-pile tests, and to install the test module in the ITER test port. The test module is to be operated under the fusion reactor environment in order to confirm high heat extraction, efficient tritium recovery, and reliability of the system. It would also be essential to conduct an electric power generation test using the test module prior to the DEMO reactor. In parallel to this

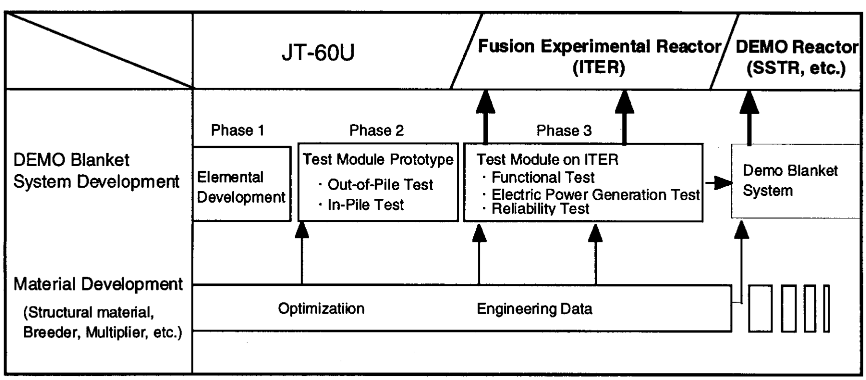

Fig. 1. Development plan for the solid breeder blanket.