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ITER BREEDING BLANKET AND DEMO RELEVANT BLANKET TEST PROGRAM

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ABSTRACT

One of the main objectives of the International Thermonuclear Experimental Reactor (ITER) is to test fusion blanket designs relevant to demonstration power plants (DEMO) which are developed in the Four Parties national programs. Demonstration of a breeding capability that would lead to tritium self sufficiency in a power plant, extraction of high grade heat, and electricity generation are the main goals of the tests. The test blanket modules will be installed in the ITER horizontal (equatorial) ports.

To achieve the different objectives of ITER, the ITER operation is divided into two phases, the Basic Performance Phase (BPP) and the Enhanced Performance Phase (EPP). The BPP has a few thousand hours of DT operation of ~ 0.3 MW.a/m^2 integrated DT neutron fluence over ten years. During this phase, test campaigns of 3-6 days will be dedicated for functional blanket tests. The EPP is also expected to last about ten years with integrated DT neutron fluence of at least 1 MW.a/m^2. Continuous test campaigns of 1-2 weeks will be dedicated for blanket tests. A breeding blanket will be installed for this phase replacing the shielding blanket of the BPP to provide the majority of the required tritium fuel with the expected external resources of 1.5 kg/year. A net tritium breeding ratio greater than 0.84 will be necessary to achieve a neutron fluence of at least 1 MW.a/m^2 during the EPP.

Four different DEMO blanket concepts are under consideration for testing in ITER: solid breeder blanket with water coolant, solid breeder blanket with helium coolant, lithium-lead breeder blanket with water coolant, and self cooled liquid lithium blanket. Also, the ITER breeding blanket will be tested during the BPP to confirm its performance for the EPP. Neutronics tests will be performed at the start of the DT operation to characterize the nuclear environment and to verify the neutronics design tools. Material testing will be done during the EPP to calibrate the fission reactor test results for fusion environment and to confirm the material performance in the ITER fluence range of 1 to 3 MW.a/m^2.

1. INTRODUCTION

One of the main objectives of ITER is to test blanket designs relevant to DEMO power plants. The tests foreseen on blanket modules include the demonstration of a breeding capability that would lead to tritium self-sufficiency in a power plant, extraction of high-grade heat, and electricity generation. The test blanket modules will be installed in 4 to 5 horizontal ports of about 1.8 m wide and 3 m high. To achieve the different objectives of ITER, the ITER operation is divided into two phases, the BPP and the EPP. The BPP has a few thousand hours of DT operation of ~0.3 MW.a/m^2 integrated DT neutron fluence over ten years. This phase will address the issues of controlled ignition, extended burn, near steady state operation, blanket testing. During this phase, test campaigns of 3-6 days will be dedicated for functional blanket tests. The tritium requirements for this phase will be provided from external resources. The EPP is also expected to last about ten years with integrated DT neutron fluence of at least 1 MW.a/m^2. This phase will address high availability operation, near steady state mode of plasma operation, material testing, and blanket testing. Continuous test campaigns of 1-2 weeks will be used for these blanket tests. A breeding blanket will be installed for this phase replacing the shielding blanket of the BPP to provide the majority of the required tritium fuel. Assessment of the ITER tritium requirements including the expected external tritium resources available at the time of operation, the ITER operational scenario, the total tritium inventory in the different ITER components, and the blanket replacement time between the two phases shows the need for a breeding blanket with a tritium breeding ratio in the range of 0.84 to 0.95 to achieve a neutron fluence in the range of 1 to 3 MW.a/m^2 during the EPP.

At present, the four parties are focusing on a limited number of blanket design options for fusion power plants. Five different blanket concepts are under consideration for testing in ITER, a breeding blanket for the second phase of ITER for tritium production, solid breeder blanket with water coolant, solid breeder blanket with helium coolant, lithium-lead breeder blanket with water coolant, and self cooled liquid lithium blanket. Neutronics tests will be performed at the start of the DT operation to characterize the nuclear environment of ITER and to verify the neutronics design tools. Also during the EPP, material testing will be done in ITER to calibrate the fission reactor test results for fusion environment and confirm the material performance in the fluence range of 1 to 3 MW.a/m^2. In preparation for testing these blankets in ITER and the development of the ITER breeding blanket, it is now essential to implement comprehensive R&D programs in each party. These programs should include the following: a) Breeder and multiplier materials development and testing in fission reactors and non-nuclear facilities, b) Structural material development and testing for operation in fusion environment, c) Functional blanket module testing in non nuclear facilities, d) Small blanket module testing in fission reactors, and e) Supporting R&D activities including tritium control and safety aspects. Elements of these programs are underway in the four parties.

The main objectives of the ITER blanket test program can be summarized as follows: 1) Evaluation of nuclear responses under fusion environment including nuclear heating and tritium production, 2) Demonstration/verification of the on-line tritium recovery and control system, 3) Demonstration of high-grade heat extraction and electricity generation, 4) Validation and calibration of design codes and data base used for the design of the blanket modules including neutronics, electromagnetic, heat transfer, hydraulics, and stress codes, 5) Confirmation and calibration of the test results obtained from fission reactors and non-nuclear facilities, 6) Demonstration of blanket module integrity and performance under thermal and electromagnetic loads, and 7) Observation of possible irradiation effects in the performance of the blanket modules.

2. ITER DESIGN FEATURES FOR BLANKET TESTING

Blanket testing has been included in the ITER design process as a fundamental design objective. This objective impacts the design configuration and the main operating parameters of ITER as can be seen in this section. ITER has twenty sectors corresponding to the number of the toroidal field coils. Each sector has a horizontal port on the outboard side between the outer legs of the toroidal field coils. These ports provide access from the outside to the plasma as shown in Fig. 1. Maintenance, plasma heating, diagnostics, and services are provided through these ports. About four to five of these ports are dedicated for blanket testing. The port size is ~ 1.6 x 2.6 m at the first wall. The blanket test articles will be inserted through these ports where it will be attached to the back plate. The coolant system and the tritium recovery system of these test articles are independent from the basic machine systems except for the ITER breeding blanket of the EPP, which is designed to use the basic machine systems. Most of the ancillary equipments are accommodated behind the test port to minimize the system response time for operational control and data collection. About 100 m^2 x 4 m space is allocated for this purpose and extra space for the large equipments is located in the reactor building. Installation, replacement, and maintenance are performed by a transfer cask. The cask has the necessary remote handling equipments and some ancillary equipments.

To perform blanket testing at relevant conditions for fusion power plants, an average neutron wall loading of about 1 MW/m^2 and an average DT neutron fluence at the first wall of at least 1 MW.a/m^2 are specified for ITER. However, the design of the ITER permanent components does not preclude achieving fluence levels up to 3 MW.a/m^2 for extensive blanket testing including blanket segment demonstration. The test ports are poloidally located where the neutron wall loading has its peak value of 1.2 MW/m^2. On the first wall of the test port, the neutron wall loading is almost constant at the peak value and the peak surface heat flux is 0.5 MW/m^2. ITER has several operating modes including pulse and near steady state operation. The reference pulse mode has a flat burn duration of 1000 seconds and dwell time of 1200 seconds. This burn time is adopted to permit the temperature of the test articles to reach and operate at thermal equilibrium conditions during each pulse. During the BPP, test campaigns of 3-6 days will be dedicated for functional blanket tests. For most of the blanket concepts, this period will be adequate for the tritium systems to reach equilibrium conditions. These test campaigns will be 1-2 weeks during the EPP.

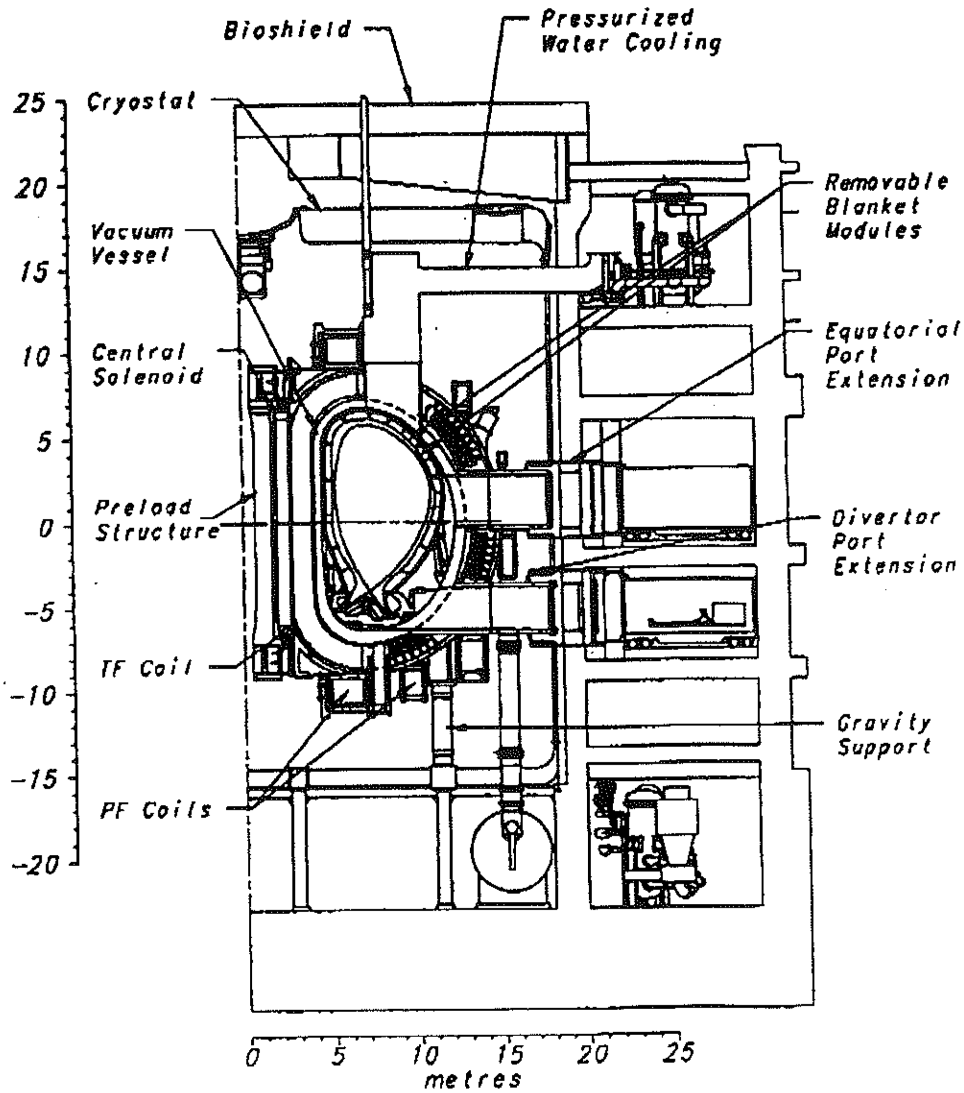

Fig. 1 ITER Vertical Cross Section

3. ITER TRITIUM BREEDING BLANKET

A breeding blanket is designed to provide the necessary tritium fuel during the EPP. It uses a ceramic breeder and water coolant for compatibility with the ITER machine design of the BPP. Li_2ZrO_3 is the selected ceramic breeder based on the current data base with enriched lithium and beryllium neutron multiplier. Both forms of beryllium material, blocks and pebbles are used at different blanket locations based on thermo-mechanical considerations and beryllium thickness requirements. As for the shielding blanket, Type 316LN austenitic steel is used as structural material and the water coolant parameters are the same for the two phases. The breeding blanket performance parameters have been checked against the ITER design requirements. The results indicate that the breeding blanket car be accommodated within the reference ITER configuration and satisfy the design requirements with adequate safety factors. Its breeding capability permits ITER to operate for a fluence goal in the range of 1 to 3 $MW.a/m^2$. This blanket and its support systems including the heat transport system, tritium processing system, and any special maintenance equipment will be tested in one of the ITER horizontal ports during the BPP.

The inboard blanket modules have two breeder zones embedded in beryllium with three coolant panels. The first wall has the first coolant panel to remove the surface heat flux and the nuclear heating from the front section of the breeding blanket. The second panel is located between the two breeder zones separated by beryllium. The last coolant panel is located at the back section of the module to remove the nuclear heating from the back section of the breeding blanket and the module structure. The outboard blanket module is similar to the inboard except it has three breeder zones instead of two to obtain a net tritium breeding ratio ≥ 0.8. The first wall consists of 13 mm thick Type 316LN stainless steel plate with built in rectangular coolant channels and a 5 mm beryllium coating for protection against plasma interaction. Tritium will be recovered by a helium purge gas (He + 0.1 to 1% H_2) during operation.

4. EUROPEAN DEMO RELEVANT BLANKET TEST PROGRAM [1,2]

The first phase of the European research and development program to identify the four most promising DEMO blanket concepts was started in 1989 and it was completed in 1995 by a comparative assessment of the four concepts. Two blanket concepts, a water-cooled lead lithium blanket (WCLL) and a helium-cooled pebble bed blanket (HCPB) were selected for the second phase of the program based on this assessment. Developments of these two blankets for fusion power plants and the delivery of blanket modules for testing in ITER are the main objectives of the second phase. The current scenario is to complete the development of the materials and technologies required for these blankets, and the detailed engineering design of the modules and their ITER interfaces by the year 2005. Fabrication and pretesting of the modules and ancillary systems will follow to start testing in ITER by the year 2010.

The WCLL blanket has liquid eutectic of lead-lithium (83Pb-17Li) as a tritium breeder and neutron multiplier. A single-steel box is used as a container for the liquid metal with radial and toroidal stiffeners. An independent pressurized water loop is used to cool the blanket box. The cooling tubes are inserted and soft brazed in toroidal channels. Also, pressurized water coolant flows in double-walled U-tubes within the lead-lithium pool. The water coolant parameters are based on the PWR systems. The water pressure is 15.5 MPa with inlet/outlet temperatu s of 265/325 and 300/325°C for the eutectic pool and the blanket box, respecti y. The selected structural material is martensitic steel to accommodate DEMO operating parameters. The lead-lithium eutectic is circulated slowly outside the blanket (< 0.005 m/s) for on-line tritium recovery. The HCPB blanket uses a lithium ceramic breeder and beryllium neutron multiplier in pebbles form. Helium is used as a coolant with ~ 8 MPa pressure and 250/450°C inlet/outlet temperature. Two independent coolant loops are used to cool the first wall and the blanket internals. The ceramic breeder and the neutron multiplier are separated by radial toroidal cooling plated. The tritium is recovered during operation by an atmospheric helium purge gas. The current reference breeder material is lithium orthosilicate, lithium zirconate and lithium titanate are also under consideration. The selected structural material is also martensitic steel.

At present, the European long term technology program has an average budget of 19 MECU/year for the period of 1995-1998, 50 MECU for blanket and 26 MECU for structural materials. The minimum estimate for the development of the two concepts is 300 MECU over 15 years. This gives an average budget of ~ 20 MECU/year, which is comparable to the current average budget of the ongoing 1995-1998 development program.

5. JAPAN DEMO RELEVANT BLANKET TEST PROGRAM [3]

Development of DEMO blankets in Japan has concentrated on the solid breeder blanket concepts with beryllium neutron multiplier. Li_2O is the first candidate breeder in pebbles form. Other ceramic breeders (Li_2ZrO_3 and Li_2TiO_3) are under consideration. As for the structural material, ferritic steel, F82H, is selected to accommodate the expected DEMO operating conditions. Heat load capability, low swelling due to neutron irradiation, and industrial experience are the main reasons for this selection. Breeder in tubes and breeder out of tubes design concepts are under development. Both forms of beryllium material, pebbles and blocks are being used in the current blanket designs. Ceramic breeder and beryllium multiplier are separated by steel structure or coolant panels to avoid material interaction.

Two coolants, water and helium are under consideration for different reasons. Water has a good materials compatibility data base, excellent heat transfer characteristics, and very low cost. Conventional PWR technology for power conversion is well established. However the thermal energy conversion is modest and high pressure containment is required. Helium coolant overcomes some of these difficulties and offers other advantages. Its chemical inertness, transparency to neutrons, and purity control are the main advantages. However its commercial deployment is much less than that of water-cooled technology. Therefore both coolants are being pursued.

A long term research and development program for DEMO relevant blankets has been proposed by Japan Atomic Energy Research Institute. The main objective of this program is to develop and construct test modules for the water- and Helium-cooled ceramic blankets. The estimated cost is 10 GYen for the period of 1997-2008. This program is being reviewed by the Fusion Council of Japan.

6. RUSSIAN FEDERATION DEMO RELEVANT BLANKET TEST PROGRAM [4]

The Russian Federation program has two blanket concepts under development, a self cooled lithium blanket with vanadium structure and a helium-cooled ceramic breeder blanket with ferritic steel structure. The first concept uses lithium as coolant and breeder material with beryllium multiplier and vanadium alloy structure. An electrical insulation is utilized to reduce the MHD pressure drop. Tritium is recovered from the lithium outside the blanket during operation. The second concept uses solid breeder material (Li_2ZrO_3 or Li_2TiO_3) with beryllium multiplier. Separate helium loops are used for heat removal and tritium recovery during operation.

For the self cooled blanket, the vanadium alloy development has concentrated on two alloys (V-4Cr-4Ti and V-5Cr-10Ti). Baseline mechanical properties, compatibility with liquid lithium, industrial fabrication and joining methods, and irradiation effects on the mechanical properties are the main elements for this development. Several electrical insulator coatings on vanadium structure have been tested in contact with liquid lithium. Purification methods to remove impurities and corrosion products from lithium loop have been tested. MHD pressure drop and heat transfer studies have been carried out for the test module design. Beryllium multiplier fabrication tests and characterization are under way for the solid breeder blanket concept.

At present, the key Russian Federation Institutes are finishing a long term plan for the development and manufacture of blanket modules for testing in ITER. The budget for this program will be determined after its approval. The current year budget for this development is 2 GRub.

7. UNITED STATE DEMO RELEVANT BLANKET TEST PROGRAM [5]

The United State fusion program has concentrated on the development of two blanket concepts since the late 1980s, a self-cooled lithium vanadium blanket and a helium-cooled solid breeder blanket with reduced low-activation structure material. The self cooled blanket concept utilizes electrical insulator (CaO or AlN) coatings to reduce the MHD pressure drop with V-4Cr-4Ti alloy as the structural material. The solid breeder blanket concept utilizes Li_2TiO_3 as the breeder material, beryllium as the neutron multiplier material, and reduced low-activation ferritic steel as the structural material. Also, SiC is under consideration as low activation structural material for future testing. A development program for the two concepts, in collaboration with other parties, is under way aiming at module testing in ITER.

At present, the United States is spending 4.1 M$/year on the development of the self cooled lithium vanadium concept. A 78% of this budget is directed toward vanadium structural material development. For the helium-cooled solid breeder with ferritic steel structure, the current spending is 1.4 M$/year where the ferritic steel development uses 78% of this budget. A similar effort of 1.3 M$ is allocated for the development of the silicon carbide based materials. These developments are aiming for submodule or module testing in ITER at different time scale. The testing of the first self cooled blanket module may start prior to the first plasma because valuable data can be obtained concerning liquid metal MHD effects related to the electrical insulator coating. Following the last year restructuring of the United States fusion program, there could be modest budget growth beyond the current level of spending if the fusion program funding is not reduced from the current level.

8. CONCLUSIONS

The development of the DEMO blanket modules and ITER breeding blanket design are progressing, which satisfy the programmatic objectives of ITER. R&D programs in the four parties have been started to provide the necessary data required for the DEMO relevant blanket modules and ITER breeding blanket for testing in ITER.

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