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Breeder and test blankets in ITER

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An overview of ITER efforts is presented in this paper in the area of blanket development and ITER test program preparation. The design of ceramic and eutectic options of the driver blanket were developed aimed on a reliable operation and adequate tritium production. With an estimated net tritium breeding ratio of 0.8–0.95 ITER is able to achieve testing fluence goal of 1 MWa/m^2 during 8 years of technology phase operation. The test program was developed to ensure realistic extrapolation to DEMO on the basis of ITER experience. Submodules for the blanket, high heat flux components and material testing are to be inserted in especially designed test ports. Tests are provided for DEMO-relevant blanket design concepts.

1. Introduction

The programmatic objective of the International Thermonuclear Experimental Reactor (ITER) is to demonstrate the scientific and technological feasibility of fusion.

The engineering and testing objectives of ITER are (1) to validate design concepts and qualify engineering components, (2) to demonstrate the reliability and maintainability of the reactor systems, and (3) to test the main nuclear technologies (blanket modules, tritium production, extraction of high-grade heat appropriate for the generation of electricity). Specifically it is hoped that the ITER device can provide (1) testing conditions with an average neutron wall loading of about 1 MW/m^2, (2) a neutron fluence of about 1 MWa/m^2, but with the design allowing for a higher neutron fluence in the range of 3 MWa/m^2, (3) a tritium breeding blanket that aims at achieving a breeding ratio as close to unity as possible, and (4) an overall availability of the ITER of at least 10%, but which should reach a level of 25% and provide continuous operation for periods lasting one to two weeks. This paper summarizes progress in the ITER driver blanket development work and efforts to define the testing program to be carried out in the machine.

2. ITER driver blanket

ITER operation scenario includes 8 years of the technology phase operation. The above mentioned objectives have to be fulfilled during this phase.

Requirements for testing fluence and those for tritium production in the driver blanket are closely connected. The decision to use a breeding blanket in ITER and the lowest level of its breeding capability are strongly dependent on the fluence goal. The cost of tritium for 1 MWa/m^2 operation comes to 500–1500 M$ with a tritium price in the range of 10^4–3×10^4 $/g. The external tritium supply required is at the level of 8 kg/y in case of no breeding blanket in ITER even with availability as low as 15% during 8 years of the technology phase.

The cost of ITER driver blanket is estimated as 150–300 M$. Specific R&D could add another 200 M$. So the upper cost for tritium production breakdown lies in the region of 0.3–1.0 MWa/m^2 even if usefulness of the driver blanket technology development and operation experience in ITER are not taken into account for the extrapolation to DEMO. If the latter advantages are considered, the cost breakdown would lower to 0.1–0.3 MWa/m^2.

Another consideration is also an estimation of possible external tritium supply. If it is possible to obtain of 1–2 kg/y externally, a fluence goal of 0.15–0.3 MWa/m^2 could be achieved. The average availability of ITER could be supported at a low level of 0.03 MWa/m^2 with the external supply only, which means that the reactor operates about 10 full power days per year. To achieve availability goal of 15% the ITER driver blanket has to produce at least 80% of tritium burned in plasma.

The above mentioned reasons show that ITER needs a driver blanket with a breeding capability not lower then 0.8 unless the fluence and availability goals are changed. High reliability is required for the driver blanket of the first experimental thermonuclear reactor. Low-temperature and low-pressure coolant helps to satisfy the requirement. Reactor relevant blanket designs are to be tested in more limited scale blanket module testing.

Five driver blanket options were considered at the beginning of ITER activity, namely:
- solid breeder lithium ceramic blankets,
- lithium–lead eutectic blankets,
- aqueous lithium salt solutions,
- ceramic particulate cooled blankets,
- He^3 breeder blanket.

As a result of conceptual designing two blanket options were recommended for the further development in ITER EDA phase: solid-breeder (SB) lithium ceramic, and lithium–lead eutectic [1]. Both blanket options use low-temperature water (e.g. 100°C) as the coolant and solution annealed type 316 austenitic steel as the structure material. In addition, the SB concept incorporates beryllium as a neutron multiplier. The SB was identified as the 'first options' for the system integration and three different designs with the solid breeder were developed. Li_2O and several ternary lithium oxides are generally considered as the leading candidates for the solid breeder blanket concepts. The desire to achieve a tritium breeding ratio of about one with limited breeding volume requires the extensive use of beryllium as a neutron multiplier. The first wall, blanket, and shield are integrated into a single unit with separate cooling systems. Poloidal and toroidal coolant flow were chosen for the inboard and outboard first wall, respectively. Both were considered in the blanket designs.

Two solid breeder configurations are considered for the detailed blanket design, viz., a multilayer configuration shown in figs. 1 and 2, and a breeder-in-tube configuration shown in fig. 3. The layered configura-

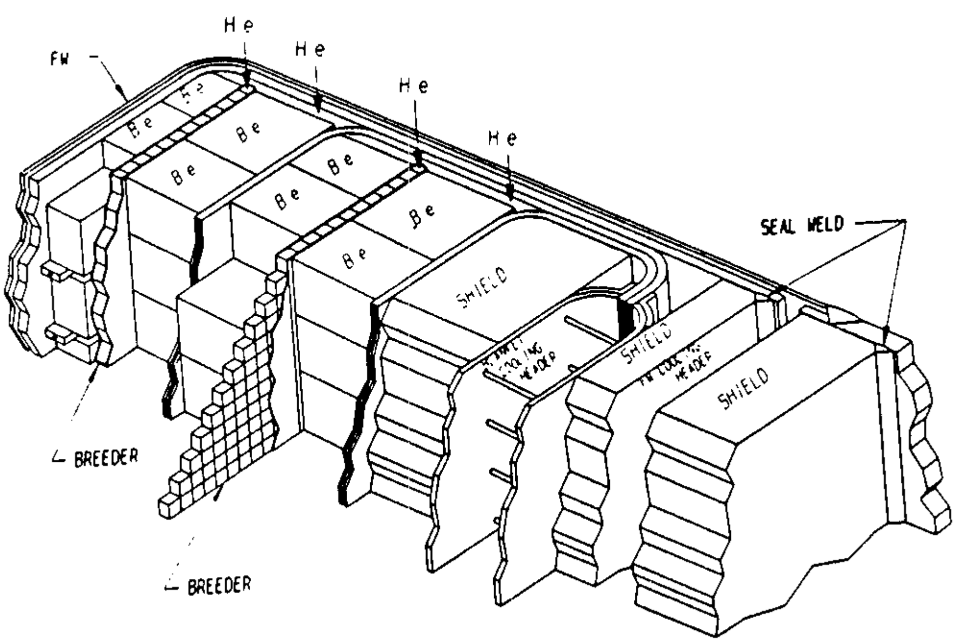

Fig. 1. Layered blocks blanket option.

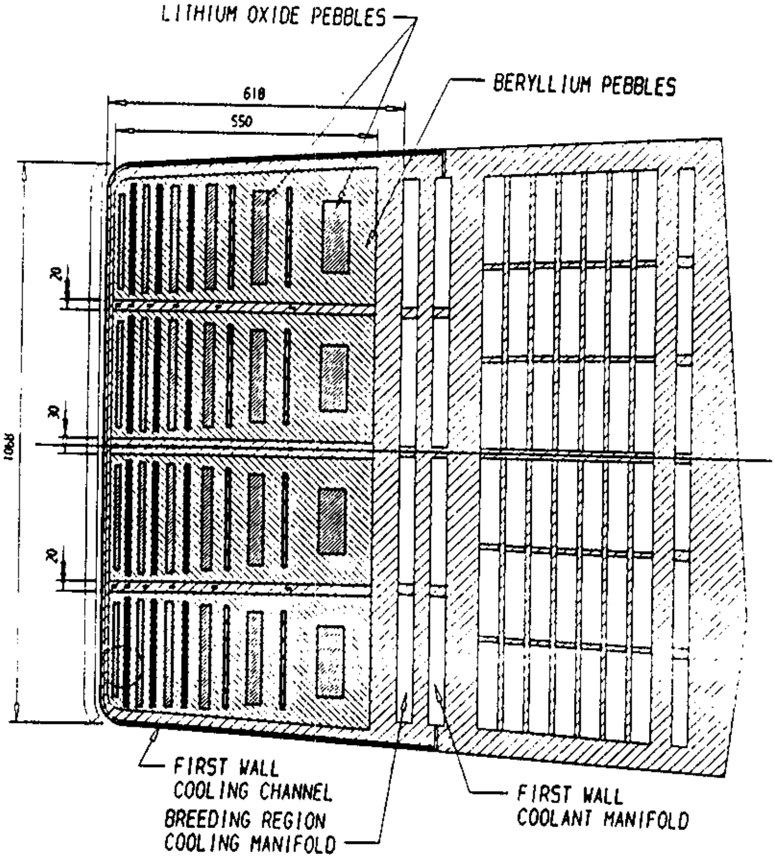

Fig. 2. Layered pebble-bed blanket option.

tion utilizes the beryllium zones to provide the desired temperature gradient between the low-temperature (60–100°C) coolant and the breeder. The temperature control in the breeder-in-tube configuration ($LiAlO_2$ breeder) is achieved by a controlled helium gas gap.

Two forms of ceramic breeder and beryllium are considered: sintered product (blocks or pellets) and small (approximately 1 mm dia) spheres. The ceramic breeder is highly enriched (50–95% ^6Li). Tritium is recovered from the breeder by a helium purge (He + 0.2–1% H_2). A net tritium breeding ratio of 0.8–0.95 is calculated. The calculated tritium inventory in the breeder can be maintained at less than 100 g. The blanket is designed with separate helium purge loop for the beryllium multiplier.

Considerable R & D efforts were performed during the last three years to study driver blanket materials and to justify the proposed designs. The latest experiments on the solid breeders have been conducted to measure the rates of lithium mass transport in the purge gas flow. Mass transport rates and a vapor pressure over Li_2O were measured as a function of temperature, He flow velocity and He/H ratio in the gas.

Results from in-reactor experiments indicate that the tritium inventory can be maintained on a relatively low level from several tens to less than one hundred grams in the ITER driver blanket. The inventory depends on the grain size, porosity and temperature of the breeder material as well as moisture pressure and He purge gas parameters.

Experiments on the compatibility of lithium oxide with 316 SS and beryllium were performed. Results show a formation of corrosion products at temperatures over 650 and 750 °C.

Simulation experiments of water leakage into Li_2O packed bed show serious damage by formation of lithium hydroxide at temperature over 470 °C.

Latest efforts were aimed on radiation resistance study of differently fabricated Be and on a study of

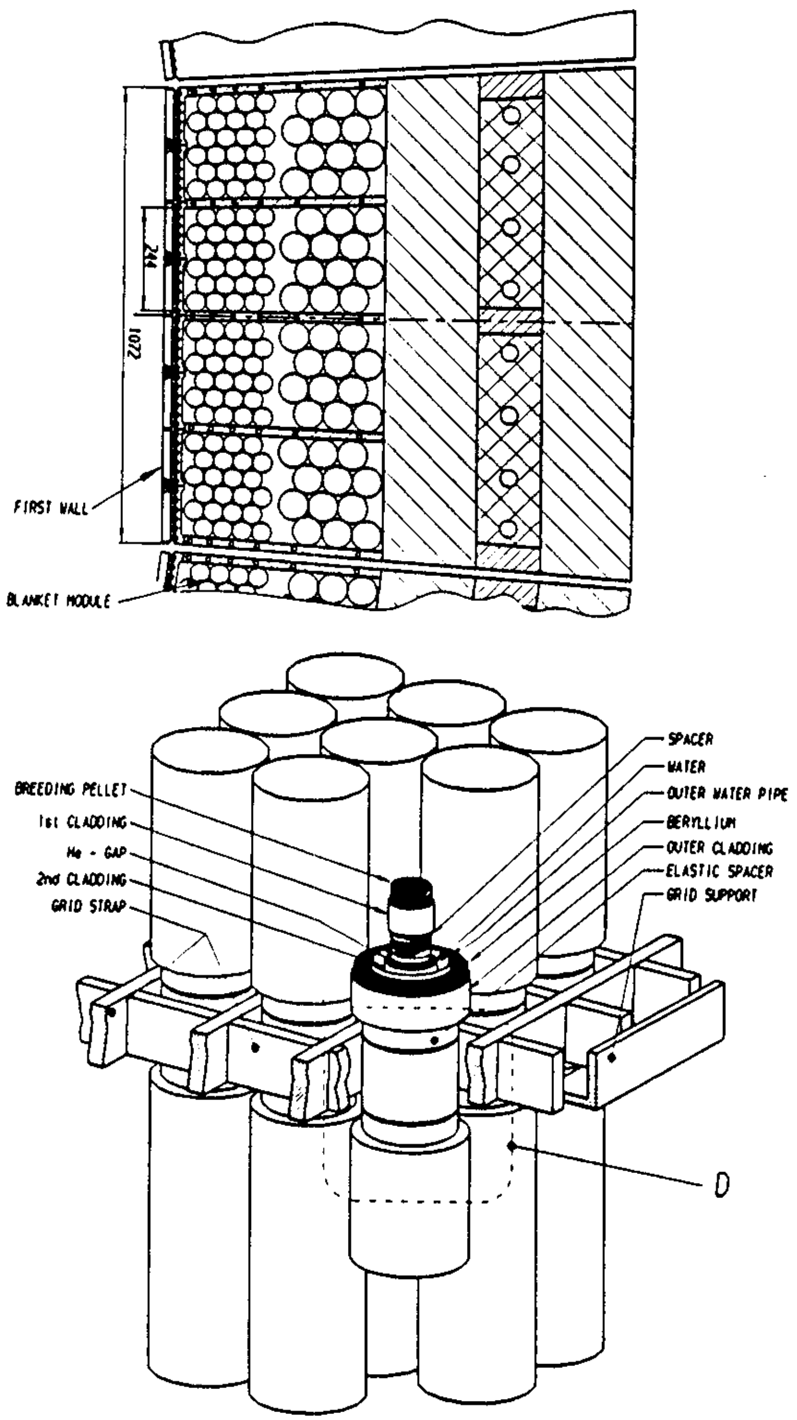

Fig. 3. Poloidal bit blanket option.

tritium retention and release from Be. Fabrication of Be samples with densities in range of 80–100% was done by now. Tritium release measurements from irradiated Be have been done for temperatures of 300–1100°C. The results show significant release from dense Be in one serie of experiments at temperatures above 600°C. Dominant release occurred at 800–1000 °C in the second serie of experiments. If Be temperature in the blanket will be kept below 500°C (which is a case for a majority of design options), most of tritium will be retained in Be, which result in a tritium inventory of over 800 g in ITER after operation to $3 MWa/m^2$.

Experiments have continued on stainless steels (mainly type 316) regarding low-cycle fatigue tests of base and weld metals, irradiation hardening, and aqueous stress corrosion. Aqueous stress corrosion tests in ITER water conditions indicate resistance of base metal 316 steel to cracking, however, significant cracking was observed in sensitized 304 steel. Some effects of crevice corrosion have also been observed in 316 steel. Data (physical properties, mechanical properties, fabrication, corrosion) has been obtained on reduced-activation, Mn-stabilized steels.

Most of R & D on ceramic breeder are adequate for ITER and DEMO blankets. They support designs developed for the driver blanket and blanket test modules to be tested in ITER.

Main critical database and design issues of the present SB blanket concepts include:
- solid breeder tritium release characteristics and methods to provide a thermal insulation between the structure and breeder materials in the SB blanket,
- beryllium irradiation effects such as swelling, tritium retention and compatibility with other materials.

An alternate blanket concept with ^{83}Pb–^{17}Li eutectic as the breeding material has also been developed as shown in fig. 4. The lithium-lead blanket has poloidal breeding channels which follow the first wall geometry. Each channel consists of coaxial pipes where eutectic is separated in individual chambers. During operation, eutectic is in solid form and it is melted for in-situ tritium recovery.

The tritium extraction is performed in a batch mode outside the reactor. The tritium extraction is necessary after about one week of full burn time. The eutectic is heated and melted before evacuation or replacement by hot gas through the coolant loop. Thermomechanical effects of the eutectic on the channel structure during transient conditions is a concern for the design. The proposed option of the eutectic driver blanket channel reduces this concern due to reduction of eutectic/channel thermomechanical interaction by melting and solidifying. The main feature of the design is the segmentation of the eutectic channel into individual chambers with connection of free space of each chamber with the lower one by means of an overflow pipe. Free space above the eutectic surface in the chamber significantly reduces thermomechanical interaction of the eutectic with channel structure and acts as a manifold for collecting helium and tritium released during melting.

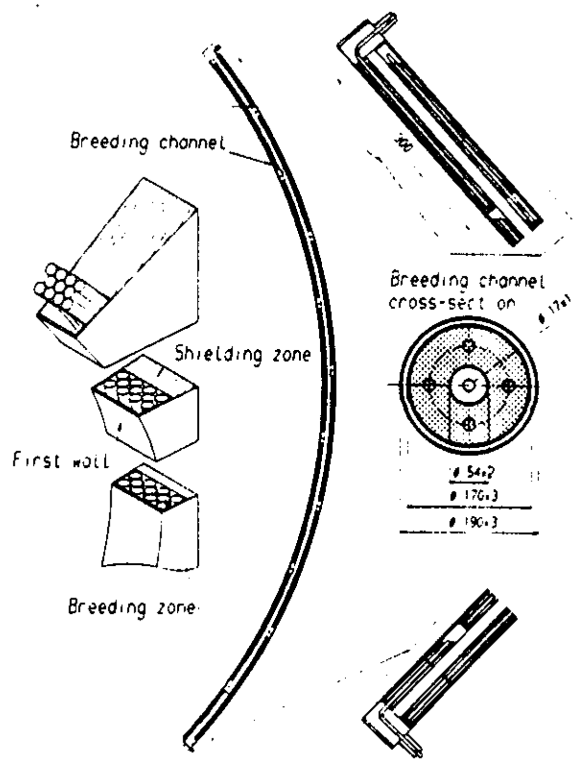

Fig. 4. Li17-Pb83 blanket option.

Design of LiPb blanket was also supported by R & D of the breeder material. Baseline physical, thermal and mechanical properties of lithium-lead eutectic were measured for both solid and liquid states. Experiments on the lithium-lead eutectic have addressed reaction kinetics with water at 350–950°C, corrosion with steel at 400–500°C, and release of tritium from irradiated material. Tests of thermomechanical interactions during solid-liquid transitions generally show favorable results. Preparation technology and handling requirements of the lithium-lead eutectic were developed including effects of structural inhomogeneity. Tests at 120°C to the fluence of 7×10^{19} n/cm^2 indicate low ^{120}Po radioactivity of ~ 400 Bq/g.

The main critical issues for LiPb blanket concept remain as tritium permeation and thermomechanical behavior of the design during melting/solidifying of the eutectic.

Blanket design development are supported by R&D on structural material. Experiments on stress corrosion cracking, He/H embrittlement, radiation creep and swelling at low temperatures are in the progress now. Preliminary results support using of 316 austenitic stainless steel as a structural material.

Combined blanket/shield design provides protection of TF coils and permits personnel access to the cryostat boundary after reactor shut-down. The preferred shielding materials are type 316 stainless steel and water. Integration issues, structural considerations, materials and shielding data base, and fabrication experience are the main reasons for this selection. Lead and boron carbide are considered to enhance the shielding performance. Their use is limited to the back of the vacuum vessel within the last 5 cm. Tungsten is used in selected areas instead of lead and boron carbide where the nuclear responses warrant.

Total nuclear heating and radiation dose to insulation are on the permissible levels of 50–70 kW and $(2-5) \times 10^9$ rad correspondingly. Dose outside the cryostat is estimated as ~ 0.5 mrem/h 24 hours after reactor shut-down in the most demanding operation scenarios. Local shielding around major penetrations with a thickness ~ 1 m is designed to provide permissible radiation level.

3. Testing in ITER

The ITER objectives and characteristics contained in [2] state that ITER will provide the data base "necessary for the design and construction of a demonstration fusion power plant". To do so ITER will serve as a test facility for blanket modules, tritium production, neutronics studies and testing advanced plasma technologies, including high-heat-flux components. An important objective will be the extraction of high-grade heat from reactor-relevant blanket modules and testing of reactor-relevant materials in a fusion environment, including advanced low activation and radiation resistant materials.

Blanket designs and some materials proposed for DEMO and fusion power reactors differ from those of the ITER driver blanket. These differences arise because the ITER driver blanket is specified to operate at less demanding performance levels (e.g. coolant temperature and pressure) than in the DEMO; the ITER design uses existing technologies and materials and puts great emphasis on safety and reliability (e.g. excluding use of liquid lithium). Reactor-relevant blankets, to be used for electricity production, should show environmental and economic attractiveness of fusion. Accordingly they have to operate at high temperatures and pressure, use advanced materials and operate at higher specific heat loads.

To obtain test information suitable for fusion demonstration reactors requires that the testing conditions be reasonably scalable. The minimum neutron first wall load and fluence for DEMO are ~ 2 MW/m^2 and ~ 10 MWa/m^2, respectively. Scaling by about a factor of two in power density and possibly by a higher factor in neutron fluence is reasonable. The majority of ITER parameters will most likely be lower than those of the DEMO and commercial reactors. Therefore, modules have been designed using engineering scaling to preserve important phenomena so that data from tests at 'scaled down' conditions can be extrapolated to reactor conditions. Engineering scaling involves altering physical dimensions (e.g. increasing the thickness of a solid breeder plate in a blanket to increase temperature differences and thermal stress) and changes in operating conditions (e.g. reducing the mass flow rate of the coolant to maintain coolant temperature rise). However, there are limits to engineering scaling. Therefore, there are minimum values for the major device parameters below which the test information is less useful, because results become difficult to extrapolate to reactor conditions.

Engineering scaling requirements are different for the various issues. In general, it is found that it is nearly impossible to design a module that can simultaneously provide testing for all the issues. Thus, several modules are generally required, with each one properly scaled to obtain useful test information for a subset of the technical issues.

The most important parameters which affect the value of testing are neutron wall load, neutron fluence, and time-related parameters (burn time, dwell time and continuous operating time). The requirements are based on extensive analysis of the behavior of nuclear components as a function of these parameters. Table 1 summarizes the recommendations. Minimum values are determined primarily from analysis of the important blanket phenomena under scaled conditions. One can show that test device parameters below the minimum value in any category will seriously limit the usefulness

Table 1
Nuclear testing requirements–summary of recommendations and reference values

Device parameter	Minimum needed for scalable tests	ITER conceptual design reference parameters (technology phase)
Average neutron wall load at the test module, MW/m^2	≥ 1	1.2
Number of ports	5	5
Minimum port size	2–3 m^2	3.74 m^2
Total test area	10 m^2	18.7 m^2
Plasma burn time	≥ 1000 s	2500 s [b]
Dwell time	TBD [a]	200–400 s
'Continuous' test duration	≥ 1 week	
Number of 'continuous' tests per year	2–3	
Average availability	10–15%	10–18%
Annual neutron fluence (at the test module), MW y/m^2	0.1	0.19
Total neutron fluence (at the test module), MW y/m^2	≥ 1	1.53

[a] Minimum acceptable dwell time is highly dependent on the design concept, and is difficult to specify. Further analysis in this area is recommended.
[b] Alternate plasma scenarios provide for steady operation.

of nuclear testing for at least one identifiable phenomenon, and thus results could be difficult to extrapolate to reactor conditions under such circumstances. Table 1 also shows the reference parameters for ITER in the long-burn hybrid operating scenario. In almost every case, the design of ITER meets or exceeds the minimum values.

Steady-state operation is a highly desirable ultimate goal for ITER during the technology testing phase, particularly from the standpoint of obtaining and sustaining equilibrium conditions during tests. Analyses show that if the dwell time lies in the range of 200 s, then the burn time should be in a range of few thousand seconds to maximize the relevance of extrapolation in some tests. Continuous test periods of 1–2 weeks have been shown to be important. Most important tests can be completed within this amount of time.

Table 2 provides a summary of fluence effects on blankets and materials in the range of values potentially available with ITER. Some low fluence effects and subsequent impact on breeder materials performance can be obtained at 0.1 MWa/m^2 The effects of creep relaxation, solid breeder sintering, cracking and swelling, helium embrittlement of all materials can be studied with fluences between 0.1 and 1.0 MW MWa/m^2. Above 1.0 MW MWa/m^2 several individual effects and interactions are expected. Also, the current test schedule provides for extensive use of sequential testing. Most tests have to be inserted and removed over periods ranging from 1–3 y.

The fluence recommendations are based also on the need to perform a sequence of blanket concept performance tests. Each of the tests would require about 0.2 MW MWa/m^2; the complete set will take roughly 3–6 y at full power and high availability (~ 25%), resulting

Table 2
Summary of fluence effects on blankets

0–0.1 MW MWa/m^2 (at test module)
Some changes in thermophysical properties of non-metals occur below 0.1 MW MWa/m^2 (e.g., thermal conductivity).
0.1–1 MW MWa/m^2 (at test module)
Several important effects become activated in the range of 0.1–1 MW MWa/m^2: radiation creep relaxation, solid breeder sintering and cracking, possible onset of breeder/multiplier swelling, He embrittlement.
Correlation of materials data with fission reactors and 14 MeV sources can be done with 1 MW MWa/m^2.
1–3 MW MWA/m^2 (at test module)
Numerous individual effects and component (element) interactions occur here, particularly for metals, e.g.: changes in DBTT, changes in fracture toughness, he embrittlement, breeder burnup effects, breeder swelling, breeder/clad interactions.

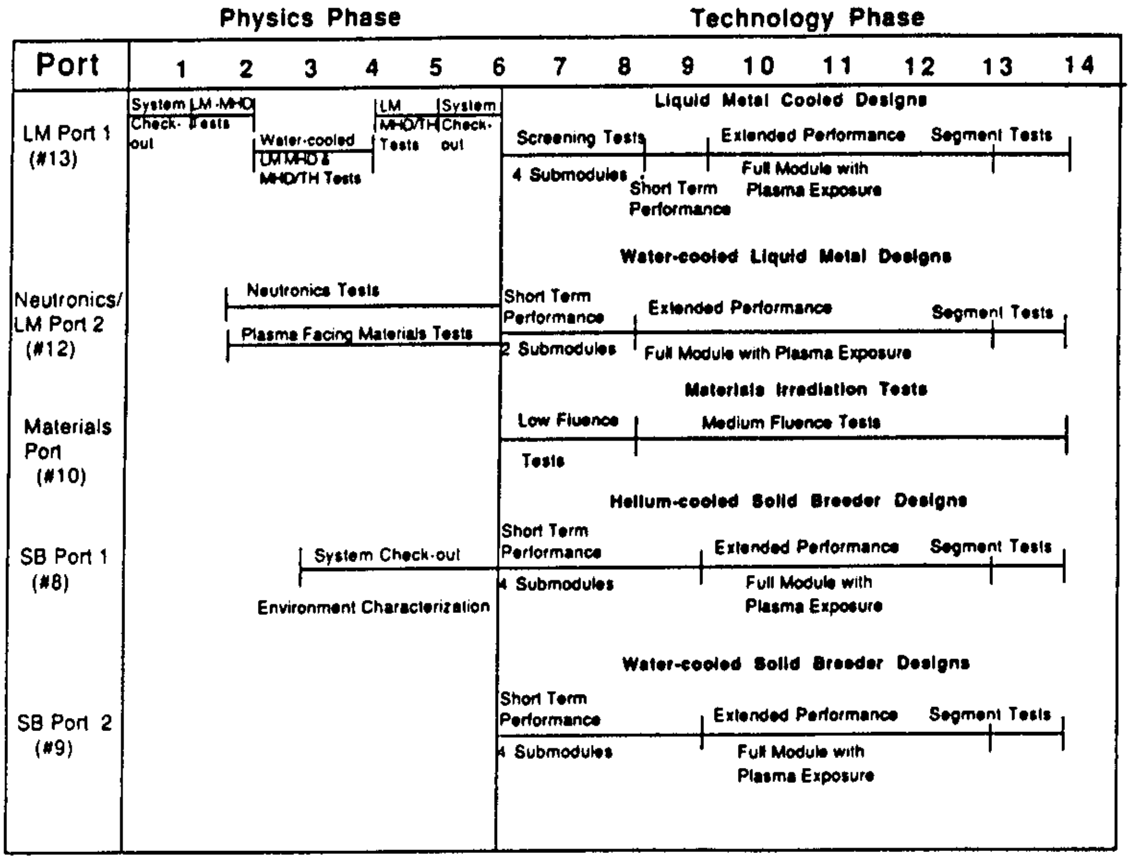

Fig. 5. Blanket test schedule.

in 1–2 MW MWa/m^2 of fluence.The overall test schedule was developed during ITER conceptional designing [3]. It is shown in fig. 5.

The most demanding in terms of space are blanket tests. The list of reactor relevant blanket types was overviewed with a worldwide participation of experts and the most promising proposals were integrated into four groups differing mainly by types of breeder and coolant, namely:

- self cooled liquid metal blankets,
- water cooled liquid metal blankets,
- helium cooled ceramic breeder blankets,
- water cooled ceramic breeder blankets.

The space which would be required to test all proposed designs in scalable models surpasses the space available on ITER. Thus a strategy was developed for sequential testing which includes (1) small size submodule performance testing (0.5–1.0 m^2 of surface area), screened from the plasma by the first wall of the basic device, (2) module (~ 3.0 m^2) extended performance testing without first wall screening at the end of test period and (3) full scale segment/sector tests using up to 1/16-th of full surface area of the torus.

The number of design options will diminish in a succession of these steps. The lead designs would then be selected for extended performance testing to determine their potential for use in advanced reactors. Finally, DEMO candidate blankets would be tested using full segments. The materials tests would consist of irradiation of many small samples in a well characterized environment. The tests would be conducted at different temperatures and neutron fluences, and the samples would be removed or replaced at relatively frequent intervals. Four central ports are required to perform all reactor relevant blanket tests. Tests will be done in parallel during the Technology Phase with a more limited number of tests during the Physics Phase. Material testing and neutronic tests require a space of one ITER port.

Some tests could be performed during the Physics Phase, in particular liquid metal MHD/thermalhydraulics tests, solid breeder system check-out and environment characterization tests, neutronics and materials tests for plasma exposure.

4. Conclusions

ITER, to act as a test bed for reactor relevant technologies, has to meet some requirements on the reliability and tritium production of its main driver blanket. Conceptional designing showed that the present envisaged technology could satisfy these requirements. Test programme, which includes advanced blanket concepts and advanced materials tests, would require 8–10 thousand hours of operation over about 8 calender years with the pulse length longer than 1000 s, the duty cycle which must be higher than 70% for at least a week, and the wall loading, which needs to be about 1 MW/m^2. ITER parameters satisfy to these requirements.

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