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Power plant conceptual studies in Europe

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Abstract
The European fusion programme is 'reactor oriented' and it is aimed at the successive demonstration of the scientific, the technological and the economic feasibility of fusion power. The European Power Plant Conceptual Study (PPCS) has been a study of conceptual designs of five commercial fusion power plants and the main emphasis was on system integration. It focused on five power plant models which are illustrative of a wider spectrum of possibilities. They are all based on the tokamak concept and they have approximately the same net electrical power output, 1500 MWe. These span a range from relatively near-term, based on limited technology and plasma physics extrapolations, to an advanced conception.

The PPCS allows one to clarify the concept of DEMO, the device that will bridge the gap between ITER and the first-of-a-kind fusion power plant. An assessment of the PPCS models with limited extrapolations highlighted a number of issues that must be addressed to establish the DEMO physics and technological basis.

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1. Background

1.1. The European fusion strategy

The aim of the European fusion programme is the exploitation of fusion as a commercial energy source. The programme is 'reactor oriented' and it is aimed at the successive demonstration of the scientific, the technological and the economic feasibility of fusion power.

A series of large tokamak devices, namely, JET (Joint European Torus), ITER and DEMO, constitutes the backbone of the European programme. The JET device, the largest tokamak in the world, is essentially devoted to the scientific phase. JET has been in operation since 1983 and the aim of its experimental programme, today, is to substantiate the ITER physics basis and to validate specific ITER engineering design options.

ITER, the 'next step' machine, will demonstrate the scientific feasibility of fusion and should strongly contribute to demonstrate its technological feasibility. These objectives will be achieved by extended burns of D–T plasma of several hundred seconds. In parallel to ITER, an International Fusion Material Irradiation Facility (IFMIF) should be constructed and operated to qualify the structural materials required for the third step of the European fusion strategy: DEMO.

DEMO will confirm the technological feasibility of fusion power and demonstrate its commercial viability. DEMO will be the first fusion device to export significant amounts of electrical power from fusion.

1.2. Rationale for the power plant studies

Models of fusion power plants are necessary to assess the safety, environmental and economic potential of fusion power. Following the definition of high level requirements for a fusion power plant, it is possible to make a number of assumptions both in physics and in technology in order to develop conceptual models of fusion power plants able to satisfy these requirements. The pertinence of the hypothesis made is the object of significant discussions within the fusion community, but if a specific characteristic can be proven to be true for fusion power plants models based on different hypothesis, its credibility is significantly enhanced. It is thus possible to assess the potential of fusion as an energy source on a fairly sound scientific basis.

For a reactor-oriented fusion development programme, it is essential to have a clear idea of the ultimate goal of the programme, namely, a series of models of fusion power plants, in order to define the correct strategy and to assess the pertinence of the on-going activities. Because it is possible to define high level requirements for a fusion power plant, it is easier to define models of fusion power plants and then of DEMO than to do the opposite. In fact, it is reasonably straightforward to define the requirements to be satisfied by DEMO after developing a model of a fusion power plant. Finally, a model of DEMO also allows assessment of the pertinence of the ITER objectives.

2. The European Power Plant Conceptual Study

From 1990 to 2000 a series of studies within the European fusion programme [1–3] examined the safety, environmental and economic potential of fusion power. These studies showed that the following hold true.

- Fusion power has well-attested and attractive inherent safety and environmental features, to address global climate change and gain public acceptance. In particular, fusion energy has the potential of becoming a clean, zero-CO_2 and inexhaustible energy source.
- The cost of fusion electricity is likely to be comparable with that from other environmentally responsible sources of electricity generation.

In the period since the establishment of the plant models developed for these earlier studies, there has been a substantial advance in the understanding of fusion plasma physics and of plasma operating regimes and progress in the development of materials and technology. Accordingly, it was decided to undertake a more comprehensive and integrated study, updated in the light of our current know-how and understanding, to serve as a better guide for the further evolution of the fusion development programme.

The European Power Plant Conceptual Study (PPCS) has been a 4-year study, between mid-2001 and mid-2005, of conceptual designs for commercial fusion power plants [4]. It focused on five power plant tokamak models, named PPCS A, B, AB, C and D, which are illustrative of a wider spectrum of possibilities. These span a range from relatively near-term, based on limited technology and plasma physics extrapolations, to an advanced conception. All five PPCS plant models differ substantially in their plasma physics, electrical output, blanket and divertor technology from the models that formed the basis of the earlier European studies. They also differ substantially from one another in their size, fusion power and materials compositions, and these differences lead to differences in economic performance and in the details of safety and environmental impacts.

3. Requirements for a fusion power plant

The requirements that a fusion power plant should satisfy to become an attractive source of electrical energy summarized hereafter were made by the EU utilities and industry [5]. The EU utilities and industry concentrated their attention on safety, waste disposal, operation and criteria for an economic assessment.

3.1. Safety and waste disposal

- There should be no need for an emergency evacuation plan, under any accident driven by in-plant energies or due to the conceivable impact of ex-plant energies;
- no active systems should be required to achieve a safe shutdown state;
- no structure should approach its melting temperature under any accidental conditions;
- ‘defence in depth’ and, in general, ALARA principles should be applied as widely as possible;
- waste transport should be confined to centres close to the production and ‘integrated’ sites, including both power plant and waste treatment/storage, should be considered;
- the fraction of waste which does not qualify for ‘clearance’ or for recycling should be minimized after an intermediate storage of less than 100 years.

3.2. Operation

- Operation should be steady state with power of about 1 GWe for base load and have a friendly man-machine interface;
- the lifetime should be about 40 years with the possibility of further extension up to 60 years for parts which are not replaceable;
- maintenance procedures and reliability should be compatible with an availability of 75–80%. Only a few short unplanned shutdowns should occur in a year.

3.3. Economics

- Since public acceptance is becoming more important than economics, economic comparison should be made with energy sources with comparable acceptability, but including the economic impact of ‘externalities’;
- licensing requirements and regulatory constraints should be strongly reduced as compared to fission plants;
- the construction time should be minimized, possibly no more than 5 years;
- standardization should be pursued as much as possible.

4. The PPCS plant models

The interrelationships of plasma performance, materials performance, engineering, economics and other factors were explored using a systems code, supplemented by the understanding gained from earlier analytical studies. The systems code studies employed the self-consistent mathematical model PROCESS [6]. This code incorporates plasma physics and engineering relationships and limits, together with improved costing models validated against the ITER costs and by comparison with similar US studies.

PROCESS varies the free parameters of the design so as to minimize the cost of electricity. The parameters arising from the PROCESS calculations were used as the basis for

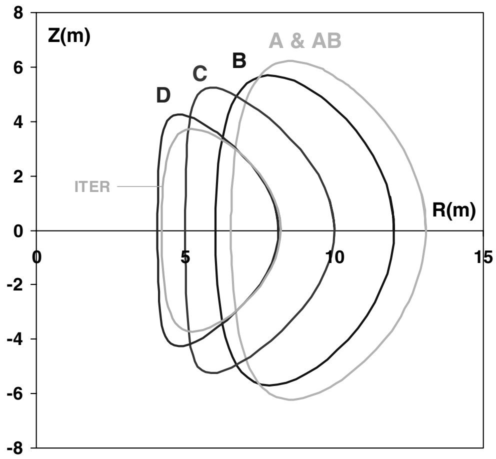

Figure 1. Illustration of the sizes and shapes of the plasmas in the PPCS models. To ease the comparison, model B has been renormalized to 1.5 GWe (size scaled with plasma volume). The original size of model B, as shown in tables 1 and 2, is 1.33 GWe. For comparison, ITER is also shown: this is very similar to model D. The axis labels denote major radius (R) and height (Z).

the conceptual design of the five models. These analyses also show which plasma, materials and engineering parameters are keys to further improving the economics.

4.1. Plasma physics basis

The PROCESS code employs a plasma physics module that, in its original form, was developed for the conceptual design activity phase of ITER and used to explore the early ITER design. This was modified to reflect further developments and has been updated to incorporate modern scaling laws [7]. The most important aspects of the plasma physics are: the use of the IPB98y2 for the energy confinement time scaling; a divertor module based on a simplified divertor model benchmarked to 2D code runs; a synchrotron reflection coefficient based on experimental measurements; and a current-drive efficiency calculated using NBI efficiency based on a modified Mikkelsen–Singer calculation.

The numerical limits used as constraints in PROCESS were based upon an assessment made for this purpose by an expert panel within the European fusion programme. For the three near-term models, PPCS-A, PPCS-B and PPCS-AB, the plasma physics scenario represents, broadly, parameters about thirty percent better than the design basis of ITER: first stability and high current-drive power, exacerbated by divertor heat load constraints, which drive these devices to larger size and higher plasma currents. PPCS-C and PPCS-D are based on progressive improvements in the level of assumed development in plasma physics, especially in relation to plasma shaping and stability, limiting density and in minimization of divertor loads without penalization of the core plasma conditions. The main parameters of the PPCS models are given in table 1.

4.2. Plant models

The technology employed in models A and B stems from the use of near-term solutions for the blanket. They are based, respectively, on the 'water-cooled lithium-lead' and the 'helium-cooled pebble bed' which are being developed in the European fusion programme [8]. All of these concepts are based on the use of a low-activation ferritic steel, which is currently being tested in the European fusion programme, as the main structural material. Associated with these are water-cooled and helium-cooled divertors. The water-cooled divertor is an extrapolation of the ITER design. The helium-cooled divertor, operating at a much higher temperature, is discussed in section 4.5 hereafter. For the balance of plant, model A is based on PWR technology, which is fully qualified. Model B relies on the technology of helium cooling, the industrial development of which is starting now, in order to achieve a higher coolant temperature and a higher thermodynamic efficiency of the power conversion system.

Following a revision of the European strategy in the development of breeding blanket concepts [9], a third near-term model based on a 'helium-cooled lithium-lead' blanket concept and a helium-cooled divertor was developed [10]. The performances of this model AB are in the same range as those of models A and B.

PPCS C and D are based on successively more advanced concepts in plasma configuration and in materials technology. In both cases the objective is to achieve even higher operating temperatures and efficiencies. Their technology stems, respectively, from a 'dual-coolant' blanket concept (helium and lithium-lead coolants with steel structures and silicon carbide insulators) and a 'self-cooled' blanket concept (lithium-lead coolant with a silicon carbide structure). In PPCS C the divertor is the same concept as in model B. In the most advanced concept, PPCS D, like the blanket, the divertor features silicon carbide as structural material and is cooled with lithium-lead. This allows the pumping power for the coolant to be minimized and the balance of plant to be simplified.

A tungsten alloy armour has been chosen for the divertor for all models. This choice allows to maximize the divertor lifetime, assumed to be at least 2 full-power-year for a thickness of the armour of about 5 mm, because of the low sputter yield. A tungsten alloy layer may be assumed on the first wall of the blanket since its erosion rate (0.1 mm per full-power-year in ITER-like conditions) is much lower than low Z materials like beryllium (about 3 mm per full-power-year in ITER-like conditions). The use of this tungsten layer does not impact the wastes issue. Under more optimistic assumptions, such as those made for model C, a bare ODS stainless steel first wall has been considered as possible.

4.3. Thermodynamic parameters and power conversion cycles

The economics of fusion power improves substantially with increase in the net electrical output of the plant. As a compromise between this factor and the disadvantages for grid integration of large unit size, the target net electrical output of all the models was chosen to be around 1.5 GWe instead of 1 GWe as recommended by the utilities. The fusion power is then determined primarily by the thermodynamic efficiency and power amplification of the blankets and by the amount of gross electrical power recirculated, in particular for current drive and for helium pumping (table 2). The result of these factors is a progressive fall in the fusion power from model A

Table 1. Main parameters of the PPCS models.

Parameter	Model A	Model AB	Model B	Model C	Model D
Unit size (GW_e)	1.55	1.50	1.33	1.45	1.53
Blanket gain	1.18	1.18	1.39	1.17	1.17
Fusion power (GW)	5.00	4.29	3.60	3.41	2.53
Aspect ratio	3.0	3.0	3.0	3.0	3.0
Elongation (95% flux)	1.7	1.7	1.7	1.9	1.9
Triangularity (95% flux)	0.25	0.27	0.25	0.47	0.47
Major radius (m)	9.55	9.56	8.6	7.5	6.1
TF on axis (T)	7.0	6.7	6.9	6.0	5.6
TF on the TF coil conductor (T)	13.1	13.4	13.2	13.6	13.4
Plasma current (MA)	30.5	30.0	28.0	20.1	14.1
β_N (thermal, total)	2.8, 3.5	2.7, 3.5	2.7, 3.4	3.4, 4.0	3.7, 4.5
Average temperature (keV)	22	21.5	20	16	12
Temperature peaking factor	1.5	1.5	1.5	1.5	1.5
Average density (10^{20} m^{-3})	1.1	1.05	1.2	1.2	1.4
Density peaking factor	0.3	0.3	0.3	0.5	0.5
H_H (IPB98y2)	1.2	1.2	1.2	1.3	1.2
Bootstrap fraction	0.45	0.43	0.43	0.63	0.76
P_{add} (MW)	246	257	270	112	71
n/n_G	1.2	1.2	1.2	1.5	1.5
Q	20	16.5	13.5	30	35
Average neutron wall load (MW m^{-2})	2.2	1.8	2.0	2.2	2.4
Divertor peak load (MW m^{-2})	15	10	10	10	5
Z_{eff}	2.5	2.6	2.7	2.2	1.6

Table 2. Thermodynamic parameters of the PPCS models.

Parameter	Model A	Model AB	Model B	Model C	Model D
Fusion power (MW)	5000	4290	3600	3410	2530
Blanket and shield power (MW)	4845	4470	4319	3408	2164
Divertor power (MW)	894	981	685	583	607
Pumping power (MW)	110	400	375	87	12
Heating power (MW)	246	257	270	112	71
H&CD efficiency	0.6	0.6	0.6	0.7	0.7
Gross electric power (MW)	2066	2385	2157	1696	1640
Net electric power (MW)	1546	1500	1332	1449	1527
Plant net efficiency[a]	0.31/0.33[b]	0.35	0.37	0.42	0.60

[a] The plant net efficiency is defined as the ratio between the net electric power output and the fusion power.

[b] Depending on the divertor concept used, either an ITER-like conception with water outlet temperature of 150 °C or a more advanced conception with water outlet temperature of 325 °C.

to model D. Given the fusion power, the plasma size and the power density are determined by the assigned constraints on plasma physics relating to restricting heat loads to the divertor. As a result of these factors, in no model does the first wall neutron load attain the limits that would have been set by the design. Taken together, these considerations lead to a reduction in the size of the plasma, from model A to model D.

Due to a relatively high pumping power in the case of helium cooling, the net efficiency calculated during the PPCS for models AB and B was disappointing. With respect to model A, which has a divertor cooled with water, the gain from helium cooling is a few percentage points (0.33 for model A, compared with 0.35 and 0.37 for, respectively, models AB and B). Moreover, such a modest gain is dependent on the successful development of a helium-cooled divertor able to support a maximum heat load of 10 MW m^{-2} instead of 15 MW m^{-2} for a water-cooled divertor, and on the development and qualification of the technology for a balance of plant using helium rather than water at PWR conditions, which is fully validated.

A more detailed assessment of the possible power conversion cycles for models AB and B was therefore carried out. The reference PPCS models use helium at 300–500 °C (input–output temperatures) and have a power conversion system using a standard Rankine cycle, the gross efficiency of which is primarily limited by the moderate He temperature. Advanced power conversion cycles have been studied taking PPCS model AB as a basis, aiming at an increase in gross efficiency. An indirect Brayton cycle using supercritical CO_2 as working fluid and a supercritical steam Rankine cycle were considered.

The primary heat transfer system layout originally considered for model AB consists of nine cooling loops for the blanket and three for the divertor. Each cooling loop transfers the heat to one steam generator (nine steam generators for the blanket, three steam superheaters for the divertor). A superheat and regenerative Rankine cycle coupled to this PHTS results in a gross efficiency (gross power/fusion power) of 55.59%. In this configuration the steam enters the high pressure section of the turbine at a temperature of 642 °C and 8.6 MPa.

Several supercritical Rankine cycles were considered [11]. Among them the so called ‘improved cycle’ seems the most promising. This cycle aims at optimizing the thermal exchange between primary and secondary circuits with a new primary heat transfer system configuration. The optimum configuration is obtained by the split of the blanket heat exchanger (HEx) units into two stages with a parallel HEx layout. A total of 18 HEx for the nine blanket loops are proposed while three HEx are maintained for the divertor loops. The following arrangement has been considered: for the blanket, nine HEx (first stage) are devoted to steam generation, seven HEx (second stage) are used for steam generation and superheating and two HEx (second stage) are used for reheating. For the divertor, two HEx are used for superheating and one HEx is used for reheating. This configuration leads to closer heat transfer curves between the primary and secondary, maximizing the thermal exchange effectiveness. It results in higher steam temperatures (increase in gross efficiency) and less steam mass flow (increase in net efficiency) compared with the other supercritical cycles. The improvement of the gross efficiency, with respect to the PPCS reference, is about 4 percentage points.

The interest for the supercritical CO_2 is its potential for high efficiency at low temperatures due to the low compression work near the critical point (7.38 MPa, 31 °C). The optimization work was focused on the recompression cycle, which improves the efficiency by reducing the heat rejection from the cycle, but no improvement of the gross efficiency was achieved. Independent supercritical CO_2 Brayton cycles were then considered [11] for the blanket and divertor cooling circuits, in order to benefit from the relatively high operating temperature of the latter. In this case, it is possible to obtain a gross efficiency similar to the one achieved with the supercritical, improved Rankine cycle.

If the net efficiency of a water-cooled plant, with water at PWR conditions (325 °C, 15.5 MPa) is limited to 33% maximum, the net efficiency of a He-cooled plant with a helium outlet temperature of 500 °C could instead reach around 40%.

4.4. Maintenance scheme and blanket segmentation

The frequency and the duration of in-vessel maintenance operations are the prime determinants of the availability of a fusion power plant. The divertor is expected to be replaced every two full-power-years because of erosion, the blanket every five full-power-years, corresponding to not more than 150 dpa of neutron damage in the steel of the first wall. A key development of the PPCS was a concept for the maintenance scheme, evolved from the ITER scheme, which is capable of supporting high availability.

ITER uses a segmentation of the blanket in 440 modules. In a power plant, such a large number of modules would result in an availability barely above 50%, which is unacceptable. To overcome this difficulty, a completely different segmentation of the power plant internals has been considered in a number of earlier power plant conceptual studies, the ARIES studies in particular [12]. Under this scheme, complete radial sectors of the tokamak are handled as individual units, the number of sectors being driven by the number of toroidal field coils. As this scheme was the only alternative available at the beginning of the PPCS, it was assessed in great detail. The engineering challenges related to its implementation are considerable and, assuming their resolution, it was assessed that the resulting availability would range between 76% and 81%.

As an alternative, a segmentation of the blanket into the smallest possible number of 'large modules' has been assessed. The maximum size of a module is determined by the size of the quasi-equatorial ports through which the modules must pass, which is limited by the magnet arrangements. The total number of modules is between 150 and 200. The feasibility of suitable blanket handling devices was investigated, and it was assessed that a plant availability of at least 75% could be achieved [13].

The larger the individual blanket element to be removed, the larger the electro-magnetic (EM) loads that will act upon that element. Already in ITER, the blanket modules are 'slit' in order to reduce the loads in case of plasma disruption. Very rough evaluations indicated that, with a blanket segmentation in 'large modules', it might not be possible to develop a mechanical attachment system able to support the loads, able to accommodate the thermal expansion between the blanket and the vessel and compatible with remote handling requirements. To accommodate these conflicting requirements, the 'multi-module' concept was proposed. The basic principle is to consider a 'strong-back' structure onto which are attached a number of smaller blanket 'modules'. The large size of the strong-back would allow for a limited number of pieces to be handled remotely, whilst the small size of the modules would reduce the EM loads to acceptable values. The overall segmentation is reminiscent of the segmentation in vertical segments adopted by the ITER CDA [14], where the multi-module segments are inserted (respectively withdrawn) from the vessel through, large vertical ports located in the upper region of the vacuum vessel.

A finite element model (FEM) has been developed for the EM analyses in case of plasma disruption. It assumes a vertical maintenance concept and a multi-module segmentation. Two toroidal segmentations have been analysed for the inboard blanket segments and two different electrical contact conditions between the strong-back and the vessel and between the strong-back and the modules have been considered. The analyses have been performed with a conservative approach, assuming the plasma as a filament placed at the plasma centre, at rest during the plasma disruption. The plasma current at the beginning of the disruption has been assumed to be 19 MA (full plasma current) and the current quench waveform has been assumed linear with a duration of 40 ms. In all cases more than 80% of EM loads are concentrated in the four modules of the equatorial region, which could be a consequence of the particular disruption considered (with the plasma at rest during the current quench). The loads are also dependent on the blanket concept itself: the difference in electrical conductivity between lithium-lead and Be-pebbles leads to an increase of a factor 2 in the EM loads.

Minimizing the EM loads and reducing the internal replacement time will be major design drivers for DEMO. The 'multi-module' concept appears promising and it shall be further assessed in the future.

4.5. Divertor

A helium-cooled divertor has been chosen for models AB, B and C to simplify the balance of plant by using the same coolant for all internal components. In model B this choice

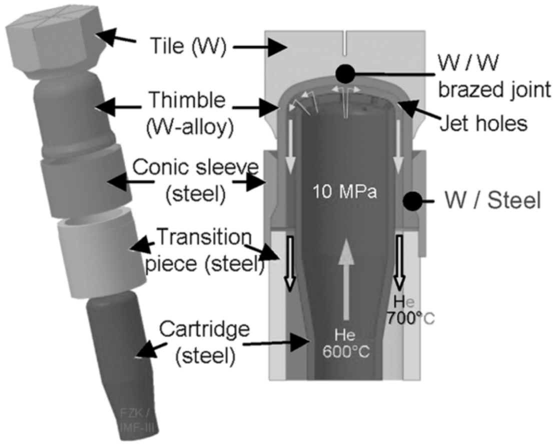

Figure 2. HEMJ concept for a He-cooled divertor.

also allows to prevent the risk of hydrogen formation resulting from the water-beryllium reaction in case of accidents. The main requirements to be satisfied by a He-cooled divertor are: (a) the ability to resist a peak heat load of at least $10\,\mathrm{MW\,m^{-2}}$, (b) a limitation of the thermal stresses by modular design; (c) the realization of a heat conduction path between the plasma-facing surface and the cooling surface as short as possible; (d) a heat transfer coefficient between the helium and plasma side structure greater than $30\,\mathrm{kW\,m^{-2}\,K^{-1}}$ and (e) the development of a structural material (refractory alloys, e.g. tungsten lanthanum oxide) with very good thermal properties and with a suitable operational temperature window, which is restricted by the ductile–brittle transition temperature and the recrystallization temperature. Moreover, to achieve a reasonable system efficiency, the pumping power should be limited to no more than 10% of the thermal power. The helium gas should operate at about 10 MPa with an inlet temperature of about 600 °C and an outlet temperature of 700 °C.

Three main concepts are currently considered [15].

- The HEMJ (helium-cooled modular divertor concept with multiple jets), in which heat transfer is enhanced through the impingement of several jets on the plasma-facing structures (figure 2). This cooling technique is already used in aircraft engines. The tungsten tiles, which act as thermal shield, are brazed on a thimble made of the tungsten alloy W–1%La2O3 (WL10). Helium flows to the thimble through holes located at the top of a steel cartridge. The cooling finger unit is fixed on a supporting structure made of ODS EUROFER using brazing and/or mechanical interlock.
- The HEMS (helium-cooled modular divertor concept with slot array) is an alternative to the HEMJ in which slots are used as turbulence promoters.
- The high efficiency thermal shield (HETS), in which heat transfer is enhanced through the impingement effects on the hemispherical surface and the effect of the centripetal acceleration as turbulence promoter when the fluid moves on the inner side of the sphere.

CFD simulations and stress analyses have shown the potential of both concepts to meet the design requirements. Mock-ups of 'single finger units' of both HEMJ and HEMS have been manufactured (figure 3) for testing in the Tsefey facility at the Efremov Institute (Russia) [16]. In some HEMJ mock-ups the armour was castellated to mitigate the effects of thermal stresses whilst no HETS mock-up has been manufactured to date. The mock-ups were tested within a heat flux range of $5\text{–}13\,\mathrm{MW\,m^{-2}}$. The helium cooling parameters were 10 MPa inlet pressure, 500–600 °C inlet temperature, and a varying mass flow rate in the range of $5\text{–}15\,\mathrm{g\,s^{-1}}$. The thermocyclic loading was simulated by means of switching the beam on and off (e.g. 60 s/60 s). The mock-ups were then subjected to destructive post-examinations, which revealed damages presumably due to micro cracks initiated during the fabrication processes. Neither brittle failure nor re-crystallization of the thimble were detected in any mock-up. Altogether, it can be said that the ability of the HEMJ and HEMS He-cooled divertor concepts to resist heat loads of $10\,\mathrm{MW\,m^{-2}}$ was confirmed. Significant efforts are still required to prove the feasibility of this concept, in particular the mechanical properties of the W-alloy before and after irradiation.

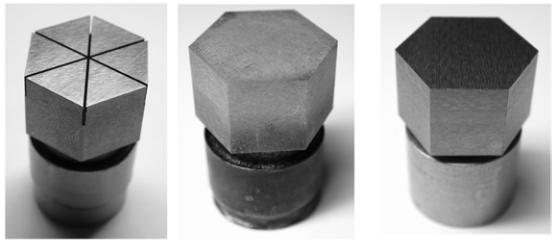

Figure 3. Mock-ups of 'single finger units' prior to testing.

The water-cooled divertor concept considered for PPCS model A consists of a limited extrapolation from the ITER divertor concept. Two alternatives were investigated. The first is very similar to the ITER concept, with CuCrZr as structural material for the cooling pipes embedded in a tungsten monobloc, which limits the maximum temperature of the water to 150 °C and, consequently, limits the plant model net efficiency to 0.31. In the second alternative stainless steel is used instead of CuCrZr. This requires an optimization of the repartition of the incident heat flux onto the steel pipes to limit the thermal stresses. The maximum water temperature is 325 °C, which allows one to increase the plant model net efficiency to 0.33.

The lithium-lead-cooled divertor considered for the PPCS model D was not developed in detail and it is not envisaged to adopt such a concept for DEMO. In practice, the overall PPCS model D was developed to a limited extent only to demonstrate the long-term potential of fusion power.

5. Assessment of the PPCS models

5.1. Safety and environmental impact

In the PPCS models, the favourable inherent features of fusion have been exploited, by appropriate design and

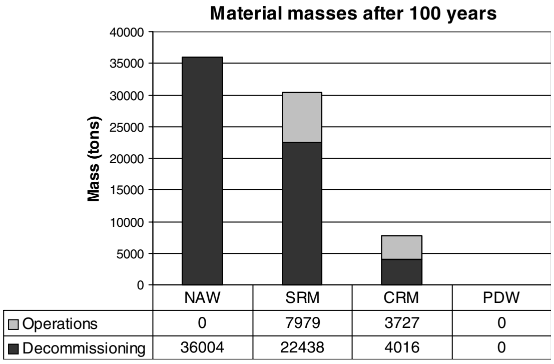

Figure 4. Categorization of all material arising from the operation and decommissioning of PPCS model B. NAW: non-active waste (to be cleared); SRM: simple recycling material (recyclable with simple remote handling procedures); CRM: complex recycling material (recyclable with complex remote handling procedures); PDW: permanent disposal waste (not recyclable).

material choices, to provide safety and environmental advantages.

- If a total loss of active cooling were to occur during the burn, the plasma would switch off passively due to impurity influx deriving from temperature rises in the walls of the reaction chamber. Any further temperature increase in the structures, due to residual decay heat, cannot lead to melting. This result is achieved without any reliance on active safety systems or operator actions.
- The maximum radiological doses to the public arising from the most severe conceivable accident driven by in-plant energies (18 mSv) would be below the level at which evacuation would be considered in many national regulations (50 mSv, the value which is also recommended by the International Commission on Radiological Protection).
- The power plant will be designed to withstand an earthquake with an intensity equal to that of the most severe historical earthquake increased by a safety margin, in accordance with the safety design rules in force.
- In case of fire, a maximum of a few grams of tritium could be released, by appropriate partitioning of the tritium inventory, which is consistent with the non-evacuation criterion.
- If there is substantial use of beryllium as an in-vessel component (approximately 560 ton are foreseen within the blanket of model B), it may be necessary to recycle it to satisfy the EU legislation on beryllium chemical toxicity.
- The radiotoxicity of the materials (namely, the biological hazard potential associated with their activation) decays by a factor ten thousand over a hundred years. All of this material, after being kept *in situ* for some decades, will be regarded as non-radioactive (clearance index <1, contact dose rate lower than 0.001 mSv h^{-1}, decay heat lower than 1 W m^{-3}) or recyclable (contact dose rate lower than 20 mSv h^{-1}, decay heat lower than 10 W m^{-3}) (figure 4). The recycling of some material could require remote handling procedures, which are still to be validated. An alternative could be a shallow land burial, after a time (approximately 100 years) depending on the nuclides contained in the materials and the local regulations. Thus the activated material from fusion power stations would not constitute a waste management burden for future generations.

Table 3. Cost of electricity for the five PPCS models.

	PPCS model				
	A	AB	B	C	D
'Internal' cost (€cents kWh^{-1})	5–9	5–9	4–8	4–7	3–5
'External' cost (€cents kWh^{-1})	0.09	0.08	0.07	0.06	0.06

For PPCS models A and B two design basis accidents (DBAs) were analysed: ex-vacuum vessel loss of coolant (ex-VV LOCA) and in-vacuum vessel loss of coolant (in-VV LOCA) induced by an ex-VV LOCA. In addition, two beyond design basis accidents (DBDAs) were analysed as well: a loss of flow (LOFA) without plasma shutdown inducing an in-VV LOCA and a loss of heat sink.

The accident analysis of DBAs showed that, in case of failure of the fast plasma shutdown system (FPSS), the plasma runs on until an in-VV LOCA would be induced by a thermal crisis of the first wall after 90 s in model A and 64 s in model B. The first wall peak temperatures in case of plasma shut down following the intervention of FPSS are 1200 °C for model A (stays above 600 °C for approximately 5s) and 850 °C for model B (stays above 600 °C for approximately 20 s). The aim of these analyses was to dimension the containment system (drain tank, suppression tank and expansion volume) and to assess their consequences in terms of radiological environmental releases (dust, tritium and activated corrosion products—ACPs). The in-VV LOCA induced by a LOFA is the accident with the largest impact in terms of dose to the population for both models A and B, as reported above. This accident was also analysed for plant model C.

5.2. Economics

'Internal cost' is the contribution to the cost of electricity from constructing, fuelling, operating, maintaining and disposing of power stations. The internal cost of electricity from the five PPCS fusion power plants was calculated by applying the codes developed in the Socio-economics Research in Fusion [3] programme. The relatively large ranges given for the internal cost estimates (table 3) correspond to differences in technological learning between a first-of-a-kind and a tenth-of a-kind reactor. Although it is notoriously difficult to estimate the future cost of electricity, in particular more than 30 years in the future, the calculated internal cost of electricity from all the models is in the range of estimates for the future costs from other sources, obtained from the literature.

The internal costs of electricity generation do not include costs such as those associated with environmental damage or adverse impacts upon health. The external costs of electricity from the PPCS plant models were estimated by scaling from the results from the European Socio-economics Research in Fusion programme. Because of fusion's safety and environmental advantages, all five PPCS models have low external costs: much lower than fossil fuels and comparable to, or lower than, wind power [17].

6. DEMO

6.1. General

One important outcome of the conceptual study of a fusion power plant (FPP) is to identify the key issues and the schedule for the resolution of these issues prior to the construction of the first-of-a-kind plant. Europe has elected to follow a 'fast track' in the development of fusion power [18], with two main devices prior to the first commercial power plant: ITER and DEMO. These devices will be accompanied by extensive R&D and by specialized machines and facilities to investigate specific aspects of plasma physics, plasma engineering, materials and fusion technology, e.g. the International Fusion Material Irradiation Facility (IFMIF). If the ITER objectives and machine design are now well established, this is not the case for DEMO. It is therefore worthwhile to reflect on the difficulties encountered during the development of the PPCS 'near-term' models, namely, models A, B and AB, prior to the selection of the main parameters and technical choices for DEMO.

This section presents some preliminary considerations following a European review of its 'fast track' fusion development strategy. The implications suggested for ITER are therefore not yet necessarily agreed among the international ITER partners.

Physics issues have been reviewed in detail [19] but are not discussed in this paper. For a given plant size and for a given blanket segmentation, the plasma current determines the loads on the internal components, which are the main drivers for the design of these components and of the main vacuum vessel. The plasma current, together with the bootstrap fraction, also determines the power required for H&CD. Even with H&CD system efficiencies above 50%, the recirculating power for H&CD severely affects the economic performances models A, B and AB. Practically, from an engineering perspective, the selection of the DEMO/FPP physics scenarios should aim at minimizing both the plasma current and the power required for H&CD.

6.2. Fusion milestones

To start construction of a FPP the following milestones have to be achieved in a timely fashion:

- qualification of materials: 120/150 dpa—steel in FW—for blanket materials, 40/60 dpa for divertor materials[10];
- qualification of in-vessel components, including welding, brazing and hipping, in DEMO (at least 100 dpa—steel in FW—for blanket and 40 dpa for divertor)[11];
- qualification of tritium systems;
- qualification of H&CD systems;
- qualification of ex-vessel components and systems if and when required (e.g. high temperature superconductors—if adopted, balance of plant (BoP) components if the FPP is cooled with helium);
- validation of the overall power plant architecture—in particular the segmentation of the internal components, and demonstration of remote handling procedures using rad-hardened equipment;
- demonstration of rad-waste recycling techniques and re-usability of recycled materials (if appropriate).

Within the European fusion development scenario, DEMO will be the key device for meeting the milestones listed above. To start construction of DEMO the following milestones have to be achieved in a timely fashion:

- demonstration of physics scenario for DEMO/FPP;
- qualification of DEMO/FPP relevant plasma-facing components (e.g. PFCs made of tungsten);
- validation of blanket functional performance (thermo-hydraulic, thermo-mechanical, tritium breeding ratio);
- validation of divertor functional performance (thermo-hydraulic, thermo-mechanical);
- qualification of materials, including welding, brazing and hipping (around 80 dpa—steel in FW—for blanket materials, around 30 dpa for divertor materials);
- qualification of tritium technology;
- demonstration of feasibility of H&CD systems able to satisfy the DEMO requirements in terms of efficiency, duty cycle and reliability.

In addition to meeting the above milestones, both DEMO and the FPP will have to satisfy a number of requirements in the areas of safety, public acceptance and economics. Some of these requirements are fundamental, e.g. the non-evacuation criteria following any accident driven by in-plant energies (DEMO and FPP) and the economic viability of the project (FPP). The discussion in this section is however limited to technological and engineering issues.

6.3. Critical issues

To develop a fusion development scenario, it is necessary to define when the above milestones will be met and, therefore, to appreciate the technological and engineering challenges behind each milestone. Previous analyses [20, 21] have highlighted the internal components (blanket and divertor) as critical and the European technology R&D programme has been amended accordingly. The focus of the on-going analysis is to assess the development needs for all components, including those outside the primary vessel. This analysis is still on-going in Europe, but a number of preliminary conclusions have already been reached.

After an extended commissioning phase, DEMO will have to operate with a high level of availability. This availability, together with the neutron wall loading, will determine the length of the DEMO operational phase required for the qualification of the internal components. A reasonable scenario can be developed by assuming an average availability in excess of 30% and an average neutron wall load of $2\,\mathrm{MW\,m^{-2}}$.

The duration of the replacement time of the internals depends on the number of elements to be replaced and on the amount of parallel work that can be implemented. The PPCS concluded that the ITER maintenance scenario is not reactor relevant and that the segmentation of the blanket,

[10] It is assumed that the structural material selected for the first FPP will be the ferritic steel Eurofer and that this material will maintain acceptable properties up to a fluence corresponding to 120/150 dpa.

[11] This corresponds to an aggressive scenario whereas FPP starts operation with in-vessel components tested up to 2/3 of the maximum fluence foreseen.

in a FPP, should not exceed 150–200 elements (there are currently 440 blanket modules in ITER). The validation of the power plant architecture and the qualification of the remote handling procedures by the complete replacement of the internal components should therefore be demonstrated in DEMO during the shutdown at the end of the first phase of DEMO operations.

The helium-cooled blanket concepts should be optimized in order to improve the net efficiency of the plant. This will require a reduction of the pumping power, in particular through a reduction of the pressure drops.

Finally, it is possible to check whether the objectives of ITER and of its accompanying programme are consistent with the assumptions made for DEMO. Preliminary considerations indicate that the following issues should be resolved prior to the start of DEMO operations (preliminary conclusions ought to be available to start the DEMO design).

- The validation of the bulk of the 'DEMO physics issues' should be completed during phase 1 of ITER operations (it is currently foreseen in phase 2). In the phase 2 of ITER operations, the DEMO relevant plasma scenario should be validated with a full tungsten first wall (the replacement of the complete first wall is currently not foreseen in ITER) [22].
- A validated breeding blanket concept able to ensure the tritium self-sufficiency of DEMO and to operate with high reliability and availability. This includes the prior qualification of the technology necessary for tritium production, extraction and control.
- A validated divertor concept able to operate in DEMO-like conditions with high reliability and availability.
- The validation of the H&CD technology for steady-state operation.

7. Conclusions

The PPCS results for the near-term models suggest that a first commercial fusion power plant—one that would be accessible by a 'fast track' route of fusion development, will be economically acceptable, with major safety and environmental advantages. These results also point out some of the key issues that should be resolved in DEMO and, more generally, help to identify the physics, engineering and technological challenges of fusion.

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