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Summary Of The International Workshop on Magnetic Fusion Energy (MFE) Roadmapping In The ITER Era

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CONFERENCE REPORT

Summary of the International Workshop on Magnetic Fusion Energy (MFE) Roadmapping in the ITER Era; 7-10 September 2011, Princeton, NJ, U.S.A.

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Abstract. With the ITER project now well under way, the countries engaged in fusion research are planning, with renewed intensity, the research and major facilities needed to develop the science and technology for harnessing fusion energy. The Workshop on MFE Roadmapping in the ITER Era was organized to provide a timely forum for an international exchange of technical information and strategic perspectives on how best to tackle the remaining challenges leading to a magnetic fusion DEMO, a nuclear fusion device or devices with a level of physics and technology integration necessary to cover the essential elements of a commercial fusion power plant. Presentations addressed issues under four topics: 1) Perspectives on DEMO and the roadmap to DEMO; 2) Technology; 3) Physics-Technology integration and optimization; and 4) Major facilities on the path to DEMO. Participants identified a set of technical issues of high strategic importance, where the development strategy strongly influences the overall roadmap, and where there are divergent understandings in the world community, namely: 1) the assumptions used in fusion design codes, 2) the strategy for fusion materials development, 3) the strategy for blanket development, 4) the strategy for plasma exhaust solution development, and 5) the requirements and state of readiness for next-step facility options. It was concluded that there is a need to continue and to focus the international discussion concerning the scientific and technical issues that determine the fusion roadmap, and it was suggested that an international activity be organized under appropriate auspices to foster international cooperation on these issues.

1. Introduction

With the ITER project now launched on its mission to achieve, for the first time, a magnetically confined burning fusion plasma on a power-plant scale, the countries engaged in fusion research are planning, with renewed intensity, the research and major facilities needed to develop the fusion nuclear science and technology for harnessing fusion energy. The Workshop on MFE Roadmapping in the ITER Era was organized to provide a timely forum for an international exchange of technical information and strategic perspectives on how best to tackle the remaining challenges leading to a magnetic fusion DEMO, a nuclear fusion device or devices with a level of physics and technology integration necessary to cover the essential elements of a commercial fusion power plant. Sixty-five researchers from 10 countries, including all the ITER partners, attended the workshop, which was held 7-10 September 2011 at Princeton University. The level of international participation reflected a widely-felt sense of urgency in the need to collaborate more closely in meeting fusion development challenges.

The workshop was organized by an international committee of 15 fusion researchers, all involved in fusion strategic planning and DEMO studies in their respective countries: M. Abdou

(USA), D. Campbell (ITER-IO), S. Cowley (UK), S. Deshpande (India), G. Federici (EU-EFDA), Y. Kamada (Japan), B. Kuteev (Russia), G. S. Lee (Korea), J. Li (China), S. Milora (USA), F. Najmabadi (USA), G. Neilson (USA), R. Wolf (Germany), H. Yamada (Japan), and H. Zohm (Germany). While attendance was open, all presentations were by invitation. In total there were 37 invited presentations (28 oral, 9 poster) and four summary presentations in four topical areas. In addition there were two general discussion sessions. All presented material is available at the workshop web site http://advprojects.pppl.gov/roadmapping.

The program focused on the science and technology issues for DEMO and the prerequisite research and development (R&D) leading to DEMO. While there have been numerous detailed studies of fusion R&D needs and power plant designs, participants at this workshop sought to identify technical issues of high strategic importance, where the development strategy strongly influences the overall roadmap, and where there are divergent understandings in the world community. It was not a goal of the workshop to propose or endorse a particular roadmap to commercial magnetic fusion, but rather to understand the major technical steps and possible branch points in the roadmap. Emphasis was given to the role of ITER and of possible major fusion nuclear facilities on the path to DEMO, as well as materials development and the physics assumptions used in fusion reactor design work.

The workshop program was organized around four topics: 1) Perspectives on DEMO and the roadmap to DEMO; 2) Technology; 3) Physics-Technology integration and optimization; and 4) Major facilities on the path to DEMO. In the following sections, we summarize some key points from the presentations and discussion in each of these topics.

2. Perspectives on DEMO and the Roadmap to DEMO

The realisation of fusion energy has been significantly advanced by the decision to build ITER. However, beyond ITER, there are still several issues that must be resolved. ITER is expected to provide a reactor relevant physics basis using a tokamak, but alternative magnetic configurations, e.g. stellarators, and alternative scenarios for tokamaks deserve further attention. The greatest challenges, however, lie in the areas of technology and engineering, and the differing approaches to these challenges are the basis for the different fusion development roadmaps presented at the workshop.

Nations engaged in magnetic fusion research and development are now examining the needed programmes to move toward DEMO, i.e., a practical demonstration of electricity generation on a power plant scale, also satisfying a range of socio-economic goals, typically including a closed tritium fuel cycle, a high level of safety, and low environmental impact. A DEMO will be major milestone toward the development of high availability fusion power plants that can be economically competitive with other energy sources. National DEMO programmes differ in their logic and time dimension, although all parties profess a similar timeframe, namely mid-century, for fusion to begin contributing to energy needs. Countries face differing energy needs in the future, and these differences drive different strategies for fusion development. In general the greater the perceived urgency for fusion energy the greater the willingness to take larger steps and larger technical risks in the interest of realizing the schedule for the energy goal. The consequences of the accident at the Fukushima Daiichi nuclear power station caused by the Great Eastern Japan Earthquake of 2011 illustrates how socio-economic and safety factors can affect the future development of fusion energy. In Japan, the "4th Science and Technology Basic Plan" covering the period 2011-2015, has been amended to put significantly more emphasis on safety, prevention of disaster, non-proliferation and nuclear security. At the workshop, the strategies were mostly discussed by leading fusion researchers from various countries providing a research community, rather than a government, perspective. All participants agreed that an increased emphasis on safety will have to be integrated in all the fusion development strategies.

The main line of fusion development in most countries is centred on the tokamak concept. The present and near term future is focussed on the exploitation of existing tokamaks and those under construction (ITER, JT-60SA). Europe and Japan envision that ITER construction will be followed by the construction and exploitation of DEMO, the last step before the first commercial power plant. A dedicated irradiation facility to qualify materials is also included in the strategy. The development of alternative configurations, especially stellarators, is also being pursued, the largest machines being LHD (Japan), with Wendelstein 7-X (Europe) under construction.

China foresees a large increase in energy demand in the next decades. Consequently, the roadmaps presented by Chinese participants are among the most ambitious, in terms of both goals and timescale for next steps. China is considering various alternatives for the development of fusion, i.e., both "pure" fusion machines and fission-fusion hybrids, and various configuration choices are considered. In the short and medium term, according to its "learn-by-doing" strategy, China is considering the construction of a DT machine. The decision to launch the Engineering Design Phase is expected in 2014 and first plasma in 2025.

South Korea has begun pre-conceptual design work on a DEMO device with the intention to start construction in the mid-2020s and begin operation in 2036. Korean participants described a strategy in which the first phase ("K-Demo-1") would be used to develop technology solutions. After a reconfiguration of the internal components, the second phase ("K-Demo-2") would be launched, leading to a full demonstration of industrial scale electricity generation from a fusion plant.

As in other countries, the U.S. and India view DEMO as the last step before commercialization. Both are considering an intermediate step between ITER and DEMO, though there is a diversity of views on the appropriate scope and mission of such a device or, considering the long time involved for an intermediate step, whether such a device is even needed. The U.S. is planning for a "fusion nuclear science facility" (FNSF), where mission elements being discussed range from basic material science to blanket testing to net electricity generation and power-plant maintenance development. India's next step, "SST-2," is intended to provide the first integrated test of some systems being developed for DEMO to act as the first step for verifying the choices being made for DEMO. Some workshop participants question the emphasis on large facilities, arguing for greater emphasis on smaller facilities, simulation, and non-nuclear environments to develop the various technologies. In any case, the role of such science and technology development programs *vs*. that of large integrated facilities in an optimum fusion roadmap needs to be clarified.

The Russian development strategy considers both "pure" fusion machines and fission-fusion hybrids. In the near term, the goal is to upgrade several existing machines (T10, T11M and T15) and to develop a fusion neutron source in preparation for DEMO.

Participation in the construction and operation of ITER is a central element in all the roadmaps presented at the workshop. While the contribution of ITER in several areas of fusion technology is undeniable, all speakers acknowledged the necessity to further develop fusion grade materials, tritium breeding, plasma exhaust solutions, heating and current drive systems (in particular their efficiency), and remote handling. Last but not least, the integration of all this technologies in order to build a reliable machine having high availability is recognised as being, probably, the greatest challenge ahead.

3. Magnetic Configurations and Operating Modes

3.1. Tokamaks

Most of the MFE reactor studies conducted to date are based on the tokamak operating in steady-state, and rely on significant extrapolations from the existing physics knowledge in terms of operation and performance. In tokamaks, steady state operation means non-inductive operation where the plasma current is driven by a combination of intrinsic ("bootstrap") current and current externally driven by heating and current drive (H&CD) systems. The former implies operation at high normalised plasma pressure (β_N), challenging the stability limits. The latter implies a large burden on efficiency since the external H&CD system has to be powered by the electrical energy generated by the plant itself. For the improvement of the overall efficiency of H&CD performance, it is imperative to develop efficient systems, that can run reliably in a steady-state, in order to reduce the resultant recirculating power requirements.

Advanced tokamak (AT) operation, i.e., plasma steady-state operation that relies on high bootstrap fractions, is attractive for reducing reactor size but challenging. Not only must the full current be driven non-inductively, but also the radial profiles of current density and pressure must be self-consistently aligned for plasma equilibrium and stability. Low disruptivity and compatibility with power exhaust requirements are also essential for successful development of any scenario. Studies of advanced tokamak power plants have been documented (ARIES-AT), but the physics basis has not yet been convincingly proven experimentally. For the progress toward steady state tokamak operation, the superconducting devices in operation (Tore Supra, EAST, KSTAR) or under construction (SST-1, JT-60-SA, ITER) will be instrumental. The NSTX and MAST Upgrades will provide progress towards advanced steady-state operation for the spherical tokamak.

Recent work has shown that, if the plasma physics assumptions are required to be consistent with those that can be reliably reached in present day experiments and, at the same time, that the recirculating power is reasonably low, a tokamak DEMO operating with steady state plasma operation is larger than ITER. For DEMO, an alternative to a steady state tokamak would be a pulsed tokamak, relaxing the requirements on the non-inductive current drive. Depending on the Ohmic flux swing and the non-inductively driven current, including the bootstrap current, pulse durations of many hours can be expected. Detailed engineering studies on the feasibility and the boundary conditions for such a device have identified lifetime limitations on major plant components due to thermal and mechanical cycling and the requirements for energy storage between pulses as key issues. For a valid risk-benefit comparison between steady-state and pulsed tokamaks, the pulsed tokamak physics and technology issues should be critically revisited taking into account the studies conducted in the past and identifying the trade-offs that lead to the optimal regime of operation.

3.2. Stellarators

Stellarators (here taken to include stellarators, heliotrons, and other helical confinement systems) have as their main advantage an intrinsically steady state magnetic field configuration. The absence of a strong plasma current and related to this the comparatively benign stability properties make profile control and current drive, at least to the extent required in tokamaks, unnecessary. Small adjustments of the rotational transform may become necessary, but the level of re-circulating power in stellarators is certainly not a major concern. On the other hand the plasma performance of stellarators is far from that of tokamaks, and stellarators still have to verify some of the basic reactor capabilities. In particular, it was noted that while tokamak designs have largely converged on one configuration (D-shaped plasma with poloidal divertor), this step is yet to happen for stellarators.

With the successful operation of the LHD in Japan and the future start of operation of Wendelstein-7X there are good opportunities to make significant advances in the next decade. Nevertheless, the technical feasibility issues associated with the geometrical complexities of stellarators were recognized at the workshop, as was the need to develop quantitative metrics on the following engineering aspects: space requirements for blanket/shield and divertor; coil spacing; bend radius; superconductor type and properties; diagnostic and heating system port and space requirements; maintenance requirements; and costing algorithms for stellarator components.

4. Physics-technology integration and optimization

Compatibility between the physics and technology of magnetic fusion energy is essential for its success as a future energy source. Thus a strong coupling between fusion engineering and plasma science is absolutely necessary. One of the workshop aims was to highlight these requirements and discuss present day knowledge and possible approaches towards a fusion power DEMO.

Concerning urgency and the level of difficulty when envisaging possible solutions, key issues that connect physics and technology very closely are: (1) system performance models, (2) steady state heating and current drive, (3) diagnostics and integrated control, (4) high density operation and fuel cycle, and (5) plasma power exhaust scenarios. Finally all these topics and the maturity of the approaches of finding solutions to them influence possible DEMO design options.

4.1. System Performance Models

The choice of the operating plasma scenario and underlying assumptions made on some important physics parameters, such as density, confinement scaling law, beta, etc. have profound implications on the layout of the device and, in particular, on its size and cost. System codes are used to determine machine parameters. A system code is a program that attempts to model an entire fusion power plant self-consistently subject to physics and technological limitations. The results should therefore represent a realistic and achievable power plant – subject to the assumptions underlying the limitations. Sensitivity to these assumptions and limitations can be tested by using a system code to, for example, compare the effects of an improvement in plasma energy confinement on the final plant design, or to show the consequences of having to limit heat transfer to the divertor to match the high heat-flux materials anticipated to be available.

Over the last decade there has been a proliferation of designs for DEMO and next-step machines with significant differences in their underlying assumptions and, consequently, in their machine parameters. Some of these design concepts were described at this workshop. The state of readiness of all the various next-step facility options needs to be assessed against a common set of criteria. There is an urgent need, in particular, to revisit the physics assumptions used in machine design and their experimental basis and appropriateness for next-step machines being planned for construction in the next ten years or so. There is a need to generate initial physics guidelines for a range of operating modes of interest for reactor design (e.g., pulsed, hybrid, advanced tokamak steady state, stellarator), for use in system codes. To the extent that designs rely on future developments, a clear understanding of the associated risks is needed. As various system codes are being used around the world, it may be worth benchmarking system codes for a number of test cases. A Europe-Japan collaboration is underway in the context of the Broader-Approach Agreement.

4.2. Steady State Heating and Current Drive (H&CD)

The economic feasibility of a steady state tokamak power plant strongly depends on whether practical and efficient current drive solutions can be found. Although steady state operation over many confinement times and several current diffusion times has been shown in present day tokamak experiments, the extrapolation to a tokamak DEMO is in question. Three main reasons can be identified for this. First, these demonstration discharges have not yet shown the level of bootstrap current fraction required for economic operation, mainly because of the limitation to the normalized plasma pressure β. Second, plasma scenarios optimized to suit the requirements to achieve a high fraction of external non-inductive current drive (e.g. low plasma density, low total plasma current) often result in plasma parameters which do not extrapolate to a fusion power plant. The third reason is that the efficiencies of current drive systems, as they are used today, are well understood and are regarded as too low for a commercial fusion power plant. While many current drive schemes nowadays use a variety of current drive systems simultaneously, it is difficult to imagine that more than one could be implemented on a DEMO or a fusion power plant. Foremost the need to breed tritium requires that the space taken up by components other than the breeding blanket be minimized.

Certainly the effort has to be increased to investigate plasma scenarios which combine DEMO relevant plasma parameters (high current, high density) and current drive schemes that extrapolate to a DEMO. This in particular includes the further improvement of the current drive efficiencies, but also the wall-plug efficiency of the CD system. This is another area of R&D where there are still very large gaps to be overcome, and which require large investments and long lead development times. Rather ingenious ideas are required, concerning both the physics of current drive and the technology of the current drive systems, to make improvements. On the technology side, the use of laser beams to neutralize the negative hydrogen ions was presented as a possibility to improve the current drive efficiency of negative ion beam injection. On the physics side the possibility to raise the efficiency of electron cyclotron current drive (ECCD) was discussed, based on an increase of the frequency together with accessing the resonance at large $k_{||}$ and an optimization of the launch angle to still ensure accessibility. For ITER parameters, and using a microwave launch at 240 GHz from the top of the plasma chamber, this scheme is predicted to increase the efficiency of ECCD into the range of neutral beam current drive, albeit at reduced experimental flexibility (which is consistent for a DEMO).

4.3. Diagnostics and integrated control

Besides the magnetic confinement experiments themselves and the plasma heating to access and maintain fusion relevant plasma parameters, diagnostics are key to the understanding of fusion plasmas. Since the early days of fusion research the advances in understanding have gone hand in hand with the progress in diagnostic techniques. Already in ITER, diagnostic design choices are strongly influenced by the requirements from plasma and neutron radiation. In a DEMO certainly many of the diagnostics, as we know them today, will not work. The harsh environment will in many cases require the use of different materials and will restrict the access to the plasma severely. As a consequence the overall diagnostic capability will be limited, in other words the sensors for plasma control will be weaker than in present day experiments. On the other side, high fusion gain in a DEMO or fusion power plant means that also the actuators are weaker. While in a tokamak the control requirements increase when going to steady state, in stellarators the 3D shaping of the magnetic field configuration makes profile control (to a very large extent) unnecessary.

Concerning the extrapolation of diagnostics and control schemes to DEMO, much could be done already in present day experiments. As a first step, diagnostic techniques should be selected according to their applicability to DEMO. Where necessary, diagnostics could be adapted to mock up the DEMO environment, even if this means downgrading their present day performance. Then control schemes could be tested based on these more limited diagnostics. Of course the simulation of high Q is not so easy in experiments which have negligible fusion power, but a possible way to achieve this could be the use of a heating method to simulate alpha heating. Integrated data analysis, based on comprehensive forward plasma models to describe combinations of measurements and subsequently using inversion algorithms to derive the desired plasma parameters, is improving the prospects of finding feasible control schemes for DEMO with a restricted diagnostic set.

4.4. High density operation and fuel cycle

In the optimum plasma temperature range, the power produced by the fusion reactions scales like β^2B^4. Thus, below the β-limit high plasma density is instrumental for keeping the temperature in the optimum range of fusion reactivity ($\sim n^2\langle\sigma v\rangle$). This means that also the cost of electricity drops with increasing density. While in stellarators the density limit is given by plasma radiation, and in the absence of impurity accumulation is set by the bremsstrahlung limit, in tokamaks the Greenwald limit is observed as a more or less generic law, but which is not well understood. It has been possible to raise the central density above the Greenwald density, e.g. by pellet injection, supporting the working hypothesis that the Greenwald limit is in fact a limit of the edge density, i.e. in H-mode the density at the pedestal top . In addition, the increase of density peaking with decreasing collisionality, explained by inward turbulence driven convection, suggests peaked density profiles in ITER and DEMO. These two arguments together improve the prospects that in a fusion power plant central density values above the Greenwald limit can be achieved and maintained.

A specific reason why high density is necessary in stellarators is the unfavourable dependence of neoclassical transport on temperature ($D \sim T^{7/2}$) and the need to keep the fast ion population low. Although in stellarators the Greenwald limit does not exist, here the concern is the tendency for impurity accumulation predicted by neoclassical effects. Stellarator confinement regimes at high density without impurity accumulation have been found, but further research is required to show how or whether they extrapolate to plasma parameters (especially collisionality) expected in a power plant. In tokamaks, on the other hand, high density will always mean operating close to operational or stability boundaries increasing the risk of a back-transition from H- to L-mode and ultimately of a disruption.

Common to both stellarators and tokamaks is the need to reduce heat flux and particle energies reaching the divertor targets. This can be achieved by detachment and a high radiation fraction, also requiring high plasma densities. Related to this, high neutral pressures in the sub-divertor volume, determined by the upstream plasma pressure, the magnetic configuration and the plasma-neutrals interaction, facilitates efficient pumping. At the workshop, ideas were presented to avoid cryogenic pumps, using a combination of diffusion and rotary pumps, which could work continuously without the need of regular regeneration cycles. Part of the investigation is to find out whether the neutral pressure can be raised to values usually observed only close to the strike zones where the plasma flows onto the target plates.

4.5. Plasma power exhaust scenarios

Heat and particle removal from the plasma are crucial for a power plant. Heat can be removed by plasma radiation, distributing the heat more evenly over the plasma vessel, or by parallel heat conduction along the open field lines onto target plates, resulting in a considerably greater concentration of the heat flux. Without any mitigation, the values projected thereby are of the order of many 10s of MW/m^2 which requires highly radiating plasma scenarios to reduce the number. For ITER, a value of 10 MW/m^2 is envisaged; for a DEMO, such values would require rather innovative design solutions, since the high neutron fluences add to the complexity of the problem. Neutron irradiation and helium production by transmutation normally degrade material properties, guaranteeing a reasonable life time of the materials only in a narrow range of elevated temperatures. In addition, the materials facing the plasma have to be compatible with the plasma-wall-interaction requirements.

Handling of the exhaust power from a tokamak power plant requires a high radiated power fraction combined with high confinement and sufficient plasma purity. In the absence of impurity radiation, the power crossing the separatrix in a power plant plasma would be a factor of ~5 greater than that in ITER, while the area of the divertor high heat flux components will probably differ by less than a factor of 2 from those of ITER. At present, the most favoured approach to power handling in a reactor combines tungsten plasma facing components, to ensure an acceptable interval between PFC replacements, with highly radiating edge/divertor plasmas to distribute the exhaust power over a sufficiently large wall area. Impurity seeding must therefore be exploited to generate sufficient edge and divertor radiation.

In tokamaks, the poloidal divertor is the reference concept for particle and power exhaust. From the plasma physics point of view there is increasing evidence that the width of the power deposition does not scale with the size of the device and even decreases with increasing magnetic field, aggravating this problem. Hence, at best the power deposition area increases with the major radius, resulting in a ratio of fusion power to major radius of $P/R \leq 60$ MW/m for a DEMO. Although an example from ASDEX Upgrade was shown where the heat flux could be limited to 4 MW/m^2 at a P/R of 15 MW/m without confinement degradation, this is still considerably below the target value. Highly radiating scenarios demonstrated in present-day machine have several differences compared to DEMO that make the results difficult to extrapolate. Firstly, to achieve a similar radiated power fraction in the edge for DEMO parameters requires a much higher radiation power density, and associated steep temperature gradients. Secondly, power handling in DEMO is achieved by reducing the target power to an acceptable level which is probably below 10 MW/m^2, not close to zero as in present experiments. This means that effects such as thermal instabilities and changes in confinement cannot be directly extrapolated to DEMO.

Additional concerns are transient heat loads. The main focus is now on edge localized modes (ELMs) and disruptions. Mitigation schemes are being developed for both to meet ITER needs. Discussions focus on the question of how to achieve high reliability, because already a small number of such events could cause serious damage to wall components. In particular runaway electrons are expected to always appear in ITER or larger devices in the course of a disruption if not avoided by some mitigation scheme.

Stellarators have the advantage that disruptions do not occur. The plasma can be terminated by a radiation collapse, but the confining magnetic field remains intact. ELMs have been observed in stellarators, but here the data base is too small to answer the question if they pose a similar problem as in tokamaks. Divertor solutions in stellarators strongly depend on the type of stellarator and the experience with such divertor concepts is rather small. LHD has just started to test the closed helical divertor which is built into the heliotron configuration with a pair of continuous helical coils. Another example is the resonant island divertor of the low magnetic shear configuration (to be tested in Wendelstein 7-X) with long connection lengths, which, compared to tokamaks, is projected to increase the power deposition width by a factor of ten. However, the heat flux varies along the helical direction and the uniformity of the distribution is very sensitive to resonant magnetic field perturbations. Localized power deposition by energetic ions, including alpha particles, on poorly confined orbits is an additional complication. Integrating a three-dimensional divertor arrangements into a feasible DEMO design is an outstanding challenge.

Generally, it was felt that fusion research is lacking experience with simultaneous high power density, steady state and hot walls operation (with or without D-T). In this context, proposals for plasma wall interaction or divertor test facilities were discussed, ranging from high-power linear or mirror devices to toroidal confinement experiments focused on plasma-material interactions. The discussion identified the possibility to develop a dedicated programme on ITER to investigate ELMs and disruptions, and not solely rely on the input until ITER starts operation, as the associated problems in ITER are of a different order of magnitude.

5. Technology: Basis, Gaps, Risks and Facility Needs

The technical basis for designing a next-step DT burning plasma experiment has greatly expanded during the last two decades thanks mainly to remarkable improvements in plasma performance and control in today's machines and advances in various areas of physics and technology. Integrating and extending these advances toward long-pulse or steady-state burning plasmas is now the focus of international tokamak research, including new devices in China and Korea, facilities under construction in India and Japan, and ITER.

However, beyond ITER there are still several technology issues that must be addressed and resolved. These include the development of sound solutions for plasma power exhaust, the qualification of neutron-tolerant materials for in-vessel components, the development of reliable power extraction and tritium breeding components and optimised remote maintenance schemes for high machine availability.

Knowledge gaps in these areas were clearly identified at the workshop and a brief summary of the status is provided below together with an analysis of the prospects. A number of testing facilities were also discussed for their potential to bridge the gap from present knowledge to that required to construct a DEMO. They are briefly mentioned below. A general concern is that attempting to cover them all in a single device may lead to unacceptable risk of failure. Some of these facilities should be available and operated well before the start of the construction of DEMO, in order to validate fundamental design choices and confirm their performance in a realistic environment.

5.1. Plasma Power Exhaust Technology

The peak power load on the divertor target is seen to be a key constraint on the design of all the next-step facility options that were discussed at the workshop and the power exhaust will ultimately determine the reactor size and choice of the operating scenario of DEMO.

Significant progress has been made during the last two decades on the development of technologies for divertor high-heat-flux-components cooled either with water or with helium. In the former case prototypes fabricated either with carbon or with tungsten have been successfully tested under cyclic loads up to 20 MW/m^2. In the latter case solutions have been found that can withstand 10 MW/m^2 for a large number of cycles. It should be recognised that these represent upper technological limits, set by the intrinsic limitations of the thermo mechanical properties of the limited numbers of materials suited for the fusion environment. In addition, in the case of exposure under large neutron irradiation fluences, such as those expected in DEMO, the power handling limits above must be prudently reduced to ~10 MW/m^2 in the case of water cooled components and to ~5 MW/m^2 in the case of He-cooled components.

As a plasma-facing first wall material, tungsten is regarded as the most promising at present. However, taking helium cooling in the divertor as an example, the combination of tungsten and presently available structural materials limits the acceptable heat flux to 5 MW/m^2. This is only half the value that is readily available from present day technology without neutron irradiation. In addition, heat load and tungsten erosion are closely coupled, whereby the erosion limit (~2 mm per 3 years) is consistent with the 5 MW/m^2, requiring in addition a low divertor plasma temperature in front of the target of less than 5 eV. To achieve such values the overall radiated power fraction must be further increased and divertor detachment is indispensable.

In addition to developing realistic high radiation scenarios it is also appropriate to look into solutions using advanced magnetic configurations (e.g., higher flux expansion and expanded boundaries), or advanced plasma facing materials targets, such as liquid metals, which could reduce the radiation requirements. Emphasis should be placed on the heat exhaust capability of the solutions proposed during normal conditions, but also with a view to the occurrence of possible uncontrolled transients. More specifically, feasibility studies aimed at objectively and rationally determining and comparing the strengths and weaknesses of the various configurations (e.g., snowflake, super-X divertor, as compared with the conventional poloidal divertor) are urgently needed. In particular, an analysis of the implications of the target configuration on the mechanical integration and maintenance scenarios, and an investigation of the design integration and engineering constraints arising from the use of large current coils in the divertor region for the case of the super-X divertor, or of high velocity liquid metal films, in the case of liquid metal targets. The final concept selection of the divertor bears strong impact on the machine design, parameter selection and operation scenario development.

Hence, this problem must be tackled from the outset, and until uncertainties are significantly reduced through a well-focused and vigorous R&D program, any conceptual design proposals remain questionable.

Another area of heat transfer that needs significant coupling with plasma operation development is the chamber wall heat flux. Maintaining adequate transparency to neutrons for tritium breeding requires a thin first wall, and the heat removal capability of a helium-cooled first wall design would be limited to ~1 MW/m^2. This means that the chamber wall of a DEMO can only withstand uniform radiation from the plasma and high power ELMs will have to be avoided.

One of the unanimous conclusions from this workshop was that initiatives are urgently needed to reassess the gaps and facility needs for the development and qualification of solutions to solve the problem of the power exhaust (including scenarios) in DEMO. This should go together with determining capabilities/options that exist to reduce these gaps using existing and planned machines including ITER.

5.2. Materials

One of the main technical challenges for the successful development of fusion energy is the development and qualification of resilient structural materials for the first-wall, high-heat flux components and breeding blanket components. The neutrons produced by fusion reactions, which are more energetic than those produced by nuclear fission, lead to unique damage problems for the materials surrounding the fusing plasma, in addition to the production of larger amount of helium that causes increased hardening and an increase in the ductile-to-brittle transition temperature (DBTT) in particular of some candidate structural steels (e.g., 9Cr steels).

Reduced Activation Ferritic-Martensitic (RAFM) steels such as Eurofer are presently the reference structural material for DEMO and for ITER TBMs. They can be manufactured on industrial scale with sufficient purity and good properties. However, RAFM steels have a narrow temperature operating window: they must operate ≥ 350 C to avoid radiation embrittlement and < 550 C to avoid loss of strength, creep, and rupture issues as well as He-induced swelling at high dpa values. The oxide dispersion strengthened (ODS) variety of RAFM steel should reach the 650 C operation limit, but the development and qualification of this material is challenging within the next 20 years.

Due to the lack of fusion test facilities that allow irradiating materials in fusion relevant environment at fluence levels close to those expected in future DEMO and power plants, only limited irradiation tests for structural materials have been performed in fission reactors. Moreover, the radiation defect accumulation behavior (and accompanying property degradation) in structural materials is greatly influenced by the neutron energy spectrum (particularly due to varying generation of H and He). In a DT fusion system, the neutron energy spectrum in the first wall and front region of the blanket are significantly harder than in fission reactors and will produce much higher levels of transmutant H and He. For example, the ferritic steels irradiation data base from fission reactors extends to ~80 dpa, but at very low H and He generation levels (only limited simulation of He in some experiments).

The knowledge on this field was thoroughly reviewed at the workshop by fusion materials specialists. In general, there was a broad consensus that setting an end-of-life irradiation fluence target of ~50 dpa would be reasonable for an early DEMO baseline design. For a commercial power plant the end-of-life target is viewed to be in the range of 150 dpa, driven by economic considerations. By aiming at ~50 dpa some of the R&D and testing requirements for structural materials would also be similar to those of many Generation IV fission concepts that also need high temperature steels, so there is opportunity for improving synergies.

With regard to the available irradiation database, consistent information was presented and discussed at the workshop, showing that sufficient irradiation effects data exists for RAFM (up to 10 dpa, up to 100 appm He) to permit reasonable prediction of performance for a DEMO with limited fluence. Helium embrittlement, irradiation creep, volumetric swelling, and phase instabilities occur at >10 dpa and data from fusion-relevant neutron sources and non-nuclear testing facilities are still needed in the intermediate-dose environment (>10 – 60 dpa). Material experts state confidence that RAFM steels will work satisfactorily to ~10 dpa and to ~100 appm He at irradiation temperature greater than 350 C. However this needs confirmation from dedicated experiments and facilities. One of the shortcomings of existing facilities is their limited capability to generate large amount of transmutations in reasonable volume and at appropriate rates.

Workshop participants advocated a much closer integration of materials science with the structural analysis and plasma physics aspects of the design process. With regard to the need to improve the engineering database for RAFM steels, the scientific community should reassess the challenges, risks, and time line associated with the International Fusion Material Irradiation Facility (IFMIF) and to begin planning which common projects are necessary for efficient progress. While IFMIF remains indispensable for qualifying material at very high fluences expected in a fusion reactor power plant, DEMO would benefit from a focussed accompanying programme exploiting fission reactors (e.g., with isotope tailoring), spallation neutron sources, ion-beams, modelling, and a dedicated exploitation of IFMIF technology being developed by Japan and Europe under the Broader Approach. Options to reduce the risks and costs of IFMIF were also discussed.

5.3. Reliability, Availability, Maintainability, and Inspectability (RAMI) of Core Components

Reliability represents a challenge to fusion, particularly for the core components (e.g., blanket/divertor and other plasma facing components). Reliability and availability evaluations should be applied from the very beginning of the design process but have only been partially addressed in the past.

The breeding blanket remains a critical component, one that must operate safely and reliably in a harsh environment. Reactor core components such as the blanket and plasma-facing components will have little or no reliability data from ITER or other facilities. Current estimates using available data from fission and aerospace for unit failure rates and using the surface area of a tokamak show probable mean-time between failures (MTBF) for the blanket in the range of only ~0.01 to 0.2 years compared to required MTBF of many years.

Blanket-related research, development, and design activities have been ongoing for many years in domestic programs, but to date no fusion blanket has ever been built or tested. Hence, their integrated function and reliability are by no means assured. ITER presents the first opportunity to test blanket materials and components in an actual fusion environment. ITER test blanket module (TBM) testing represents a critical step toward establishing the principles and technologies of tritium self-sufficiency and energy extraction, on which the feasibility of deuterium-tritium fusion energy production relies. At the same time the ITER TBMs and corresponding ancillary systems can provide initial components and operational reliability data for different DEMO blanket concepts. Nonetheless, further blanket development beyond ITER TBM toward long time performance and higher neutron fluence, high reliability, and lifetime capability will be needed.

There was a wide consensus on the need to establish a vigorous reliability growth and risk minimization program for the design and development of in-vessel components for DEMO (beyond obviously demonstrating their engineering feasibility), to identify synergies with the ITER RAMI program, and to investigate what is done in other fields (e.g., nuclear, aerospace). However, presentations at the workshop showed that consensus on the best approach is lacking at present. Some in the US community propose to pursue reliability development based on testing in a fusion nuclear science facility, while other emphasize the use of smaller facilities, test stands, and computer simulation in a distributed program and the use of Technical Readiness Level (TRL, commonly used in other fields) as a metric for assessing development progress and providing a framework for R&D and involvement of industry. China favors instead a more direct approach by building a next-step fusion nuclear facility as soon as possible without extensive optimization, and using it to develop both the technology and operating experience.

Independent of the approach, there is the need to begin thorough quantitative RAMI assessments as it may have significant impact on the design. While it is clear that some of the components are one-of-a-kind, lack reliability data, or have not yet been prototyped or life-tested, much can be learned from an analysis based on engineering judgment and experience with similar components. Comparison of a bottom-up calculation (even if partially based on assumptions) of reliability with the top down reliability allocations for the system is essential if major system, subsystem and/or component design perturbations or maintenance requirements are to be avoided in the future. Usual kinds of fault analysis, e.g. failure modes and effects analysis (FMEA), would determine the requirements, e.g. redundancy of equipment to achieve a set level of availability. All of this needs to be thought through and taken into account in the final design choices.

Remote maintainability, which has a strong impact on machine availability, will affect in depth the design of many components and interfaces and must be given sufficient attention from the beginning, with proper resources and involvement of industry to conduct a full review of the requirements and solutions for remote maintenance as well as an evaluation of the impact on machine availability. Proposed design solutions must be fully remotely maintainable and a significant amount of time-consuming demonstration R&D is required, often involving design iteration and changes before fabrication can begin.

In addition, the validation of the structural components of DEMO requires design criteria developed specifically for those components and the unique conditions at which they are operated. Licensing and validation of the design must be a necessary consideration throughout the DEMO design development. The validation of the structural components of DEMO requires design criteria and one has to engage early on with the ASME (American Society of Mechanical Engineers), AFCEN (Association Française pour les Règles de Conception, de Construction et de Surveillance en Exploitation des Matériels des Chaudières Electro Nucléaires), or other fusion specific design code standards from the outset to drive the evolution of design criteria, as well as to understand data requirements.

6. Major Facilities on the Path to DEMO

For the next two decades, the central and most important element of MFE fusion research will be ITER. By creating and controlling a 500 MW burning plasma for extended durations, eventually approaching continuous operation, it is expected that ITER will make large scientific and technological advances toward the realization of fusion energy. Successful construction and operation of ITER will provide very valuable information for DEMO relevant physics and technologies, such as superconducting magnets, cryogenic systems, heating and current drive, fuelling, tritium handling, remote handling, plasma control, and burning plasma diagnostics. Even so, ITER will not be the last development step before a commercial fusion power plant can be constructed. Both with regard to improvements in physics and technology, even more so in efficiency and RAMI, further progress will have to be made and, ultimately, demonstrated in DEMO which will be the essential link towards a commercial power plant.

At the workshop, several major new facilities were discussed that would take a major step toward DEMO by addressing scientific or technical problems that cannot be fully validated in ITER. These facilities focus on major DEMO technical issues, such as managing the plasma material interface under high heat flux condition, divertor power and particle handling, component testing, materials development, power plant maintenance, and net electricity generation. Stellarator approaches are also proposed, in view of their significant advantages for a fusion reactor, such as disruption-free and steady-state operation, compared with tokamaks. Any of these new facilities could be very useful for narrowing the gaps and mitigating some risks.

Further comparison of the proposed facilities should be conducted to identify their roles, missions, strengths, and weaknesses relative to one another and to evaluate any further risks which might compromise the fusion development path to DEMO. In planning the roadmap to DEMO, a key challenge is to identify the needed major integrated facilities besides ITER, and to clarify their roles in an optimum development program *vs.* those of smaller facilities and programs needed to development the physics and technology basis.

6.1. ITER and existing facilities

Successful construction and operation of ITER is viewed as a prerequisite to any follow-up device and/or project. However, there is a need to critically evaluate the information expected during operation of ITER in some critical DEMO design areas, and the leverage that one still has on the operation of ITER to address key questions for DEMO. For example the use of tungsten in DEMO would probably require a full demonstration of a full tungsten wall in ITER, but in this case, the feasibility of such a wall exchange in ITER needs to be demonstrated. Also, the role, albeit complementary, of some of the existing tokamaks to help address some of these gaps is still open to discussion.

The substantial time and cost requirements for ITER's scientific goals require that the plans for ITER construction and for future experimental campaigns must be carefully targeted for optimum scientific value, and must be validated for a high prospect of success. Efforts must be made within ITER-IO and parties towards successful construction of ITER within the planned schedule and budget. A well-defined set of specialised, more efficient and flexible fusion devices as well as technology and computing facilities, is needed and should be adapted to the ITER requirements. The main existing facilities could be very useful for solidifying the ITER physics basis in such areas as advanced tokamak operation, ELM and disruption control, plasma-wall interaction, and steady-state operation. JET will play a key role with its new ITER-like wall regime, as will medium sized tokamaks of similar geometry such as ASDEX Upgrade and DIII-D. The performance of a tritium campaign in JET would represent a step in the development and design validation of fuel cycle components for DEMO. Newly operating superconducting-magnet tokamaks, such as EAST and KSTAR, could explore advanced tokamak physics under long pulse and even steady-state condition in the future, together with JT-60SA. The LHD and Wendelstein 7-X experiments will advance the physics understanding of stellarators. Some fusion technology facilities and upgrades are also useful for crucial R&D to ITER on materials, fuel cycle and remote handling.

6.2. DEMO Divertor Test Bed

A prime uncertainty in next-generation devices is the divertor performance. For ITER, the divertor power handling limit has played a more critical role in the operational scenario definition than beta limit and energy confinement constraints. Thus a test bed for divertor, for conditions as close as possible to those expected in DEMO, would be highly desirable. The main mission requirement of this facility would be to demonstrate for a given geometry and plasma scenario that the heat flux during normal operation and off-normal transients can be controlled with existing technologies or modest extrapolation thereof. This facility would need to have the capability to enable integrated physics and technology testing and there is a need to address from the outset the problem of the geometrical flexibility needed in a testing device to accommodate some of the innovative solutions or plasma configurations being considered.

6.3. 14 MeV Neutron Irradiation Facilities

The proposed International Fusion Materials Irradiation Facility (IFMIF) remains an important facility to investigate radiation damage in materials. However, limiting the end-of-life irradiation requirements for a first set of core components in a DEMO to ~50 dpa would relax some of its irradiation testing prerequisites in contrast to those for a fusion power plant (>150 dpa). The construction of such a reduced-scope DEMO without the results from IFMIF could entail some risk, but that the general consensus at the workshop was that the level of risk may be acceptable. Further analyses would be needed to determine if IFMIF is on the critical path to such a DEMO and to determine its implication for the step from DEMO to the power plant. However, independent of IFMIF, a DEMO design and R&D programme would benefit from an accompanying focussed material irradiation programme exploiting fission reactors (e.g., with isotope tailoring), ion-beams, modelling, and the Japan-Europe Engineering Validation and Engineering Design Activity (EVEDA) for IFMIF. Additionally, a much closer integration of materials science with the structural analysis-design process is necessary.

6.4. Next-step Fusion Nuclear Facility

Design options for the next step towards fusion energy were presented at the workshop. The designs and plans varied in their degree of maturity and also reflected divergent opinions on how to bridge the gap between ITER and a fully functional fusion power plant. This included DEMO facilities in which the capabilities would developed in stages, fusion nuclear science facilities (FNSF), pilot plants, the helical advanced stellarator (HELIAS), the heliotron reactor (FFHR), and fusion-fission hybrids. These options all envision a continuously operating DT burning plasma with high neutron wall load ($\geq$1 MW/m^2), but they vary in the scope of their mission from materials science and component testing in fusion plasma environment, to net electricity and maintenance prototyping. A comparison of the different design concepts should be conducted to identify strengths and weaknesses of the various options, and their ramifications for the development path to DEMO. The evaluation of the mission of such a facility and its role in reliability development is necessary, together with a feasibility study, to evaluate the risks and the maturity of solutions proposed to the problems of power handling (divertor in particular), steady-state operation, tritium breeding and processing, power extraction, availability, and maintenance.

The physics basis of these facilities was discussed in some detail. However, all of them constitute a significant extrapolation from the existing physics base on which ITER is being built. Examples are advanced tokamak FNSF or DEMO, the component test facility based on the spherical tokamak, or a pilot plant based on the compact stellarator. The HELIAS and FFHR are extrapolations from Wendelstein 7-X and LHD, respectively; their objectives are to fill the physics basis for any future development in the stellarator direction. For the tokamak line, a key issue is whether to base the design on a pulsed or steady-state operating scenario. The former would be the more conservative choice, requiring less extrapolation from the ITER basis, while the latter may lead to a more attractive end product. For the stellarator the physics and technology basis overall is less developed. While the stellarator offers more degrees of freedom to find an optimum configuration, it has the additional complication that for each configuration its flexibility is much smaller (for instance the magnetic shear is basically given by the coil configuration). Thus, one has to be much more careful in selecting the optimization or design criteria when building a stellarator.

In this context the discussion also included ideas for for building future nuclear devices in a staged approach, aiming at further developing the plasma physics, materials science, and technology while gaining experience from operating such a device and also extending its nuclear capability step by step (e.g. upgrade of chamber wall and blanket, divertor, materials, heating and current drive systems, etc.), similar to what has been done in the past with most magnetic confinement experiments. A key consideration for next-step facilities is the degree to which the capabilities can be staged to carry out a series of missions, or to take advantage of new knowledge, in the same facility. The need to include sufficient flexibility in the design to accommodate improvements in plasma performance and design improvements of core components was generally supported, but is not clear to what extent this can be achieved. A good example of staging is JET, in which substantial increases in auxiliary power, an internal poloidal divertor and significant remote handling capability were foreseen in the design phase. Existing devices have been successfully operated, and performance substantially improved through significant design modifications and machine upgrades throughout their lifetimes, but for a nuclear fusion reactor flexibility is likely to be more limited. Careful design studies are needed to determine the realistic possibilities for flexibility and staging.

7. Conclusions

The workshop was instrumental in identifying a short list of important issues, for which there is a need and an opportunity to follow up internationally with further discussion and joint work among specialists.

The assumptions used in fusion design codes.

Fusion reactor designs depend sensitively on physics and technology assumptions used in the design. For example, assumptions about the bootstrap current fraction, overall current drive efficiency (wall-plug to plasma), maximum divertor heat fluxes, radiation fraction, and whether or not a tokamak can operate well above the no-wall stability limit have high leverage on the design. Some assumptions presume large advances over the long-term. There is need to clarify which assumptions can be used as a design basis for next-step facilities that could be ready to start construction in the next ten years or so.

The strategy for fusion materials development.

Irradiation testing for material qualification is a necessity, and may determine the critical path, for developing structural and plasma-facing materials for DEMO. The fusion community has long embraced the idea of an International Fusion Materials Irradiation Facility (IFMIF) to provide a fusion-relevant neutron source. The construction of prototypes of its main units (ie., prototype accelerator, lithium target and test cells) is in progress as part of the IFMIF-EVEDA (Engineering Validation Engineering Design Activities) activities under the Broader Approach. The irradiation testing requirements to satisfy materials prerequisites for next-step fusion nuclear facilities may be much less than for DEMO, depending on their mission, and may be satisfied with facilities that can be made available in the near term. There is a need for fusion facility planners and materials specialists to identify areas where structural code design methodologies are missing (high temperature and irradiated), and the paths that must be resolved to support regulatory licensing acceptance and to develop a plan for materials development and facility construction that is self-consistent.

The strategy for blanket development.

Tritium self-sufficiency is a requirement for fusion development beyond ITER, so breeding blankets will be a necessity for essentially any next-step fusion nuclear facility, regardless of its mission. The blankets and associated tritium processing systems comprise a complex system with multiple functions, materials, loads, and environmental conditions. There is a need to devise a strategy for blanket technology development that will lead to self-consistent solutions, addressing both materials and engineering issues and extracting maximum benefit from the ITER Test Blanket Module (TBM) program.

The strategy for plasma exhaust solution development.

The heat and particle exhaust requirements for high duty-factor fusion devices go well beyond those of ITER. There is a need to develop the physics and technology of plasma exhaust, including materials, divertor configurations, neutral gas pumping, and operating scenarios, leading to solutions that are both self-consistent and compatible with both plasma performance and tritium breeding. The roles of existing plasma devices, new non-nuclear facilities, and fusion nuclear devices in an optimum development strategy need to be understood.

The requirements and state-of-readiness for the various next-step facility options.

Fusion community representatives from all the ITER partners presented plans for next-step fusion nuclear facilities, some with strong government encouragement. Most plans call for construction to start in the 2020s and proceed in parallel with ITER operation. There were also presentations addressing the readiness gaps and the development needs for key fusion technologies, but a self-consistent plan for closing the gaps in time to support the facility schedules does not exist. While the policy environment and resource availability can vary from country to country, the question of technical readiness will be judged by the international community of fusion scientists and engineers. There is a need for national programs to develop their design options in more detail, and for the international community to begin a critical examination of both the facility plans and technology programs, and foster work that will reconcile the two.

The high level of international cooperation in the planning of the workshop, and the valuable exchange of information that took place, showed that there is a need to continue and to focus the international discussion concerning the scientific and technical issues that determine the fusion roadmap. As a first step it was suggested that an international activity be organized under appropriate auspices to continue the discussion and foster international cooperation on these issues. Taking note of the interest in the workshop, and based on these conclusions, the International Atomic Energy Agency (IAEA) organized a consultancy meeting, seeking advice on actions that IAEA could take to promote international cooperation in developing fusion energy. Looking ahead, the scale of programs and facilities needed to advance toward DEMO readiness motivate a broadening of the scope of international collaboration in fusion development. Certainly success in ITER is essential, not only because of the importance of its technical mission, but also to demonstrate a capability to successfully collaborate internationally in a large, strategically important step in fusion development. Effective models for international collaboration in the planning and execution of a broad range of programs to advance fusion also need continued development.

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