Fusion nuclear technology and materials: status and R&D needs

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Abstract

The importance of fusion nuclear technology has grown in recent years due to advances in large tokamaks and a strong international movement towards a “next step” device such as ITER. Programs around the world have made major advances in the development of in-vessel reactor components. However, much further R&D will be required in order to develop attractive DEMO components. The realization that testing in a next-step fusion test reactor may be required in as little as 10 years has provided a new sense of urgency. In this paper, the current status and R&D needs are surveyed for the key in-vessel components, including blankets, plasma-facing components, and tritium systems. Special needs in the areas of neutronics, materials development and safety are highlighted.

1. Introduction

The Fourth IAEA Technical Committee Meeting and Workshop in 1986 marked the first time that fusion nuclear technology (FNT) was included as a separate and distinct topic for discussion, indicating the increased importance and effort devoted to this expanding discipline [1]. Since 1986, ITER has emerged as a major focal point of fusion programs worldwide. While ITER was specifically excluded as a topic of this workshop, it has broad-reaching effects over all aspects of fusion reactor design and technology. The near-term goal of ITER construction provides a new urgency to the development of fusion reactor nuclear technologies. During the period of time since the Fourth IAEA Workshop, DEMO reactor component and systems developments also have flourished. DEMO R&D programs are impacted by the ITER activity as well, since ITER is expected to provide one of the few opportunities for integrated testing of DEMO components.

Important discoveries and successes have given us increased insight into fusion nuclear component behavior for magnetic fusion energy. This includes first walls and blankets, tritium fuel systems, radiation shielding and plasma-facing components. These advances have significant implications on the feasibility of fusion reactors; however, a great deal more work is needed to
develop components with reliability and safety features that will prove acceptable for ITER, DEMO and future power reactors.

For example, recent successes on solid breeder blanket experiments and model development show that tritium release is far less serious a concern than was previously suggested. Novel techniques for thermal control have been proposed and demonstrated. Most of the remaining concerns for this class of blanket relate to attractiveness (i.e., reliability and performance limits) rather than feasibility.

Great strides also have been made with liquid metal blankets. The most serious feasibility issue for self-cooled liquid metal blankets is magneto-hydrodynamic (MHD) effects. Numerous experiments have been performed on most of the major geometric elements of self-cooled blankets. Together with strong model development programs, these have helped reduce uncertainties in predicting the MHD pressure drop restriction on heat removal. Observations of new phenomena, such as enhanced diffusivity near the walls, indicate that performance may be better than expected in some cases. A major change in direction for many of the liquid metal R&D programs is the use of electrical insulators to reduce the MHD pressure drop. This has been motivated primarily by increasing magnetic field strengths of reactor designs. Insulating coatings bring a range of new issues to the forefront of liquid metal blanket R&D, including chemical kinetics, radiation damage effects and insulated duct MHD.

Neutronics activities during the past decade have helped to reduce uncertainties in predicting the various critical neutronics parameters, such as the tritium production rate, bulk heating and activation. In some cases, large discrepancies have been observed between experimental predictions and the experiments themselves. In cases where these discrepancies have a major impact on reactor performance and safety, new experiments have been proposed to quantify margins to be implemented in design. While neutronics is a mature field of study, a large amount of further R&D is required to guarantee tritium fuel self-sufficiency and to provide an acceptable margin of safety on the key performance, environmental and safety parameters.

Most of the work on plasma-facing components (PFCs) has been undertaken in conjunction with plasma confinement experiments, such as JET, TFTR, Tore Supra, Textor, etc. Because of this, the emphasis has been on providing components which perform satisfactorily at zero neutron fluence over short periods of time and very low availability. The technology for long-pulse operation at reactor-relevant power densities does not exist. For example, in ITER-CDA, the average heat load to the divertor was about 20 MW m⁻². No credible solution exists to remove this amount of heat in a component that can withstand high erosion rates over a reasonable lifetime. Recent trends have been to reduce the requirements on the divertor. For example, the radiative divertor concept promises to reduce peak power densities by a factor of 10. As with other nuclear components, the key to success is developing components that can operate reliably over a reasonable lifetime in the fusion environment.

Tritium processing system R&D is probably the most mature area within the field of FNT. Major experiments in the USA, Japan and Europe have demonstrated many of the key technologies for tritium fuel processing. Both JET and TFTR now have actual operating experience with tritium in tokamaks. The most serious remaining issues are related to the extraction of tritium from the various breeder materials still under consideration.

In general, commercially available materials will not perform satisfactorily in a fusion power plant environment. These include structural materials, breeders, plasma-facing materials and specialty materials such as electrical insulators. The development of new materials is a major undertaking; the demanding fusion environment places a greater challenge on materials development than on any previous energy technology. At present, candidate materials have been identified; however, in the absence of fusion irradiation data, it is impossible to extrapolate to reactor-relevant fluence goals.

Safety is increasingly becoming a major factor in design and R&D. This has led to more emphasis on helium as a coolant and on reduced activation systems. In parallel with advances in safety, efforts on improving reliability and maintainabilit-
ity have increased, together with better tools to predict reactor plant availability and licensability. Again, ITER has forced us to assess the real possibility of building a fusion reactor, with all the practical concerns of a large, central-station power plant.

Fusion development pathways have become a major topic of concern relative to FNT. Neutron test facilities are mandatory for fusion nuclear technology R&D, and will be major cost and timing drivers for the future of fusion development. While ITER is a highly desirable step, it is generally agreed that one or more complementary facilities will be required prior to DEMO. The need for a fusion materials test facility has been clearly documented in the past. New emphasis is being placed on complementary component testing in volumetric neutron source facilities, such as VNS.

In the remaining sections, the status and remaining R&D needs are summarized for the primary fusion nuclear components and disciplines, including solid breeder blankets, liquid metal blankets, neutronics, PFCs, tritium processing systems and materials. This summary covers a broad range of topics and was compiled using the expert judgment of several contributors. As it is not intended to be a thorough review article, only minimal references have been provided.

2. Solid breeder blanket status and R&D needs

Several material and design configuration options still exist for solid breeder blankets. The past seven years have been marked by an increasing level of detail in the designs and analysis, rather than a continued widening of the options considered. Whereas in the past most of the blanket R&D focused on materials tests and simple element geometries, significant component R&D programs have been initiated in the past decade, and a number of important results have emerged already.

Table 1 summarizes the major material candidates for the coolant, breeder and structural materials. In Japan, layered pebble-bed concepts are under development for both ITER and power reactor applications [2]. Both water and helium coolants are considered for power reactors. In Europe, both pin-type BIT and layered BOT options are still under investigation [3]. In the USA and Russia, solid breeder blanket design efforts are less focused, and a number of design options still exist.

The major classes of issues for solid breeder blankets include the following:

- breeder and multiplier tritium inventory, recovery and containment;
- breeder–multiplier–structure mechanical interactions;
- thermal control;
- purge flow;
- structural behavior, failure modes and reliability;
- corrosion and mass transfer;
- off-normal and accident conditions.

Significant results have been achieved in many of these areas. In particular, several new tritium-release experiments have been performed in the past few years, such as BEATRIX-II (USA/Japan/Canada), VOM (Japan), CRITIC-II (Canada), EXOTIC (EU), LIBRETTO (EU) and COMPLIMENT (EU). Results of these experiments show that tritium release is better than expected from early predictions, partly because of the use of isotopic swamping. Sophisticated models have been developed, including many of the complex time-dependent transport and surface processes involved. There is currently considerable confidence that tritium-release rates and inventories will be acceptable. Further R&D is needed to resolve burn-up and lifetime effects.

Previously, basic thermophysical and mechanical properties were still an issue; recent work has moved beyond basic properties and begun to investigate the effects of irradiation in breeders and
beryllium, and the effects of element interactions in submodule configurations. R&D results have tended to favor pebble-beds over sintered blocks for the breeder and multiplier. Several recent experiments have measured the thermal and mechanical behaviors of beryllium and ceramic breeder pebble-beds. Thermal control is an important issue, because the elements of a solid breeder blanket have a variety of temperature limits which must be maintained throughout operation, including some allowance for spatial gradients and fractional power operation.

Based on more detailed investigations, many researchers now agree that beryllium will be required for fuel self-sufficiency for all solid breeder designs. Partly because of the greater perceived need for beryllium and the extensive use of it in ITER (including the first-wall armor), beryllium R&D programs have flourished in recent years. Studies of fabrication, bonding techniques, tritium release, radiation damage, and thermal and mechanical performance have been undertaken.

One of the most serious concerns with solid breeder blankets is reliability. Most designs involve a large number of parts (coolant channels, purge channels, manifolding) which require extensive weldments within the radiation field. Designs which minimize complexity are essential. An accurate assessment of failure modes and rates will probably require an extensive component test program within a fusion environment.

3. Liquid metal blanket status and R&D needs

The R&D issues of liquid metal blankets are largely different from the issues involving solid breeder blankets. Liquid metals are immune to radiation damage and do not require in-situ tritium extraction. Therefore, liquid metal breeder blankets do not require an irradiation test program as large as that required for solid breeder blankets. Another important difference is that, in general, no beryllium multiplier is needed for liquid metal breeder blankets using either lithium or Pb-Li as the breeder material.

The crucial point for liquid metal breeder blankets, especially for self-cooled concepts, is the large influence of the strong magnetic field on the liquid metal flow. MHD effects can cause large pressure drop, influence flow profiles and distribution, and degrade heat transfer. Therefore, an extensive research program is required to investigate the relevant MHD issues, both theoretical and experimental. Other areas requiring R&D work are tritium control, purification of the liquid metal breeder, and the behavior of the liquid metal during plasma disruptions.

The main R&D issues of liquid metal breeder blankets are described below and can be summarized as follows:

- **MHD**—insulators; flow distribution; heat transfer;
- **tritium control**—extraction (lithium); permeation (Pb-17Li);
- **purification**—corrosion products; polonium and bismuth; chemistry for self-healing coatings; transient electromagnetics.

### 3.1. Magnetohydrodynamics

It is generally agreed that self-cooled liquid metal blankets need electrical insulators to decouple the flowing liquid metal from the load-carrying wall. The preferred solution is an electrically insulating coating at the duct-wall surface. The development of such coatings is a crucial issue, involving fabrication technology, testing in flowing liquid metal, and irradiation tests. The main issues of insulator development can be summarized as follows:

- candidate materials;
- technology;
- imperfections—MHD issues; generation rate; healing;
- radiation-induced degradation—damaging irradiation; ionizing irradiation; electric field.

During the last 10 years, liquid metal MHD research on fusion applications has been concentrated on problems characterized by thin conducting walls. This has led to computer models based on a core flow method, describing the pressure drop and flow distribution. An extension of these models to cases where the duct walls are electrically insulated and its experimental verification are required for application in blankets employing insulating coatings. With regard to heat transfer,
it is generally assumed that the strong magnetic field in a fusion blanket suppresses all turbulence in the flowing liquid metal. However, there are indications that, under certain conditions, a special kind of two-dimensional turbulence may improve the heat transfer considerably. This could lead to more attractive blanket concepts, especially if the eutectic alloy Pb–17Li, characterized by a rather low thermal conductivity, serves as the breeder material. Suitable experiments and model development are required to investigate methods for such heat transfer enhancement.

3.2. Tritium control

A distinction has to be made between the two candidate breeder materials, i.e. Li and LiPb, because the tritium solubility is different by many orders of magnitude. The high solubility in lithium results in a very low tritium partial pressure, avoiding the problem of tritium permeation losses. At the same time, however, it is very difficult to keep the tritium inventory small enough for safety reasons. The development of suitable tritium extraction methods is a major R&D task for lithium blankets.

The tritium control problems in the case of Pb–17Li are reversed. Because of the very low solubility, tritium inventory is of no concern. Extraction methods are known which result in a total tritium inventory of less than 100 g, in the breeder material of a power reactor. However, the low solubility results in a relatively high partial pressure, which could lead to high permeation losses in the heat exchanger and in the piping system. Permeation-reducing coatings are required in blankets employing Pb–Li as the breeder material, especially if the breeder is cooled by water or helium (separately cooled concepts). The technology of such coatings has to be developed and tested, taking into account cyclic stresses and the effect of irradiation.

3.3. Purification of the liquid metal breeder

Impurities that lead to induced activation represent both a safety and a maintenance issue. Therefore, their inventory has to be minimized by suitable on-line extraction methods. One kind of impurity is corrosion products. For both candidate breeder materials, it is desirable to remove corrosion products continuously to avoid plugging of loop components and to minimize activation. Another kind of impurity is certain tracer elements in the liquid metal which will become activated in the blanket. These elements can be original constituents or may be generated by the neutron irradiation. Such impurities are of no concern in lithium.

In the Pb–Li alloy, the most crucial element is the α-emitter 210Po, which is a transmutation product of bismuth. Bismuth is an original constituent of lead and a transmutation product of lead generated by neutron capture. The potential for polonium release in the case of a liquid metal spill has to be minimized by controlling its concentration to very low levels. Therefore, online methods have to be developed for the removal of polonium and/or bismuth from the Pb–Li alloy.

An additional problem in the liquid metal breeder purification system is the requirement to allow for self-healing of the electrically insulating coatings at the duct walls. This requires either a certain oxygen (oxide layers in Pb–17Li-cooled blankets) or nitrogen concentration (nitride layers in lithium cooled blankets). The situation has to be avoided in which this content interacts with the extraction methods for tritium.

3.4. Transient electromagnetics

Large electrical currents can be induced in the breeding blankets in the case of plasma disruptions and when combined with the magnetic field, these may lead to large forces and stresses in the blanket segments. Liquid metal breeders have the potential to enhance the current flow, as a result of the high electrical conductivity of such breeders. For a quantitative description of these effects, new methods have to be developed and verified. These models are not identical to those used for solid bodies, because liquid metals do not transfer shear stresses but, in contrast, can transmit pressure pulses in all directions.
4. Neutronics status and R&D needs

An extensive neutronics R&D program is needed for the successful construction of future experimental and commercial fusion reactors. There are a number of neutronics issues remaining that need to be addressed and resolved with R&D. Important design responses of interest are the shielding of nuclear components and personnel; tritium production rate in the blanket that satisfies tritium self-sufficiency; nuclear heat generation; induced radioactivity, and decay heat. The main elements of the R&D program that ensure that the accuracy of the predictions of these nuclear responses are within tolerable levels are (1) basic nuclear data measurements and evaluation, (2) transport and response codes development, (3) integral experiments and analyses, and (4) experimental and measuring techniques development. Executing this R&D program will reduce the large costs associated with large safety factors normally introduced to compensate for uncertainties, and will provide the experimental verification required for approval and licensing of the device.

There has been extensive effort in the area of differential data measurements and evaluation. These basic measured and/or evaluated data now constitute a number of latest databases, among which are ENDF/B-VI, JENDL-3, JEF-2, BROND and CENDL. The FENDL library is based on all these databases and is in the process of preparation by IAEA [4]. As for the transport codes based on deterministic methods, there are several of them available in the USA (such as TORT and DORT, 1-, 2- and 3-DANT), Japan (such as DOT-DD) and the EU (such as BISTRO). Examples of Monte Carlo transport codes are MCNP (USA), MORSE-DD, GMVP (Japan), TRIPOLI (EU), and BLANK (Russia).

The integral experiments themselves can be classified as benchmark experiments and engineering-oriented experiments [5]. The first class of experiments are usually simple in geometry and material selection, so as to be best used in intercomparing the performance of transport codes and/or identifying discrepancies in basic nuclear data. Examples of benchmark experiments can be found in Ref. [6]. In particular, there have been recent experiments to examine beryllium data owing to beryllium’s importance in fusion blankets as a neutron multiplier. However, engineering-oriented integral experiments simulate in some detail the engineering features of a fusion blanket—shield module found in a fusion reactor. They aim to evaluate the overall potential of a particular blanket type and the mean prediction uncertainties in key nuclear responses, and to generate design safety factors for designers to use.

The USDOE/JAERI collaborative program on fusion neutronics is an example of a dedicated research program on design-oriented integral experiments conducted at the FNS facility at JAERI, Japan [7]. Tritium production rates (TPRs), induced activity, nuclear heating and in-system spectrum measurements are performed on Li2O test assemblies placed in various geometrical arrangements. The program consisted of three phases. Other design-oriented fusion integral experiment have been conducted at the LOTUS facility in Switzerland [8] and at Osaka University, Japan [9].

With regard to the prediction capability for tritium breeding, a wide range of calculated-to-experimental (C/E) values were observed in all the experiments performed in the USDOE/JAERI collaboration for local TPRs (as well as line-integrated TPRs—a quantity closer to TBR in a blanket) from Li-6 (T6), from Li-7 (T7) and from Li-natural (Tn). First-hand estimates for the design margins have been obtained from the observed C/E values for the line-integrated TPRs and these could be as large as about 30% more than that predicted from calculations. Also generated during the program was a substantial amount of data on radioactivity and decay heat, and nuclear heating in ITER-related materials. Wide divergence between the computations and the measurements for a number of materials were found, showing the need for updating the decay data and cross-section of the associated libraries used in the activation calculations. The JENDL activation file and REAC3 (175-g) give the most desirable results among the libraries tested. As for heat deposition, nuclear heating rates in several different materials (Li2CO3, graphite, Ti, Ni, Zr, Nb, Mo, Sn, Pb and W) have also been measured.
by the microcalorimetric technique. For almost all these materials, the C/E values lie in a band extending from 0.5 to 2.0.

With regard to shielding capability, a number of bulk shielding integral experiments are currently in progress or planned using the 14 MeV neutron sources at the FNS [10], the Frascati Neutron Generator (FNG) [11] and the Technical University at Dresden (TUD) [12]. The objectives of these experiments performed on various shield thicknesses are to quantify the design margins for the protection of superconducting coils and to give realistic estimates of peaking factors originated from radiation streaming and the existence of opening paths.

Among the urgent and long-term items for the ITER neutronics and shielding R&D program are the following:

(a) the need for a comprehensive transport data library for multigroup and pointwise discrete ordinates, and Monte Carlo calculations—FENDL is preferred as the reference library but further effort is needed to complete and test the library against existing benchmarks;
(b) further updating and testing of existing radioactivity codes and activation libraries is needed, and a more comprehensive database is required;
(c) measuring techniques need further development, particularly in the area of nuclear heating;
(d) quantification of safety factors for blanket and shield design of ITER through integral experiments is urgently needed, and this quantification should include effects such as heterogeneity, existence of void, gap and openings.

5. Plasma-facing component status and R&D needs

PFCs present a major challenge for fusion reactor development. PFCs include the divertor, main limiters, bumper limiters and protective tiles, and the heating and current drive systems. Some aspects of the first wall are often considered part of this area. PFC testing is complicated by the close coupling with the overall physics operation of the test device. To date, virtually all the R&D for PFCs has been performed in support of plasma operations on experimental tokamaks.

Divertors tend to receive the most attention, because they are subjected to the most challenging array of conditions, including very high heat fluxes, high particle fluxes and intense neutron radiation. The most popular reactor divertor designs usually incorporate an actively cooled substrate and a separate plasma-facing armor material. These are sometimes referred to as “duplex structures”. The armor may be bolted to the substrate, brazed or attached via some other high-conductivity joint. Maintaining acceptable thermal stresses in a “compliant” joint, while still maintaining high thermal conductivity and lifetime, is a major challenge. The ability to replace rapidly the armor with in-situ maintenance operations may be required if armor lifetimes cannot be increased.

Several armor materials are candidates, including beryllium, various blends of graphite, high-Z refractory metals (such as tungsten), and composites of beryllium and carbon. The choice of armor material depends primarily on plasma performance issues.

The actively cooled substrate design functions primarily as a high heat flux, heat exchanger. Different coolants are still under consideration, including water, helium, liquid metals and even gas-solid suspension flows. One option which has received attention at a low level for many years is the free-surface liquid metal concept, in which a liquid metal is directly exposed to the edge plasma.

R&D needs can be classified as either plasma-physics-related or engineering-related tests, and the plasma-physics-related tests have been eliminated from consideration here. Engineering R&D is similar in scope to that of blankets. Specific issues and R&D needs in this category are as follows.

(1) Thermal hydraulics performance. In some cases, heat removal rates of 20 MW m⁻² or higher are required for divertor plates. The capability to remove heat efficiently from the large surfaces of the impurity control system requires innovative design solutions.
(2) **Thermomechanical behavior.** Divertor plates subjected to high heat fluxes and a high number of cycles will be subjected to high thermal stresses, probably exceeding the yield point of materials. Fracture mechanics and fatigue are important concerns.

(3) **Plasma–surface interactions.** Erosion and redeposition will alter the surface characteristics and cause macroscopic thinning of the plasma-facing material. The behavior of these modified surfaces under high heat fluxes and stresses is unknown.

(4) **Tritium permeation and retention.** Tritium transport is important for safety reasons, and also as an element in the overall tritium fuel cycle. Large amounts of tritium may become trapped in PFCs or transported into the coolant systems. Strict regulatory requirements may be imposed on the tracking and accounting for tritium throughout the plant. With the large throughput and high concentrations of tritium in PFCs, this may become a critical issue.

(5) **Radiation damage effects.** Over a long period of time, the effects of surface erosion and neutron radiation will strongly affect the divertor lifetime. Extended tests are needed to assess the capability for acceptable lifetimes.

(6) **Transient response.** The response of the impurity control system to both normal and off-normal transients needs to be determined. In particular, surface erosion and electromagnetic forces during disruptions could severely reduce the operating life.

(7) **Advanced divertor designs.** Alternative designs, such as liquid metal droplet divertors, have been suggested. Such approaches would eliminate the normal divertor plate. Many of the engineering concerns with standard plate designs would be eliminated, but new concerns will arise and the alternative concepts will need to be tested.

In addition to the divertor, other plasma-interactive components also need to be considered carefully. For example, r.f. launcher arrays are large (about 1.5 m × 2 m), complex components very close to the plasma, requiring active cooling. Nuclear effects can have a significant impact on their performance, including electromagnetic spectrum degradation, reduced plasma coupling and thermomechanical problems. Shielding is a major issue with these components, since open paths are required to transport the wave energy. Much more effort will be required to develop these components.

FNT R&D for PFCs is still in its infancy, and is driven primarily by next-generation devices which include significant nuclear phenomena. Since PFCs still offer critical feasibility issues, a much-expanded R&D program is acutely needed.

6. **Tritium processing systems status and R&D needs**

Tritium processing systems are in a more advanced stage of development than are other areas of FNT research. There are two fundamental reasons for this:

1. Tritium processing research can be based on the research of weapons programs in the USA, Russia and the EU;
2. Most of the work on tritium research does not depend on 14 MeV neutrons.

Thus, a system such as TSTA can be designed and operated on a scale similar to that of an experimental reactor. All the major parties in fusion research have major research programs on tritium processing, such as TPL (Tritium Process Laboratory) in Japan, Tritium Laboratory Karlsruhe in KfK, and ETHEL (European Tritium Handling Experimental Laboratory) in Ispra. Also, large tritium expertise exists in Canada from the CANDU program and in France from the weapons program there. Major activities must exist in Russia and China from their weapons programs, but the level of effort devoted to fusion is not certain.

There is no doubt that hydrogen isotopes can be separated into individual isotopes with acceptable purity for fusion applications. The method for isotope separation is by cryogenic distillation (CD). This method has been well demonstrated, and computer codes are available to calculate the size, cost and tritium inventory of the unit under different operating conditions. The key problem is
to minimize the tritium inventory of the system while simultaneously maintaining the operating stability of the system. It is well known that many days are required to bring TSTA from the start of operation to steady state operation, i.e. the isotopic composition from each product streams will remain constant. Even after it reaches the steady state, it is difficult to control the system to operate at this condition. Thus, there is no assurance at this time that reliable supply of the fuel to the fusion device can be ensured.

The problem arises that the CD systems used by fusion are very small. A small distillation system is difficult to control. If the tritium inventory is reduced, then the stability problem may increase. Thus, effort is required to improve control of the CD system. Also, the operation of the CD systems for a pulsed reactor, such as ITER, has to be demonstrated.

There are different problems associated with tritium recovery from the different types of blanket as follows.

(1) Solid breeders. The R&D effort on solid breeders is concentrated on the purge gas behavior on tritium removal from solid breeders. Based on the information generated, it can be concluded that tritium can be recovered from solid breeders with a reasonable tritium inventory, if the proper temperature of the solid breeder and the proper chemical conditions of the purge gas can be maintained. However, the mechanical design of the purge channel is a key concern. The purge channel is usually a small channel about 10 m long. It is imbedded into the solid breeder region. With radiation and thermomechanical effects, the dimensions of the solid breeder will certainly change. Under these conditions, it is difficult to foresee whether or not the purge channel will remain open for an extended period of time.

(2) Liquid lithium. The tritium solubility in the lithium is very high. Thus, it is very difficult to remove the lithium to a tritium concentration of about 1 appm. The molten salt tritium recovery method has been the only process which has demonstrated the recovery of tritium to this concentration. However, salt will dissolve in the lithium, which has caused concerns over the corrosion characteristics of the salt on the structural material. Cold trapping may be an effective method to remove the salt from lithium, but this has not been demonstrated. Recently, distillation has been proposed to recover tritium from lithium. However, the distillation system requires high power (about 100 MW) and high temperature operation (about 800–1100°C). There has been no experimental demonstration that distillation can be used to recover tritium on a large scale.

(3) Liquid Li–Pb. The problem associated with tritium recovery from Li–Pb is very different from that with lithium. The key concern of tritium recovery from lithium is to reduce the tritium inventory, while the key problem associated with tritium recovery from Li–Pb is to reduce the tritium partial pressure. A permeation window is an effective way to recover tritium from Li–Pb. However, the resulting tritium partial pressure is still quite high. This partial pressure is acceptable if the reactor temperature is low and an effective tritium diffusion barrier can be established.

The key goals for the tritium system R&D are as follows:
(1) ensure that tritium is available at the composition required;
(2) minimize the tritium inventory;
(3) minimize the tritium partial pressure for permeation.

All these have to be accomplished with acceptable cost of the system.

Thus, the following list contains some of the key R&D directions for tritium systems in the area of FNT.
(1) For the tritium processing system, develop a control technique which will ensure stable operation of the CD systems with minimum tritium inventory, and with both steady state and pulsed input from the plasma.
(2) For solid breeders, expand the R&D from pure chemistry to an engineering system to ensure the tritium can be recovered with the proper purge rate, purge chemistry as well as purge configuration.
(3) For liquid lithium, demonstrate tritium recovery, including tritium inventory, and the effect of the tritium recovery process on the reliability of the blanket.
(4) For liquid Li–Pb, demonstrate the tritium process with a residual tritium partial pressure over the Li–Pb which is acceptable for the blanket condition typical of a power reactor.

7. Materials development status and R&D needs

7.1. Status of materials development for fusion

It is generally agreed that commercially available materials will not perform satisfactorily in a fusion power plant environment. The development of materials for fusion is a greater challenge than for any other energy-producing system. The operating environment includes the following: radiation damage which results in atomic displacement and transmutation products; chemical interactions with coolants, tritium and tritium breeders; mechanical stresses resulting from primary, thermal and cyclic, disruption, and magnetic loads; elevated operating temperatures; complex structures.

The fusion radiation environment results in swelling, irradiation creep and degradation of thermophysical properties. Coolant, tritium and breeder material compatibility issues include corrosion and mass transfer, hydrogen embrittlement and degradation of thermomechanical properties. A database of structural material responses to various loads is essential to determine the allowable tensile, fatigue, crack growth, creep and fracture toughness characteristics. Elevated temperature operation of PFCs pushes the limits of material performance, driven by time-dependent deformations (elevated temperature and irradiation-enhanced creep). The toroidal geometry of tokamak fusion power plants requires complex component structures, which raises issues of fabrication, welding and maintainability.

The first step in new materials development is establishing the feasibility of candidate materials. Until 14 MeV neutron test facilities became available, the first step may not be fully realized. The second step in developing fusion materials entails microstructure tailoring to minimize the effects of radiation damage. After microstructure optimization, structural material design qualifications have to be developed for licensing large-scale production of materials.

Currently, design codes, such as the ASME pressure vessel codes, are used to evaluate material performance. These design codes cover materials, construction, method of fabrication, inspection and safety devices. For a material to be approved, it must first be approved by ASTM. The data required for approving a base metal and welded joints include the following: ultimate strength, yield strength, reduction in area, elongation, strain fatigue, creep strength, notch toughness, stress rupture strength, brittle fracture characteristics, neutron irradiation effects, and service experience (if available) over a temperature range. The extensive databases needed to develop design codes must be generated for fusion power plants.

The multifunctionality of the blanket requires integrated testing of the system as a whole. A 14 MeV neutron source volume of several liters is a minimum requirement to carry out integrated testing of blanket unit cells. While breeder materials, multipliers and liquid metals may not require R&D in a 14 MeV neutron environment, blanket structural materials, electrical insulating materials, coatings and tritium diffusion barrier coatings do.

To provide code-validated licensable materials, this program should evolve according to several steps. For example, once candidate materials have been identified, it takes anywhere from 15 to 25 years to develop a metallic structural material according to the following logic:

(a) demonstrate feasibility 5 years
(b) develop fusion-specific materials 10 years
(c) develop damage correlations 10 years
(d) develop design methodology (codes) 10 years

Steps (b) and (c) can overlap, but step (d) requires results from step (c) to develop component design methodologies similar to the ASME boiler pressure vessel codes.

Radiation databases generally preclude other environmental conditions, such as coolant interactions, welds, mechanical joints, effects of damage gradients on large structures, fatigue, thermal cycling, etc. These are generally developed during phases (c) and (d) of material development.
"Coding" of load-bearing structures for fusion radiation environments has never been carried out and may take several years ("coding" of materials serves to satisfy the insurer that, if a component fails, it will not lead to destruction of property). Generally, only 0.1% of the concentration levels of constituents can be removed from an alloy without requiring recoding.

Fusion materials R&D involves several international collaborations [13], including the ITER program, several bilateral agreements, the IEA fusion materials agreement, and informal collaborations. The ITER program dominates international fusion material development programs, concentrating on structural materials and special-function ceramic materials for blanket and shield, PFCs, heating and current drive, and diagnostics. First-wall and blanket-related efforts concentrate on establishing the properties of irradiated austenitic stainless steels (60–400°C and up to 20 dpa), studying environmentally assisted crack growth, and rewelding irradiated austenitic steels. In the area of advanced blanket module R&D efforts, irradiation tests on ferritic–martensitic steels and vanadium alloys are planned. For divertor applications, oxide-dispersion-strengthened (ODS) copper alloys are being irradiated and niobium alloys are being assessed. Ceramics for diagnostic and plasma-heating systems are being studied by in-situ measurements of the effects of ionizing and displacive radiation on their electrical and optical properties.

The IEA sponsors collaborations through the "Implementing agreement on a programme of research and development on fusion materials", between the USA, EU, Japan, Canada and Switzerland. There are active programs on the response of ceramic tritium breeding materials, and a joint program on developing ferritic–martensitic steels (Japan is to provide two 50 tonne heats of "superclean"-reduced activation ferritic steels for an IEA-coordinated research program). The IEA fusion materials collaboration has sponsored numerous workshops, encouraging sharing of information on insulating ceramics, modelling irradiation damage processes, low activation criteria and materials, and databases. The IEA also pursues the development of neutron sources: IEA-sponsored workshops and evaluation programs have concluded that an accelerator-based D–Li source is the best choice to meet program needs.

7.2. R&D needs for structural materials and ceramics

**Structural materials.** Major candidate structural materials, such as austenitic steels (e.g. 316 SS), ferritic–martensitic steels (e.g. modified HT-9), vanadium alloys and SiC–SiC composites, still have several neutron interaction issues to be resolved. Among them are swelling, creep, strength, ductility, corrosion and radioactivation.

While the austenitic steels have limited DEMO relevance, because of high temperature thermomechanical properties limitations, these materials have the largest database for ITER use at fluences up to 10 dpa. The ferritic alloys must demonstrate adequate ductile-to-brittle transition temperatures after irradiation, and magnetic permeability issues need to be resolved. Furthermore, although ferritics have a significant fabrication database, welding of irradiated materials needs to be developed (containing high He concentrations after irradiation). Vanadium alloys show promise, but they have undergone limited industrial experience, with few commercial suppliers. Also, tritium solubility in vanadium alloys is an issue. SiC–SiC composites have the lowest known activation and show promising high temperature properties. However, radiation damage and performance feasibility have not been demonstrated, and cost and manufacturing issues remain unresolved.

**Non-structural ceramics.** Issues here included electrical conductivity; dielectric breakdown; radiation-induced electrical degradation; and optical absorption and luminescence. Relatively few data are available on neutron damage effects, which may cause major problems for ITER.

**Breeder materials.** Concerns with breeding materials include thermomechanical behavior and chemical compatibility. While neutron interaction with Li–ceramic breeding materials has been well documented, module performance has to be
demonstrated. For flowing self-cooled lithium or Li–Pb, neutron interaction is not among the most critical issues; however, neutron-stable electrical insulating coatings inside coolant tubes have to be developed.

**Divertor and PFCs.** Among the structural issues are degradation of thermal conductivity, embrittlement, tritium retention, duplex materials and structural integrity. The primary materials under consideration are beryllium, graphite, high-Z refractory metals, Cu and Ni, and SiC–SiC composites. The issues are too numerous to count here.

**Superconducting magnets.** Critical-current reduction and insulation failure are the most important issues for superconducting coils. Superconductors, such as NbTi and Nb3Sn, have low radiation tolerances, and ceramic insulators cannot yet be specified with confidence.

In conclusion, commercially available materials will not be able to satisfy the long-term performance requirements of fusion power plants. The stringent plasma operating conditions impose near-zero failure tolerances on large components inside the plasma chamber. An aggressive materials R&D program with much greater scope is required to achieve successful DEMO operation by the year 2025. A 14 MeV neutron source with adequate testing volumes would have to be in place by the year 2000. Integrated blanket testing in a 14 MeV neutron environment is essential for component development.

### 8. Safety status and R&D needs

Since the last IAEA Technical Committee Meeting and Workshop on Fusion Reactor Design and Technology, at Yalta in 1986, planning within the worldwide fusion community has been focused on the requirements of the International Thermonuclear Experimental Reactor (ITER). The requirements for analysing the safety of ITER and for ensuring that it meets the needs of its eventual host country also have driven the needs for safety-related technical data. These needs can be divided roughly into four categories.

#### 8.1. Tritium transport

A wide range of experiments have been performed on the solubility and permeation of tritium in unirradiated materials. However, few material samples have been irradiated with fusion-like neutron fluxes. The tritium transport through these radiation-damaged materials may be significantly different from that through the unirradiated material.

Codes for the modelling of tritium migration pathways are fairly well developed. TMAP4, which models the transport of tritium through several in-reactor and external pathways, has been verified for the correctness of its coding, and validated by comparison with experimental data [14]. Raskob at the Kernforschungszentrum Karlsruhe [15] has developed UFOTRI for modelling the atmospheric and biological transport of tritium.

Barriers to hinder tritium permeation could be of great benefit, particularly in preventing tritium migration through heat exchanger surfaces. The development of durable barriers is a major R&D requirement.

#### 8.2. Activation products

The release of activated first-wall and blanket materials from ITER is the major pathway for radiation exposure of the public. Past experiments have shown that the material released from an alloy, such as stainless steel, is quite different from the original constituents of the alloy. Some elements, such as manganese, are released preferentially, as a result of the formation of volatile oxides.

Experiments have been performed in which heated samples of candidate first-wall materials are exposed to air or steam [16]. The oxidation and release of the constituent elements, and the corresponding release rates, are used to estimate the off-site doses following a reactor accident.

To date, all these experiments have been performed using unirradiated materials. In these experiments, radioactive activation products, which would accumulate in the irradiated material, have been simulated by the inclusion of stable isotopes in the sample coupons. Some of these surrogate activation products are difficult to incorporate into the candidate alloys, because of solubility limits in
the molten state. Better modeling of activation product behavior will require either irradiated samples or powder metallurgy techniques to obtain the required sample constituents.

Radiation damage also will undoubtedly alter the mechanisms for the release of activation products from the samples.

The transport of vapors and aerosols out of a damaged reactor will be reduced through the plate-out of vapors on cold surfaces and through settling of aerosols. Data on the rates of plate-out and settling will allow safety analysts to bound more confidently the doses to the general public in case of a fusion reactor accident.

8.3. Dose codes

Codes have been developed to model the dose at the site boundary and that to the surrounding population from the release of tritium and activation products. The MACCS code [17], originally developed to calculate doses resulting from fission product releases, has been expanded to include all significant activation product isotopes [18]. Transport through the food chain is modelled by the COMIDA code [19]. The EU has developed the COSYMA code for accident consequence analysis [20]. These codes calculate doses on the basis of the effective dose equivalent (EDE), in keeping with international standards.

8.4. Regulatory framework

Most members of the worldwide fusion community have agreed that direct application of the regulatory framework now used for fission reactors will not reflect the inherently different characteristics of fusion. However, in the absence of positive action on the part of the fusion community, these fission regulations will be applied by default. ITER will be the first facility for which these issues of regulatory framework must be addressed. Since there already exists a regulatory framework in most countries for the construction and operation of experimental nuclear facilities, that framework will serve as the starting point for the licensing of ITER; deviations from that framework will be sought when they appear to be justified. In the USA, a DOE order 5480.FUS is being developed for the regulation of fusion facilities.

9. Conclusions

The past few years have seen significant progress made in several aspects of FNT. The R&D to date has focused on “low-cost”, high-leverage items which can resolve critical issues and help in concept screening. Many of the remaining tasks will require much greater effort and resources to develop attractive components for power reactors. As we proceed into the “era of ITER”, FNT is expected to play an increasingly important role in the world fusion programs. By the time of the next IAEA workshop, we may be preparing for construction of the first ever fusion nuclear reactor.

References


