

7 Materials and Technology

7.1 FUSION MATERIALS MODELLING AND VALIDATION

The novelty of scientific developments in mathematical modelling of materials was the defining feature of research work carried out in 2010/2011. The research effort involved the development of a range of new fundamental concepts, mathematical methods, and models needed for the quantitative interpretation of experimental data on microstructural evolution of neutron - and ion - irradiated materials, and for rationalizing the observed high temperature behaviour of steels, tungsten, and various technologically relevant alloys.

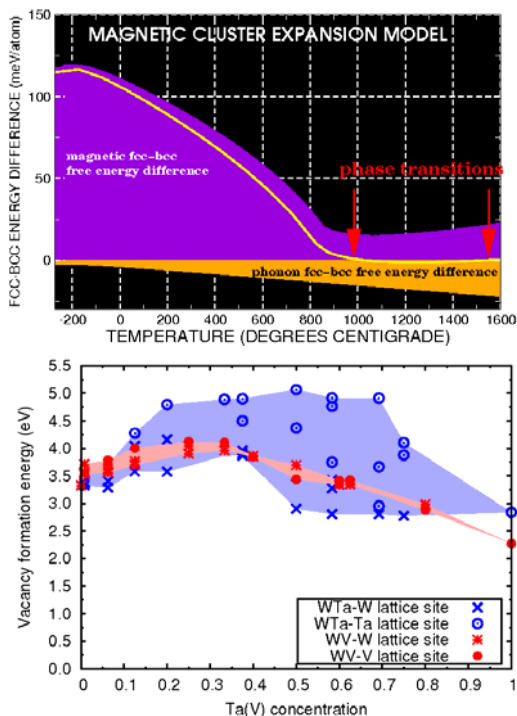


Figure 7.1: Variation of magnetic and phonon free energies that defines the temperatures of α - γ and γ - α phase transitions in pure iron, predicted by Magnetic Cluster Expansion simulations (upper). Vacancy formation energies for tungsten-tungsten and tungsten-tantalum alloys derived from *ab initio* density functional calculations (lower).

Following earlier work that showed that thermal magnetic fluctuations had a significant effect on high-temperature properties of ferritic-martensitic steels, a quantitative model for the phase stability of iron and iron-chromium alloys (Magnetic Cluster Expansion) was developed and applied to the treatment of phase transitions in these materials. Fig. 7.1 (upper) shows free energies of face-centred - cubic (fcc) and body-centred-cubic (bcc) phases of iron, and the α - γ bcc-fcc and γ - α fcc-bcc phase transitions resulting from the excitation of magnetic and vibrational degrees of freedom. The figure indicates that it is the difference between the *magnetic* free energies of

austenitic (fcc) and ferritic (bcc) phases that gives the dominant contribution to the total free energy difference between the two phases, the interplay between which in turn determines the martensitic microstructure of steels.

The development of new tungsten alloys for divertor applications has until recently remained a relatively unexplored area of fusion materials development. *Ab initio* density functional calculations, combined with small-scale specimen tests involving ion-irradiated model alloys, provide a way of screening the alloys to determine their suitability as structural materials for high-temperature applications, and their compatibility with other materials. Extensive *ab initio* simulations of phase stability of tungsten-tantalum and tungsten-vanadium binary alloys were carried out at CCFE in collaboration with Polish and Swiss Associations. Fig. 7.1 (lower) illustrates some of the findings derived from this work, related to the phase stability of tungsten alloys and spontaneous formation of vacancy defects at elevated temperature. The figure shows that the vacancy formation energy in W-Ta alloys depends sensitively on the lattice site at which a vacancy is formed, whereas in W-V alloys the vacancy formation energy is almost independent of the position of a vacancy in the alloy. The results show the possibility of designing alloys where vacancies form within a desired range of temperatures, with potential implications for the development of alloys with improved stability under irradiation.

Understanding the dynamics of formation, migration, interaction, and coalescence of defects in iron alloys and steels at high temperatures is a question central to the interpretation of experimental data on microstructural evolution of alloys and steels under fusion operating conditions. The Magnetic Cluster Expansion (MCE) simulations above show that magnetic excitations in steels dominate the stability of α and γ phases, as well as the stability of defects formed in steels under irradiation. Understanding the time-dependent dynamics of magnetism requires extending the MCE simulations into the time domain. This is achieved through the development of spin-lattice dynamics, which generalises molecular dynamics to the case of magnetic metals and alloys. One of the fundamental unresolved questions in spin lattice dynamics simulations was the dynamic definition of temperature for the magnetic subsystem, an issue that in the context of molecular dynamics is addressed directly by monitoring the kinetic energy of moving atoms. The complexity of the question is illustrated by the fact that kinetic temperature of atoms is a positive definite quantity whereas the temperature of magnetic moments can be positive or negative. Research work performed at CCFE in collaboration with a group based at Hong Kong has fully resolved the question, and has enabled developing a mathematical algorithm for dynamic monitoring the magnetic temperature. Fig. 7.2 shows how temperature of magnetic moments evolves if magnetic and lattice subsystems are brought into contact, and how well magnetic temperature at equilibrium, defined using the newly developed algorithm, agrees with the kinetic temperature of atoms defined via their average kinetic energy.

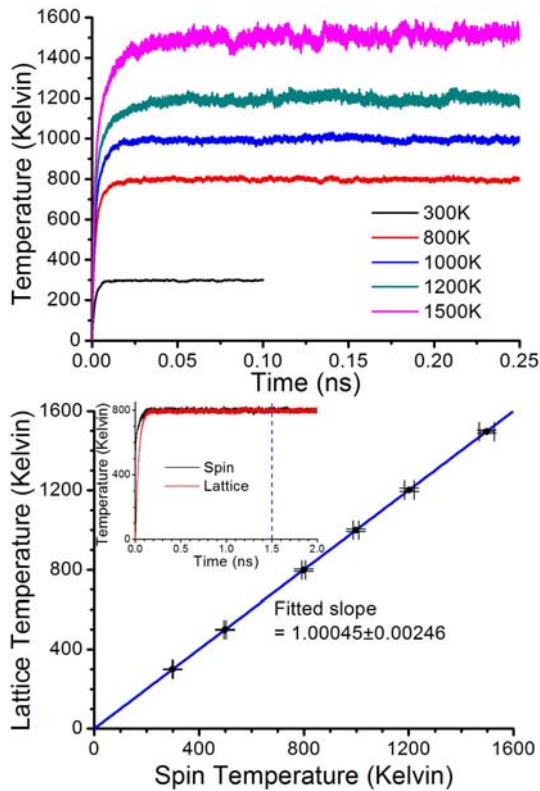


Figure 7.2: Spin-lattice dynamics simulations showing the transient dynamics of thermalization of magnetic moments in iron (upper). Temperature of magnetic moments, evaluated dynamically, plotted versus kinetic temperature of moving atoms at equilibrium (lower).

Transmutation effects in materials, including production of helium and hydrogen in nuclear reactions initiated by fusion neutron irradiation, is an issue of major concern in relation to the structural integrity of components of a fusion power plant. The stability of materials with respect to transmutations can be assessed either by exposing materials to high energy neutrons, or by performing simulations involving self-consistently determined neutron fields evaluated for the appropriate realistic geometrical configuration of components in a power plant. The advantage of the latter is that simulations can predict transmutation rates for various parts of the plant much faster and at almost zero cost in comparison with the effort required for conducting experimental tests on a similar scale.

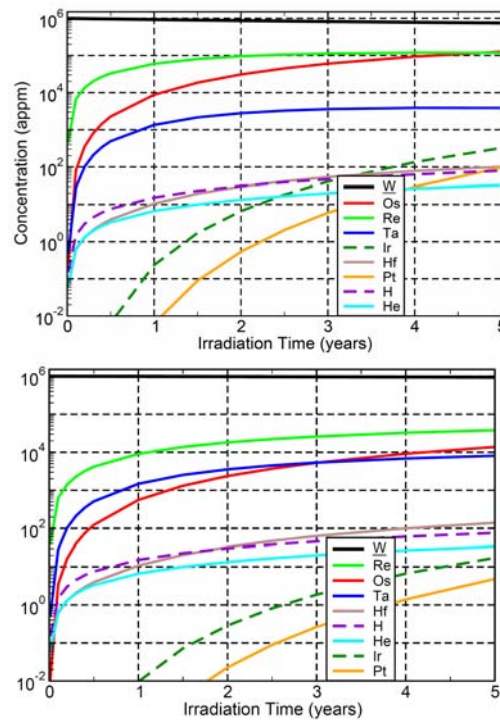


Figure 7.3: Calculated rates of accumulation of transmutation products in the initially pure tungsten (natural isotope abundance assumed). Transmutation rates evaluated taking no account of multiple neutron scattering self-shielding effects (upper). Transmutation rates evaluated taking into account the self-shielding effects (lower).

Fig. 7.3 illustrates the difference between rates of nuclear transmutations in tungsten irradiated with fusion neutrons, assuming that they come directly from the plasma (this also corresponds to irradiation conditions which will be obtained in the International Fusion Materials Irradiation Facility (IFMIF) or a similar neutron source, in the limit of small specimen size), and transmutation rates characterizing tungsten taken as a part of a realistic-size component structure in a fusion power plant. The non-linear modification of the neutron spectrum resulting from resonance interaction of neutrons with other parts of the structure, and suppression of nuclear reactions due to the absorption of neutrons with energies close to the resonance peaks of the reaction, results in the dramatic reduction of transmutation rates, which could be lower by a factor of 3 or 4 in a power plant in comparison with observations performed using small specimens or thin foils in a model neutron irradiation experiment.

7.2 NUCLEAR DATA AND CODES FOR TRANSMUTATION-DECAY MODELLING

High quality activation data and transmutation-decay inventory codes are of fundamental importance in the development of new nuclear technology such as fusion energy. Neutrons carry 80% of the 17.6 MeV energy released in each DT fusion reaction taking place in a tokamak burning plasma. In order to provide technological support to the design of ITER components and next step devices like DEMO, it

is important to know where these neutrons are transported to and how they interact with and transmute the materials that are present. Since such devices are expected to generate more than 10^{20} high energy neutrons per second, being able to individually simulate where each neutron will go and how it will interact with materials is currently a very challenging computation to perform. However, with advanced acceleration tools combined with parallel computation it is possible, in some important regions of tokamak devices, to gain an assessment of average neutronic quantities such as flux, nuclear heating, neutron damage and activation of materials, for example.

In the steps towards an operational fusion power plant the behaviour of new materials and components need to be simulated in the intense neutron fields so that one may predict the levels of activation necessary, for example, to calculate dose budgets for operations and maintenance, to predict waste inventories for decommissioning strategies, and to estimate effects such as material swelling and embrittlement. Such simulations are required to design, optimise and ultimately select promising candidate materials for components which require further qualification and testing in high neutron flux irradiation fields, such as the materials testing facility IFMIF and component testing facilities (CTF).

The CCFE Neutronics and Nuclear Data Group maintain and actively develop the capability and tools required to perform such simulations. One of the key areas of expertise lays with the production of the European Activation File (EAF), which has been developed and continuously improved for more than 20 years, and last year the latest and most improved version, EAF-2010, was produced at CCFE. This year work continued on a new version of the transmutation–decay inventory code FISPACT, which, with EAF, is a component part of the European Activation System (EASY). The new release of FISPACT is expected later in 2011.

7.2.1 EXPERIMENTAL PROGRAMME ON NUCLEAR DATA

The completeness and accuracy of radiation transport, activation and decay nuclear data required for simulations of neutronic effects are fundamental for the assurance of the accuracy of such simulations, and the best evaluations and most up to date data available are used at CCFE for this research. However much of the nuclear data required for fusion calculations is not very well known. Some of the underlying cross section data, have large uncertainties due to lack of supporting experimental data; and in many cases data must be derived from nuclear interaction modelling codes, such as the European code called TALYS.

To progress fusion technology, access to high flux, high energy experimental neutron facilities is urgently required. There are currently no facilities which can irradiate materials or components with the necessary intensity and spectrum, although some have been proposed such as IFMIF or CTF. Until such facilities exist, the reliance on the best available data obtained from experiments using

existing neutron facilities or from TALYS and used with the FISPACT inventory code, is crucial to the development of fusion nuclear technology.

In support of the technological requirements for fusion, and to address some of the data problems highlighted, the UK fusion technology programme is conducting several experiments using the ASP DT neutron irradiation facility at Aldermaston (see Fig 7.4). The facility is recognised as the UK's standard for 14 MeV neutron emission and fluence. The facility consists of a deuteron ion accelerator and a tritium target (see Fig 7.5). The pure ion beam enters a solenoid focusing lens that precedes the main accelerating tube. An insulated transformer provides up to 300 kV of linear electric potential to the accelerating tube that consists of a series of alternate ceramic insulators and aluminium accelerating electrodes. This electric field imparts kinetic energy (up to 300 keV) to the deuterons to initiate a DT fusion reaction when the particle impacts a target consisting of a tritiated titanium coating applied to a copper disc. The resultant neutron emission from the target is essentially isotropic with a small energy-angular variation around 14.1 MeV.

The present experimental programme involves CCFE in collaboration with AWE, NPL and York University, and covers two key areas of fusion technology development: (1) improving the quality of nuclear cross-section data required for fusion-related materials; and (2) benchmarking fusion nuclear analysis tools such, as the 3D shutdown activation dose system MC-R2S. In the future, it is intended to extend the programme in order to design and execute experiments to support the development of essential fusion technology systems such as tritium breeding blankets.

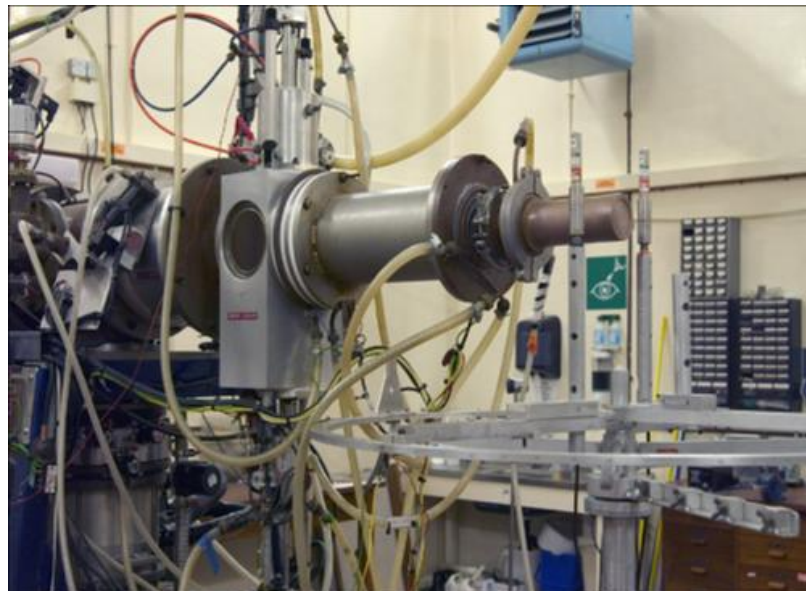


Figure 7.4: The ASP target area and irradiation cell (photo courtesy of AWE).

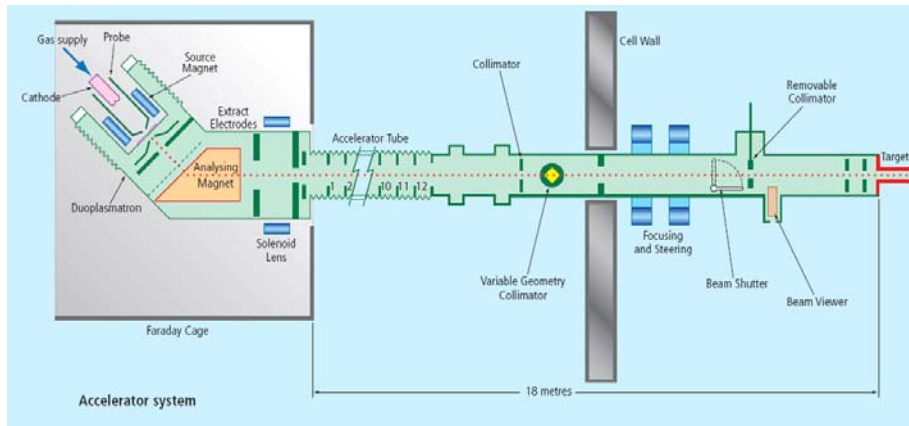


Figure 7.5: Schematic showing features of the ASP accelerator (figure courtesy of AWE).

7.3 PLASMA-FACING MATERIALS IN JET

7.3.1. TILE EXCHANGE DURING THE SHUTDOWN TO INSTALL THE ITER-LIKE WALL

In the first half of 2010 all the plasma-facing tiles (over 4000, all of which were carbon-based) in JET were removed, to be replaced with new tiles for the ITER-like Wall (ILW). The vessel has been cleaned, and a mixture of bulk Be, Be-coated inconel, W-coated carbon and solid W tiles have been installed. Marker tiles (described in the 2010 Annual Report) have been successfully installed to monitor erosion/deposition in the forthcoming JET campaigns amongst each of these types of tiles. There will be a special shutdown in the second half of 2012 to remove these tiles: measurement of erosion/deposition and H-isotope retention for JET with a metal wall compared to the previous carbon wall is one of the prime objectives of the ILW project.

There is a large analysis programme for representative tiles from the carbon-based tiles that have been removed during 2010, and the first stages of this programme have been completed. About 30 of the tiles removed are marker tiles that were mounted in the vessel either in 2004 or in 2006. Tile profiling began in February 2011, after measurement of all the tiles due to be mounted during the ILW shutdown had been completed. One of the first tiles to be re-profiled was Inner Wall Guard Limiter (IWGL) tile 3X11L, which was exposed in JET 2004-2009. This tile was located in a similar poloidal position to tile 7X12L (tile 11 is adjacent to tile 12 in the limiter), though in Octant 3 instead of octant 7, so not in a neutral beam shine-through region. In the 2008-9 Annual Report we described how a tungsten (W) coating deposited on tile 7X11L had been completely eroded away on the section of the tile closest to the plasma. It was speculated that this was due to the central part of the IWGL limiters being in contact with the plasma during plasma start-up, and nothing to do with whether or not the tile was in a shine-through region.



Figure 7.6: Photographs of two IWGL tiles : 7X12L (right - exposed 2004-7) and 3X11L (left - exposed 2004-2009).

Fig 7.6 shows photographs of two IWGL tiles: 7X12L (on the right - exposed 2004-7) and 3X11L (on the left - exposed 2004-2009). The erosion/deposition patterns are identical, even though the former tile was in a neutral beam shine-through region and the latter was not, confirming that the erosion is indeed due to plasma contact. Most of tile 7X12L had initially been coated with W or rhenium (Re), whereas tile 3X11L just had a W-coating in a band ~15mm wide at the top edge of the tile. It was not possible to quantify the erosion on 7X12L as the tile had not been profiled before exposure in JET, however tile 3X11L is a marker tile. Prior to exposure, the surface co-ordinates were recorded at a matrix of 30x15 points (with 5mm spacing between points). Tile profiling has been repeated at the same points following removal from JET, and the differences between the co-ordinates are plotted in Fig 7.7. Red bars represent the extent of erosion, whilst green represent deposition. The maximum amount of erosion determined in the region closest to the plasma is about 200 microns.

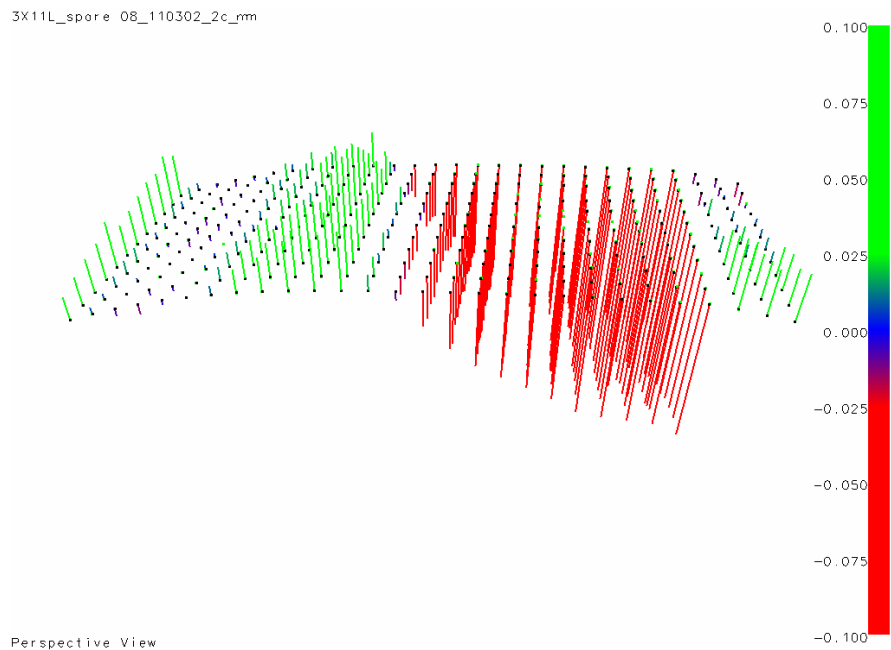


Figure 7.7: Net erosion/deposition on IWGL tile 3X11L during exposure in JET 2004-9 (by subtracting tile profile measurements after exposure from measurements before - red is erosion, green is deposition).

The central part of the IWGL is in contact with the plasma for up to 10s as the plasma current is ramped up from 0 to a maximum of 5 MA (more typically ~ 3 MA) without any additional heating. Tungsten coatings in the divertor are intended to survive plasma contact for similar periods at full current and with up to 40MW additional heating in the plasma. Fig 7.8 is a photograph of a Load Bearing tile that was coated with 10 microns W, after exposure in JET 2006-9 at the most common location of the outer strike point (see inset diagram)

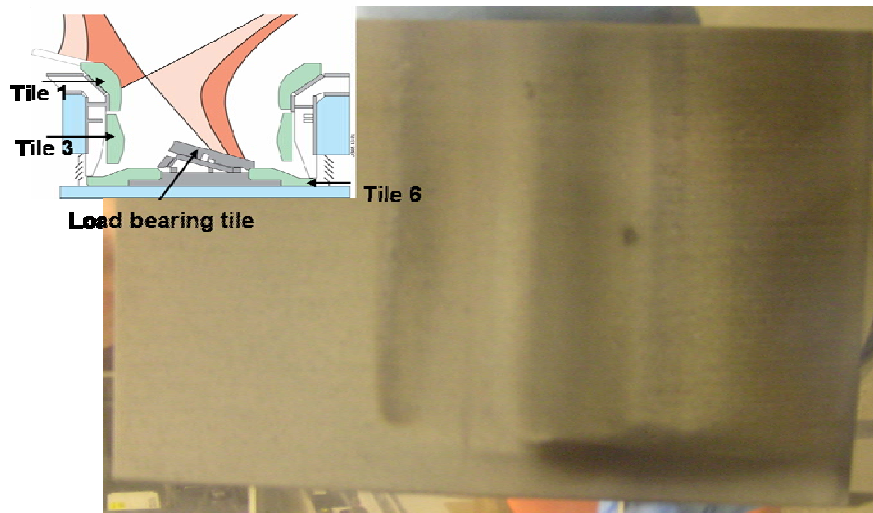


Figure 7.8: Load bearing tile 14RW (exposed 2007-9), with inset to show position in the divertor and a common plasma configuration with the strike point on these tiles.

The dark marks on the tile are regions where there is a thin deposited film on the surface (mostly carbon) – the pattern varies with strike point position. However, the tile is still covered everywhere with the W coating to a depth of several microns. It was not possible to tell by Ion Beam Analysis if the coating is a little thinner in some places due to erosion, but this will be determined by further analysis of the tile. This test of a coated tile in the divertor is thus very reassuring for the ILW; the coatings in the divertor are unlikely to fail due to erosion during the ILW campaigns. However, it provides an interesting contrast to the experience of W coatings on the IWGL. If plasma is in contact with a limiter, atoms sputtered from the surface enter directly into the confined plasma, and are transported away from the limiter to ultimately be deposited somewhere in the plasma boundary, whereas an atom sputtered from a surface in the divertor has a very high probability of being re-deposited locally, so that the net loss of material from the divertor tile is maybe two orders of magnitude less than from a limiter tile under a similar incident flux. It will be important for ITER to ensure an adequate lifetime of limiters during the start-up phase.

One of the important issues for the ITER safety case is the nature and amount of dust that will accumulate during operation, and consequently the ITER team are very interested in the dust that is produced in JET. The shutdown for the installation of the ILW has provided a unique opportunity to collect the dust from within JET:

planning for the collection was described in the 2010 Annual report. The dust from each of the five regions of the divertor was collected in a separate container with a closure valve. By weighing the containers before and after the dust collection, the amount of dust from each region can be determined: the measurements have now been completed and the results are shown in Fig 7.9.

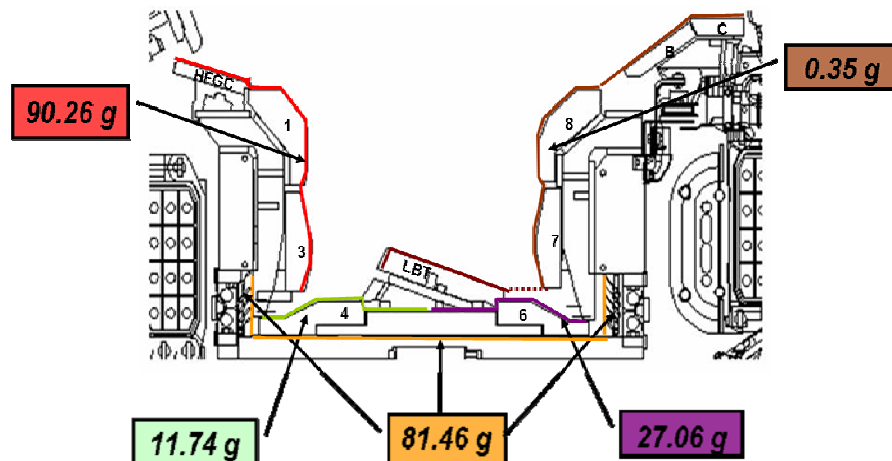


Figure 7.9: Amounts of dust and flakes collected from the divertor during the shutdown to install the ILW.

The total amount of dust collected was ~211g, which represents the amount that has accumulated in the JET divertor during the 2005-6 and 2007-9 campaigns. This number can be extrapolated to give an estimate of the rate of build-up of dust in ITER if it operated with a carbon first wall, but account should also be taken of the deposits that have built up on the surfaces of the divertor tiles. Although these deposits are still adherent in JET, the much greater accumulation to be expected in ITER would eventually result in spallation. We estimate that the mass of deposited films in the divertor increases by ~500g during each JET operational period: this figure should be included when estimating the rate of accumulation of dust/flakes in ITER.

7.4 TECHNOLOGY

7.4.1 DEMO AND POWER PLANT STUDIES

The design of DEMO remains to be determined and even its mode of operation is presently uncertain. In particular, there remains a debate between a truly steady-state device which, in a tokamak design, needs efficient and reliable current drive systems, and the alternative of a pulsed device in which the electrical output is maintained in steady-state by the use of energy storage technologies. The design of these two options differs by much more than the presence or absence of the current drive and energy storage systems of course, and this has been investigated in more detail this year.

A Pulsed DEMO

The challenges of a pulsed DEMO and a steady state DEMO are very different, with pulsing reducing the need for current drive systems, to maintain the plasma in a steady-state but leading to additional problems for components due to the cyclic stresses, which can shorten the lifetime of components such as the first wall.

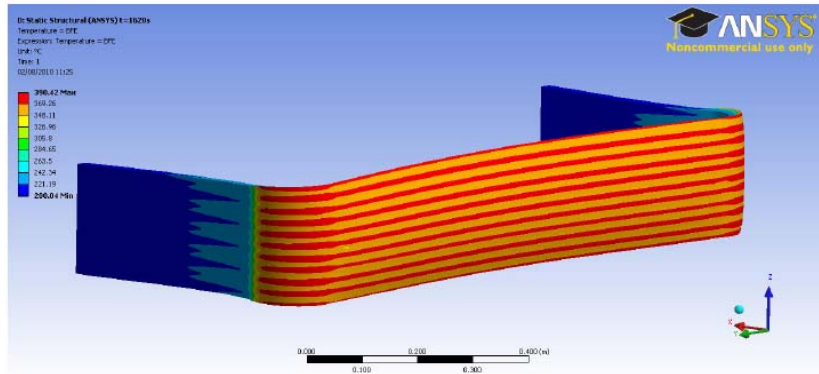


Figure 7.10: Model of a first wall segment used in analysing the cyclic fatigue expected in a pulsed version of DEMO.

Earlier results examining the effect of pulsed heat loads on the first wall of DEMO suggested that there could be a problem with the first wall lifetime since, even in the absence of neutron damage to materials, the cyclic loads could weaken the wall too quickly. This has been followed up in much more detail and results obtained for 3 different models of the first wall (one of which is illustrated in Fig 7.10), each using 2 different coolants; water or helium. With an improved design and more careful analysis, it appears that water cooled versions of the first wall are likely to achieve lifetimes which can allow operation of pulsed DEMO without the need for too frequent changes of the blanket and first wall, however the helium equivalents – which operate at higher temperatures, still potentially present a problem. Although this appears to suggest that water coolant is a better option than helium for use in a pulsed version of DEMO, this is looking at only one facet of the design in isolation and, although important, this is not sufficient evidence to drive the choice of coolant – for instance it is known that the neutron damage to materials, particularly embrittlement, will be more harmful at the lower temperatures characterised by the water coolant than at the higher temperature of helium. It is necessary to include the full range of important effects in order to optimise the design.

B Steady-state DEMO

In addition to developing models of a pulsed DEMO design, work has continued to determine more of the details of an optimised steady-state device. The results of 2 studies are reported here both of which relate to the amount of heat in the plasma and how best to remove it

without damaging material structures. In this way they are the steady-state equivalent of the analysis in section A and will allow us further insight into the best mode of operating a fusion power plant.

The first study concerns further analysis of how to drive the plasma current in a steady-state device without relying on high performance of the plasma, in particular considering ways to reduce the amount of power needed to keep the plasma current flowing for months at a time. Two main aspects of this have been examined here, the beam energy and the plasma rotation. At a given rotation speed, the amount of power needed to support the plasma current increases as the energy of the beam decreases. Fig 7.11 illustrates this for a particular case and also shows how the plasma rotation affects the required current drive power. For instance at a high forward rotation, labelled 0.5 in the figure, the required neutral beam power is significantly increased but, if the rotation could be reversed, the power could be decreased.

In fact DEMO is not expected to rotate at these high speeds, although ways of inducing such rotation are being investigated, however these results are likely to be important when applied to a Component Test Facility (CTF) where rotation speeds are expected to be substantial. These results are therefore of more general interest.

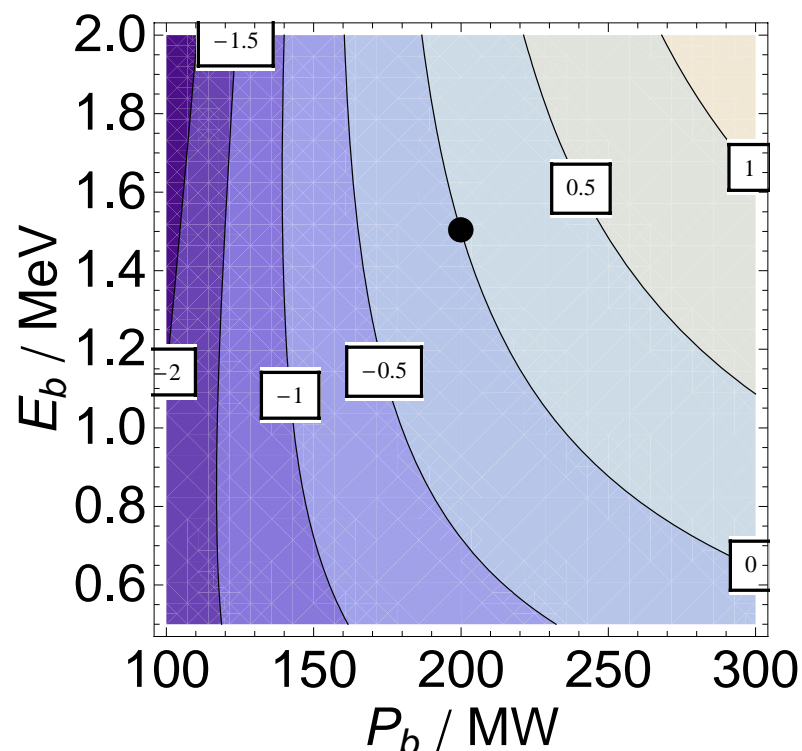


Figure 7.11: The figure shows that, at a given rotation speed, reducing the energy of the neutral beam increases the power needed to keep the plasma current flowing. The different contours are at different rotation speeds and show that rotation too is an important factor in determining the required neutral beam power. If the rotation could be controlled, then current drive would be more easily achieved in a steady-state tokamak DEMO.

Having looked at possibilities to control the heating power required in a steady-state DEMO device, we must also think about ways to control the exhaust power. The second study concerns the optimal way to remove the heat from the plasma without damaging the materials, and is related to the study of pulsed heat loads presented above. In those calculations it was assumed that a substantial fraction (up to 100%) of the heat coming from the plasma will fall on the first wall, rather than being conducted to the divertor and this is expected to be achieved through radiation. The main alternative to this approach is to develop other techniques for power handling, such as novel divertor designs. This approach is also being investigated (for instance the expanded divertor designed for use in MAST-U) and this is described elsewhere in this report.

Modelling of radiation in a DEMO device is more complex than is sometimes realised. DEMO is likely to operate under conditions where the intrinsic radiation is already quite high and, by deliberately seeding the fusion plasma with impurities, we can increase this level almost without limit – although if it becomes too high we will excessively cool the plasma and prevent fusion happening. This complex behaviour is expected to play a key role in controlling the power exhaust from DEMO and is being analysed through experiments, where possible, and modelling to extrapolate behaviour to DEMO conditions.

A measure of the complexity is illustrated in Fig 7.12 which shows the modelled radiation profile when a substantial amount of argon impurity is introduced into the plasma. The amount and the location of the radiation are both important and both depend on the way that the impurities move within the plasma coupled to the local plasma conditions. In this particular case it is shown that, although we might want the majority of radiation to come from the edge of the plasma to avoid cooling the main fusion fuel, this will not be the case, and we must allow for the cooling effect when we model the core plasma. We do this through systems code studies which incorporate the radiation loss from the core as an additional cooling mechanism and this is an important facet of the design of DEMO. The need to carry out sophisticated modelling of one aspect of the plasma behaviour, here radiation, and then incorporate that into a systems study to see the effect on other key parameters such as plasma temperature and heat loads to the divertor and the first wall, is a good illustration of the approach that we take to DEMO design activities. Such studies also then suggest new experimental tests that may be carried out in existing fusion devices.

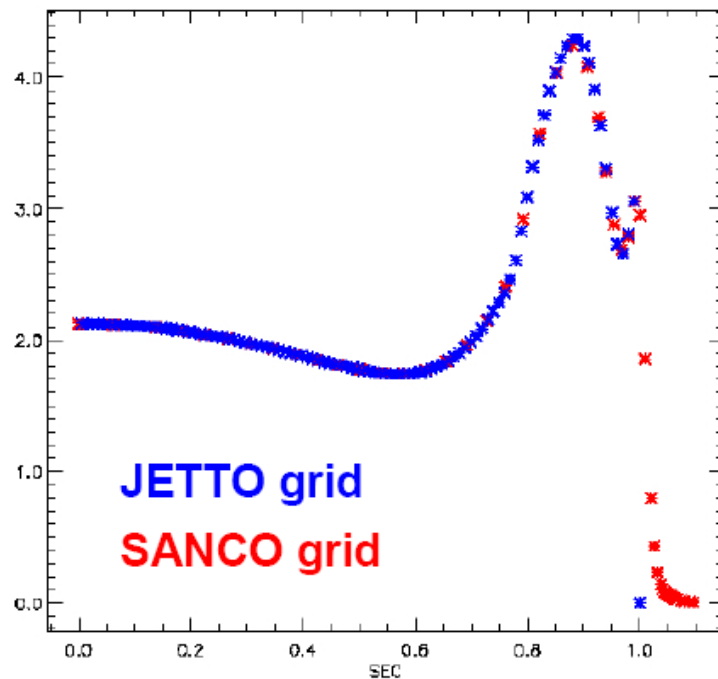


Figure 7.12: Illustrating how impurity radiation may come from a distributed region, here the whole plasma, rather than just from a narrow region at the edge. Argon was deliberately introduced into the plasma in this case, with the express purpose of reducing the heat flow out to the divertor. The simulation here compares outputs from 2 computer codes: JETTO models the core plasma whilst SANCO emphasises more the modelling of the edge region.

C DEMO Heating and Current Drive

The assessment of the maturity of the technology for Heating and Current Drive (H&CD) systems relevant for DEMO has continued from the previous year. The neutral beam study has concentrated on the design options for the injectors and their influence on deployment on the tokamak. The results show that circular beam extraction geometries use space most efficiently (in terms of minimising the volume occupied) and large (ITER-like) rectangular geometries are least efficient. This is inconvenient for concepts such as the photoneutralizer and energy recovery that favour a larger aspect ratio rectangular beam. The geometric arrangement of the injectors around the tokamak is determined by the acceptable envelope in the plasma for optimum current drive and minimum power shine-through. This has been determined by a simulation using the TRANSP code to be a rhomboid in the plasma mid-plane as shown the solid lines in Fig 7.13. Neutral beams falling outside this area either do not contribute to current drive or show excessive power shine-through onto the vessel wall. Various design options for the neutral beam injectors, such as injector size and number, beamline length (equivalent to inclination angle) beam divergence and steering have been analysed and the positions of the beam centres and indicative dimensions are plotted on the chart.

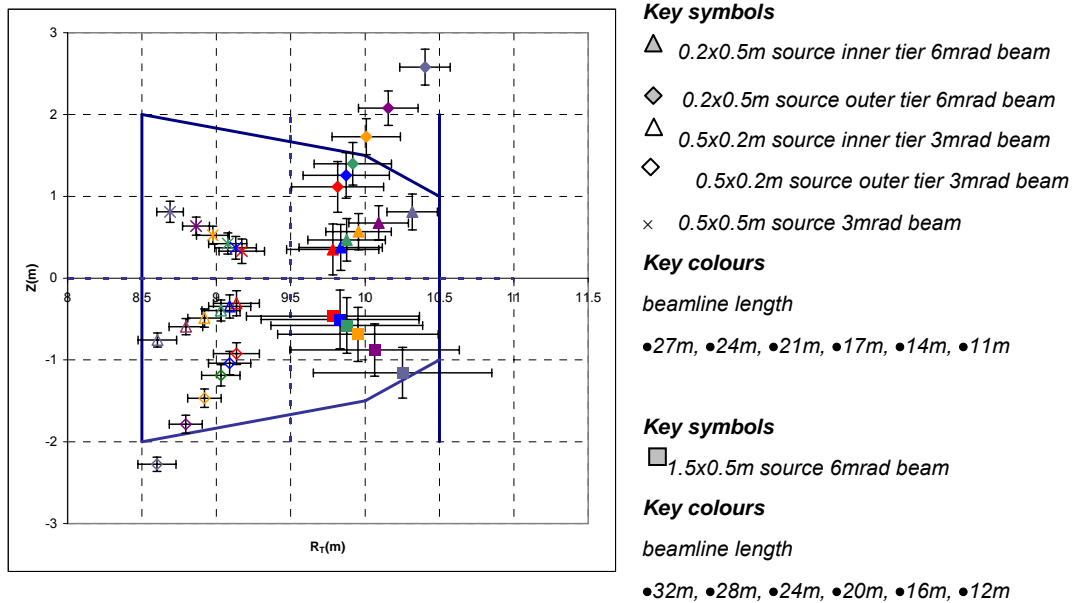


Figure 7.13: Beam positions in the plasma mid-plane for different injector sizes and geometries. The four quadrants are as follows: Upper Left- 0.5x0.5m source 3mrad beam divergence, Upper Right – 0.2x0.5m sources 6mrad beam divergence, Lower Right – 1.5x0.5m sources 6mrad beam divergence, Lower Left – 0.5x0.2m sources 3mrad beam divergence. Colours denote beamline length as given in key.

Only one quadrant per injector design is shown; in reality each quadrant could be filled by installing four (or eight) injectors onto the port. In each case the beamline length varies from approximately 10m to 30m with the injectors inclined to achieve a crossing point of the beam centres shown. Clockwise from top left the geometries are: square injector of dimension 0.5m, two rows of injectors of dimensions 0.2x0.5m (JET-like), large (ITER-like) injector of 1.5mx0.5m and two rows of JET-like injectors (0.5x0.2m) but with reduced beam divergence of 3mrad. For other parameters see key. Shorter beamlines (less than 20m) are clearly limited in application and the benefits of reducing the beam divergence are plain. The figure also indicates that the minimum practical spacing of the beam centres in the plasma mid-plane is ~1m, approximately twice the spacing indicated by TRANSP simulation for a q-profile giving plasma stability. It is therefore probable that at least two ports will be required even if large 1.5x0.5m injectors delivering ~50MW of power each are used. The issue is complicated further by the need to distribute the injected power across the different tangency radii and the possible introduction of plasma rotation, although the results show that several options are feasible for NBI on DEMO. However, if large injectors are selected the re-ionisation losses place further restrictions on the number of injectors per beamline and the beamline length. Minimising the power on the beamline wall favours multiple small injectors with lower total power per beamline but a higher number of ports. This is not particularly problematic as the beam envelope size

is less than 10% of that for the large source so the interruption to the blanket is not compromised.

The necessary length of the beamline is not in itself problematic (although it increases the size of the “nuclear island”) as studies of the neutron transport, irradiation damage and activation of materials commonly used in beamlines show that for distances greater than 10m from the first wall the neutron flux is reduced by approximately 1.5 orders of magnitude and damage and activation is acceptable, with the possible exception of the SmCo magnets used on the ion source. The issue of plasma facing materials for RF systems has also been considered with gold and thin film rhodium being particularly attractive for antennas and mirrors.

C1 Energy recovery

One aspect of neutral beam performance not addressed in the injector deployment study is electrical efficiency and, more specifically, the improvement of the ratio of injected power to consumed electrical power in the neutral beam system. The development of the photoneutralizer had been previously identified as a key technology for future systems but, in the interests of risk mitigation, it was decided to also review the concept of energy recovery. This is a familiar technique employed for the positive ion neutral beam injectors on Tore Supra, whereby the un-neutralized fraction of the beam is collected by a decelerating electrode system and the current re-circulated through the main HV power supply, which then only provides the equivalent current of the neutral beam.

A design for the deceleration system is shown in Fig 7.14 for a negative ion 1MeV beam. This features more electrodes than in other designs, giving better control of the particle trajectories and resulting in collection efficiency of ~98%. The ions are collected at an energy of 25keV and secondary electron suppression is achieved by the penultimate electrode at -1000kV.

For negative ion precursor beams the un-neutralized fraction of the beam is composed of both negative and positive ions and conventionally it has been regarded as unfeasible to collect and re-circulate the positive ion current due to the polarity of the HV power supply. This has always limited the efficiency improvement achievable for negative ion systems but the review has demonstrated that the residual positive ion beam can be usefully converted into DC electrical power, either low voltage, high current or high voltage, low current depending on the configuration of the converter, with an efficiency of ~90%. Recovering both residual beams boosts the electrical efficiency of the injector system from 27% (no recovery) to ~40% but will require the elimination of the beam halo.

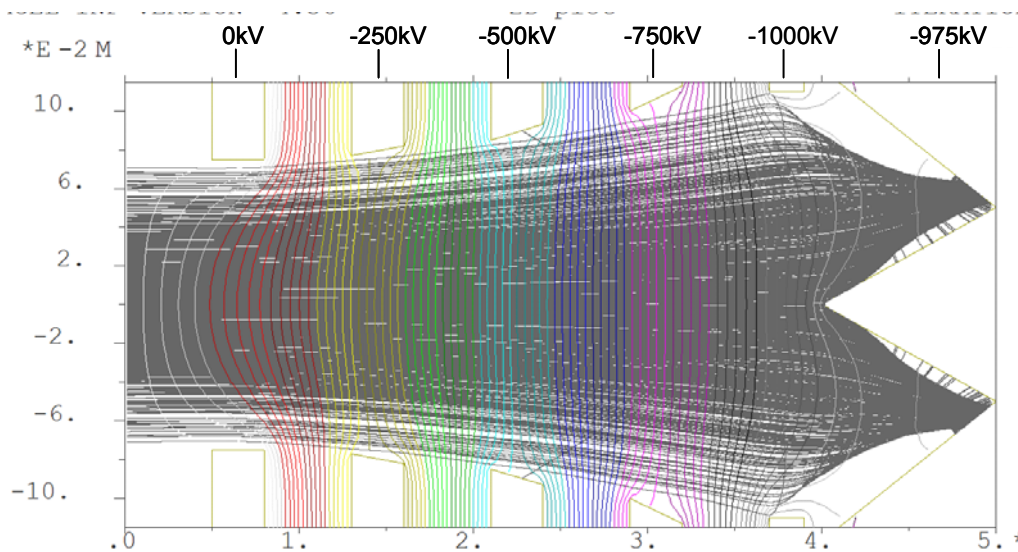


Figure 7.14: Design of the high efficiency decelerating collector of the energy recovery system and trajectories for a 1MeV beam.

C2 Negative ion sources

The programme to understand the role of caesium in negative ion sources has yielded useful insights into the physics behind surface negative ion production. An analytical model of the plasma sheath region in the presence of surface produced negative ions has shown the formation of a virtual cathode that restricts the flux of negative ions transported across the sheath. Biasing the surface negatively with respect to the plasma can eliminate the virtual cathode except for cases where the surface yield is high, when the virtual cathode persists at all biases. The implication of this observation is that to increase negative ion current density in a surface source it is necessary to increase the positive ion density as this will reduce the depth of the virtual cathode. Applying the model to the ITER prototype source shows that the virtual cathode is persistent, implying that improvements will require an increase in plasma density.

This model will now be linked to the one dimensional fluid model of an ion source to create a flexible design tool for the negative ion programme. The source model has been thoroughly validated against various experimental data and source designs and has been used to predict the positive ion species ratio of the chequerboard PINI operated in tritium in preparation for a JET D-T campaign anticipated in the next 5 years.

D Neutronics activities

A major part of CCFE's strong programme in Neutronics is carried out in connection with grants and contracts with the ITER International Organisation (IO) and the European Domestic Agency for ITER, Fusion for Energy (F4E), for studies for ITER design topics. This work is described in chapter 8 of the Annual Report.

In addition to these analysis tasks, ground-breaking development work is being done in several areas of strategic importance to fusion nuclear analyses:

- An innovative acceleration technique for fusion Monte Carlo radiation transport analyses is being developed and tested. A paper on this issue was presented at the SOFT-26 conference, and more conference presentations are intended in 2011;
- In the area of CAD-based modelling tools, CCFE has participated in the production of ITER reference analysis models, where it is recognised that we have leading strengths. In addition, testing has recently started on a novel CAD-to-MCNP technique based on arbitrary tetrahedral meshing. It is intended to present papers to conferences on this topic in 2011;
- Several enhancements in CCFE's leading MC-R2S code have been implemented, such as the parallelisation of the MCNP decay photon source and the extraction of ancillary radiological information;
- Modern 3D plotting and visualisation tools are now being explored, and the possibility to adapt alternative Monte Carlo tools such as FLUKA for fusion nuclear analyses is being investigated, in an exercise where CCFE hopes to co-operate with UK universities.