CURRENT STATUS OF NUCLEAR FUSION ENERGY RESEARCH IN KOREA

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The history of nuclear fusion research in Korea is rather short compared to that of advanced countries. However, since the mid-1990s, at which time the construction of KSTAR was about to commence, fusion research in Korea has been actively carried out in a wide range of areas, from basic plasma physics to fusion reactor design. The flourishing of fusion research partly owes to the fact that industrial technologies in Korea including those related to the nuclear field have been fully matured, with their quality being highly ranked in the world. Successive pivotal programs such as KSTAR and ITER have provided diverse opportunities to address new scientific and technological problems in fusion as well as to draw young researchers into related fields. The frame of the Korean nuclear fusion program is now changing from a small laboratory scale to a large national agenda. Coordinated strategies from different views and a holistic approach are necessary in order to achieve optimal efficiency and effectiveness. Upon this background, the present paper reflects upon the road taken to arrive at this point and looks ahead at the coming future in nuclear fusion research activities in Korea.

KEYWORDS : Nuclear Fusion, Plasma, Tokamak, KSTAR, ITER, DEMO

1. INTRODUCTION

Nuclear energy research started during World War II in the Western world. With the harnessing of nuclear power having been realized, scientists became interested in utilizing the enormous power from the nuclear fission reaction to provide electricity for the commercial market. Fusion energy also drew strong interest over the possibility to make a 'super bomb' and to win dominance in the Cold War era. The first attempt to use fusion energy peacefully, however, came from the idea to utilize the tremendous explosive power of the hydrogen bomb directly to make an alternative canal in the Suez area or an artificial harbor in Alaska [1]. Behind the curtain, secret researches to make nuclear fusion reactions in the laboratory were being carried out and a number of concepts were being tested, such as pinches, stellarators, and tokamaks. Nuclear fusion research remained secretive in the early phase because most of the activities were closely related to research on the hydrogen bomb. At the Second International Conference on the Peaceful Uses of Atomic Energy held in Geneva in 1958, controlled nuclear fusion research was openly discussed and the latest endeavors of leading countries were revealed. Declassification was acknowledged as the major contribution of this conference. Since this conference, controlled fusion research has shown great progress through the exchange of ideas through communication and competition among research groups and institutions.

Nuclear fusion and basic plasma research activities in Korea started in the 1970s with small-scaled laboratory plasma experiments. The first tokamak, called SNUT-79, was constructed in Seoul National University during the late 1970s. Afterwards, small scale fusion devices such as KT-1 in the Korea Atomic Energy Research Institute (KAERI), KAIST-Tokamak in Korea Advanced Institute of Science and Technology (KAIST), and the Hanbit

Table 1. Requirements for Korean DEMO

Goals	Criteria
Power generation	· 1.0~1.5 GWe · Power plant lifetime ~ 60 yrs · Tokamak type
Fuel self-sustainability	· D-T fuel · Tritium breeding ratio > 1.1
Continuous operation	· Steady-state tokamak operation
No evacuation	· Tritium handling · Waste treatment · No high level radioactive waste
Few unscheduled shutdowns	· Avoidance of plasma disruptions · Maintainability

mirror device in the National Fusion Research Institute (NFRI) were constructed or refurbished for fusion research. In 1995, the Korean government launched the Korean Superconducting Tokamak Advanced Research (KSTAR) project [2,3] as the national core fusion programme in the framework of a so-called "mid-entry strategy for high-tech development". Roughly 30 industries participated in the design, fabrication, and assembly of the KSTAR tokamak, including heavy industries and nuclear companies. Korea was finally able to join the International Thermonuclear Experimental Reactor (ITER) consortium in 2003.

The Korea National Science Council adopted the Basic Plan for Fusion Energy Development in 2005 and the Korea National Congress promulgated the National Development Act of Nuclear Fusion Energy in 2006. Korea is now aiming at developing a commercial fusion power plant in the 2040s by adopting the "fast track approach" suggested by the EU, Japan, and the US. In parallel with KSTAR and ITER, a pre-conceptual study of the Korean DEMO fusion power plant started in 2006 for utilization of fusion energy in an economically viable way [4]. The technical requirements of the Korean DEMO were investigated in order to set major goals and criteria, as listed in Table 1. Although more detailed and thorough research should be conducted to find the optimum design parameters satisfying these technical requirements simultaneously, guidelines and reference design parameters were obtained through a fundamental system analysis.

The self-reliance of technologies to the Korean DEMO requires active research and development in the following four fields: a) high performance plasma operations and control, b) blanket and nuclear power conversion, c) burning plasma and tritium self-sufficiency, and d) reliable materials. KSTAR will be a test bed for tackling the first requirement by pursuing long-pulse and high-performance advanced tokamak (AT) mode of operation with reliable and robust plasma control techniques. ITER and the separate fusion engineering development programs will contribute to resolving the latter problems. Basic research is also necessary to increase the understanding of the fusion plasma behavior and thereupon provide concepts of alternatives and improve the predictive simulation capability.

In this review paper, principles of nuclear fusion power generation and research status for the reactor core are reported in Ch. 2, activities in ancillary system developments are given in Ch. 3, and research in the reactor study is described in Ch. 4. A brief description of the roadmap towards the Korean DEMO is provided in Ch. 5, followed by a summary.

2. CORE SYSTEM

2.1 Fusion Principles and Core Plasma

2.1.1 Fusion Reaction and Ignition

The aim of fusion research is to design schemes in which light nuclei approach each other frequently within such small separations that there is a high likelihood of numerous fusion reactions taking place, as shown in Fig. 1. A hot gas, where nuclei and electrons are no longer bound together, is called plasma. Even in plasma, however, the nuclei do not come close enough to react due to mutually repulsive forces. By heating the plasma to an even higher temperature, the ions acquire an even higher velocity, or kinetic energy, and can then overcome this repulsive force. Clearly, the number of fusion reactions that take place will depend on the plasma temperature and plasma density.

The production of plasma and its subsequent heating require energy. A successful fusion power plant requires that the power produced by the fusion reaction exceed the power required to produce and heat the plasma. The ratio of the power generated to that consumed (the fusion power amplification factor) is called the Q value. Initially, the plasma will be heated by various external sources. With increasing temperature, however, the number of fusion reactions also increases and the fusion reaction itself heats the plasma due to the production of energetic helium atoms (actually ions, α particles). The kinetic energy of the helium nuclei exceeds the average kinetic energy of the nuclei of the fuel (deuterium and tritium) by orders of magnitudes. The energy is distributed to the fuel nuclei via collisions. In fact, a point can be reached - termed ignition - when external heating is no longer necessary and the value of Q goes to infinity. In practice, however, power plant operation would likely correspond to a Q value of 20-40.

The state of hot plasma and its nearness to the ignition condition can be characterized by the triple product of the temperature, the density, and the so-called 'energy confinement time'. The latter value describes the ability of the plasma to maintain its high temperature; in other

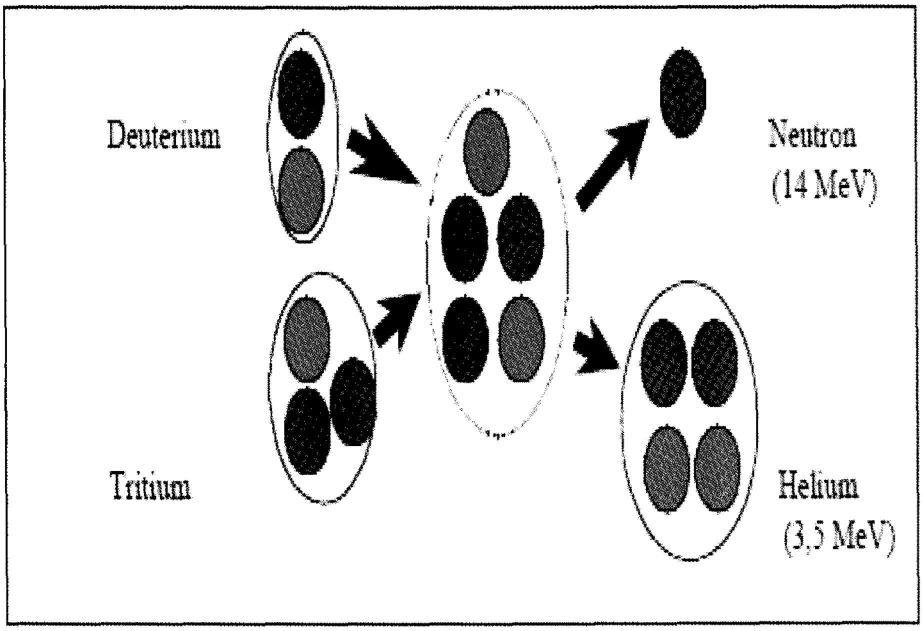


Fig. 1. Fusion Reaction between Deuterium and Tritium

words, it is a measure of the degree of insulation of the plasma. Ignition can only be achieved when this "fusion triple product" exceeds a certain value.

2.1.2 Magnetic Confinement Fusion

The temperatures necessary to ignite plasma are between 100-200 million °C. Obviously no solid material is able to confine a medium with such a high temperature. This dilemma is solved by the attribute that in the plasma, all the particles carry an electrical charge and can thus be confined by a magnetic field. (The charged particles gyrate around the magnetic field lines.) It transpires that a doughnut-shaped configuration of the magnetic field "cage" is appropriate for this purpose, although the magnetic field lines not only have to be doughnut-shaped, they also need to have a helical twist. This scheme is referred to as magnetic confinement.

Different proposals have been set forth to produce helically-wound doughnut-shaped magnetic field cages. The most successful has been the tokamak, first realized in Russia [5]. A sketch is shown in Fig. 2. The magnetic field is the sum of the toroidal magnetic field produced by the coils shown and the magnetic field produced by a current in the plasma. The problem associated with the tokamak concept is driving the current in the plasma. The most important concept applied today is to place another magnetic coil in the centre of the tokamak (see Fig. 2: solenoid magnet) and to ramp the current in this coil up or down. This will produce a varying magnetic field in the coil, which in turn induces a voltage in the plasma (the principle of induction). This voltage can only be sustained for a limited time - one or two hours at the very most. Base load electricity plants need of course to produce power under steady state conditions. Many current R&D activities are directed towards finding alternative ways of driving the current in the plasma (via microwave heating or particle beam injection).

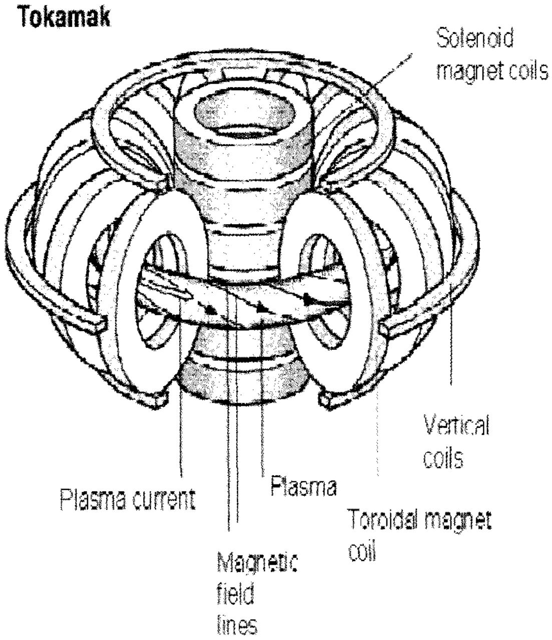


Fig. 2. The magnetic Confinement Scheme of Tokamak

2.1.3 Basic Components of Fusion Power Plant

The various features such as the steam generator, turbine, and current generator will be the same as in conventional nuclear or fossil-fuelled power plants. A flow chart of the energy and material flows in a fusion plant are depicted in Fig. 3. The fuel - deuterium and tritium - is injected into the plasma in the form of a frozen pellet so that it will penetrate deeply into the centre. The neutrons leave the plasma and are stopped in so-called blankets, which are modules surrounding the plasma. The neutrons deposit all their kinetic energy as heat in the blanket. The blankets also contain lithium in order to breed fresh supplies of tritium via a nuclear reaction. The "ash" of the fusion reaction - helium - is removed via the divertor. This is the section of the containing vessel where the particles leaving the plasma hit the outer wall. The outer magnetic field lines of the tokamak are shaped such that they intersect the wall at certain places, namely, the divertor plates. Only a small fraction of the fuel is "burnt" so that deuterium and tritium are also found in the "exhaust" and can be re-cycled. The tritium produced in the blankets is extracted with a flushing gas - most likely helium - and delivered to the fuel cycle. The heat produced in the blanket and the divertor is transported via water or helium to the steam generator and used to produce electricity to feed to the grid. A small fraction is used to supply electricity to the various components in the plant itself. Electrical power is required mainly for the cryogenic system that provides low temperature helium to the superconducting magnets, for the current in the magnets, and for the plasma heating and current drive systems.

The reactor core is arranged in different layers, like an onion. The inner region is the plasma, surrounded by the first wall and blanket. All this is contained in the vacuum vessel. Outside the vacuum vessel are the coils for the magnetic field. Since the magnets operate at very low temperatures (superconductors), the whole core is located inside a cryostat.

2.1.4 Major Progresses in Core Plasma Physics in Korea

Progress on the path to ignition in magnetic confinement fusion research is best characterized by the improvement in the triple product. As described above, the triple product is the product of plasma temperature, plasma density, and energy confinement time. Figure 4 depicts the increase of the triple product by five orders of magnitude in the last four decades. Only a factor of 5-6 remains to be overcome before ignition is reached. The first promising results were achieved in the Russian tokamak T3, following which tokamaks were constructed in many countries from the beginning of the 1970s. As an example, construction of the Joint European Torus (JET) started at the end of the seventies and it went into operation in 1983. JET demonstrated considerable power production in 1997 through experiments with deuterium and tritium [6]. A total of 16.1 MW fusion power was produced for about one second and about 4 MW for a few seconds. The fusion reaction produced for a short period nearly as much energy (65%) as was delivered to the system in the form of

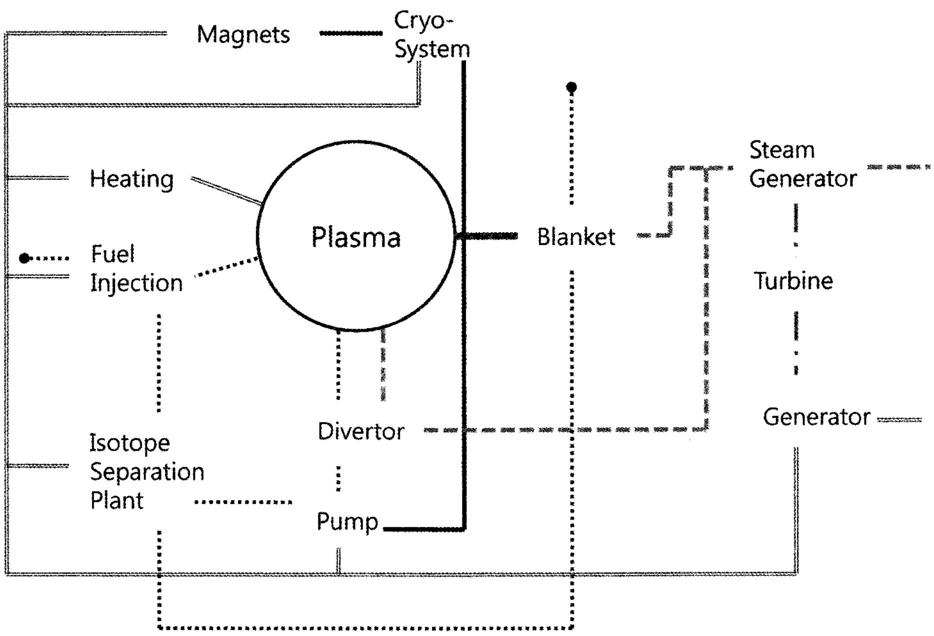


Fig. 3. Flow Chart Power for a Future Fusion Reactor : Fuel(····), Electrical (═), Heat(---), Neutron(—), Mechanical Power(-··-) and Cooled Helium(━)

external heating, corresponding to Q = 0.65. While this is an outstanding result in itself, it also demonstrated the principle of alpha-particle heating, as described above.

While significant progress has been made in terms of fusion plasma performance, there still remain many issues of physics that should be resolved to approach ignition. Among them, particularly important issues are: improvement of the energy confinement time, plasma stability, particle and power exhaust, and α particle (helium nuclei) heating. Here, the confinement time is - as mentioned above - a measure of the heat insulation of the plasma core. In a magnetic confinement fusion device, such as a tokamak, plasma core energy is typically observed to be lost much faster than expected from the usual collisional transport process. This anomalous heat transport is now well recognized as being due to plasma turbulences that occur spontaneously inside plasma. With respect to improving the energy confinement time there are basically two approaches. One is simply to increase the machine size, since a larger amount of plasma will insulate the core better than a smaller amount of plasma. The main problem in this case is the high construction cost, which increases rapidly with the machine size. The other is to reduce the anomalous heat transport rate by suppressing the turbulence level.

The issue of α particle heating has seen the least amount of study thus far among the aforementioned areas of concern. The main question is how the plasma confinement and stability described above are affected when main heating occurs by internal α particles, rather than via external heating systems such as microwaves or a neutral beam. While many theoretical studies have been carried out in relation to α particle heating few experimental works have been performed due to the difficulty of D-T experiments in existing devices. It should thus be noted that this issue is indeed one of the main research goals of ITER, where a substantial fusion power of about 500MW will be routinely produced. This ITER target value of fusion power is anticipated on the basis of the empirical scaling law from existing devices and as such it can also be affected by the α particle effect.

Here, we briefly introduce the physics research goal of the KSTAR device [2], which entered into the operation phase with successful first plasma generation in 2008. The main research objective of KSTAR is to demonstrate the steady-state operation of the high-performance advanced tokamak mode. Here, the 'advanced tokamak mode' or AT-mode refers to the operation mode that provides improved confinement and stability properties up to the level required for an economic fusion reactor. Another important element for the AT-mode is a high bootstrap-current fraction in a range of 80-90% of the total plasma current. The bootstrap current is a self-generated current inside the plasma, and thus a high fraction thereof means that the grid power to drive the required plasma current externally can be substantially reduced, resulting in an increase of Q. Figure 5 shows the target space of KSTAR operation mode in terms of normalized plasma beta (β_N) and pulse length. In comparison with other devices, the maximum target values of KSTAR are nearest to those required for an advanced DEMO reactor, which is expected to follow ITER as the final test device for electricity generation before the commercial fusion power plant.

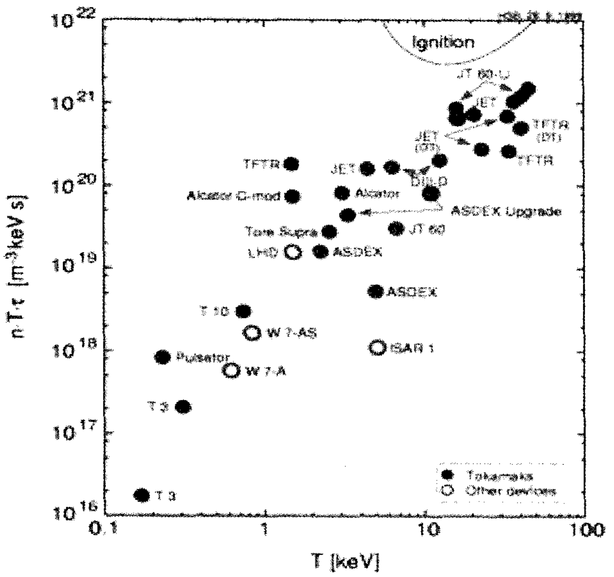


Fig. 4. The Development of the Triple Product of Plasma Temperature, Plasma Density, and Energy Confinement Time in the Last Three Decades

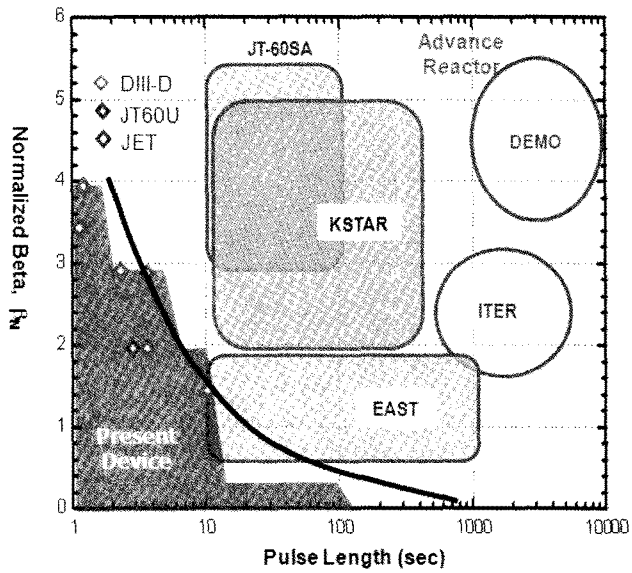


Fig. 5. Target Operation Space of KSTAR, Compared with Other Devices. EAST (China) Started Operation from 2007 and JT-60SA (Japan) is Planning to Operate from around 2015

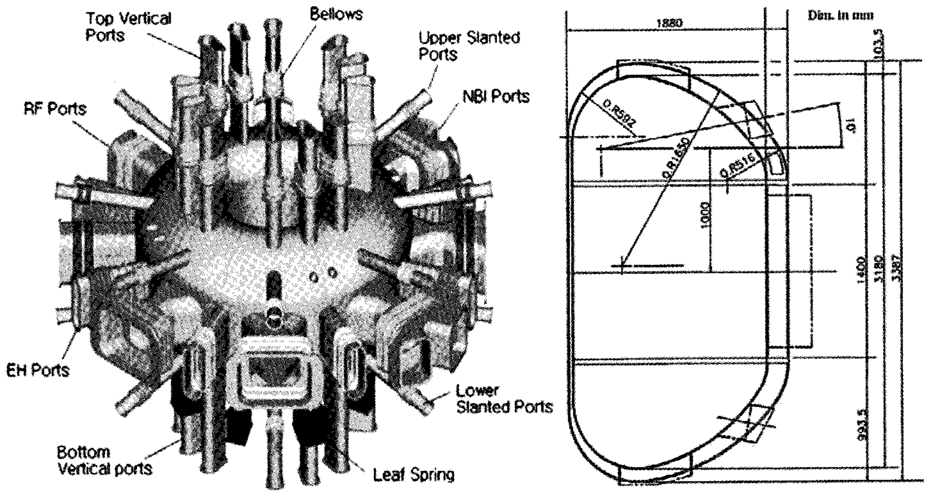


Fig. 6. Schematic and Cross-sectional View of the VV Body

Finally, we note that in order to achieve this research goal KSTAR still needs to be upgraded substantially, particularly in the ancillary systems, such as the heating and current drive system. In addition, the capability of fusion plasma theory and simulation needs to be improved significantly for accurate analysis and prediction of experimental results from KSTAR.

2.2 Main Structure and Mechanical Systems

The main structure of the magnetically confined fusion device is normally composed of a vacuum vessel (VV), a cryostat and thermal shield for the superconducting device, support structures, and a vacuum pumping system. Due to absence of active fusion research activities prior to the KSTAR project, development of the main structures was launched at the start of the KSTAR project. Consequently, most of the main structures were implemented during the design, fabrication, and assembly and integration stages of KSTAR. As a result, intermediate-sized, main structures of the tokamak device have been successfully developed through the KSTAR project and several key technologies were established for future fusion researches and reactors [7].

The vacuum vessel of the fusion device provides an essential vacuum environment for plasma generation and fusion reaction. The vacuum vessel (VV) also plays a role in the support structure for the in-vessel component. Because the vacuum vessel carries tremendous current during the plasma disruption, it should be mechanically robust under huge electromagnetic forces. Moreover, the current flow in the vacuum vessel decisively affects the magnetic field configuration. Therefore, the vacuum vessel is the most important and fundamental structure for fusion devices. The KSTAR VV consists of a double-walled, D-shaped body structure that has 72 ports with bellows and leaf-spring-style VV supports. The D-shaped

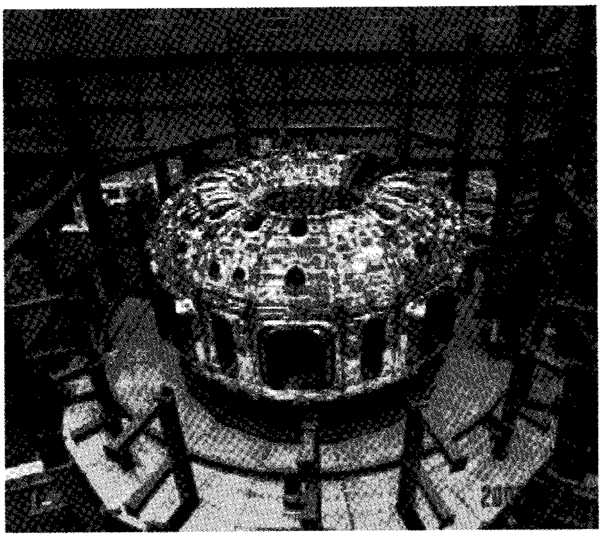


Fig. 7. The KSTAR VV Thermal Shield Mounted on the VV

VV is made of SA240-316LN. For reinforcement of the double-wall structure of the vacuum vessel, 32 equally spaced poloidal ribs and two toroidal rings are installed in the inter-space of the two shells. The two shells, the port stub walls, and the ribs form a water flow channel for the vessel baking and cooling operations. The volume of the VV including the ports reaches 100 m^3. Figure 6 shows a schematic and cross-sectional view of the VV body. Note that several technological challenges for development of a large-scale vacuum vessel for the fusion reactor must still be overcome-these issues will be addressed as part of the ITER project.

The cryostat is a single-walled vacuum vessel that provides a vacuum environment for the entire space in which all of the SC magnets and sub-systems are contained. It requires a target vacuum pressure ($<1 \times 10^{-2}$ Pa) to allow the cool-down start of the cold mass. A cool-down experiment of KSTAR has shown that several key engineering aspects were satisfactorily established for a large-scale vacuum chamber. The vacuum tightness of the cryostat was finally checked after KSTAR cool-down, in which the vacuum pressure reached below 3×10^{-6} Pa. The development of technologies for the cryostat for the fusion reactor, however, entails considerably different engineering aspects.

The thermal shield is a unique system required in superconducting fusion devices. Therefore, evaluations of various thermal shields for KSTAR comprised the first trial in Korean fusion research. The KSTAR thermal shield is strongly expected to be a prototype of the ITER thermal shield. There are three different KSTAR thermal shields: a vacuum vessel thermal shield (VVTS), a cryostat thermal shield (CTS), and a port thermal shield (PTS). The main shield panel is fabricated from 1 316L plate 3 mm in thickness and stainless steel pipe with a 7 mm ID. A wall with a thickness of 1.5 mm is welded on the panel. The shield panel has roughly two types of supports, which are made of epoxy glass to minimize heat transfer by conduction. As the VVTS is placed in a narrow gap between the TF superconducting magnet and the vacuum vessel, the VVTS panel is coated with silver at a thickness of 10 micro-meters in place of the use of multi-layer insulation (MLI). The CTS is comprised of 30 layers MLI to mitigate the effects of thermal radiation from the cryostat surface to the superconducting coils. Figure 7 shows the VVTS independently installed on the VV surface. The cool-down process fully verified that the KSTAR thermal shield was successfully designed and functioning. Therefore, one of the key technologies for a superconducting fusion reactor has been established.

The highlights of the engineering facets of the superconducting tokamak construction are the assembly of the main structures and the system integration. Given that KSTAR was the first fully superconducting tokamak ever built in the world, there is no reference for the machine assembly. Therefore, various key strategies and technologies that have been developed during KSTAR assembly can provide outstanding examples for future fusion device construction. Notably, Korea will contribute 100% of ITER assembly tooling.

2.3 Superconducting Magnet System

The major task for superconducting magnets are on the development of the KSTAR superconducting magnet system and the ITER TF conductor.

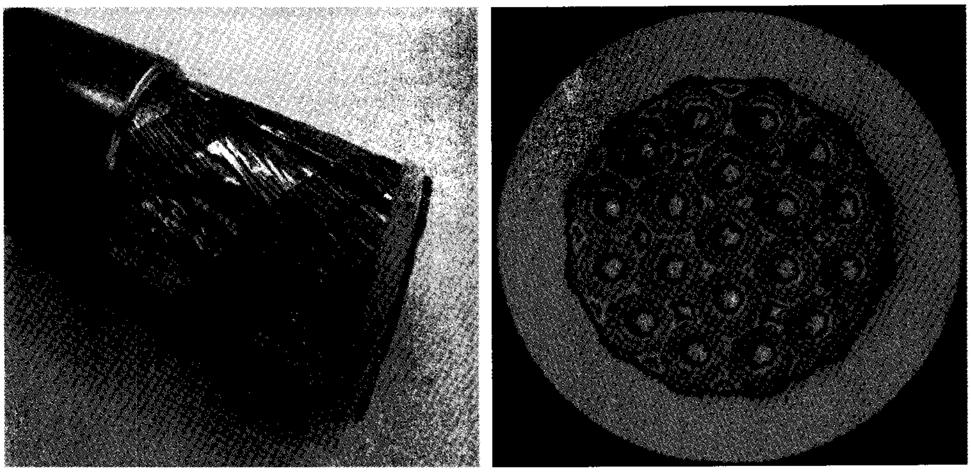
2.3.1 KSTAR Superconducting Magnet System

The KSTAR superconducting magnet system consists of 16 TF (Toroidal Field) coils and 14 PF (Poloidal Field) coils [8,9]. Both the TF and PF coil systems use internally cooled superconductors. The TF coil system provides a field of 3.5 T at a plasma center, with a peak flux density at the TF coils of 7.5 T, and the stored energy is 470 MJ. Incoloy 908 conduit and Nb3Sn superconducting cable are used for the TF CICC (Cable-In-Conduit Conductor). The nominal current of the TF coils is 35.2 kA with all coils in series. The PF coil system provides 17 V-sec and sustains plasma current of 2 MA for 20 seconds inductively. The system consists of 8 coils in the CS (Central Solenoid) coil system and 6 outer PF coils. The PF 6-7 coils use NbTi CICC with a modified stainless steel, 316LN (STS316LN+), while other PF coils use Nb3Sn in an Incoloy 908 conduit.

The dimensions and materials of the conductors are summarized in Ref. [10-12]. A continuous winding scheme without internal joints is adopted to reduce the joint losses and engineering efforts for the jointing work. The design parameters of the TF and PF coils are summarized in Ref [13]. The total cold mass of the TF magnet is about 150 tons. The coolant of the TF coils is supercritical helium with an inlet temperature of 4.5 K and an inlet pressure of 5 bars. There are four cooling channels per TF coil with a total mass flow rate in 16 TF coils of 300 g/s. The designed peak currents of the PF coils are 25 kA and 20 kA for the Nb3Sn conductor and NbTi conductor, respectively. The CS coils are segmented by four pairs of solenoid coils with different numbers of turns and will be operated with different current values to meet the strong requirement of plasma shaping. The cooling conditions for the CS and PF coils are similar to those of TF coils. The total helium mass flow rate in the CS and PF coils is about 250 g/s.

An extensive stability analysis for the SC magnet system has been performed. The set of developed simulation codes includes 1-D and Quasi 3-D transient quench analyses of the CICC magnet system and operation analyses of the Tokamak PF and TF coil systems. The results from the code will be used for determining operation parameters of the desired operation scenarios before real operation of the tokamak.

The procedures of coil fabrication are as follows: (i) CICC leak test; (ii) grit blasting; (iii) coil winding; (iv) attachment of helium feed-throughs and joint terminations; (v) heat treatment; (vi) insulation taping and ground wrapping; (vii) vacuum pressure impregnation (VPI); (viii) encasing; and (ix) test and delivery. The TF magnet structure consists of a case, inner inter-coil structure (IIS), outer inter-coil structure (OIS), cooling line, joint box, and other interfacing structures [14]. On each TF coil an in-plane magnetic force of 15 MN is stressed by TF charging and out-of-plane force by CS, PF, and plasma current. To sustain these magnetic forces, the TF coil has a wedge-shaped structure on the inboard leg and an inter-coil structure with shear keys. The cooling routes of the TF structure are connected to TF coils in series. The



(a) ITER TF Conductor (b) Nb3Sn Strand

Fig. 8. ITER TF Conductor and the Cross-section of the Nb3Sn Strand

cooling line is embedded between the TF structure and cooling pad, which is brazed on the TF structure. The CS structure consists of inner and outer shells, top and bottom blocks, flexible joints, and stoppers. The major functions of the CS structure are both a mechanical support and a structure for supplying pre-compression of about 15 MN on the CS coils [15]. The cooling lines of the CS structure are connected to CS coils in series. The peak stress including pre-compression is about 500 MPa at the neck part of the inner shell during operation.

A full size TF model coil and CS model coil were developed and tested in the large coil test facility in NFRI. The test results were successful and all of the KSTAR superconducting magnets were successfully fabricated. During the first commissioning of KSTAR tokamak, the magnet system performed well, meeting expected goals.

2.3.2 ITER TF Superconducting CICC

Korea is participating in the ITER project and are planning to deliver 20.18% of the ITER TF conductor. Preparations for the development of the ITER TF conductor have been actively carried out since 2004. An extensive study has been performed for the development of the Nb3Sn superconducting strand for the ITER TF conductor [16]. In particular, the theoretical understanding of the characteristics of the superconducting strand has provided a sound background for te further development of a high performance superconducting strand [17-21].

The ITER TF conductor consists of 900 superconducting strands and 522 Oxygen Free copper strands. Two types of ITER TF conductors were developed in Korea and tested in the CRPP Sultan test facility. The test results of the prototype conductors became the base for the activity of design change. Two types of TF conductors were developed based on the new design. Fig. 8 shows the

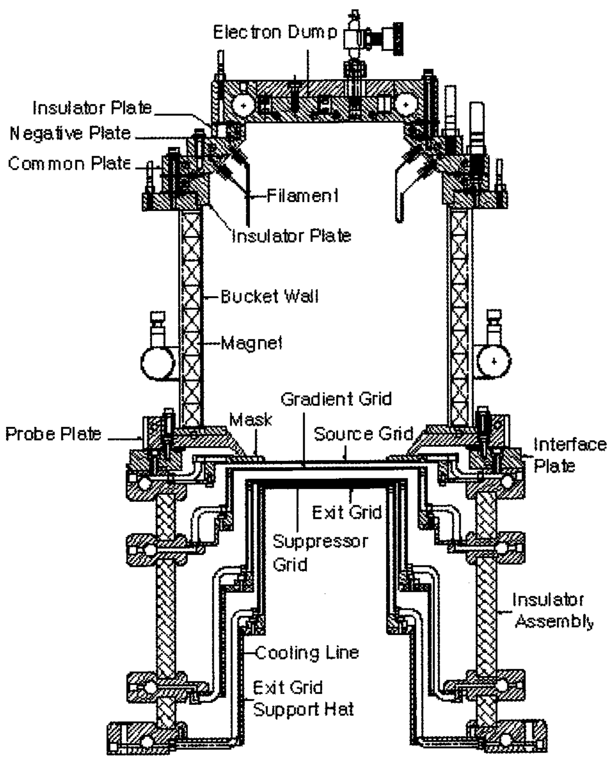


Fig. 9. Prototype KSTAR Ion Source

ITER TF conductor and the cross-section of the Nb3Sn strand developed for the qualification test. The CPQS (Conductor Performance Qualification Sample) test results show that the current sharing temperature of the Korean conductor is 6.3~6.6 K, which satisfies the ITER requirement (> 5.7 K).

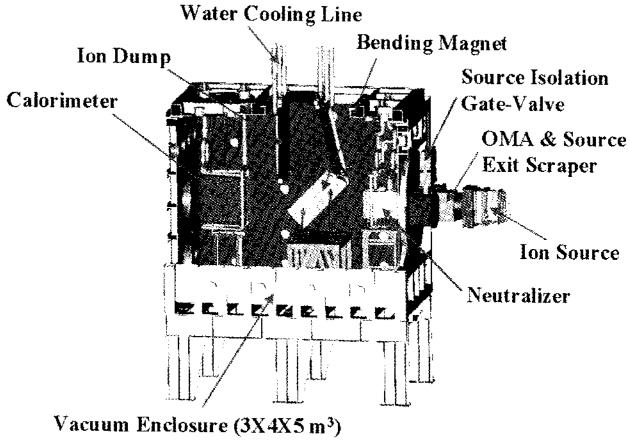


Fig. 10. Prototype KSTAR NB System

3. ANCILLARY SYSTEM

3.1 Heating Systems

The four dominant means of heating and current drive are neutral beam injection (NBI), ion cyclotron (IC) wave, electron cyclotron (EC) wave, and lower-hybrid (LH) wave. Among these methods, the ion and electron cyclotron waves and the lower-hybrid waves are usually called radio-frequency (rf) waves. In NBI, an energetic neutral beam is injected from the external accelerator into the plasmas while in radio-frequency heating, high-frequency waves are generated by oscillators outside the tokamak. If the injected particles (NBI) have momentum and the applied waves (IC, EC, LH) have a particular frequency (or wavelength), their energy can be transferred to the charged particles in the plasma, which in turn collide with other plasma particles, thus increasing the temperature of the bulk plasma and for certain conditions generating non-inductive plasma current.

3.1.1 Neutral Beam Injection (NBI) Heating System

Two neutral beam (NB) lines will provide plasma heating, current drive, core fueling, pressure profile control, and active beam diagnostics in KSTAR. The deuterium beam power per a beam line is 8 MW with a maximum particle energy of 120 KeV. In order to construct the required NB system, all of the NB components, the ion source, and the related power supplies should be developed to cover a distinctive KSTAR parameter, a 300-s operation time, on the bases of proven technology. The ion source [22,23], the acceleration power supply [24], the cryo-pumping system [25,26], and the cooling module for the beam line components are the critical components of this system. A prototype KSTAR neutral beam system, the main purposed of which is to test the developed ion source (Fig. 9) and beam-line components (Fig. 10), such as the calorimeter, neutralizer, bending magnet, ion dumps, and cryo-sorption pump, has been constructed.

Recent activities have been concentrated on making high energy, high current, and a long beam during the test. The critical component to determine the experimental conditions is the ion source. The maximum energy that has been tested in the system is 100 keV at 52 A of ion beam current during 1 second. The measured ion ratios [27] were 82 %, 5 %, and 13 % for the beams of H^+, H_2^+ and H_3^+, respectively, and the optimum beam divergence was proved to be less than 1 degree. The present achievement in the beam current is still below the required specification, and thus further R&D work must be performed to increase the beam current such that it meets the final goal. This target could be achieved by upgrading the plasma generator and improving the accelerator geometry (for optics and transparency).

3.1.2 Ion Cyclotron Resonance Frequency (ICRF) Heating System

In Korea, rf application to the plasma has began from the area of low temperature processing plasma, which is widely used in the semiconductor industry, and ICRF heating experiments in KT-1 and Hanbit were also reported. Design of the ICRF heating system in KSTAR commenced with cooperation with ORNL in 1996. KAERI is serving as the main institute for developing the system. Kyunggi University and Seoul National University conducted modeling work for ICRF heating. In addition, a preliminary experimental study on ICRF-assisted discharge cleaning and tokamak startup was carried out from 2005-2007 through participation in ICRF experiments on ASDEX-U and DIII-D, which are similar in torus dimension to KSTAR.

The KSTAR ICRF system was designed to deliver 6 MW of rf power to the plasma and it provides heating and on/off axis current drives for various operating scenarios over a wide range of magnetic fields with a frequency range of 25-60 MHz. The antenna consists of four current-straps and the capability of changing the current drive efficiency to control the current profile is provided by changing the phasing between the current straps.

An antenna was designed and fabricated in 2002 after successful, detailed tests of a proto-type antenna built in 1999 [28]. The major modifications include: implementation of water-cooling in the antenna, a water-sealing method, the material of the Faraday shield, and the assembly procedure. For 300-sec operation at a high power of 6 MW, the antenna has many cooling channels inside the current strap, Faraday shield, cavity wall, and vacuum transmission line (VTL) to remove the dissipated RF power and incoming plasma heat loads. The high power and long pulse capabilities of the antenna were experimentally estimated by performing two series of RF tests in 2004: with and without water-cooling. Significant improvement in the antenna performance through active cooling was evident.

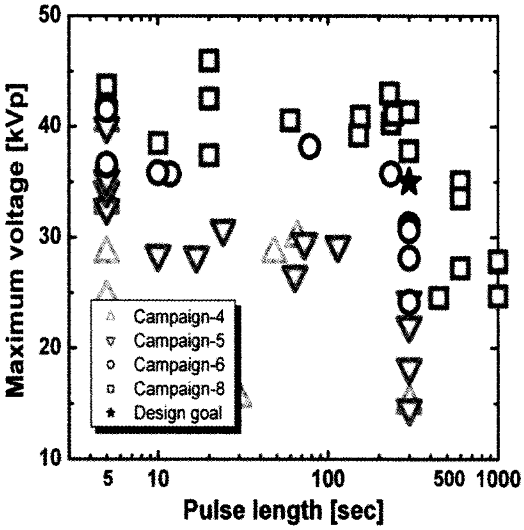


Fig. 11. Achieved RF Voltage vs. Pulse Length

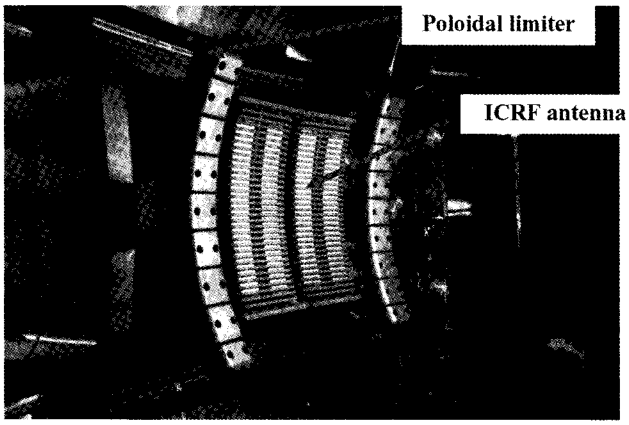


Fig. 12. ICRF Antenna Installed at KSTAR

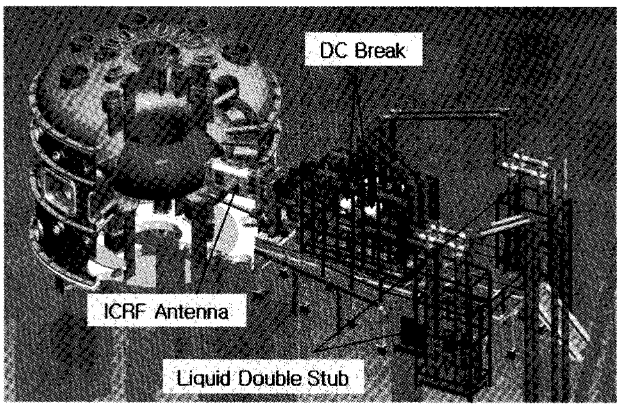


Fig. 13. 3-D View of Final KSTAR ICRF System

During the test campaign in 2005, we performed RF tests with active cooling of the antenna and transmission line. Standoff voltages of 41.3 kV for 300 sec and 46.0 kV for 20 sec, which exceed the design requirement of KSTAR ICRF system, were achieved. In addition, in order to estimate the performance of the antenna in a steady state operation, we extended the pulse length up to 600 sec and 1000 sec, which are much longer than the design requirement of 300 sec. As a result, we achieved a standoff voltage of 35.0 kV for 600 sec and 27.9kV for 1000 sec without encountering any problems [29]. Figure 11 is a plot of the achieved voltage vs. pulse length. Finally, the antenna was installed in 2007, as shown in Fig. 12, with other ICRF components and it contributed KSTAR first plasma via ICRF assisted discharge cleaning.

The ICRF heating and current drive scenario requires 4 units of 2 MW transmitters with a frequency range from 25 to 60 MHz. In order to develop the ICRF source, 100kW and 300 kW transmitters have been developed via collaboration with an industrial partner in Korea. For the KSTAR transmitter system, after the fabrication of the cavity and power supply was completed in 2004, the site acceptance test was performed until achieving 1.9 MW for 300 s at 33 MHz in 2007[30]. Although this is a very encouraging result for the development of an ICRF transmitter for ITER (International Thermonuclear Experimental Reactor), continued efforts for reliable operation are required to achieve the final goals of the KSTAR and ITER ICRF systems. For high power, long pulse operation, relevant ICRF components have been developed in the areas of liquid stub tuner, vacuum feed-through with DC breaks, and air/water cooled transmission line.

Figure 13 shows a 3-D view of the final version of the ICRF system including the antenna, resonant loop, and their supporting structure. With developing efforts in ICRF heating and the current drive experiment in KSTAR, it is necessary to participate in development of the ITER ICRF system and heating experiments to prepare the Demo ICRF system, where effects of neutrons and high energy ions are dominant.

3.1.3 Microwave Heating and Current Drive

The EC and LH waves are also called microwaves; the electron cyclotron wave has very high frequency of range of tens of GHz (GHz = 10^9 Hz), and the lower-hybrid wave has a frequency of range of several GHz. In high density and high temperature tokamaks, EC frequencies range from 84 GHz to 170 GHz and LH frequencies range from 2 GHz to 5 GHz.

The EC wave is a high power rf at millimeter wavelength, where it is regarded as a quasi-optical beam, and hence it propagates directly into the plasma without attenuation or interaction with the plasma edge, in contrast with IC and LH waves. The high frequency EC wave is generated by an oscillator called a "gyrotron". The

Table 2. The Current Status of the EC H&CD System in Tokamaks

Devices	DIII-D	JT-60U	Tore-Supra	TCV	T-10	ASDEX-U	KSTAR	ITER
Frequency (GHz)	110	110	118	82.7/118	140/140.45	140	84/170*	170
Power (MW)	6.0	4.0	1.0	3.0/1.5	0.6/0.5	4.0	0.5/3*	24
Pulse length (s)	10	10	-	-	-	2	2/300*	1000

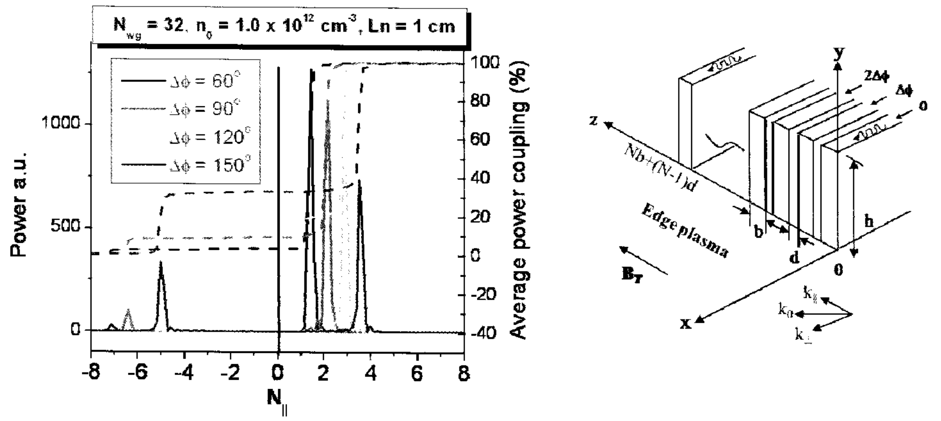


Fig. 14. The Grill of the KSTAR LH Launcher and Radiated Power Spectrum. The Number of Waveguides is 32 and the Edge Plasma Density and Density Gradient are Assumed as 1 × 10^{12} cm^{-3} and 1 × 10^{12} cm^{-4}, Respectively. The Grill Waveguide Width, b = 5.5 mm, and Thickness d is 1.5 mm, and the Height is 55.0 mm. The Directivity is 95 % for 60 deg of Phase Shift, 90 % for 90 deg, 80 % for 120 deg, and 65 % for 150 deg

generated high power EC wave is transmitted through a low-loss transmission line to the plasma. Since it is a quasi-optical beam, the launching structure does not have to be in close proximity to the plasma. The launcher is usually a mirror structure that varies the injection angle of the EC wave via focusing to a local position. The waves continue to propagate smoothly within the plasma until they encounter resonance and are locally absorbed. Absorption is generally complete and does not lead to the formation of an energetic tail or other non-linear effects. This localized absorption property allows control over the deposition profile and lends this heating method flexibility for such applications as on-axis and off-axis heating and current drive, MHD control (resistive MHD mode of neo-classical tearing mode), transport studies, and pre-ionization and assisted startup for the saving of volt-sec. Recently, local current profile control by an electron cyclotron current drive (ECCD) has been recognized as a highly promising scheme to stabilize neo-classical tearing modes (NTMs) for stable high β operation.

The EC wave is widely used in many tokamaks over the world, because it has advantages of local heating and current drive. Table 2 shows the current status of the EC H&CD systems in the KSTAR and other tokamaks. Note the all number in power is considered as the source power.

There are several important technology issues in the EC H&CD system for ITER and the reactor. The most critical components are the gyrotron with a frequency of 170 GHz, output power of 1 MW, and CW, and the torus window.

An 84 GHz, 0.5 MW ECH was developed and installed during the KSTAR construction phase, and it was used for the KSTAR first plasma campaign [31]. The initial KSTAR ECH system was equipped with one gyrotron at 84 GHz. This ECH system was successfully used for the assisted startup using the second harmonic pre-ionization in the KSTAR first plasma campaign [32].

In fusion energy the main goal that researchers are striving for is a continuously working steady state tokamak reactor, which can only be achieved with an external current drive. The efficiency of the current drive is a critical feature for the commercial use of fusion power. The recirculating power fraction should be kept as low as possible. Of the four H&CD methods, the lower-hybrid waves are the most efficient [33]. Lower-hybrid (LH) waves, excited in the outer plasma layers by phased waveguide arrays, propagate to the center of the torus with net parallel phase velocity, where they interact resonantly with the hot plasma electrons, forming a unidirectional hot-electron tail, which can carry all of the plasma current.

The waves have net parallel momentum, which upon being absorbed by electrons traveling with the wave parallel phase velocity, exerts a force that drives an electric current. The current is mainly carried by these resonant high velocity electrons because, being relatively collisionless, they retain momentum longer than bulk electrons. The current interest for LH waves lies mainly in their ability to control and shape the plasma current profile rather than in their plasma heating capabilities.

For advanced KSTAR operations, the 5 GHz LHCD system, which is the same frequency as ITER LH, would play a key role, concerning the controllability of shaping the plasma current profiles rather than plasma heating.

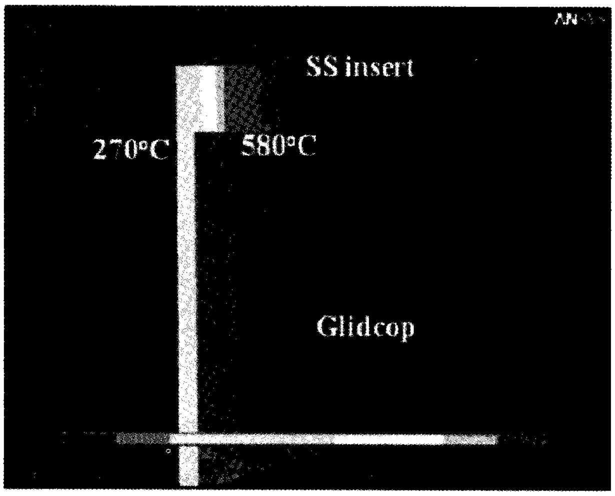


Fig. 15. The Steady-state Grill Design and Thermal Analysis for the KSTAR 5 GHz LH Launcher. The Water Cooling in the Top and Bottom of the Glidcop Septa within 5 mm of Front Gives Sufficient Cooling for Uniform Heat Flux of 100 W/cm^2 from the Steady-state KSTAR Plasma

We obtained good results through R&D efforts since the beginning of the KSTAR project. R&D mainly includes the 5 GHz steady-state high power klystron and launcher with international collaboration. Consequently, a prototype of a 5 GHz, 500 kW CW klystron has been developed with a power rating of 460 kW for 10 s and 300 kW for 720 s.

Also, a 5 GHz steady-state high power launcher was designed with similar features as the Alcator C-Mod LH launcher in collaboration with Princeton Plasma Physics Laboratory (PPPL). The launcher is composed of a front coupler facing the plasma and the waveguide channels. The front coupler is a fully active phased array antenna with phase shift between adjacent waveguides. It is called a "grill". Figure 14 shows the radiated LH wave power spectrum from the grill as a function of the parallel refractive index, which corresponds to the wave phase velocity. The characteristics of the power spectrum, the directivity, and main lobe width depend on the number of waveguides, the waveguide width, and the phase shift, as shown in Fig. 14. Heat removal is a key issue in the design of the front coupler (grill) for the steady-state KSTAR operation, but a preliminary design was recently proposed for the grill structure. Figure 15 shows the steady-state grill design and the thermal analysis results.

3.2 Operation and Control

The first integrated control system for a nuclear fusion facility has been developed for KSTAR operation [34]. The control system of the KSTAR superconducting tokamak is divided into the integrated control system (ICS) for plant operation [35] and a plasma control system (PCS) for shot operation [36].

The ICS is a network-based distributed control system aiming at integration of all plant system I&Cs, achievement of synchronized operation, and machine protection. An Experimental Physics and Industrial Control System

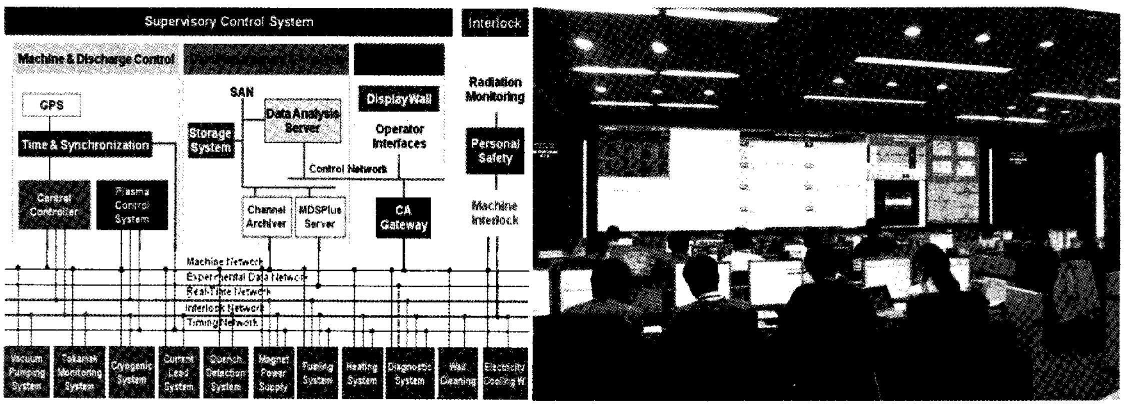


Fig. 16. Schematics of the KSTAR Integrated Control System (Left) and Picture of the KSTAR Control Room During the Plasma Experiments (Right)

(EPICS) is adopted as middleware for the ICS for the communication framework and software standard. The data of the device and experiment communicate through 5 different networks between the central control system and the distributed local control systems depending on the specific purposes. The schematics of the ICS and the central control room for the KSTAR operation are shown in Fig. 16. A time synchronization system and a plasma control system have been developed for synchronized operation and real-time feedback control of power supplies, the heating system, diagnostic system, and gas fueling system for the plasma discharge and control. The control system will be upgraded to adapt to collaborative operation requirements for higher plasma performance achievement.

Due to the superconducting nature of the KSTAR device, the machine operation requires vacuum commissioning and magnet cool-down to cryogenic temperature for about one or two months prior to the plasma experiments. All these steps require an integrated control system to monitor the machine status and achieve control as dictated by the operators. In the first campaign of the KSTAR operation, the KSTAR machine showed reliable operation for 5 months without failure, especially in the vacuum system and cryogenic facility [37]. The plasma control system controls plasmas in terms of shape, position, instabilities, and profile. Eventually, KSTAR will be accessed and operated from anywhere through the future control system as an international collaboration device to exploit high performance and steady-state plasma operation, which are essential for a fusion power plant.

3.3 Diagnostics R&D

The key parameters of the burning plasma should be measured at a very high level of reliability for machine protection and plasma control in the course of operation of the fusion reactor [38]. Therefore, the development of the diagnostic core technologies is important in fusion study for a fusion reactor such as DEMO. Korean R&D activities on diagnostics have been mainly taken for KSTAR and ITER.

KSTAR diagnostics are to be installed in a phased manner, prioritized into basic, baseline, and advanced sets according to the operation scenarios and the availability and development of related technologies. The basic diagnostics consist of a minimal set for the first plasma operation, which involves most of the magnetic diagnostics including Rogowski coils, flux loops, magnetic probes, diamagnetic loops, etc. and relatively simple systems for measuring basic plasma parameters such as electron temperature and density, impurity behavior, and neutral pressure. Baseline diagnostics including charge exchange spectroscopy, motional Stark effect, X-ray spectrometer, reflectometer, FIR interferometer, Thomson scattering, etc. are designed for the later operation campaigns of physics experiments with intensive heating capabilities and a divertor configuration. Profile measurements of

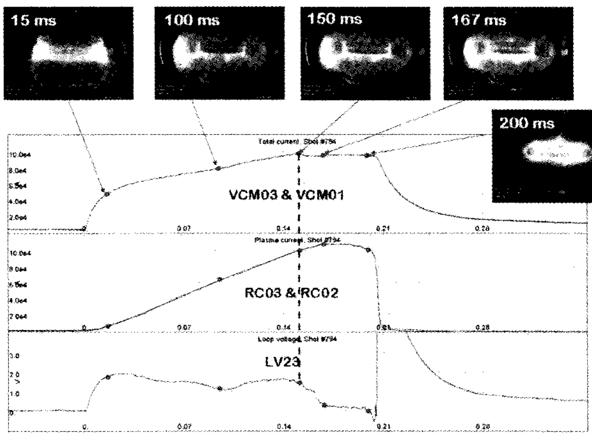
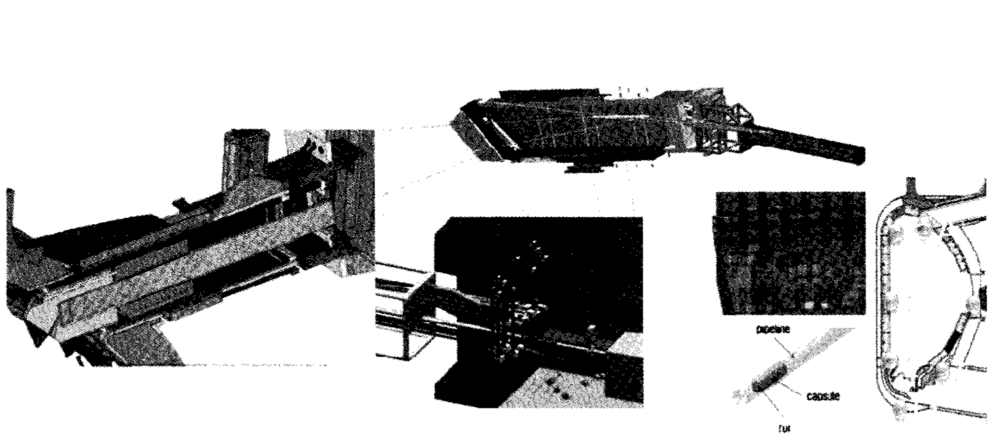


Fig. 17. Time Evolution of Plasma Parameters Measured During a Plasma Discharge (Shot # 794): Total Current from VCMs, Plasma Current from a Rogowski Coils and Loop Voltage from a Flux Loop



(a) Upper Port Plug (b) VUV Spectrometer (c) Neutron Activation System

Fig. 18. Korean ITER Diagnostic Systems

some important parameters will be implemented during these campaigns. Advanced diagnostics are those that are neither urgent nor sufficiently realized the technologies, and thus their development for implementation will be determined according to the demands and necessities during the future KSTAR operation and research.

During the first plasma campaign of KSTAR in 2008, the basic diagnostics consisting of magnetic diagnostics, a mm-wave interferometer, an electron cyclotron emission radiometer, a visible survey spectrometer, a Ha monitor, Visible TV, and an inspection illuminator were installed by using long reentrant cassettes for better access to the plasma region and the parameters of the first plasma were measured, as shown in Fig. 17.

The development of ITER diagnostics in Korea was initiated to prepare procurement for the Korea-assigned diagnostic systems. The ITER diagnostics, consisting of about 40 individual systems, are designed to measure about 45 parameters of the burning plasma in order to control, optimize, and evaluate long pulse plasma discharges. As in the case of KSTAR, the measurements in ITER are categorized into three groups according to their different roles in the ITER operational program: (1) those necessary for machine protection and basic plasma control; (2) those which can potentially be used for advanced plasma control; and (3) additional measurements for evaluating the plasma performance and for understanding important physical phenomena.

The procurement allocation of the ITER diagnostic systems among the seven Members was based mainly on the port-based diagnostic packages, which are divided into 32 procurement packages in total. Each port-based package consists of one port plug, one lead diagnostic, and other diagnostic systems assigned to the same port. As shown in Fig. 18, the Korean domestic agency (KO-DA) will procure one diagnostic upper port plug [39] and two diagnostic systems including the VUV spectrometer and the neutron activation system, which corresponds to 3.5% of the total ITER diagnostic credit. Three kinds of VUV spectrometers for monitoring impurity species in the ITER plasma will be installed at upper port 18 and equatorial port 11, respectively. The upper system is designed as a grazing incidence VUV (21~26 nm) imaging spectrometer with cylindrical and toroidal collimating mirrors and a flat-field 2-D imaging spectrograph with an MCP/CCD detector. The equatorial system employs a VUV survey spectrometer with 5 channels for wide-range coverage of the VUV/XUV spectrum (2.4~160 nm), so that high resolution of spectral lines and real time data acquisition could be realized. The divertor VUV spectrometer is designed to have two channels (12~48 nm). The neutron activation system [40] measures the neutron fluence on the first wall and the total fusion power by using encapsulated metal foils (~1 cm), which are transferred pneumatically from irradiation locations to a remote analysis station. The system measures gamma rays from the metal foil samples, which are activated by the fusion neutron flux.

3.4 Electric Power System and Power Supply Systems

The development of technologies and components for power handling has played a major role in tokamak fusion experiments. In order to confine and stabilize the plasma ring, the toroidal field coils had to generate a strong static magnetic field. A large electric power was requested to supply the magnetic energy and the joule losses in the copper windings. The duration of the experiments was limited because of the limited maximum power demand from the grid. For pulse operation, high current density had to adapt in the copper coils in order to reduce winding size. The heat in the coils during the

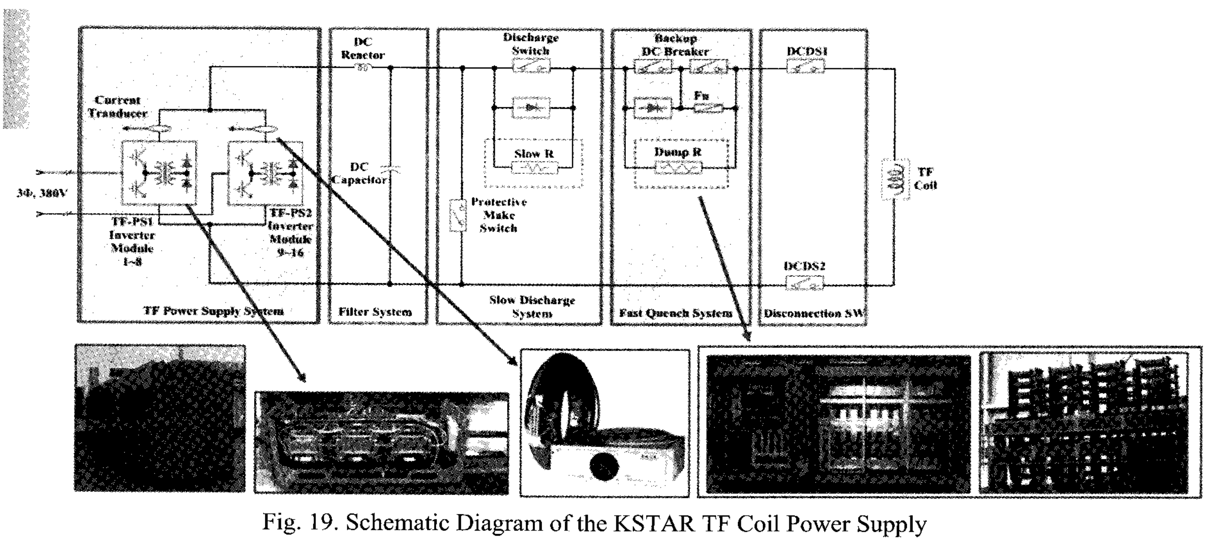


Fig. 19. Schematic Diagram of the KSTAR TF Coil Power Supply

pulse was dissipated outside between shots. To maximally dissipate the energy during biasing, the set up of the magnetic field was fast and the associated power supplies absorb electric power well over the ohmic losses. This was manifested in JET operation, where about 300MW is necessary during the plasma phase and more than 500MW for the plasma setting up [41].

In order to use a higher magnetic field, a large superconducting coil system has recently been developed. For long pulse or steady state operation, superconductors can be used to reduce losses and the electric power necessary to feed the coils. The winding is slowly charged with a large current and stores the necessary magnetic energy. The current in the coils can be maintained stationary for days or weeks, and thus the charging phase can be as long as necessary to decrease the absorbed power from the grid to low values. For ITER, considering a charging duration of 30 min, the peak active power is about 60MW, around one-tenth that of JET [42].

In the major fusion experiments a large fraction of the electric power is taken directly from the main high voltage transmission grid. However, the electric load has pulsed profiles, with durations from a few seconds to several minutes and a repetition period lasting from some minutes to hours. In supplying such loads, the main issues to be addressed are local effects on other neighboring loads and the global effects on the utility grid [43]. When the grid is not able to sustain such a load, an energy storage device, usually a flywheel generator, will be necessary. This device stores part of the energy needed for the pulse during the latency time and delivers high power during a short pulse time. The voltage distortion caused by the harmonics generated by the operation of the thyristor converter power supply should be considered [44]. Mitigation of the negative effects can be achieved with a Reactive Power Compensator & Harmonic Filter (RPC&HF) system [45]. Systems with increased additional heating power and for plasma feedback control need fast switching due to stringent requirements in terms of reduced output ripple, fast response time, and fast power flux interruption. For fast plasma control Gate Turn Off (GTO), an Insulated Gate Bipolar Transistor (IGBT) and, recently, an Integrated Gate Commutated Thyristor (IGCT) inverters are used [46].

The KSTAR tokamak requires 3.5 Tesla of TF for plasma confinement, and requires a strong poloidal flux swing to generate an inductive voltage to produce and sustain the plasma. These magnetic fields are generated by supplying extremely high DC current to TF and PF superconducting coils in a controlled manner using various specially designed power supply systems. Figure 19 shows the schematics of the KSTAR TF coil power supplies. The KSTAR power supply was developed by considering the Pohang Accelerator Laboratory (PAL) power system, which has high accuracy performance of 5 ppm at a level of several hundred amperes. The TF coil power supply has a capacity of 25 V, 40 kA class with IGBT inverter technology, which is the rectifying line voltage, and generates high frequency AC. Thus, we could achieve compactness, lighter weight, and higher efficiency. The PF coil power supply system consists of 7 independent supplies to apply current to the 7 pairs of PF coils up to 20 or 25 kA. Each PF power supply is composed of bi-directional 12 phase thyristor units and a blip resistor insertion system (BRIS) can produce a fast flux swing from a rapid change in the current achieved by using an IGCT switching device [47]. Electric power of 50 MVA and a reactive power compensation system of 7.15 MVA were used for the 1st operation of the KSTAR. For high power and long pulse operation, a 200 MVA flywheel generator and a reactive power compensation system of 60 MVA will be newly developed.

4. FUSION REACTOR STUDIES

4.1 Reactor Design

A DEMO plant for demonstration of fusion power is regarded as the last step before the development of a commercial fusion reactor. The primary requirements for the DEMO reactor can be summarized as follows: First, it should demonstrate net electric power generation. Second, it should demonstrate tritium self sufficiency. Lastly, it should demonstrate the safety aspects of a power plant and should be licensable as a power plant.

To develop a concept of a tokamak fusion reactor and to identify the design parameters, tokamak system analysis code has been under development [48] at KAERI (Korea Atomic Energy Research Institute). Various components of a reactor system are strongly inter-related and the system analysis code finds the design parameters that satisfy the plasma physics and technology constraints simultaneously. The results arising from the system analysis are used to define the concept of the reactor and identify which areas of plasma physics and technologies should be developed, and to what extent, for realization of the reactor concept. To explore the range of concepts of the DEMO reactor and a fusion power plant, assumptions on the level of physical and technological development have to be made. There will be many reactor models depending on the assumptions, from the least ambitious plasma physics combined with the least ambitious technologies to the most ambitious in all areas. In the system analysis code, prospects of the development of plasma physics and the relevant technology are included in a simple mathematical model, i.e., the overall plant power balance equation and the plasma power balance equation.

Using the tokamak reactor system analysis code, we investigated [49] the performance of the DEMO reactor in terms of the plasma parameters that characterize the performance, i.e. normalized beta value, β_N, confinement

Table 3. Radial Build of a DEMO Reactor

Component	Thickness (m)	Radius (m)
Machine Center	0.0	0.000
Bore	1.600	1.600
Central Solenoid	0.348	1.948
Gap	0.080	2.028
TF Coil	1.372	3.400
Vacuum Vessel	0.150	3.550
Gap	0.260	3.810
Inboard Shield	0.500	4.310
Inboard Blanket	0.500	4.810
Inboard First Wall	0.040	4.850
Scrape Off Layer	0.150	5.000
Plasma Center	2.500	7.500
Plasma Edge	2.500	10.000
Scrape Off Layer	0.150	10.150
Outboard First Wall	0.040	10.190
Outboard Blanket	0.800	10.990
OutboardShield	0.800	11.790
Gap	0.800	12.590
Vacuum Vessel	0.150	12.740
TF Coil	2.470	15.210

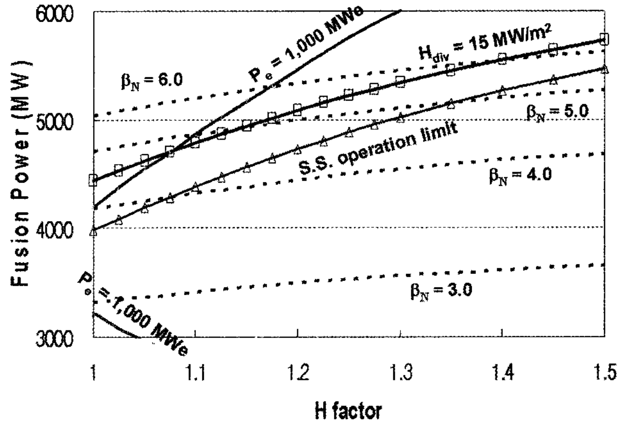


Fig. 20. Allowable Plasma Space of a DEMO Reactor

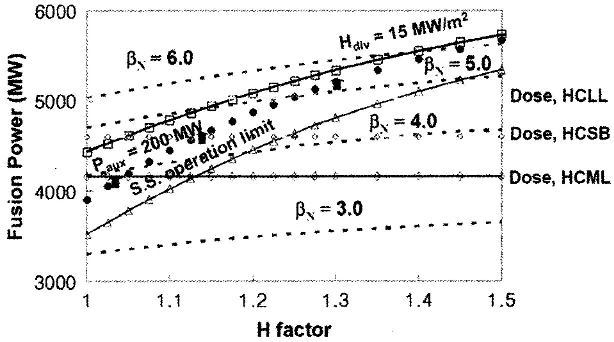


Fig. 21. Allowable Plasma Space of a DEMO Reactor with Insulator Dose Limit

improvement factor for the H-mode, H, and the ratio of the Greenwald density limit n/n_G. They are assumed to be improved beyond those of ITER, i. e., $\beta_N \geq 2.0$, $H \geq 1.0$, and $n/n_G \geq 1.0$, since these regimes must be achieved in a fusion reactor. We chose the major radius $R_o = 7.5$ m, the minor radius a = 2.0 m, the triangularity d = 0.7, the elongation $\kappa = 2.1$ and the edge safety factor $q_{95} = 4.0$, and we looked for the performance region where both net electric power of 1,000 MWe and steady-state operation are possible. We assumed the plasma density is above the density limit, $n/n_G = 1.2$, and the divertor is protected from an excessive heat load by radiation through impurity (Fe) seeding of the divertor and the main plasma. For the technology conditions, a maximum magnetic field of 15 T, thermal efficiency of 40%, and current drive efficiency of 60% are considered to be achievable in the near future. For steady-state operation, the plasma current is driven by a combination of the bootstrap current and current driven by external heating. A plasma current ramp-up is assumed to be provided by the magnetic flux of the central solenoid coil and the external current drive is provided by NBI with a beam energy of 2.5 MeV. For a breeding blanket, a He Cooled Molten Lithium (HCML) blanket is used. The HCML blanket uses He as a coolant, Li as a tritium breeder, and ferritic steel as a structural material, and the neutronic calculation was performed as presented in Ref. [50].

Table 3 shows the radial build of various sub-components of the DEMO reactor calculated from a system analysis with the aforementioned physics and technology conditions. Figure 20 shows that an electric power generation of Pe = 1,000 MWe is possible in the right hand side region of the curve "Pe = 1,000 MWe". In the upper region above the curve "Pe = 1,000 MWe", more heating and current drive power is required and thus the re-circulating power increases. Below the curve "H_{div} = 15 MW/m^2". the divertor heat load H_{div} is smaller than 15 MW/m^2. Steady-state operation is possible in the region above the curve "S.S. operation limit". For the region below the curve "S.S. operation limit", the current drive power is not sufficient to fully drive the current required for steady-state operation. The operational space where both steady-state operation and a divertor heat load H_{div} smaller than 15 MW/m^2 are possible is very narrow and found in the region $\beta_N \geq 4.0$. Hence, to expand the operational space for steady-state operation, improved technology to increase the current drive efficiency needs to be developed.

A reactor concept with small size for given performance is desirable from an economic point of view. The design of the blanket and shield, respectively, plays a key role in determining the size of the reactor, since it constraints the various reactor components and it is natural that the thickness of the blanket and shield should be determined self-consistently. For this, we coupled the system analysis with the one-dimensional neutronic calculation to determine the reactor parameters in a self-consistent manner. Neutronic optimization was performed from the aspects of the tritium breeding ratio (TBR), nuclear heating, radiation damage to the toroidal field, etc. With the coupled system analysis and one-dimensional neutronic calculation, we assessed the impact of various types of blanket concepts such as a HCML, HCSB, and HCLL blanket model on the reactor design. It was found that for given plasma performance, i.e., normalized plasma beta, plasma confinement, and plasma density, the minimum reactor size for each blanket concept can be determined by the self-consistent system analysis. Thus, with the insulator radiation dose limit (10^7 Gy), to access a plasma performance characterized by $\beta_N = 4.8$, $H = 1.2$ and $n/n_G = 1.2$, it was found that $R_0 = 7.81$ m for a HCML blanket, $R_0 = 7.60$ m for a HCSB blanket, and $R_0 = 7.47$ m for a HCLL blanket are the minimum values. In Fig. 21, we investigate the allowable operation space for each blanket model with other parameters corresponding to those in Fig. 20. It is shown that the shielding capability of the blanket and shield is the main constraint in determining the reactor performance.

From the system analysis we found that to access the operational space where steady-state operation and higher electric power are possible, the following areas need to be further developed.

1. The bootstrap current must be high enough and the current drive efficiency must be substantially higher than those anticipated in the ITER
2. Both improved plasma physics and technology are required to handle a high heat load on the divertor
3. The shielding capability of the blanket and shield needs to be improved

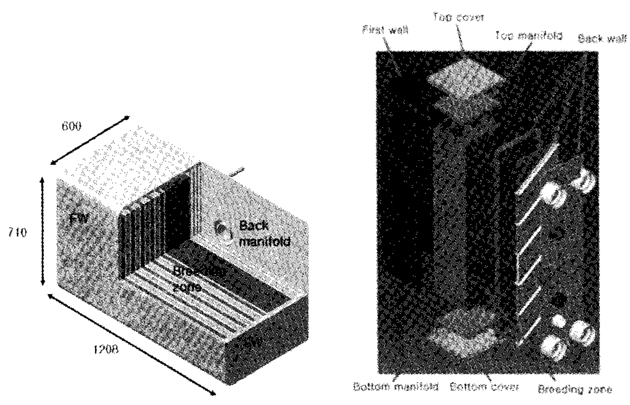
4.2 Blanket R&D

The breeding blanket concept has to be defined prior to the DEMO study, since it is essential for characterizing the DEMO. The concept of HCSB (He Cooled Solid Breeder) and HCML (He Cooled Molten Lithium) blankets were developed as options for the DEMO blanket. A system analysis has been performed to investigate their relevancy to DEMO blanket concepts, since tritium breeding capability, neutron energy multiplication, shielding capability (nuclear heating rate in inboard toroidal field (TF) coil, insulator dose at TF coil winding pack), etc. have impacts on other parameters of the reactor components.

The design and R&D on key technologies have progressed for testing the blanket concepts in ITER, through a TBM (Test Blanket module) and shield blanket. The technologies of heat removal, joining, and fabrication of the blanket first wall will be verified through the ITER shield blanket development. The tritium breeding capability and heat removal capability of the breeding blanket will be tested in the ITER TBM program, since one of the main objectives of ITER is to test and validate the design concepts of tritium breeding blankets relevant to a power producing reactor.

The ITER shield blanket consists of the first wall panel and shield block, providing the main thermal and nuclear shielding to the vessel and external machine components. The basic concept is a modular configuration with a mechanical attachment system, attached directly to the vacuum vessel. The first wall panel is mounted on the shield block [51,52]. The first wall employs beryllium tile as a plasma facing material, joined to the SUS structure by HIP (Hot Isostatic Pressing) joining technology with a copper interlayer.

The HCSB blanket uses He as a coolant, Beryllium (Be) as a neutron multiplier, and Ferritic/Martensitic Steel (FMS) as a structural material. For the ceramic breeder, Li_4SiO_4 (Li_2TiO_3 as an option) and Be are used in a pebble-bed form in order to accommodate any possible geometrical changes during neutron irradiation. He coolant is at a static pressure of 8 MPa with an inlet temperature of 300°C and an outlet temperature up to 500°C depending on the operating conditions. He with 1% hydrogen is used as a purge gas for tritium extraction from the ceramic breeder. In the HCSB design, the amount of Be is reduced by replacing some of it with graphite as a reflector. The thick graphite reflector is advantageous in that it can play a role of a heat sink in



(a) HCSB Concept **(b) HCML Concept**

Fig. 22. Two Breeding Blanket Concepts for ITER TBM: (a) Solid Breeder Type HCSB, (b) Liquid Breeder Type HCML

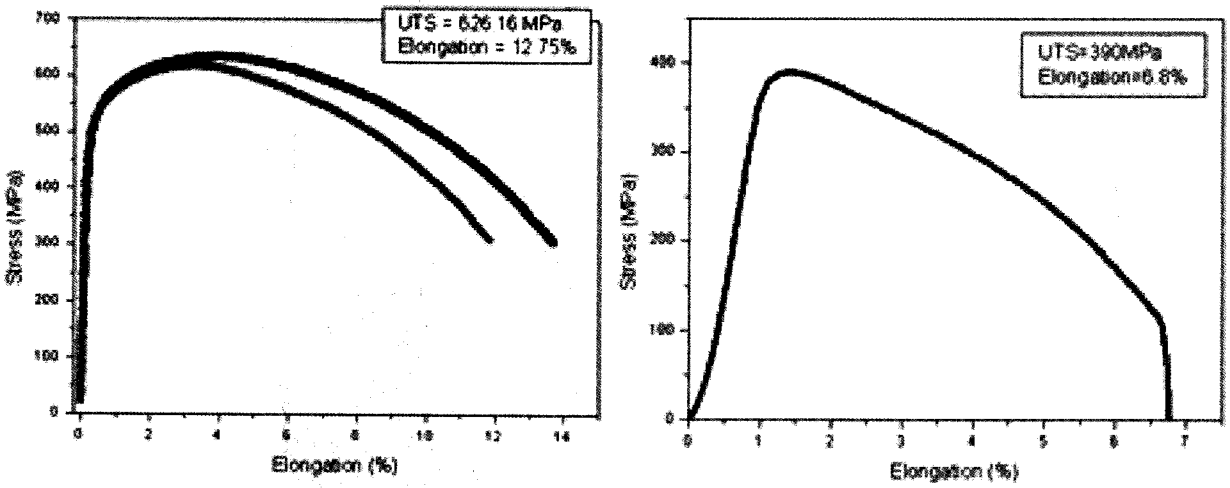


Fig. 23. Mechanical Properties Measured at Room Temperature and 550 °C

the case of a loss of coolant accident [53-55].

The HCML TBM uses He as a coolant and Li as a tritium breeder. Ferritic/Martensitic Steel (FMS) is used as a structural material and graphite is used as a reflector in the breeder zone to increase the Tritium Breeding Ratio (TBR) and the shielding performances. In the HCML blanket, the neutron leakage from the blanket to the shield region is minimized by using a graphite reflector and no special neutron multiplier such as Be is necessary. It was shown that in the graphite-reflected blanket, TBR is maximized around the natural Li composition and a local TBR greater than 1.315 with the coverage fraction of the first wall, 80%, can be easily obtained. It was also shown that the shielding capability increases by combining moderators (such as TiC and ZrH2) and absorbers in the shield region. The potential advantages of the HCML TBM design in terms of being ultimately utilizing in the DEMO blanket design are as follows: virtually no concern for tritium permeation into the coolant system; a simplified high-performance system with a He-direct cycle; alleviated material problems due to a very slow Li flow speed; no concern for Li fire in an inert gas environment; marginal MHD (Magneto-Hydro-Dynamics) effects due to a very slow Li flow; no Po-210 & Hg-204 generation; and a Li loop as a redundant cooling circuit in the case of a He loss accident [56]. Two TBM concepts are shown in Fig.22.

Various R&D on key technologies such as joining and fabrication technology, breeder development and characterization, He cooling technology, tritium technology, etc. has been carried out to resolve key issues in each concept and to validate their safety when installed in ITER. More R&D will be performed to validate the breeding blanket technologies, including tritium extraction and self-sufficiency, high grade heat extraction, material compatibility, manufacturability, reliability, availability, and safety.

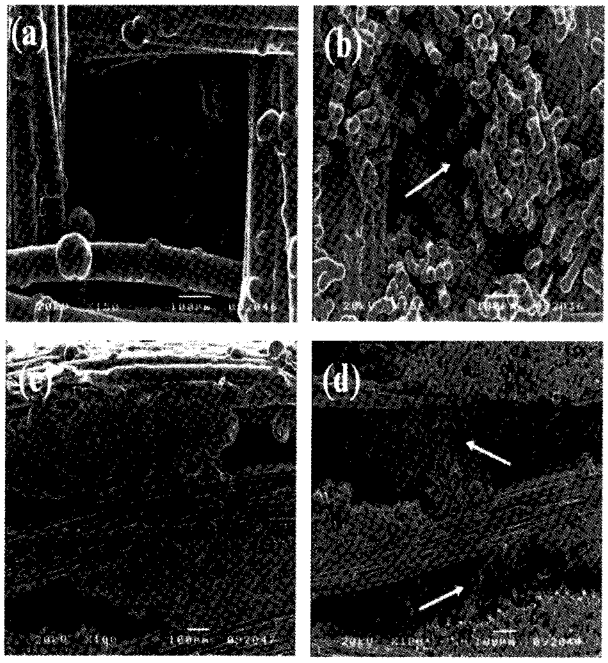


Fig. 24. Typical Microstructure of SiC_f/SiC Composite Prepared by (a),(c) Conventional CVI Process and (b),(d) WA-CVI Process

4.3 Status of Material Research and Development in Korea

The severe environment associated with a fusion reactor (radiation, heat flux, chemical compatibility, thermo-mechanical stresses) poses numerous materials science and technology challenges, and will require the development of materials with excellent physical and mechanical properties as well as stability of these properties in the adverse fusion operating environments. In particular, the development of materials for in-vessel systems will provide acceptably high performance and reliability and exhibit favorable safety and environmental features for establishing fusion reactors as a viable energy source [57,58].

The material R&D for fusion energy systems including the ITER TBM is considered integral to the overall project and is being planned as a part of the fusion energy system R&D program in Korea. The following fusion reactor related material technologies were developed and are being developed for the next generation nuclear energy system, even though the primary target is not a fusion application; 1) characterization of low activation ferritic/martensitic (FM) steels such as EUROFER and F82H in NFRI, and the development of FM steel with a focus on investigation and improvement of the thermal and creep properties and evaluation and simulation of the irradiation behaviors (in KAERI) [59], and development of FM steels for application to the structure material of DEMO (in KIST); and 2) fabrication and characterization of SiC_f/SiC composite and C/SiC FGM layer coated C_f/C composite (in KAERI) [60-62], and 3) development of nano-powder technologies and the fabrication of pebbles of Li-based oxide (in KAERI) [63-65]. Additionally, the development of HIP joining technology between Be/Cu-alloy/Stainless steel or FM steel is being performed for ITER first wall qualification mock-ups (in KAERI) [66] and tungsten(W) and/or W-alloy coating technologies are also under development for application to the plasma-facing first wall material of DEMO (in KIST) [67]. A preliminary study on modeling and simulation of irradiation damage for fusion materials is also being carried out in KAERI, NFRI, and KIST in cooperation with several universities [68,69].

Korea FM steel was fabricated by vacuum induction melting in a laboratory scale. The ingots were placed in the 1150 °C furnace for 24 hours for homogenization, and forged at the same temperature to remove solidified microstructure. Hot rolling at 1050 °C up to 75 % was followed to control the size of the specimen for microstructure and mechanical property analyses. The specimen was heat treated under the same conditions as the reference heat treatment of FM steel [70]. The ultimate tensile strength was 626 MPa at room temperature and changed to 390 MPa at 550 °C. The elongation was decreased from 12.8 % to 6.8 % as the experimental temperature was increased from room temperature to 550 °C, as shown in Fig. 23.

Various fabrication processes of SiC_f/SiC composites are being developed. The thermal and mechanical properties up to 1200°C and the performance behaviors including ion irradiation and corrosion are being evaluated. To obtain a dense SiC_f/SiC composite, a new modified CVI process, called whisker growing assisted CVI (WA-CVI), was developed. Typical microstructure of CVI and WA-CVI SiC composites is shown in Fig. 24.

4.4 Licensing

World wide, licensing requirements for fusion power plants have not yet been defined clearly. ITER, which is a landmark in the path of fusion power engineering, will be the first instance of licensing a representative fusion reactor [71]. Since fusion standards are not available, fission experience provides the framework basis for licensing the construction and operation of fusion reactors [72]. Recently, ASME Sec. III Div. 4, assigned for a fusion code, has been under discussion for development in a working group.

Study about fusion reactor licensing in Korea has been initiated by the KSTAR construction project, which was carried out from 1995 to 2008. The safety analysis report of the KSTAR facility shows that exposure dose from neutron and secondary gamma rays during operation is estimated as 4.96 mSv per year, which is less than the occupational dose limit, 20 mSv. Thus, KSTAR is classified as a radiation generating device and its operation license was issued in December 2007.

ITER will be constructed in France and it will have to comply with the French requirements in terms of safety, quality, and licensing. ITER has been classified as a basic nuclear installation. French quality order requires that ITER should develop a quality assurance (QA) program and submit to the safety authority for approval. All the domestic agencies (DAs), which are mainly responsible for in-kind procurement of ITER, should submit their QA programs to the ITER organization for approval. The licensing process starts by examination of the safety options. The preliminary safety report is submitted to the French regulatory bodies, which launch public hearings and safety reviews. The main safety concerns are the confinement of tritium, activated dust in vacuum vessel, and activated corrosion products in the coolant of plasma-facing components [73].

The other relevant issues may be reliability, availability, maintainability, and inspectability (RAMI) of the machine during operation. The ITER RAMI program started at the end of 2007. It is characterized by functional breakdown and deployment, failure mode, effect and criticality analysis (FMECA), and simulations using reliability block diagrams for all systems. ITER is attempting to implement RAMI and diffuse it throughout the organization including DAs. In KSTAR, a high degree of RAMI is also required and adopting the RAMI approach is desirable.

5. ROADMAP

The first technology roadmap for fusion energy was developed as a part of the nuclear technology roadmap (NuTRM) by the Korea Nuclear Society in 2003 [74]. After completing the construction of KSTAR along with the major commitment to the ITER project as an official partner, the national fusion energy development roadmap was upgraded by focusing on a clear mission - developing technical bases for its own fusion demonstration device. Experts from fission as well as fusion communities worked together and the roadmap was updated as a national fusion energy development roadmap, titled "National technology roadmap for fusion energy 2036" with the more concrete target of developing technologies for the Korean Demonstration reactor [75]. The new technology roadmap emphasizes self-sustaining technological bases for constructing a fusion power plant in a similar time window to other advanced countries in the fusion community. This challenging roadmap is considered to be formidable, because Korea can exploit its well-established nuclear technologies in power plant design, construction, and operation. Key technologies related to the tokamak as a fusion reactor core will be developed through intensive and cooperative research programs on KSTAR and ITER with elevated research capacities of universities as well as research institutes. The Korean government and the legislature recognized the importance of the program by establishing a special law not only for supporting these endeavors but also to lend them high priority in their R&D agenda.

6. CONCLUSIONS

The research history of the Korean nuclear fusion energy development is considerably short compared to that of Western countries and Japan. Nevertheless, the vision of the frontiers and the concentrated planning and support of the government in tandem with the high quality of industrial technologies will enable us to catch up to the level of the leading countries in a relatively short period in the areas of component manufacturing. Korea is, however, still behind in the areas of reactor design, fusion science, and nuclear fusion technology. Required human resources are also insufficient. Coordinated efforts by academic and research institutions as well as industries and the government should be made for further improvements in speed and efficiency. Korea is highly acclaimed as a world leading country in nuclear technology. Adopting the well-developed nuclear technologies and taking advantage of the vast experiences of power plant operation would give a synergistic effect to nuclear fusion energy development. Collaborative researches will be possible between fission and fusion in the areas of materials, thermo-hydraulics, neutronics calculation, heat transfer, plant engineering, and licensing. This will also be beneficial to the domestic nuclear industries as new fields of application of well-developed technologies will be created.

Development of nuclear fusion energy will have a fresh impact on the Korean nuclear community by creating new fields of research and development and new applications of well-developed nuclear technologies, and by providing a wide variety of chances for creation of synergy in green energy development.

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REFERENCES

[1] Charles Seife, *Sun in a bottle*, Viking Penguin, New York (2008).
[2] G.S. Lee *et al.*, "The KSTAR project: An advanced steady state superconducting tokamak experiment," *Nucl. Fusion*, **40**, 575 (2000).
[3] G.S. Lee *et al.*, "Design and construction of the KSTAR tokamak," *Nucl. Fusion*, **41**, 1515 (2001).
[4] M. Kwon *et al.*, "A strategic plan of Korea for developing fusion energy beyond ITER," *Fusion Eng. Design*, **83**, 883 (2008).
[5] L.A. Artsimovich, "Tokamak devices," *Nucl. Fusion*, **12**, 215 (1972).
[6] Keilhacker M. and the JET team, "High fusion performance from deuterium-tritium plasmas in JET," *Nucl. Fusion*, **39**, 209 (1999).
[7] H. L. Yang, and et al., "Construction, Assembly and Commissioning of KSTAR Main Structure," *Nucl. Eng. Technol.*, **40**, 439 (2008).
[8] Y. K. Oh, et al., "Design Overview of KSTAR Magnet Structures," *19th Symposium on Fusion Engineering*, **19**, 400 (2002).
[9] Y. K. Oh, et al., "KSTAR Magnet Structure Design," *IEEE Trans. Applied Superconductivity*, **11**, 2066 (2001).
[10] S. Lee, et al "Fabrication of cables for the background-field magnet system of SSTF," *IEEE Trans. on Applied Superconductivity*, **12**, 583 (2002).
[11] B. Lim, et al., "Fabrication of the KSTAR superconducting CICC," *IEEE Trans. on Applied Superconductivity*, **12**, 591 (2002).
[12] B. Lim, et al., "Development of CICC for KSTAR TF Coil System," *IEEE Trans. on Applied Superconductivity*, **13**, 1496 (2003).
[13] K. Kim, et al., "Status of the KSTAR Superconducting Magnet System Development," *Nucl. Fusion*, **45**, 783 (2005).
[14] H. J. Ahn, et al., "Engineering Design Status of KSTAR TF Coil Structure," *IEEE Trans. Applied Superconductivity*, **12**, 492 (2002).
[15] C. H. Choi, et al., "Electromagnetic Loads on the KSTAR Magnet System," *IEEE Trans. on Applied Superconductivity*, **12**, 534 (2002).
[16] Pyeongyeol Park and Keeman Kim, "Status of Nb3Sn strand development in Korea," *Cryogenics*, **48**, 347 (2008).
[17] S. Oh, et al., "A Variable Temperature Walters Spiral Probe for the Critical Current Measurement of Superconducting Strands," *IEEE Trans. on Applied Superconductivity*. **18**, 1063 (2008).
[18] S. Oh, et al., "Field dependence of the n-value and its relation with the critical current of Nb3Sn strands," *Superconductor Science and Technology*, **20**, 851 (2007).
[19] S. Oh, et al., "A Study on the Temperature Dependence of the Ginzburg-Landau Parameter, Thermodynamic and Upper Critical Field of Nb3Sn Strands," *IEEE Trans. on Applied Superconductivity*, **17**, 2635 (2007).
[20] Sangjun Oh and Keeman Kim, "A Scaling Law for the critical current of Nb3Sn Stands Based on Strong Coupling

Theory of Superconductivity," *Journal of Applied Physics*, **99**, 033909 (2006).

[21] Sangjun Oh and Keeman Kim, "A Consistent Description of Scaling Law for Flux Pinning in Nb3Sn Strands Based on the Kramer Model," *IEEE Trans. on Applied Superconductivity*, **16**, 1216 (2006).

[22] Byung-Hoon Oh, Doo-Hee chang, Seung Ho Jeong, Kwang-Won Lee, Sang-Ryul In, Byung-Joo Yoon, Min-Seok Seo, Jung-Tae Jin, Bum-Ryul Kim, Dae-Sik Chang, "Long pulse beam extraction with a prototype ion source for the KSTAR neutral beam system," *Rev. Sci. Instrum.* **79**, 1140 (2008).

[23] Doo-Hee Chang, Seung-Ho Jeong, Byung-Hoon Oh, Kwang-Won Lee, and Chang-Seog Seo, "Arc discharge efficiency of a multi-megawatt long pulse ion source for the KSTAR neutral beam injector," *Plasma Sources Sci. and Technol.* **14**, 336 (2005).

[24] B.H. Oh, K.W. Lee, S.H. Jeong, I.H. Song, H.S. Shin, C.H. Choi, "Development of an Acceleration Power Supply for the KSTAR Tokamak", 3rd IAEA Technical Committee Meeting on Steady-State Operation of Magnetic Fusion Devices, Greifswald, Germany, 2002.

[25] S.R. In, B.J. Yoon, B.Y. Kim and T.S. Kim, "Ion beam optics in the bending magnet of the KSTAR NBI system," *J. Korean Phys. Soc.* **49**, S320 (2006).

[26] S.R. In, H.J. Shim, "Neutral beam evolution in the KSTAR NBI test stand", *J. Kor. Vac. Sci. Technol.* **7**, 27 (2003).

[27] Tae-Seong Kim, Jinchoon Kim, Sang Ryul In, Seung Ho Jeong, "Calculation of extracted ion beam particle distribution including within-extractor collisions from H-alpha Doppler shift measurements", *Rev. Sci. Instrum.* **79**, 704 (2008).

[28] Y. D. Bae et al., "Development of a KSTAR ICRF antenna for long pulse operation," *Nucl. Fusion*, **43**, 805(2003).

[29] J. G. Kwak et al., "Progress in the development of heating systems towards long pulse operation for KSTAR," *Nucl. Fusion*, **47**, 463(2007) .

[30] J.G. Kwak et al., "Achievement of 1.9 MW for 300s during the commissioning of the KSTAR ICRF transmitter," *Fusion Eng. Des.*, (2009) to be published.

[31] Y. S. Bae et al., "KSTAR ECH system; status and plan," *Fusion Sci. Tech.* **52**, 321 (2007).

[32] Y. S. Bae et al., "ECH pre-ionization and assisted startup in the fully superconducting KSTAR tokamak using second harmonic," *Nucl. Fusion*, **49**, 022001 (2009).

[33] ITER Physics Expert Group on Energetic Particles, Heating and Current Drive, "Chapter 6: Plasma auxiliary heating and current drive," *Nucl. Fusion* **39**, 2495 (1999).

[34] M. Kwon, I. S. Choi, J. W. Choi, J. S. Hong, M. C. Keum, et al, "The control system of KSTAR," *Fusion Eng. Des.*, **71**, 17 (2004).

[35] M.K. Park, et al., "Overview of KSTAR integrated control system," *Nucl. Eng. Technol.*, **40**, 451 (2008).

[36] S. H. Hahn, et al., "Plasma control system for "Day-one" operation of KSTAR tokamak," *Fusion Eng.Des.*, doi:10.1016/j.fusengdes.2008.12.082 (2009).

[37] Y. K. Oh, et al., "Commissioning and initial operation of KSTAR superconducting tokamak," *Fusion Eng. Des.*, doi:10.1016/j.fusengdes.2008.12.099 (2009).

[38] A.J.H. Donne, et. al., "Progress in the ITER Physics Basis: Chapter 7: Diagnostics," *Nucl. Fusion* **47**, 337 (2007).

[39] S.I. Pak, H.G. Lee, K.J. Jung, C.I. Walker, D.G. Kim, and K.S. Choi, "A conceptual design of the ITER upper port plug structure," *Fusion Eng. Des.*, **83**, 1594 (2008).

[40] M.S. Cheon, S. Pak, H.G. Lee, L. Bertalot, and C. Walker, "In-vessel design of ITER diagnostic neutron activation system," *Rev. Sci. Instrum.* **79**, 10E505 (2008).

[41] E. Bertolini, P.L. Mondino, P. Noll, "The JET magnet power supplies and plasma control systems," *Fusion Technol.* **11**, 71 (1987).

[42] ITER Final Design Report, July 2001

[43] H. Smolleck, et al., "Effects of pulsed-power loads upon an electric power grid," *IEEE Trans. Power Del.* **6**, 1305 (1991).

[44] P. Bettini, A. De Lorenzi, "400 kV-network harmonic impedance evaluation from harmonic distortion measurements," *ETEP*, **8**, 31 (1998).

[45] H. Akagi, "Large static converters for industry and utility applications," *Proc. IEEE* **89**, 976 (2001).

[46] G. Rostagni , "The electric power handling from present machines to fusion power stations," *Fusion Eng. Des.*, **74**, 87 (2005).

[47] J. H. Choi, "Overview of the superconducting magnet power supply system for the KSTAR 1st Plasma experiment," *Nucl. Eng.Technol.*, **40**, 459 (2008) .

[48] B.G. Hong et al., "Tokamak Reactor System Analysis Code for the Conceptual Development of DEMO Reactor," *Nucl. Eng. Technol.*.**40**, 1 (2008).

[49] B.G. Hong, D.W. Lee, S.K. Kim, Y. Kim, "Development of a tokamak reactor system code and its application for concept development of a demo reactor," *Fusion Eng. Des.*, **83**, 1615(2008).

[50] Y. Kim, B.G. Hong, C.H. Kim, "A neutronic investigation of He-cooled liquid Li-breeder blankets for fusion power reactor," *Fusion Eng. Des.*, **75-79**, 1067(2005).

[51] Duck-Hoi Kim, et. al., "Current status on the detailed design and development of fabrication techniques for the ITER blanket shield block in Korea," *Fusion Eng. Des.*, **83**, 1181 (2008).

[52] Duck Hoi KIM, et. al., "Welding technology development for the fabrication of ITER blanket shield block in KO," *Proc. Of the 18th Topical Meeting on the Technology of Fusion Energy*, Sep. 2008.

[53] Seungyon Cho, et. al., "Current status of design and analysis of Korean Helium-Cooled Solid Breeder Test Blanket Module," *Fusion Eng. Des.*, **83**, 1163 (2008).

[54] Mu-Young Ahn, et. al., "Preliminary safety analysis of Korea Helium Cooled Solid Breeder Test Blanket Module," *Fusion Eng. Des.*, **83**, 1753 (2008).

[55] Sunghwan Yun, et. al., "Depletion analysis of a solid-type blanket design for ITER," *Fusion Eng. Des.*, **83**, 1654 (2008).

[56] D. W. Lee et. al., "Current status and R&D plan on ITER TBMs of Korea," *J. Korean Phys. Soc.*, **49**, S340 (2006).

[57] S.J. Zinkle, M. Victoria, K. Abe, "Scientific and engineering advances from fusion materials R&D," *J. Nucl. Mater.* **307-311**, 31 (2002).

[58] E.E. Bloom, J.T. Busby, C.E. Duty, P.J. Maziasz, T.E. McGreevy, B.E. Nelson, B.A. Pint, P.F. Tortorelli, S.J. Zinkle, "Critical questions in materials science and engineering for successful development of fusion power," *J. Nucl. Mater.* **367-370**, 1 (2007).

[59] W.S. Ryu, et al., "Characterization and Improvement of Nuclear Materials," KAERI/RR-2793/2006 (2006).

[60] W.-J. Kim, S.M. Kang, J.Y. Park, W.S. Ryu, "Effect of SiC whisker formation on the densification of Tyranno SA/SiC composites fabricated by CVI process," *Fusion Eng. Des.*, **81**, 931-936 (2006) .

[61] J.Y. Park, S.M. Kang, W.J. Kim and A. Kohyama, "Characterization of MeV ion-irradiated SiC_f/SiC composites prepared with different methods," *Advanced Mater. Res.* **59**, 257 (2009).

[62] J.I. Kim, W.J. Kim, D.J. Choi, J.Y. Park, W.S. Ryu "Design of a C/SiC functionally graded coating for the oxidation protection of C/C composites," *Carbon*, **43**, 1749 (2005).

[63] J.Y. Park, C.H. Jung, H.K. Park, S.J. Oh, and Y.S. Kim, "Preparation of Li_2ZrO_3 Powders by the Combustion Process," *Ceram. Trans.*, **85**, 55 (1997).

[64] C.H. Jung, J.Y. Park, W.J. Kim, W.S. Ryu and S.J. Lee, "Characterizations of Li_2TiO_3 prepared by a solution combustion synthesis and fabrication of spherical particles by dry-rolling granulation process," *Fusion Eng. Des.* **81**, 1039 (2006) .

[65] M.H. Lee, C.-H. Jung and S.J. Lee, "Highly sinterable lithium titanate powders fabricated by an organic-inorganic solution route," *Solid State Phen.*, **807**, 124 (2007).

[66] J.-Y. Park, "Fabrication and destructive tests of ITER first wall qualification mock-ups," presentation in Korea-Japan Blanket Workshop, Feb. 26-27, 2009, Seoul, Korea.

[67] H.K. Seok, K.H. Jung, Y.C. Kim, J.H. Shim, D.I. Kim, S.H. Han, K.H. Baik, P.R. Cha, "Evaluation of W-Si-C thick coating as a plasma facing material," Accepted in *J. Nucl. Materials*.

[68] G.-G. Lee, J.H. Kwon and C.S. Shin, "An Overview of multi-scale modeling approach to radiation effects on materials," presentation in Korea-Japan Blanket Workshop, Feb. 26-27, 2009, Seoul, Korea.

[69] N.Y. Park, Y.C. Kim, H.K. Seok, S.H. Han, S. Cho, P.R. Cha, "Molecular dynamics simulation of irradiation damage in tungsten," *Nucl. Instru. Meth. Phys. Res.* **B 265**, 547 (2007).

[70] R. Coppola, M. Magnami, R. May and A. Moeslang, "He bubble growth in a martensitic steel for fusion reactors", *J. Appl. Cryst.*, **33**, 469-472 (2000).

[71] B.N. Kolbasov, "Safety aspects of fusion power plants," *Nucl. Fusion* **47**, editorial (2007).

[72] J.Ph. Girard, et. al., "Summary of the 8th IAEA technical meeting on fusion power plant safety," *Nucl. Fusion*, **48**, Conference Report, 1(2008).

[73] J.-Ph. Girard, et. al., "ITER, safety and licensing," *Fusion Eng. Des.*, **82**, 506 (2008).

[74] Nuclear Technology Roadmap (NuTRM) - Vision E, Korea Nuclear Society (2005)

[75] Fusion Development Roadmap 2007-2036, Ministry of Education, Science and Technology (2007)