The KSTAR Tokamak *

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Abstract--The KSTAR (Korea Superconducting Tokamak Advanced Research) project is the major effort of the Korean National Fusion Program to design, construct, and operate a steady-state-capable superconducting tokamak. The project is led by Korea Basic Science Institute and shared by national laboratories, universities, and industry along with international collaboration. It is in the conceptual design phase and aims for the first plasma by mid 2002. The key design features of KSTAR are: major radius 1.8 m, minor radius 0.5 m, toroidal field 3.5 T, plasma current 2 MA, and flexible plasma shaping (elongation 2.0; triangularity 0.8; double-null poloidal divertor). Both the toroidal and the poloidal field magnets are superconducting coils. The device is configured to be initially capable of 20s pulse operation and then to be upgraded for operation up to 300s with non-inductive current drive. The auxiliary heating and current drive system consists of neutral beam, ICRF, lower hybrid, and ECRF. Deuterium operation is planned with a full radiation shielding.

I. INTRODUCTION

A. Background

The Korean National Fusion Program, which was approved in December 1995, calls for the construction of a steady-state-capable superconducting tokamak, KSTAR (Korea Superconducting Tokamak Advanced Research) by the middle of 2002. The mission of the KSTAR project is to develop a steady-state-capable advanced superconducting tokamak to establish the scientific and technological bases for an attractive future energy source. Korea Basic Science Institute was designated as the lead organization and as the site of the tokamak device. Other national laboratories, universities, and industries involved in the project include Korea Atomic Energy Research Institute, Korea Advanced Institute of Science and Technology, Korea Research Institute of Standards and Science, Samsung, Hanjung, Pohang University of Science and Technology, Seoul National University, just to name the major partners. Conceptual design activity is being carried out in collaboration with Princeton Plasma Physics Laboratory, who coordinates a team of staff from GA, LLNL, MIT, ORNL and PPPL with previous experience in TPX [1] design work. The physics portion of the conceptual design review was completed in June 1997 as reviewed by representatives of world-wide fusion laboratories. The tokamak systems engineering review is scheduled for December 1997.

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B. Research Objectives of KSTAR Tokamak

To support the KSTAR mission stated above, the research objectives have been established as (i) to extend the present stability and performance boundaries of tokamak operation through active control of plasma profiles and transport; (ii) to explore methods to achieve steady-state operation for tokamak fusion reactors using non-inductive current drive; (iii) to combine the optimized plasma performance and the continuous operation capability as a step toward an attractive tokamak fusion reactor. The design of KSTAR tokamak features all superconducting magnets; long pulse operation capability (300 sec); flexible plasma shape/position control; flexible current and pressure profile control; advanced profile and control diagnostics.

The KSTAR program is staged into two phases: the baseline and the upgrade. In Phase I (2002 - 2005), the inductively driven 20-s operation with the baseline auxiliary system (15 MW) is to be conducted whereas the 300-s operation with the upgraded auxiliary system (40 MW) for full non-inductive current drive at high-beta regimes is planned for Phase II (2006 - 2010).

II. PHYSICS DESIGN AND GOALS

A. Physics Goals and Plasma Parameters

Profile control of the plasma current and pressure will be the primary tool for the advanced tokamak operating modes for KSTAR as learned from the existing devices, including DIII-D, JET, JT-60U, and TFTR. The goal of improving energy confinement and pressure limits led to the selection of a double-null reference plasma configuration (including the single-null capability) with strong cross-section shaping (high elongation and high triangularity) [2] for optimum plasma performance. Operation of such *advanced tokamak modes* under a steady state condition is the ultimate goal of KSTAR physics. The machine will be operable in either hydrogen or deuterium. The shielding and refrigeration systems are designed for a peak neutron rate of 3.5×10^{16} s^{-1}. The design-basis lifetime neutron yield is 1.9×10^{21}. Neither remote maintenance nor low-activation materials is necessary.

The major parameters of the initial KSTAR machine and auxiliary heating systems are summarized in Table I. The initial system is designed for a pulse length of 20 seconds, long enough to study advanced core, edge, and divertor physics on time scales well beyond energy and particle confinement times, but short enough to permit the use of bolted tiles and between-shot cooling of the divertor. Plasma performance can be increased by expanding the heating systems as shown in the Upgrade column of Table I and by installing actively cooled divertors. The pulse length of 300 s was chosen from the consideration of neutron budget limit and machine cost. On the other hand, the pulse length is long enough (as compared with the skin time and the particle wall recycling time) to allow the investigation of steady-state physics.

As described in the following section, the TF coils and the vacuum vessel are designed over-sized in contrast to the plasma size in order to reduce the toroidal field ripple and to maximize neutral beam access. Therefore, a close-fitting conducting structure is provided for passive stabilization of the vertical instability as well as of non-axisymmetric modes.

Table I
Major Parameters of KSTAR

Parameter	Baseline	Upgrade
Toroidal field, B_T (T)	3.5	
Plasma current, I_P (MA)	2.0	
Major radius, R_0 (m)	1.8	
Minor radius, a (m)	0.5	
Elongation, κ_x	2.0	
Triangularity, δ_x	0.8	
Poloidal divertor nulls	2	
Pulse length (s)	20	300
Plasma heating power (MW)		
Neutral beam	8	24
Ion cyclotron	6	12
Lower hybrid	1.5	4.5
Electron cyclotron	0.5	TBD
Peak DD neutron source rate (s^{-1})	3.5×10^{16}	
Annual deuterium operating time (s)	20,000	
Total number of pulses expected	50,000	

B. Operational Scenarios and Plasma Control

The KSTAR physics goal requires a wide operating space in terms of the internal inductance ($0.4 \le \ell_i \le 1.3$) and the normalized beta ($1.5 \le \beta_N \le 5$) [3]. Flexibility is desired for plasma startup scenario and the single-null capability. KSTAR reference discharges have been examined using TSC code [4] for various modes of operation, including high-β H-mode, high-ℓ_i mode, reversed shear mode, and high bootstrap Aries-I mode [5].

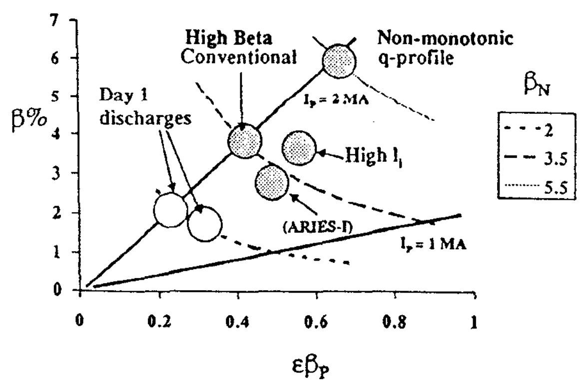

Fig.1. Various reference discharges

In Fig. 1, various standard reference discharges (R=1.8 m, a=0.5 m, κ=2.0, δ=0.8, B_T=3.5 T) are illustrated for varied heating power and plasma current (open circles for Phase I and shaded circles for Phase II). High-ℓ_i, reverse shear, and Aries-I modes seem quite suitable for steady-state operation.

Configuration of PF coils and the plasma is shown in Fig. 2. Fourteen PF coils are connected in an up-down symmetric set of seven circuits for DN discharges [6]. For shape and position control on a fast time scale (10-20 ms), two sets of normal copper coils are provided between the vacuum vessel and the passive stabilizer, while the superconducting poloidal field coils are used for slow control (~1 s). Although the PF system is capable of providing a flux swing of 14 volt-seconds (~12 V breakdown voltage), an electron cyclotron heating power of up to 0.5 MW is planned to assist the plasma startup [7] at a lower loop voltage (6 V).

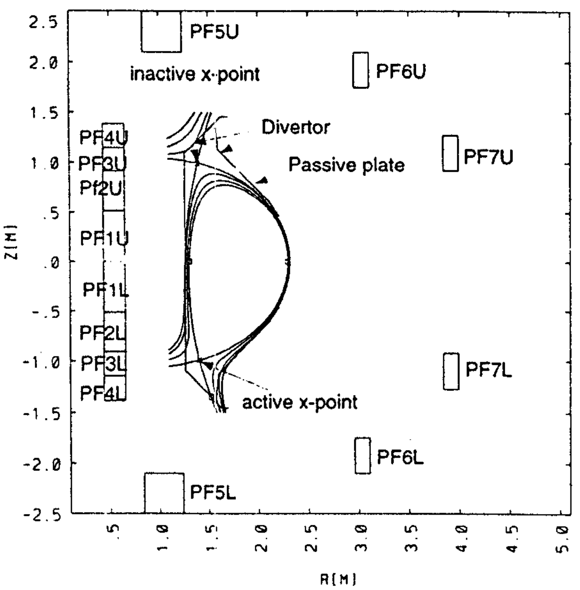

Fig.2. PF coil configuration in KSTAR

Control of low-order quasi-static field errors is necessary to avoid locked modes [8]. A system of 12 picture-frame-shape error field correction coils (superconducting) will be installed.

C. Plasma Diagnostics

To fulfill its mission, it is essential that KSTAR should have as complete a set of diagnostics as is now customary on devices such as DIII-D, JET, TFTR, and JT-60U. But the budget and scheduling limitations have led to a staged classification of diagnostics [9]. The basic diagnostics are needed from day one for tokamak operation, control, and close integration with in-vessel components. The baseline diagnostics should come on line next for confinement and heating performance, divertor performance, pulse length extension, and profile control. Emphasis is placed on profile diagnostics which are essential for the operation of *advanced tokamak modes*. Integration of profile measurements into control system is an important development area of future tokamak research and KSTAR will take a full advantage of its development as time goes on. A dedicated neutral beam system for diagnostics is being examined. A layout of horizontal ports for diagnostics is shown in Fig. 3.

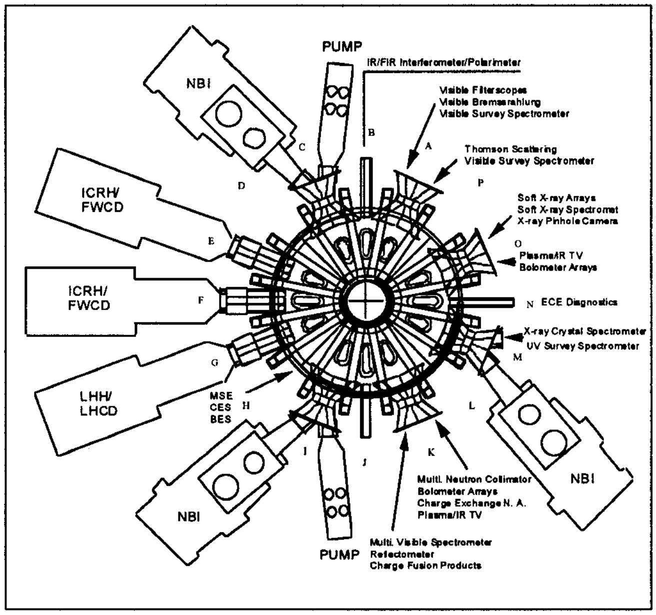

Fig.3. A layout of horizontal ports for diagnostics

III. ENGINEERING DESIGN OF MAIN COMPONENTS

A. Overall Machine Configuration [10]

An isometric view of the KSTAR tokamak is shown in Fig. 4. The magnet system consists of 16 superconducting toroidal field (TF) coils and 13 superconducting poloidal (PF) coils, symmetrically located about the plasma midplane. The seven inner PF coils form the central solenoid (CS) assembly. All the PF coils are located outside the TF coils. The vacuum vessel is a double walled structure located within the bore of the TF coils. A cryostat is employed as is in the ITER design. The cryostat and the vacuum vessel form a vacuum boundary for the superconducting TF and PF coils mitigating the convective heat loading to the cryogenic structure. The vacuum vessel is supported off the cryostat base by four reinforced vertical ports. All the ports and utility ducts penetrate through the cryostat via vacuum seal. The actual plasma is bounded by a close fitting plasma-facing components; inboard limiter, outboard passive stabilizer, and divertor plates both top and bottom as shown in Fig. 5.

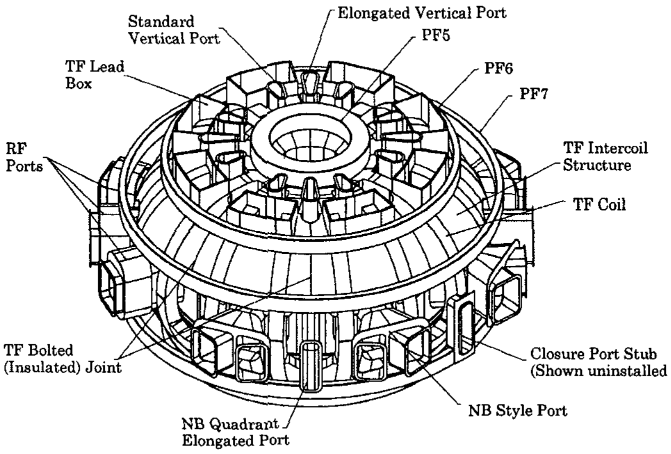

Fig.4. Isometric view of KSTAR structure

B. Superconducting Magnet System [11]

The TF coil is wound as 8 pancakes of 7 layers each in a D-shape , 4.2 m high and 3 m wide. The TF coils provide a field of 3.5 T at the plasma axis R=1.8 m, with a peak field at the coils of 7.5 T. All the coils are connected in series, and thus the total flux is 31.5 MA-Turn at the nominal current of 35.2 kA. Two coils are combined into a welded assembly and two such assemblies are to be threaded into a vacuum vessel quadrant. Toroidal field ripple at the plasma edge is maintained quite low (< 0.13% at r/a=1) by using 16 large-bore D-shape coils.

All the superconducting conductors are cable-in-conduit conductors (CICC). The Nb_3Sn strand selected for KSTAR is the ITER HP-I type [12], taking advantage of the fact that it has been thoroughly tested under the ITER development program. The large-bore PF6 and PF7, on the other hand, uses NbTi, which does not require such an elaborate heat treatment needed for Nb_3Sn conductors. The conduit materials are Incoloy 908 and/or SS316LN. Conductor details are given in Table II.

Table II. TF and PF Conductor Parameters

Parameter	Units	TF	PF1-5	PF6-7
Conductor		Nb_3Sn	Nb_3Sn	NbTi
Cu/Non-Cu		1.5:1	1.5:1	3.5:1
$A_{conduit}$	(mm^2)	233	179.2	179.2
D_{strand}	(mm)	0.81	0.81	0.81
n_{strand}		486	360	360
$n_{cu\text{-}strand}$		162	120	120
L_{cable}	(km)	9.86	6.78	7.33
$M_{sc\text{-}strand}$	(tons)	13.4	6.84	7.5

The TF case is 316LN, while PF coils do not have cases minimizing the eddy current losses. The TF and PF coils are cooled by supercritical helium that is supplied at a pressure of 5 atmospheres. The TF assembly is mounted to the tokamak support structure through the cryostat base structure and the PF 5-7 are bolted to the TF cases through mounts permitting relative radial motion. The CS is hung off the TF structure and is allowed to move radially. Both internally co-wound wire sensors and fiber optic temperature sensors are employed for reliable quench detection.

C. Vacuum Vessel [13]

KSTAR vacuum vessel consists of two shells separated and held together by a number of poloidal ribs. Toroidal rings above and below the horizontal ports provide additional structural rigidity. The shell material is SS316 LN. The space between the shells varies along the poloidal direction, about 5 cm at inboard where space is very precious and 30 cm at the expansive outboard. Filled with low-flow water, the vessel wall acts as both neutron shielding and vessel cooling. For vessel baking at 350 °C, the water is drained from the wall and replaced with hot nitrogen gas flow (400 °C).

As shown in Fig. 5, the wall is composed of inboard and outboard cylinders, two conic sections, and two circular forms, thus providing the ease of fabrication as compared to a free form vessel. The vessel is toroidally divided into four quadrants which will be weld-closed at assembly. Three of the quadrants are identical, featuring two NBI-type ports with a small port in between. The fourth quadrant, utilized for RF launchers, has three identical rectangular horizontal ports. Vertical ports are located at the top and bottom of the vacuum vessel (12 top-12 bottom) with centerlines aligned with the horizontal ports. The lower central vertical port in each quadrant is used to support vertical loads on the vacuum vessel.

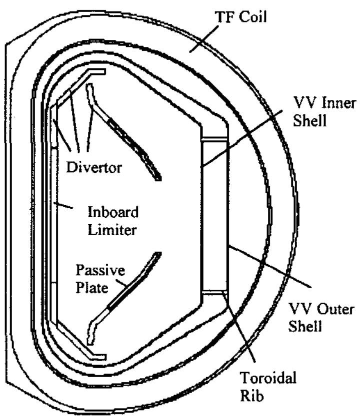

Fig.5. Poloidal cross section of KSTAR showing the plasma facing components and the vacuum vessel

D. Plasma Facing Components and Divertors [14]

The plasma facing components consist of the inboard limiter, the upper and lower outer passive stabilizers, and the upper and lower divertors. The plasma side is covered with bolted graphite tiles (carbon fiber composite tiles for the divertor plates). To protect from heat and particle loading, a bumper limiter frame is provided around the RF launchers, which are located in the same quadrant of the vacuum vessel. The inboard limiter and the passive stabilizer assemblies are designed to handle the full heating power (40 MW) for the maximum pulse length (300s). Because of a high heat flux anticipated for KSTAR, however, the divertors are designed to be inertially cooled for the baseline 20s operation. Simulation analyses using both UEDGE code [15] and B2 code [16] show that the peak power density is below the engineering limit of 3.5 MW/m^2. For the 300s operation, high conductivity CFC tiles will be brazed onto the backing plate rather than bolted to.

Particle exhaust will be maintained by two 50,000 l/sec LHe-cooled cryopumps, one each in the upper and lower divertor plenum in addition to the nominal base pumping by turbomolecular pumps [17]. These cryopumps will exhaust neutrals produced by recycling at the divertor strike points. An internal baffle will limit gas leakage from this pumping plenum. Optimum location of the cryopump and the baffle is being studied by using DEGAS code [18]. The overall design of the cryopump ring will be fashioned after the DIII-D cryopump.

The effectiveness of the close-fitting passive stabilizer plate (high conductivity, dispersion strengthened copper alloy) has been examined extensively in terms of the stability factor, plasma startup interference, and voltage breakdown. The current design is that each stabilizer will have one toroidal break and the upper and lower stabilizers are connected in a saddle configuration similar to ASDEX-U [19]. The gap between the stabilizer and the last closed flux surface is set at about 4 cm through trade-off between the stability factor and the safety margin for the plasma wall loading.

IV. HEATING AND CURRENT DRIVE SYSTEMS

A. Neutral Beam Injection system

Among available heating methods in tokamak plasmas, neutral beam injection (NBI) has been the most successful and reliable method. And naturally the NBI will be the primary heating and current drive (H&CD) system for KSTAR. Three beamlines are allocated for KSTAR NBI system. The major deviation from the existing tokamak NBI systems is the capability of steady-state operation. The availability of steady-state ion source is of concern, and the closest to this application would be the US Common Long Pulse Source (12x48 cm^2 extraction geometry) [20] in use for TFTR and DIII-D. Therefore the reference system was modeled after the TFTR beamline, which houses three ion sources side by side with the long dimension aligned vertically conforming to the vertically elongated opening of the torus. The beam energy of 120 keV has been chosen for better central beam fueling at higher operating density. The first beamline in the same direction to the plasma current can produce about 8 MW of injected neutral power.

Using the same beam characteristics, an idea of *localized* NBI in the flux surface coordinate has been investigated. It can be done by rotating the ion sources by 90° (i.e., the long dimension of the beam is now horizontally aligned) and by aiming off median with a tilt angle but still in co-direction. Analysis of such NBI variation (on axis, off-axis, more off-axis injection with 6 MW each) using TRANSP code [21] is shown in Fig. 6. The localization effect is remarkable. In Fig. 6(a), the beam driven plasma current profiles show a distinct profiles, implying the feasibility of driving the off-axis plasma current. Beam fueling profile is also controllable as shown in Fig. 6(b). It is also possible to have an extremely peaked beam fueling profile (H_{ne}~6 where H is the peakedness parameter), which is a benefit according to a scaling law based on beam fueling profile [22]. In view of the above result, the rest of the beamlines (beyond the baseline phase) could be designed with this alignment albeit the penalty of putting the long dimension of the beam through the narrow dimension of the duct.

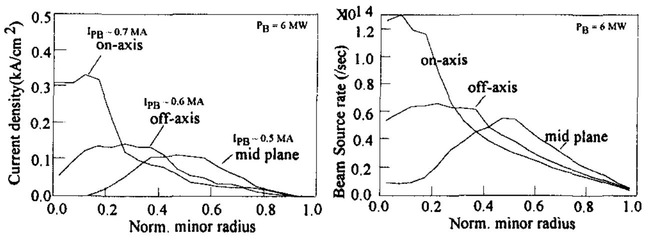

Fig.6. (a) Calculated beam driven plasma current profiles for different aiming; (b) Calculated beam fueling profiles for different aiming.

B. ICRF Heating and Current Drive

Heating and current drive by ICRF fast waves are one of the main features in the operation of reversed shear plasmas on KSTAR. The ICH/FWCD system shall initially be configured to provide 6 MW of ICH/FWCD power to the plasma with one antenna. The system can be upgraded to 12 MW of power to the plasma with the addition of a second system.

Three operating scenarios for heating and current drive have been studied using the global wave code TORIC [23]. *H minority heating in D majority plasma* requires 50 MHz at 3.5 T. Over 80% of total absorbed power is going to the plasma ions through the fundamental cyclotron resonance of H and the second harmonic resonance of D. *Current drive in D majority plasma with H minority* requires 38 MHz. Operation at 75 MHz may be possible for high frequency FWCD in H-only plasma. To cover these operating frequencies, RF transmitters must be operated over the 30 to 80 MHz frequency range. Numerical calculations of ICRF heating and current drive scenarios have been performed [24].

The ICH/FWCD launcher consists of a large antenna mounted through a standard port of KSTAR vacuum vessel. This antenna consists of four current straps, each of which is grounded at the center, and has a coaxial feed line connected to each end of the current strap. The portion of the current strap that couples power to plasma is 65 cm long, and is located 1 cm from the back surface of the Faraday shield. Each strap is 9 cm-wide and 2 cm-thick and dimensions are chosen to maximize the loading resistance.

C. Lower Hybrid and Electron Cyclotron

The lower hybrid frequency range (LH) has been experimentally shown to be quite effective in driving non-inductive plasma current. The LH driven discharge length has been extended up to ~120 sec with relatively high plasma density on Tore Supra [25]. LH is required to provide off-axis current profile control for KSTAR, initially 1.5 MW and upgradable to 4.5 MW. The selection of the frequency at 3.7 GHz is justified in terms of the density limit, above which damping is mostly on ions drastically reducing the current drive efficiency, and also by the fact that high power Klystrons are available at that frequency.

Initially the electron cyclotron frequency system (EC) will have a minimal power (0.5 MW at 80 GHz) to assist the plasma initiation in KSTAR to allow a low voltage startup (6 V). The upgrade route for the EC system and the experimental scenario have not been fully examined at this writing.

VI. CONCLUSIONS

An interim report of the physics and engineering design features of KSTAR tokamak is presented here while the design activity is still on-going. The plasma control and exhaust capabilities needed for exploration of *advanced* performance scenarios are provided, as are the flexibility and diagnostics required for a good physics experiment. It is hoped that the steady-state tokamak design based on all superconducting magnets would make KSTAR a premier facility for the development of continuous, high-performance modes of tokamak operation in the next decade, and the KSTAR program welcomes world-wide collaboration.

Although the heat load, ignition, and nuclear issues of the future fusion reactors (e.g., ITER) are not addressed in the KSTAR program, the physics result and the plasma control aspect of steady-state operation from KSTAR can be of a direct benefit to the future steady-state tokamak research and operation (e.g., ITER), in particular, the operation of the superconducting CS. It is also anticipated that the advanced physics results from KSTAR can be cross-compared at about the same time with the superconducting stellarators (LHD and Wendelstein-7X) now under construction as well as new spherical tokamak experiments (NSTX and MAST).

It is also worth pointing out that the KSTAR program will also significantly enhance scientific and technological capabilities in Korea. Advances in infrastructure technologies will result from large-scale superconducting magnet design and manufacturing, large-scale high-vacuum vessel fabrication, technology involved in high power neutral beam and microwave generation, state-of-the-art plasma diagnostics and control, and sophisticated computational methods.

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