The KSTAR project: An advanced steady state superconducting tokamak experiment

G.S. Lee, J. Kim, S.M. Hwang, C.S. Chang[a], H.Y. Chang[a], M.H. Cho[b], B.H. Choi[c], K. Kim[d], K.W. Cho, S. Cho, K.K. Choh, C.H. Choi, J.H. Choi, J.W. Choi, I.S. Choi, C.J. Do, T.H. Ha, J.H. Han, J.S. Hong, K.H. Hong, N.I. Hur, I.S. Hwang, K.H. Im, H.G. Jhang, Y.S. Jung, B.C. Kim, D.L. Kim, G.H. Kim, H.S. Kim, J.S. Kim, J.Y. Kim, W.C. Kim, Y.S. Kim, K.H. Kwon, M.C. Kyum, B.J. Lee, D.K. Lee, H.G. Lee, J.M. Lee, S.G. Lee, H.G. Na, Y.K. Oh, J.H. Park, H.C. Ri, Y.S. Ryoo, K.Y. Song, H.L. Yang, J.G. Yang, B.J. Yoo, S.J. Yoo, N.S. Yoon, S.B. Yoon, G.H. You, K.I. You, W. Choe[a], D.-I. Choi[a], S.G. Jeong[a], D.Y. Lee[a], Y.S. Bae[b], H.S. Kang[b], G.N. Kim[b], I.S. Ko[b], W. Namkung[b], J.S. Oh[b], Y.D. Bae[c], Y.S. Cho[c], B.G. Hong[c], G. Hong[c], C.K. Hwang[c], S.R. In[c], M.H. Ju[c], H.J. Lee[c], B.H. Oh[c], B.J. Yoon[c], S. Baang[d], H.J. Choi[d], J. Hwang[d], M.G. Kim[d], Y.J. Kim[d], S.I. Lee[d], J. Yee[d], C.S. Yoon[d], K.-H. Chung[e], S.H. Hong[e], Y.S. Hwang[e], S.H. Kim[e], Y.H. Kim[e], K.H. Chung[f], J.Y. Lim[f], D.W. Ha[g], S.S. Oh[g], K.S. Ryu[g], Q.L. Wang[g], T.K. Ko[h], J. Joo[i], S. Suh[i], C.H. Choi[j], J.H. Lee[j], Y.W. Lee[j], H.S. Shin[j], I.H. Song[j], J. Baek[k], I.Y. Han[k], Y. Koh[k], P.Y. Park[l], C. Ryu[m], J.J. Cho[n], D.M. Hwang[n], Y.S. Kim[n], J.A. Schmidt[o], H.K. Park[o], G.H. Neilson[o], W.T. Reiersen[o], R.T. Simmons[o], S. Bernabei[o], F. Dahlgren[o], L.R. Grisham[o], S.C. Jardin[o], C.E. Kessel[o], J. Manickam[o], S.S. Medley[o], N. Pomphrey[o], J.C. Sinnis[o], T.G. Brown[o], R.B. White[o], K.M. Young[o], J. Schultz[p], P.W. Wang[p], L. Sevier[q], M.D. Carter[r], P.M. Ryan[r], D.W. Swain[r], D.N. Hill[s], W.M. Nevins[s], B.J. Braams[t]

National Fusion R&D Center,
Korea Basic Science Institute,
Taejeon, Republic of Korea

[a] Korea Advanced Institute of Science and Technology, Taejeon, Republic of Korea
[b] Pohang University of Science and Technology, Pohang, Republic of Korea
[c] Korea Atomic Energy Research Institute, Taejeon, Republic of Korea
[d] Samsung Advanced Institute of Technology, Taejeon, Republic of Korea
[e] Seoul National University, Seoul, Republic of Korea
[f] Korea Research Institute of Standards and Science, Taejeon, Republic of Korea
[g] Korea Electrotechnology Research Institute, Changwon, Republic of Korea
[h] Yonsei University, Seoul, Republic of Korea
[i] Sungkyunkoan University, Seoul, Republic of Korea
[j] POSCON Corp., Pohang, Republic of Korea
[k] Dae-Sung Cable Co., Seoul, Republic of Korea
[l] Korea Iron Steel Wire Ltd., Seoul, Republic of Korea
[m] CVE Corp., Seoul, Republic of Korea
[n] Samsung Aerospace Corp., Changwon, Republic of Korea
[o] Princeton Plasma Physics Laboratory, Princeton University, Princeton, New Jersey, USA
[p] Massachusetts Institute of Technology, Cambridge, Massachusetts, USA
[q] General Atomics, San Diego, California, USA
[r] Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA
[s] Lawrence Livermore National Laboratory, Livermore, California, USA
[t] New York University, New York, N.Y., USA

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Abstract. The Korea Superconducting Tokamak Advanced Research (KSTAR) project is the major effort of the national fusion programme of the Republic of Korea. Its aim is to develop a steady state capable advanced superconducting tokamak to establish a scientific and technological basis for an attractive fusion reactor. The major parameters of the tokamak are: major radius 1.8 m, minor radius 0.5 m, toroidal field 3.5 T and plasma current 2 MA, with a strongly shaped plasma cross-section and double null divertor. The initial pulse length provided by the poloidal magnet system is 20 s, but the pulse length can be increased to 300 s through non-inductive current drive. The plasma heating and current drive system consists of neutral beams, ion cyclotron waves, lower hybrid waves and electron cyclotron waves for flexible profile control in advanced tokamak operating modes. A comprehensive set of diagnostics is planned for plasma control, performance evaluation and physics understanding. The project has completed its conceptual design and moved to the engineering design and construction phase. The target date for the first plasma is 2002.

1. Introduction

1.1. Project mission

The mission of the Korea Superconducting Tokamak Advanced Research (KSTAR) project, the major effort of the national fusion programme of the Republic of Korea, is to develop a steady state capable advanced superconducting tokamak to establish a scientific and technological basis for an attractive fusion reactor [1]. To support this project mission, three major research objectives have been established: (i) to extend the present stability and performance boundaries of tokamak operation through active control of profiles and transport, (ii) to explore methods to achieve steady state operation for tokamak fusion reactors using non-inductive current drive, and (iii) to integrate optimized plasma performance and continuous operation as a step towards an attractive tokamak fusion reactor.

1.2. Design features

To meet the mission and research objectives of KSTAR, key design features have been established:

- Fully superconducting magnets,
- Long pulse operation capability with non-inductive current drive,
- Flexible pressure and current profile control,
- Flexible plasma shape and position control,
- Advanced diagnostics.

The KSTAR tokamak and its ancillary systems are designed for long pulse operation. Global current relaxation times are estimated to be in the range 20–60 s. The complicated nature of plasma–wall interactions makes it difficult to estimate plasma–wall timescales, but ∼100 s is a reasonable estimate based on constant energy flow to the walls and static plasma positioning. Considering practical engineering constraints, system cost and conventional facility requirements, the KSTAR facility is designed for a pulse length of 300 s. However, since initial operation will focus on advanced tokamak physics study which does not require long pulses, the initial configuration will provide a pulse length of 20 s driven by the poloidal magnet system. To develop steady state, high performance plasma operating scenarios, capable plasma control tools are required. The KSTAR facility will have a plasma heating system that will heat the plasma to high temperature and high β, drive the current non-inductively, and control current and pressure profiles. Many technologies are used to meet these requirements: neutral beams, ion cyclotron waves, electron cyclotron waves and lower hybrid waves.

Since high elongation and triangularity in plasma cross-section shaping are important for improving performance and stability limits, the poloidal coils and divertor are based on a strongly shaped, double null divertor plasma configuration. Flexibility is provided to explore a wide range of pressures (β_N) and current profile shapes (ℓ_i) in double null as well as single null plasmas. To control the MHD behaviour of high β plasmas, in-vessel conducting structures are provided for passive stabilization, as well as two sets of in-vessel coils for active position control and ex-vessel superconducting modular coils for field error correction. An advanced diagnostic system will be employed to measure current and pressure profile variations and to assess performance and stability.

1.3. Major parameters

The major parameters of the initial KSTAR tokamak and auxiliary heating systems are summarized in the 'Baseline' column of Table 1. The machine will be operable with either hydrogen or deuterium, but

Table 1. Major parameters of the KSTAR device

	Baseline	Upgrade	Extended option
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	
Heating power (MW)			$\leq$27.5
Neutral beam	8	14	
Ion cyclotron	6	6	
Lower hybrid	1.5	1.5	
Electron cyclotron	0.5	0.5	ECCD
Peak DD neutron source rate (s^{-1})	3.5×10^{16}		
Annual deuterium operating time (s)	20 000		
Number of pulses	50 000		

deuterium operation time will be limited to allow personnel access to the SS-316LN based vacuum vessel interior after reasonable cooldown periods.

Extending the pulse length to 300 s requires replacing the initial inertially cooled divertor structures with an actively cooled system. The divertors can also be modified to test new configurations or materials. Plasma performance can be increased by expanding the heating systems to the ratings shown in the 'Upgrade' column of Table 1. In addition, it is expected that the diagnostic complement will be expanded throughout the operating life of the experiment. The device and facility have been designed with sufficient port access to simultaneously accommodate the upgrade heating systems and a comprehensive diagnostic set, as well as a cooling water supply passage for the upgrade. Although the detailed descriptions are not shown in the 'Extended option' column of Table 1, options exist to add more power up to 27.5 MW in the future, if it becomes advantageous to do so. Such extended heating options, to which there is currently no programmatic commitment, would require a rearrangement of the diagnostic system. For example, one more ion cyclotron resonance heating (ICRH) system could be added or the lower hybrid (LH) system could be expanded or changed to a higher frequency, or a counter-injected neutral beam heating system could be installed during an advanced operation phase reflecting experimental outcomes and physics issues. Although the poloidal field (PF) system is capable of providing a flux swing of 14 V·s, an electron cyclotron heating (ECH) power of 0.5 MW at 84 GHz will be installed to assist the plasma initiation in the KSTAR tokamak to allow a low voltage startup at 6 V. The upgrade route for the electron cyclotron heating and current drive (ECCD) system will also be considered as an 'Extended' heating option.

2. Physics design

The key plasma performance measures are the plasma beta, β, poloidal beta, β_p, and energy confinement time τ_E. Steady state reactor considerations motivate tokamak designs with β values of 5% or more (for high power density) and with most of the toroidal current driven by the bootstrap effect (to reduce recirculating power). Tokamak plasma configurations with high bootstrap fraction f_{bs} have $\varepsilon\beta_p$ values approaching unity (where ε is the ratio of minor to major radius). In the $\varepsilon\beta_p$–β plane shown in Fig. 1, the KSTAR operating space is bounded by a line corresponding to its design current (I_p = 2 MA) and minimum q_{95} (3.7), and a curve of constant $\varepsilon\beta_p \times \beta$, which is proportional to $(\beta_N)^2$ where $\beta_N = \beta/(I_p/aB)$. Several candidate operating modes are identified. Increasing β_N expands the region available for reliable high β operation. Moving along a constant β_N curve away from the I_p = 2 MA boundary increases the bootstrap fraction and the margin against low q disruptions but reduces the current. Since τ_E scales approximately in

Table 2. KSTAR AT operating modes

	Day 1 SS	Day 1 H mode	High β H mode	ARIES-I	Reversed shear	High ℓ_i
Plasma current I_p (MA)	1.5	2.0	2.0	1.4	2.0	1.5
Heating power P_{heat} (MW)	15.5	15.5	27.5	27.5	27.5	27.5
$\varepsilon\beta_p$	0.3	0.23	0.35	0.54	0.57	0.58
β (%)	1.7	2.3	3.5	2.6	5.7	3.25
β_N (%-m-T/MA)	2.0	2.0	3.1	3.3	5.0	3.8
q_{95}	5.0	3.7	3.7	5.3	3.7	5.0
I_{bs}/I_p (f_{bs})	0.31	0.23	0.30	0.66	0.88	0.59
H_{93H}	0.84	0.86	1.06	1.13	1.79	1.25
τ_E (ms)	98	150	119	85	210	95
$T_e(0)/T_i(0)$ (keV)	7.9/9.1	5.0/5.3	9.8/12.8	5.5/7.1	10.2/13.1	9.1/14.9
$n_e(0)$ (10^{20} m^{-3})	0.7	1.7	1.1	1.6	2.2	1.3

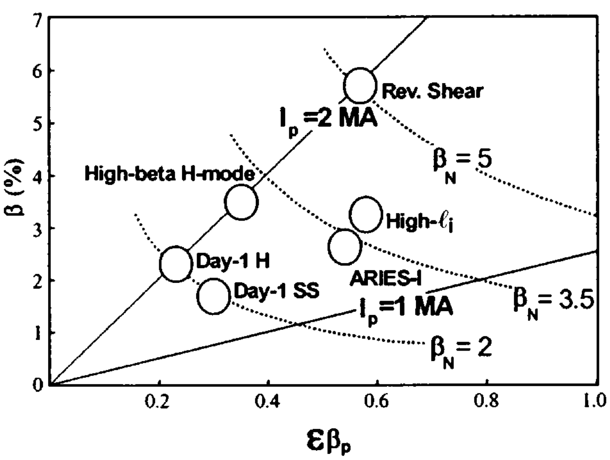

Figure 1. Advanced tokamak operating diagram of KSTAR showing several candidate operating modes (circles), and contours of constant current (solid lines) and constant β_N (broken curves).

proportion to the plasma current in tokamak scalings, e.g. $\tau_{ITER\text{-}93H}$, increasing the confinement enhancement factor $H_{93H} = \tau_E/\tau_{ITER\text{-}93H}$ is thus beneficial for plasma reliability as well as steady state operation.

The KSTAR tokamak is designed to have sufficient size, magnet performance and flexibility to accommodate a range of advanced tokamak (AT) operating modes. Its plasma dimensions are similar to those used in DIII-D AT experiments, while its higher toroidal magnetic field strength ($B_t = 3.5$ T versus 2 T) permits greater flexibility. Most importantly, KSTAR is engineered for long pulse lengths (20–300 s) to facilitate a significant extension of the duration of AT modes beyond that available in existing AT facilities. The JET DT experiment has achieved an AT mode combining an internal transport barrier (ITB) with an ELMy H mode to achieve $\beta_N \approx 1.5$ for a pulse length of ~2 s [2]. The DIII-D experiment has recently achieved a regime similar to reversed shear with $\beta_N > 3.5$ and roughly twice H mode confinement for ~1 s [3]. Continued improvements in AT performance and duration are expected from today's facilities, but a facility like KSTAR is needed to extend pulse lengths beyond 10 s and towards steady state operation. Parameters for the operating modes identified in Fig. 1 are tabulated in Table 2.

The KSTAR magnetic control system is designed to meet requirements for AT research in long pulse regimes. The main components are the superconducting PF coil system, field error correction coils, internal control coils and internal passive stabilizers.

An inductively driven reference discharge scenario with full current ($I_p = 2$ MA) and toroidal field ($B_t = 3.5$ T) and a 20 s flat-top interval will be available, assuming the plasma is maintained at $\beta_N = 3.5$ by auxiliary heating. Plasma initiation is accomplished with a 6 V inductive breakdown, assisted by ECH and by the use of the internal control coils to create a large field null. These capabilities of the basic KSTAR tokamak system provide a robust, reliable long pulse plasma scenario to serve as a platform for AT research.

The device also has the flexibility to accommodate long pulse (up to 300 s) AT operation over a range of pressures and profile shapes, bounded by the rectangle in Fig. 2, centred on the reference scenario flat-top configuration. The device can operate for full length pulses with β_N up to 5, but only

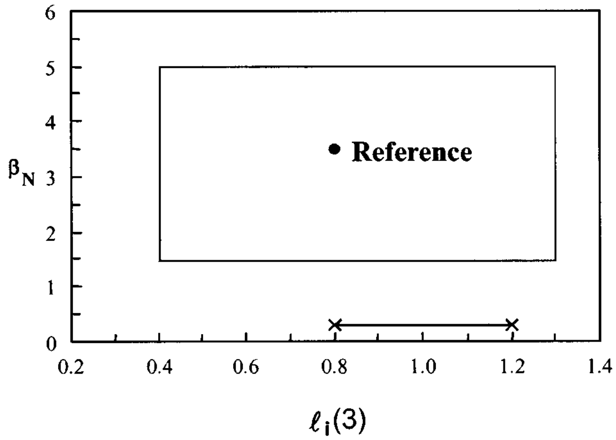

Figure 2. Steady state operating space for KSTAR at maximum B_t and I_p. The 'Reference' point denotes the flat-top condition for the reference discharge scenario. The line (×–×) denotes the available flexibility in ℓ_i under ohmic heating conditions at the end of the current ramp-up to allow for a range of inductive startup scenarios.

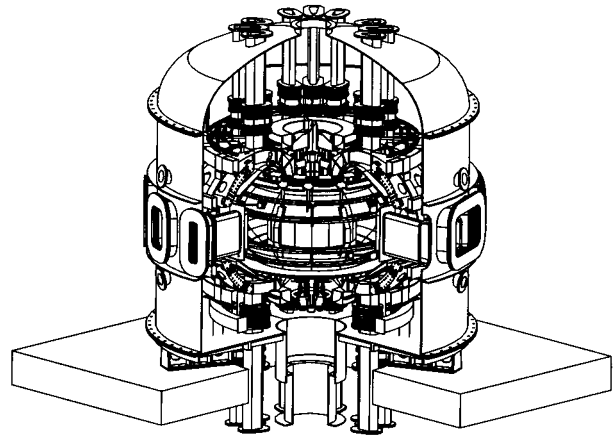

Figure 3. Overall machine configuration of KSTAR.

short pulse operation is available for $\beta_N < 1.5$. The available range of ℓ_i (0.4–1.3) is designed to accommodate current profile shapes ranging from those of reversed shear to high ℓ_i tokamak modes. Access to the interior of the operating rectangle is via inductive current drive (allowing for a range of possible ℓ_i at the end of the ohmic heating phase), while long pulse sustainment at any ℓ_i–β_N operating point requires auxiliary current drive and profile control.

The KSTAR plasma is surrounded by plasma facing structures that support power and particle exhaust (limiters, pumped divertors), stability (passive stabilizer plates) and heating (radiofrequency (RF) wave launchers). Since the effectiveness of these structures depends on the distance between them and critical points on the plasma boundary, controlling these distances is a prerequisite for overall plasma control. The PF and internal control coil systems are designed to control these interfaces within ∼1 cm on slow (∼1 s) and fast (∼10–20 ms) timescales. The double null divertor configuration facilitates the study of strongly shaped plasmas. The PF coil system provides the flexibility to produce both single and double null configurations over the full ℓ_i–β_N operating space. Since magnetic islands due to resonant field error harmonics are detrimental to plasma performance, a system of field error correction coils is available to control low order field error harmonics due to coil imperfections, leads and installation errors.

The KSTAR power handling system is designed to accommodate the initial heating power and pulse length (16 MW for 20 s) requirements. The divertor is designed to handle both high recycling and radiative divertor operating scenarios, providing the flexibility needed to study and optimize divertor performance and establish edge conditions compatible with high AT core performance. All power handling surfaces (divertor targets, limiters, local neutral beam and ripple loss armour) are faced with carbon tiles, with expected peak fluxes (in the divertor) of 3.5 MW/m^2. The power handling system can be upgraded (up to 27.5 MW for 300 s) as the needs of the research programme evolve.

3. Engineering design

3.1. Vacuum systems

A cutaway view of the overall machine configuration of KSTAR is shown in Fig. 3. The in-vessel hardware is built around the reference plasma equilibrium, as shown in Fig. 4, with geometrical parameters specified in Table 1. Nominal scrape-off zones (2 cm wide on the outboard side and 4 cm on the inboard side) and divertor X point to target distance (0.38 m) in the poloidal plane limit the placement of the surrounding hardware. The dimensions are chosen so as to facilitate unobstructed transport of heat and particles to the divertor.

The plasma is surrounded by plasma facing components and RF wave launchers. The inner limiter is a cylinder about 1.4 m tall, with its surface tangent to the inner scrape-off boundary. The divertor

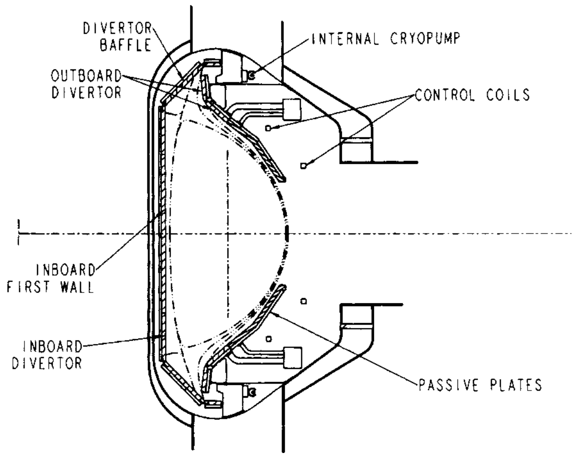

Figure 4. Cross-sectional view of KSTAR.

structures each consist of three high heat flux targets — inner, central and outer. The inner target surface is geometrically an extension of the inner limiter. The centre target surface and the divertor separatrix intersect each other at an angle of 25°. This design reduces the peak heat fluxes onto the target surface, while at the same time permitting a simple target geometry. A gap between the centre and the outer target is provided for particle exhaust into the pumping plenum. The outer target is contoured to form a narrow vee near the strike point and intersect the scrape-off plasma on a vertical surface to restrict neutral particle reflux back to the core plasma. Particle exhaust will be maintained by two liquid helium cooled cryopumps, one each in the upper and lower divertor plenums.

Toroidal limiters are placed outboard of the plasma, one above and one below the midplane. These are attached to the passive stabilizers and are designed to absorb plasma losses due to startup, radiation and possible neutral beam shine-through. Their placement relative to the plasma is determined by a trade-off between vertical stabilization, which favours placement of the passive structure as close to the plasma as possible, and impurity control, which favours moving the toroidal limiters further outward to reduce sputtering.

A 0.88 m high vertical opening between the outboard limiters provides access for plasma heating and diagnostics. The height is set by neutral beam access requirements. Two poloidal limiters conforming to the +2 cm surface, along with bumper segments attached to the toroidal limiters, form a 'picture frame' around the IC and LH wave launchers to provide an optimum environment for their operation. Both launchers can be placed with their front surfaces as close to the plasma as the +2 cm surface but can be moved further back if desired.

The vacuum vessel is a double walled structure located within the bore of the toroidal field (TF) coils. The vacuum vessel consists of four quadrants, which are field welded together at assembly. The vessel material is SS-316LN. Ribs serve to attach the inner and outer walls as well as provide rigidity. Toroidal rings above and below the horizontal ports provide additional structural rigidity.

Each vacuum vessel quadrant features three horizontal ports for heating and current drive systems, diagnostics, vacuum pumping and maintenance access. Three of the quadrants are identical, featuring two neutral beam style ports with a small port in between. The fourth quadrant, utilized for RF launchers, features three identical rectangular horizontal ports. Vertical ports are provided for additional diagnostic access and support of the vacuum vessel. Additional small horizontal ports are provided for routing cooling lines to the divertors.

Vertical ports are located at the top and bottom of the vacuum vessel with centrelines aligned with the horizontal ports. Vertical ports are not provided on the planes where the quadrants are joined together. The lower central vertical port in each quadrant is used to support vertical loads on the vacuum vessel. It is mounted to the cryostat base with a sliding support to accommodate temperature differences between the vacuum vessel and cryostat. The other vertical ports provide access for diagnostic viewing.

During operation the space between the vacuum vessel inner and outer shells is filled with water. The water is borated with enriched boric acid and thermalizes and absorbs fusion neutrons produced during deuterium operation. For bake-out, the water is drained from the wall interspace and replaced with hot nitrogen gas at 350°C to heat the vessel walls and internal components.

The cryostat is a single walled vacuum vessel consisting of a central cylindrical section and two end closures. The cylindrical section is an 8.5 m nominal internal diameter cylinder reinforced with external ribs. The cryostat provides the necessary thermal barrier between the ambient temperature of the experimental hall and the cryogenically cooled magnets.

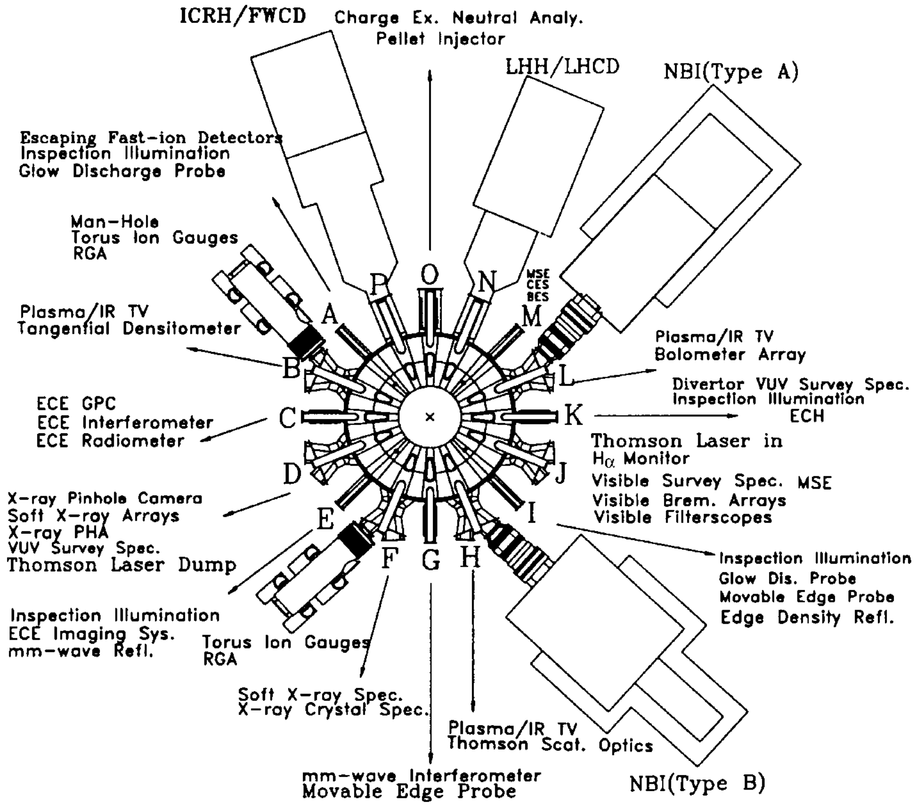

Figure 5. Layout of the KSTAR heating and diagnostic systems. GPC: grating polychromator; PHA: pulse height analyser; RGA: residual gas analyser.

3.2. Superconducting magnet systems

A toroidal array of 16 TF coils produces the 3.5 T toroidal field at the nominal plasma centre. The design requirements of low ripple and tangential neutral beam access have been leading factors in determining the size of the TF coils. The TF system is designed such that the inward magnetic forces on the inner legs of the TF coils are reacted by wedging of the inner legs. The TF coils are assembled in four-coil quadrants. The factory assembled four-coil quadrant reduces the final installation process at the nose of the coil to four (90°) wedged interface surfaces.

The TF and PF conductors are internally cooled, cable-in-conduit superconductors. The conductor for the TF coils and PF1 through PF5 is Nb_3Sn, whereas the PF6 and PF7 conductor is NbTi. One strand in each triplet of the first subcable is pure copper. The Nb_3Sn strand selected for the TF and PF coils is HP-III strand based on ITER superconducting strand specifications [4].

There are seven pairs of PF coils located symmetrically about the horizontal midplane. All PF coils are circular and are formed with a winding fixture similar to that for the TF coils. The Nb_3Sn coils PF1–PF5 are reacted, insulated and cured. The eight inner PF coils (upper and lower PF1–PF4s) form the central solenoid (CS) assembly. The CS assembly is attached to the TF coil assembly at the top. The magnet system and the rest of the tokamak systems are housed in a common cylindrical cryostat, which is evacuated prior to cooling down the superconducting coils. In order to verify the characteristics of the KSTAR CS coil, two CS model coils have been fabricated and successfully tested [5].

3.3. Ancillary systems

The design of the KSTAR diagnostic system, which is critical to the physics mission of KSTAR, was discussed in Ref. [6]. With the specific mission of long pulse operation, there is a strong requirement for diagnostics that can provide real time data for control over a long pulse duration. Thus these diagnostics have to operate with reliability and stable calibration, and their output must be integrated into the control system in addition to providing data for physics analysis. Integration of profile measurements into control systems is expected to be an area of significant research in the period prior to KSTAR operation.

The heating system of KSTAR consists of neutral beam injection (NBI) and RF systems. The flexibility to provide a range of control functions, including current drive and profile control, derives from the use of multiple heating technologies: tangential NBI (energy $<$ 120 keV, 8 MW), IC waves (frequency range 20–60 MHz, 6 MW) and LH waves (frequency 3.7 GHz, 1.5 MW). The launched wave spectra can be controlled to provide flexibility in the heating and current drive profiles. The system can be upgraded to 21.5 MW with the addition of 6 MW NBI or, if necessary, up to 27.5 MW by installing additional NBI or RF units and rearranging other ancillary hardware. The NBI system will be designed to provide a local heating capability (mainly ions) at a constant plasma density and profile shape. The ICRF capabilities will allow physics experiments over a range of magnetic fields and provide electron heating and drive current near the axis. The LH antenna design will provide a wavenumber spectrum optimized for localization of electron heating and current drive off-axis.

Figure 5 shows the layout of the main diagnostic systems distributed on the various horizontal ports of the tokamak. Figure 5 also shows the allocation of space for the main heating systems and the layout depicted for the period when the diagnostics and heating systems have been fully implemented. The integrated control technique of long pulse high β plasmas is in its initial development stage, utilizing specifically designed heating and diagnostic systems.

4. Conclusions

The results of the KSTAR conceptual design define a machine with a unique set of capabilities. The steady state tokamak design based on all superconducting magnets will make it the premier facility for the development of continuous, high performance modes of tokamak operation. 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. The combination of performance and design features will make KSTAR the right next step for tokamak research in the next decade.

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E-mail address of G.S. Lee: gslee@comp.kbsi.re.kr

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