Design and construction of the KSTAR tokamak

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Abstract. The extensive design effort for KSTAR has been focused on two major aspects of the KSTAR project mission — steady-state-operation capability and advanced tokamak physics. The steady state aspect of the mission is reflected in the choice of superconducting magnets, provision of actively cooled in-vessel components, and long pulse current drive and heating systems. The advanced tokamak aspect of the mission is incorporated in the design features associated with flexible plasma shaping, double null divertor and passive stabilizers, internal control coils and a comprehensive set of diagnostics. Substantial progress in engineering has been made on superconducting magnets, the vacuum vessel, plasma facing components and power supplies. The new KSTAR experimental facility with cryogenic system and deionized water cooling and main power systems has been designed, and the construction work is under way for completion in 2004.

1. Introduction

The mission of the Korea Superconducting Tokamak Advanced Research (KSTAR) project is to develop a steady-state-capable advanced superconducting tokamak to establish a scientific and technological basis for an attractive fusion reactor [1, 2]. To support this project mission, three major research objectives have been established: (i) to extend 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.

1.1. Design features

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

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

The KSTAR tokamak and its ancillary systems are designed for long pulse operation to explore the physics of steady state fusion plasmas. Global current relaxation times are estimated to be in the range 20–60 s. Considering practical engineering constraints, the activation issue, system cost and conventional facility requirements, the KSTAR tokamak is designed for a pulse length of 300 s. However, since initial operation will focus on advanced tokamak (AT) physics study which does not require long pulse operation, 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, advanced 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 in-vessel modular coils for field error correction and resistive wall mode (RWM) stabilization. An advanced diagnostic system will be employed to measure current and pressure profile variations and to assess performance and stability. The overall tokamak configuration and a cross-sectional view of KSTAR are shown in Figs 1 and 2.

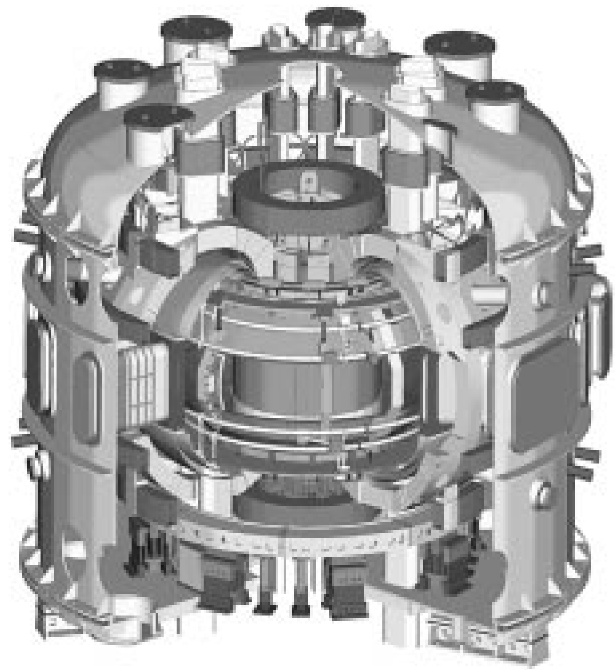

Figure 1. KSTAR tokamak configuration.

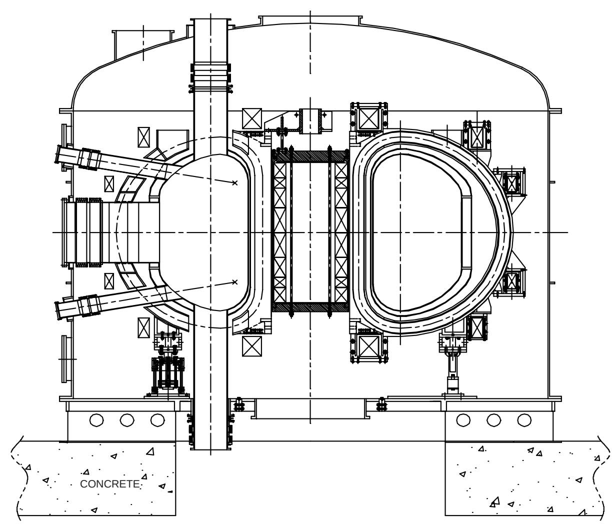

Figure 2. Cross-sectional view of KSTAR.

1.2. Major parameters

During the first phase of the project that ended in 1998, extensive physics and engineering design efforts resulted in the choice of major machine parameters, performance requirements and critical design features. 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 deuterium operation time will be limited to allow personnel access to the SUS-316LN based vacuum vessel interior after reasonable cooldown periods. To minimize activation of the tokamak structure, the cobalt content in the structural material is reduced.

Extending the pulse length to 300 s requires replacing the initial inertially cooled divertor structures with an actively cooled system. 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 in phased implementation. 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 would require a rearrangement of the diagnostic system. For example, one more ion cyclotron resonance heating (ICRH) system could be added, 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 17 V s, an electron cyclotron heating (ECH) power of 0.5 MW at 84 GHz will be installed to assist the plasma initiation in KSTAR 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 in the extended heating option.

2. Physics design and modelling

To demonstrate long pulse, high performance AT operating modes in KSTAR, a range of target operating modes have been identified, such as the H mode, the reversed shear mode and the high l_i mode. There are several physics design issues to be considered for the realization of these target operating modes in KSTAR, such as MHD stability, equilibrium, plasma control, heating and current drive, and heat and particle removal.

In KSTAR, high β MHD stability is provided mainly by strong plasma shaping and the conducting passive plate, which are known to be effective for stabilizing the high n ballooning and the low n external kink modes, respectively. Detailed stability analyses indicated that high β MHD stability is possible for the KSTAR target operating modes. In particular, the reversed shear mode can be stable up to $\beta_N = 5.0$, with a high bootstrap current fraction of $f_{BS} \approx 0.88$ (the stability limit reduces to $\beta_N = 2.5$ without a conducting passive plate). Because RWMs can be excited in this intermediate β_N range, an active control system is thus implemented in KSTAR for the feedback stabilization of RWMs.

For high performance operation through active profile controls, KSTAR is designed to have a wide range of operation flexibility in plasma pressure and current profile space. Figure 3(a) shows the target operating space in the $\beta_N - l_i$ plane. The superconducting PF coil system in KSTAR should be designed to provide these target equilibria, meeting all the superconducting limits, and extensive equilibrium calculations have been performed to find the optimum PF coil dimensions. Figure 3(b) shows the operation windows in terms of the flux linkage through the geometric centre ($R = 1.8$ m) at the seven corners of the operating space, which was obtained for the optimized PF coil dimensions. The KSTAR PF coil system is also designed to have single null equilibrium flexibility by allowing an up–down independent power supply system for four pairs of coils (PF 3–6) out of the seven pairs of the PF coil system.

A reliable and powerful plasma control system is essential for successful operation of long pulse AT modes in next generation tokamaks. There are several plasma control issues to be considered in KSTAR, such as plasma position and shape control, field error correction, disruption avoidance and mitigation, and feedback stabilization of slow timescale MHD modes. Another control issue, especially important for a long pulse device like KSTAR, is the control of plasma profiles and transport. Up to now, most of the design work for the plasma control system in KSTAR has been concentrated on the magnetics control because of its immediate impact on machine design. KSTAR currently has three major components for magnetics control: (i) two pairs of internal control (IC) coils for fast timescale (∼10 ms) vertical and radial position control, (ii) seven independent pairs of superconducting PF coils for slow timescale (∼1 s) plasma shape and current control, and (iii) the field error correction (FEC) RWM coil system for non-axisymmetric field error correction and RWM control. The requirements of the IC coils have been evaluated from dynamic simulations of position control in various model cases for a sudden shift of position and a drop in β of up to 10% caused by minor disruption or ELM-like events in fast time recovery. Meanwhile, the shape control capability of the KSTAR PF coil system has been assessed through dynamic discharge simulation of a high β

Table 1. KSTAR major parameters

	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	1 and 2	
Pulse length (s)	20	300	
Heating power (MW)			$\leq$27.5
Neutral beam	8.0	16.0	
Ion cyclotron	6.0	6.0	
Lower hybrid	1.5	3.0	
Electron cyclotron	0.5	1.0	
Peak DD neutron source rate (s^{-1})	1.5×10^{16}	2.5×10^{16}	

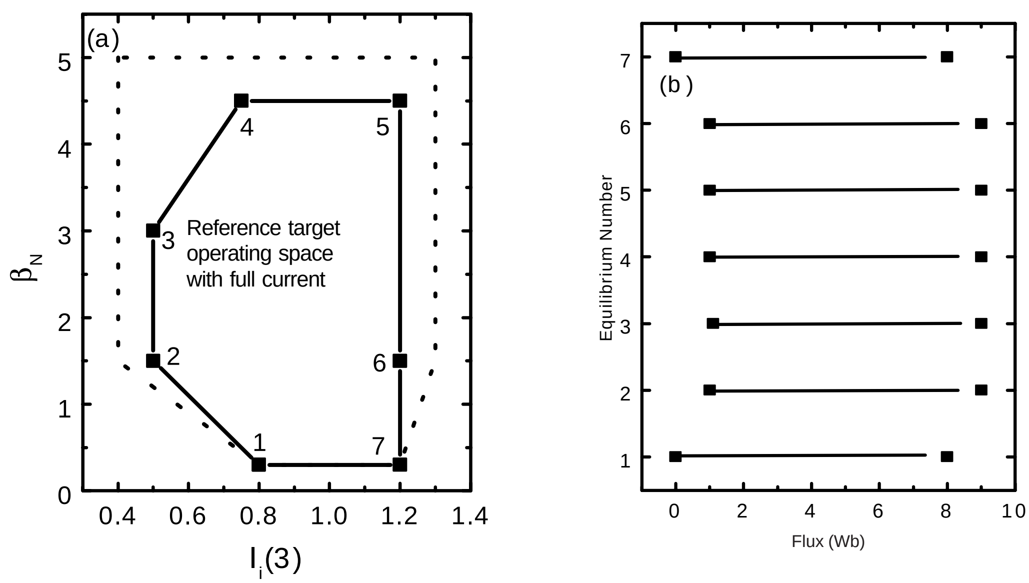

Figure 3. (a) Updated steady state target operating space in the β_N–l_i plane. Inside the solid line is the target operating space with full plasma current (2 MA) and full shape, while inside the dotted line is the extended operating space with possibly reduced plasma current. (b) Operation window at the seven corners of the target operating space.

plasma ($\beta_N = 5.0$) undergoing a 10% permanent drop in β, showing that proper shape control is possible with an isoflux control scheme with a standard proportional–integral–derivative (PID) control law.

In KSTAR, the field error correction (to avoid the locked modes) and the RWM control are realized by utilizing a set of FEC/RWM coils, which are located inside the vacuum vessel. For the assessment of field error magnitude, the various field error sources, such as coil misalignment, coil winding irregularities, bus and lead lines, and vacuum vessel welding, have been considered. With the engineering requirements ($\rho = 2$ mm, $\mu = 1.1$, where ρ is the standard deviation of misalignment and μ is the magnetic permeability of a welded part), it is found that the maximum 20 kA of FEC coil current is required to correct the field error to below the critical value. The feasibility of using the FEC coil to control the RWM has been investigated in a cylindrical limit, showing that the RWM can be controlled with large proportional gains, and the required feedback current is about 7.5 kA [3].

Development of optimum plasma discharge scenarios is also an important physics design issue for achieving the target AT modes in KSTAR. For a reliable plasma initiation, KSTAR utilizes a 0.5 MW ECH assisted 6 V startup scenario. High quality field null (<3 mT) over a large region is achieved by utilizing PF and IC coils to compensate stray fields due to eddy currents. The KSTAR CS coil size has been optimized to supply the required volt-seconds for the reference inductive discharge scenario with 20 s flat-top operation at $I_p = 2$ MA and $\beta_N = 3.5$, and 4 s ramp-up and ramp-down. In order to reduce the quench risk of the CS coil system during the current ramp-up phase, initial magnetization flux is maximized at 8 Wb and the flux swing requirement on PF1 is shifted onto PF2 and PF5. Preliminary discharge simulations have been performed to estimate the non-inductive operation capability of target AT modes. Also, noting that the usual fast ramp-up scenario to make the reversed shear q profile may be difficult in KSTAR with superconducting PF coils, a new scheme has been devised which utilizes two steps of current ramp-up and off-axis current drive by LHCD. For a more self-consistent discharge simulation of these KSTAR target AT modes, an integrated discharge simulation code is being developed. The ASTRA (Automatic System of Transport Analysis in a Tokamak) code has been chosen as a main transport code, and NBI, ICRH/FWCD and LHCD modules have been incorporated into the ASTRA code [4].

3. Tokamak system engineering

The KSTAR tokamak system consists of a vacuum vessel, in-vessel components, cryostat, thermal shield, superconducting magnets and magnet supporting structures. These systems are in the final stage of engineering design with the involvement of industrial manufacturers. The overall configuration and the detailed dimensions of the KSTAR structure have been determined. In addition, the fabrication and installation documents covering fabrication, inspections and assembly procedures are being prepared for the final review process [5].

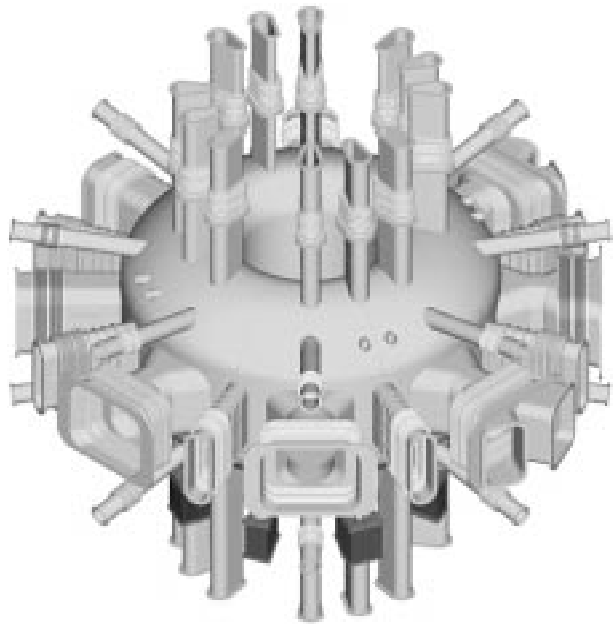

Figure 4. KSTAR vacuum vessel.

3.1. Vacuum vessel

The vacuum vessel is a double walled structure located within the bore of the toroidal field (TF) coils, and consists of the inner and outer shells; horizontal, vertical and slanted ports; and the leaf spring style vessel supports with various types of bellows. Double walls are connected by poloidal and toroidal ribs and are filled with water for cooling and neutron shielding. The overall external dimensions of the main body are 3.4 m height, 1.1 m inner radius and 3.0 m outer radius. The vessel material is SUS-316LN. A 3-D CAD view of the KSTAR vacuum vessel is shown in Fig. 4. The vacuum vessel is designed to be capable of achieving the base pressure of 1×10^{-8} torr, and also to be structurally capable of sustaining the vacuum pressure, the baking gas pressure between shells, the electromagnetic (EM) loads during plasma disruption and the thermal loads during bake-out. Extensive stress analyses have been performed on the vacuum vessel, cryostat and magnet supporting structure under various load conditions, including static (dead weight, coolant and vacuum pressure), thermal (baking gas and cooldown temperature) and dynamic (EM and seismic forces) loads, as well as their combined loads, using the ANSYS code. The analysis results showed that the maximum stresses are within the allowable limits of KSTAR structural design criteria based on the ASME design code.

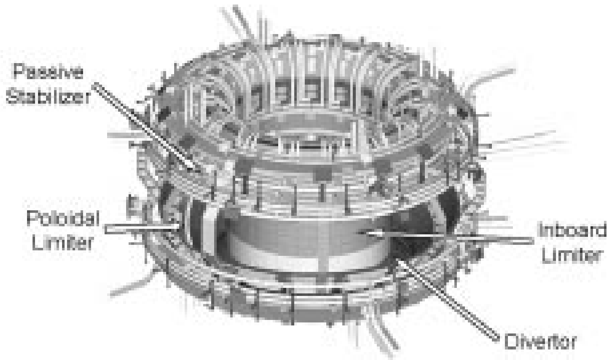

Figure 5. KSTAR in-vessel components.

3.2. In-vessel components

The in-vessel components consist of divertors, inboard limiter, passive stabilizer (including ripple armour), neutral beam shine-through armour, poloidal limiters, in-vessel cryopumps, IC coils and FEC coils.

The baseline plasma facing components (PFCs) are designed with bolted graphite or carbon fibre composite (CFC) tiles supported by SUS-316LN (for divertor, inboard limiter, neutral beam shine-through armour and poloidal limiter) and CuCrZrMg (for the passive stabilizer) back-plates. The back-plates are attached to the vacuum vessel through the PFC supports except for the poloidal limiter, which rests on the mechanical support of the passive stabilizer. The back-plates of divertors, inboard limiter and passive stabilizer are divided into 16 sectors to meet required coolant and baking pipe size, and for easy assembly and maintenance.

All baseline PFCs will be water cooled during plasma operation to maintain the surface temperatures of graphite and CFC tiles at less than 600 and 1200°C, respectively. The baking temperature of the PFCs, set to 350°C, can be achieved within 24 hours and the operation scenario has been established by thermohydraulic analysis. Coolant and baking gas requirements on operation and bake-out have been obtained and the baking/cooling channel design has been carried out. Stress analyses for the situations of plasma disruption, coolant pressure and bake-out have been carried out using the ANSYS code. The contribution of EM loads was found to be much less than that of the thermal loads generated during bake-out. Thermal analyses on the carbon tiles have been performed to determine the required thermo-mechanical properties and to select the proper materials.

A cryopump with a capacity of over 50 torr L/s at 1 mtorr will be installed in the divertor pumping plenum. The cryo-surface at less than 4.3 K is maintained with 3.7 K two phase liquid helium, and regeneration will be done within 10 min for 20 s of baseline operation.

In addition, the engineering design of the in-vessel coils based on hollow copper conductors with SUS-316LN jacket is in progress. The overall configuration of KSTAR in-vessel components is shown in Fig. 5.

3.3. Cryostat

The KSTAR cryostat is an 8.8 m diameter, single walled cylindrical vacuum vessel with a dome shaped lid that provides the vacuum boundary to protect the superconducting magnets. Electrical and mechanical penetrations with bellows have been designed to restrict the displacements of ports due to EM loads and thermal loads within the allowable limits. The cryostat design has been executed to satisfy the performance and operation requirements, such as a base pressure of 1×10^{-5} torr. The cryostat vessel has also been designed to be structurally capable of sustaining the atmospheric pressure plus the dead weight of the vacuum vessel, in-vessel components and magnet, and the dynamic EM loads under all normal and abnormal conditions by performing modal, buckling and stress analyses.

3.4. Thermal shield

The purpose of the thermal shield is to reduce the thermal radiation from the room temperature side to the coil temperature (4.5 K) region. There are two types of thermal shield; one is the vacuum vessel thermal shield (VVTS) located 4 cm off the vacuum vessel outer wall and the other is the cryostat thermal shield (CTS) located 15 cm off the inside cryostat. Both shields act as heat barriers between superconducting magnets operating at 4.5 K and the surfaces of the cryostat and vacuum vessel whose temperature is 300 K. The thermal shield is composed of multilayer insulation (MLI), cryopanels and supports. Aluminized Kapton and aluminized Mylar are used as MLI materials for the VVTS and the CTS, respectively. Carbon fibre reinforced plastic (CFRP) and graphite fibre reinforced plastic (GFRP) are selected as VVTS and CTS support materials, respectively.

The cryopanel was designed to maintain a maximum temperature of 80 K during normal operation and 100 K during bake-out. The TS coolant is gaseous helium operating at 20 bar and its inlet and outlet temperatures are 60 and 80 K, respectively, during normal operation.

3.5. Superconducting magnets

The KSTAR superconducting magnet system consists of 16 TF coils, 4 pairs of CS coils and 3 pairs of outer PF coils.

The TF and PF magnets, using cable-in-conduit type conductors, are cooled with forced flow supercritical helium. The conductor for the TF coils and for PF1–PF5 is Nb_3Sn superconductor with Incoloy 908 conduit, whereas the PF6 and PF7 conductor is NbTi superconductor with SUS-316LN conduit. The Nb_3Sn strand selected for the TF and PF coils is HP-3 strand based on the ITER superconducting strand specification.

A toroidal array of 16 TF coils connected in series produces the 3.5 T toroidal field at the nominal plasma centre with a maximum field on the conductor of 7.2 T and a stored magnetic energy of about 500 MJ. 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 magnet has a D shape with overall dimensions of 3.0 m width and 4.2 m height. The TF system is designed so 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 an octant configuration with two coil assembly. To maintain the structural integrity of the magnet system, the TF structure consists of magnet case, intercoil structure and interoctant joints with insulated shear keys.

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–PF4) form the CS assembly. The CS assembly, consisting of inner tension rods, outer shell, and top and bottom blocks, is attached to the TF coil assembly. All outer PF coils are bolted to the TF coil assembly by a pin-connect type of support structure that can sustain the gravitational and EM loads, with the capability of allowing relative radial motion during cooldown.

3.6. Magnet support structure

The magnet gravity support structure consists of a toroidal ring, eight magnet support posts and eight vertical limiters. The functions of the gravity support are to support the weight of TF, CS and PF coils and structures such as to permit relative radial motion during cooldown and to restrict the vertical, toroidal and off-centre motion. The toroidal support ring located under the TF structure serves as a reference plane of the TF coil system. To prevent eddy current heating, there are eight insulation breaks in the toroidal direction. The vertical load on the coil system is estimated to be 330 t for the coil weight and 320 t for the vertical disruption load. The posts are designed to sustain applied loads and permit radial motion during the cooldown of the coil system. To minimize the heat conduction from cryostat to coil, stainless steel plates with CFRP plates are installed in the posts. A prototype post has been fabricated and tested. The test results are consistent with the design analysis and proved stability for static and dynamic loads of 80 t up to 5000 cycles at 80 K. The vertical limiter is a redundant demountable structure to protect the coil and supporting post in the event of a large vertical disruption or an earthquake. It is made of stainless steel with an internal gap to prevent heat conduction.

The lateral loads in a tokamak could be generated by plasma disruptions and localized halo current flows through the PFCs or vacuum vessel. The estimated peak lateral load in KSTAR is about 1.3 MN and is applied on the vacuum vessel relative to the TF coils. The lateral loads are supported partly by the gravity support structures and partly by tie rod type lateral load support structures. To limit the relative lateral motion of the vacuum vessel to the TF system, 16 lateral load support structures are installed on the top of the vacuum vessel between the top vertical ports and joint boxes of the TF structure at eight places.

4. Ancillary system design and engineering

The design and development of the KSTAR diagnostic system, which is critical to the physics mission of KSTAR, is 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

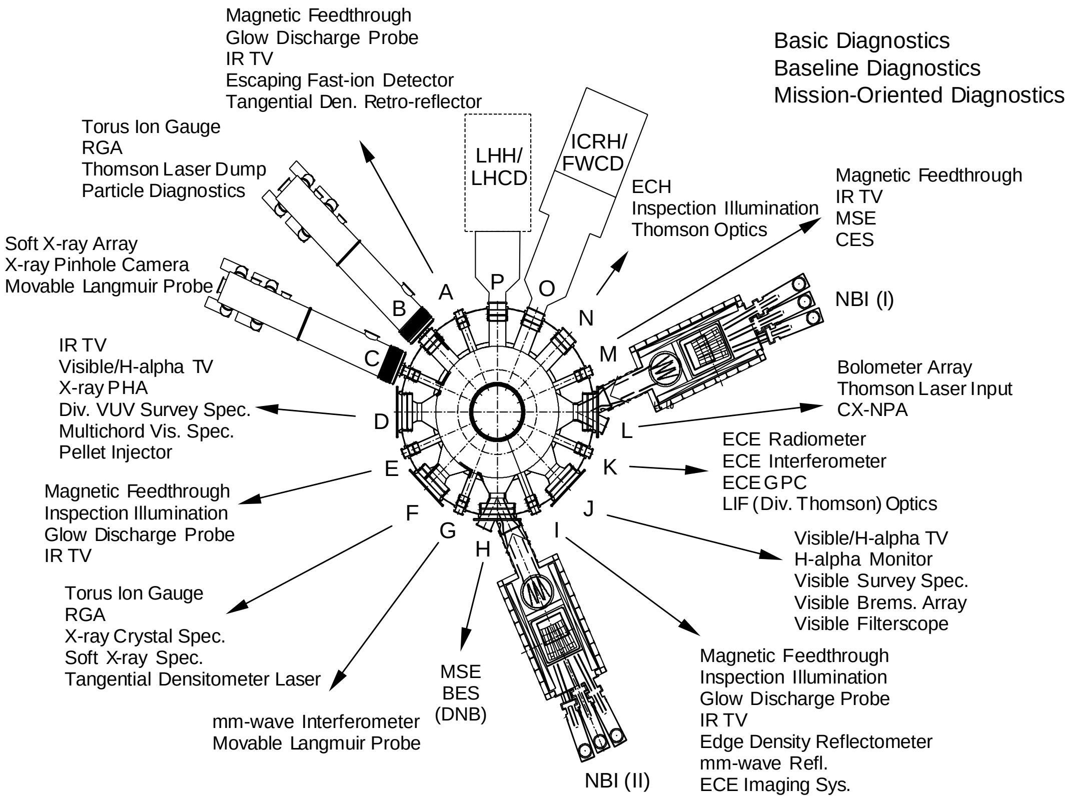

Figure 6. Layout of the KSTAR heating and diagnostic systems.

an area of significant research in the period prior to KSTAR operation.

The baseline heating and current drive system on KSTAR consists of neutral beam injection (NBI) and radiofrequency (RF) systems [7]. 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 25–60 MHz, 6 MW) and LH waves (frequency 5 GHz, 1.5 MW). The launched wave spectra can be controlled to provide flexibility in the heating and current drive profiles. 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 will 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 6 shows the layout of the main diagnostic system distributed over various horizontal ports of the tokamak. Figure 6 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 plasma is in its initial development stage, utilizing specifically designed heating and diagnostic systems.

5. Facility construction

The design of the KSTAR experimental facility, with high-bay buildings for the machine hall, mechanical and electrical conventional utilities, and a 10 kW class cryogenic system, was completed in 1998. The facility construction was started in early 1999, and the completion of the experimental building with beneficial occupancy for machine assembly is expected in early 2002. The construction of special utilities will begin upon completion of building construction and is targeted for completion and commissioning in the latter part of 2004.

6. Conclusions

The results of the design and engineering work defined a machine with a unique set of capabilities.

The AT design based on a fully superconducting magnet system will make KSTAR a premier facility for development of steady state, high performance modes of tokamak operation in this decade.

Acknowledgement

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