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Design and Construction of the KSTAR Tokamak

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Abstract. The extensive design effort 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 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, vacuum vessel, plasma facing components, and power supplies. The new KSTAR experimental facility with cryogenic system and de-ionized water-cooling and main power systems has been designed, and the construction work has been on-going for completion in year 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 toward 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 physics of steady-state fusion plasmas. Global current relaxation times are estimated to be in the range of 20-60 s. Considering practical engineering constraints, 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 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, capable plasma control tools are required. The KSTAR will have a plasma heating system that will heat the plasma to high temperature and high beta, 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 (b_N) and current-profile shapes (l_i) in double-null as well as single-null plasmas. To control the MHD behavior of high-beta plasmas, in-vessel conducting structures are provided for passive stabilization, two sets of in-vessel coils for active position control, and in-vessel modular coils for field-error correction and resistive wall mode stabilization. An advanced diagnostic system will be employed to measure current and pressure profile variations and to assess performance and stability.

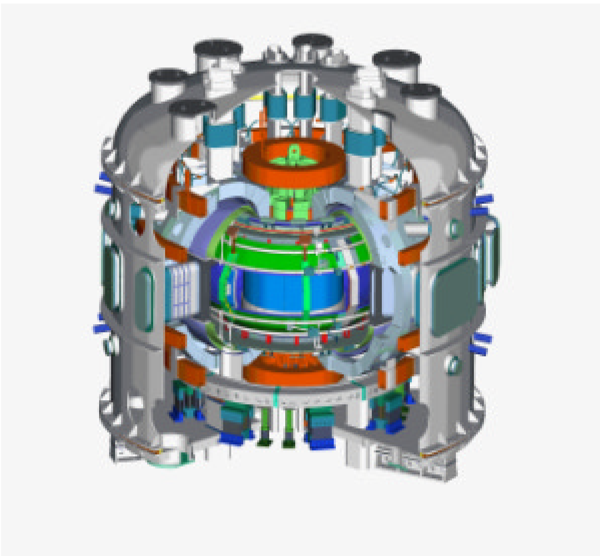

FIG. 1. KSTAR Tokamak Configuration

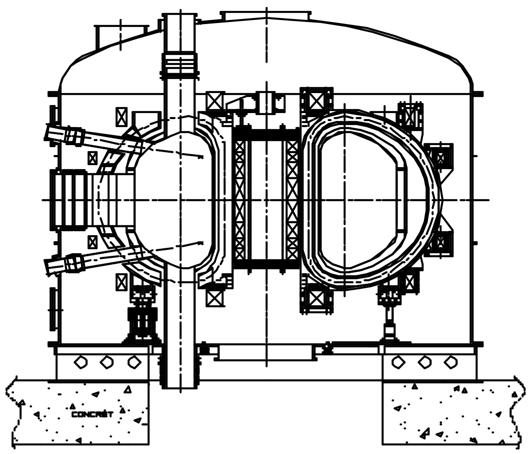

FIG. 2. Cross-sectional View of KSTAR

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 I. The machine will be operable in 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 cool-down periods. To minimize activation of tokamak structure, the reduction of Cobalt content in structual material is utilized.

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 I. 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 cooling water supply passage for upgrade. Although the detailed descriptions are not shown in the “Extended” column of Table I, 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 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 PF system is capable of providing a flux swing of 17 V-sec, 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.

TABLE I. KSTAR MAJOR PARAMETERS

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 & 2	
Pulse length (s)	20	300	
Heating power (MW)			≤ 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}	

2. Physics Design and Modeling

To demonstrate a long-pulse, high-performance advanced tokamak (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-li 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, heat and particle removal.

In KSTAR, the 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 the high-β MHD stability, are possible for the KSTAR target operating modes. Especially, the reversed-shear mode can be stable up to β_N =5.0, with the high bootstrap-current fraction of f_{BS} ~ 0.88 (the stability limit reduces to β_N =2.5 without conducting passive plate). Because the resistive wall modes can be excited in this intermediate β_N -range, an active control system is thus implemented in KSTAR for the feedback stabilization of the resistive wall mode.

For the 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 β_N - l_i plane. The superconducting poloidal field (PF) coil system in KSTAR should be designed to provide this target equilibria, meeting all the superconducting limits, and extensive equilibrium calculations have been performed to find optimum PF coil dimensions. Figure 3 (b) shows the operation windows in terms of the flux-linkage through the geometric center (R=1.8m) at seven corners of operating space, obtained for the optimized PF coil dimensions. KSTAR PF coil system is also designed to have the single-null equilibrium flexibility by allowing up down independent power supply system for the four pairs coils (PF 3-6) among the seven pairs of PF coil system.

A reliable and powerful plasma control system is essential for successful operations 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, feedback stabilization of slow time-scale MHD modes. Another control issue, especially important for the long-pulse device, like KSTAR, is the control of plasma profiles and transport. Up to now, most design work of plasma control system in KSTAR has been concentrated on the magnetics control because of its immediate impact on machine design. KSTAR has currently three major components for magnetics control: (i) two pairs of internal control (IC) coils for fast time scale (~10 msec) vertical and radial position controls, (ii) seven independent pairs of superconducting PF coils for slow time scale (~1 sec) plasma shape and current control, and (iii) Field Error Correction (FEC)/Resistive Wall Mode (RWM) coil system for non-axisymmetric field error correction and resistive wall mode control. The requirements of the IC coils have been evaluated from dynamic simulations of position control in various model cases for sudden shift of position and up to 10% beta-drop by minor-disruption or ELM-like events in fast time recovery. Meanwhile, the shape control capability of KSTAR PF coil system has been assessed through dynamic discharge simulation of the high-b plasma (b_N=5.0) undergoing 10% permanent beta-drop, showing that a proper shape control is possible with isoflux control scheme with standard 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 (r =2mm, m =1.1, where r is the standard deviation of misalignment and m is the magnetic permeability of welded part), it is found that the maximum 20 kA of FEC coil current is required to correct the field error below the critical value. The feasibility of using FEC coil to control 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 to achieve the target AT modes in KSTAR. For a reliable plasma initiation, KSTAR utilizes 0.5 MW ECH-assisted 6 V start-up 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. KSTAR CS coil size has been optimized to supply the required volt-second for the reference inductive discharge scenario with 20 s flattop operation at Ip =2MA and b_N =3.5, and 4 s ramp-up and ramp-down. In order to reduce the quench risk of CS coil system during the current ramp-up phase, initial magnetization flux is maximized to 8 Wb and a burden of 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 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 implemented into the ASTRA code [4].

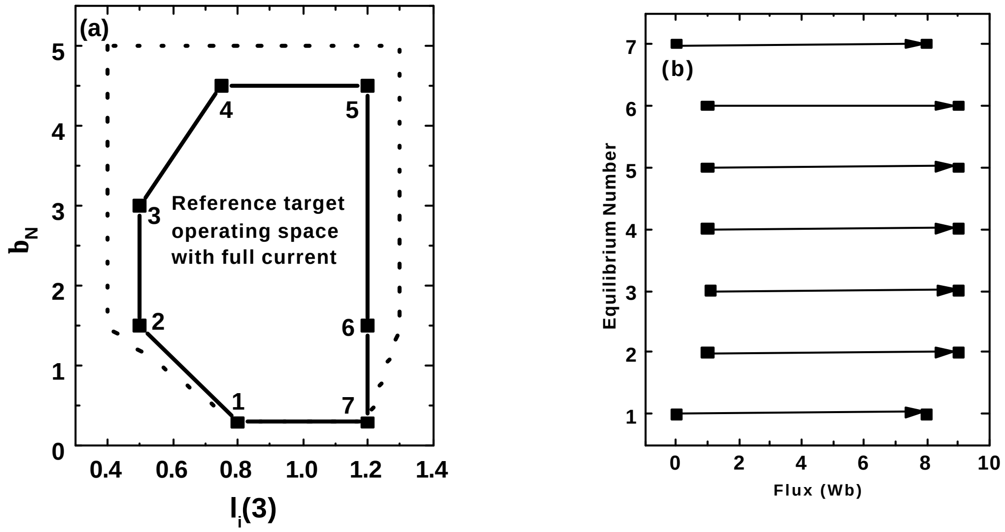

FIG. 3. (a) An updated steady-state target operating space in l_i-b_N plane. Inside solid line is the target operating space with full plasma current (2 MA) and full shape, while inside dotted line represents the extended operating space with possibly reduced plasma current. (b) Operation window at seven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

3. Tokamak System Engineering

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

3.1 Vacuum Vessel

The vacuum vessel (VV) 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 filled with water for cooling and neutron shielding. The overall external dimensions of the main body are 3.4 m high, 1.1 m inner radius, and 3.0 m outer radius. The vessel material is SUS 316LN. The vacuum vessel is designed to be capable of achieving the base pressure of 1×10^{-8} Torr, and also be structurally capable of sustaining the vacuum pressure, baking gas pressure between shells, and the electromagnetic (EM) and thermal loads during plasma disruption and bakeout, respectively. The extensive stress analyses have been performed on the vacuum vessel, cryostat, magnet supporting structure under various load conditions including static (dead weight, coolant and vacuum pressure), thermal (baking gas and cool down temperature), dynamic (EM and seismic forces) and their combined loads using ANSYS code. The analysis results showed the maximum stresses are within allowable limits of ASME design-code based the KSTAR structural design criteria.

3.2 In-vessel Components

The in-vessel components consist of divertors, inboard limiter, passive stabilizer (including ripple armor), neutral beam shinethrough armors, poloidal limiters, in-vessel cryopumps, internal control coils, and field error correction coils.

The baseline plasma facing components (PFC) are designed with bolted graphite or carbon-fiber-composite (CFC) tiles supported by SUS 316LN (for divertor, inboard limiter, neutral beam shinethrough armor, and poloidal limiter) and CuCrZrMg (for passive stabilizer) back-plates. The back-plates are attached to the VV through the PFC supports except for the poloidal limiter which resides on the mechanical support of 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 easy assembly and maintenance.

All baseline PFCs will be water-cooled during plasma operation to maintain the surface temperatures of graphite and CFC tiles less than 600 C and 1,200 C, respectively. The baking temperature of the PFCs that has been set to 350 C, can be achieved within 24 hours and its operation scenario has been established by thermo-hydraulic analysis. Coolant and baking gas requirements on operation and bakeout, have been obtained and the baking/cooling channel design has been carried out. Stress analyses for the situations of plasma disruption, coolant pressure and bakeout have been carried out using ANSYS code. The contribution of EM loads was found to be much less than the thermal loads generated during bakeout. Thermal analyses on the carbon tiles have been performed to determine the required thermo-mechanical properties and to select the proper materials.

Cryopump of over 50 Torr·l/s at 1 mTorr will be installed in the divertor pumping plenum. The cryo-surface of less than 4.3 K is maintained with 3.7 K two-phase liquid helium and regeneration will be done within 10 minutes for 20 seconds of baseline operation.

Also, the engineering design of the in-vessel coils based on hollow-copper conductors with SUS 316LN jacket is in progress.

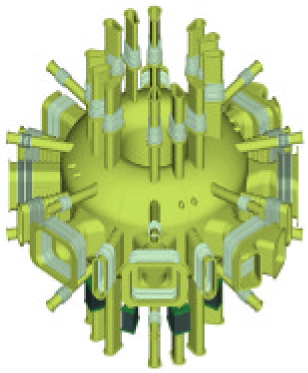

FIG.4. KSTAR Vacuum Vessel

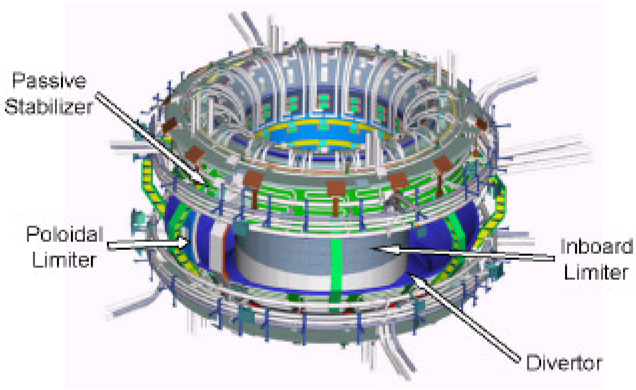

FIG. 5. KSTAR In-vessel Components

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 base pressure of 1×10^{-5} Torr. The cryostat vessel has also been designed to be structurally capable of sustaining the atmospheric pressure plus a dead weight of the vacuum vessel, in-vessel componets, 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 (TS) 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 shields; one is the vacuum vessel thermal shield (VVTS) located 4 cm off the VV 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 that operate at 4.5 K and the surfaces of cryostat and VV whose temperature is 300 K. The TS 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. Also CFRP and GFRP are selected as VVTS and CTS support materials, respectively.

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

3.5 Superconducting Magnets

The KSTAR superconducting magnet system consists of 16 Toroidal Field (TF) coils, 4 pairs of Central Solenoid (CS) coils, and 3 pairs of outer Poloidal Field (PF) coils.

The TF and PF magnets using cable-in-conduit type conductors (CICC) are cooled with forced-flow supercritical helium. The conductor for the TF and PF1 through 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 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 center with the maximum field on the conductor at 7.2 T and stored magnetic energy is 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 Dee-shape with overall demension in 3.0m width and 4.2 m hight. 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 octant configuration with two-coil assembly. To maintain structual integrity of magnet system, the TF structure consists of magnet case, inter-coil structure, inter-octant 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-PF4s) form the central solenoid (CS) assembly. The CS assembly consisting of inner tension rods, outer shell, top and bottom blocks, is attached to the TF coil assembly. All outer PF coils are bolted to the TF coil assembly by pin-connect type support structure that could sustain gravity and electromagnetic loads, with capability of allowing relative radial motion during cool-down.

3.6 Magnet Support Structure

The magnet gravity support structure consists of a toroidal ring, 8 magnet support posts, and 8 vertical limiters. The functions of the gravity support are to support the weight of TF, CS, PF coils and structures, to permit the relative radial motion during cool-down, and to restrict the vertical, toroidal, and off-center motion. The toroidal support ring located under TF structure serve as a reference plane of TF coil system. To prevent the eddy current heating, there are 8 insulation breaks in the toroidal direction. The vertical load on coil system is estimated to be 330 tons for the coil weight and 320 tons for vertical disruption load. The posts are designed to sustain applied loads, and permit the radial motion during the cool-down of coil system. To minimize the heat conduction from cryostat to coil, stainless steel plates with carbon fiber reinforced plastic (CFRP) plates are installed in posts. A prototype post has been fabricated and tested. The test results are consistent with design analysis, and proved to be stable for static and dynamic loads of 80 tons up to 5,000 cycles at 80 K. The vertical limiter is a redundant demountable structure to protect coil and supporting post in the event of large vertical disruption and earthquake. It is made of stainless steel with intermediate gap to prevent the heat conduction.

The lateral loads in tokamak could be generated by plasma disruptions and localized halo current flows through the plasma facing components (PFC) or vacuum vessel. The estimated peak lateral load in KSTAR is about 1.3 MN, and is applied on vacuum vessel relative to TF coils. The lateral loads are supported partly by gravity support structures and partly by tie-rod type lateral load support structure. To limit the relative lateral motion of vacuum vessel to TF system, 16 lateral load support structures are installed on the top of vacuum vessel between top vertical ports and joint boxes of TF structure at 8 places.

4. Ancillary Systems Design and Engineering

The design and development of the KSTAR diagnostic system, which is critical to the physics mission of KSTAR, is discussed in elsewhere [6]. With the specific mission of long-pulse operation, there is a strong requirement for diagnostics that can provide real-time data for control in 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 baseline heating and current drive system on the KSTAR consists of neutral beam injection (NBI) and radio-frequency (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 of < 120 keV, 8 MW), ion-cyclotron waves (frequency range of 25-60 MHz, 6MW), and lower-hybrid waves (frequency of 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 experiment over a range of magnetic fields and provide electron heating and drive current near the axis. The LH antenna design will provide a wave-number spectrum optimized for localization of electron heating and current drive off-axis.

Figure 6 shows a layout of the main diagnostic system distributed on the various horizontal ports of the tokamak. Figure 6 also includes the allocation of space for the main heating systems, and the layout drawing depicted for the period when the diagnostics and heating systems have been fully implemented. The integrated control technique of long-pulse high-beta plasma is in its initial development stage, utilizing specifically designed heating and diagnostic system.

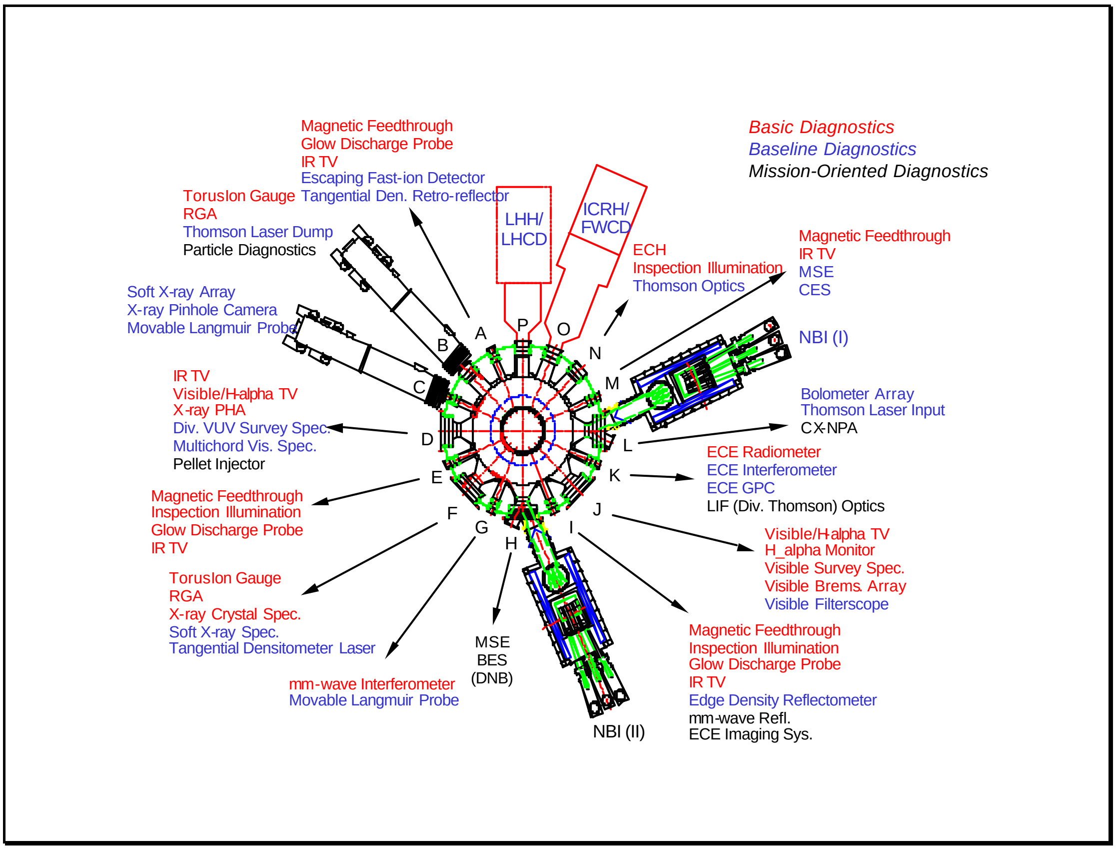

FIG. 6. Layout of the KSTAR Heating and Diagnostic Systems

5. Facility Construction

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

6. Conclusions

The results of the design and engineering works defined a machine with unique set of capabilities. The advanced tokamak 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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