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THE ITER POLOIDAL FIELD SYSTEM*

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Abstract:

The ITER poloidal field (PF) system uses superconducting coils to provide the plasma equilibrium fields, slow equilibrium control and plasma flux linkage (V-s) needed for the ITER Operations and Research Programme. Double-null (DN) divertor plasmas and operation scenarios for 22 MA Physics (high-Q/ignition) and 15 MA Technology (high-fluence testing) phases are provided. For 22 MA plasmas, total PF flux swing is 333 V-s. This provides inductive current drive (CD) for start-up with 66 V-s of resistive loss and 440-s (330-s minimum) sustained burn. The PF system also allows plasma start-up and shutdown scenarios, and can maintain the plasma configuration during burn over a range of current and pressure profiles. Other capabilities include increased plasma current (25 MA with inductive CD; 28 MA with non-inductive CD assist), divertor separatrix sweeping, and semi-DN and single-null plasmas.

Introduction:

The ITER PF system meets a complex set of requirements. These requirements relate to the function of the PF system in isolation -- plasma equilibrium, equilibrium control and flux linkage -- and to the interaction of the PF system with other ITER systems. In addition, since the central solenoid (CS) of the PF system is a major component of the ITER tokamak core, PF system requirements and CS magnet performance affect the overall size of the ITER.

The PF system must also provide experimental flexibility for the full ITER Operations and Research Programme. Here the PF system must support a range of operational scenarios for Physics experiments and Technology testing, accommodate a range of plasma current density and pressure profiles within a given scenario, and also allow operation with DN, semi-DN and single-null (SN) plasma configurations.

Design Requirements:

Design of the PF system and the overall ITER configuration has been an iterative process. The design requirements and key physics and engineering considerations that emerge from this process are:

Magnet Technology. Sustained fusion burn and possibility of steady-state operation are essential aspects of ITER design. Considerations [1] of maximum PF magnet performance and minimum power demand during operation mandate superconducting technology. Nb_3Sn superconductor and radially-graded structural reinforcement in the CS coils [1] minimize overall ITER device size [2].

PF Coil Configuration. Assembly and maintenance of the TF and PF magnets require the PF coils to be outboard of the TF magnets. Access gaps for assembly of in-magnet and in-vessel components and for plasma heating/CD systems and diagnostics limit where the PF coils can be located. The need to produce a symmetric DN plasma and considerations of TF and PF coil system structural symmetry lead to a PF coil system with up/down symmetry.

Plasma Configuration. Physics considerations [3] of divertor geometry, maximum plasma elongation, minimum MHD q and plasma current (I_p) required for ignition determine the plasma configuration. Divertor channel lengths (x-point to strike point) $\geq$0.6 m inboard and $\geq$1.5 m outboard are required to provide acceptably low divertor plasma temperature and helium exhaust. This leads to a plasma with moderately high triangularity: $d_x = 0.60$ at the x-point, and $d_{95} = 0.38$ at 95% poloidal flux. This triangularity is consistent with the range $d_{95} = 0.3$-0.5 needed for MHD stability at high β [4].

Considerations of obtaining sufficient plasma current for ignition [3] and minimizing device size [2] favor a plasma with maximum vertical elongation. An upper limit on elongation of $k_x \cong 2.25$ and $k_{95} \cong 2.0$ is set by passive and active stabilization of the plasma vertical position [5]. A lower limit on edge safety factor of $q_{95} \geq 3.0$ is set by MHD stability and avoidance of disruptions [4]. Final plasma configuration parameters are $R_o = 6.00$ m, $a = 2.15$ m, $R_x = 4.71$ m, $Z_x = +/- 4.78$ m, and $k_{95} = 1.98$ and $q_{95} = 3.00$ at $I_p = 22$ MA. This configuration is common to all of the operational scenarios presented below.

Operational Scenarios and Experimental Flexibility. The PF system is nominally designed for the 22 MA Physics (A1.) operation scenario. This scenario provides sufficient inductive CD capability for a 200-s burn anywhere within the broad operational regime needed to explore the physics characteristics of a burning plasma. The A1. scenario requirements establish the basic PF coil parameters (coil locations and CS radial build and maximum ampere-turn capabilities). Subsequent analysis has shown that with minor increases in the ampere-turn capabilities of the outboard PF coils, alternate Physics scenarios with increased plasma current (A2. and A3.) and Technology scenarios with extended burn duration (B1. and B2.) are also feasible. Scenario and operational parameters used for PF design are summarized in Table I.

Plasma Profile Parameters. The PF currents required for plasma equilibrium and the flux linkage required for a given equilibrium condition depend on the plasma current density profile j(r) and pressure p. For PF design, these quantities are specified in terms of two profile parameters: internal inductivity, $l_i(3) = 4U_p/\mu_o R_o I_p^2$, and poloidal beta, $\beta_p(3) = 2\mu_o\langle p\rangle/\langle\langle B_p\rangle\rangle^2$, where U_p is the poloidal magnetic energy within the plasma, B_p is the poloidal field, $\langle p\rangle$ is the volume-average pressure, and $\langle\langle B_p\rangle\rangle$ is the poloidal line average of B_p on the plasma boundary.

The inductivity range for all scenarios is $0.55 \leq l_i(3) \leq 0.75$. This range bounds the j(r) profiles with acceptable MHD stability at high toroidal β [6]. The plasma pressure range is approximately $0.7 \leq \beta_p/\beta_p(\text{ref}) \leq 1.4$, where

β_p(ref) (Table I) is the nominal value of β_p for the respective scenario. This pressure range provides a two-fold decrease or increase in plasma reactivity for nominal plasma composition (Z_{eff}, impurity and helium fractions), or alternately, constant reactivity with corresponding variations in plasma composition. Nominal fusion power is ~1.1 GW for the Physics scenarios and ~0.9 and ~1.6 GW for B1. and B2. Technology operation. These parameters apply from start-of-burn (SOB) to end-of-burn (EOB)] with nominal plasma temperature, density and composition. The profile ranges for the pre-burn start-of-current flattop (SOFT) state are $0.55 \leq l_i(3) \leq 0.75$ and $0 \leq \beta_p(3) \leq 0.2$.

Plasma Flux Linkage and V-s Requirement.

The PF system must provide sufficient variation in flux linkage at the nominal plasma axis to support the inductive and resistive flux required to establish the SOFT state, and to inductively sustain the plasma configuration through EOB. The flux variations required are evaluated using:

$$\Delta\Psi_{PF}(SOFT) = L_p^{SOFT}(l_i,\beta_p)I_p + 0.4\mu_0 R_0 I_p \quad , \qquad (1)$$

and

$$\Delta\Psi_{PF}(SOB) = L_p^{SOB}(l_i,\beta_p)I_p + 0.4\mu_0 R_0 I_p + 10 \text{ V-s} \quad , \qquad (2)$$

where L_p^{SOFT} and L_p^{SOB} are the plasma self-inductances for the corresponding SOFT and SOB profile parameters. Resistive loss (volume-integrated) at SOFT is estimated using the Ejima scaling formula [6] $\Delta\Psi_{res}(SOFT) = 0.4\,\mu_0 R_0 I_p$. The feasibility of obtaining this near-minimum resistive loss has been verified in ITER near-term R&D experiments conducted in JET, JT-60 and DIII-D, and in numerical simulations of ITER start-up.

Resistive loss during heating to burn is estimated to be 10 V-s. Additional resistive flux for sustaining the burn is estimated using

$$\Delta\Psi_{PF}(EOB) = \Delta\Psi_{PF}(SOB) + U_{loop}(I_{ind})t_{burn} \quad , \qquad (4)$$

where

$$U_{loop} = (4.3 - 0.6\,R_o/a)\,2.15 \times 10^{-3}\,(Z_{eff}I_{ind}R_o/k_{95}a^2)/T_{10}^{1.5} \qquad (5)$$

is the loop voltage during burn [3], and t_{burn} is the burn duration. In Eq. (4), $I_{ind} = I_p - I_{BS}$ is the net inductively-driven current, I_{BS} is the bootstrap current, and T_{10} is the density-weighted average plasma temperature in units of 10 keV. For 22 MA Physics operation with $Z_{eff} = 1.66$, $T_{10} = 1.0$ and $I_{BS} = 3$ MA, $U_{loop} = 0.117$ V. For a burn duration of 200 s, 23.4 V-s of resistive flux are required. The PF system is sized to provide sufficient inductive flux linkage for a) $t_{burn} \geq 200$ s for $0.55 \leq l_i(3) \leq 0.75$ and $0.4 \leq \beta_p(3) \leq 0.8$, and b) sufficient total V-s to provide $\Delta\Psi(\text{burn}) \geq 0.1\Delta\Psi(\text{EOB})$. Requirement (b) sets the minimum PF V-s capability, which is 327 V-s for the A1. scenario.

PF System Design Description and Performance:

A set of 14 superconducting (SC) PF coils establishes the plasma equilibrium and provides inductive current drive and slow ($t > 1$ s) equilibrium control. These coils, positioned in 7 up/down symmetric pairs (Table II), are located outboard of the TF coils. Conducting structures for passive stabilization of the plasma vertical position and a pair of resistive coils for active control of the plasma vertical position are located within the vacuum vessel. Details are given in [5]. Details of the engineering design of the SC PF coils are given in [1].

Dimensions and maximum ampere-turn and voltage requirements for the PF coils are given in Table III. Maximum currents are set by flux linkage and equilibrium requirements within the various scenarios. The maximum design currents impose minor limitations on the A3. and B1. scenarios (see Table I).

The currents and voltages given in Table II are extremes for the scenarios and profile parameters in Table I. Maximum voltages are set by requirements for an open-circuit plasma loop voltage of 25 V/turn [3] and equilibrium control in the initial phase of the current ramp-up.

The coil parameters given in Table III describe the current-carrying portion of the winding pack. The CS coils (PF1-PF4) have a radially-graded pack [1], in which the amount of structural reinforcement varies over the coil radius. Flux and peak CS field at maximum initial magnetization (IM) are respectively 132.8 V-s and 13.5 T. Magnetic energy at IM and EOB is typically 13 GJ. Maximum V-s capability at EOB is obtained with $I_{PF1} = I_{PF2} = 22.8$ MA-turns, the limit set by conductor quench protection requirements [1].

Total PF V-s capability at EOB and V-s available for burn vary with scenario and also with $l_i(3)$ and $\beta_p(3)$. PF V-s capabilities are summarized in Table III. For the A1. scenario with $l_i(3) = 0.65$ and $\beta_p(3) = 0.6$, V-s capability is 333 V-s. This gives a 6 V-s margin relative to the 10% minimum burn flux requirement, and a 28 V-s margin relative to the estimated V-s for a 200-s burn.

Additional Design Considerations:

PF Optimization: The number, configuration and location of the PF coils are optimized to meet operational requirements. The number of coil groups (7) and the subdivision of the CS into 4 groups are set by the need to control the equilibrium configuration over a range of pressure and j(r) profiles and to provide the range of plasma equilibria (circular -> DN divertor -> circular) needed for plasma start-up, burn and shutdown. The location and size of the gap between PF5 and PF6 is set by access needed for in-vessel assembly and maintenance. The gap between the PF7 coils is set by access for plasma heating and diagnostics. Variations in coil position within these limitations have only a minor effect on PF performance (< 10% variation in maximum magnetic energy).

Alternate Plasma Configurations: Limiter-defined start-up and shutdown sequences on either the inside or outside first-wall and slightly up/down asymmetric DN (semi-DN, SDN) and SN plasmas can be produced. In the SDN mode, a 1-2 cm vertical displacement of the plasma magnetic axis relative to the device midplane splits the inside and outside separatrices by radial distances (measured at the outside midplane) comparable to the power scrape-off decay length (~0.5 cm). Larger vertical displacements (to ~0.2 m, limited by the in-vessel component geometry) result in a SN configuration. Here the radial splitting between the separatices is ~5 cm and power flow will be solely to the divertor connected to the inside separatrix.

Shape Control and Divertor X-point Sweeping: Plasma shape control during the burn and start-up/shutdown phases and divertor x-point sweeping (δR_x = +/- 3 cm at 0.3 Hz) will be accomplished by control of the currents in the SC PF coil set. All major plasma shape parameters (R-a, R+a, R_x, Z_x), DN->SN symmetry and I_p can be independently controlled by appropriate combinations (non-interacting modes) of PF current perturbations. Feedback control from magnetic and other diagnostic data is planned. Maximum control voltages at the PF coils are typically 2-5 kV, and peak PF power demand for control is less than 300 MW. A control accuracy of ~+/- 0.5 cm appears feasible, limited mainly by magnetic data accuracy and residual errors in the plasma flux configuration reconstruction algorithms. The control time scale of >1 s is set by the penetration time of the control fields through the vacuum vessel and limitation of PF power demand.

References:

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Table I

Operation Scenarios for PF Design (all with $0.55 < l_i(3) < 0.75$)

Scenario	Description	I_p(MA)	β_p(ref)	β_p Range	Feature or [Limitation]
A1.	Physics	22	0.6	0.4-0.8	>200-s burn @ $l_i(3)=0.75$
A2.	Physics	25	0.6	0.4-0.8	~50-s burn @ $l_i(3) = 0.75$
A3.	Physics	28	0.5	0.4-0.6	[$0.6 \leq l_i(3) \leq 0.7$], [20 V-s CD assist]
B1.	Technology	15.4	1.4	1.0-2.0	~130 V-s for burn [$\beta_p < 1.8$ for $l_i(3) = 0.55$]
B2.	Technology	21.4	1.0	0.6-1.4	wall load ~1.5 MW/m^2

Table II

PF Coil System Parameters

Coil	R (m)	+/-Z (m)	ΔR (m)	ΔZ (m)	N[a]	I_{max}(MA)	V_{max}(kV)
PF1U/L	1.725[b]	0.950	0.650[b]	1.840	520	22.8	12
PF2U/L	1.725[b]	2.850	0.650[b]	1.840	520	22.8	12
PF3U/L	1.725[b]	4.750	0.650[b]	1.840	520	22.8	20
PF4U/L	1.725[b]	6.650	0.650[b]	1.840	520	22.8	20
PF5U/L	3.900	9.000	0.900	0.899	480	18.5	20
PF6U/L	11.500	6.000	0.499	1.499	338	16.5	20
PF7U/L	11.500	3.000	0.499	0.900	208	9.7	20

[a]Number of turns in winding pack

[b]Radially graded winding pack

Table III

PF System V-s Capabilities at Reference β_p

Scenario	I_p(MA)	Total Flux Swing(V-s)			Burn Flux Swing (V-s)		
	$l_i(3) =$	0.55	0.65	0.75	0.55	0.65	0.75
A1.	22.0	333.5	333.1	330.0	65.8	51.8	38.9
A2.	25.0	340.9	338.5	336.9	38.0	22.2	7.4
A3.	28.0[a]	343.6	341.2	339.2	27.7[a]	10.2[a]	-6.7[a]
B1.	15.4	335.6	334.8	334.6	135.6	126.5	118.0
B2.	22.0	348.9	346.5	344.9	74.4	60.1	46.8

[a]With 20 V-s non-inductive assist during current ramp-up

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