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ITER PLASMA SAFETY INTERFACE MODELS AND ASSESSMENTS

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Physics models and requirements to be used as a basis for safety analysis studies are developed and physics results motivated by safety considerations are presented for the ITER design. Physics specifications are provided for enveloping plasma dynamic events for Category I (operational event), Category II (likely event), and Category III (unlikely event). A safety analysis code SAFALY has been developed to investigate plasma anomaly events. The plasma response to ex-vessel component failure and machine response to plasma tansients are considered.

1. INTRODUCTION

ITER[1] is designed to operate with D-T plasma, producing 1–1.5 GW of fusion power for an ignited burn pulse duration of 1000 s or more. Because of the uncertainties in plasma physics, the safety approach in ITER is to take minimum credit from plasma physics. However, it is necessary to address various plasma physics phenomena in investigation of potential accident sequences to demonstrate that ITER design has sufficient provisions to withstand these sequences without violating safety and criteria.

Section 2 summarizes the physics guidelines and specifications to be used in safety analysis for enveloping plasma dynamic events for safety Category I (operatinal event), II (likely event), and III (unlikely event).[3] The basic physics guidance developed here are implemented in a safety analysis code SAFALY[4] to investigate safety related events ranging from plasma transients to thermal behavior of in-vessel components· Sample results from SAFALY are discussed in Sect. 3.

The effect of plasma transients on machine and safety are important. Runaway electrons are considered as an example in Sect. 4. Plasma response to first wall LOCA and the possibility for passive shutdown due to Be evaporation are covered in Sect. 5.

2. PHYSICS GUIDELINES FOR SAFETY ANALYSIS

The main physics issues have been assessed by the ITER JCT, the Home Teams, and the ITER Physics Expert Groups on the basis of data from present tokamak experiments. The physics basis and design guidelines are developed from reasonable extrapolations of this database.[1,2] Representative ITER plasma and device parameters, derived from guidelines are given in Table 1. A brief summary of physics guidelines to be used as a basis for safety studies,[3] related to enveloping plasma events for safety Category I (operational event), Category II (likely event), and Category III (unlikely event), are presented here. In all expressions, units are mks, MA, MW, with κ,δ average values at 95% flux, and $n_{20} = n_e/10^{20}\ m^{-3}$, $T_{10} = \langle T/10\ keV\rangle$, $A_i =$ atomic mass. H-mode profiles: $n,T \sim (1 - r^2/a^2)^{\alpha}$, with $\alpha_n \approx 0.1$ and $\alpha_T \approx 1.0$ as nominal values.

Table 1
Nominal ITER device and plasma parameters

Parameter	Symbol	Value
Major/minor radius	R/a	8.14 m/2.8 m
Plasma configuration	—	Single-null
Plasma elongation	κ_{95}, κ_x	~1.6, ~1.75
Plasma triangularity	δ_{95}	~ 0.24
Nominal plasma current	I	21 MA
Toroidal field	B	5.68 T (at R = 8.14 m)
MHD safety factor	q_{95}	3.05
Fusion power (nominal)	P_{fus}	1.5 GW
Fusion power excursion	—	±20%
Plasma thermal energy	W_{th}	1.14 GJ nominal
Plasma magnetic energy	W_{mag}	1.19 GJ
Average wall loading	Γ_n	~1 MW/m^2
Inductive pulse flat-top (ignited conditions)	t_{pulse}	1000 s

DISCLAIMER

Confinement: Plasma energy confinement must be sufficient to achieve ignition and sustained burn at fusion power of ~1.5 GW, under empirical scalings for ELMy H-mode plasma energy confinement and corresponding He and impurity concentrations (Z_{eff} ~ 1.5, n_{DT}/n_e ~ 0.6).

Energy confinement: ELMy H-mode

τ_E(required) = τ_E(ELMy H-mode)
= $H_H \times [0.85 \times \tau_E$(ELM-free H-mode)

Here H_H = H-mode scale factor with respect to 0.85 times ELM-free H-mode [H_H = 1 ELMy H-mode, $H_H < 1$ degraded H-mode, H > 1 "advance scenario"].

τ_E(ELM-free H-mode) = τ_E(ITER93H) = $\tau_{E\text{-}93H}$
$= 0.053 I^{1.06} R^{1.9} a^{-0.11} n_{20}^{0.17} B^{0.32} A_i^{0.41} \kappa_x^{0.66} P^{-0.67}$

For simulations of plasma transients or accident scenarios, it would be necessary to bound the energy confinement time to avoid numerically possible but physically implausible results. Here, we impose the following restriction

τ_E = min [τ_E(neoclassical); τ_E(OH); τ_E(ELMy)]
≈ min [τ_E(OH); τ_E(ELMy H-mode)]

$\tau_E(OH) = 0.07\,[n_{20}]\,a\,R^2\,q_\psi$ Neo-Alcator scaling

Particle confinement:
ELMy H-mode with $\tau^*_{He}/\tau_E = 10$.
Here, $\tau^*_{He} = \tau_{pHe}/(1 - R_{He})$, τ_{pHe} = He particle confinement time, R_{He} = He recycling coefficient.

Impurity content: Beryllium $n_{Be}/n_e = 2\%$

Auxiliary power: $P_{aux} \geq P_{thr}$(L-H threshold power)

Safety factor: $q_{\psi 95} \geq 3.0$

Beta Limit: $\beta_{max}(\%) = g\,(I/aB) = \beta_N\,(I/aB)$

- $\beta_N \leq 2.5$ nominal operation (ignition studies)

Note that confinement and beta limits deteriorate and disruptivity increases with low-q operation. The experimentally observed effective stability limit is β_N/q_ψ ~ 0.7–0.9. Here we use the effective stability limit β_N/q_ψ as a measure of disruptivity. Table 2 summarizes the guidance for beta limit disruptions, expressed as β_N/q (normalized beta/q).

Density Limit: The density limit imposes an upper limit on the plasma edge density. The density limit in H-modes generally prompts a return to the L-mode. In ITER, the return to L-mode will produce a fast decay of the fusion power, faster than the density decay, and a disruption will be highly probable in the absence of sufficient auxiliary power. Here we introduce two density limits: Greenwald [n_{GR}] and Borrass [n_{BR}] density limits. Set points for density limit disruptions in ITER-class (reactor) plasmas will be taken as $n_e \leq k_n\, n_{crit} = k_n \times [n_{GR}$ and $n_{BR}]$. Table 2 summarizes the guidance.

Table 2
Set points for beta and density limit disruptions

Event	Beta Guidance $[(\beta_N/q_\psi)_{crit}]$	Density Guidance $n_e/n_{crit} < k_n$
Cat I	~0.7–0.9	(1.4–1.5)
Cat II	~1	1.75
Cat III	~1.2	2

Greenwald density limit is $n_{20}^{GR} = \kappa J(MA/m^2) = I/(\pi a^2)$. The Borrass density limit adapted for a single-null divertor configuration of ITER is:[2]

$$n_{20} = C(n_e/n_{es}) Q_\perp^{5/8} B^{5/16} (1 - f_{rad}^{div})^{11/16} / (q_\psi R)^{1/16}$$

where n_{es} = plasma electron density at the separatrix, $Q_\perp(MW/m^2)$ = mean power flux crossing the separatrix, $f_{rad}^{div} = P_{rad}^{div}/(4\pi^2 R a \kappa^{0.5} Q_\perp)$ is the divertor impurity radiative fraction, $q_\psi = q_\psi(95\%)$, and C ≈ 2.37. The value of n_{es}/n_e (~ 0.4–0.7) depends on particle transport and fueling at the edge.

H-mode Threshold Power: H-mode is reached above a certain threshold power ($P_{L\text{-}H}$). A reverse H-to-L transition will occur when power crossing the separatrix falls below roughly half of $P_{L\text{-}H}$.
L-to-H transition: $P_{sep} \geq P_{L\text{-}H}$.
$P_{L\text{-}H0} = 0.044\, n_{20}\, B\, S$
$P_{L\text{-}H1} = 0.3 n_{20} B R^{2.5}$ $P_{L\text{-}H2} = 0.016 n_{20}^{0.75} B S$
$P_{L\text{-}H3} = 0.036 n_{20} B^{0.6} S$ $P_{L\text{-}H4} = 0.025 n_{20} B S$.

where P_{sep} (= $P_{heat} - P_{rad} - \partial W/\partial t$; $P_{heat} = P_\alpha + P_{OH} + P_{aux}$) = power crossing the separatrix, and S = plasma surface area. For physics-safety studies, because plasma startup and nominal operation are not safety issues, it is reasonable to assume that ITER plasma reaches H-mode instantaneously, without any consideration of L-H power threshold.

H-to-L transition: Typically, $P_{H\text{-}L} \sim 0.5 \times P_{L\text{-}H}$. For physics-safety studies, considering the variations in L-H power expressions and lower boundaries of experimental data points, the recommended expression for the H-L transition power threshold is

$$P_{H\text{-}L} = \min[0.2 \times (P_{L\text{-}H0}, P_{L\text{-}H1}, P_{L\text{-}H2}, P_{L\text{-}H3}, P_{L\text{-}H4})].$$

3. LOSS OF PLASMA CONTROL

Examples for loss of plasma control are sudden change (increase/decrease) in fueling rate or sudden application of available auxiliary power into an ignited plasma. These transients are studied with the SAFALY code. For conservatism, no mitigative action or active controls are assumed. A sudden change in (doubling of) the confinement time is also investigated to cover uncertainties in plasma physics. Plasma density and β-limit disruptions are assumed at Cat. II and III levels (optimistic levels) to conservatively assess the maximum fusion power transients.

For sudden improvement of confinement time (factor of two increase), plasma density and fusion power increases. At about 2.3 s (3.6 s), plasma beta exceeds the Cat II (Cat III) β-limit disruption set points. Prior to disruption, up to 2.5 GW (3 GW) of fusion power is reached transiently. Maximum divertor heat loads reach 20 MW/m^2. Because of short time scales [overpower continuance time scale << time scale of structure thermal behavior], damage to in-vessel components are not expected.

In the case of accidental coupling of 100 MW auxiliary power into an ignited plasma, confinement time decreases quickly (because of power dependence) divertor loads rapidly increase and reach 15 MW/m^2 within 5 s (~ energy confinement time) and continue for about 10 s. Divertor target surface temperature increases up to 2800, and plasma is terminated due to fuel dilution from impurities (excess radiation) and reverse transition into the L-mode.

4. RESPONSE TO PLASMA TRANSIENTS

An example in this group is the runaway electrons. The runaway electrons (Table 3) produced during plasma disruptions can cause serious damage to the plasma facing components. The issue is whether the runaway electrons can cause (in one disruption) damage to a large number of the blanket modules and result in an accident accompanied with a large rate of the water leak. In Table 3, runaway heat load on FW is estimated by assuming that plasma moves vertically and sweeps the FW over a short poloidal distance (~0.5 m). If the plasma moves slowly and radially, touching FW at the same poloidal position, the local heat load could be as high as 100 MJ/m^2, and the total energy of runaway electrons is sufficient to melt a narrow ditch along the toroidal direction with a depth about 1 cm in all blanket modules. However, this is only possible if the FW/ blanket modules are perfectly aligned.

Table 3
Parameters of the runaway electrons

Parameter	Symbol [Unit]	Value
Pre disruption current	I [MA]	21
Predicted runaway current	I_{run} [MA]	12
Lifetime	[s]	10
Time to hit conducting wall	[s]	1
Total energy in runaways	W_r [MJ]	30
Diameter of current channel	d_{run} [m]	~4
Runaway flux on first wall	[MJ/m^2]	5

Estimates of heat load and probability: The angle between escaping runaway electrons and FW can be estimated from $\alpha = v/c = \Delta/2\pi Rq$, where v is the velocity of electrons normal to the FW, and Δ is the thickness of runaway SOL. Poloidal width of wetted area: $h \sim (2a\Delta)^{0.5}$ assuming a worst case that plasma does not move vertically and the poloidal width of the wetted area is defined by SOL thickness. Estimates of the minimum and maximum values are given below:[5]

minimum	maximum
$v \sim 30$ m/s	$v \sim B_p/(4\pi nm)^{0.5}$
plasma motion	MHD instabilities
$\alpha \sim 10^{-7}$	$\alpha \sim 10^{-3}$
$\Delta \sim 1.5\ 10^{-3}$ cm	$\Delta \sim 15$ cm
[$h \sim 0.9$ cm]	[$h \sim 90$ cm]

For $10^{-4} < \alpha < 10^{-3}$, Δ is greater than the FW/ blanket module alignment [of $\delta_w \sim \pm 1$ cm], and all blanket modules along the toroidal direction will be wetted. However, heat load is not very large because of poloidal distribution [$h \sim$ 30-90 cm]. Peak heat load is $Q \sim 0.7 - 2$ MJ/m^2. At very small angles, $\Delta/\delta_w < 1$, the probability that only n blanket modules are wetted by the runaway electrons (assuming each of the N = 60 blanket modules are distributed randomly within the range $\delta_w \sim \pm 1$ cm) is:[5]

$$P(n) \approx \left(\frac{\Delta}{\delta w}\right)^n \frac{N!}{n!(N-n)!}$$

and the heat load is:

$$Q = \frac{W_{RA} N}{2\pi R n (4\pi q a R \alpha)^{1/2}}$$

Note that heat loads of 30-40 MJ/m^2 (dangerous for FW) can only be reached at very small α's, at which the probability for a large number of modules being wetted by runaway electrons is very low. The probability that more than 10 modules will be wetted by runaways is less than 10^{-6}. Although safety credit cannot be taken for random misalignment, it is possible to make a deliberate

FW modulation in the toroidal direction (by placing a few blanket modules by ~3-5 cm closer to the plasma then the rest of them) to avoid an accident with all blanket modules. Note that deliberate FW modulation is not a part of the ITER design basis.

5. PLASMA RESPONSE TO EX-VESSEL LOCA

In an accident event where cooling system of the FW fails, the FW temperature continues to rise as long as the ignited state of the core plasma continues. It is interesting to look at the issue of Be evaporation to see if plasma can have an inherent self-healing effect (by terminating the burn due to excessive Be concentration) and limit FW temperature [to <1000°C] in response to ex-vessel LOCA. Plasma response has been investigated using various methods and models, ranging from simple estimates to consideration of diffusion models (Bohm and neoclassical diffusion),[6] to the use of the hybrid code SAFALY.[4] From a simple analysis, for plasma termination, needed Be density [from power balance, assuming perfect confinement] is $\sim 2 \times 10^{19}$ m^{-3}. Assuming probability/retention of R ~ 0.01 [1 out of 100 Be particles transported to plasma], total number of Be needed to be released from FW is about 4×10^{24}. With FW area affected by ex-vessel LOCA ~300 m^2 and residence time of order τ_p ~5–10 s, needed evaporation flux is $\sim 10^{21}$ atoms/m^2/s, which is equivalent to evaporation rate ~340 mm/y, corresponding to the FW temperature of ~1000°C. Results are summarized in Fig. 1, indicating that passive plasma shutdown appears possible in the temperature range of 1050 to 1300°C for R values in the range of 10^{-2} to 10^{-4}.

Note that the evaporated Be is ionized in the scrape off layer (SOL) plasma, and most of Be ions are swept into the divertor chamber, while a small portion diffuses into the core plasma. The estimated ionization and radiation losses by Be in the SOL and the energy lost to the divertor exceeds the alpha heating power (300MW) when the SOL temperature is greater than 20 eV.[6] Energy losses in the SOL will result in radiation (thermal) collapse (disruption), and/or loss of H-mode (return to L-mode). Similar results (thermal collapse of the edge plasma) are obtained with the 2-D, UEDGE code.[7]

In calculations with SAFALY,[5] postulated LOCA occurs at 1 s after the simulation starts. The plasma is terminated passively at about 180 s after the LOCA by a combination of density limit disruption and power balance failure. At ~150 s, the surface temperature just before the disruption is about 1100°C, and the coolant tube temperature is about the same. The FW coolant tube is SS316 and the tube will not melt at these temperatures. However, Copper is used as a heat sink around the tube, so the heat sink could melt.

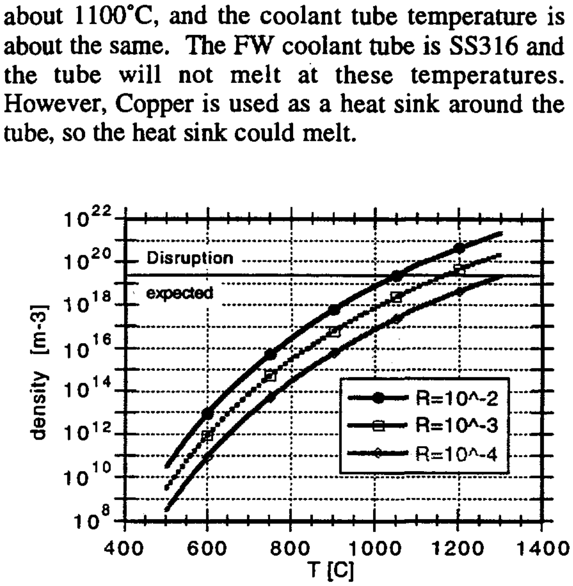

Figure 1: Beryllium concentration inside the plasma due to evaporation of Be. Assumed FW area 300 m^2. R = fraction of evaporated Be atoms that are transported into the core plasma. The residence time of a Be atom inside the plasma is taken as 5 seconds.

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