ITER PHYSICS-SAFETY INTERFACE: MODELS AND ASSESSMENTS

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ABSTRACT/SUMMARY

Plasma operation conditions and physics 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 guidelines and specifications for enveloping plasma dynamic events for Category I (operational event), Category II (likely event), and Category III (unlikely event) are characterized. Safety related physics areas that are considered are: (i) effect of plasma on machine and safety (disruptions, runaway electrons, fast plasma shutdown) and (ii) plasma response to ex-vessel LOCA from first wall providing a potential passive plasma shutdown due to Be evaporation. Physics models and expressions developed are implemented in safety analysis code (SAFALY, couples 0-D dynamic plasma model to thermal response of the in-vessel components). Results from SAFALY are presented.

I. INTRODUCTION

ITER^{1,2} is designed to operate with D-T plasma, producing 1.5 GW of fusion power for an ignited burn pulse durations of 1000 s or more. Because of the uncertainties in plasma physics and difficulties for simple robust/precise quantifications of safety related physics areas, the safety approach in ITER is to take minimum credit from plasma physics. However, it is necessary to address various plasma physics phenomena in safety analysis. These include: (i) disruptions (including vertical displacement events VDE's), (ii) runaway electrons, (iii) off normal fusion power and plasma shutdown, and (iv) fusion power transients. In this paper we present results from some of these areas as motivated by safety consideration.

The physics basis and design guidelines are developed from reasonable extrapolations of the tokamak database and

guidance of tokamak theory.¹⁻³ In Sect. II, we describe the physics guidelines and specifications for enveloping plasma dynamic events.⁴⁻⁵ Here, we classify these guidelines as nominal operational bound, rare but possible (very few, if any data points), and upper bound (no data points). In safety language, these may be labeled as Category I, II, and III events, respectively. The basic physics models and equations developed here are implemented in a safety analysis code (SAFALY)⁶ to calculate anomaly sequences of fusion reactors initiated by plasma events. SAFALY code couples 0-D dynamic plasma model to the thermal response of the in-vessel components.⁶ The effect of plasma on machine and safety are discussed in Sect. III. Here three examples are considered: disruptions, runaway electrons, and fast plasma shutdown. Results from analysis are presented and, when appropriate, mitigation techniques are discussed. The plasma response to FW LOCA and possibility for passive shutdown of plasma by Be evaporation are covered in Sect. IV.

II. PHYSICS GUIDELINES AND SPECIFICATIONS 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 and guidance of tokamak theory.¹⁻³ The main plasma and device parameters (Table I) have been derived from the performance goals using these guidelines. The plasma operation conditions and physics requirements to be used as a basis for safety studies are further investigated.^{4,5} A brief summary of these guidelines, related to enveloping plasma events for safety Category I (operational event), Category II (likely event), and Category III (unlikely event) are presented here.

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Table I. Nominal ITER device and plasma parameters to be used for safety analysis

Parameter	Symbol	Value
Major radius	R	8.14 m
Minor radius	a	2.80 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
Pulse repetition time	t_{rep}	2200 s

Units are mks, MA, MW, with κ, δ average values at 95% flux, and $n_{20} = n_e/10^{20} \text{ m}^{-3}$ = line-average electron density, $T_{10} = \langle T/10 \text{ keV} \rangle$ = density-weighted average temperature ($T \approx T_e \approx T_i$), A_i = average atomic mass. H-mode profiles: $n, T \sim (1 - r^2/a^2)^{\alpha_n, T}$, with $\alpha_n \approx 0.1$ and $\alpha_T \approx 1.0$ as nominal values. Nominal ignited operating point: 1.5 GW, $n_e \sim 1.3 \times 10^{20} \text{ m}^{-3}$, $T \sim 10.5 \text{ keV}$.

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 ($Z_{eff} \sim 1.5$, $n_D T/n_e \sim 0.6$). Reference assumptions/models are as follows:

Energy confinement: ELMy H-mode

$$\begin{aligned}\tau_E(\text{required}) &= \tau_E(\text{ELMy H-mode}) \\ &= H_H \times [0.85 \times \tau_E(\text{ELM-free H-mode}) \\ &\quad \sim H_L \times \tau_E(\text{ITER89P L-mode})]\end{aligned}$$

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, $H_H > 1$ improved "advance scenario" modes], $H_L = H_{89}$ = L-mode enhancement factor.

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

$$\tau_E(\text{ELMy H-mode}) = H_H \times 0.85 \times \tau_{E-93H}$$

$$\begin{aligned}\tau_E(\text{L-mode}) &= \tau_E(\text{ITER89-P L-mode}) = \tau_{E-89P} \\ &= 0.048 A_i^{0.5} I^{0.85} R^{1.2} a^{0.3} \kappa_x^{0.5} n_{20}^{0.1} B^{0.2} P^{-0.5}\end{aligned}$$

In many 0-D or 1-D plasma simulations, empirical scaling expressions for energy confinement time are used. 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^{3,5}

$$\begin{aligned}\tau_E &= \min [\tau_E(\text{neoclassical}); \tau_E(\text{OH}); \tau_E(\text{ELMy H-mode})] \\ &\approx \min [\tau_E(\text{OH}); \tau_E(\text{ELMy H-mode})]\end{aligned}$$

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

Particle confinement: ELMy H-mode with $\tau^*_{He}/\tau_E = 10$. Here $\tau^*_{He} = \tau_{pHe}/(1 - R_{He})$, where τ_{pHe} = He particle confinement time and 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: $q_{95} \geq 3.0$

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

- $\beta_N \leq 2.5$ nominal operation (ignition studies)
- $\beta_N (> 2.5?)$ advanced scenarios with reversed shear

Current profile control is possible for steady state, advanced scenarios (with reversed shear), resulting in a potentially higher β_N -value. Note that confinement and beta limits deteriorate and disruptivity increases with low-q operation. Experimentally observed effective stability limit is $\beta_N/q_{\psi} \sim 0.7-0.9$, indicating that ITER parameters ($\beta_N/q_{\psi} \sim 2.5/3$) may be marginal. Thus, it would be difficult to attain high- β_N values (>3) without raising q substantially. Here we recommended to use the effective stability limit β_N/q_{ψ} as a measure of disruptivity. Table II summarizes the guidance for beta limit disruptions,⁵ expressed as β_N/q (normalized beta/q) is $\beta_N/q_{\psi} \leq (\beta_N/q_{\psi})_{crit}$.

Table II. Set points for beta limit disruptions [$\beta_N/q_{\psi} \leq (\beta_N/q_{\psi})_{crit}$]

Event/Description	Guidance [(β_N/q_{ψ}) _{crit}]
Cat I: nominal operation range	~0.7-0.9
Cat II: rare but possible	~1
Cat III: upper bound (bounding limit)	~1.2

Density Limit: The density limit imposes an upper limit on the plasma edge density. The density limit in L-mode plasmas is generally disruptive, the limit in H-modes generally prompts a return to the L-mode and a disruption is not inevitable. 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

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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 $n_e \leq n_{crit}$ with $n_{crit} = k \times [n_{GR} \text{ and } n_{BR}]$. Table III summarizes the guidance for Cat I, II, and III.⁵

Table III. Set points for density limit disruptions
[$n_e \leq n_{crit} = k \times [n_{GR} \text{ and } n_{BR}]$ $n_e/n_{crit} \leq k$]

Event/Description	Guidance [$n_e/n_{crit} \leq k$]
Cat I: nominal operation range	(1.4–1.5)
Cat II: rare but possible	1.75
Cat III: upper bound (bounding limit)	2

Greenwald density limit is $n_{20}^{GR} = \kappa J(\text{MA}/\text{m}^2) = I/(\pi a^2)$. Note that the ITER nominal density [at 1500 MW ignited burn] is higher than the Greenwald limit (by ~1.4–1.5 times--Cat I). In several experiments with pellet fueling (although in L-mode), it was possible to exceed the Greenwald limit by as much as a factor of 2 (to 3). However, ITER can not take credit for $(2-3) \times n_{GR}$ because this is only possible transiently with deep, central pellet fueling resulting in a very peaked density profiles. In ITER (or in a reactor scale plasma), it is not possible to have a deep central fueling (unless the compact toroids are used for fueling and proven to be a viable option) to get peaked density profiles. At the present time, ITER is not using compact toroids for fueling, and pellet velocities is only around 1 km/s (pellet penetration is only at the edge).

A tentative characterization of the Borrass density limit adapted for a single-null divertor configuration of ITER is:²

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

where n_{es} = plasma electron density at the separatrix, $Q_{\perp}(\text{MW}/\text{m}^2)$ = 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 $C \approx 2.37$. The value of n_{es}/n_e (~0.4–0.7) depends on particle transport and fueling at the plasma periphery.

H-mode Threshold Power: H-mode is reached above a certain threshold power (P_{L-H}). A reverse H-to-L-mode transition will occur when the power crossing the separatrix falls below a value roughly half of P_{L-H} .

L-to-H transition: Initially ITER used $P_{sep} \geq P_{L-H0} = 0.044 n_{20} B S$, where $P_{sep} (= P_{heat} - P_{rad} - \partial W/\partial t; P_{heat} = P_{\alpha} + P_{OH} + P_{aux})$ is the power crossing the separatrix and S is the plasma surface area. For nominal ITER parameters, $P_{L-H0} \sim 300 n_{20}$ (MW). During startup, n_{20}

~0.3–0.5, access to H-mode will require 100–150 MW. Recent analysis of the ITER H-mode threshold database, suggest the following expressions:⁷

$$\begin{aligned} P_{L-H1} &= 0.3 n_{20} B R^{2.5} & P_{L-H2} &= 0.016 n_{20}^{0.75} B S \\ P_{L-H3} &= 0.036 n_{20} B^{0.6} S & P_{L-H4} &= 0.025 n_{20} B S. \end{aligned}$$

In all cases, the threshold is assumed to be given by the lower boundary of the data points. At $n_{20} \sim 0.5$, the extrapolation of these expressions to ITER gives about $P_{L-H1} = 150$ MW, $P_{L-H2} = 65$ MW, $P_{L-H3} = 60$ MW, and $P_{L-H4} = 85$ MW. Isotope scaling of the threshold power is not yet well established. A lower threshold in tritium than in deuterium is possible. Note that P_{L-H0} expression is the most conservative for ITER..

For physics-safety studies, because the plasma startup and nominal plasma operation are not safety issues, it is reasonable to assume that ITER plasma reaches H-mode instantaneously, without any consideration of the L-H power threshold. Alternatively, it may be more realistic to consider the following [one-half of minimum value]:

$$P_{L-H} = \min [0.5 \times (P_{L-H0}, P_{L-H1}, P_{L-H2}, P_{L-H3}, P_{L-H4})]$$

H-to-L transition: $P_{H-L} \sim 0.5 \times P_{L-H}$. For nominal ITER parameters, P_{H-L} (MW) ~ $150 \times n_{20}$ [about 180 MW at $n_{20} \sim 1.2$, indicating that radiated power fraction, P_{rad}/P_{α} , cannot exceed ~40% (for 1500 MW nominal fusion power) without likely loss of H-mode and likely loss of sustained burn]. For physics-safety studies, considering the variations in L-H power expressions and lower boundaries of experimental data points, recommended expression for the H-L transition power threshold is

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

Fusion Power: The nominal value of the fusion power is 1500 MW, of which 1200 MW is the neutron power (300 MW alpha power). A power excursion of $\pm 20\%$ lasting for ~10 s is anticipated. Fusion power is obtained by integrating D-T fusion reactivity over plasma profiles [$P_{fus} \sim \int n^2 \langle \sigma v \rangle E_{fus} dV$]. Expression for $\langle \sigma v \rangle$ is⁸

$$\begin{aligned} \langle \sigma v \rangle &= 10^{-6} \times C1 \times \theta \times \exp(-3\xi) \times [\xi / (m_r c^2 T^3)]^{0.5} \\ \theta &= T / (1 - T(C2 + T(C4 + T \times C6)) / (1.0 + T(C3 + T(C5 + T \times C7)))) \\ \xi &= [(B_G)^2 / (40)]^{1/3} \end{aligned}$$

where $\sigma v(\text{m}^3/\text{s})$ = reactivity, $T(\text{keV})$ = temperature. For D-T: fuel, $B_G[(\text{keV})^{0.5}]$ = parameter = 34.3827, $m_r c^2(\text{keV})$ = rest mass energy = 1.124656×10^6 and C 's [C1 through C7] are the fitting vectors: $C1 = 1.17302 \times 10^{-9}$, $C2 = 1.51361 \times 10^{-2}$, $C3 = 7.51886 \times 10^{-2}$, $C4 = 4.60643 \times 10^{-3}$, $C5 = 1.35 \times 10^{-2}$, $C6 = -1.06750 \times 10^{-4}$, $C7 = 1.366 \times 10^{-5}$.

Edge and Divertor Radiation Power Partitioning and Scaling: Nominal operational scenario is an ignited case [$P_{aux} = 0$] with fusion power, $P_{fus} = 1500$ MW. Nominal (ignited operation) values are as follows.

Power (in MW)		
Alpha heating power, P_α		300
Radiation from core plasma, $P_{core-rad}$		100
Power crossing scrape-off,		
$PSOL = P_\alpha - P_{core-rad}$	200	
Radiation from edge, $P_{edge-rad}$	50	
Power to the divertor (separatrix),		
$P_{div} = P_{sep} = PSOL - P_{edge-rad}$	150	
Radiation from divertor region, $P_{div-rad}$	100	
Power to divertor target,	50	
$P_{div-target} = P_{div} - P_{div-rad}$		

During plasma transients, when the fusion power, density, or impurity fractions change, scaling relationships given in Table IV can be used. Physics Expert Groups will make an attempt to verify these scalings or will try to provide a better scaling expressions in the near future.

Table IV.

Power loss partition for nominal 1500 MW ignited operation ($P_{aux} = 0$) and scaling relationships for transients [for driven operation, $P_{aux} \neq 0$, replace P_α with $P_\alpha + P_{aux}$]

Parameter	Scaling/proportionality relationship	Nom. (MW)
Alpha power, P_α	$\propto f_{DT}^2 n_e^2 \langle \sigma v \rangle$, $f_{DT} = n_{DT}/n_e$	300
Core radiation, $P_{core-rad}$	$= P_{brems} + P_{syn}$ $P_{brems} \propto n_e^2 T_e^{0.5} Z_{eff}$	~100
Power crossing the scrape-off layer, $PSOL$	$= P_\alpha - P_{core-rad}$ $= P_\alpha + P_{aux} - P_{core-rad}$ for $P_{aux} \neq 0$	~200
Edge line radiation, $P_{edge-rad}$	$\propto (f_{z-edge})^{0.75} (n_{edge})^{1.5}$ $f_{z-edge} = n_{z-edge}/n_{edge}$ n_{edge} = edge density assume n_{edge}/n_e constant	~50
Power to the divertor/separatrix $P_{div} = P_{sep}$	$= PSOL - P_{edge-rad}$	~150
Power radiated at divertor region (separatrix) $P_{div-rad}$	$\propto (f_{z-div})^{0.75} (n_{es}) (P_{sep})^{0.5}$ $f_{z-div} = n_{z-div}/n_{edge}$ n_{es} = separatrix density $= n_{edge}$ assume n_{es}/n_e constant	~100
Power load at divertor target, $P_{div-target}$	$= PSOL - P_{edge-rad} - P_{div-rad}$ $= P_{div} - P_{div-rad}$ $= P_{sep} - P_{div-rad}$	~50
Auxiliary power, P_{aux}	$P_{aux} = 0$ -- ignition scenario $P_{aux} \neq 0$ -- driven scenario [up to 100 MW available]	0 ≤100

Beryllium Evaporation and First Wall/Divertor Impurity Retention Probability: Evaporation/sublimation is a concern in fusion devices during both normal and abnormal operations. Evaporation yield⁹

$$\Gamma_i = \frac{2.6 \cdot 10^{24}}{\sqrt{M_i T [K]}} 10^{B_i - A_i/T [K]}, \frac{\text{atoms}}{m^2 \cdot \text{sec}}$$

where T = surface temperature in K, M_i = atomic mass, and α = sticking coefficient [for metals $\alpha \approx 0.6 - 1.0$]. with

$$M_i = \text{Be, C, W} = [9, 12, 183.85]$$

$$A_i = \text{Be, C, W} = [16720, 40181, 44485]$$

$$B_i = \text{Be, C, W} = [11.61, 14.80, 12.74]$$

Figure 2 shows evaporation data [in mm/year or atoms/m²/s] for Be, C, and Tungsten as a function of the target temperature.

Figure 2: Evaporation rate (mm/year) or (atoms/m²/s) for Be, C, and W as a function of T (°C, K)

The plasma impurity concentration and erosion/evaporation etc. from the wall or the divertor plates/targets are related through a complicated transport process, involving ionization of the atoms, their streaming along magnetic field lines, and their diffusion perpendicular to the field lines. From experimental results,¹⁰⁻¹³ we find that a probability of an Be atom produced at the wall to penetrate into the main plasma is in the range of 0.1 -to- 0.01 [i.e., 1 in 10 to 1 in 100 atoms are transported into the plasma]. A similar probability for Be atom produced at the divertor neutralizer plate to penetrate into the main plasma is in the range of 10^{-2} -to- 10^{-3} [i.e., 1 in 100 to 1 in 1000 are transported into the plasma].

III. DISRUPTIONS, RUNAWAY ELECTRONS AND FAST PLASMA SHUTDOWN

A. Disruption and VDEs

Disruptions of ITER plasmas containing ~ 1 GJ of both thermal and magnetic energy (Table I) produce large forces and stresses on all in-vessel components and are the determining factor in their structural design. The design must allow for the thermal and electromagnetic loading effects of major disruptions and the vertical displacement events (VDEs) that will follow these disruptions. Disruptions can be grouped into three classes:² Case I—'Radial Disruption', involving a rapid thermal quench (~ 1 –10 ms) followed by a rapid current quench (10–25 ms); Case II—'Vertical Disruption', involving a rapid thermal quench (~ 1 –10 ms) followed by a slow current quench (25–300 ms), with an associated vertical drift phase (VDE); and Case III—'VDE', involving an initial slow vertical drift phase and onset of plasma-wall contact (50–500 ms) followed by onset of a disruption and/or rapid loss of remaining plasma thermal energy, beginning when a plasma edge safety factor of ~ 1.5 is reached. Both Case II and Case III will result in contact of the plasma with the first wall and the onset of appreciable (up to 8 MA) poloidal 'halo' currents in the in-vessel structures, resulting in $I_{\text{halo}} \times B$ forces in the structures. To minimize the number of Case II and III disruptions, impurity pellets ('killer pellets') can be injected to convert them into Case I disruptions. Design basis specifications for VDEs and their associated halo currents are given in Fig. 1. In Fig. 1 displays the toroidal peaking factor vs halo current envelope for Cat I, II, and III events/load conditions.

Figure 1. Summary of halo current data (see ref. 2) and recommended load condition correlation and domains.

B. Runaway Electrons

The runaway electrons (Table V) produced during plasma disruption can cause a serious damage of the plasma facing components in ITER. The issue is whether the runaway electrons can cause (in one disruption) a damage to a large number (or all) of blanket modules and result in an accident accompanied with large rate of the water leak.

Table V. Parameters of the runaway electrons produced during plasma disruption in ITER

Pre disruption current	I [MA]	21
Predicted runaway current	I_{run} [MA]	12
Energy spectrum [$E_0 = 12.5$ MeV]		$\exp(-E/E_0)$
Internal inductance	l_i	1
Lifetime (neglecting wall interaction)	[s]	10
Time to hit ITER conducting wall	[s]	1
Total energy in runaways	W_r [MJ]	30
Diameter of runaway current channel	d_{run} [m]	~ 4
Runaway energy flux on first wall	[MJ/m ²]	5

In Table V, runaway heat load on the first wall is estimated under the assumption that plasma is moving vertically and sweeping the first wall (FW) over a short poloidal distance (~ 0.5 m). This estimation gives a typical but not the maximum possible heat load. If the plasma moves slowly and radially, touching the FW at the same poloidal position, the local heat load could be as high as 100 MJ/m² and the total energy of runaway electron is sufficient to melt a narrow ditch along toroidal direction in all blanket modules with the depth about 1 cm. However, this is only possible if the first wall/blanket modules are perfectly aligned in toroidal direction.

Estimates of heat load and probability:¹¹ The angle between escaping runaway electron and the first wall can be estimated from $\alpha = v/c = \Delta/2\pi Rq$, where v is the velocity of electrons normal to the FW, Δ is the thickness of runaway SOL. Poloidal width of wetted area: $h \sim (2a\Delta)^{0.5}$ assuming worst case that plasma does not move vertically and poloidal width of the wetted area is defined by SOL thickness. Estimates of the minimum and maximum values are given below:¹¹

minimum	maximum
$v \sim 30$ m/s plasma motion	$v \sim B_p/(4\pi nm)^{0.5}$ MHD instabilities
$\alpha \sim 10^{-7}$	$\alpha \sim 10^{-3}$
$\Delta \sim 1.5 \cdot 10^{-3}$ cm [$h \sim 0.9$ cm]	$\Delta \sim 15$ 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 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/m². 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:¹¹

$$P(n) = \left(\frac{\Delta}{\delta_w}\right)^n \frac{N!}{n!(N-n)!} \sum_{j=1}^{\delta/\Delta-1} (1 - j\Delta/\delta_w)^{N-n}$$

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/m² can be reached only at very small α 's. The probability that more than 10 modules will be wetted by runaway electrons is very low $p < 10^{-6}$. Although the safety credit can not be taken for random misalignment, it is possible to make a deliberate first wall modulation in toroidal direction (by placing a few blanket modules by $\sim 3-5$ cm closer to the plasma than the rest of them) to avoid accident with all blanket modules.

C. Fast Plasma Shutdown--Killer Pellets

The requirements for the fast shutdown system for ITER is given in Table VI. An important application of the ITER fast shutdown system is mitigation of plasma disruptions. Injection of impurities ('killer pellets') was proposed as a candidate for the fast plasma shutdown in ITER.^{1,2} Impurity radiation can provide the required termination time and distribute the plasma thermal and magnetic energy over the large first wall surface area. Modeling (see ref. 12) of the fast termination of the ITER plasma shows that impurity pellet injection can remove the plasma thermal energy and reduce the plasma current sufficiently during the required time. However, the presence of high-Z impurities in the plasma leads to the formation of large runaway currents (50-75% of initial current) -- potentially dangerous for the first wall. A possible solution may be massive injection of a low-Z (D or He) which makes the plasma less susceptible to runaway formation and at the same time can provide the required termination times -- required amount of D is 20-100 g.

Table VI. Fast plasma shutdown requirements

	Minimum	Maximum
Thermal quench time	0.01 (Melting of the surface layer of the first wall)	3 (To avoid local melting of the PFCs)
Current quench time	0.02 (Large mechanical loads on blanket)	3 (Upper estimate for VDE time)

IV. PLASMA RESPONSE TO EX-VESSEL LOCA

In an accident event where the cooling system of the first wall (FW), 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 $< 1000^\circ\text{C}$] in ex-vessel LOCA. Plasma response has been investigated using various methods and models, ranging from use of simple estimates,^{5,13} to consideration of diffusion models (Bohm and neoclassical diffusion),¹⁴ to use of hybrid code SAFALY.⁶ From a simple analysis, one needs the following for plasma termination in ITER:⁵

Needed Be density [determined from power balance]	$2 \times 10^{19} \text{ m}^{-3}$ [$P_\alpha \sim P_{\text{core-rad}}$, assuming perfect confinement]
Total number of Be needed in the plasma	$2 \times 10^{19} \times \text{Volume} \sim 4 \times 10^{22}$
Probability/retention (R)	0.01 [1 out of 100 Be particles transported to plasma]
Total number Be needed to be released from FW	4×10^{24} Be atoms
FW surface area affected by ex-vessel LOCA	$\sim 300 \text{ m}^2$
Residence time of Be	$\sim \tau_p \sim 5-10 \text{ s}$
Needed Evaporation Flux	$10^{21} \text{ atoms/m}^2/\text{s}$

Required evaporation flux ($10^{21} \text{ atoms/m}^2/\text{s}$) is equivalent to needed evaporation rate of $\sim 340 \text{ mm/year}$, corresponding to first wall temperature of more than 1000°C . Similar results¹³ are summarized in Fig. 3.

Figure 3: Beryllium concentration inside the plasma due to evaporation of beryllium. Assumed FW area is 300 m^2 . R is the fraction of evaporated beryllium atoms that are transported into the core plasma. The residence time of a Be atom inside the plasma was assumed as 5 seconds.

Results from Fig. 3 indicate 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} (fraction of evaporated Be atoms transported into the core plasma). It will be difficult to limit FW temperatures <1000°C.

Note that the evaporated Be are 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.^{xx} Energy losses in the SOL will result in radiation collapse (disruption), and/or loss of H-mode (return to L-mode).

Plasma response to ex-vessel LOCA in the first wall (FW) cooling system is also investigated using a hybrid code, SAFALY. SAFALY treats the interaction of the plasma with plasma facing components (PFCs). The initial conditions in the simulations were:

Plasma: nominal ITER ignited plasma (1.5 GW)
Retention probability: 0.01 for FW, 0.001 for DV
Structure : FW – Be (a part of baffle is W)
BL – SS316 (coolant, H₂O)
DV – CFC and W as PFC

Coolant of FW: 185° C, 20 kW/m²/K

Coolant data during LOCA: from an hydraulic code.

In the calculation, no burn control is imposed (fueling rate keeps the steady state) and the LOCA occurs at 1 s after the simulation starts. Time evolution of plasma and structure parameters are displayed in Figs. 4 and 5. The plasma is terminated passively at about 180 s after the LOCA by a combination of density limit disruption and power balance failure, Fig. 4. At about 150 s, the surface temperature is about 900°C (Fig. 5) and Be concentration in the plasma start to become significant (Fig. 4). Here, the time delay of impurity retention by the energy condiment time is assumed. The surface temperature just before the disruption is about 1100°C and the coolant tube temperature is about the same.

Figure 4. Time evolution of plasma parameters after ex-vessel LOCA in FW coolant.

Figure 5. Time evolution of structure parameters after ex-vessel LOCA in FW coolant.

The coolant tube of FW is made by SS316 and the tube will not melt at the temperatures shown in Fig.5 (~1000°C). But, Copper is used as a heat sink at around the tube, so the heat sink could melt. During the accident, the surface temperature of divertor also increases due to radiation increase from FW, but it is not as serious. As to divertor LOCA, maybe, we can not expect the passive shutdown, because the EDA reference adopts W and CFC as PFC material of the divertor.

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