# Characterization of Runaway Electrons in ITER

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**Abstract**. Progress in development of the integrated simulator of current quench (CQ) with runaway electrons (RE) is presented. First results on expected characteristics of REs in ITER are as follows. Significant fraction of plasma current can be converted in RE current in ITER CQ phase of the disruption. Crucial role of the impurities and neutrals in this conversion has been demonstrated. Overwhelming majority of VDE with RE are uncontrollable in ITER and accompanied by the high heat loads on the FW. The distribution of the loads is weakly affected by MHD perturbations. Efficiency of repetitive gas injection (RGI) as RE mitigation technique was demonstrated with use of the simplified model. Nonlinear MHD stability analysis of the plasma with RE and associated radial transport of relativistic electrons needs further development.

## 1. Introduction

Previous modeling of Runaway Electrons (RE) has shown that almost all disruptions in ITER will be accompanied by a formation of massive RE current. Relativistic electrons with the electron energy of a few tens of MeV and the total current up to 10 MA can result in a significant reduction of a life time of plasma facing components in ITER and, therefore, RE must be suppressed or mitigated. The main mechanism of RE generation in a high current tokamak such as ITER will be RE avalanche. This phenomenon has been studied earlier theoretically [1,2,3] and experimentally [4,5, and references therein]. It is widely appreciated that current quench (CQ) dynamics, accompanied by conversion of the plasma current into runaway electron current, is governed by the complex interplay of various physical phenomena including the effects of MHD activity, plasma motion and deformation, effect of strong impurity and neutral influx into the after thermal quench plasma, influence of ITER passive structures and active control systems. Therefore, predictive modeling of this stage of the discharge requires an integrated approach [4,5]. Progress in development of the integrated simulator of CQ with runaways is presented in Section 2. Simulation results on the runaway current dependence on the plasma parameters, impurity content, controllability of VDE with runaways, loss and distribution of RE over ITER First Wall along with preliminary results of simplified modeling of the RE mitigation by means of repetitive gas injection are reported in Section 3. Summary and conclusions are given in Section 4.

## 2. Simulation approach

Well validated DINA code [3,6] is used as an integrating core for the CQ simulator development. Whenever possible the DINA results are verified by the ASTRA code [7] simulations. In the course of the study currently employed in the DINA and ASTRA model of the RE current dynamics based on the [1] can be substituted by a newly developed kinetic module similar to the ARENA code [2] calculating the evolution of the RE distribution function in a realistic ITER equilibrium. Impurity charge state dynamics, radiation and transport are calculated by the ZIMPUR code [8]. Analysis of the MHD stability of the plasma with runaway electrons, pellets etc. is performed with use of the KINX [9] – ideal MHD stability code and DELTAPCYL code [10] - to estimate growth rates and amplitudes of the tearing modes. The orbit Following Monte Carlo code DRIFT [11] was recently upgraded [12] to allow calculations of the drift orbits in the presence of arbitrary 3D perturbations. This code has been modified for calculating the relativistic electron trajectories and then used for evaluation of the distribution of the lost REs over the ITER first wall as well as for estimating the values of the radial diffusion coefficient associated with MHD activity with the given spectrum and amplitudes. All numerical codes in this study are based on free boundary equilibria calculated with the use of the SPIDER code.

New Monte Carlo solver for the kinetic equation governing the formation and evolution of RE has been recently developed. Algorithms used in the new code are basically the same as those of the ARENA code [2]. Principal advantage of the new code is in the proper accounting for the realistic geometry of ITER equilibrium in bounce averaging of the kinetic equation. New kinetic module calculates the evolution of 2D distribution (in velocity space energy-magnetic moment) function of REs. In the near future the kinetic module should be integrated with DINA code for selfconsistent treatment of RE dynamics and evolution of the parallel electric field. At the present the simplest equation of the form $E = E_c + (j_0 - j_{re})/\sigma$ is used, where $E_c = (4\pi e^3 n_e \ln\Lambda)/(mc^2)$, $j_0$ and $j_{re}$ are total plasma and RE currents, respectively and $\sigma$ is the plasma conductivity. This, along with the switching off the radial transport of REs and neglecting the finite RE orbit width in the present version of the module, make the radial variable in kinetic module to be just a parameter determining the amplitudes of the coefficients in kinetic equation. Also, to speed up kinetic calculations, the lower electron energy limit is set to cut off thermal component and consider relativistic part of the distribution only.

Energy and pitch angle spectra of RE's have been calculated by the kinetic code for typical scenarios of plasma disruption. It has been confirmed that, similarly to findings of [1,2,13] the energy spectrum of RE electrons is not sensitive to the plasma parameters such as plasma resistance and loop voltage at high plasma current when avalanche is a dominant mechanism of RE multiplication. Indeed both acceleration and avalanche multiplication time (and energy loss on avalanche) are proportional to toroidal electric field and, hence, the spectrum should be insensitive to the strength of the electric field. In the relativistic energy range RE form beam like distribution with mean velocity directed along the accelerating electrical field. As expected kinetic calculations approved reliability of theoretical results of [1] in description of RE current growth rate due to avalanche mechanism, average energy of the REs in the range 10-30MeV and total RE kinetic energy content of few tens MJ, which is proportional to RE current amplitude. Then the details of RE distribution function in the present study were used at the final stage of analysis only, when FW loading due to RE loss was calculated. While for CQ scenario analysis we relied on the Rosenbluth-Putvinski formula [1] for the RE current growth rate.

Detailed 1.5D time dependent transport simulations of the bulk plasma responses to disruption, taking into account the runaway electron generation and the impurity dynamics and radiation, were performed with the ASTRA transport code [7]. The basic set of transport equations included the equations for electron density $n_e$, electron $T_e$ and ion $T_i$ temperatures and poloidal flux $\psi$. For simulation of the plasma equilibrium we used 2D Grad-Shafranov equation solved in ASTRA in 3-momentum approximation with the fixed boundary.

Modeling of impurity behaviour was performed by the ZIMPUR code [8] simulating the transport, dynamics of the charge states and radiation of impurities. Dynamics of the concentrations of impurity ions, $n_k$, in various charge states $k$, ($0 \le k \le Z$) was described by the set of equations:

$$\frac{\partial (V' n_k)}{\partial t} + \frac{\partial}{\partial \rho}\left[V' < (\nabla\rho)^2 > \Gamma_k\right] = V' n_e \{ I_{k-1} n_{k-1} - (I_k + R_k) n_k + R_{k+1} n_{k+1} \}, \qquad (1)$$

where $\rho$ is the radial coordinate, $V(\rho)$ is the volume inside the magnetic surface, $\Gamma_k$ are the radial fluxes of particles, $I_k$ is ionization rate, and $R_k$ consists of the sum of radiation and dielectronic recombination rates, which is enlarged on the value $R_k^{cx} = <\sigma V>_k^{cx}\ n^0/n_e$ describing the charge exchange of impurity ions with hydrogen isotope atoms. The radiation of each sort of impurity was determined by the summation over all possible charge states, $P_Z = \Sigma L_k n_k$, where coefficients $L_k$ include bremstrahlung, linear and recombination radiations. It was assumed that electron temperature could not be lower than 2 eV, below which the radiation data were assumed to be not completely reliable.

The relations between impurity ion fluxes $\Gamma_k^x$ and ion densities $n_k^x$ at the plasma surface $\rho_n = \rho/\rho_{max} = 1$ were used as the boundary conditions, $\Gamma_k^x = V_\perp n_k^x$. The value $V_\perp$ was taken to be equal to the velocity of the plasma electrons escaping the plasma column and was calculated with the electron density equation. The impurity source was defined as the impurity neutral flux on the plasma boundary. The value of this source was fitted to provide the desirable impurity contamination in plasma. Impurity flux to the plasma column boundary was fitted to produce the necessary impurity contamination at the start of simulations and then kept fixed.

For present modeling the DINA code calculated two-dimensional free boundary plasma equilibrium equation in external magnetic fields together with one-dimensional (averaged on magnetic surfaces) set of transport equations for electron and ion energy and diffusion equation for the poloidal magnetic flux. Halo currents generation was not taken into account. Circuit equations were solved for the vacuum vessel, passive structures and active coils currents. DINA simulations were performed with poloidal field system and passive structures corresponding to the "Baseline 2010" [14]. Poloidal coils were assumed to be short circuited during VDE. Present modeling did not include the effect of plasma current density flattening in case of decrease of the value of safety factor in plasma axis below 1 (no sawtooth model) during runaway generation.

RE population growth rate in ASTRA-DINA modeling takes the form

$$\frac{dn_r}{dt} = \left(\frac{dn_r}{dt}\right)_D + \left(\frac{dn_r}{dt}\right)_A + \left(\frac{dn_r}{dt}\right)_T, \qquad (2)$$

where the first term in r.h.s of eq. (2) corresponds to the Dreicer generation, second one – to the avalanche due to close collisions and the third one is a RE source due to Tritium decay, $\left(\frac{dn_r}{dt}\right)_T = \frac{n_T}{2\tau_\beta} f_\parallel$, where $n_T$ is the Tritium density, $\tau_\beta = 17.7$ years and $f_\parallel$ is a passing electron fraction. Direct estimation shows that at $n_T = 0.5\ 10^{20}\mathrm{m}^{-3}$ the source rate due to Tritium decay is $ec(dn_r/dt)_T \approx 2\ \mathrm{A\ m^{-2}\ s^{-1}}$. This rate is sufficiently small, and can be important only when "standard" (Dreicer) mechanism is strongly suppressed, i.e. in the low temperature high density plasma. Seed RE current due to relict tail of relativistic electrons was neglected in the present modeling as we have no theoretical background to assume better confinement of the "fast" electrons compared to that of bulk plasma ones at the thermal quench phase of disruption.

ASTRA and DINA models have been carefully cross-benchmarked, showing perfect agreement at the model assumptions for circular and elongated plasmas, and verified against representative JET discharge (shot #11050 [15, 16]) with major disruption provoked by intense gas puff to achieve the density limit conditions and disruption initiation. Plasma current behaviour during current quench in that shot has a plateau together with intense bursts of the hard X-rays indicating the runaway generation [16]. Acceptable agreement between the experimental and modelled plasma current time evolution was obtained with use of electron temperature values 10 eV and electron density – $2.24{\cdot}10^{19}m^{-3}$. Simulations of JET experiments showed that the low temperature could be achieved in plasma either due to very high anomalous energy losses with the effective coefficient of ~ $10^4$-$10^5$ $m^2s^{-1}$ or due to strong plasma impurity radiation. Since we have no reasons to assume very high thermal conductivity, we consider the impurity radiation to be the main mechanism of the energy losses after TQ in this report.

## 3. Simulation results

Series of ASTRA-DINA simulations were done to examine CQ characteristics and RE generation depending on ITER after thermal quench plasma parameters and impurity content. Control ability of VDE with REs and distribution of the FW loads due to RE loss at VDE were studied. Preliminary analysis of the RE mitigation technique based on the repetitive gas injection scheme proposed in [17] was performed. The details of the calculations are given in the following subsections.

### *3.1 Sensitivity study of runaway electron formation*

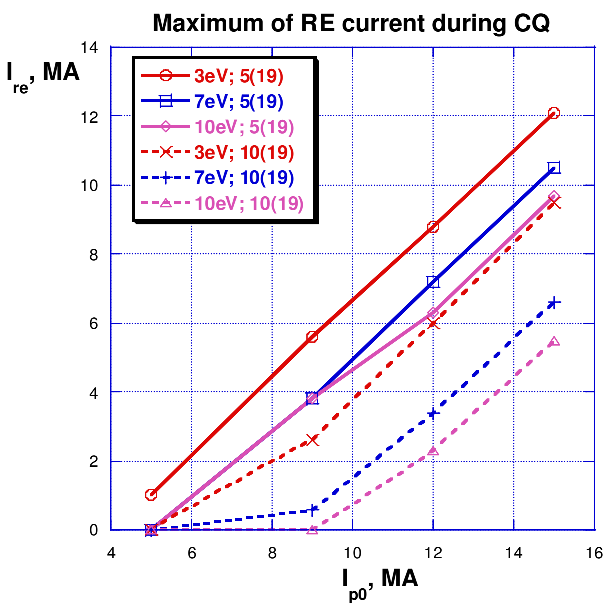


**Fig. 1.** Maximum RE current as function of plasma current before current quench for initial current profile with $l_i$=1. Curves correspond to different electron temperature and electron density of disruptive plasma

Sensitivity study of the plasma current conversion into the RE current has been done with use of the DINA code under simplified assumptions on plasma density and temperature profiles, which were assumed to be flat and constant in time. Parameters of background plasma have been varied in the following levels: $\boldsymbol{n_e}$ – $5{\cdot}10^{19}$, $10{\cdot}10^{19}$ $m^{-3}$; $\boldsymbol{T_e}$ – 3, 7, 10 eV; plasma current before thermal quench $\boldsymbol{I_{p0}}$ – 5, 9, 12, 15 MA.

Initial plasma current profile corresponds to the internal plasma inductance value equal $l_i(3)$~ 1.

It was shown that the avalanche of runaway electrons results in a generation of massive RE current in ITER even during disruptions of discharges with plasma current significantly lower then 15 MA.

Figure 1 shows maximum RE current (assuming no VDE and no loss of RE's) as function of initial plasma current before a start of the current quench. Curves correspond to different assumptions about plasma temperature and density. One can see that significant RE's are expected during disruption with plasma current > 6-8 MA. If the value of plasma current before thermal quench is smaller than 5 MA there is no runaway generation during the current quench for parameters of background plasma $T_e \geq 7$ eV and $n_e \geq 5 \cdot 10^{19} m^{-3}$. Calculations have demonstrated that if $I_{p0} < 12$ MA the runaway current profile is very picked. In that case the value of $l_i$ can be more than 3.

### *3.2 Controlability of VDE with REs*

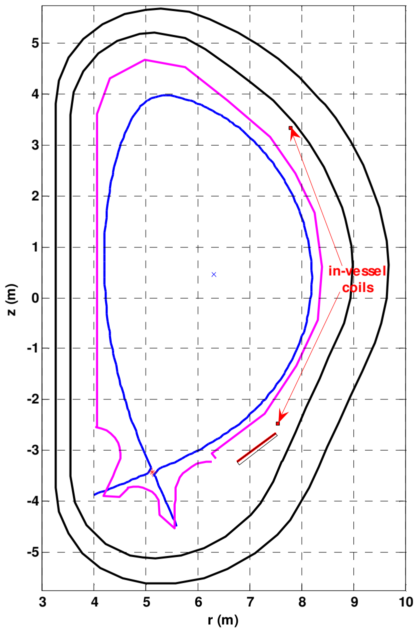


**Fig. 2**. Plasma configuration at SOF ($l_i$ = 0.7, $\beta_p$ = 0.1) used for the study of plasma vertical stabilization by in-vessel coils during disruption with REs.

Simulations of plasma equilibrium control of RE discharges suggested earlier as a mean to mitigate RE heat loads on the wall have been carried out to investigate controllability of the RE discharges after major disruptions.

Current drop of few MA in a short time results in fast plasma vertical displacement which has to be stopped by ITER in-vessel coils (see Fig. 2) used for the plasma vertical stabilization with feedback control. The upper and lower coils are connected to one power supply in such a way to produce in the plasma region radial magnetic field. The maximum voltage produced by the power supply of the coils is 0.5 kV/t and the maximum current is 240 kAt. The inputs of the feedback controller for the plasma vertical stabilization include the derivative of the plasma vertical position and in-vessel coil current, the output is the control voltage applied to the in-vessel coils with the delay of a few milliseconds according to the power supply model.

Modelling has shown that ITER system for vertical stabilization is capable to control only limited range of RE current profiles. In particular, it was found that critical value of plasma current drop is $\Delta I_p \approx 4$ MA (from 15 MA to 11 MA during 80 ms). At the plasma current drop of $\Delta I_p \approx 5$ MA with the same rate the vertical position control is lost because the passive vertical stabilization deteriorates significantly due to the faster plasma displacement inward. This system is not sufficient to ensure reliable control of post thermal quench plasmas with RE's. Therefore ITER should rely on RE suppression rather than their control.

### *3.3 RE radial transport and loss to the FW*

Analysis of the RE loss and of the associated heat load distribution over ITER first wall has been done with use of modified DRIFT code [11,12]. Evolution of equilibrium configuration during VDE was evaluated by DINA code. Relativistic electron orbits were calculated in the presence of external 3D magnetic perturbations due to ELM control coils, ripples, NBI ports and test blanket modules (TBMs). Algorithm of FW load calculations is described in details in [12]. Sequential plasma configurations and corresponding incident positions of the RE orbits at the FW are shown at the Fig.3. Main results of the calculations can be summarized as follows. The distribution of the heat loads over FW is determined first of all by the position of the plasma with RE relative to the wall. 3D wall shaping provides toroidal peaking factor ~ 3. Additional toroidal peaking factor of about 2 appears due to RMP perturbation (reference RMP currents for inductive reference scenario provide m/n=3/2 to be the dominant harmonic, i.e. additional toroidal peaking repeats the dominant mode structure in the vicinity of the FW). Ripples, TBMs and NBI ports weakly contribute to the toroidal peaking of the loads. RE loss to the wall results in extremely high heat loads (of several MJ/m2) and, therefore, VDE with REs should be avoided for safety ITER operation.

Radial transport of the REs due to magnetic perturbation was found to be extremely high, ($D>100m^2/s$) in fully stochastic fields only. Our preliminary analysis has shown that amplitude of D is independent of E ( for E>1MeV), radial profile of D better coincides with magnetic surface structure at lower RE energy (orbit width effect), D decreases with grows of magnetic moment μ, but still remains rather high (from $10^2$ to $10^1$ $m^2/s$).

RMP perturbations have a limited applicability as a tool to be used for suppression of RE generation. Variation of RMP coil currents do change the positions and widths of stochastic regions inside the plasma, however this structure is more sensitive to the equilibrium current profile (which is hardly controllable at the current quench phase) than to the distribution of the external currents in RMP coils. In all numerous variants of equilibria and RMP currents considered narrow stochastic regions were always separated by the layers with regular magnetic surfaces. Additional study is necessary to develop an effective and consistent with equilibrium evolution model for the RE radial transport for kinetic code.

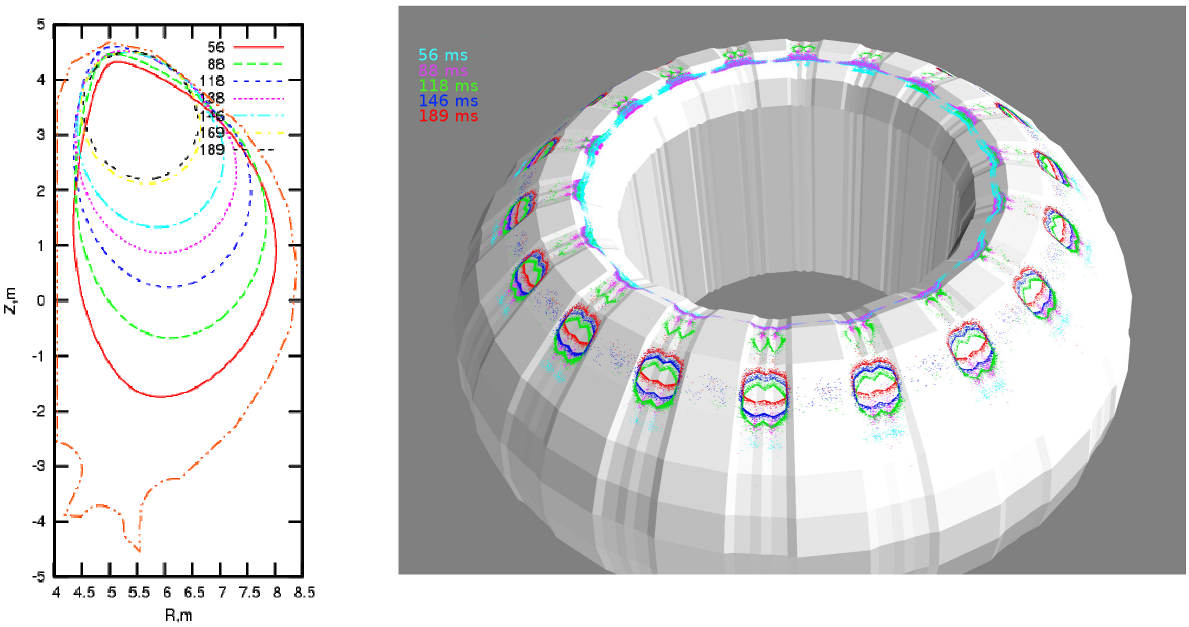


Fig.3 Plasma boundary (left) and incident point positions (right) for the lost RE at different time slices of VDE

### *3.4 Impurity effect on RE formation*

In Subsection 3.1 plasma temperature and density was considered to be time independent. Here we present results of selfconsistent modeling where evolution of the plasma profiles is governed by power balance equations including impurity radiation effect. The first wall in ITER is covered by Beryllium (Be). That is why we use Be as the main impurity in the modelling. Since there is no good model of plasma-wall interaction in the course of disruption the impurity contamination was set as a parameter. Our simulations show that it is difficult to reach the plasma temperatures lower than 100 eV in the central plasma region with a single Be impurity due to low Be radiation. It results in very slow CQ and in very weak RE generation. Even for sufficiently high Be contamination (~12%) which provides about half electron density it was found that Be is fully ionized in the central plasma regions and the radiated power is not sufficient for effective cooling the plasma. Electron temperature in the plasma core in this case was found to be higher than 100eV, provided small plasma resistance and suppression of RE generation. Be radiation was high at the plasma column periphery resulting in current channel contraction simultaneously with the reduction of the plasma current. These simulations showed that Be coating of ITER wall helps to prevent RE generation after the major disruption in this device. Cooling of the plasma periphery can help to reduce plasma-wall interaction due to the VDE. However, the radiation of Be can be significantly increased due to charge exchange with hydrogen isotope neutrals coming into plasma from the wall after the thermal quench. Essential effect of the neutrals was seen at rather high neutral density (Fig.4). At low Be density RE generation is suppressed by insufficient plasma cooling. At higher density suppression of avalanche multiplication is accompanied by the shortening of the CQ phase.

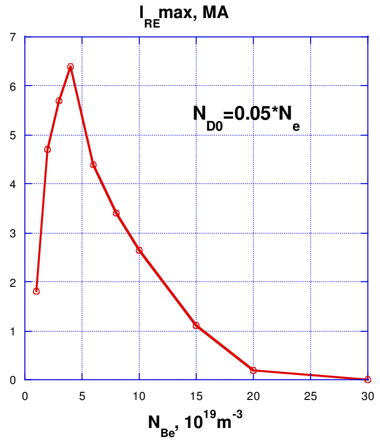


**Fig.4** $I_{re}^{max}$ vs. Be concentration at $n_D^0$=5%$n_e$

On the other hand, high-Z impurities can cause high radiation in the plasma core, cooling the plasma and inducing effective RE generation. DINA simulation of CQ with VDE in the presence of Ar impurity (7,10 and 14% of ion density) demonstrated effective RE generation (8, 10, and 11.5MA) and increase of the CQ duration (from 90 to 150ms). For Ar concentration higher than 10% upper X-point is formed during CQ and VDE. Ne and W impurity effect on RE current generation was also studied. Rising heavy impurity density results in essential increase of the plasma current decay rate before its conversion to RE current and speed up VDE velocity.

***3.5 RE mitigation by repetitive gas injection***

DINA code in the plasma fix boundary mode was used in preliminary study of the runaway electron (RE) suppression by means of repetitive gas injection (RGI). Speed of gas (hydrogen) injection was assumed to be $V_{inj}$=1000 m/s. It was assumed that the gas injection resulted in the increase of electron density in the each magnetic surfaces along the motion path on the level of $\Delta n=2\cdot10^{19}$ m$^{-3}$. Gas injection was lasted during the time period $\Delta t_{inj}$ when the gas reached the q=2 magnetic surface. After that the both n and T profiles are mixed uniformly in whole plasma volume with amplitudes determined from the simplest balance relation $(n+\Delta n)\cdot(T-\Delta T)$=const. Time period between the repetitive gas injection shots was varied on the levels Δt= 10, 20 and 30 ms. Effect of “microdisruptions” on RE confinement when cooling front reached q=2 surface was modeled by effective rise of RE loss ($\tau_{re}$ was decreased from 50 to 1 ms). Evolution of total plasma and RE currents during RGI with Δt= 10ms is shown at the Fig. 5. Increase of the injection interval to Δt= 20 and 30 ms resulted in the increase of the duration of CQ time to 200 and 250ms, and of the maximum RE current after the first blip to 5 and 6MA, respectively. In case of 10 times higher gas jet density RE was suppressed completely after the very first bleep. CQ duration then reduced to 30ms.

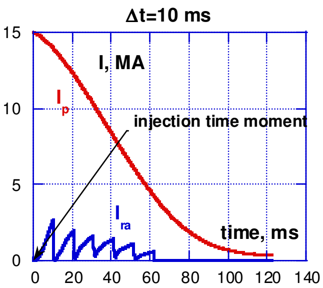


Fig.5 RE mitigation by RGI

Ideal MHD stability analysis of the plasma with RE and propagating cooling front has shown that the only ideal MHD instability that can be driven unstable for low beta plasma after thermal quench is the external peeling mode localized near the plasma boundary and driven by finite current density at the plasma edge. Internal resistive tearing mode are driven strongly unstable by the spike in current density at the cooling front. However the estimates for island width due to the spike passing through rational surfaces are quite moderate w~0.05m e.g. compared to the RMP generated islands in ITER.

## 4. Summary and conclusions

Significant fraction of plasma current can be converted in RE current in ITER CQ phase of the disruption. Crucial role of the impurities and neutrals in this conversion has been demonstrated. Overwhelming majority of VDE with RE are uncontrollable in ITER and accompanied by the high heat loads on the FW. The distribution of the loads are weakly affected by MHD perturbations unless the conducting plasma current channel shrinks due to the gas injection and the current driven modes go strongly unstable due to the spike in current density at the edge. Efficiency of RGI as RE mitigation technique was demonstrated with use of the simplified model. The nonlinear coupling of external and internal MHD modes should play a major role in the mitigation scenario. Nonlinear MHD stability analysis of the plasma with RE and associated radial transport of relativistic electrons needs further development.

The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

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