THE EVALUATION OF THE HEAT LOADING FROM STEADY, TRANSIENT AND OFF-NORMAL CONDITIONS IN ARIES POWER PLANTS*

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The heat loading on plasma facing components (PFCs) provides a critical limitation for design and operation of the first wall, divertor, and other special components. Power plants will have high power entering the scrape-off layer and transporting to the first wall and divertor. Although the design for steady heat loads is understood, the approach for transient and off-normal loading is not. The characterization of heat loads developed for ITER[1] can be applied to power plants to better develop the operating space of viable solutions and point to research focus areas.

1. Introduction

As part of a more detailed treatment of plasma facing components and the scrape-off layer plasma in ARIES power plant studies, the development of heat loading criteria is pursued. Recently a heat loading compilation was completed for ITER[1] in order to guarantee that all heating sources were treated with the most up-to-date guidance. Using this as a template, ARIES appropriate loadings were determined. The heating sources are divided into nominal or steady state, transient, and off-normal. The transient source will concentrate on ELMs and the off-normal will concentrate on disruptions. The data available are examined to provide heating, timing, deposition, and heating type for the divertor and first wall (FW). A particular power plant design is used, referred to as ACT1. Table 1 provides parameters relevant to the heat loading for the ARIES-ACT1 power plant and ITER.

2. Steady Heat Loading

The power that is available for heating the PFCs is the alpha power from fusion reactions and the injected heating/current drive (H/CD) power. Some portion of this power will be radiated from the plasma in the form of bremsstrahlung, cyclotron, and line radiation, which is emitted isotropically from locations inside the plasma, and will deposit mostly on the first wall, with a small portion entering the divertor opening. This level of radiation is intended to be controlled in order to reduce the power that is transported to the divertor, so intentional impurities are introduced (such as neon or argon). The remaining power, $P_{SOL} = P_\alpha + P_{aux} - P_{rad}$, crosses the plasma separatrix and flows toward the divertor. Some of this power can transport to the first wall, but this is generally small for steady conditions and nominal plasma-to-wall distances (>10 cm). The power decay length, λ_q, is defined at the outboard midplane (R+a, OMP), and this width expands as the magnetic field lines travel toward the X-point and into the divertor. The peak surface heat flux on the divertor target is expressed as,

$$q_{peak,div}\ (MW/m^2) = P_{SOL}\, f_{IB,OB}\, f_{vert} \times \left[(1-f_{div,rad}) / A_{IB/OB,div,cond} + f_{div,rad}/A_{IB/OB,div,rad} \right]$$

where $f_{div,rad}$ is the fraction of SOL power that is radiated in the divertor (due to partial or full detachment regimes), $f_{IB,OB}$ is the fraction of power transported to the inboard and outboard divertors (for DN this is 20%/80%, respectively), f_{vert} is the fraction of power that goes to the upper or lower divertor as a result of imbalance in the plasma vertical position (for DN this is 65%).

The $A_{div,cond}$ is the divertor conduction target area and the $A_{div,rad}$ is the area available to absorb power radiated in the divertor and is generally much larger than the conduction target area. For an outboard divertor these can be approximated based on typical designs as $2\pi(R-a/2)\lambda_q f_\psi f_{tilt}$ and as $2\pi a(R-a/2)$, respectively. On the inboard these are $2\pi(R-a)\lambda_q f_\psi f_{tilt}$ and $\pi a(R-a)$. f_ψ is the poloidal flux expansion, and f_{tilt} represents the divertor target angular tilt relative to the magnetic flux line ($1/\sin\theta$). The product of f_ψ and f_{tilt} is taken to be 10. The power decay length is determined from a correlation developed experimentally on tokamaks, mostly on JET[2], and can be considered fairly uncertain and the subject of considerable ongoing research[3],

$$\lambda_q\ (m) = 7.25\times10^{-2} q_{95}^{0.75} n_L^{0.15} / (P_{SOL}^{0.4} B_T).$$

For this discussion we will use a particular power plant design originating from systems analysis optimization. This design is considered an aggressive physics and aggressive technology extrapolation, referred to as ACT1[4]. The parameters of this tokamak are R = 6.25 m, a = 1.56 m, κ = 2.2, δ = 0.7, I_p = 10.9

MA, B_T = 6.0 T, β_N^{total} = 5.75, n/n_{Gr} = 1.0 ($n_{Gr} = I_p/\pi a^2$), q_{95} = 4.5, which generates 1800 MW of fusion power, with auxiliary H/CD power of 45 MW, ultimately making 1000 MW of electric power. The alpha power is then 360 MW, the radiated power is about 115 MW, giving a SOL power of 290 MW. These parameters and other heat load relevant parameters can be found in Table 1 along with those for ITER. Since this is a DN configuration, the corresponding peak heat flux in one of the outboard divertors is 13.5 MW/m^2, assuming a 90% radiated power fraction in the divertor, and where the power decay length at the OMP is 4.0 mm. The corresponding inboard peak heat flux is 4.7 MW/m^2.

Table 1. ARIES-ACT1 and ITER Parameters

	ARIES-ACT1	ITER
Steady/nominal		
Divertor/material	DN/W	SN/W
P_{fusion}, MW	1800	500
P_{aux}, MW	45	45
P_{rad}, MW	115	44
$f_{rad,divertor}$	0.9	0.7
λ_q, mm	4.0	5.0
P_{SOL}^{SS}, MW	290	100
$q_{div,OB,IB}^{peak}$, MW/m^2	13.5, 4.7	13.1, 8.1*
$A_{FW,OB}$, m^2	396	454
$A_{FW,IB}$, m^2	99	230
Transient/ELM		
W_{ped}, MJ	117	105
T_{ped}, keV	4.5	4.5
n_{ped}, /m^3	1.0	0.7
H_{98}	1.65	1.0
ΔW_{ELM}^{large}, MJ	24	21
ΔW_{ELM}^{small}, MJ	5.9	5.3
$A_{ELM,div}^{OB,IB}$ m^2	1.38, 1.17	1.67,1.34*
$\tau_{\|\|}$, ms	0.22	0.25
$\Delta t_{ELM,rise}$, ms	0.44	0.5
f_{ELM}	~4-20	~1-6
$q_{div,OB,IB}^{peak}$ (inter-ELM), MW/m^2	9.5, 3.3	10.5, 6.4*
$q_{div,OB,IB}^{peak}$ (ELM) , MW/m^2	4250, 0	1600, 4040*
Off-normal/disruption		
W_{th}, MJ	690	350
$A_{dis,div}^{OB,IB}$, m^2	13.8,11.7	16.7,13.4*
$A_{dis,FW,}^{OB,IB}$, m^2	198, 99	227,115*
Δt_{TQ}, ms	2.0	~1.0
$\Delta t_{TQ,full}$, ms	8.0	4.0
W_{mag}, MJ	280	520*
Δt_{CQ}, ms	25	40

*using analysis presented here

The 90% radiated power fraction in the divertor is high, although detailed analysis with UEDGE[5] using fluid neutrals indicates this is accessible in the detached regime. The heat flux on the first wall from radiated power is 0.20 MW/m^2 average, with an outboard midplane peak value of 0.30 MW/m^2. This constitutes the steady power loading and will serve as reference when we consider transient and off-normal loading. ITER has also identified other loading sources such as charge exchange neutrals, stationary MARFE's and fast alpha losses.

3. Transient Heat Loading

Although there are long timescale transients associated with power plant startup and other operational features, which have timescales long or similar compared to the PFC thermal time constants, we will concentrate on transients created by edge localized modes (ELMs), which are very fast and repetitive thermal transients. The former were reported in a separate paper[6]. ELMs are associated with the H-mode plasma confinement regime, and provide the periodic release of energy and particles from the plasma. Although this is desirable for sustaining this plasma regime, the ELMs disperse large heat and particle loads to the plasma facing surfaces. There are different types of ELMs, however, the best plasma energy confinement is generally associated with the largest ELMs. The time scale for the ELM energy pulse to reach the divertor target in our power plant is about 200-250 micro-seconds. Experiments on tokamaks have been trying to characterize the ELM heat loading, particularly in the divertor, with IR imaging, imbedded thermocouples, and Langmuir probes on JET and ASDEX-U[7-16]. It is found that the time evolution of the peak power and temperature in the divertor scale with a parallel ion transit time, $\tau_{||} = 2\pi R q_{95}/c_{s,ped}$, where c_s is the sound speed, which is about 220 micro-seconds for our power plant. The energy pulse rises over about 2 of these time scales and then drops over about 4 of these timescales, giving an asymmetric triangular waveform in time. The total energy released during an ELM is determined by calculating the stored energy in the pedestal, and using an experimental correlation for the energy released $\Delta W_{ELM}/W_{ped}$ as a function of the plasma collisionality. For our power plant the collisionality in the pedestal region is about 0.09, which gives a $\Delta W_{ELM}/W_{ped}$ of 0.15-0.2. The pedestal pressure, determined by an MHD stability model[17,18] is ~ 116-140 kPa, and using W_{ped} = $3/2 p_{ped} V_{plasma}$, we find the value for the pedestal stored energy of 97-117 MJ. With the higher value the total energy released in an ELM is 19.4-23.4 MJ. Another correlation exists with the parallel ion flow time, which gives lower values of $\Delta W_{ELM}/W_{ped}$ = 0.05-0.12, or with the lower value ΔW_{ELM} = 4.9-5.9 MJ. It is observed that 50-100% of the total ELM energy goes to the divertor, depending on the ELMs relative size $\Delta W_{ELM}/W_{ped}$, and that 20-40% of the energy that goes to the divertor appears in the rise phase, depending on plasma collisionality.

For our power plant the pedestal temperature and density are about 4.5 keV and 1.0×10^{20} /m^3 consistent with 1.5D plasma simulations, corresponding to our pedestal pressure of about ~140 kPa. Based on the above observations a calculation of the ELM energy to the divertor and first wall will be made for large ($\Delta W_{ELM}/W_{ped}$ = 0.2, ΔW_{ELM} = 23.4 MJ) and small ELMs ($\Delta W_{ELM}/W_{ped}$ = 0.05, ΔW_{ELM} = 4.9 MJ). Here we assume all the ELM energy goes to the outboard side for a DN divertor configuration. Based on experiments [11], for the large ELM we assume 50% of the total energy released goes to the divertor, 65% to each divertor, and 40% of that arrives in the shorter rise phase. The remaining 50% of the total ELM energy released goes to the first wall. For the small ELM we will assume that 80% of the total energy released goes to the divertor, 65% to each divertor, and that 20% of this energy arrives in the shorter rise phase. The remaining 20% of the total ELM energy released goes to the FW. It is found that the rise phase energy deposition contributes dominantly to the rapid temperature rise, so this fraction of the energy arriving in the divertor will be used. Because of the short deposition time we will use the semi-infinite heat conduction solution to determine the temperature rise of the PFC surface, since the component cannot conduct this energy into the bulk on this timescale. In addition, since this formulation assumes a step rise in the energy, while it is actually a ramp, a factor of 2/3 is applied based on simulations[12,13]. The expression for the temperature rise is,

$$\Delta T_{rise}\ (^oK) = 2/3\ (2\ \alpha^{1/2}\ \Delta W_{ELM}^{div,rise})\ /\ [\pi^{1/2}\ k\ A_{div,ELM}\ (2\ \tau_{||})^{1/2}],$$

$$= 2/3\ C_{material}\ \Delta W_{ELM}^{div,rise}\ /\ A_{div,ELM}\ (2\ \tau_{||})^{1/2}$$

where α is the ratio $k/\rho C_p$, k is the heat conductivity, $\Delta W_{ELM}^{div,rise}$ is the energy reaching the divertor in the rise phase, $A_{div,ELM}$ is the area of deposition on the divertor target which is considered the same as between ELMs (although we will address this later), and 2 $\tau_{||}$ is the rise phase time frame. The $C_{material}$ is 62 for tungsten at 1000^{o}C and 85 for ferritic steel at 650^{o}C, for $\Delta W_{ELM}^{div,rise}$ in MJ. SiC is the structural material for the ARIES-ACT1 design, and would therefore constitute the first wall, which has a similar $C_{material}$ as Fe (83), so those temperature rises can be used for this material. For the large ELMs and a tungsten divertor target we obtain a temperature rise of 4360 ^{o}K. For a tungsten melting temperature of ~3400 ^{o}C, we expect melting regardless of its operating temperature. For the small ELM the temperature rise is 730 ^{o}K, which would not result in melting. The typical operating temperature range considered for tungsten in power plants is ~800-1300 ^{o}C, determined by brittle behavior and recrystallization, respectively. Recent experiments[15,16] have identified that the wetted area on the divertor target plate during an ELM broadens relative to its value between ELMs. For large ELMs this broadening is a factor of ~ 4-6, while for smaller ELMs this is a factor of ~ 1.5. For the large ELM this (4x) reduces the temperature rise to 1090 ^{o}K, which would keep the material from melting.

Turning to the first wall, experiments have indicated a peaking in the heat loading to the first wall of ~ 4x, and we use only the outboard first wall area since we assume the entire ELM is exhausted to the outboard side. The outboard first wall area is 396 m^2, which becomes 99 m^2 with peaking. Using the same formulation as for the divertor, with changes to the deposition area and energy, while taking all of the energy to arrive in the 3 x (2 $\tau_{||}$) time scale, and removing the 2/3, the large ELM results in a temperature rise of 203 ^{o}K for tungsten, and 278 ^{o}K for Fe (SiC), while small ELMs would have temperature increases of 17 ^{o}K for tungsten and 23 ^{o}K for Fe (SiC). The temperature rises for tungsten are below melting for typical operating temperatures, as are those for Fe, representing a low activation ferritic steel. Although the melting temperature of ferritic steel is ~ 1500 ^{o}C, the expected operating temperature range for low activation ferritic steel is 550-650 ^{o}C, depending on the alloy and its radiation resistant modifications. The melting (decomposition) temperature of SiC is ~ 2400 ^{o}C, and the operating temperature is 1000^{o}C. Depending on the thermal cycling, an armor layer may be preferable on the FW, whether Fe steel or tungsten, to accommodate material losses from erosion or cracking. A design concept has been developed incorporating tungsten plugs that penetrate through the ferritic steel or SiC, straight to the first wall coolant, providing a parallel conductivity path[6]. Simultaneously, the volume of tungsten or Fe steel must be minimized in order to have the least impact on tritium breeding.

The transient ELM loading has additional features that should be considered. The ELM frequency is derived from a correlation, from tokamak experiments[9], that gives $f_{ELM}\Delta W_{ELM} \sim 0.2\text{-}0.4 \times P_{SOL}$, giving for our power plant example a frequency of about 3.7 /s for large ELMs and 18 /s for small ELMs. These frequencies lead to $1\text{-}6\times10^{8}$ cycles in a year (which represents a typical time frame for power plants between routine maintenance). Even if melting is avoided, this many cycles may lead to crack growth in the region with the temperature rise. Experiments on the cycling behavior are ongoing, but indicate that cracks may appear even without melting after 10^{5} cycles[19,20]. Operating above the DBTT appears to be required to minimize cracking, but only very low energy pulses may be tolerable with little to no cracking. If melting is present, but can be tolerated due to benign material movement and very shallow melt layers, it may still accelerate material losses. The particle loading during ELMs is much more poorly diagnosed, but what has been observed is a nearly constant fractional release of the pedestal density $\triangle N_{ELM}/N_{ped}$ of about 4%, over a wide range of plasma collisionalities. This gives about 2.2×10^{21} particles released per ELM for our power plant. Whether there is peaking is not known. In addition, the ELM loading on the first wall is distributed along magnetic field lines, essentially in bands, that randomly shift around with each ELM, indicating that the same location on the first wall will not necessarily see the next ELM. There is also variability in ELM parameters, such as for the released ELM energy, the average value +/-10% occurs 87% of the time, +/-20% occur about 30% of the time, and +/-40% can occur about 5-10% of the time[10]. Finally, in addition to the ELM itself there is a period called the inter-ELM phase, where power is also exhausted from the plasma. The steady state power exhaust would be an average over a full ELM cycle, and is normally reported experimentally as an average over many ELM cycles. The power released continuously during the inter-ELM period is given by,

$$P_{SOL}^{inter\text{-}ELM} = [\ P_{SOL}^{SS}(\Delta t_{ELM} + \Delta t_{inter\text{-}ELM}) - \Delta W_{ELM}\ ] / \Delta t_{inter\text{-}ELM}$$

where for our large ELM power plant example, $\Delta t_{ELM} = 1.3$ ms, $\Delta t_{inter\text{-}ELM} = 0.27$ s, $P_{SOL}^{SS} = 290$ MW, and $\Delta W_{ELM} = 23.4$ MJ, giving an inter-ELM power exhaust of 205 MW, leading to an outboard divertor peak heat flux of 9.5 MW/m^2 (including radiation). This can be compared to 13.5 MW/m^2 determined for a steady state heat load, so the ELMs are reducing the power output of the plasma over the long inter-ELM periods, by releasing energy periodically in short bursts with a heat flux of 4250 MW/m^2.

4. Off-Normal Heat Loading

The presence of off-normal heat loading is generally attributed to major disruptions, which are due to rapid loss of the plasma stored energy, followed by the quenching of the plasma current. There are minor disruptions, which are losses of confinement from a high to a low regime, but do not end in the destruction of the plasma configuration. Major disruptions will be the subject of this section. The two disruption scenarios appropriate for a power plant are the vertical displacement event (VDE) and the midplane disruption (MD). In the former, the plasma drifts vertically due to a loss of control, makes contact with the first wall transitioning from a high to a low confinement plasma regime, followed by a complete loss of the plasma stored energy (thermal quench, TQ) when the edge safety factor reaches about 2. In the MD the plasma loses all its stored energy (TQ) while in its nominal location at full size and shape. In both cases this is followed by a current quench (CQ), which induces the largest toroidal eddy currents in the conducting structures in FW, blanket, shield, and vacuum vessel, and halo currents between the plasma and structures in the poloidal direction. Poloidal currents will also be induced by the rapid decreases in the plasma stored energies. Tokamak experiments have been pursuing the identification of power flows during disruptions to provide guidance for ITER[21-29].

For our disruption types the fraction of the plasma stored energy that is released in the thermal quench ranges from 65-100%. VDEs have a release of about half the stored energy at wall contact over energy confinement time scales of 1-2 s, and then the rest is lost in the TQ, while in the MD it can be assumed that up to 100% of the plasma stored energy is lost in the TQ, particularly for our aggressive plasma configurations[21]. The timescale for the TQ has been correlated with plasma volume, giving a range of $\Delta t_{TQ} = 1.5\text{-}2.75$ ms[26]. Just like ELMs, the heat load has an asymmetric triangle waveform, rising in about Δt_{TQ}, and falling in ~2-4 of these time scales. This time scale can vary widely even within the same device due to complex dynamics of the thermal quench, which can occur in steps rather than in one drop. JET experiments indicate that about 10-50% of the energy released in the thermal quench can go to the divertor, the remainder will be assumed to end up on the first wall. Some small fraction (15%) of energy has been observed to be radiated from the plasma during the TQ with a peaking factor of 3.5x. About 25% of the energy released arrives in the rise phase Δt_{TQ}, with the remaining 75% arriving over 2-4 Δt_{TQ}. The deposition area on the divertor target is observed to expand, beyond the steady heat load area, during the thermal quench by large factors of 5-10x.

Using the prescriptions derived from tokamak experiments the heat loading in the divertor and on the first wall will be approximated. The plasma stored energy is 690 MJ, so for the MD the thermal quench releases all of this, while for the VDE, the thermal quench only releases about half of this. For the MD we have 69-345 MJ going to the divertor, and correspondingly 621-345 MJ going to the first wall. It is possible to have up to 15% radiated, which would end up on the first wall, but we will not consider this here. The divertor target area to receive the energy is 1.38 m^2 with no expansion, and 7-14 m^2 with. The area of the outboard first wall to receive the energy is 396 m^2, and with a peaking factor of 2x, we reduce this to 198 m^2. Since the heat fluxes are so large, reaching levels where melting is anticipated, the timescale to be used in the expression for the temperature rise should be the full duration of the "above melting threshold", and so for these calculations we use $4x\Delta t_{TQ}$ for the duration and remove the 2/3 factor associated with the rise phase[26]. Using the same semi-infinite expression for the temperature rise, we find values in the divertor (assuming 10x expansion of deposition area, Δt_{TQ} = 2 ms, and 65% to each divertor) of 2250-11260 ^oK for tungsten, well beyond melting at the high end, while just reaching melting at the low end. On the first wall (outboard only) the temperature rise values are 1210-2170 ^oK for tungsten and 1660-2980^oK for Fe (SiC). For the VDE disruption these temperature rises can be divided by 2, the other half of their stored energy is released over a longer time scale (a plasma energy confinement time, ~ 1-2 s) prior to the thermal quench, most likely elevating the divertor and FW temperatures above their normal operating temperature, making melting a difficult situation to avoid for tungsten or ferritic steel.

The current quench phase is where the plasma releases its magnetic energy ($\sim 1/2 L_{int} I_p^2 + 0.2 \times 1/2 L_{ext} I_p^2$), causing the plasma current to decrease. Here we describe the magnetic energy available as the energy internal to the plasma plus an additional amount to account for the external inductive energy between the plasma and the vacuum vessel (or other primary conducting boundary). The time scale for the CQ is 15-25 ms, based on a tokamak database relating $\Delta t_{CQ}/A_{p,cross}$ (plasma cross-sectional area, $\pi a^2 \kappa$) to the plasma current, which shows the fastest disruptions at ~ 1.8 ms/m^2 [30]. This is initiated by the thermal quench, which lowers the plasma temperature to very low values (few-10 eV). The plasma radiates this power primarily to the first wall, approximately 40-80%. Some of the magnetic energy ends up generating eddy currents in the surrounding conducting structures (10-30%), and some can also be conducted or convected to the first wall (0-30%). If runaway electrons (very high energy electrons) are generated the power split can change, with 50-60% of the magnetic energy released with I_P current decay prior to and following the runaways, 10-30% of energy released in runaway electron impact on the first wall and conduction/convection to the first wall, and into generation of eddy currents (10-30%)[28]. The runaway electron impacts are highly localized and can cause significant damage. Considering the non-runaway case, the magnetic energy available, using $L = L_{int} + L_{ext} = \mu_o R(li/2 + \ln(8R/a) - 2)$, for our power plant with li = 0.6, we get 140 MJ internal and 685 MJ external, or a disruption magnetic energy of 280 MJ. For the case here we will take 20% of this energy to be dissipated inducing currents in surrounding conductors. Radiated powers are considered to be deposited 20% to the inboard and 80% to the outboard, while conducted/convected powers are only to the outboard for DN. The radiation to the first wall ranges from 112-224 MJ with the remainder in conducted/convected. With OB/IB weighting for radiation and 2x peaking for both radiation and cond/conv channels, the combined loading gives temperature rises of 340-355 ^oK for tungsten and 470-490 ^oK for Fe (SiC) on the outboard, and 89-177 ^oK for tungsten and 122-243 ^oK for Fe (SIC) on the inboard. The temperature increases do not lead to melting for tungsten, Fe or SiC. These temperature increases are smaller overall because the time scale for deposition is so long compared to ELMs or the TQ, and the total magnetic energy available is small compared to the plasma stored energy.

If runaway electrons are produced in the current quench, as a result of the strong electric field created by the thermal quench, the first wall damage can be severe due to the local deposition of these particles which have very high energies, as high as ~ 1-20 MeV. Experimental data on runaway electrons deposition is scarce, and difficult to project precisely to future devices. The examination of runaway electrons in JET experiments[28] indicates that when runaway electrons are generated, the heat loading changes relative to a current quench without runaways. The magnetic energy in the runaway plasma ranges from 15-50% of the original pre-disruption plasma and the current profile peaks with values of li ~ 2.5. Using the 50% value, and our li = 0.6 changing to 2.5, we derive a runaway current of ~ 6.2 MA for our power plant, although more detailed estimates can be made[31]. A significant fraction of the magnetic energy can be spent driving eddy currents in the surrounding conductors, and here we assume ~20%[22,28]. While the plasma current decays from 10.9 to 6.25 MA, it mainly radiates the magnetic energy to the first wall similar to a CQ without runaways. The runaway current forms and we assume it occupies all the plasma current, during which there is no longer any radiated power. When the runaway current terminates the magnetic energy in the plasma during the runaway phase either is converted into kinetic energy of runaway electrons (20-60% for JET) impinging on the FW, or ohmically heats the surrounding plasma, or becomes cond/conv to the FW. Finally, a thermal plasma re-emerges after loss of the runaway electrons, which resumes radiating the remaining magnetic energy. The deposition area for runaways is a largely unknown parameter, the ITER projections are 0.3-0.6 m^2. Taking the 10-30% of the original magnetic energy converted to kinetic energy of runaway electrons, this would give 28-84 MJ available, and over ~ 1 ms deposition times, would lead to tremendous heat loads. JET experiments[27] indicate the temperature rise of the first wall behaves as ohmic heating, not surface heating like an ELM, and scales with the runaway current squared. Using Potts formula for perpendicular impingement, the depth of penetration for 12.5 MeV electrons is 7.2 mm for tungsten and 1.8 cm for Fe, which would be much lower for more grazing angles expected with magnetic field lines. Regardless, it is hard to imagine how any runaway electrons would be tolerable in a power plant, without significant armor that would conflict with thermal conversion and tritium breeding.

Mitigation of disruptions refers to the injection of particles, whether in the form of pellets, liquids or gases, in order to 1) diminish the conducted/convected heat loads during the TQ, 2) reduce the halo currents and subsequently the electromagnetic forces associated with them, and 3) avoid the generation of runaway electrons. In terms of heat loads we are interested here in those loads during the TQ and the heat loads from runaway electrons. The idea of mitigation is to inject particles before the TQ occurs by detecting some signal of the imminent disruption, although the injection itself induces a TQ. Experiments[29,32-34] show that about 90-100% of the plasma stored energy is radiated with massive gas injection (MGI) of noble gases (Ar, Ne) mixed with deuterium, the technique that has been studied the most. Correspondingly, the heat loads measured in the divertor are only a few percent of the plasma's stored energy. In addition, the current quench time is typically reduced, which must be carefully monitored since this drives stronger EM forces. The halo currents are typically reduced by factors of 2 or larger. Our power plant plasma has a stored energy of 690 MJ, and radiating this to the first wall area, with 80/20 split for the OB/IB, a peaking of 2x assumed, and over the same time scale as a TQ of 4 x Δt_{TQ}, we obtain a temperature rise of 1930 ^oK on the outboard and 970 ^oK on the inboard for tungsten. We obtain 2650 ^oK and 1330 ^oK for outboard and inboard, respectively, of Fe (SIC). The values for tungsten are under melting, and those for ferritic steel and SiC are too high. The mitigation of runaway electrons requires much larger numbers of particles input than the mitigation of the thermal quench, although these theoretical projections are uncertain. Although TQ mitigation has been demonstrated with next discharge recovery, the runaway electron mitigation is expected to shutdown the device in order to re-condition the PFCs. Other techniques for runaway electron mitigation may prove feasible.

DISCUSSION

It should be made clear that, in spite of excellent experimental activities, the estimates used in projecting heat loading are still quite uncertain. The ELM prescription, which is probably the most developed, shows that the splitting of energy deposition is complex and variable. However, the increased scrutiny given to the ELM thermal loading has shown that earlier estimates were probably over-estimates. Direct experiments on the thermal loading both in offline and tokamak devices are the only way to establish such complicated descriptions. It is important that these activities are integrated into PFC design and SOL plasma physics research.

There are several differences between ITER and our power plant example, including 1) the power plant has a high duty cycle and must operate for ~ 1 year continuously, 2) it will operate at higher material temperatures to maximize thermal conversion efficiency (1000 ^oC), 3) aggressive physics designs will likely have lower plasma current, and broader current profiles, but larger disruption TQ loading, 4) stronger plasma shaping will lead to DN divertor geometries, 5) it only needs to support startup and a single reference plasma configuration,

6) neutron irradiation will likely alter the material responses to these loadings significantly, 7) will require high radiated power fractions in the divertor, 8) high thermal conversion and tritium breeding will constrain the use of thick armors on the first wall, and 9) magnetic stored energies can be lower in a power plant depending on the plasma current and current profile.

Even if steady heat loading conditions could be guaranteed, there are areas that require a better description. In the DN configuration, the vertical position displacements give rise to a varying thermal loading between the upper and lower divertor, which should be clarified. The large radiated power fractions required in the divertor should be further established. Establishing the benefits of higher steady maximum heat flux divertor designs should be quantified as part of the broader power plant design, and in conjunction with possible transient scenarios. Disruptions in a power plant, appear to generate a damage level requiring shutdown and replacement of the PFCs. Even a mitigated TQ disruption would likely compromise the FW PFCs. If this led strictly to PFC damage requiring only replacement, this might be tolerable economically one to a few times in a plants operating lifetime (40 FPYs). However, if this led to, or enhanced the probability of, an accident the implications are more severe, for example runaway electrons penetrating to the high pressure FW coolant. It appears that runaway electrons will not likely be tolerable under any circumstances. ELMs provide a difficult loading environment, although it appears avoiding melting may be possible even for the largest ELMs. Unfortunately, the cycling (~ 10^8 per year) associated with these transients will likely drive the allowed MJ/m^2 to very low values to avoid cracking. Since these transients affect a narrow region near the PFC surface, and not the bulk material, actual component lifetimes with cracking should be established with very high cycle exposures. The power plant environment will provide a much stronger nuclear component, affecting the PFC materials, that will certainly modify the material behavior to these thermal loadings compared to those seen in non-nuclear testing facilities and tokamaks.

The first attempts to examine observed transient and off-normal heat loading from tokamak experiments with a power plant design have raised several important questions, which cannot be directly answered quantitatively right now. Research to eliminate disruptions, or their effects, is clearly of the highest priority. In addition, the combined efforts of developing high heat flux PFC designs, reducing the magnitude of transient loading like ELMs, and optimizing the tungsten, or other, plasma facing material are all likely required for a workable solution to thermal loading. A continued inclusion of physics criteria for engineering design of PFCs is recommended.

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