Мотивация МО ИТЭР предлагаемых изменений:

1. Снизить ЭМ нагрузки на ПС за счет исключения электрического замыкания пальцев при использовании менее оплавляемого вольфрама

2. Снизить риски изготовления технологически более сложных панелей с бериллиевой облицовкой

3. В отсутствии бериллия облегчить обслуживание и ремонт панелей внутри камеры

4. Сократить простои реактора на планируемую ранее замену бериллиевой стенки на вольфрамовую а также затраты на изготовление второго комплекта ПС

5. Сократить затраты на бериллиевую лабораторию

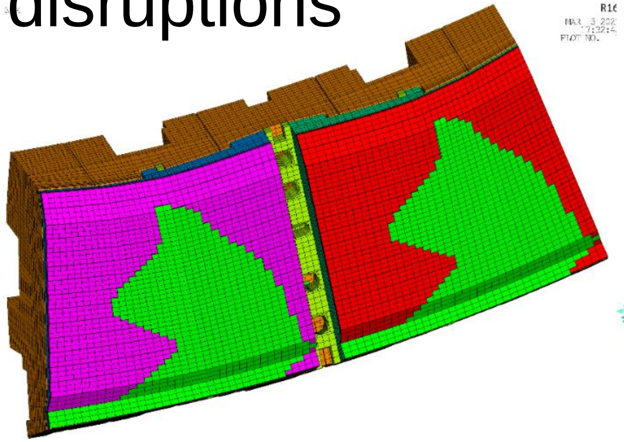
6. На новой фазе AFP отработать режимы контроля переходных процессов на не охлаждаемых более дешевых имитаторах ПС.

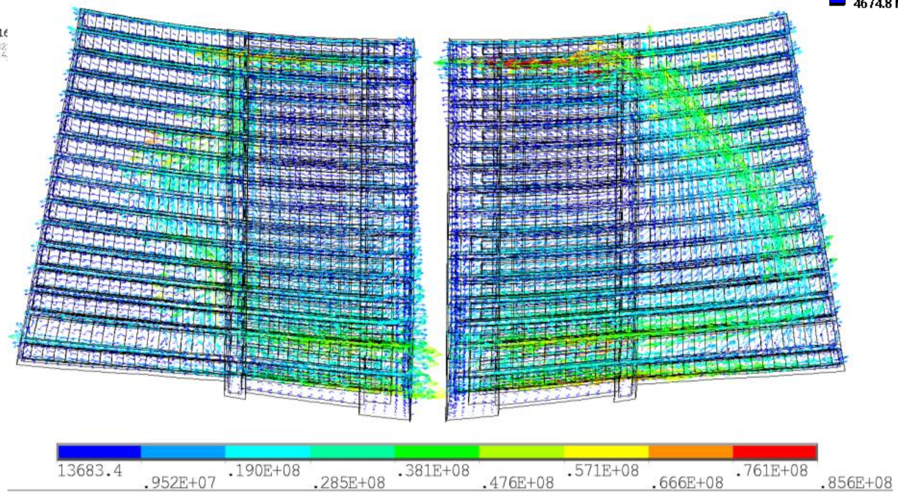
7. Снизить количество образующейся пыли и накопленного в бериллии трития

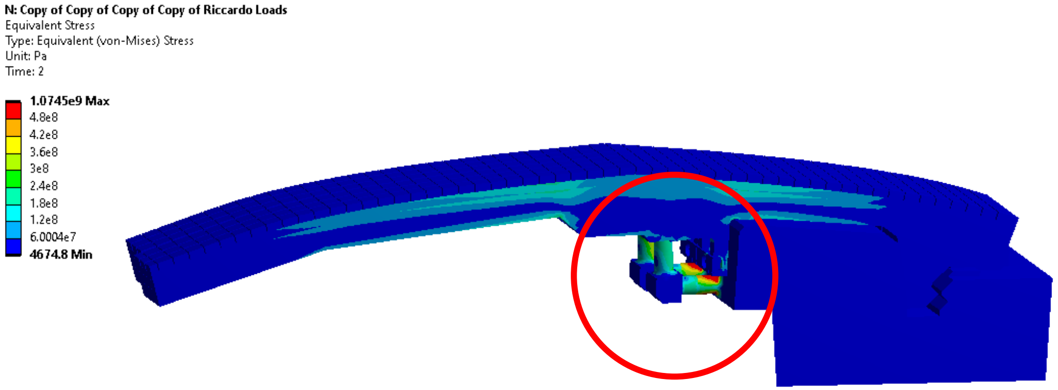
В случае обширного оплавления бериллия может произойти закорачивание нескольких пальцев ПС, что приведет к увеличению ЭМ нагрузки на зону соединения пальца с основанием панели ПС.

НИИЭФА
РОСАТОМ

EM analysis: compute eddy currents and distribution of forces and moments in shorted panel during subsequent mitigated disruptions

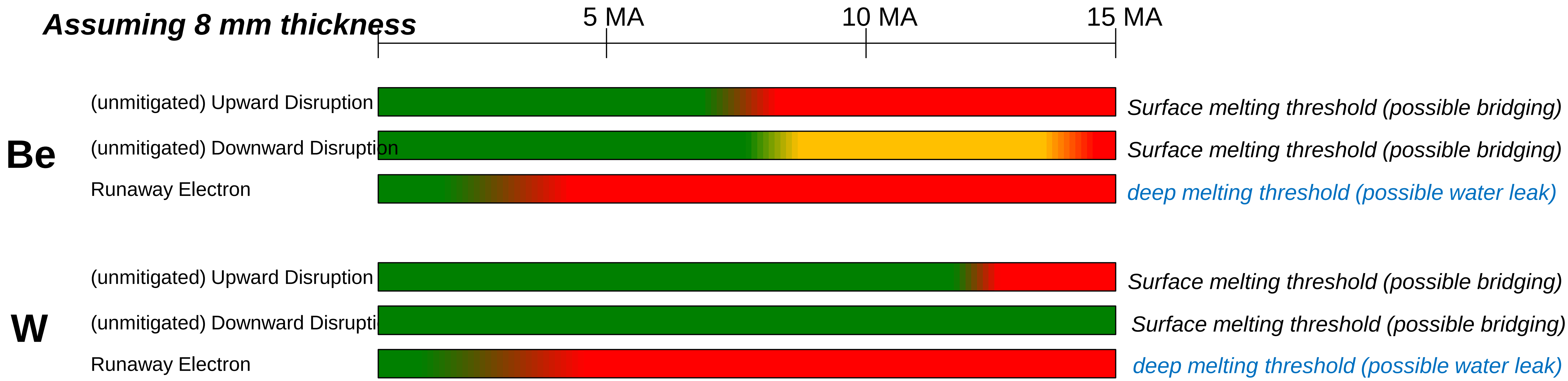






Once shorting occurs, hydraulic connections highly stressed and will not resist the load in subsequent high I_p current quenches

(Qualitative) comparison on transient events



So W looks better for upwards and downwards disruptions. Neither is fully suitable to handle Runaway Electros (with current thickness)

Оценка плюсов и минусов. Готовность поставщиков Be стенки

НИИЭФА
РОСАТОМ

Qualification Program (1/2)

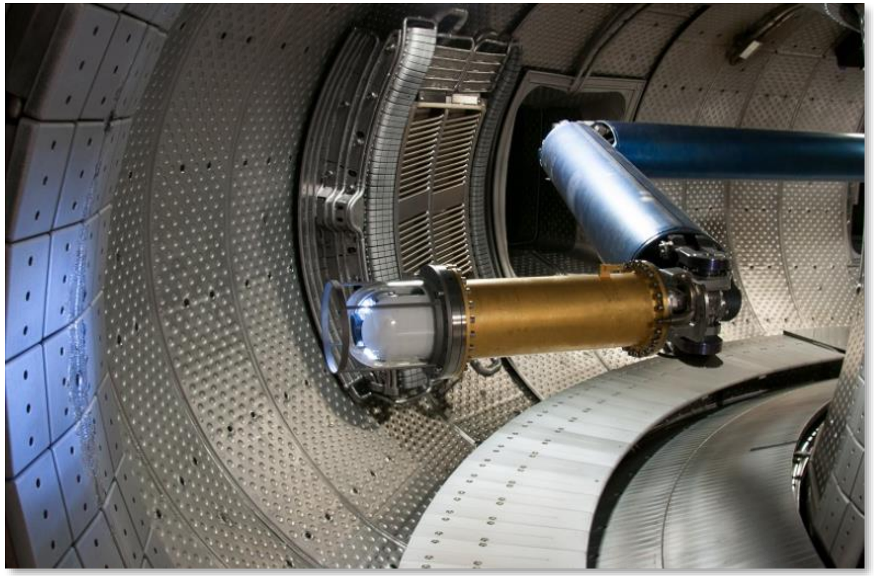
- Joining Be onto Cu is intrinsically challenging due to the formation of brittle intermetallics. This is not the case for the W to Cu joint.

- Despite successful joining qualification activities on small-scale mock-ups in the past years, severe technological issues have been recently identified in components of more relevant size.
- In one DA, 40% of the manufactured fingers that have so far been tested had unacceptable defects
- In another DA, several Be tiles fell off during high heat flux testing
- The third DA was successful but tested only 6 fingers at 4.7 MW/m2 for 1000 cycles.

Based on these recent unsatisfactory results, it is likely that the series production of a Be First Wall will face financial and schedule issues that have been so far underestimated.

Soon ready for next campaign C8 of WEST Phase 2

НИИЭФА
РОСАТОМ



Boronization

- Starting operation without boronization in a W-machine possible but challenging (WEST,AUG,C-Mod,etc.) → plasma performance is poor due to lowZ impurities/associated W sputtering
- Metal wall components ,boronization done with diborane (diluted) and GDCand TF off(with C components Carborane used–EAST) → Toroidally distributed GDC electrodes (+gas injection) provides uniform distributions of ~ 50-100nm boron films in several hours.
- Effects of boronization: a)reduction of lowZ content and b) covering of W main wall surfaces:
    - a) LowZ content reduction linked to remote areas and no direct plasma exposure
    - b) Covering W surfaces linked to plasma contact with main W wall (duration~100's sec) → not relevant for ITER (large amount of B)
- Typical H,D,T retention in B amorphous layers=0.3-0.6 (fuel atom per Boron atom)
- Large surface area of ITER → amount of boron per boronization is not small
    - → Wall coverage: 750m2x50nmx2,300kg/m3 = 86g
- Use of boron powder dropper effective in covering plasma exposed surfaces (less clear for remote areas) and expected amounts of B for ITER are very large (many g/shot) → optimization for reduction of B possible?

Содержание

ИТЭР.
Требуется определить мех. нагрузку на гидравлический коннектор ППС с бланкетом в случае повышенных электромагнитных нагрузок в связи с оплавлением бериллия и замыканием мостиками.

ТРТ
Бериллий (Z =4) для плазмы лучше чем бор, углерод и другие элементы, исключая литий.
Т.е. среди твердых материалов
К моменту первой плазмы на ИТЭР с вольфрамом (2034 г.) есть шанс запустить ТРТ с бериллиевой стенкой – как альтернатива.
Поэтапный вывод ТРТ на номинальный режим как у всех. «Стенка» может видоизменяться по мере роста мощности в плазме.

Содержание

Summary on ITER physics aspects relevant to New Base Line strategy development.

FINDINGS

Plan B comprising Augmented First Plasma (AFP) with FPO stage (FPO = DT1+ DT2 +) provides significant advantages of the ITER Research Plan (IRP) accelerating achievements of the ITER goals:

Demonstration of the Q=10, 300-500s (DT1) at total neutron fluence < ~10^{25} neutrons

Demonstration of the Q=5 long puls (~1000s) operation (DT2) and

High duty operation with total fluence up to 1027 neutrons.

Plan B is effective regardless to which PFC material (W or Be) is chosen.

High-Z PFC and W in particular, essentially increase the risks for ITER to fulfill its missions. Principal problems are:

High core radiation of W. Extremely low amount of W (~10^{-5}Ne) is known to prevent ignition. W penetration into the plasma core from divertor apparently can be controlled. The W source from the wall is unknown, thus present simulations can determine the fatal concentrations of W instead of predict its realistic value.

Plasma start-up from W limiter (inner wall) is much problematic. Even if possible it significantly narrows (Ip, Ne) operation space.

B (or other Light-Z) material coating of W walls proved to be very effective in the present tokamaks (AUG, WEST, C-Mod, EAST, FTU) is not considered as a possible mean to mitigate W penetration in the core plasma in ITER

An advantage of the W cf. Be wall declared as a principal one is due to higher W melting temperature. However, melting threat associated with accidental transients (VDEs and Disruptions). At full plasma current regimes W wall would melt similarly to Be one. Moreover, W bridges

Proposed auxiliary heating mix provides mostly electron heating at least at plasma densities expected for the AFP experiments. It means that W income to plasma would be much smaller in AFP than in DT1 due to low ion temperature.

Controllability of the plasma with W walls should be lower than with Be ones. (Number of actuators remains the same, but number of parameters to control increases)

RECCOMENDATIONS

Modeling performed and experimental experience accumulated up to date pointed out rather on the incompatibility of the High-Z PFC with high (reactor grade) performance plasma than on the recommendation of their use in fusion devices.

W – Be replacement significantly increases risks for ITER to reach its goals

ITER with Be-W FW-Divertor and ITER with fully W PFC are principally different devices. If for the former JET experiments already paved the road to project goals, for the second numerous R&D with unknown duration and cost are necessary.

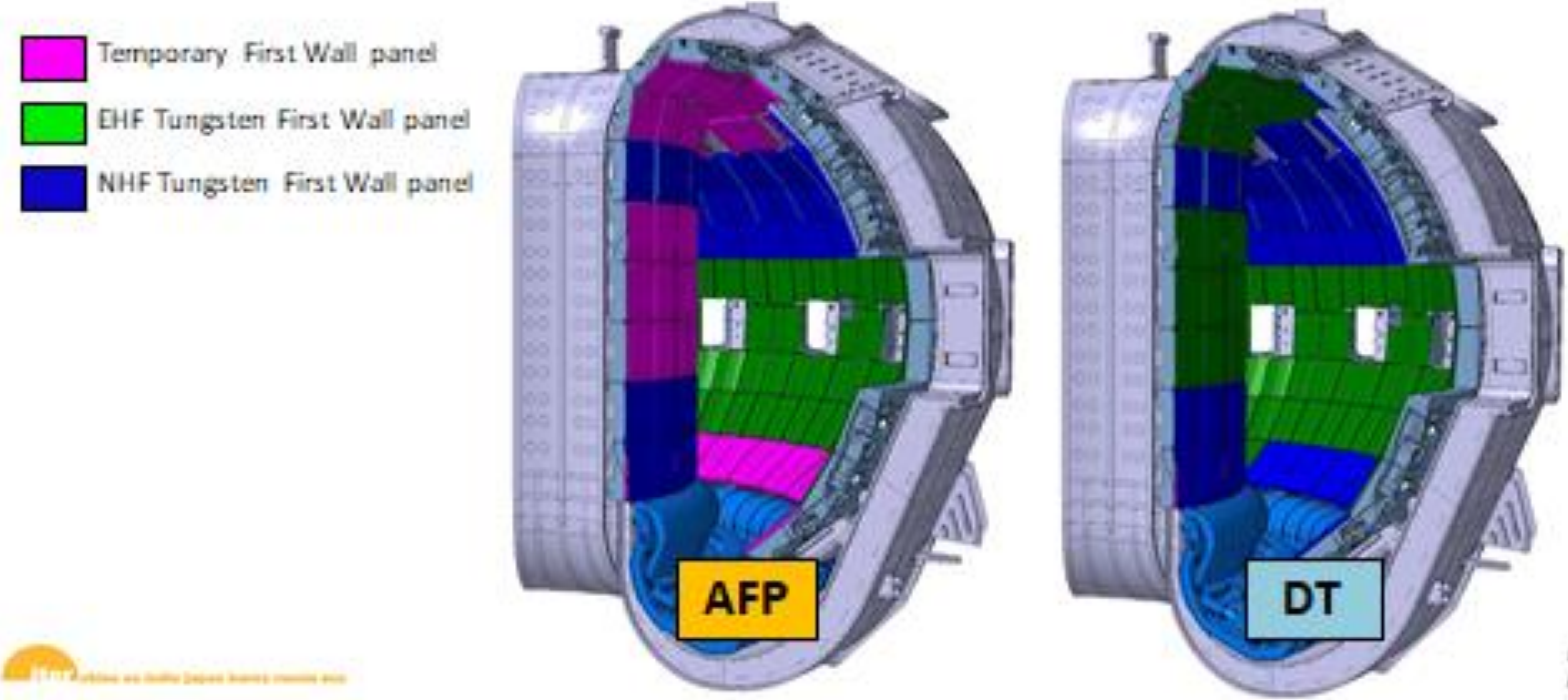
CONCLUSION

Proposed replacement of the Be by W FW is much risky and has no solid (sufficient) physical justification.

Содержание

НИИЭФА
РОСАТОМ

Propose to install 275 (final) FW panels during AFP



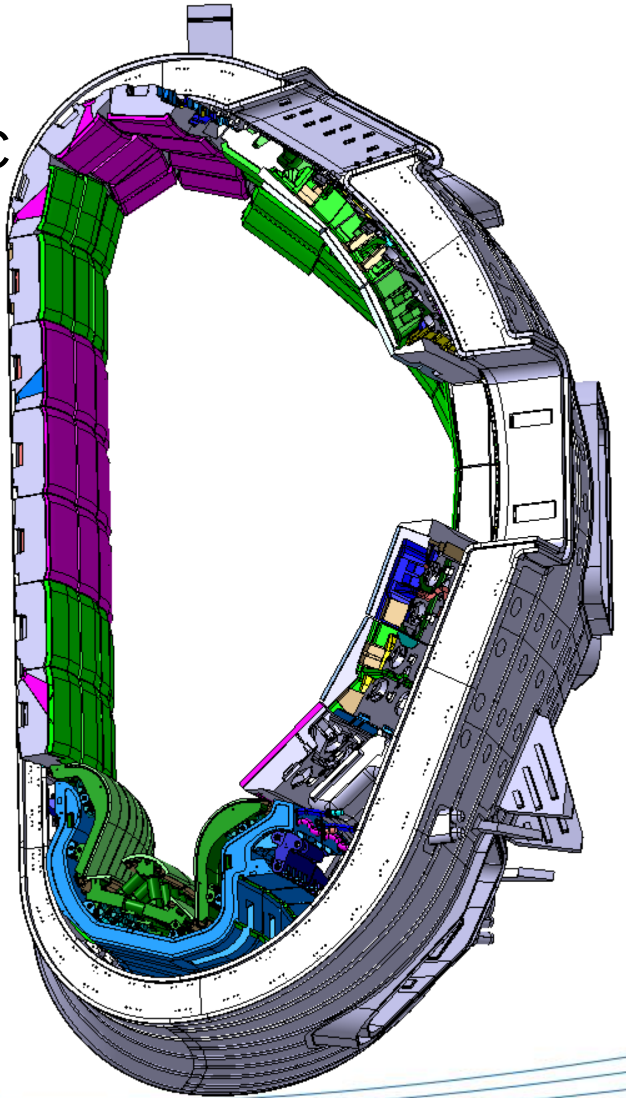
Investigating possibility to install some FW panels during A-FP

НИИЭФА
РОСАТОМ

Full coverage inertial PFC

Tungsten First Wall panel

Shield Block



	no Steel Proposal		Rationale	
Row	A-FP config	W	Stationary Loading	Likelihood of damage during A-FP (or after)
1	Final FW	18	Low	Low
2	Final FW	18	Low	Low
3	temporary	18	High	Low
4	temporary	18	High	Low
5	temporary	18	High	Low
6	Final FW	18	Low	Low
7	temporary	18	Low	medium
8	temporary	18	High	High
9	temporary	18	High	High
10	temporary	18	UP ports	medium
11	Final FW	36	Low	Low
12	Final FW	36	Low	Low
13	Final FW	36	Low	Low
14	Final FW	22	EQ ports	Low
15	Final FW	22	EQ ports	Low
16	Final FW	36	Low	Low
17	Final FW	36	Low	Low
18	temporary	36	Low	High
Total		**440**		

Note: this would potentially ‘kill’ the partial-coverage concept

Боронизация

Required R&D for scenarios B and C

- Boronization
  - Optimization of boronization (minimum B, uniform wall coverage, etc.)
  - Physics mechanisms determining lifetime of gettering effect
  - “Refreshing” of boronization getter effect by GDC/ICWC
  - Fuel retention evaluation and in-situ removal schemes

- W wall
  - Operation with W wall (+B in recessed areas) → impact on tokamak operational space (L-mode and H-mode)
  - Physics identification of mechanisms leading to core W contamination for wall source (plasma-driven source, C-X source, pedestal transport, core transport)
  - Optimization of W wall interaction and plasma transport to minimize core W contamination from wall (Wall clearance, ELM suppression, ECH, low rotation, etc.)

БОРОНИЗАЦИЯ

A possible alternative (though, unlike standard boronization which is routine on almost all tokamaks, this has been tried on only few current devices) is the injection of boron powder during tokamak discharges [Bortolon 2019]. This has been demonstrated recently on the full W ASDEX Upgrade and EAST with Mo wall and W divertor. A yet-unexplored alternative would be to use EC plasmas similar to those which may be used on ITER for wall conditioning between plasma pulses, but this would require experimental demonstration before being considered for ITER and it is far from sure that it could be made to work. Moreover, diborane gas, which is the likely medium which would have to be used for the EC plasma deposition, is hazardous (flammable and toxic) and requires very strictly controlled gas handling systems to be put in place. The boron powder injection experiments in ASDEX Upgrade showed that a good wall coverage to minimize W wall erosion and allow for low collisionality operation required injection of 340 mg of powder per discharge [Bortolon 2019]. Scaling with surface area and discharge duration leads to the injection of ~50 g per 100 s discharge in ITER. Since this is an accumulative effect, the routine use of this technique during PFPO is likely to lead to thick deposits being formed and thus the potential for significant production of dust and subsequent in-vessel tritium retention, unless these deposits are removed after PFPO-2 and boronization is not used in FPO, as discussed below.

БОРОНИЗАЦИЯ

It is very important to note that the use of boronization is unavoidably accompanied by similar problems as those associated with the use of carbon, since, during tokamak discharges, boron forms volatile products with hydrogen (or D and T) which can migrate to and deposit on remote and often inaccessible areas of the vacuum vessel/in-vessel components. As with carbon and Be, boron eroded from the walls by plasma action will also be ionized and migrated by the edge plasma, co-depositing directly on PFC surfaces, trapping fuel. There has been very little R&D on boron co-deposition (compared to the database for carbon and Be). Most published papers consider the case of amorphous H-(B+C) layers, since most of the work was performed on tokamaks with carbon walls using boronization. Laboratory studies of pure boron films [Annen 1997], including thermal desorption spectroscopy, show that a-BH and a-CH layers are very similar, so that H will be bound at roughly the same strength, and thus more strongly than in Be-H layers. As a consequence, codeposition rates with B will be similar or higher to those with Be, but the trapped fuel in the H-B layers will not be released at any baking temperature accessible to ITER. For example, the ~50 g of boron power injection per 100 s discharge in ITER mentioned above would lead to a trapping of 2.7 g of T, if applied in DT plasmas.

Первый отчет

ПЕРВЫЙ ОТЧЕТ

НИИЭФА
РОСАТОМ

Given the possible risks to the IRP, a solid risk mitigation strategy to the use of a W FW in ITER would require demonstration of high performance H-mode plasmas in a device more closely approaching the ITER parameters than is possible on medium-sized tokamaks such as ASDEX Upgrade, EAST etc., and with similar capabilities to those of ITER for ELM control and core W transport control (ECH). Given that JET now has a very finite lifetime (projected end of operations by end 2023 latest) and has neither ECH nor an in-vessel ELM control coil system, the only device suitable for such a demonstration would be JT-60SA. However, operations on this machine will begin with carbon walls and divertor and a switch to full W is foreseen only in 2030 when the main mission goal of high-beta, full non-inductive steadystate operations has been achieved. This is probably too late for the procurement of FW panels in ITER to be ready for Assembly II. It is also worth noting that JT-60SA has made this staged PFC material strategy choice for precisely the same reason that ITER adopted Be for the start of PFPO operations.

НИИЭФА
РОСАТОМ

A.7. Considerations related to fusion community issues

Considerable effort has been dedicated by the scientific community to address the issues related to the specific choice of plasma-facing materials adopted for ITER (namely a Be first wall and a W divertor). This has involved a large-scale project at JET lasting more than a decade, but also supporting experiments elsewhere (e.g. PISCES-B). As a consequence of these efforts, it is now widely accepted that the Be+W combination is a low risk choice for ITER to obtain clean high performance H-mode plasmas with Pinput < 2-3 PLH (as will be the case for ITER, even during burning plasmas), while avoiding W accumulation in the core plasma due to the beneficial screening effects of the divertor. It also offers a much lower (more than a factor of 10 [Brezinsek 2013]) level of fuel retention by co-deposition than carbon and the most recent studies of the scaling for this Be co-deposition [Zaloznik 2022] show that the retention has a strong chance of being maintained below in-vessel regulatory limits throughout the planned FPO-1-3 campaigns, even without active fuel recovery techniques.

This month, JET reported on the achievement of significant fusion power production (~10 MW with Q ~ 0.3) and low T retention, clearly demonstrating the adequacy of the ITER baseline material choice and vindicating more than a decade of R&D effort to establish high power operation in the Be/W environment. Various public statements in connection with this announcement, including those from the ITER Organization, have appropriately made direct reference to the importance of the JET DT results in the sense that they have been achieved with the FW and divertor materials which are currently planned to be used in ITER.

ПЕРВЫЙ ОТЧЕТ

Proposing a change to a full W wall for the start of ITER operation will be met with strong scepticism by large sectors of the fusion community, since it puts at risk the timely achievement of ITER’s fusion power goal. No device with a W wall has demonstrated high performance Hmode operation at low values of Pinput/PLH, typical of ITER, in the absence of strong coating of the wall by low Z material (usually boron, but also lithium) which are not viable in ITER due to the associated T retention issues. This is why in the IRP, the change of the FW to W was considered after ITER had demonstrated its fusion power goals (namely after FPO-3), and enjoys the support of the scientific community and the IC STAC. By the end of FPO-3 the main wallplasma interaction in ITER should be understood and controlled (e.g. ELMs would be

52

suppressed), thus minimizing the impact of a W wall on plasma performance. There have been no significant advances in the field since ITER construction began which invalidate this logic or make the risks associated with a W wall lower than originally foreseen. On the contrary, the past decade of operation in JET with the ILW, culminating in the new DT results just released, have consolidated the case for the ITER material choice and provided an excellent foundation to expect that the IRP can be executed as planned.

ПЕРВЫЙ ОТЧЕТ

НИИЭФА
РОСАТОМ

The issues associated with the use of Be in tokamaks have been well known since the late 1980’s when it was introduced at JET, after initial studies in the ISX-B tokamak at ORNL. Similarly, it has always been recognized in the scientific community that Be is not a suitable material for the FW of a DEMO (reactor). The rationale for ITER has always been to begin with a material choice that offers the fastest/lower risk route to high Q operation and then switch to a more relevant material once the main mission goals have been achieved. This was the primary motivation for the effort made at the 2007 ITER Design Review to provide the capability for FW panel exchange. Moreover, there are many systems in ITER whose application on DEMO is very questionable (in-vessel coils, NBI heating, etc.).
While the complications associated with the use of Be might not have been fully appreciated at ITER until recently, it will be difficult to explain why we are now realizing this, following more than 30 years of successful experience at JET, since none of the facts related to Be have changed since the ITER 1998 FDR design. Any decision to switch from Be □ W on the ITER FW would be both heavily disputed by many sectors of the R&D community and is very likely to take a significant toll on the technical and scientific credibility of the ITER Organization towards the fusion community. This latter point deserves serious consideration, independently of the technical/financial (not scientific) merits of abandoning Be.

ПЕРВЫЙ ОТЧЕТ

НИИЭФА
РОСАТОМ

Another major concern with the use of Be is related to its Uranium (U) impurity content. The ITER specification allows 30 ppm of natural U in Be. This U under 1014 n/cm2/s will generate fission products including gas and volatiles outgassed from the Be-dust and Blanket FW that need to be managed with halogen traps, charcoal filters and decay tanks during normal operation as well as accident scenarios. Natural U derived fission products outgassed from irradiated Be was witnessed at
Petten Joint Research Centre (JRC) in Netherlands. As a mitigation action, Be with a lower U impurity concentration (say less than 5 ppm) may also be envisioned but with additional costs. To be noted that Be is already a quite expensive material (about 5000 Euro/kg).

Содержание

Development of conditioning schemes to decrease oxygen content and to reduce the surface area of a W wall exposed to the plasma. In the majority of present devices, boronization is performed with diborane/He gas mixtures in a glow discharge cleaning (GDC) plasma – this is plasma chemical vapour deposition. Experience on these machines is that the effect of boronization lasts from ~100 s (for wall coverage) to ~1000 s (for oxygen decrease), so that GDC, which requires that the toroidal field be switched off, is not an option for ITER.

ПЕРВЫЙ ОТЧЕТ

In conclusion, removing Be from the ITER machine would lead to substantial advantages in several areas like safety and licensing, assembly, manufacturing, RH, integration. The overall cost saving for the project is substantial and could reach one billion Euro, including the avoidance of a 33 months machine shutdown for the complete FW panels replacement, although up to 2 years would be needed to replace specific rows which is expected to be required.

The major drawback is related to the high-Z impurity influxes to the plasma, and its significant expected impact on the plasma performance. This can slow down significantly the progress towards high Q operation counter-balancing the saving in shutdown time. Secondly, the impact on the ongoing Procurement Arrangements for the Blanket FW panels shall also be assessed.

Физика

Содержание

During H-mode operation, the W influxes both from the divertor and the main wall are dominated by ELMs. This is due to the high energy of the particles expelled by the ELMs (with typical pedestal temperatures of several 100’s eV on current devices to several keVs on ITER), which exceed the sputtering threshold for high Z material (see Fig. 6 from [Kallenbach 2005]). While at low temperatures (typical of conditions between ELMs) the presence of impurities in the plasma dominates the sputtered high-Z influx, for higher temperatures (typical of ELMs) the sputtering of high-Z material by main plasma ions is a major contributor. These conclusions remain valid for hydrogen plasmas, despite the fact that sputtering thresholds are higher and yields lower in comparison with deuterium and helium (the projectile species in Fig. 6).

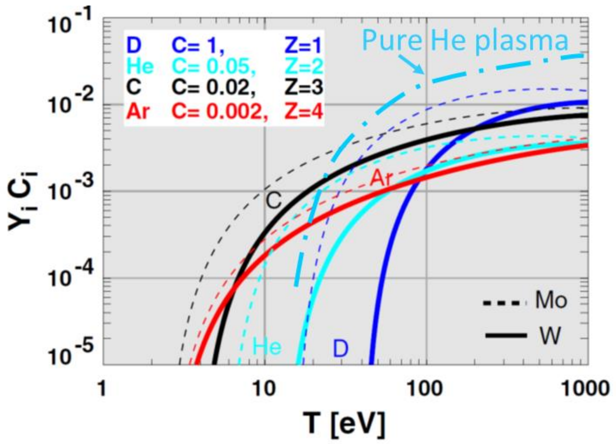
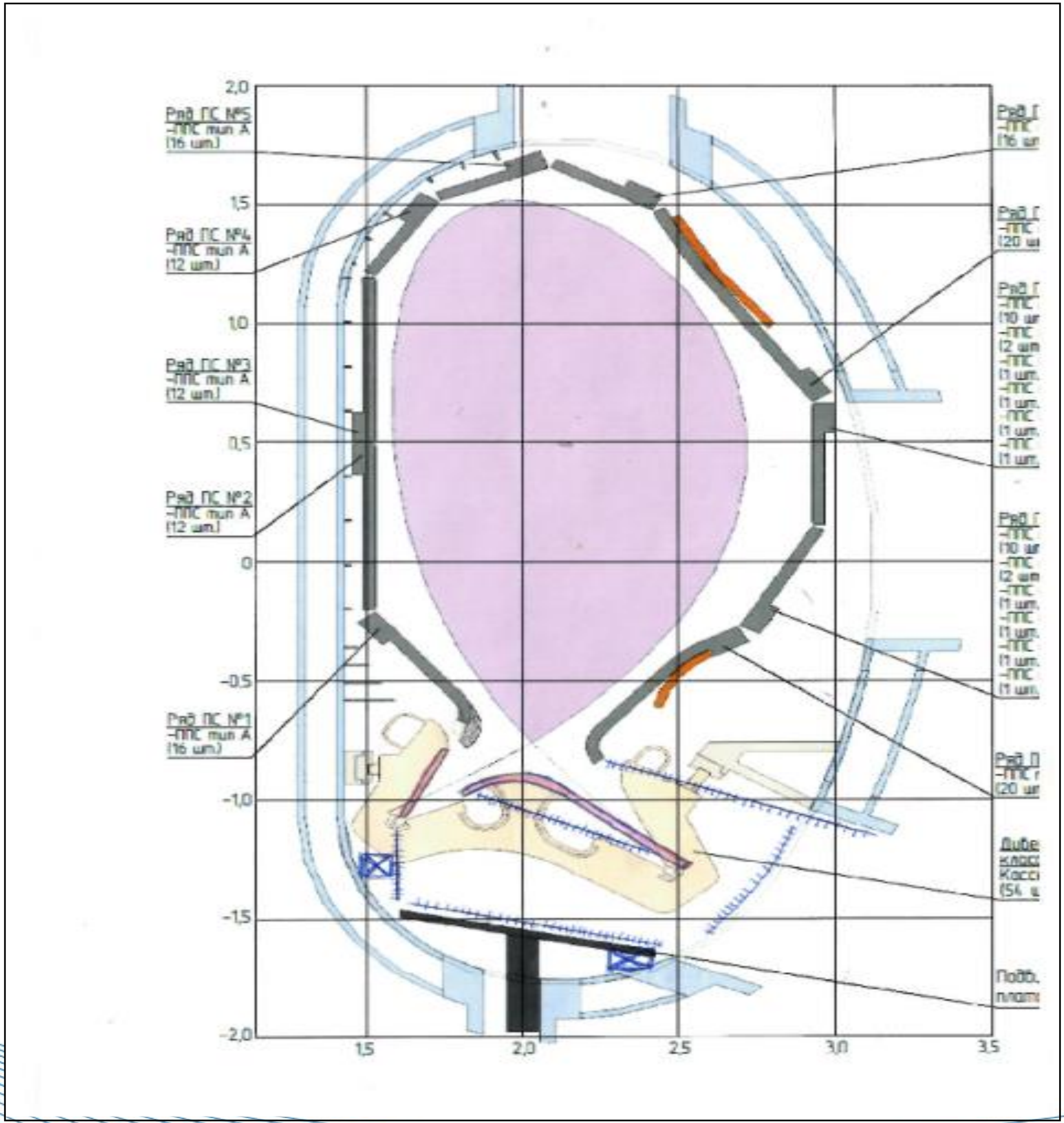


Figure 6. Sputtering yields for W and Mo by deuterium, helium, carbon and argon. The impact energy is assumed to be E = 3ZTe + 2Ti assuming Ti = Te. The impurity yields are multiplied by the assumed concentrations and impurity charge states for a better comparability of their effects.

Раздел ТРТ



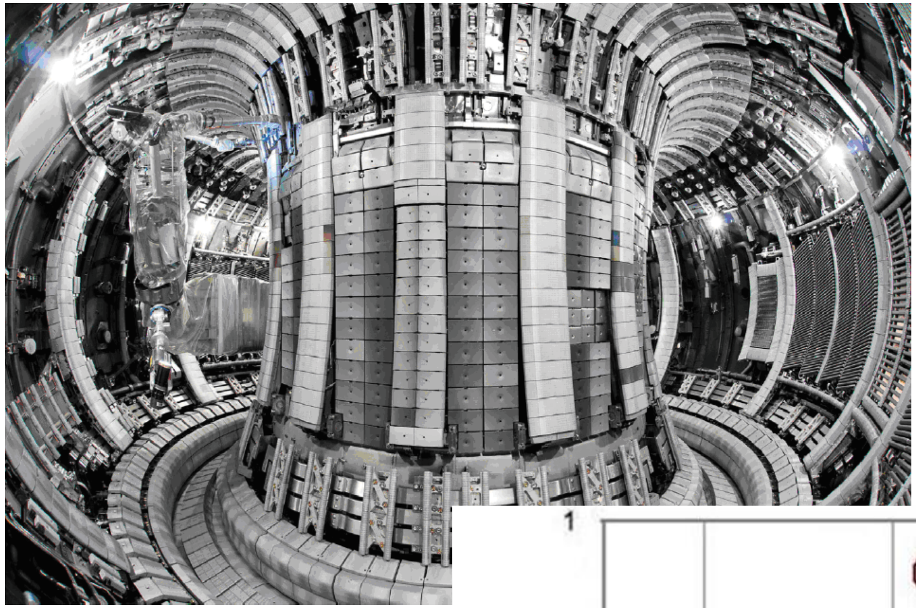
Новое положение мишеней

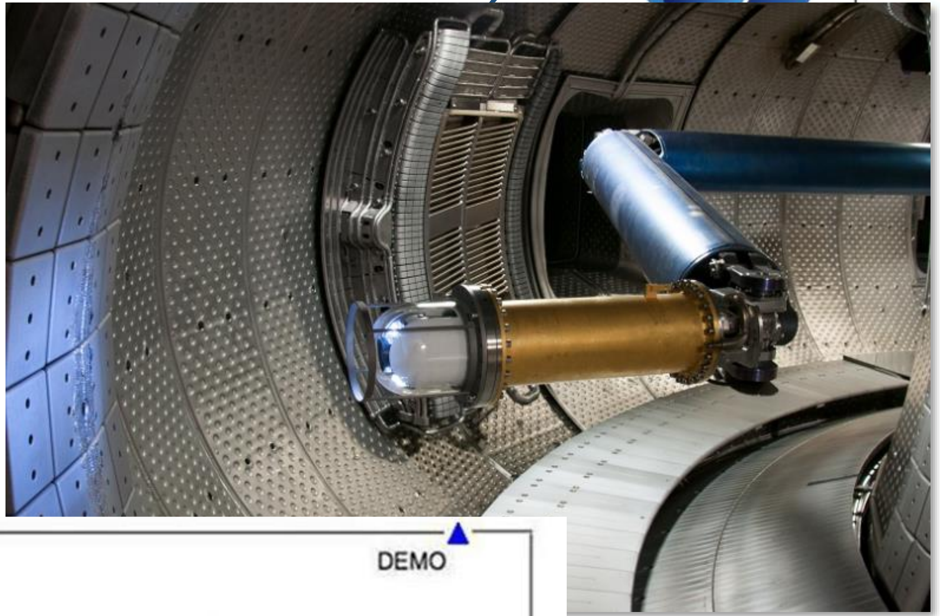
Откачивающие поверхности

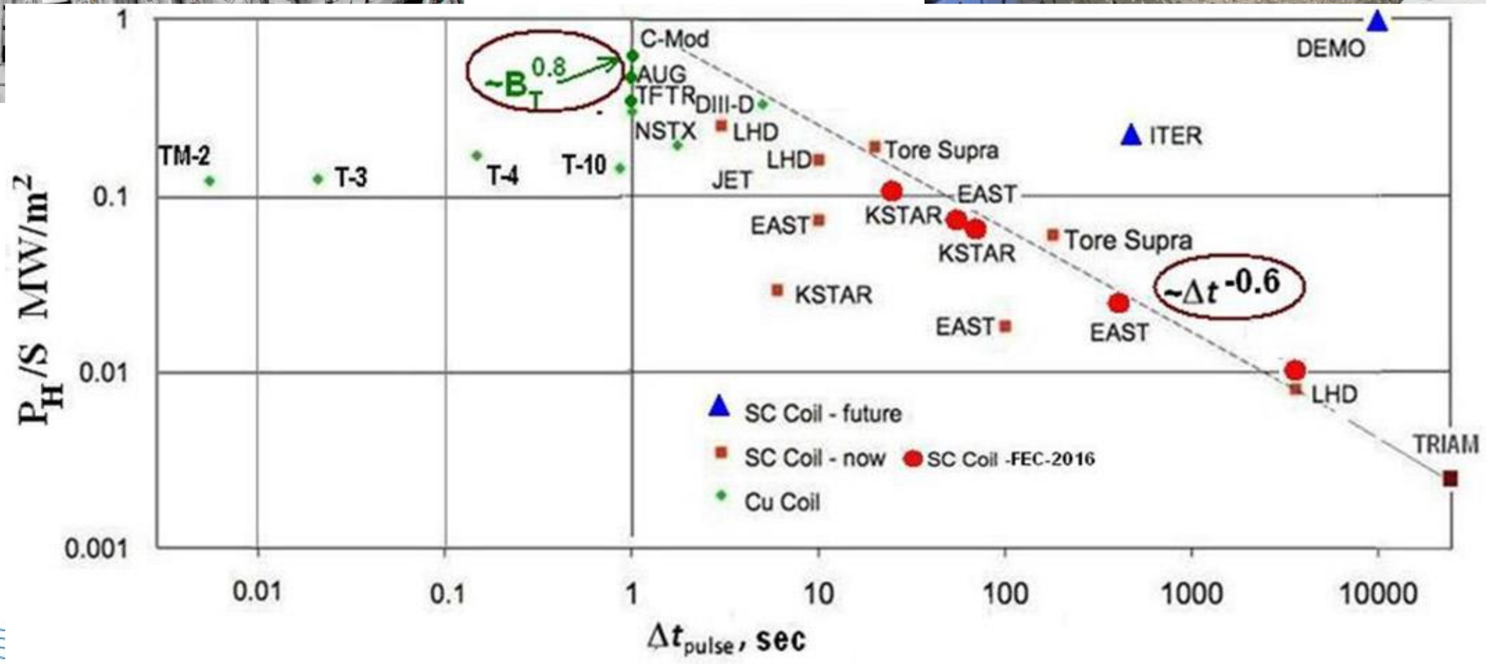
Вариант 2. Твердотельная мишень в зоне сепаратриссы (+качание) –розовая. Поглощающие панели (синие с ворсинками) в тени от основного потока тепла выполнены на основе жидкого лития или нераспыляемых геттеров.

«Первая стенка» JET (*«Шанхай», нет охлаждения*)
VS
«первая стенка» WEST (*активное охлаждение, «аскетизм»*)

НИИЭФА
РОСАТОМ

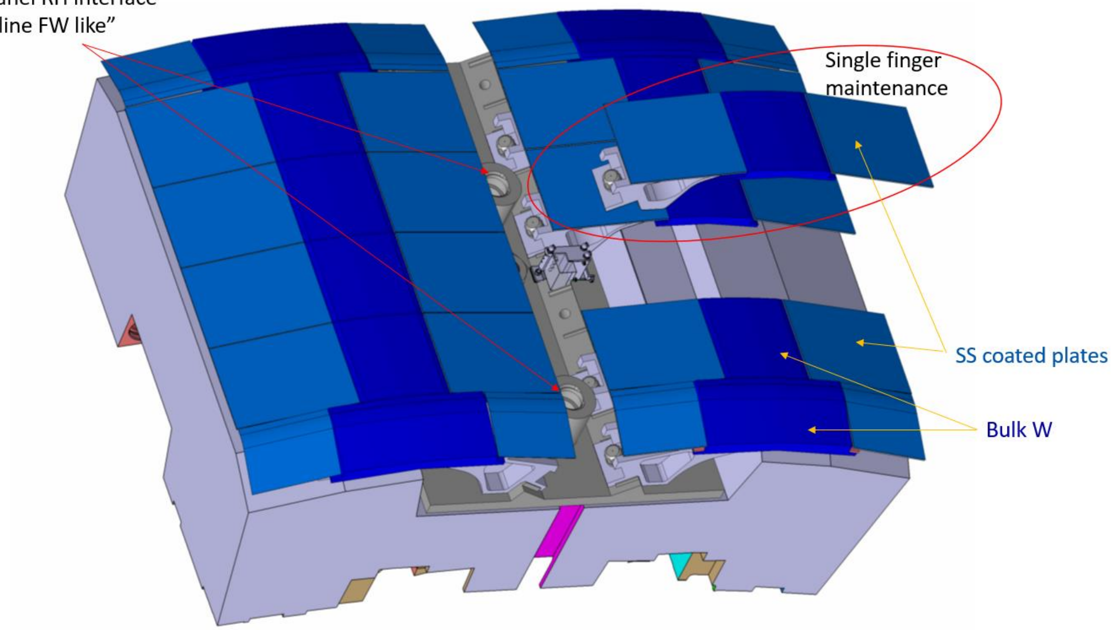


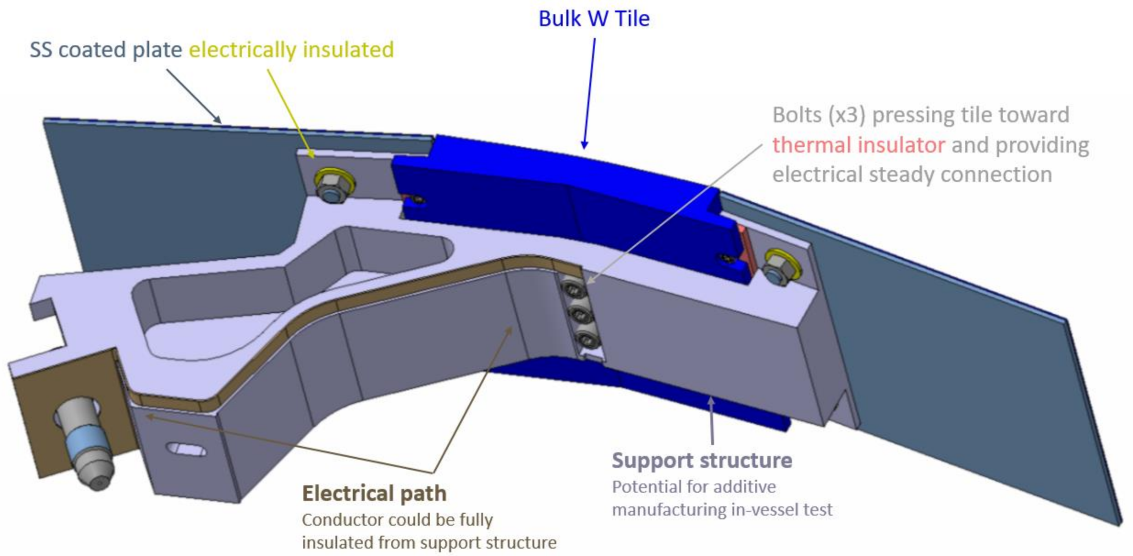


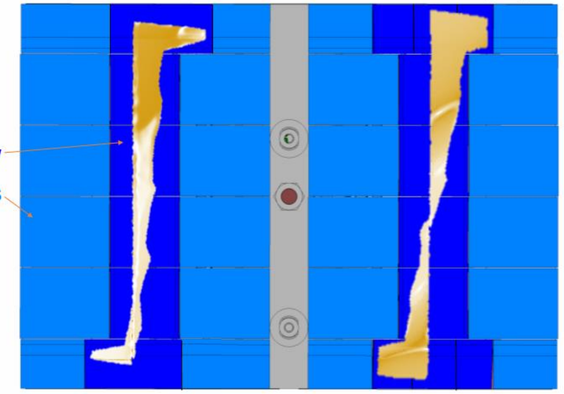


Резервные картинки

Temporary First Wall concept design







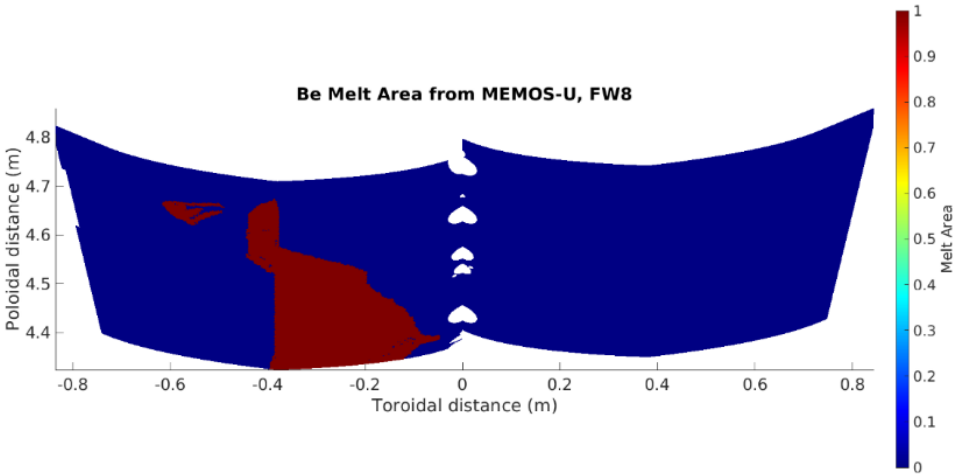
Bulk/Coated division could be determined
based zoning via Field-Line Tracing analysis

Melt area under 15MA Upwards VDE

**NOT considering edge heat flux*

FW08

Be



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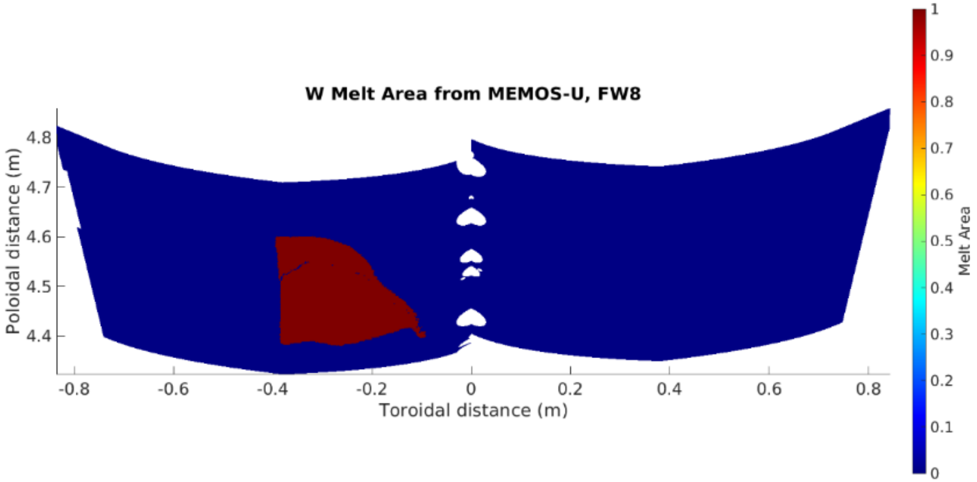
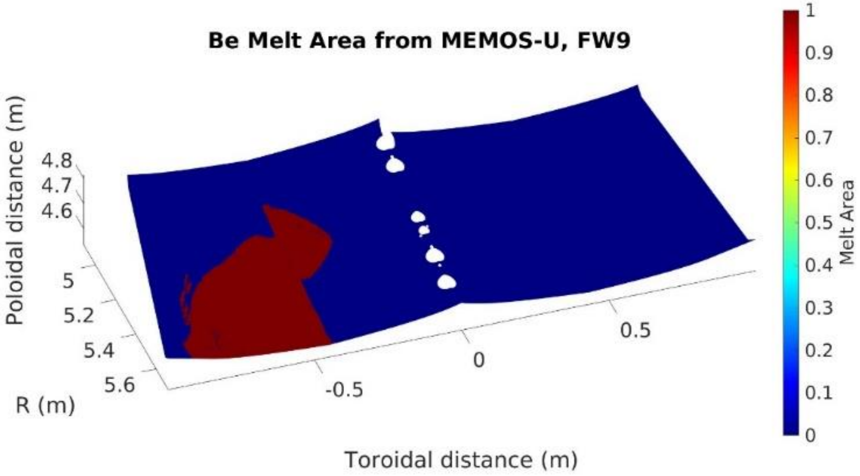


FW09

Be



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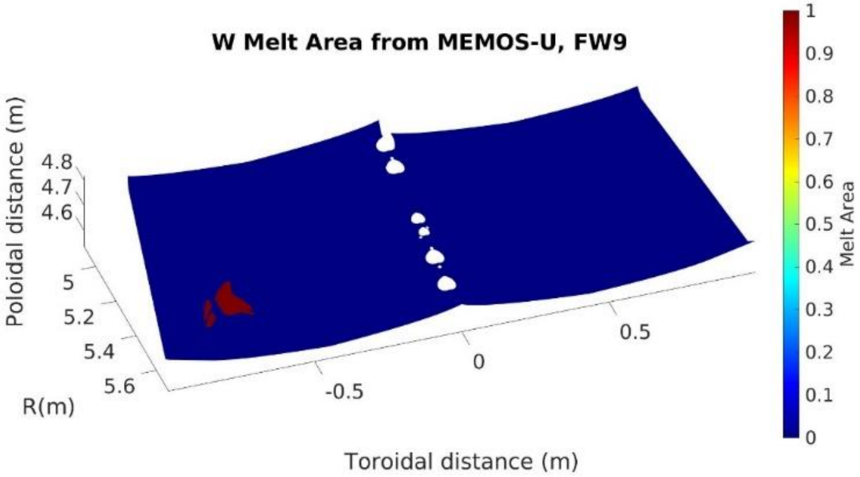


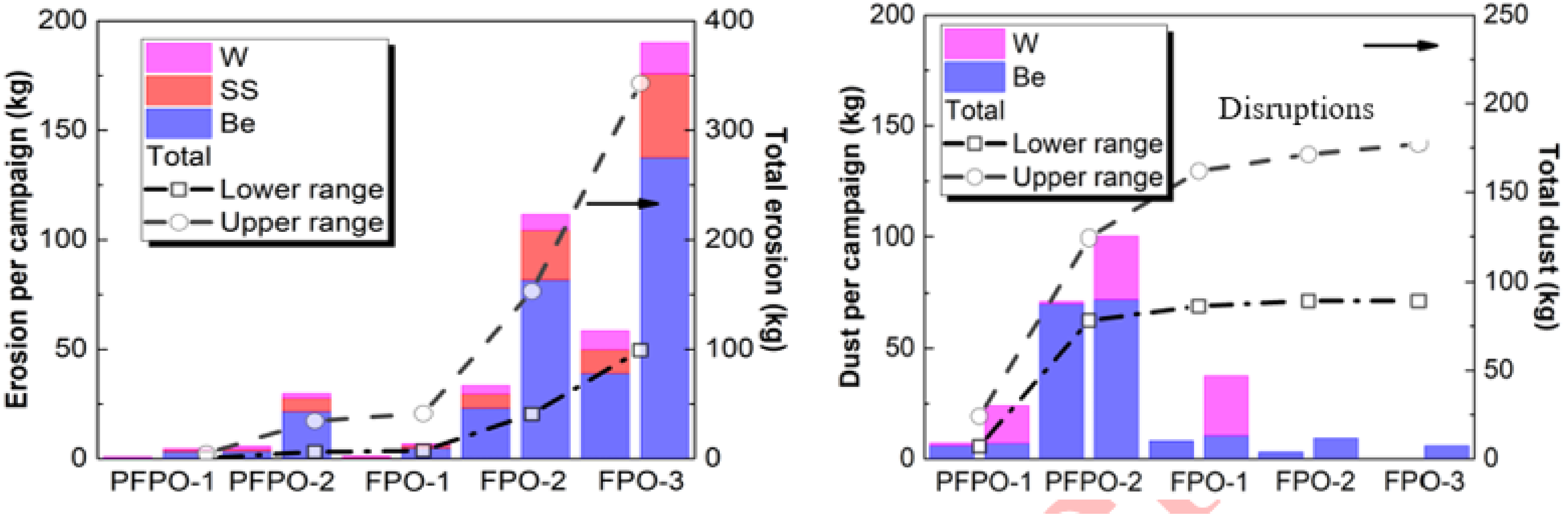
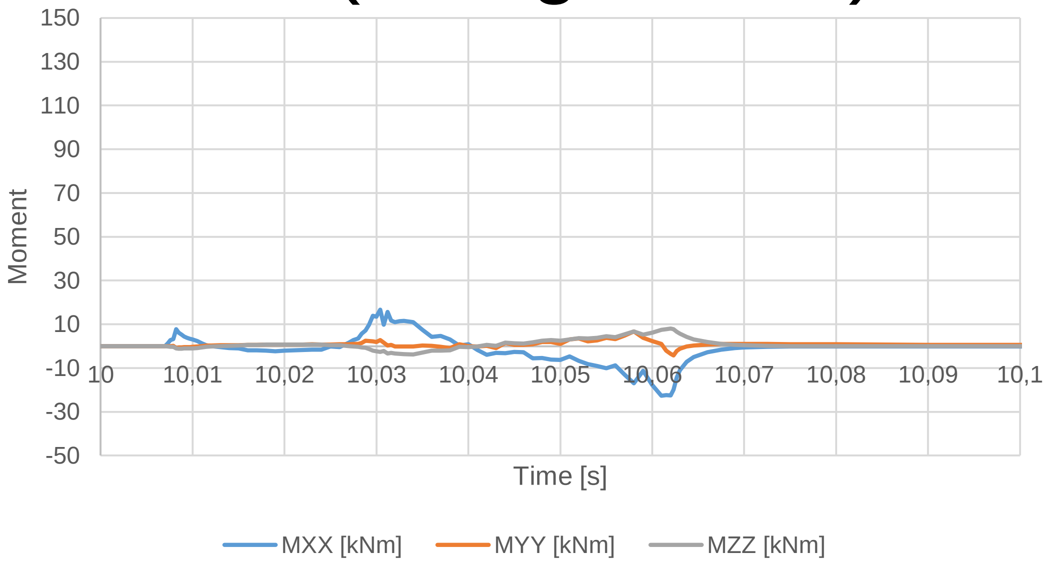
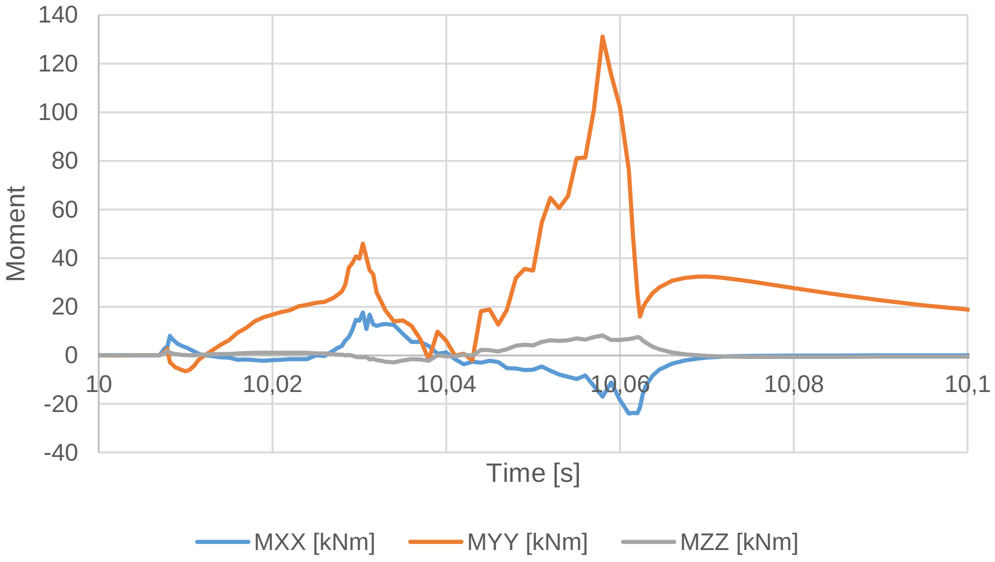
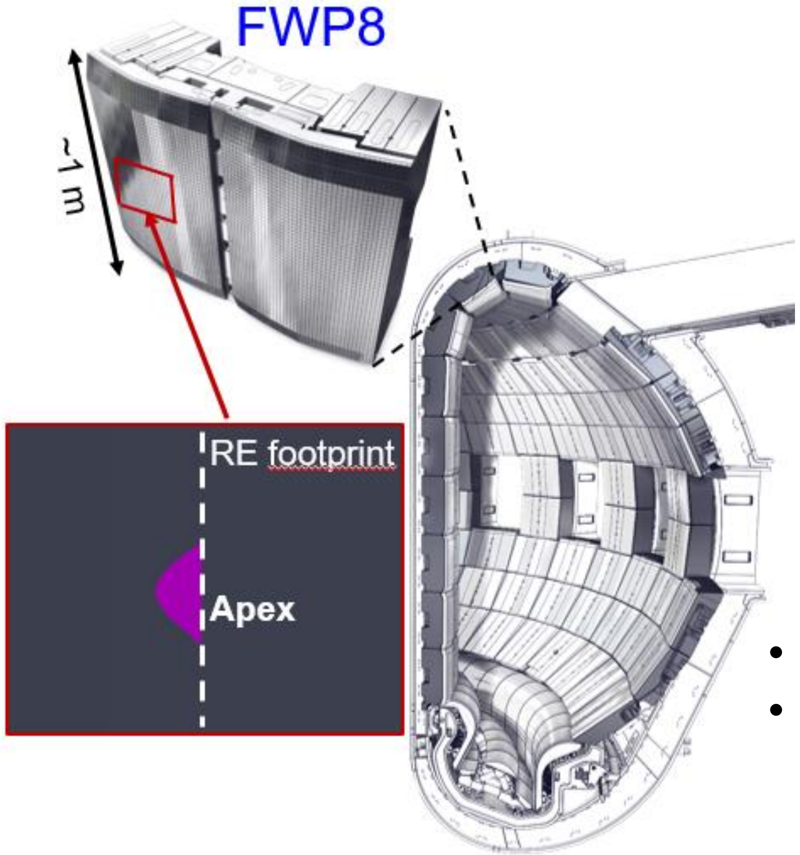


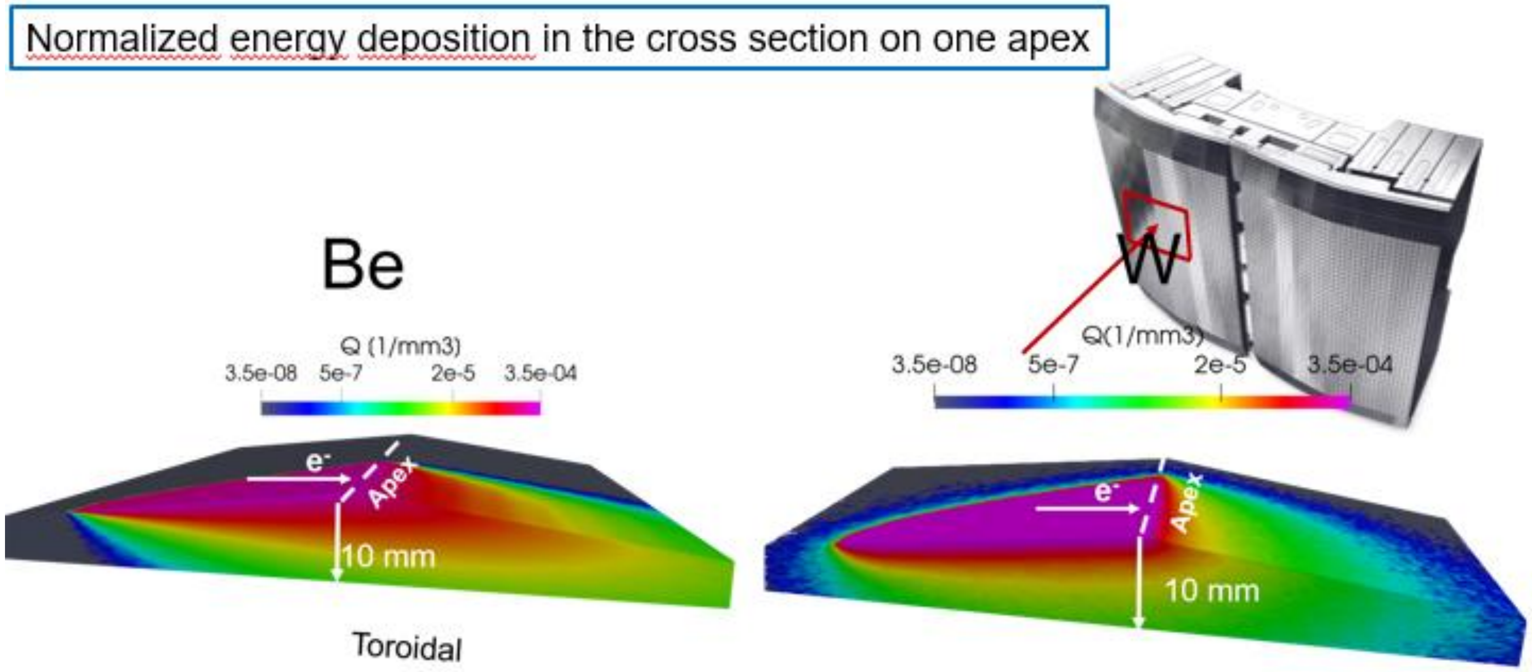
Figure 9. Dust production in different plasma phases

	FW09	FW09
Be	Without melt (Design Case) 	With melt 

Runaway electron assessment: Be vs W comparison

НИИЭФА
РОСАТОМ





- Disruptions can generate runaway electrons → deep melting for Be and W
- W is more efficient at stopping electrons than Be→ more localized heat loads

Содержание

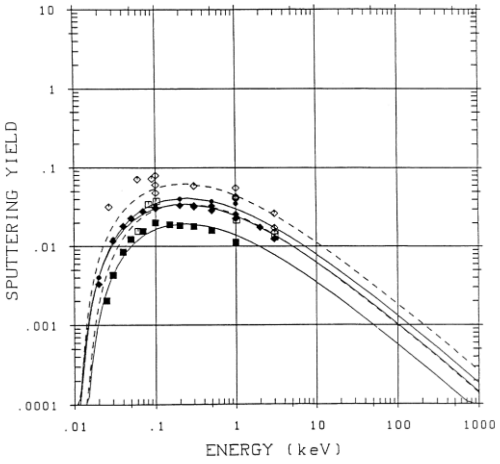


Fig. 13: Energy dependence of the sputtering yield of Be with H, D and T. The experimental data were measured at 650°C to avoid surface oxidation. The threshold energy of the experimental data for D and H was taken from calculated data for fitting. The data are published in [31,32,33].

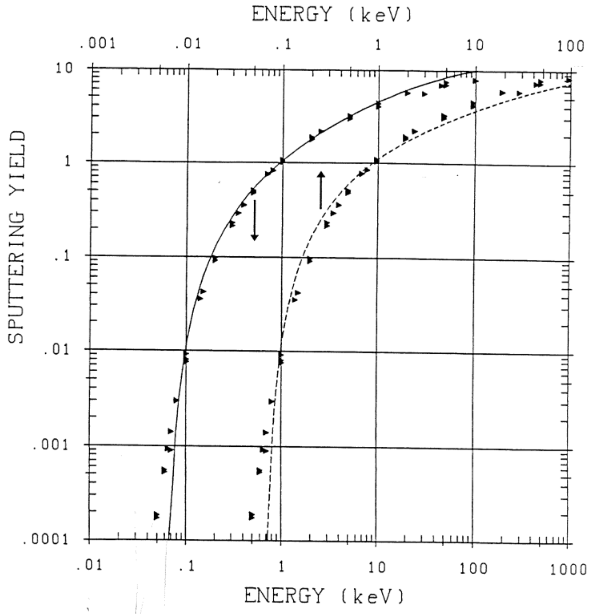


Fig. 5: Energy dependence of the tungsten selfsputtering yield fitted with the *Bohdansky* formula (4) (dashed line) and with the revised *Bohdansky* formula (9) (solid line).

Calculated Data

Target	Projectile	Angle	Symbol	$E_{th}(eV)$	Q(atoms/ion)
Be	O	0	▼	47.0	1.71
Be	Be	0	►	23.0	0.770

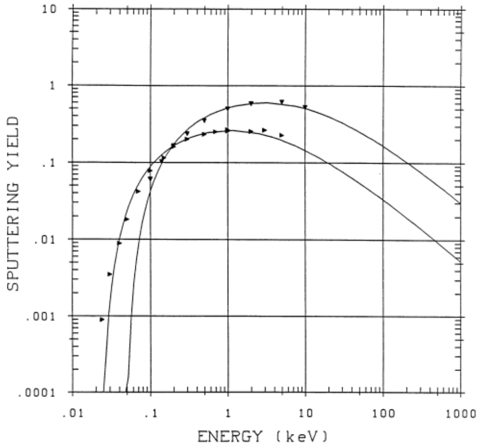


Fig. 16: Energy dependence of the sputtering yield of Be with O and Be. The data for O were calculated only for clean targets without formation of BeO (no dynamic calculation). The data are published in [33,35-38].

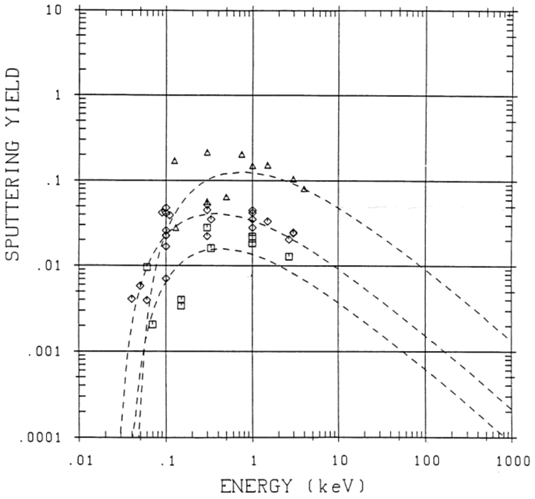


Fig. 14: Energy dependence of the sputtering yield of Be with H, D and ^{4}He measured at room temperature. The similarity with the results obtained for BeO (Fig. 52) as well as the large scatter of the data points indicates ion induced oxidation during sputtering. These results are published in [33,34].

Experimental Data

Target	Projectile	Angle	Symbol	$E_{th}(eV)$	Q(atoms/ion)
W	H	0	□	429.	$0.700 \cdot 10^{-2}$
W	D	0	◇	178.	0.0179
W	^{12}C	0	□	27.6	0.780
W	W	0	▹	59.0	30.9

Calculated Data

Target	Projectile	Angle	Symbol	$E_{th}(eV)$	Q(atoms/ion)
W	D	0	♦	201.	0.0345
W	T	0	•	129.	0.0654
W	^{12}C	0	■	47.2	1.42
W	W	0	►	63.0	32.2

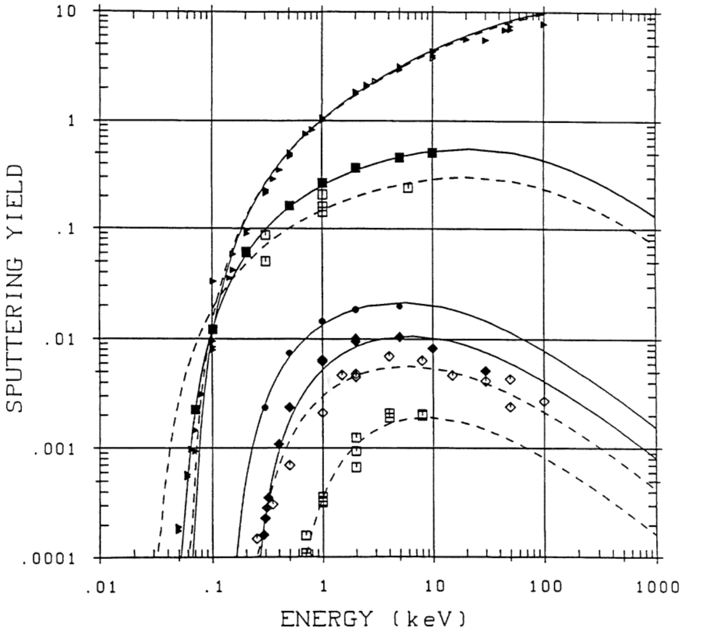
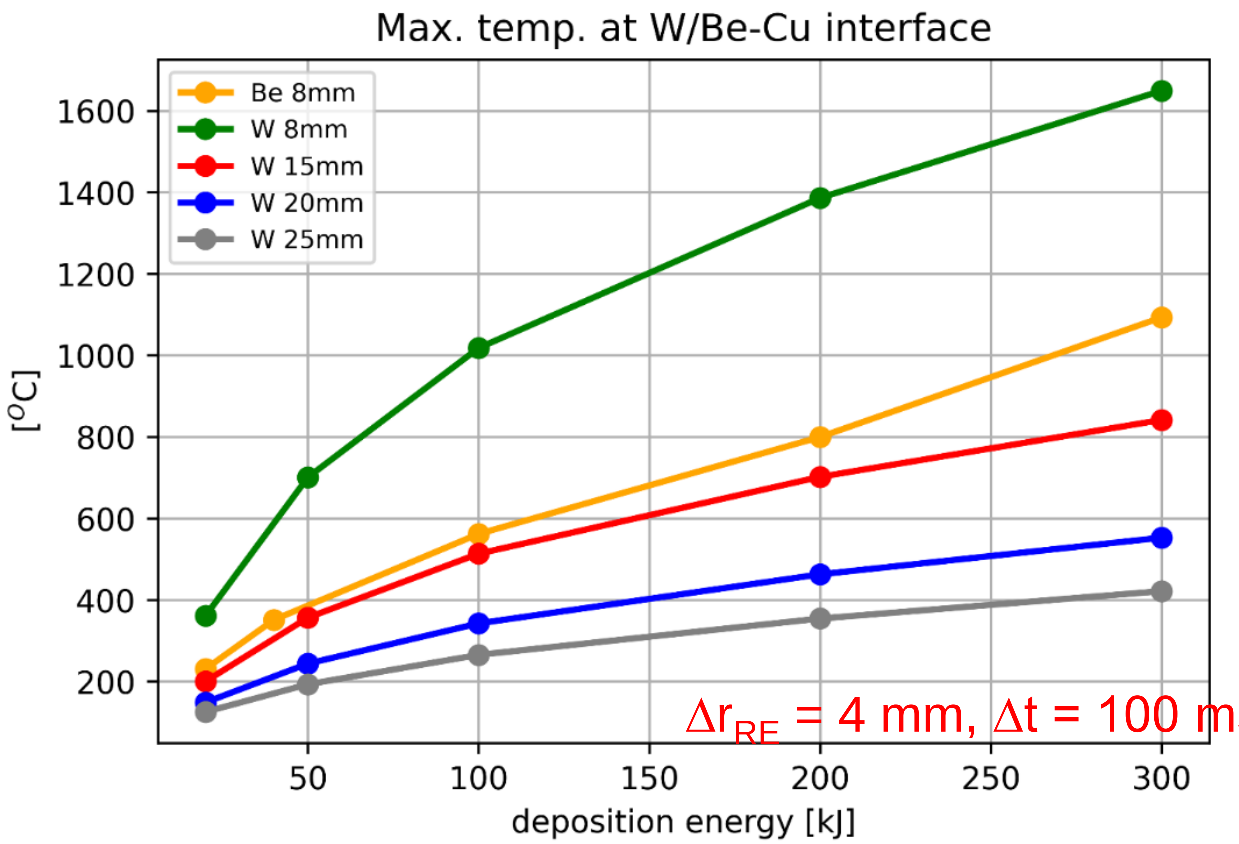
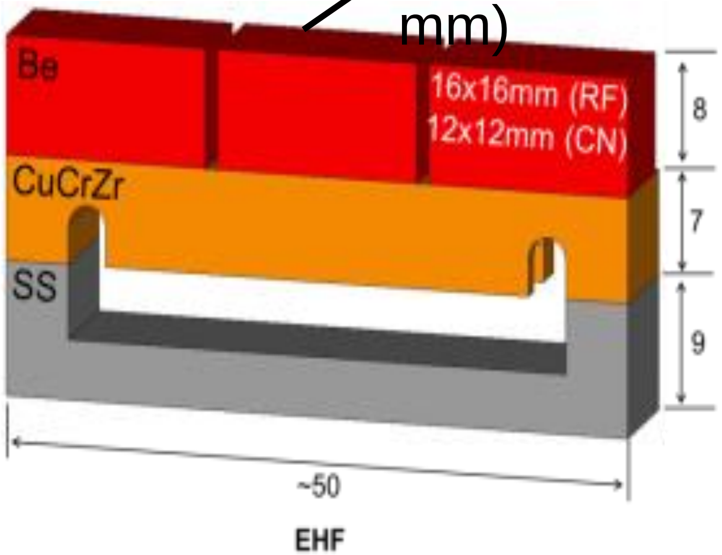


Fig. 45: Energy dependence of the sputtering yield of W with H, D, T, C and W. The data with C were measured at temperatures > 1200°C to avoid built-up of carbon layers. The data are partly published

Runaways Electron assessment



increasing the thickness of W layer from 8 mm to (?? mm)



The max. temperature at the W-Cu interface decreases when increasing W thickness => next step to select the adequate W thickness

New proposal: increase the Tungsten apex thickness to protect from Runaway Electrons

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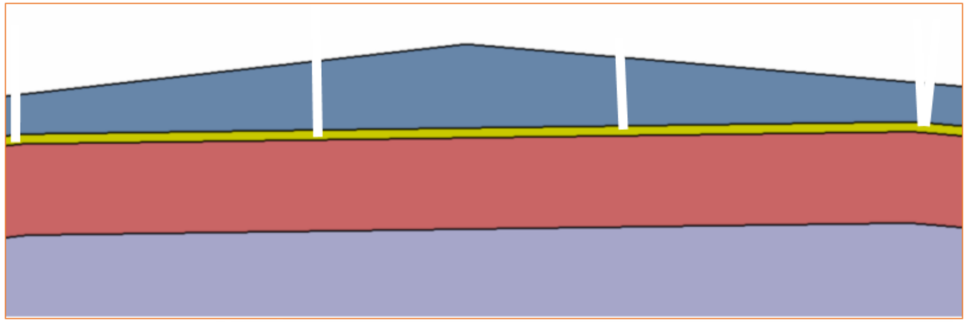
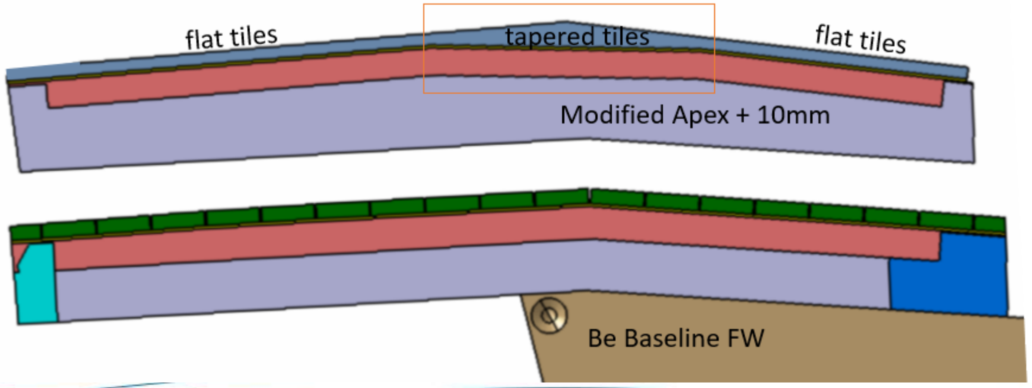
It is thanks to Tungsten’s higher melting point (and structural integrity at higher temperature) that we can consider better protection against RE. This was not possible with Be, as we were up against the max. allowable temperature in Be.

Looking for engineering compromise between cost reduction and manufacturing impact, to achieve the desired benefit of improved protection



The current approach is:

- to keep the structural part as it is
- Modify plasma shaping, and design a variable thickness over toroidal length



*This has cost implications on the manufacturing complexity

Содержание

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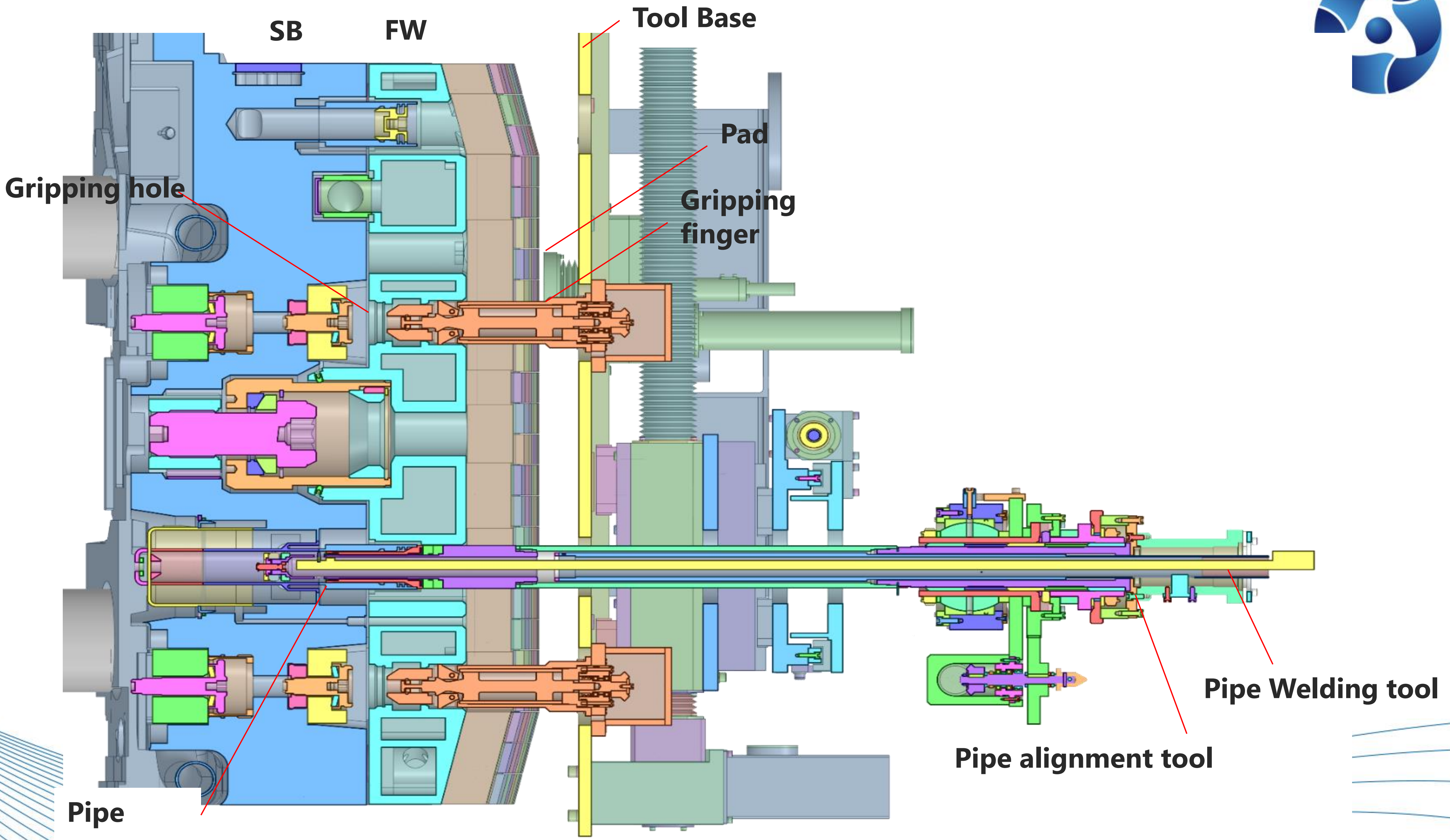
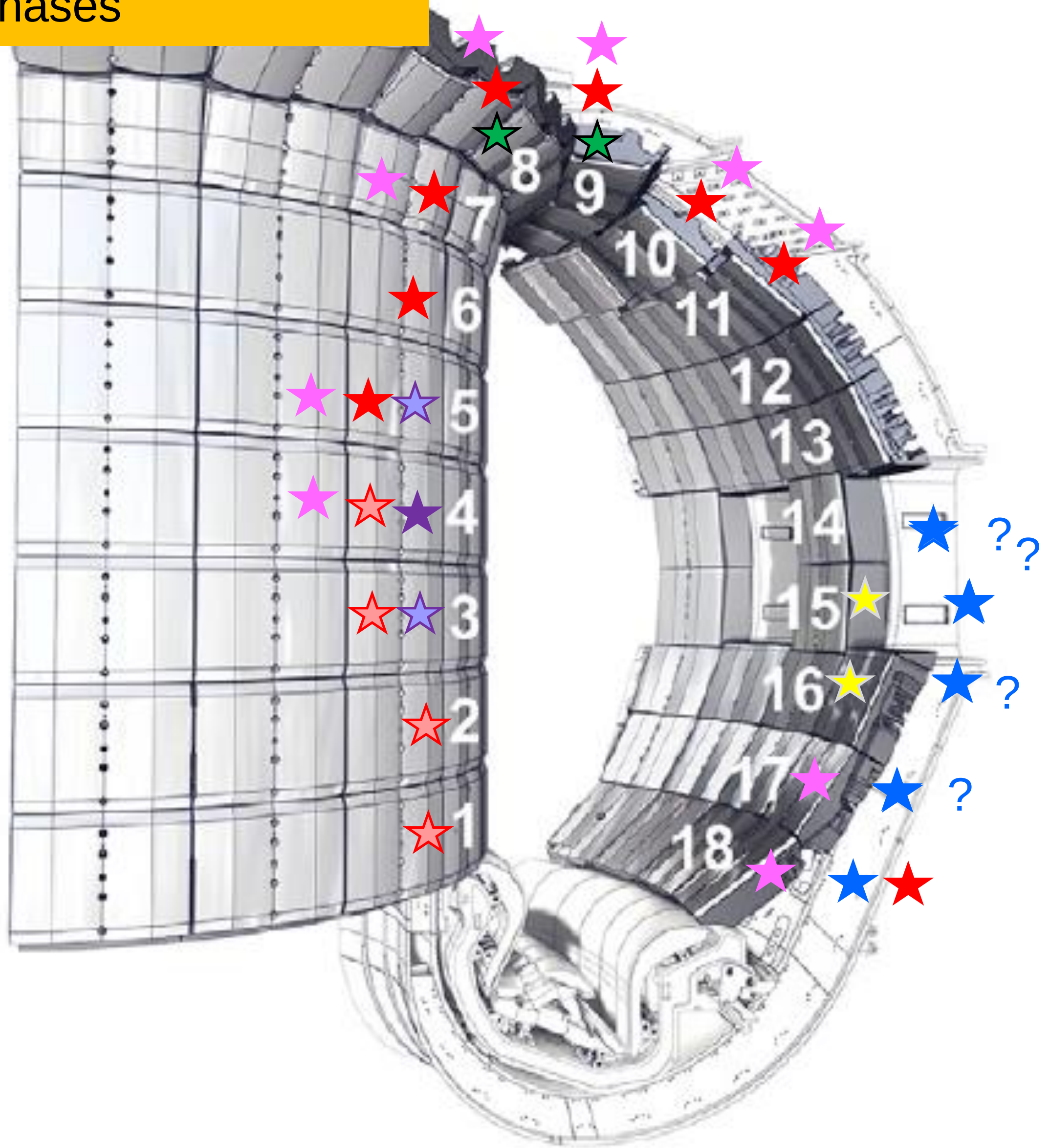


For post-FPA DT phases



Summary of expected plasma thermal loading:

- strongly solicited during Stationary (SS) diverted operation
- strongly solicited during Start-Up (SU) operation
- solicited during Start-Up operation
- possible interaction / loading during Ramp-Down (RD)
- can be subjected to high/moderate fast transient loads (CQ/TQ MD or VDEs)
- chance of RE impact
- possible slideaway RE interaction during S-Up
- NBI shine-through loading on specific FWs

ECRH, ELM filaments?

Hypothesis:

- RD well-controlled to maintain diverted configuration as long as possible when Ip decreases, to avoid too much interactions on the outboard wall
- Inner wall SU (on FW#4) considered as the current baseline strategy

From rationale in ITER_D_3BL4TU - Physics recommendations for required number of First Wall Panel spares by R. A. Pitts

Содержание