Summary for FTP, ITR, and SEE

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Fusion Energy Division

Oak Ridge National Laboratory

USA

24th IAEA Fusion Energy Conference

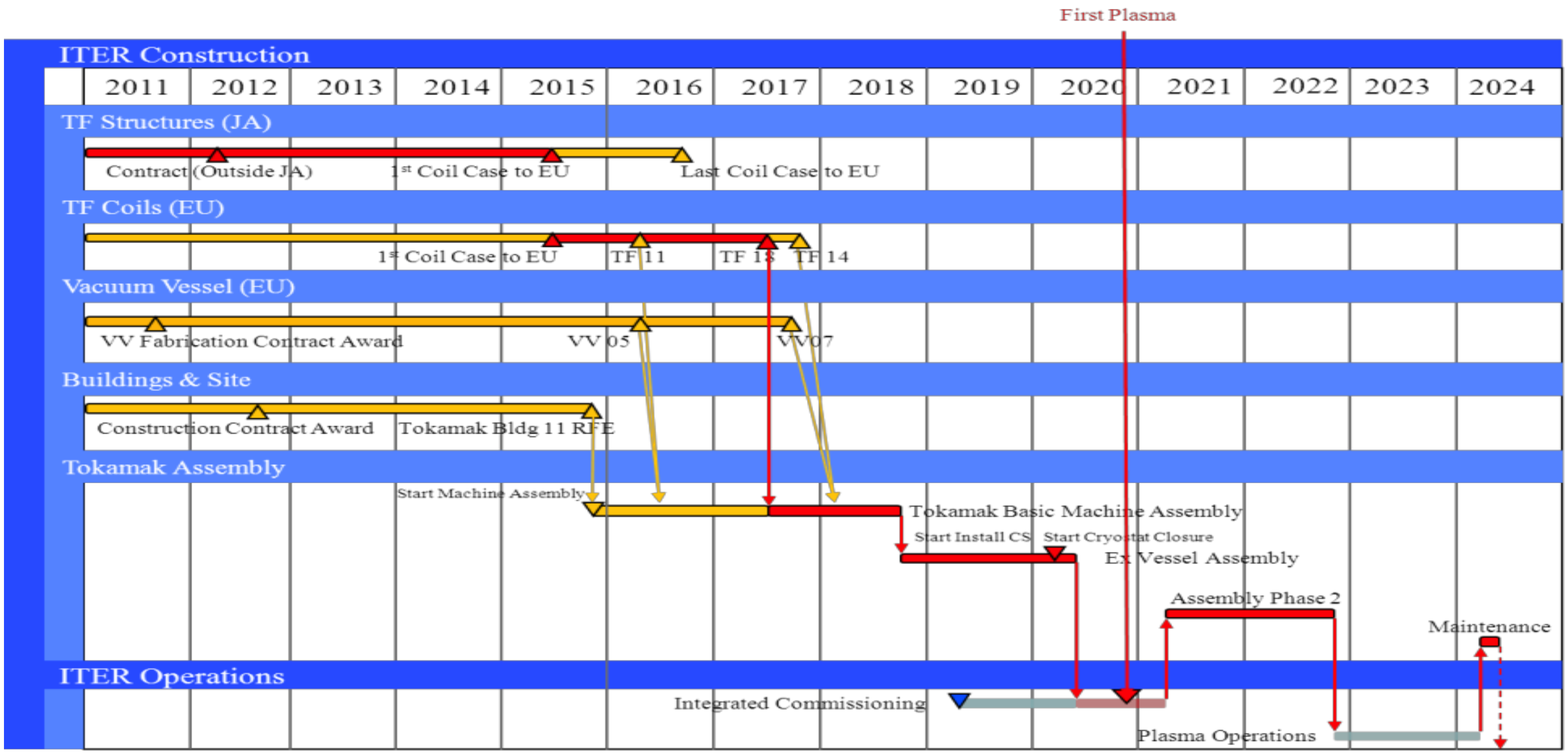
October 13, 2012

Outline of presentation

- Status of ITER and the Broader Approach
- Plasma surface interactions and divertor solutions
- Materials and blanket research
- Road mapping activities and related future facilities
- Concluding remarks

Schedule
ITER Level 0 Reference Schedule

- Machine completion and First Plasma November 2020
- D-T Operation 2027
- Level 0 Reference Schedule within Baseline Boundaries
- We are addressing some potential schedule delays (Building, VV, PF Coils etc.) and working with the DAs on resolving them.
- IO and DAs as UIT are working hard to maintain this schedule.

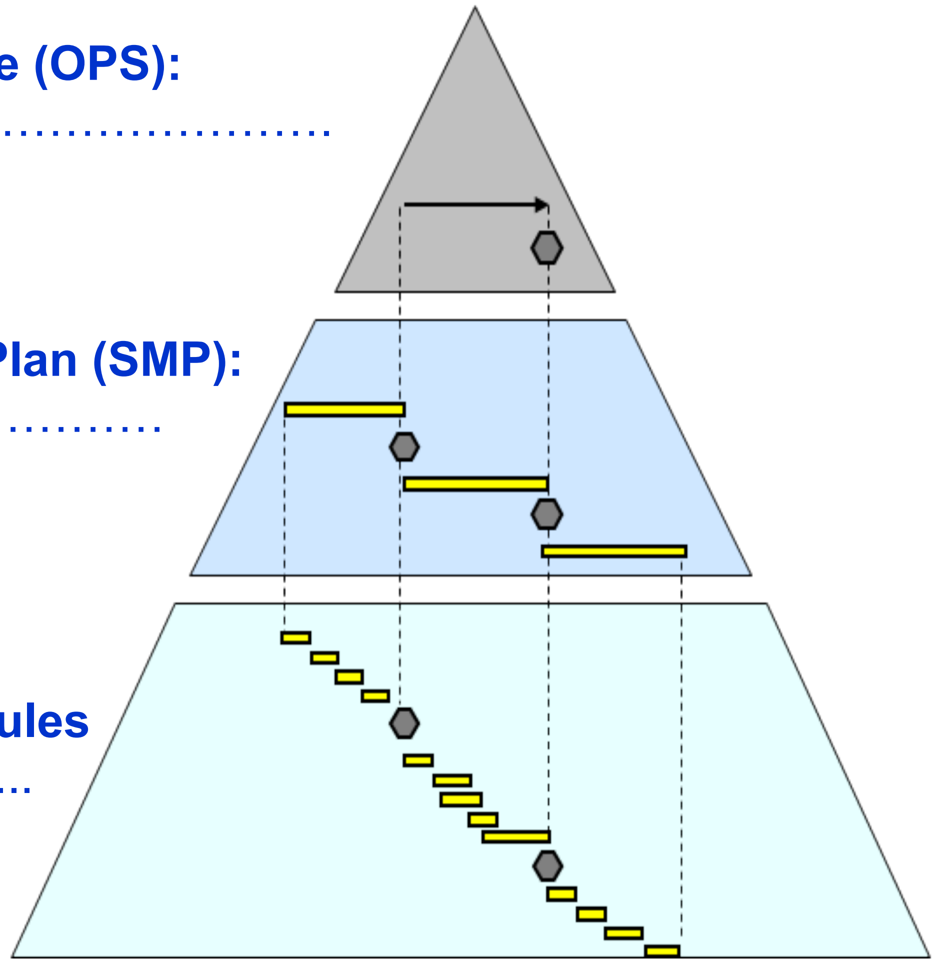


Achievement of Schedule Structure and Management
Completion of DWS

Level 0 Overall Project Schedule (OPS):
~ 50 Activities……………..………………………….

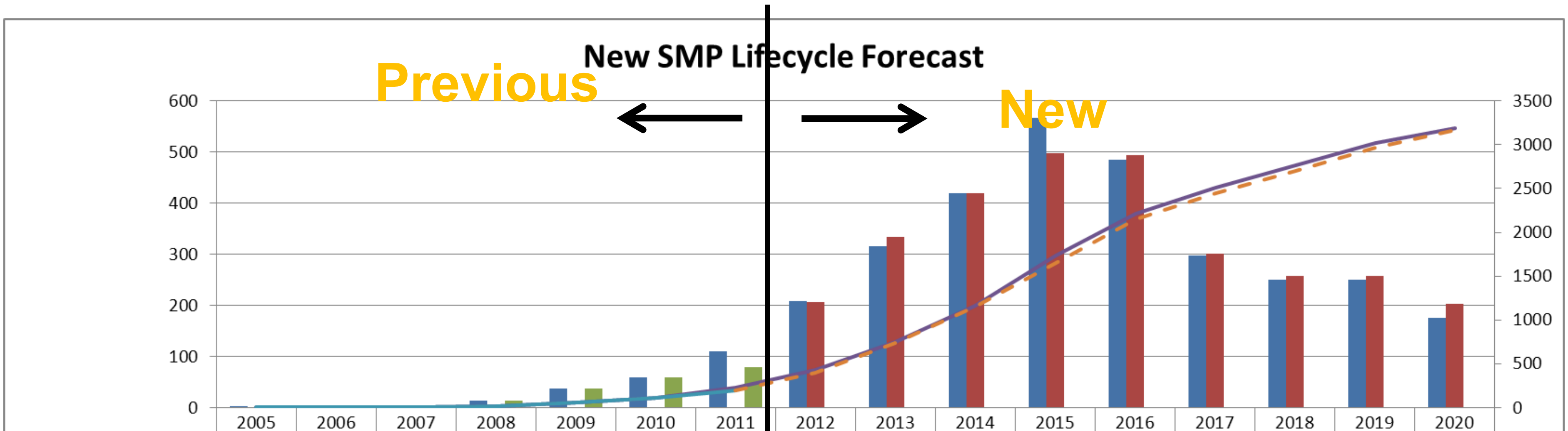
Level 1 Strategic Management Plan (SMP):
~ 3000 Activities ……………………………

Level 2 Detailed Working Schedules (DWS): ~ 150k Activities……………..
(Including IO and DAs)



DWS has been completed on 28th June and reviewed
Milestones are monitored and become controlled.

New SMP Lifecycle Forecast



	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014	2015	2016	2017	2018	2019	2020
Reference	1		2	13	38	60	110	208	315	419	566	484	298	251	251	175
Forecast								207	333	420	497	494	301	258	258	202
Achieved	1		2	13	38	60	79									
Cum Ref	1	1	3	16	54	114	224	432	747	1166	1732	2216	2514	2765	3016	3191
Cum Achieved	1	1	3	16	54	114	193									
Cum Forecast							193	400	733	1153	1650	2144	2445	2703	2961	3163

- Using the new SMP post 2012
- IO and DAs Implementing strong schedule recovery action

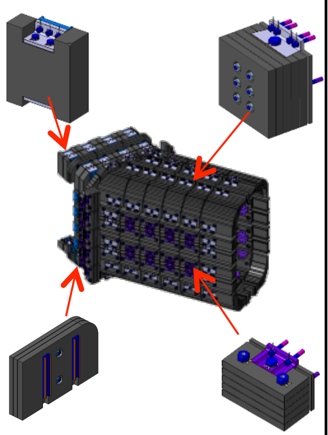
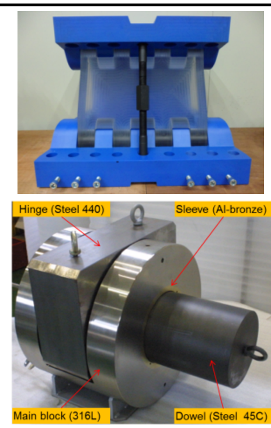
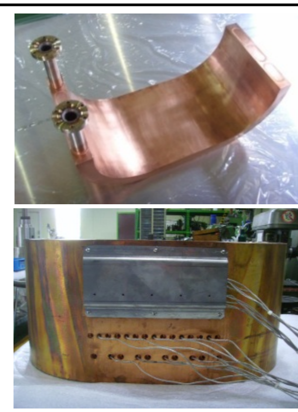
Procurement Arrangements (PA) are important milestones

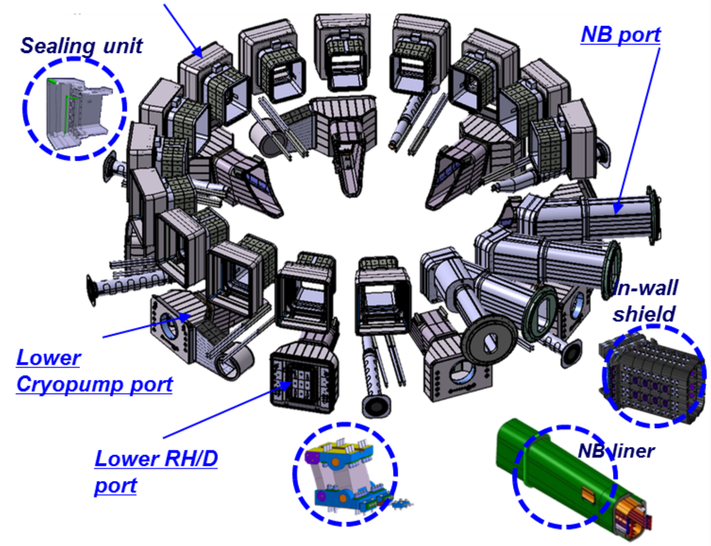
- Total number of signed PA is now **80 out of a total of 137**
- The achieved value to date is 2338.86109 kIUA out of a total In-Kind project value of 2880.52391 kIUA; **81.2 % of value achieved**

Progress on Manufacturing of the ITER VV Equatorial and Lower Ports in Korea (ITR/P5-01)

J.S. Sa

- □ **Manufacturing preparation for the main port is in its final stages and will be completed by the end of 2012**
  - Material procurement, fabrication design and fabrication procedure qualifications
  - Full scale mock-up for lower port stub extension (more than 30 tons) is under fabrication
- □ **Detail designs and fabrication feasibility studies including mock-up fabrication have been performed for the In-wall shield of NB port, VV gravity support and NB duct liner**

IWS of NB port	VVGS	NBDL
 <Detail design>	 <Mock-up for VVGS and sleeve>	 <Full scale mock-up for NBDL>



<ITER VV E/L Ports supplied by KODA>

<Fabrication of full scale mock-up for lower PSE – rib welding on the inner shell>

ITER Magnet Systems – from Qualification to Full Scale Construction

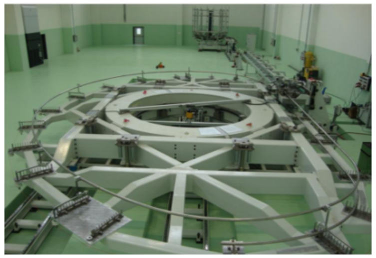
H. Nakajima (JAEA) et, al.

- The qualifications and constructions of ITER superconducting magnets are going well in collaboration between the ITER organization and 6 Domestic Agencies toward the main goal of first plasma in 2020.
- The qualification phase for TF conductors is over and full scale construction of 32% conductors has been completed as shown in the Table.
- The trial manufacture of actual size radial plates for TF winding and full size mock-ups of TF coil structure have been completed. EU and JA are ready for full scale construction.
- Preparations of the manufacturing facilities for windings of the ITER magnet systems are under way in 4 DAs. The final qualifications with actual size prototypes will start soon.

Supplying DA	Total	
Amount of TF strands (tons)	437	
Completion as of 2012.9	348	79.4%
Number of TF Cables	133	
Completion as of 2012.9	49	36.8%
Number of TF Conductors	133	
Completion as of 2012.9	43	32.3%

Radial plate prototype(EU)

TF structure mock-up (JA)

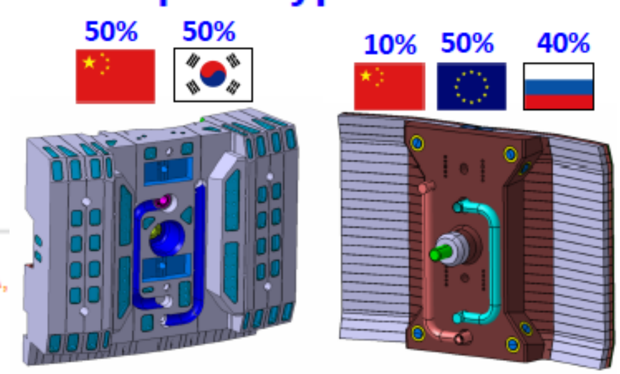


TF winding facility (EU)

CC winding facility (CN)

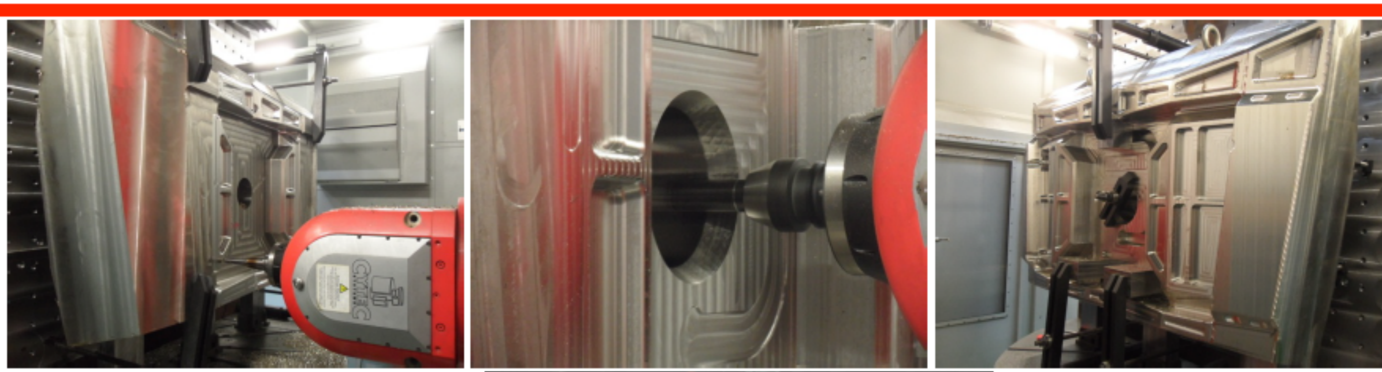
The ITER Blanket System Design Challenge
Raffray ITR 2/6

- The Blanket design is extremely challenging, having to accommodate high heat fluxes from the plasma, large EM loads during off-normal events and demanding interfaces with many key components (in particular the VV and IVC) and the plasma.
- Substantial re-design following the ITER Design Review of 2007. The Blanket CDR and PDR have confirmed the correctness of this re-design.
- Effort now focused on finalizing the design work .
- Parallel R&D program and formal qualification process by the manufacturing and testing of full-scale or semi-prototypes.
- Key milestones:
  - Final Design Review in spring 2013.
  - Procurement to start in late 2013.

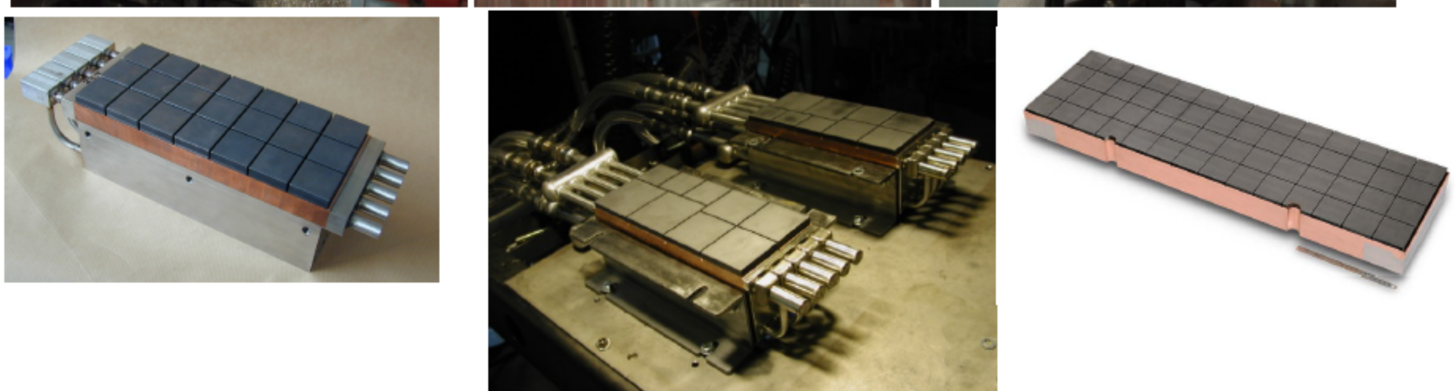


iter china eu india japan korea russia usa | 24th IAEA Fusion Energy Conference, San Diego, CA,

Mock-Ups and Prototypes Are Being Manufactured as Part of the Qualification Programs



Shield Block by KODA



First Wall by EUDA

First Wall by RFDA

Technology R&D Activities for the ITER full-W Divertor

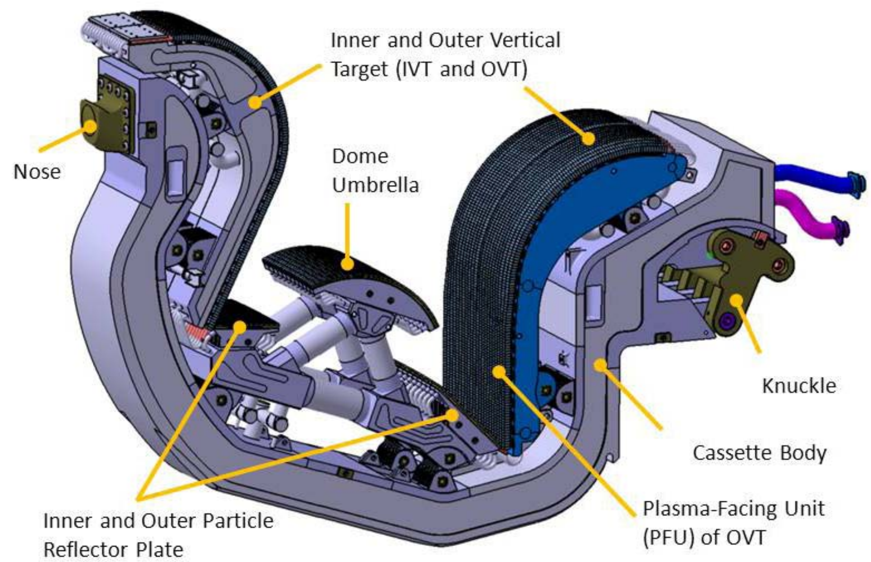
Lorenzetto ITR/2-3

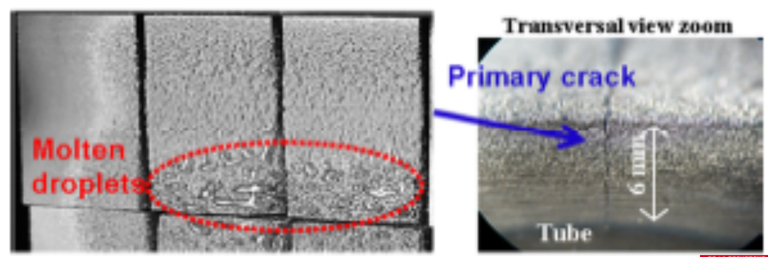
- ➢ To decrease costs, the ITER Organization has proposed to reduce by one the number of divertors planned during the ITER lifetime and to begin operations with a full-tungsten (W) variant in place of the original first divertor armoured with carbon fibre composite in the high heat flux regions;

- ➢ The design of the W divertor is being prepared by the IO;

- ➢ The fabrication technologies developed by the procuring Domestic Agencies, Europe, Japan and the Russian Federation, are being adapted to the new design requirements;

- ➢ A decision on which divertor to manufacture for ITER is scheduled to be taken by the end of 2013.





W monoblocks after 20 MW/m² thermal fatigue

Achievement of ITER relevant parameters with RF gyrotron

In the last five years four gyrotron prototypes were fabricated and tested.

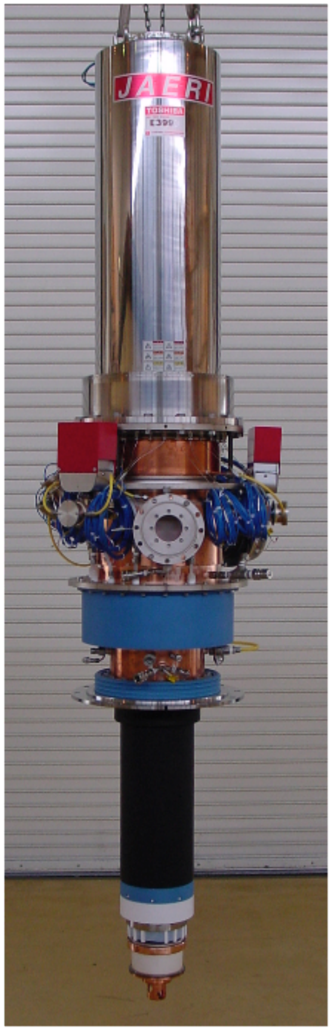
Gyrotrons V-10, V-11 were tested in 2010 and 2011 respectively with CRYOMAGNETICS LHe –free magnet.

It is important to note that two last gyrotrons (V-10 and V-11) demonstrate very similar output parameters (see Table below).

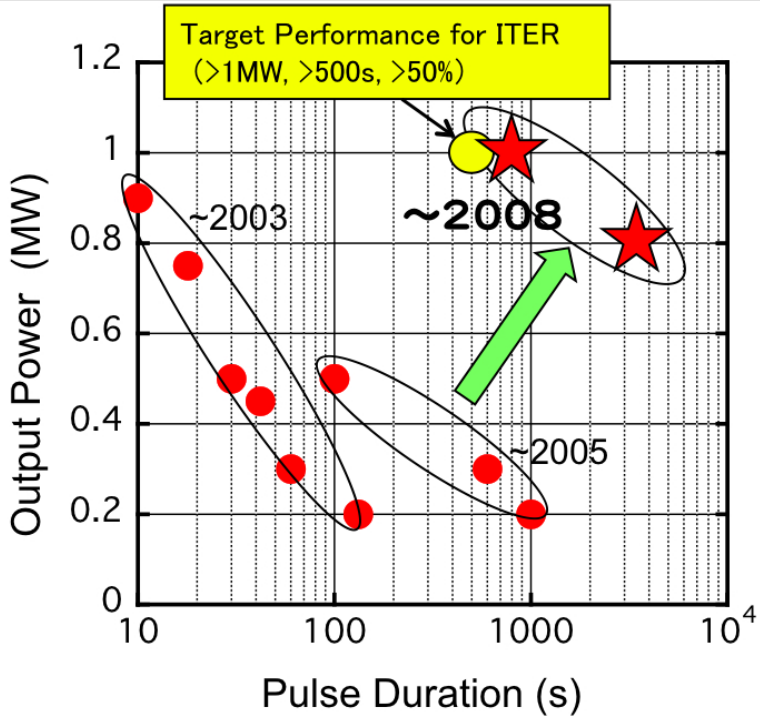
Gyrotron	Beam voltage kV	Beam current A	Retarding voltage kV	Power * kW	Efficiency %	Pulse duration sec
V-10 V-10 V-11	71 71 70	34 34 39.5	30.5 30 30	~750 ~750 ~850	~54 ~54 ~53	1000 600 ** 1000
V-10	70	45	31.5	~960	~55	400 (serial pulses)
V-11	**70.5**	**45**	**31.5**	**~960**	**~55**	**1000**

170GHz Gyrotron Development in JAEA

Kajiwara FTP/1-3Rb)



170GHz gyrotron



<u>TE31,8 mode gyrotron</u>

- 1MW/800s
- 0.8MW/1hr operation
- Max. efficiency: ~60%
- Total output energy: >250GJ

- <u>For power increase, higher mode oscillation has been challenging</u>

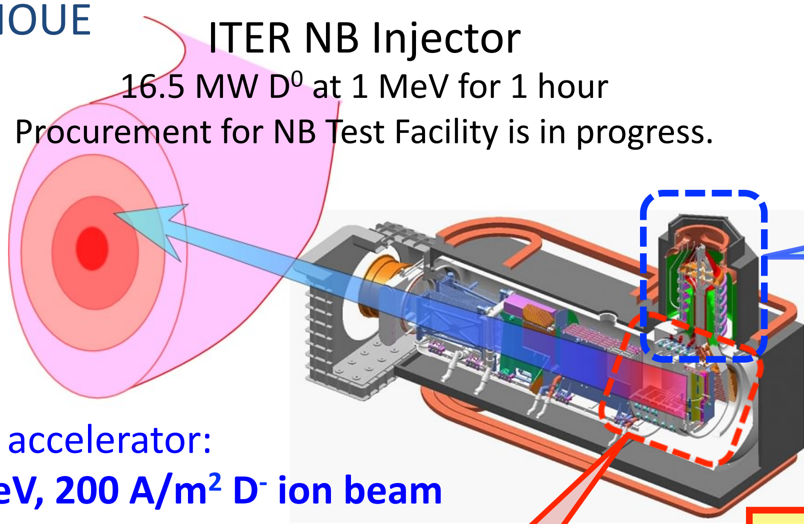
TE31,11 mode gyrotron (J7)
(Multi-frequency operation, >1.3MW operation)

- 5kHz power modulation is improved using novel anode switching

Acceleration of 1 MeV H^- Ion Beams at ITER NB-relevant High Current Density

FTP/1-2
T. INOUE

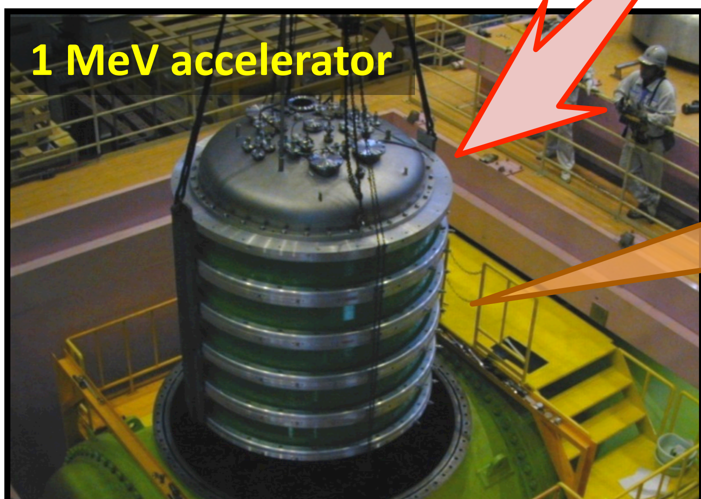
ITER NB Injector
16.5 MW D^0 at 1 MeV for 1 hour
Procurement for NB Test Facility is in progress.



A single stage full-size mockup
achieved 240 kV (120 %) for 2 hours.

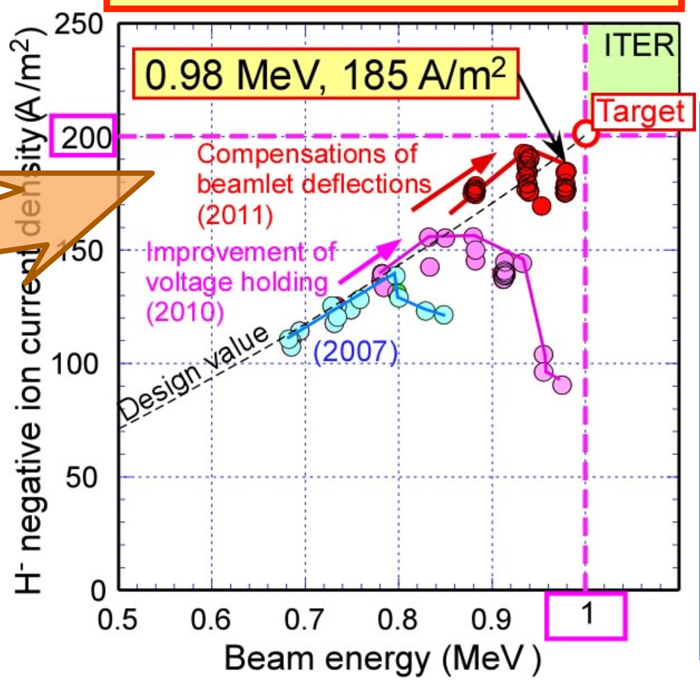
A two stage full-size mockup, test interrupted by 3.11.

ITER accelerator:
1 MeV, 200 A/m^2 D^- ion beam
(total: 40 A from 1280 apertures)



Achievement by 2012:
0.98 MeV, 185 A/m^2 H^- ion beam
(total current: 0.46 A from 15 apertures)

Status of accelerator R&D



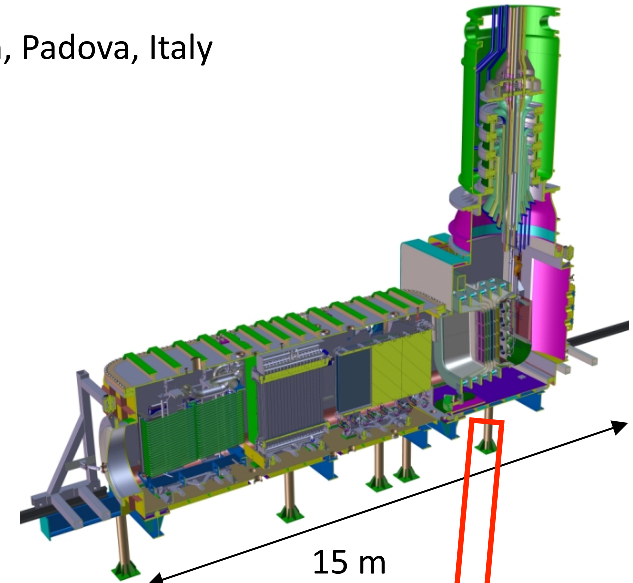
Next steps

- **Long pulses (~ 60 s) with:**
  - low grid heat load
  - High reliability
- **Beam focusing**
- **Detail design of ITER accelerator.**

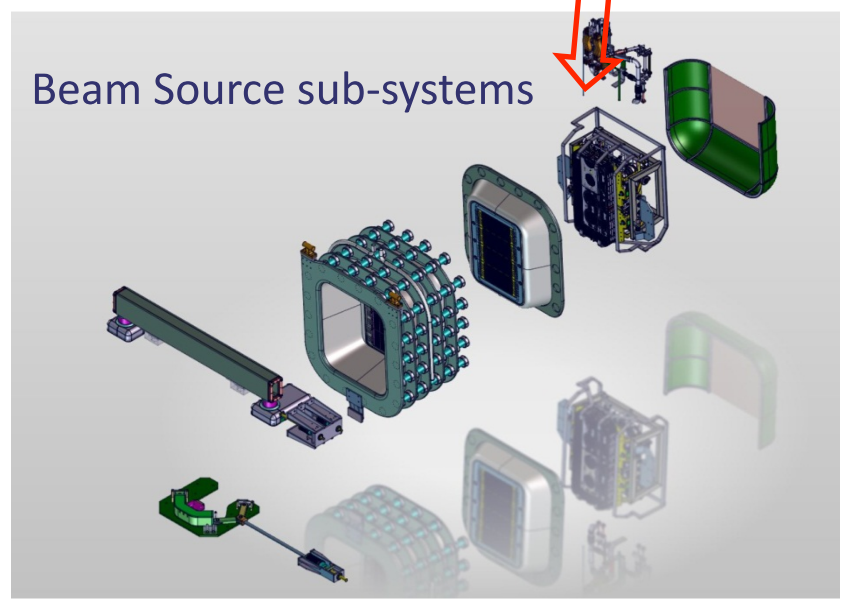
ITR/1-3

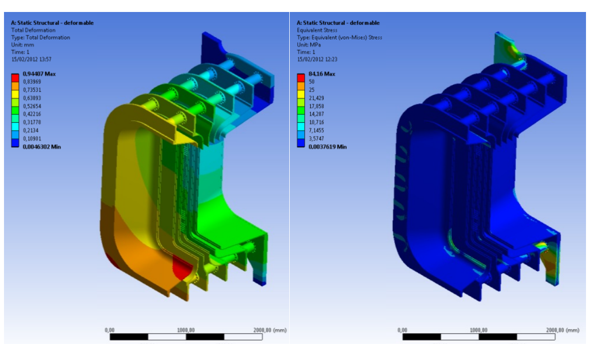
Design of the MITICA neutral beam injector: from physics analysis to engineering design

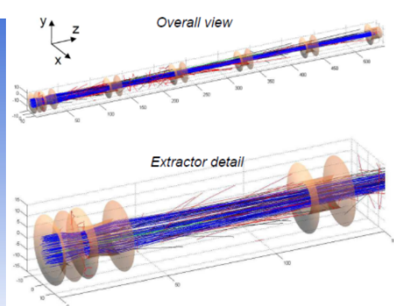
P. Sonato, et al.
Consorzio RFX, Euratom-ENEA association, Padova, Italy



- The paper presents the development of the design of MITICA, the prototype heating neutral beam injector for ITER, starting from physic requirements up to the development of the engineering requirements by using a number of original and commercial numerical analysis codes.
- An integrated design approach is necessary and involves all the design stages of all the components and plant systems. The progress of the design foresees a continuous cyclic check of the requirement consistency towards a robust design fulfilling all physics needs and constraints and all engineering practices in all the technology fields.
- In 2012 the design is converging towards the completion of the main design activities of all the main items and main plant systems of MITICA.

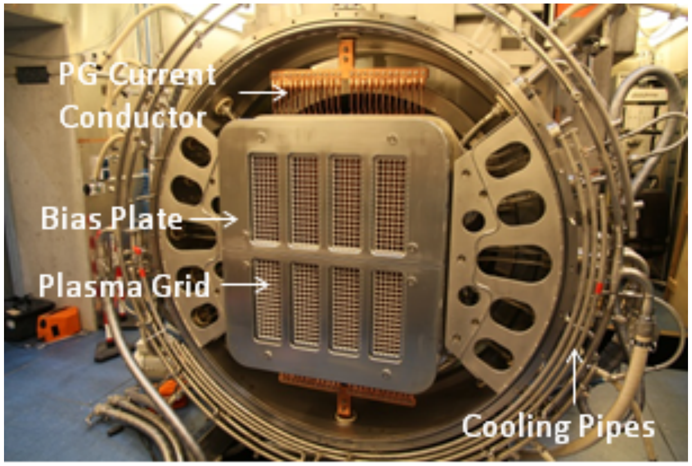






The IPP NNBI test facility ELISE:

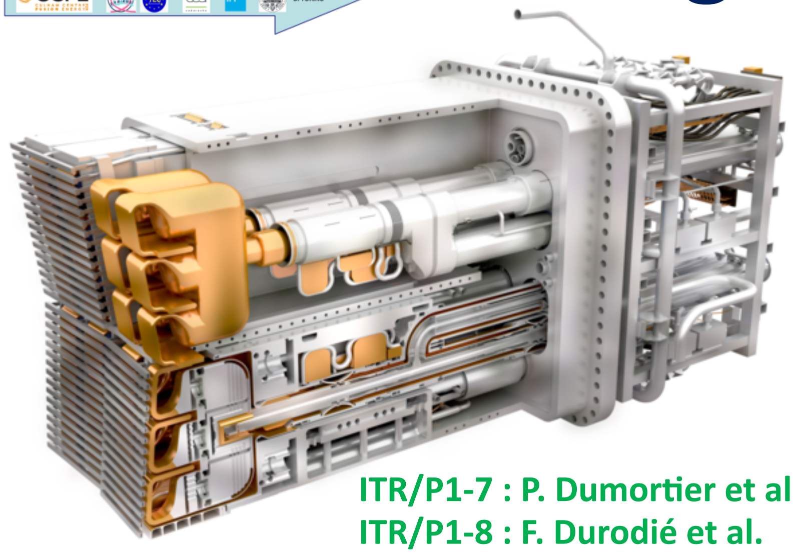
- important intermediate step of the development of the ITER NBI system
- allows gaining an early experience of the performance and operation of large RF driven sources for negative hydrogen ions
- will give an important input for the commissioning and the design of the SPIDER and MITICA test facilities at Padua and the ITER neutral beam system.
- is now ready to go into operation



Extraction system assembly within the HV flange and the plasma grid current conductors.

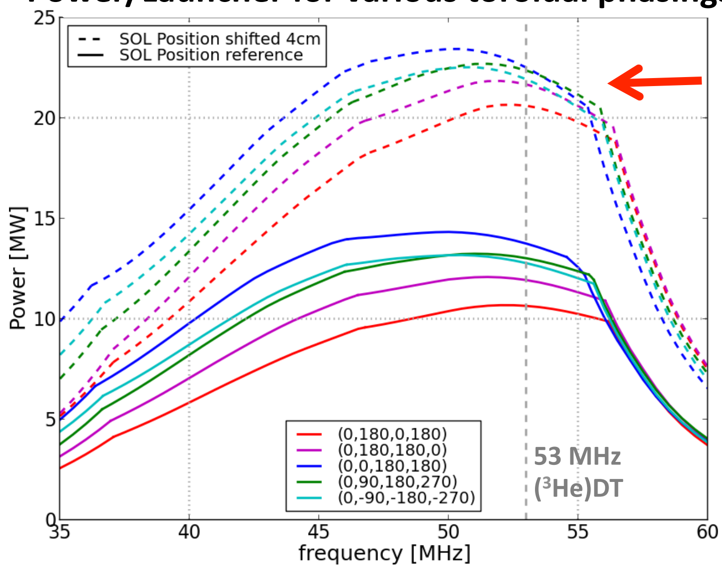
This early start is an important prerequisite for establishing the neutral beam system at ITER in-time for the first plasma pulse.

RF Design of the ITER ICRF Launcher



ITR/P1-7 : P. Dumortier et al.
ITR/P1-8 : F. Durodié et al.

Power/Launcher for various toroidal phasings



ITER ICRF System : Two Launchers

- 20 MW / launcher capped by 45 kV system voltage
  - Phase 1 : 20 MW total from 24 MW installed generator power
- Designed on behalf of IO by CYCLE *
  - Consortium of EU institutions partially funded by IO through F4E (EU DA)

RF Design completed

- Successful Preliminary Design Review (May 2012)
- Performance depends crucially on SOL :
  - TOPICA coupling code : validated on JET-ILA, DIII-D, TS, …
  - Worst case SOL profile received from IO assumed
  - Expect to be able to move the SOL profile towards the antenna
- RF properties extensively validated on Mock-ups
- RF Sheaths under assessment

Mechanical Design

- Approaching detailed design phase
- Closure of preliminary design issues expected 3/2013

R&D Program outstanding

- Vacuum Ceramic Windows, RF Contacts, Faraday Screen Bars, …

The project F4E-2009-GRT-026 has been funded with support from Fusion for Energy (F4E). The views and opinions expressed herein do not necessary reflect those of Fusion For Energy or European Commission or ITER Organisation. F4E is not liable for the use which might be made of the information in this publication.

*

ITER Diagnostics – Technology and Integration Challenges
Johnson ITR/2-1

- In the last 2 years, development has entered a new phase, with design responsibility transitioning from the ITER Organization to the DAs.
- Significant progress in the integration of the diagnostics into the ports.
  - Modular approach for port plug integration using standard interfaces promotes parallel diagnostic front-end designs and planning for remote handling.
  - Nuclear dose-rate allocation budget strategy and plan to reassess during design process.
  - Compatible diagnostic first wall design
  - Standardized modular approach to interspace and port cell support structures.
  - Mature plan for qualification testing of integrated port plug in standard test facility.
- Conceptual Design Reviews have identified numerous measurement challenges
  - Development of radiation-tolerant magnetic sensors
  - 1st mirror degradation for optical diagnostics
  - Strategies to suppress reflected light
  - High expected T_e poses challenges in interpretation for some techniques.
  - ECE radiometer needs to accommodate 2 views and full harmonic spectrum, to assess electron energy distribution
  - Vulnerability to damage from stray microwave radiation
- Refinement of requirements continues as Procurement Arrangements are signed, providing solid framework for DAs to begin Preliminary Design and R&D phase.

Development of a new laser system for plasma diagnostics with the world's highest performance

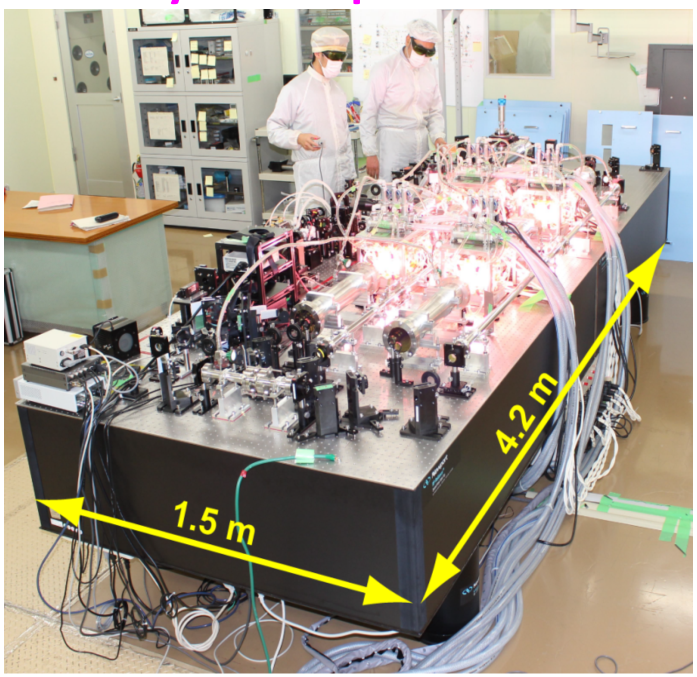
- ITER edge Thomson scattering system requires a pulsed YAG laser with high output energy of 5 J and high repetition rate of 100 Hz (average power of 500 W)

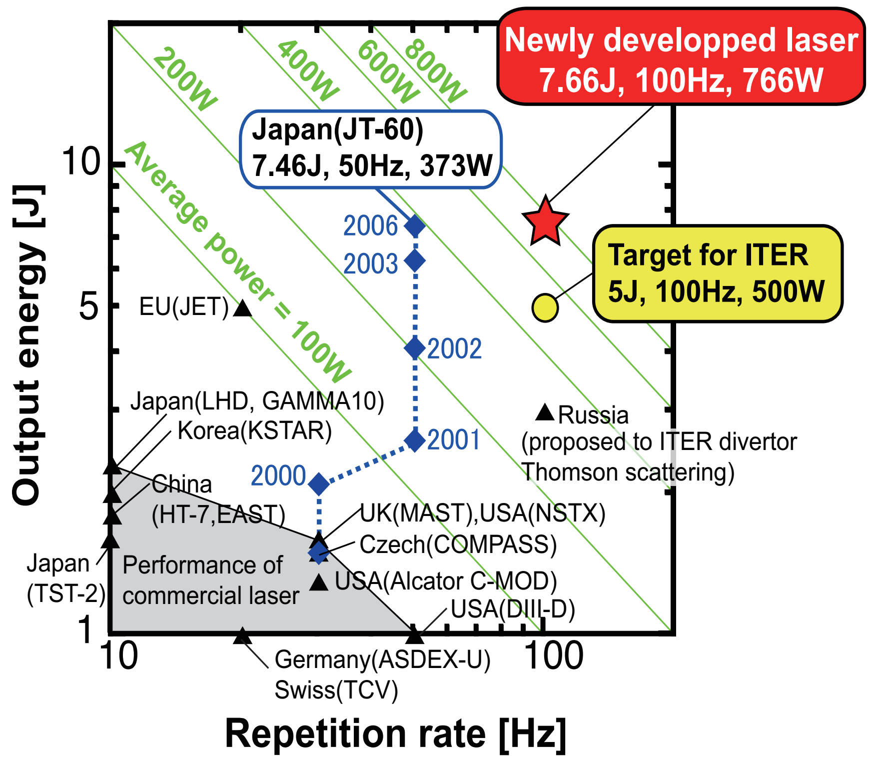
➔ Development of a new laser system

- Output energy of 7.66 J and repetition rate of 100 Hz have been achieved (average power of 766 W, which is twice the previous record)

➔ Required laser performance for ITER has been successfully exceeded

Newly developed YAG laser





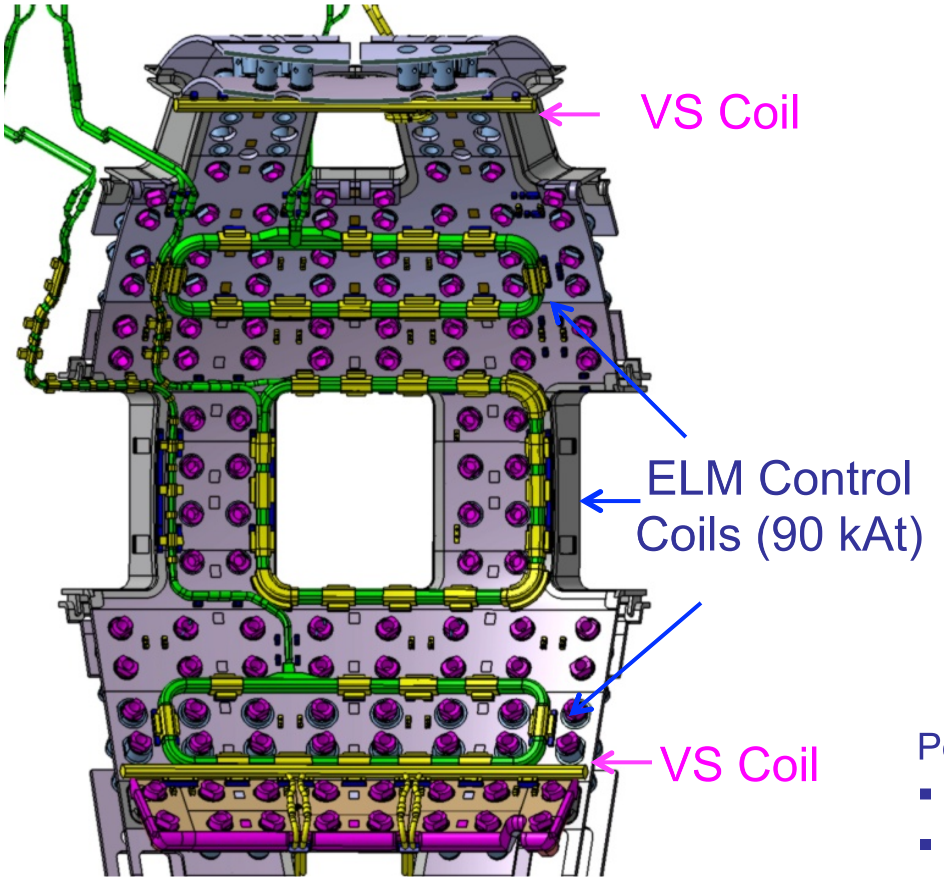
Y. KAWANO et al., ITR/ P5-35

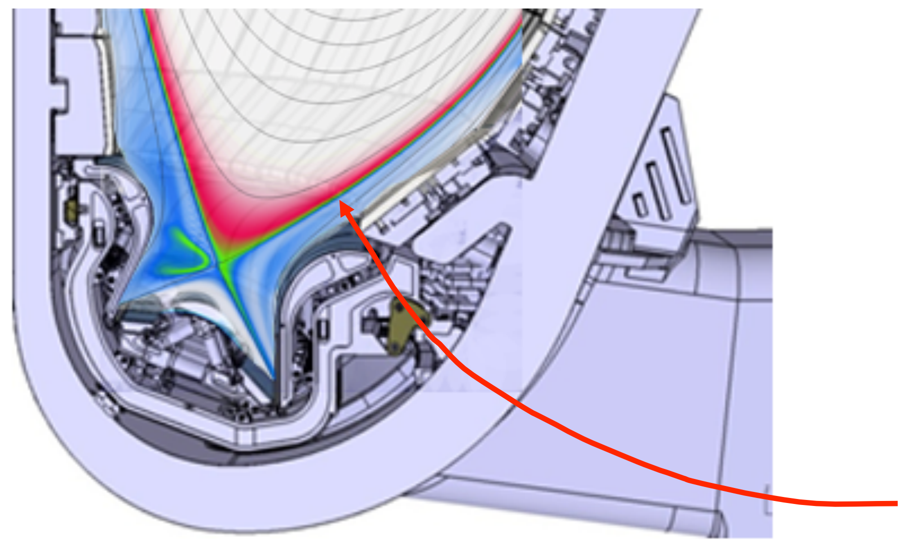
Technical challenges requiring fusion community engagement (many of these have fusion technology components)

- Significant progress in modelling of physics processes underlying ELMs and ELM control
- Disruption Mitigation System Conceptual Design
- Full revision of predicted heat loads on PFCs in support of First Wall and Tungsten Divertor design
  - The final design phase is planned to be completed by the end of 2013
- Conceptual design of Plasma Control System moving towards completion
- Extensive experimental and modeling R&D continuing in support of development of ITER disruption/ runaway electron mitigation systems

ELM Control in ITER : Schemes

- Two systems for ELM control in ITER : ELM Control Coils & Pellet Pacing
  - ELM Control Coils → Aimed at suppression of Type I ELMs → $\Delta W_{ELM} \sim 0$
  - Pellet injection → Controlled triggering of Type I ELMs → $\Delta W_{ELM} < \Delta W_{ELM}^{controlled}$





Maruyama - ITR/P5-24

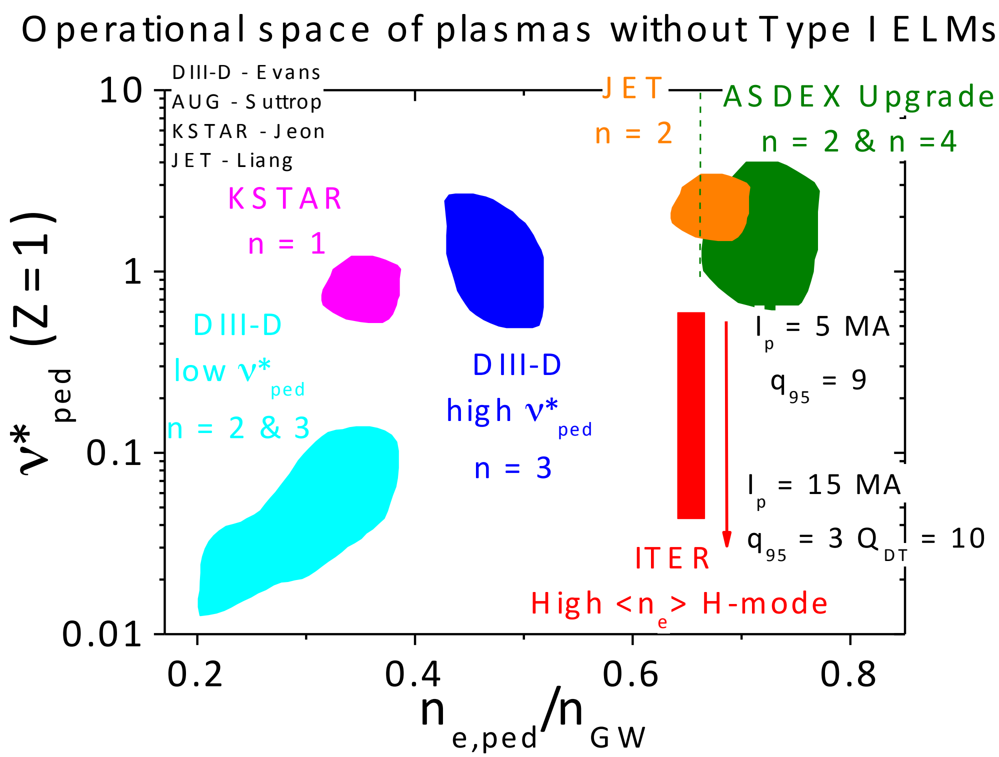
Pellet parameters :

- v_{pellet} = 300 – 500 ms^{-1}
- V_{pellet} = 17-34 mm^3(pacing)/50-90 mm^3(core fuelling)
- f_{pellet} (per injector) = 4-16 Hz

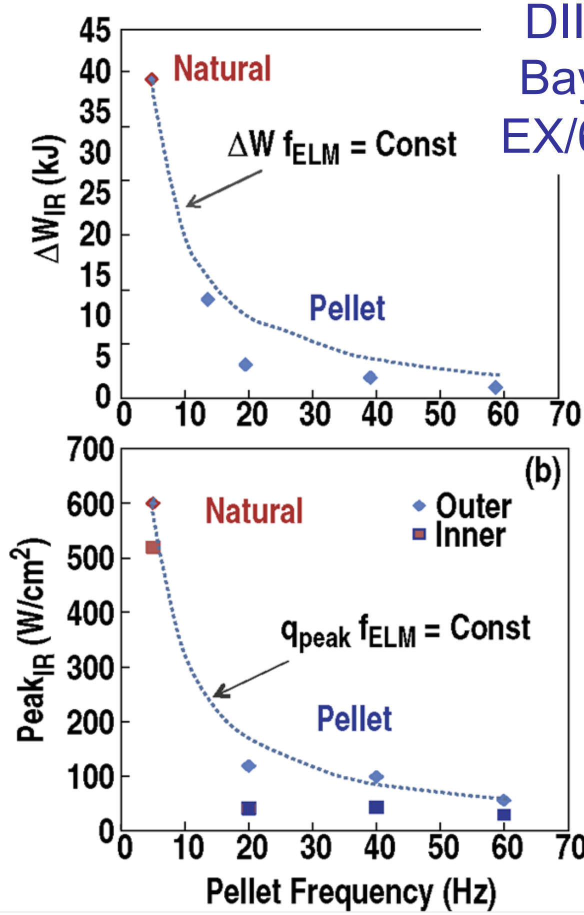
ELM Control in ITER : Summary of Experimental Progress

➢ Major progress on application of both techniques to suppress or control Type I ELMs:

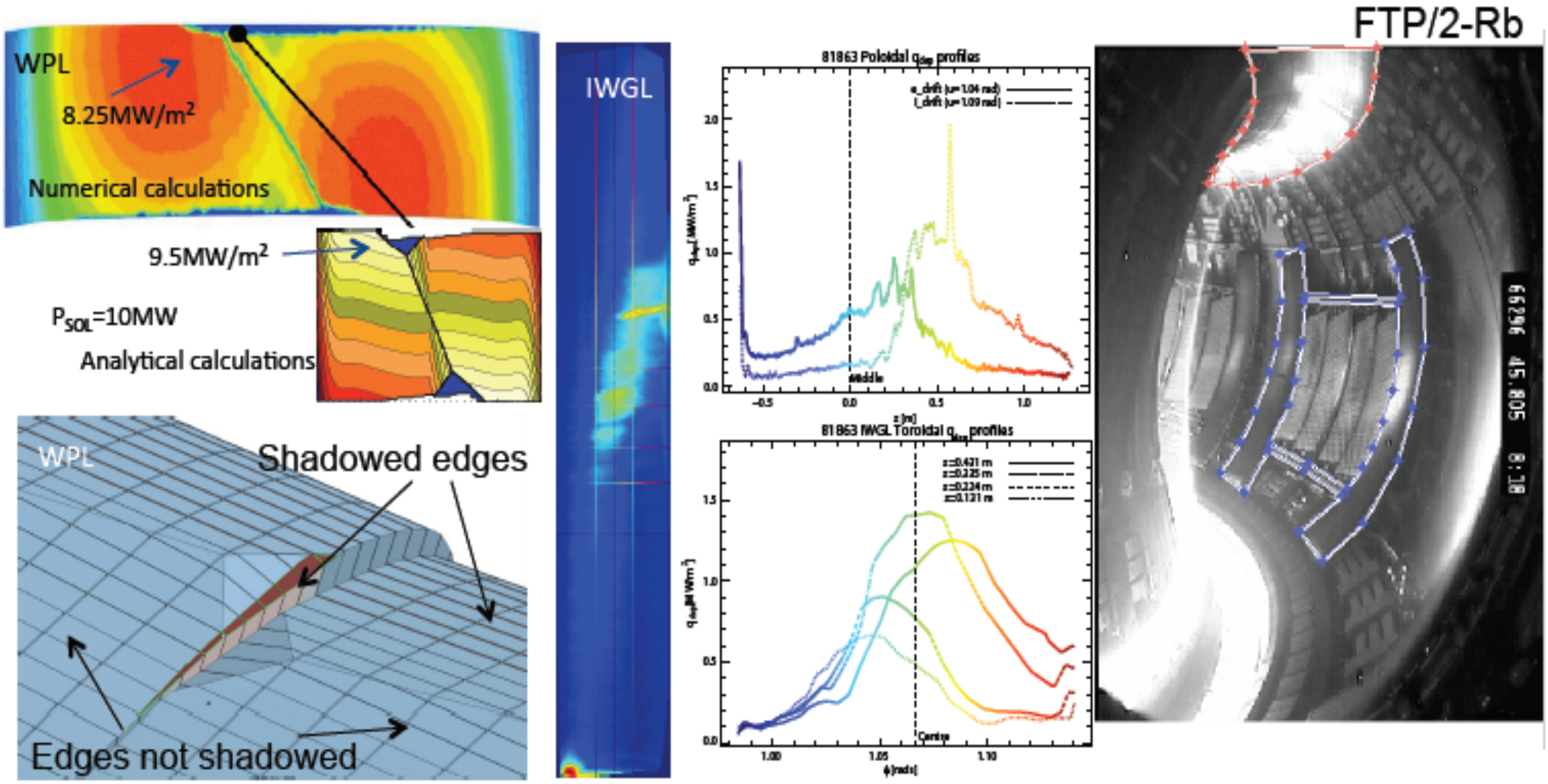
- Elimination of Type I ELMs with coils observed in DIII-D, AUG, KSTAR and JET
- ELM control by pellets demonstrated in AUG, JET and DIII-D → latest DIII-D experiments have accessed ITER relevant range of ELM control



DIII-D Baylor EX/6-02



Be tile power handling and main wall protection



- Be tiles optmised for high heat fluxes and long pulse by surface shaping to achieve constant power distribution on the tile and shadowing of exposed edges larger than 40μm.
- Analytical and numerical calculations validated experimentally. Good power handling achieved as expected.
- Active protection successful in preventing and terminating plasmas that can lead to Be melting.

ipfn

Sustained Regulation of Current Profile and β_N Demonstrated in DIII-D Using Model-Based Profile Control Algorithms

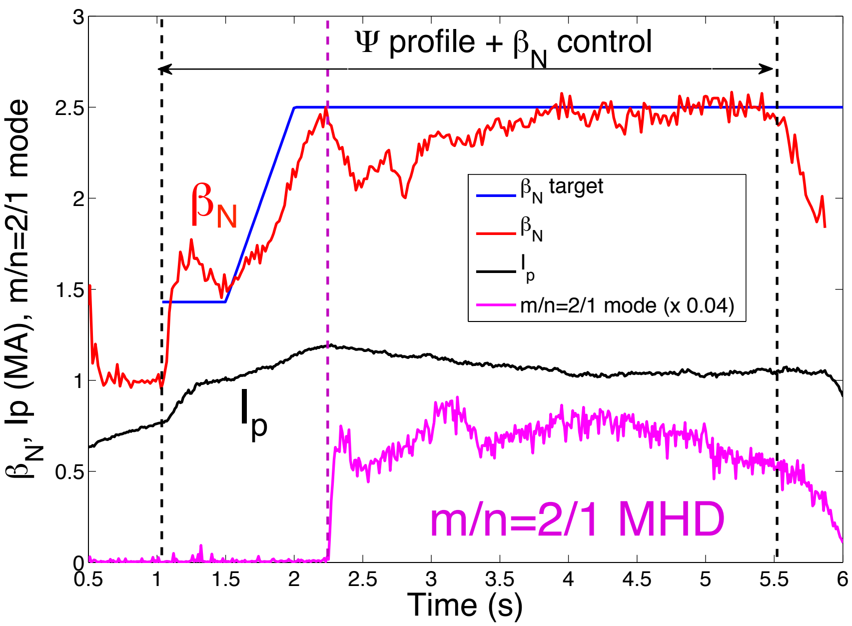
******
NEW
******

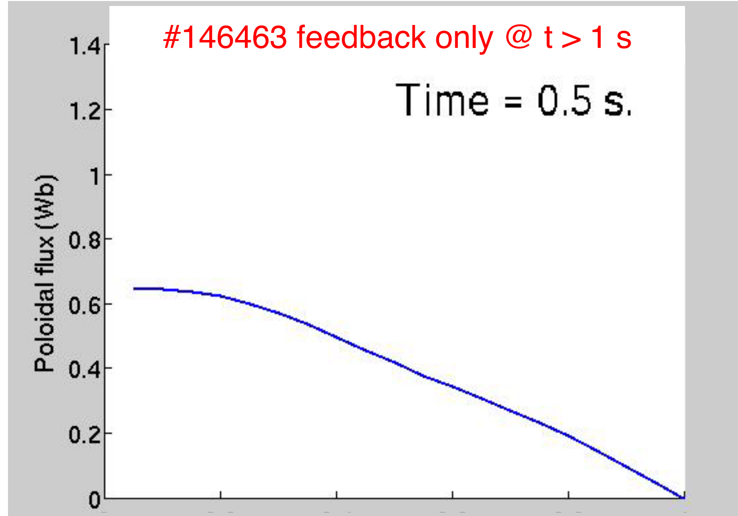
CONTROL ACTUATORS

1. On-axis co-current NBI
2. Off-axis co-current NBI
3. Counter-current NBI
4. Balanced NBI
5. Off-axis ECCD
6. Surface loop voltage

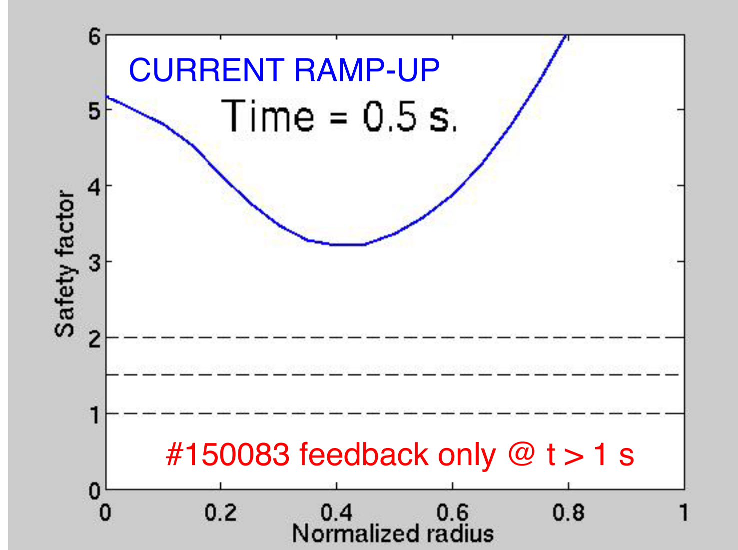
$\Psi(x)$ + β_N control

Shot #146463





ψ-profile movie



q-profile movie

D. Moreau, M. Walker, J. Ferron, F. Liu, E. Schuster, et al.

ITR/P1-20

FTP/4-3

Initial results of the large liquid lithium test loop for the IFMIF target

H. Kondo[1], T. Furukawa[1], Y. Hirakawa[1], T. Kanemura[1], M. Ida[1], K. Watanabe[1], K. Nakamura[1],
H. Horiike [2], N. Yamaoka [2], I. Matsushita [3], H. Iuchi [3], F. Groeschel [4], and E. Wakai [1]
[1]Japan Atomic Energy Agency, [2] Osaka University,
[3]Mitsubishi Heavy Industries Mechatoronics Systems, [4]IFMIF/EVEDA Project Team

Construction and initial performance tests of the EVEDA (Engineering Validation and Engineering Design Activities) Lithium Test Loop (ELTL) were completed (a Li target flow at a velocity of 5 m/s was successfully achieved in the target assembly).

One of the major milestones in the engineering validation toward the IFMIF (International Fusion Materials Irradiation Facility) was accomplished.

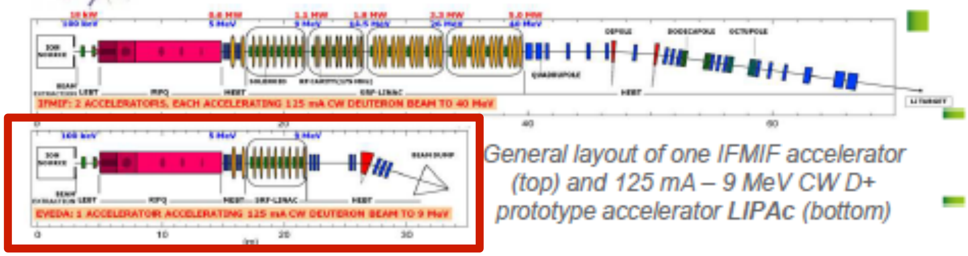
Front view of the EVEDA Li Test Loop (ELTL)

Li target flow at 735 l/min (5m/s)

OVERVIEW ON CEA CONTRIBUTIONS TO THE BROADER APPROACH PROJECTS

354 - FTP/P1-29

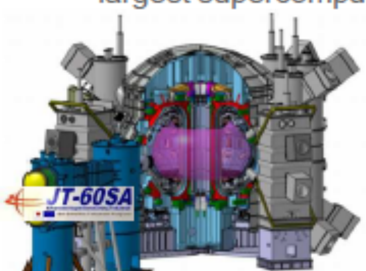
3 large scale Projects jointly implemented by Europe and Japan since 2007 in support to Iter and preparation to DEMO



General layout of one IFMIF accelerator (top) and 125 mA – 9 MeV CW D+ prototype accelerator ***LIPAc*** *(bottom)*

- IFMIF: Eng. Design and Validation of a 14 MeV ṅ irradiation facility for material testing
  - CEA contribution: from design to supply of sub-systems of LIPAc prototype
  - LIPAc operation at nominal 1 MW beam at Rokkasho site by June 2017

- IFERC: DEMO R&D activities, Iter Remote Experimentation and Supercomputer Centre for large scale simulation activities.
  - Petaflops class « Helios » supercomputer (Bull) incl. operation and maintenance (2012-2016) provided by CEA
  - In operation since Jan. 2012, ranked #12 in the latest Top500 list, the largest supercomputer in the world dedicated to a single scientific



JT-60SA "artist view"
~ JET size, 1/5 of Iter Ip

- JT-60SA: Fully SC tokamak to contribute to the early realization of fusion energy by supporting the exploitation of ITER and research towards DEMO.
  - CEA responsible for the delivery of 9 of the 18 NbTi SC Toroidal coils and cold test of the whole ➔ last tested coil at Naka site by begining of 2017.
  - Procurement of the 9 kW @ 4,5 K cryoplant delivered on site by end of 2016
  - Coordination of the Eu. activity for revisiting the JT-60SA research plan and activities of scenario simulation, jointly with the JT-60SA team.

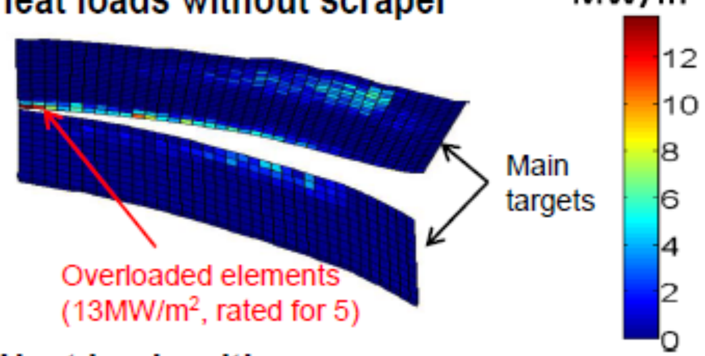
PLASMA SURFACE INTERACTIONS AND DIVERTOR SOLUTIONS

New W7-X divertor component will prevent overload of main targets

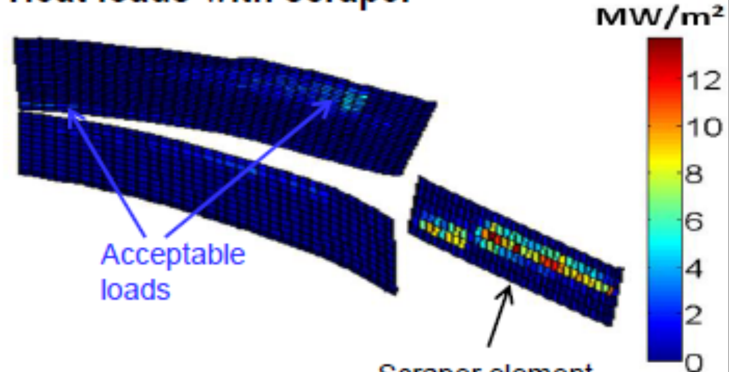
Lore FTP/P1-2

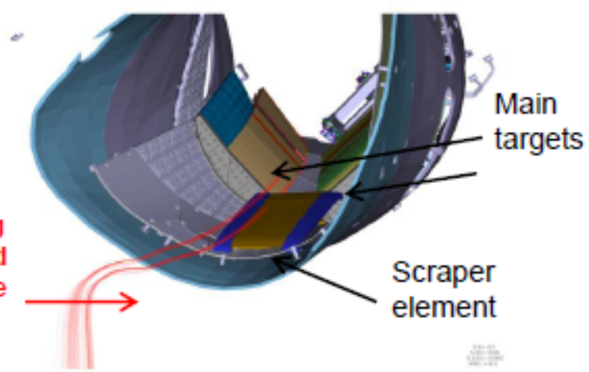
- **Problem: W7-X magnetic configuration changes as plasma temperature, density is increased**
  - Configuration evolution results in overload of main target edges
  - ~11MW/m^2 on sections rated for 5MW/m^2
  - Occurs on ~40s timescale, effectively steady-state thermal loads
- **Solution: New 'scraper element' intercepts heat and particle fluxes to protect main targets**
  - Element will be built using actively cooled carbon fiber composite (CFC) monoblocks of type qualified for ITER
  - Steady-state heat fluxes ~13MW/m^2 (rated for 20MW/m^2)

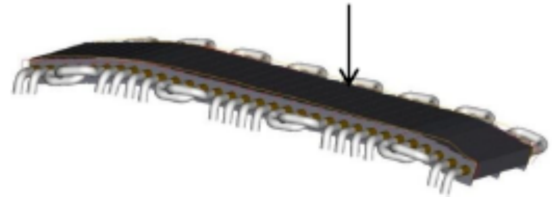
Heat loads without scraper



Heat loads with scraper



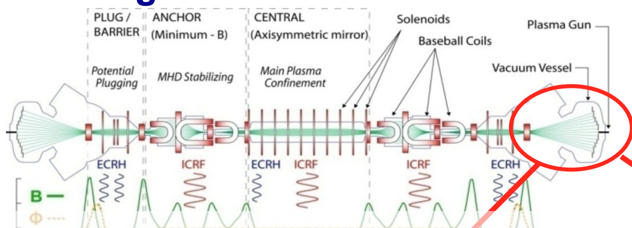




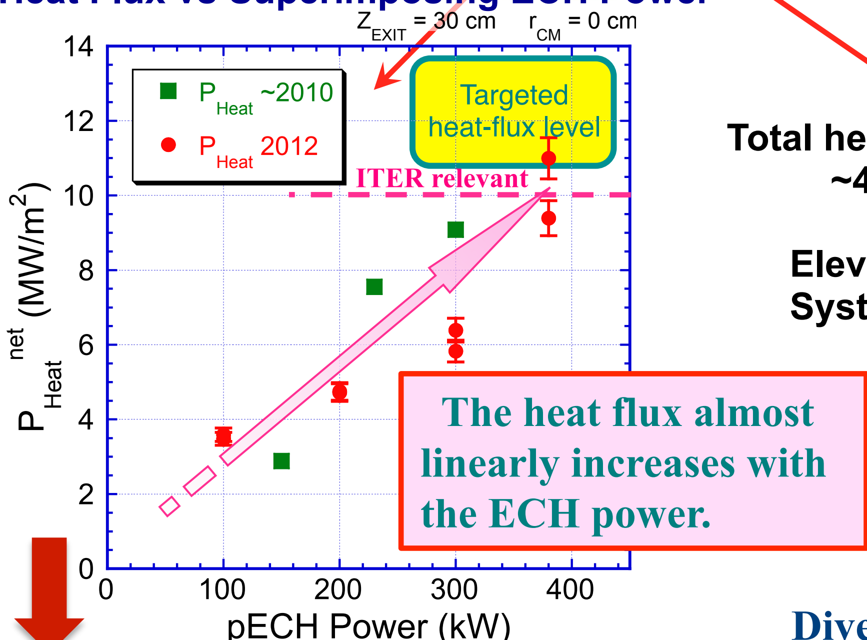
Plasma Characteristics of the End-cell of the GAMMA 10 Tandem Mirror for the Divertor Simulation Experiment

Y. Nakashima, et al., Plasma Research Center, University of Tsukuba

World Largest Tandem Mirror GAMMA 10/PDX

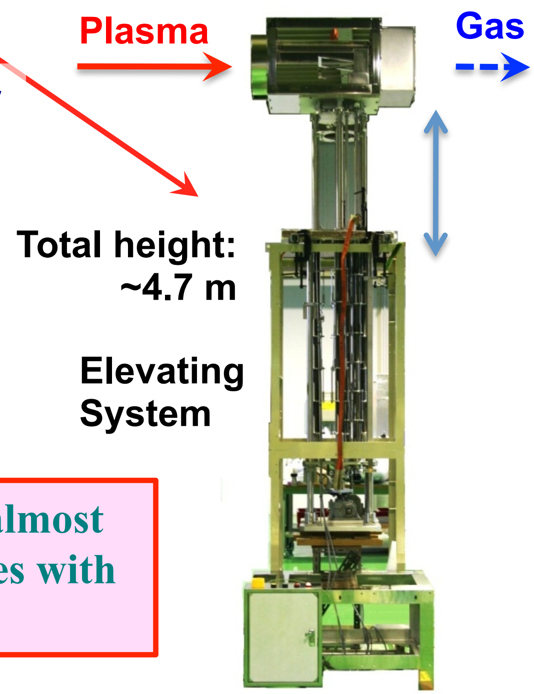


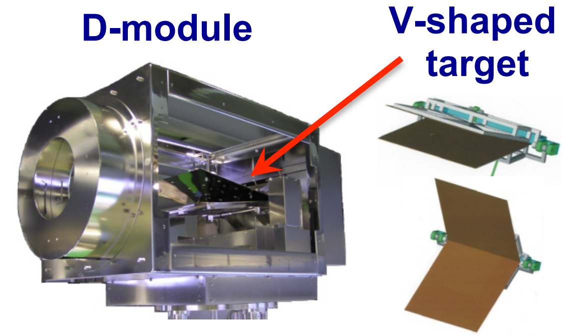
Heat Flux vs Superimposing ECH Power



A short ECH pulse of 380 kW into ICRF-heated plasmas attained the highest heat-flux density comparable to the heat load of ITER divertor plate (≥ 10 MW/m^2).

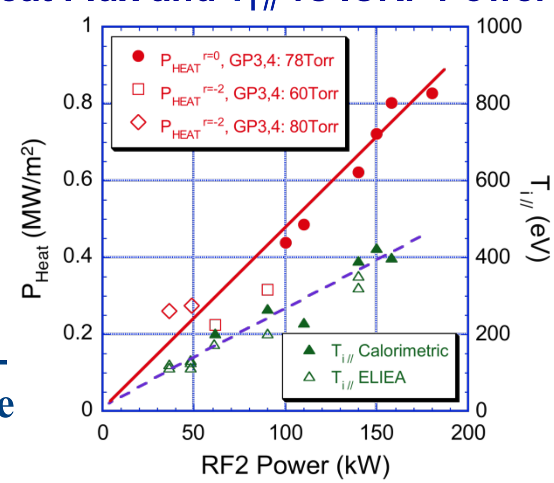
A Large-scale Divertor Simulation Experimental Module (D-module) has been completed this spring.





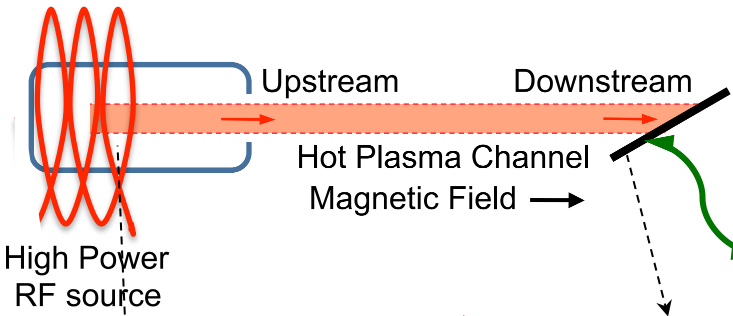
Divertor simulation experiments using D-module have been extensively started.

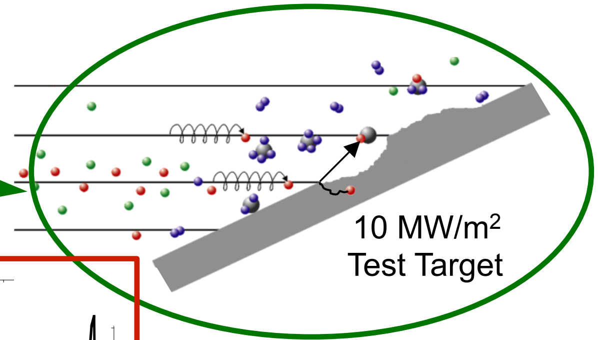
Heat Flux and $T_{i//}$ vs ICRF Power

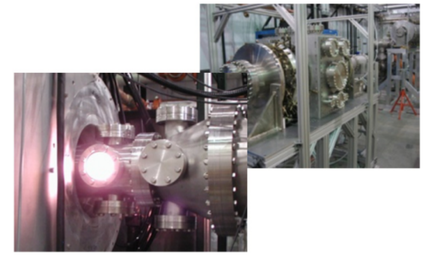


Plasma characterization proved that GAMMA 10 has extraordinary high performance, such as high heat flux, ion temperature and its controllability.

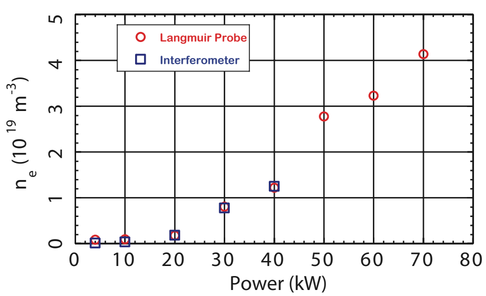
PMTS: new plasma generator to address PMI of materials in fusion environment



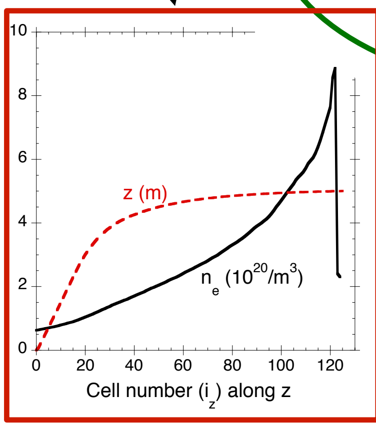




ORNL expertise in RF technologies

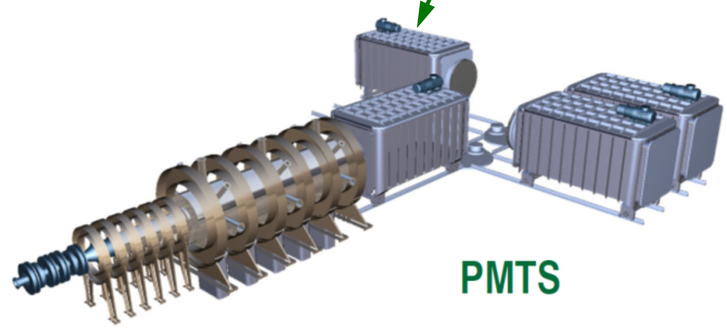


ORNL's RF source demonstrated densities of ~4 x 10^{19} m^{-3}



B2-Eirene simulations predict ITER plasma conditions can be reached at the target

Development of novel plasma production techniques

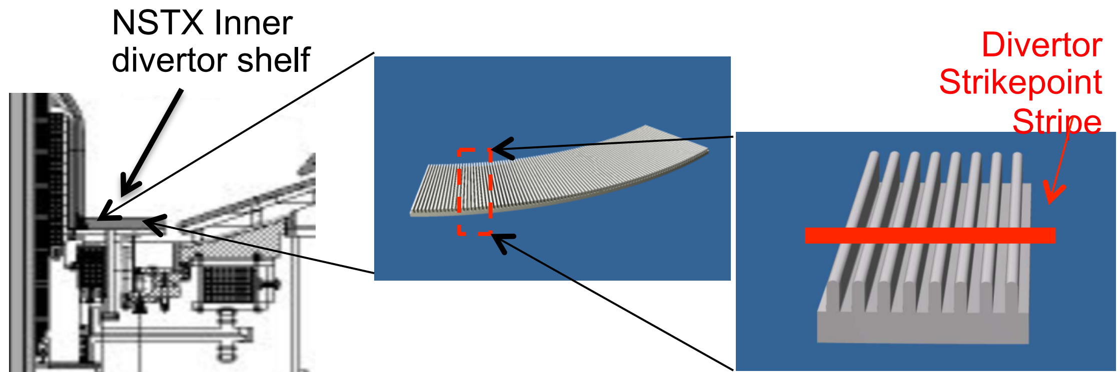


PMTS

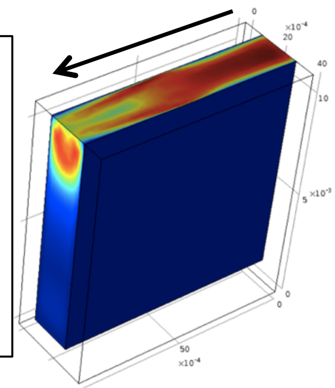
PMTS will be a cost effective device to carry out a critical R&D program to advance the PMI science and the PFC development on the road to FNSF and DEMO

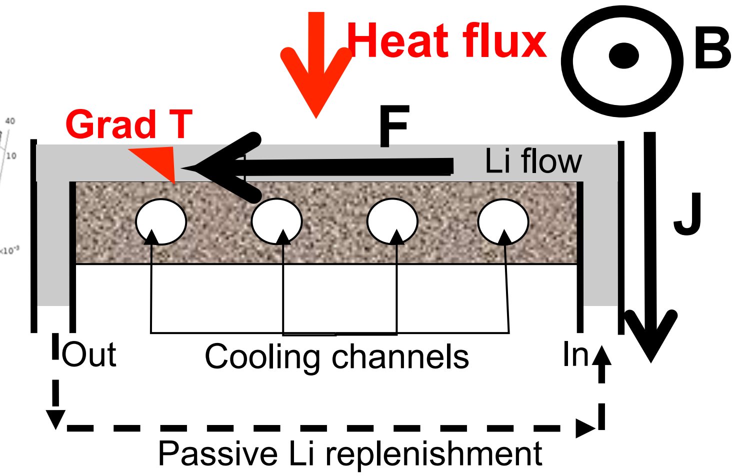
LiMIT: Lithium-Metal Infused Trenches

- **LiMIT concept**: metal (stainless steel or Moly) tiles with radial trenches containing liquid lithium
- The trenches run in the radial (polodial) direction such that they lie primarily perpendicular to the toroidal magnetic field
- Li flows in the slots of the metal plate powered by the vertical temperature gradient

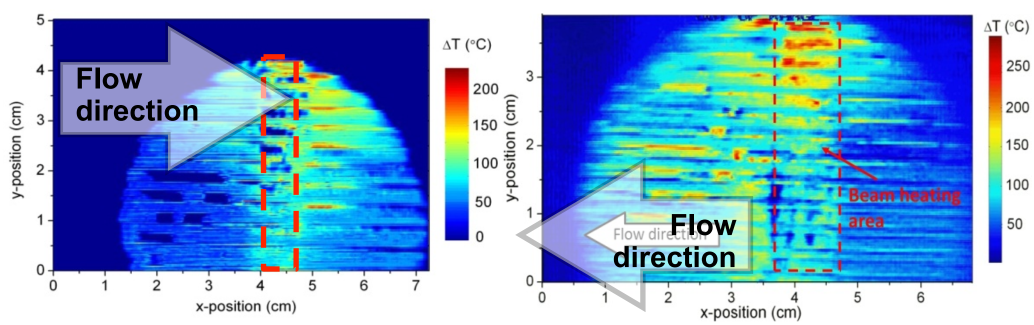


The vertical temperature gradient generates vertical thermoelectric currents within each trench of LiMIT, which when JxB-"crossed" by the toroidal magnetic field, will create a radial force on the Li driving it along the slot.
This is self-driven flow.





- The Li flow transfers the heat from the strike point to other portions of the divertor plate
- The bulk of the metal plate can be actively cooled for a long-pulsed device or passively
- New Li is continuously replenished facing the plasma
- Under the plate the Li flows back naturally



M.A. Jaworski, et al. Phys. Rev. Lett. 104, 094503 (2010)
D.N. Ruzic, et al. Nucl. Fusion Lett. 51 (2011) 102002

FTP/P1-13: **Super-X Divertor Simulation for HCSB-DEMO**

G.Y. Zheng, et al.
Southwestern Institute of Physics, China

- EFIT equilibrium HCSB-DEMO (R=7.2m, a=2.1m, I_p=14.6MA).
- **Inner divertor**: a snowflake-like configuration near separatrix.
- **Innovative outer divertor**: super-X leg to large R combined with local X-point near target → larger wetted divertor surface area.

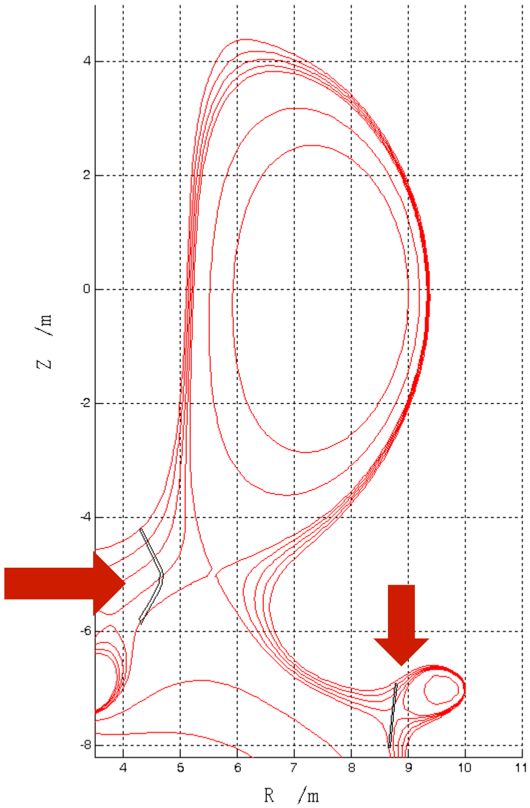


Figure 1. Super-X Divertor configuration

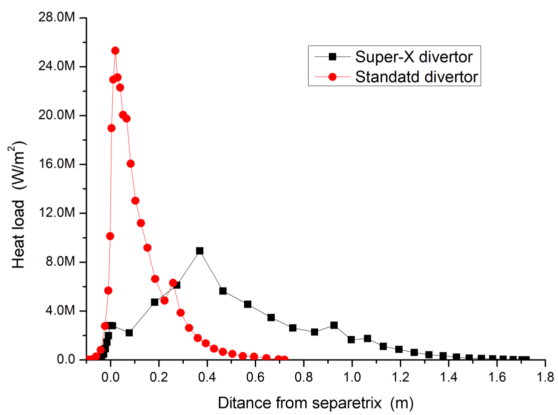


Figure 2. The heat load at inner targets

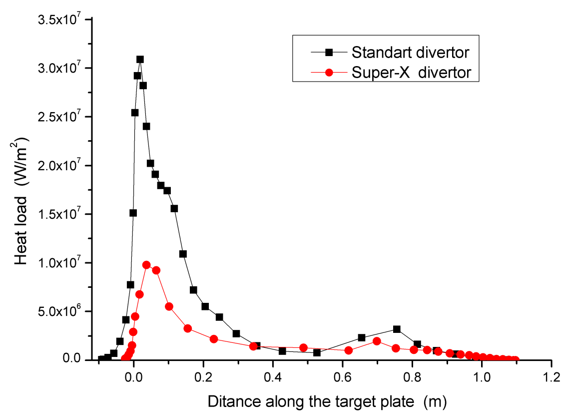


Figure 3. The heat load at outer targets

- SOLPS5.0, B2.5-Eirene simulation; SOL $n_e=3.5\times10^{19}/m^3$ & P=600MW into divertors.
- **Inner divertor**: peak heat load reduced from 25.3MW/m² to 9.2MW/m².
- **Outer divertor**: peak hat load reduced from to 31.0MW/m² to 10.0MW/m².

Divertor design and physics issues of huge power handling for Demo reactor:

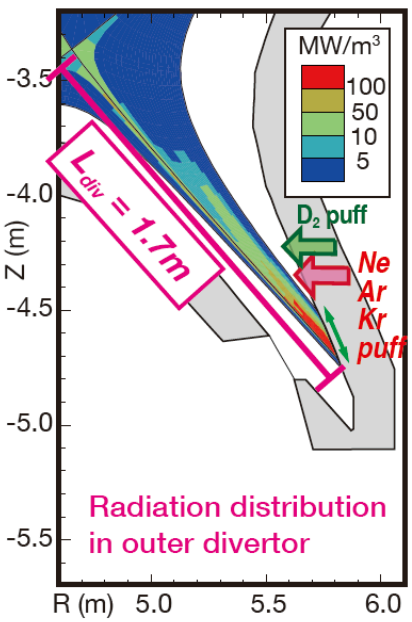
FTP/P7-32 N. Asakura, K. Shimizu, K. Hoshino, et

JAEA IFERC

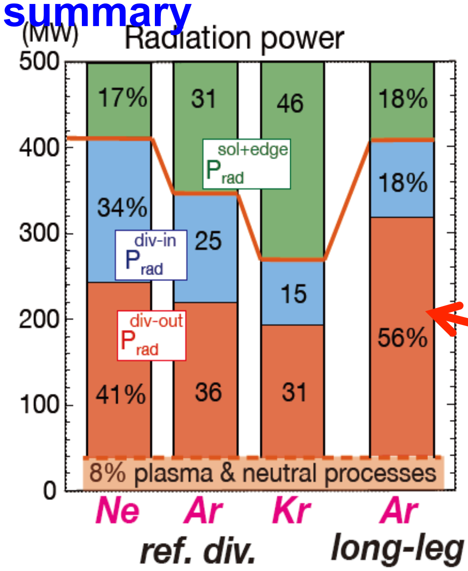
Enhance P_{rad}^{div} and $P_{rad}^{SOL+edge}$ for "Full detachment" of divertor plasma needed.

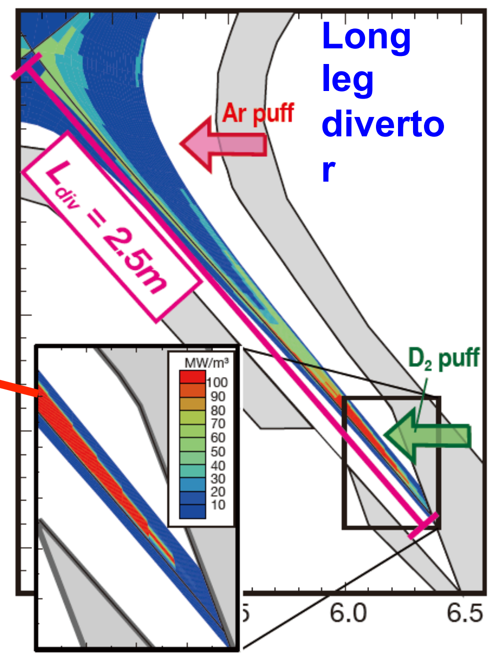
SONIC code + Monte Carlo impurity transport **applied to DEMO divertor detachment.**

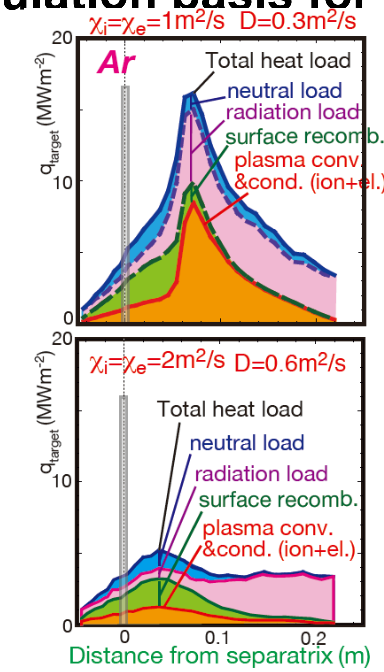
- **Seeding impurity: Ar/Kr preferred to increase P_{rad}^{SOL} ⇒ research issues: restrict impurity to ensure core plasma confinement and low contamination**
- **Long divertor leg: enhance P_{rad}^{div} and helps full detachment ⇒ research issues: optimize He exhaust with reduced ion and neutral fluxes**
- **Cross-field diffusion: strongly increases detachment and energy dissipation ⇒ research Issues: enhance global/local diffusion; modeling/simulation basis for Demo**



Power distribution summary







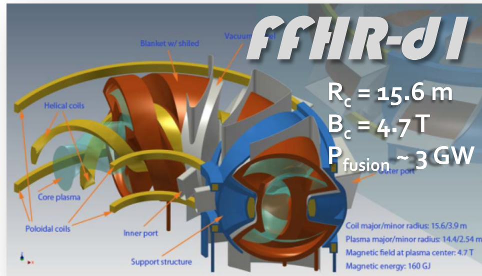
FTP/P7-34

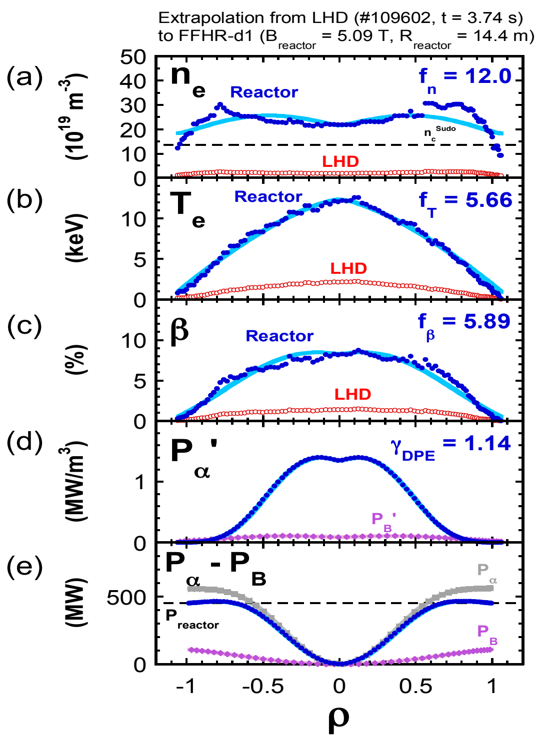
Multifarious Physics Analyses of the Core Plasma Properties in a Helical DEMO Reactor FFHR-d1

J. Miyazawa, et al.

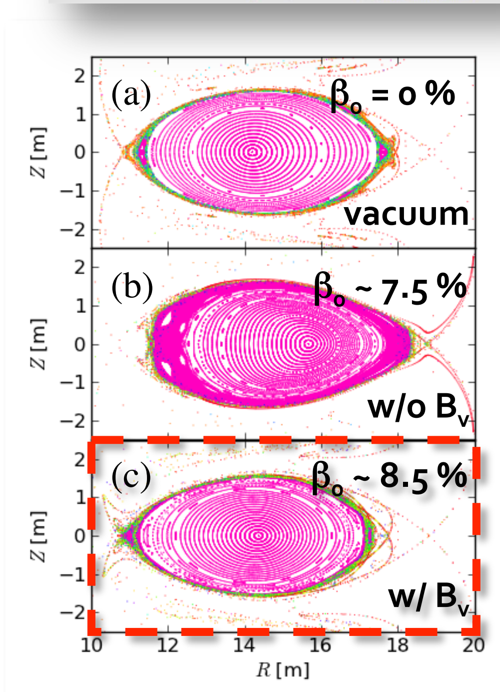
National Institute for Fusion Science

Japan

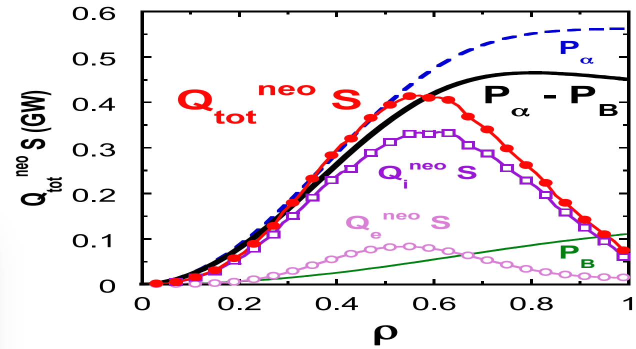




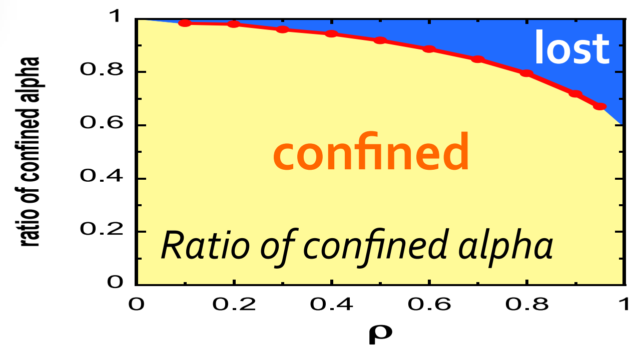
1. Radial profiles are extended directly from LHD using the gyro-Bohm model



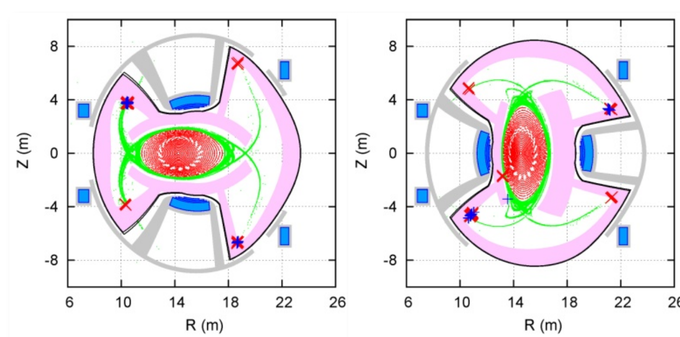
2. Magnetic surfaces at high beta can be maintained by applying B_v



3. Neoclassical thermal transport allows sustained burning plasma



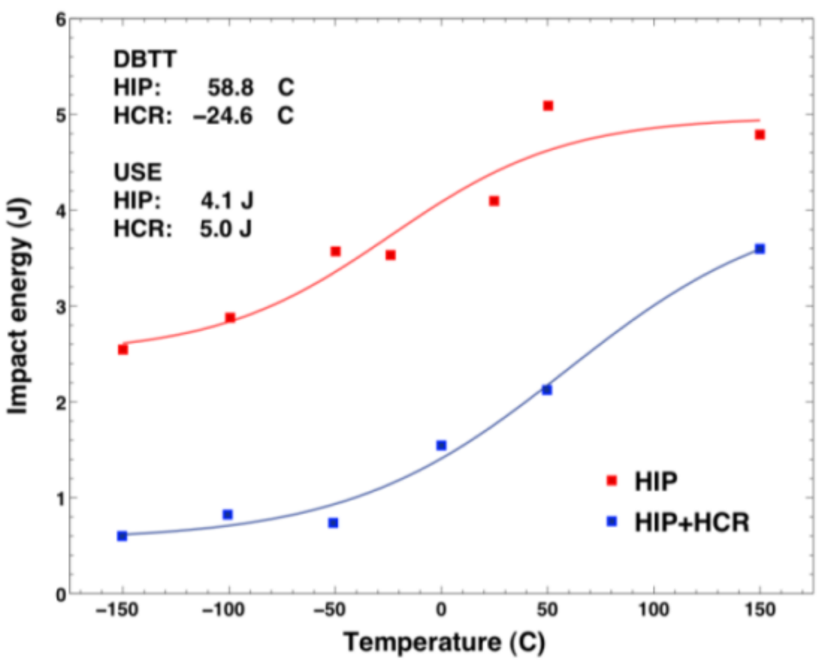
4. Direct loss of alpha particles is tolerable and reaches the divertor behind blanket



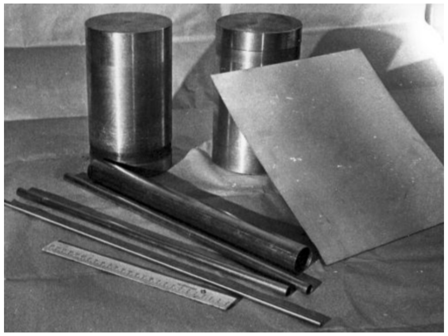
MATERIALS AND BLANKETS

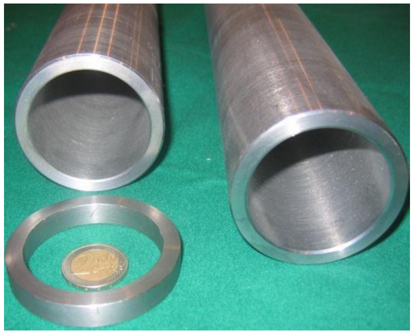
FTP/4-5Ra: Opt. production of nanostructured ODS ferritic steels
P. Unifantowicz, J. Fikar, et al., CRPP-EPFL, Switzerland
FTP/4-5Rb: Low activation V alloys for fusion reactors
VM Chernov et al., Bochvar Inst., Moscow, Russia

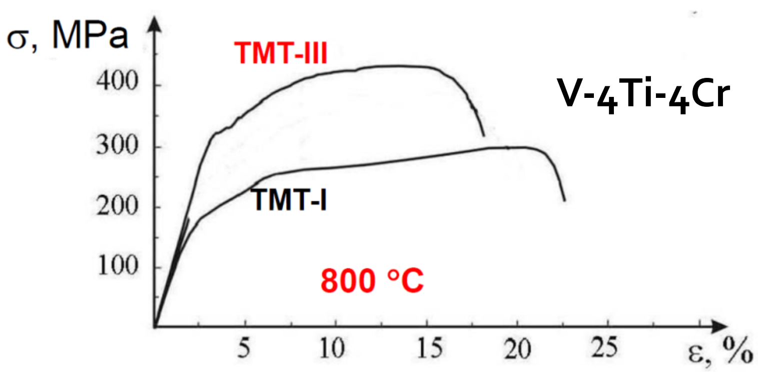
- **ODS-FS: Multiple Hot Cross Rolling (HCR) to 65-80% Reduction of Thickness (ROT) & higher purity of substrate powder improve strength and DBTT.**



- **Reference V-4Ti-4Cr @2012:**
  - **Fusion appl. for <100 dpa & 350C – 800C**
- **Optimized V-4Ti-4Cr @2020:**
  - **B600 tests for <150 dpa & 380C – 700C**
  - **Corrosion tests: Li, Na, Pb, Pb-Li**
- **Advanced V-Cr-W-Zr-C-O:**
  - **Further optimization tests in progress**



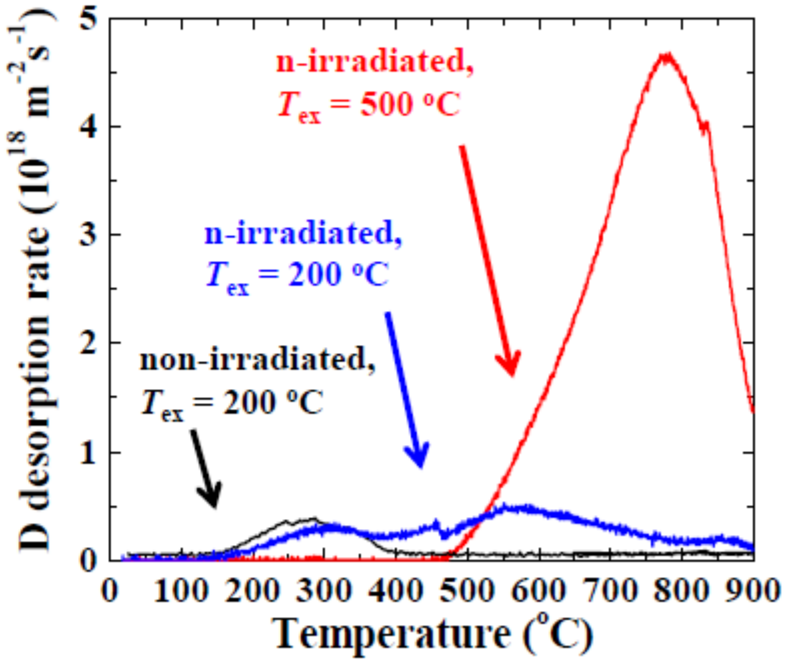




FTP/4-1 Hydrogen Isotope Trapping at Defects Created with Neutron- and Ion-Irradiation in Tungsten (W)

Y. Hatano (U. Toyama, Japan) et al. (*hatano@ctg.u-toyama.ac.jp*)

- As key information to evaluate tritium inventory in fusion reactor cores, deuterium (D) retention in neutron-irradiated W was studied in Japan-US Joint Project TITAN.
- Even at low dose (0.025 dpa), neutron-irradiation resulted in formation of strong trapping sites and significant increase in D retention.
  Detrapping energy: ca. 1.8 eV
  Trap Density: ~0.5 at.%
- Increased D retention with increase in temp. of plasma exposure due to deep penetration of D (50–100 μm at 500 °C).
- Based on these data, tritium removal techniques from neutron-irradiated W are discussed. Isotope exchange with H removes 2/3



Thermal desorption spectra of D from neutron-irradiated and non-irradiated W after exposure to D plasma to 10^{26} D m^{-2}. Exposure temperature, T_{ex}, was 200 and 500 °C.

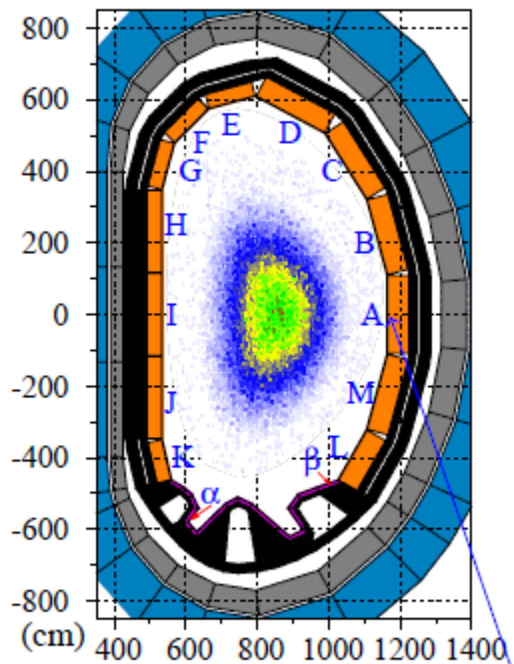
Integrated modelling of material-damage

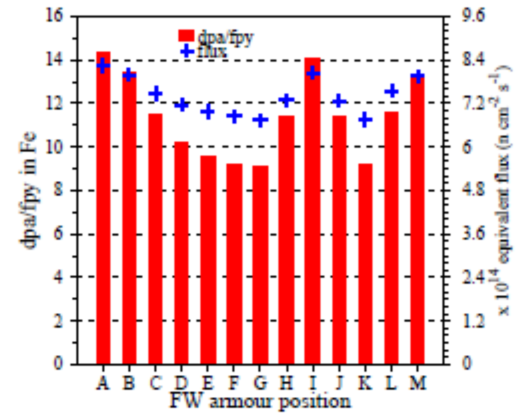
- A new approach to produce predictions of how neutron-irradiation will change materials

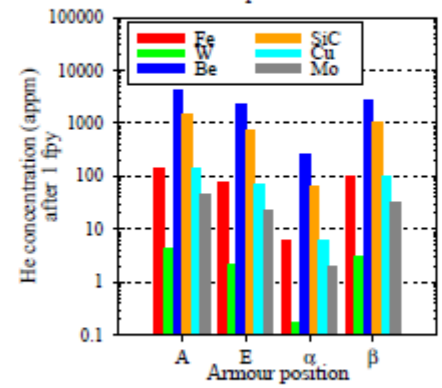
(1) Neutron transport simulations (flux, spectrum + dpa)

(2) Inventory calculations (transmutation + gas production)

(3) Modelling material behaviour (e.g. helium embrittlement of grain boundaries)







Element	Critical times and dpa for GB embrittlement by helium at A			
	FW armour		blanket at depth of 17-19 cm	
Fe	4 years	57.47 dpa	18 years	86.13 dpa
W	244 years	1084.49 dpa	300+ years	470.0+dpa
Be	22 days	0.43 dpa	69 days	0.56 dpa

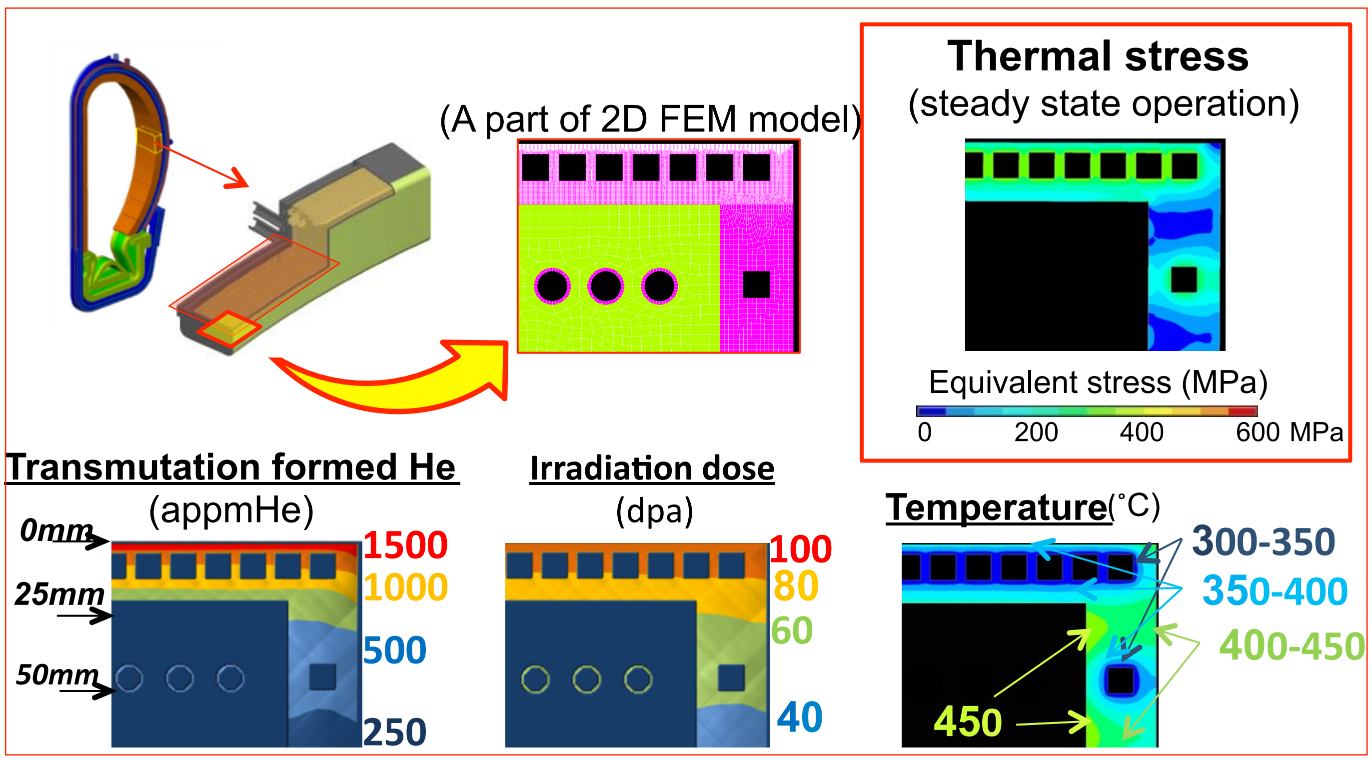
Gilbert *et al.* *Nucl. Fus.* **52** (2012) 083019 & **51** (2011) 043005

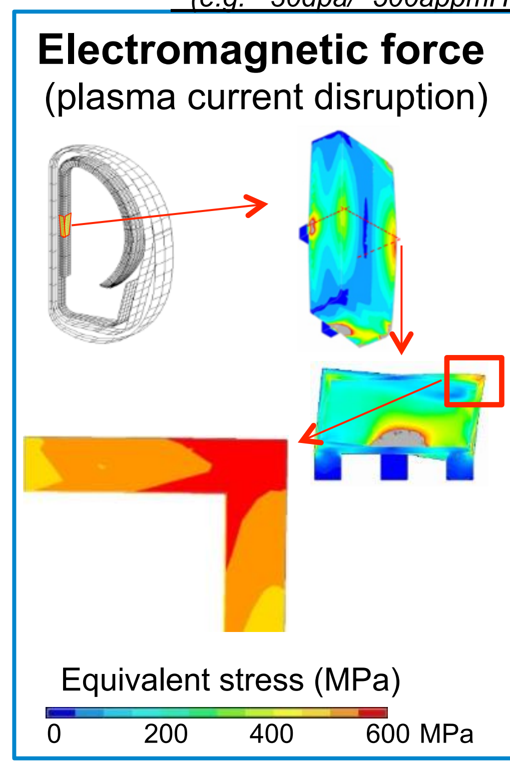
FTP/P7-17 Research and Development Status of Reduced Activation Ferritic/Martensitic Steels Corresponding to DEMO Design Requirement

H. Tanigawa (JAEA), et.al.

- Applicability and technical issues of RAFM steel as the DEMO blanket structural material have to be discussed in view of the DEMO blanket structural soundness in various operation modes.
- In the case of water cooled / ceramic breeder blanket of SlimCS design, RAFM hardening and embrittlement is expected just around cooling channels, and the discussion on the structural soundness of the blanket and applicability of RAFM should regard these thin structure.
- ***Estimation of the critical condition* and validation of 14 MeV neutron irradiation effects up to that condition in early stage of DEMO engineering design phase is essential.***

* (e.g. ~30dpa/~500appmHe)





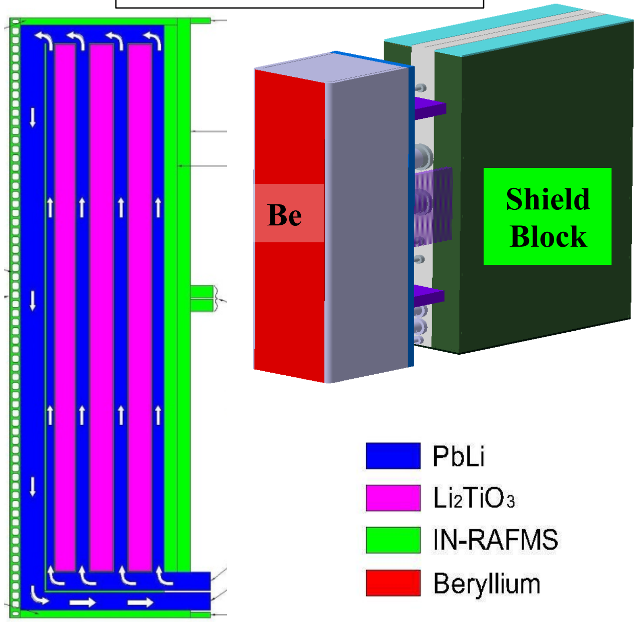
FTP/4-1Rabc: Kumar, Giancomo, Feng, et al

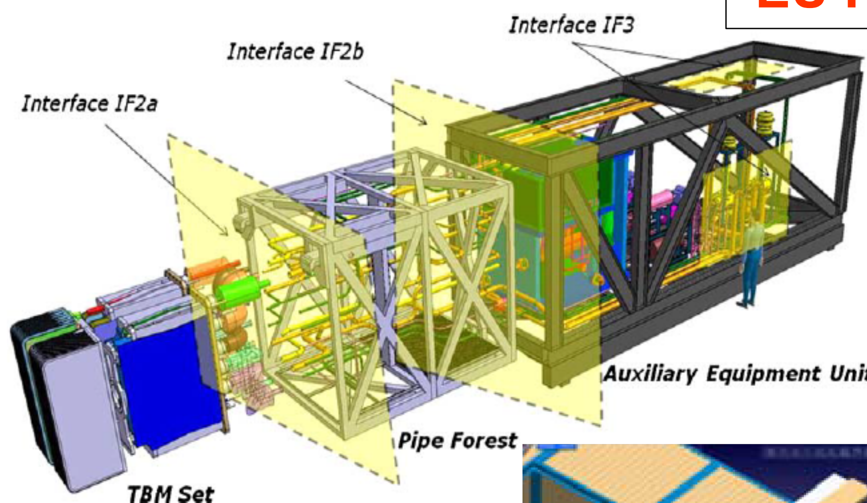
ITER-TBM Program activities in India, Europe, and China

- **Goals**: **"Demonstrate the scientific and technological feasibility of fusion power for peaceful purposes"** via testing of nuclear technology in integrated fusion environment.
- **To deliver the TBM Systems** for the first phase of ITER.
- Materials and technology **R&D in full swing**.

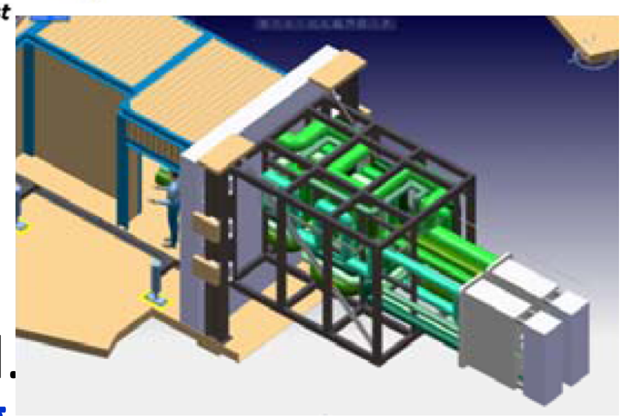
EU HCLL

Indian LLCB





China HCCB



- Generic TBM Agreement has been approved. **Partners' TBMAs being prepared for signing.**

ROAD MAPPING ACTIVITIES AND RELATED FUTURE FACILITIES

HN Nielson et al: SEE/1-1:

On the Path to MFE DEMO

1. With ITER under construction, MFE has crossed a threshold to a DEMO programme.
2. The science and technology issues for DEMO are known. Now we are developing strategies to resolve them:
   - Making ITER succeed.
   - New fusion nuclear facilities
   - What else is needed?
3. A new IAEA DEMO Programme Workshop series can help all parties define, plan, and coordinate DEMO programme activities.

FTP/3-3: **On the Physics Guidelines for a Tokamak DEMO**

H. Zohm, et al., MPI fur Plasmaphysik, et al., E.U.

DEMO Physics Design Guidelines: issues

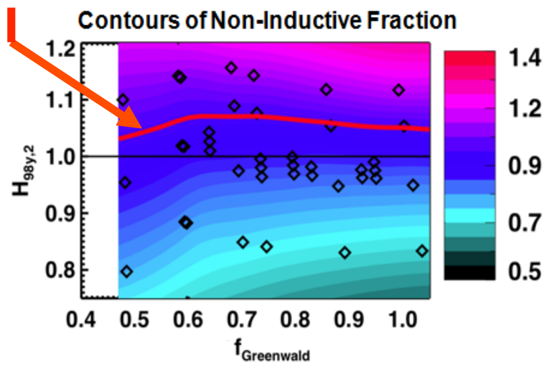
Area	Issues
Equilibrium	Shaping vs. A
MHD stability	Fast particle effects on $\beta_{max,th}$, stability of fast particle driven modes
Energy balance	Transport at high β, n/n_{GW} and P_{rad}, T_e profile shape
Particle balance	Density profile shapes @ n/n_{GW}, impurity transport
Power exhaust	Detachment physics, λ_{power} in detached divertor, other divertor concepts, ELM mitigation
H&CD physics	Full optimisation NBCD localisation
Dynamical phases	Realistic times for recharge
Disruptions	Timescales

Color code
Sound physics base
Restricted applicability
Major progress needed

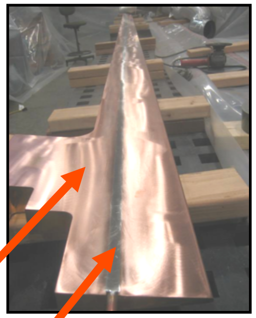
Summary of FTP/3-4: "Progress on developing the spherical tokamak for fusion applications" by J. Menard, et al.

- NSTX Upgrade aims to address understanding gaps to next-step STs
  - Confinement, stability vs. collisionality
  - Full non-inductive start-up, ramp-up
  - Solutions for very high heat flux, such as snowflake divertor, and lithium (future)

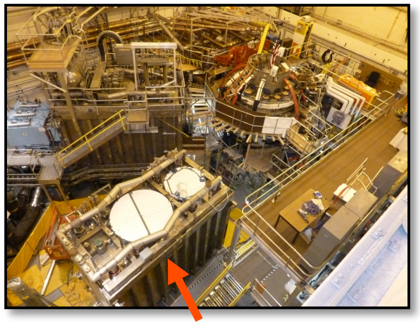
100% non-inductive current projected



- Good progress made in overcoming increased EM loads of Upgrade
  - Project on schedule & budget, ~50% complete
  - Aiming for project completion in summer 2014

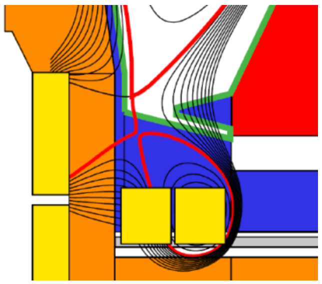


Friction-stir-welded TF joints
Improved TF cooling-tube soldering

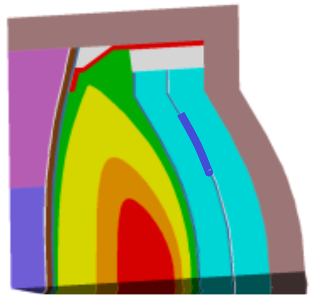


2[nd] NBI in NSTX-U test cell

- ST-FNSF studies quantifying performance and achievable missions vs. device size
  - Building on achieved/projected NSTX/NSTX-U performance and design
  - Incorporating high triangularity, advanced divertors, TBR ~ 1 option, good maintainability



High-δ snowflake

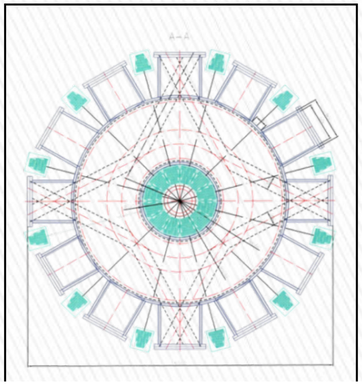


TBR~1 for R=1.6m

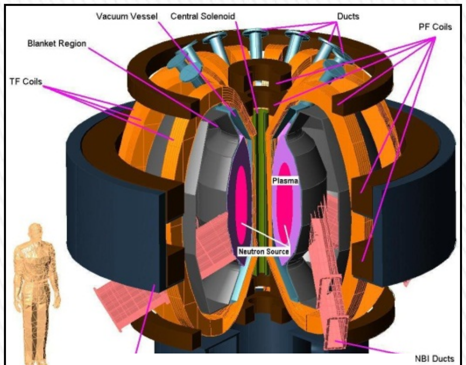
FTP/4-6: EA Azizov, et al, Development of magnetic Fusion Neutron Source (FNS) and fusion-fission hybrid systems in Russia

Database, analysis, modeling and advanced technologies provide basis to:

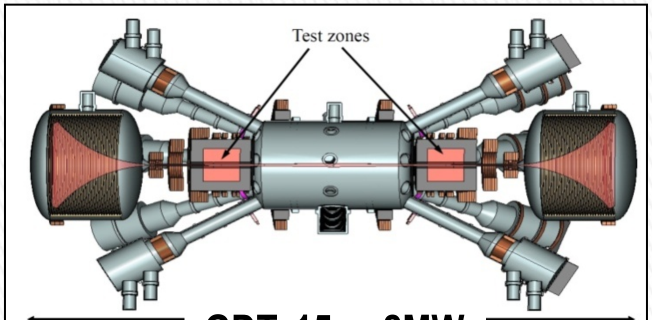
- Develop a compact FNS based on tokamak from ST to conventional aspect ratios
- Expect risks in physics and technology when extending FNS to hybrid systems
- Define key efforts of Russian fusion program: ITER, research FNS, hybrid systems



SC FNS, 1.9m, 20MW

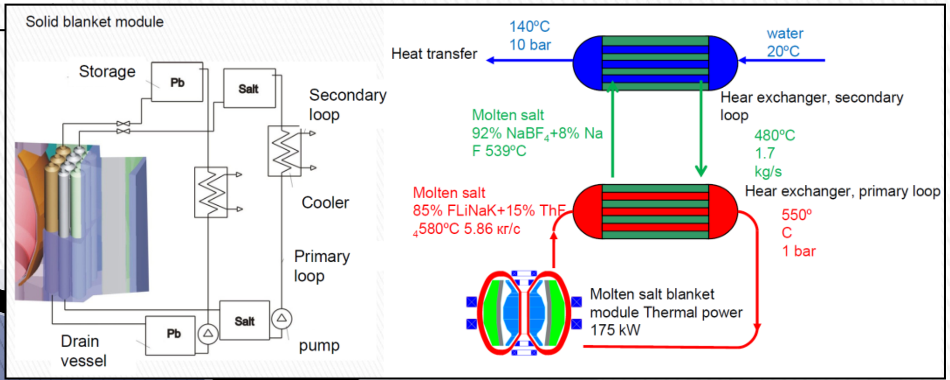


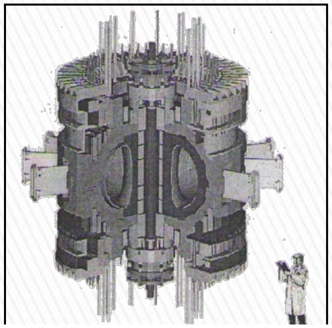
FNS-ST, 0.6m, 5MW



GDT, 15m, 2MW

Solid and molten-salt blanket options for hybrid





Ignitor, 1.32m, 95MW

Concluding remarks

- Progress on ITER since the last meeting has been impressive. Design is at an advanced stage and production and prototyping of machine components has begun in the DAs.
- Our research community has been fully engaged in solving the technical challenges posed by ITER thereby advancing the state of the art in plasma control technologies(magnets, H&CD, fueling, PFCs) and expanding the R&D boundaries to include fusion nuclear technologies(fuel cycle, TBMs).
- The momentum must be continued across the board to address the more demanding requirements for DEMO (all of the above plus high temperature radiation resistant materials). In addition to engaging existing facilities and recent and future additions (superconducting tokamaks, W7-X, IFMIF), we will need an increased emphasis on dedicated test facilities for technologies.

BACKUP

What will we have learned from ITER

- ***Reactor scale technology demonstrations***
  - Superconducting magnets
  - IC, EC, NB for heating/current drive **(but not yet optimized for efficiency)**
  - First assessment of tritium breeding and high temperature operation in mockups deploying RAFM steel and various breeding materials and coolants at ~1 MW/m^2 neutron wall loading **(but low fluence and short pulse and difficult to asses TBR)**
  - Strong core fueling to maintain DT density consistent with production of several hundreds of MW of fusion power **(DEMO requires even higher density).**
  - Divertor with 10 MW/m^2 steady state heat removal capability operating in semi detached , tightly coupled regime with low net erosion and plasma contamination. **(DEMO will operate at higher power density).**
  - Technologies to mitigate the effects of ELMs and disruptions **(pellets OK, reactor relevance of in vessel coils?)**
  - Torus pumping, exhaust gas cleanup, isotope separation systems of unprecedented scale **(but lower throughput in batch processing mode)**
  - Will have a fairly high inherent availability for research program, but overall duty factor < 10% **(plasma control technologies will not have the luxury of generous routine maintenance)**

ITR/1-1 Eich

Multi-machine scaling of SOL H-mode power width, λ_q

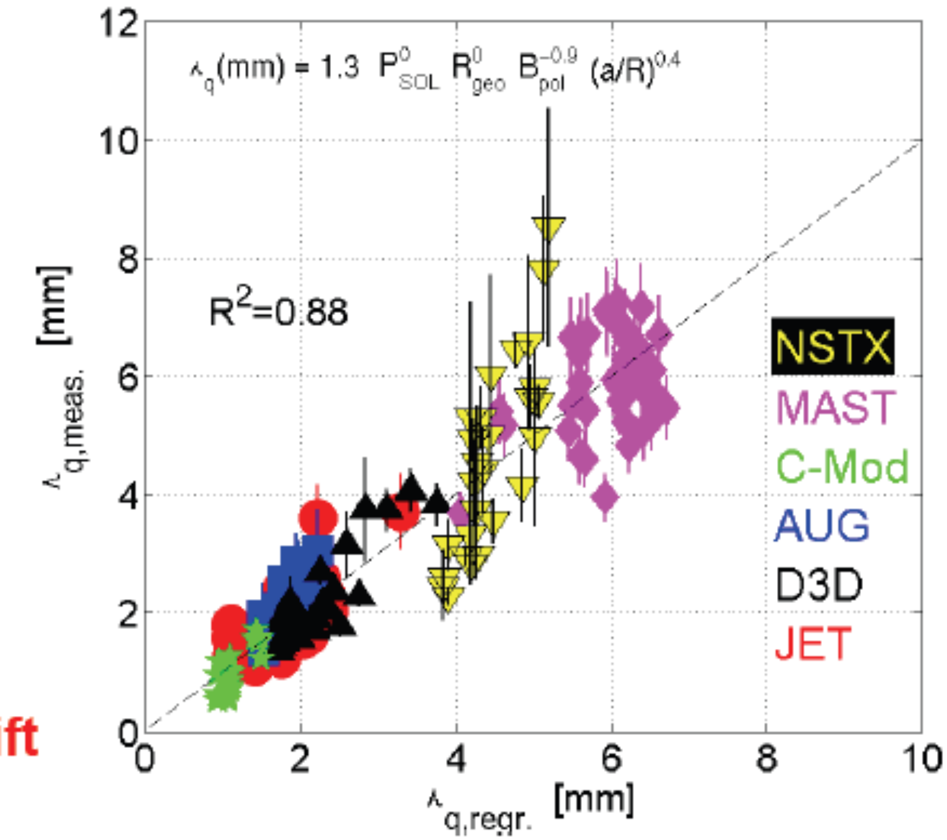
ITER Physics Basis scaling for burning plasma λ_q may be too optimistic

Multi-machine λ_q database from IR measurements at outer divertor targets in attached H-mode → all analysed in the same way

Regression finds <u>no machine size</u> (major radius) scaling and $\lambda_q \propto 1/I_p$

Predicts $\lambda_{q,ITER} \cong 1$ mm at Q = 10

Consistent with recent heuristic drift based model



Implied increased divertor heat flux density on ITER may be mitigated by dissipation (partial detachment/impurity radiation)

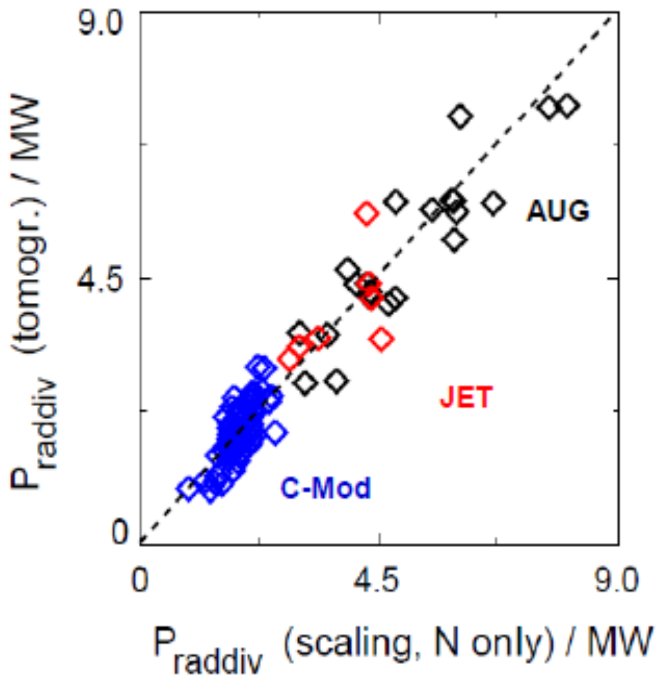
Multi-machine comparisons of divertor heat flux mitigation by radiative cooling with nitrogen

ITR/P1-28

Kallenbach

• Scaling obtained for divertor radiated power in AUG, C-Mod and JET-C (N-seeded data only)

$$P_{raddiv,Nonly} = 1720\, p_{0,div}^{0.47}\, (Z_{eff}-1)^{0.31}\, R^{1.095}\, \lambda_q^{1.148} \quad [MW],\ [Pa],\ [m],\ [m]$$



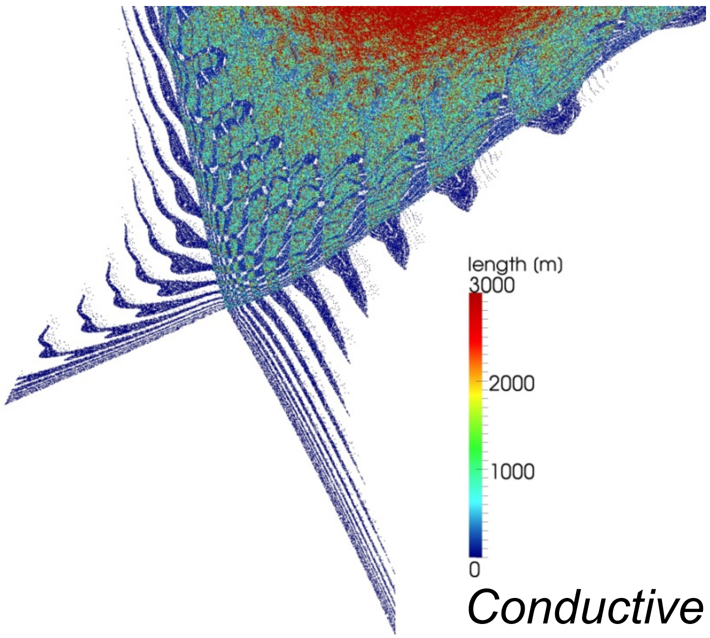
$P/R^{1.1}$ as metric for divertor radiation
→ not very favourable extrapolation to large devices

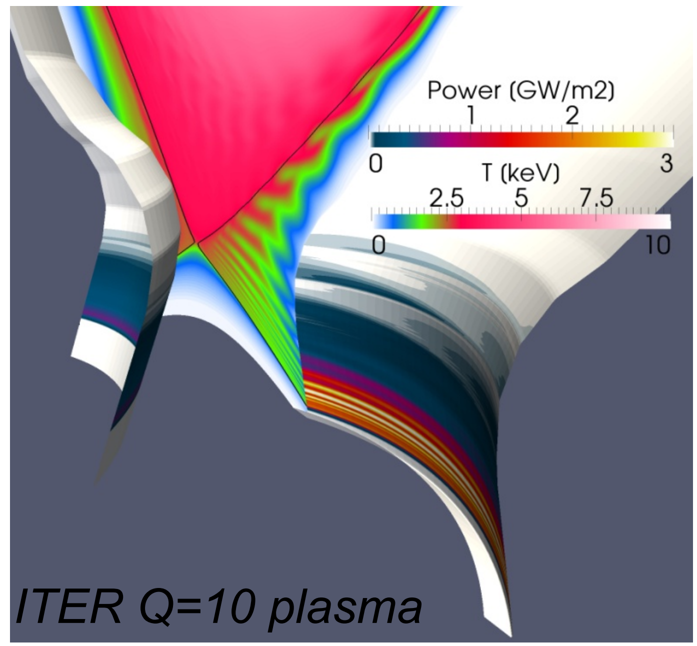
Results in line with analytical expectation for divertor radiation scaling (2-point model)

Extrapolation to ITER depends critically on power width λ_q (at radiating zone, mapped to omp) for λ_q=1.6 mm, 25 MW divertor radiation are predicted for δZ_{eff}=1 (nitrogen), but a factor 2 higher λ_q value is expected for ITER at the radiating zone (T. Eich, ITR/1.1) → 55 MW

Non-linear Simulation of ELM Energy Deposition

- **ELM simulations in ITER Q=10 plasma scenario (JOREK code)**
- **Conductive ELMs :**
  - large magnetic perturbation leads to ergodic edge with homoclinic tangles
  - parallel conduction on perturbed magnetic field lines
  - broadening of energy deposition scales with magnetic perturbation
- **Convective ELMs:**
  - filament formation, energy density losses in parallel direction
  - broadening energy deposition due to radial distance travelled relative to parallel transport



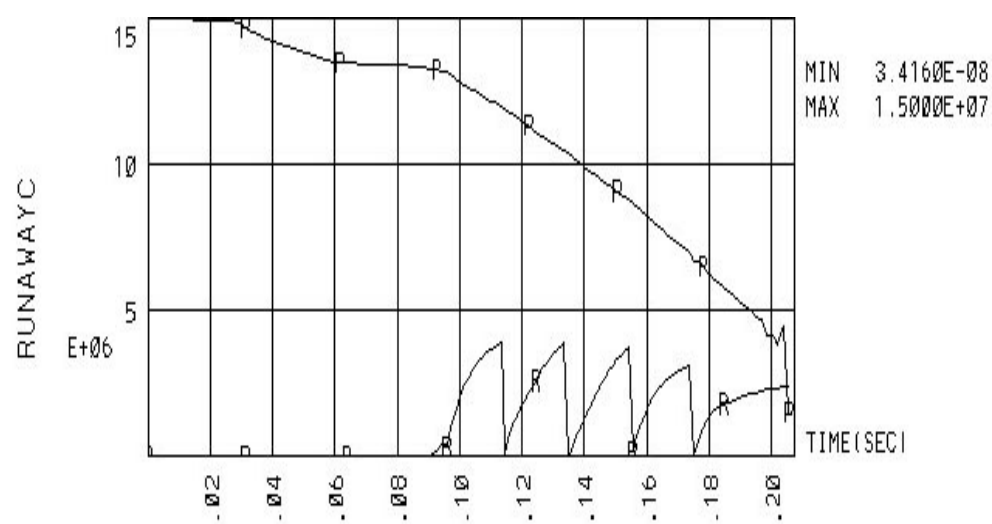


Conductive ELM in ITER Q=10 plasma

Modelling of ITER plasma shutdown with runaway mitigation using TSC

I.Bandyopadhyay, A.K. Singh, M. Sugihara and S. C. Jardin

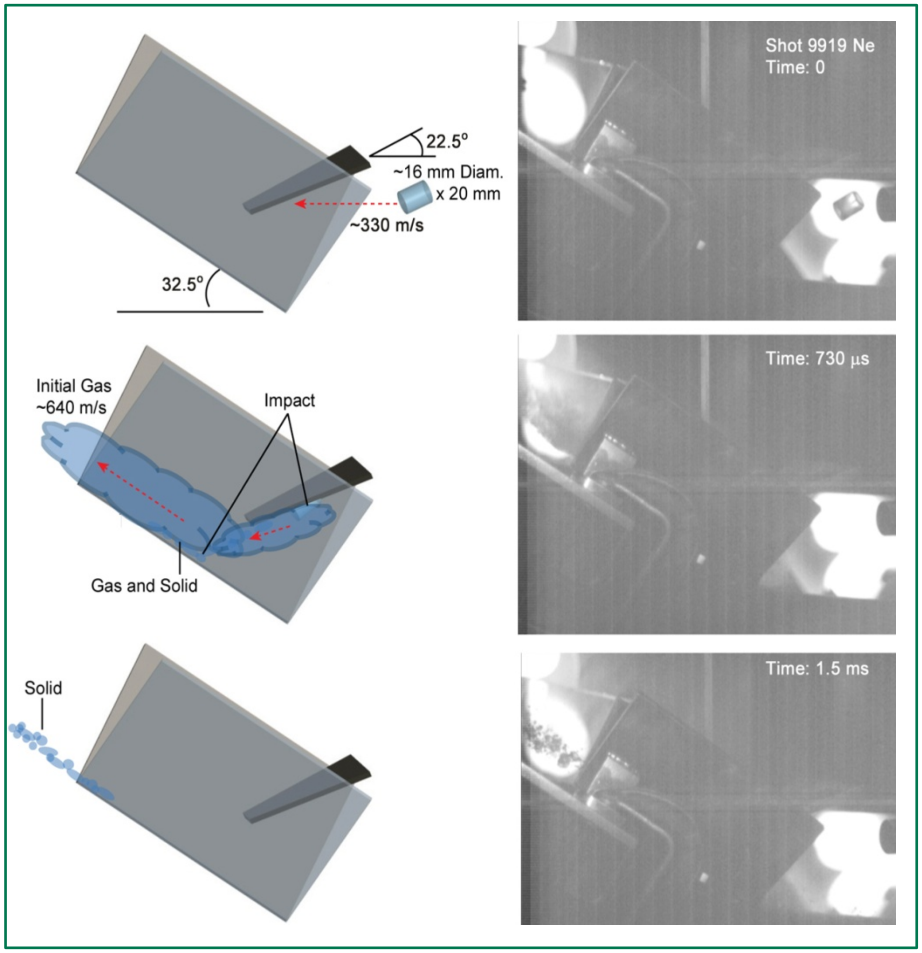
- Model of runaway destabilization with a train pellets launched in a plasma with impending disruption, but before thermal collapse
- Series of pellets of radii between 5-25mm launched at speed between 500-1000m/s and paced between 50-200Hz can trigger MHD events, which can help deconfine and limit runaway currents to acceptable values
- Tradeoff between large pellets with slow pacing and smaller pellets with faster pacing
- Experiments should be carried out to confirm proposed runaway mitigation



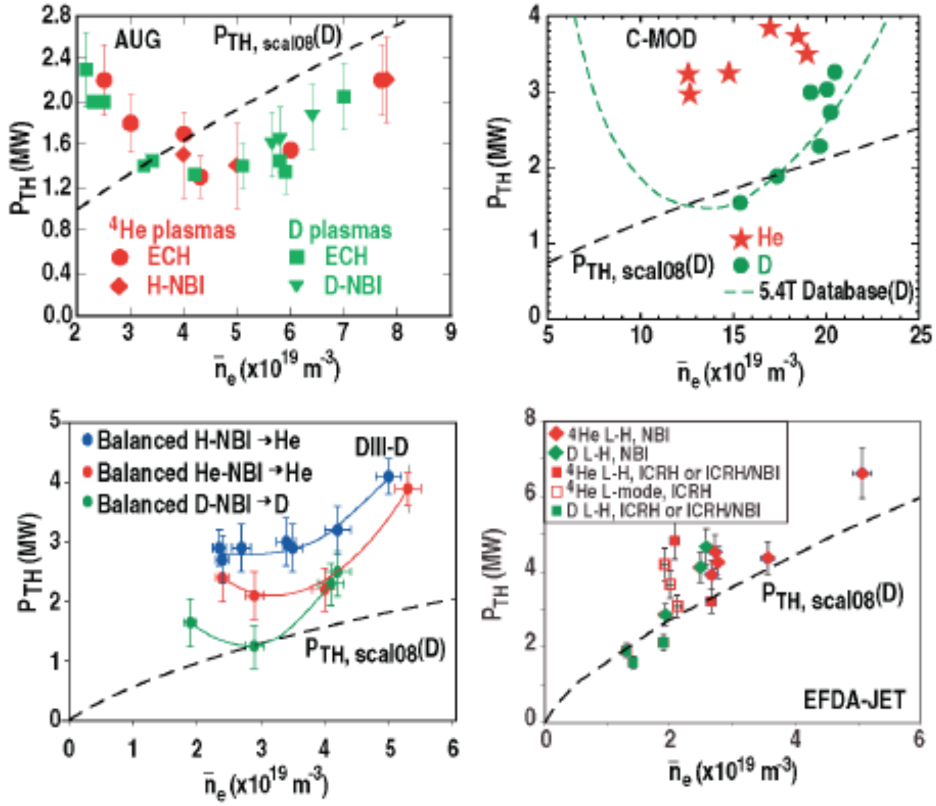
Development and Testing of Plasma Disruption Mitigation Systems Applicable for ITER

- Disruptions on large tokamaks present challenges to handle the intense heat flux, the large forces from halo currents, and the potential first wall damage from multi-MeV runaway electrons
- ORNL is developing the technology to inject sufficient material (Ne, D_2) deeply into plasmas for a rapid shutdown and runaway electron collisional suppression
- Massive gas injection via ORNL fast magnetic valves has been used on DIII-D and Alcator C-Mod to successfully mitigate some of the deleterious effects
- Injection of large shattered pellets is a new technique developed at ORNL, and experiments on DIII-D resulted in deeper penetration and much higher density with a single pellet containing ~3×10^{23} atoms than with massive gas injection of the same
- Important capability needed for maintaining successful operation of ITER and future reactors

Illustrations and photos of a pellet shattering on double-impact prototype target (same target design used inside DIII-D vessel)



The H-mode Power Threshold for Helium Approaches that for Deuterium on the High Density Side



(Gohil et al., ITR/P1-36)

- This behavior on the high density side of the density dependence of P_{TH} is favorable for ITER

- Other experiments show the H-mode power threshold increases with the application of 3D (non-axisymmetric) fields

Going beyond ITER

$\boxed{\beta = \beta_N I / aB}$ $\boxed{P_{fusion} \propto \beta^2 B^4}$

	I_p (MA)	B (T)	β_N	β (%)	P_{fusion} (MW)	P_{elec} (MW_e)	Plant Efficiency
ITER Scenario 2	15	5.3	1.8	2.4	400	-	-
ITER AT DEMO	9	5.6	5	3.8	1240	~ 400	~0.3
EU Model D reactor	14.1	5.6	4.5	5.4	2530	1518	0.6

Optimized current and pressure profiles: flexible plasma heating fueling and current drive tools, edge density control

Higher plasma density ($2x10^{20}/m^2$): efficient core plasma fueling and particle exhaust
Low plasma contamination and low net erosion: high density divertor operation, divertor solution?
Fully developed blanket systems for tritium self sufficiency
Low activation materials (F82H, Eurofer…), ductile W
Higher magnetic field: HTS magnets?

Low recirculating power: high efficiency high power density H&CD systems.
High thermal efficiencies: advanced high temperature blanket (DCLL) and divertor, and materials ODS steels, SiC/SiC). High temperature Brayton cycle.

High duty factor: reliability, control/mitigation of transient events, maintainability and RAMI