Virtual Laboratory for Technology FES FY2017 First Quarter Report

Phil Ferguson
for the VLT members

Director's Corner

- The VLT (formed in 1998, reinvigorated in May 2016) continues to conduct high impact research in support of the FES program mission and represent the technology program within the fusion community
- The VLT Team (currently 8 universities, 9 national labs and 1 private company) is pleased to start submitting quarterly highlights to FES, starting with this FY2017 first quarter report
- Bi-monthly teleconferences have been initiated, with good representation at each call
- The VLT is assembling information from previous studies identifying engaging scientific leadership areas and opportunities over the next ten years, which may be useful for the approaching National Academies Study
- This and future highlights will cover research highlights from the recent VLT publications and the following main VLT research areas:
  - Magnet Systems; Heating & Current Drive; Plasma Fueling/ ELM Pacing/Disruption Mitigation
  - Plasma Facing Components; Plasma Materials Interactions; Structural Materials
  - Design/Systems studies; Power Handling; Fusion Safety; Fuel Cycle Research; Blanket Technology; Vacuum Systems

If you have any questions on the information in this report, please don't hesitate to contact us.

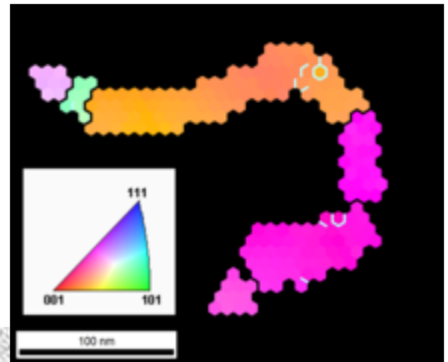
VLT
Virtual Laboratory for Technology
For Fusion Energy Science

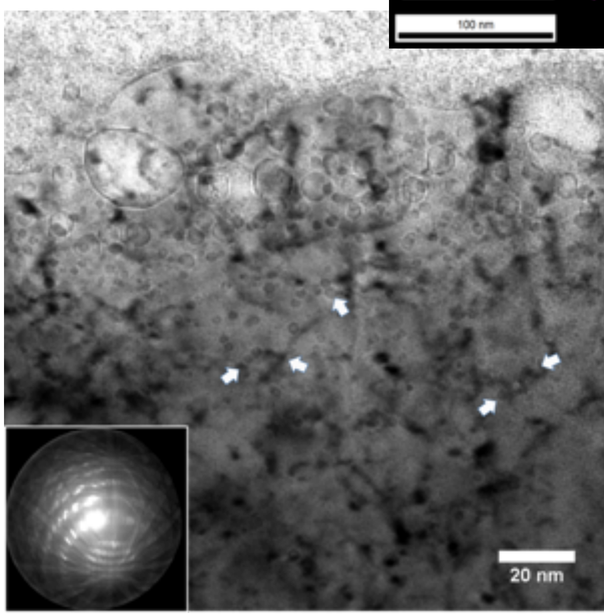
Advanced Microscopy Methods Reveal Details of the Materials Response to Plasma Interactions

Research Highlight

Microstructural Analysis

Crystallographic map of one tendril shows it is composed of individual W grains as small as 16 nm wide. Color indicates crystal orientation on insert key.





STEM image locates helium nanobubbles trapped on dislocation lines near a tungsten surface

Scientific Achievement

This study used advanced electron microscopy to characterize fine-scale defects in plasma-exposed tungsten.

Significance and Impact

This provides new nanoscale information to help validate computational models and eventually design new plasma-tolerant materials.

Research Details

- Transmission Kikuchi diffraction (tKD) was used to measure grain orientations and grain boundary character in isolated nanofuzz tendrils.
- Scanning transmission electron microscopy (STEM) was used to relate helium bubble size and evolution to preexisting defects (dislocations, grain boundaries) in plasma-exposed tungsten

Application

These methods will allow us to characterize new alloy or material concepts to improve upon tungsten's performance under plasma exposure.

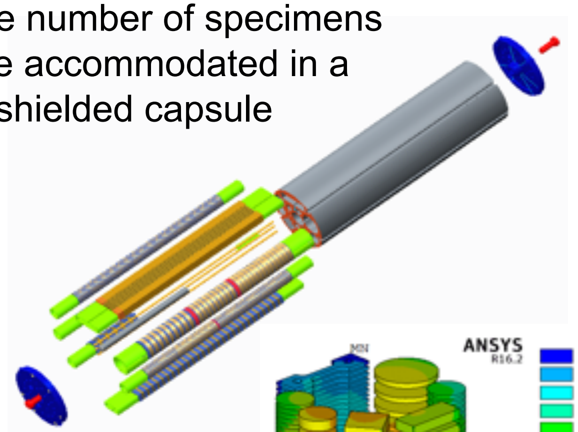
C.M. Parish, K. Wang, R.P. Doerner, M.J. Baldwin, Scripta Materialia, 127 (2017) 132-135.
K. Wang, M.E. Bannister, F.W. Meyer, C.M. Parish, Acta Materialia, 124 (2017) 556-567
(Early Career Award research program)

The PHENIX Collaboration Irradiated Large Number of Tungsten Materials for PFM Studies

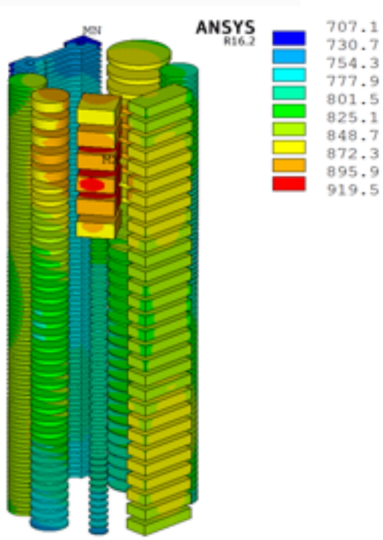
Research Highlight

HFIR Capsule Design Approach

Multiple holders containing a large number of specimens were accommodated in a Gd-shielded capsule



Thermal 3-D analysis calculated temperature distributions in specimen stacks



Challenge

The properties of neutron-irradiated tungsten are needed to determine feasibility and enable design of fusion plasma-facing components. Irradiations in HFIR play a critical role. Achieving the fusion-relevant solid transmutation rates is a challenge.

Experiment Design

The RB-19J irradiation experiment was designed for a reactor reflector position, adjacent to the outer fuel element. It incorporated a Gd neutron absorber to reduce the flux of thermal neutrons, thus controlling the W -> Re,Os transmutations. Temperatures were set by gas gap design, controlled by gas mixtures, and measured by thermocouples.

1,130 specimens included >10 material variables and multiple specimen types for physical and mechanical property determination and microstructural examination after irradiation.

Experiment Operation

RB-19J was in HFIR for four cycles during 2016, with W temperatures close to the design points of 500, 800 and 1,100ºC.

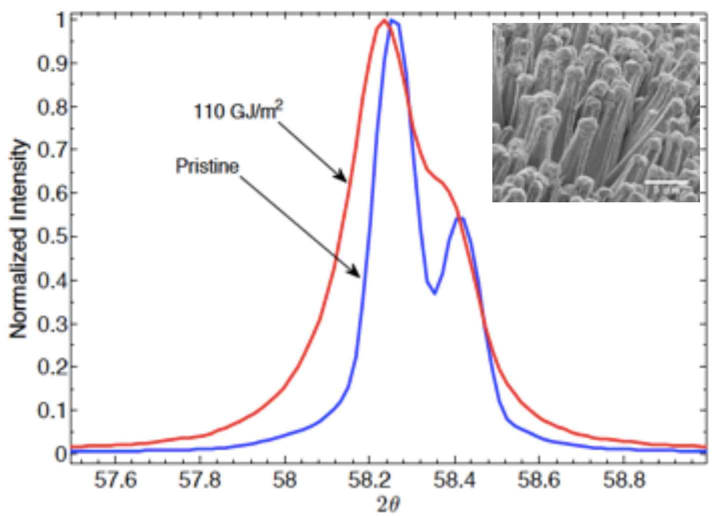
Achievement

Evaluation will provide a wealth of data on early-stage irradiation effects on properties of tungsten in PFM/PFC applications

Design Team: J. McDuffee, Y. Katoh, C. Petrie, W. Geringer, R. Howard

Experimental Measurements of Surface Damage and Residual Stresses in Micro-engineered Plasma Facing Materials

Research Highlight



(200) peak FWHM before and after plasma exposure. Peak broadening is observed as a function of repeated plasma exposure, and is a qualitative measure of residual plastic strain.

David Rivera, Richard Wirz, Nasr M. Ghoniem, "Experimental Measurements of Surface Damage and Residual Stresses in Micro-engineered Plasma Facing Materials," *J. Nuclear Materials*, under review, 2017.

Scientific Achievement

- Refractory metals with micro-engineered surfaces are developed as heat shields for materials in high heat flux applications.
- Tungsten surfaces with micro-pillar type surface architecture are shown to have significantly reduced residual thermal stresses after plasma exposure, as compared to those with flat surfaces.

Significance and Impact

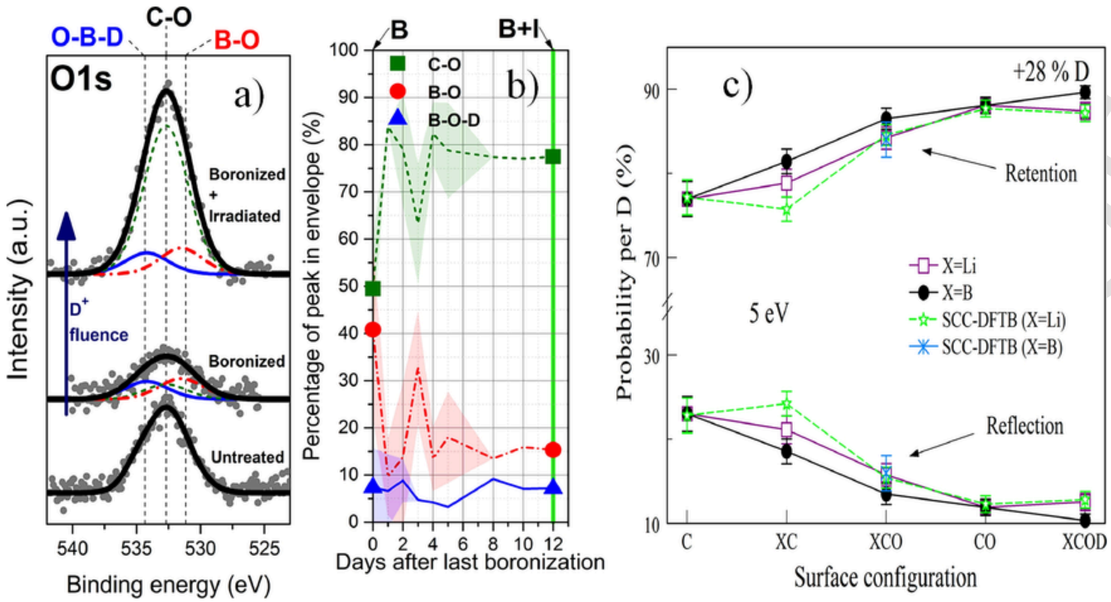
Development of resilient plasma facing materials will revolutionize the design and operation of divetors in fusion power systems.

Research Details

- X-ray diffraction (XRD) spectra of the W-(110) peak reveal that broadening of the Full Width at Half Maximum (FWHM) for micro-engineered samples is substantially smaller than corresponding flat surfaces.
- Spectral shifts of XRD signals show that subsequent cyclic plasma heat loading anneals out most of the built-up residual stresses in micro-engineered surfaces.

Research Highlight

In-situ PMI measurements and modeling of complex boronized surfaces with MAPP in NSTX-U



a) O1s XPS region showing retention of B retention (area of O-B-D peak) of D in NSTX-U, b) Computer simulations results of D uptake and reflection probabilities for D impact at various surface configurations with lithium and boron. XCOD surface has about 30% of D.

J. Gonzalez, F. Bedoya et al. "Quantification of irradiation defects in beta-silicon carbide using Raman spectroscopy." *Nucl. Mat. Energy* (2017) accepted.

Scientific Achievement

This work is the first PMI measurements in NSTX-U revealing the complex chemistry of boronized surfaces at the outboard divertor region in NSTX-U

Significance and Impact

In this work we demonstrated the importance of the role of oxygen in the retention of deuterium with boron mixed surfaces and graphite. Advanced computational modeling predicted the role of oxygen and B-C bonding.

Research Details

- In-situ XPS data with MAPP diagnostic revealed the irradiation-driven mechanisms that can drive oxygen to the surface and bind deuterium
- Advanced simulations with Predrag Krstic's group continue to complement work including ex-vessel surface chemistry studies

TPE Advances Understanding of Hydrogen Species Retention in Neutron Damaged Tungsten

Research Highlight

Fusion Safety Program's Mission:

- Assist the US fusion community in developing the inherent safety and environmental potential of fusion power by producing fusion nuclear and tritium licensing data at the Safety and Tritium Applied Research (STAR) facility and developing safety analysis computer codes

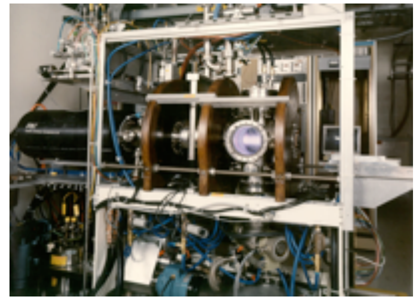
Recent STAR research highlights:

- Deuterium depth profiling results from Nuclear Reaction Analysis (NRA, SNL) reveal deuterium retention in plasma exposed single crystal HFIR neutron-irradiated tungsten (ORNL) under US-Japan PHENIX collaboration
- Results advance the understanding of types of defects and retention of hydrogen species in polycrystalline tungsten by eliminating the grain boundary defects created by neutrons in polycrystalline tungsten
- Glow-discharge optical emission spectroscopy (GD-OES, INL) on un-irradiated polycrystalline tungsten gives a baseline for future studying of HFIR irradiated PHENIX polycrystalline tungsten

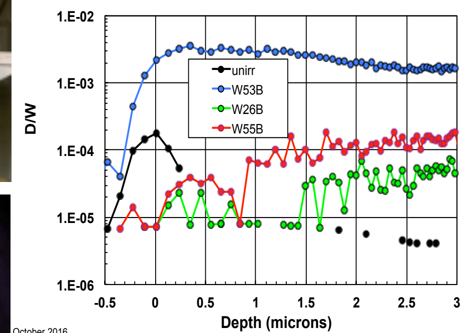
STAR Publication

- M. Shimada and B.J. Merrill, "Tritium Decay Helium-3 Effects in Tungsten", Nuclear Materials and Energy, DOI: 10.1016/j.nme.2016.11.006

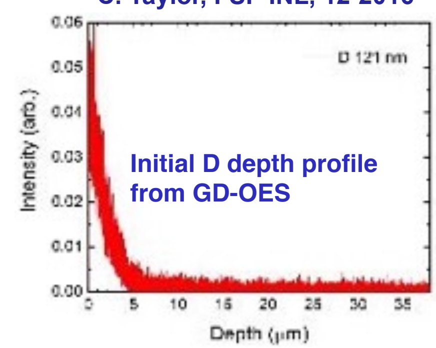
Tritium Plasma Experiment (TPE)



D depth profile in 0.1 dpa SCW
W. R. Wampler, SNL, 10-2016



C. Taylor, FSP INL, 12-2016



Structural and Chemical Evolution in Neutron Irradiated & He-injected Ferritic ODS PM2000 Alloy

Research Highlight

Science Objective

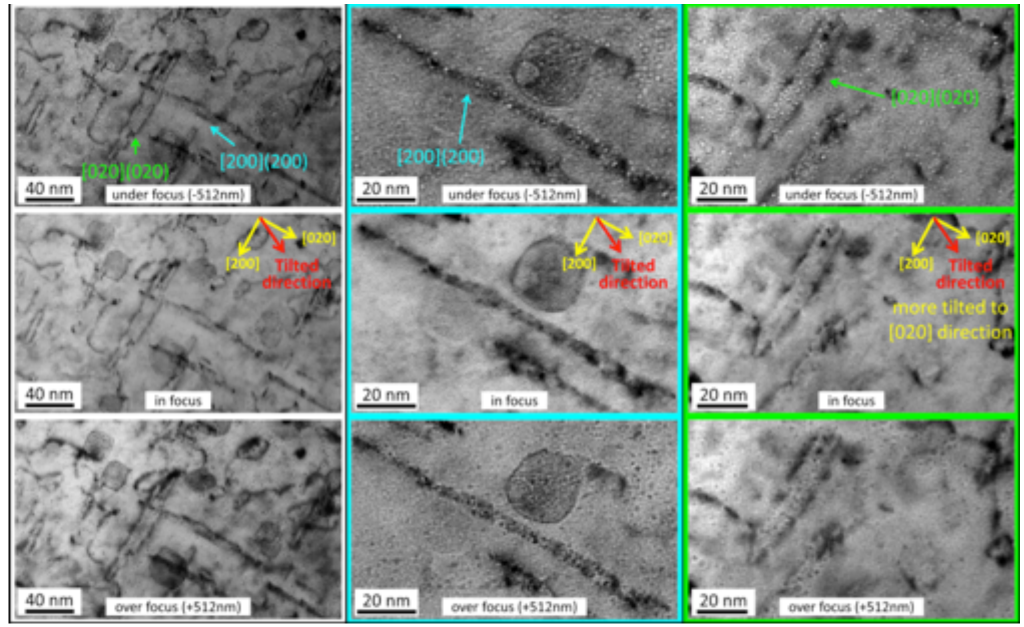
Generate data to assess simultaneous He-injection and neutron irradiation on microstructural evolution in a large grain-sized ferritic alloy to compare with nano-structured ferritic alloys

Why it Matters

In situ He injection (ISHI) studies effectively simulate the fusion environment. This study revealed a surprising synergism between He and dislocations loops that effectively controlled the He bubble distribution to yield a low swelling FeCrAl alloy.

Method

Neutron irradiation in HFIR using Ni/Al coatings on any substrate provide simultaneous He injection.



He bubble and void diameter data from transmission electron microscopy examination show that He bubble number density is 10x void number density and that voids are only observed within large ODS particles. The surprise finding is the ability of the dislocation loop microstructure to create and maintain a fine He bubble microstructure with an average diameter of 1.3-nm at a density of 10^{23} bubbles per m^3. The images show fine He bubbles (seen in overfocused and underfocused contrast) associated with interstitial dislocation loops. A large void can be seen in the ODS particle in the center images above.

Hee Joon Jung, Dan Edwards, Rick Kurtz (PNNL)
& T Yamamoto, Y Wu, GR Odette (UCSB)
J. Nuclear Materials **484** (2017) 68-80; http://dx.doi.org/10.1016/j.jnucmat.2016.11.02

U.S. DEPARTMENT OF ENERGY | Office of Science

Pacific Northwest NATIONAL LABORATORY

UCSB

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Vacancy Effects on the Formation of He & Kr Cavities in 3C-SiC Irradiated and Annealed at Elevated Temperatures

Research Highlight

Science Objective

Generate data that can be used to assess He impact on cavity swelling and Kr diffusion in SiC. This study finds that presence of He in 3C-SiC significantly promotes cavity growth. The results also show that long-range Kr diffusion does not occur in SiC up to 1600 °C.

Why it Matters

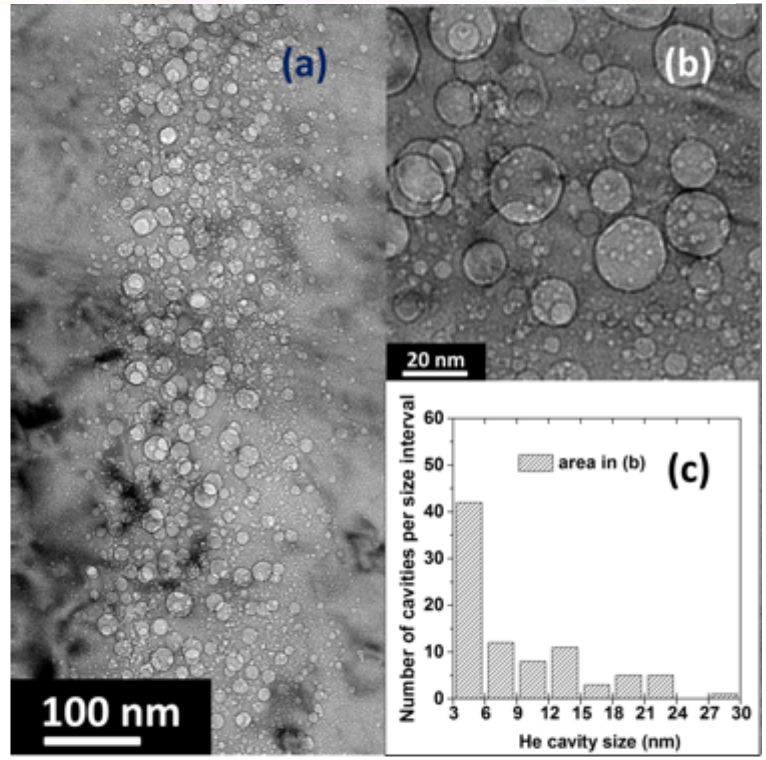
Helium cavity swelling in SiC affects its structural stability and thermomechanical properties, and could limit its service lifetime in fusion reactors. Fission gas Kr could be confined in SiC up to 1600 °C.

Method

Sequential He and Kr ion implantation in 3C-SiC, followed by thermal annealing at higher temperatures; microstructural examinations by transmission electron microscopy (TEM) and depth profiling of Kr.

Hang Zang, Weilin Jiang, Wenbo Liu, Arun Devaraj, Danny, Edwards, Charles Henager Jr., Richard Kurtz, Tao Li, Chaohui He, Di Yun, and Zhiguang Wang [PNNL Stafff underlined]

Nucl. Instr. Meth. Phys. Res. B 389-390 (2016) 40-47.
http://dx.doi.org/10.1016/j.nimb.2016.11.017



(a) Bright-field TEM image of He cavities in 3C-SiC sequentially implanted with He and Kr ions with latter penetrating the He depth region, (b) high-resolution image near the depth of He peak, and (c) size distribution of helium cavities estimated from (b), excluding those smaller than 3 nm.

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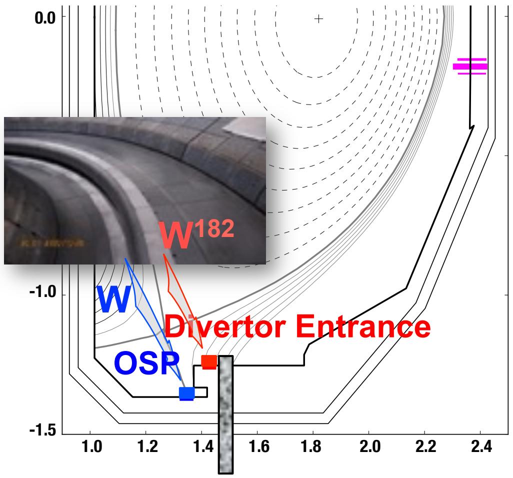
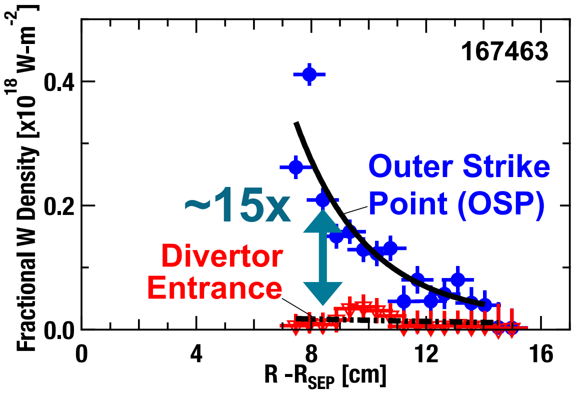
Research Highlight

Insight into high-Z plasma contamination from two divertor regions achieved via isotopically enriched W coatings



MiMES



W transported into main chamber

DiMES

Significance and Impact

The need to reduce W impurity content to <50 ppm in a fusion device necessitates a thorough understanding of how W is transported from the divertor to the core. This study identifies the dominant location of W impurity sourcing- the strike point region, for high performance plasmas.

Scientific Achievement

A unique DIII-D experiment involving isotopically enriched W coatings in the divertor indicates that W impurities are predominantly transported to the midplane from the OSP in high performance, advanced tokamak (AT) plasmas.

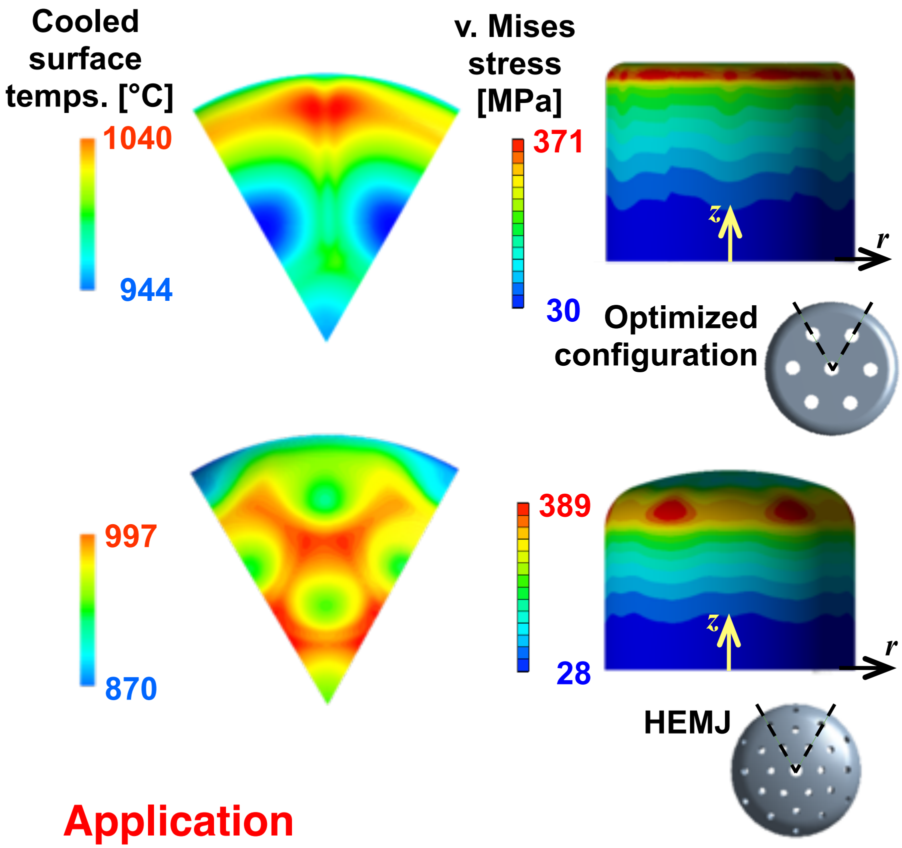
Research Details

- Toroidally symmetric isotopically enriched W coatings installed in two divertor locations.
- Inductively coupled plasma mass spectrometry utilized to distinguish W isotopes on the upstream **MiMES** probe to assess W transport from the divertor.
- Local W PMI and migration is monitored by **DiMES** probe near divertor W sources.
- ~3x more W is produced at the OSP than from the divertor entrance, but ~15x more Watoms from the SP are detected at the midplane by MiMES.

T. Abrams et al., APS-DPP Conference 2016
E.A. Unterberg et al., APS-DPP Conference 2016

Research Highlight

Thermal-Hydraulics Studies and Optimization of He-Cooled Divertors



Engineering Achievement

Numerical optimizations suggest that the complex geometry of the He-cooled modular divertor with multiple jets (HEMJ) can be simplified without compromising thermal-hydraulics performance.

Significance and Impact

The HEMJ, the He-cooled divertor design proposed for DEMO, is difficult to manufacture. A simpler, more robust design should improve reliability and reduce cost.

Research Details

- Numerical simulations in ANSYS with one-way coupling
- Configuration with a flat cooled surface and an array of 7 jets (*vs.* the curved surface and array of 25 jets for HEMJ) can achieve 6.5% higher heat transfer coefficient of 37.5 kW/(m^2-K) and 4.8% lower maximum von Mises stress on cooled surface of 371 MPa
- Max. cooled surface temperature of 1040 °C is, however, 42 °C higher than HEMJ

B. Zhao, S.A. Musa, S.I. Abdel-Khalik and M. Yoda (2017) "Optimization of multiple jet arrays for helium-cooled finger-type divertors," to appear in *Fusion Science and Technology*

Application

Helium-cooled solid divertors leading candidate for solving power and particle exhaust issues in long-pulse fusion magnetic fusion devices.

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Georgia Tech

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Efficiency Improvement of High Power ECH Transmission Lines

Research Highlight

Scientific Achievement

This research provided experimental evidence of the efficacy of a method of eliminating unwanted modes in high power ECH transmission lines, thus improving their efficiency and safety.

Significance and Impact

ECH power, transported through highly oversized corrugated metallic waveguides, is lost due to diffraction at bends and misalignments. Lost power reduces the available plasma heating power and causes a heat load that can lead to serious damage.

Research Details

- This new concept uses two miter bends with mirrors that tilt and convert the unwanted modes into the primary mode.
- The concept was successfully demonstrated by collaborators in Japan at QST, Naka.

Application

The new technique can be used to reduce unwanted modes in ECH transmission lines. It may find application in the worldwide ECH program, including ITER.

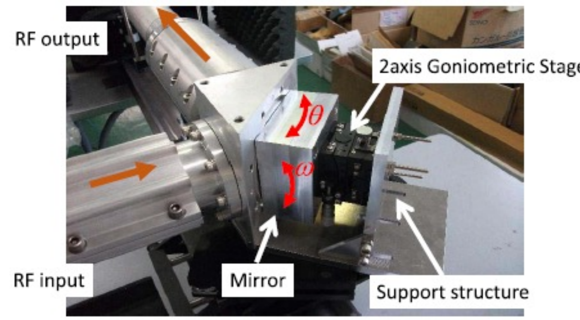
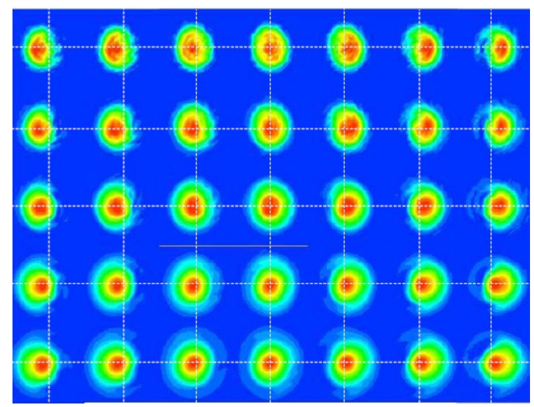


Photo of the waveguide miter bend with a tiltable mirror.



Measured radiated microwave power. Left to right: Effect of mirror tilt; Top to Bottom: Distance from waveguide to detector

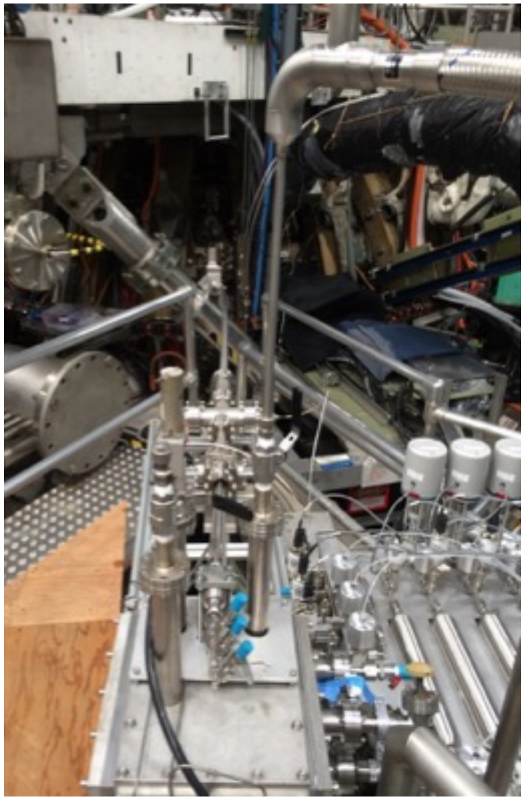
Y. Oda, R. Ikeda, M. Fukunari, T. Ikeyama, K. Takahashi, K Kajiwara, T. Kobayashi, S. Moriyama, K. Sakamoto, M. A. Shapiro and R. J. Temkin, IEEE Trans. Plasma Science 44, pp. 3392-3397 (2016).

MIT: S. Jawla, H. Hoffmann, S. Schaub, M. Shapiro and R. Temkin; supported by DOE VLT and US IPO

New ITER-like Shattered Pellet Injector Installed on DIII-D for key Disruption Mitigation Studies

- **A three-barrel shattered pellet injector (SPI2) was fabricated and tested at ORNL for use on DIII-D in support of the ITER disruption mitigation system**
  - Based on the SPI design developed for ITER.
  - Tested at ORNL for D2, Ne and mixed pellets of 7 and 8.5mm sizes**.
- **Key experiments for ITER disruption mitigation issues are planned.**
  - DIII-D remains the only machine with SPI and now has 2 systems to study simultaneous injection from multiple locations.
  - Experiments on thermal mitigation and runaway electron dissipation will be performed in 2017.
- **Installation and interfacing to DIII-D was completed in 1st QTR 2017.**
- **Commissioning to be performed in Jan 2017 with experiments commencing in Feb 2017.**

DIII-D



SPI2

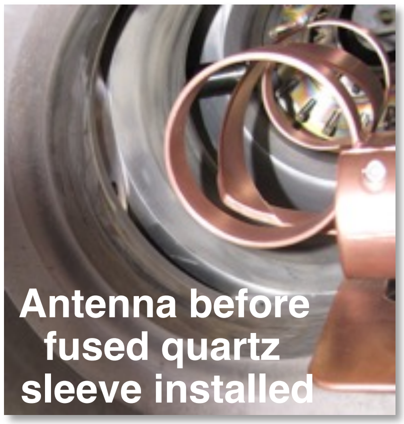
S. Meitner, T. Bjornholm, D. Shiraki, L. Baylor

U.S. ITER/ORNL has responsibility for the ITER disruption mitigation system plasma

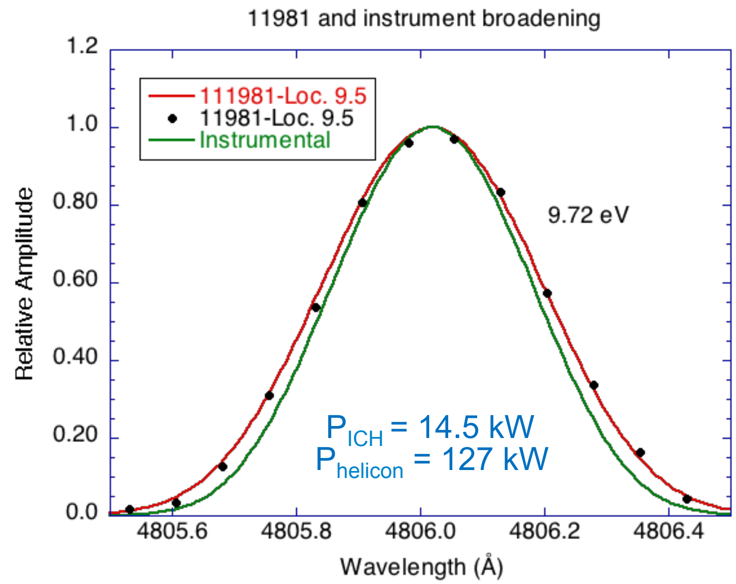
***S.J. Meitner et al., Fusion Sci. Tech., to be published*

DIII-D National Fusion Facility | Oak Ridge National Laboratory | VLT Virtual Laboratory for Technology For Fusion Energy Science

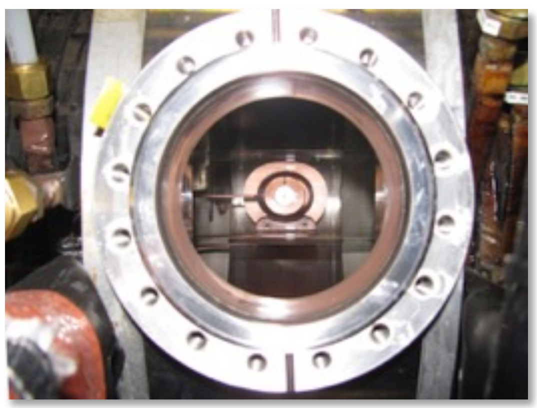
Proto-MPEX Demonstrates Direct Ion Heating with ICH



8.4 MHz ICH
14 kW coupled
Deuterium plasma with Argon trace seeding



Vacuum feedthrough, feeds, and fused quartz sleeve



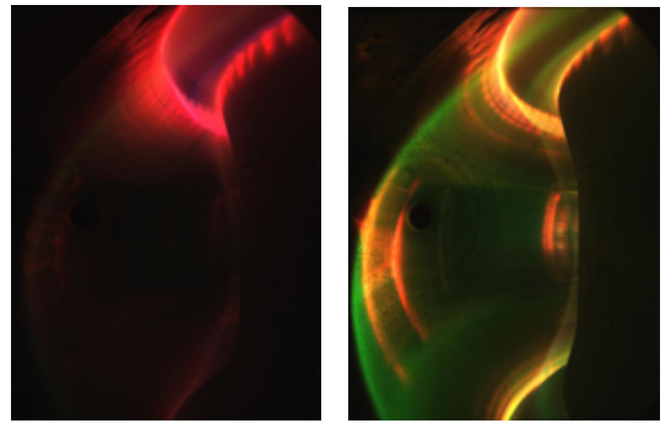
Ion temperatures measured downstream: ~ 3 – 13 eV

R. Goulding, R. Isler, T. Biewer, J. Rapp (ORNL) & J. Beers (UT student)

THE UNIVERSITY OF TENNESSEE KNOXVILLE
OAK RIDGE National Laboratory
VLT Virtual Laboratory for Technology For Fusion Energy Science

Collaborative experiments conducted using Li delivery technologies on EAST in Dec'2016

- Inserted flowing liquid Li (FLiLi) driven by EM pump
- FLiLi compatible with auxiliary heated plasmas: 0-4.5 MW
  - Improved plasma performance including full-field ohmic H-mode
  - No obvious limiter surface damage
- Used Li dropper and granule injector for ELM control
- Multi-institutional US team integrated with ASIPP PWI team to run experiments

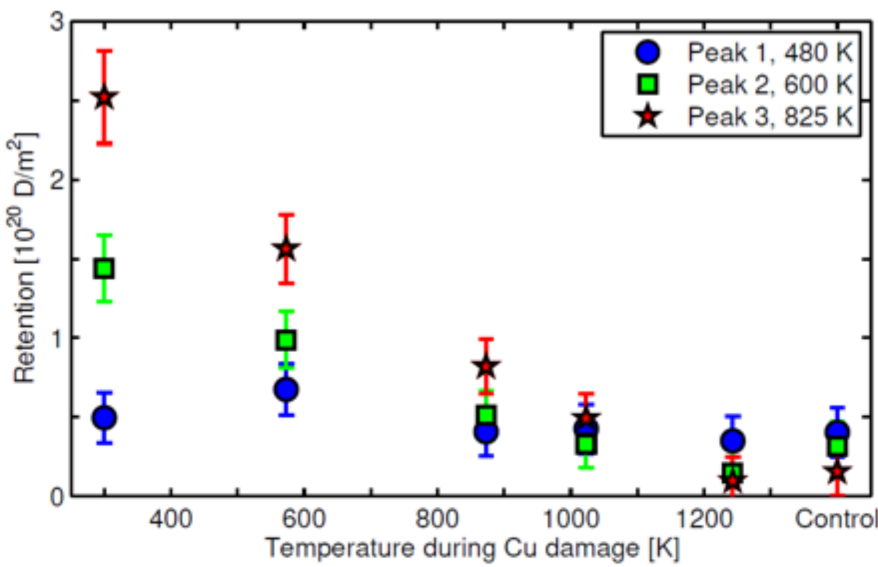


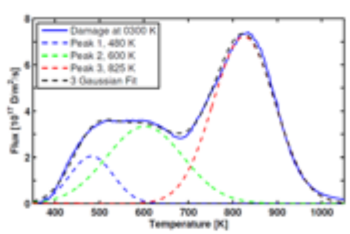
Without FLiLi **With FLiLi**

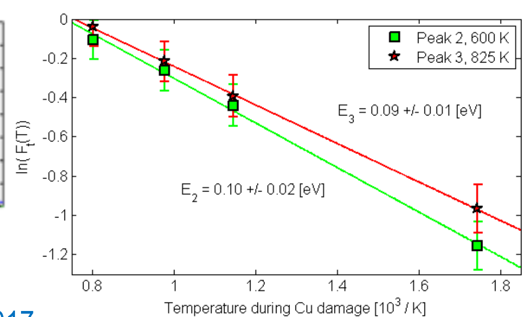
US-ASIPP Team members-see photo

ASIPP
OAK RIDGE National Laboratory
GENERAL ATOMICS
JOHNS HOPKINS UNIVERSITY
PPPL PRINCETON PLASMA PHYSICS LABORATORY
VLT Virtual Laboratory for Technology For Fusion Energy Science

Heavy ion beam irradiation at elevated temperature leads to annealing of induced damage







M. Simmonds et al.,
submitted to J. Nucl. Mater. 2017.

- 5 MeV Cu ions are used to simulate neutron damage in W at elevated temperature, followed by D plasma exposure at 383 K
- Ion beam displacement damage results in an increase in higher energy D trapping sites (i.e. peaks 2&3)
- Low energy traps (peak 1) are unchanged with ion irradiation
- At ~1000-1200 K, ion-induced damage anneals out and intrinsic trap levels are recovered
- Analysis of TDS data reveals an activation energy for damage recovery of ~ 0.1 eV, consistent with self-interstitial atom mobility
- Measurements need to be repeated with the presence of H/He

UCSD/LANL collaboration supported in part by UC Office of the President Research Fund under award #12-LR-237801.

UCSD University of California San Diego
Los Alamos NATIONAL LABORATORY EST. 1943
VLT Virtual Laboratory for Technology For Fusion Energy Science

SNL Continues to Advance Understanding of PMI & PFC design and Support ITER R&D needs

Science based understanding of PSI

- How do high fluxes of H/D/T and He interact with materials? *(TPE & PISCES collaborations)*
  Sandia hosted US-Japan Workshop on PMI / HHF (Nov. 2016)
- What is the dynamic response of surfaces? *(UT & PISCES collaborations)*
  Kolasinski, invited talks at CAARI 2016 and DRM workshop 2016

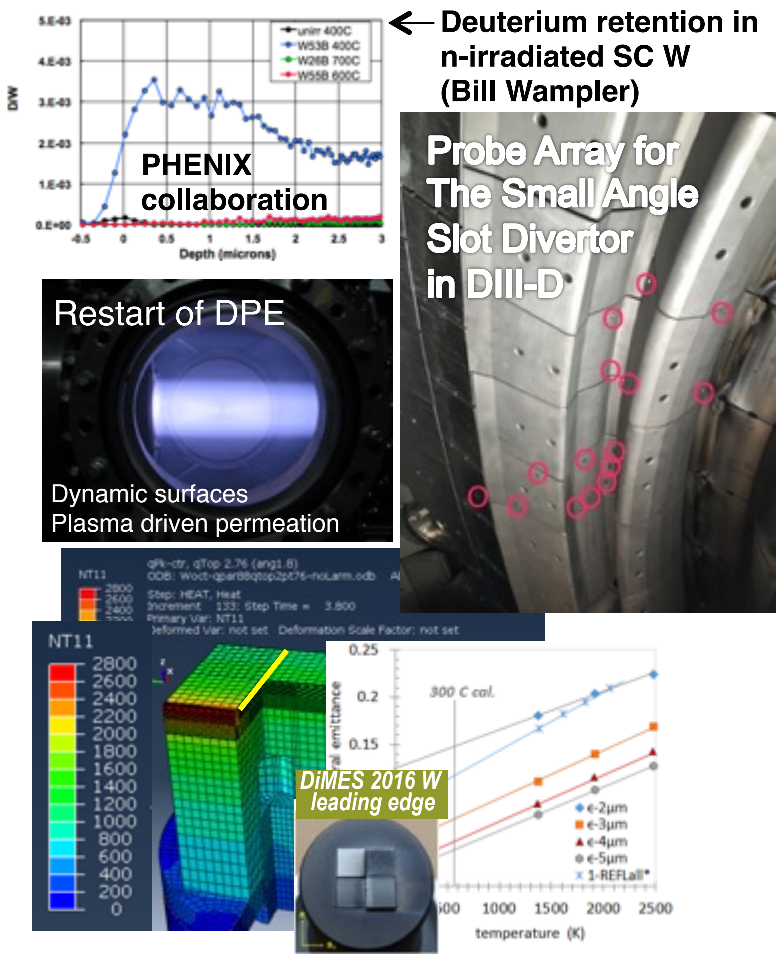
Improved plasma edge measurements

- Measure erosion/redeposition accurately (DIII-D, NSTX-U, EAST collaborations)
  Heat and particle flux during tungsten ring exp.; Barton, APS 2016
- Develop H neutral and other sensors
  Buchenauer, Talin, Friedmann

Science-based engineering

- PFC thermal models help us understand* power handling and how PFCs fail. *(DIII-D, NSTX-U)*
  **Temperature from IRTV for W in DiMES is hundreds of degrees lower than real temperature after correcting emittance for temperature and wave length.*
- Developing novel materials and designs

Nygren et al, (invited oral, new vision of materials) ISFNT, FED Apr 2016, Nygren, Tabares, (invited review, liquid surfaces) Part 1 Nucl. Mat. Energy **9** 2016 6-21; Nygren, Part 2 NME (in review); Nygren, PFMC, Phys. Scr. **T167** 2016



Sandia National Laboratories

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Use of Graphitic Foams Explored for Monoblock Fusion Divertor Components

Achievement

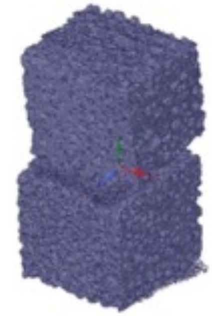
- Demonstrated that a nearly isotropic high-conductivity, low-Z plasma facing material is possible by combining pyrolytic graphitic ligaments with an isotropic engineered microstructure

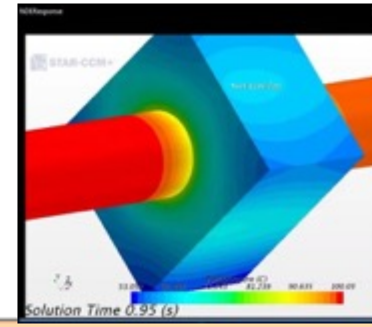
Significance and Impact

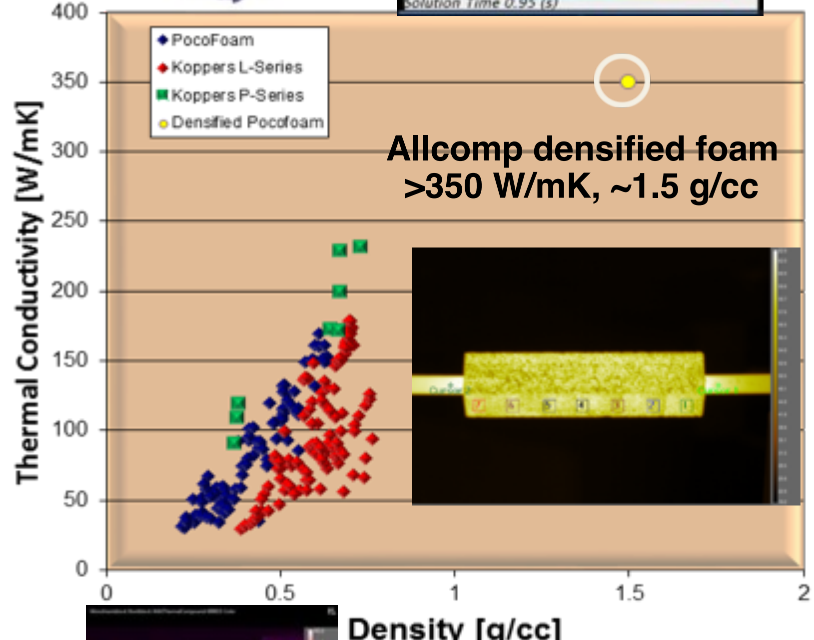
- For the first time, the thermal efficiency of low-Z armor is comparable to the copper heat sink
- Max Planck IPP interested in fielding the monoblock in the W7-X stellarator for potential use in divertor scraper element.

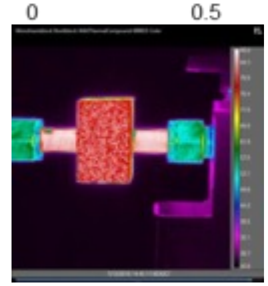
Research Details

- Densified graphitic foam and mock-ups produced
- Thermal properties measured, k=265 W/mK to date, expect 350 W/mK
- Robust braze joint obtained on CuCrZr tubes
- Hot water IR thermography showed that using no braze joint actually performs better thermally
- Thermal modeling and tomography of microstructure completed
- Samples for W7-X multipurpose probe fabricated for plasma exposure during OP1.2a.









D. Youchison, A. Lumsdaine, J. Klett, R. Dinwiddie, P. Bingham

Oak Ridge National Laboratory

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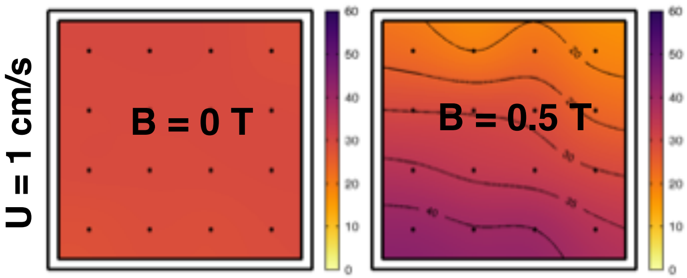
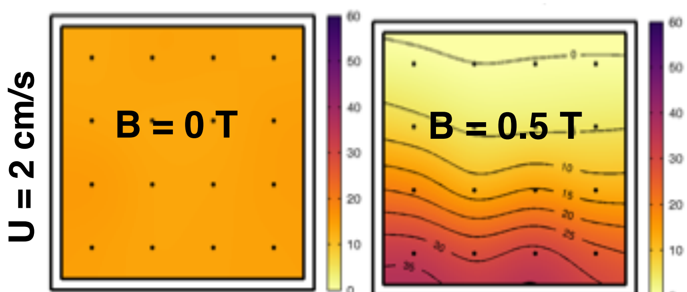
UCLA Highlights from Blanket LM MHD and Pebble Bed Thermomechanics Studies

1. Heat Transfer in MHD PbLi Flows (in collaboration with IPR, India)
2. Effect of Pre-compaction and Cyclic on Pebble Bed Stress State (in collaboration with NFRI, Korea)

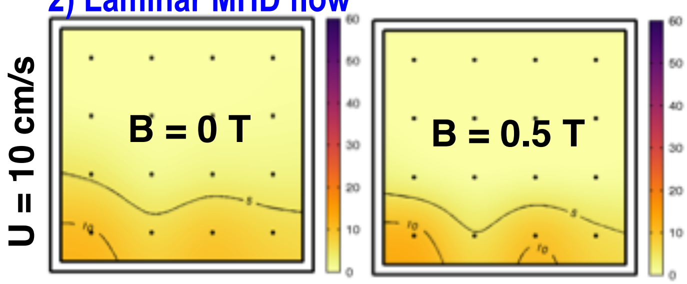
Experiments in MaPLE suggest 3 possible thermal convection flow regimes in a PbLi blanket



1) **MHD thermogravitational convection**



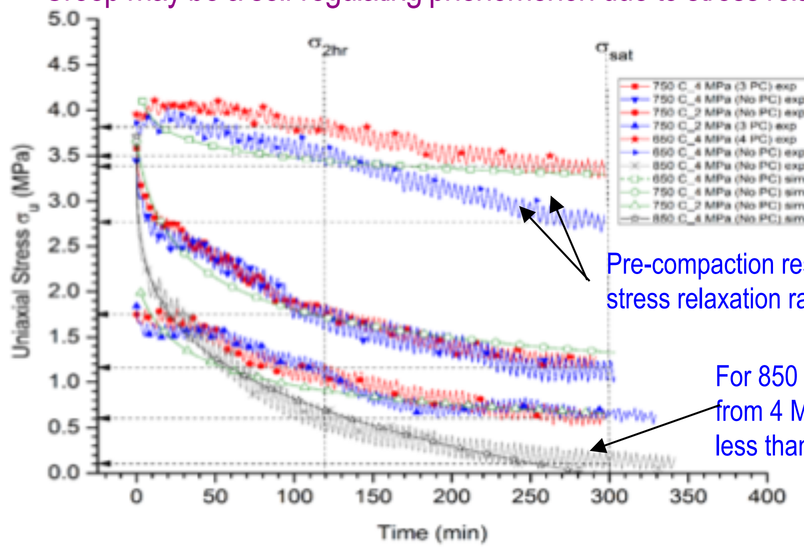
2) **Laminar MHD flow**



3) **Turbulent MHD flow**

Regions of pebble bed where creep mechanisms activate (T > 650 °C) will experience stress relaxation

- Stress relaxation rate strongly depends on temperature
- Creep may be a self-regulating phenomenon due to stress relaxation



Pre-compaction results in lower stress relaxation rate for T< 650 °C.

For 850 °C, stress dropped from 4 MPa to 0.1 MPa in less than 4 hrs.

M. Abdou, A. Ying, N. Morley, S. Smolentsev et al.

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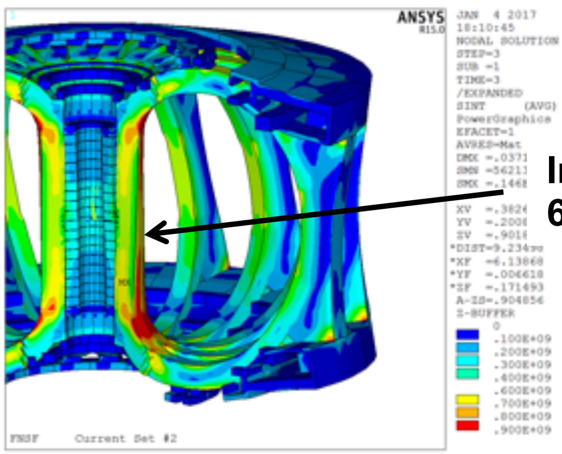
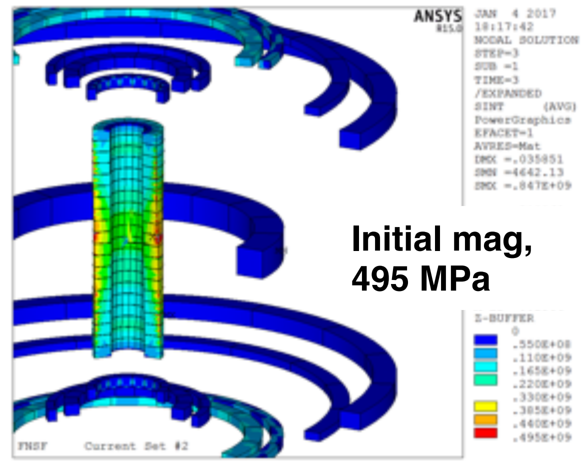
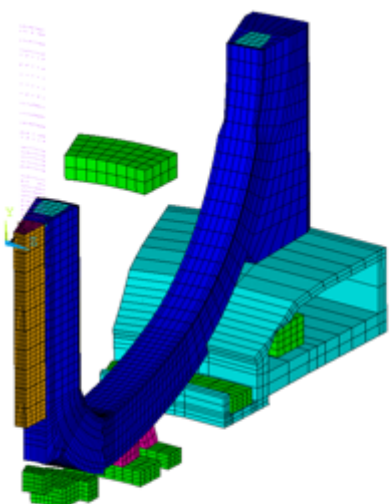
Bucked and Wedged CS/TF Solution Found to Keep TF and CS Coil Stresses Acceptable in FNSF



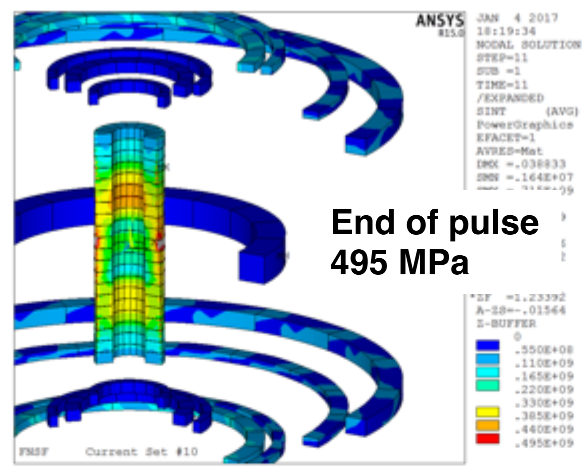
Multi-fiducial state analysis of CS/PF coil stresses





LTSC chosen for FNSF TF and CS/PF, high performance conductor
Ternary Nb_3Sn
1200 A/mm^2, 4.2 K, 16 T

TF coil Super-structure on outboard to support large outer TF leg and allow horizontal maintenance



P. Titus, PPPL

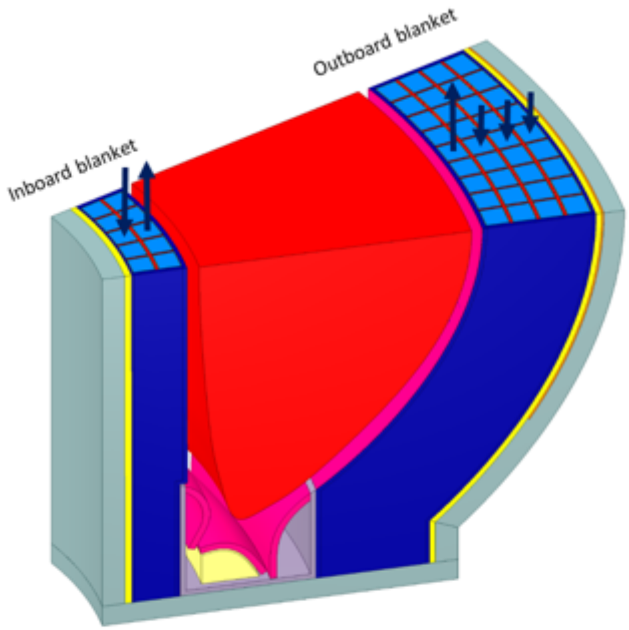
Fusion Energy Systems Studies

PPPL PRINCETON PLASMA PHYSICS LABORATORY

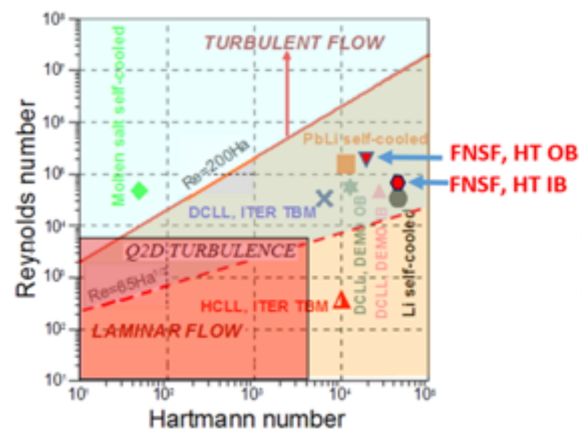
VLT Virtual Laboratory for Technology For Fusion Energy Science

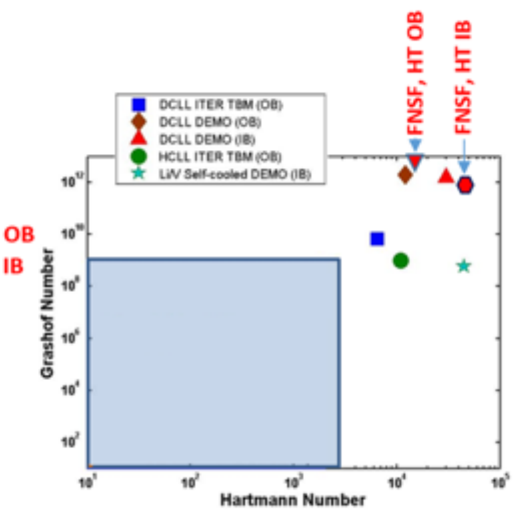
MHD Flow Model with 2D & 3D MHD Solvers are Used to Determine Pressure Drops in PbLi flows in FNSF DCLL Blanket

For 518 MW of fusion power and a peak neutron wall load of 1.75 MW/m2, a SiC Flow Channel Insert (FCI) was found to have acceptable pressure drops in the blanket, while the sandwich FCI did not unless the fusion power could be reduced.



The dimensionless numbers, *Re*, *Ha*, *Gr* can be used to identify flow regimes, and to serve as metrics to measure R&D progress on the pathway from the FNSF to DEMO. The FNSF provides DEMO-prototypic parameters.





Fusion Energy Systems Studies

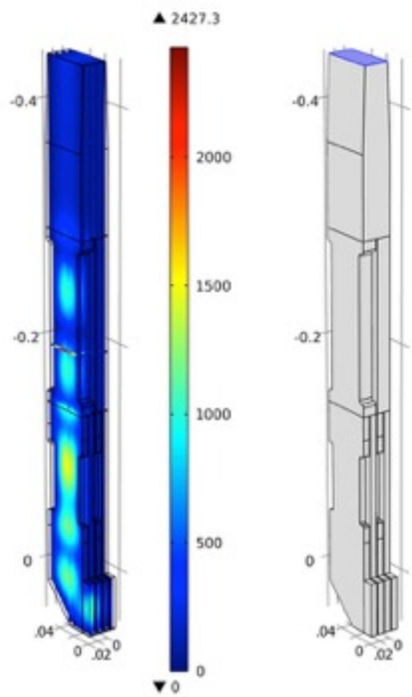
S. Smolentsev

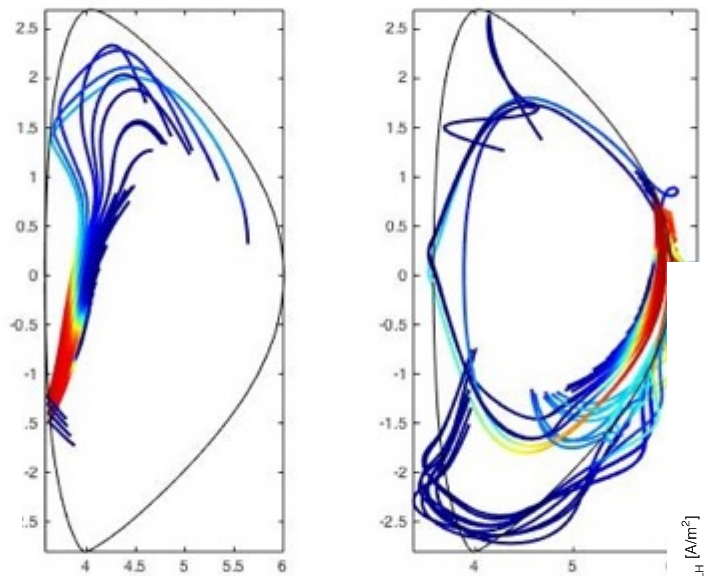
Ucla

VLT Virtual Laboratory for Technology For Fusion Energy Science

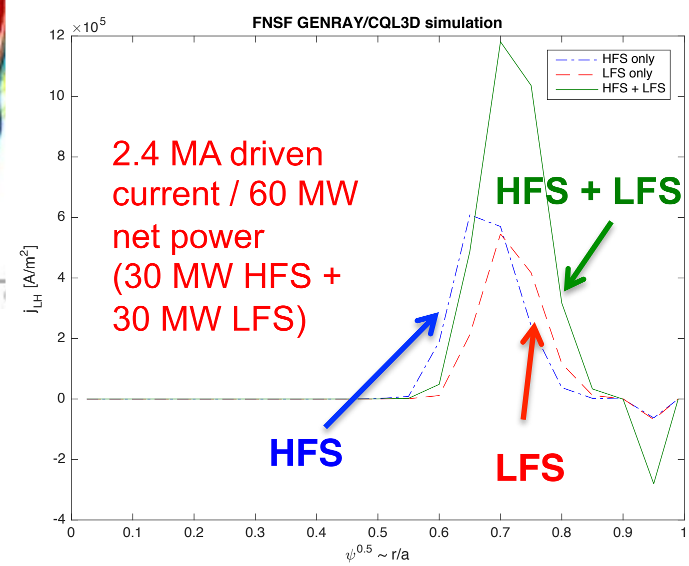
Lower Hybrid Current Drive Analysis for FNSF Shows High CD Efficiency and Effective Launch Locations on HFS & LFS

➔HFS antenna takes advantage of quiescent plasma and high B-field to obtain deeper wave penetration to core plasma





🡐LFS antenna launches from above the midplane to increase its h_{CD}, but avoiding interference with passive stabilizer plates



Ray-tracing results from GENRAY-CQL3D, power is very low after red rays

🡐 Compact multijunction antenna concept for use with HFS launch of LH waves, and same approach can reduce LFS LH footprint in breeding blanket

G. Wallace

Fusion Energy Systems Studies

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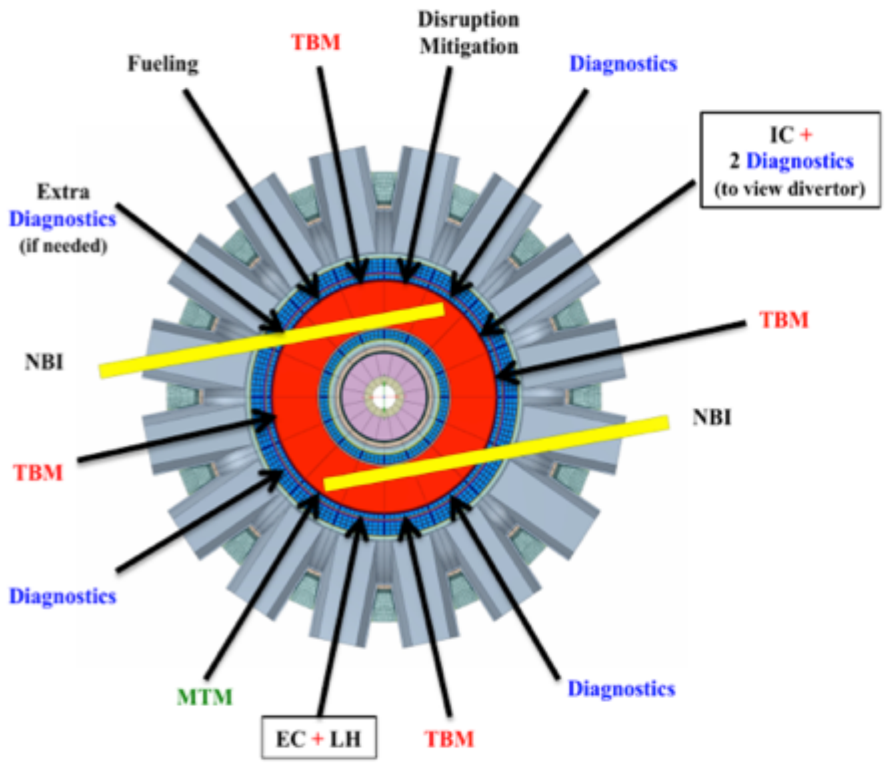
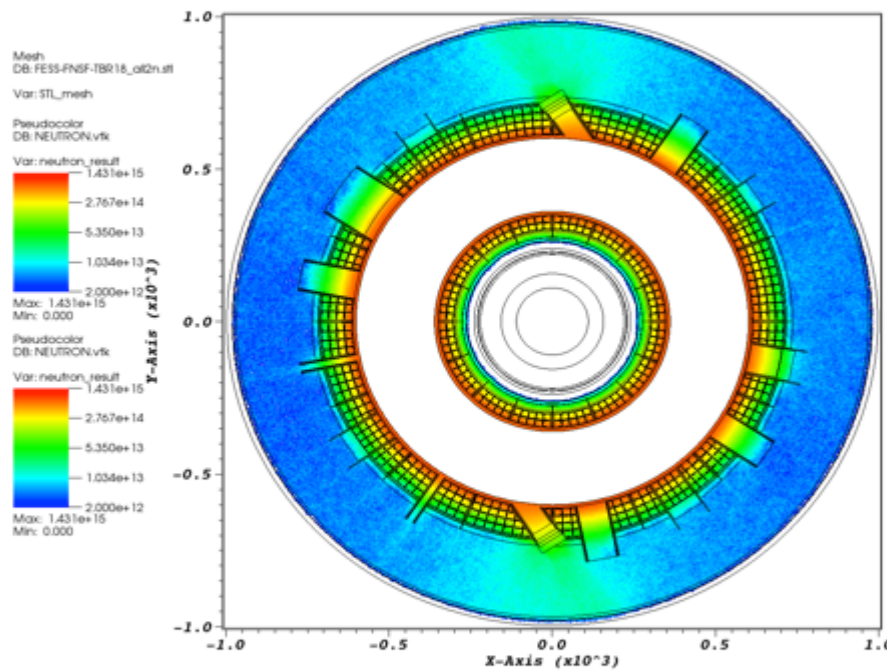
3D Nuclear Analysis First to Analyze Wide Range of Sectors to Determine Tritium Breeding Ratio for FNSF

Top view with worst case penetrations; H/CD, diagnostics, TBM, MTM



Neutron flux, all sectors together

TBR = 1.068



A. Davis, L. El-Guebaly, M. Harb, UW

Fusion Energy Systems Studies

WISCONSIN

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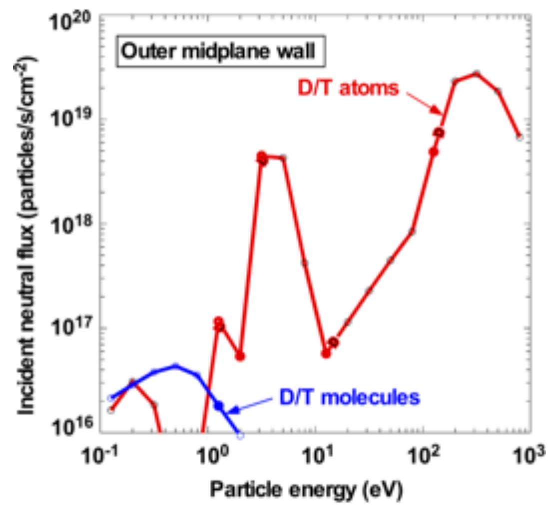
Design/Assessment of Divertors for FNSF Study

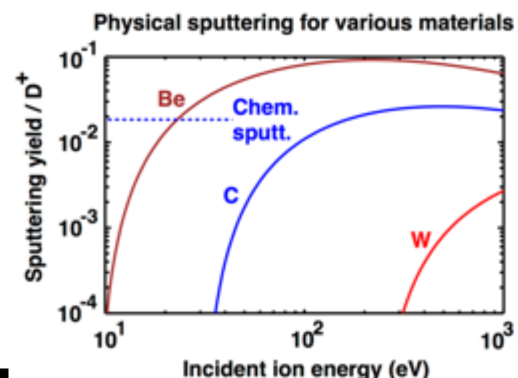
- Provided divertor plasma conditions for highly radiative detachment conditions for FNSF using UEDGE.
- Computed corresponding neutral charge-exchange (CX) flux on walls using DEGAS 2 (see figures on right).
- Assessing implications for sputtering of tungsten (W) from high-energy CX tail including prompt redeposition.
- Refined and utilized molecular DT gas model to complement atomic gas model for computing divertor neutral pressure.

T.D. Rognlien, M.E. Rensink, *Fusion Eng. Design, in prep.*

Tom Rognlien and Marv Rensink (LLNL FWP SCW1122)

Charge-exchange neutral energy distribution for FNSF at midplane wall





Fusion Energy Systems Studies

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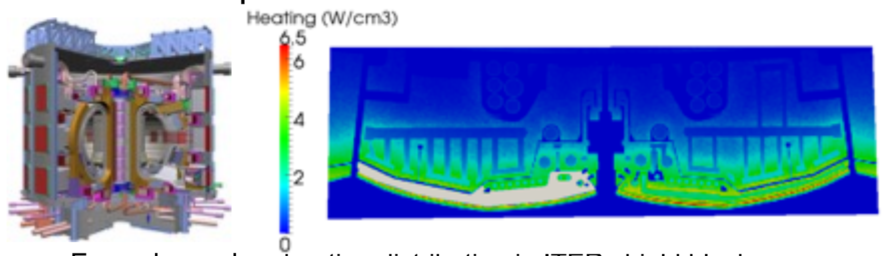
Updated Nuclear Data Impacts Fusion Neutronics Calculations

Both Monte Carlo and Deterministic radiation transport codes are important tools in designing fusion reactors

- they provide the distribution and level of radiation e.g. neutron flux/fluence, radiation damage (dpa), helium production, tritium production, radiation dose (n +g), nuclear heating (n+g), and shutdown dose/activation

The radiation transport codes need to have accurate nuclear data to produce accurate results



Example: nuclear heating distribution in ITER shield block

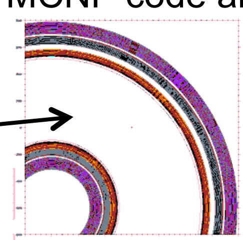
Fusion Evaluated Nuclear Data Library (FENDL):

- International effort coordinated by the IAEA that assembles a collection of best nuclear data from national cross section libraries, e.g. ENDF/B (US), JENDL (Japan), JEFF (Europe), BROND (Russia)
- Coordinates experimental and calculational benchmarking of resulting library

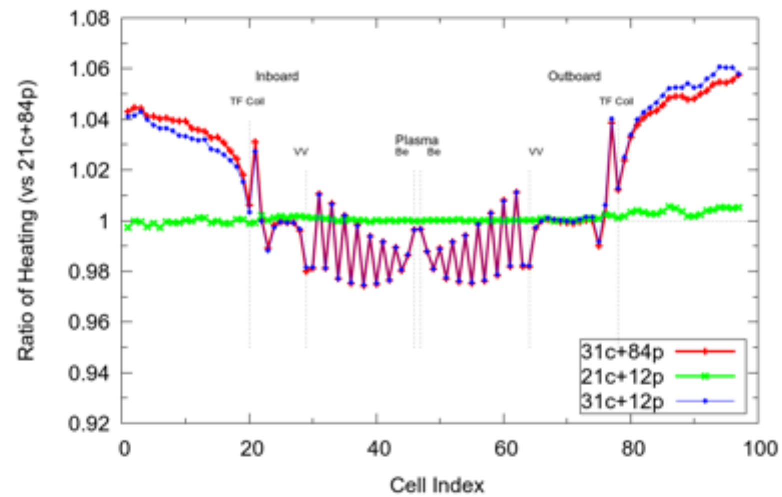
Recently released version 3.1b

Evaluated radiation levels using a simple, yet realistic 1-D model of ITER with the MCNP code and updated nuclear and photo-atomic data

Plasma region

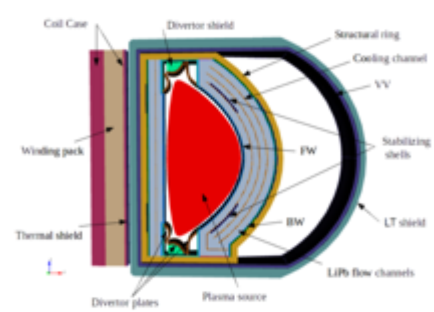


➢ Results show nuclear heating of toroidal field coil increases up to 6% with new library

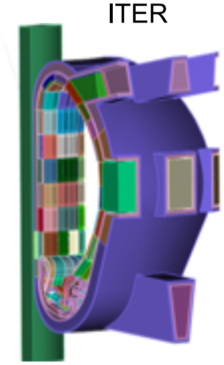


Future work will examine the impact using detailed 3-D CAD models of ITER & FESS-FNSF

FESS-FNSF



ITER



T.D. Bohm, M.E. Sawan

Tritium Processing Development for Magnetic Fusion

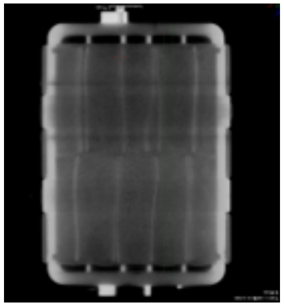
Application

Development and testing of tritium fuel cycle components and data-mining of TSTA historical documentation.

Scientific Achievement

Preliminary performance data has been collected for a uranium hydride storage bed design for tritium. Initial development of a database for TSTA historical documentation.

Pre-hydride STACI Bed



Post-hydride STACI Bed

Significance and Impact

- The fundamental engineering parameters study for a uranium hydride storage bed has been completed. This work baselined the performance of the Self-assaying Tritium Accountability and Containment Unit (STACI) design for ITER
- Completion of 31,000 pages of TSTA documents and over 4000 handwritten notebook pages have been scanned and uploaded to a searchable database (dtSearch)

Research Details

- 14 hydride and dehydride test were performed on a previously unhydrided STACI unit using hydrogen as a surrogate for tritium. Engineering parameters have been baselined to establish; U:H ratio (storage capacity), loading and unloading time, performance with mixed gas flows, and CT images of volume expansion of uranium material (see images).
- Initial database has been completed on a standalone computer. All file converted to a pdf searchable format. Test searches have been completed .

W. Kirk Hollis, Dave Dogruel, Brain Arko:
STACI experiments in collaboration with ITER IO, NFE-14-0030

Los Alamos NATIONAL LABORATORY EST.1943

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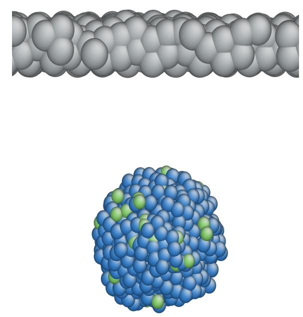
Atomistic modeling of He-H synergies performed

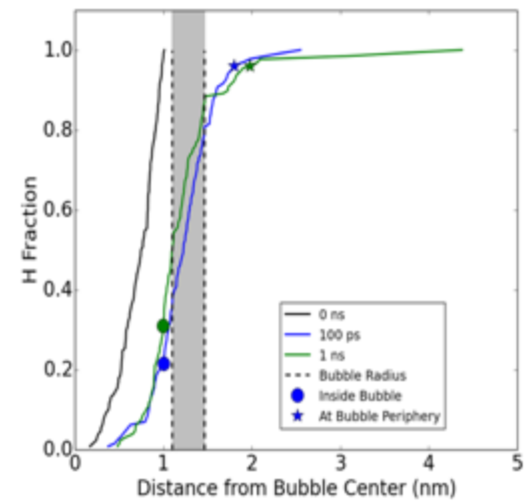
- MD simulations performed for 2 nm diameter bubble containing high pressure He (3 He/vac) and random distribution of H (0.5 H/vac) at 1800K
- H is observed to rapidly migrate to bubble periphery and remains 'trapped' at the bubble interface
- H trapping energies to the bubble periphery have been quantified as ~2.3 eV, implying a desorption energy > 1000 K
- Raises question about potential for tritium trapping/inventory
  - artifact of interatomic potential
  - short time MD simulations

0 ps

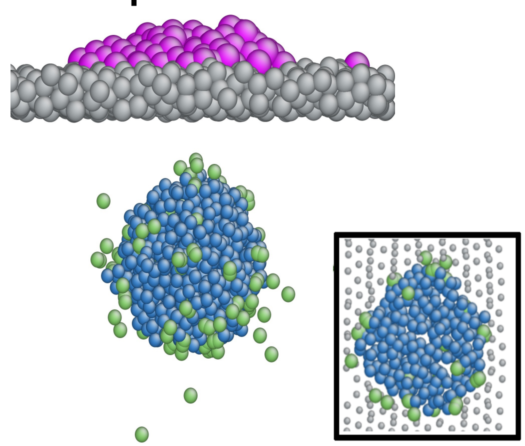


Green: Hydrogen
Blue: Helium
Grey: Surface Tungsten
Magenta: Adatoms

H Distribution for the (111) 1800 K with 3 He/V and 0.5 H/V



100 ps



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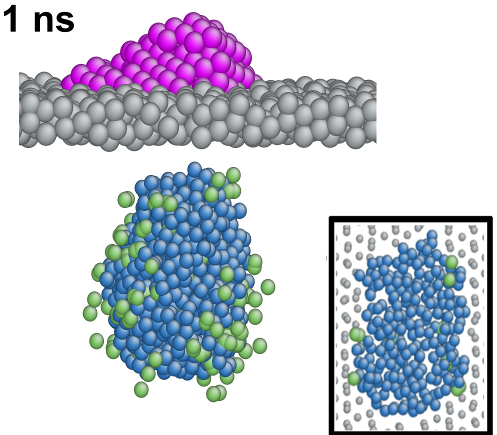

* Bergstrom, Cusentino and Wirth, *Fusion Sci. & Technology;* dx.doi.org/10.13182/FST16-121 *(2017) in press.*

THE UNIVERSITY OF TENNESSEE KNOXVILLE

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LiMIT: Progress on New Design for EAST Tokamak & COMSOL modeling of trench structures

(LiMIT: Liquid-Metal Infused Trenches)

Research Details

- Advance the reactor-relevant applicability of the LiMIT system by developing a full-scale limiter system that can be placed on the EAST tokamak in Hefei, China.

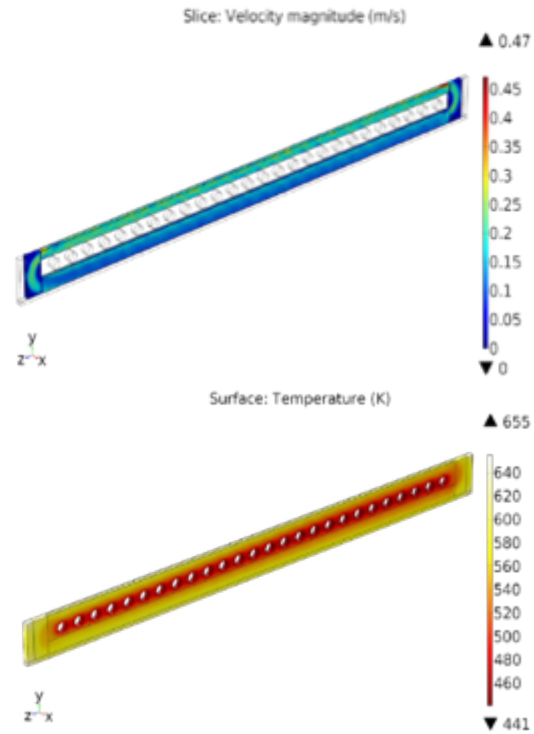
Scientific Achievement

- Heat flux is wide Gaussian on the top surface that provides at least 0.5 MW/m2 on middle 15 cm and 0.2 MW/m2 on entire bottom surface to mimic backside heating
- Strong velocity (25 cm/s) and steady flow conditions
- Simulations maintain reasonable steady state values for the lithium temperature, less than 350 degrees Celsius, which is well within operating limits

Significance and Impact

- Design ready for fabrication, and the experiment schedules for planned installation of the integrated module in EAST is being finalized. These tests will verify the long pulse operation capacity of the flowing lithium PFC module

COMSOL Multiphysics simulations of trench flow



D. Ruzic et al

CPMI http://cpmi.uiuc.edu Center for Plasma-Material Interactions
ILLINOIS
VLT Virtual Laboratory for Technology For Fusion Energy Science

Estimation of Fuel Retention in Liquid Metal PFCs – experiments with lithium

Research Details

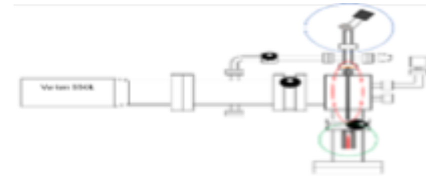
- Thermal dissociation studies of LiH, estimation of H evolution by residual gas analyzers and time resolved partial pressure measurements.

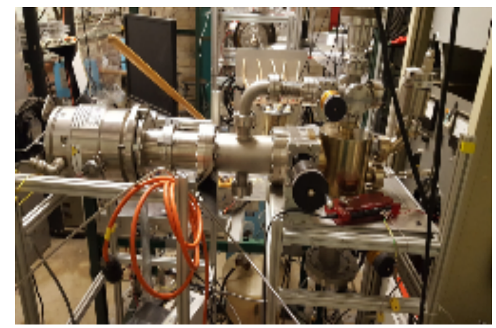
Scientific Achievement

- The rate of hydrogen evolution begins to dramatically increase around 535 °C
- Hydrogen partial pressures were found from a series of analog scans taken with the Magnetic Sector RGA.
- Measurements performed at FLIRE in Illinois for hydrogen absorption in pure flowing liquid Li showed retention at 0.1-0.2% D concentration in exposed Li samples.
- The rate of hydrogen evolution begins to dramatically increase around 535 °C
- Hydrogen partial pressures were found from a series of analog scans taken with the Magnetic Sector RGA.
- Measurements performed at FLIRE in Illinois for hydrogen absorption in pure flowing liquid Li showed retention at 0.1-0.2% D concentration in exposed Li samples.

Significance and Impact

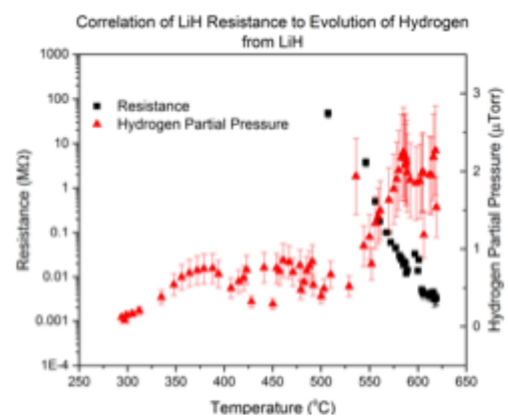
- The temperature range for which thermal distillation to recycle absorbed fuel is possible was established from the preliminary results.





(**Top**) A block diagram illustrating the full inner workings of the TDS chamber used to measure hydrogen degassing rates from LiH. The Odyssey QMS, circled in blue, is offset from the heating stage, circled in green, by 52 cm. A sniffer tube, circled in red, extends the line-of-sight visibility of the Odyssey down to 6.5 cm above the heating

(**Bottom**) An actual image of the real TDS system.



Correlation of LiH resistance vs temperature

D. Ruzic et al