INL/MIS-24-77135-Revision-0

Metallic fuel PIE at INL: from harvesting legacy materials to ATR/TREAT experiments

July 2024

Changing the World's Energy Future

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Idaho National Laboratory

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Metallic fuel PIE at INL: from harvesting legacy materials to ATR/TREAT experiments

Luca Capriotti

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Facilities: *HFEF, IMCL, EML*
Idaho National Laboratory

JAEA/INL CNWG Advanced Fuels Technical Experts, Idaho falls, March 11 - 13, 2024

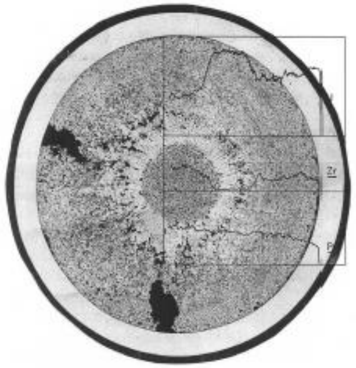
Advanced Fuels Campaign

U.S. DEPARTMENT OF ENERGY | Office of NUCLEAR ENERGY

Key Features & Benefits of Metallic Fuels

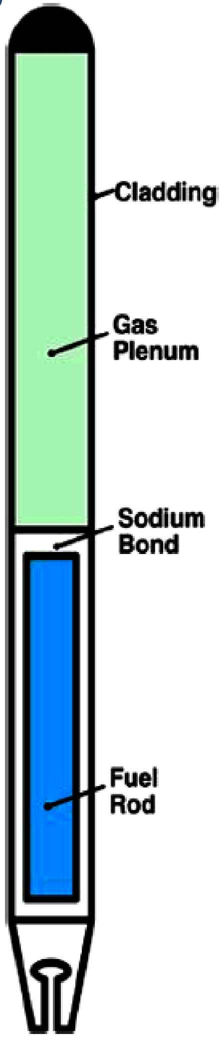
- Historic benefits
  - Higher breeding ratio (fissile and fertile)
  - Benign response to accident condition
  - Hard neutronic spectrum
  - Outstanding fuel reliability to high burnup (~20 at.%)
  - Compatibility with proliferation-resistant electrochemical recycle
  - Simple, compact (demonstrated remote) fabrication processes
  - Synergistic with passive approach to reactor safety

- Metal fuel characteristics
  - U-Zr/U-Pu-Zr alloy base (good irradiation stability)
  - 75% smeared density (accommodate fuel swelling, mitigate FCMI)
  - Large fission gas plenum (accommodate high gas release)
  - Na bond in fuel-cladding gap (keep fuel temperatures low)
  - Low-swelling FMS cladding (minimize cladding/duct dimensional changes)



G.L Hofman and L. C. Walters, "Metallic Fast Reactor Fuels," Materials Science and Technology Vol. 10A, 1994.

Schematic of a metallic, Na bonded, fast reactor element
Carmack et al., J. Nucl. Mater. 392 (2009)



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AFC Metallic Fuel R&D Priorities – over the years and decades

2003-2017

- Transmutation metallic fuel alloys (non and fertile)
- Comparison cases between ATR vs EBR-II vs Phenix
  - AFC-1 & -2 series
  - FUTURIX-FTA

2012-2020

- Innovative metallic fuel alloys testing
  - AFC-3 & 4 series

2019-up to now

- Qualification case for U-10Zr / HT9, fill in primary gaps (FCCI)
- Re-established transient testing and PIE capabilities
- Accelerating Fuel Qualification (AFQ)

Through the years:

- Enlarge database of EBR-II fuel PIE (many experiments were left without PIE examination) & (limited) on MFF experiments
- Advanced characterization with state-of-the-art instruments – comparison of old vs new data
- V&V for performance codes

(Synergistic and complementary efforts from other programs such as NSUF, NEUP, VTR, and more.)

PIE Strategy & techniques

Moving towards a mechanistic understanding of nuclear fuel performance

Scale of Exam

Macro → <1 µm

TOOLS

Visual Exams	Neutron Radiography Dimensional Exams	Gamma Spectrometry Fission Gas Release Optical Microscopy Chemistry Analysis	Electron Microscopy SEM, TEM, EPMA Focused Ion Beam Atom Probe

Fuel Performance

Did it Fail Catastrophically	How much did the geometry change	Fission Product Migration Microstructure, Restructuring, FCCI Burnup, Actinide Balance	Elemental distribution 3D microstructure

Number of Samples

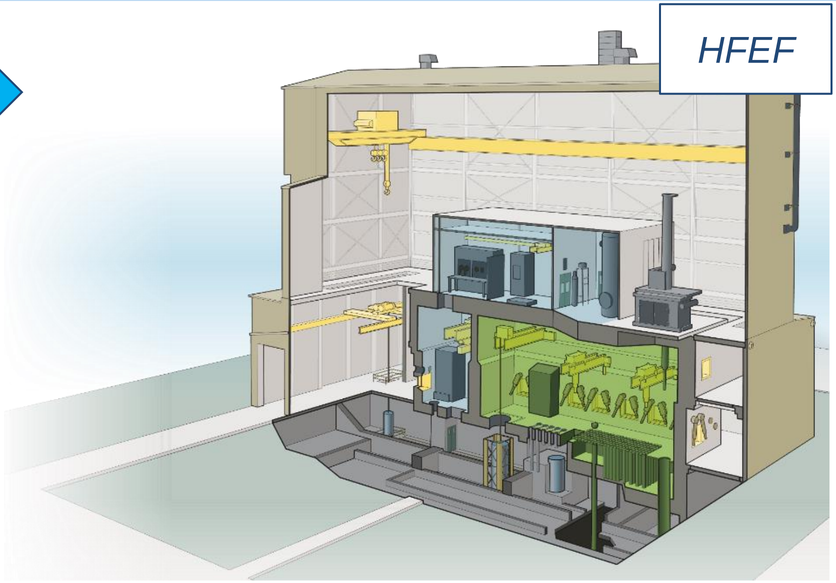
A few pins

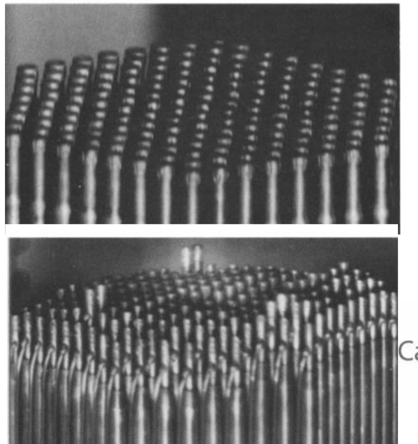
The historic approach

Section pins

Select samples to EM

FIB Sample Preparation





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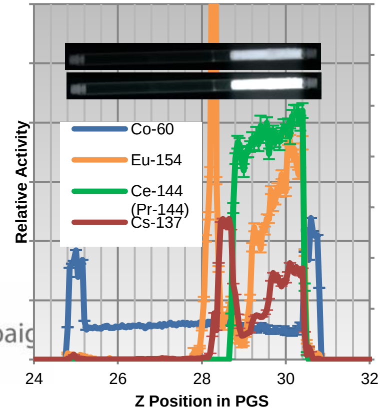
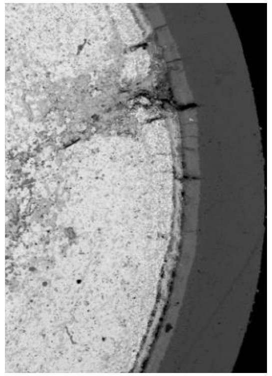
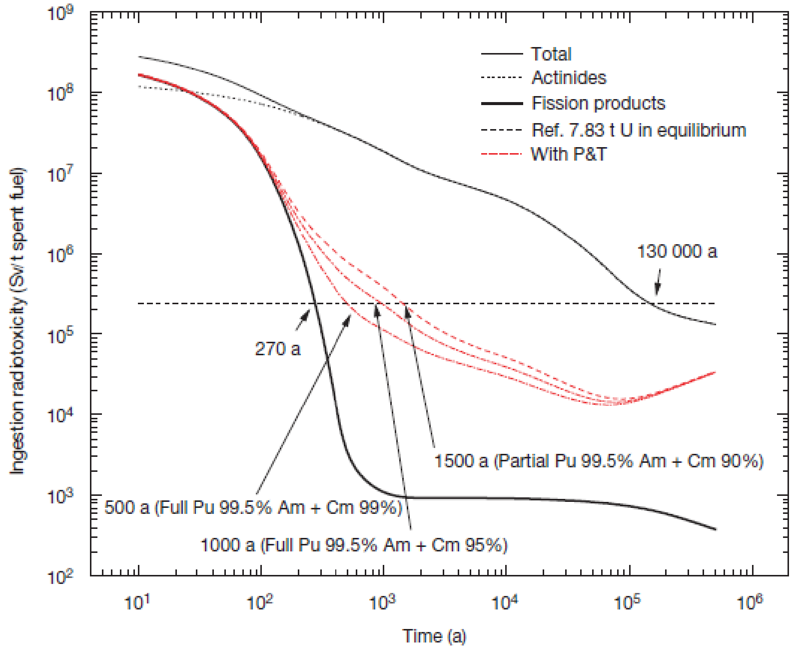




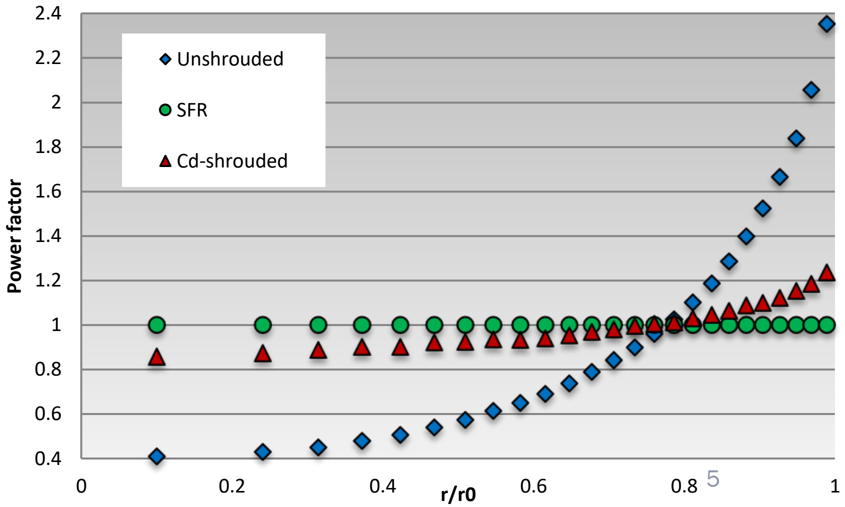
HFEF: Hot Fuel Examination Facility
IMCL: Irradiated Material Characterization Lab

Transmutation metallic fuel alloys and spectrum comparison

- **Transmutation metallic fuel for fast reactor**
  - LWR fuel takes 130,000 years to reach the radiotoxicity of natural U
  - If minor actinides take out of the waste and recycle them in a fast reactor radiotoxicity reaches natural U ~10,000 years
  - Why metallic fuels: Pyroprocessing, Ease of hot-cell fabrication



- **Case study to validate Cd shroud irradiation in ATR**
  - Comparison between true fast reactor spectrum vs ATR. Proper temperature radial profile is possible to create.
  - Fuel performance phenomena primary dependent on temperature are possible to compare and study
  - Spectrum Comparison report, INL/EXT-17-41677



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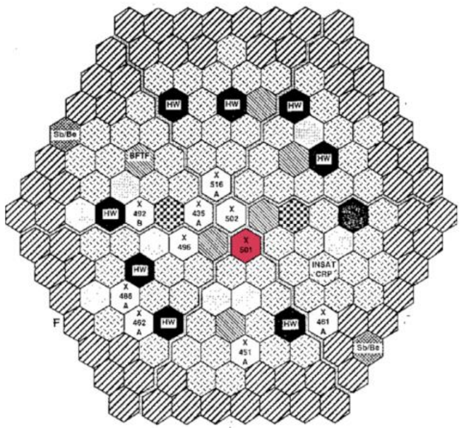
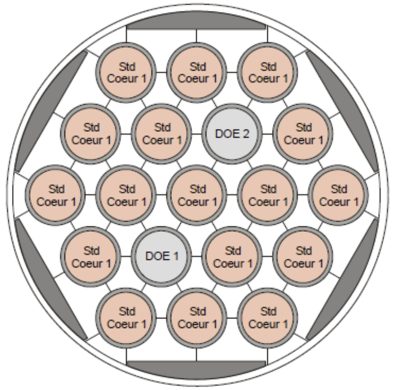
Transmutation experiments

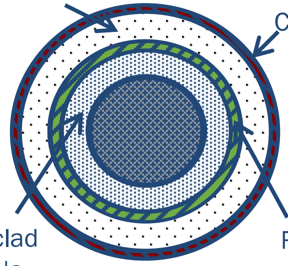
- Several minor actinide bearing metallic fuel irradiations have been performed to understand the effect of MA on fuel performance
  - X501, EBR-II: injection casting, Am volatility.
  - FUTURIX-FTA, Phénix: arc melting.
  - AFC-1, ATR: Cd shrouded position / double encapsulation, arc melting.

Experiment	Fuel / clad composition*	Fission density (f/cm3)	Burnup (at.% HM)	Peak temp cladding (°C)
X501	U-20Pu-10Zr-2.1Am-1.3Np / HT-9	2.1 x 10^{21}	6.1	~ 540
FUTURIX-FTA/DOE1	U-28Pu-31Zr-3.8Am-2.1Np / AIM1	2.08 x 10^{21}	9.1	~ 550
AFC-1H	U-28Pu-31Zr-3.8Am-2.1Np / HT-9	3.91 x 10^{21}	26.68	~ 495

* Non fertile metallic alloys exist as well as part of FUTURIX and AFC-1. Nitride fertile and non fertile alloy were part of FUTURIX and AFC-1 experiments campaign







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Status PIE performed

Examination	X501	FUTURIX-DOE1	AFC-1H	FUTURIX-DOE2 (non fertile)	AFC-1 (no fertile)
Visual	X	X	X	X	X
Neutron radiography	X	X	X	X	X
Profilometry	X	X	X	X	X
Gamma scan	X	X	X	X	X
Fission gas analysis	X	X	X	X	X
Sectioning	X	X	X	X	X
Chemistry / burnup	X	X	X	X	X
Metallography	X	X	X	X	X
SEM	X	X	X	Not planned	Not planned
EPMA	Not planned	X	X	Not planned	Not planned
FIB/TEM	X	X	Not planned	X	X

- Neutron Radiography
  - Low axial growth
  - Homogenous fuel density
- Gamma Spectrometry
  - Ru-106 flat profile (in X501 all decayed)
  - Cs-137 dissolved in Na and migrate in the plenum
  - Mn-54/Co-60 signal cladding endcaps
  - Eu-154 fission product in fuel and migration to the plenum
- Fission gas release (Xe+Kr)
  - AFC-1 /1H fission gas release corresponds well with EBR-II tests.
  - FUTURIX-FTA & X501 fission gas release corresponds well to EBRII tests as well.
- He release
  - X501-G591, 91%.
  - FUTURIX-FTA pins and AFC-1 rodlets: lower value of 60%.

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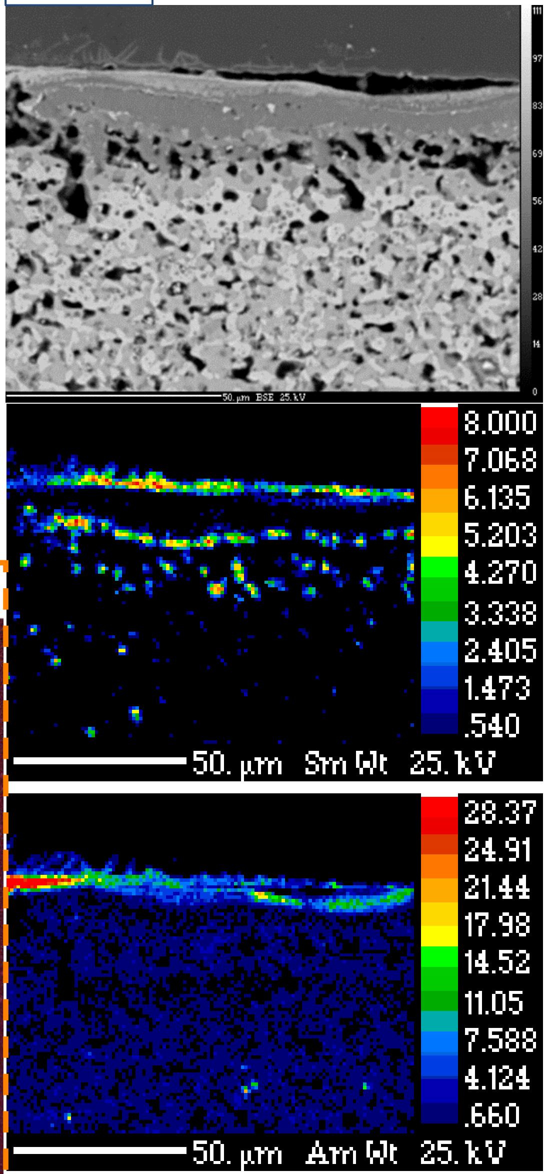
NSUF Nuclear Science User Facilities

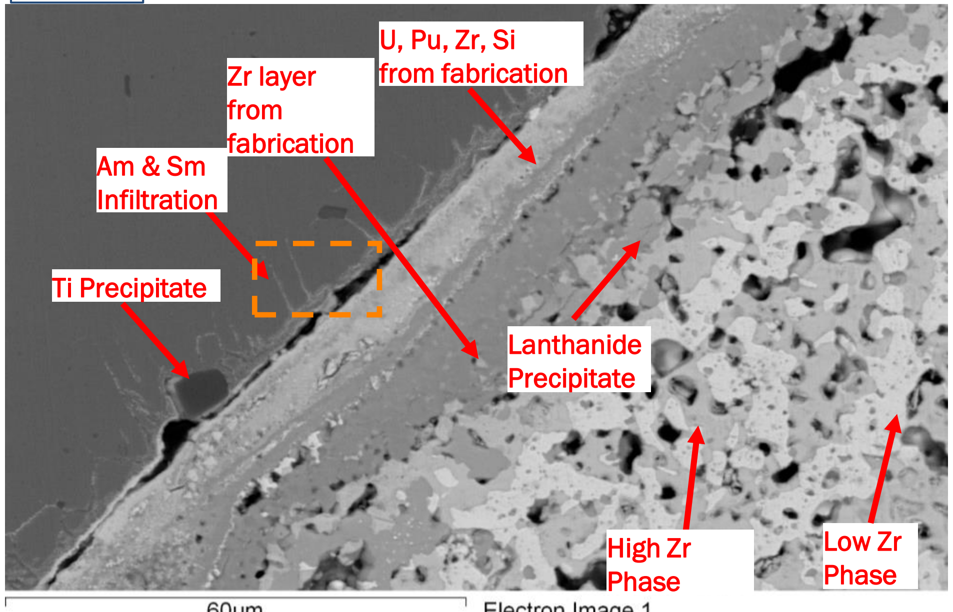
FCCI DOE1 – correlative microscopy

- U, Pu, Zr, Si, and possibly some Am layer from quartz mold
  - Observed after fabrication
- The Zr layer from fabrication
- Lanthanide precipitates (La, Ce, Nd) near the Zr layer.
  - No cladding attack from the major lanthanides (La, Ce, Nd).
- Infiltration of Am and Sm into the cladding.
  - high vapor pressure assisted phenomenon

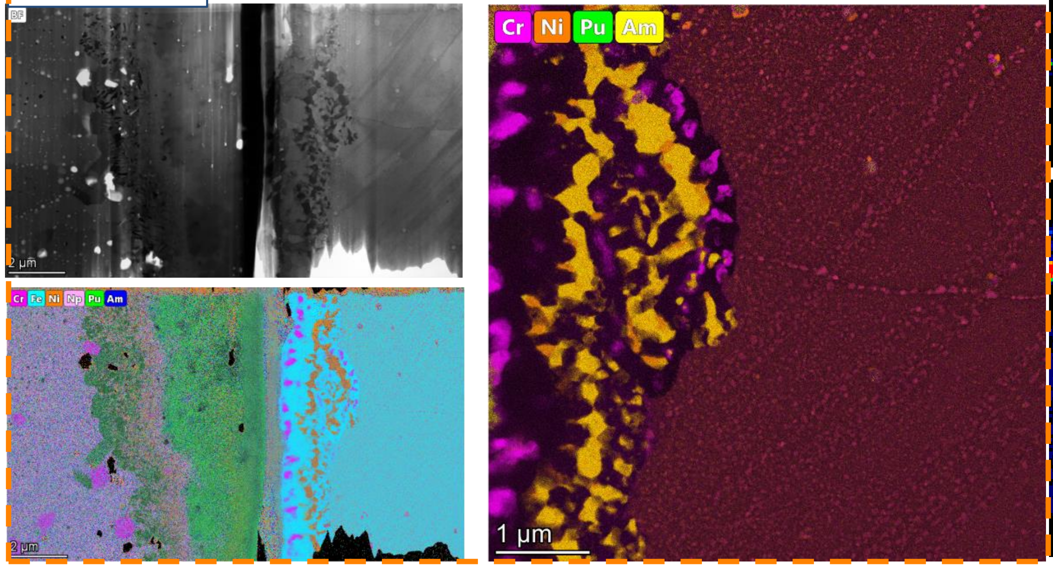
EPMA



SEM

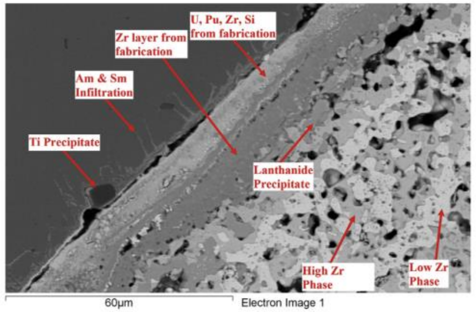


STEM EDS

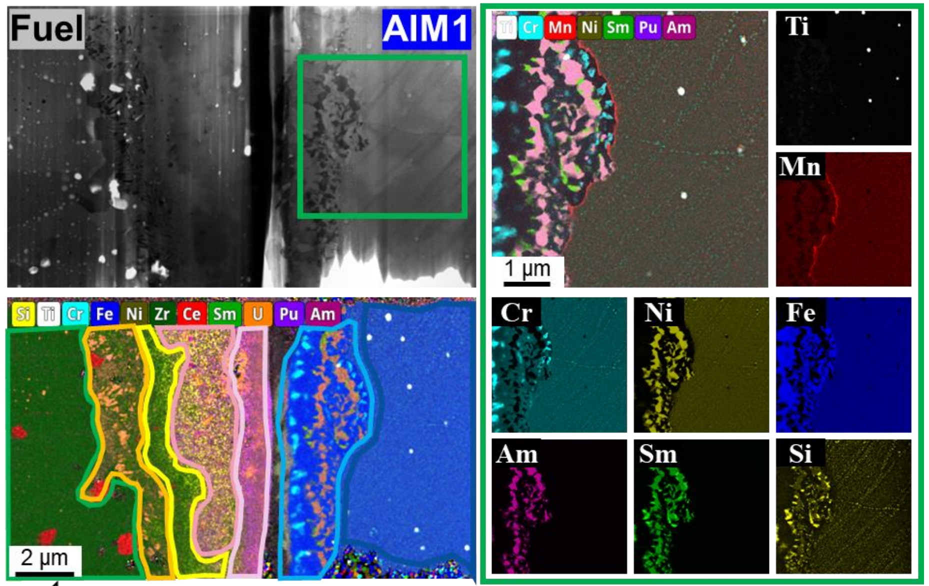


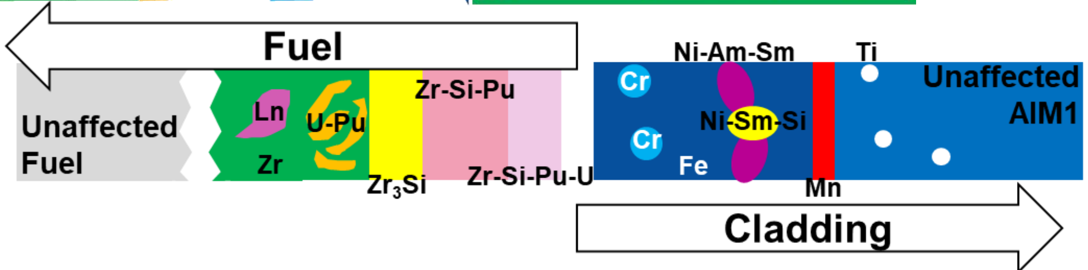
FCCI - TEM

FUTURIX-FTA-DOE-1

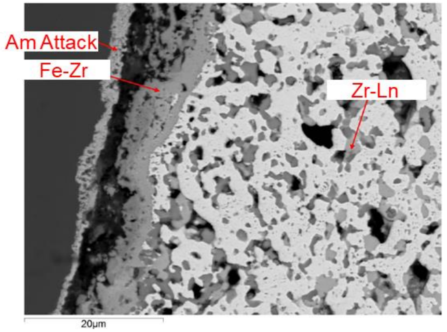


- **Fuel-side:** α-Zr, U-Pu precipitation, Zr_3Si, Zr-Si-Pu-U, no Fe penetration
- **Cladding-side:** **2-3 μm thick** interaction:
  - **Sm**, **Am** penetration into cladding (no Nd/Ce, little Pu/Np)
  - Interaction made of Fe rich layer, and Cr, Ni-Am-Sm, and Ni-Sm-Si rich precipitates

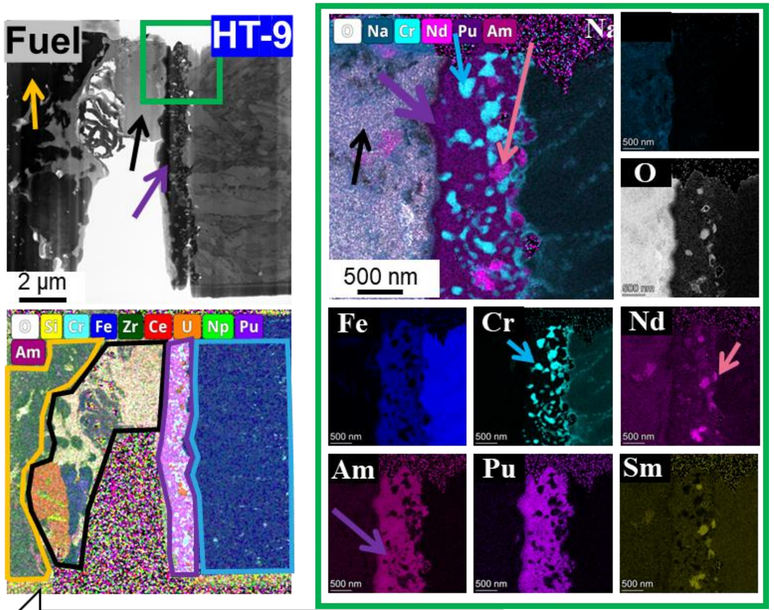


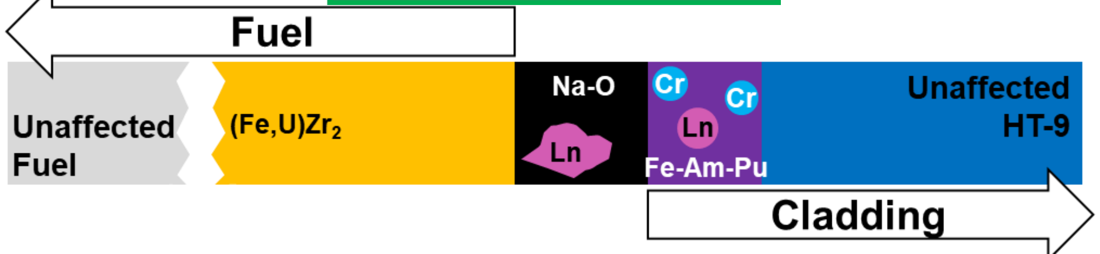


X501-G591



- **Fuel-side:** Fe penetration into Fuel → $FeZr_2$ phase
- **At interface:** Sodium, Oxygen and fission products.
- **Cladding-side:** **1 μm thick** interaction:
  - **Am, Pu** penetration into cladding (little Np, no U)
  - Interaction made of Fe-Am-Pu rich layer, Cr rich, and Lanthanide (Nd, Ce, La, Sm) rich precipitates.





Innovative metallic fuel concepts testing

- Historical Fuel Performance Issues
  - Swelling - limited burnup to 3 at. %, Solved with lowering smeared Density to 75% to allow for interconnected porosity releasing fission gas, solid fission product build-up limits fuel to 15-20 at.% burnup
  - Alloying elements to raise the fuel melting temperature and tailor the phase of U or U+Pu in the fuel (Zr, Fs, Mo, Ti)
  - Fuel Cladding Chemical Interaction (FCCI)
    - FCCI occurs at nominal operating conditions in U and U-Mo fuels and limits burnup to 10at. % (U-Fe, U-Ni interaction typically)
    - FCCI occurs at nominal operation conditions in U-Zr and U-Pu-Zr fuels beyond 10at.% burnup (Lanthanide – Fe interaction typically)
  - Fuel Constituent Redistribution – an effect of phase transitions
    - U, U-5Fs, and U-10Mo do not redistribute
    - U-10Zr does redistributes where Zr migrates to the center of the fuel
    - U-Pu-10Zr redistributes with Zr migrating to the central region and the periphery

New concepts

Annular / low smear density

New alloys

Additives (and liners)

New alloys

Fs – 49.8Mo-38Ru-6Rh-4Pd-2Zr-0.2Nb

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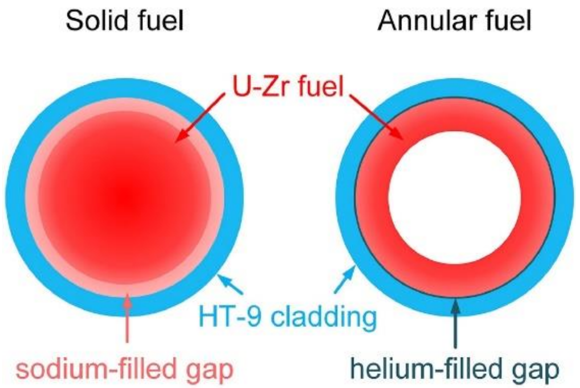
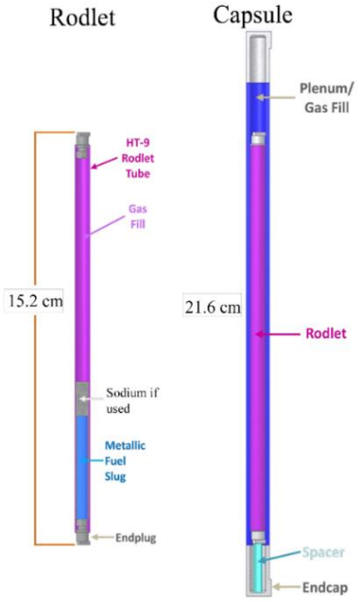


Advanced fuel alloys experiments in ATR

- AFC-3 and -4 are alloys exploration tests
  - Alternate alloys and forms to U-10Zr: U-10Mo
  - Pd additive to mitigate FCCI
  - Liner, e.g. Cr
  - Annular forms to eliminate Na treatment issues (He bonded)
  - Lower smear density
- Irradiation in ATR from 2-4 to ~10-12 at.% burnup
  - Rodlet – capsule – Cd basket system: proper temperature profile is created in ATR irradiations
  - This allows for the study of fuel performance phenomena that are primarily dependent upon temp. / temp. gradient
  - AFC-3C had irradiation cladding temperature exceeded 600°C for 3 or 4 rodlets.
  - AFC-3D had reasonable PICT temperatures and appears to have better performance
- Irradiation Issues with “early” annual fuel exp.
  - Capsule fabrication 3A/B
  - Reactor power uncertainty



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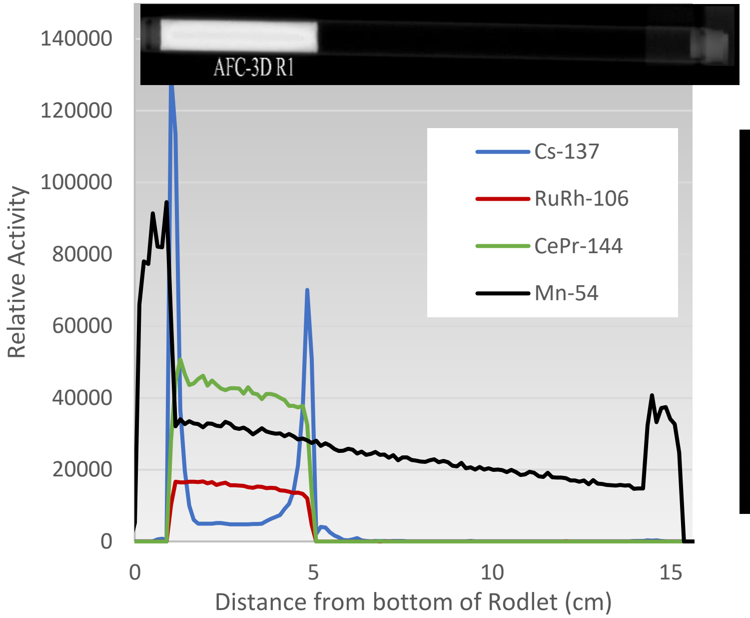
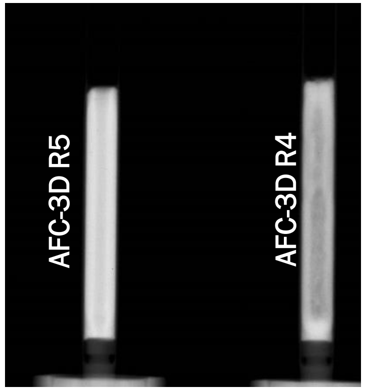
Rodlet ID	Alloy	Fuel Form	Bond Material	Nominal Smear Density
3A-R1	U-10Mo	Solid	Sodium	75%
3A-R2	U-10Mo	Annular	Helium	55%
3A-R4	U-10Zr	Annular	Helium	55%
3A-R5A	U-1Pd-10Zr	Solid	Sodium	75%
3A-R5B	U-2Pd-10Zr	Solid	Sodium	75%
3B-R1	U-4Pd-10Zr	Solid	Sodium	55%
3B-R2	U-4Pd-10Zr	Annular	Helium	55%
3B-R4	U-10Mo	Solid	Sodium	55%
3B-R5	U-10Mo	Solid	Sodium	55%

Rodlet ID	Alloy	Fuel Form	Bond Material	Nominal Smear Density
3C-R1	U-10Mo	Solid	Sodium	75%
3C-R2	U-10Mo	Annular	Helium	55%
3C-R3	U-10Zr	Sodium	Solid	65%
3C-R4	U-10Zr	Annular	Helium	55%
3C-R5A	U-1Pd-13Zr	Solid	Sodium	75%
3C-R5B	U-2Pd-13Zr	Solid	Sodium	75%
3D-R1	U-10Zr	Annular	Helium	55%
3D-R2	U-4Pd-13Zr	Solid	Sodium	55%
3D-R3	U-10Mo	Solid	Sodium	55%
3D-R4	U-10Mo	Annular	Helium	55%
3D-R5	U-4Pd-13Zr	Annular	Helium	55%

AFC-3C/3D NDE PIE highlights

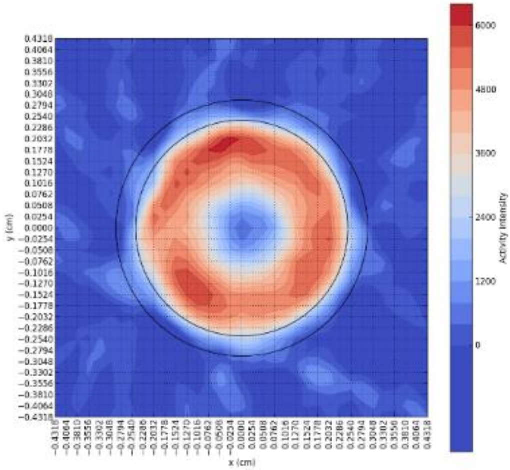
- Profilometry
  - AFC-3C presents some degree of diametrical strain (max 0.7% strain)
  - AFC-3D no measurable change in diameter
- Thermal neutron radiography
  - Annular fuel pins (e.g. 3D-R5) maintained their annuli
  - Visible change in grey scales shows variation in density
- Gamma spectrometry
  - RhRu-106 flat distribution
  - Annular fuel: Cs migrates towards cooler axial ends
  - Ce-144 some axial migration
  - Tomography: RhRu-106 location may indicate highest fission density
  - Tomography: RhRu-106 signal indicates annuli is still open (cf. metallography)





U-10Zr 55% SD, annular

RhRu-106



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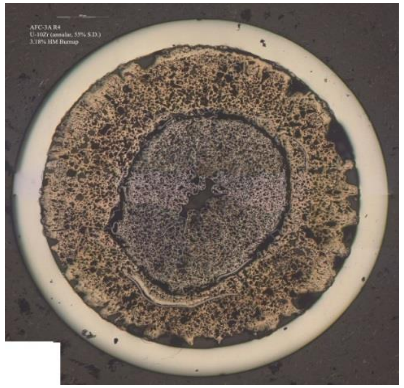
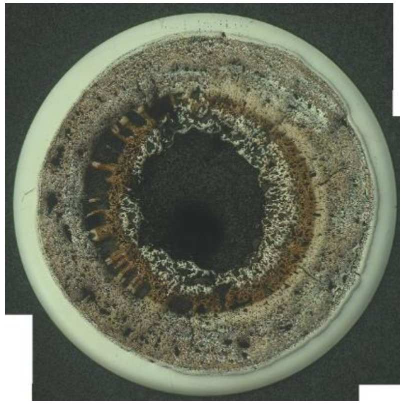
AFC-3C/3D PIE: metallography and SEM U-10Zr annular comparison

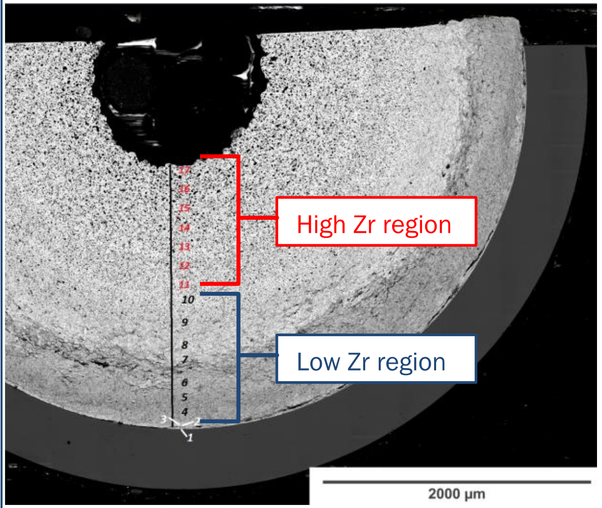
AFC-3A R4*



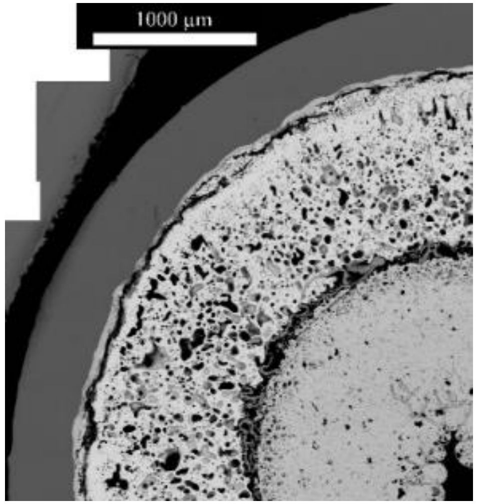
AFC-3C



AFC-3D



AFC-3D



AFC-3A R4*

- Metallography
  - U-10Zr annular: AFC-3D behaves better compared to -3A-3C
  - Machining, gap ~50µm AFC-3A R4 vs 17µm AFC-3C R4, temp 177°C vs 60°C
  - Periphery temperature maintained below the critical temperature for Zr migration (Beta phase)

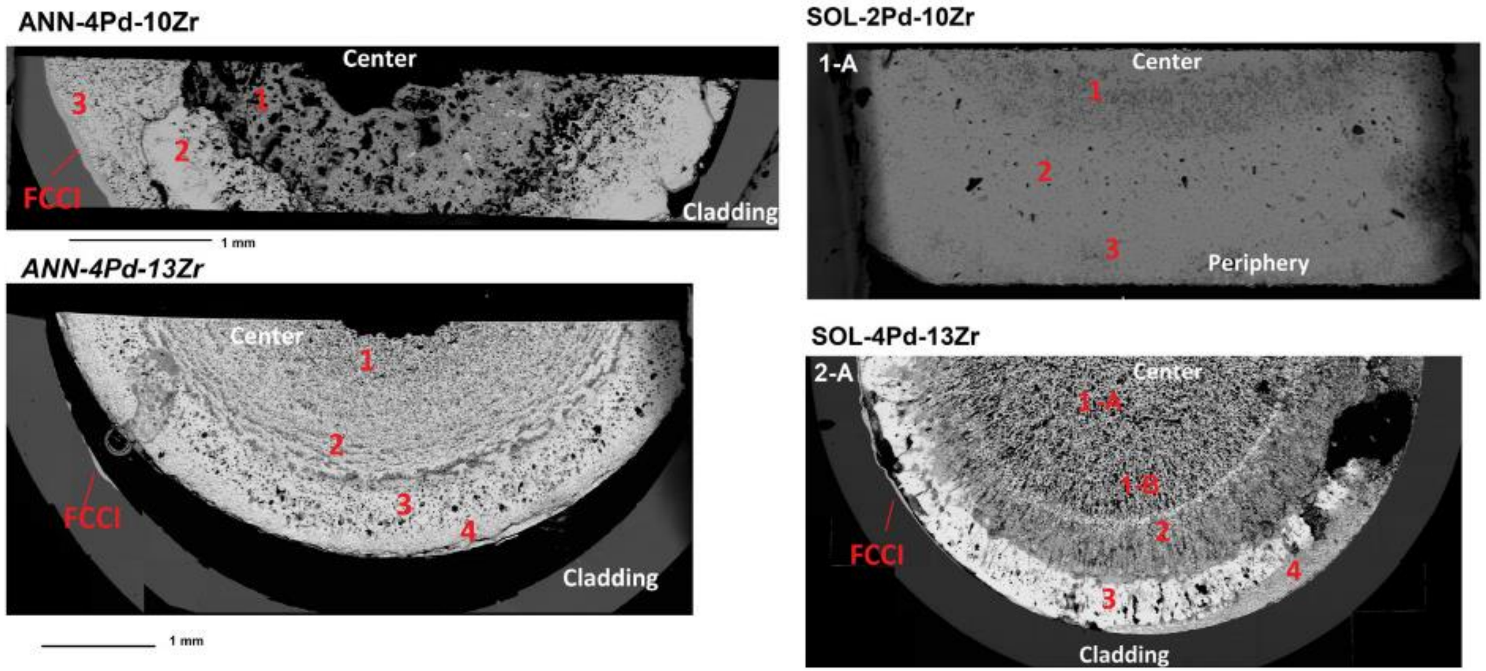
[*AFC-3A-3B: U-10Zr, bu 3.2 at.%, PICT 530 ° C / U-1/2Pd-10Zr, bu 2.5 at.%, PICT 585 ° C]

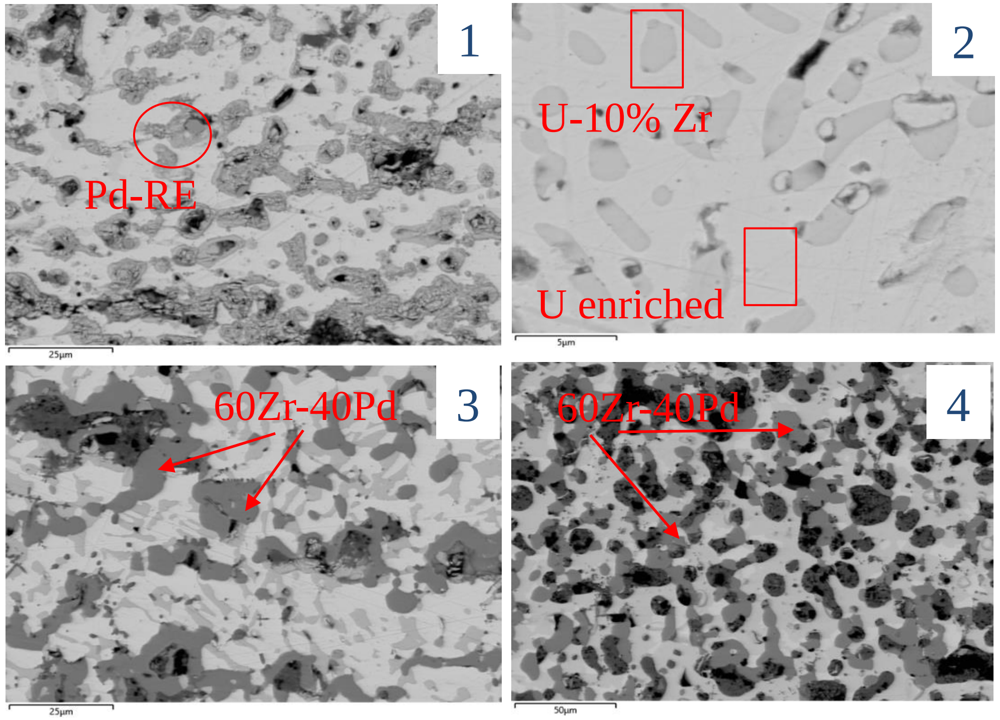
- SEM/EDS
  - AFC-3D-R1, annular U-Zr performed exceptionally well
  - Two contributing factors
    - Appropriate PICT always during irradiation
    - Well machined fuel slugs
  - No noticeable wastage inside the cladding
  - 2-region redistribution (U, Zr) as expected
  - Comparison with AFC-3A-R4: high FCCI (U-Fe interaction), higher PICT.

U-Zr with Pd additive

Sample Name	AFC ID	Composition(%wt.)	Fuel Form	BondElement	Nominal smeareddensity (%)
ANN-4Pd-10Zr	*AFC-3B-R2*	U-4Pd-10Zr	Annular	Helium	55
ANN-4Pd-13Zr	*AFC-3D-R5*	U-4Pd-13Zr	Annular	Helium	55
SOL-2Pd-10Zr	*AFC-3A-R5B*	U-2Pd-10Zr	Solid	Sodium	75
SOL-4Pd-13Zr	*AFC-3D-R2*	U-4Pd-13Zr	Solid	Sodium	55

- Pd has seen to preferentially bind to Ln and immobilize them
- The behavior of Zr and Pd are important for this fuel form to well perform
  - Zr needs to stayed alloyed with U for fuel performance
  - Zr and Pd intermetallics need to be avoided (e.g. PdZr2), so too much Pd – detrimental (ANN-4Pd-10Zr)
- Increase Zr to 13%wt. to compensate for Zr-Pd phase formation



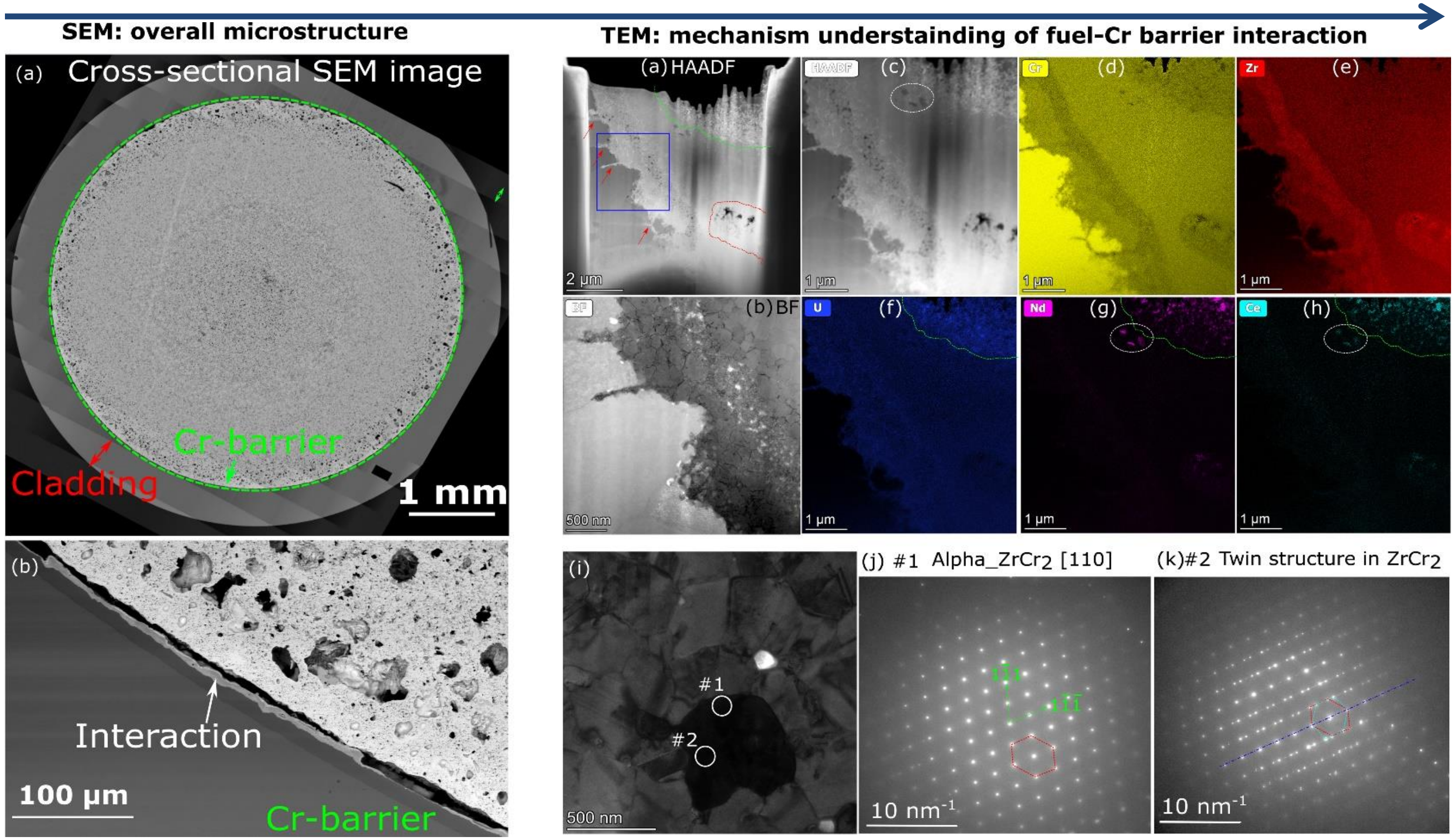


Di Lemma et al. Journal of Nuclear Materials (2021), 153403, https://doi.org/10.1016/j.jnucmat.2021.153403

ATR irradiated U-10Zr (solid) with Cr-coating

- In-depth post irradiation examination (PIE) was conducted on an ATR-irradiated HT-9 clad U-10Zr fuel with Cr barrier (provided by KAERI) to
  1) evaluate barrier irradiation performance
  2) characterize the potential interactions, microstructural and compositional changes in the Cr barrier
  3) evaluate the micromechanical properties
- Gained insight into the microstructural and compositional stability of the electroplated Cr diffusion barrier during in-reactor irradiation
- Shedding light on a promising solution to mitigate/prevent the FCCI in HT-9 clad U-10Zr fuels, benefiting the adoption of metallic fuel for SFRs at high or ultra-high burnup
- SEM examination reveals good Cr barrier integrity against FCCI between the fuel and the clad (left)
- TEM characterization provides mechanism understanding on the interaction ➔ interactive diffusion of Zr in the Cr barrier, forming α_$ZrCr_2$ intermetallic phases and consuming Cr barrier (right)

Micron to Nano Scale



AFC-4C-R5 selected sample ATR irradiated, 8.7% HM burnup, 540-550 °C PICT

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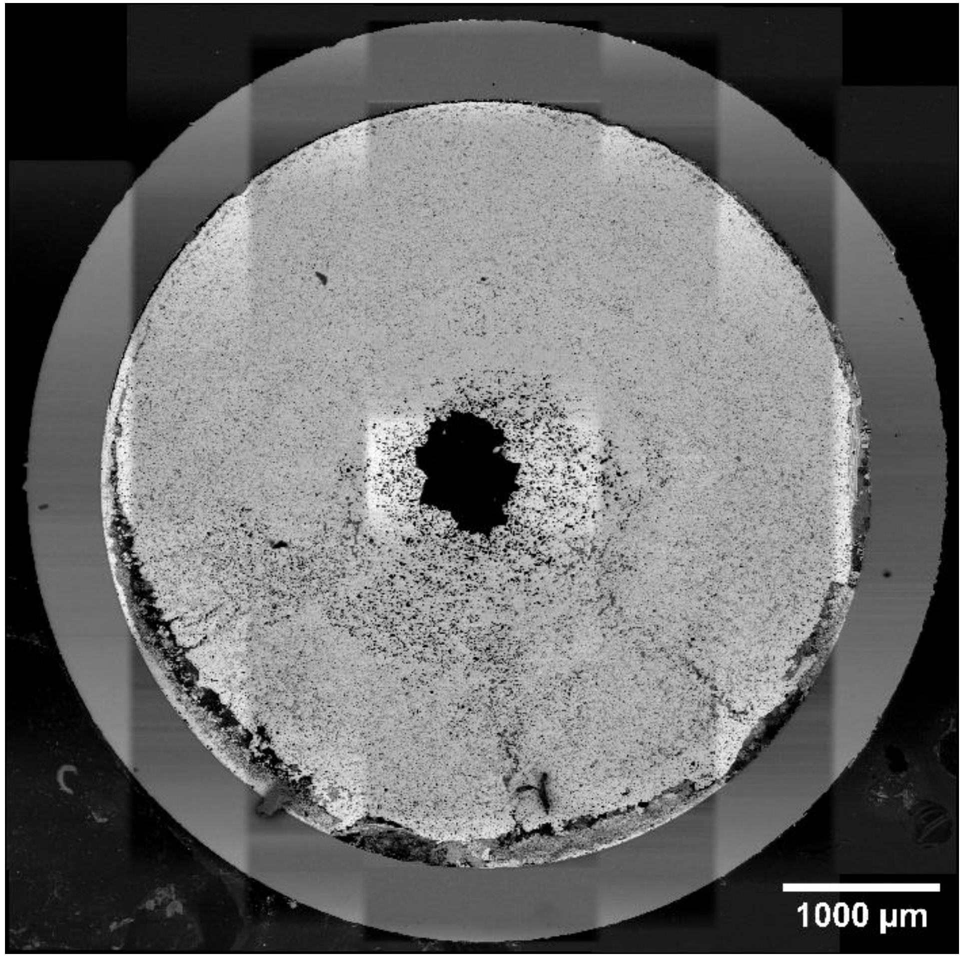
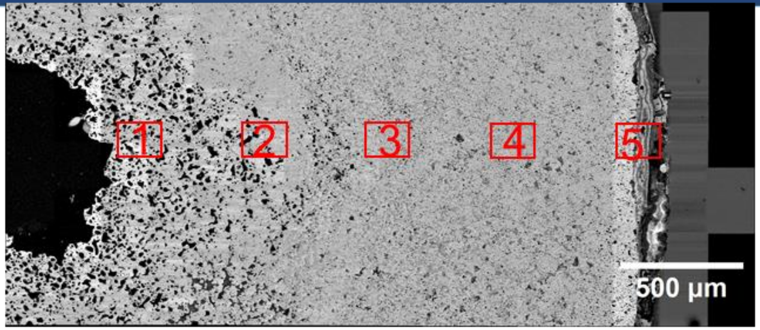
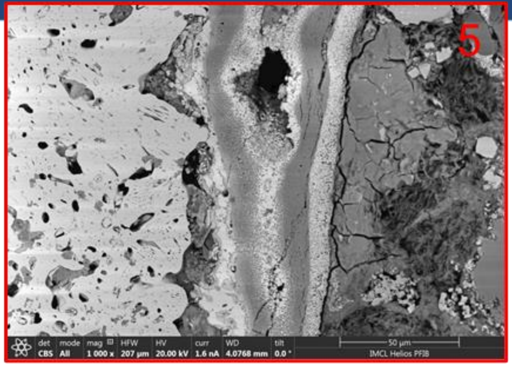
Wang et al., under review (almost accepted) JNM

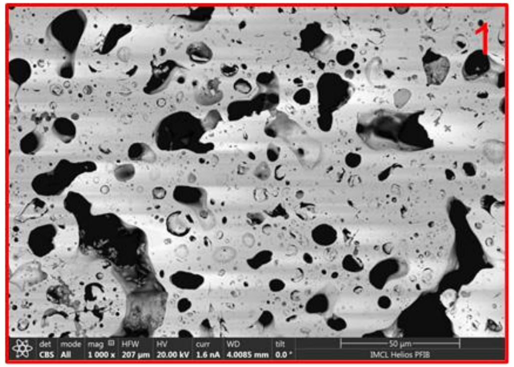
U-10Zr Annular fuel with Cr diffusion barrier

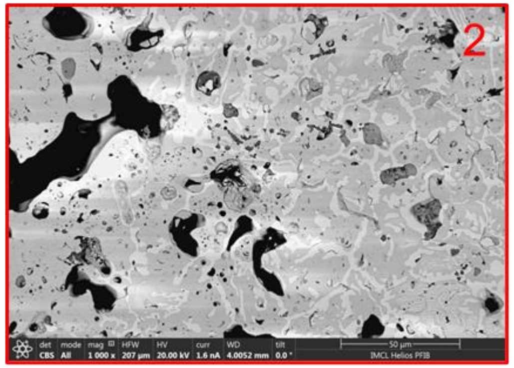
Rodlet ID	Comp %wt. / cladding / smear density	Fuel Form / Bond material	Burn up %FIMA	PICT (°C)
AFC-4C-R1	U-10Zr /HT9 + Cr / 55%	Annular / He	12	515

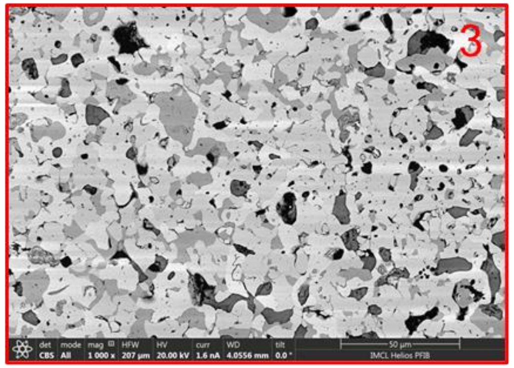


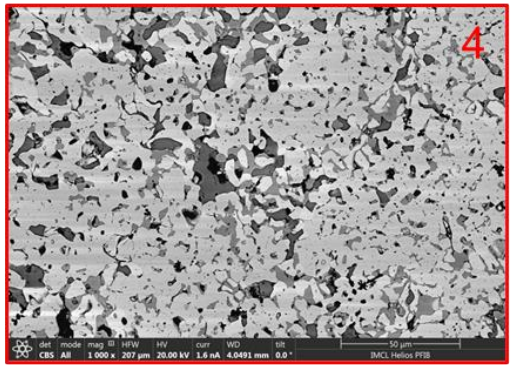




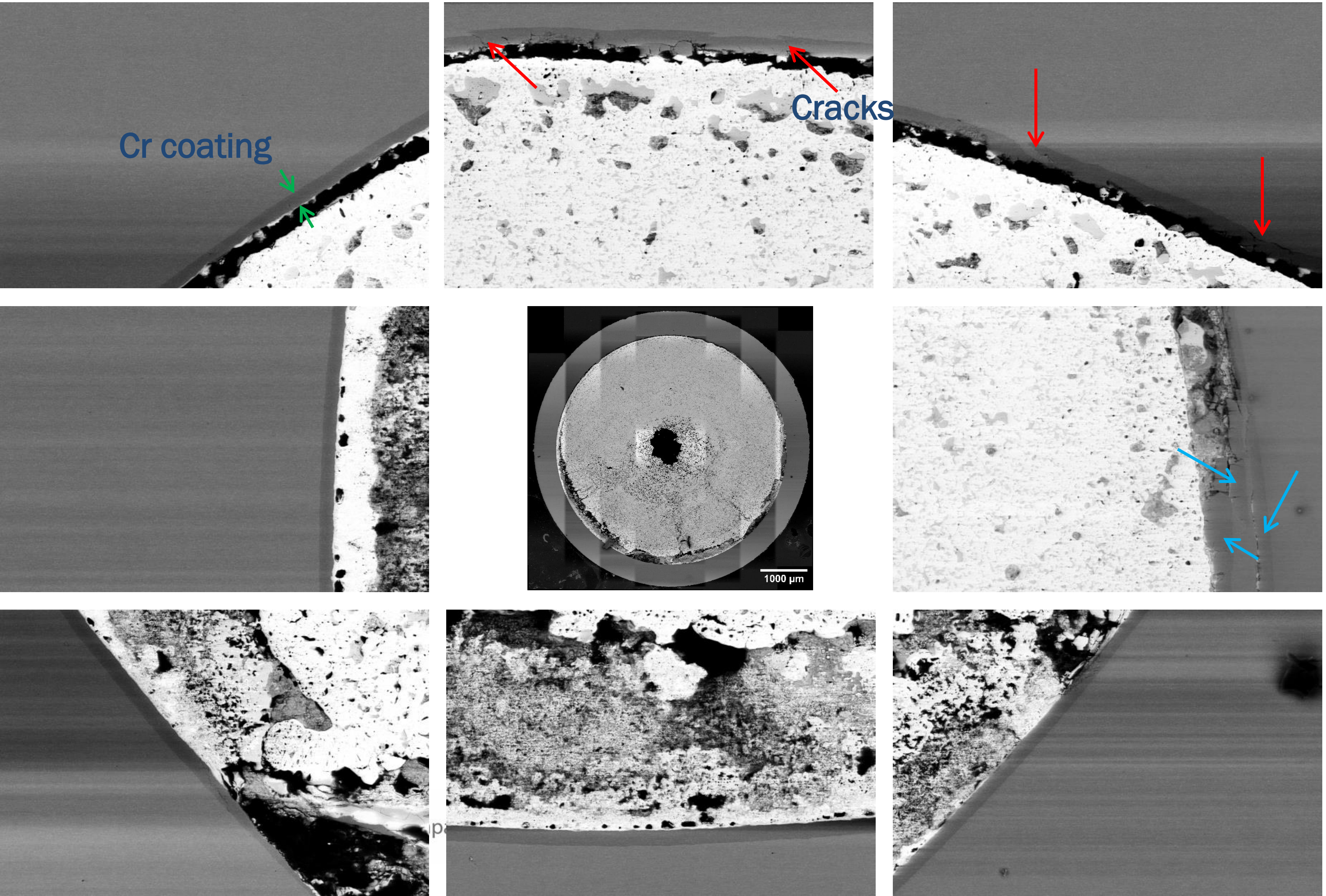








U-10Zr Annular fuel with Cr diffusion barrier



- Cr coating generally well adherent to the cladding.
- Local cracks, but not associated with FCCI
- Some rare and "weird" structures are present (double coating with HT-9 in between, see middle-right picture)
- Average thickness = 14.3 ± 1.5 µm based on 21 manual measurements across interface

Qualification U-10Zr, Transient capability and AFQ

- Qualification case for U-10Zr / HT9, fill in primary gaps
  - Fuel-cladding chemical interaction database
  - Fuel physical properties and improved models

Harvesting legacy materials from FFTF MFF experiments

- Transient fuel performance evaluation
  - *Re-establish capabilities and devices for transient test*
  - *Initiate THOR-C and THOR-M series*

Pre and Post characterization of pre-irradiated EBR-II pins

- Accelerating Fuel Qualification (AFQ)
  - Establishing methodology and framework
  - Fission Accelerate Steady State irradiation (FAST)
  - Initial assessment and PIE

Qualifying U-10Zr: Harvesting FFTF MFF legacy materials

Fuel-cladding chemical interaction database

Fuel physical properties and improved models

MFF Irradiation Test

BISON Simulations

Samples Selection

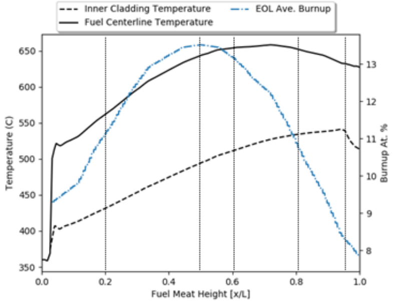
Pin Cutting and Cross-Section Preparation

SEM – BSE/EDS

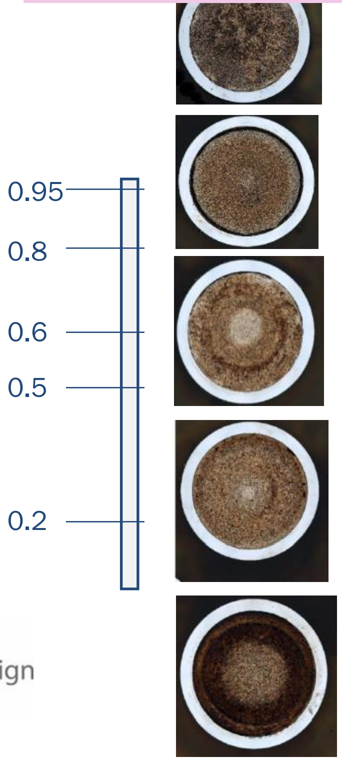
ML/AI development and application

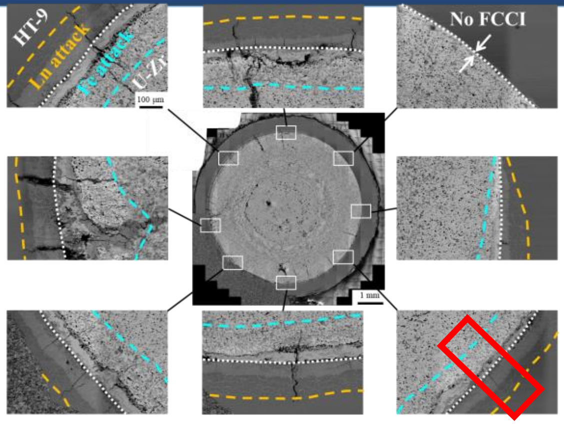
FCCI Characterization

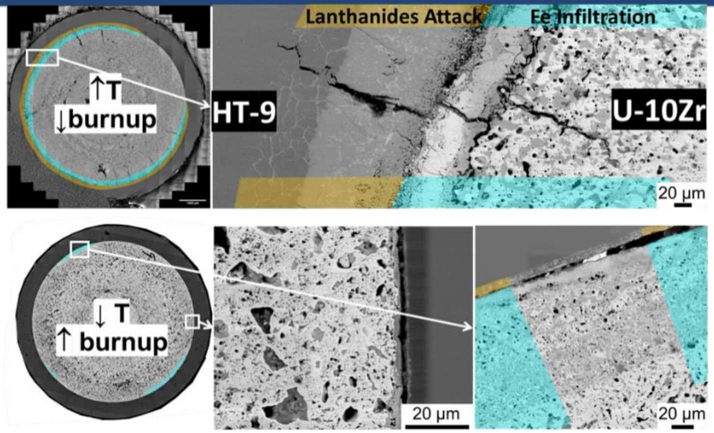
T, burnup

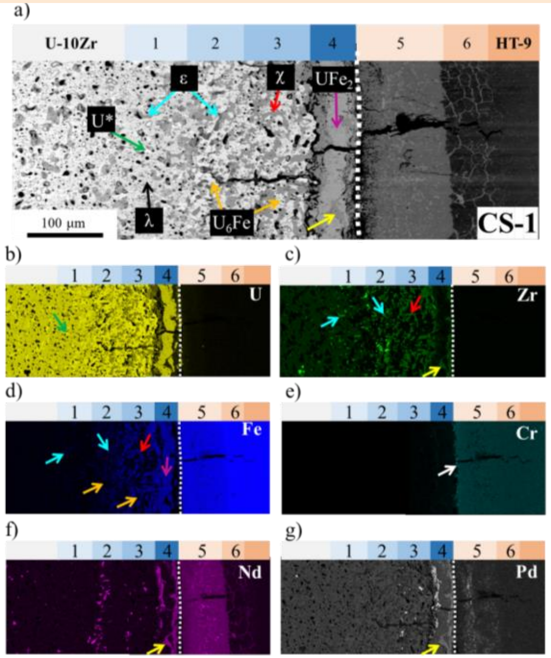


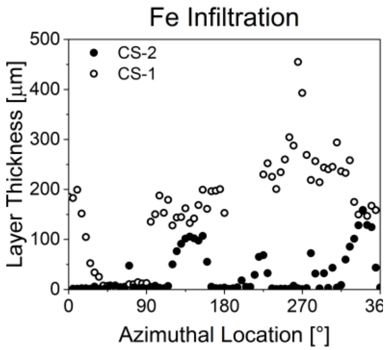
Pin 192167

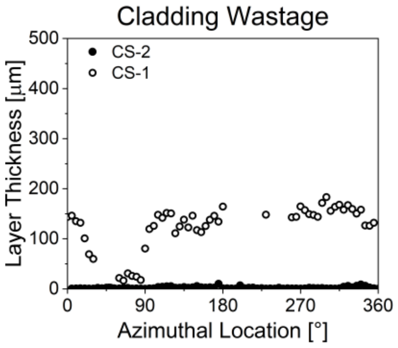








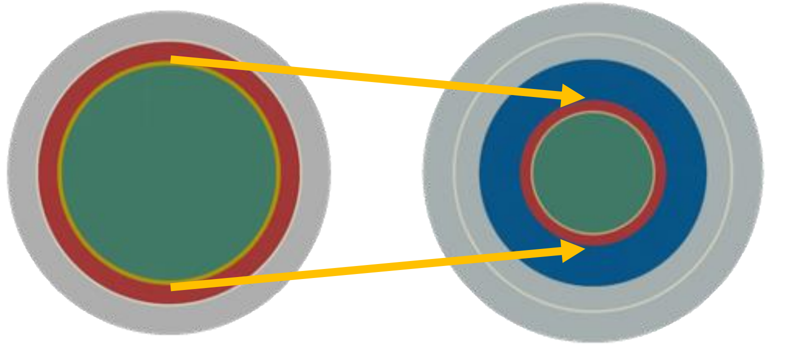




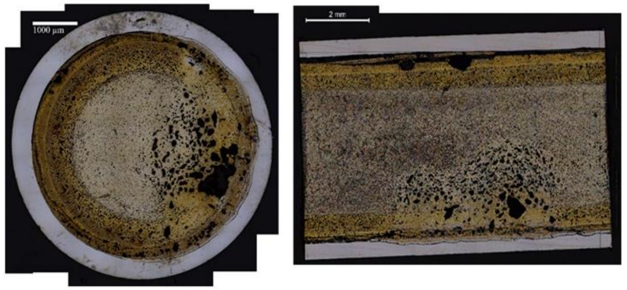
Advanced Fuels Campaign

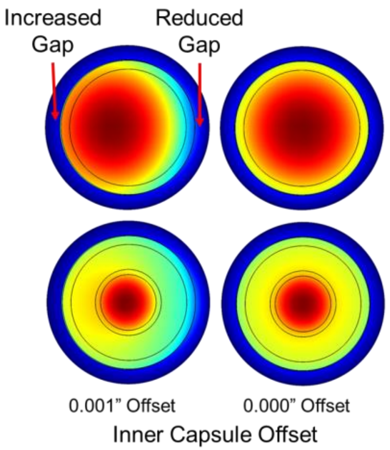
AFQ-FAST: A Revised Capsule Design

- Rekindling a small test performed in the 1960's, a FASTer approach to testing was developed
- The Fission Accelerated Steady-state Test (FAST) utilizes a reduced diameter fuel pin to achieve two objectives:



1. Improve experiment reliability: reduced sensitivity to fabrication tolerances and capsule/pin eccentricity





2. Increase burnup rate for fuel experiments: reduce time to achieve high burnup

Given

$$Q_0 = \frac{LHGR_0}{\pi r_0^2}$$

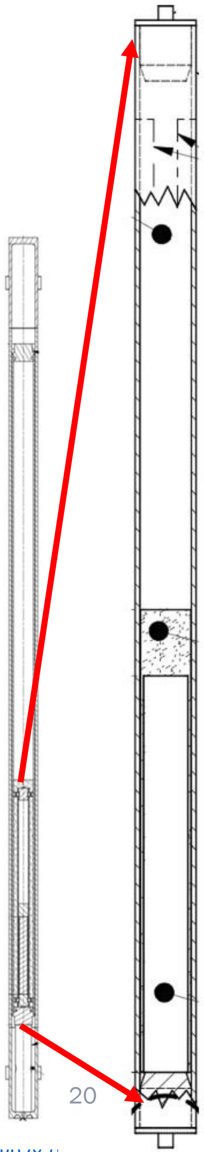
if $r = \alpha r_0$ and $LHGR = LHGR_0$, then

$$Q = \frac{Q_0}{\alpha^2}$$

For $\alpha = \frac{1}{2}$,

$$Q = 4Q_0$$

$$t \sim Q^{-1} \therefore t \sim \frac{t_0}{4}$$



Advanced Fuels Campaign

https://doi.org/10.1016/B978-0

FAST Metal Fuel Test Matrix

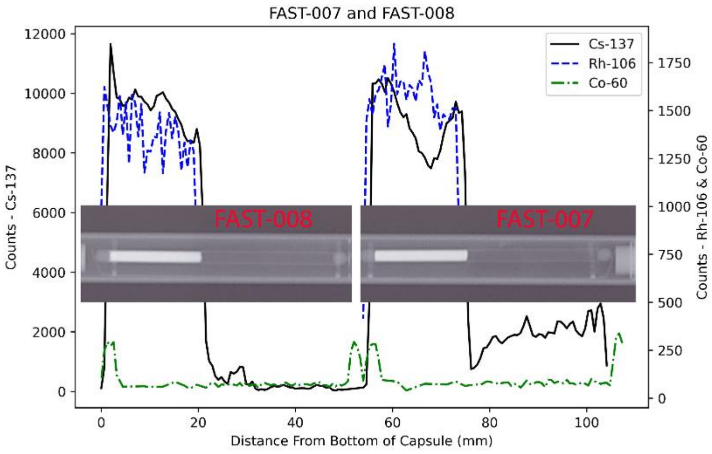
- Each capsule in the small-I positions contains a novel experiment and control experiment
  - Controls are solid, 75% SD U-10Zr in HT9
- Experiments include
  - He-bonded annular fuel
  - Additives: Pd, Sb, & Sn
  - Zr liners
- Engineering PIE completed for all low burnup pins (green)
- Recently received in HFEF and PIE started on 8-10 at% rodlets (yellow)

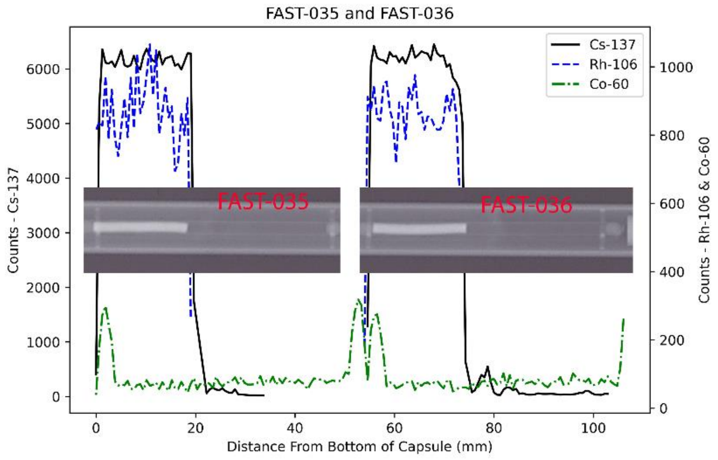
Capsule	Rodlet ID	Fuel Comp	Geometry	Bond	Liner	Target BU
AFC-FAST-016	FAST-035	U-10Zr	Solid	Na	-	2.0%
	FAST-036	U-10Zr	Solid	Na	-	2.0%
AFC-FAST-005	FAST-007	U-10Zr	Annular	He	-	4%
	FAST-008	U-10Zr	Solid	Na	-	4%
AFC-FAST-009	FAST-025	U-10Zr	Solid	Na	Zr	8%
	FAST-051	U-10Zr	Solid	Na	-	8%
AFC-FAST-006	FAST-015	U-10Zr	Annular	He	-	8%
	FAST-016	U-10Zr	Solid	Na	-	8%
AFC-FAST-014	FAST-039	U-10Zr	Solid	Na	-	10%
	FAST-040	U-3Pd-10Zr	Solid	Na	-	10%
AFC-FAST-013	FAST-031	U-10Zr	Solid	Na	-	10%
	FAST-032	U-3Sn-10Zr	Solid	Na	-	10%
AFC-FAST-015	FAST-045	U-10Zr	Solid	Na	-	10%
	FAST-046	U-3Sb-10Zr	Solid	Na	-	10%
AFC-FAST-003	FAST-003 (OA)	U-10Zr	Solid	Na	-	12%
AFC-FAST-010	FAST-026	U-10Zr	Solid	Na	Zr	12%
	FAST-052	U-10Zr	Solid	Na	-	12%
AFC-FAST-007	FAST-047	U-10Zr	Annular	He	-	12%
	FAST-048	U-10Zr	Solid	Na	-	12%
AFC-FAST-011	FAST-027	U-10Zr	Solid	Na	Zr	16%
	FAST-053	U-10Zr	Solid	Na	-	16%
AFC-FAST-008	FAST-049	U-10Zr	Annular	He	-	16%
	FAST-050	U-10Zr	Solid	Na	-	16%
AFC-FAST-012	FAST-028	U-10Zr	Solid	Na	Zr	20%
	FAST-054	U-10Zr	Solid	Na	-	20%

Advanced Fuels Campaign

Initial PIE Efforts

- NRAD: capsules do not show anything un-usual. Capsule 05 seems to have swell all the way to cladding compared to Capsule 16
  - Low temp rodlets have not contacted the cladding
  - High temp rodlets do not have annulus closed
- PGS: Capsule 16 (FAST 035 and 036) present a “early” life fission products behavior, Cs does not seem to have migrated yet to the plenum. Bottom rodlet a small peak is present (highlighted in green)





Rodlet	dL (%)
FAST-007	8.63
FAST-008	6.72
FAST-035	3.06
FAST-036	3.19
X425A-T423	10.3%
X425B-T424	10.2%

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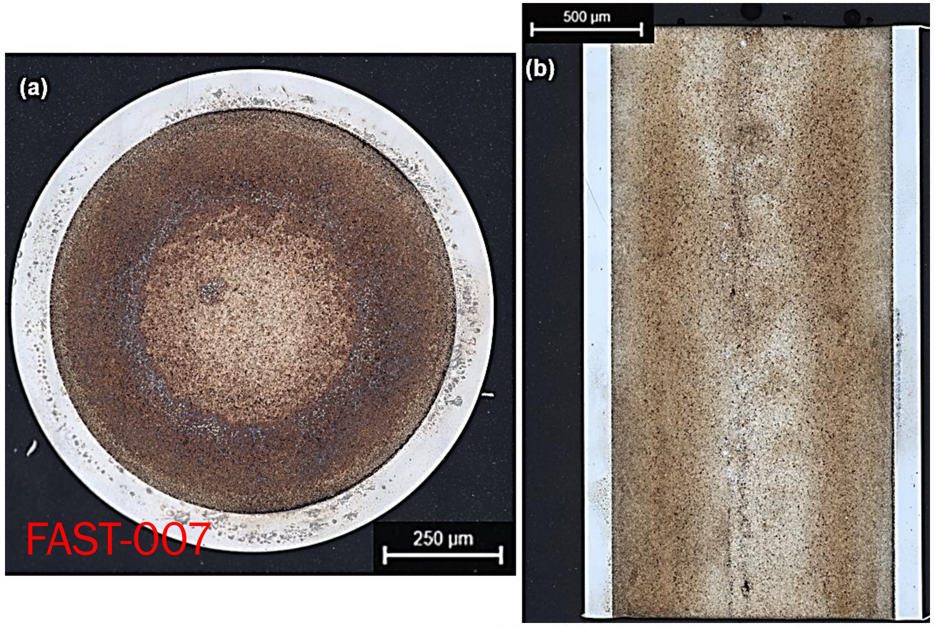
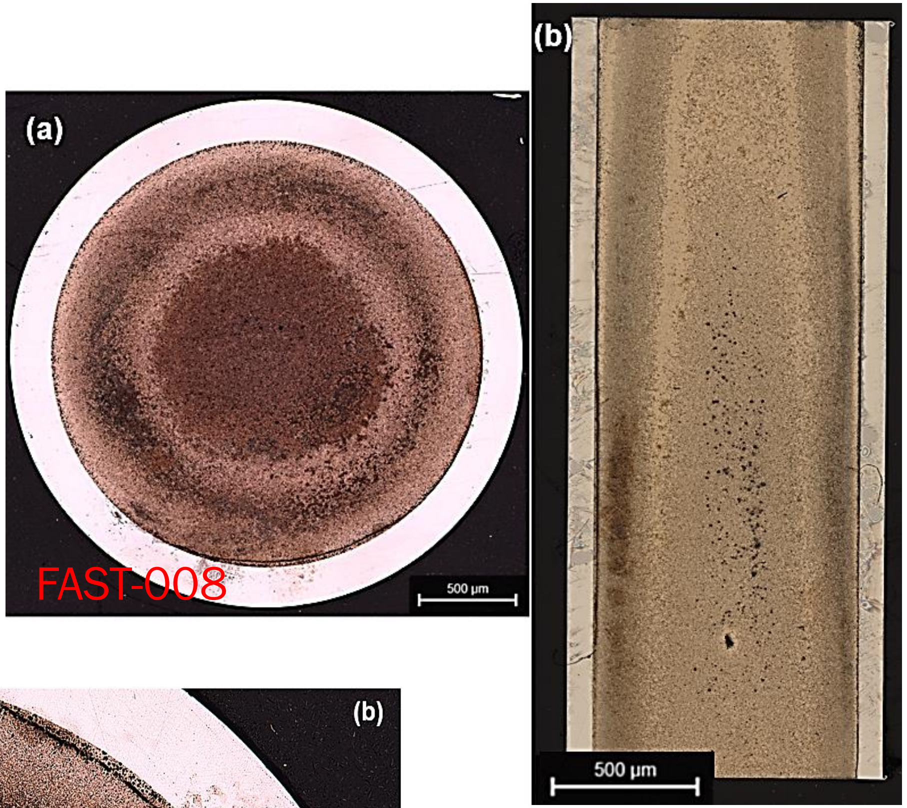
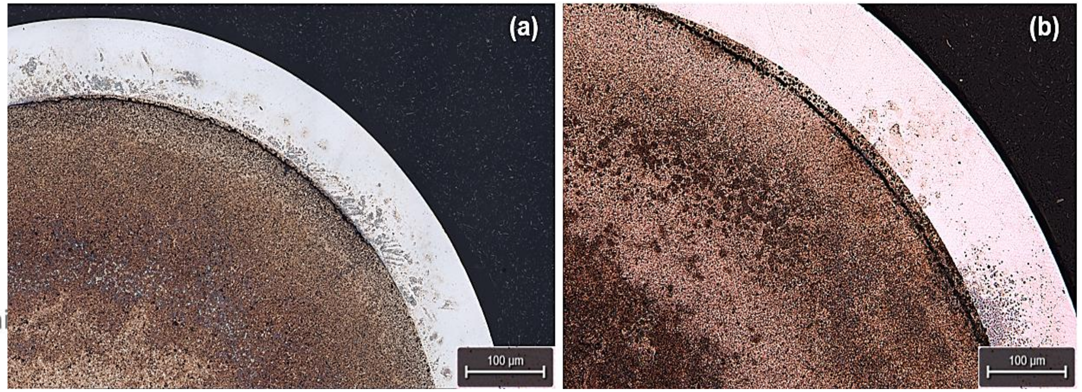
Metallography – FAST-007 & -008

- FAST-007: 75% SD, Annular, He-bonded, U-10Zr
- FAST-008: 75% SD, Solid, Na-bonded, U-10Zr







EBR-II PIE database, advanced chara. and V&V

- Legacy EBR-II metallic fuel pins selection and PIE status

Examination	X501-G591	X430 (multiple)	X521 (multiple)	X486-J555	X496 (multiple)	X441 (multiple)	X512 (multiple)	X429B (multiple)	X510 (multiple)
Visual	X	X	X	X	X	X	X	X	X
Neutron radiography	X	X	X	X	X	X	X	X	X
Profilometry	From old data	X	X	X	X	X	X	X	X
Gamma scan	X	X	X	X	X	X	X	X	X
Fission gas analysis	X	X	X	X	X	X	X	Planned	Several
Sectioning	X	X	X	X	X	X	X	Planned	Planned
Chemistry / burnup	X	X	X			X			Planned
Metallography	X	X	Partially	X	X	X	X	Planned	Planned
SEM	X					Several	1 sample		Planned
EPMA						Several	1 sample		
FIB/TEM	X					Several			

Advanced Fuels Campaign

NSUF Nuclear Science User Facilities

X512: Understanding top of the fuel microstructural evolution

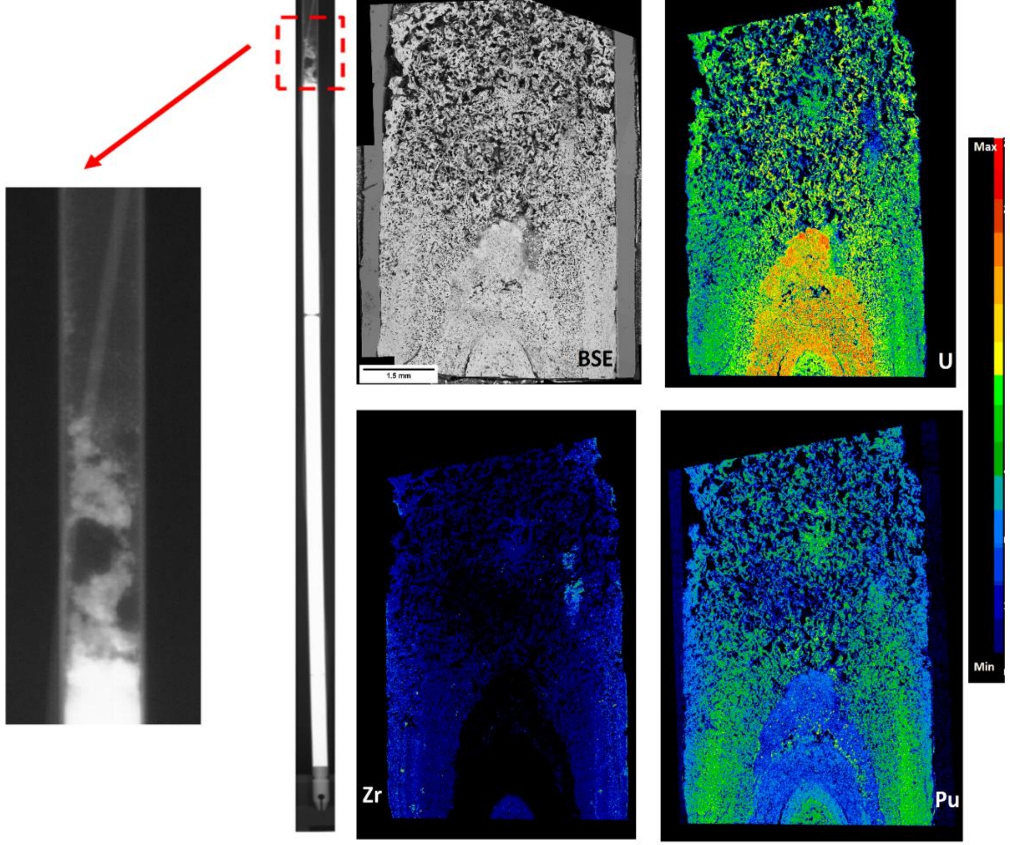
- X512 (OPT-1): U-Zr and U-Pu-Zr overpower test (32%) in EBR-II
  - U-19Pu-10Zr: Qualitatively the same microstructure
  - Middle axial height: 3 phases structure

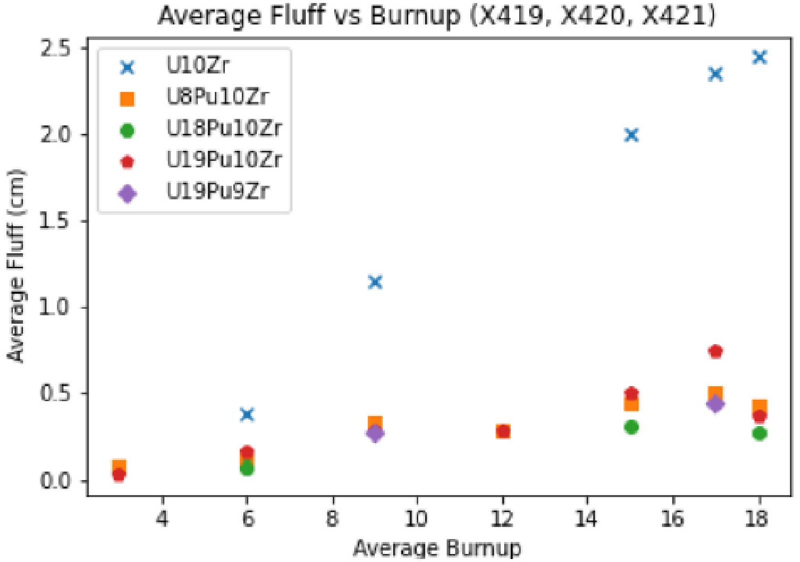
A porous structure is observed on the top of EBR-II fuel, here named “Fluff” structure:

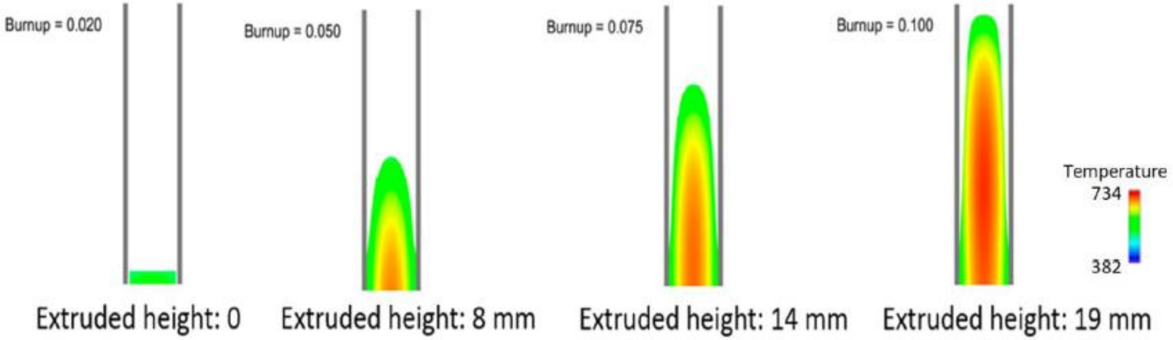
- Recent EPMA analyses showed this to contain not only FP but also fuel
- Such structure may be relevant to reactivity calculations

Analyses of EBR-II database (FIPD & IMIS) have been performed to determine the extension of this phenomenon and to correlate its formation to reactor operation parameters:

- The largest contributors to fluff formation to be burnup and composition.
- Moreover, higher pin operating temperature show to decreased fluff formation.



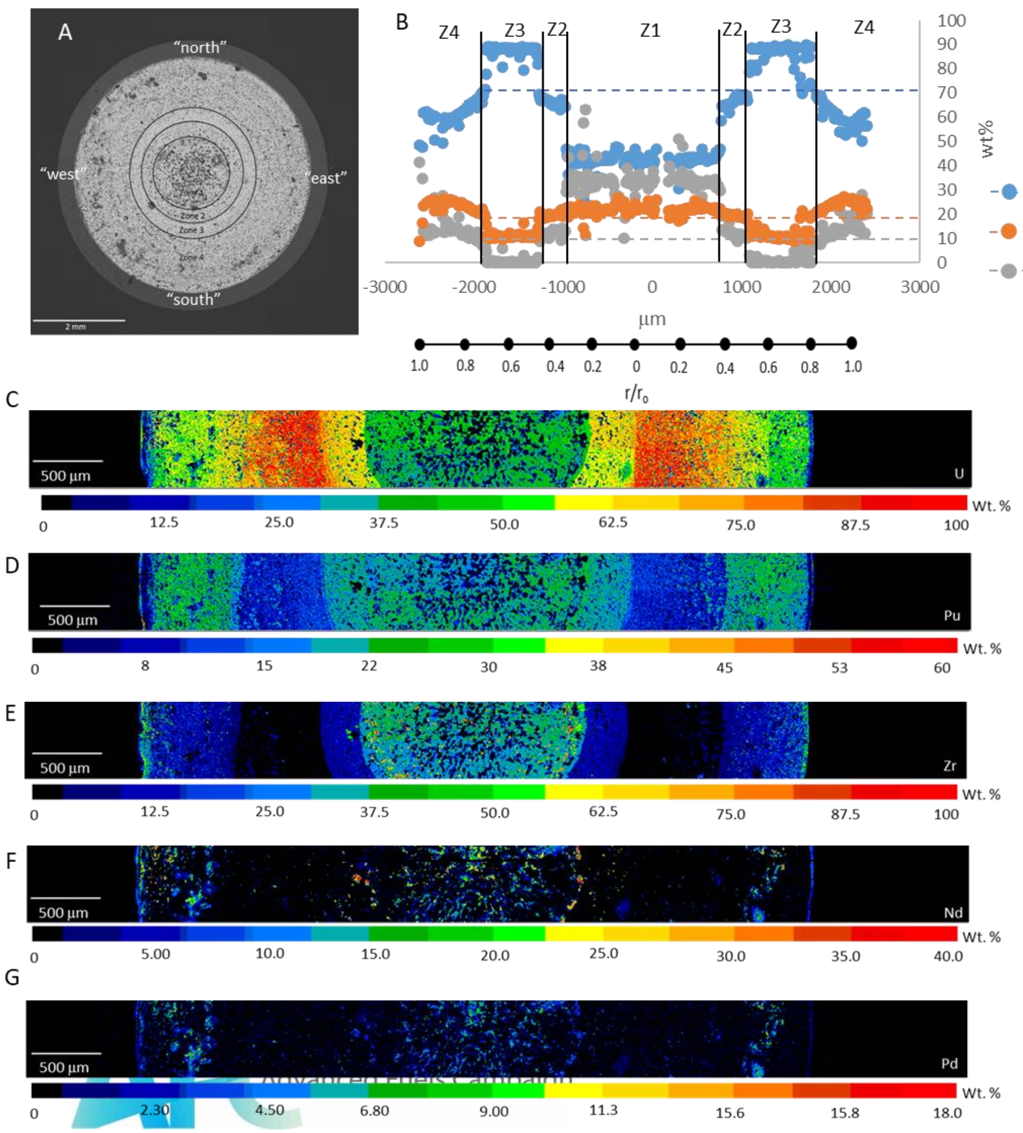




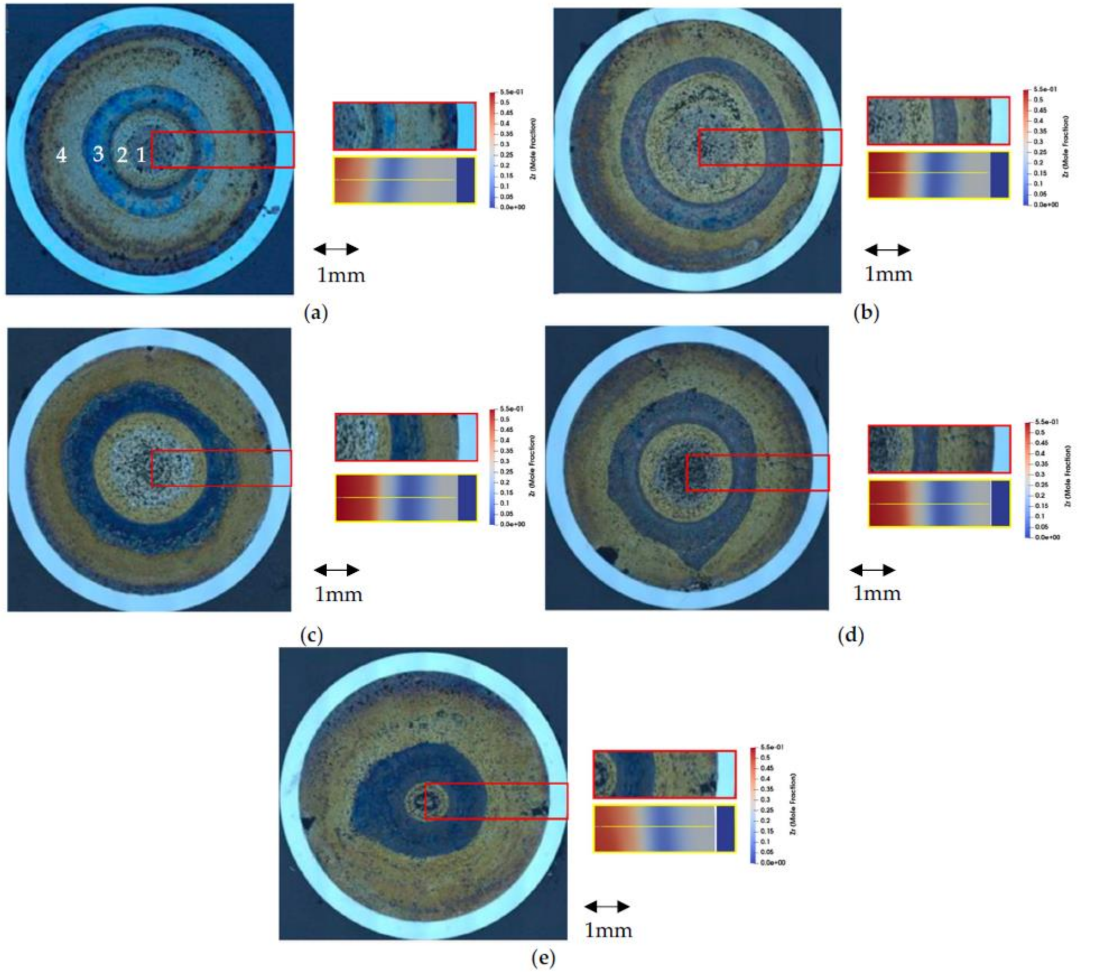
Idaho National Laboratory

X441 EPMA and BISON validation

U-19Pu-10Zr



- X441: U-Pu-Zr design parameter experiment (Zr wt.% variable)
  - Variable content of Zr change redistribution regions
  - Newest and larger EPMA dataset acquired since the '80
  - V&V both with BISON and on going with MARMOT



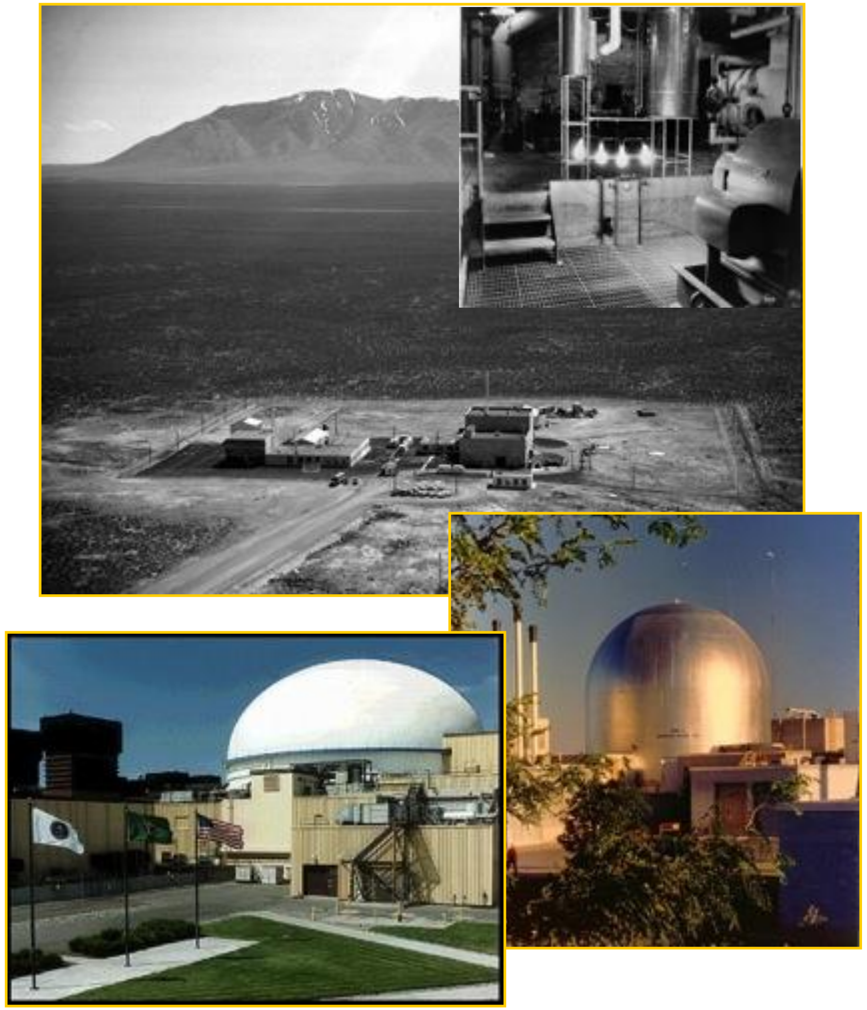
Conclusion

- AFC Metallic fuel R&D priorities shifting over years and 2 decades
  - Transmutation fuel alloys and spectrum comparison
  - Innovative metallic fuel alloys testing
  - Qualifying U-10Zr
  - Re-establish transient testing PIE capabilities
  - AFQ
- Multi year projects in harvesting EBR-II materials with several technical scopes
  - Enlarge database of EBR-II fuel PIE (many experiments were left without PIE examination)
  - Advanced characterization with state-of-the-art instruments – comparison of old vs new data
  - V&V for performance code, BISON and MARMOT
  - Transient fuel performance evaluation - Initiate THOR-C and THOR-M series
- Recently re-started harvesting FFTF MFF materials
  - U-Zr reference design – “closure” knowledge gaps (FCCI and thermophysical prop.)
  - Holistic approach to maximize data, minimize materials used, complementary modelling evaluation
- Complementary and synergistic efforts from other programs:
  - VTR – safety case / qualification for U-Pu-Zr
  - NSUF, NEUP – investigating “science” questions, low TRL concepts and separate effects for V&V

Back up slides

History of Metallic Fuels in Fast Reactor

- EBR-I (1951)
  - Unalloyed U
  - U-2Zr
  - Pu-1.25Al
- UK Dounreay Fast Reactor (1963)
  - U-0.1Cr
  - U-7Mo
  - U-9Mo
- Enrico Fermi FBR (1963)
  - U-10Mo
- EBR-II (1964)
  - U-5Fs
  - U-10Zr
  - U-20Pu-10Zr
- FFTF (1982)
  - Qualification of U-10Zr
  - Assembly testing of U-20Pu-10Zr



Harvesting EBR-II legacy materials

Advanced characterization with state-of-the-art instruments – comparison of old vs new data

V&V for performance code, BISON and MARMOT

- X441: U-Pu-Zr design parameter experiment (Zr wt.% variable)
  - Variable content of Zr change redistribution regions
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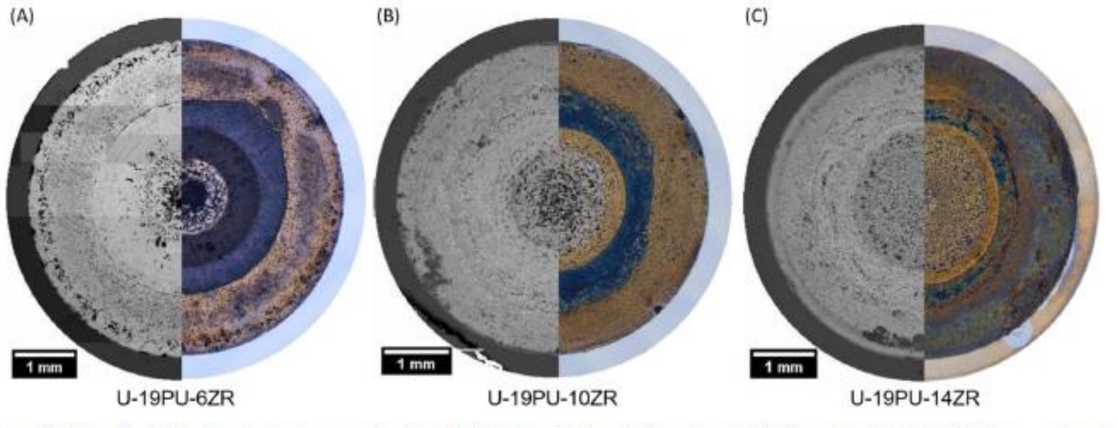

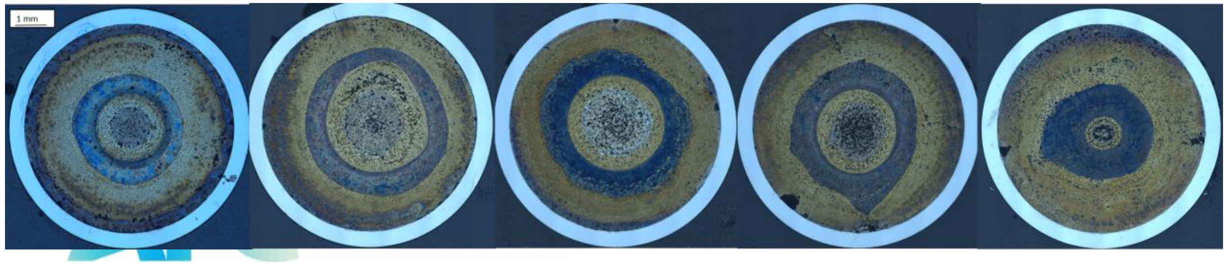

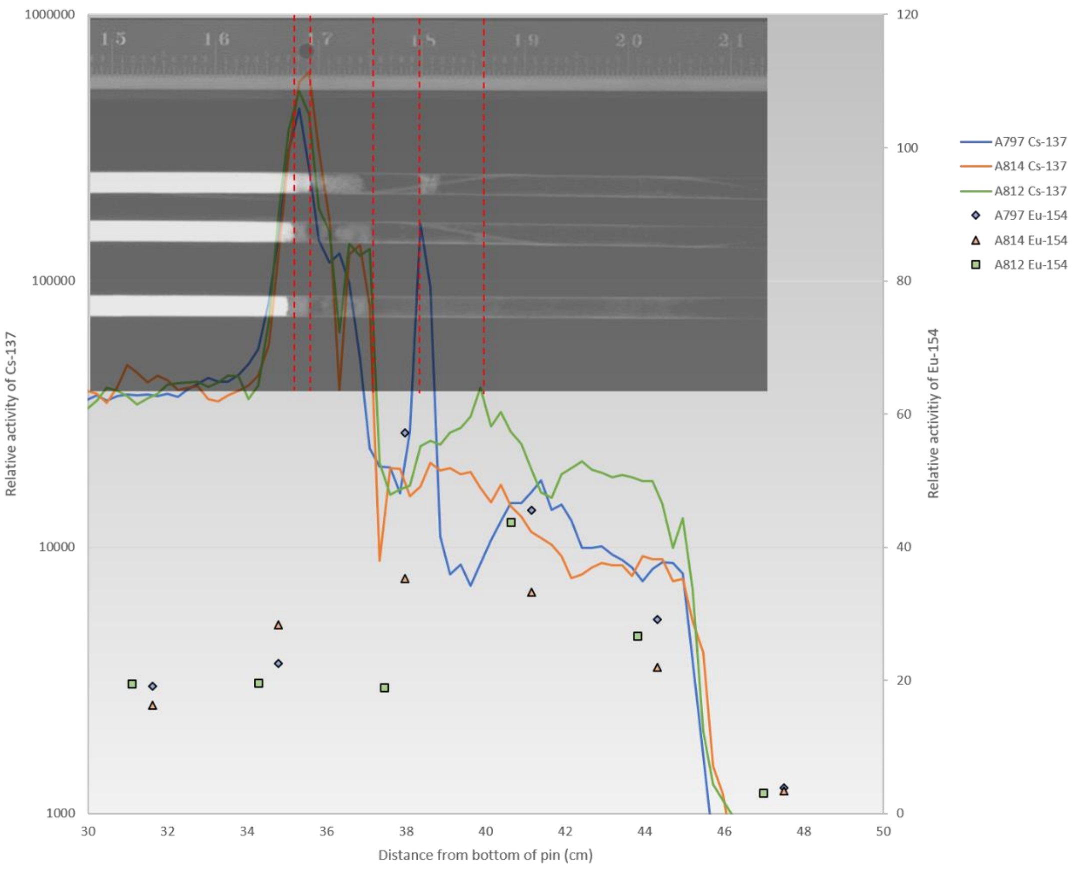

History of AFC irradiations 2003 - 2019