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NATRIUM

Metal Fuels Operating Experience

a TerraPower & GE-Hitachi technology

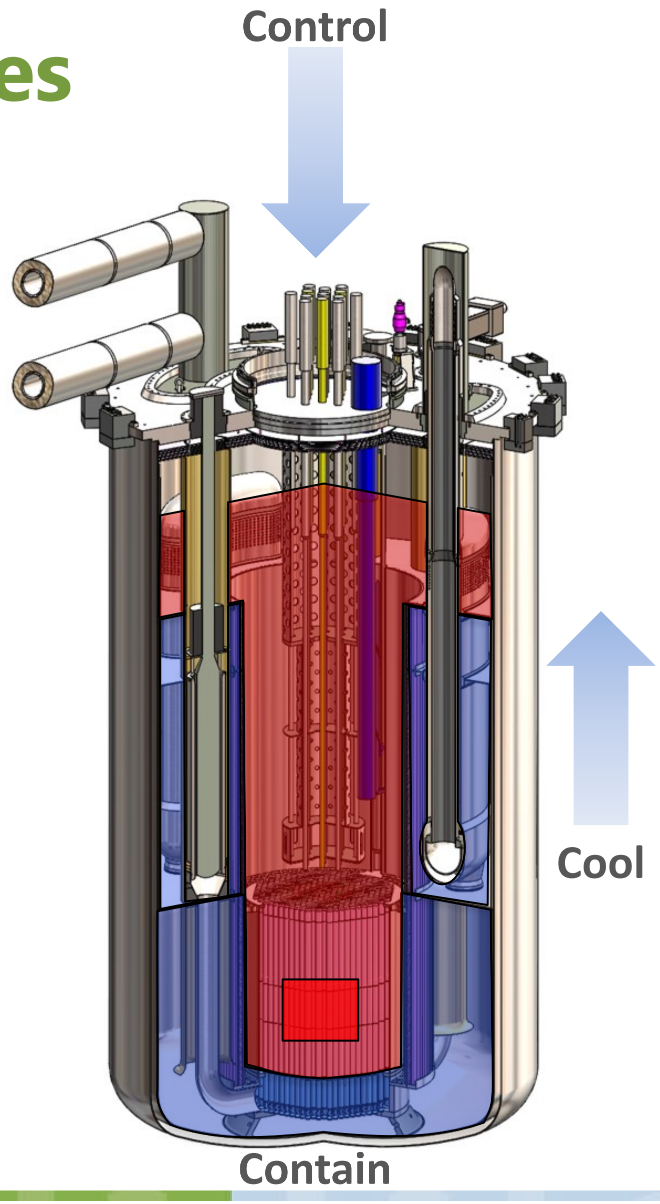
NATD-LIC-PRSNT-0039

Natrium™ Reactor Overview

- The Natrium project is demonstrating the ability to design, license, construct, startup and operate the Natrium plant within a seven-year timeframe
- Pre-application interactions are intended to reduce regulatory uncertainty and facilitate the NRC's understanding of the Natrium design and its safety case

Natrium Safety Features

- Pool-type Metal Fuel SFR with Molten Salt Energy Island
  - Metallic fuel and sodium have high compatibility
  - No sodium-water reaction in steam generator
  - Large thermal inertia enables simplified response to abnormal events
- Simplified Response to Abnormal Events
  - Reliable reactor shutdown
  - Transition to coolant natural circulation
  - Indefinite passive emergency decay heat removal
  - Low pressure functional containment
  - No reliance on Energy Island for safety functions
- No Safety-Related Operator Actions or AC power
- Technology Based on U.S. SFR Experience
  - EBR-I, EBR-II, FFTF, TREAT
  - SFR inherent safety characteristics demonstrated through testing in EBR-II and FFTF



Control

- Motor-driven control rod runback
- Gravity-driven control rod scram
- Inherently stable with increased power or temperature

Cool

- In-vessel primary sodium heat transport (limited penetrations)
- Intermediate air cooling natural draft flow
- Reactor air cooling natural draft flow – always on

Contain

- Low primary and secondary pressure
- Sodium affinity for radionuclides
- Multiple radionuclides retention boundaries

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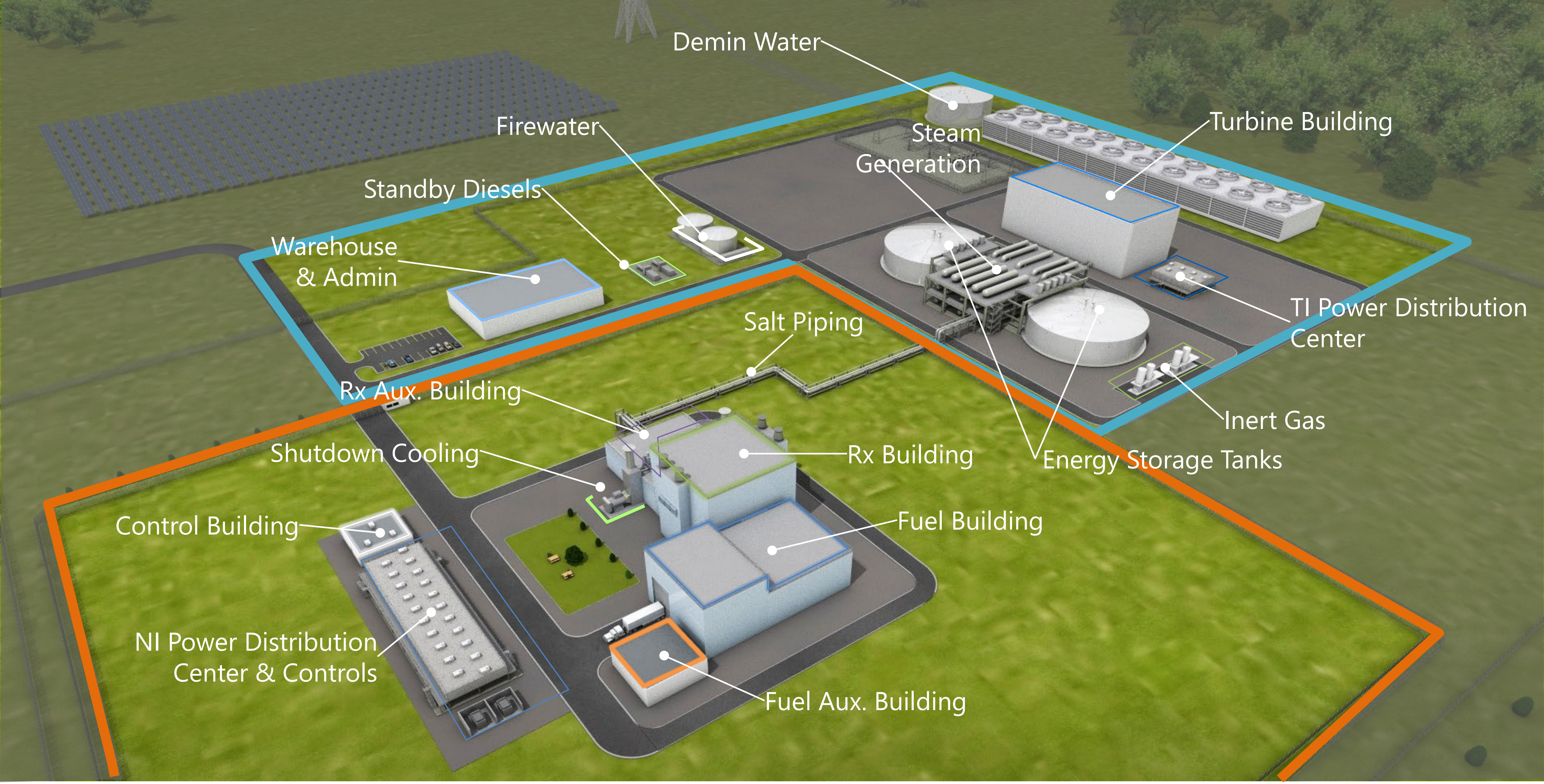
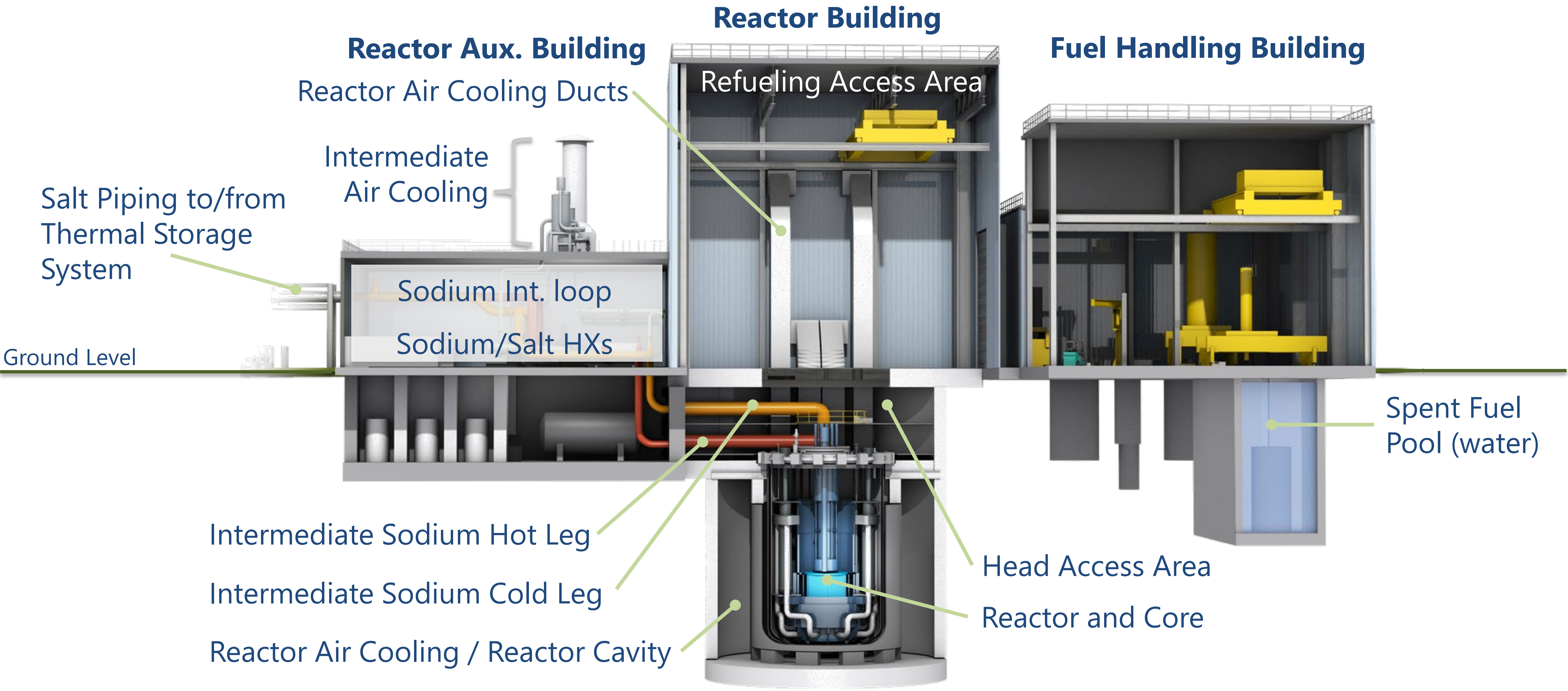

Demin Water
Firewater
Turbine Building
Steam Generation
Standby Diesels
Warehouse & Admin
TI Power Distribution Center
Salt Piping
Rx Aux. Building
Inert Gas
Shutdown Cooling
Rx Building
Energy Storage Tanks
Control Building
Fuel Building
NI Power Distribution Center & Controls
Fuel Aux. Building



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Presentation Outline Metallic Fuel Operating Experience

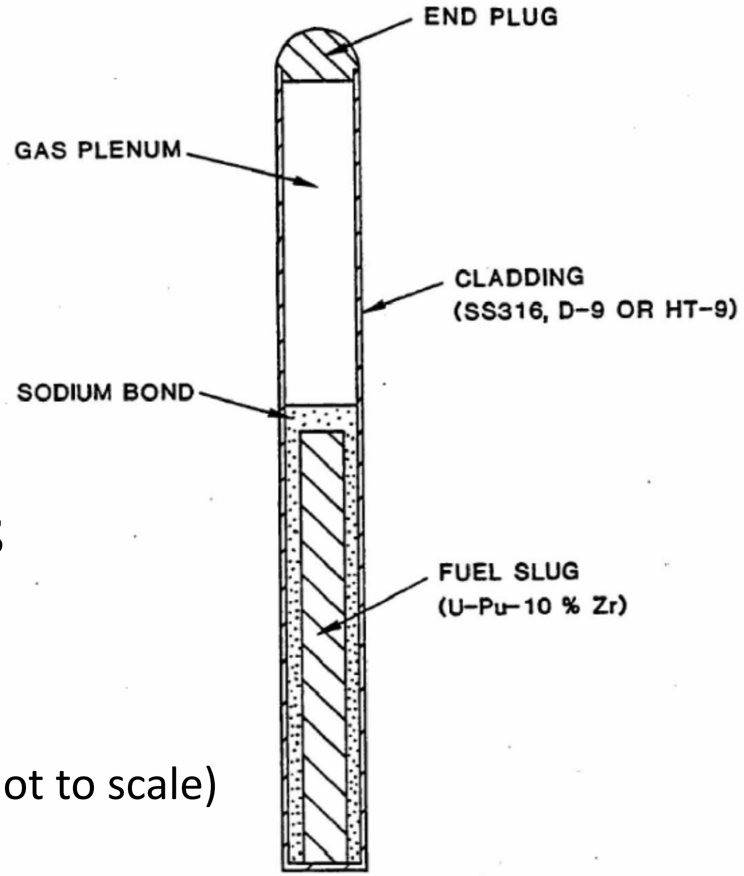
- Summary of the history of metallic fuel design and operation
- Key design parameters of metallic fuel designs and design evolution
- EBR-I and EBR-II metallic fuel operating history
- Important phenomena under steady-state conditions
- FFTF MFF metallic fuel operating history
- Relevant phenomena under transient conditions
- Questions and discussion

History of Metal Fuels Operating Experience

Reactor (Year Started)	Major Design Features and Behavior
EBR-I (1951) Mark I,II,III,IV • Unalloyed U • U-2Zr • Pu-1.25Al UK Dounreay (DFR) (1963) • U-0.1Cr • U-7Mo • U-9Mo EBR-II (1964) Mark I • U-5Fs (Fs: 2.4Mo-1.9Ru-0.3Rh-0.2Pd-0.1Zr-0.01Nb)	1st Era Designs (EBR-I, DFR, EBR-II Mark-I/IA) 3 at.% burnup • Metallic fuel used due to its compatibility with sodium, good conductivity, fabricability, reprocessing, reliable performance within operating limits • Metal fuel swelling due to fission product gases limited burnup • Uranium alloys behave similarly • Alloying can improve some performance characteristics; does not eliminate fission gas induced swelling
EBR-II (1964) Mark II, III • U-5Fs (Fs: 2.4Mo-1.9Ru-0.3Rh-0.2Pd-0.1Zr-0.01Nb) • U-10Zr • U-20Pu-10Zr	2nd Era Designs (EBR-II Mark II, Mark III) to 10 at.% burnup • 75% fuel smeared density to increase fission gas release (FGR) • Increased plenum size to accommodate higher FGR-maintain pressure/stress • Increased cladding thickness • Cladding materials 316, Cold Worked (CW) 316, CW D9
EBR-II (1964) Mark IV, V • U-10Zr • U-20Pu-10Zr FFTF (1972) • Assembly testing of U-10Zr • Assembly testing of U-20Pu-10Zr	3rd Era Designs (EBR-II Mark IV, Mark V, FFTF MFF) to 15-20 at.% • Improved cladding alloys with very low swelling • Ferritic-martensitic stainless steels – HT9 • Increased cladding thickness

Key Design Parameters of Metallic Fuel Rod

- Smear density
- Plenum size
- Fuel alloying
- Sodium bond
- Cladding materials
- Cladding thickness
- Diameter
- Length



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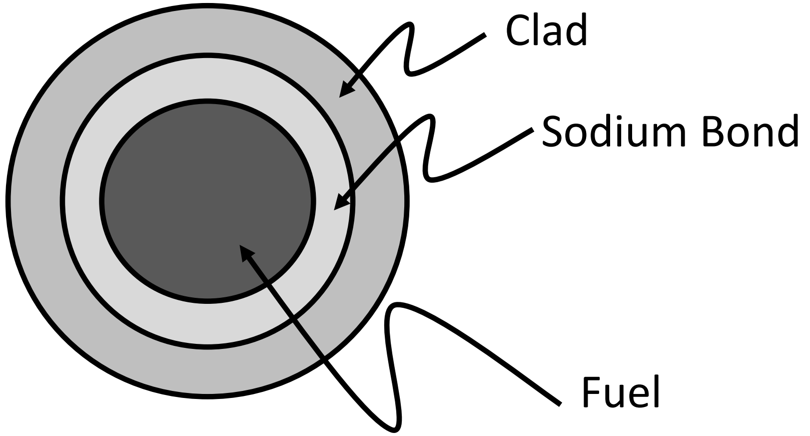

$$Smear\ Density = \frac{Area\ of\ \ Fuel}{Area\ Within\ Clad}$$

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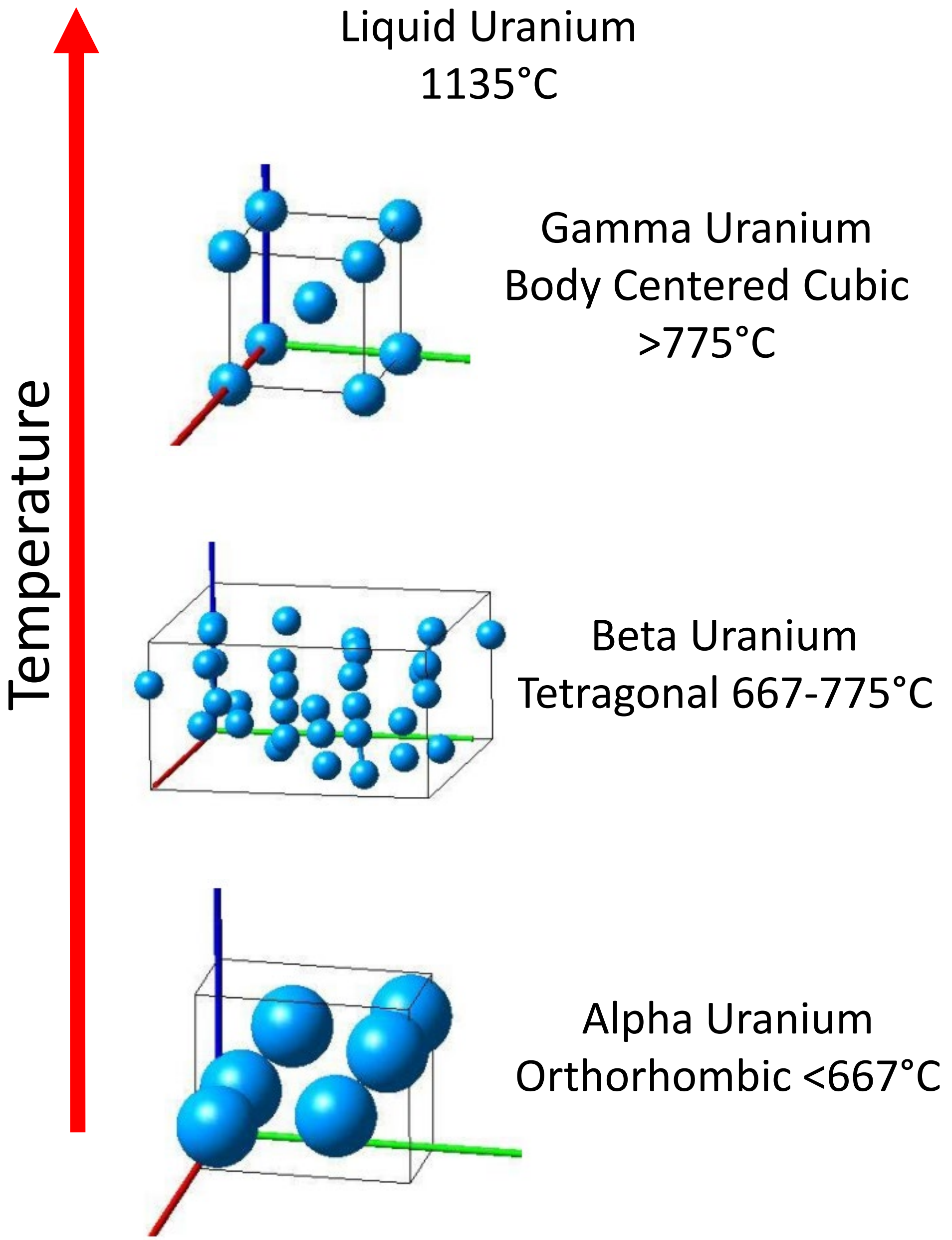
Metal Uranium Fuel

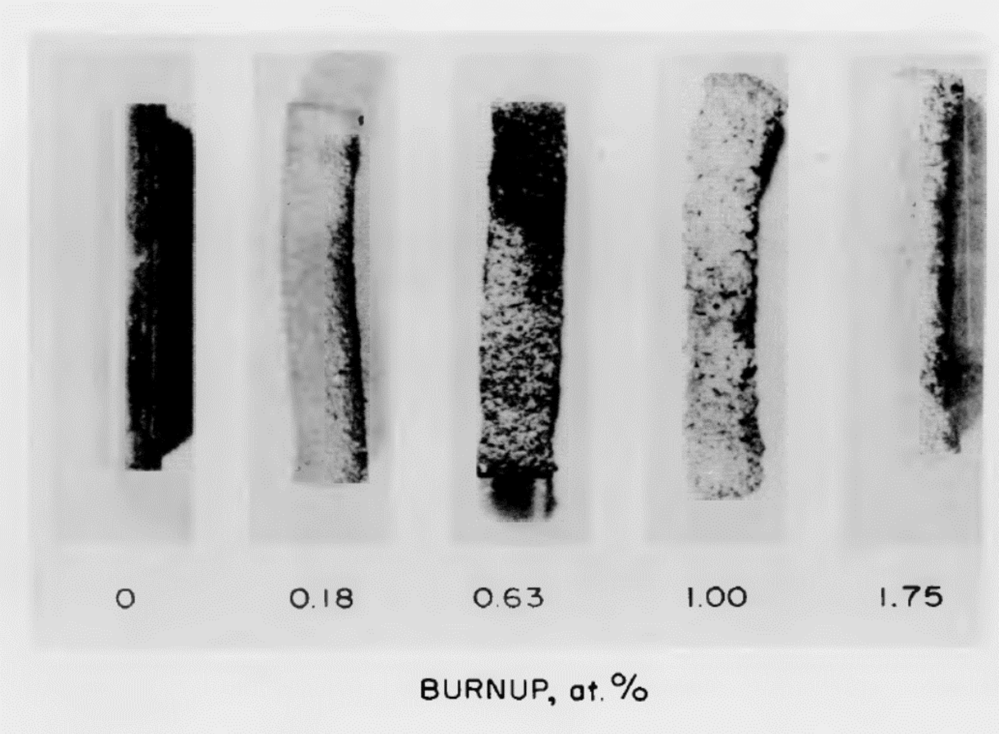
Why metal uranium?

- The Good
  - Uranium fissions
  - Uranium can be turned into plutonium which fissions
  - Highest concentration of uranium atoms
  - High thermal conductivity: 30-50 W/mK
  - Good compatibility with sodium coolant

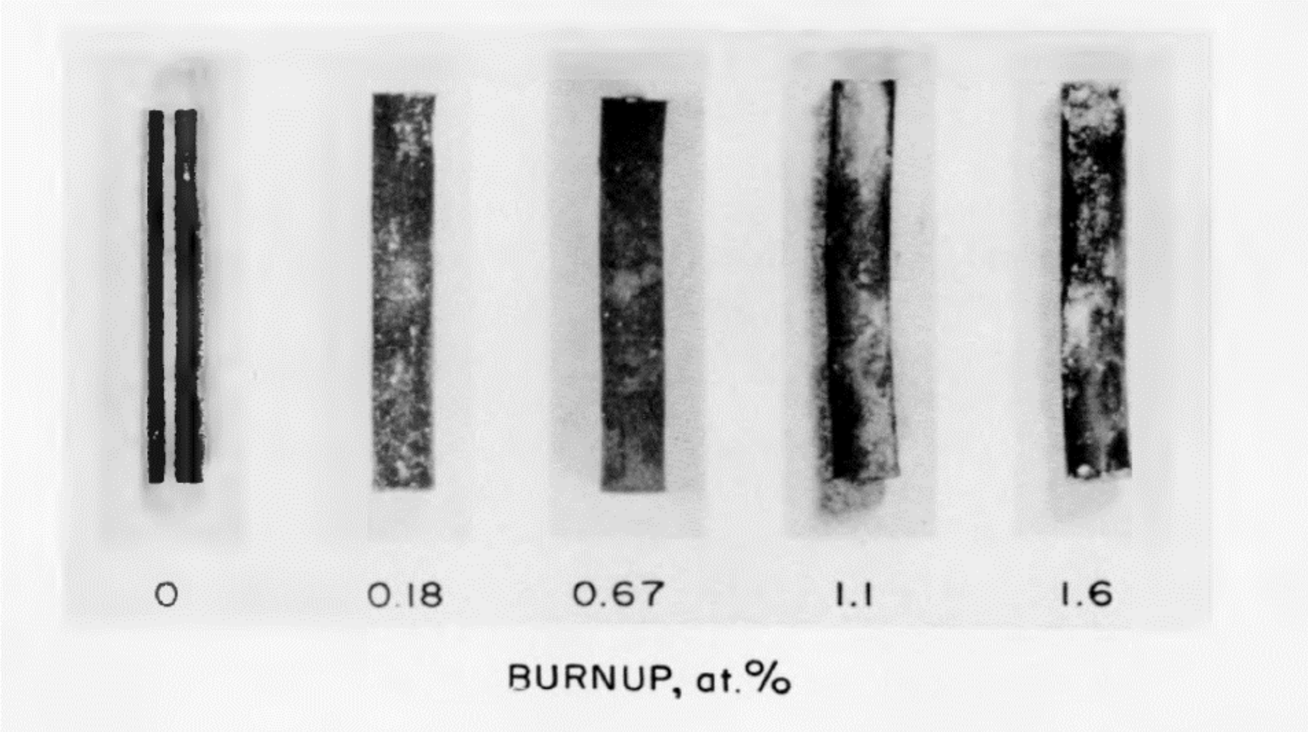
- The Bad
  - Poor dimensional stability – Asymmetric Structure
    - Swelling (change in volume)
    - Growth (change in shape)
  - Multiple solid phases (allotropes) or crystal structures
  - Fuel-clad interaction: eutectics, fission products



Efforts to Reduce Fuel Swelling and Distortion



Effects of irradiation on uranium specimens hot-rolled at 300°C and quenched from the beta phase



Effects of irradiation below 350°C on cast uranium-1.62 wt% zirconium

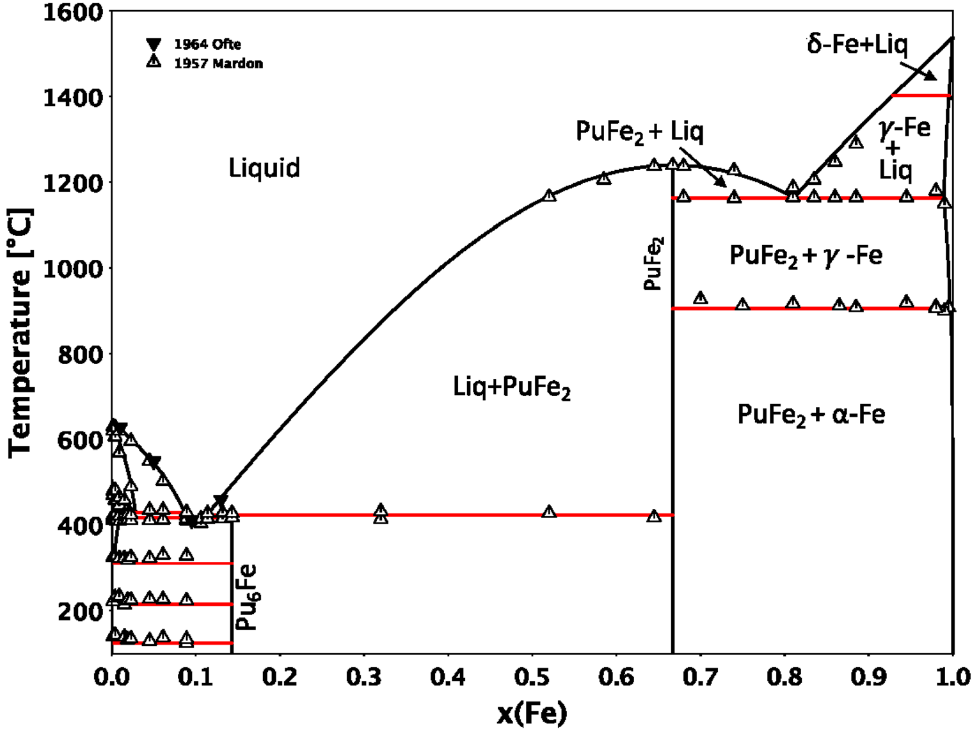
Heat treatments and alloy additions help, but do not stop fission gas induced swelling; burnups were still quite limited.

Kittel and Paine, "Effects of Irradiation on Fuel Materials," 1959

Why U-10Zr and U-Pu-10Zr Alloys for the IFR Program?

- The Integral Fast Reactor (IFR) program envisioned a closed fuel cycle with plutonium breeding and recycling
- Plutonium forms a eutectic with iron at 415°C
- Uranium forms a eutectic with iron at 725°C
- U-15Pu-10Zr forms a eutectic with stainless steel at 825°C
- The addition of zirconium to the alloy allowed for acceptable eutectic formation temperatures
- In addition, zirconium improves dimensional stability and increases the solidus temperature of the fuel

Plutonium-Iron Phase Diagram



Metallic Fuel Design Evolution

Parameter	Mark-I/IA	Mark-II	Mark-III	Mark-IV	Mark-I/IA	Mark-II	Mark-IIC	Mark-IICS	Mark-III	Mark-IIIA	Mark-IV	(MFF-1/1A,IFR-1)	Series II (MFF-2,-3)	Series IIIb (MFF-4,-5,-6)
	EBR-I				EBR-II							FFTF		
	1st Design Era					2nd Design Era					3rd Design Era			
Fuel alloy (wt%)	U	U-2Zr	U-2Zr	Pu-1.25Al	U-5Fs	U-5Fs	U-10Zr	U-10Zr	U-10Zr	U-10Zr	U-10Zr	U-10Zr,U-Pu-10Zr	U-10Zr	U-10Zr
Enrichment weight (%235U)	93	93	93		52	67	78	78	66.9	66.9	69.6	25.2,17.5,4.5	32.4	31
Cladding	SS347L-ribbed	SS347	Zry-2 ribbed	Zry-2 ribbed	SS304L	SS316	SS316	SS316	CWi D9	CW SS316	HT9	HT9	HT9	HT9
Fuel smear density (%)	81.2	90.3	100.0	83.6	85	75	75	75	75	75	75	75	75	75
Fuel slug diameter (cm)	0.925	0.975	0.925	0.597	0.366	0.33			0.439		0.427	0.498	0.498	0.498
Cladding wall thickness (cm)	0.0508		0.0508	0.05334	0.023	0.03	0.03	0.03	0.038	0.038	0.046	0.0559	0.0559	0.0559
Cladding outer diameter (cm)	1.138	1.138	1.026	0.759	0.442	0.442	0.442	0.442	0.584	0.584	0.584	0.686	0.686	0.686
Length (cm)			53.815	60.960	46	61.2	63	53.6	74.9	74.9	74.9	238.1	238.1	237.1
Spacer wire diameter (cm)			0.117	0.140	0.124	0.124	0.124	0.124	0.107	0.107	0.107	0.135	0.135	0.135
Burnup limit (at. %)	0.2	0.062	0.2	0.2	2.6	8	8.9	6.4	10	10	N/Aii	3.8,9.4	13.8,14.3	10.1,14.1
Plenum/fuel vol. ratio					0.18	0.83	1.01	0.68	1.45	1.45	1.45	1.2	1.63	1.79
Approximate number fabricated					97,399	104,501	incl in Mk-II	incl in Mk-II	16,104	incl in Mk-III	400	182	338	507
Dates Irradiation	1951	1954	1957	1962	1964	1973			1984			1986	1988	1990
Dates Fabrication	1951	1954	1957	1962	1964	1973	1981	1987	1987	1989	1987			

- Limited burnup <1 at.% due to fission product gas induced swelling
- Uranium alloys behave similarly
- Alloying can improve some performance; does not eliminate fission gas induced swelling

- 75% fuel smeared density to increase fission gas release (FGR) and alleviate FCMI
- Increased plenum size
- Increased cladding thickness
- Cladding materials 316, (CW) 316, D9

- Improved cladding alloys with very low swelling
- Ferritic-martensitic stainless steels – HT9
- Increased cladding thickness
- Increased diameter, and length

EBR-I AND EBR-II METALLIC FUEL OPERATING EXPERIENCE

1st Design Era Metallic Fuel Operating Experience in EBR-I and EBR-II

- First generation of EBR-I fuel used a uranium-2wt% zirconium alloy, Type 347 stainless steel cladding, and NaK bond (same as primary coolant); burnups were limited to less than 0.35 at% Walters et al., “Performance of Metallic Fuels and Blankets in Liquid-Metal Fast Breeder Reactors,” 1984
- Second generation of EBR-I fuel used high enriched uranium, Type 347 stainless steel cladding, and NaK bond (same as primary coolant); maximum burnups was 0.1 at% Walters et al., “Performance of Metallic Fuels and Blankets in Liquid-Metal Fast Breeder Reactors,” 1984
- Third generation of EBR-I fuel used Zircaloy-2 cladding co-extruded with a uranium-2wt% zirconium alloy; burnups were limited to less than 0.3 at% Rice et al., “EBR-I Mark III Design Report,” 1958
- Fourth generation of EBR-I fuel with plutonium-1.25wt% aluminum alloy and Zircaloy-2 cladding, increased cladding thickness, and demonstrated breeding ratio 1.27; burnups were limited to less than 1 at% ANL-6301, 1960 and ASME, “Experimental Breeder Reactor I,” 1979
- First generation of EBR-II fuel introduced uranium-5wt% fissium alloy, larger plenum, stainless steel cladding; burnups were limited to less than 3 at% Walters et al., “Performance of Metallic Fuels and Blankets in Liquid-Metal Fast Breeder Reactors,” 1984

Work of Blake (1961) and Barnes (1964)

- Attempts to develop non-swelling fuel alloys were not successful

- Blake
  - Swelling is caused by the accumulation of fission products
  - If the fuel is contained within sufficiently strong cladding and space is provided with the cladding to accommodate fission products, high burnup can be achieved
- Barnes
  - Fission gases (Xe and Kr) are insoluble in uranium
  - At pore volume fractions above about 30%, pores interconnect and fission gases are released
- New Design Concept
  - These observations led to “modern” metallic fuel designs
  - Smear densities of 75% (annular gap), sodium-bonded, large plenums

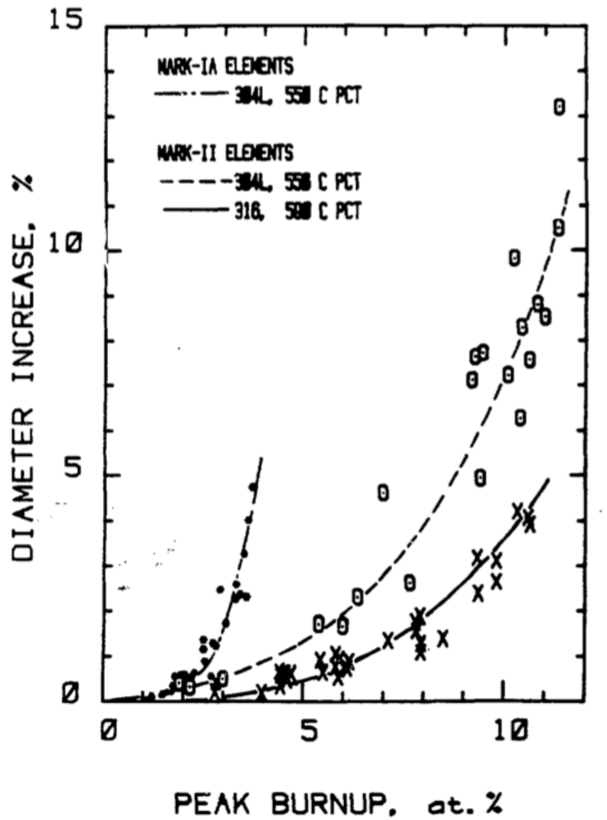
EBR-II Mark-II Fuel Design Evolution

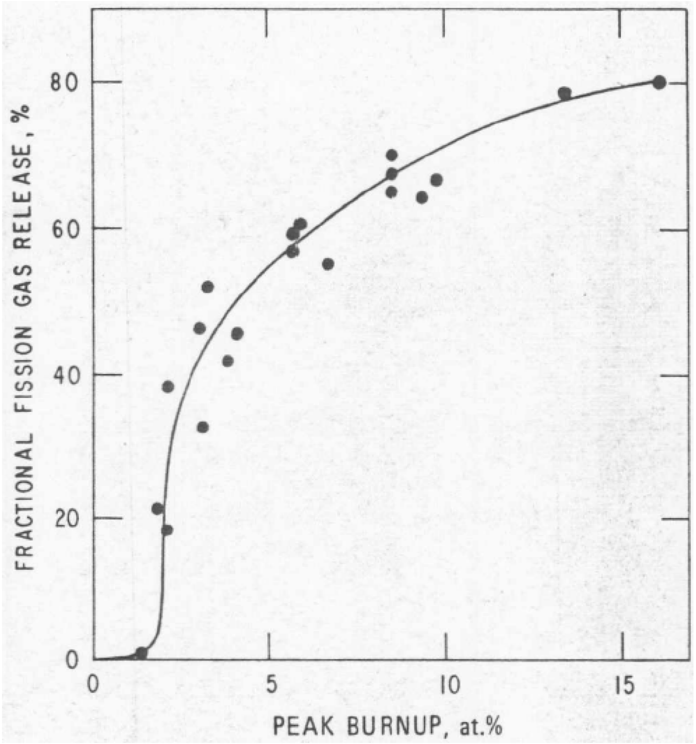
- Fuel-cladding gap was increased from 0.15 to 0.25 mm radially to allow unrestrained isotropic fuel swelling to a volume increase of 35%.
- Fission gas induced porosity of ~ 30% associated with this amount of swelling is largely interconnected.
- Volume of the gas plenum above the fuel was increased four-fold to accommodate this released gas at a reasonably low pressure.
- Increasing the cladding wall thickness by 33% to 0.3 mm for further reduction in cladding stress.

EBR-II Mark-II Sample of Fuel Qualification for Large Design Changes

- Largest design change to 75% smear density, plenum size
- Total test time 7 yrs: Run No. 27, Feb 1968 to Run No. 75, Feb 1975
- 1 LTA with 37 Encapsulated fuel pins
  - Peak burnup, 16 at.%
  - 11 interim examinations
- 6 LTA each with 91 unencapsulated fuel pins
  - Run No. 36, Jun. 1969 until Run No. 58, Aug. 1972
  - Peak burnup, 6 at.%
- 1 LTA reconstituted with 91 unencapsulated fuel pins
  - Run No. 65, Aug. 1973 until Run No. 74, Oct. 1974
  - Peak burnup, 9.5 at.%

EBR-II Metal Fuel Mark-II Design Performance



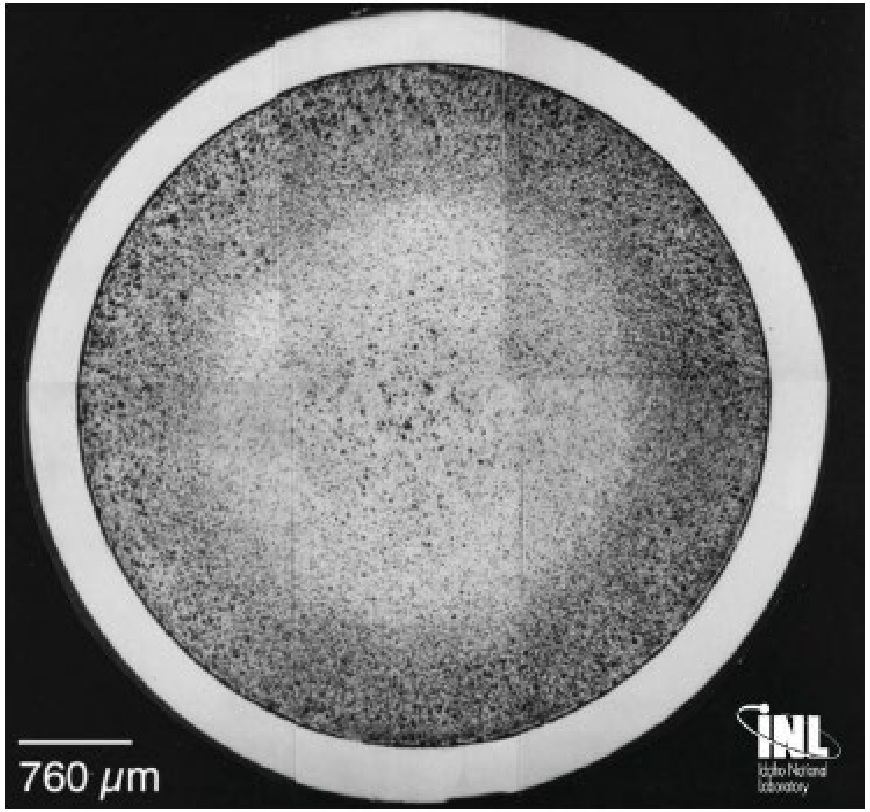


EBR-II Mark-II diameter axial profiles showing improved performance of 316 compared to 304L.
Mark-II U-5Fs 75% smeared density fuel demonstrating high fission gas release.

Hofman, "Irradiation Behavior of Experimental Mark-II Experimental Breeder Reactor II Driver," 1980

Steady-State Fuel Performance: Key Phenomena

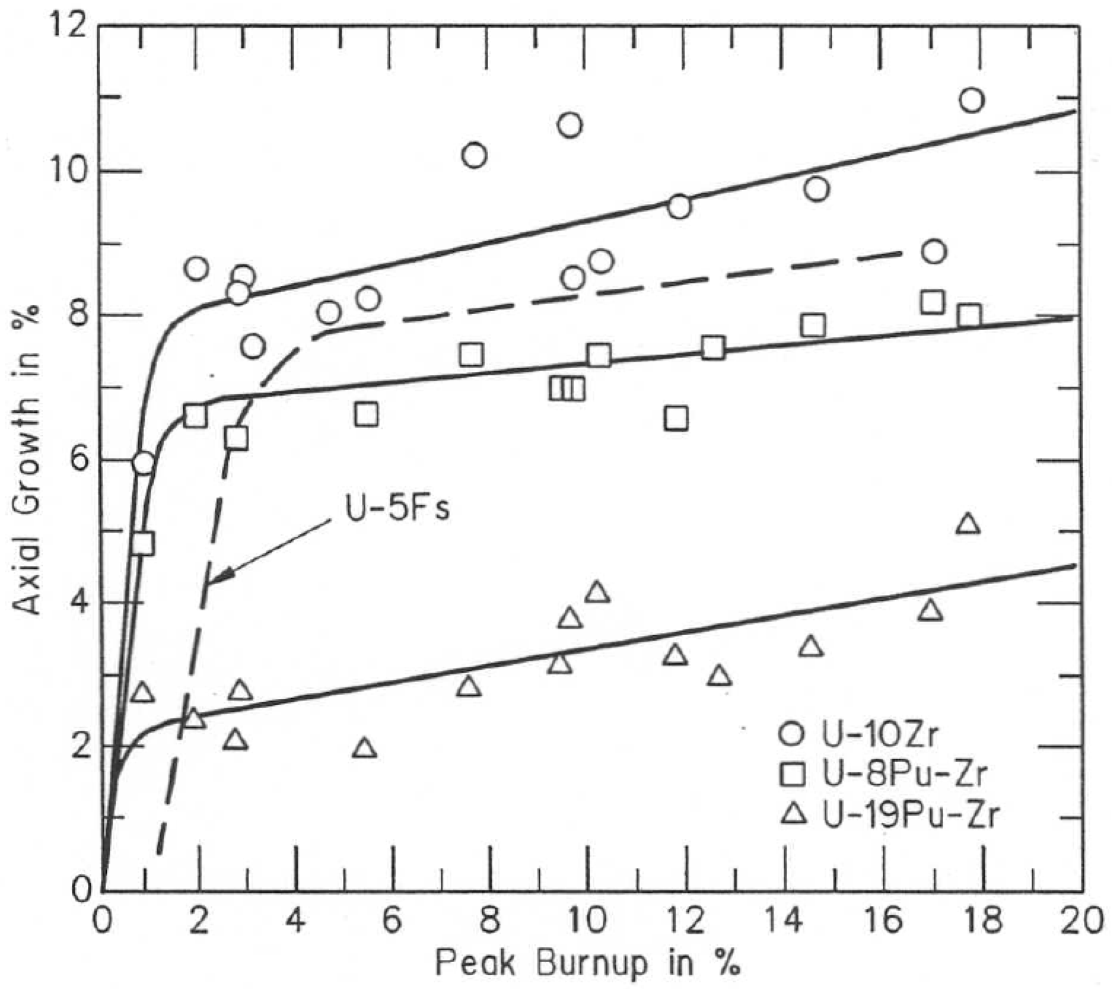
- Fuel axial growth
- Fission gas release
- Cladding irradiation creep and swelling
- Fuel-cladding chemical interaction
- Constituent redistribution



Sodium-bonded, 75% smear density,
U-10wt% Zr at about 5 at% burnup.

Fuel Axial Growth

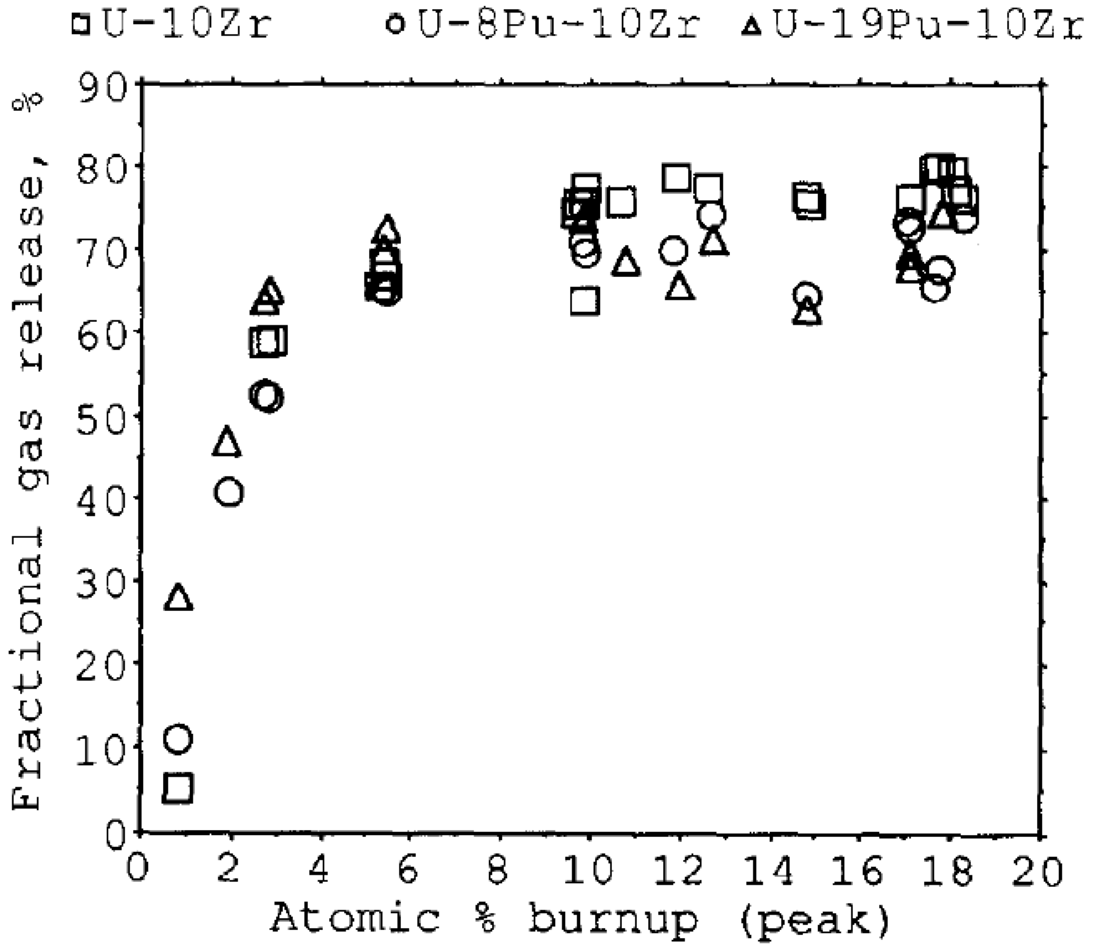
- At 75% smear density, U-Pu-10Zr fuel contacts the cladding at 1-2 at% burnup
- Prior to contact with the cladding, the fuel grows substantial amounts (2-8%) in the axial direction
- After contact, the fuel grows another 2-3% in the axial direction by 20 at% burnup
- Fuel swelling and growth increases fuel porosity and reduces uranium density resulting in a decrease in core reactivity



G. Hofman and L.C. Walters, "Metallic Fast Reactor Fuels," (1994)

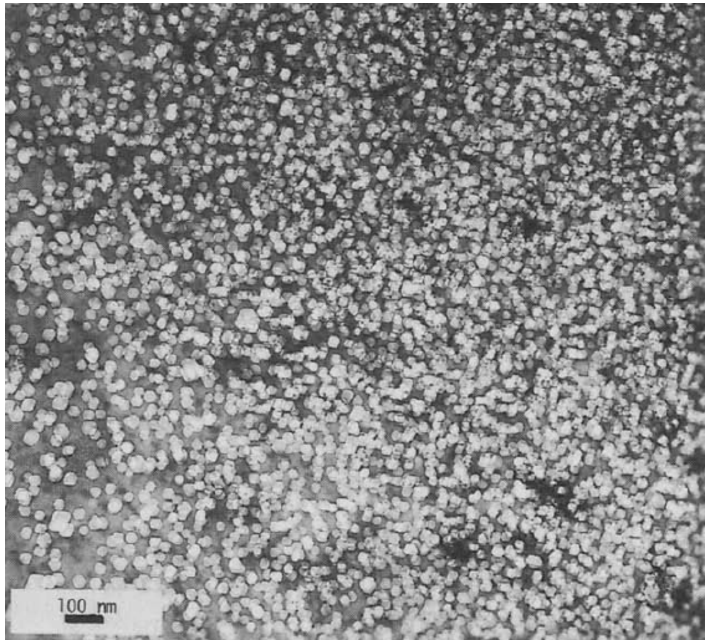
Fuel Swelling and Fission Product Gas Release

- Fission gases in isolated pores drive initial fuel swelling
- As fuel continues to swell, more porosity becomes interconnected, allowing fission gas to access the plenum
- Load on cladding increases with burnup
  - Porosity is filled with solid fission products
  - More fission gas in the isolated and interconnected porosity → higher gas pressures

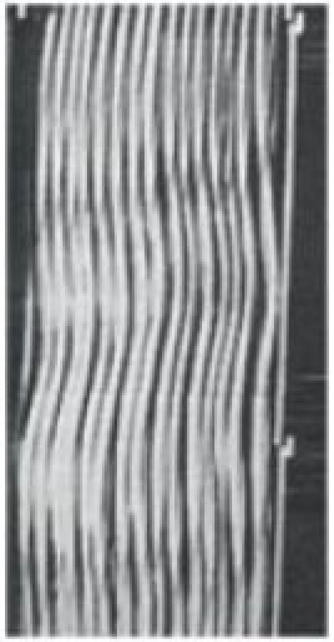


R.G. Pahl, et al., "Irradiation behavior of metallic fast reactor fuels," 188 (1992)

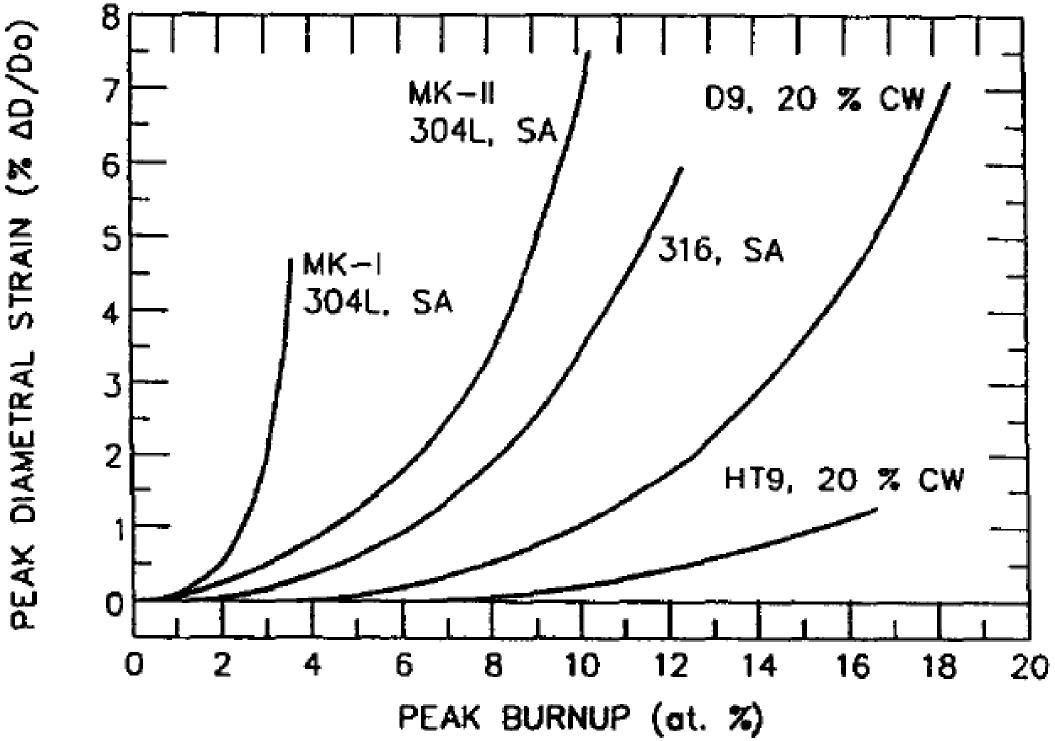
Irradiation-Induced Swelling in Cladding Alloys



Micrograph of irradiation-induced swelling in 304 stainless steel irradiated to about 30 DPA at 400 °C



Distortion of 316 clad fuel pins caused by irradiation creep and swelling

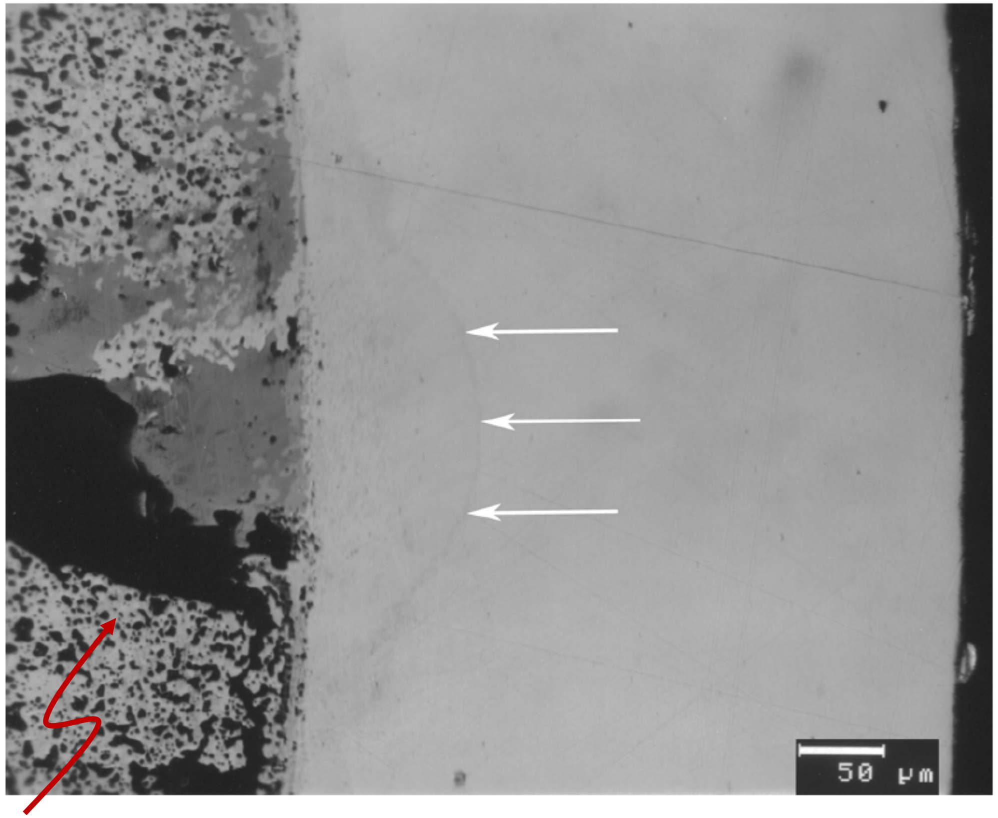


In addition to cladding irradiation creep caused by fuel and fission product loading, swelling of the cladding caused by void formation has historically been a significant contributor to fuel system deformation. *Ferritic-martensitic* alloys like HT9 are now the primary focus of cladding development efforts because of their superior swelling resistance.

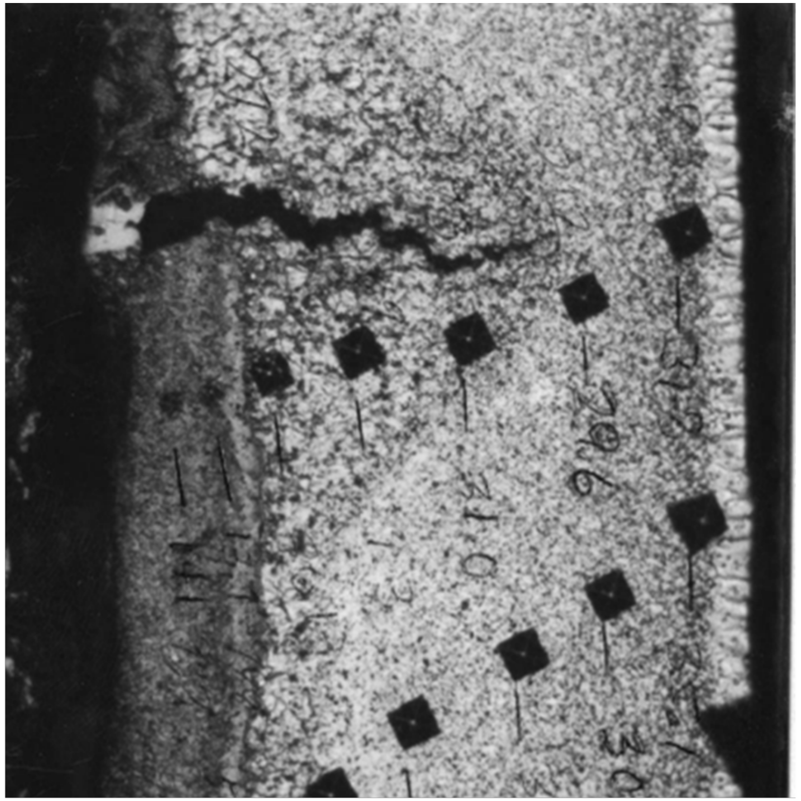
J.H. Kittel, B.R.T. Frost, J.P. Mustelier, K.Q. Bagley, G.C. Crittenen, and J. Van Dievoet, “History of Fast Reactor Fuel Development,” Journal of Nuclear Materials 204 (1993)

G. Hofman, L.C. Walters, and T.H. Bauer, “Metallic Fast Reactor Fuels,” Progress in Nuclear Energy 31 (1997)

Fuel-Cladding Chemical Interaction (FCCI)



Deposit of lanthanide fission products diffusing into cladding
Keiser, "Fuel-Cladding Interaction Layers in Irradiated U-Zr and U-Pu-Zr Fuel Elements," 2006

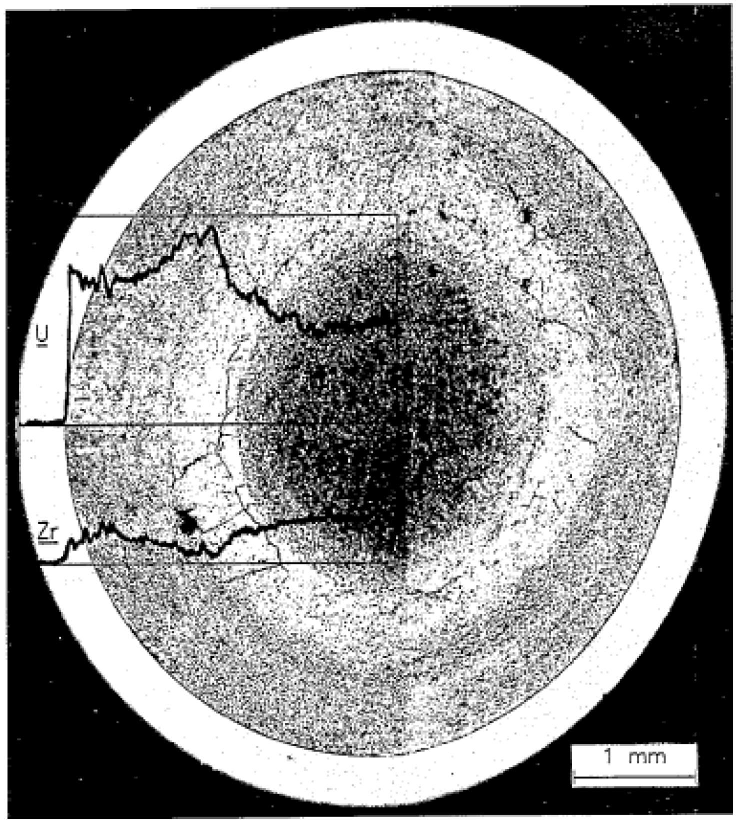


Indents show an increase in hardness in the interaction zone; embrittlement led to clad cracking

FCCI is the phenomenon that typically limits the lifetime of metallic fuels.

Constituent Redistribution

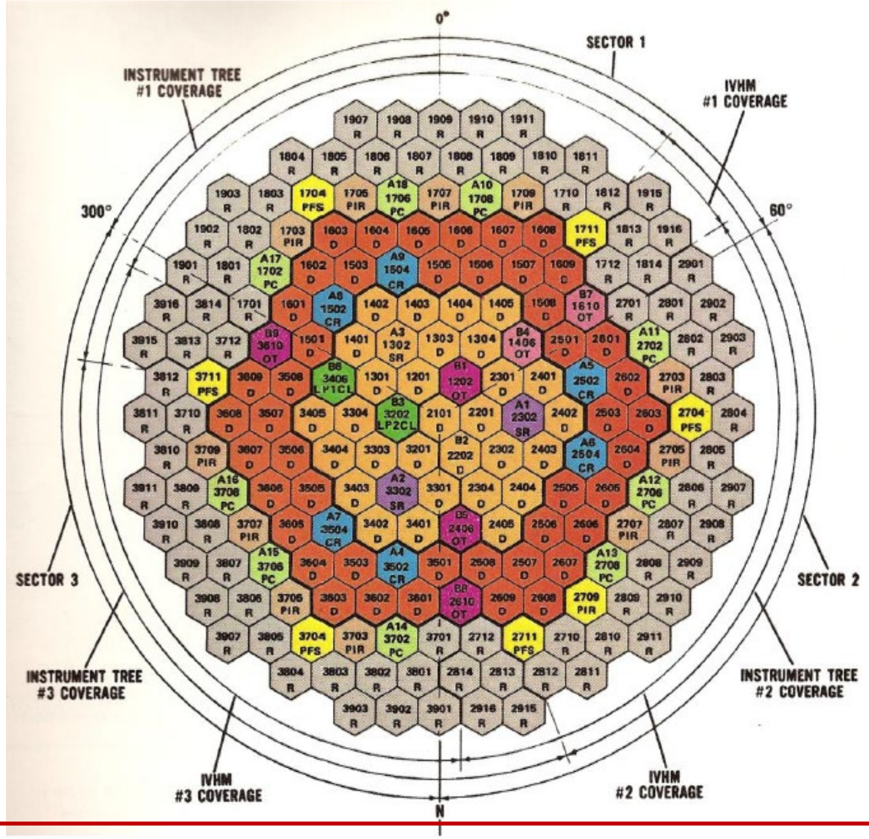
- Under a typical temperature gradient, three distinct radial regions may form in U-10Zr and U-Pu-Zr fuel
- Zr concentration is lower at mid-radius but higher at center and edge regions
- An equilibrium profile is established at ~5 at.%
- Fundamental mechanism is fairly well understood and has been implemented in predictive models (Ogawa et al., 1990; Hofman et al., 1996; Sohn et al., 2000; Ishida et al., 1993; Kim et al., 2006; Hirschhorn et al., 2020)



U-10Zr (DP81) zirconium is lower at mid radius but higher at center and periphery regions. Hofman, 1996

FFTF METALLIC FUEL OPERATING EXPERIENCE

Schematic of Fast Flux Test Facility Core



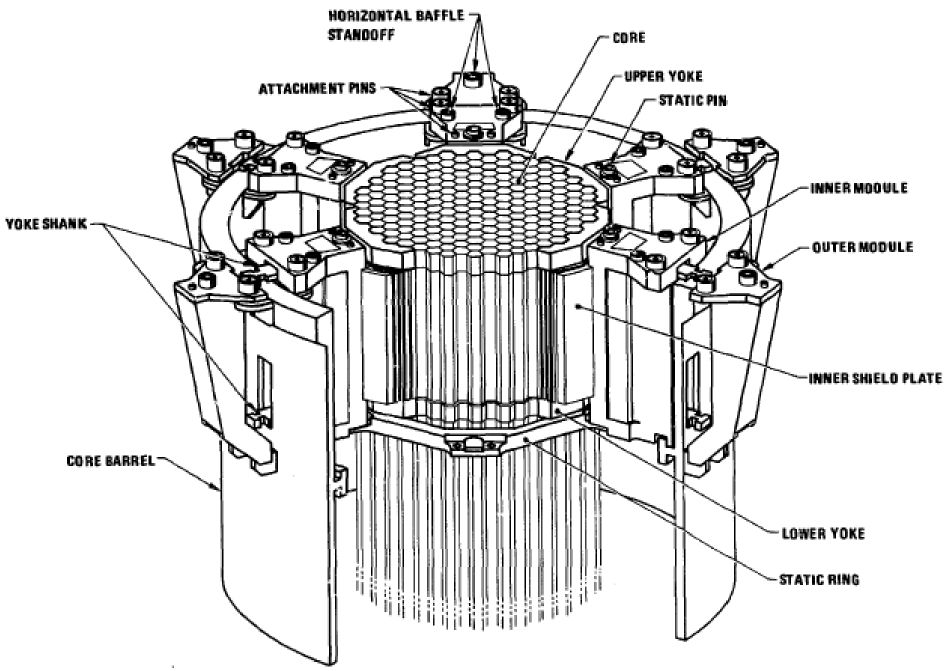
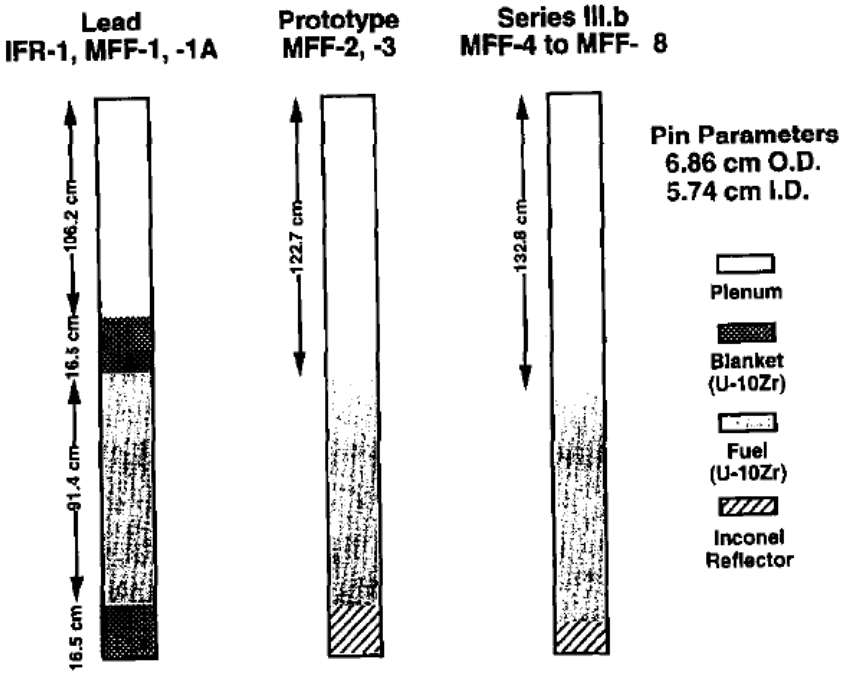
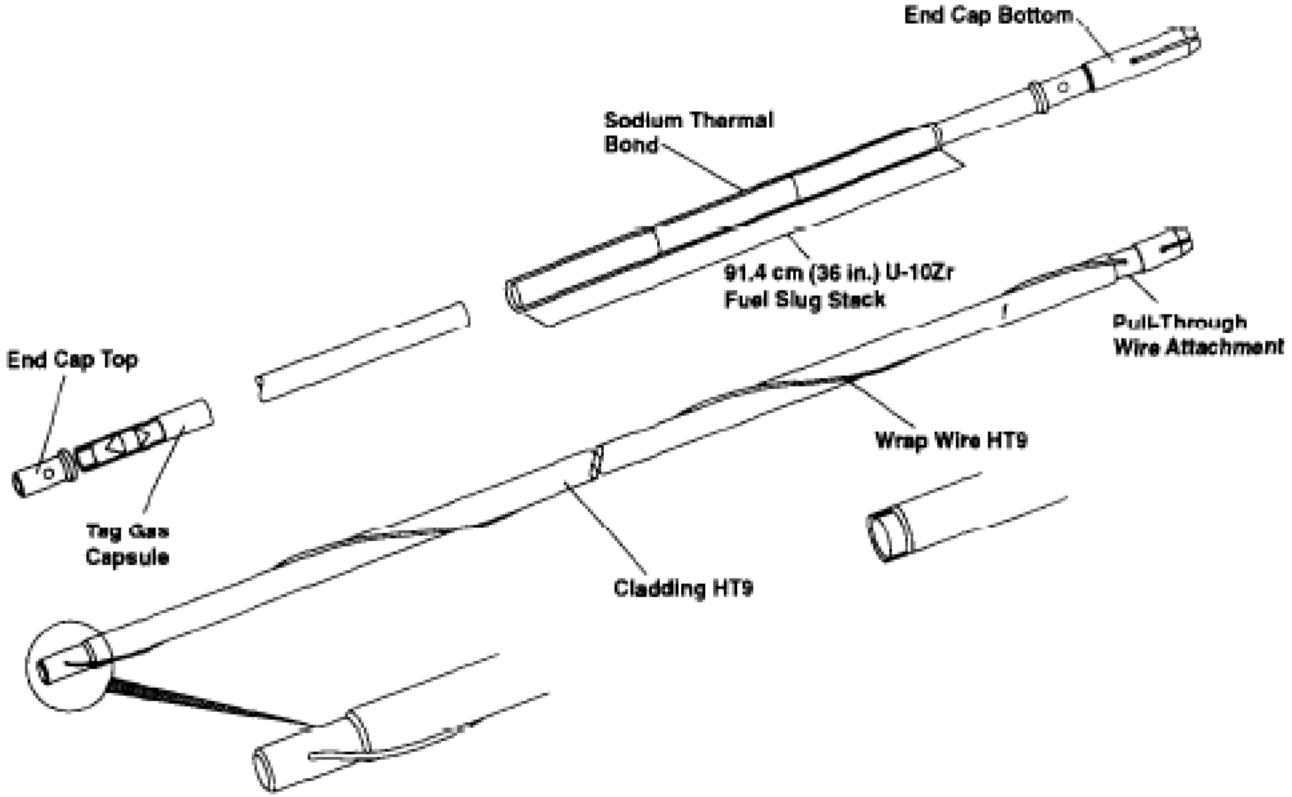


Figure II-3. Core Restraint Modules

FFTF core was oxide fuel with the plan to convert to metallic fuel. A metallic fuel qualification test program was in progress when reactor was shutdown in 1994.

FFTF MFF Metal Fuel Design

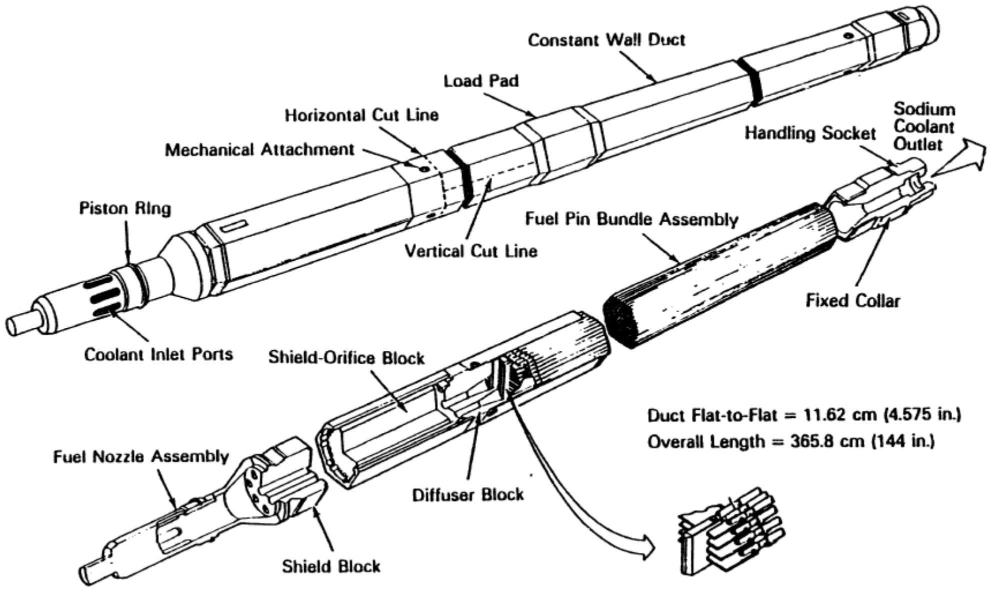


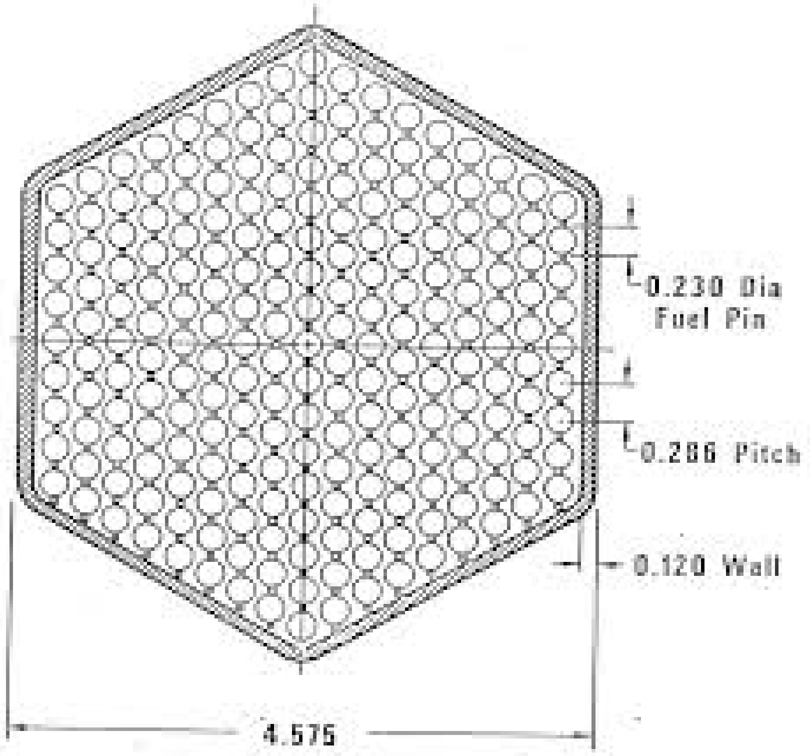


FFTF metallic fuel designs: 75% smeared density uranium-10wt% zirconium and uranium-plutonium-10wt% zirconium alloy, larger plenum, thicker cladding, D9 and HT9 cladding; burnups tested to 15 at%

Pitner, "Metal fuel test program in the FFTF," 1993

FFTF Fuel Assembly Design





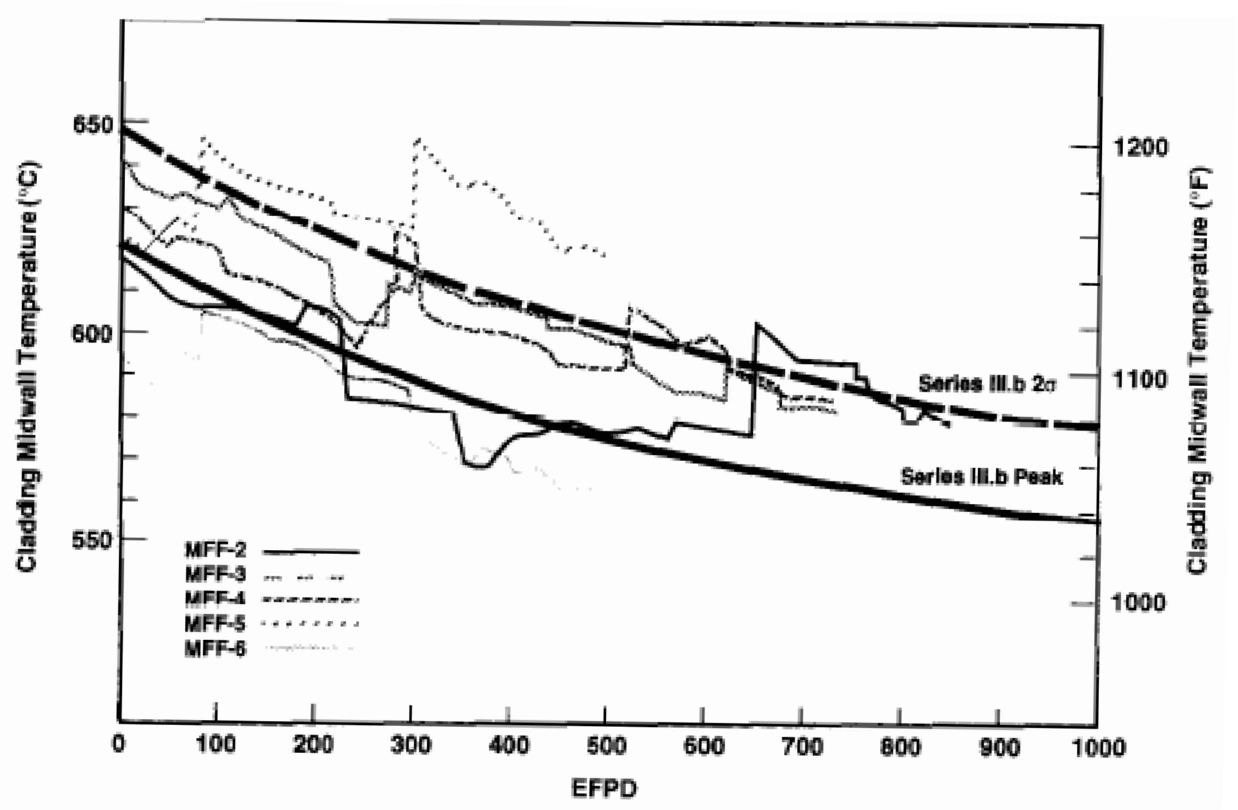
FFTF metallic fuel designs more relevant to commercial plant designs compared to EBR-II with larger diameter, longer length fuel pins and assembly with 169 fuel pins compared to 61 pins in EBR-II.

FFTF Metal Fuel Tests

Test	Fuel [a]	Cladding	Peak linear power (kW/m–kW/ft)	Peak clad temp. (°C–°F)	Exp. EFPD	Peak burnup (MWd/kgM)	Fast fluence (10^{22} n/cm^2)
IFR-1	B & T	D9	49.2–15.0	604–1120	620	94	15.4
MFF-1A	B	HT9	42.7–13.0	577–1070	250	38	5.6
MFF-1	B	HT9	43.0–13.1	577–1070	685	95	17.3
MFF-2	B	HT9	54.1–16.5	618–1145	853	143	19.9
MFF-3	B	HT9	59.1–18.0	643–1190	726	138	19.2
MFF-4	B	HT9	56.8–17.3	618–1145	726	135	19.0
MFF-5	B	HT9	55.8–17.0	649–1200	503	101	14.0
MFF-6	B	HT9	55.8–17.0	588–1090	503	95	12.8
ALMR (Design)	T	HT9	31.5– 9.6	545–1013	~ 1400	141	33.0

[a] B = binary (U–Zr), T = ternary (U–Pu–Zr).

- U-10Zr in all assemblies; U-Pu-10Zr in IFR-1
- HT9 cladding and IFR-1 with D9 cladding
- Common fuel design
  - fuel pin length 2.4 m (8 ft)
  - fuel column length of 91.4 cm (36 in.)
  - diameter of 6.86 mm (0.270 in.)
- Range of temperatures representing nominal to 2-sigma HCF (peak cladding at BOL, 577 – 649˚C)
- Peak burnups ~10-14 at.% at 500 – 850 EFPD



A.L. Pitner and R.B. Baker, “Metal fuel test program in the FFTF,” 204 (1993)

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FFTF Metal Fuel Fission Gas Release (Porter, 2012; Carmack, 2016)

Test ID	IFR-1	IFR-1	IFR-1	IFR-1	IFR-1	MFF-3	MFF-5
Pin number	181193	181165	181154	181180	181175	193045	195011
Fuel type	(19Pu)	(19Pu)	(8Pu)	(0Pu)	(0Pu)	U-10Zr	U-10Zr
Peak BU (at.%)	9.42	9.65	9.60	9.72	9.66	13.8	10.1
Avg. BU (at.%)	7.85	8.04	8.00	8.10	8.05	10.85	7.61
Fission gas generated(mol)	0.0209	0.0215	0.0213	0.0215	0.0215	0.0314	0.0233
Pin plenum volume (cm3)	20.1	20.6	19.8	20.3	19.4	21.7	24.8
Pin gas pressure (kPa)	1970	1940	2040	2050	2090	2223	1877
Kr + Xe collected (STPcm3)	336.9	333.8	339.0	348.8	334.2	425.6	412.16
Fission gas release (%)	72	69	71	73	69	61	79

FFTF metallic fuel gas release is generally consistent with that measured for short-pins (34.3-cm fuel column) for testing performed on fuels of similar composition in EBR-II.

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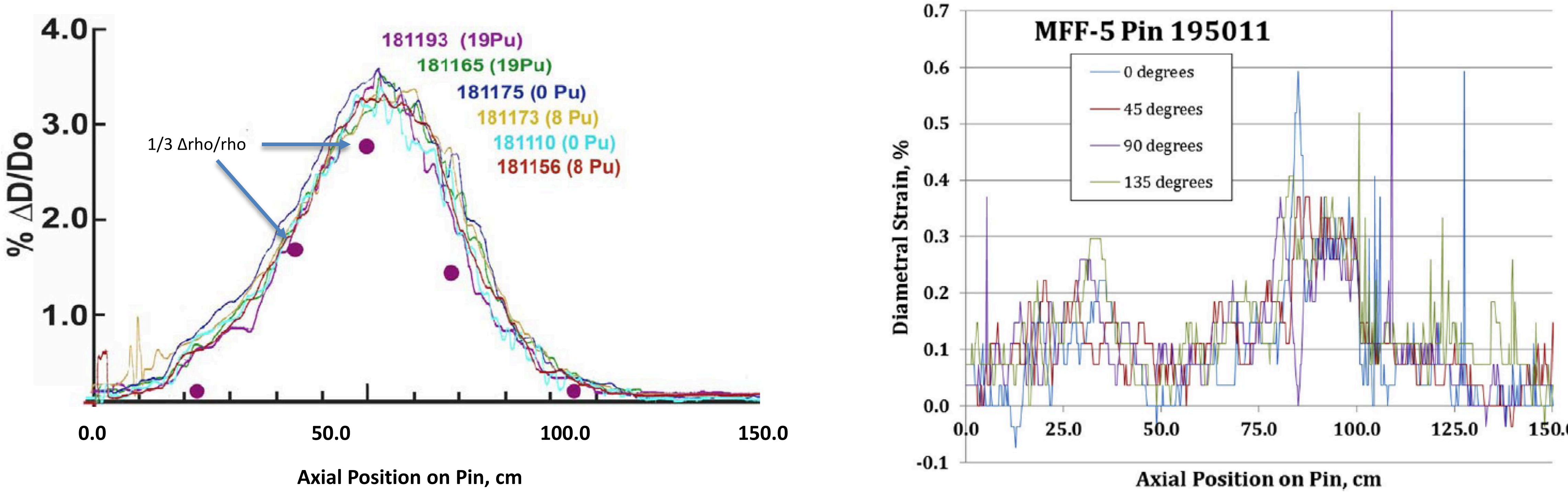
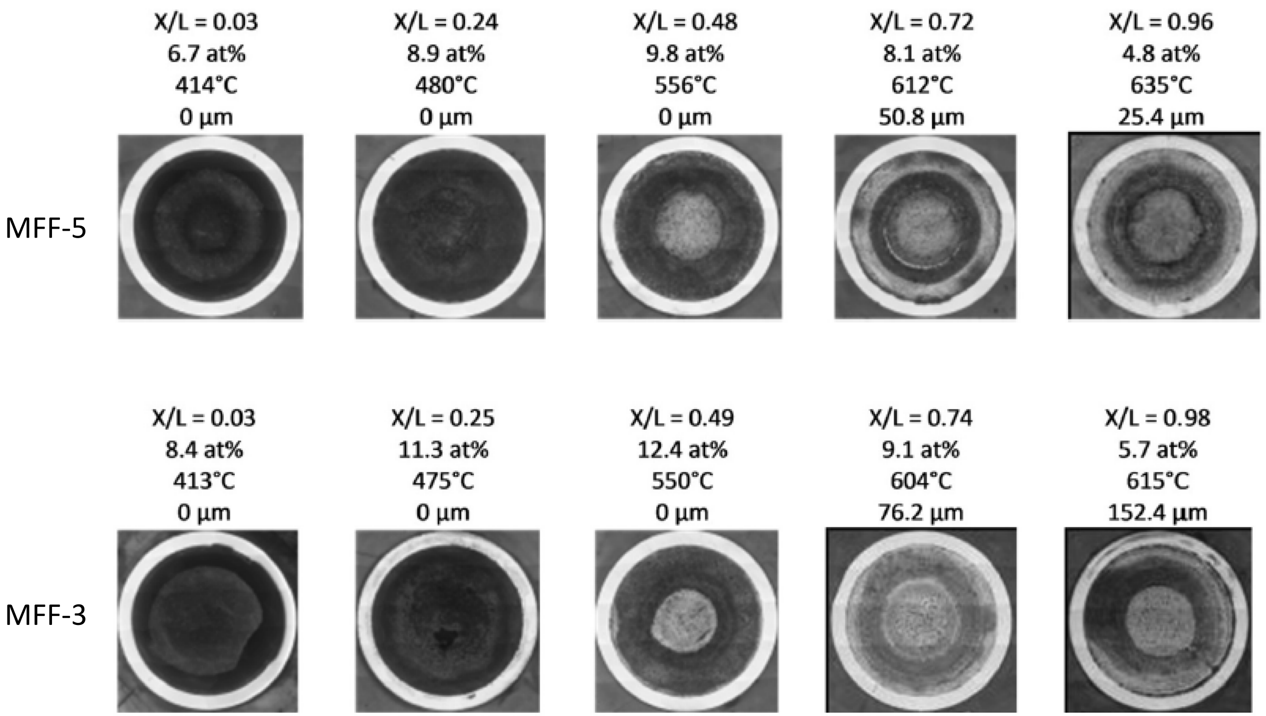
FFTF Metallic Fuel Diameter Axial Profiles for D9 (IFR-1) and HT9 (MFF-5) Cladding at 10at.% (Porter, 2011; Carmack, 2016)



FFTF IFR-1 showed similar profile for fuel alloys with different Pu consistent with short-pin (34.3-cm fuel column) testing performed on fuels of similar composition in EBR-II;
FFTF HT-9 (MFF-5) cladding diametral strain is 6-10X less than D9 (IFR-1); diametral strain attributed to HT9's much lower swelling (N.B. different y-axis scales).

FFTF Metallic Fuel FCCI of HT9 (MFF-5, MFF-3) Cladding at 10 and 14 at.% (Carmack, 2016)



- FFTF U-10Zr metallic fuel FCCI time-temperature data was consistent with short-pin (34.3-cm fuel column) testing performed on fuels of similar composition in EBR-II
- FCCI was not uniform around the circumference of the pin at each elevation
- For two FFTF pins (MFF-5, IFR-1), the peak FCCI axial location was at 0.7 L/Lo compared to near the top of the fuel column. In other fuel pins, the peak FCCI axial location was at ~0.9 L/Lo
- FCCI is strongly temperature dependent so, for EBR-II, it was always largest at the top of the fuel where cladding temperatures were highest

METALLIC FUEL RELIABILITY

EBR-II Metallic Fuel Reliability and Failure

- End of life reliability was assessed by run-to-failure tests for Mark IA, Mark II, and Mark III/IV designs
- Mark IA run-to-failure experiments resulted in 39 failures (Olson, 1976)
  - No significant effect of design variables or irradiation conditions
  - Characterized by intergranular crack propagation initiated on outside surface of cladding
  - A burnup limit of 3 at.% was determined by Weibull analysis
- EBR-II Mark II had 13 failures out of 30000 pins, primarily in the restrainer (dimple); removed from designs and eliminated this failure mode (Walters, 1980)
- In EBR-II there were 22 natural breached pins reported out of 13,600 U-Zr and U-Pu-Zr fuel pins (Crawford, 2007)
  - 19 of the breaches occurred in the in Mark III/IIIA/IV tests of X420, X421 and X447 (run to failure or very high temperature)
  - 2 natural breaches occurred in X429 and X453
  - 16 welds, three (3) in plenum of unknown cause and three (3) in fuel column region due to creep failure of the cladding

R.G. Pahl et al., 1990; R.G. Pahl et al., 1993; G.L. Batte' et al., 1990; D.C. Crawford et al., 2007

EBR-II Metallic Fuel Run Beyond Cladding Breach Tests

- Seven (7) run beyond cladding breach (RBCB) tests
  - U-Fs, U-Pu-Zr, and U-Zr were tested
  - The number of days irradiated after breach ranged from 34 to 233 days
  - The response behavior in all tests with a fuel region breach follows a similar pattern
    - initial breach is identified by both delayed-neutron (DN) and fission-gas signals
    - initial elevated DN signal will last around 10-20 minutes and is attributed to bond sodium expulsion
    - a significant fraction of the fission product cesium is released concomitantly with the sodium
    - at time of breach, almost all the released fission gas in the plenum and connected open porosity is vented to the primary system
    - the long-term release is exhibited by the monitored fission gas isotopes as they are produced during the RBCB operation
    - the venting of the released fission gas results in very little cladding stress after a breach
  - observed post-breach behavior was benign as shown by very little difference in fuel and cladding structure from intact fuel pins and little difference in breach size and features for the different post-breach operating time

Seidel et al., 1990; G.L. Batte' et al., 1990; G.L. Hofman et al., 1997; D.C. Crawford et al., 2007

METALLIC FUEL TRANSIENT OPERATION

Transient Fuel Performance: Key Phenomena

- Uranium-iron eutectic formation
- Cladding creep rupture
- Post-failure fuel behavior

The Transient Test Reactor (TREAT) at INL

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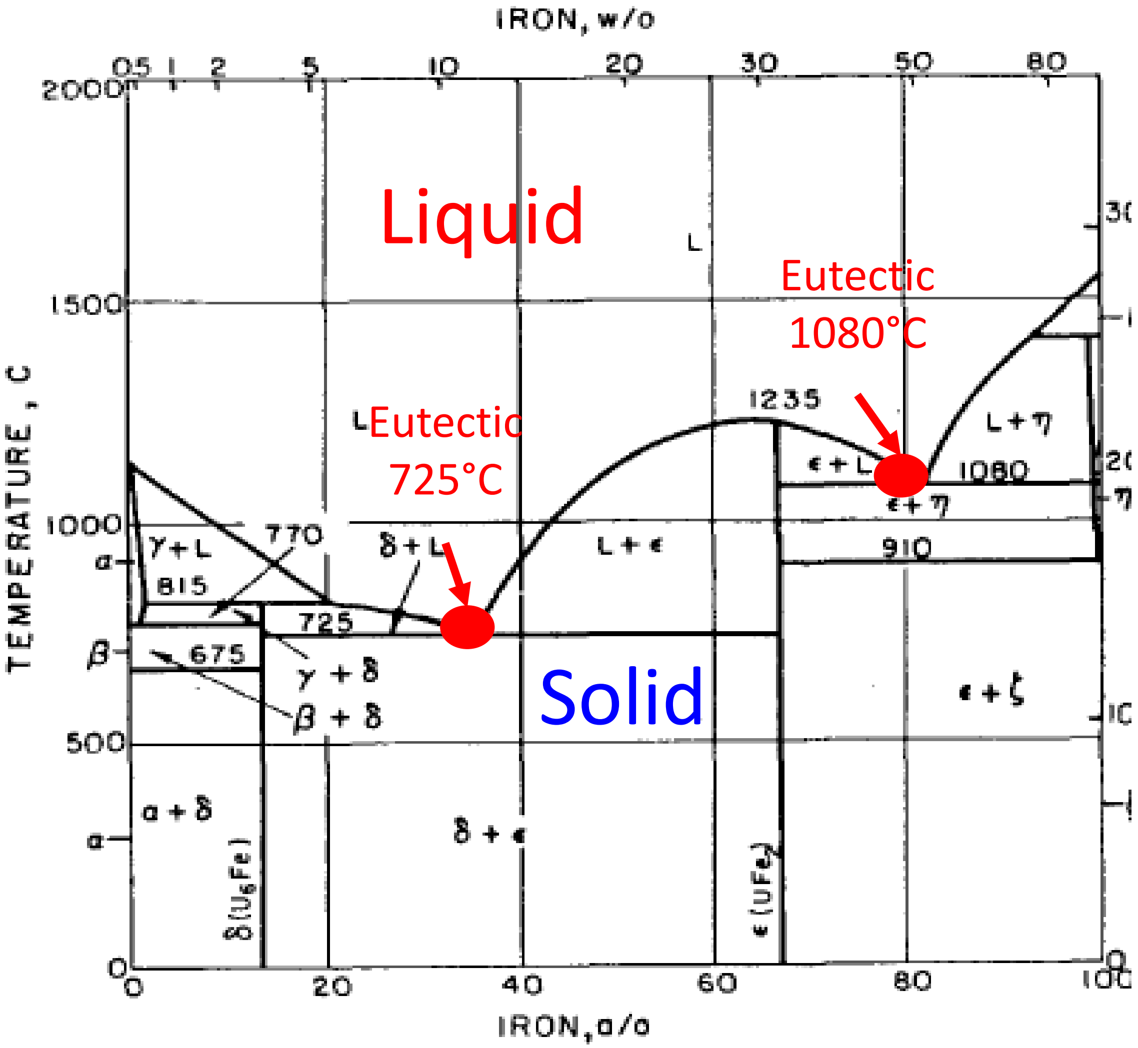
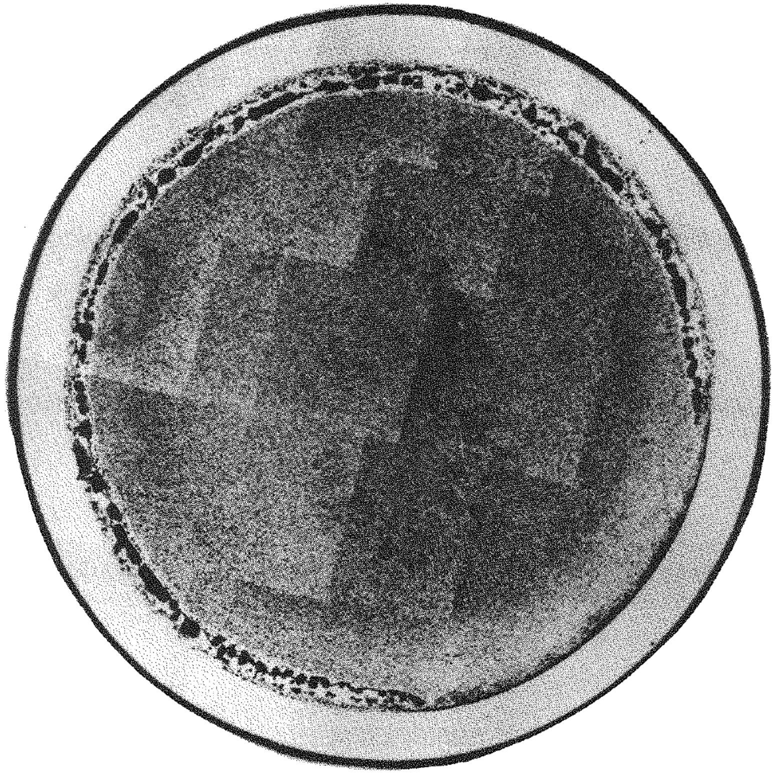
Eutectic Reactions – Uranium and Iron

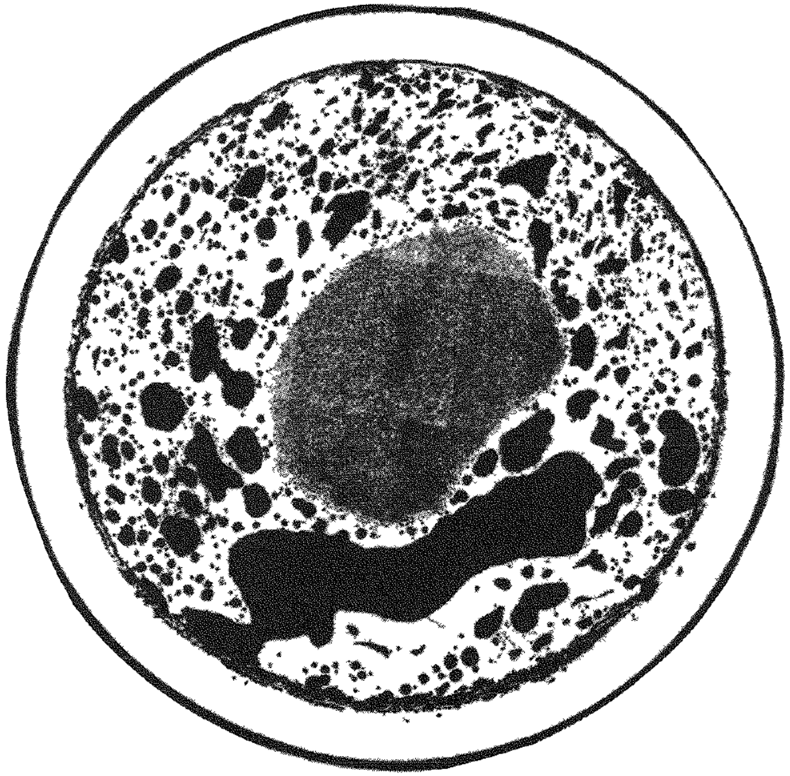
- Note that the U-Fe eutectic at 725°C is much lower than the melting point of uranium at 1132°C
- The reaction rate at 725°C is on the order of hours, so a time at temperature analysis is needed to assess damage
- The reaction rate at 1080°C is on the order of seconds



Eutectic Reactions (continued)



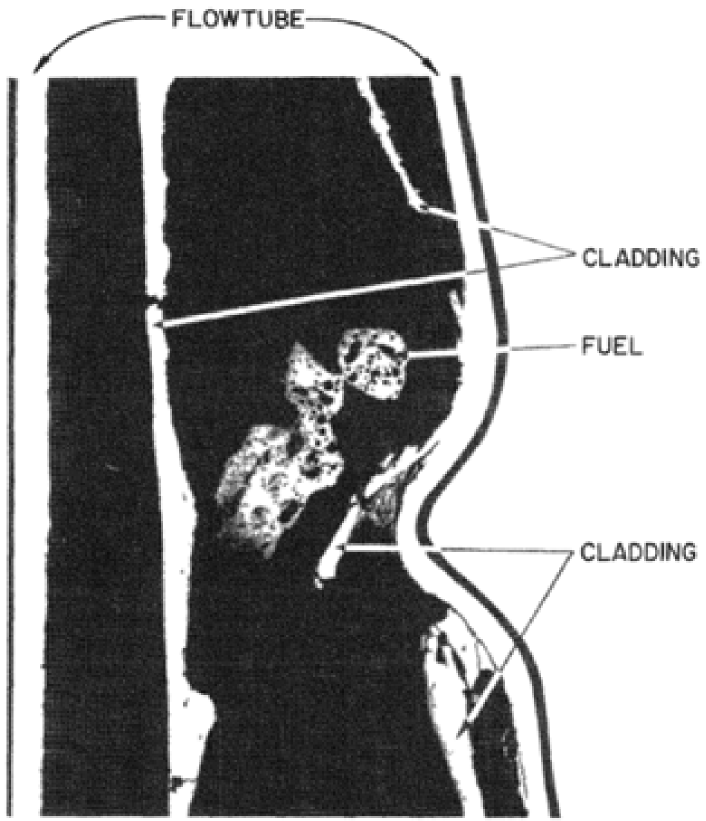
Results from isothermal furnace testing at 750°C for one hour



Results from isothermal furnace testing at 800°C for one hour

Given sufficient time at temperature, the formation of low melting point eutectics can result in cladding penetration; eutectic formation is generally the limiting failure mechanism at low burnups.

Cladding Rupture and Post-Failure Fuel Behavior



Cladding rupture after extensive deformation in TREAT

Partial flow blockage from fuel that had been expelled from the cladding and subsequently re-solidified

Cladding creep rupture is limiting at higher burnups; understanding post-failure behavior is important to ensure coolability under severe beyond-design-basis events (i.e., ~4X nominal power).

EBR-II Transient Testing

- Operational Reliability Tests
  - 56 low-ramp-rate transients (1.6% power increase per second)
  - 13 high-ramp-rate tests (4 MW/s)
- Shutdown Heat Removal Test (SHRT) program
  - Supported by fully-instrumented and calibrated in-core fueled and non-fueled assemblies (XX07, XX08, XX09 and XX10)
  - Qualify the fuel for transient operation by high-temperature irradiation test assembly irradiated in EBR-II for 42 min with cladding temperatures reaching as high as 800˚C
  - Demonstrated metal fueled core can safely operate with loss-of-flow-without-scram (LOFWS) events and loss-of-heat-sink-without-scram events (LOHSWS) with no core damage

EBR-II Metallic Fuel Overpower Transient Testing-TREAT

- Various combinations of fuel and cladding alloys were evaluated at TREAT to determine transient-overpower margin to failure, pre-failure axial fuel expansion, and post-failure fuel and coolant behavior.
- 15 pins of various combinations of fuel (U-Fs, U-Zr, U-Pu-Zr) with cladding (SS316, D9, HT9).
- Metal fuel failed at 4-4.5 times nominal peak power under the relatively fast transient conditions used in the tests. Oxide fuel tested under the same conditions failed at just 2.5-3 times nominal peak power, demonstrating the larger margin to cladding failure for metal fuel.

EBR-II Transient Testing – TREAT (after Crawford, 2007)

Test ID	Fuel Type	Cladding Type	Fuel Design	Burnup (at. %)	Posttest Cladding Condition	Fuel Length Increase(%)
M2	U–5Fs	SS316	Mark-II	0.3	Intact	16
	U–5Fs	SS316	Mark-II	4.4	Failed	10-15
	U–5Fs	SS316	Mark-II	7.9	Failed	3
M3	U–5Fs	SS316	Mark-II	0.3	Intact	18
	U–5Fs	SS316	Mark-II	4.4	Intact	4
	U–5Fs	SS316	Mark-II	7.9	Intact	4
M4	U–5Fs	SS316	Mark-II	Fresh	Intact	1
	U–5Fs	SS316	Mark-II	2.4	Failed	8
	U–5Fs	SS316	Mark-II	4.4	Intact	4

9 of the 15 total transient overpower tests of metallic fuel were U-5Fs Mark II fuel design. Pre-failure axial expansions were large (around 15-20 %) in low-burnup fuel but decreased rapidly to ~ 4 % with increasing fuel burnup.

EBR-II Transient Testing – TREAT (after Crawford, 2007)

Test ID	Fuel Type	Cladding Type	Fuel Design	Burnup (at. %)	Post-test Cladding Condition	Fuel Length Increase(%)
M5	U–19Pu–10Zr	D9	X419,420,421	0.8	Intact	-0.8, -0.4
	U–19Pu–10Zr	D9	X419,420,421	1.9	Intact	0.0, 2.5
M6	U–19Pu–10Zr	D9	X419,420,421	1.9	Intact	3.0
	U–19Pu–10Zr	D9	X419,420,421	5.3	Failed	NA
M7	U–19Pu–10Zr	D9	X419,420,421	9.8	Failed	NA
	U–10Zr	HT9	X425	2.9	Intact	3.7

A total of 15 transient overpower tests of metallic fuel were performed and show similar trends. The differences between alloys and samples is largely explained by retained fission gas and melting temperature.

T.H. Bauer, A.E. Wright, W.R. Robinson, J.W. Holland, and E.A. Rhodes, “Behavior of Modern Metallic Fuel in TREAT Transient Overpower Tests,” 92 (1990)

EBR-II Metallic Fuel Overpower Transient Testing-TREAT

- All fuel rod breaches in the metal fuel
  - located at the top of the fuel column.
  - due to cladding rupture induced by at-temperature pin-plenum pressure and cladding thinning due to eutectic-like formation of a molten fuel/cladding phase that penetrated the cladding wall.
- Mitigating behavior of metallic fuel
  - Axial fuel expansion that occurs pre-failure has the benefit of adding negative reactivity to the core during an overpower transient for the zirconium-alloyed fuels (similar effects were observed with higher-burnup U-5Fs fuel); this effect has no parallel in oxide fuel.
  - Rapid fuel dispersal (i.e., half of fuel inventory being ejected from the fuel rod) that occurred post-failure was also found to beneficially add negative reactivity to the core.

Summary

1) Early designs focused on breeding but were limited in burnups due to high-smear density and thin cladding
2) Reducing smear density and providing plenum to accumulate fission gas dramatically increased achievable burnup
3) Progression of different cladding materials leading to reduced cladding strains with FM steels providing the best performance
4) Broad range of fuel alloys successfully irradiated with quite similar fuel behaviors (FGR, axial growth, radial swelling)
5) Key fuel performance phenomena are well understood
6) Excellent reliability with many thousands of pins irradiated over decades with a broad number of designs and experiments
7) Excellent transient performance showing more margin than oxide SFR fuels and good compatibility of fuel and coolant in the event of a breach

SUBJECT TO DOE COOPERATIVE AGREEMENT NO. DE-NE0009054

Acronyms

ASME – American Society of Mechanical Engineers
BU – burnup
CW – cold worked
DFR - Dounreay Fast Reactor
DN – delayed neutron
EBR - Experimental Breeder Reactor
FCCI – fuel-cladding chemical interaction
FFTF – Fast Flux Test Facility
FGR – fission gas release
IFR – Integrated Fast Reactor
LOFWS – loss of flow without scram
LOHSWS – loss of heat sink without scram
LTA – lead test assembly
MFF – mechanical fuel failure
RBCB – run beyond cladding breach
SFR - sodium-cooled fast reactor
SHRT – shutdown heat removal test
TREAT - Transient Reactor Test

Questions?