Post irradiation examination of Fuel and Core Structural Materials irradiated in FBTR

N G Muralidharan, C.N.Venkiteswaran, V.Karthik, V.Anandraj, Jojo Joseph,
K.V.Kasiviswanathan, S.Venugopal, T.Jayakumar, Baldev Raj
Indira Gandhi Centre for Atomic Research, Kalpakkam, India

IAEA TECHNICAL MEETING ON HOT CELL POST IRRADIATION EXAMINATION & POOLSIDE INSPECTION, May 23-27, 2011.

SCOPE OF PRESENTATION

PIE facility in IGCAR

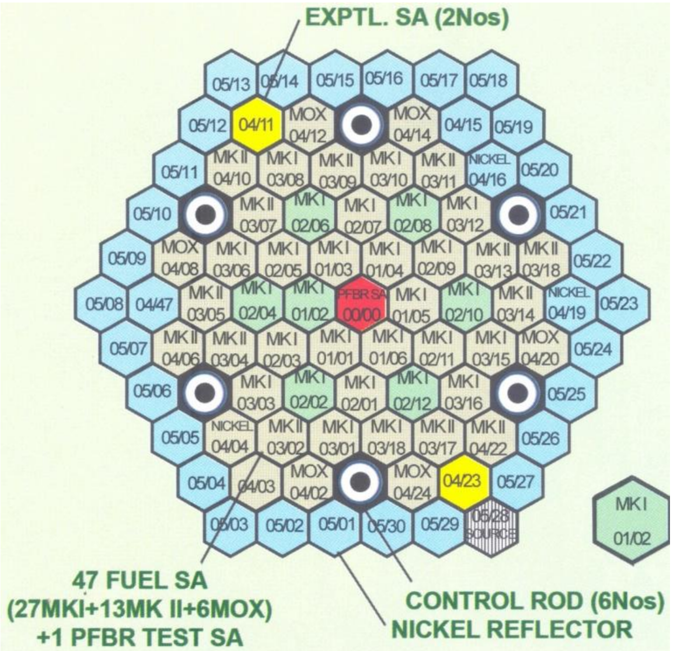
Performance Assessment of

- **Mixed carbide fuel of FBTR**
- **Control Rod subassembly**
- **Nickel reflector subassembly**

Experimental Irradiations in FBTR

- **PFBR MOX fuel**
- **FBTR Grid Plate material**

40 MWt Fast Breeder Test Reactor - FBTR



Full core
65 Fuel subassemblies

- ♦ Indigenous development of Fast Breeder Reactor Technology
- ♦ Test bed for fuel & Structural materials irradiation
- ♦ Development of FBR Fuel cycle technology

Reactor Core

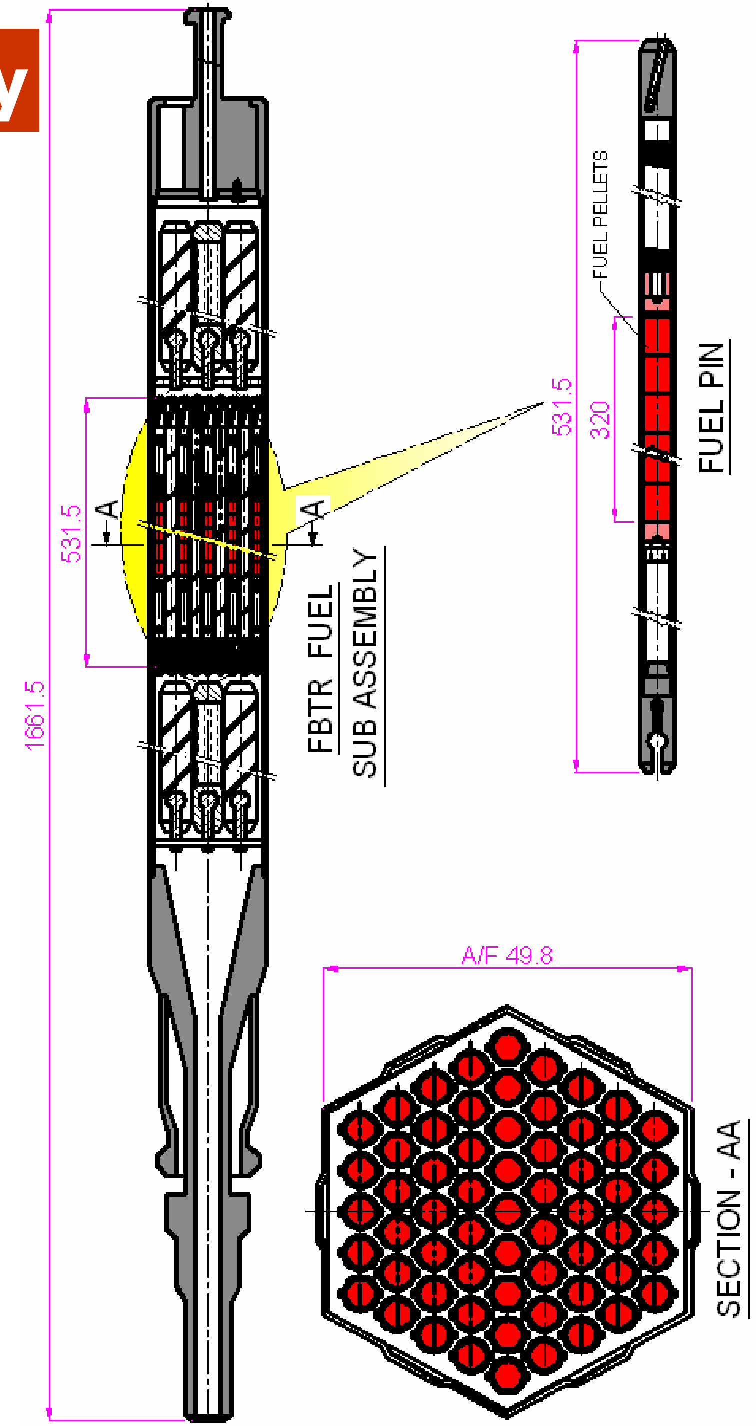
- ✓ <u>Mixed Carbide:</u> Mark I - $(U_{0.3}Pu_{0.7})C$ & Mark II $(U_{0.45}Pu_{0.55})C$
- ✓ Hybrid fuel in expanded core – 27 Mark I, 13 Mark II and 8 nos. of high Pu MOX subassemblies (44% PuO_2- 56% UO_2)- inducted into the core in 2007
- ✓ 20 % cold worked ASS 316 as wrapper & clad material

Peak fast neutron flux
~ 2.3×10^{15} n $cm^{-2}s^{-1}$ (> 0.1 MeV)
Peak power attained– 18 MWt ,3.2 MWe
Inlet Sodium Temperature ~ 380 C
Outlet Sodium Temperature ~ 480 C

- ❖ **Mark – I Fuel has crossed a burn-up of 165 GWd/t**
- ❖ **PFBR MOX fuel $(U_{0.72}Pu_{0.28})O_2$ Attained 112 GWd/t Burn-up & PIE in progress**

Details of FBTR Fuel Subassembly

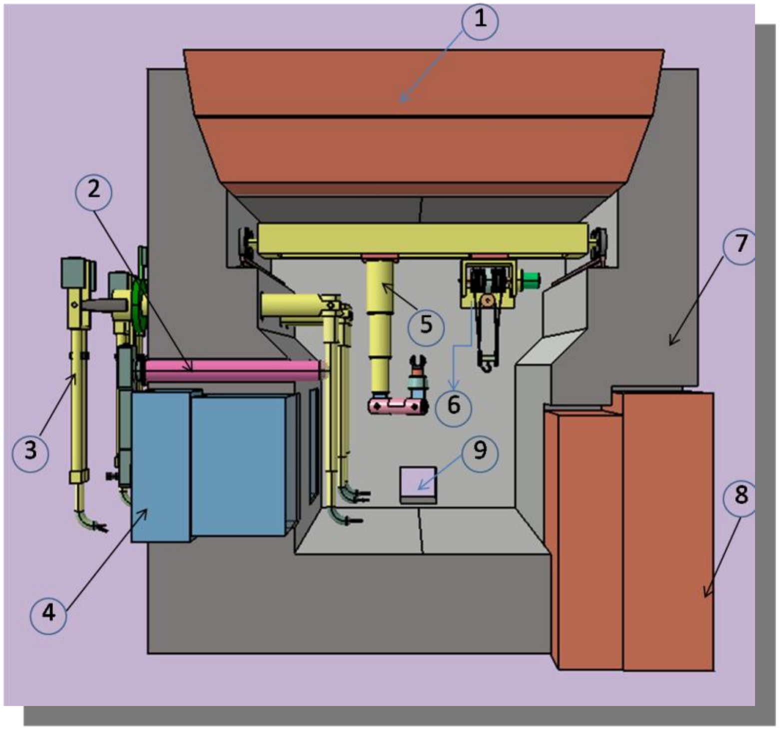
No. of fuel pins in a FSA	61
Fuel composition	$(U_{0.3}Pu_{0.7})C$
M_2C_3 content	5 – 12 %
Fuel stack length	320 mm
Type of bond	Helium
Fuel Density	90 % of T.D
Smear density	83 % of T.D
Pellet diameter	4.18 mm
OD of fuel pin	5.1 mm
ID of fuel pin	4.36 mm
Clad & wrapper material	20 % CW 316 stainless steel
Irradiation details	
Peak linear power	400 W/cm
Peak Burnup	165 GWdt
Maximum 'dpa'	83

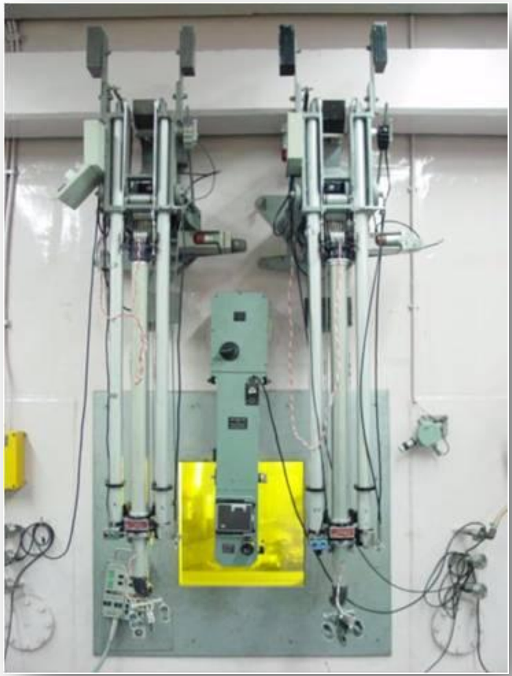


HOT CELL FACILITY FOR PIE

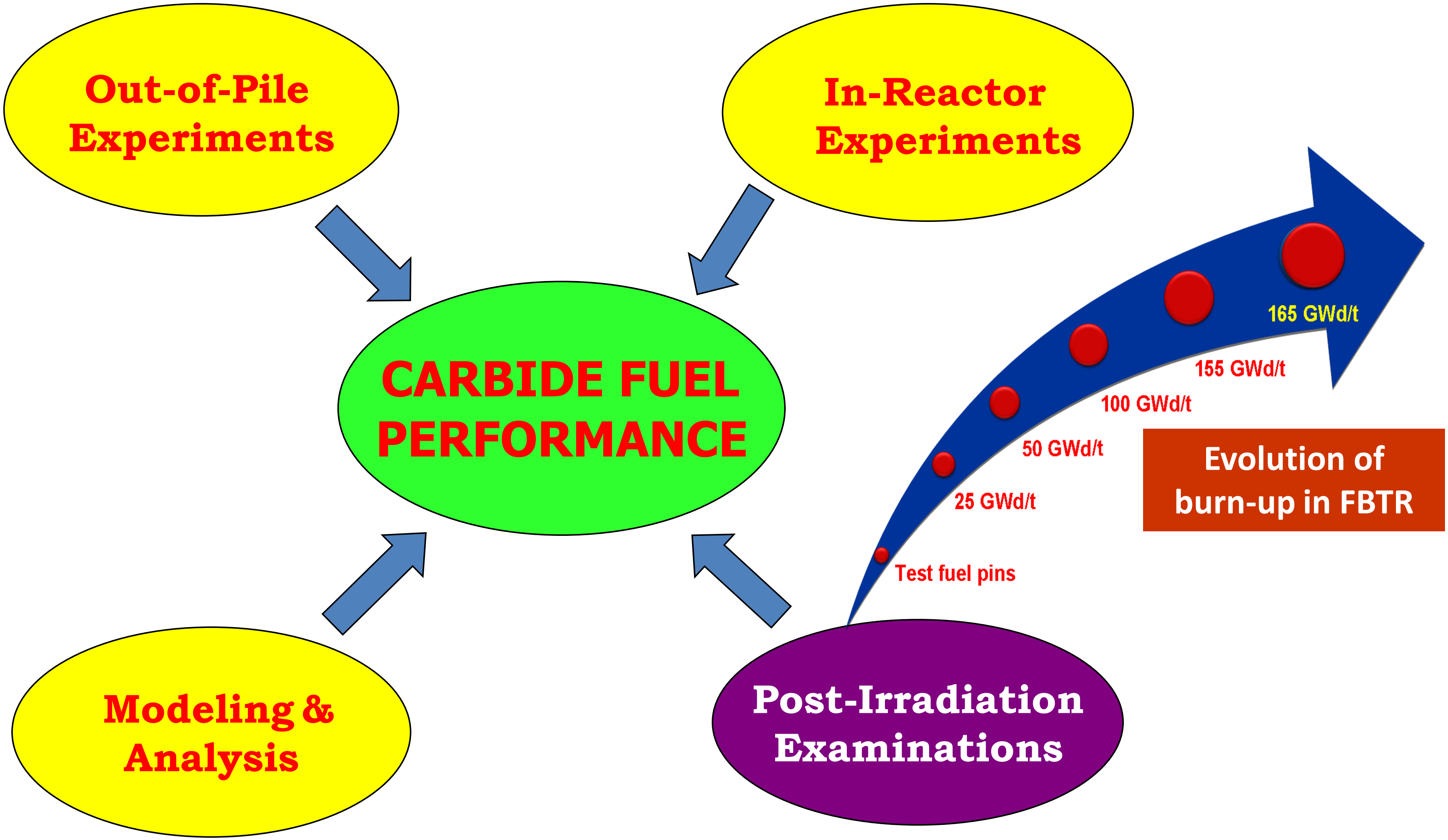
FEATURES OF RML HOT CELLS

No. of Concrete cells	- 7
No. of Lead cells	- 2
Type	- α, β, γ
Atmosphere	- Nitrogen
Max. activity	- 1.5×10^5 Ci (1 MeV)

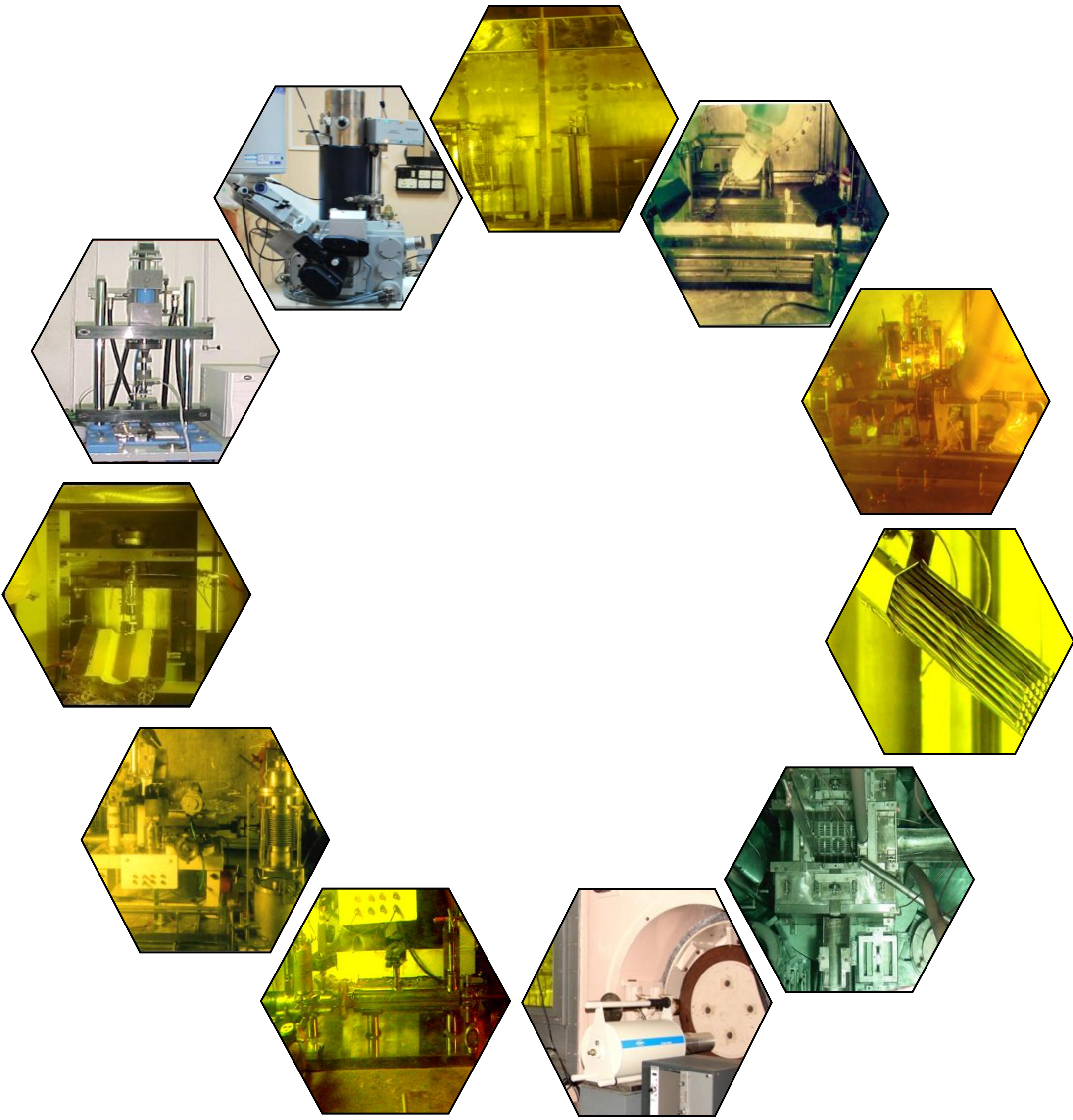




Performance assessment & burn-up enhancement of FBTR carbide Fuel



PIE techniques and facilities



Assessment of Fuel Performance

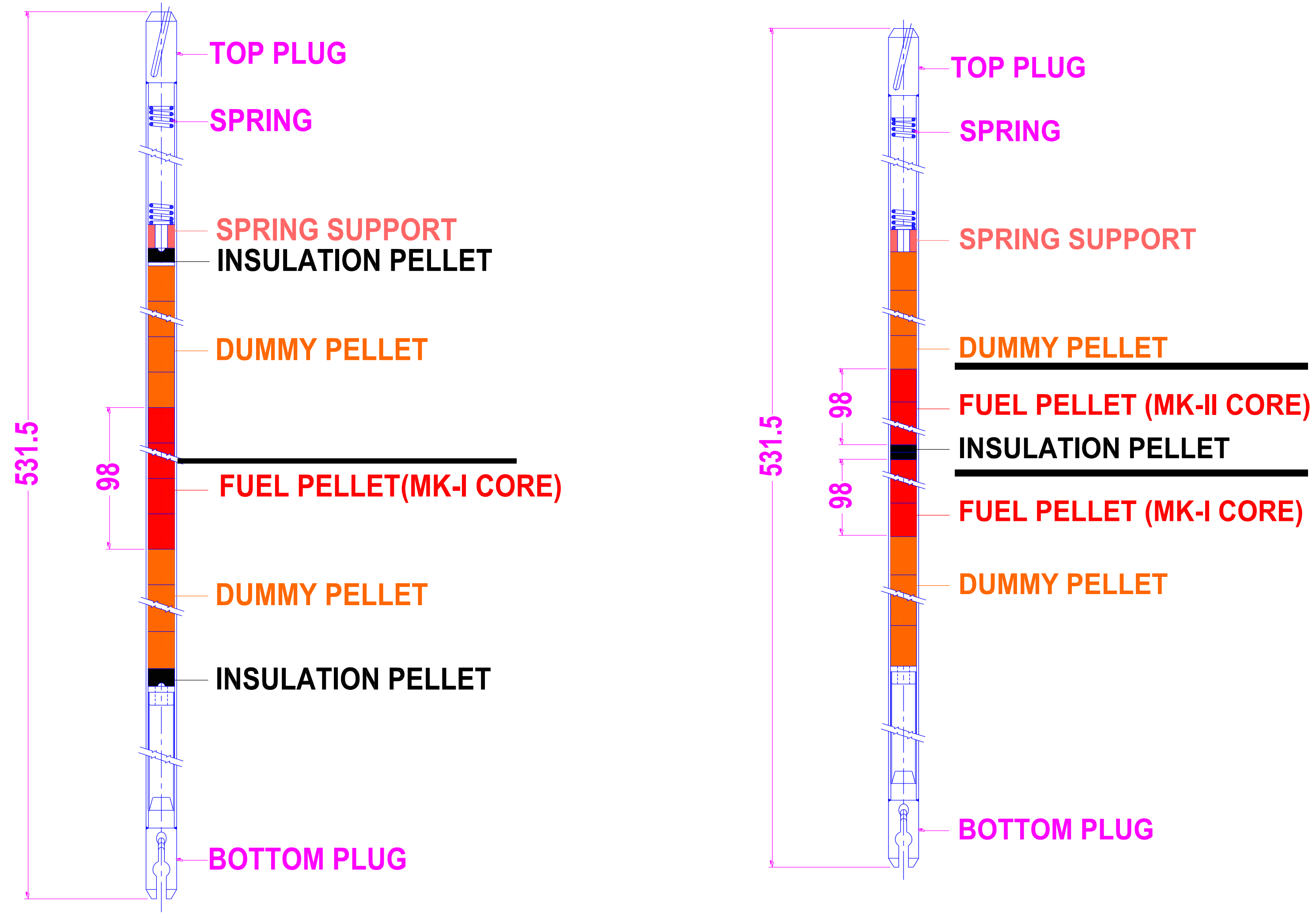
- X – Radiography
- Neutron Radiography
- Ceramography
- Fission Gas extraction & Analysis
- Gamma scanning

Assessment of Structural material

- Metrology
- Eddy current testing
- Void Swelling estimation
- Tensile Properties
- Small specimen tests
- Microscopy

Experimental fuel pins

AIM: To Understand the "beginning-of-life performance" of $(U_{0.3}Pu_{0.7})C$ & $(U_{0.45}Pu_{0.55})C$ compositions

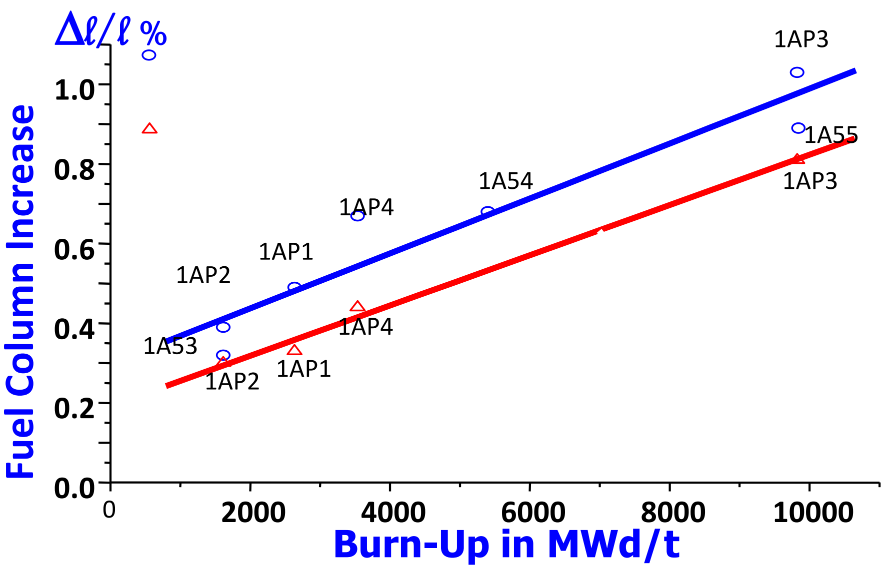


Experimental fuel pin Type-A **Experimental fuel pin Type-B**

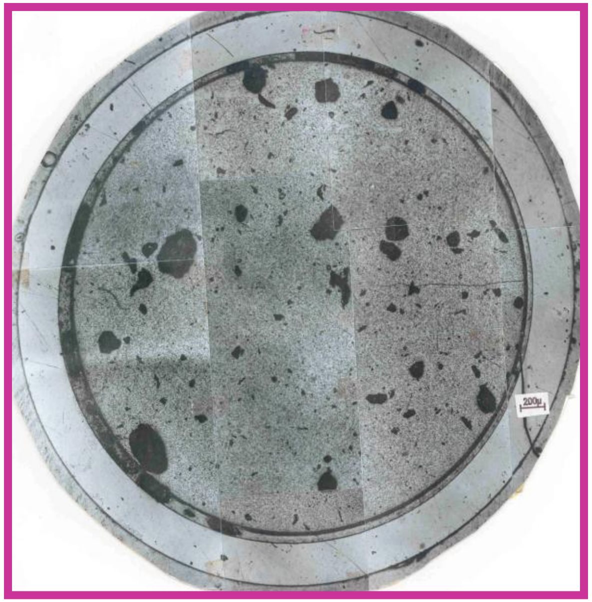
BEGINNING-OF-LIFE PERFORMANCE ASSESSMENT OF CARBIDE FUEL

EXAMINATION OF SEVEN EXPERIMENTAL FUEL PINS (1500 – 10000 MWd/t BU)

- **Fuel Crack Initiation at Low Burn-up of ~1600 MWd/t (16 EFPD)**
- **Reduction in The Fuel-clad Gap**
- **Amenability to Higher Linear Power for Fresh Fuel**
- **Reduction in the Swelling Rate For Mark - II Fuel**

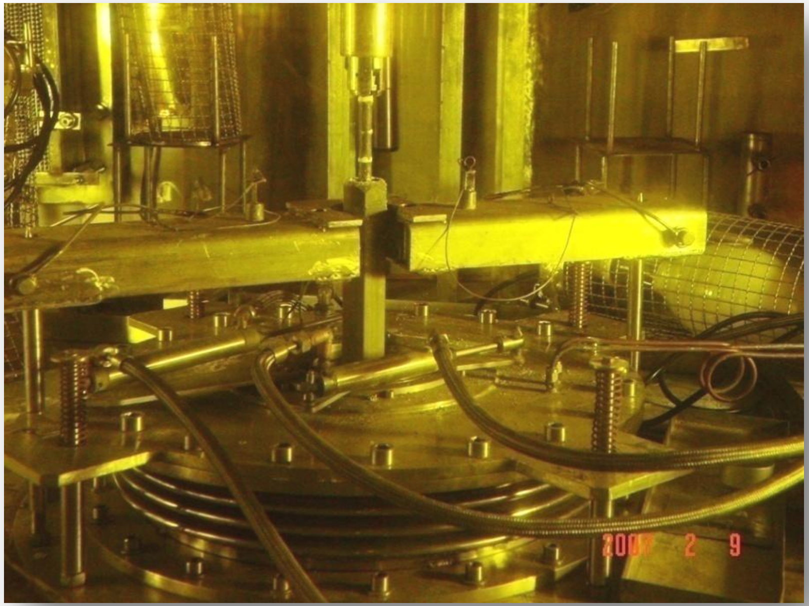


Comparison of swelling of Mark I & Mark II fuel

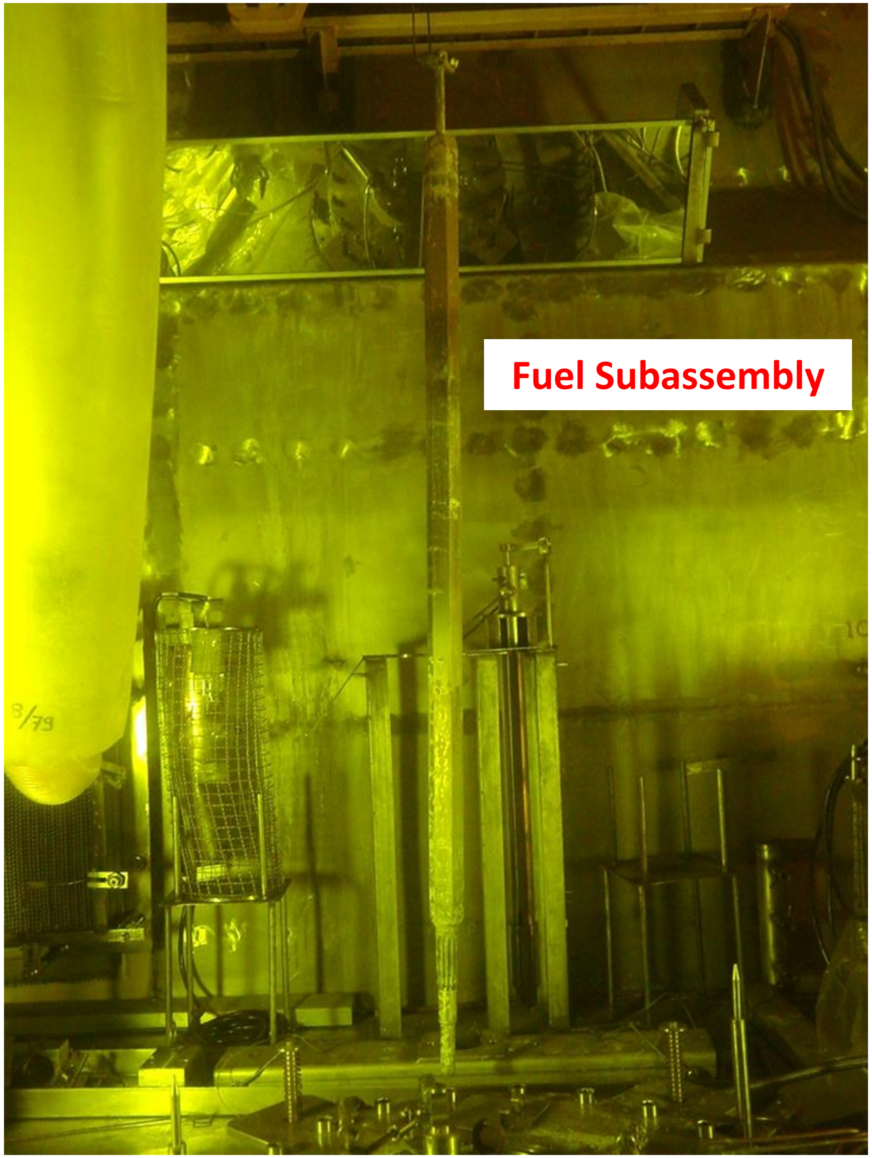


Photomosaic of fuel cross section after 16 EFPD *(MARK-I)*

Visual Examination of 155 GWd/t Fuel Subassembly and Fuel pin Bundle

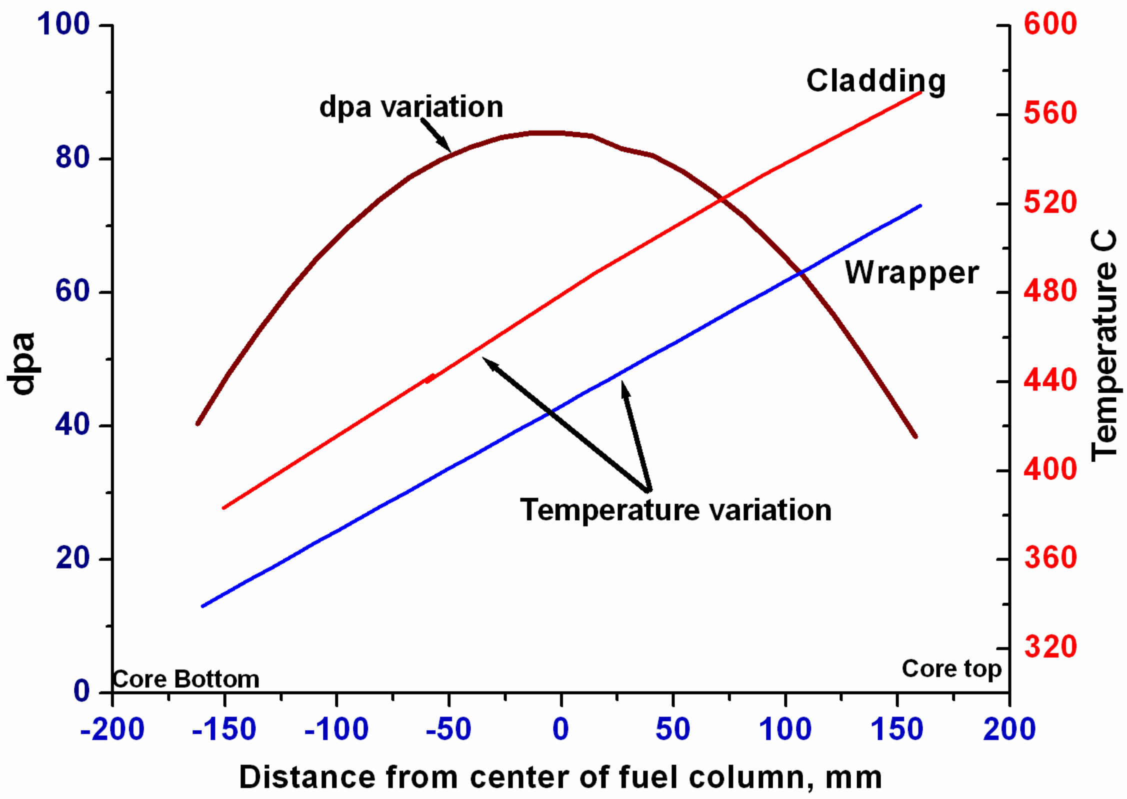


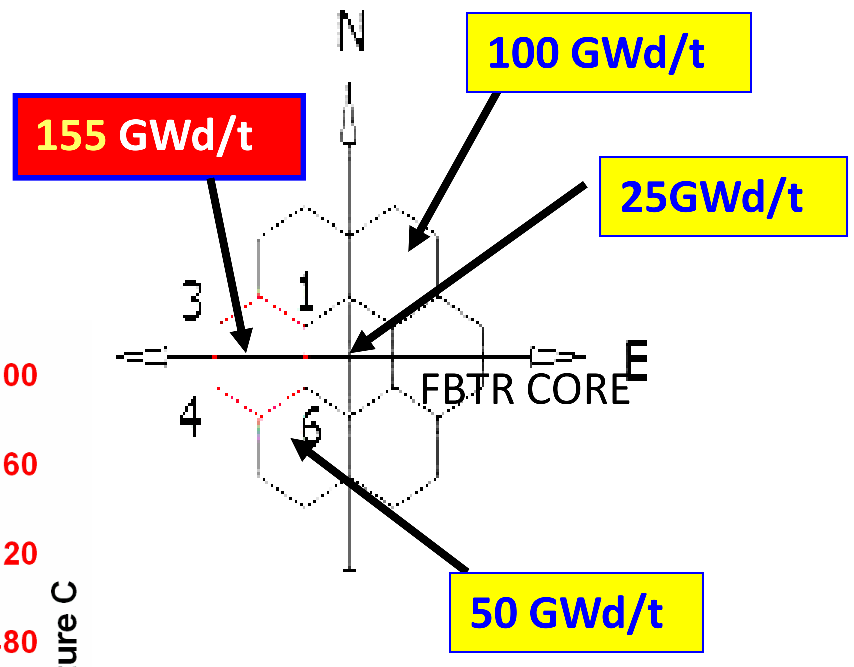
Receipt of fuel into hot cell



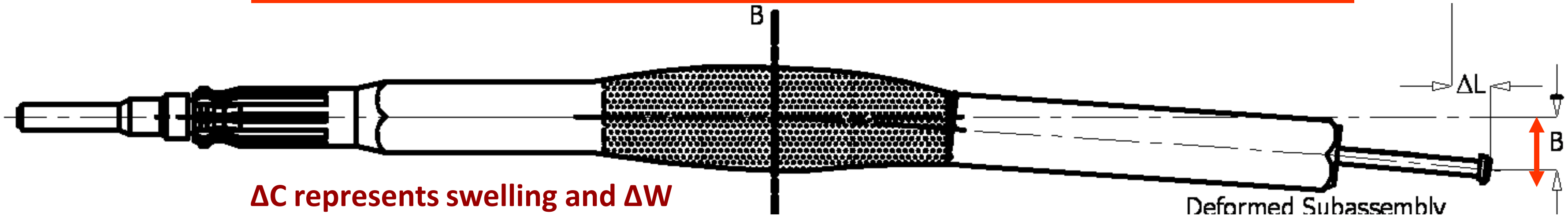
Irradiation temperature & 'dpa' along the FSA

Burnup (peak)	Fluence (peak)	dpa (peak)
100 GWd/t	0.9×10^{23} n/cm²	56
155 GWd/t	1.3×10^{23} n/cm²	83

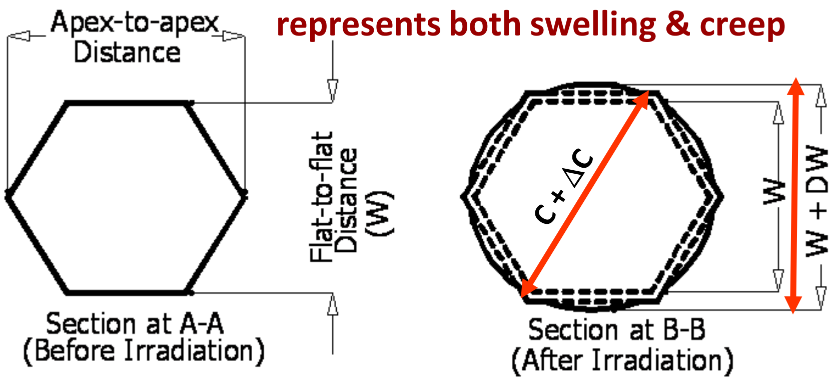


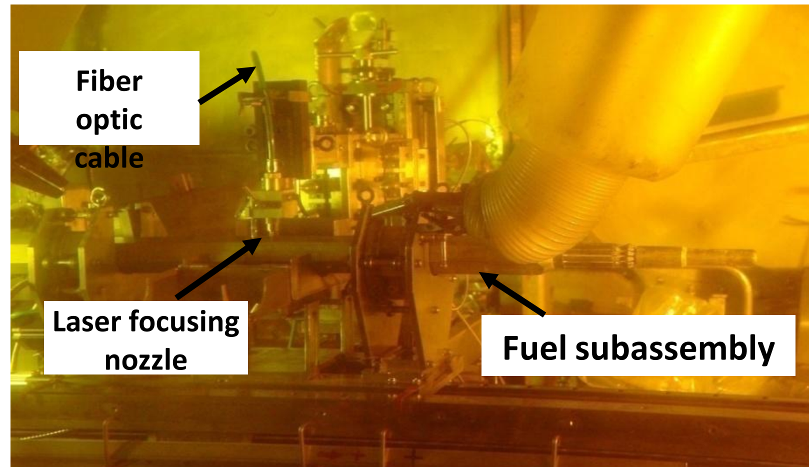


DIMENSIONAL CHANGES IN WRAPPER

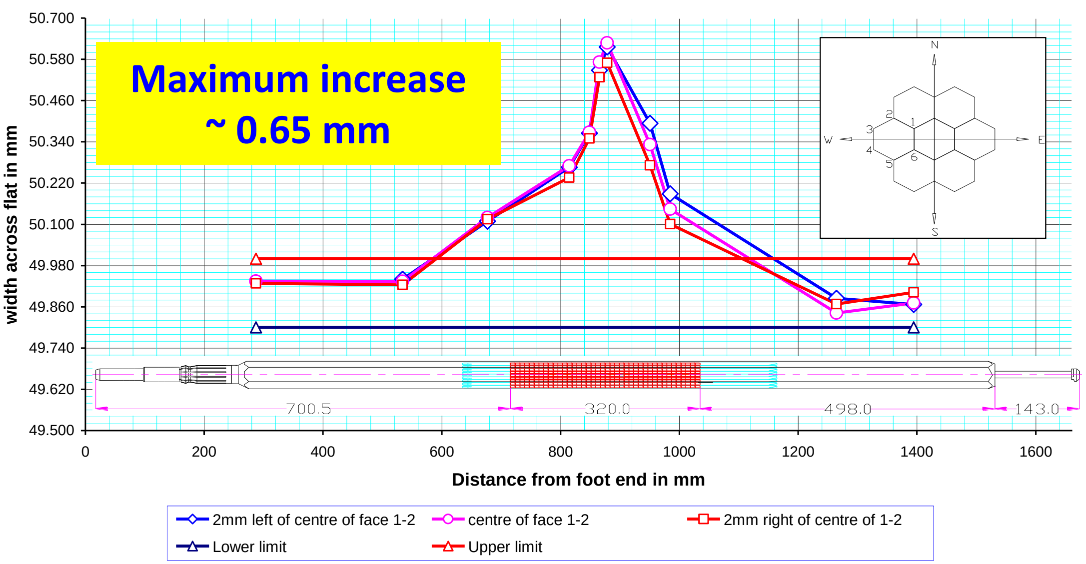


ΔC represents swelling and ΔW represents both swelling & creep



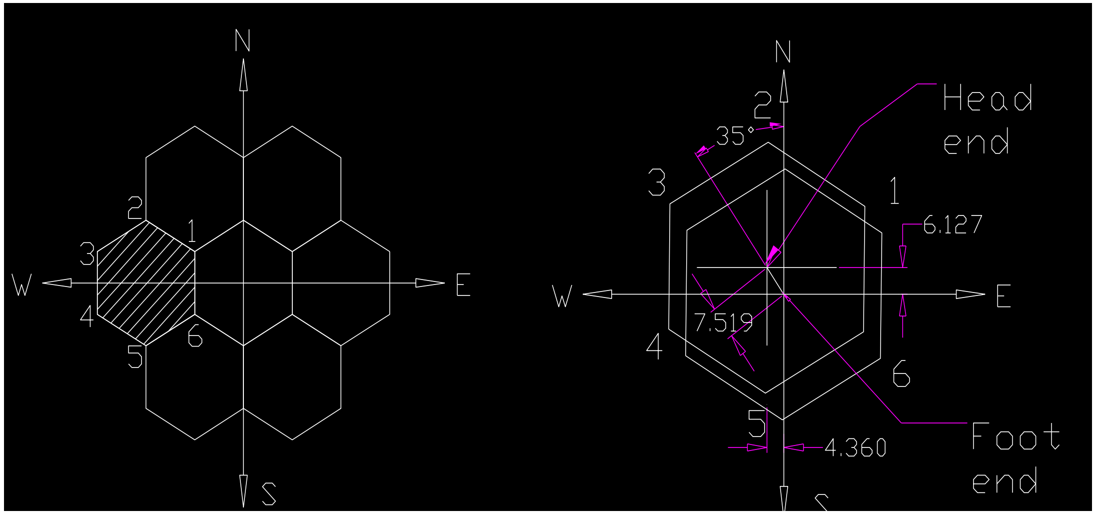


Computer Controlled Automated Profilometer & dismantling machine



Variation of width across flat along the length of FSA

DIMENSIONAL CHANGES IN WRAPPER



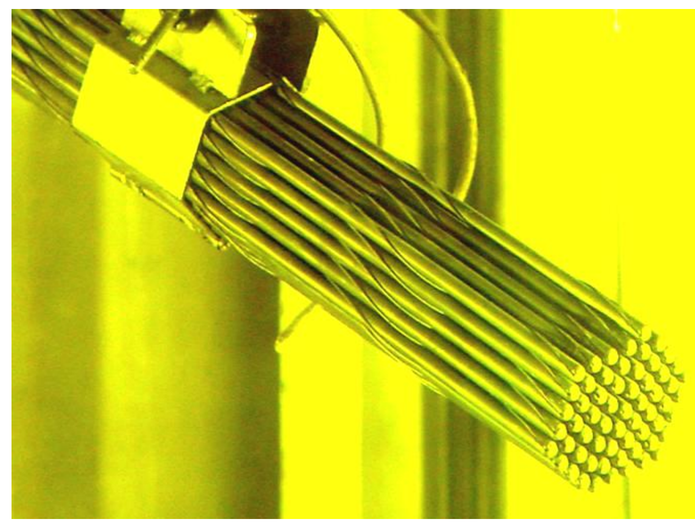
Burn-up GWd/t	Head- to-foot misalignment (Bowing)
25	--
50	0.8 mm
100	4.3 mm
155	7.5 mm

DISMANTLING OF FUEL SUBASSEMBLY

PULSED Nd-YAG LASER SYSTEM

Laser power	150 watts (av.)
Maximum pulse energy	50 J
Laser spot Diameter	0.4mm

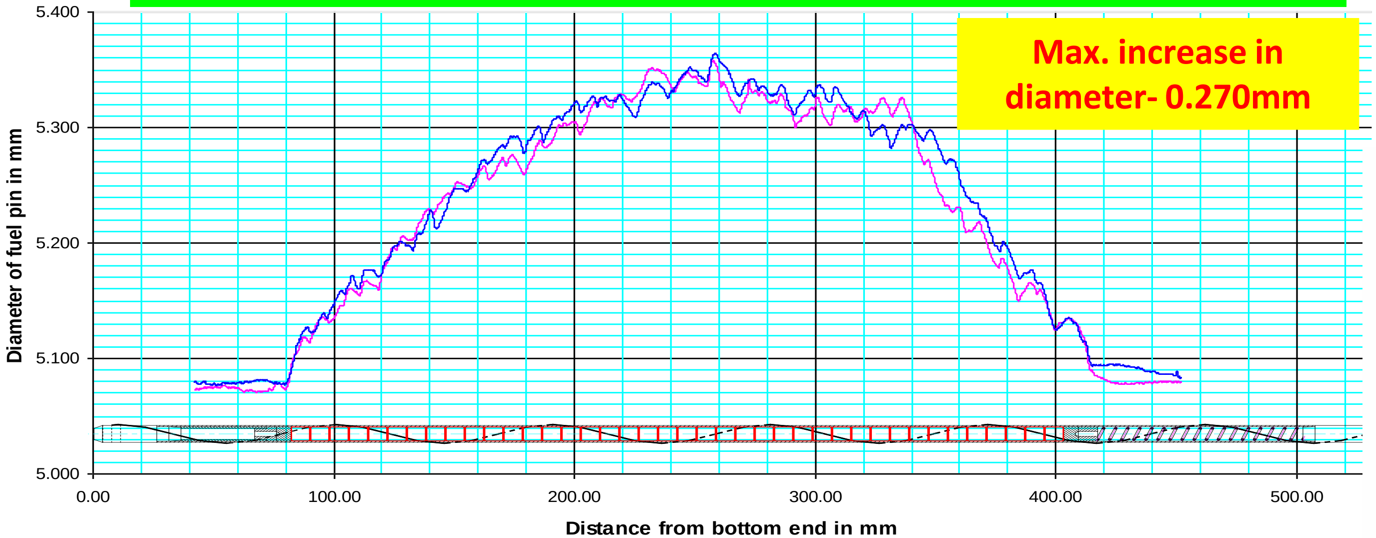
Nitrogen as purge gas

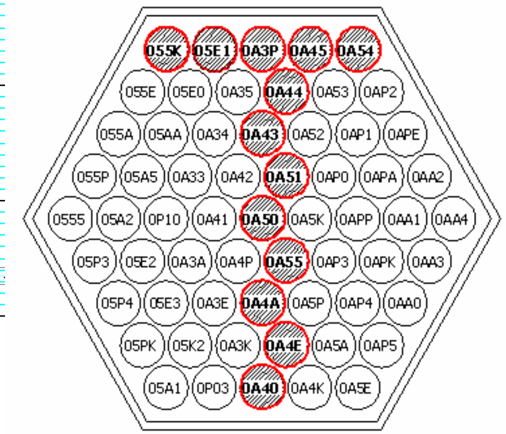


No Waviness in Fuel pin bundle indicating Absence of Bundle-to-Duct Interaction

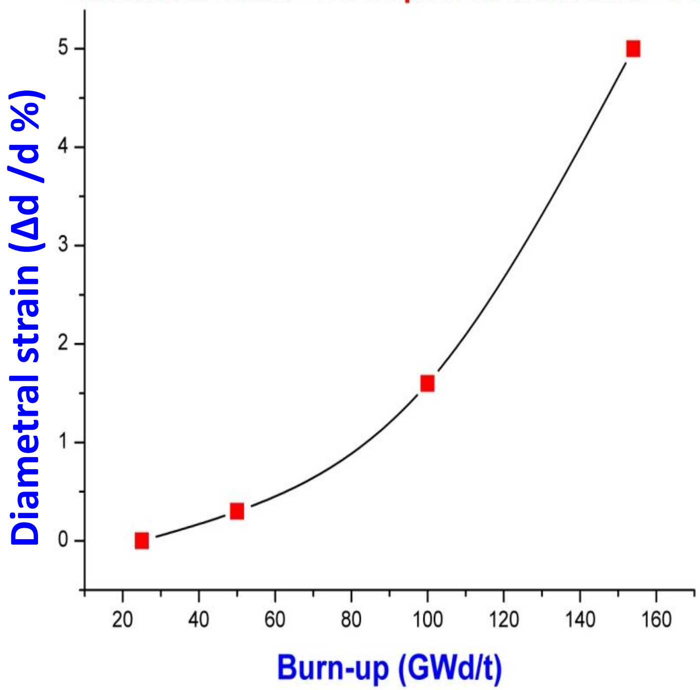
DIAMETER & LENGTH MEASUREMENTS OF FUEL PINS

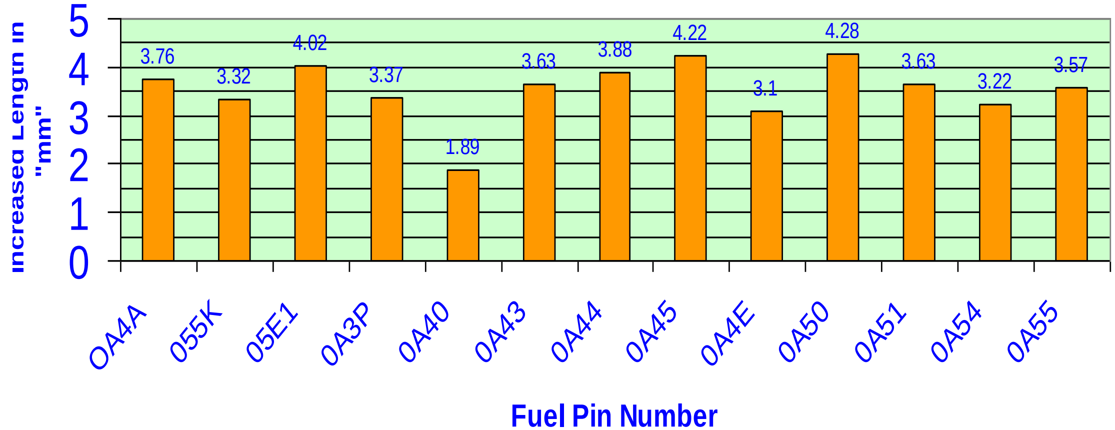
Typical Diameter Profile of a 155 GWd/t burn-up fuel pin





Diametral strain on fuel pins as a function of burn-up





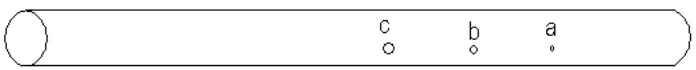
Length increase in fuel pins

EDDY CURRENT EXAMINATION OF CLAD TUBES

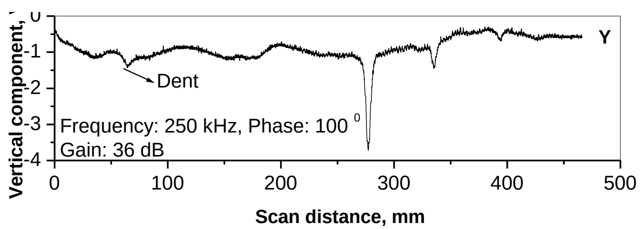
- Single Frequency Eddy Current Instrument & Encircling Eddy Current Probe
- Calibration Pin with Reference Defects of 0.351, 0.47 & 1.04 mm Diameter Hole
- Computerized Data Acquisition & Data Analysis

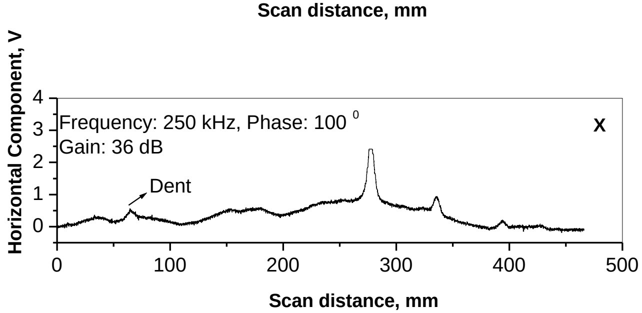
Optimized Test Parameters

Test frequency	250 kHz
Gain	36 dB
Reference phase angle	100 °
Inspection speed	2 mm/s



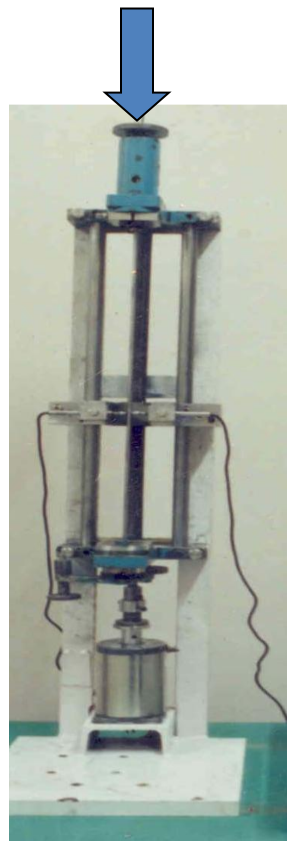
CALIBRATION PIN

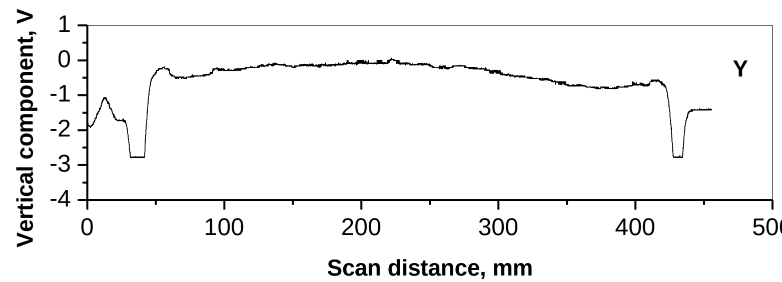


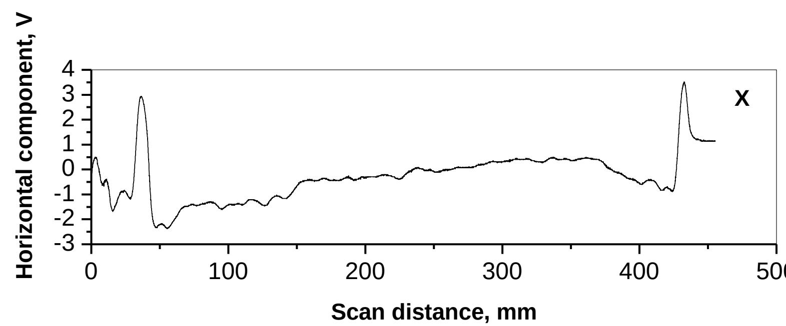


Eddy Current Signals From Standard Defects

Stepper motor driven system for EC testing & fuel pin profilometry



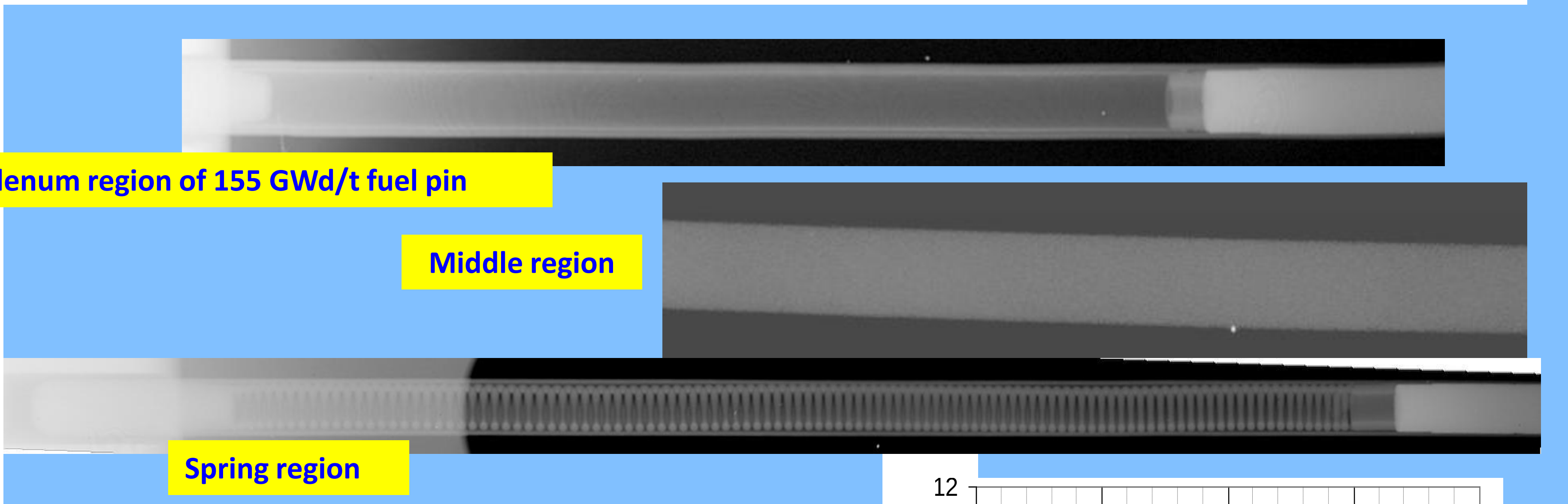


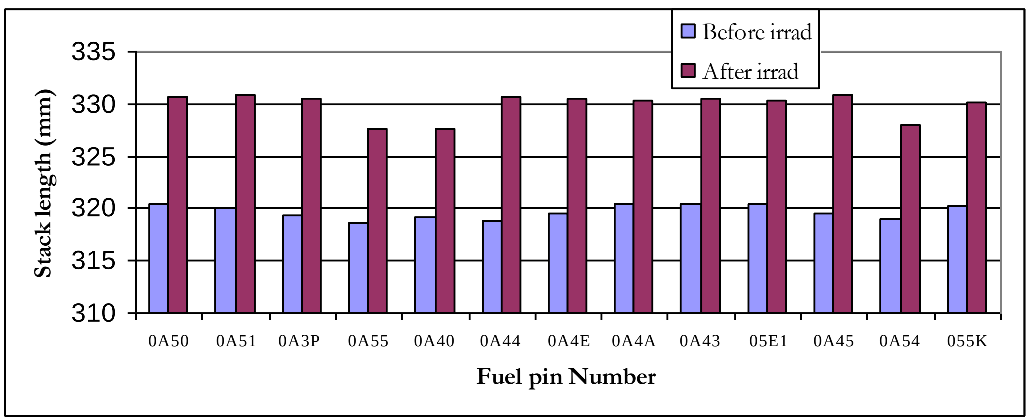


Typical Eddy Current Signals From Irradiated Fuel Pin Indicating no clad defects larger than 0.35 mm

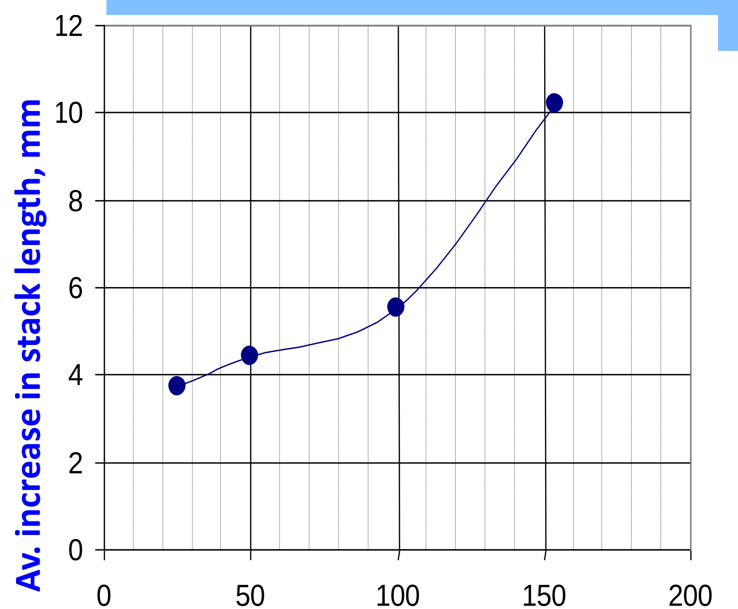
PERFORMANCE OF FBTR CARBIDE FUEL

X –radiography of Fuel



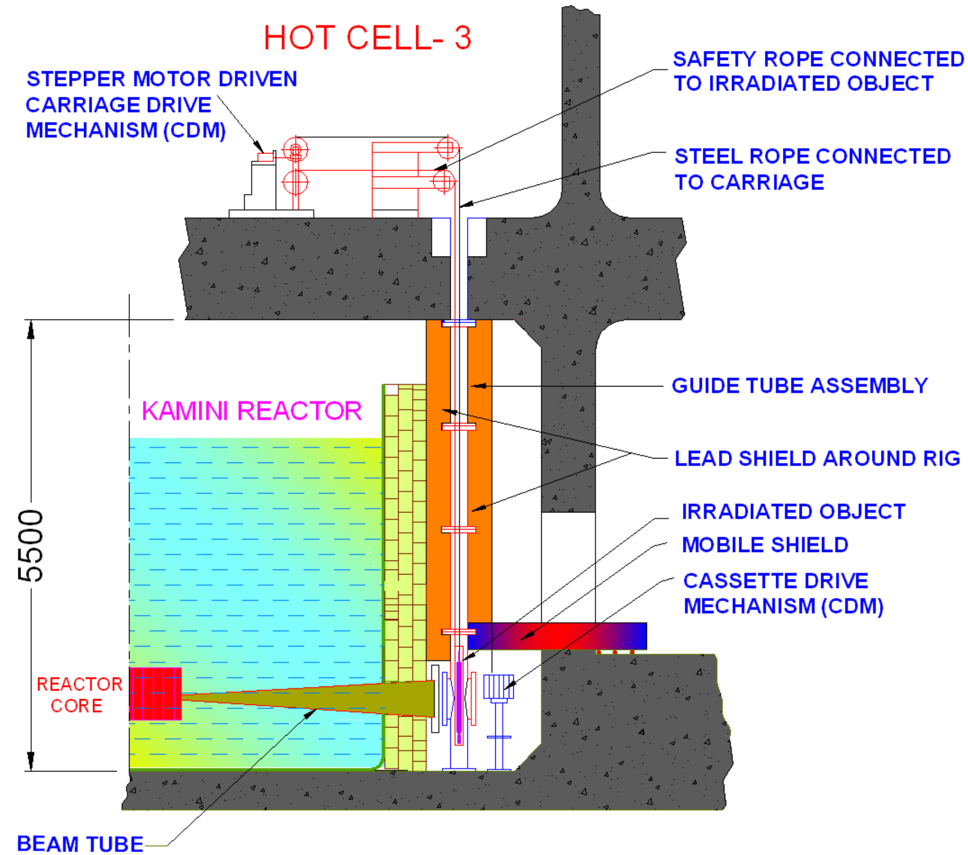


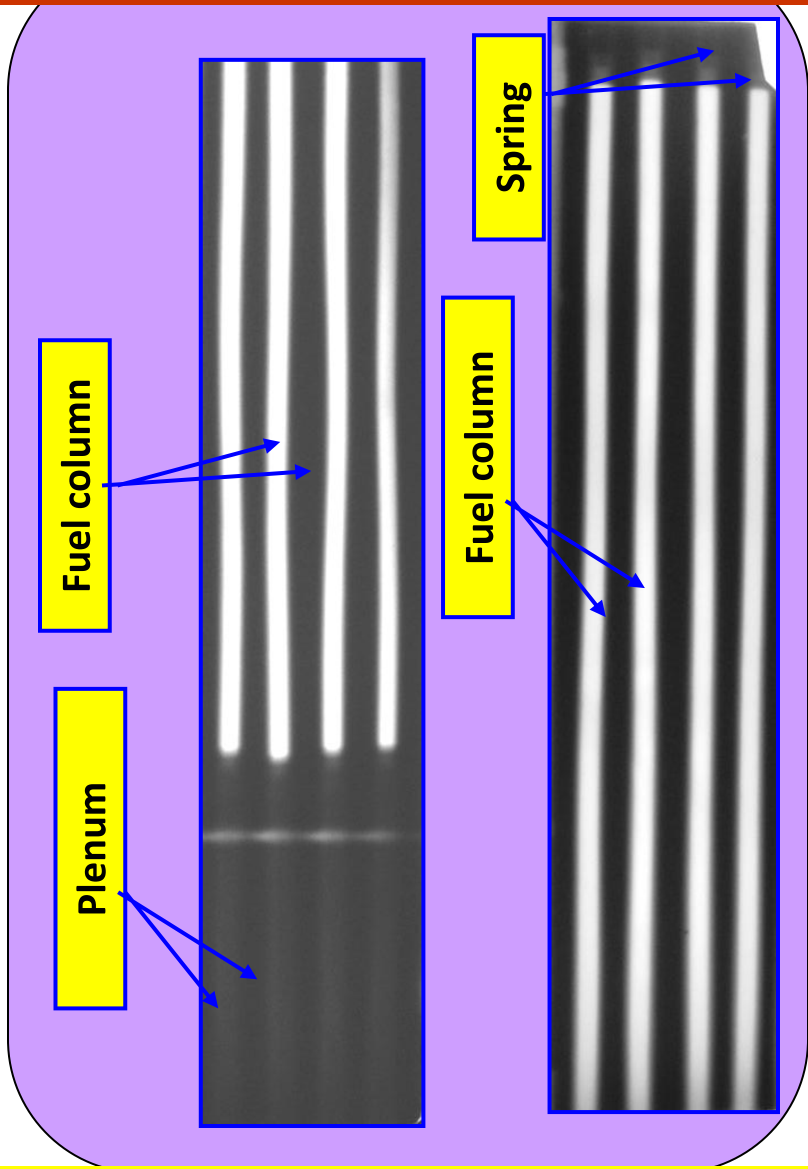
Variation in fuel stack length after 155 GWd/t burn-up



Average stack length increase with burn-up from X-radiography

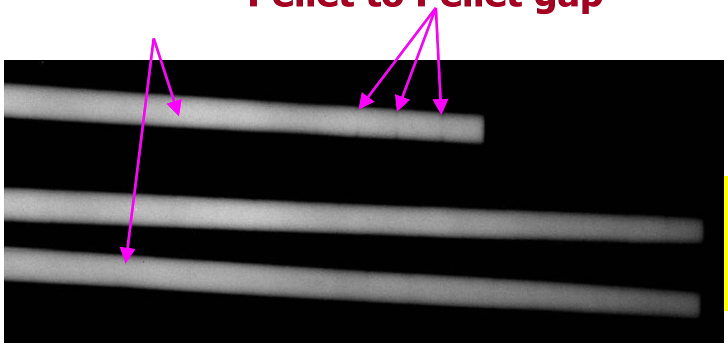
Neutron Radiography using KAMINI Reactor





Neutron - radiography of fuel 50 GWd/t

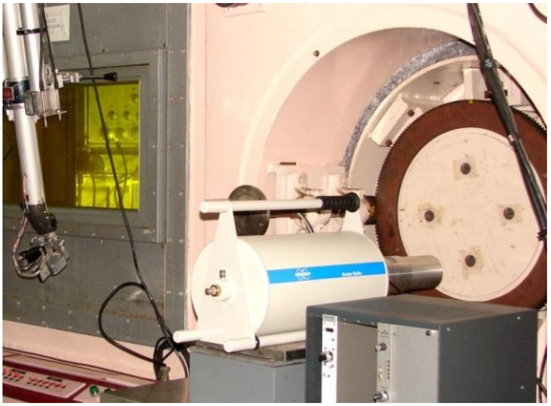
Pellet to Pellet gap



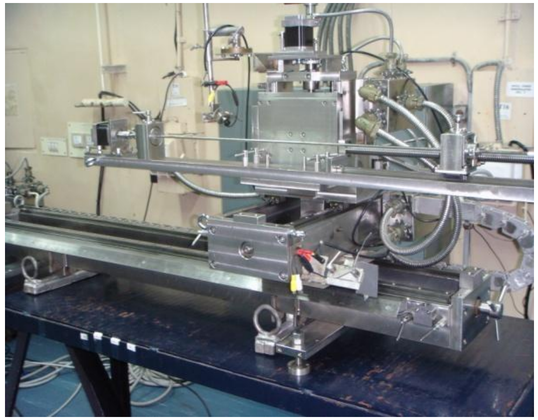
Neutron radiograph of four 155 GWd/t fuel pins showing the fuel column and the spring portion and plenum

AXIAL GAMMA SCANNING OF FUEL PINS

- 4- axis motorised movement (X, Y, Z & θ)
- Remotely replaceable modules for each axis
- PC controlled stage movements with software for data acquisition
- CCD camera based remote viewing system

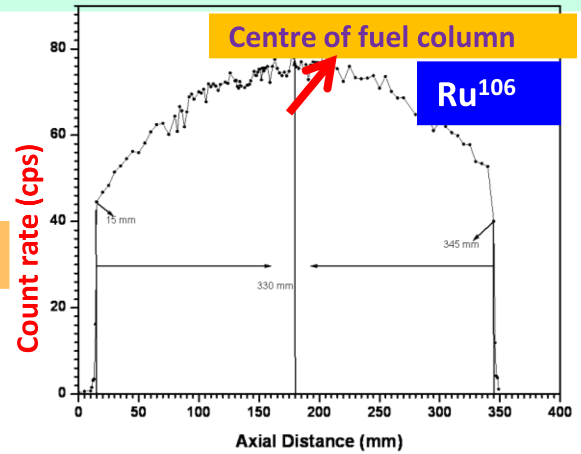


Collimator – Turret assembly & HPGe detector

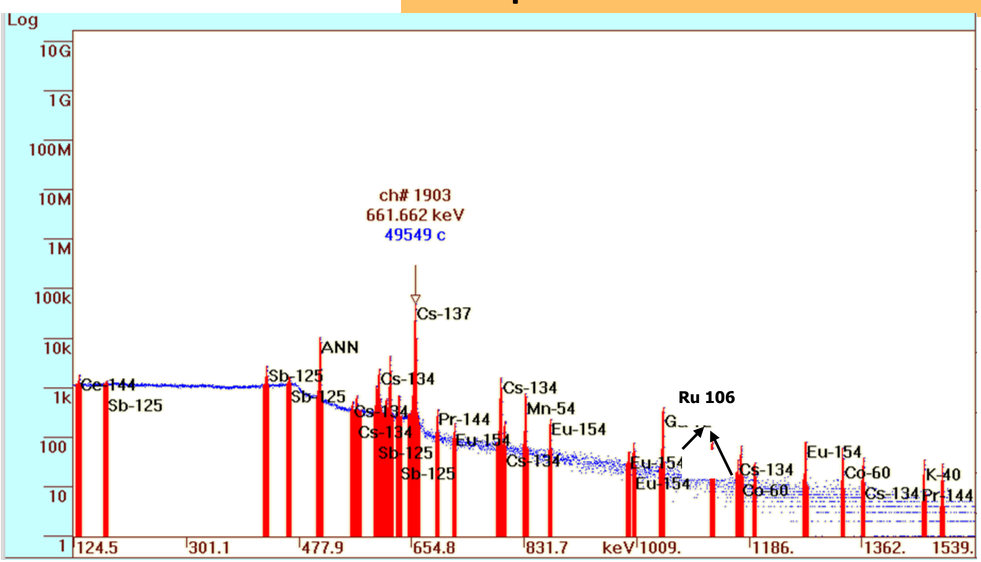


Four axis gamma scanning bench

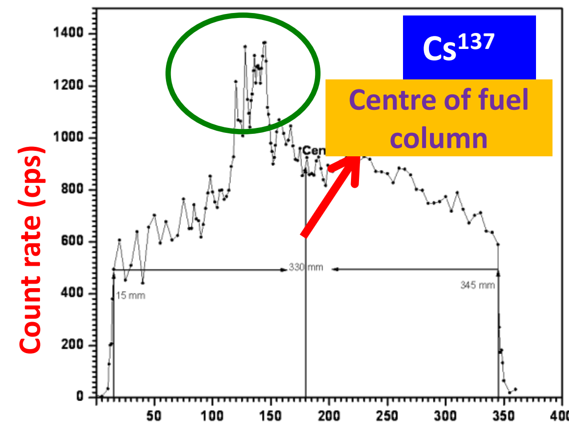
Axial profile of Ruthenium & Cesium



Axial distance from the bottom of fuel column



Typical gamma spectrum from 155 GWd/t burnup fuel pin

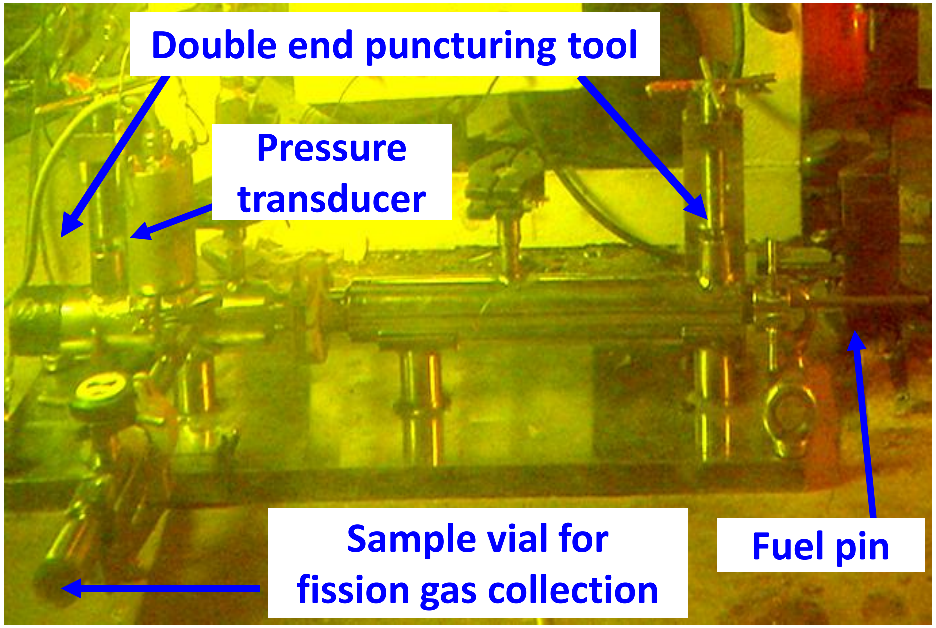


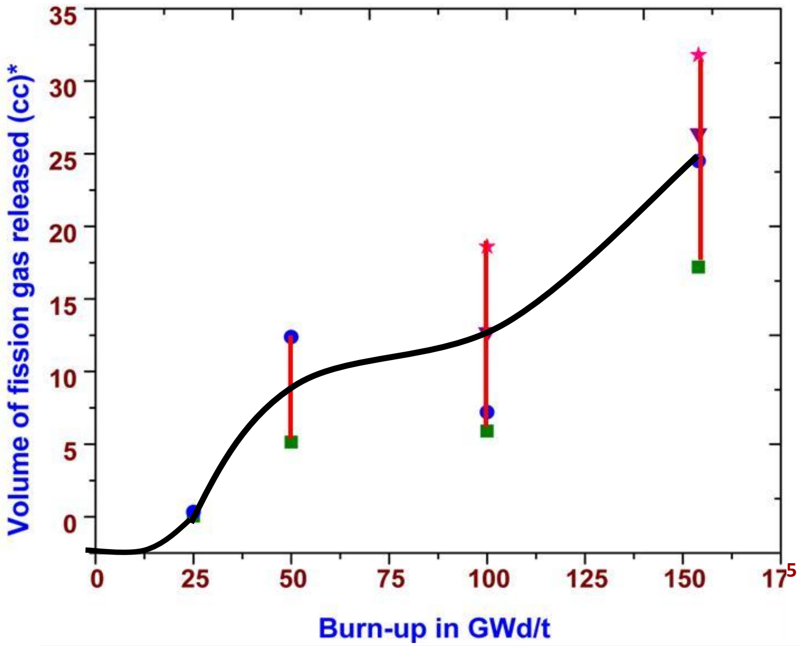
Axial distance from the bottom of fuel column

FISSION GAS RELEASE BEHAVIOUR

- Double End Fission Gas Extraction System
- Fission Gas Analysis Using Gas Chromatograph

Xe/Kr ratio – 13.1



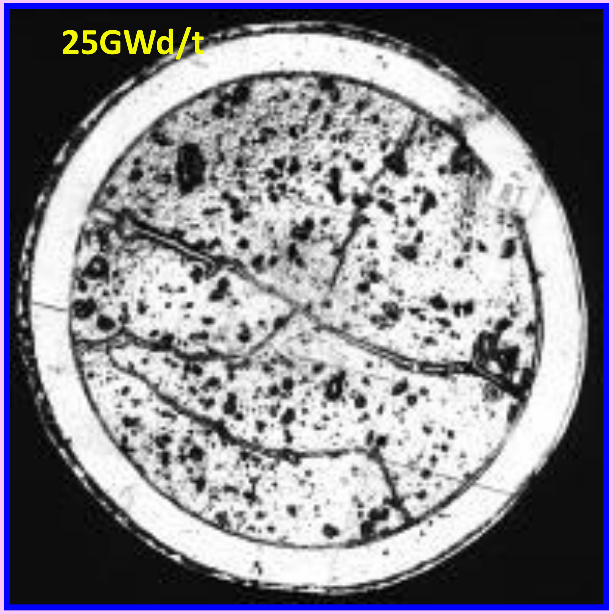


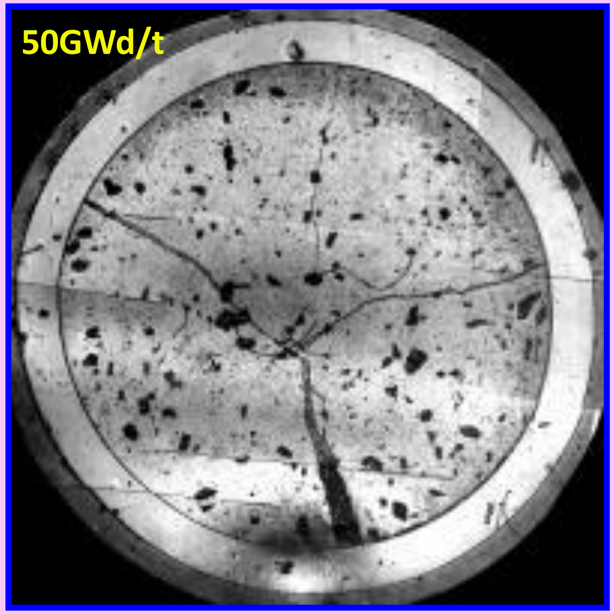
Fission Gas Release with Burn-up

Burnup	% Release	Max plenum Pressure
25 GWd/t	1 %	-
50 GWd/t	8 – 18 %	8 bars
100 GWd/t	4 - 14 %	14 bars
155 GWd/t	8 - 16 %	20 bars

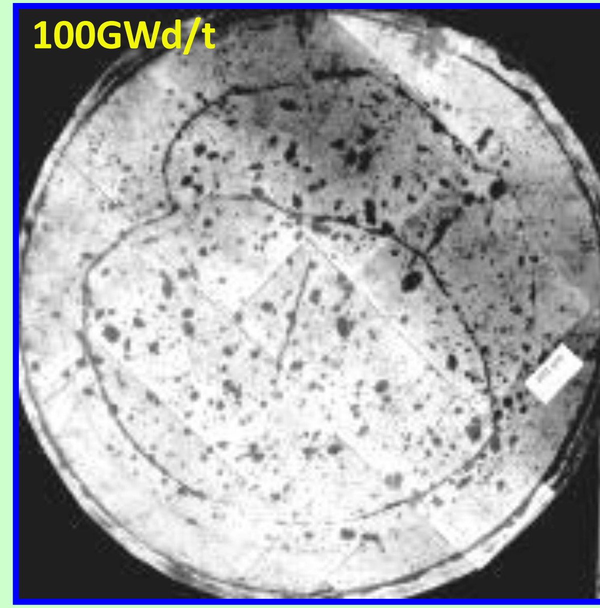
PLENUM CONSTITUENTS (%)	
Xenon	82
Krypton	6.2
Helium	11.7

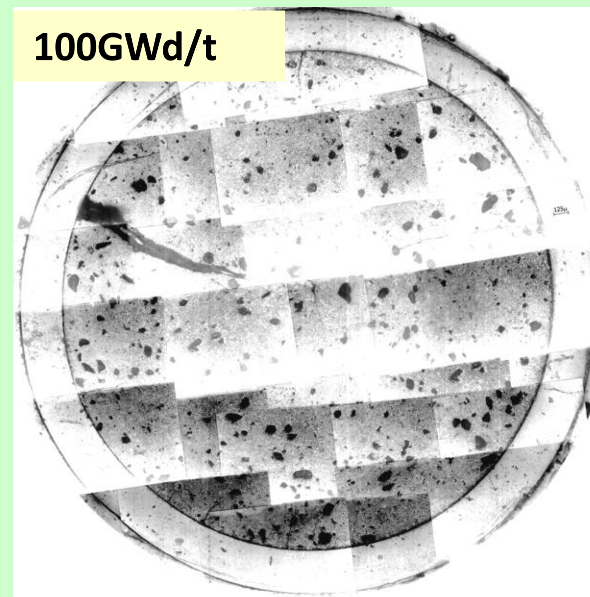
CERAMOGRAPHY OF FBTR CARBIDE FUEL





Micrographs of fuel pin cross section at the *centre of fuel column* after 25 & 50 burn-up



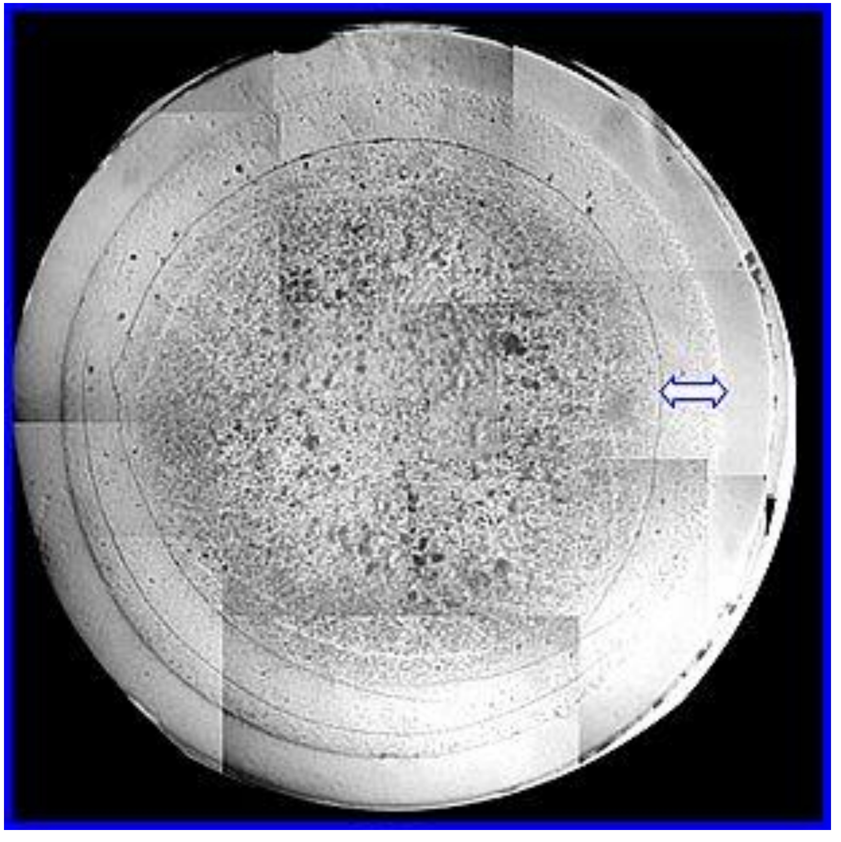


Micrographs at centre and end of fuel column after 100 GWd/t burnup

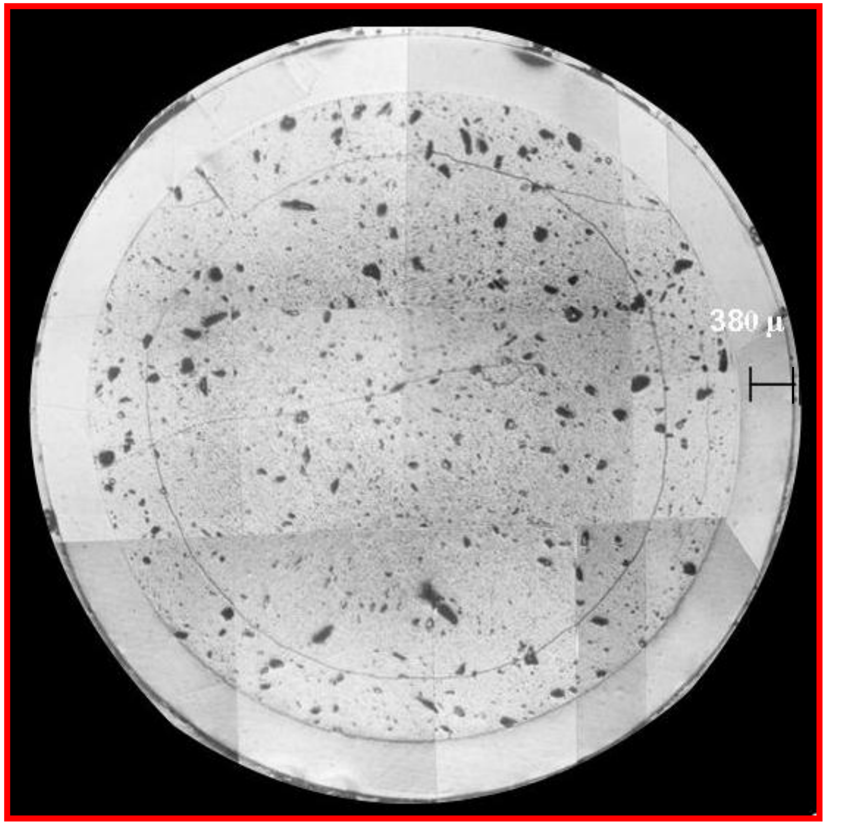
- **Radial cracking at low burn-ups in free swelling regime**
- **Fuel clad gap reduces gradually with burn-up**
- **Change of cracking pattern from radial to circumferential with closure of fuel clad gap**

Micrographs of fuel pin cross section after 155 GWd/t burnup

155 GWd/t – CENTRE of the fuel column



155 GWd/t – END of the fuel column



- Complete closure of fuel-clad gap along the entire fuel column at 155 GWd/t burn-up
- Porosity free dense zone at the outer rim of the fuel
- Swelling of fuel is further accommodated by porosities & clad swelling

Fuel Swelling Rate (free swelling)

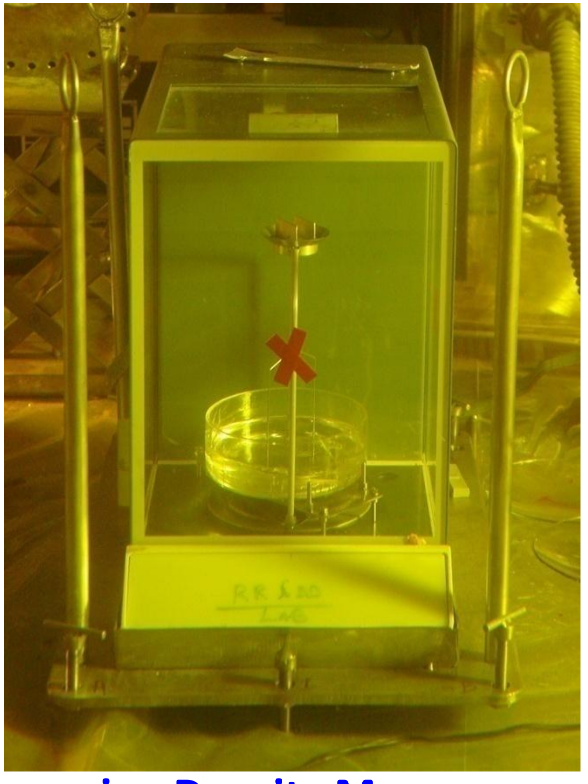
Burn-up GWd/t	Ceramography $(\Delta D / D)$ %	Vol. Swelling $(\Delta V / V)$%
25	1.43	4.29
50	1.7	5.1

Pellet & clad dimensions under restrained swelling

Burn-up & location w.r.t. fuel column		Clad diameter 'mm' (post-irradiation)		Pellet diameter	ΔA/A %
		OD	ID		
100 GWd/t	Centre	5.21	4.46	4.46	3.6
	End	5.1	4.36	4.35	
155 GWd/t	Centre	5.335	4.58	4.58	6.93
	End	5.1	4.36	4.36	

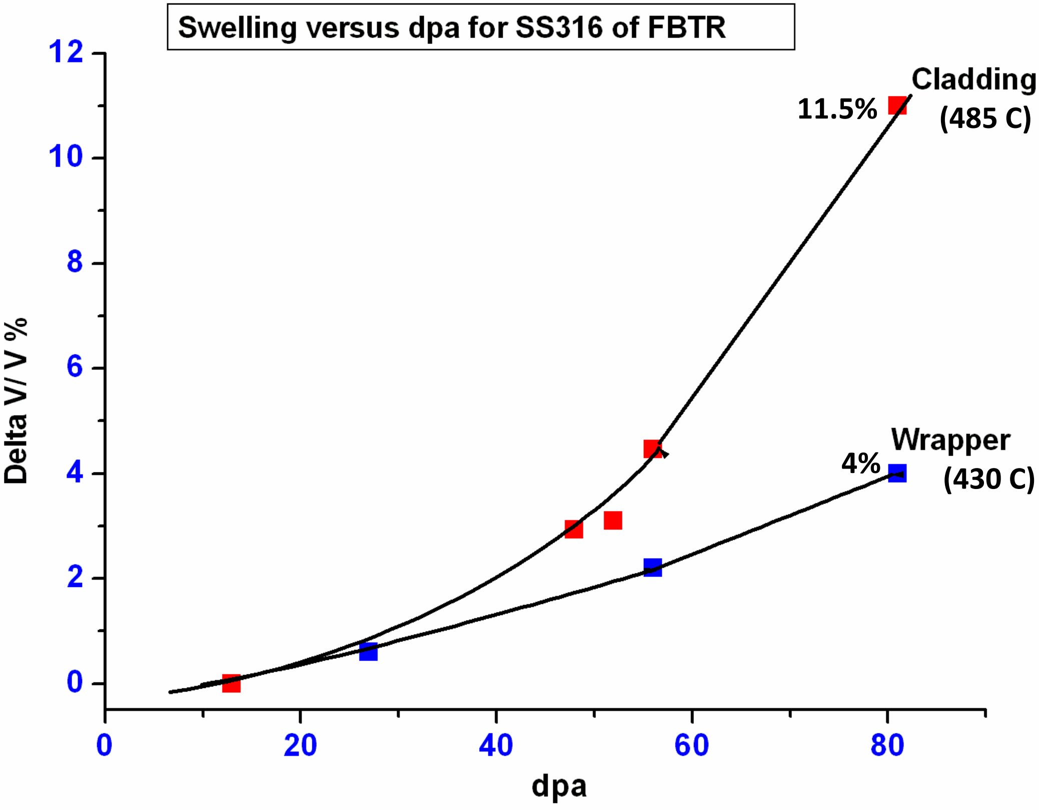
Pre-irradiation Clad O.D - 5.1 mm ; I.D 4.36 mm Pellet OD – 4.18 mm

Void Swelling of Clad / Wrapper - ΔV/V



Immersion Density Measurements

Change in density measured



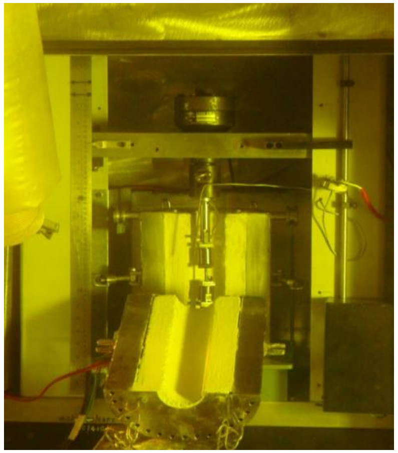
ΔV / V (~ 12 %)	⟷	3 ΔD / D (~ 15 %)

Most of the Cladding and Wrapper Strains are the contributions from

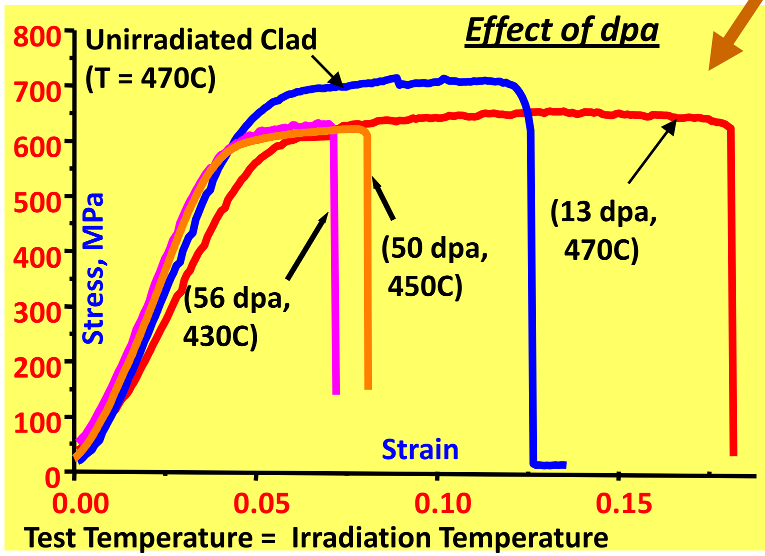
Void Swelling !

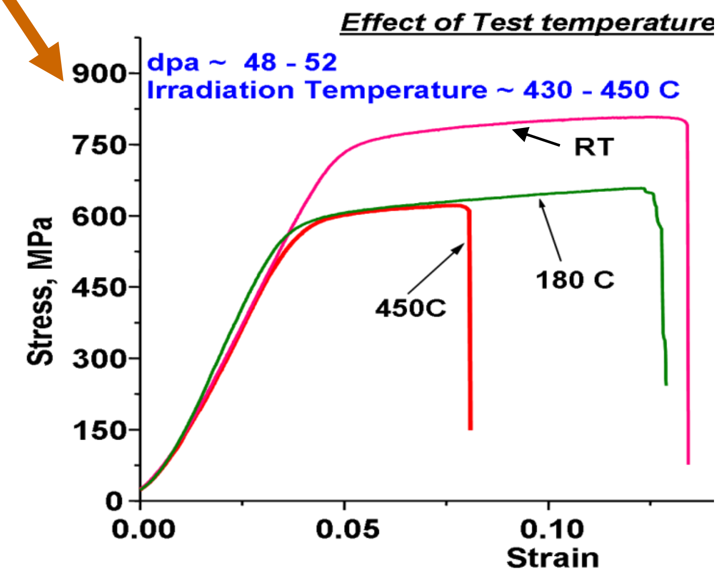
Mechanical Properties of Irradiated 20 % CW SS 316 Clad

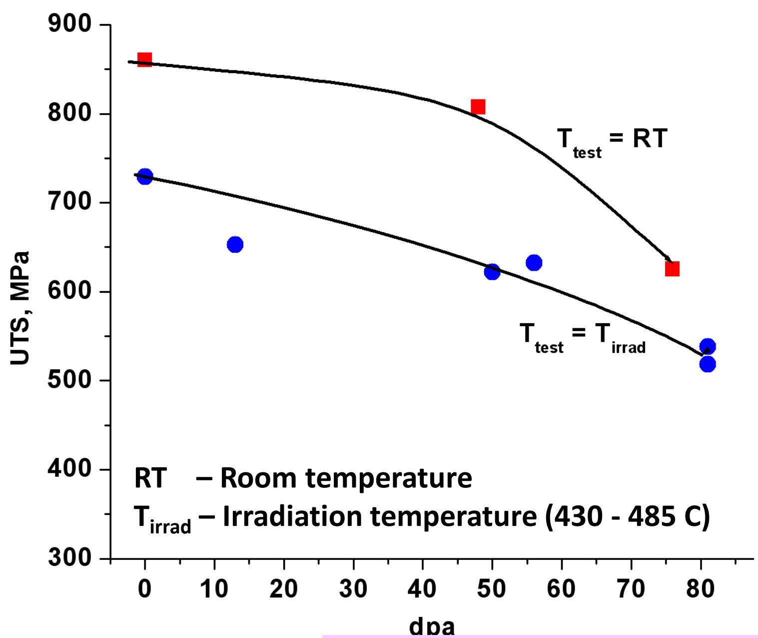
Stress-strain curve for various dpa and test temperatures

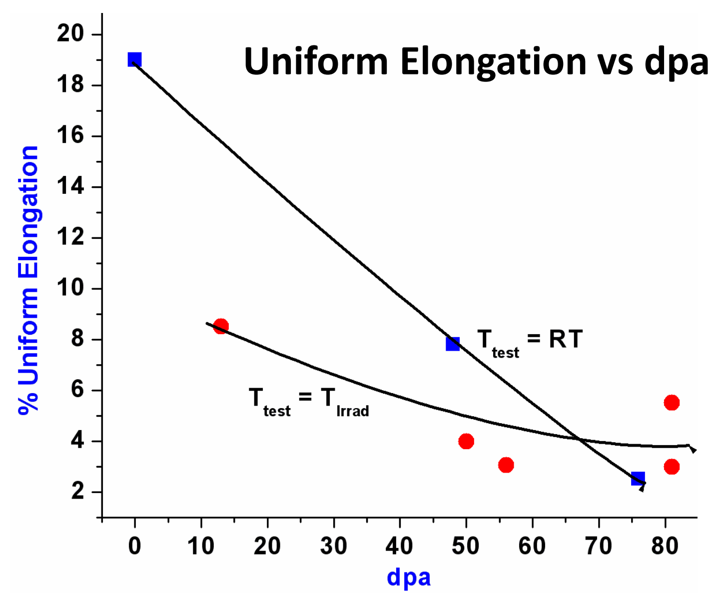


Remote tensile test system



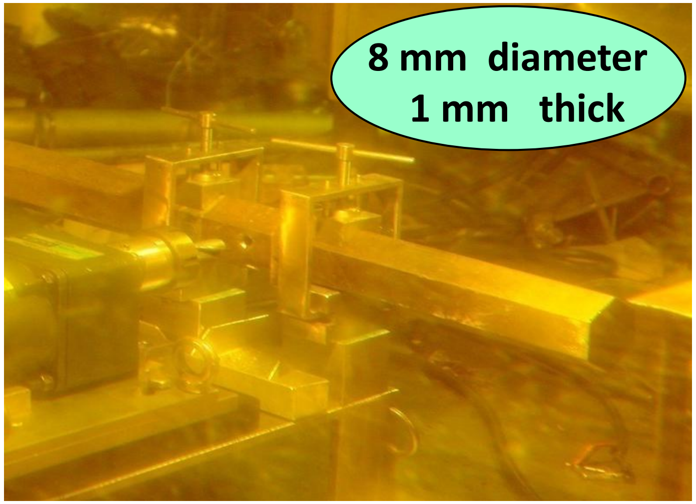






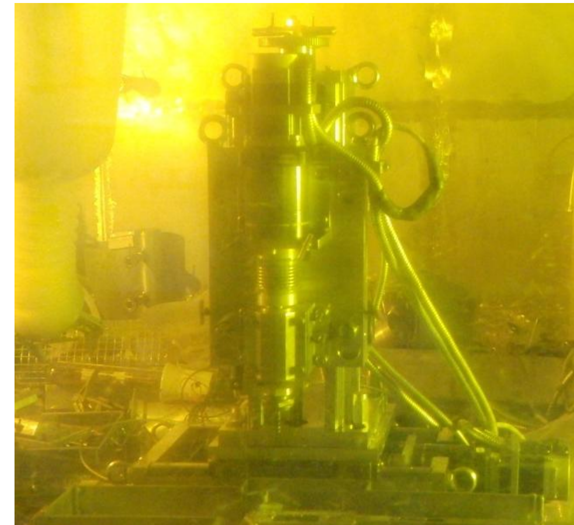
Trends in Strength and Ductility of SS316 with 'dpa'

Tensile Properties of Irradiated Wrapper – by Shear Punch Testing

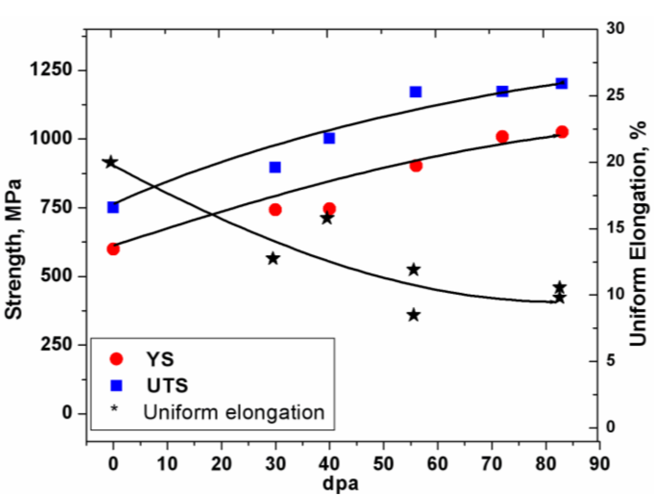


Extraction of small specimens from irradiated wrapper

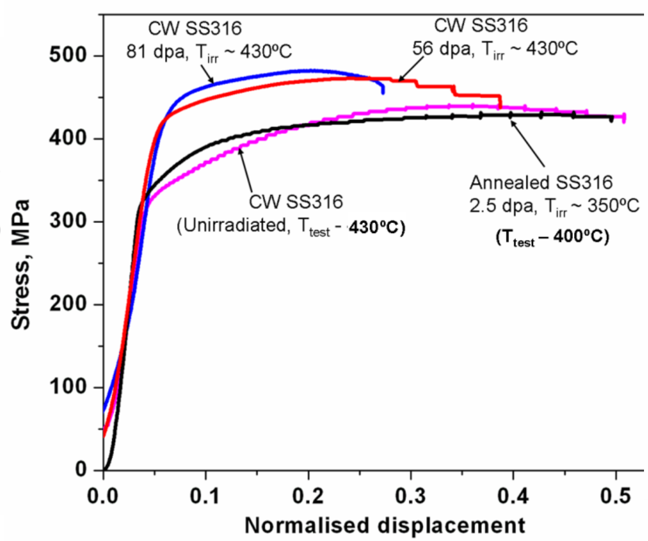
Shear punch test fixture



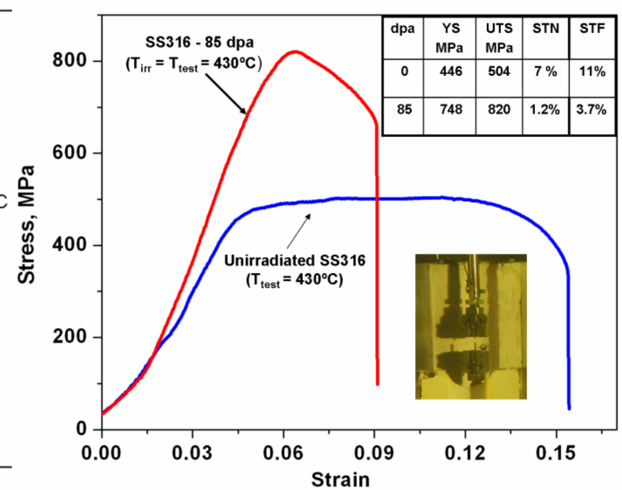
CNC machine for tensile specimen extraction



Variation in room temperature tensile properties of hexagonal wrapper with dpa



High temperature shear punch test of wrapper

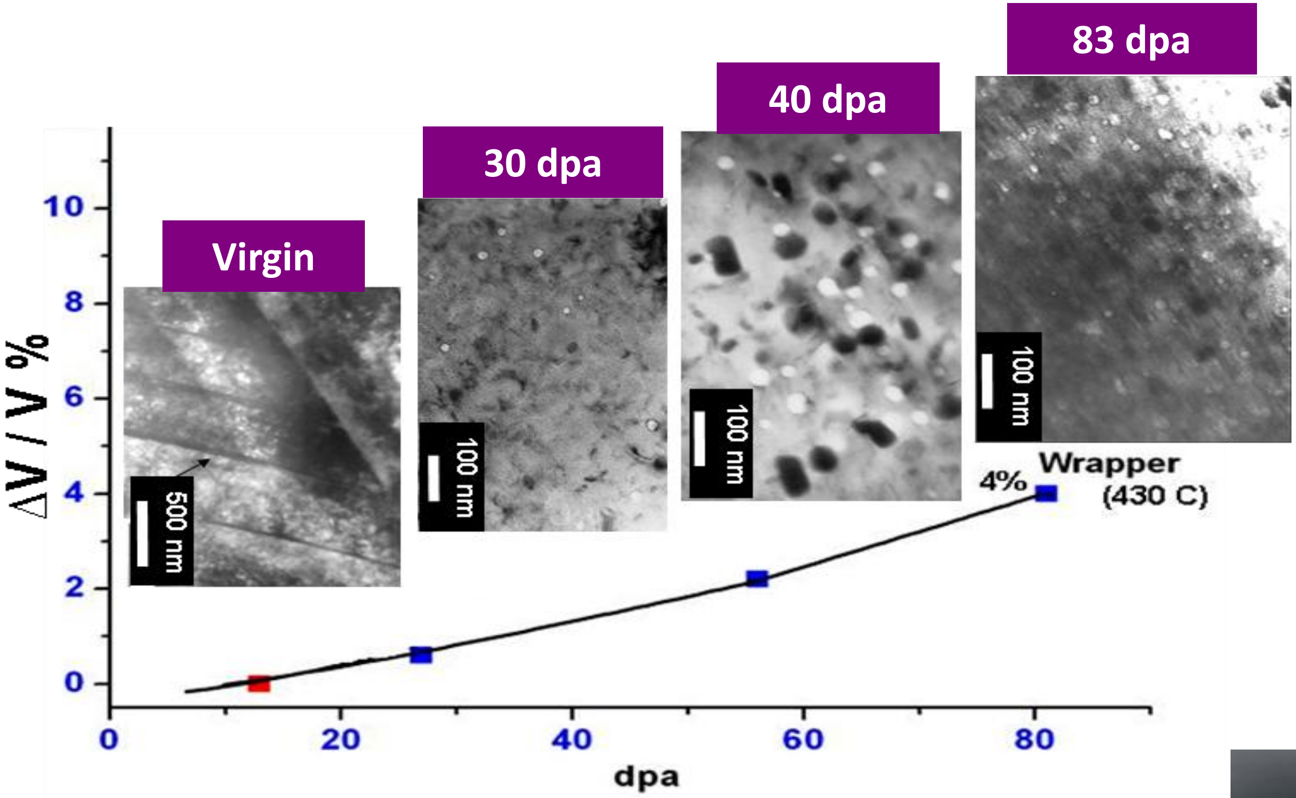


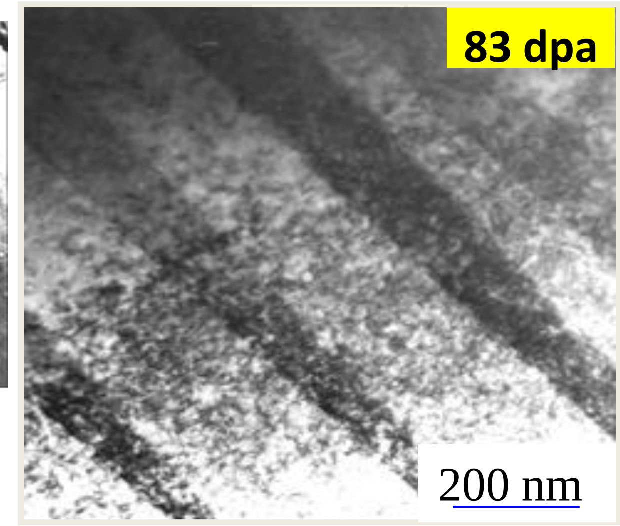
dpa	YS MPa	UTS MPa	STN	STF
0	446	504	7 %	11%
85	748	820	1.2%	3.7%

Note : STN – Strain to Necking calculated as strain from yield stress to point of neck initiation
STF – Strain to Fracture calculated as strain from yield stress to point of failure.

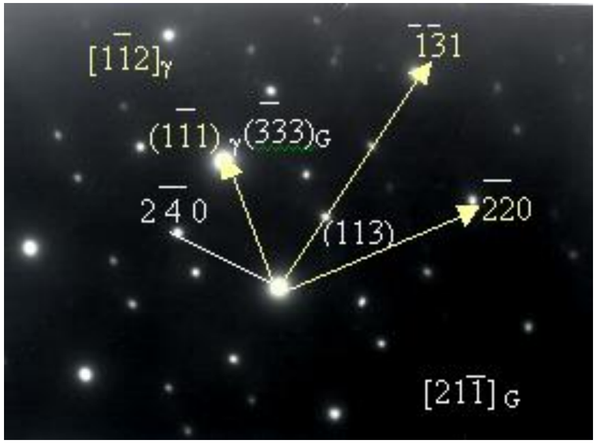
High temperature tensile test of wrapper (83 dpa)

Microstructural Evolution in Irradiated CWSS316 Wrapper





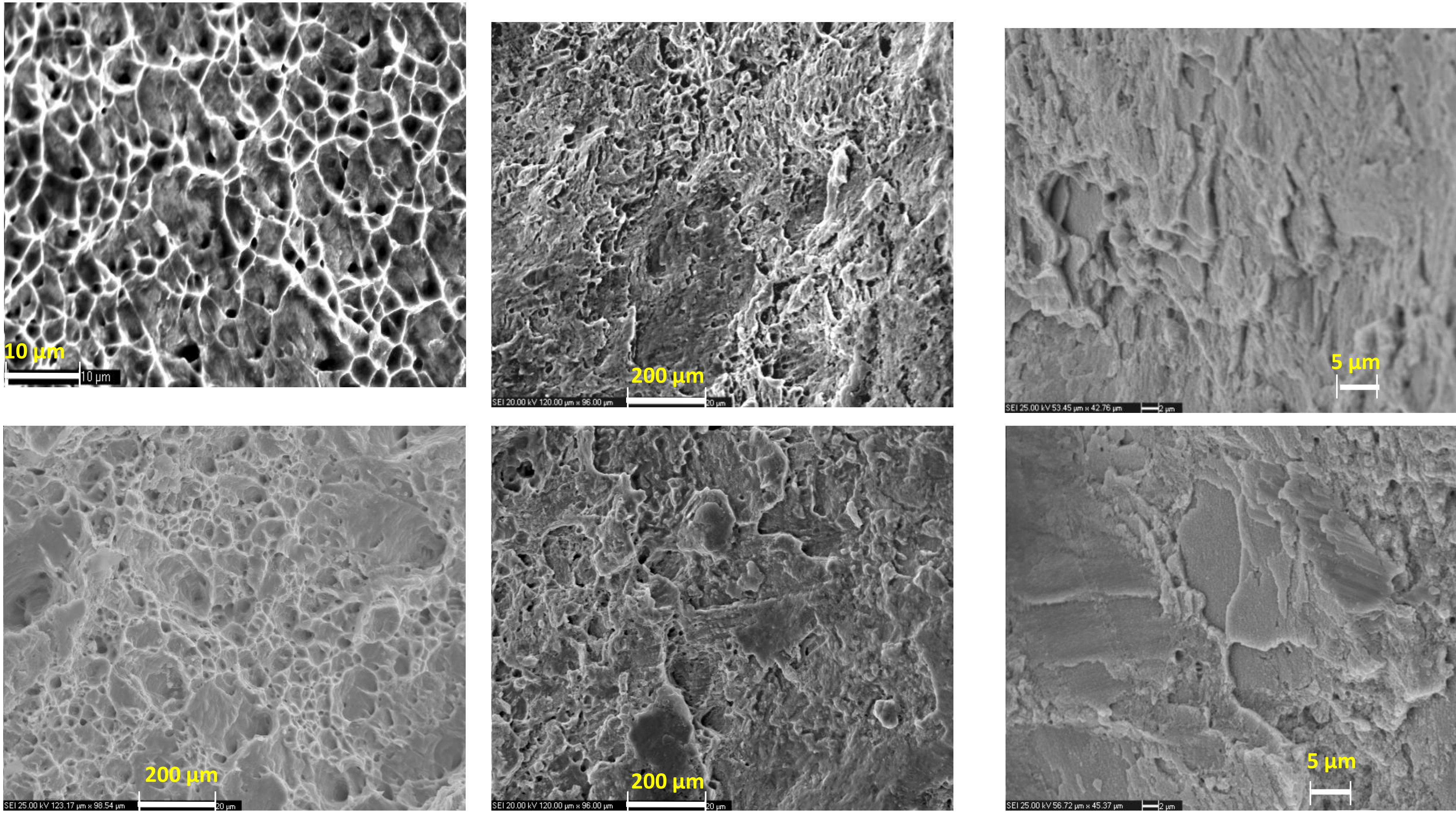
Retention of Cold Work structure



Radiation induced Precipitates - G Phase $M_6(Ni,Co)_{16}Si_7$

- ✓ Extensive Void formation beyond 40 dpa
- ✓ Microstructure with Precipitates in addition to dislocation loops
- ✓ Additional Radiation enhanced and radiation induced Ni and Si rich precipitates observed

Fractographic studies on irradiated cladding at various dpa



Ductile fracture in 13 dpa sample

Mixed mode fracture in 56 dpa sample

Cleavage facets in 83 dpa sample

Irradiation behaviour of FBTR Clad & Wrapper

FBTR CLADDING

- ✓ Strength Reduction with increasing dpa (Softening)
- ✓ Void Swelling ~ 11.5 %
- ✓ ΔD/D ~ 5 %
- ✓ Ductility ~ 3 % at Op Temp

FBTR WRAPPER

- ✓ High Strength Levels (Hardening)
- ✓ Void Swelling ~ 4 %
- ✓ Inter SA gap Reducing
- ✓ Ductility ~ 8 % (Room Temp)

Limits w.r.t performance/safety

1. Cladding Integrity - FCMI & Loss of Mechanical Properties
2. Burn-up limitation mostly comes from Wrapper/Duct Behavior
3. Limits imposed based on Void Swelling and related effects on Fuel Handling, Mechanical Property Degradation
4. 6 % Void Swelling - Set as limits for Phenix Ducts

Continuous Performance Assessment for Optimum Burn-up

- DESIGN BURN-UP LIMIT OF FBTR FUEL - 50 GWd/t

- **EXPERIMENTAL FUEL PINS**
  - ✓ Fuel Crack Initiation and reduction in the Fuel-clad Gap at Low Burn-up
  - ✓ Amenability for operation at Higher Linear Power

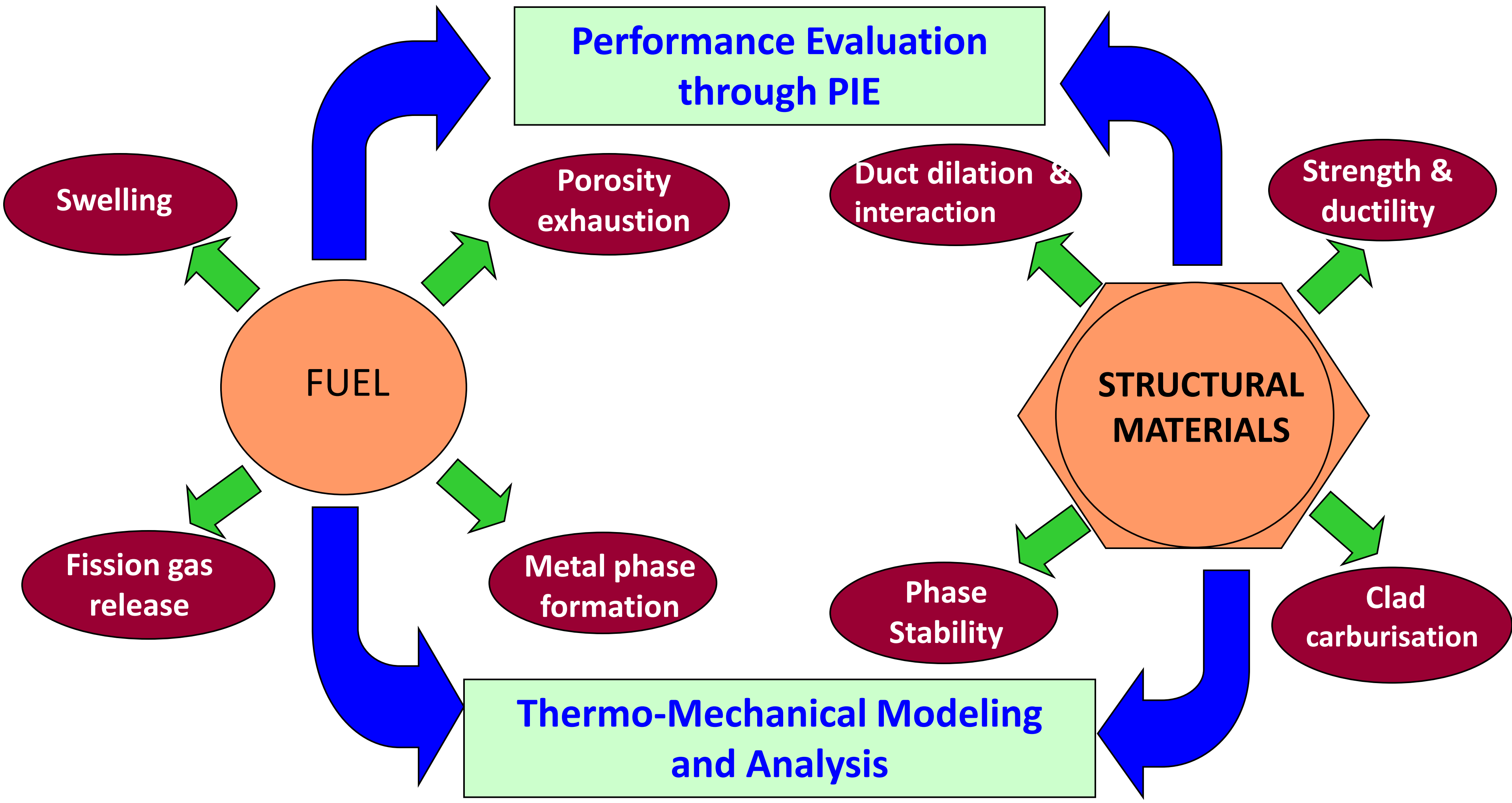
- **PIE AFTER 25 & 50 GWd/t BURN-UP**
  - ✓ Lower Fuel Swelling Rate
  - ✓ Absence Of FCMI till 50 GWd/t (Fuel- Clad Gap available)
  - ✓ Negligible Wrapper & Cladding Strain

- **PIE AFTER 100 GWd/t BURN-UP**
  - ✓ Gap Closure at Fuel Centre (Initiation of FCMI)
  - ✓ Low Fission Gas release & pressure
  - ✓ Significant Increase in Dimensions of Cladding & Wrapper
  - ✓ Sufficient Strength & residual ductility

- **PIE AFTER 155 GWd/t BURN-UP**
  - ✓ Gap Closure along Entire Fuel Column (Restrained Swelling)
  - ✓ Higher Rates of Increase in Cladding/Wrapper Dimensions
  - ✓ Substantial Decrease in Strength & Ductility of Cladding

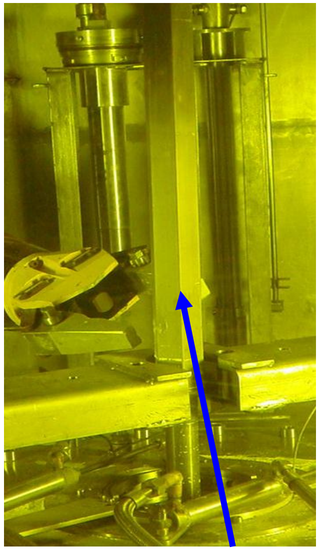
PIE Results & thermo-mechanical modeling indicate a possibility of only a marginal increase in burn-up. The burn-up of one lead fuel subassembly is taken beyond 155 GWd/t for testing endurance limits

Life Extension of FBTR fuel

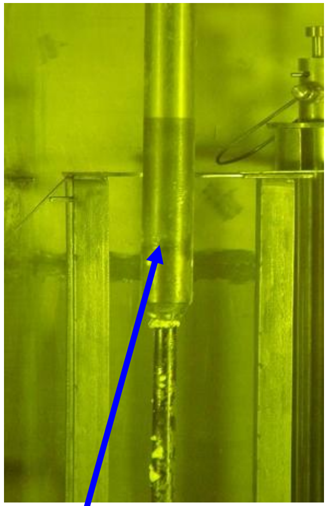


PIE of Control Rod Assembly of FBTR

Objectives: To examine
- Interaction of CR with the outer sheath
- Integrity of the B_4C pellets
- Swelling behaviour
- Depletion of B^{10}



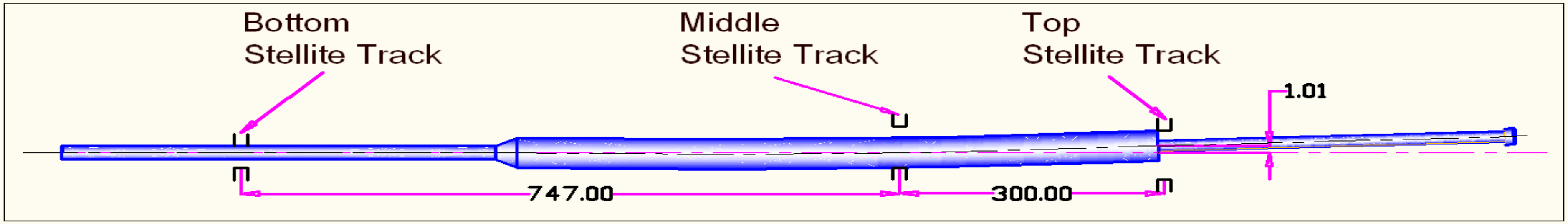
Outer Sheath

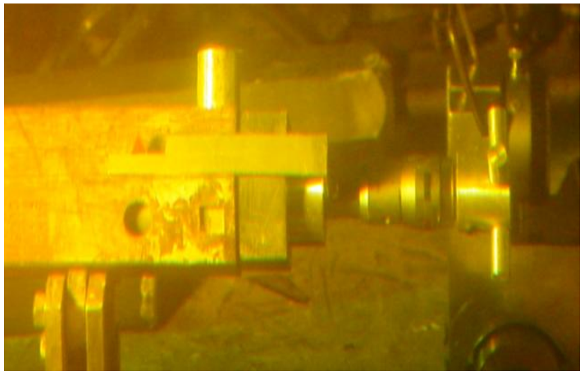


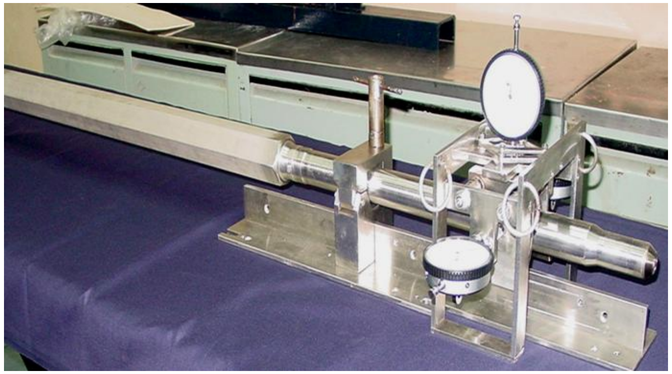
Control rod (CR)

Design & irradiation data

Maximum burn-up of B_4C pellet (at %)	3.4 % ~ 9×10^{27} capture/m^3
EFPD of operation	725.6 days @ 250W/cm
B^{10} enrichment	90.59 0.02 %
B_4C pellet diameter	38 0.15 mm
Pre-irradiation density of B_4C pellet	2.28 – 2.3 g/cc
Cladding material	SS316 L, 20-25% CW
Clad OD & Wall thickness	OD- 45.1 mm, WT- 1.2 mm
Maximum clad neutron dose (dpa)	40.7
B_4C pellet temperature (design)	T_c - 1850 C ; T_s - 540 C
Total length of control rod	1651.5 mm
Total length of B_4C pellet region & number of B_4C pellets	430 mm, 9 pellets

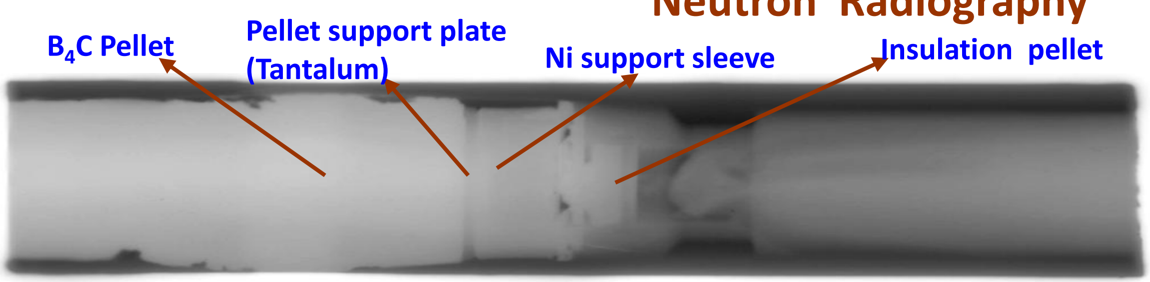



Internal micrometer for measuring the ID of the satellite track


Outer Profile scanning device

- Diameter of control rod clad within the original tolerance limits
- Maximum bow between the axes of the control rod and outer sheath varied from 0.83 mm to 1.01mm
- Shift in longitudinal axes can result in interference during raising of control rod

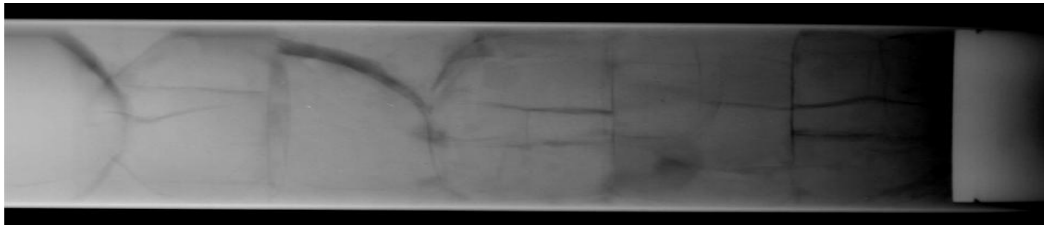
Neutron Radiography



- No evidence of gross depletion of B^{10}
- All internals & Insulation pellet intact

PIE of Control Rod Assembly (Contd)

X- Radiography

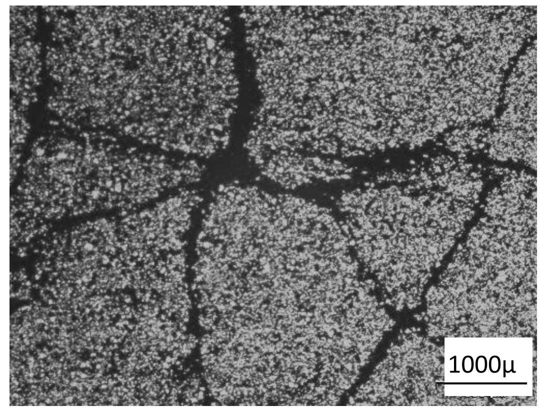


- X-radiographs indicated extensive fragmentation of bottom pellets
- No significant change in dimensions of pellets

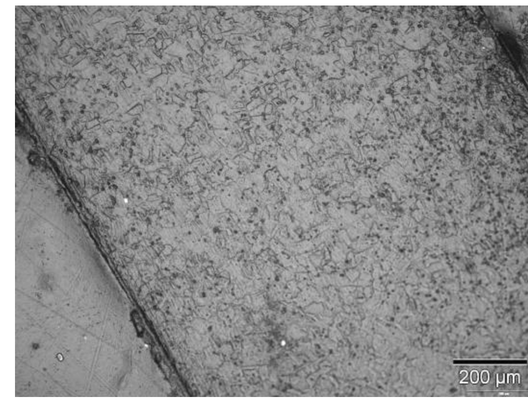
Bottom pellet showing infiltration of sodium

Photomicrographs of

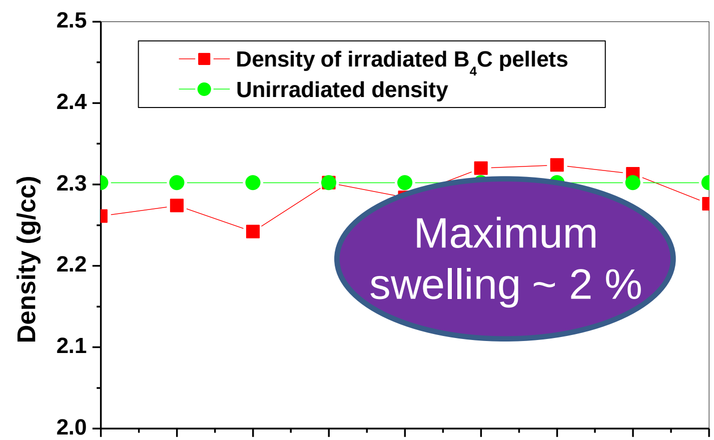
B_4C pellet



Clad Cross section



^{10}B / ^{11}B ratio determined from Laser mass spectrometer
^{10}B depletion in all the pellets was found to be less than 1 %.



Density variation in B_4C pellets

CONCLUSIONS

B_4C pellets and SS clad have not reached life limiting conditions

&

Direct reuse of pellets not possible due to loss of integrity

PIE of Nickel Reflector subassembly

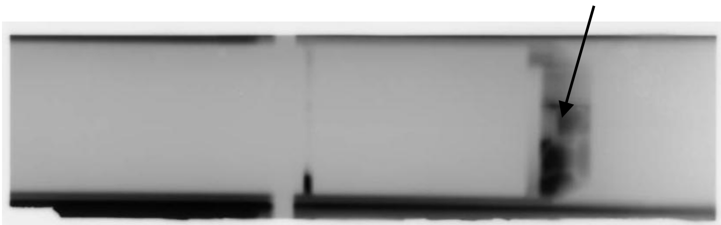
Design fluence limit – 1.14 x 10^{23} n/cm^2/s

PIE to assess

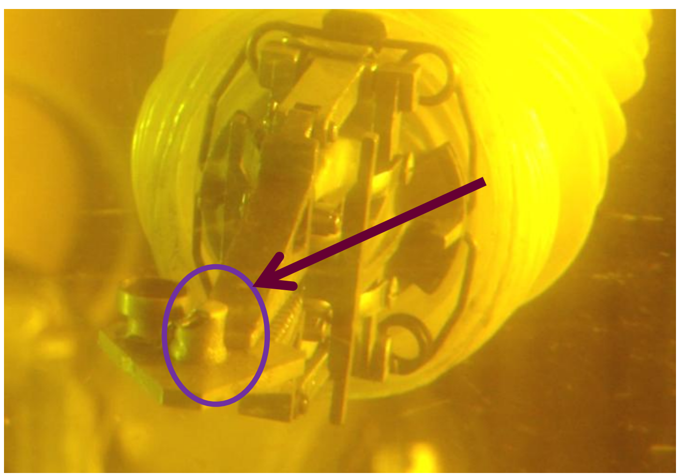
- **Swelling of nickel blocks**
- **Condition of the collapsible tube**
- **Dimensional changes in the nickel blocks**

Impurity content in Nickel : 0.6 %

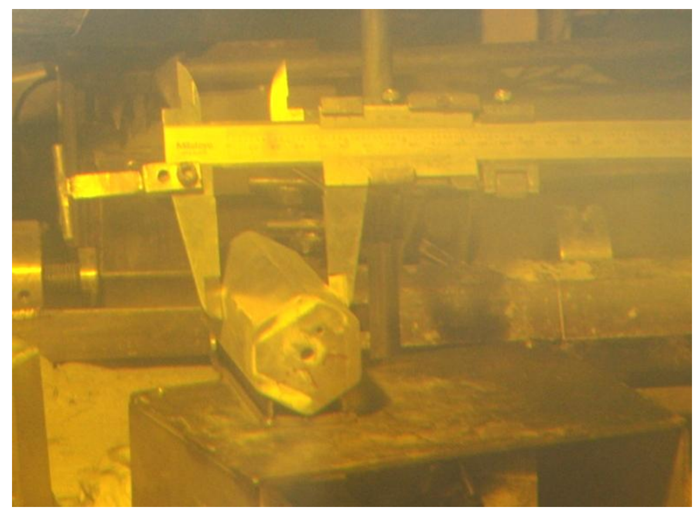
Fluence exposed : 1.09 E23 n/cm^2/s



Neutron radiographs Showing the Collapsible tube and Nickel blocks

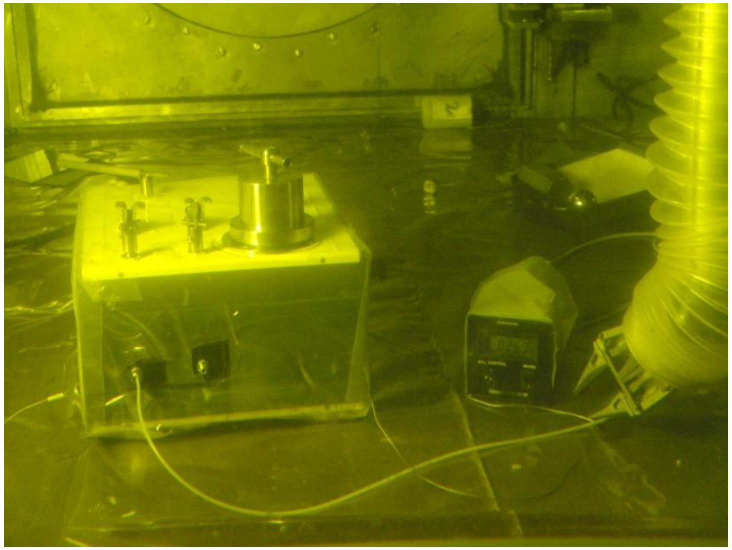


Collapsible tube inside Ni subassembly retrieved after dismantling

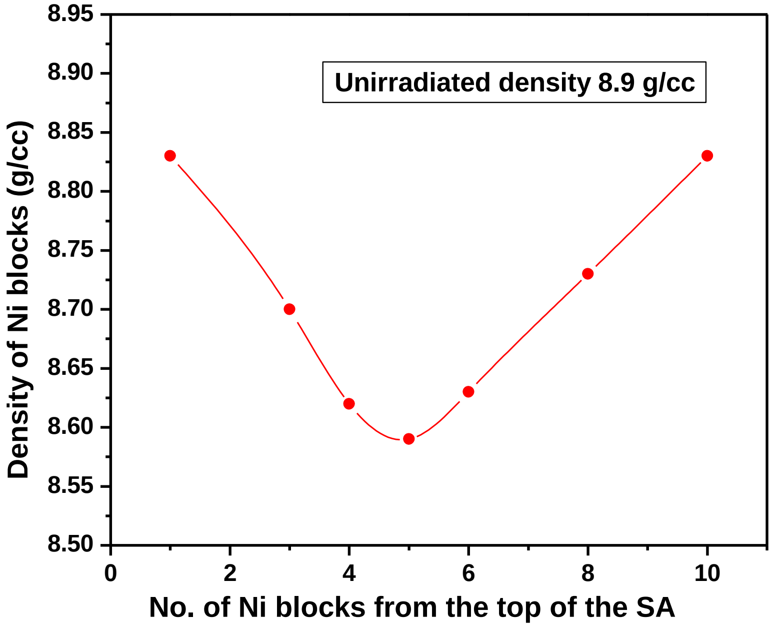


Dimensional measurements on the nickel blocks

PIE of Nickel reflector assembly of FBTR



Density measurements of nickel blocks



Swelling of irradiated nickel blocks

CONCLUSIONS

- Collapsible tube has undergone deformation due to axial expansion of the nickel blocks (~7 mm)
- Maximum volumetric swelling of the nickel block is 3.6 %
- Radial gap between nickel blocks and wrapper has reduced from 3 mm to 2mm
- Residual life of reflector subassemblies can be extended further without any concern on mechanical interaction between nickel blocks & wrapper

FBTR AS TEST BED

Experiments Irradiation in FBTR

- Testing of Zr / Zr-Nb alloy pre-pressuised capsules
- PFBR MOX Fuel
- Testing of FBTR Grid plate material
- Testing of Advanced Structural Materials (D9, D9I, 9Cr-1Mo)
- Sr^{89} for Medical uses

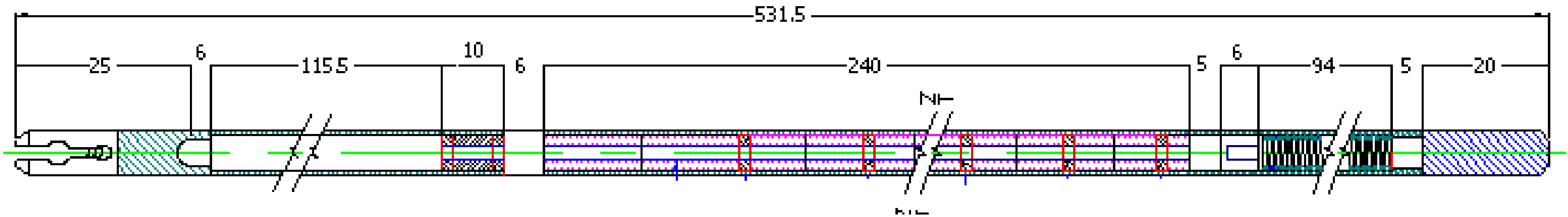
PIE of PFBR MOX test fuel pin

MIXED OXIDE FUEL IS PROPOSED TO BE USED IN PFBR

TARGET LINEAR POWER & BURNUP IS 450 W/cm & 100 GWD/t

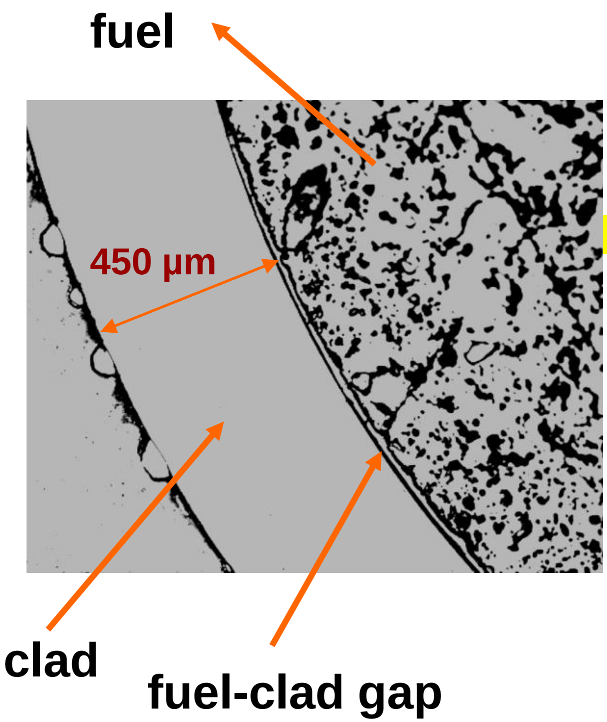
TEST IRRADIATION OF MOX FUEL PIN & PIE TO EVALUATE

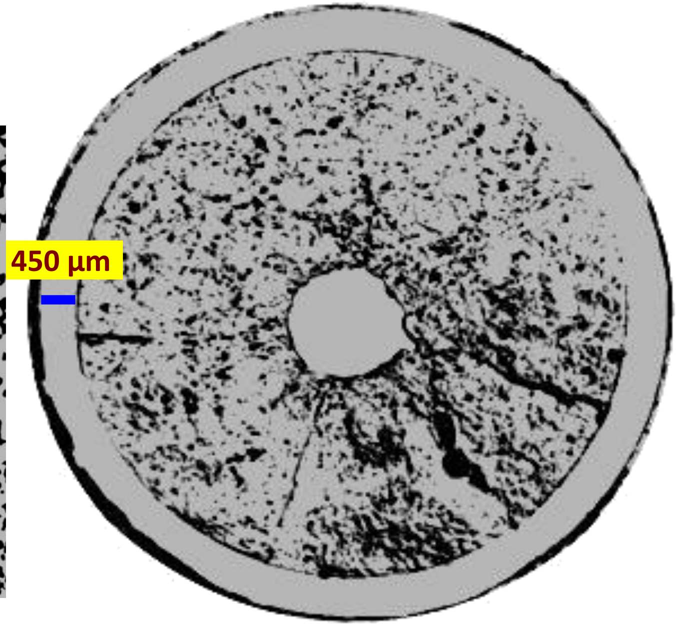
- **Beginning of life gap closure behaviour**
- **Optimising the duration of preconditioning at linear power of 400W/cm**



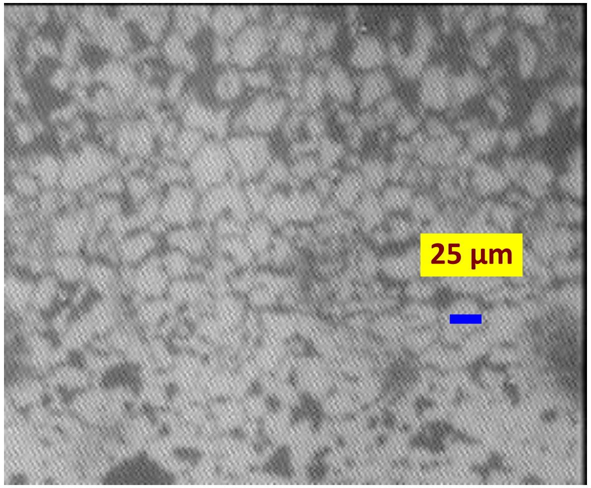
Design & Irradiation data	
Fuel composition	$(U_{0.71}\ Pu_{0.29})O_2$ U^{233} – 53.5 %
Fuel column length	240 mm
Pellet diameter	OD – 5.52 mm ; ID – 1.6 mm
Clad diameter	OD – 6.6 mm ; ID – 5.7 mm
Irradiation duration	13 EFPD (2700 MWd/t)
Linear heat rating	400 W/cm

PIE of PFBR MOX test fuel pin





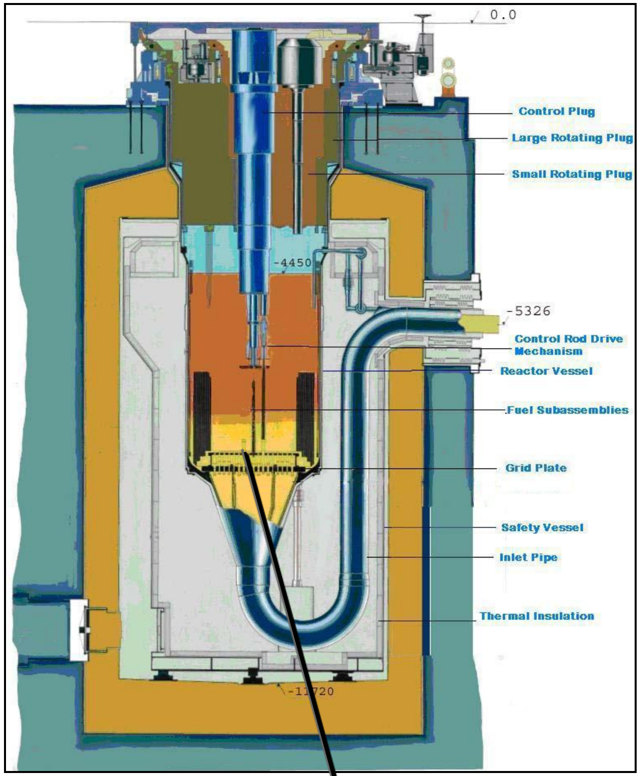
Photomicrograph of the cross-section of PFBR MOX experimental fuel pin



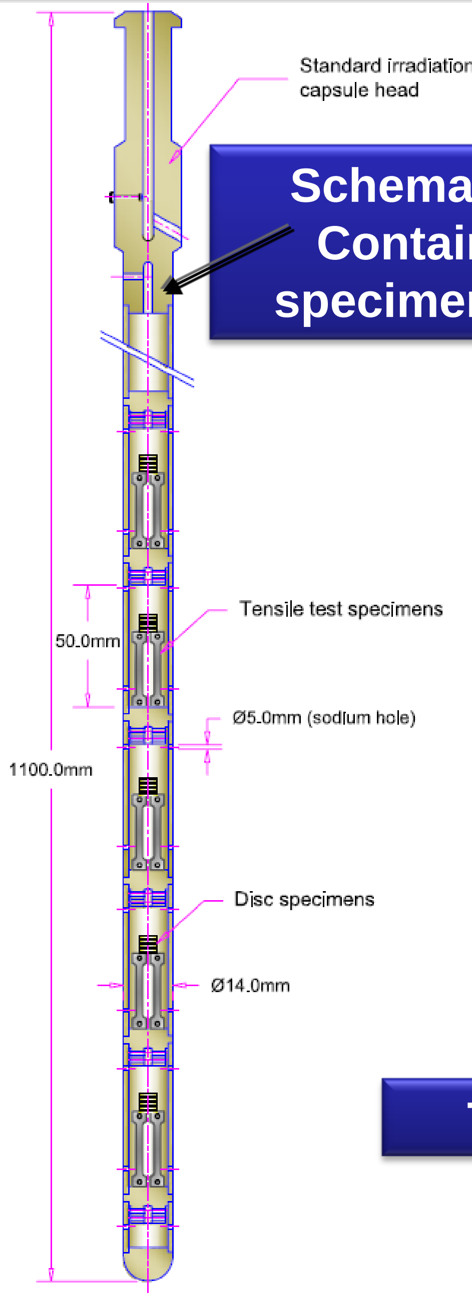
Microstructure of MOX fuel

❖ Fuel-Clad gap reduction observed throughout the fuel column (Reduction from 90 - 110 μm radial gap to 13 μm)

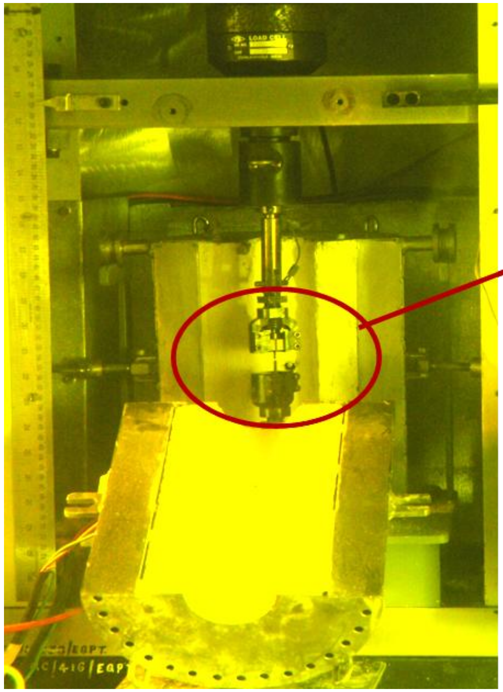
LIFE EXTENSION OF FBTR

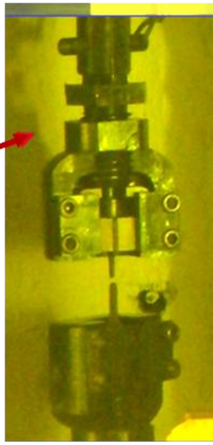


GRID PLATE



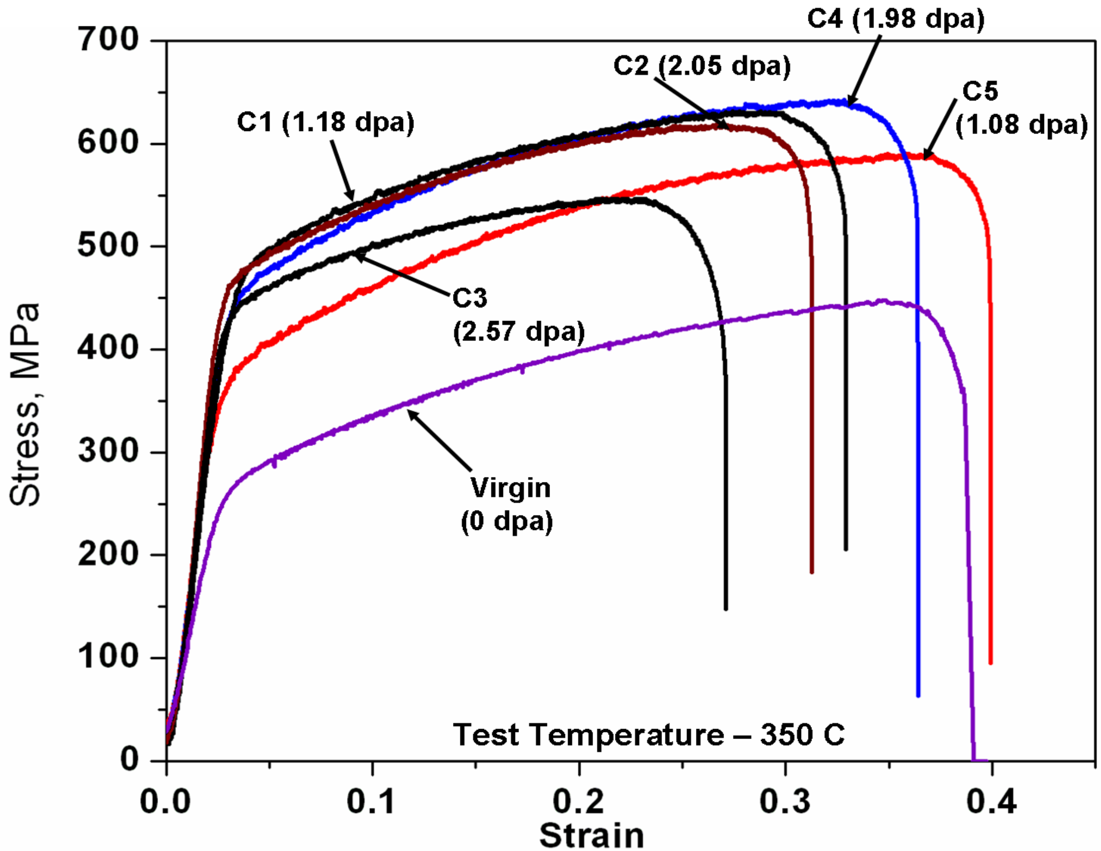
Schematic of Irradiation capsule Containing Grid plate material specimens for irradiation in FBTR





Testing in RML hot cells

Irradiation Testing of FBTR Grid Plate Material



Dose: 0 - 2.6 dpa (dpa values estimated from the flux and spectrum

Tensile specimens: 3 X 1 X 12.7 mm

Tensile test temp: 350 C

Nominal strain rate - 4 x $10^{-4}s^{-4}$

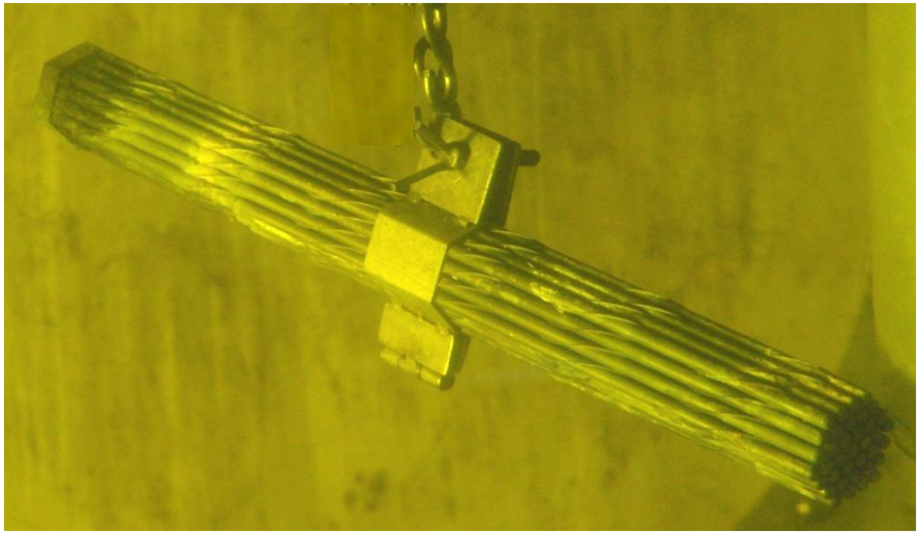
PIE results :

- Increase in YS: 246 to 430 MPa
- Increase in UTS: 447 to 640 MPa
- Decrease in unif. Elong.: 32% to 20%
- Decrease in total Elong.: 35% to 24%

Post Irradiation Examination in IGCAR – ROAD AHEAD !

PIE OF

- PFBR OXIDE FUEL SUBASSEMBLY AFTER 112 GWd/t BURNUP
- METALLIC & SPHERE-PAC FUEL IRRADIATED IN FBTR
- IRRADIATED ADVANCED STRUCTURAL MATERIAL D9, D9I, ODS
  (Pre-pressurized capsules, small specimens)
- REFABRICATION AND RE-IRRADIATION OF FUEL PINS



FUEL PIN CLUSTER RETRIEVED FROM PFBR MOX TEST SUBASSEMBLY


Thank You