Coated Particle Fuel
FCM Replacement Fuel for LWRs

OECD-NEA EGATFL

Mar. 3 ~ 5, ’15, OECD

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Contents

I. Introduction

❑ Lessons from Fukushima Event

- Fukushima-like Consequence is **UNLIKELY**, if Fuel remains Integral to retain Radioactivity and Hydrogen Generation is avoided
- Safety is more than ever the Future of Nuclear
  - Accident Risk Cost: 0.45 ~ 0.89 yen/kWh (JAEC, Nov. '11)

❑ US Senate Budget Appropriation

- *(Jul. '13) "The Committee is encouraged … to develop enhanced accident tolerant fuels … to cope with beyond design-basis accidents. … The Committee supports … development on concepts that target reduced heat and hydrogen production … and which provide additional barriers to fission product release, thus limiting the possibility of offsite contamination in the event of catastrophic accidents."*
- *(Jun. '14) "The Committee directs … to continue implementation of the accident tolerant fuels development program, the goal of which is development of meltdown-resistant nuclear fuels leading to in-reactor testing and utilization in 10 years. …"*

I-NERI Project

- Joint I-NERI Project between ORNL/USNC and KAERI
  - Objective: Feasibility of FCM Replacement Fuel for LWRs
- Project Overview
  - Period: 3 years (Nov. 15, '11 ~ Nov. 14, '14)
  - Project Tasks
    - Task 1: Neutronics Exploration (KAERI)
    - Task 2: Core TH Assessment (KAERI)
    - Task 3: Safety Assessment (KAERI)
    - Task 4: Fuel Qualification (ORNL/KAERI)
      - Qualification (ORNL)
      - Performance (KAERI)
    - USNC Analysis Supports All Tasks

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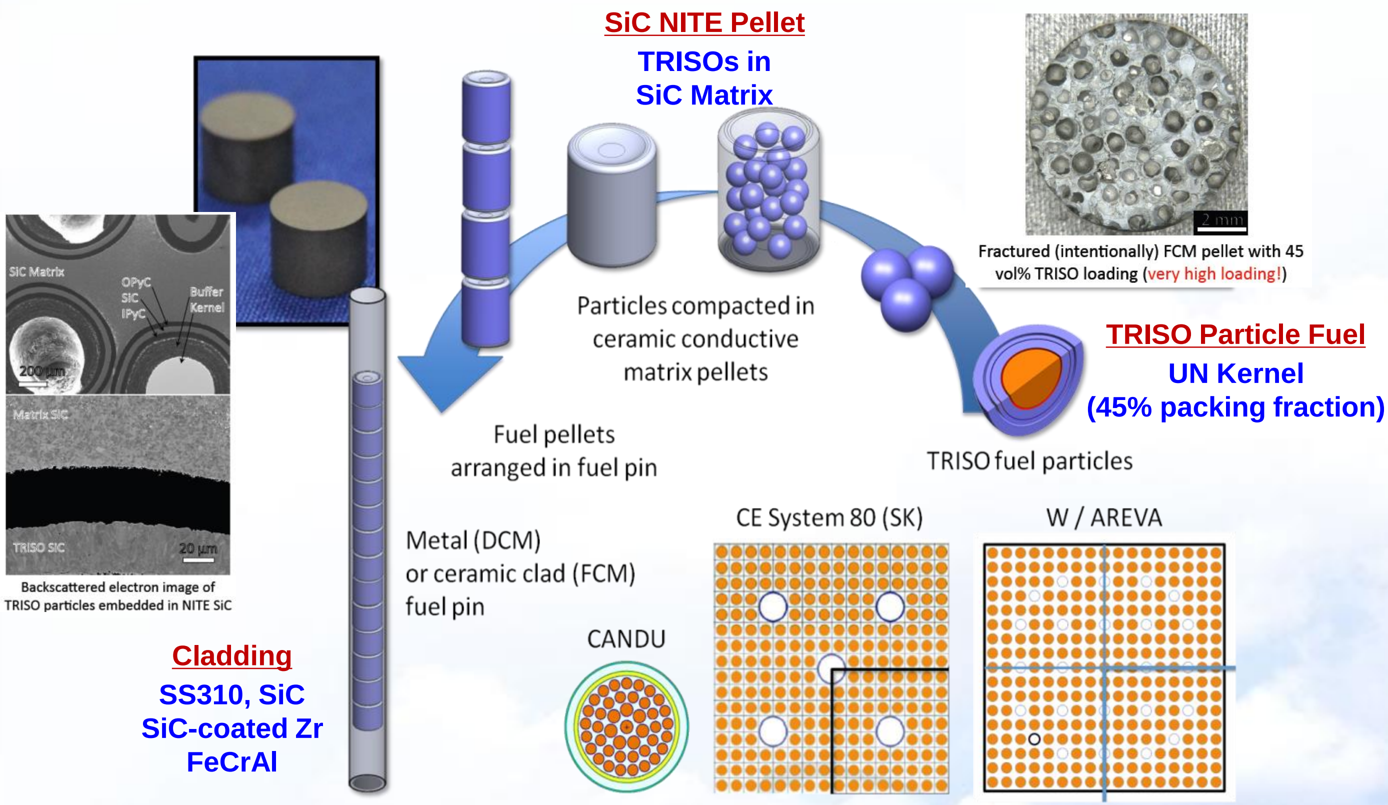
I-NERI Project Milestone

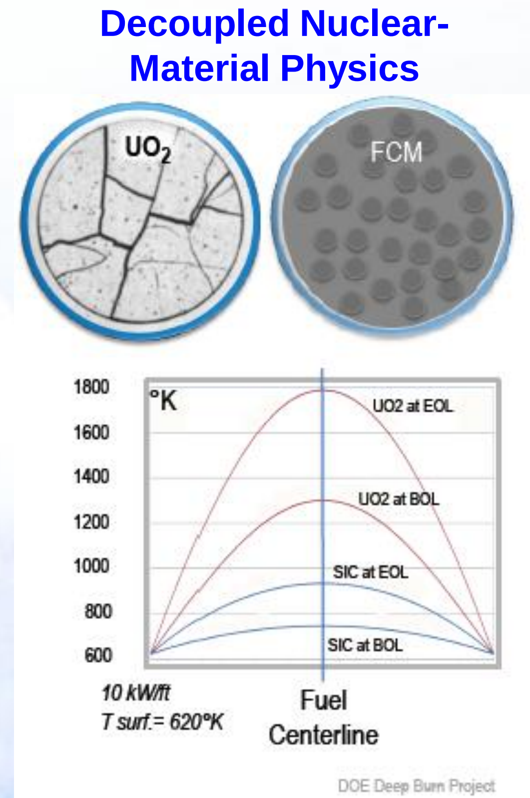
2011	I-NERI FY 2012	I-NERI FY 2013	I-NERI FY 2014	2nd Phase I-NERI
FCM Fuel Concept				
Design Analysis Tasks (KAERI/ USNC)	FA Design Selection, Analysis System Pre. Scoping (ND/TH/Safety)	FCM* Transition/ Equilibrium Core LP, Scoping Analysis (ND/TH/Safety)	Mixed** Transition/ Equilibrium Core LP, Design Analysis Margin Assessment (ND/TH/Safety/Fuel)	Feasibility to SMR Cores
Fuel Qualification Tasks (ORNL/ KAERI/USNC)	FCM Fuel Sample, Analysis Models (Fuel)	Irradiation Test (HFIR), Analysis System (Fuel)	Post-Irradiation Test, Fuel Performance Analysis (Fuel)	Extended Qualification and Fabrication
	COMPLETED			PENDED

* FCM Core: Core initially fully loaded and reloaded with FCM FAs
** Mixed Core: Core initially fully loaded with UO_2 FAs and reloaded with FCM FAs

II. FCM Fuel

- Fully Ceramic Microencapsulated (FCM) Fuel
  - Coated Particle Fuel: TRISO (Proven from HTGR)
  - Pellet: TRISOs in High Density SiC Matrix
  - Cladding Options: SS310, SiC, SiC-coated Zr, FeCrAl





Features of FCM Fuel

❑ Accident Tolerance

- Resistance to Meltdown: TRISO (< 1800^{o}C), SiC Matrix (< 2700^{o}C)
- Resistance to Fission Product Release: TRISO & SiC Matrix Barriers
- Resistance to Hydrogen Generation: Low Reaction Rate Cladding
- Resistance to Fuel Thermo-Mechanical Degradation
  - Highly Conductive Fuel: Low Fuel Centerline Temp (~ 800^{o}C)
  - Mechanical Integrity of Cladding during LOCA: Low Internal Press.

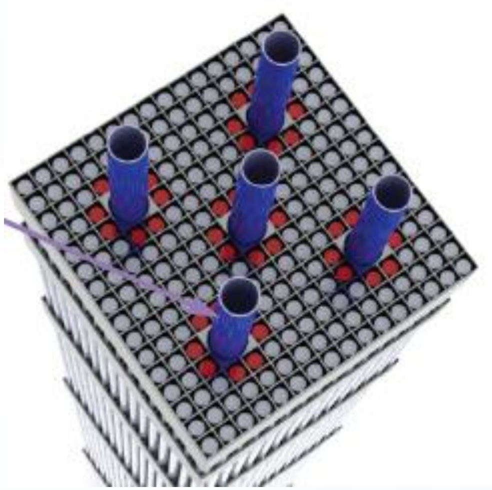
❑ Additional Features

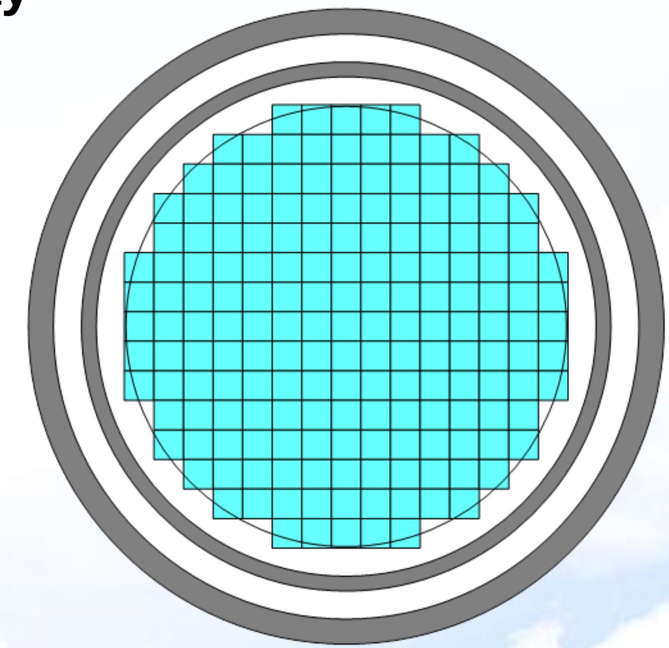
- Benefit in Spent Fuel Management and Disposal
  - Deep-Burn of Recycled Spent Fuel
  - Dry Disposal and Long-Term Containment of Radioactivity
- Proliferation Resistance
  - Difficulty in Pu Extraction and Low Quality W-Pu in Spent Fuel
- Flexible Applications to Other Reactor Types like HTGR, CANDU, …

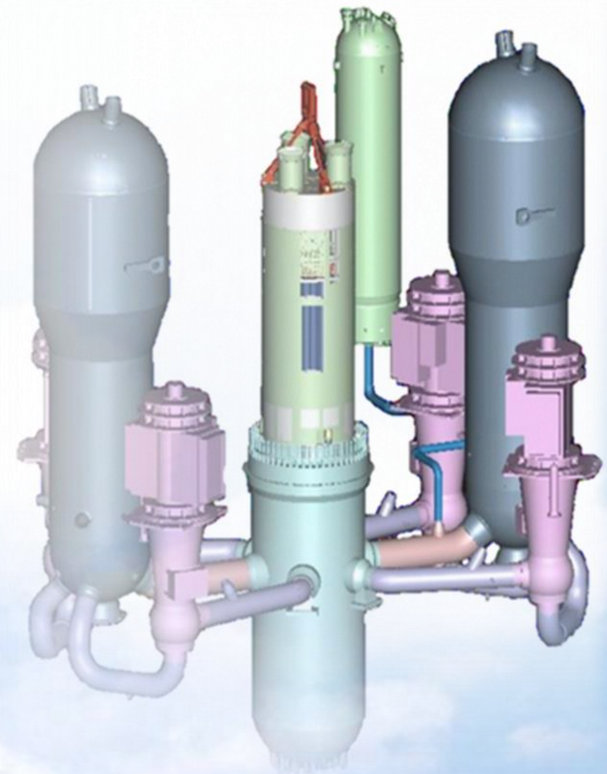
III. Work Progress

Reference PWR: OPR-1000

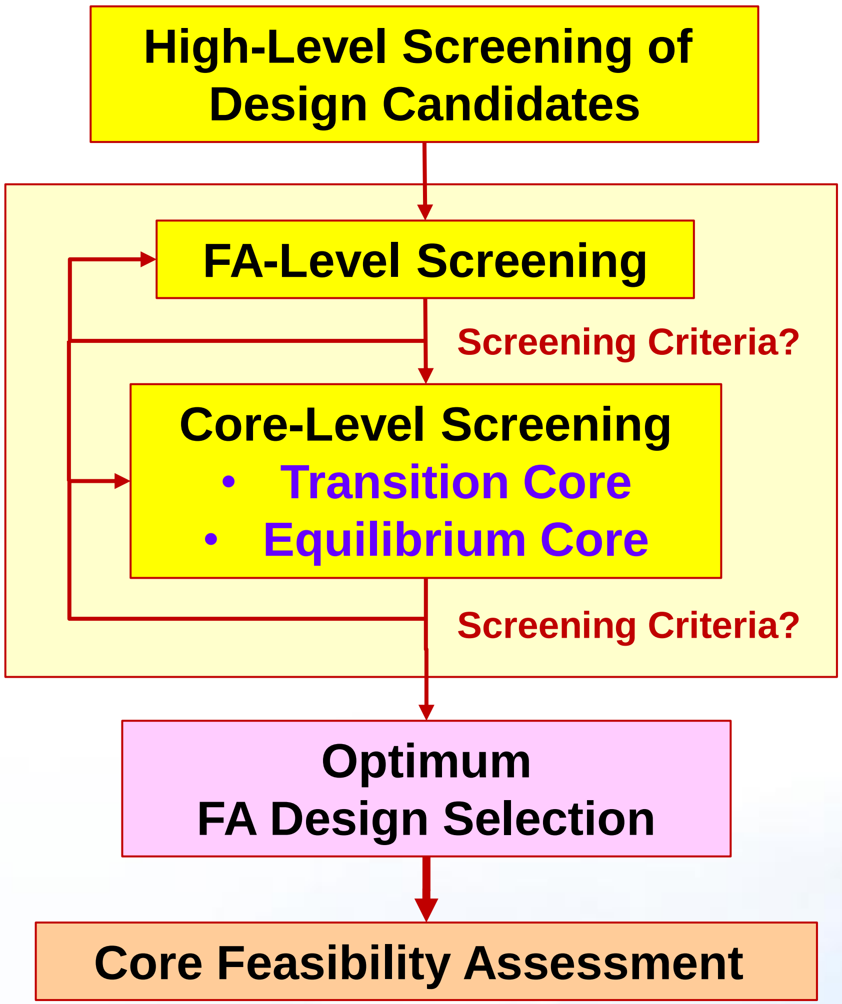
- 1000MWe Korean Standard PWR
  - Upgraded Design of ABB-CE System 80
- Core consists of 177 FA's
  - 16x16 FA Array







III-1. Design Selection of FCM FA

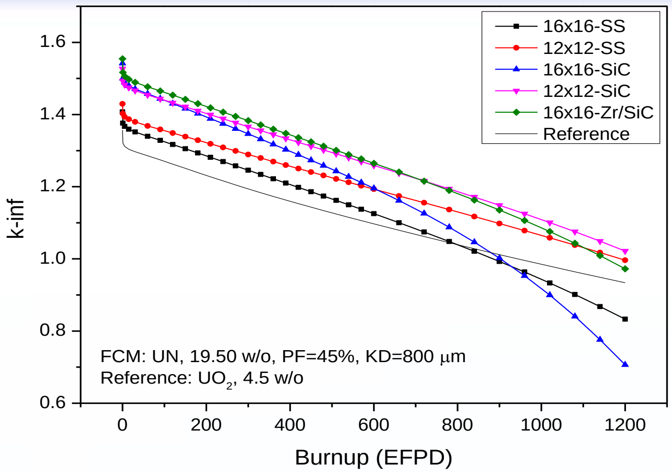


Key Screening Criteria

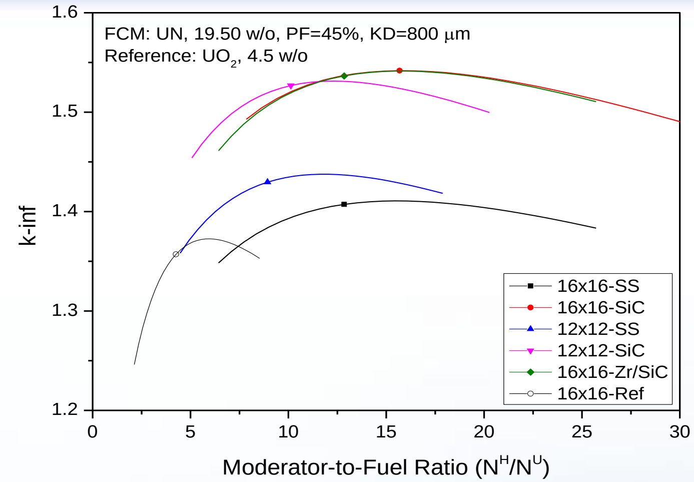
- Nuclear, Thermal-Hydraulic and Mechanical Compatibility with Conventional UO_2 Core
- Enhanced Accident Tolerance

Design Parameter	Criteria	Remarks
FA Array	Compatibility	Mech.
Fissile Inventory	UN, UO_2, UCO	Nucl.
Enrichment	up to 20 w/o	Non-Prolifer.
Kernel Diameter	500 ~ 800μm	Nucl./Mech.
Buffer Layer	50 ~ 100μm	Nucl./Mech.
Packing Fraction	up to 45%	Nucl./Mech.
Cycle Length	18 month	Nucl.
Kinetic Parameters	Compatibility	Nucl.
Cladding Material	SS310, SiC, Zr-SiC, FeCrAl	Nucl./Safety
Cladding Thick.	No Buckling	Mech.
ΔP/Lift Force	< 120% of Ref.	TH
DNBR	Compatibility	TH
Fuel Rod Gap	> 2mm	Mech.

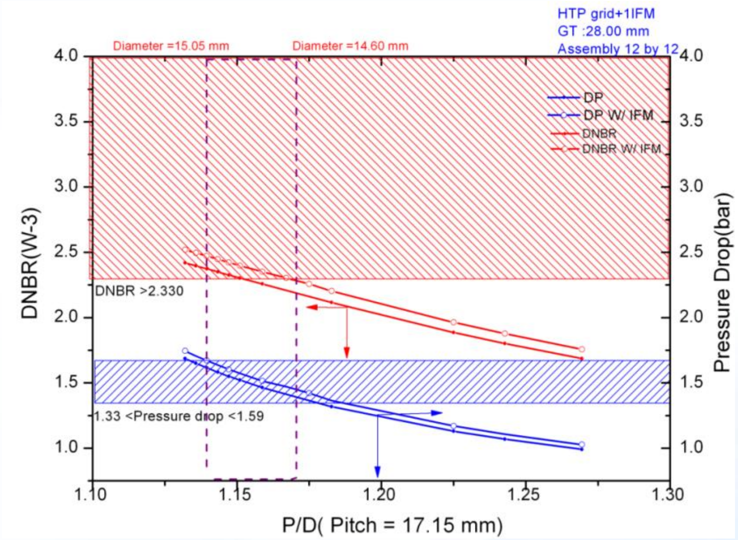
FA-Level Compatibility



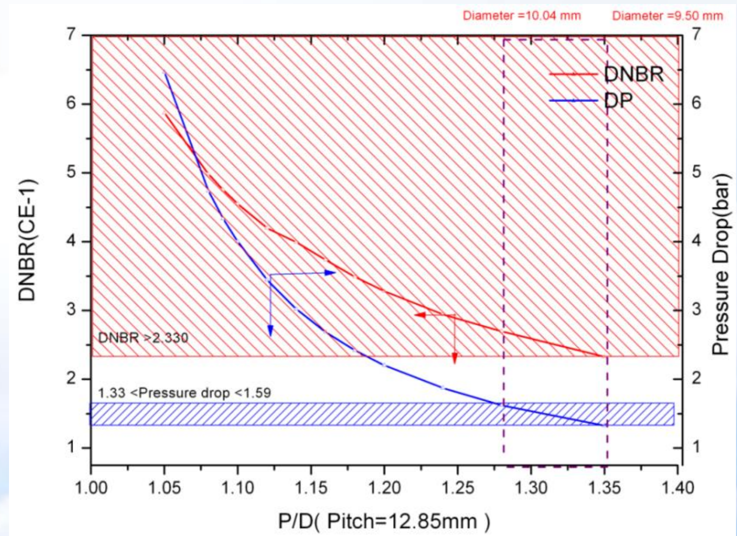
k-infinite vs. Burn-up



Metal-to-Water Ratio

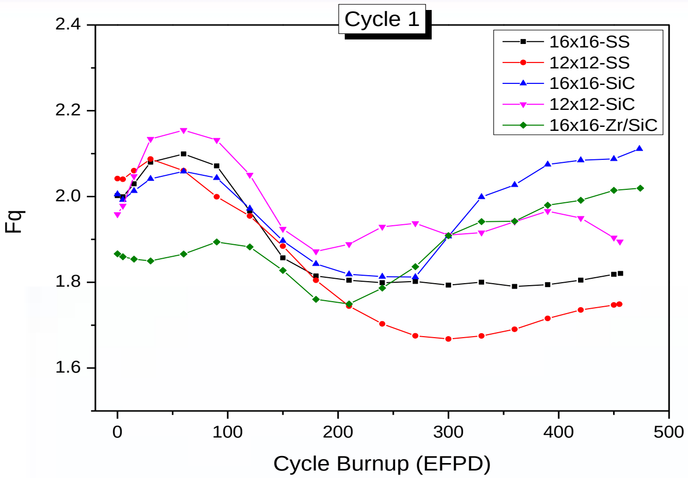


DNBR and ΔP vs. P/D Ratio for 12x12 FA

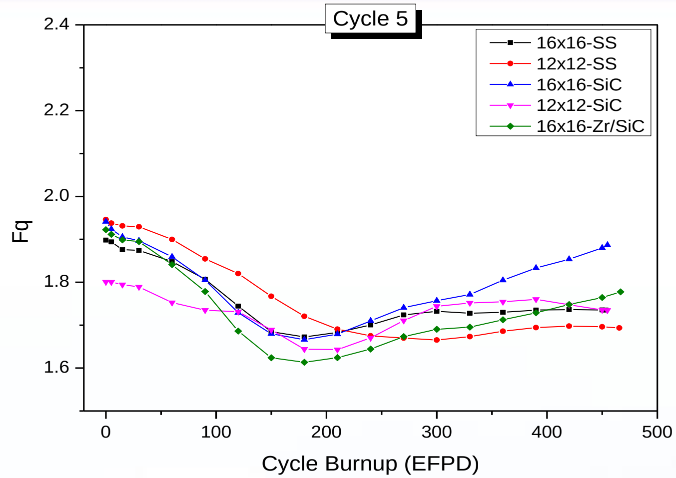


DNBR and ΔP vs. P/D Ratio for 16x16 FA

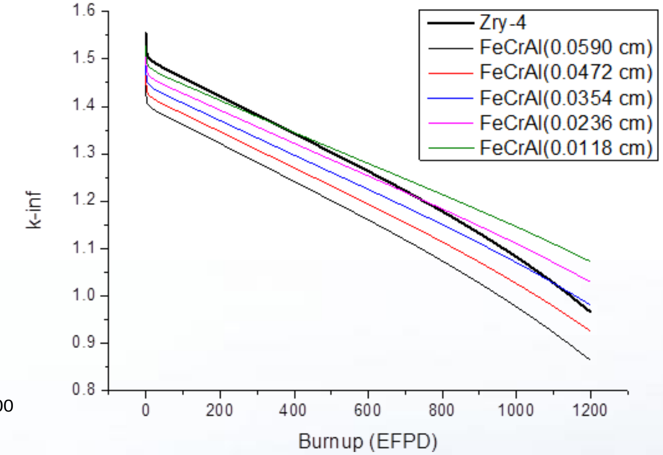
Core-Level Compatibility and Enhanced Safety



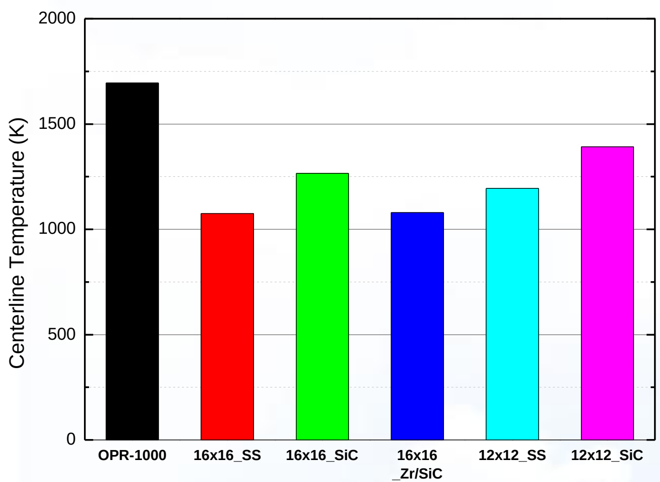
Power Peaking Factors (Initial Core)



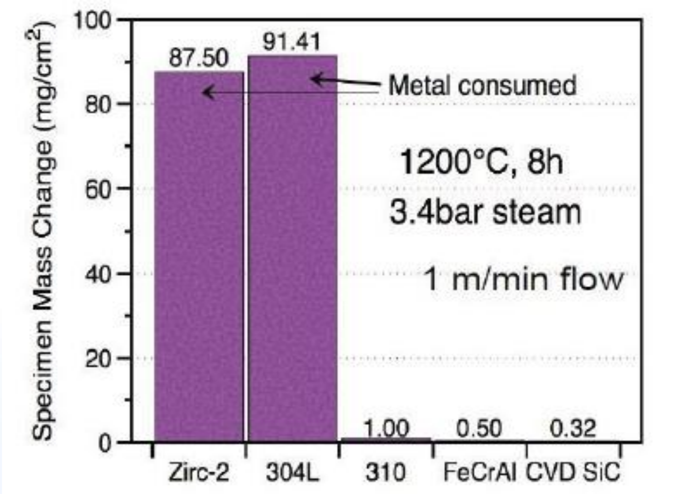
Power Peaking Factors (Equilibrium Core)



FeCrAl vs. Zircaloy Cladding



Fuel Initial Stored Energy



Oxidation Resistance of Cladding Material

Accident Scenario	Key Safety Parameter	Gain in Margin
LOFA	MDNBR	0.09
LBLOCA	PCT	82K
CEA Ejection	PCT	302K
LBLOA w/o SI	Grace Time	1028 s
SBO	Grace Time	112 min

Gain in Safety Margin by FCM FA

Optimized FCM FA Design

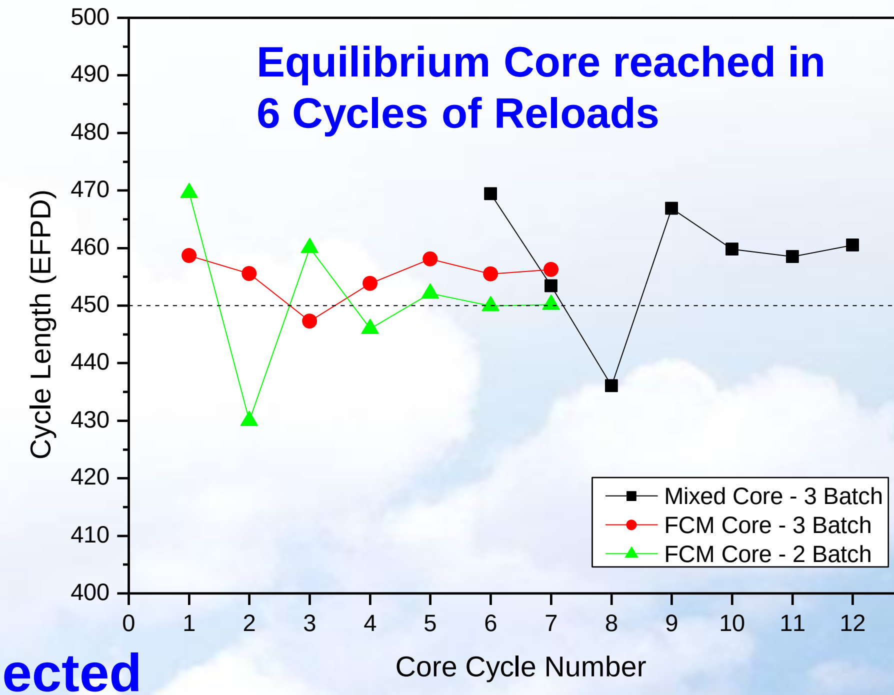
- Optimized FCM FA Design was selected from 3-Step Screening
  - 16x16 FA
  - FeCrAl Cladding
- Optimized FCM FA Design is used for Core Feasibility Assessment
  - Core Neutronics
  - Core Thermal-Hydraulics
  - Fuel Performance
  - Core Safety

Item		Unit	16x16-Ref.	16x16-FCA
Assembly Pitch		cm	20.78	←
Cell Pitch		cm	1.285	1.285
TRISO Coated Particle	Fuel Material / Enrichment		UO_2 / < 5%	UN / < 20%
	Fuel Density	g/cc	10.176	14.32
	Fuel Kernel Diameter	μm	-	800
	Buffer Layer	μm	-	50
	Inner PyC Coating Layer	μm	-	35
	SiC Coating Layer	μm	-	35
	Outer PyC Coating Layer	μm	-	20
Fuel Rod	Pellet/Compact Material		UO_2	TRISO/SiC Matrix
	Pellet/Compact Density	g/cc	10.176	3.18
	Pellet/Compact Radius	cm	0.4095	0.4541
	TRISO Packing Fraction	%	-	45
	Clad Material		Zry-4	FeCrAl
	Clad Density	g/cc	6.55	7.10
	Clad Inner Radius	cm	0.4180	0.4626
	Clad Outer Radius	cm	0.4750	0.5000
	Clad Thickness	cm	0.0570	0.0374
	Gap Thickness	cm	0.0085	←
	Rod to Rod Spacing	cm	0.3350	0.2850
Heavy Metal Loading, 1/4		g/cm	279.0	94.5

III-2. Task 1: Neutronics Exploration

❑ Core Loading Pattern Search to Equilibrium Core

- Nuclear Analysis System
  - DeCART-2D (2D Transport Lattice) / MASTER (3D Nodal Diffusion)
- Among 2 Types of Core Configuration
  - FCM Core
    - Core initially fully loaded and reloaded with FCM FA's
    - 2-/3-Batch, 18 Months Cycle
  - Mixed Core
    - Core initially fully loaded with UO_2 FA's and reloaded with FCM FA's from Cycle 6
    - 3-Batch, 18 Months Cycle



- **Mixed Core Configuration was selected as a Reference Core for Further Assessment**

Core Physics Parameters

❑ Core Loading Pattern – Equilibrium Core

Mixed, 3-Batch

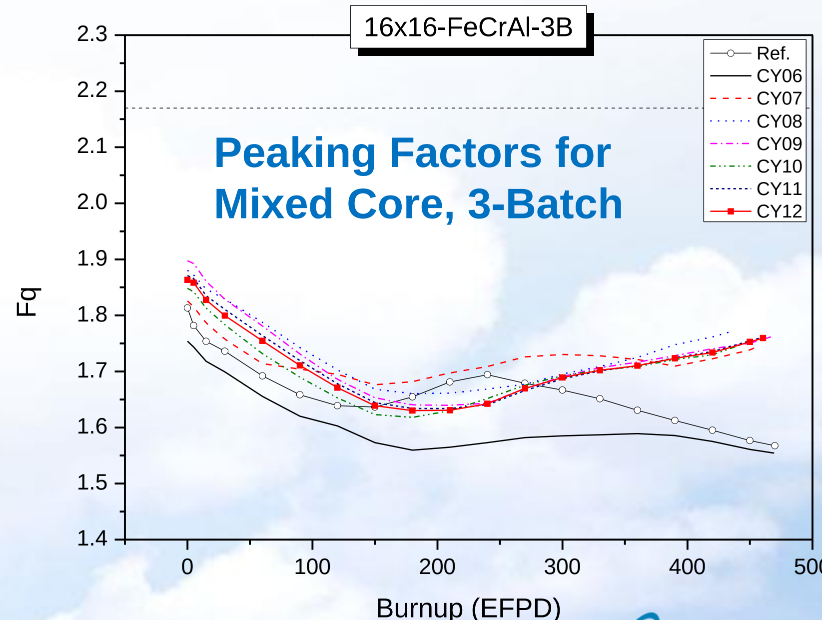
T3	O3	F3	O3	O1	O3	F3	O2
	F2	T2	F2	T3	F3	T1	F1
		O3	T2	O2	O1	F2	T1
			F3	O3	O2	F1	
				F3	O1	T2	
					F1		

FA Type	# of FAs	Batch Total
T1/T2/T3	20/20/9	49
O1/O2/O3	20/20/24	64
F1/F2/F3	20/20/24	64

FA Type	No. of Fuel Rods	Enrichment (w/o U-235) [N/Z/A]	No. of BA Rods	BISO PF (%)
F1	136/100	18.9/17.1/18.1	236	0.8
F2	184/52	18.9/17.1/18.5	236	2.4
F3	136/100	18.9/17.1/18.1	236	2.8

❑ Generation of Core Physics Parameters

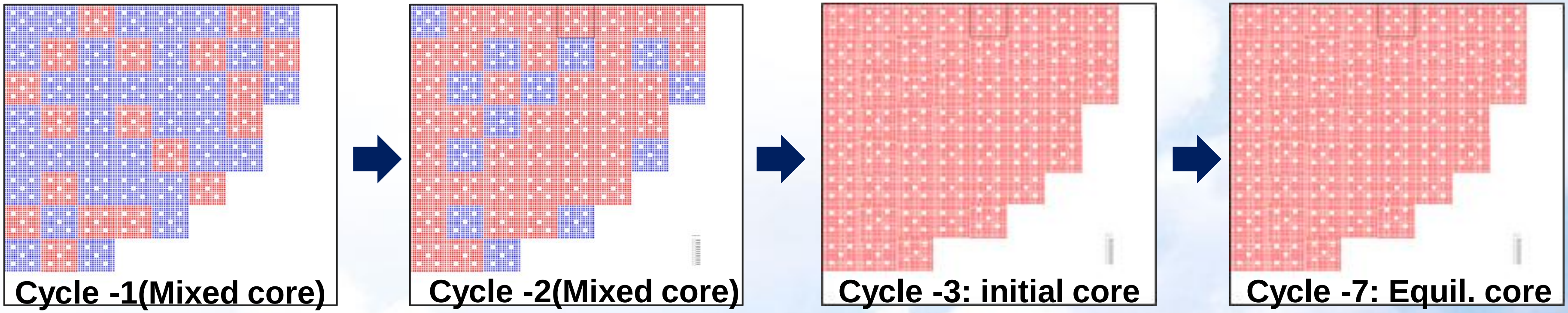
- **Power Distributions, Shutdown Margin, Kinetic Parameters, …**
- **All Parameters meet Interim Design Acceptance Criteria**



III-3. Task 2: Core TH Assessment

Core TH Margin Assessment for a Mixed Core Configuration

- Core TH Design Analysis System
  - MATRA (Pin-by-Pin Full Core Subchannel TH (Parallel Processing))
  - Subchannel Cross Flow Loss Coefficient Model for a Mixed Core
- Core TH Assessment Model from Mixed to Equilibrium Cycles
  - Pin-by-Pin Quarter Core Model
  - Core Physics Parameters obtained from Core Neutronics Exploration



Loading pattern (Blue: UO_2 FA (PLUS-7), Red: FCM FA)

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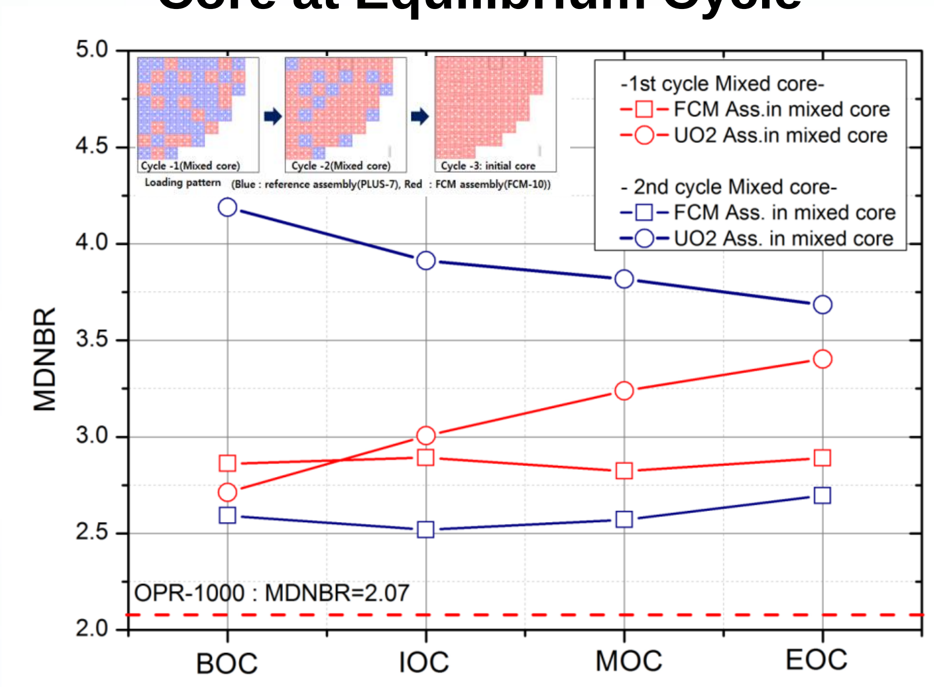
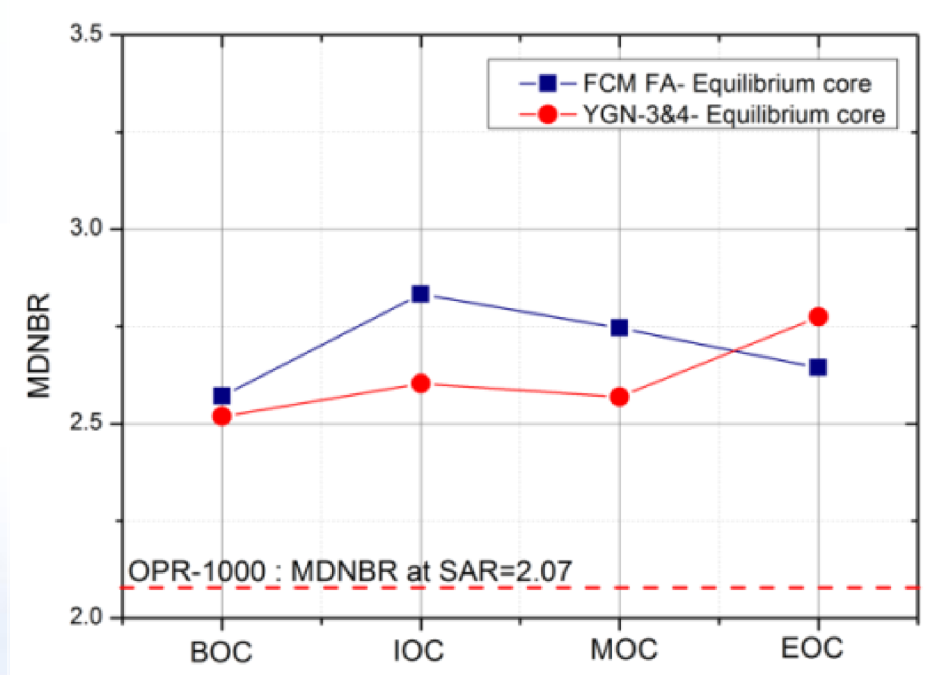
Core DNBR Margins from Mixed to Equilibrium Cycles

- MDNBR Margins available from Mixed to Equilibrium Cycles
- Comparable Margins in FCM Core with those of a Reference UO_2 Core at Equilibrium Cycle



MDNBR of FCM and UO_2 FAs in Mixed Transition Cycles 1 & 2



Comparison of MDNBR with Reference UO_2 Core in Equilibrium Cycle

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III-4. Task 3: Safety Assessment

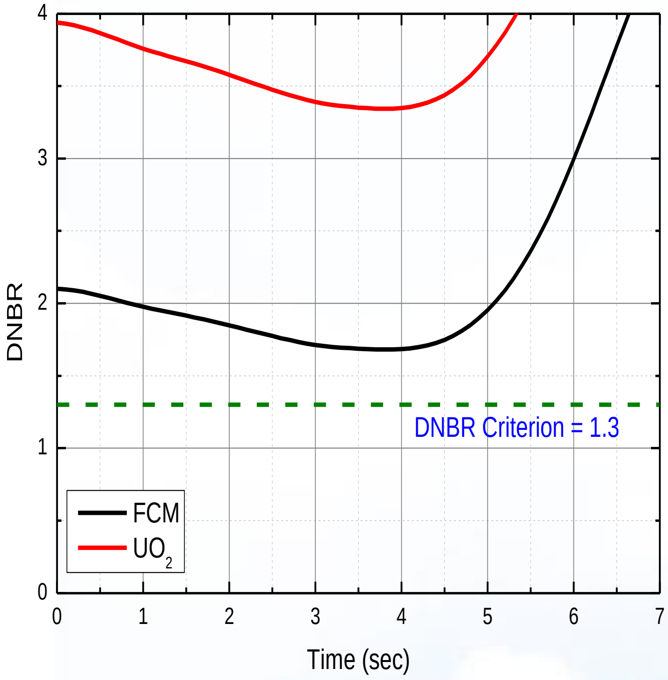
- **Safety Margin Assessment for a Mixed Core Configuration**
  - Accident Analysis System
    - MARS (System TH)
    - Thermal Conductivity Model of Irradiated FCM Pellet
  - Safety Margin Assessment from Mixed to Equilibrium Cycles
    - Limiting Design Basis Accidents
      - Complete Loss of Flow Accident (LOFA)
      - CEA Ejection Accident (REA)
      - Large Break Loss of Coolant (LBLOCA)
    - Beyond Design Basis Accidents
      - LBLOCA without Safety Injection (SI)
      - Station Blackout (SBO)
    - Core Physics Parameters obtained from Core Neutronics Exploration

Design Basis Accidents

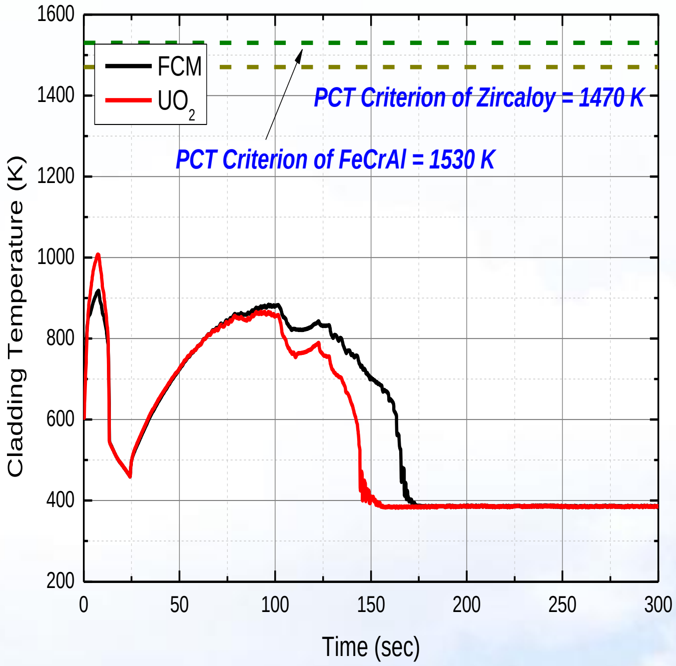
❑ Mixed Transition Cycles (Reload Cycle 1 and 2)

- Safety Margins available for both FCM and UO_2 Fuels

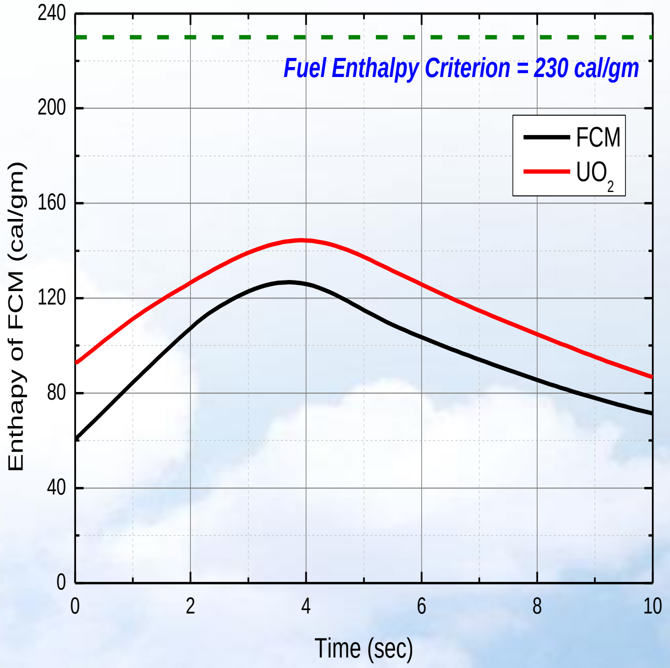
LOFA (1st Cycle, Mixed Core)



LBLOCA (2nd Cycle, Mixed Core)



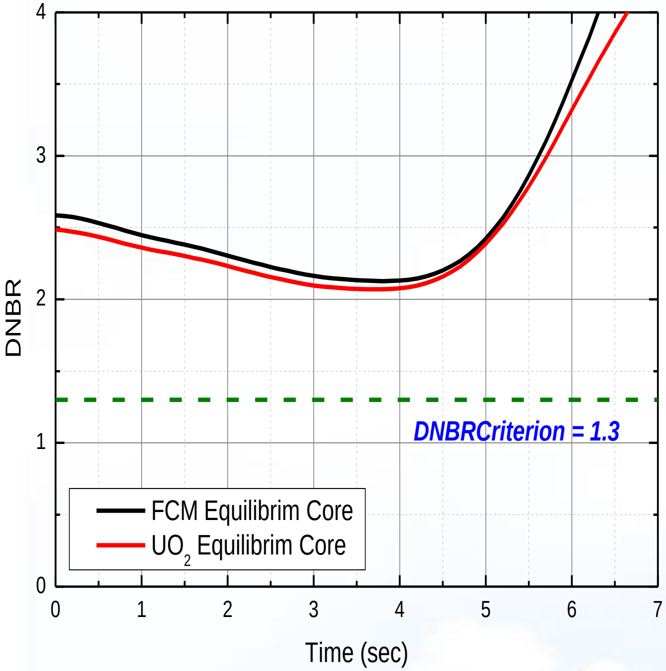
REA (2nd Cycle, Mixed Core)



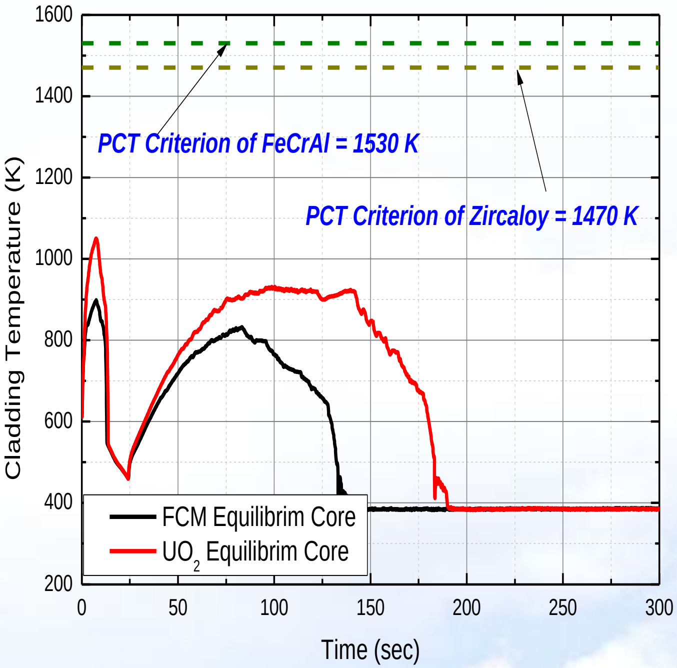
❑ Equilibrium Cycle (Full FCM Core)

- Increased Safety Margins in FCM Core when compared with those of a Reference UO_2 Core

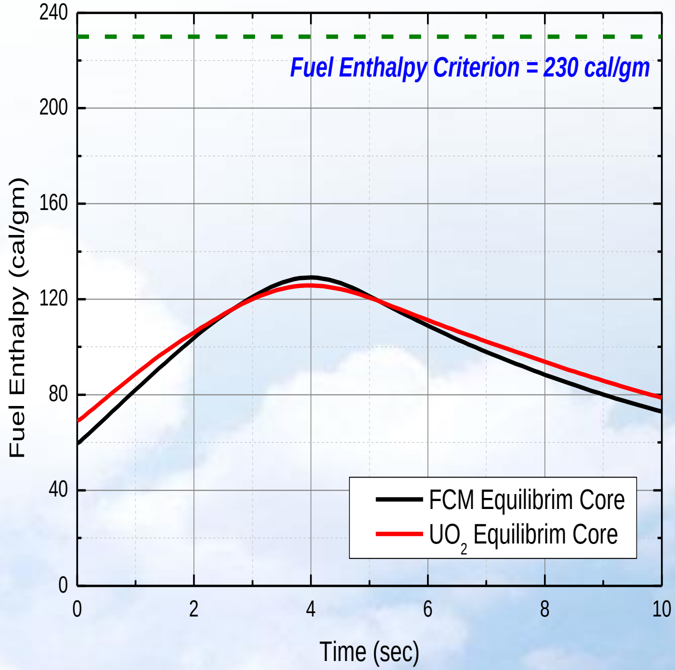
LOFA, Equilibrium Core



LBLOCA, Equilibrium Core



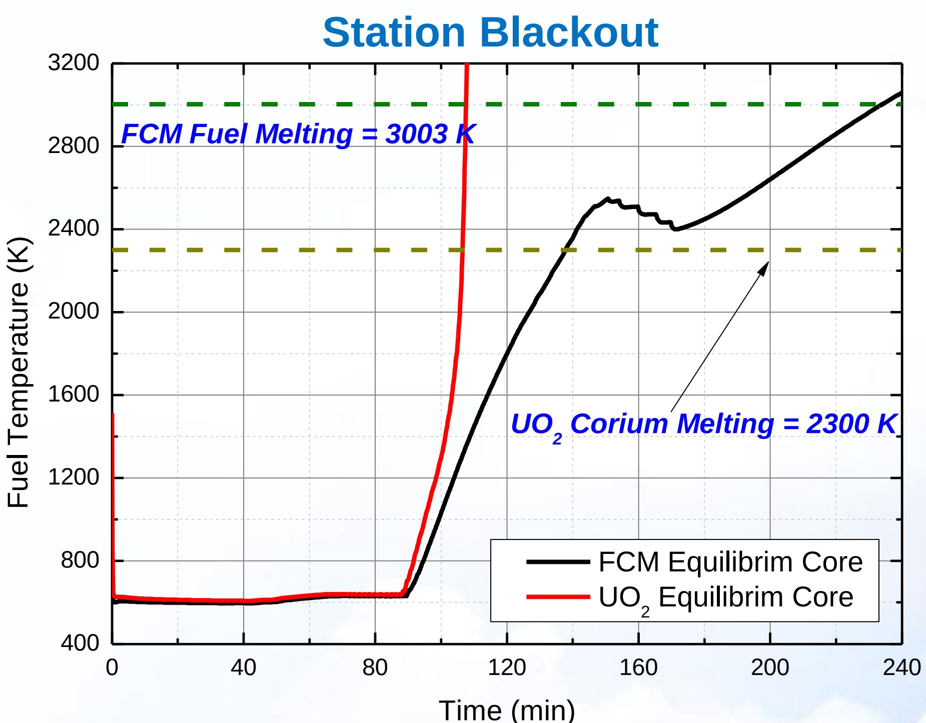
REA, Equilibrium Core

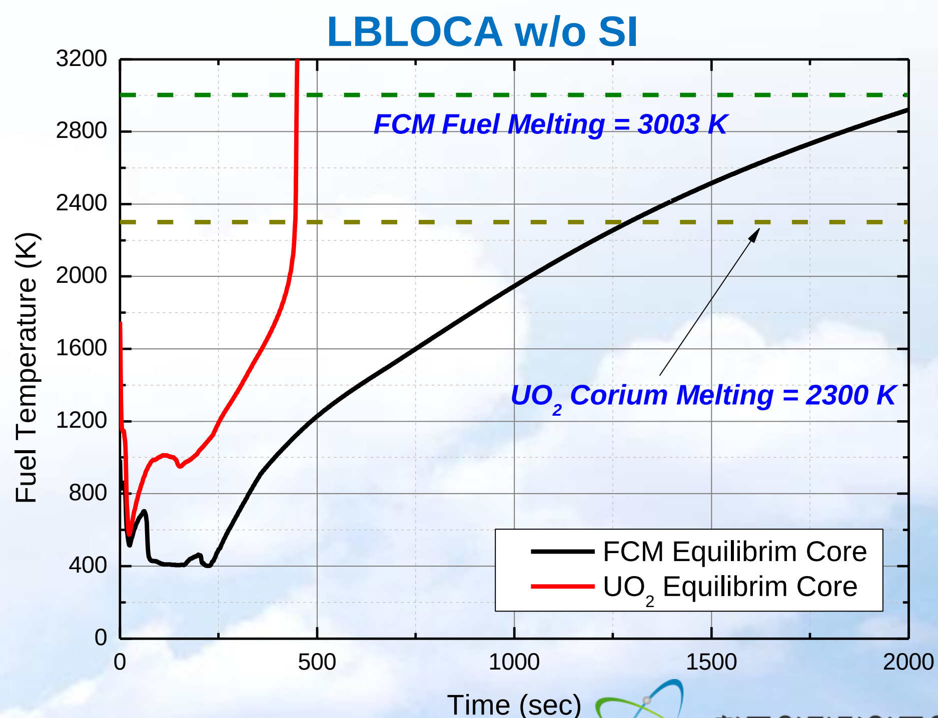


Beyond Design Basis Accidents

❑ Equilibrium Cycle (Full FCM Core)

- FCM Core allows Longer Grace Times for Operator Action when compared with those of a Reference UO_2 Core
  - SBO: 127 min Longer
  - LBLOCA w/o SI: 26 min Longer





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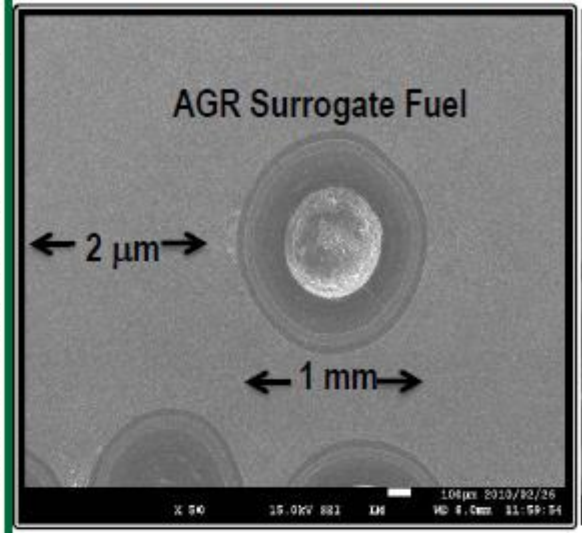
III-5. Task 4: Fuel Qualification

- **Fuel Qualification (ORNL)**
  - FCM Fuel Fabrication
    - UN Fuel Kernel Fabrication: ~ 96% dense $UN_{0.99}C_{0.01}$, ~ 800μm Dia.
    - R&D of Coating Processes for UN Kernels
    - FCM Pellet Fabrication
      - Overcoating TRISO with NITE SiC
  - FCM Fuel Irradiation
    - Surrogate FCM Fuel irradiation in HFIR
    - Possible Irradiation of UN FCM in ATR: Plan

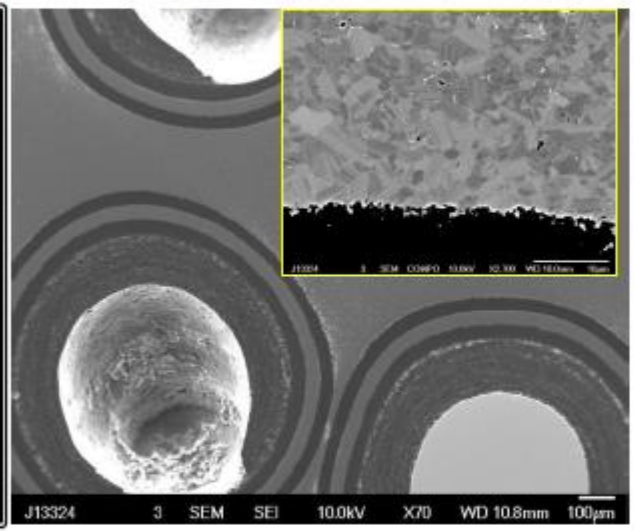
- **Fuel Performance at OPR-1000 Operating Conditions**
  - Coated Particle Performance Analysis
    - COPA: TRISO Performance Analysis System
  - FA Performance Analysis
    - Fuel Rod Buckling and Fuel Assembly Vibration

Fully Ceramic Matrix Fuel Development

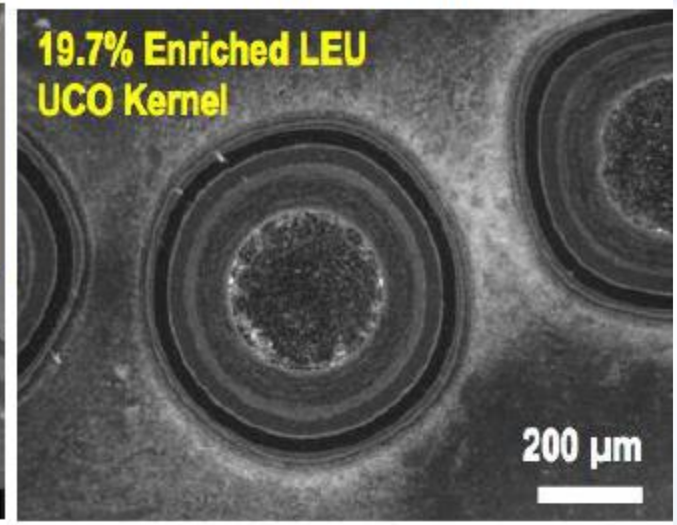
Q2/2010
FCM Introduction (surrogate)



Q1/2011
Optimized Matrix

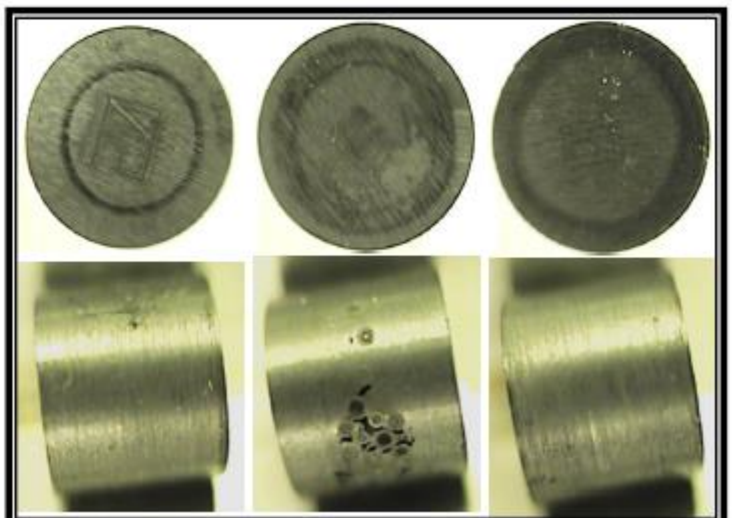


2Q/2011-Ongoing
U-FCM Production

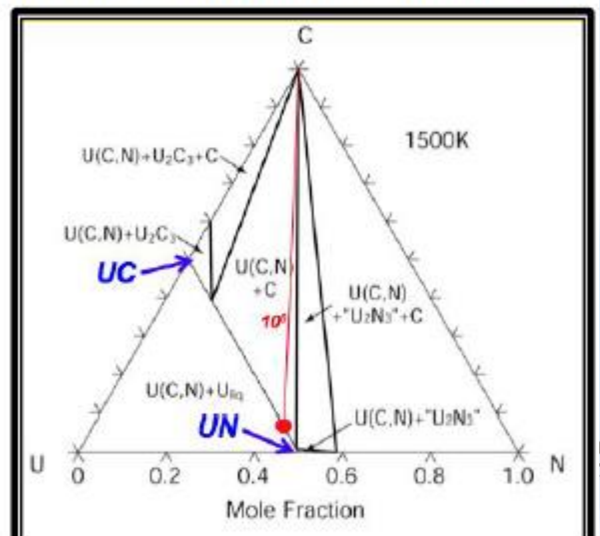


3Q/2011
Surrogate Irradiation

1Q/2012-Ongoing
PIE of Surrogate FCM

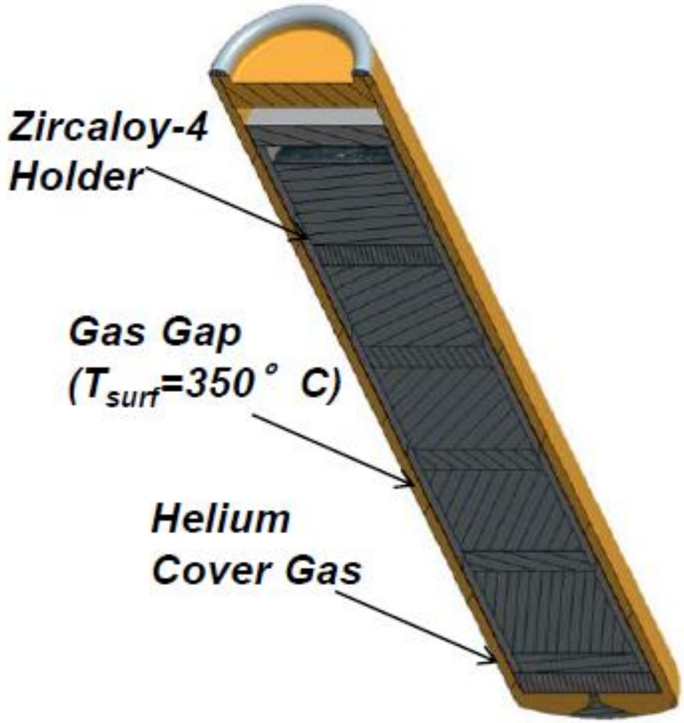


2Q/2012-Ongoing
Model, Demonstrate, Produce UN



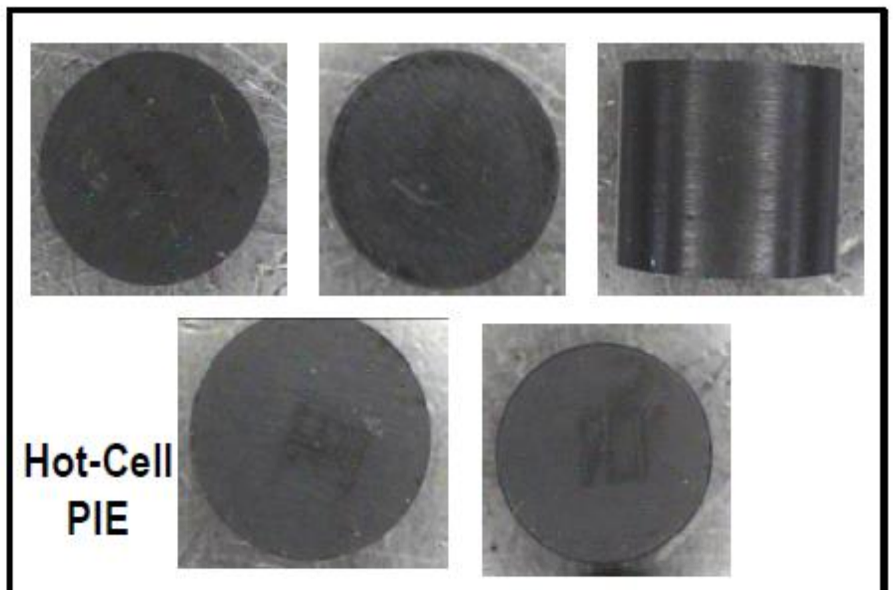
FCM HFIR Irradiation

- 20 FCM pellets with variable loading of surrogate particles irradiated in 4 capsules (to 130% LWR lifetime)
- TRISO loading fractions : 0 - 41 vol%
- FCM pellets are in 17x17 PWR rod geometry and are clad in Zircaloy-4
- Pellet temperature : 425 °C



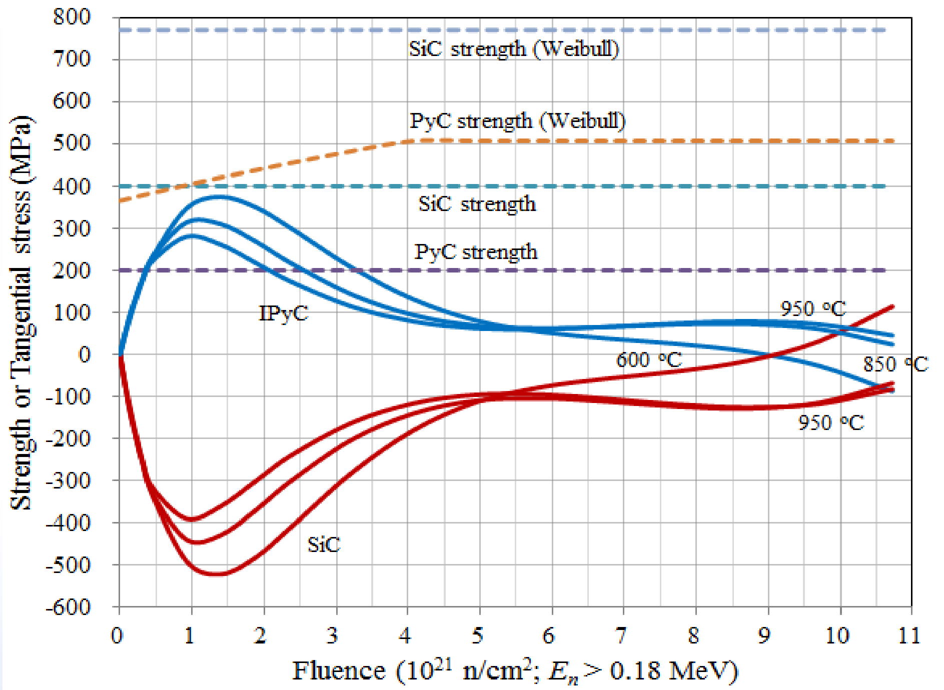
No visual change in appearance

No reaction with Zircaloy



Coated Fuel Particle Performance

- Coated Fuel Particle Performance over Fuel Lifetime
  - Fast Neutron Fluence (> 0.18MeV): up to 10.7x10^{21}n/cm^2
  - Internal Fission Gas Pressure
    - Negligible at PWR Conditions (< 850^{o}C)
    - Little Cs-Release under 910^{o}C
  - SiC Layer Integrity
    - No Failure up to 10.7x10^{21}n/cm^2
    - At Higher Fluence, SiC Layer Integrity may be challenged by Swelling of iPyC Layer, however, it may survive if iPyC fails
    - Irradiation Tests @ PWR Conditions are required to remove Uncertainties involved in iPyC Deformation Model



Tangential Stresses on Coating Layers (No Failure of iPyC assumed)

IV. Summary and Conclusions

- **Accident Tolerant FCM Replacement Fuel Design for OPR-1000 has been proposed and demonstrated Feasible**
  - Optimum FCM FA Design: 16x16 FA with FeCrAl Cladding
  - Nuclear, Thermal-Hydraulic and Mechanical Compatibility
  - Enhanced Accident Tolerance
  - Fuel Fabrication and Qualification through Irradiation Tests (ORNL)

- **Further Works required for Deployment**
  - Characterization and Qualification of FCM Fuel through extensive Irradiation and Experiment Program: Performance and Safety
  - Scale-up of Fuel Fabrication Technologies to Engineering Scale
  - Collaboration with Nuclear Industries
  - Extended Application of FCM Fuel to SMRs, CANDUs, HTGRs, …

… thank you for your kind attention