CNL Lattice Physics Assessments of Alternative/Advanced Fuels for PWR-SMRs (Draft Presentation for SPANS 2024)

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Talk #7: CNL Lattice Physics Assessments of Alternative/Advanced Fuels for PWR-SMRs

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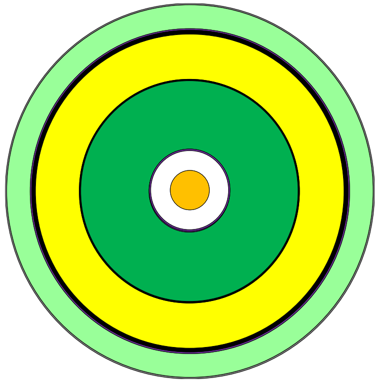

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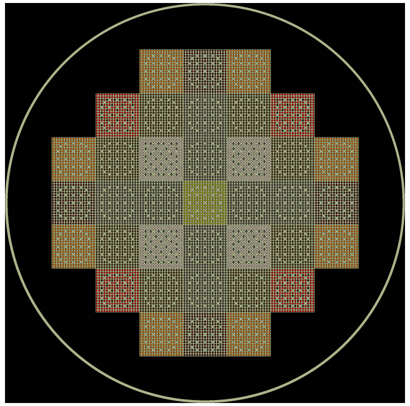
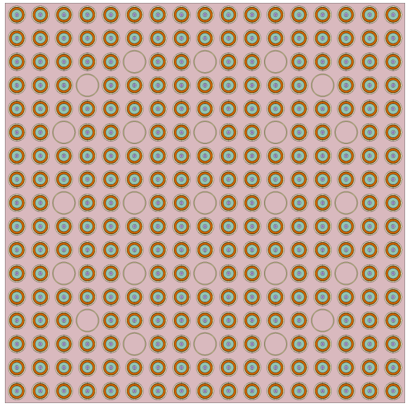




Introduction (1/3)

- **Small Modular Reactors (SMRs) based on light water reactor technologies (LWR) at high Technology Readiness Level (TRL).**
- **LWR-SMR Vendors looking to build in Canada**
  - **GEH-Hitachi (BWRX-300), NuScale (iPWR)**
- **Conventional fuel:**
  - **NuScale iPWR: 4.05 wt% U-235/U, UO2, Zircaloy-4 clad.**

- **Can we do better? Can Alternative/Advanced fuels help?**
  - **Higher burnup, longer fuel lifetime?**
  - **Improved utilization of nuclear fuel resources?**
  - **Improved resilience and toughness?**
  - **Better thermal margins? Lower peak fuel temperatures?**

Introduction (2/3)

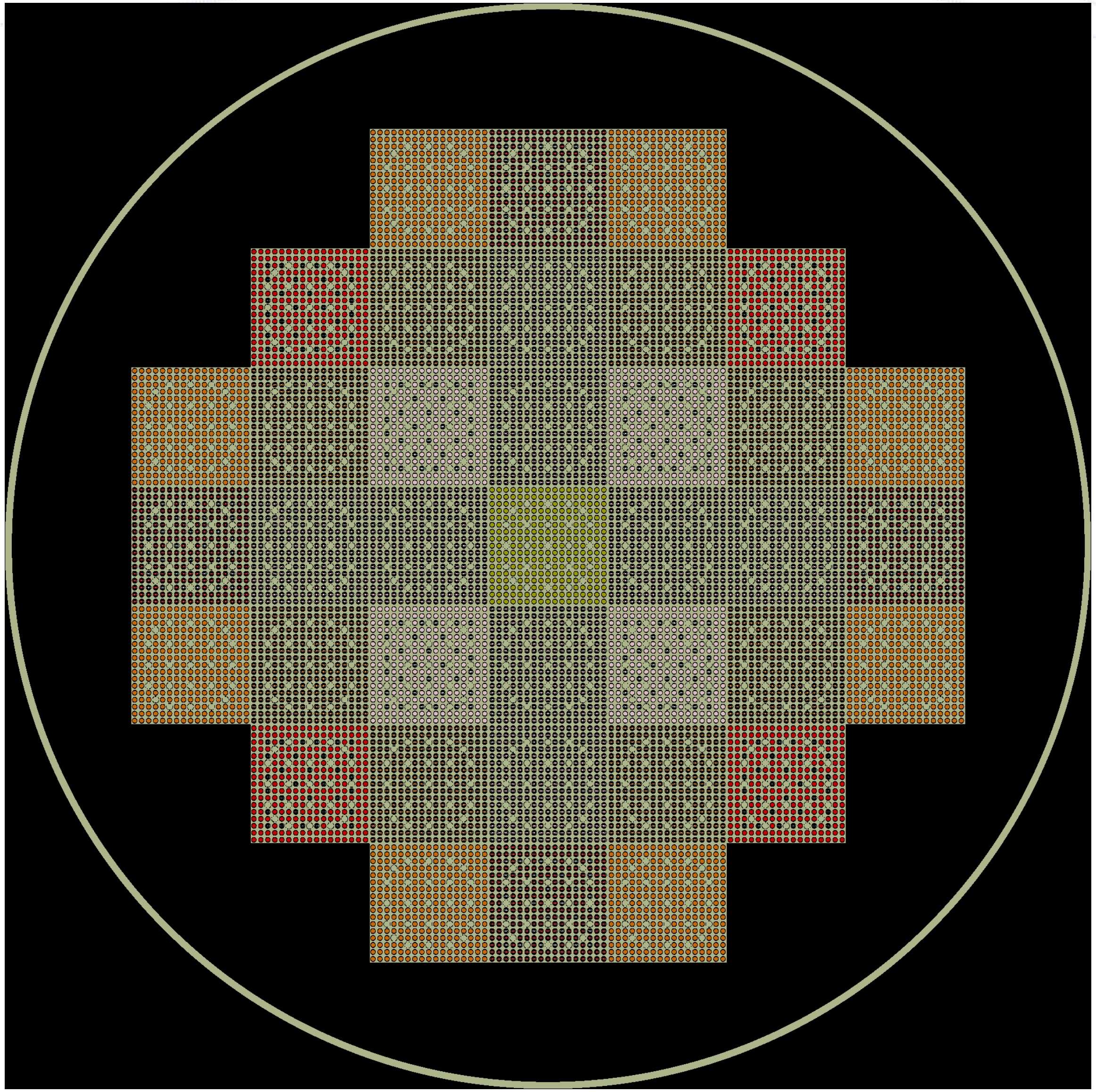
- **Motivation for Study:**
  - Long-term nuclear energy sustainability.
  - Alternative fuels could enhance performance characteristics (burnup, fuel lifetime).
  - Alternative fuels could improve resilience, if pushed to higher burnups.
  - Harness other fertile fuels (such as thorium), and consume stockpiles of plutonium found in PWR, BWR and PT-HWR fuel.
- **Practical Reason for Study:**
  - Alternative fuels give more flexibility and options from a reactor physics perspective.
  - Potential for better performance (with better long-term economics)
- **What we are trying to learn:**
  - What are the performance characteristics? How are they better than conventional fuel?
- **Potential for Uptake:**
  - Reactor vendors and utilities looking at deploying PWR-SMRs will benefit from this work.
  - Supports government policies to ensure long-term nuclear energy sustainability.

Introduction (3/3)

- **Objective: Use Serpent to evaluate alternative fuels for PWR-SMR (~NuScale iPWR)**
  - **Carry out lattice physics calculations – we can infer performance for full-core.**
  - **Enhance performance/safety characteristics (ATF-like, TRISO-like)**
  - **Results could be applicable to iPWR and BWRX-300.**
- **Geometry of Fuel Assembly (FA) is same, but fuel materials, cladding, and coatings modified.**
- **Heterogeneous, multi-clad, multi-region annular fuel design.**
- **Tests to Evaluate:**
  - **Oxides, nitrides, carbides, oxy-carbides, and silicides (higher densities, better conductivity)**
  - **LEU (5 wt% U-235/U), LEU+ (10 wt% U-235/U), HALEU, (19.75 wt% U-235/U),**
  - **(U,Th), (Pu,Th), (Pu,Th,DU)**
  - **PWR-RGPu (~67 wt% Pu-fissile), HWR-RGPu (~72.5 wt% Pu-fissile/Pu)**
- **Evaluate performance characteristics**
  - **Exit burnup, fuel lifetime, fissile utilization, fuel composition.**

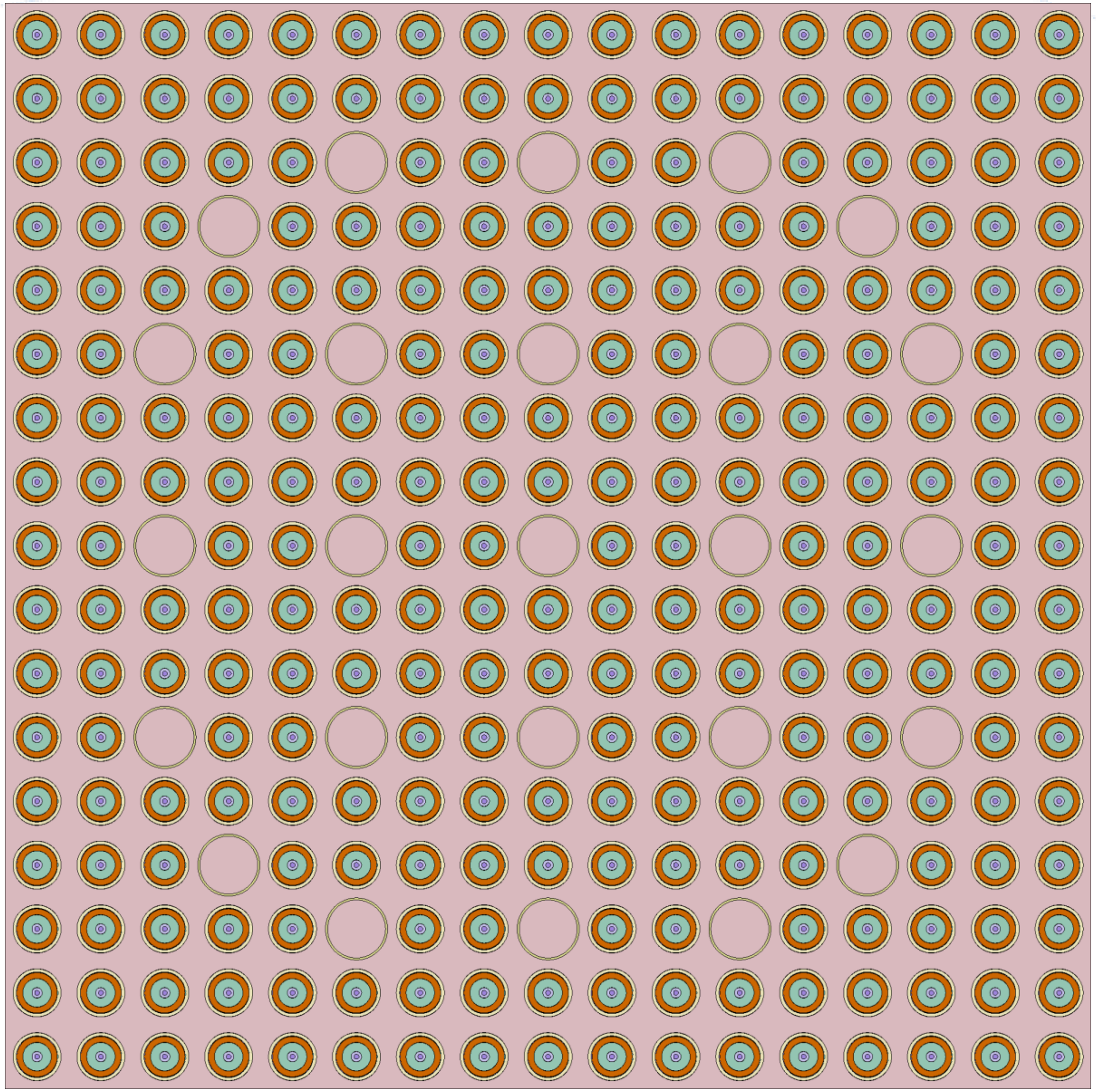
PWR-SMR – NuScale iPWR

- 160-MWth / 50-MWe (older design)
- 37 FA's, ~4.324 MWth/FA
- 200-cm height bare core
- ~146.8-cm, effective diameter of bare core
- Nominal design: 4.05 wt% U-235/U UO_2
- Other assemblies use fuel elements with 4.55 wt% U-235/U UO_2 mixed with Gd_2O_3.
  - Burnable neutron absorber (Gd_2O_3) to reduce boric acid requirements to control excess reactivity.
- 12.8 MPa, 258°C inlet, 314 °C outlet
- Natural circulation drives flow in core
  - No pumps needed; chimney effect



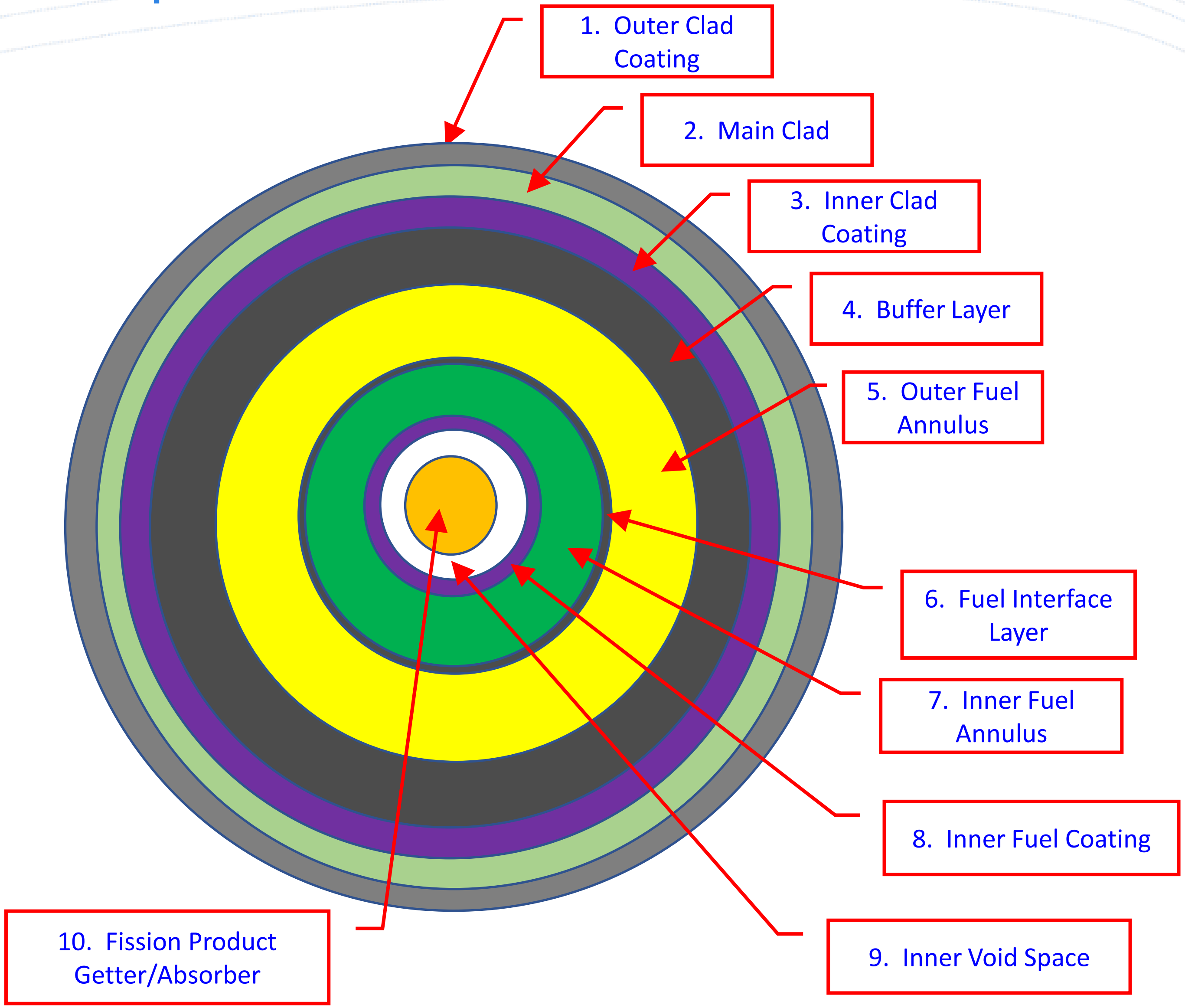
PWR-SMR Fuel Assembly

- **Modification of the 17×17 fuel assembly used in the NuScale iPWR**
- **Same fuel assembly geometry as NuScale iPWR**
- **Same outer element diameter (9.5-mm)**
- **264 Fuel Elements**
- **24 Water Holes / 1 Instrumentation Tube**
- **1.26-cm square pitch pin cell**
- **21.4-cm square pitch FA, 200-cm height FA**
- **H_2O Moderator/Coolant at ~0.757 g/cm^3, (12.8 MPa, 284 °C)**
- **Geometry/materials inside fuel modified.**
  - **Heterogeneous design.**
  - **Multi-layer.**
  - **Multi-clad**



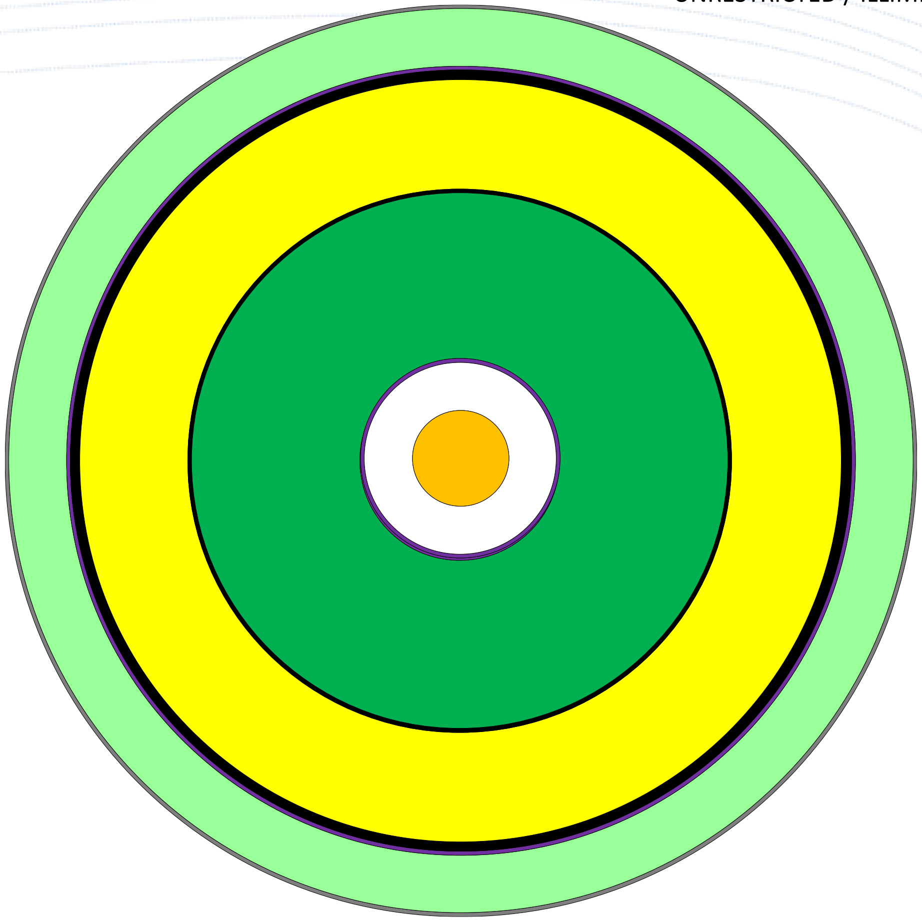
Modified Fuel Element Design Concept

- **Additional coating and barrier regions**
- **Protection of Zircaloy-4 clad.**
- **Enhanced resistance to fission products.**
- **Annular fuel with central void space.**
- **Potential for "getter material", MgF_2.**
- **Two fuel annuli – option for duplex fuel.**
  - **Outer annulus vol: ~12,859 cm^3 per FA.**
  - **Inner annulus vol: ~11,118 cm^3 per FA**
  - **Different fuels in outer/inner annuli.**
- **"ATF-like" and "TRISO-like" features**
- **Can fuel be fabricated? Yes. Leverages:**
  - **UK AGR Fuel (annular)**
  - **Historical work on Duplex-type fuel.**
  - **ATF-type fuel (coatings)**
  - **CANDU fuel (coatings for pellets)**
  - **Gas-cooled Fast Reactor fuel (annular)**
  - **TRISO fabrication methods.**



Design Concept Variants

- **Matrix Materials**
  - **Oxide, Nitride, Carbide, Oxy-Carbide, Silicides**
  - **Potential for higher densities, better conductivity.**
- **Fuels**
  - **LEU (5 wt% U-235/U) → Focus for Today**
  - **HALEU (19.75 wt%U-235/U)**
  - **(LEU+,Th), (HALEU,Th); LEU+ (10 wt% U-235/U)**
  - **(Pu,Th), (Pu,Th,DU)**
    - **7.5 wt% Pu/(Pu+Th)**
    - **15 wt%, 20 wt%, 30 wt% Pu/(Pu+Th)**
- **Two types of Plutonium tested:**
  - **HWR-RGPu (~72.5 wt% Pu-fissile/Pu)**
  - **PWR-RGPu (~67 wt% Pu-fissile/Pu)**
- **Up to 90 test cases.**
  - **Homogeneous and heterogeneous fuel pellets.**



Heterogeneous Multi-layer, Multi-Clad Annular Fuel Element (to Scale)

Lattice Physics Calculations with Serpent 2.1.31

- **3-D Monte Carlo lattice physics calculations of FA.**
  - Nuclear Data based on ENDF/B-VII.0
  - Reflecting boundary conditions.
  - 1000 generations (800 active, 200 inactive),
  - 1 million neutron histories/generation
  - Statistical uncertainty in kinf ~± 0.09 mk.

$$k_{eff} = \frac{\nu\Sigma_{f1} + \nu\Sigma_{f2}\dfrac{\Sigma_{S(1\to2)}}{(D_2B^2 + \Sigma_{R2})}}{(D_1B^2 + \Sigma_{R1}) - \Sigma_{S(2\to1)}\dfrac{\Sigma_{S(1\to2)}}{(D_2B^2 + \Sigma_{R2})}}$$

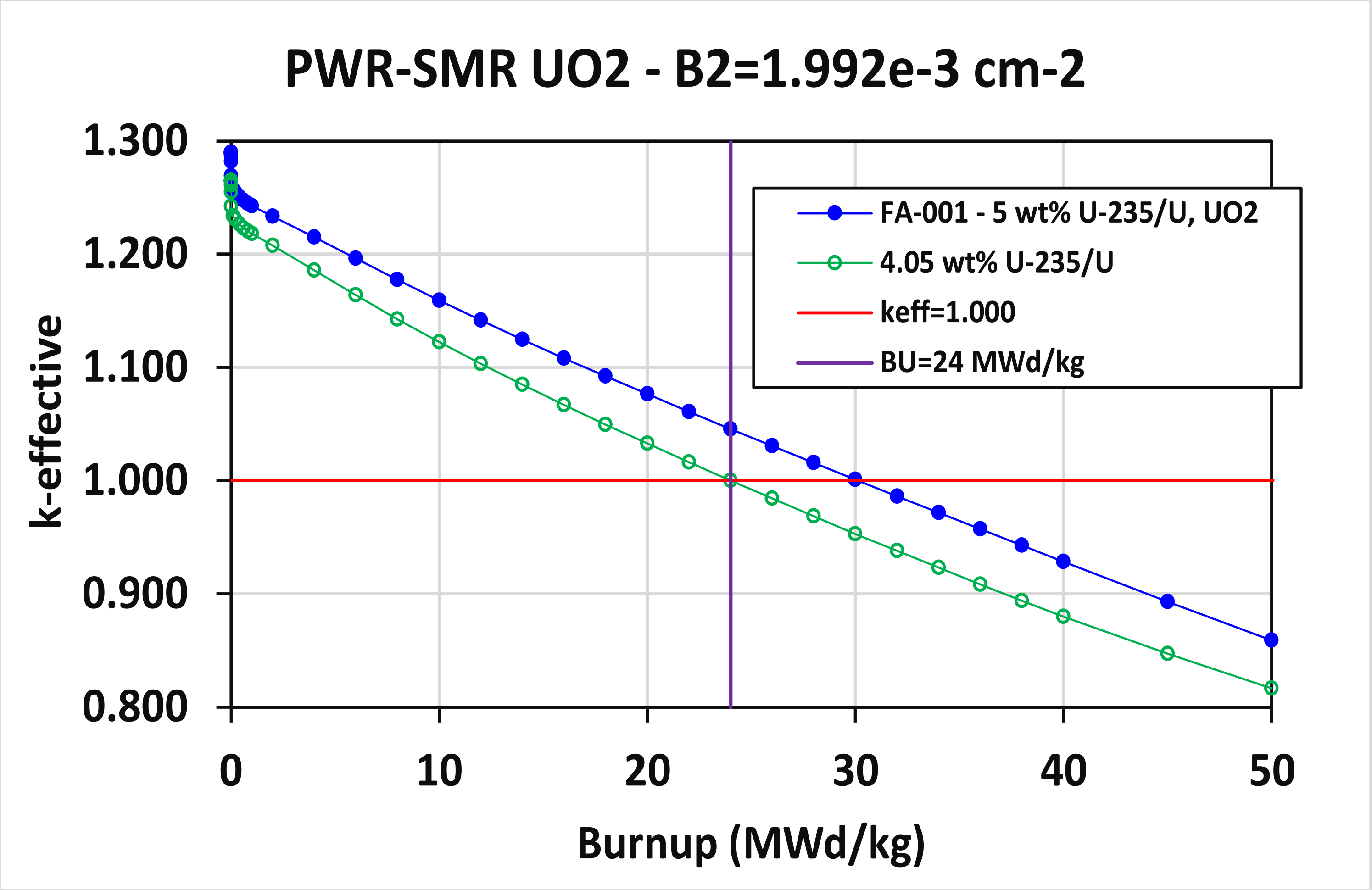
- **Prediction of neutron flux, spectrum, and power distributions, kinf.**
- **Generate homogenized two-group diffusion data.**
- **Input buckling (B^2) :**
  - Bare-core dimensions: $B^2 = (2.405/73.4)^2 + (\pi/200)^2 = 1.317E\text{-}3\ cm^{-2}$.
  - Matching NuScale data for burnup: $B^2 = 1.992e\text{-}3\ cm^{-2}$
    - 4.05 wt% U-235/U, BU~24 MWd/kg.

$$B^2 = \left(\frac{2.405}{R_a}\right)^2 + \left(\frac{\pi}{H_a}\right)^2$$

- **Neutron leakage = kinf – keff**
- **We can infer full-core behavior from lattice physics results.**

Adjust Buckling to Match Exit Burnup for NuScale iPWR Fuel with 4.05 wt% U-235/U UO2 Fuel

- B2 = 1.992e-3 cm-2
- Gives keff=1.000 at BU=24 MWd/kg for NuScale Fuel.
- Fissile utilization:
  - ~592 MWd/kg-fissile
  - RFU~0.561
- Use this value for all subsequent calculations of keff.



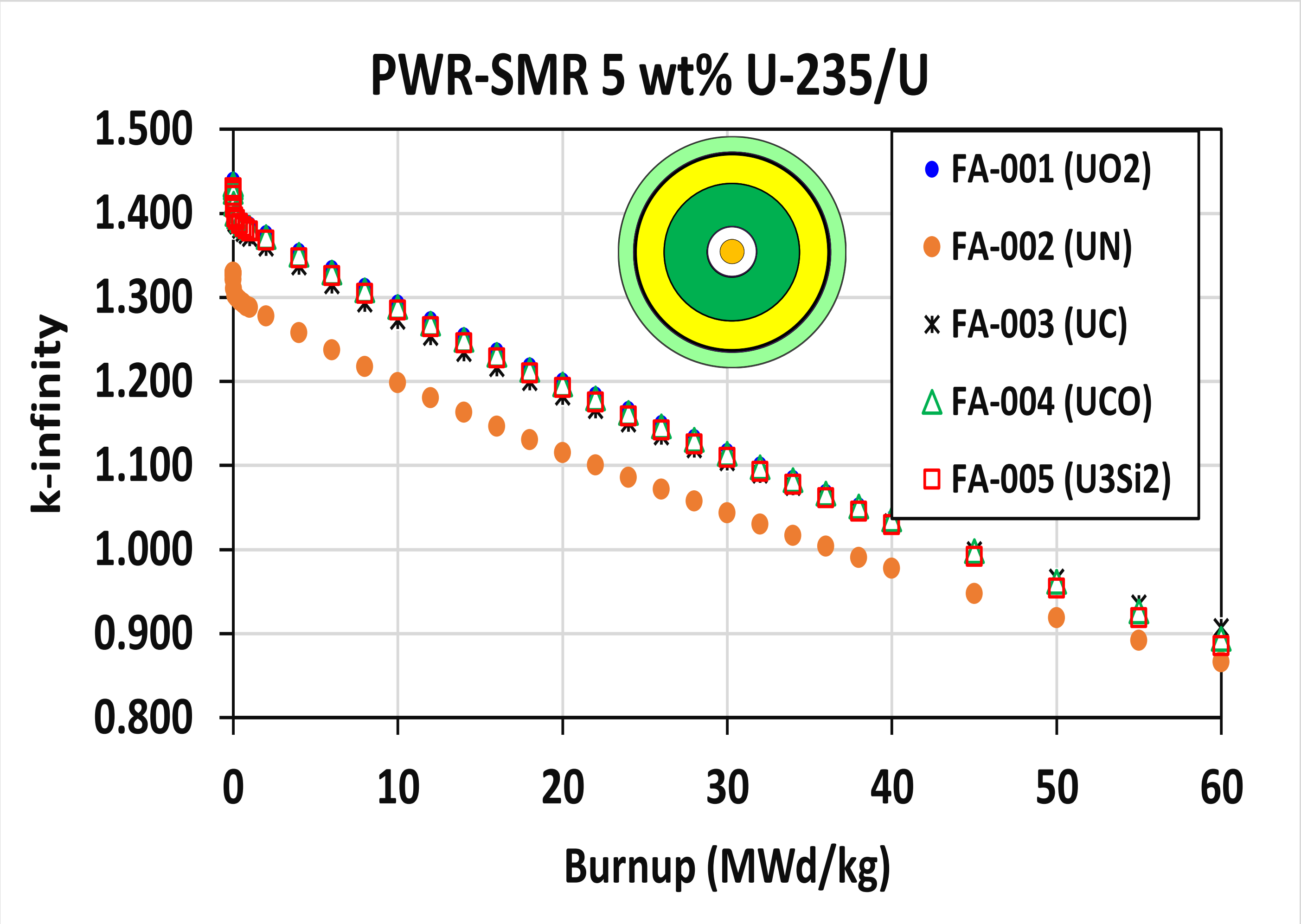
Metrics to Evaluate

- kinf vs. BU, keff vs. BU, keff vs. Time, Leakage (kinf-keff) vs. BU, or Time
- 1-Batch Exit Burnup, BU(n=1), BU when keff =1.000
- 3-Batch Exit Burnup (use linear reactivity model)
  - BU(n) = BU(1) x 2n / (n+1). BU(3) = 1.5 x BU(1)
- Core lifetime = BU(1)/ specific power, or BU(3) / specific power
- Specific power varies from fuel to fuel because of differing initial HM Mass
  - Fuel assembly power = 4.324 MW/FA (core average)
  - Specific power = 4.324 MW / FA HM mass.
- Fissile Utilization (FU) = BU / initial fissile mass fraction
  - A metric that is analogous to uranium utilization
- Relative Fissile Utilization = FU / 1,054 MWd/kg-fiss,
  - 1,054 MWd/kg = 7.5 MWd/kg / 0.00711 (for large CANDU with NU)
  - RFU=1.0 (as good as CANDU)
- Composition of Fuel, variation with burnup / time

Sample Results – 5 wt% U-235/U

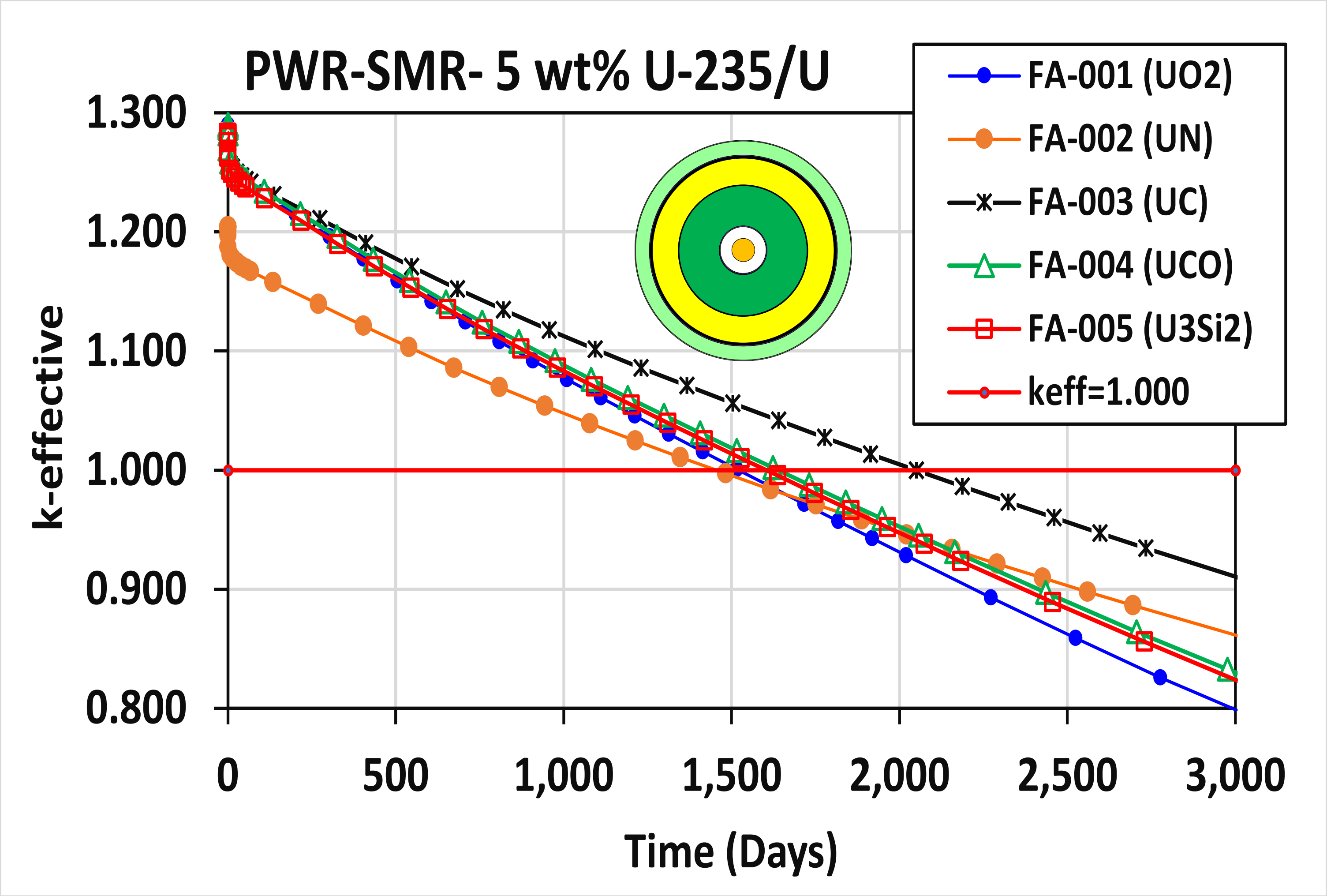
- **kinf vs. BU**

- **FA-001 – Oxide, UO_2**
- **FA-002 – Nitride, UN**
  - Reduced kinf due to N-14
- **FA-003 – Carbide UC**
- **FA-004 – Oxycarbide**
  - $UC_{0.395}O_{1.44}$
- **FA-005 – Silicide, U_3Si_2**
- **All fuel elements use same geometry – same multi-layer, multi-clad, heterogeneous design.**



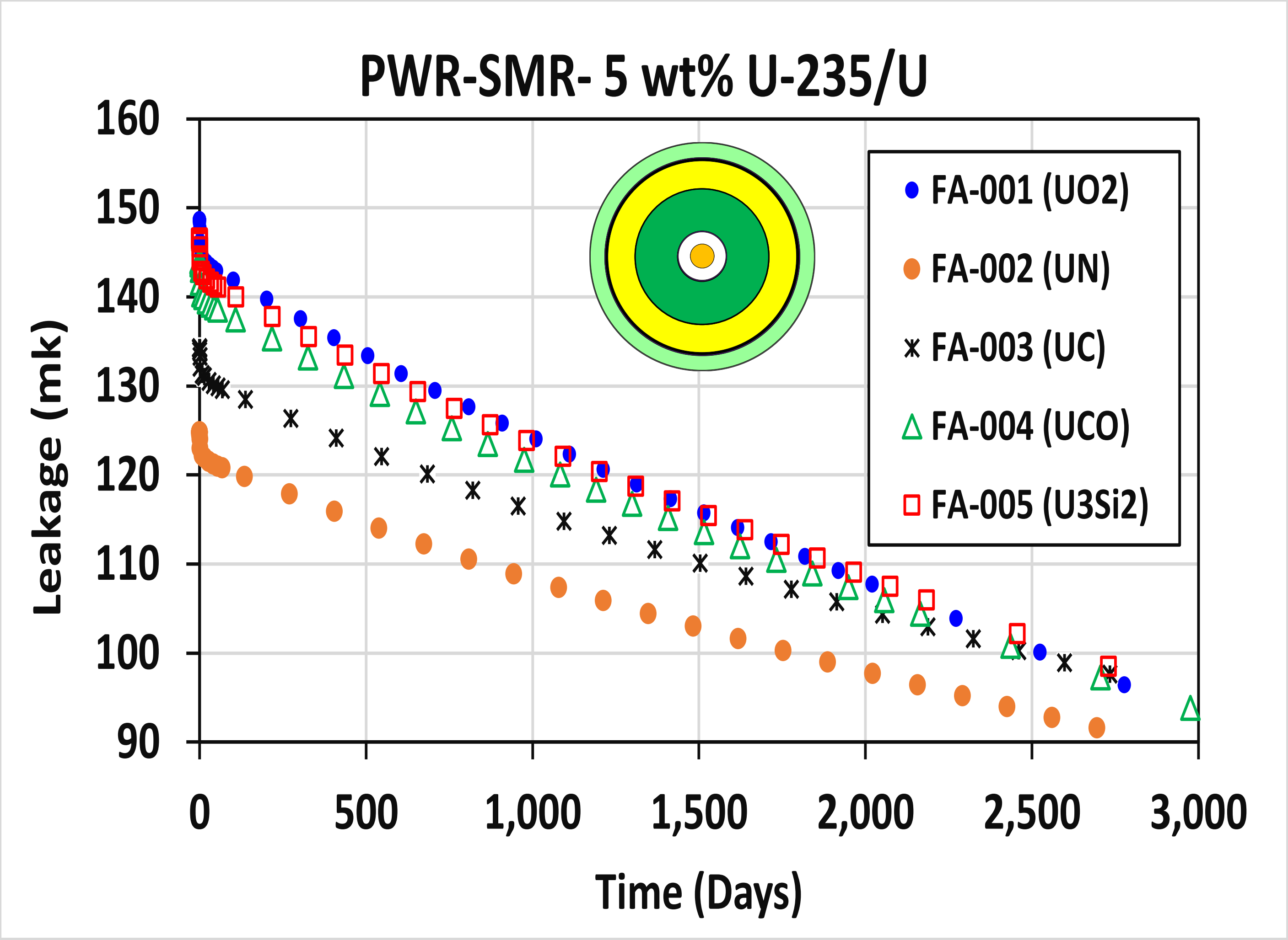
Sample Results – 5 wt% U-235/U

- **keff vs. Time**
- **1-Batch life-time:**
  - **1,400 to 2,100 days.**
- **Carbide longer life.**
- **Nitride comparable, even with neutron absorption.**
- **Oxide, Oxy-carbide, and Silicide comparable**



Sample Results – 5 wt% U-235/U

- **Leakage vs. Time**
- **150 mk to 90 mk**
  - Very large.
  - Large scale PWR ~30 mk
- **Comparable for oxides, oxy-carbides, and silicides**
- **Lower for carbides.**
- **Lowest for nitrides (due to neutron absorption)**



Sample Results – 5 wt% U-235/U – 1-Batch Burnup

- **1-Batch BU (21 to 31 MWd/kg), Lifetime (4 to 5.6 years),**
- **Fissile Utilization (430 to 605 MWd/kg-fiss), and RFU (0.31 to 0.57) – better than NuScale**
- **Nitride fuel takes a hit on BU – due to neutron absorption in N-14**
- **Other matrix materials are comparable – similar BU, lifetime, utilization.**
- **Oxycarbide (UCO) has best burnup and fissile utilization (30.3 MWd/kg, RFU~0.57)**
- **However, Carbide fuel (UC) has the longest lifetime (5.6 years).**

Matrix Type	Specific Power (kW/kgHM) at BOC	Estimate of Initial Fissile Content (wt%fiss /IHM)	1-Batch Burnup (MWd/kg)	1-Batch Lifetime (Days)	1-Batch Lifetime (Months)	1-Batch Lifetime (Years)	1-Batch Fissile Utilization (MWd/kg-fiss)	1-Batch RFU**
Oxide	19.800	0.05	30.12	1521.14	50.70	4.17	602.4	0.571
Nitride	14.838	0.05	21.58	1454.30	48.48	3.98	431.6	0.409
Carbide	14.629	0.05	29.98	2049.13	68.30	5.61	599.5	0.568
Oxy-Carbide	18.477	0.05	30.26	1637.83	54.59	4.49	605.3	0.574
Silicide	18.316	0.05	29.37	1603.47	53.45	4.39	587.4	0.557

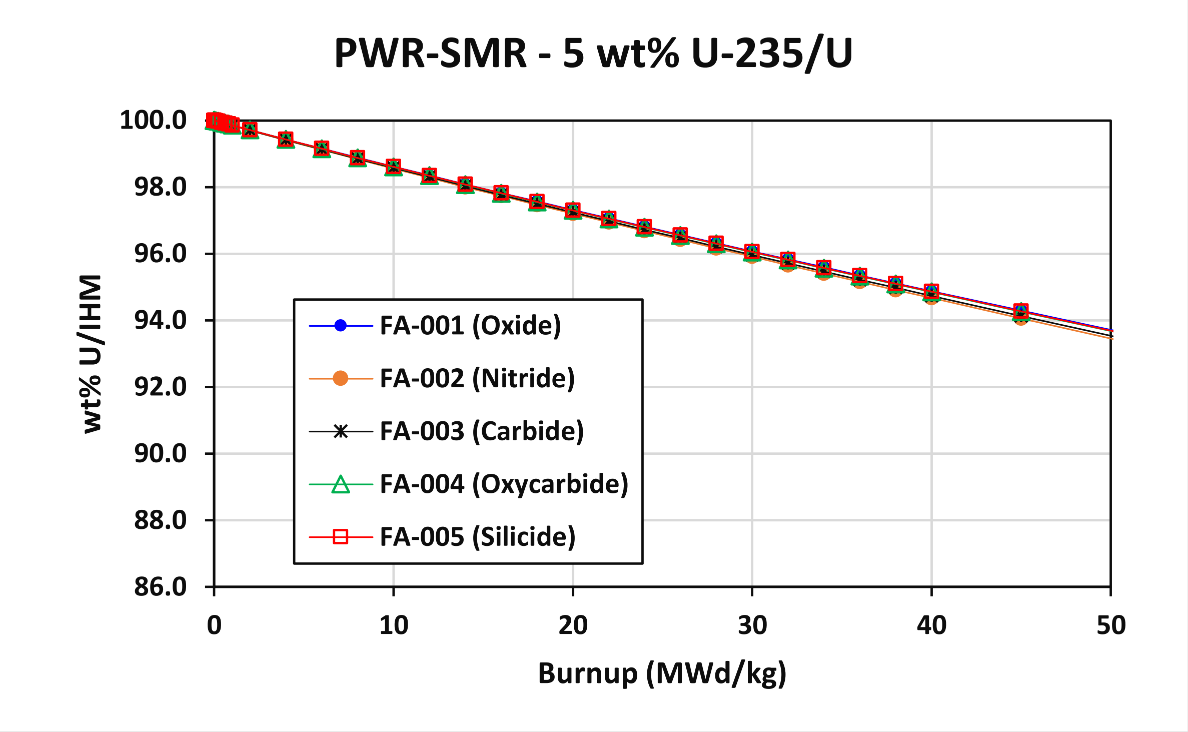
Sample Results – 5 wt% U-235/U – 3-Batch Burnup

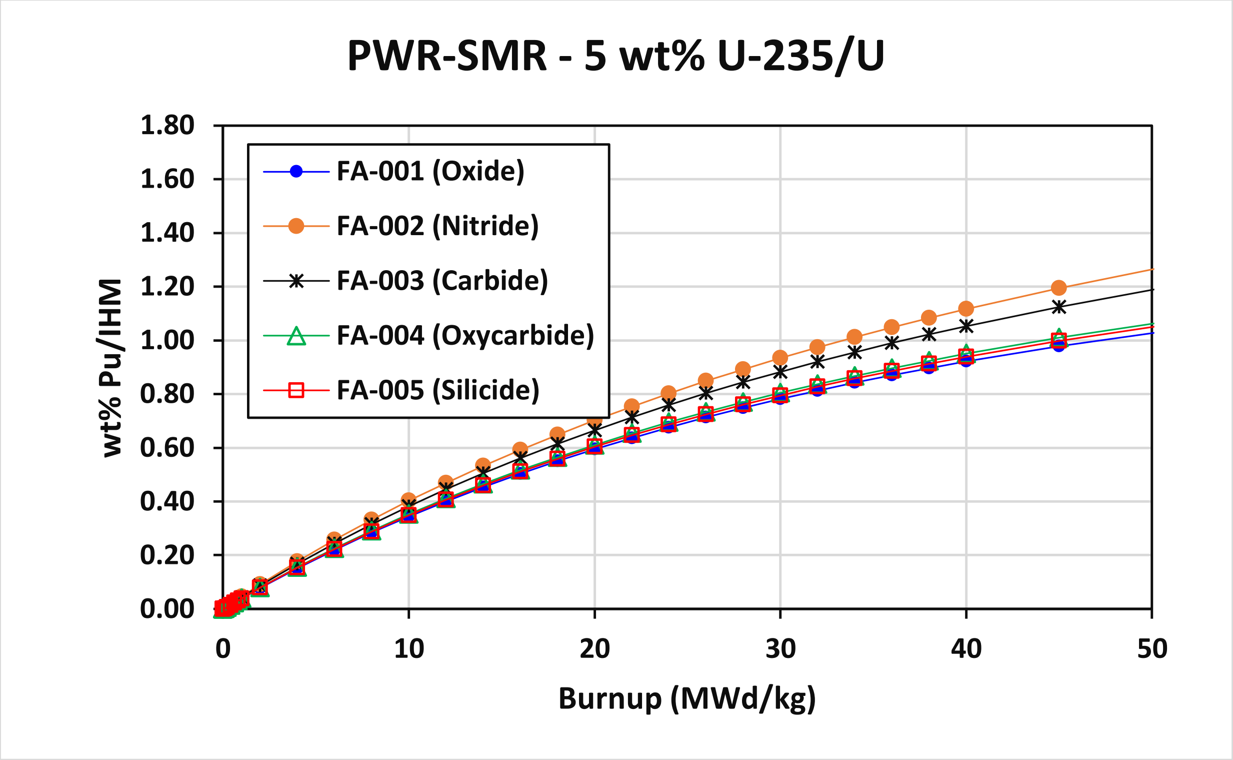
- **3-Batch BU (32 to 45 MWd/kg), Lifetime (6 to 8.4 years),**
- **Fissile Utilization (647 to 907 MWd/kg-fiss), and RFU (0.61 to 0.86)**
- **Nitride fuel takes a hit on BU – due to neutron absorption in N-14**
- **Other matrix materials are comparable – similar BU, lifetime, utilization.**
- **Oxycarbide (UCO) has best burnup and fissile utilization (45.5 MWd/kg, RFU~0.86)**
- **However, Carbide fuel (UC) has the longest lifetime (8.4 years).**

Matrix Type	Specific Power (kW/kgHM) at BOC	Estimate of Initial Fissile Content (wt%fissile / IHM)	3-Batch Burnup (MWd/kg)	3-Batch Lifetime (Days)	3-Batch Lifetime (Months)	1-of-3 Batch Cycle Length (Months)	3-Batch Lifetime (Years)	3-Batch Fissile Utilization (MWd / kg-fiss)	3-Batch RFU**
	5 wt% U-235/U in Outer and Inner Annulus								
Oxide	19.800	0.05	45.18	2281.70	76.06	25.35	6.25	903.5	0.857
Nitride	14.838	0.05	32.37	2181.45	72.71	24.24	5.98	647.4	0.614
Carbide	14.629	0.05	44.97	3073.70	102.46	34.15	8.42	899.3	0.853
Oxy-Carbide	18.477	0.05	45.39	2456.75	81.89	27.30	6.73	907.9	0.861
Silicide	18.316	0.05	44.05	2405.20	80.17	26.72	6.59	881.1	0.835

Sample Results – 5 wt% U-235/U – wt% U, Pu vs. Burnup

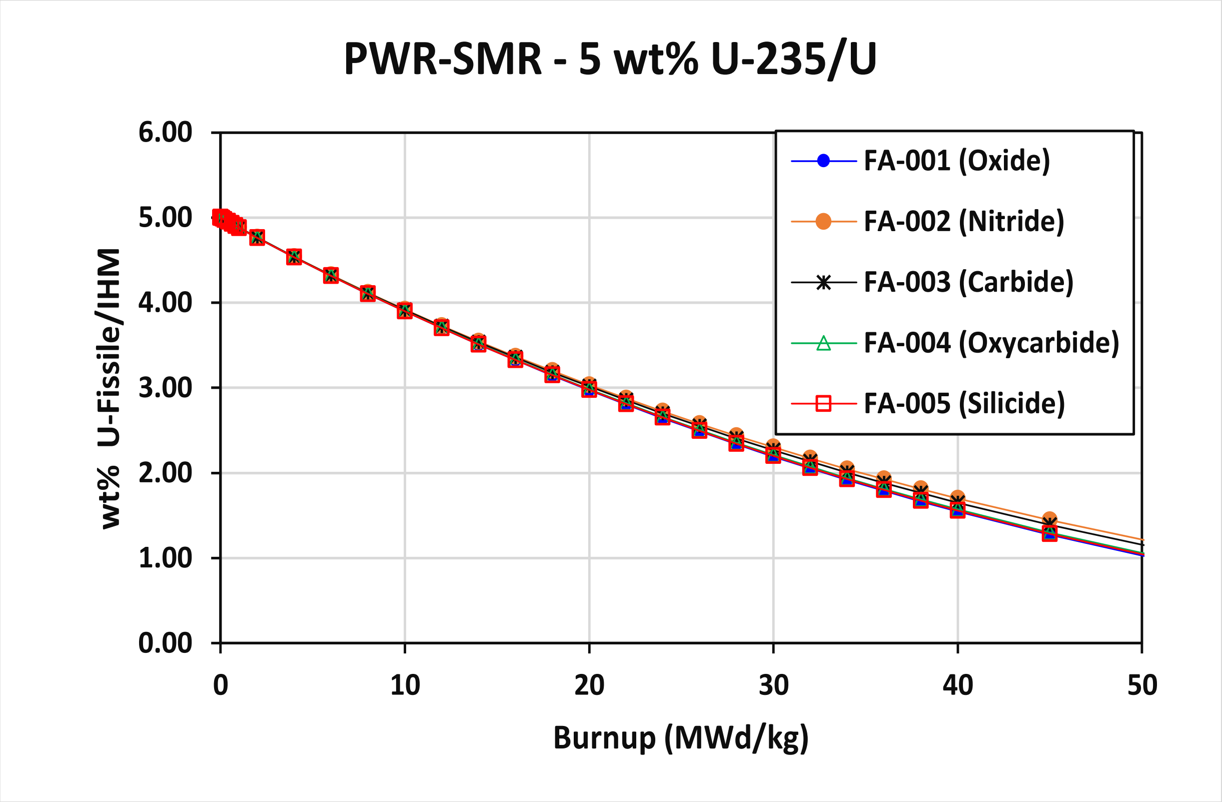
- **Total U drops from 100 wt% U/IHM down to ~94 wt% U/IHM at ~45 MWd/kg.**
- **Total Pu content increases to ~1.0 to 1.2 wt% Pu/IHM**
  - **Highest for carbide and nitride fuels.**

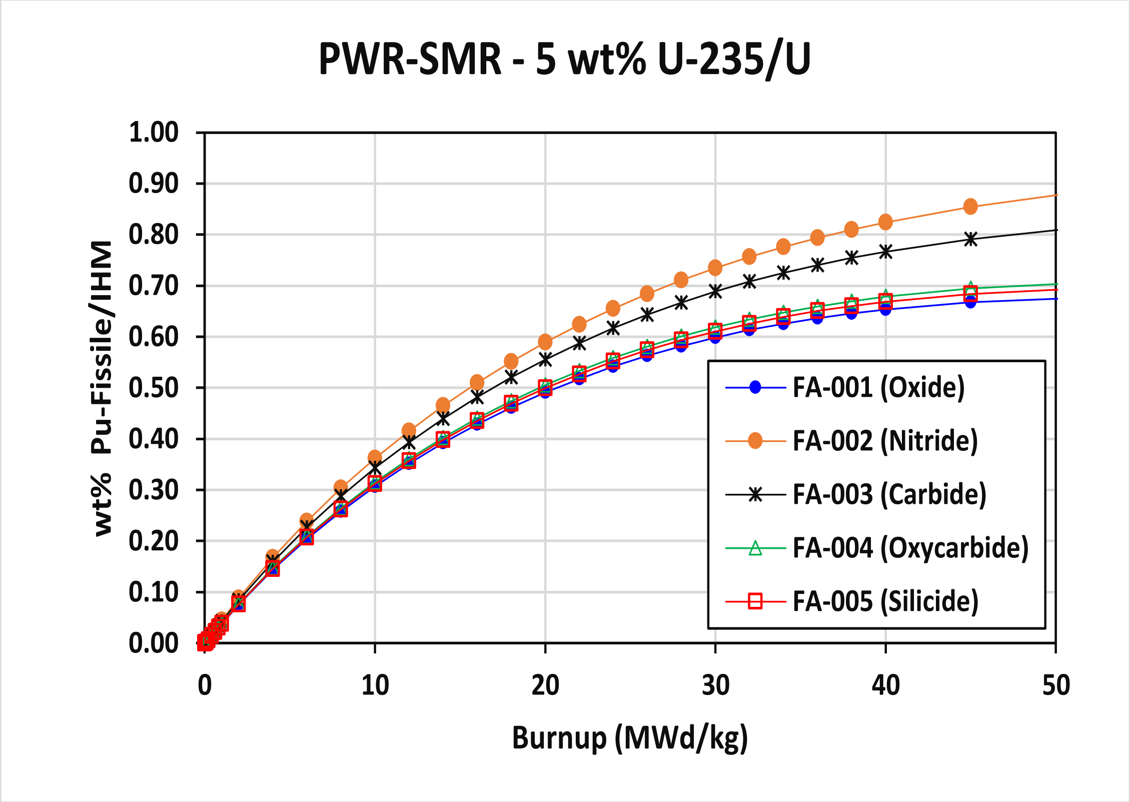




Sample Results – 5 wt% U-235/U – wt% Fissile vs. BU

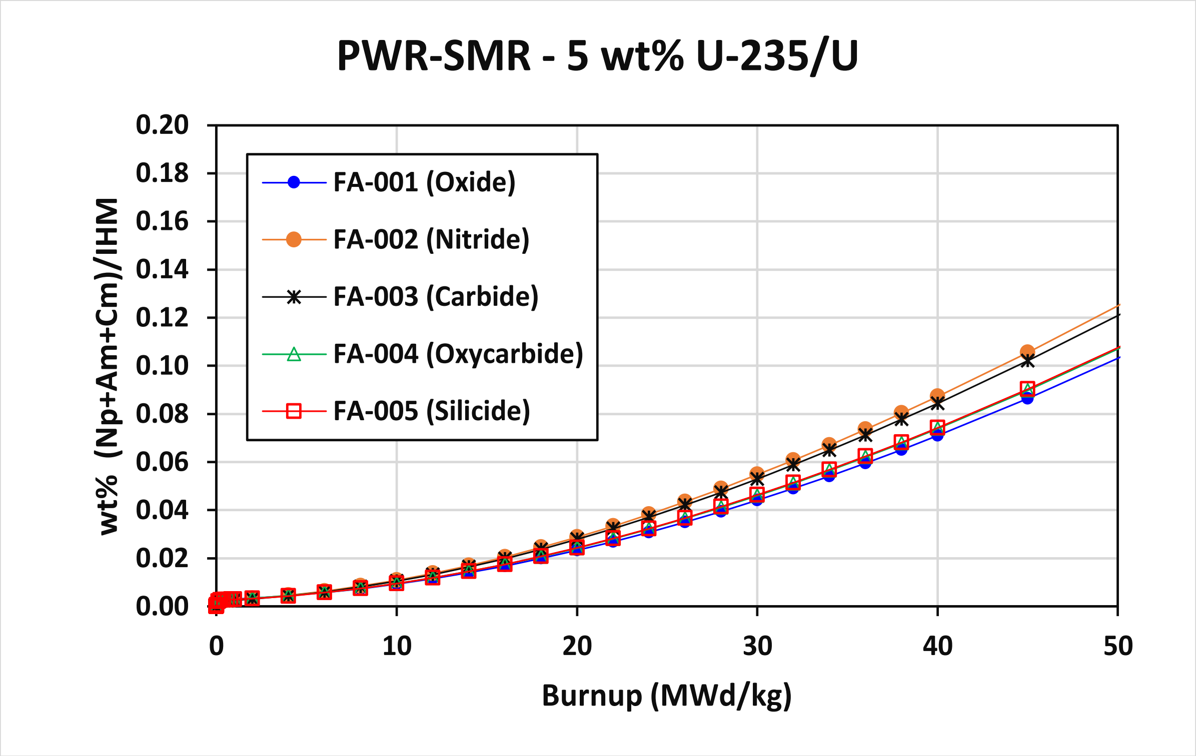
- **Total fissile drops from 5 wt% Fissile/IHM to ~2 to 2.5 wt% Fissile/IHM at ~45 MWd/kg.**
- **U-Fissile/IHM drops from 5 wt% U-235/IHM to ~1.5 wt% U-235/IHM at ~45 MWd/kg.**
- **Pu-Fissile/IHM increases to ~0.7 to 0.85 wt% Pu-Fissile/IHM at ~45 MWd/kg.**

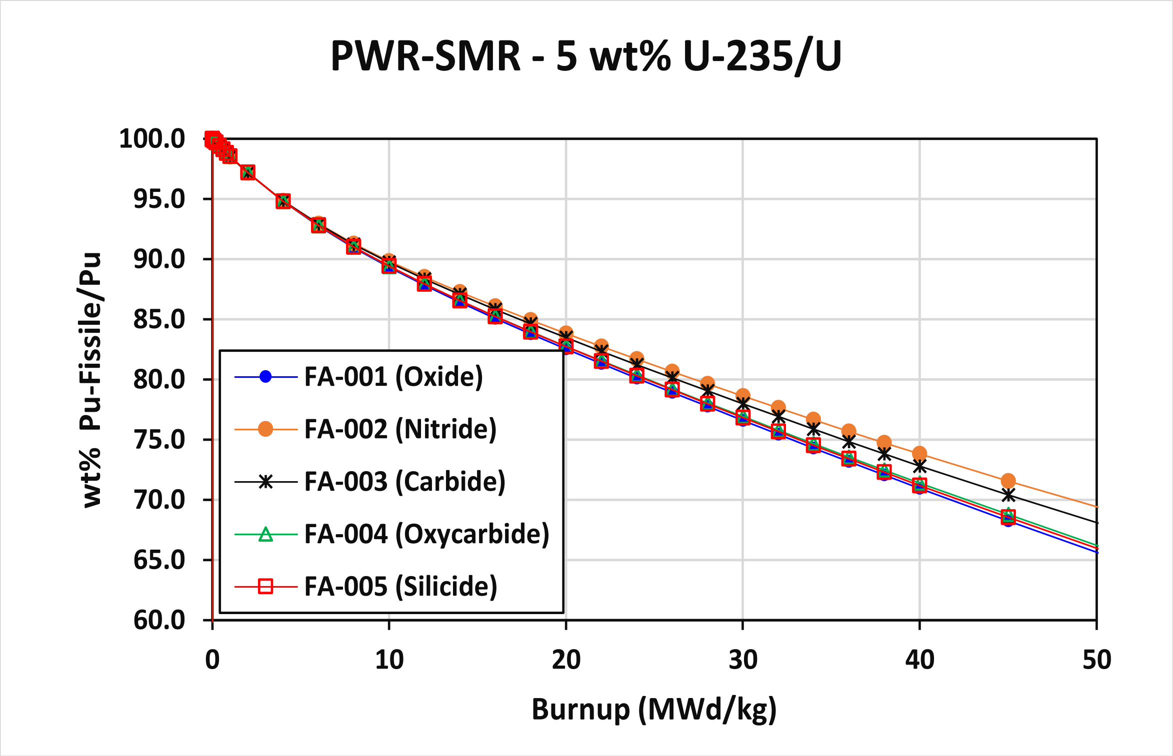




Sample Results – 5 wt% U-235/U – wt% MA vs. Burnup

- **Np+Am+Cm content increases to ~0.06 to 0.10 wt%/IHM at ~45 MWd/kg**
  - **Mostly Np-237, from neutron capture on U-235, U-236**
- **Pu-Fissile/Pu drops to ~68 wt% to 72 wt% at ~45 MWd/kg.**





Sample Results – 5 wt% U-235/U – Content at 3-Batch Burnup

- **The content of Pu and Minor Actinides (Np, Am, and Cm) has implications for spent fuel storage, recycling, and proliferation concerns.**

Matrix Type	Fuel Mass IHM (kg)	3-Batch BU (MWd/kg)	wt% Th/IHM	wt% Pa/IHM	wt% U/IHM	wt% Np/IHM	wt% Pu/IHM	wt% Am/IHM	wt% Cm/IHM	wt% (Np+Am+Cm)/IHM
Oxide	218.387	45.18	4.32E-07	1.057E-07	94.271	6.14E-02	0.980	2.00E-02	5.47E-03	0.087
Nitride	291.420	32.37	4.44E-07	1.054E-07	95.614	**4.71E-02**	**0.981**	**1.26E-02**	**2.17E-03**	**0.062**
Carbide	295.574	44.97	5.38E-07	1.550E-07	94.135	**6.91E-02**	**1.123**	**2.60E-02**	**6.87E-03**	**0.102**
Oxy-Carbide	234.015	45.39	4.56E-07	1.161E-07	94.210	6.35E-02	1.014	2.16E-02	5.94E-03	0.091
Silicide	236.081	44.05	4.48E-07	1.139E-07	94.379	6.17E-02	0.987	2.01E-02	5.40E-03	0.087

Sample Results – 5 wt% U-235/U – Fissile Content at 3-Batch Burnup

- **Fissile content in fuel at exit burnup has implications for recycling.**
  - **High enough to be used in PT-HWRs.**
- **Pu-240/Pu content has implications for weapons proliferation – Pu is not suitable.**

Matrix Type	Fuel Mass IHM (kg)	3-Batch BU (MWd/kg)	wt% Fissile/IHM	wt% U-Fissile/IHM	wt% Pu-Fissile/IHM	wt% U-fissile/U	wt% Pu-fissile/Pu	FIR	wt% U-233/IHM	wt% Pu-240/Pu
Oxide	218.387	45.18	**1.931**	1.263	**0.668**	1.340	**68.162**	0.386	3.79E-07	**23.962**
Nitride	291.420	32.37	**2.907**	2.147	0.760	2.245	**77.456**	0.581	4.92E-07	18.678
Carbide	295.574	44.97	2.182	1.390	**0.791**	1.477	70.430	0.436	5.13E-07	22.248
Oxy-Carbide	234.015	45.39	1.976	1.281	0.695	1.359	68.567	0.395	4.05E-07	23.588
Silicide	236.081	44.05	2.018	1.337	0.681	1.416	69.053	0.404	3.93E-07	23.452

Conclusions (1/2)

- **A multi-clad, multi-region heterogeneous fuel element design concept for use in a 17x17 PWR-SMR fuel assembly based on the NuScale iPWR design was tested**
  - **Anticipate enhanced resilience, similar to ATF-type fuels, and TRISO-type fuels.**
    - **Fuel performance assessment studies underway at CNL.**
  - **Current and past fabrication methods can be leveraged.**
    - **Implementation of duplex fuel, duplex clad manufacturing, coating methods for fuel pellets and clad (for CANDU, PWR, BWR, AGR, Gas-cooled Fast Reactors (GFR), etc.) – work ongoing at other laboratories (Idaho National Laboratory) and private sector companies (General Atomics, Westinghouse, NFDC (Japan), and others), previous US-DOE NERI Program.**
- **Different fuel types, including U, (U,Th) and (Pu,Th) fuels were evaluated.**
- **Different fuel matrix materials, including oxides, nitrides, carbides, oxy-carbides, and silicides, were evaluated**
- **Up to 90 test cases evaluated.**
  - **Today, we focused mainly on results from 5 cases, 5 wt% U-235/U, FA-001 to FA-005.**
  - **Some discussion of other fuels (U,Th), (Pu,Th)**

Conclusions (2/2)

- **Lattice physics calculations carried out to evaluate kinf, keff, neutron leakage, burnup, core life time, and spent fuel compositions.**
- **Many viable options exist to achieve BU comparable or better than reference design of NuScale (~24 MWd/kg, ~48 months (1-batch))**
- **Neutron leakage impact is large (97 to 141 mk) vs. ~30 mk for large PWR.**
  - **SMRs have reduced burnup and fissile utilization, due to neutron leakage.**
- **≥ 5 wt% U-235/U suitable.**
- **≥ 5 wt% U-235/(U+Th) suitable (not as good as pure uranium for SMR)**
- **≥ 15 wt% Pu/(Pu+Th) suitable.**
- **Carbide fuels most attractive; higher loading density, good conductivity.**
- **Nitrides good, if using N-15 enrichment.**

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Questions?

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Publications: https://www.researchgate.net/profile/Blair-Bromley/research

Extra Slides

More Information

Supplementary Details

"Skate to where the puck is going to be, not where it has been."

- *Walter Gretzky*

Options for Future Work and Improvements (1/2)

- Test (Pu,Th, DU) fuels:
  - 10 wt% Pu, 10 wt% DU, 80 wt% Th / 15 wt% Pu, 10 wt% DU, 75 wt% Th
  - 20 wt% Pu, 10 wt% DU, 70 wt% Th / 30 wt% Pu, 10 wt% DU, 60 wt% Th
- Test UN, and (U,Th)N fuels with enriched nitrogen (~95 at% N-15/N).
- Test FeCrAl coating instead of SS310
- Test SiC instead of PyC for interface.
- Test ZrC and ZrN for interface between Zircaloy-4 and UC and UN fuels.
  - High-temperature protective coating.
- Test use of enriched isotopes for structural components
  - Zr Alloys (95 at% Zr-90/Zr)
  - Fe-Ni Alloys (95 at% Ni-64/Ni)
  - Improve neutron economy, increase fuel burnup and fuel lifetime.

Options for Future Work and Improvements (2/2)

- Full-core PWR-SMR reactor physics analyses.
  - More accurate evaluation of leakage effects and fuel burnup.
- Test (U,Th) fuels using recovered uranium (RU)
  - Use recycled U (≥95 wt% U-fissile/U) from (Pu,Th) fuels.
- Examine use of thorium in control rods to hold down excess reactivity.
  - Currently, boric acid is used.
  - Don't waste neutrons by boron absorption. Make them work for you.
- Test D_2O and BeO reflectors.
  - Help reduce neutron leakage and increase fuel burnup / lifetime.
- Examine multi-stage recycling,
  - Reprocess and recycle to make new PWR-SMR Fuel.
  - Implement with (U,Th) and (Pu,Th), and (Pu,DU) fuels. Determine support ratio.
- Examine tandem recycling – Use SUNF from PWR-SMR in other reactors
  - Large-scale PWR (AP1000) or PT-HWR (CANDU-6/EC6)

Summary for Uranium and Uranium-Thorium Fuels

- 5 wt% U-fissile/IHM → BU(1) ~ 25 to 30 MWd/kg, RFU ~ 0.4 to 0.6.
- NuScale iPWR design with 4.05 wt% U-235/U UO2 fuel (~24 MWd/kg).
- Thorium penalizes neutron economy at low burnups.
- Pure uranium is better than (U,Th), for small reactors.
- Reducing neutron leakage increases burnup, making thorium more viable.
- Carbides are most attractive. Larger uranium loading density.
- 5 wt% U-235/U with UC:
  - 1-batch lifetime of ~68 months (35% higher); UO2 ~ 51 months .
- Nitrides possible, but only if 95 at% N-15/N.

Summary Plutonium-Thorium Fuels with ~ 7.5 wt% Pu/(Pu+Th)

- 1-batch burnups in the range 3 to 18 MWd/kg (rather low).
- Fuel lifetimes in the range of 10 to 23 months.
- Relative fissile utilization in the range of 0.1 to 0.25,
- Silicides give highest burnup, but low HM loading density.
- Oxycarbides give highest lifetime. Slightly better than oxides or carbides.
- High Pu-240 content in HWR-RGPu gives lower lifetime than PWR-RGPu.
- Neutron capture in Pu-240, and others (Th-232, Pu-238, Pu-242) limits burnup and lifetime.
- Small amounts of DU makes things worse; neutron capture in U-238.
- Neutron leakage causes huge penalty.
- Must use higher Pu content to compensate.

Summary Plutonium-Thorium Fuels with ≥ 15 wt% Pu/(Pu+Th)

- 1-batch burnups range from 30 to 47 MWd/kg,
- 1-batch fuel lifetime ranges from 44 to 73 months (3.7 to 6.1 years)
- Relative fissile utilization in the range of 0.29 to 0.36
- Even better performance is achieved with 20 wt% and 30 wt% Pu/(Pu+Th).
- Use of HWR-RGPu is better than PWR-RGPu (more Am-241 from Pu-241 decay) at higher burnups.
- (Pu,Th) Silicides have highest burnup, but lower HM loading density, and so lower lifetimes.
- (Pu,Th)C carbide fuels are best for good burnup and long lifetime.
- (Pu,Th)N nitride fuels with 95 at% N-15/N → even higher lifetimes.

Implications for 1-Batch and 3-Batch Refuelling

- U and (U,Th) fuels with ~5 wt% U-235/IHM, and with ~10 wt% U-235/IHM suitable for 3-batch refuelling.
  - Increase of fissile content (4.05 wt% → 5.0 wt%) helps relative to conventional NuScale Fuel.
  - Compensates for reduced fuel volume, and presence of SS310 clad coating.
- ~7.5 wt% Pu/(Pu+Th) not suitable (too low burnup), unless used in 3-batch refuelling.
- 15 to 20 wt% Pu/(Pu+Th) well-suited for 3-batch refuelling.
- High fissile content fuel (~20 wt% fissile/IHM) can be used for 1-batch refuelling
  - HALEU, or by using 30 wt% Pu/(Pu,Th), using either PWR-RGPu or HWR-RGPu
  - UC with HALEU
  - 1-batch burnup of ~122 MWd/kg, fuel lifetime ~22.9 years.
  - (Pu,Th)N, ~30 wt% Pu/(Pu+Th) with HWR-RGPu, 95 at% N-15/N
  - 1-batch burnup of ~98 MWd/kg, and a fuel lifetime of ~14.7 years.

Impacts of Neutron Leakage in Small-sized PWR-SMR Core

- Neutron leakage has a severe impact on kinf, keff, burnup and core lifetime.
- NuScale VOYGR iPWR is a very small-sized core.
- Neutron leakage varies between 97 mk and 141 mk.
- Large-scale PWR (AP1000) has much smaller leakage (~18 mk to 30 mk).
- Leakage can reduce burnup/lifetime by 30% or more.
- Impact more significant for (U,Th) & (Pu,Th) fuels. Cannot take full advantage of U-233 bred.
- Potential improvements:
  - Use a better neutron reflector in PWR-SMR core.
  - 30-cm thickness of heavy water (D2O) in a sealed Zircaloy-4 tank.
  - 30-cm thickness of blocks of BeO clad with Zircaloy-4.
  - But, this may require a larger pressure vessel for the PWR-SMR.

Implications of Spent Fuel Compositions

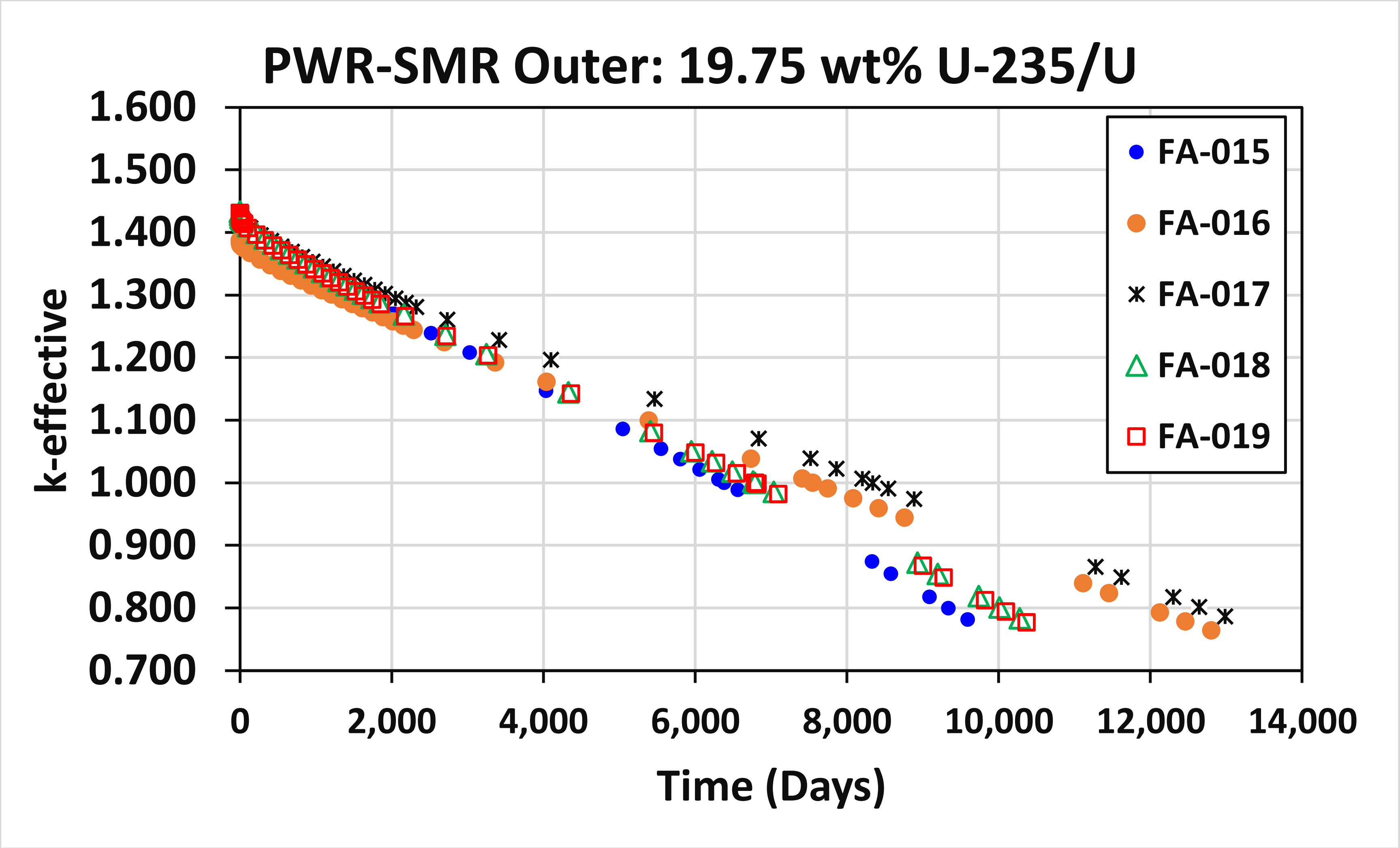
- Relatively high content of fissile fuel due to the low exit burnup.
- At 3-batch exit burnups, the fissile content ranges between ~ 2 and 7 wt% fissile/IHM
  - Particularly if using HALEU, or ≥ 15 wt% Pu/(Pu+Th)
- PWR-SMR spent fuel is *"Slightly Utilized Nuclear Fuel (SUNF)".*
- Sufficient fissile content leftover – recycle in large-scale PT-HWR (CANDU).
- May be possible to do minimal reprocessing and fuel re-fabrication.
- Minor actinides (Np, Am, and Cm) ranges between 0.04 and 0.5 wt% MA/IHM.
- U and (U,Th) fuels: mostly Np-237 (n-capture on U-235 → U-236 → U-237 → Np-237)
- (Pu,Th) fuels: mostly Am-241 (from decay of Pu-241)
  - HWR-RGPu is preferable to PWR-RGPu to avoid Minor Actinide (MA) production.

Implications for Proliferation

- Pu-240: > 18 wt% Pu-240/Pu
  - Pu in SUNF unattractive for the proliferation
  - Weapons-grade Pu requires ≥90 wt% Pu-239/Pu, ≤10 wt% Pu-240
- For (Pu,Th) fuels, fissile U > 95 wt% U-fissile/U (mainly U-233)
  - Potentially attractive, but mitigated by trace amounts of U-232.
- Small amounts of DU (~10 wt% DU/(Pu+Th+DU) can denature U-233.
  - But resonance absorption in U-238.
- Future studies: fuels with ≥ 15 wt% Pu/(Pu+Th+DU)
  - Use 10 wt% DU/(Pu+Th+DU)
  - Ensures U-233 will be denatured.
  - Impact of U-238 capture will be offset by high Pu content.

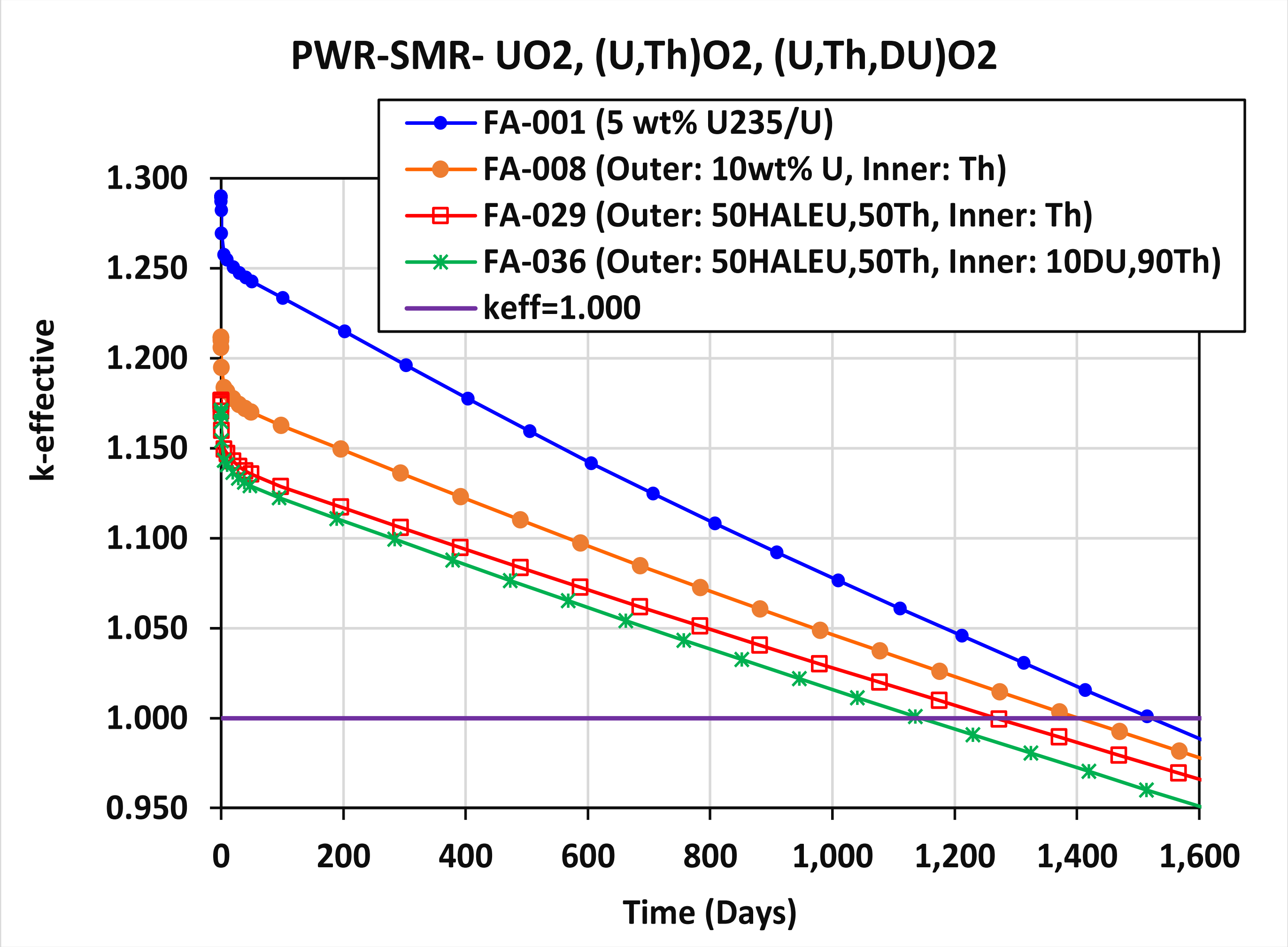
Keff vs. Time – HALEU (inner and outer annulus)

- Carbide Fuel (FA-017):
- 1-Batch Burnup ~122 MWd/kg
- 1-Batch Lifetime
  - 8340 Days
  - 278 Months
  - 22.85 Years



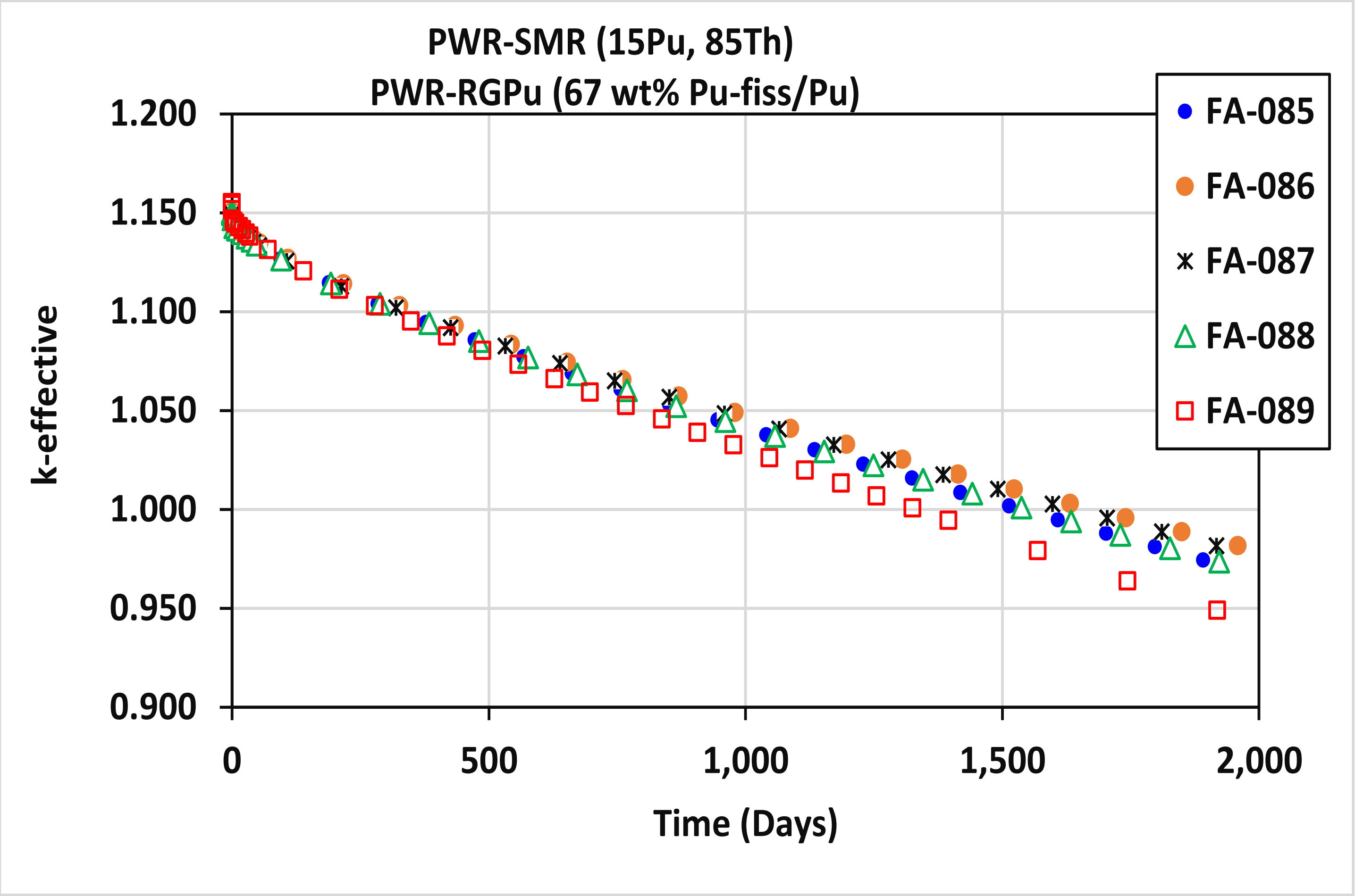
Keff vs. Time: UO2, (U,Th)O2, (U,Th,DU)O2 with ~5 wt% U-Fissile/IHM

- Using pure 5 wt% U-235/U gives highest burnup and lifetime.
- Neutron leakage must be reduced to that of a very large reactor (~20 mk) before thorium-based fuels (U,Th)O2 exceed performance of pure UO2 fuels with same initial fissile content



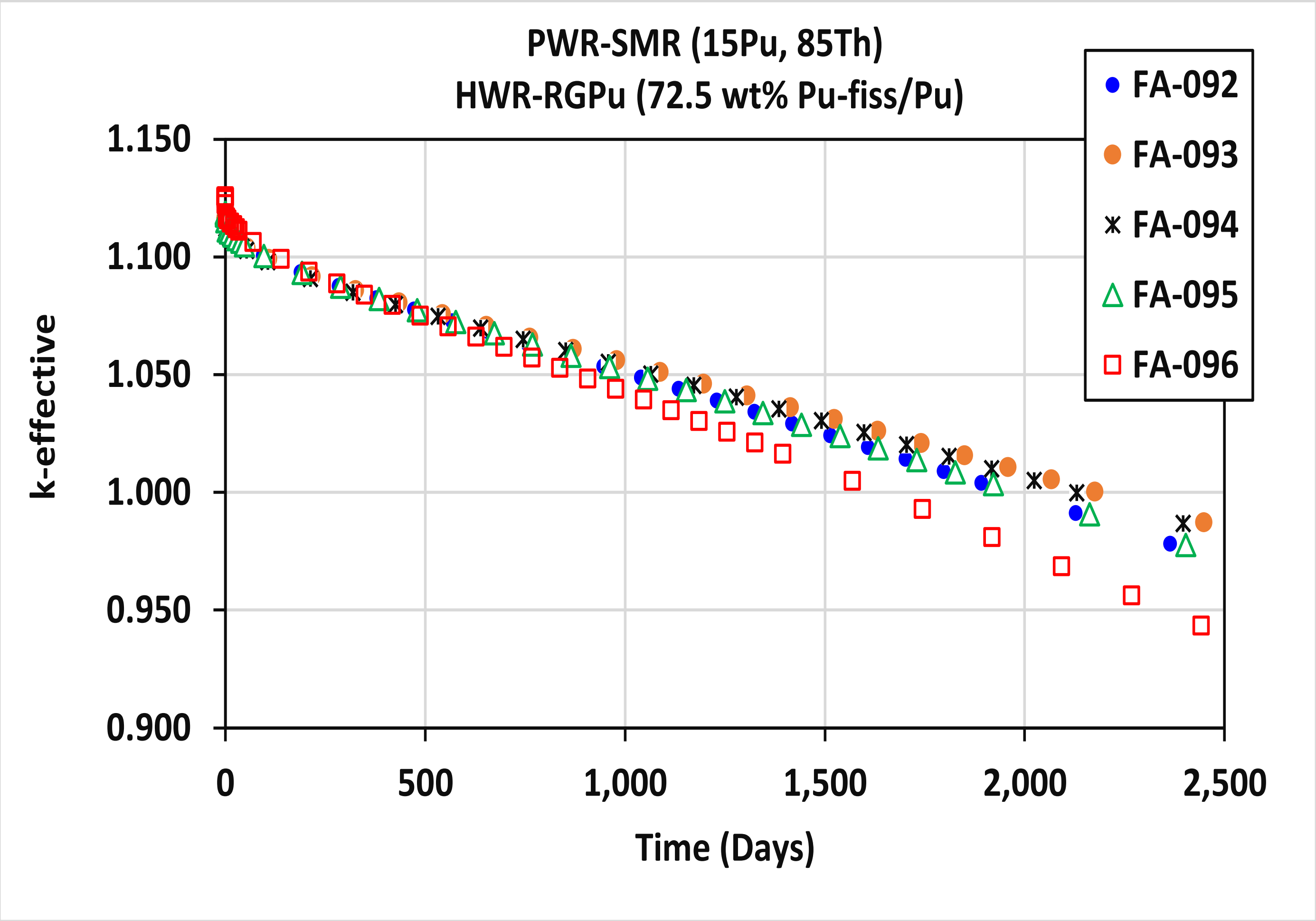
Keff vs. Time
(Pu,Th) with 15 wt% Pu/(Pu+Th), PWR-RGPu

- Enriched nitride and carbides give longest life.
- Nitride (1-Batch) (FA-086):
  - 1675 days, 4.59 years
  - 30.79 MWd/kg
- Carbide (1-Batch) (FA-087):
  - 1639 days, 4.49 years.
  - 30.77 MWd/kg



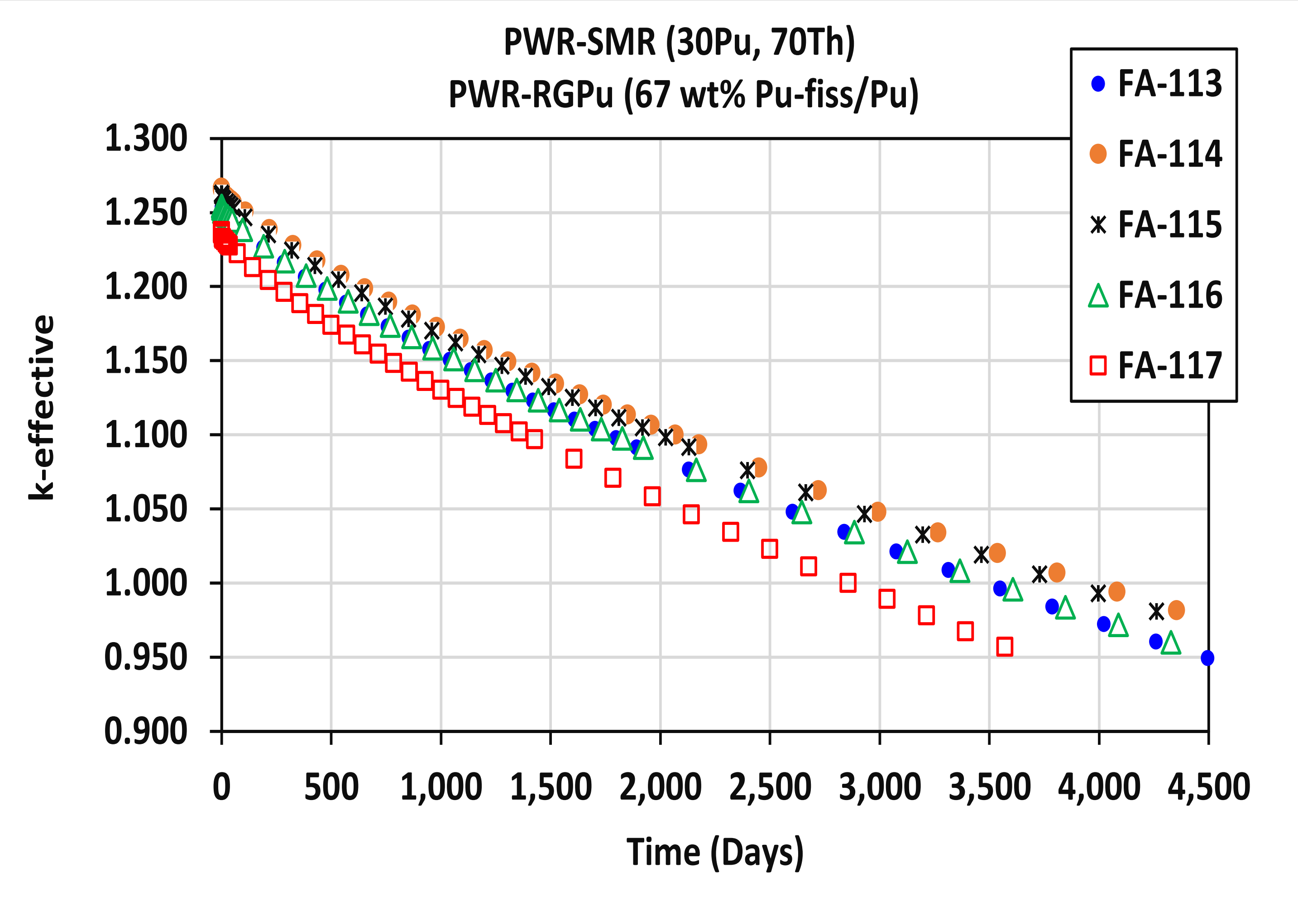
Keff vs. Time
(Pu,Th) with 15 wt% Pu/(Pu+Th), HWR-RGPu

- Enriched nitride and carbides give longest life.
- Nitride (1-Batch): (FA-093)
  - 2180 days, 5.97 years
  - 40.07 MWd/kg
- Carbide (1-Batch): (FA-094)
  - 2127 days, 5.83 years.
  - 39.94 MWd/kg



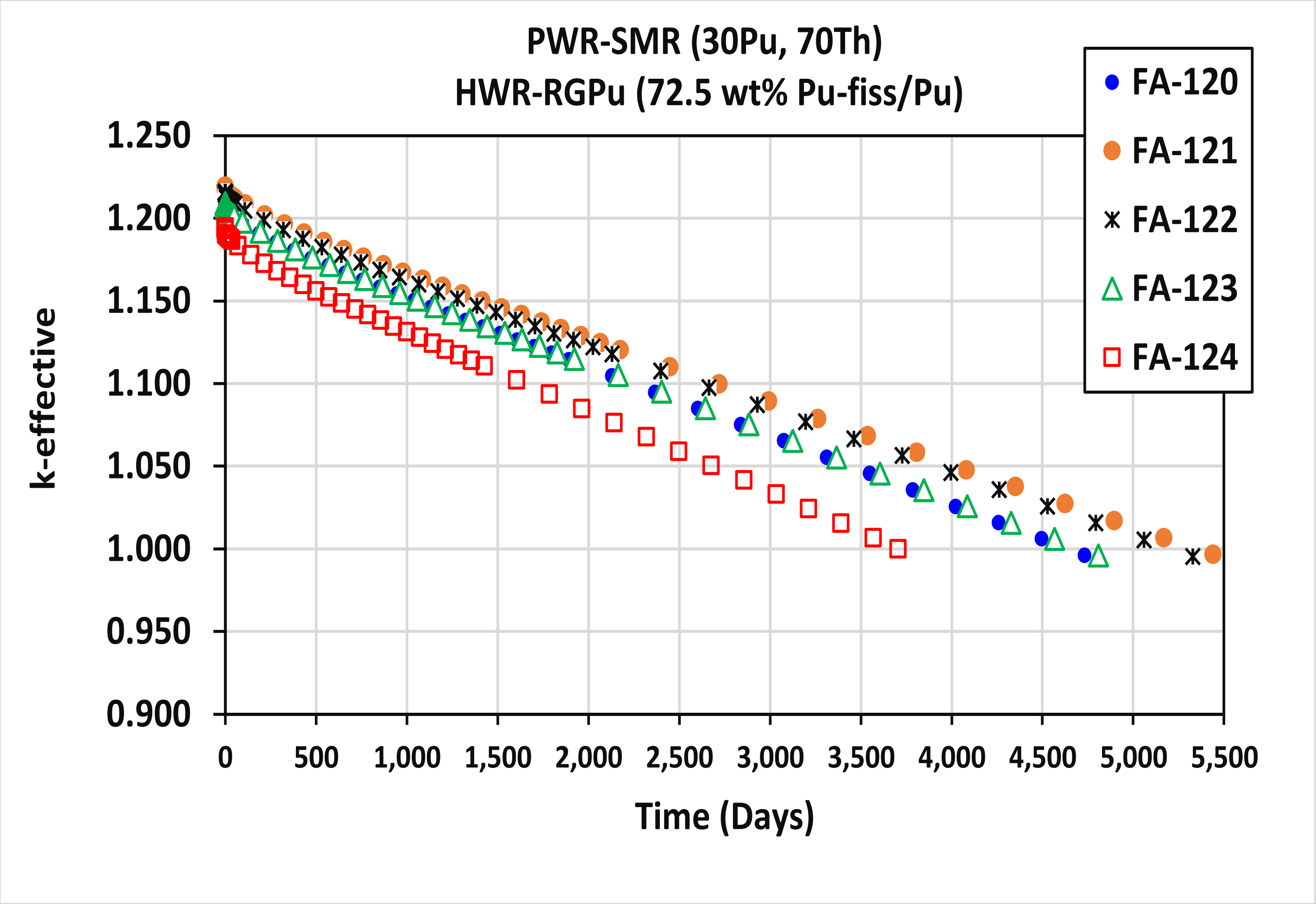
Keff vs. Time
(Pu,Th) with 30 wt% Pu/(Pu+Th), PWR-RGPu

- Enriched nitride and carbides give longest life.
- Nitride (1-Batch): (FA-114)
  - 3955 days, 10.8 years
  - 72.68 MWd/kg
- Carbide (1-Batch): (FA-115)
  - 3852 days, 10.6 years.
  - 72.3 MWd/kg



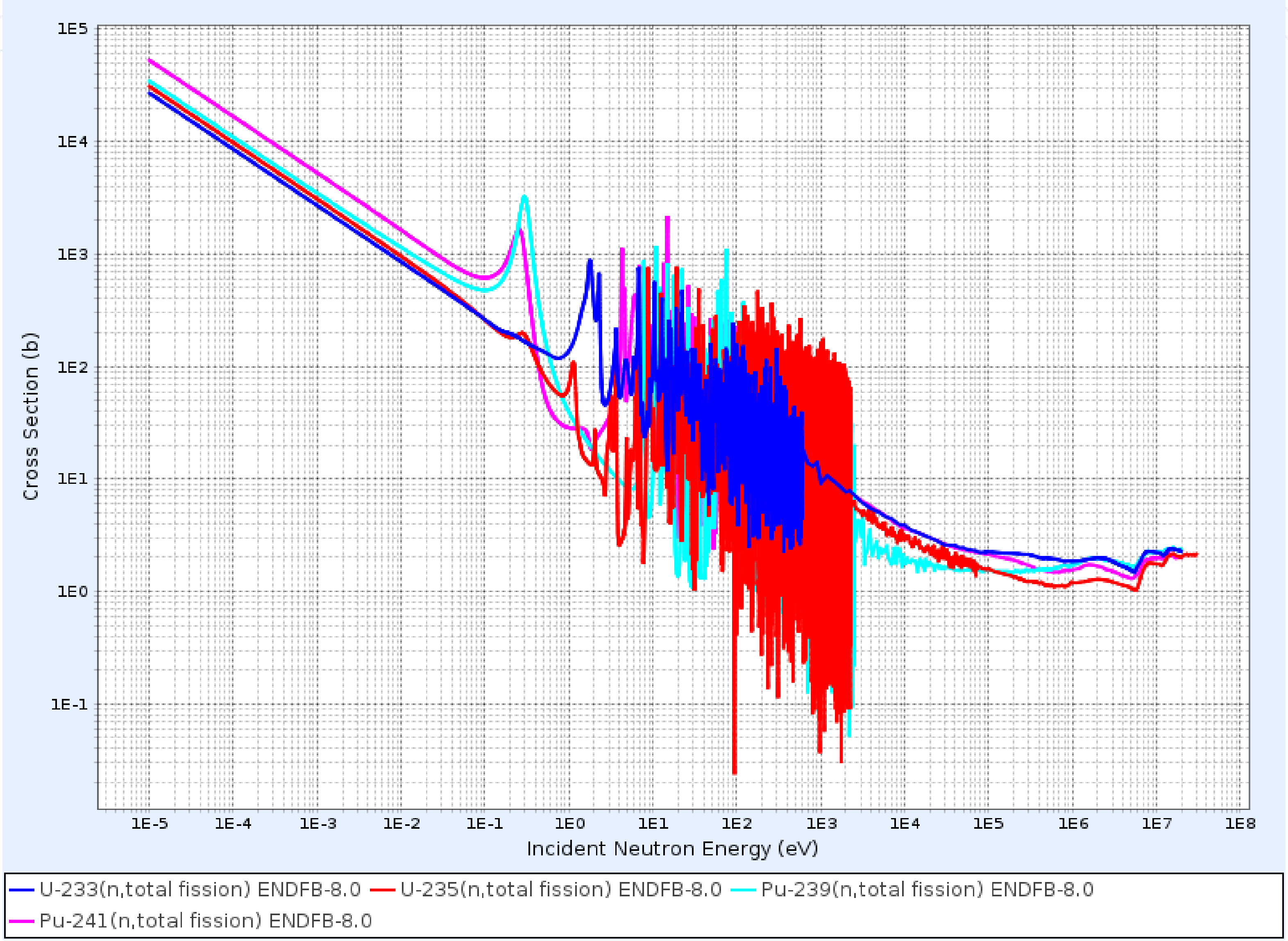
Keff vs. Time
(Pu,Th) with 30 wt% Pu/(Pu+Th), HWR-RGPu

- Enriched nitride and carbides give longest life.
- Nitride (1-Batch) (FA-121):
  - 5348 days, 14.7 years
  - 98.27 MWd/kg
- Carbide (1-Batch) (FA-122):
  - 5203 days, 14.4 years.
  - 97.67 MWd/kg



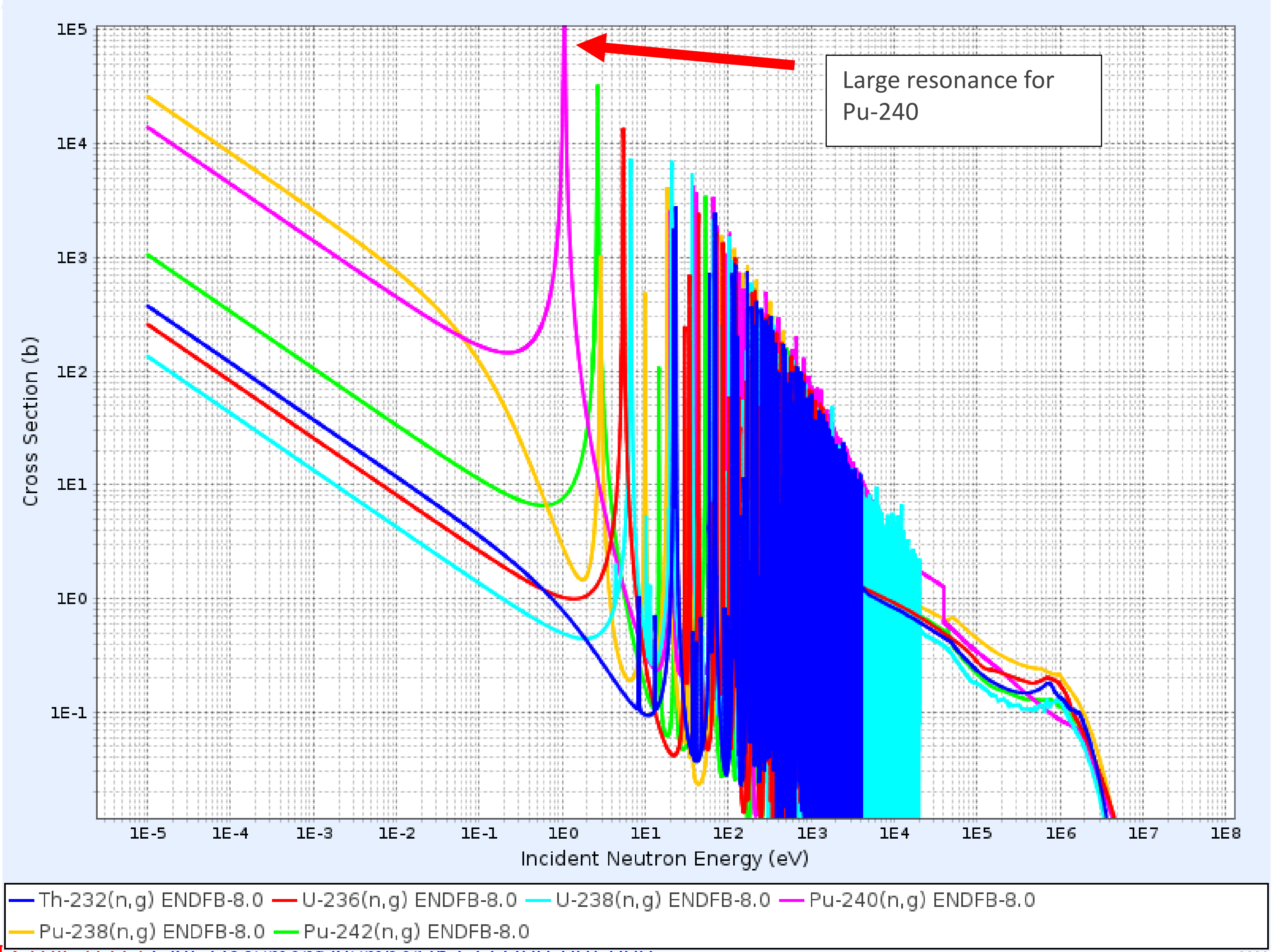
Cross Section Considerations

- High thermal fission cross sections for U-233, U-235, Pu-239 and Pu-241.



Cross Section Data for Key Absorbers

- Large thermal capture cross sections for fertile isotopes (Th-232 and U-238) and non-fissile Pu isotopes.
- Large neutron capture resonances for non-fissile Pu isotopes, particularly Pu-240



Fertile / Fissionable Isotopes – Neutron Absorption

- Non-fissile minor actinides isotopes, particularly Am-241 and Np-237 have a large thermal capture xsec
- Non-fissile isotopes Pu-238 and Pu-240 have large thermal neutron capture xsec.
- Resonance integrals for certain isotopes very large, such as Pu-240, followed by Am-234, Am-241, and Pu-242.
  - Explains lower than expected kinf and burnup in (Pu,Th) fuels
  - HWR-RGPu has a large Pu-240 content
  - PWR-RGPu has a large Pu-241 content, which leads to larger Am-241 content with burnup.

Isotope	Thermal Neutron Capture (barns)	Resonance Integral Neutron Capture (barns)
Th-232	6.532	84.35
U-236	4.703	345.6
U-238	2.414	278.1
Np-237	144.0	661
Pu-238*	**458**	**153.6**
Pu-240**	**263.6**	**8103**
Pu-242	16.82	1130
Am-241	532.0	1305
Am-243	70.46	1823
Cm-244	13.35	660