Preliminary Studies of Pressure-Tube Blanket Lattices with Thorium-Based Fuels for Hybrid Fusion-Fission Reactors

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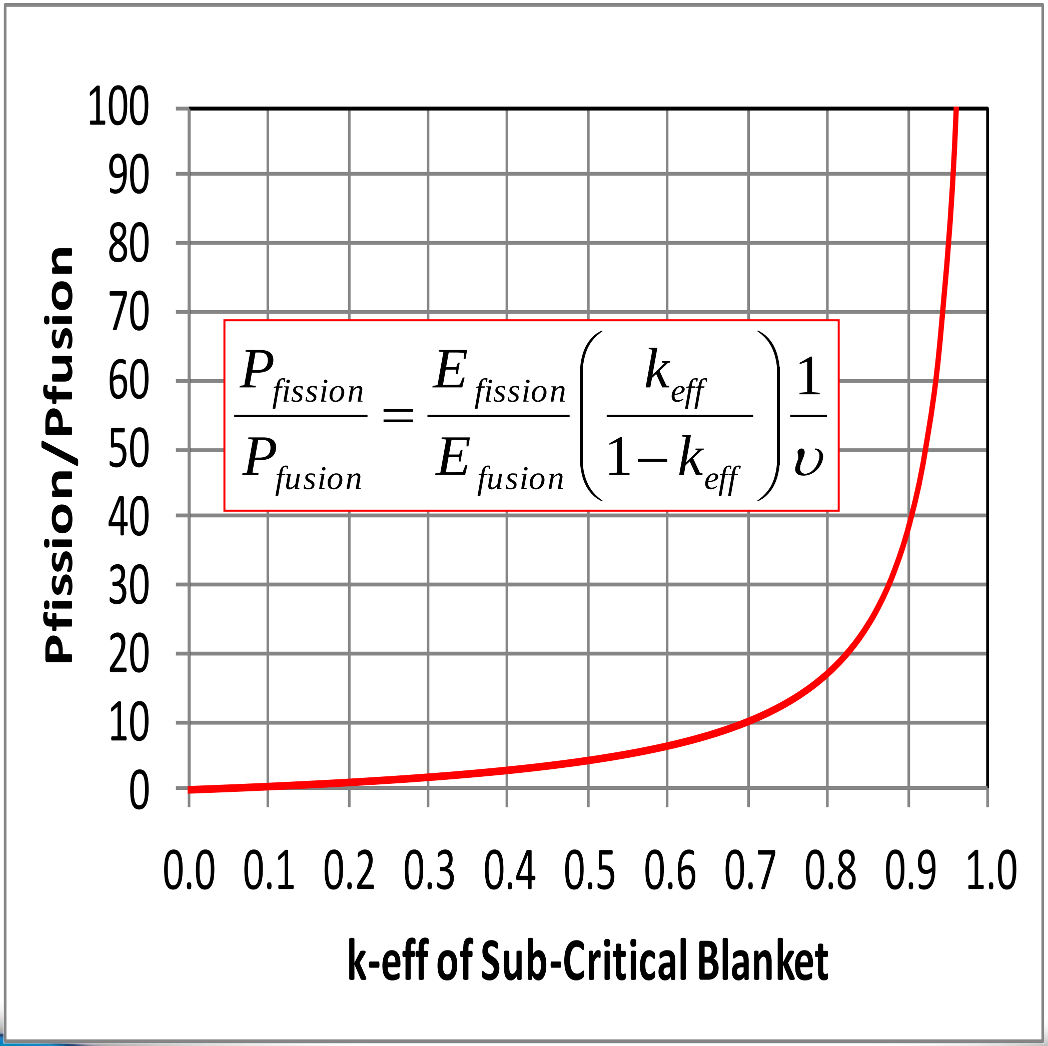
Introduction

- Hybrid Fusion Fission Reactor (HFFR) – ongoing efforts to develop.
  - Early, practical application for first-generation fusion reactors.
  - Fusion neutron source, fissile/fertile blanket.
  - Breed fuel, generate power, (consume minor actinides).
- Alternative concept for HFFR blanket – cylindrical geometry.
  - Repeating lattice of fuel channels with fuel bundles.
  - Similar to pressure tube heavy water reactors (PT-HWR).
    - Online re-fuelling capability.
  - Thorium-based fuel for U-233 breeding and power.
  - Gas-coolant for high-temperature operation (like AGR's in UK).
    - Also similarities to GCHWRs: EL-4 (France), KKN (Germany)
- Preliminary heat transfer and neutronic studies.
  - Approximate methods to evaluate performance.
  - Potential improvements; more rigorous methods later.

Power Gain in HFFR Blanket

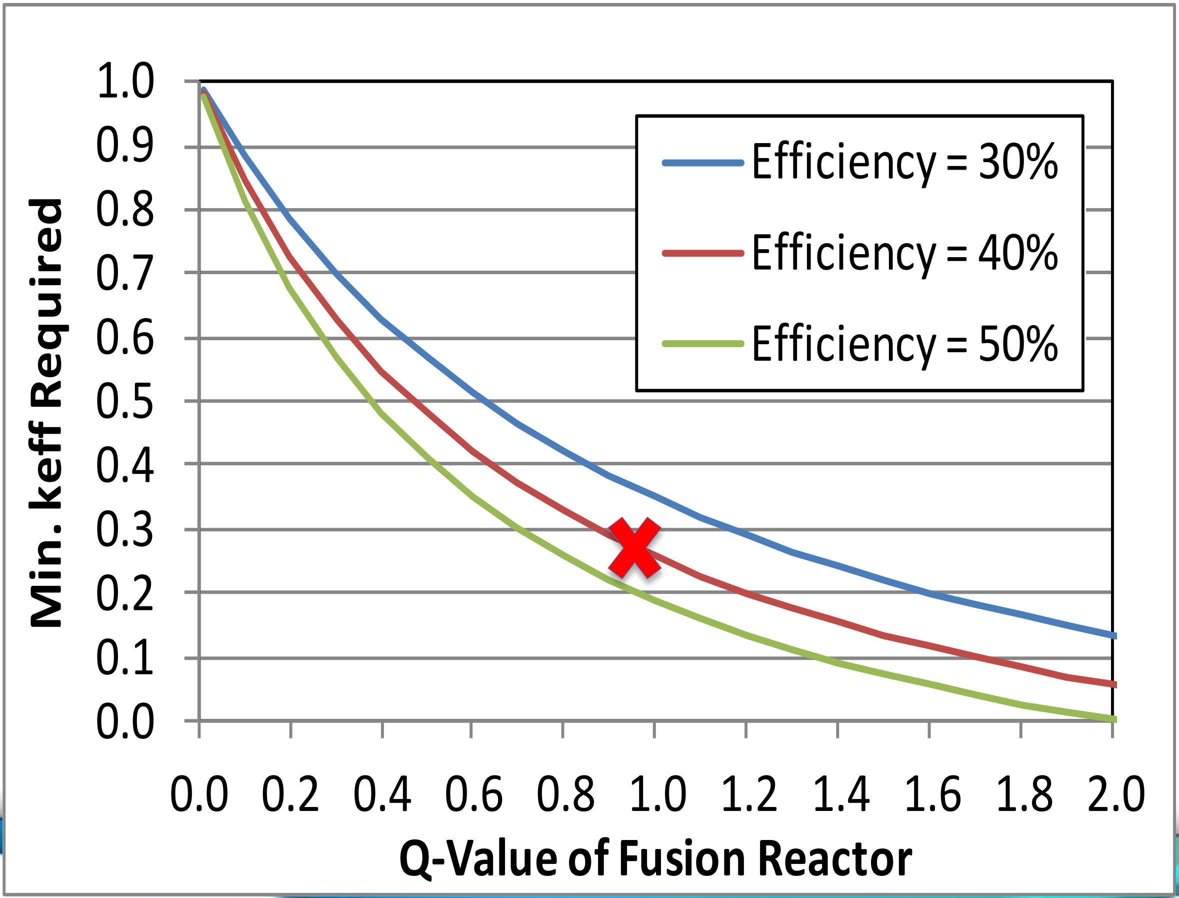
- Simple, approximate analytical relationship, based on 1-group neutron transport.
- $E_{fission}$ ~190 MeV (U-233)
- E_{fusion} ~ 17.6 MeV (D-T)
- ν ~ 2.5 neutrons/fission
- K_{eff} is multiplication factor for sub-critical blanket.
- Power gain ≥ 10, if
  - $K_{eff} \geq 0.70$
- Power gain ≥ 1.5, if
  - $K_{eff} \geq 0.26$



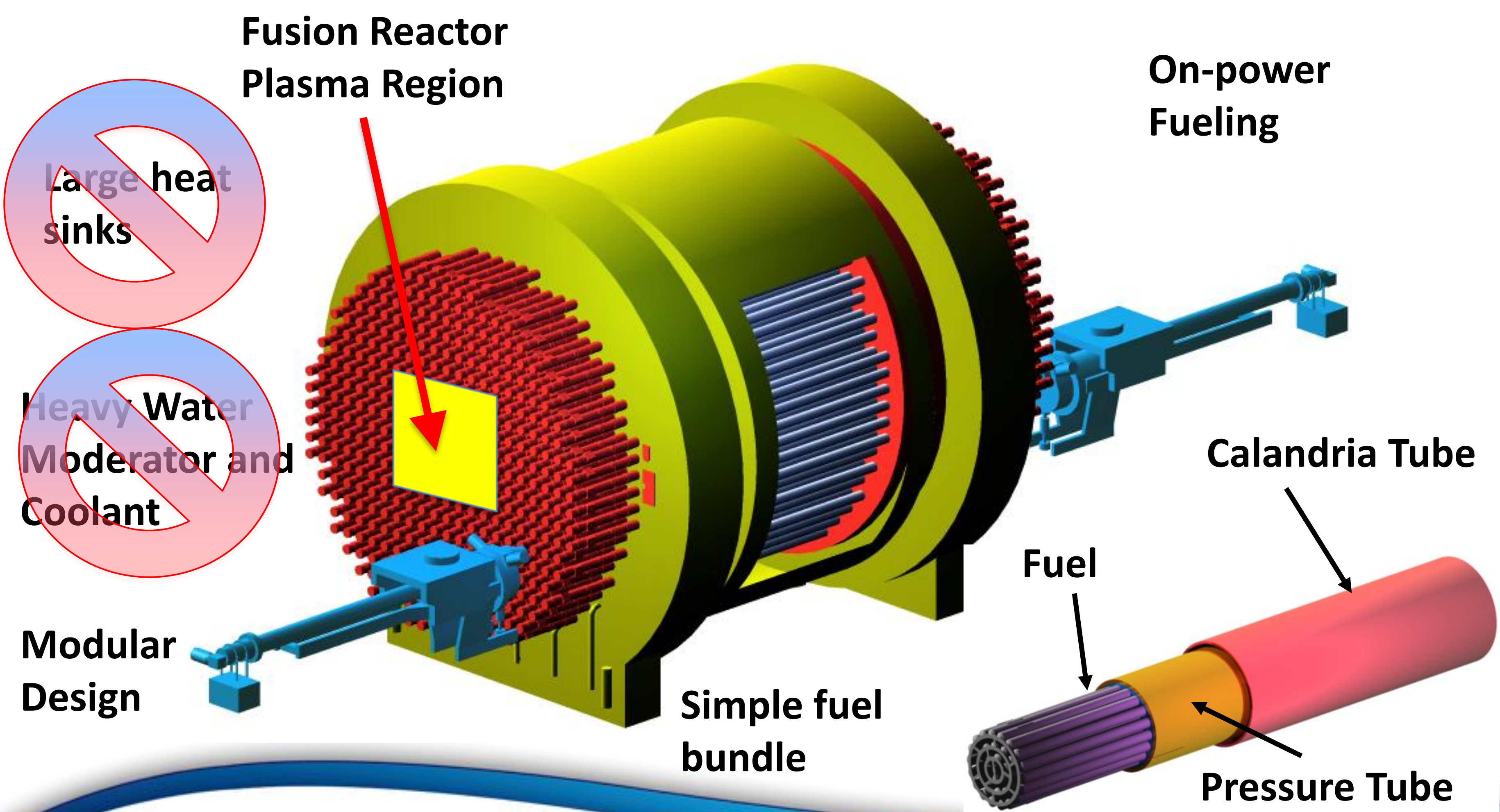
Min. Keff for Net Power in HFFR

- $Q = P_{fusion}/P_{input}$
  - Confine, heat plasma.
- HFFR can operate with a low-Q fusion reactor.
- Heat from fusion power is captured.
- For $P_{net} \geq 0.0$
  - $Q = 1.0$
  - $\eta_{th} = 40\%$
  - $P_{fiss}/P_{fus} \geq 1.5$
  - $k_{eff} \geq 0.26$
- Wide design space.
- Adjust thickness and fissile content of blanket as required.

$$P_{net} = \eta_{th} \times (P_{fission} + P_{fusion}) - P_{fusion}/Q \geq 0.0$$

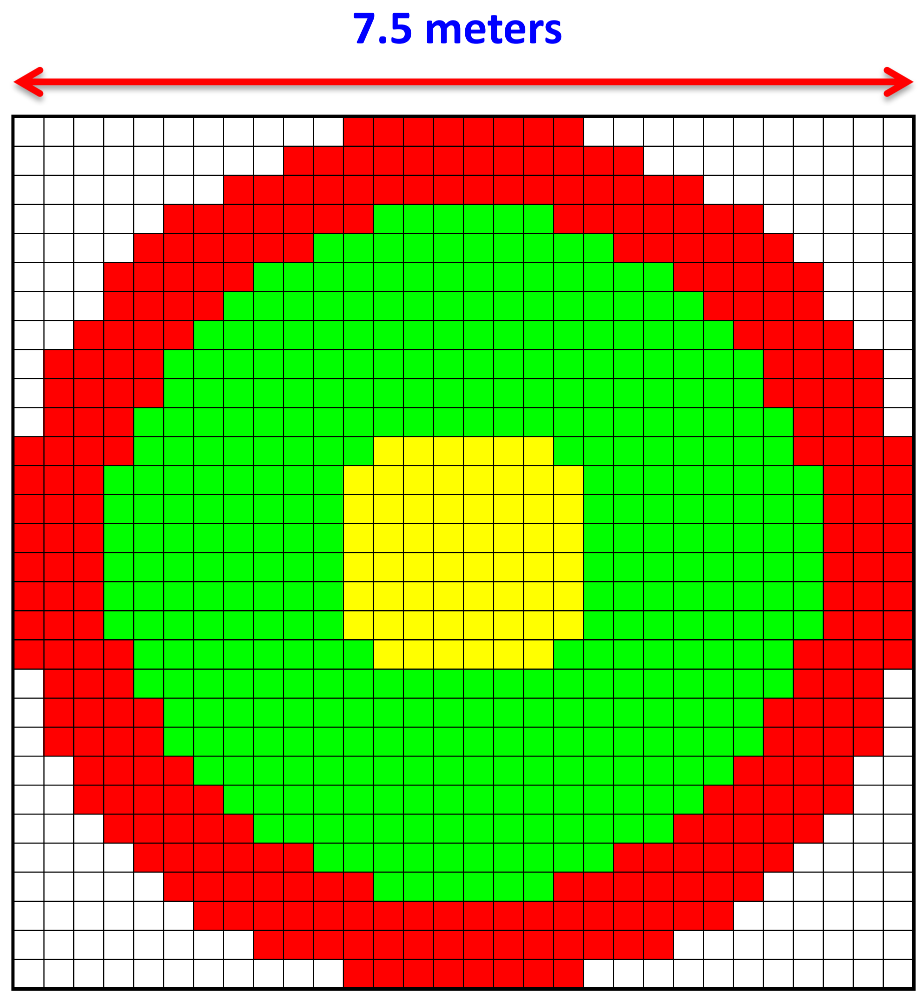


Adapt PT-HWR Technology for HFFR



HFFR Blanket Concept

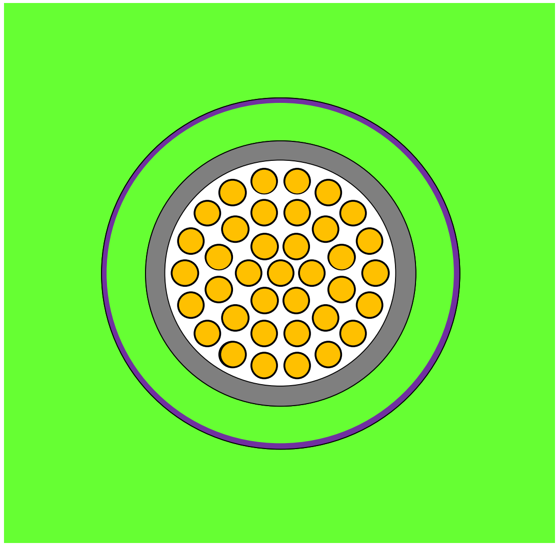
- **Engineering feasibility.**
- **Central fusion plasma region**
  - **200 cm diameter.**
  - **10 meters long.**
  - **Magnetic mirror, or other.**
  - **Field coils outside.**
- **Cylindrical blanket region**
  - **100 cm ≤ R ≤ 400 cm**
  - **Repeating 25-cm lattice.**
  - **~380 fuel channels (for R=300 cm).**
- **Sufficient space for:**
  - **On-line refuelling machines.**
  - **Feeders / headers to fuel channels.**



(yellow)	Vacuum / Plasma / Fusion Region
(green)	Blanket Region
(red)	Reflector / Shielding Region

Blanket Lattice Concepts

- Analogous to that of PT-HWR, with changes:
  - 19, or 37-element bundle, 50-cm long
  - 20 bundles per 10-meter channel.
  - (U-233,Th)O_2, 0 to 10 wt% U-233.
  - SS316 Clad (0.4 mm)
  - SS316 Pressure tube (PT) (10.5 cm / 0.9 cm)
  - SS316 Calandria Tube (CT), (15.8 cm / 0.25 cm)
  - Coolant/Fuel (CF) Area Ratio ~ 1.0 to 3.2.
  - Coolant: CO_2 at 11 MPa (250°C to 750°C).
    - Neon is potential alternative.
  - CO_2 for PT/CT gap insulation.
  - No moderator, use CO_2 at 1 atm for filler.
    - Maintain a fast spectrum.
    - Suppress fission for better breeding.

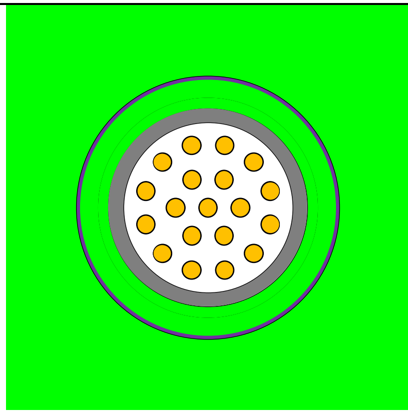


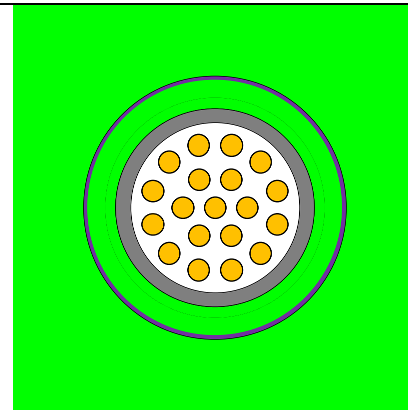
Lattice Pitch = 25 cm

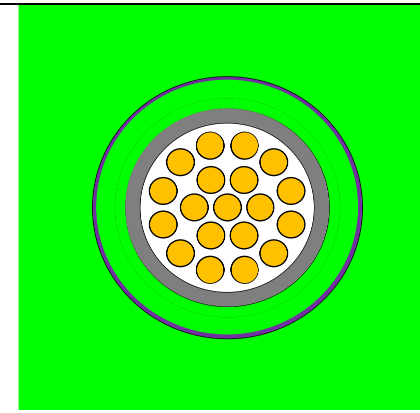
(U-233,Th) mass
~ 7 kg to 17 kg

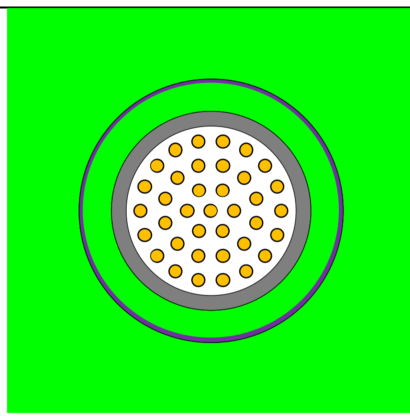
Blanket Lattice Concepts

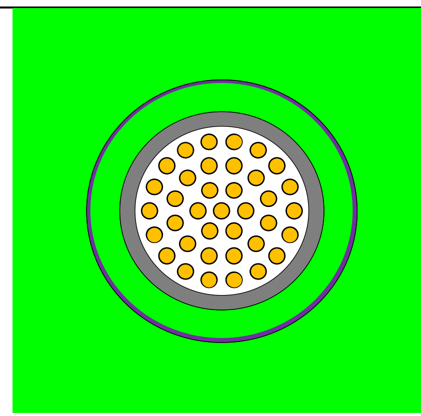
- **Variation of # pins (19, 37), and coolant/fuel area ratio (1.0, 2.0, 3.2)**

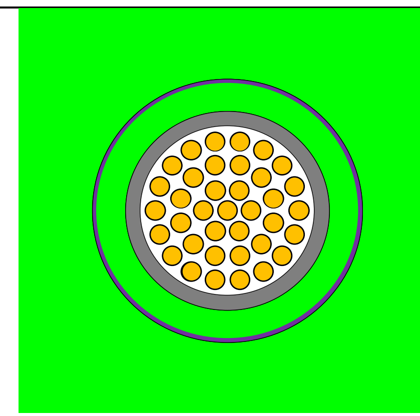

19-CF32


19-CF20


19-CF10

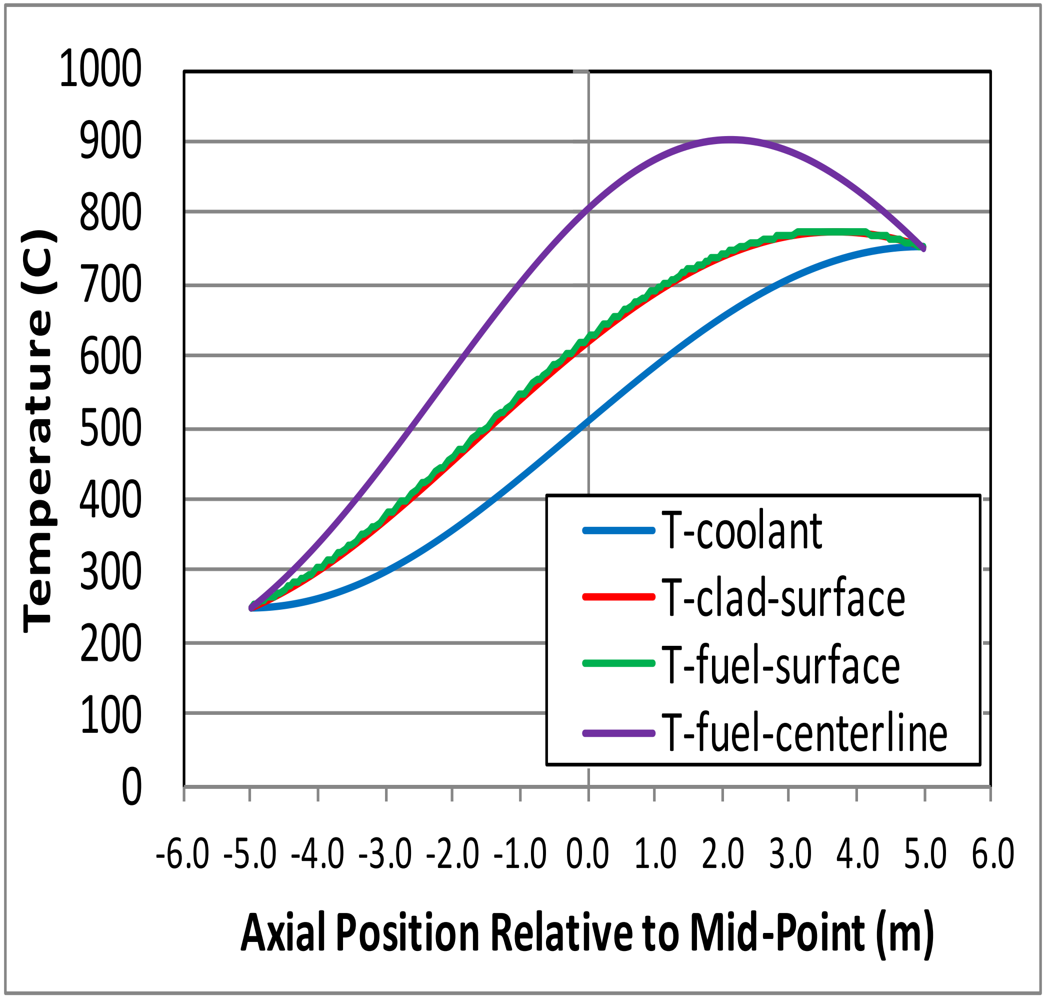

37-CF32


37-CF20


37-CF10

Temperature Profile in 3,000 kW Channel

- **Textbook analytical solutions.**
  - **Convection/conduction.**
  - **Single, smooth pin analyses.**
  - **Cosine power profile.**
- **T_{center} ≤ 903°C, T_{clad} ≤ 773°C**
  - **SS-316 melts @ 1,375°C .**
- **ΔP ≤ 130 kPa (out of 11 MPa).**
  - **Neglect effect of spacers and appendages.**
- **Pump power ≤ 9.2 kW**
  - **≤ 1% channel (3,000 kW).**
  - **High pressure (11 MPa).**
  - **Large coolant ΔT (500° C).**



Temp., Pressure Drop, Pumping Power

- **Well below expected tolerance limits. Pump power ≤ 1% Channel**

Lattice	T_{clad} Max. (°C)	$T_{fuel\text{-}CL}$ Max. (°C)	T_{pellet} (°C)	ΔP (kPa)	Pump Power (kW)
19-CF10	796	1151	733	93.8	6.7
19-CF20	863	1287	814	33.5	2.4
19-CF32	941	1417	890	19.7	1.4
37-CF10	773	903	634	129.6	9.2
37-CF20	807	970	685	45.8	3.3
37-CF32	854	1044	737	25.8	1.8

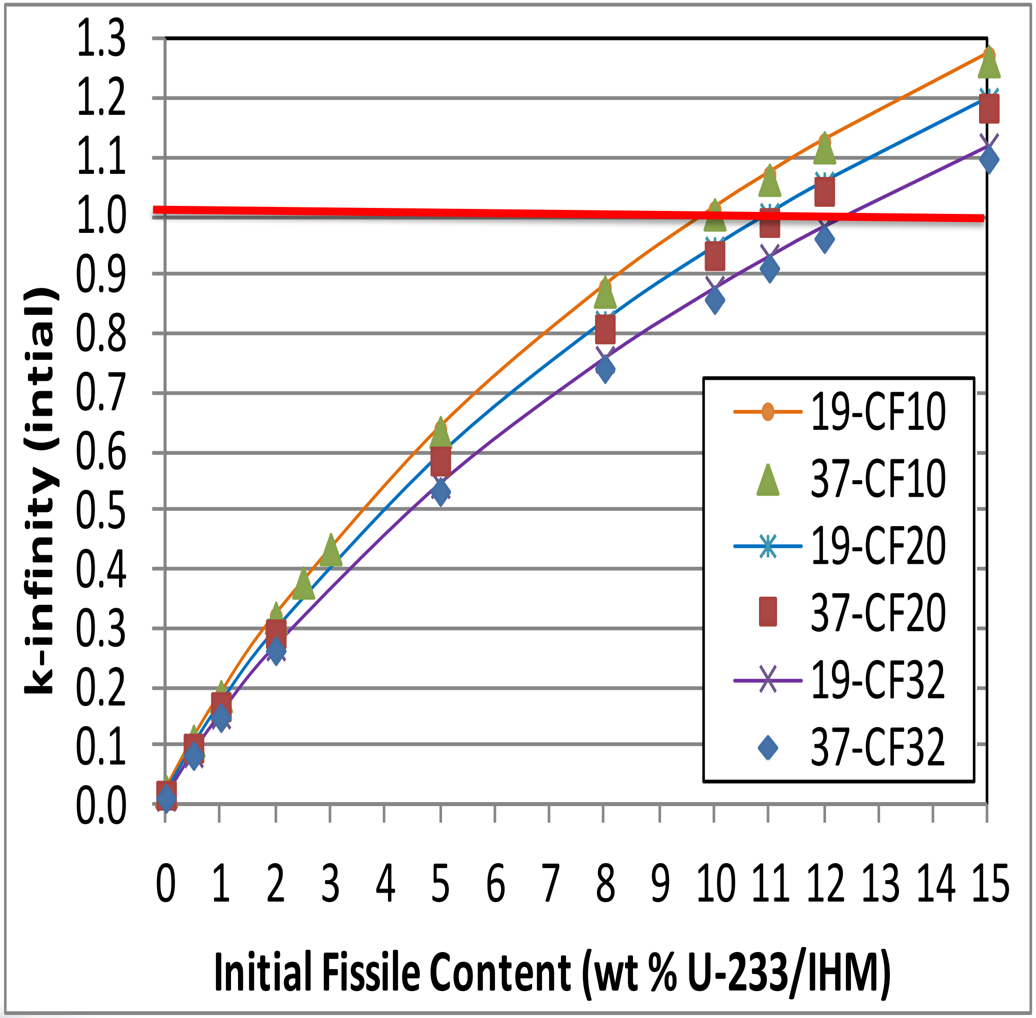
Blanket Lattice Physics Modeling

- **WIMS-AECL Version 3.1 – Lattice Physics**
  - **2-D collision-probability integral neutron transport.**
  - **89-group nuclear data library, ENDF/B-VII.0.**
  - **Single-cell lattice with burnup in a infinite spectrum.**
  - **Quick, *but approximate* calculations for fast spectrum.**
  - **Can extract 1-group homogenized diffusion data.**

- **Burnup with constant, nominal specific power:**
  - **19.5 kW/kg = 150 kW / 7.7 kg (19-CF32 bundle) .**
  - **150 kW = 3,000 kW / 20 bundles .**

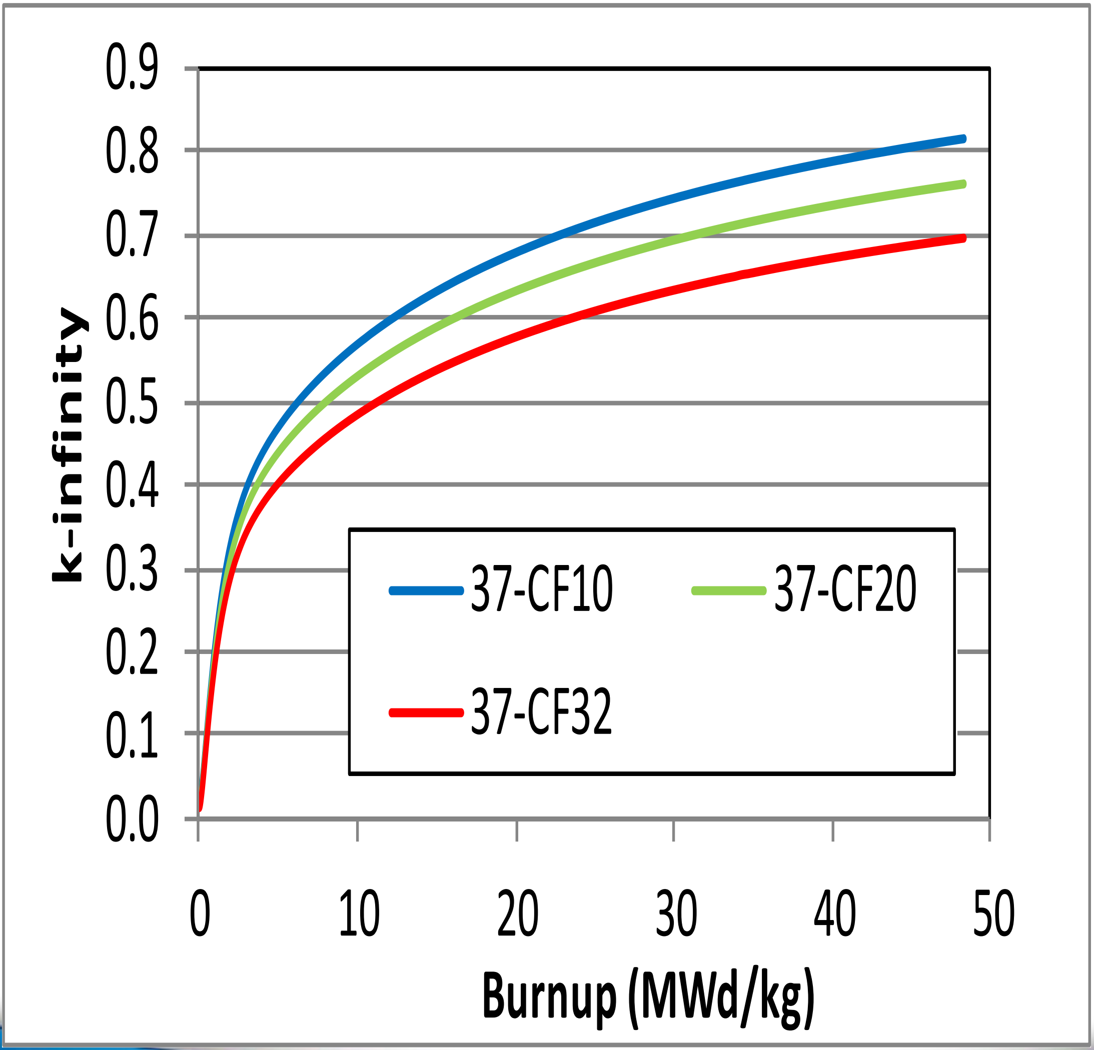
Lattice Reactivity vs. Fissile Content

- To get k-inf ≥ 1.000
  - Need ≥ 10 wt% U-233
  - More fuel mass with:
    - 19-CF10 or
    - 37-CF10
- Large Diffusion Coeff:
  - D = 5.5 - 6.5 cm (CF10)
  - D = 6.7 - 7.5 cm (CF20)
  - D = 7.5 - 8.3 cm (CF32)
- Large Migration Length:
  - M = 80 - 90 cm (CF10)
  - M = 100 – 110 cm (CF20)
  - M = 115 – 125 cm (CF32)



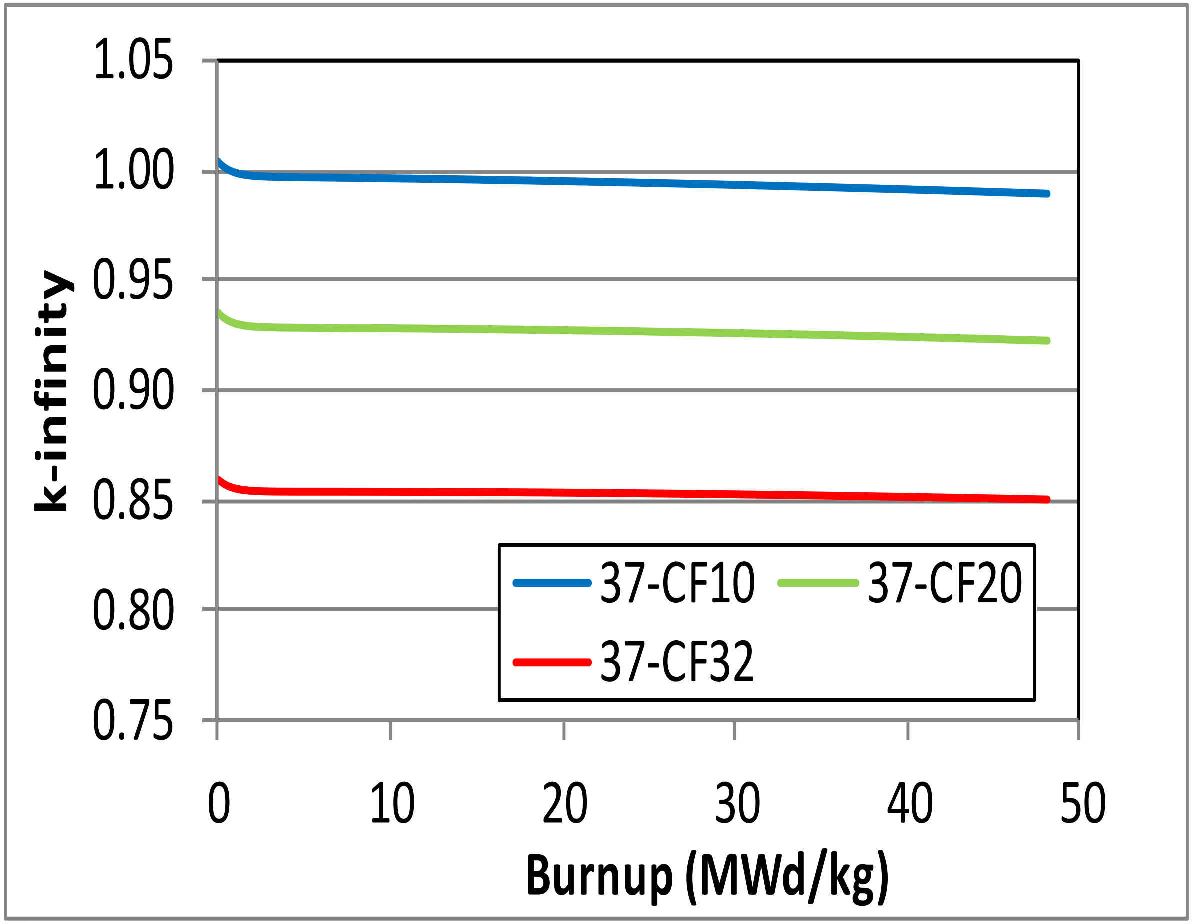
Lattice Reactivity vs. Burnup

- For 0 to 2 wt% fissile
  - $K_{inf} \lesssim 0.3$ (0 MWd/kg)
  - $K_{inf} \lesssim 0.8$ (45 MWd/kg).
- For 0 wt% fissile
  - Ideal for breeding.
  - By 10 MWd/kg.
  - $K_{inf} \rightarrow 0.550$.
- 19-CF10, 37-CF10:
  - More reactivity.
  - More fuel mass.



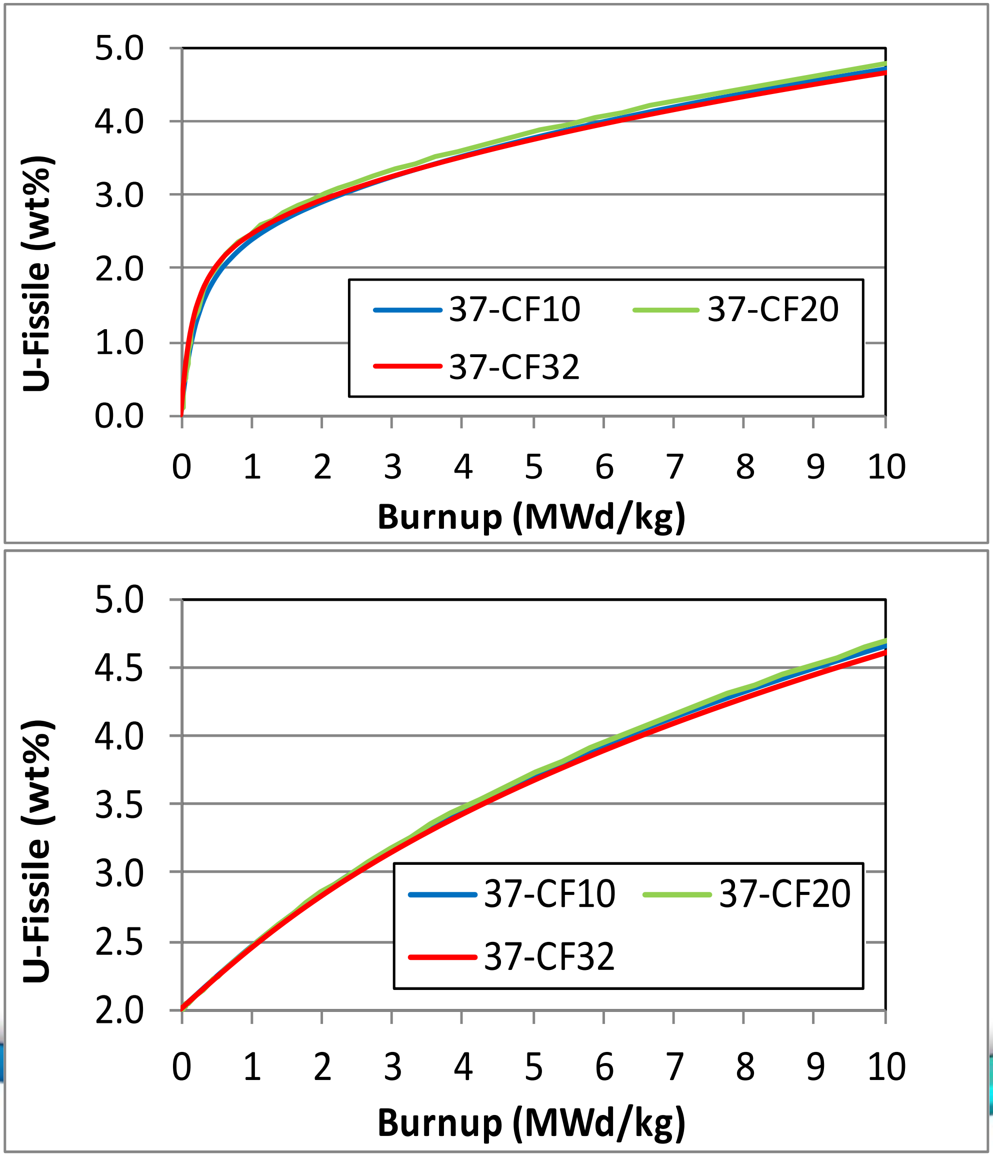
Lattice Reactivity vs. Burnup

- For 10 wt% fissile
  - K_{inf} ~ constant.
  - Ideal for power.
  - Leave in core long time.
  - K_{inf} ~ 1.000 for
    - 19-CF10
    - 37-CF10



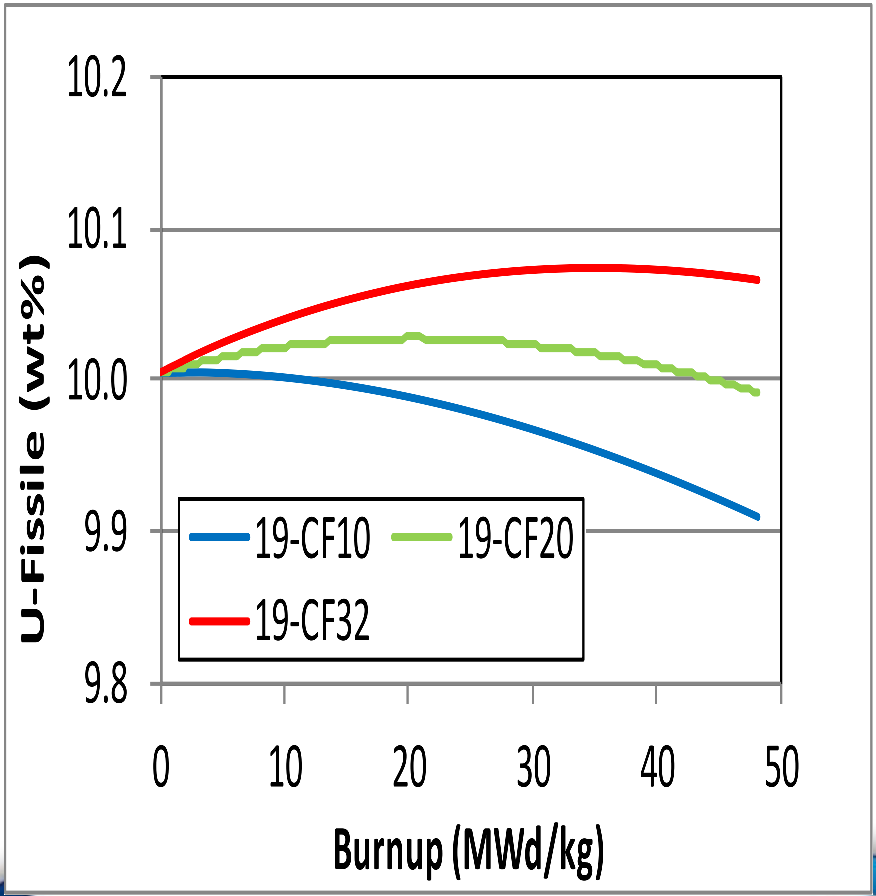
U-Fissile Content vs. Burnup

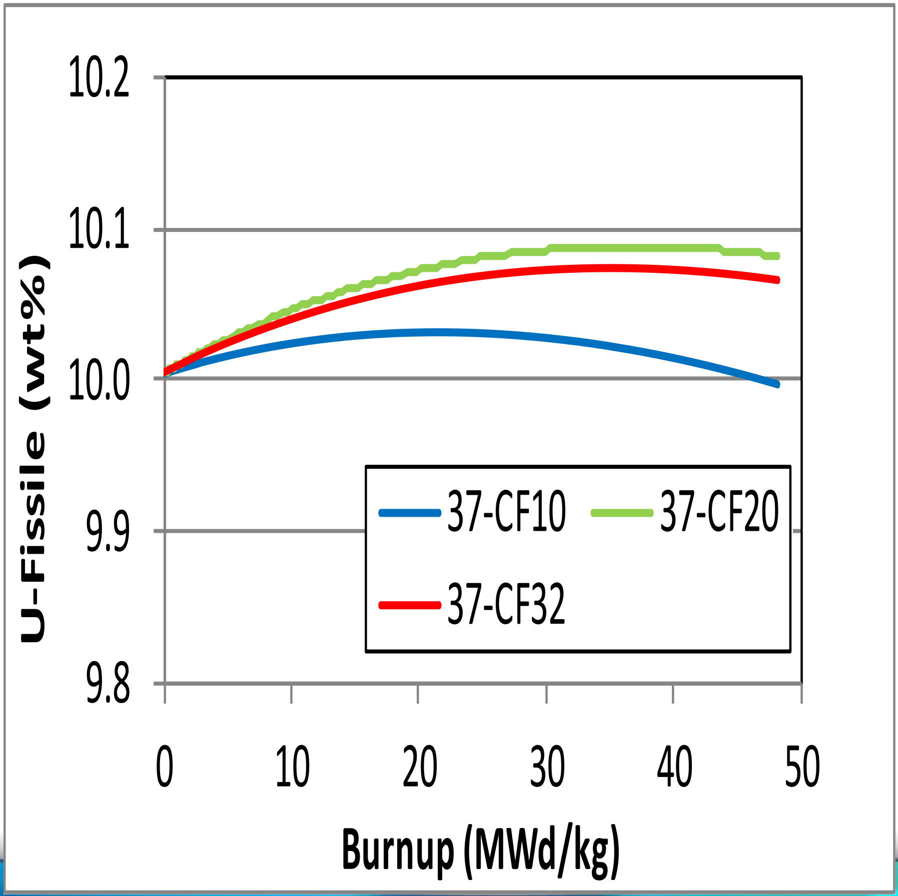
- U-233 + Pa-233 + U-235.
- Same trends as k_{inf}.
- For 0 to 2.0 wt% initial,
  - At ≥50 MWd/kg
  - 7.5 wt% to 8.0 wt%
- For 0 wt% initial, content at low burnup (≤ 10 MWd/kg) will depend more on power/flux history.
  - Calcs. with 19.5 kW/kg.
- Remove blanket ≤ 10 MWd/kg.
  - 2.5 wt% to 5.0 wt%
  - Re-process or direct use.



U-Fissile Content vs. Burnup

- **For 10 wt% fissile ~ constant.**
  - **Can stay in blanket indefinitely, or until structural integrity limit.**





Blanket Modeling

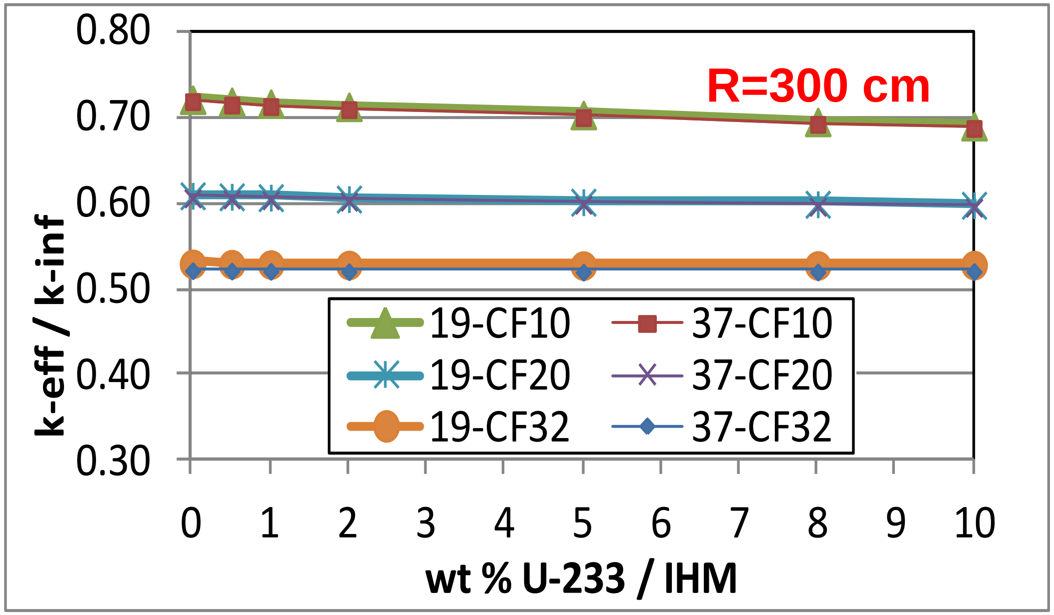
- **Use 1-group diffusion theory as first approximation.**
  - **Gain quick, *approximate understanding* to guide future analyses.**
  - **2-D (r,z) model of homogeneous blanket. Ignore outer radial reflector.**
  - **Use homogenized diffusion data from lattice calcs at zero burnup.**
    - **Provided from WIMS-AECL calculations.**
- **Solve for coefficients and constants in:**

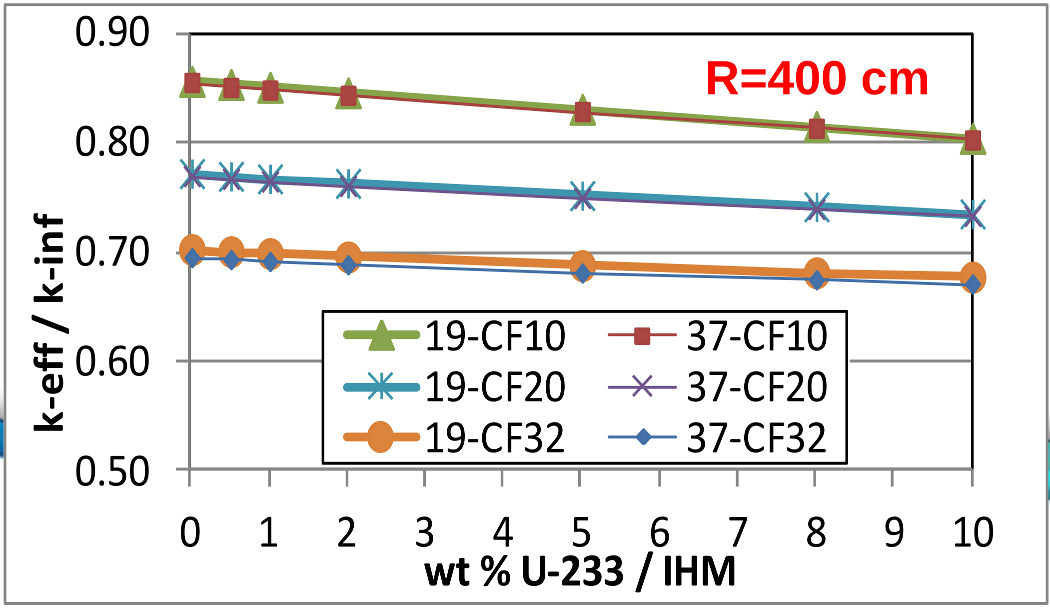
$$\phi(r,z) = \cos(\alpha z)\times\left(A_1 I_0(\beta r) + A_2 K_0(\beta r)\right)$$

  - $\phi(r = R_{outer} + d_{ext}, z) = 0$; $\phi(r, z = \pm(H/2 + d_{ext})) = 0.0$
  - $\alpha^2 = (\pi/H_{ext})^2$; $\beta^2 = B^2 - \alpha^2$; $B^2 = (\nu\Sigma_f - \Sigma_a)/D$; $\beta^2 \leq 0.0$
  - **Fixed neutron current with cosine axial profile at inner radius.**
- **Model neglects first-flight collisions of 14-MeV D-T neutrons.**
  - **Improve later with more rigorous modeling.**

Non-Leakage for Different Blanket Sizes

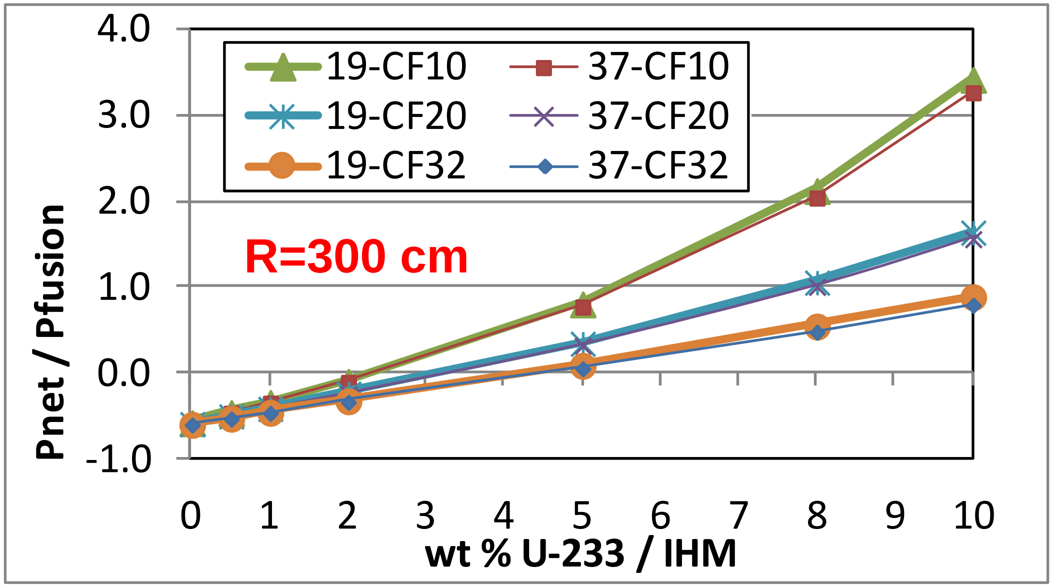
- Effect of blanket thickness.
- Lattice very leaky.
- K_{eff}/k_{inf}:
  - 0.53 to 0.72 for 300 cm.
  - 0.65 to 0.85 for 400 cm.
- Recall: need $k_{eff} \geq 0.26$:
  - $\geq$ 2.5 wt% U-233 (fresh).
  - $\geq$ 300 cm outer radius.
  - If $k_{eff}/k_{inf} = 0.72$
    - BU-ave $k_{inf} \geq 0.36$

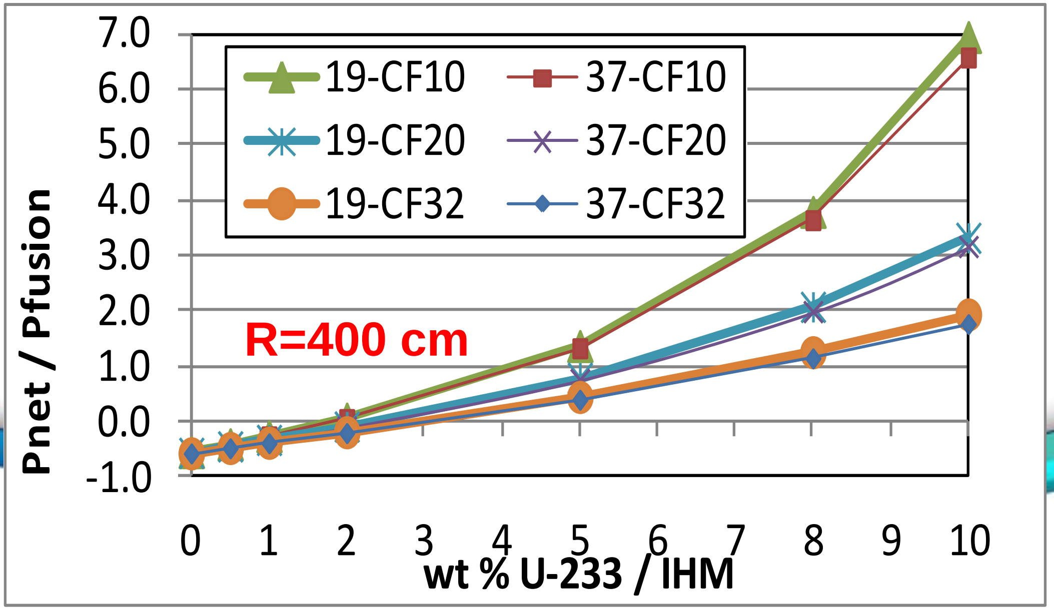




Net Power Production in HFFR

- η ~ 40% (similar to AGR).
  - Feasible with 750°C outlet coolant temp.
- $Q \geq 1.0$
  - First generation fusion.
- $P_{net} = 0.4\times(P_{fiss}+P_{fus}) - P_{fus}/Q$.
- For $P_{net} \geq 0$, 19-CF10, 37-CF10.
  - Fissile ≥ 2.5 wt% U-233
  - Higher for CF20, CF32 fuels.
- For P_{fus} = 100 MW,
  - 400-cm outer radius
  - 10 wt% U-233 / Th
  - P_{net} ~ 660 MWe





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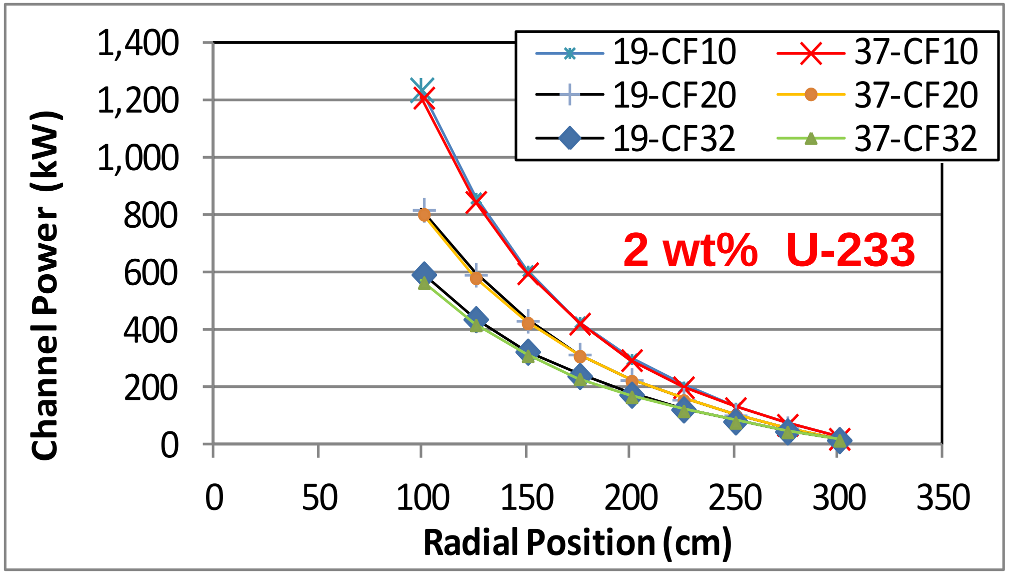
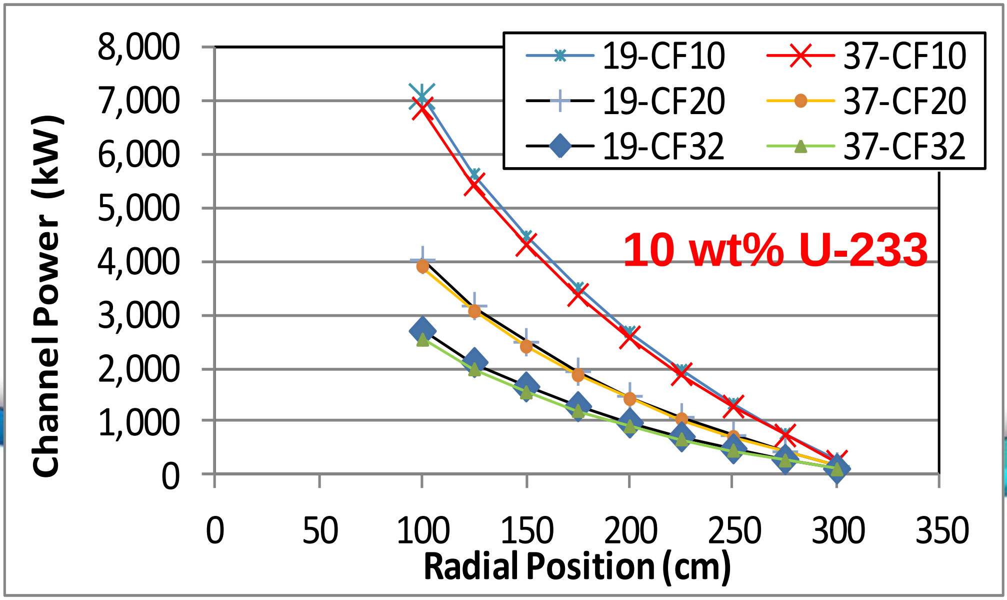
Radial Power Distribution in Blanket

- Homogeneous blanket.
- 100-cm inner radius.
- 300-cm outer radius.
- P_{fusion} = 100 MW.
- Peak neutron flux:
  - 9.0×10^{14} n/cm^2/s to
  - 1.5×10^{15} n/cm^2/s
- $P_{channel}$ = 0 to 7,000 kW
- Coolant mass flow rate needs to be adjusted for each channel for ΔT=500°C.
- For 7,000 kW, pumping power ~100 kW (1.4%).





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Break-even HFFR with Th-232 Fuel

- **19-CF10 and 37-CF10 superior in performance.**
  - **Fuel must be burned to ≥ 5.8 MWd/kg (19.5 kW/kg).**
  - **Fissile content at discharge ≥ 3.8 wt% U-233.**
  - **At P_{fusion} = 100 MW, 365 to 379 kg/yr U-233 produced.**

Lattice	k_{eff} Req.	k_{eff}/k_{inf} (1)	k_{inf} Req. (2)	Burnup at k_{inf} (MWd/t)	wt% U-233 at Burnup	BU-Ave. wt% U-233	$P_{fission}$ (MW) (3)	Fuel Used (kg/day)	Net Fissile Produce (kg/yr)
19-CF10	0.258	0.72	0.358	5,770	3.84	3.01	150	26.0	364.8
37-CF10	0.258	0.72	0.358	5,771	3.99	3.13	150	26.0	378.7
19-CF20	0.258	0.61	0.423	11,243	4.83	3.74	150	13.3	235.2
37-CF20	0.258	0.61	0.423	11,636	5.05	3.92	150	12.9	237.8
19-CF32	0.258	0.53	0.486	23,351	6.18	4.74	150	6.4	144.9
37-CF32	0.258	0.53	0.486	25,693	6.54	5.03	150	5.8	139.3

Conclusions

- **HFFR Concept Developed:**
  - **Using ideas from PT-HWR technology (also AGR, GCHWR).**
  - **Q=1.0 fusion reactor required as neutron source.**
  - **Cylindrical geometry (variant of a Magnetic Mirror expected).**
  - **Operate with U-233/ThO_2 fuel for power and breeding.**
- **Pros and cons of pressure-tube blanket for HFFR.**
  - **Pros: modularity, on-line refuelling, constant power, practical.**
  - **Cons: many pipes; larger neutron leakage; blanket less compact.**
- **Alternative approaches:**
  - **Tighter lattice pitches, but harder to engineer**
    - **Less clearance for inlet, outlet header pipes.**
    - **Less clearance for refuelling machines.**
  - **Revert to pressure vessel approach with off-line, batch refuelling.**
    - **Complexities/challenges of refuelling gas-cooled fast reactors.**

Conclusions (cont.)

- Potential for direct use of irradiated HFFR fuel
  - Use in conventional fission reactors without reprocessing.
  - Fissile content would range from 2 wt% to 5 wt% U-233.
  - Rejuvenate spent fuel (0.5 to 1.5 wt% fissile) in HFFR.
    - Breed in HFFR, burn in fission reactor.
- If using pure ThO_2 blanket as feed,
  - Core-average fissile must be ≥ 2.5 wt% for net power.
    - Lattice results showed BU ≥ 5.7 MWd/kg required.
    - Potential effects due to power history.
    - 37-CF10 and 19-CF10 are preferred lattices.
  - If discharged at ≤ 4 MWd/kg, or ≤ 3.5 wt% U-233 in exit,
    - Blanket-average reactivity will be too low.
    - Insufficient neutron and power multiplication.
    - Insufficient reactivity for HFFR to be energy self-sufficient.

Future Options

- **Look at more heterogeneous blankets**
  - **Inner blanket with neutron multiplier / power region.**
    - **Use 10 wt% U-233/Th or NU; reactivity and power boost.**
  - **Outer blanket with 0 to 2 wt% U-233/Th-232 for breeding.**
- **Other permutations:**
  - **Tighter-pitch lattices, alternative bundle designs.**
  - **Bundles containing U, Pu, MA's, Li (for breeding tritium).**
- **More rigorous computational tools / data:**
  - **Full 3-D source-driven transport calculations w/ burnup.**
  - **MCNP6, Serpent, Monteburns/VESTA, or others *(underway).***
- **Develop solution for equilibrium blanket with on-line refuelling.**
  - **Analogous to what is done for PT-HWRs.**

Acknowledgements

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Questions?

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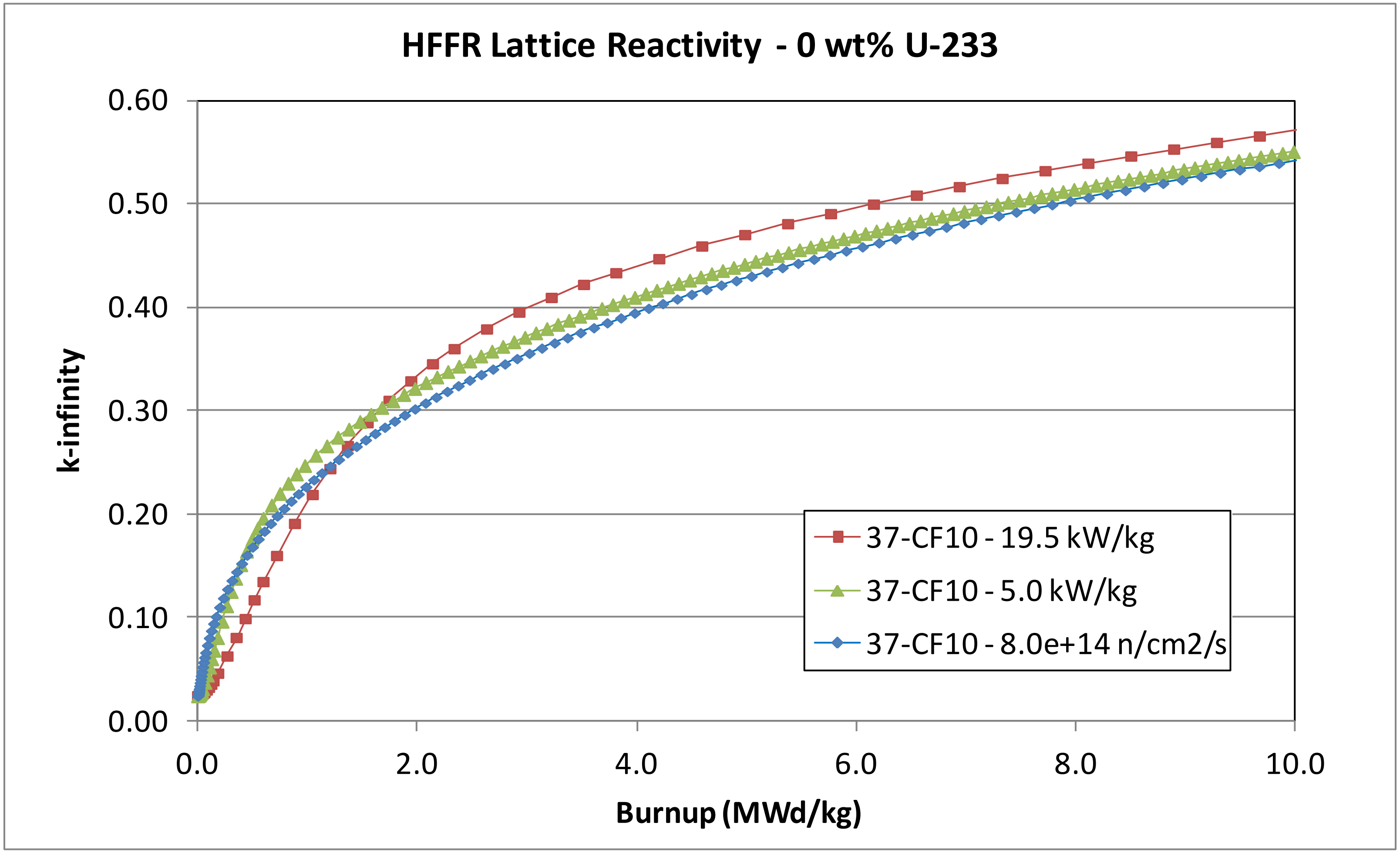
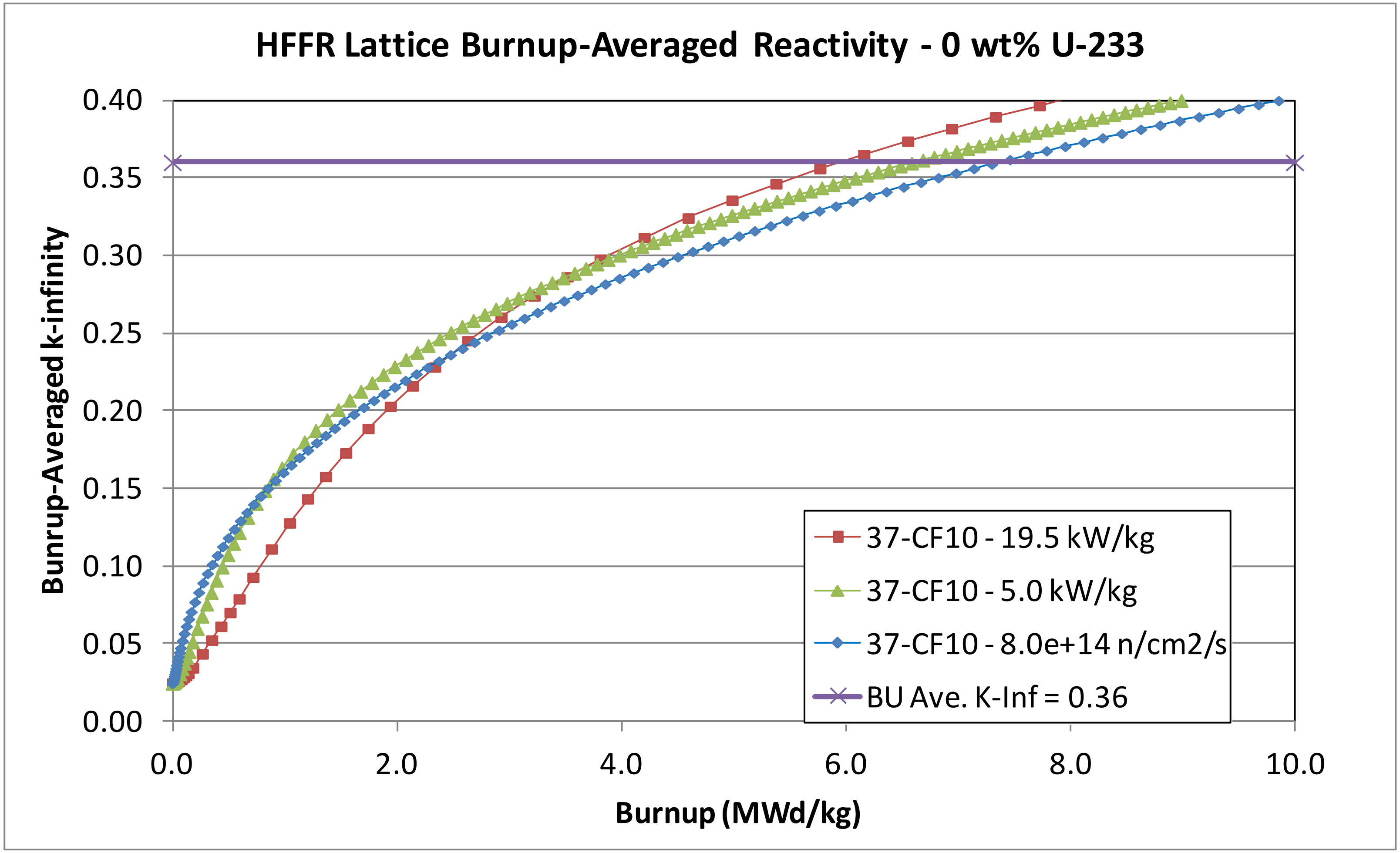
Extra Info - History Effects

- **K-inf vs. Burnup**



Extra Info - History Effects

- **Burnup-averaged k-inf vs. Burnup.**
- **BU-ave k-inf = 0.37 at BU = 5.8 to 8 MWd/kg, depending on history.**



Extra Info - History Effects

- U-Fissile vs. Burnup.
- Difference of ~0.5 wt% depending on history.