# SABR

# SUBCRITICAL ADVANCED BURNER REACTOR

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# ACKNOWLEDGEMENT

SABR is the sixth in a series of fast transmutation reactor concepts that have been developed in faculty-student design projects at Georgia Tech.  The contributions of E. Hoffman, R. Johnson, J. Lackey, J. Mandrekas, C. De Oliviera, W. Van Rooijen, D. Tedder and numerous students in the Nuclear & Radiological Engineering Design classes is gratefully acknowledged.

# Motivation

- GNEP calls for building pure TRU-fuel ‘Advanced Burner Reactors’ (ABRs) to fission the Transuranics (TRU) in spent nuclear fuel

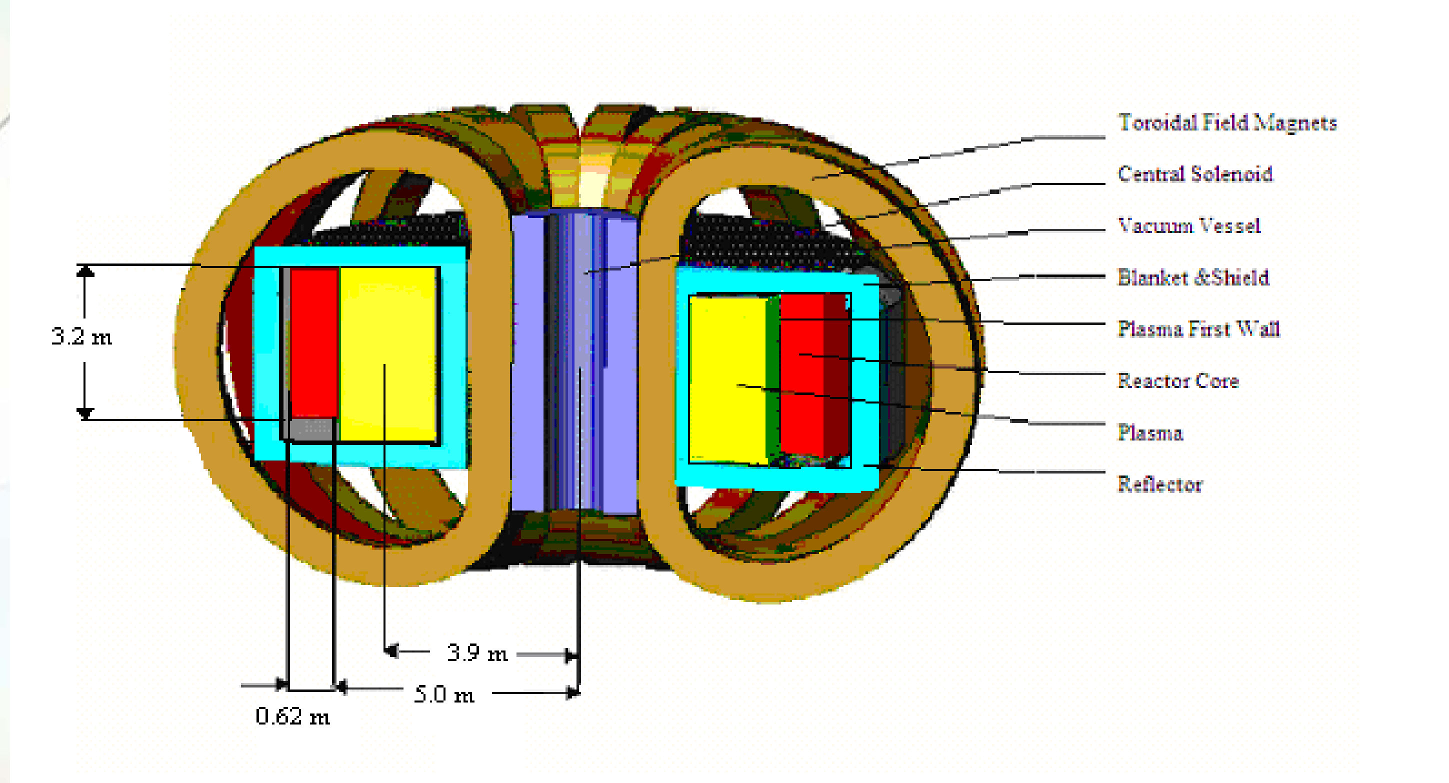
- Pure TRU-fuel transmutation reactors present safety & fuel cycle challenges that can be met by sub-critical operation

- ***SMALLER β:*** **operating sub-critical, $\rho < 0$, increases margin to prompt critical from $\beta$ to $\beta + |\rho|$, which more than compensates for the much smaller delayed neutron fraction, $\beta$, for TRU than for U235.**

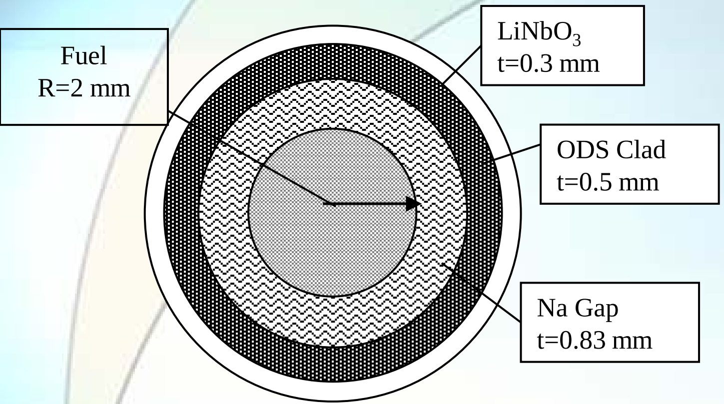
- ***SMALLER DOPPLER:*** **added margin to prompt critical $\beta + |\rho|$ with subcritical operation in part compensates the very small, probably positive, fuel Doppler temperature coefficient of reactivity in the absence of U238 in pure TRU fuel.**

- ***LARGER BURNUP ρ Decrement:*** **neutron source strength can be increased to offset large burnup reactivity decrement of pure TRU fuel (No U238), greatly increasing achievable TRU burnup per burn cycle**

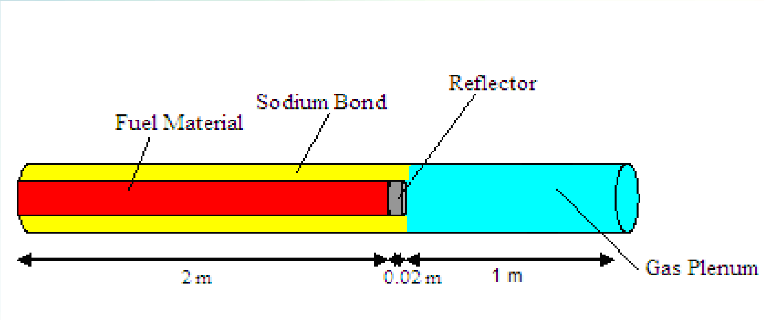
$$P_{fis} = C_1 \frac{kS_n}{(1-k)} = C_2 \frac{kP_{fus}}{(1-k)}$$




**Annular Core**
**Metal TRU-ZR Fuel**
**Sodium Cooled**
**ODS Structure**
**3000 MWt FAST REACTOR**
**4-batch Fuel Cycle**
**PYRO-processing**
**> 90% Burnup of TRU**
**200 MWt TOKAMAK Neutron Source**
**Based on ITER Physics & Technology**
**Tritium Self-sufficient**
**Operational 2035-40**




# FUEL

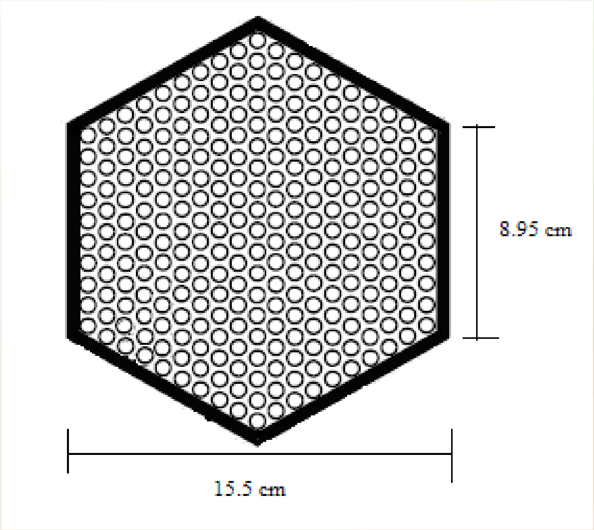



Axial View of Fuel Pin

**Composition 40Zr-10Am-10Np-40Pu (w/o)**
(Under development at ANL)

## Design Parameters of Fuel Pin and Assembly

| | | | |
|---|---|---|---|
| Length rods (m) | 3.2 | Total pins in core | 248778 |
| Length of fuel material (m) | 2 | Diameter_Flats (cm) | 15.5 |
| Length of plenum (m) | 1 | Diameter_Points (cm) | 17.9 |
| Length of reflector (m) | 0.2 | Length of Side (cm) | 8.95 |
| Radius of fuel material (mm) | 2 | Pitch (mm) | 9.41 |
| Thickness of clad (mm) | 0.5 | Pitch-to-Diameter ratio | 1.3 |
| Thickness of Na gap (mm) | 0.83 | Total Assemblies | 918 |
| Thickness of $LiNbO_3$ (mm) | 0.3 | Pins per Assembly | 271 |
| Radius Rod w/clad (mm) | 3.63 | Flow Tube Thickness (mm) | 2 |
| Mass of fuel material per rod (g) | 241 | Wire Wrap Diameter (mm) | 2.24 |
| $Volume_{Plenum}$ / $Volume_{fm}$ | 1 | Coolant Flow Area/ assy ($cm^2$) | 75 |

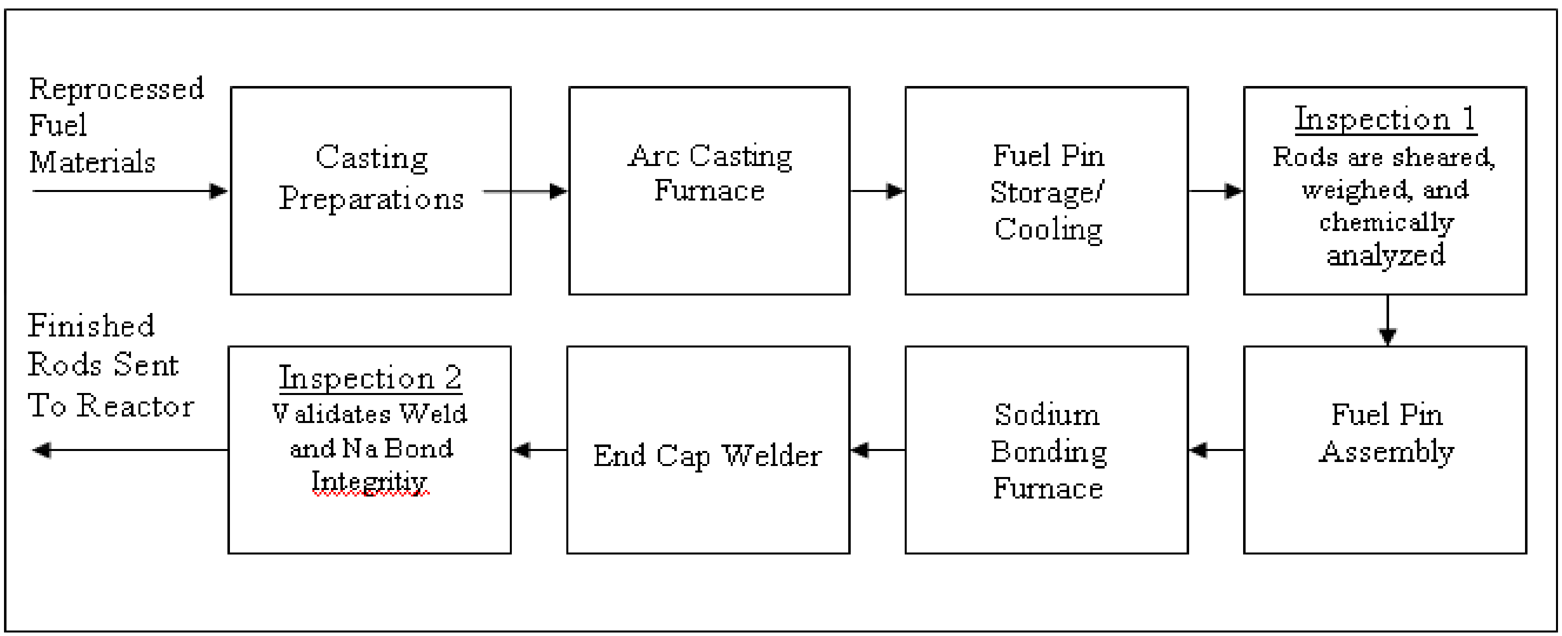



Cross-Sectional View Fuel Assembly

# Fuel Fabrication Facility

(Based on ongoing ANL R&D)

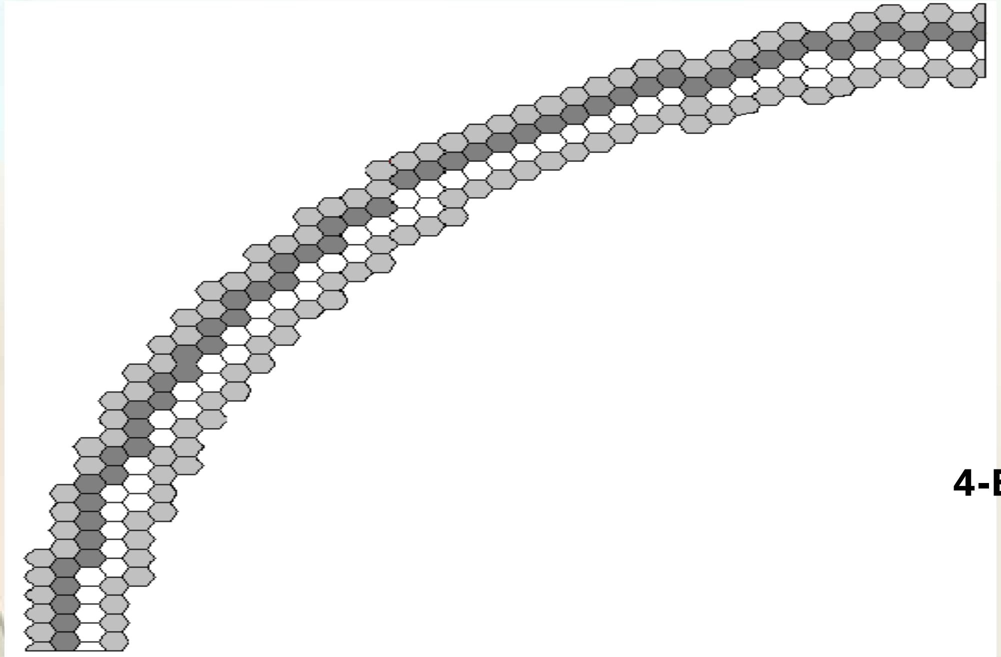
- Assuming downtime of 33%, one facility could produce rods containing 8,760 kg TRU/yr
- The initial fuel loading for SABR (4 batches) requires 35,996 kg TRU
- To fabricate the initial fuel loading would require either 4 years - using 1 fabrication facility, or 1 year - using 4 facilities



# Neutronics

**CODES**

| | |
|---|---|
| **EVENT** | Multigroup, 2D Spherical Harmonics (238 and 27 GRPS) |
| **MCNP** | Continuous Energy Monte Carlo |
| **CSAS** | Calculates Event X-SECTS |
| **NJOY** | Doppler Broaden ENDF/B-VI.6 and –VII Libraries |
| **SCALE/ORIGEN** | Isotopic Burnup |

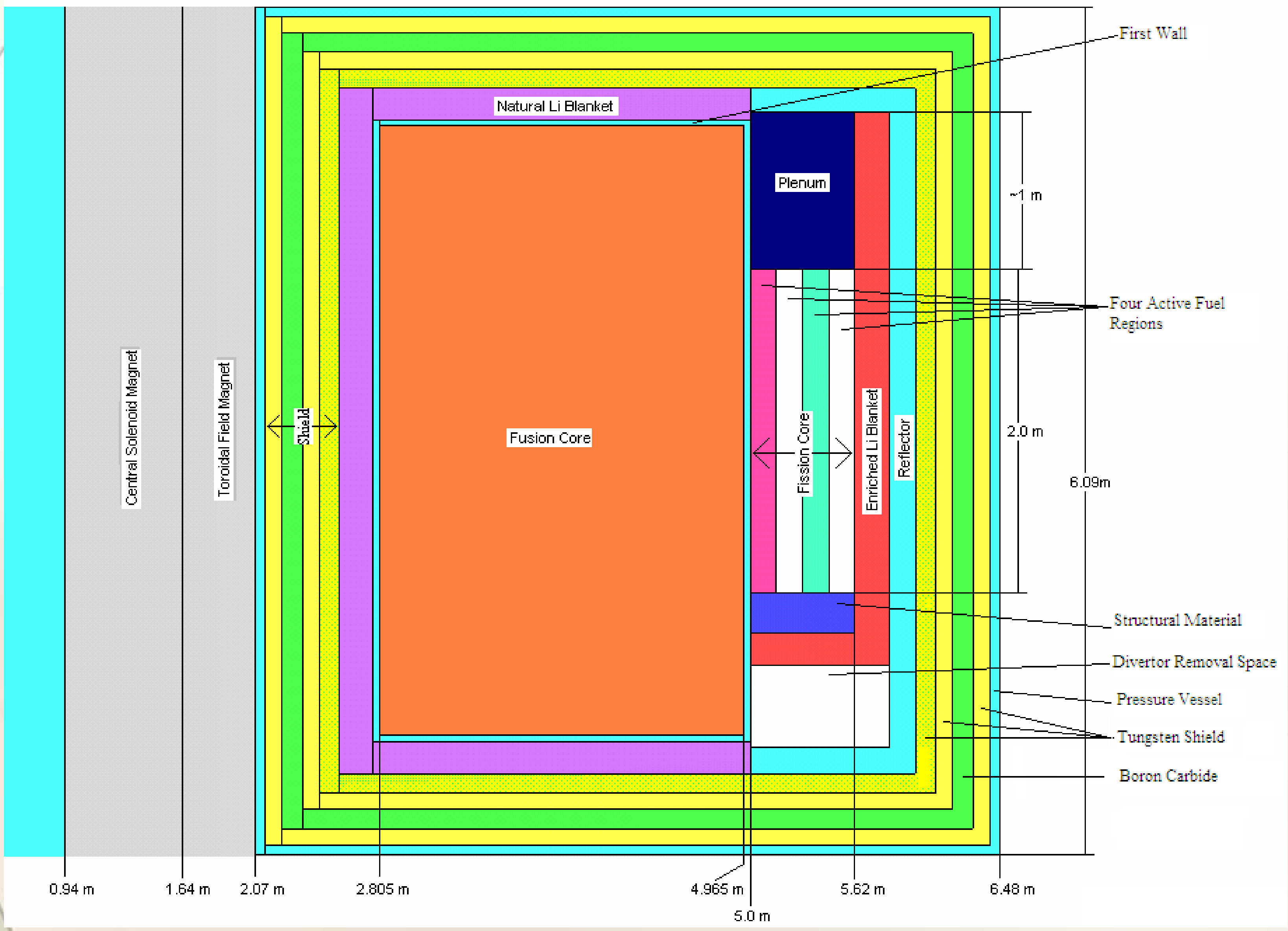


**4-Batch Layout of Fuel Assemblies**

Initial Loading of 36 MT of Fresh TRU Yields $K_{eff}$ = 0.95.

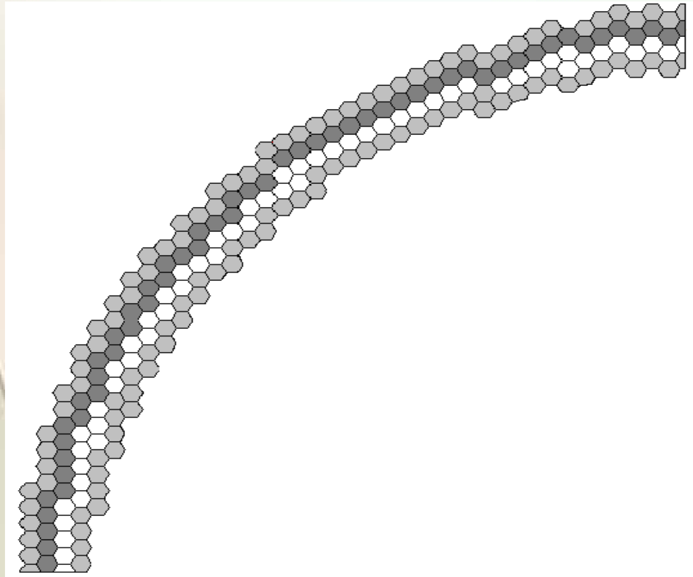
16 $B_4C$ Control Assemblies Worth 9$.

# R-Z Cross section SABR calculation model



# 4-BATCH FUEL CYCLE

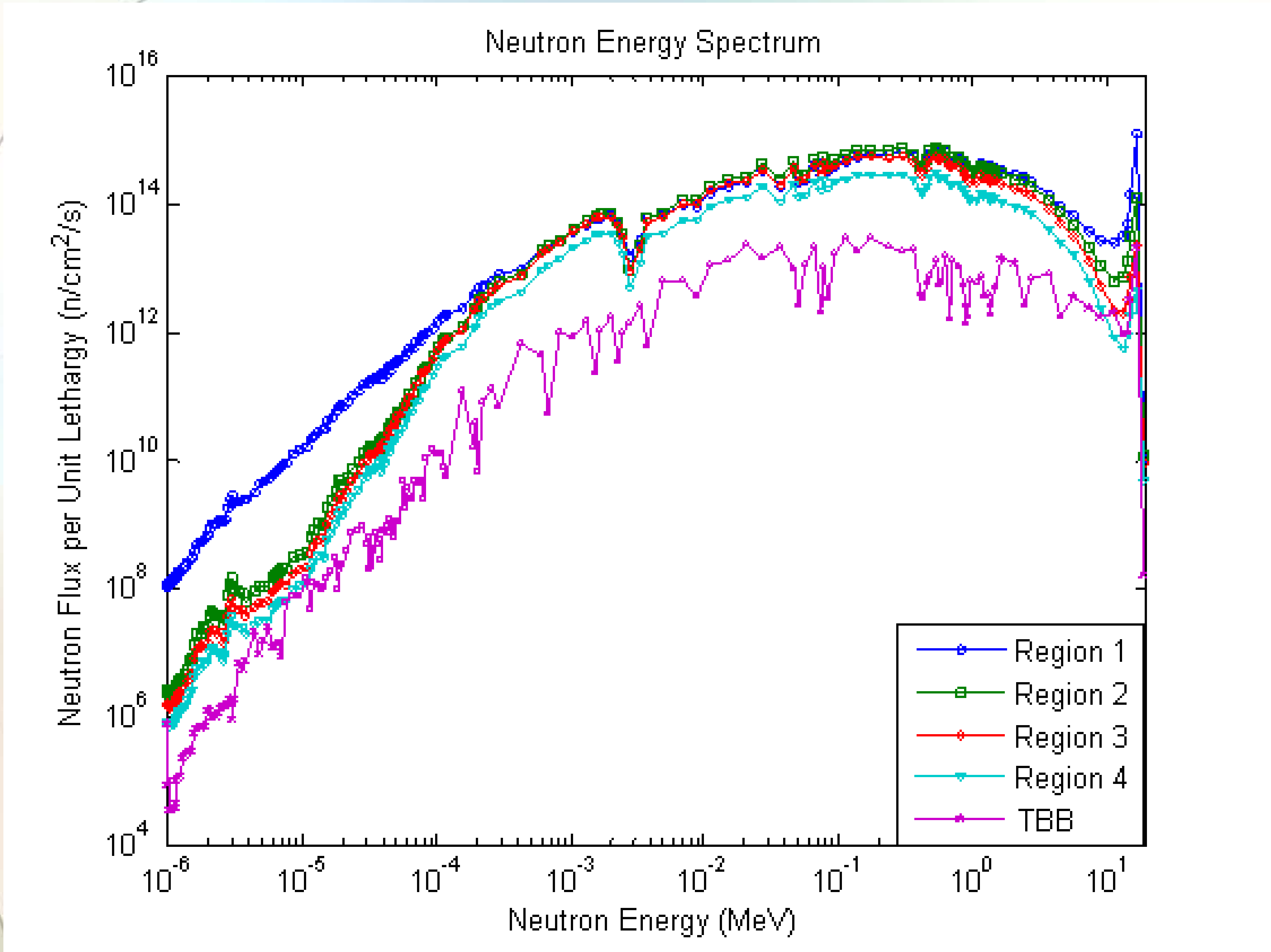
- 4 750-d burn cycles
- 3000 d (8.2 yr) total residence
- $k_{eff}$ = 0.95 fresh TRU (BOL)
- $k_{eff}$ = 0.89 (BOC) to 0.83 (EOC)
- $P_{fus}$(MW)= 99 (BOC) to 164 (EOC)
- 25% TRU burnup per 4-batch burn cycle, >90% with repeated recycling
- $P_{fis}$ = 3000MWt transmutes 1.06 MT TRU/FPY
- 1000 MWe LWR produces 0.2 MT TRU/yr
- Fuel cycle constrained by 200 dpa (8.4 FPY) clad radiation damage lifetime.

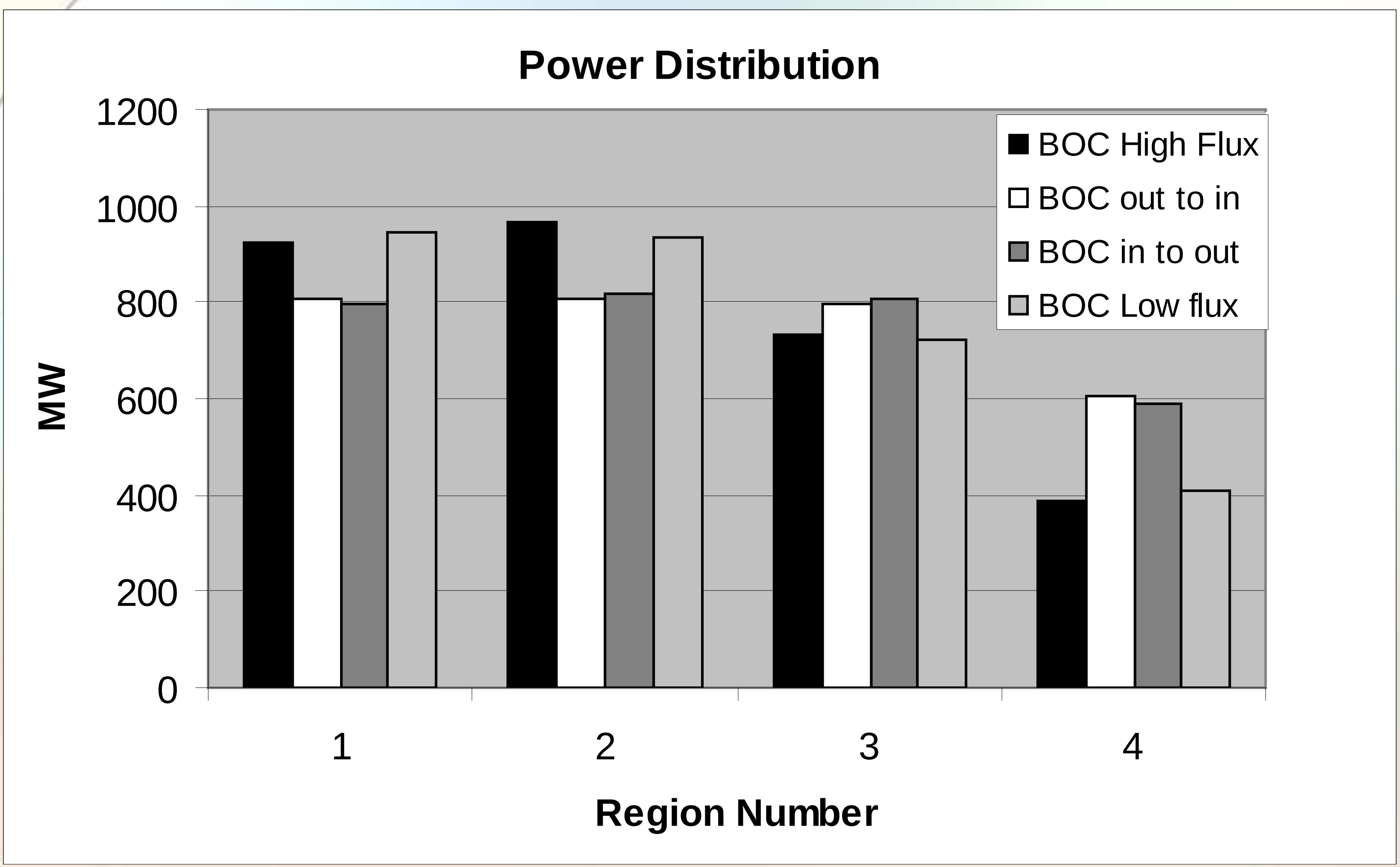


ANNULAR CORE CONFIGURATION

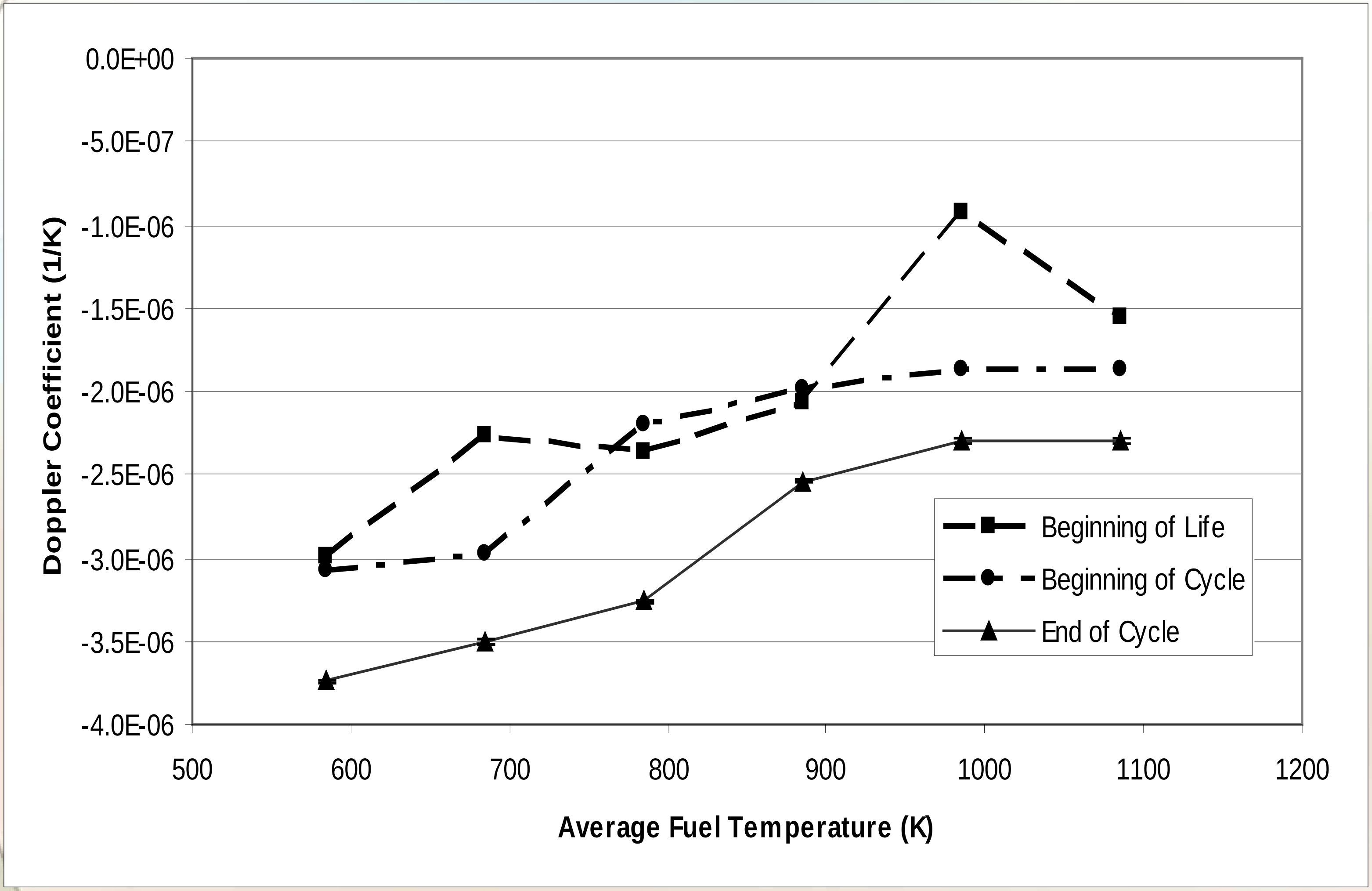
**TRU FUEL COMPOSITION**

| Isotope | Beginning of Life | Beginning of Cycle | End of Cycle |
|---|---|---|---|
| Np-237 | 16.67 | 15.52 | 14.77 |
| Pu-238 | 1.33 | 4.50 | 6.52 |
| Pu-239 | 38.67 | 34.57 | 32.04 |
| Pu-240 | 17.33 | 19.31 | 20.56 |
| Pu-241 | 6.67 | 5.73 | 5.17 |
| Pu-242 | 2.67 | 3.37 | 3.82 |
| Am-241 | 13.83 | 13.33 | 13.00 |
| Am-242m | 0.00 | 0.18 | 0.32 |
| Am-243 | 2.83 | 2.86 | 2.88 |
| Cm-242 | 0.00 | 0.26 | 0.34 |
| Cm-243 | 0.00 | 0.01 | 0.01 |
| Cm-244 | 0.00 | 0.33 | 0.53 |
| Cm-245 | 0.00 | 0.02 | 0.04 |

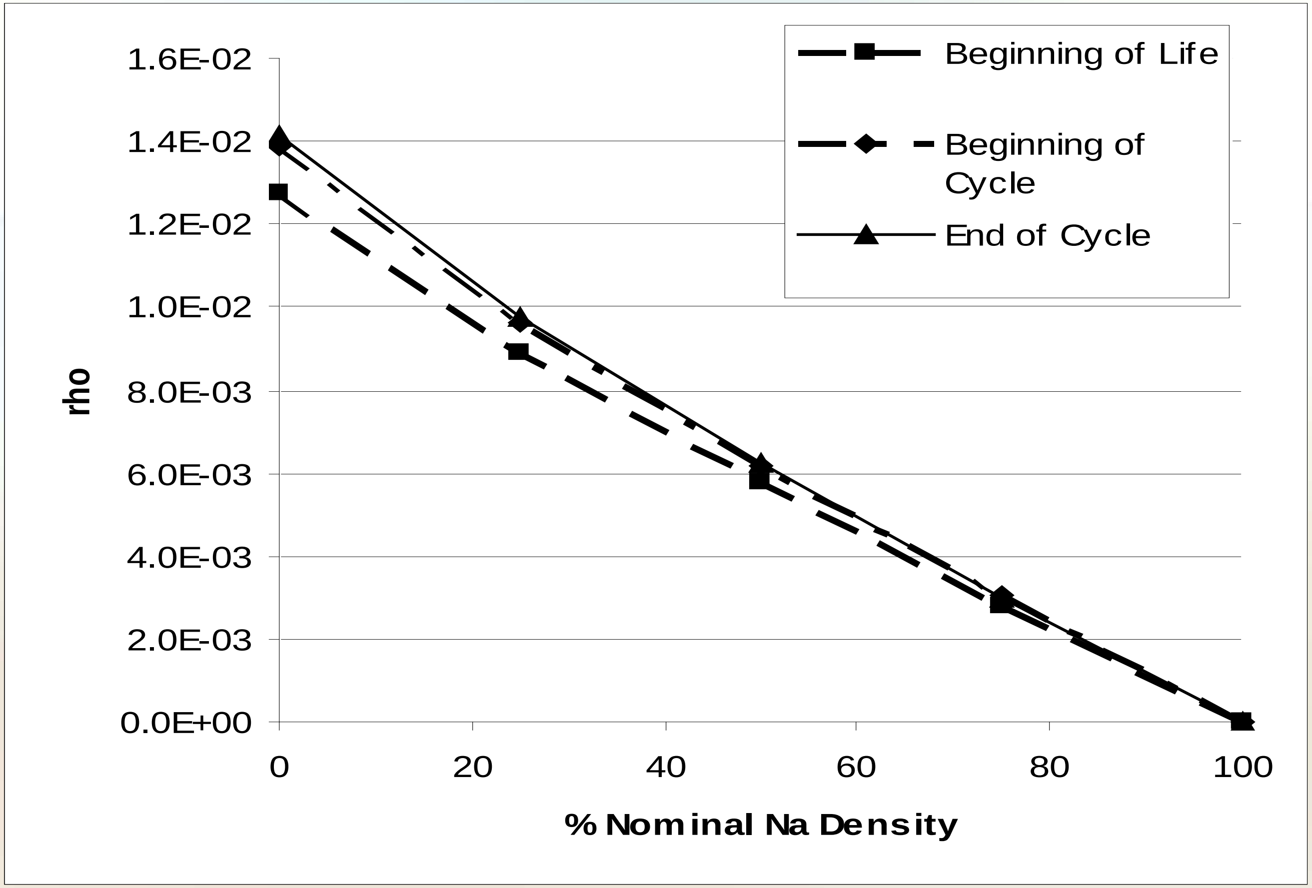

Neutron Energy Spectrum
Neutron Flux per Unit Lethargy (n/cm2/s)
10^16
10^14
10^12
10^10
10^8
10^6
10^4
10^-6
10^-5
10^-4
10^-3
10^-2
10^-1
10^0
10^1
Neutron Energy (MeV)
Region 1
Region 2
Region 3
Region 4
TBB


Power Distribution
1200
1000
800
600
400
200
0
MW
1
2
3
4
Region Number
BOC High Flux
BOC out to in
BOC in to out
BOC Low flux

# Doppler Coefficient vs Average Fuel Temperature



# Sodium Voiding Reactivity

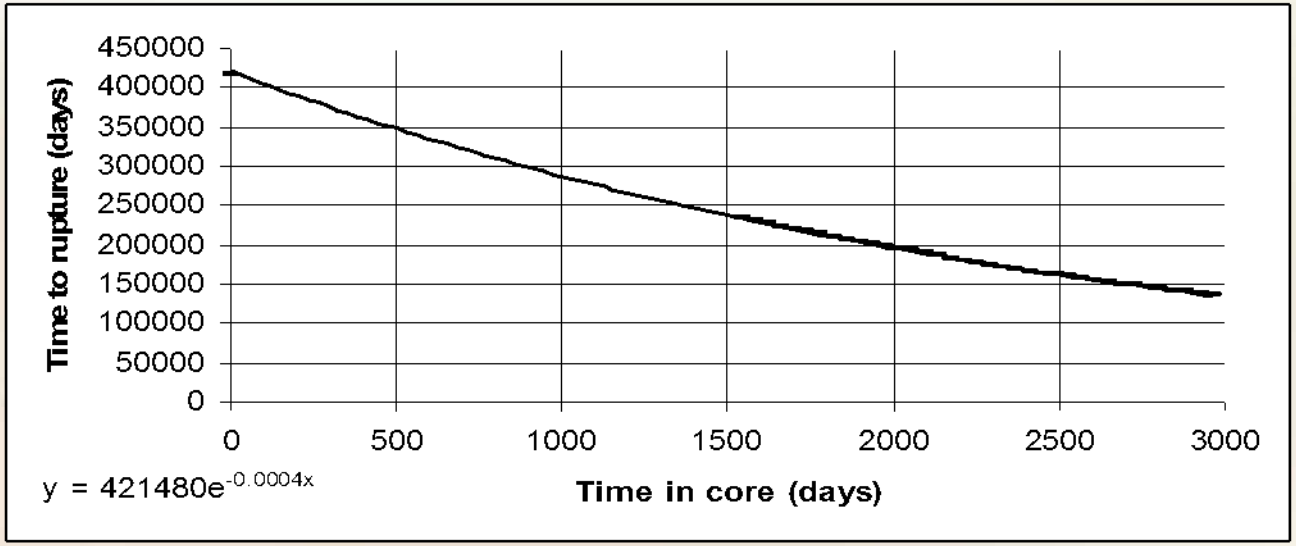


# Fuel Pin Analysis

- Fuel pin designed to a clad radiation damage lifetime of 200 dpa. At fast neutron fluence of $6.23 \times 10^{22}$ n/cm$^2$ per FPY (23.7 dpa/FPY), radiation damage lifetime is 8.44 FPY.
- Fuel plenum designed to withstand gas pressure buildup for 8.44 FPY and not exceed creep strain limit of 1%. Based on ORIGEN calculation of gas buildup, the pressure at 8.44 FPY will be 11.1 MPa, for which the creep strain < 1%.

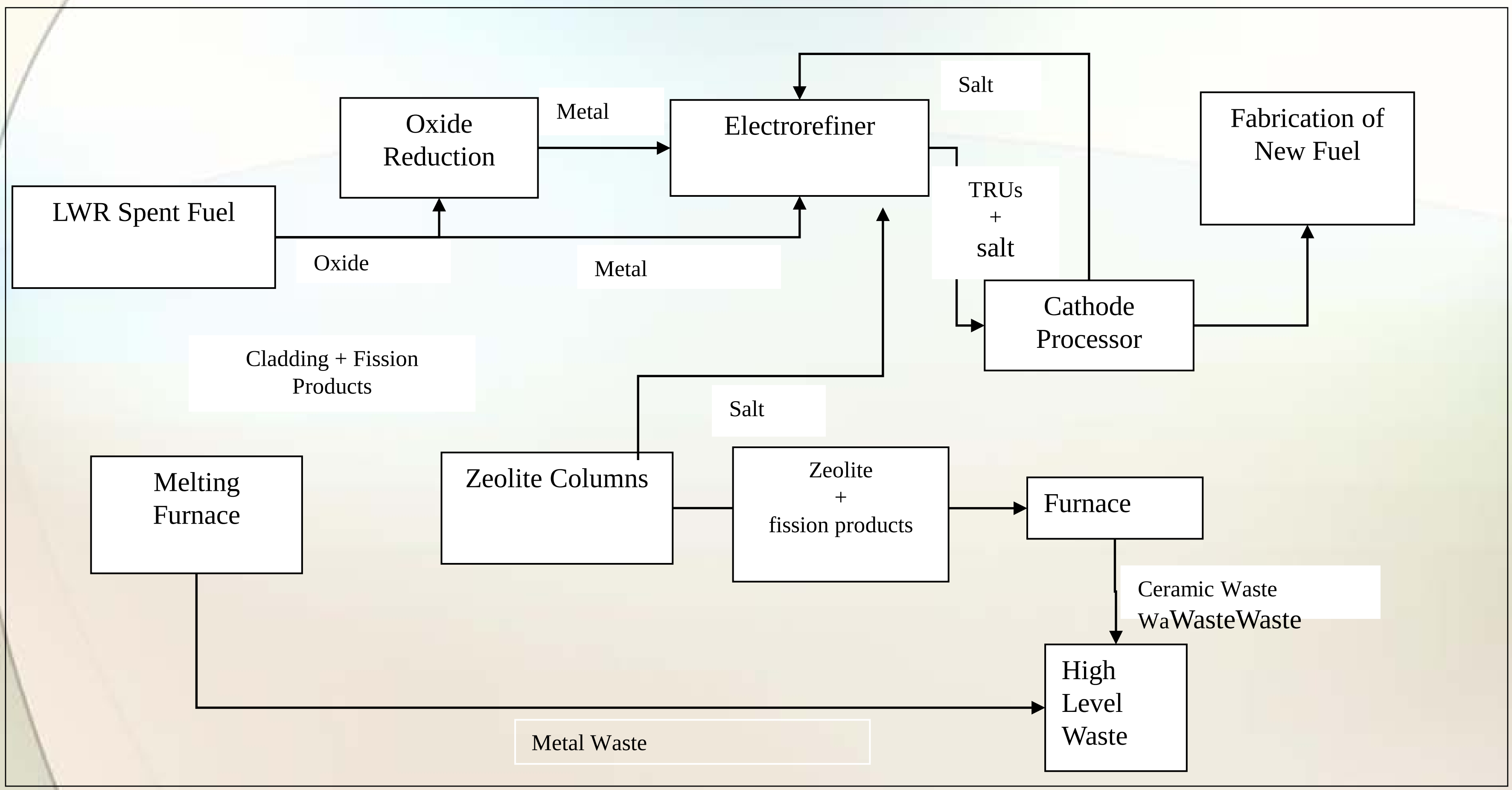
| HELIUM | BROMINE | HYDROGEN | IODINE | KRYPTON | XENON |
|---|---|---|---|---|---|
| 2.03E-3 | 1.70E-4 | 9.33E-5 | 1.93E-3 | 2.66E-3 | 3.93E-2 |

- C*umulative Damage Fraction* analysis indicates that the mean time to rupture is much greater than the actual time of the pin in the core throughout the fuel cycle.



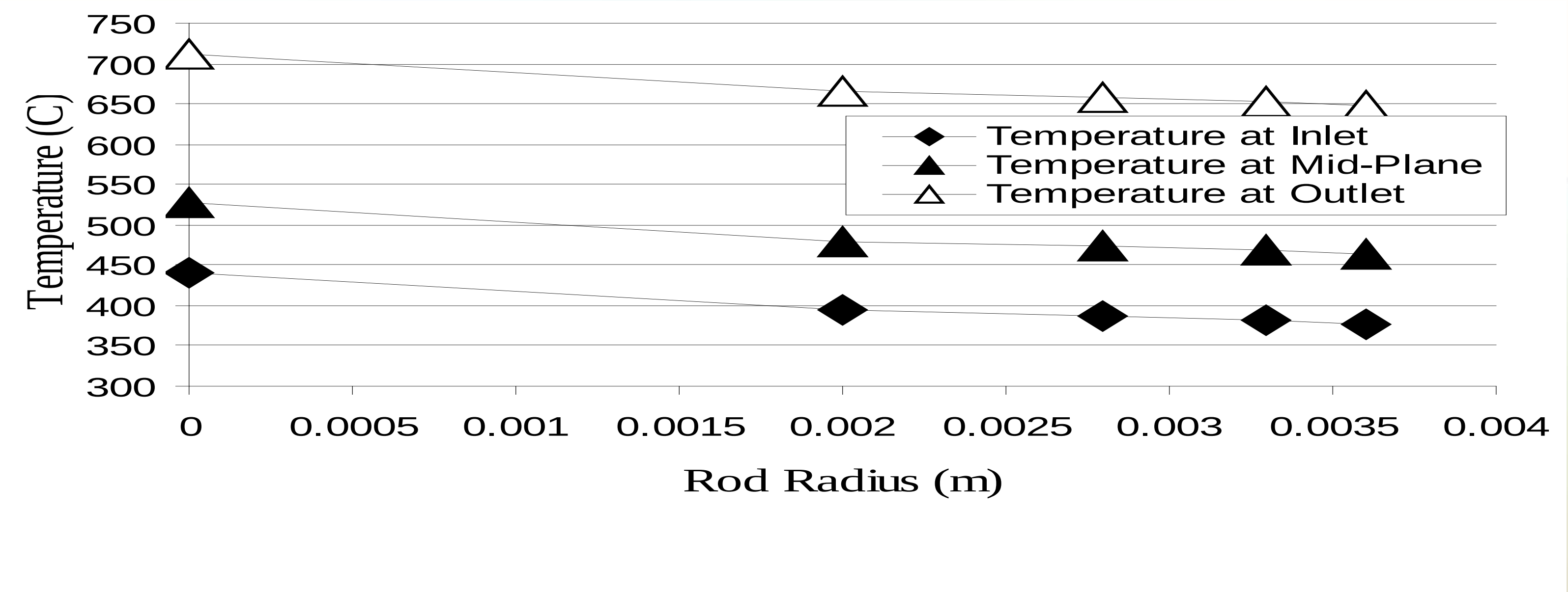
# Flowchart of Pyroprocessing Facilities

(ADAPTED FROM ONGOING ANL R&D)




RECOVERY RATES: Pu and Np 99.85%, Am 99.97% and Cm 99.95%.

# Core Thermal Analysis




Temperature Distribution in Fuel Pin (fuel 0.0-0.2cm, Na-gap 0.2-0.28cm, clad 0.28-0.33cm, $LiNbO_3$ 0.33-0.36cm)

# Core Thermal Analysis (cont.)

## Core Thermal and Heat Removal Parameters

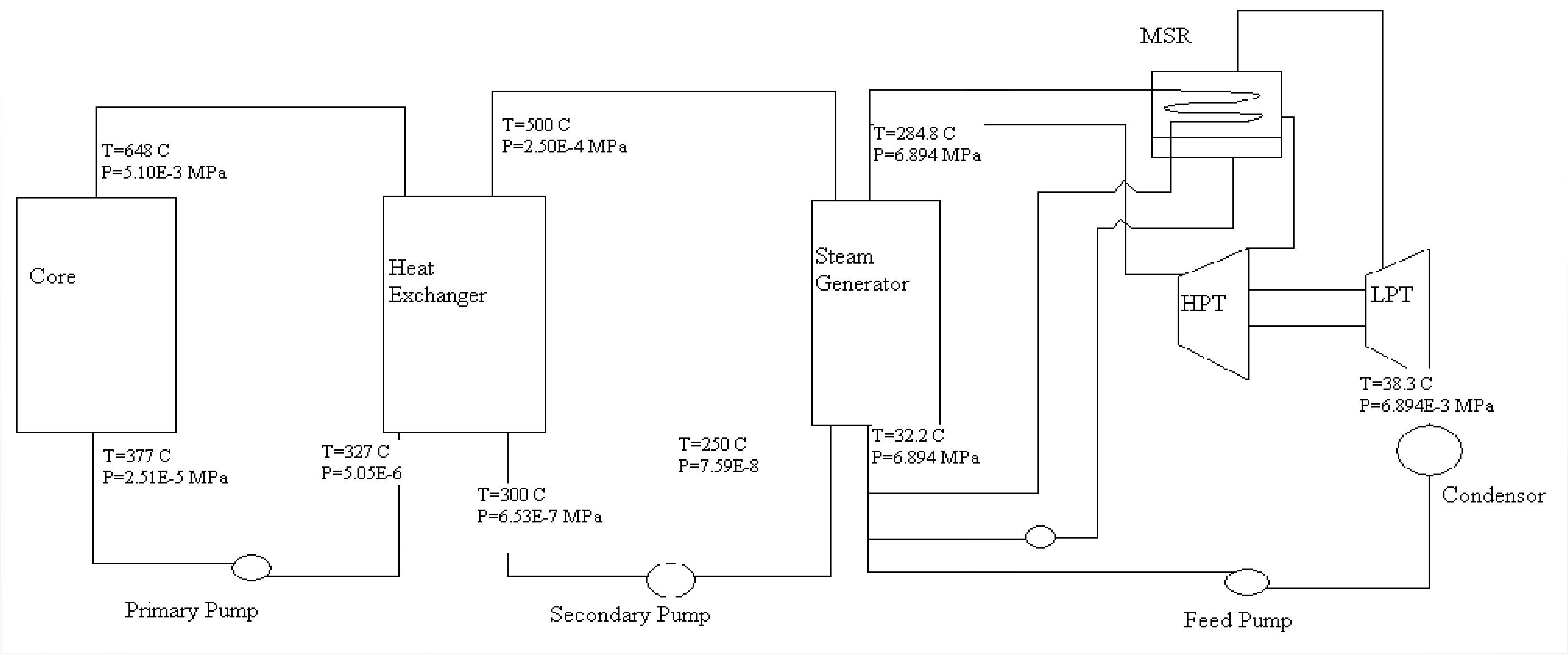
| | |
|---|---|
| Power Density | 73 $MW/m^3$ |
| Linear Pin Power | 6 kW/m |
| Coolant $T_{in}$ | 377 °C |
| Coolant $T_{out}$ | 650 °C |
| Min. Centerline Temp | 442 °C |
| Max Centerline Temp | 715 °C |
| Mass Flow Rate( $\dot{m}$ ) | 8700 Kg/s |
| Coolant Velocity(v) | 1.4 m/s |
| Total Pumping Power | 454 KW* |

In the absence of a lithium niobate electrically insulating coating on all metallic surfaces in the fuel assemblies, an MHD pressure drop of 68 MPa would be generated, requiring a pumping power of 847 MW.

# Core Heat Removal and Power Conversion

**Heat Removal and Power Generation Cycle**

- **Primary and intermediate Na loops**
- **Secondary water Rankine cycle**




| | |
|---|---|
| **THERMAL POWER GENERATED** | **3000 MWt** |
| **ELECTRICAL POWER PRODUCED** | **1049 MWe** |
| **ELECTRICAL POWER USED** | **128 MWe** |
| **NET ELECTRICAL POWER** | **921 MWe** |
| **ELECTRICAL CONVERSION EFFICIENCY** | **30.7 %** |

# Relationship Between Fusion Power and Reactor k

The multiplication constant of the fissionable fuel, k, decreases with fuel burnup, but the fusion neutron source (power) can be increased with TRU burnup to compensate reduction in k.

$$P_{fis} = const. \times \frac{k\,P_{fus}}{(1-k)}$$

Thus, the maximum $P_{fus}$ determines the minimum k for which the reactor can maintain a given fission power output, hence the TRU burnup in a fuel cycle.

**EQUILIBRIUM FUEL CYCLE PARAMETERS FOR $P_{fis}$ = 3000 MWt**

| **FUEL CYCLE** | **8.2 FPY** | **16.4 FPY** | **24.7 FPY** | **32.9 FPY** |
|---|---|---|---|---|
| TRU BURNUP | 24.9% | 49.7% | 72.4% | 94.9% |
| RAD. DAM.* | 194 dpa | 388 dpa | 585 dpa | 779 dpa |
| $k$ (BOC) | 0.987 | 0.917 | 0.856 | 0.671 |
| $k$ (EOC) | 0.927 | 0.815 | 0.714 | 0.611 |
| $P_{fus}$ (BOC) | 13 MW | 83 MW | 144 MW | 329 MW |
| $P_{fus}$ (EOC) | 73 MW | 185 MW | 286 MW | 389 MW |

*depends on spectrum and material.

# Neutron Source Design Parameters

Physics (stability, confinement, etc), Engineering (stress, radiation protection, etc) and Radial Build Constraints determine allowable design space.

The design parameters for a Tokamak neutron source for transmutation are similar to those for ITER.

Operation of ITER will serve as a prototype for a Tokamak fusion neutron source

# Neutron Source Design Parameters (cont.)

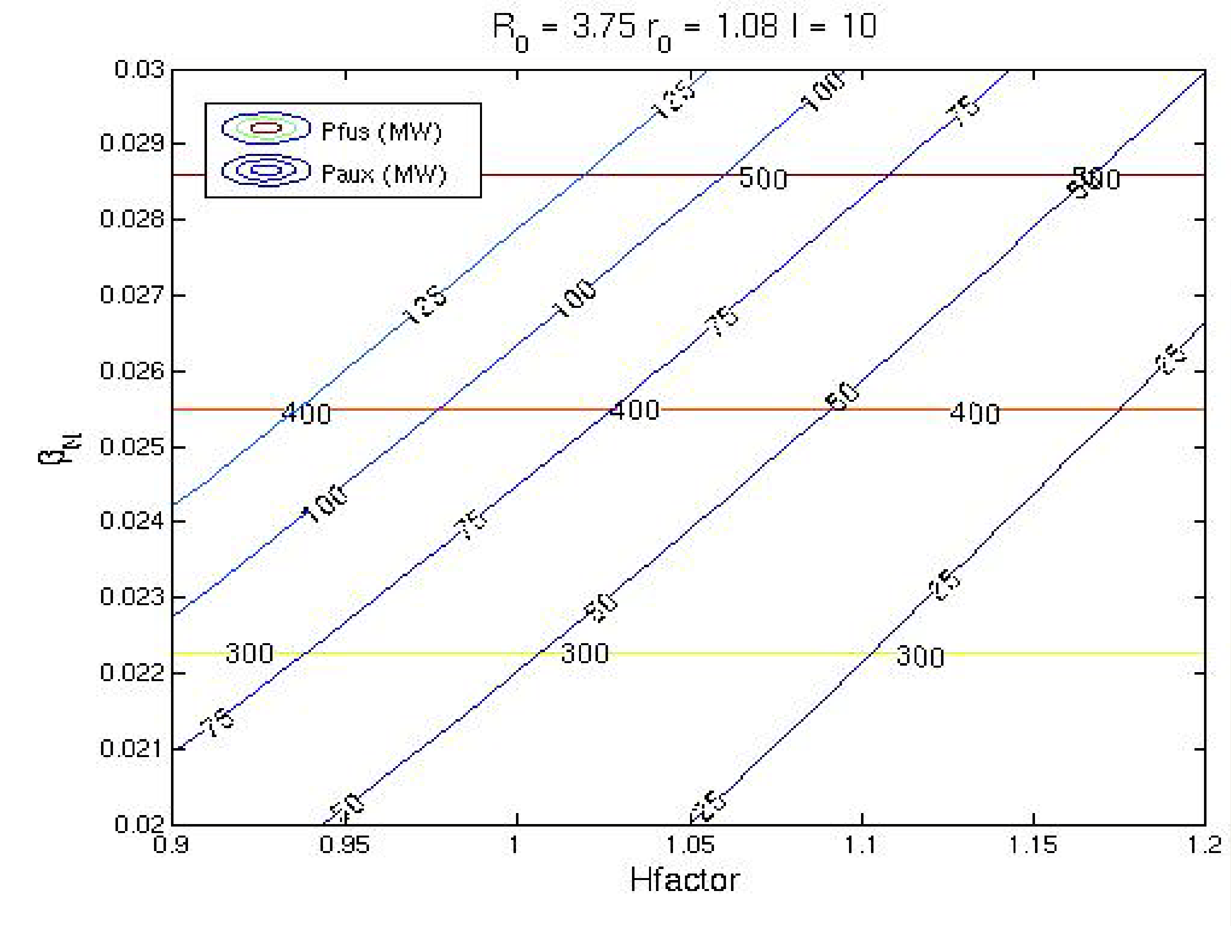
**SABR**
**TOKAMAK NEUTRON SOURCE PARAMETERS**

| Parameter | Nominal | Extended | ITER |
|---|---|---|---|
| Current, I (MA) | 8.3 | 10.0 | 15.0 |
| $P_{fus}$ (MW) | 180 | 500 | 410 |
| $S_{neut}(10^{19}$ #/s) | 7.1 | 17.6 | 14.4 |
| Major radius, R (m) | 3.7 | 3.7 | 6.2 |
| Aspect ratio, A | 3.4 | 3.4 | 3.1 |
| Elongation, κ | 1.7 | 1.7 | 1.8 |
| Magnetic field, B (T) | 5.7 | 5.9 | 5.3 |
| $B_{TFC}/B_{OH}$ | 11.8/13.5 | 11.8/13.5 | 11.8/13.5 |
| Safety factor, $q_{95}$ | 3.0 | 4.0 | |
| $H_{IPB98}(y,2)$ | 1.0 | 1.06 | 1.0 |
| Normalized beta, $\beta_N$ | 2.0 | 2.85 | 1.8 |
| Plasma Mult., $Q_p$ | 3.1 | 5.1 | 10 |
| H&CD Power, MW | 100 | 100 | 110 |
| $\gamma_{cd}$ ($10^{-20}$ A/W$m^2$) | 0.61* | 0.58* | |
| Bootstrap current, $f_{bs}$ | 0.31 | 0.26 | |
| Neutron$\Gamma_n$ (MW/$m^2$) | 0.6 | 1.8 | 0.5 |
| FW $q_{fw}$ MW/$m^2$) | 0.23 | 0.65 | 0.15 |
| Availability (%) | ≥ 50** | ≥50** | |

*May Require Extension Beyond ITER

***Definitely Requires Extension Beyond ITER

# 400-500 MW Operation Space at 10 MA



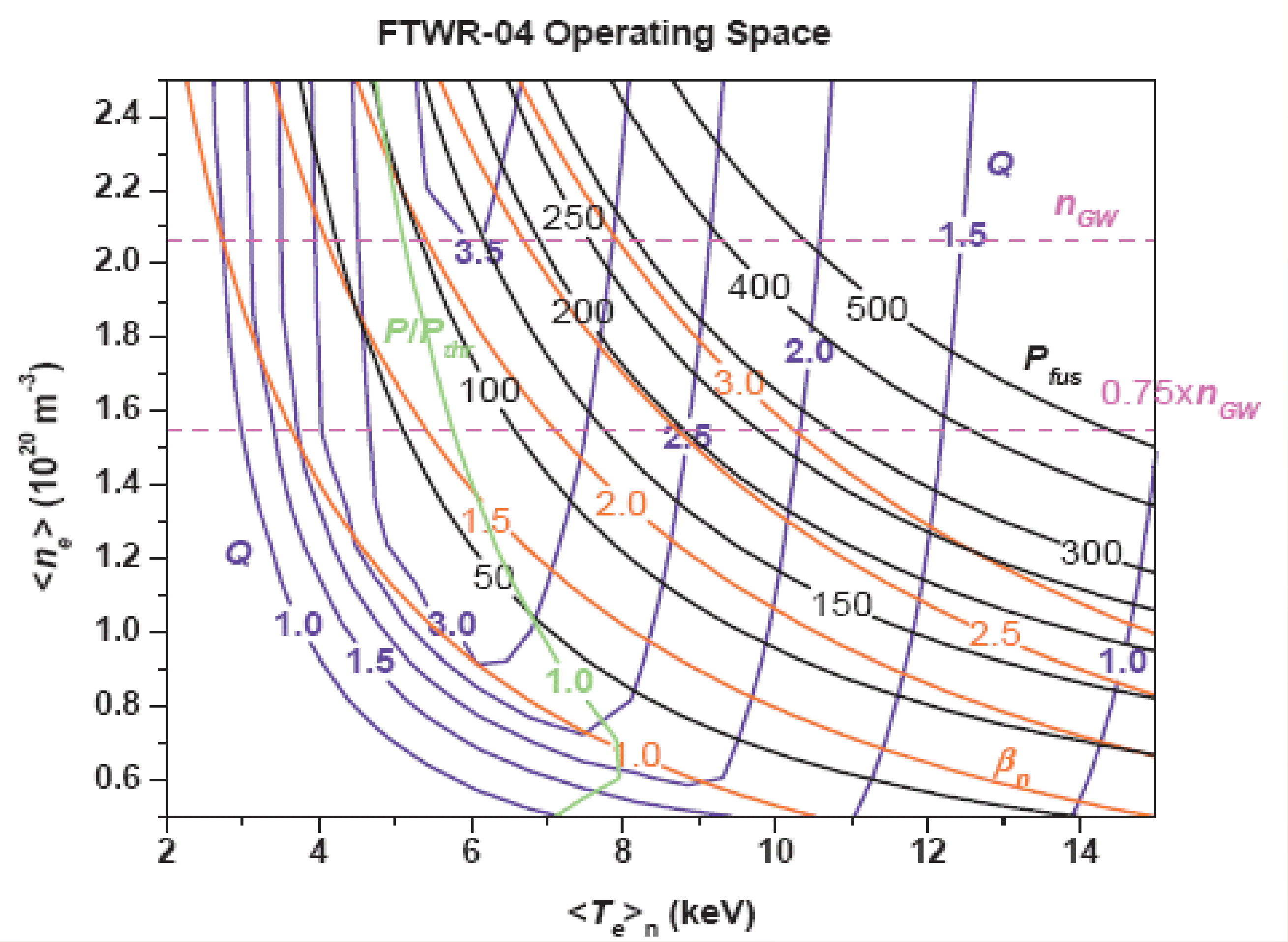

**Operational space of SABR at 10 MA[14]**

(*Horizontal lines indicate $Pf_{us}$ and slanted lines $P_{aux}$*)

There is a broad range of operating parameters that would achieve the 10 MA, 400-500 MW operating point.

# 150-200 MW Operating Space

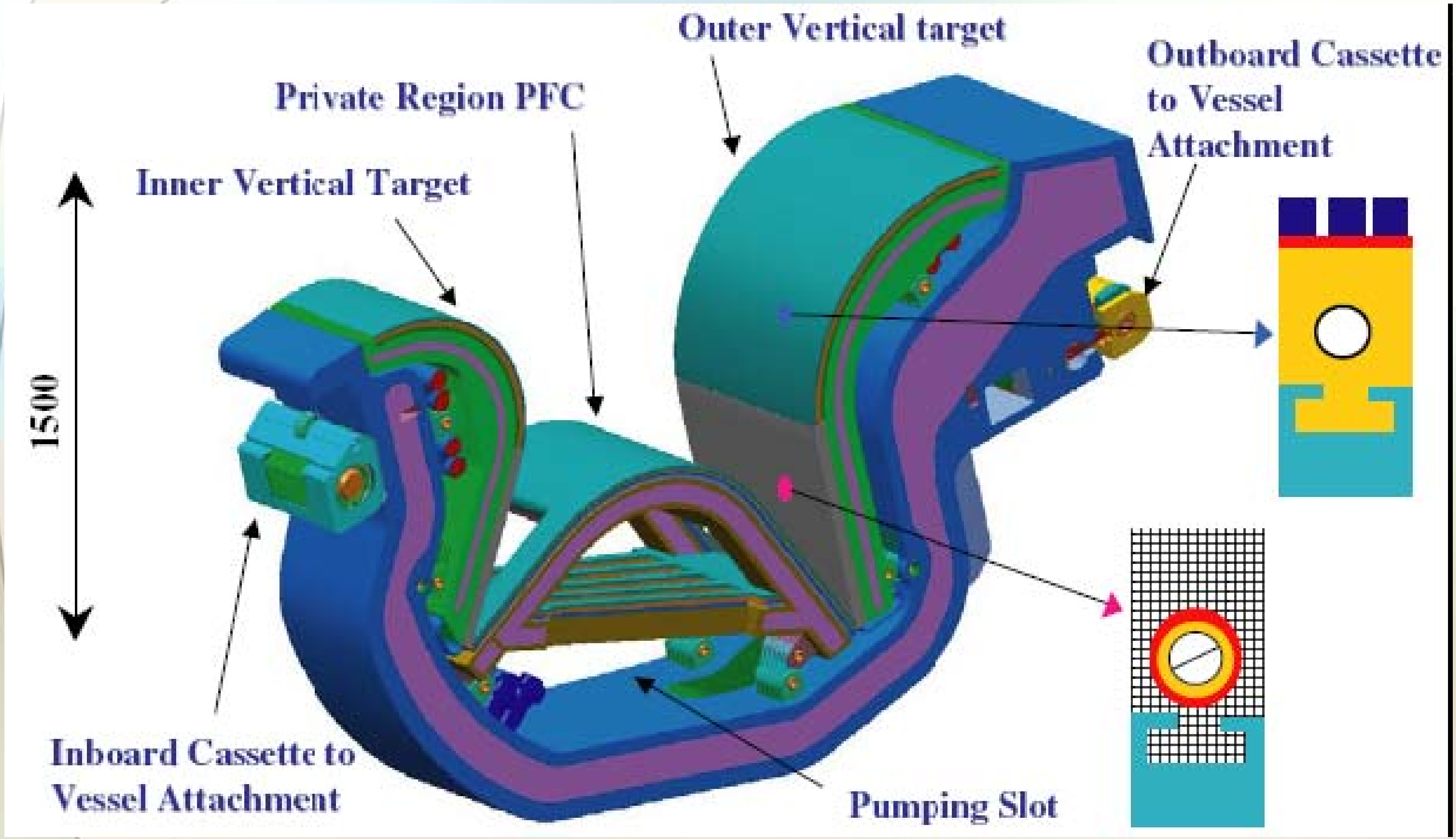
Physics (stability, confinement, etc) and Radial Build Constraints determine operating space.




*POPCON for SABR reference design parameters (I =7.2MA)*

There is a broad operating parameter range for achieving the nominal design objective of $P_{fus}$ = 150-200 MW.

# Heat Removal from Fusion Neutron Source




-- Design for 500 MWt plasma

-- ITER designs adapted for Na

-- 50%/50% first wall/divertor

-- FLUENT/GAMBIT calculations

# Heat Removal from Fusion Neutron Source (cont.)

**First Wall**

- Be coated ODS (3.5 cm plasma to Na)
- Design peak heat flux 0.5-1.0 MW/m$^2$
- Nominal peak heat flux 0.25 MW/m$^2$
- Temperature range 600-700 C (1200 C max)
- $T_{in}$ = 293 C, $T_{out}$ = 600 C
- Coolant mass flow 0.06 kg/s
- 4x10$^{22}$ (n/cm$^2$)/FPY = 33 dpa/FPY
- Radiation damage life 200 dpa =

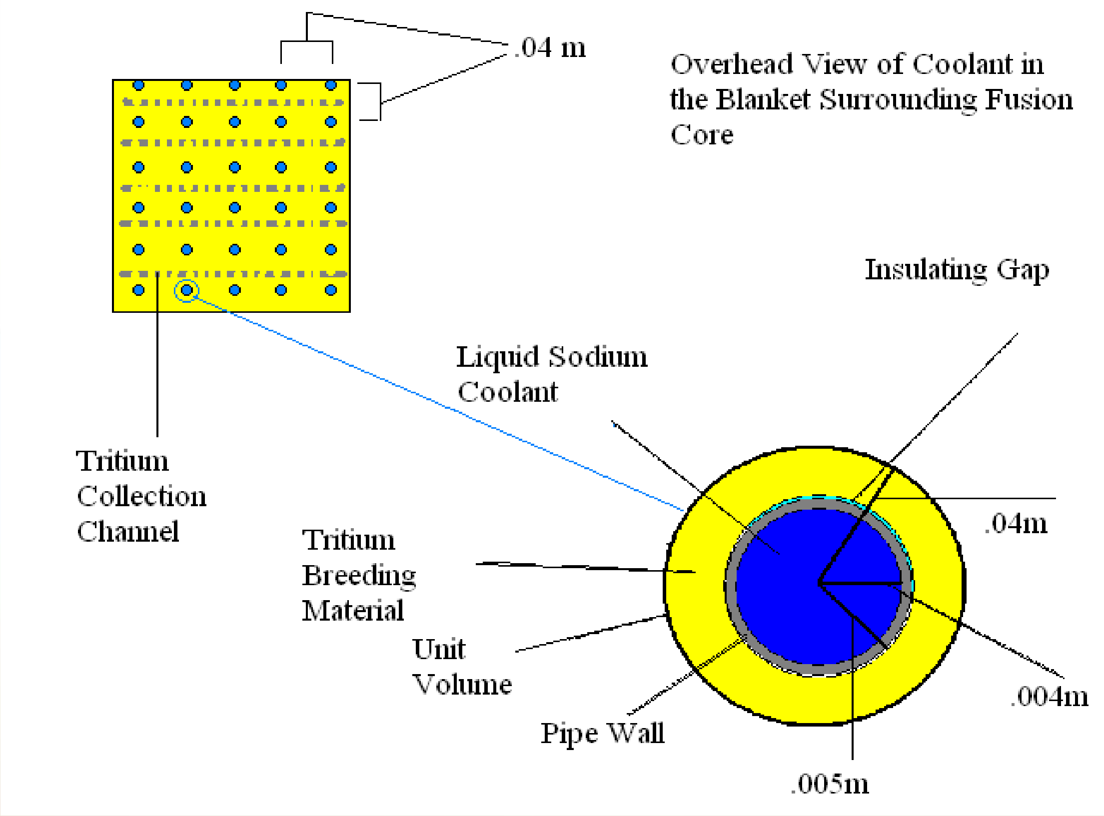
  8.1 yr @ 500 MW & 75%

  20.2 yr @ 200 MW & 75%

**Divertor Module**

- Cubic W (10mm) bonded to CuCrZr
- Na in same ITER coolant channels
- Design Peak heat flux 1 – 8 MW/m$^2$ (ITER < 10 MW/m$^2$)
- $T_{in}$ = 293 C, $T_{out}$ = 756 C
- Coolant mass flow 0.09 kg/s
- Lifetime - erosion

# Li4SiO4 Tritium Breeding Blanket

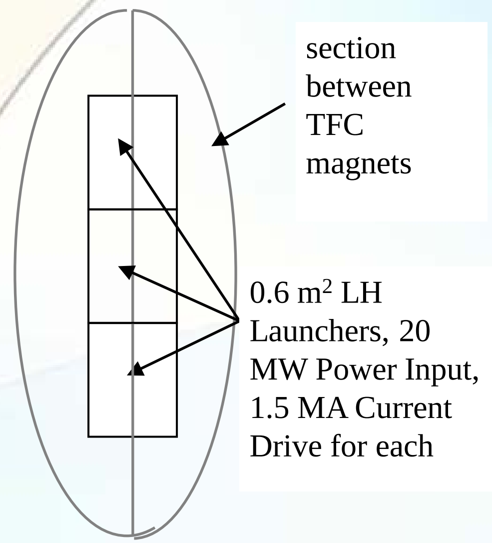



15 cm Thick Blanket Around Plasma (Natural LI) and Reactor Core (90% Enriched LI) Achieves TBR = 1.16.

NA-Cooled to Operate in the Temperature Window 420-640 C.
Online Tritium Removal by He Purge Gas System.

Dynamic Tritium Inventory Calculations for 750 d Burn Cycle Indicated More Than Adequate Tritium Production.

# SABR Lower Hybrid Heating & CD System

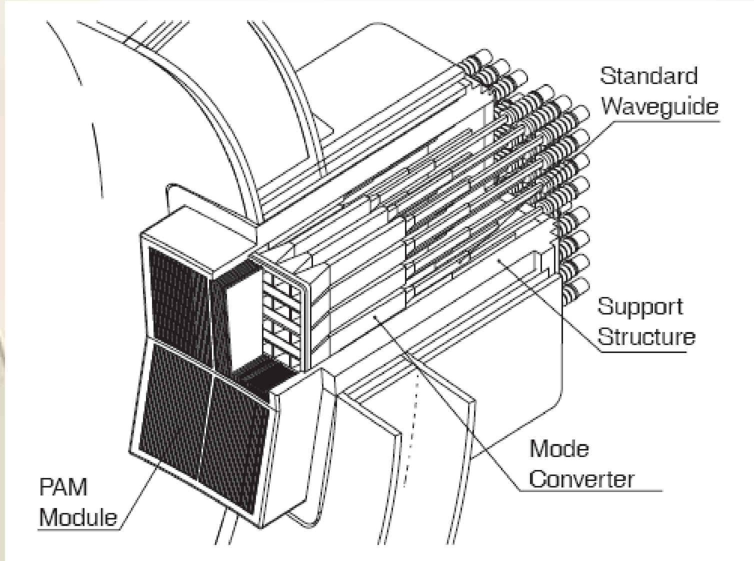



2 SETS of 3 PORTS @ 180°
20 MW Per 0.6 m² PORT

H&CD SYSTEM PROPERTIES

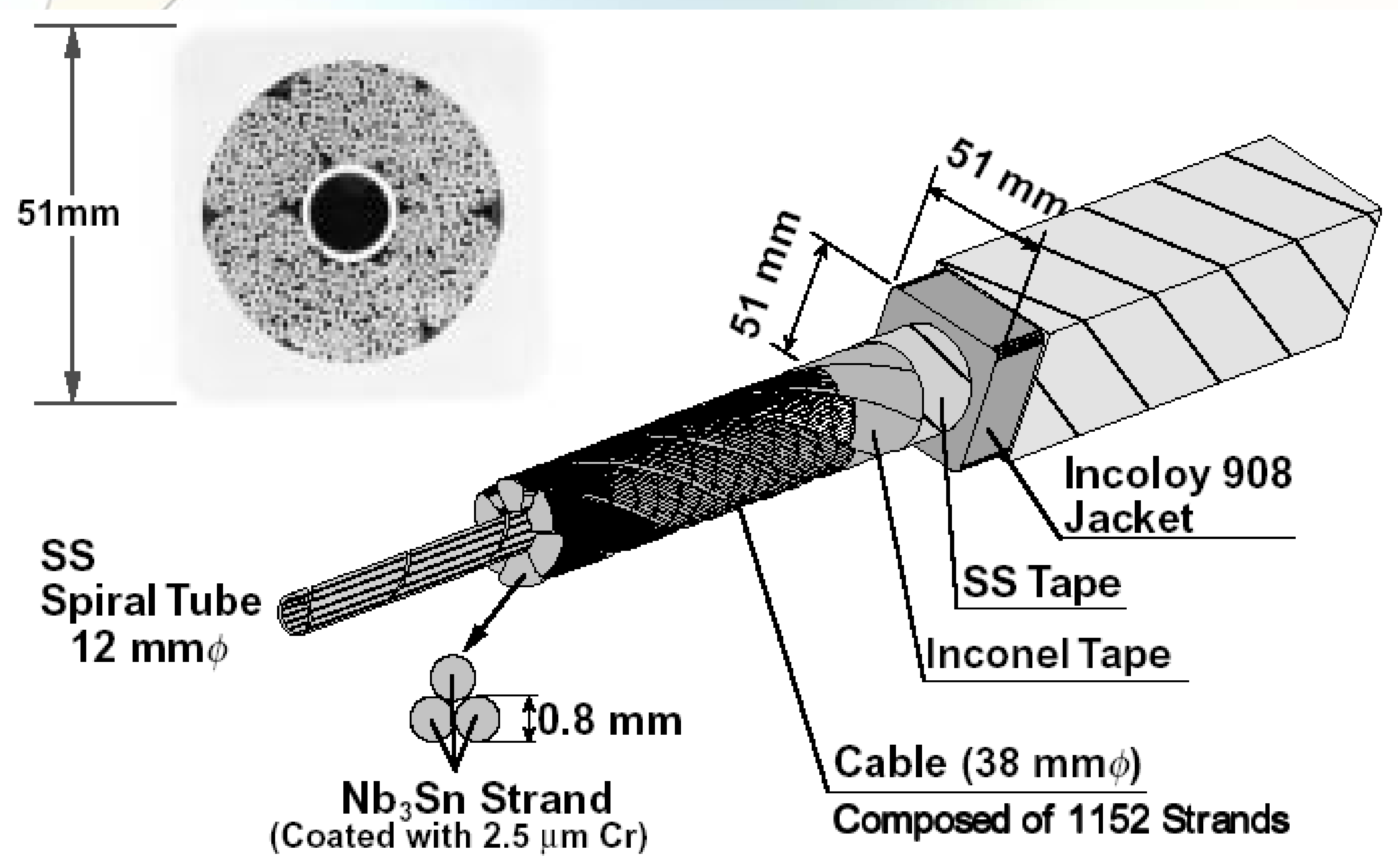
| Property | SABR | ITER |
|---|---|---|
| $I_{bs}$ (MA) | 2.5 | ~7.5 |
| $f_{bs}$ (%) | 25 | ~50 |
| $I_p$ (MA) | 10 | 15 |
| $P_{aux}$(MW) | 100 | 110 |
| $P_{tot}$(MW) | 120 | 130 |
| # Port Plugs | 6 | 10* |
| PD ($MW/m^2$) | 33 | 9.2 ** |

** 4 equatorial, 3 upper, 3 NBI, ** ICRH power density




Used ITER LH Launcher Design

# SABR S/C Magnet Design Adapted from ITER




Detailed cross section of CS cable-in-conduit conductor

# SABR S/C Magnet Design Adapted from ITER (cont.)

### Central Solenoid Parameters

| CS Conductor Parameters | |
|---|---|
| Superconductor | $Nb_3Sn$ |
| Operating Current (kA) IM/EOB | 41.8 / 46.0 |
| Nominal B Field (T) IM/EOB | 12.4 / 13.5 |
| Flux Core Radius, $R_{fc}$ (m) | 0.66 |
| CS Coil thickness, $\Delta_{OH}$ (m) | 0.70 |
| $VS_{start}$ (V-s) design/needed | 87.7/82.5 |
| $\sigma_{CS}$ (MPa) IM/EOB | 194. / 230. |
| $\sigma_{max}$ (MPa) (ITER) | 430. |
| $f_{struct}$ | 0.564 |

### TF coil parameters

| Parameters | |
|---|---|
| Radial Thickness, $\Delta_{TF}$ (m) | 0.43 |
| Number of TF Coils, $N_{TF}$ | 16 |
| Bore h x w (m) | 8.4x5.4 |
| Current per Coil (MA), $I_{TF}$ | 6.4 |
| Number of Conductors per Coil (turns), $N_{cond}$ | 120 |
| Conductor Diameter (mm), $d_{TF}$ | 43.4 |
| Superconductor Material | $Nb_3Sn$ |
| $I_{cond}$, Current per Conductor (kA) | 68 |
| $B_{max}$, Maximum Magnetic Field (T) | 11.8 |
| Radius of Maximum Field (m) | 2.21 |
| $B_0$, Magnetic Field on Axis (T) | 6.29 |

# SHIELD

Shield Layers and Compositions

| Layer | Material | Thickness | Density |
|---|---|---|---|
| Reflector | ODS Steel (12YWT) | 16 cm | 7.8 g/cm$^3$ |
| Cooling CH A | Sodium-22 | 1cm | 0.927 g/cm$^3$ |
| 1 | Tungsten HA (SDD185) | 12 cm | 18.25 g/cm$^3$ |
| Cooling CH B | Sodium-22 | 1cm | 0.927 g/cm$^3$ |
| 2 | Tungsten HA (SDD185) | 10 cm | 18.25 g/cm$^3$ |
| Cooling CH C | Sodium-22 | 1cm | 0.927 g/cm$^3$ |
| 3 | Boron Carbide ($B_4C$) | 12 cm | 2.52 g/cm$^3$ |
| Cooling CH D | Sodium-22 | 1cm | 0.927 g/cm$^3$ |
| 4 | Tungsten HA (SDD185 | 10 cm | 18.25 g/cm$^3$ |

SHIELD DESIGNED TO PROTECT MAGNETS

MAX FAST NEUTRON FLUENCE TO S/C = $10^{19}$ n/cm$^2$

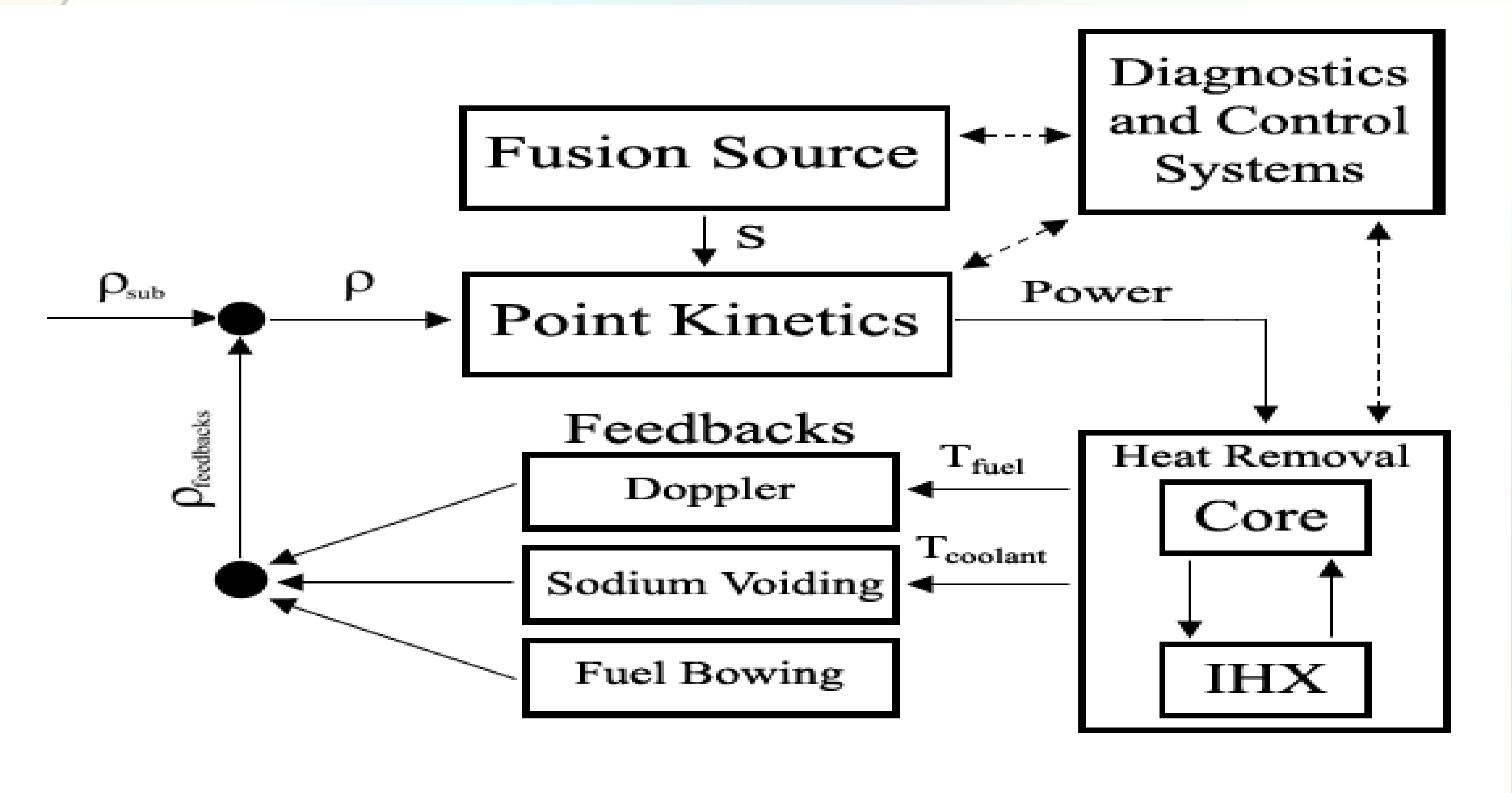
MAX ABSORBED DOSE TO INSULATOR $10^9$ /$10^{10}$ RADS (ORG/CER)

CALCULATED IRRADIATION IN 40 YEARS AT $P_{FUS}$ = 500 MW AND 75% AVAILABILITY

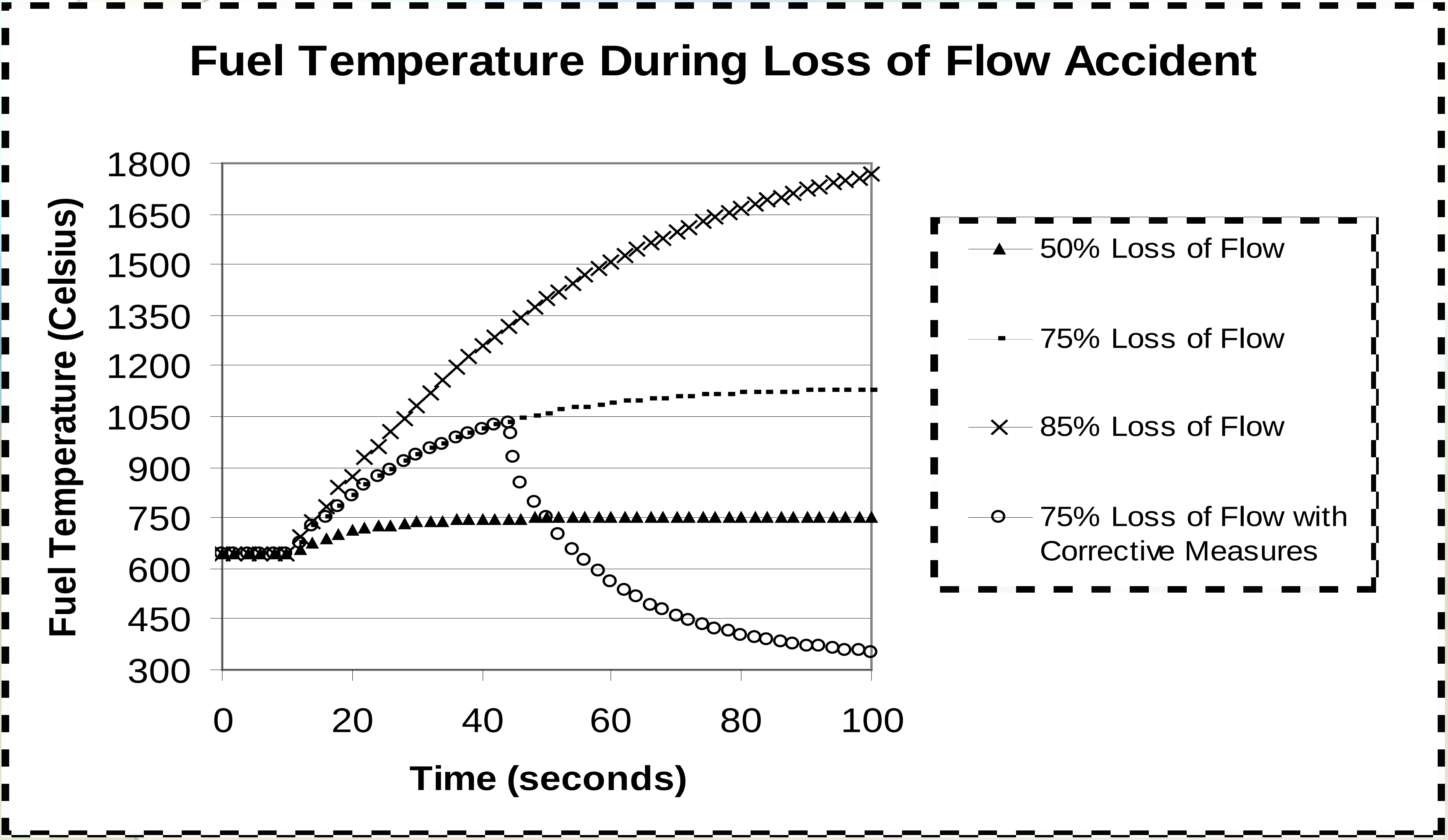
FAST NEUTRON FLUENCE TO S/C = $6.9 \times 10^{18}$ n/cm$^2$

ABSORBED DOSE TO INSULATOR = $7.2 \times 10^7$ RADS

# Dynamic Analysis of Loss of Flow



# Dynamic Analysis of Loss of Flow (cont.)




THE SUBCRITICAL REACTIVITY MARGIN PROVIDES 10'S SECONDS FOR CORRECTIVE CONTROL ACTION.

# SUMMARY & CONCLUSIONS

- The GNEP concept of a pure TRU-fuel burner reactor is challenging because of large burnup reactivity decrement, small delayed neutron fraction and small Doppler coefficient in the absence of U238.
- SABR, a subcritical, TRU-ZR fuel, NA-cooled fast reactor design concept has been developed, based on current nuclear technology R&D.
- A Tokamak DT fusion neutron source, based on ITER physics and technology, has been shown to be adequate to support the subcritical reactor.
- Fuel residence time in SABR is limited by clad failure at 200 dpa to 8.4 FPY.
- a 4-batch, 8.2 FPY fuel cycle burns 25% of the TRU fuel in SABR, with $k_{eff}$ =0.83 and $P_{fus}$ = 164 MWt at EOC.
- > 90% burnup can be achieved in SABR by repeated recycling, with reprocessing.
- Dynamic analysis of loss-of-flow accident indicates that the SABR sub-criticality margin provides 10's of seconds to initiate control action.