A SUPPLEMENTAL FUSION-FISSION HYBRID PATH TO FUSION POWER DEVELOPMENT

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By

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For the Georgia Tech SABR Design Team

Outline

- Fusion R&D for electrical power production.
- What are fusion-fission hybrids (FFHs) & what is their raison d'etre?
- What is the time scale for developing the fusion neutron source for a FFH?
- The SABR conceptual design for a FFH burner reactor.
- SABR transmutation fuel cycle studies.
- SABR (preliminary) dynamic safety studies.
- R&D requirements for developing fusion power, with and without FFH.
- Schedules for developing fusion power, with and without FFH.
- Some technical issues with combining fusion and fission.
- Three recommendations.

MAGNETIC FUSION R&D LEADING TO A COMMERCIAL POWER REACTOR

Assessment of R&D Needed for Fusion Power Production 4 Levels of Performance Questions

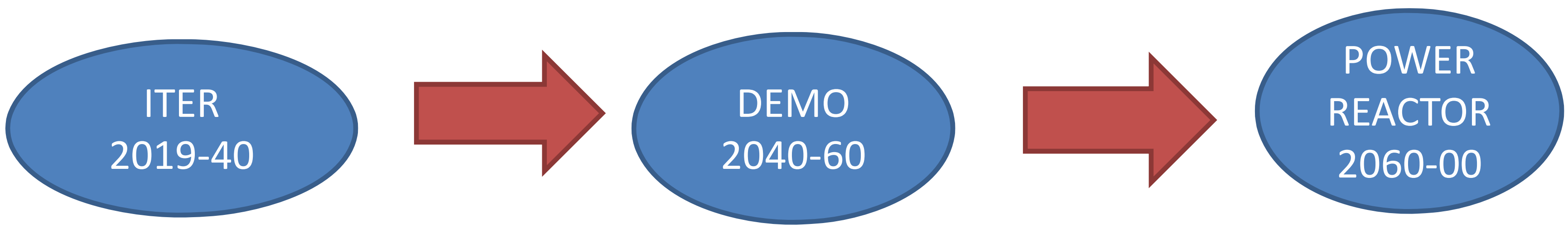
1. What must be done to *achieve the required level of individual physics and technology performance parameters*? (physics and technology experiments)
2. What further must be done to *achieve* the required levels of all the different individual physics and technology performance parameters *simultaneously*? (component test facilities & experimental reactors, e.g. ITER)
3. What further must be done to *achieve* the required level of all the individual physics and technology performance parameters simultaneously and *reliably over long periods of continuous operation?* (advanced physics experiments, component test facilities & demonstration reactors)
4. What further must be done to *demonstrate the economic competitiveness* of the power that will be produced?(prototype reactors)

Status of Magnetic Fusion R&D

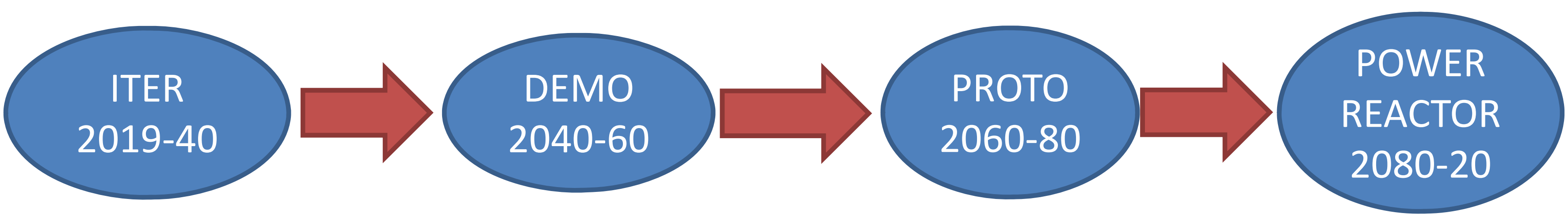
1. The tokamak is the leading plasma physics confinement concept.
   - ~100 tokamaks worldwide since 1957.
   - Physics performance parameters achieved at or near lower limit of reactor relevance.
   - Large, world-wide physics & technology programs supporting ITER (initial operation 2019).
   - ITER will achieve reactor-relevant physics and technology parameters simultaneously, produce 500 MWth and investigate very long-pulse operation.
2. Many other confinement concepts (e.g. mirror, bumpy torus) have fallen by the wayside or remain on the backburner.
3. A few other confinement concepts (e.g. stellarator, spherical torus) have some attractive features, which justifies their continued development. However, the performance parameters are at least 1-2 orders of magnitude below what is required for a power reactor, and at least 25 years would be required to advance any other concept to the present tokamak level.
4. Plasma support technology (SC magnets, heating, fueling, vacuum, etc.) for the tokamak is at the reactor-relevant level, due to the large ITER R&D effort.
5. Fusion nuclear technology (tritium production, recovery and processing) has had a low priority within fusion R&D. ITER will test fusion tritium breeding blanket modules.
6. The continued lack of a radiation damage resistant structural material would greatly complicate fusion experiments beyond the ITER level (e.g. DEMO) and might make a fusion reactor uneconomical, if not altogether impractical.

An Unofficial Fusion Development Schedule

Canonical



More Likely?



THE FUSION-FISSION HYBRID REACTOR

- What is it?
- Mission?
- Rationale?
- Choice of technologies?

The Fusion-Fission Hybrid

- *What is it?*

  A Fusion-Fission Hybrid (FFH) is a sub-critical fission reactor with a variable strength fusion neutron source.

- *Mission?*

  Supporting the sustainable expansion of nuclear power in the USA and worldwide by helping to close the nuclear fuel cycle.

SUSTAINABLE NUCLEAR POWER EXPANSION

- The present ‘once-through’ LWR fuel cycle utilizes < 1% of the potential uranium fuel resource and leaves a substantial amount of long-term radioactive transuranics (TRU) in the spent nuclear fuel. The TRU produced by the present USA LWR fleet will require a new Yucca Mountain HLWR every 30 years, and a significant expansion of nuclear power would require new HLWRs even more frequently.
- A significant expansion of nuclear power worldwide would deplete the known uranium supply within 50 years at the present <1% utilization.
- Fast ‘burner’ reactors can in principle solve the spent fuel accumulation problem by fissioning the transuranics in spent nuclear fuel, thus reducing the number of HLWRs needed to store them, while at the same time utilizing more of the uranium energy content.
- Fast ‘breeder’ reactors can in principle solve the uranium fuel supply issue by transmuting U238 into fissionable (in LWRs and fast reactors) transuranics (plutonium and the higher ‘minor actinides’), leading to the utilization of >90% of the potential energy content of uranium.
- Fast reactors can not be fueled entirely with transuranics because the reactivity safety margin to prompt critical would be too small, and the requirement to remain critical requires periodic removal and reprocessing of the fuel. Operating fast reactors subcritical with a variable-strength fusion neutron source can solve both of these problems, resulting in fewer fast burner reactors and fewer HLWRs.

Rationale for FFH Fast Burner Reactors

- Fast Burner reactors could dramatically *reduce* the required number of high-level waste *repositories* by fissioning the transuranics in LWR SNF.

- The potential advantages of FFH burner reactors over critical burner reactors are:

  1) *fewer reprocessing steps*, hence fewer reprocessing facilities and HLWR repositories[a]—no criticality constraint, so the TRU fuel can remain in the FFH for deeper burnup to the radiation damage limit.

  2) *larger LWR support ratio*---FFH can be fueled with 100% TRU, since sub-criticality provides a large reactivity safety margin to prompt critical, so fewer burner reactors would be needed.

[a] separation of transuranics from fission products is not perfect, and a small fraction of the TRU will go with the fission products to the HLWR *on each reprocessing*.

Choice of Fission Technologies for FFH Fast Burner Reactor

- Sodium-cooled fast reactor is the *most developed* burner reactor *technology*, and most of the world-wide fast reactor R&D is being devoted to it (deploy 15-20yr).
  1. The metal-fuel fast reactor (IFR) and associated pyroprocessing separation and actinide fuel fabrication technologies are the most highly developed in the USA. The IFR is *passively safe* against LOCA & LOHSA . The IFR fuel cycle is *proliferation resistant*.
  2. The sodium-cooled, oxide fuel FR with aqueous separation technologies are highly developed in France, Russia, Japan and the USA.
- Gas-cooled fast reactor is a much less developed backup technology.
  1. With oxide fuel and aqueous reprocessing.
  2. With TRISO fuel (burn and bury). Radiation damage would limit TRISO in fast flux, and it is probably not possible to reprocess.
- Other liquid metal coolants, Pb, Pb-Li, Li.
- Molten salt fuel would simplify refueling, but there are issues. (Molten salt coolant only?)

Choice of Fusion Technologies for the FFH Fast Burner Reactor

- The tokamak is the *most developed* fusion neutron source *technology*, most of the world-wide fusion physics and technology R&D is being devoted to it, and ITER will demonstrate much of the physics and technology performance needed for a FFH (deploy 20-25 yr).

- Other magnetic confinement concepts promise some advantages relative to the tokamak, but their choice for a FFH would require a massive redirection of the fusion R&D program (not presently justified by their performance).
    1. Stellarator, spherical torus, etc. are at least 25 years behind the tokamak in physics and technology (deploy 40-50 yr).
    2. Mirror could probably be deployed in 20-25 years, but would require redirection of the fusion R&D program into a dead-end technology that would not lead to a power reactor.

SABR FFH Burner Reactor Design Concept

SABR FFH DESIGN APPROACH

1. Use insofar as possible the physics and technologies, and adapt the designs, that have been developed for the Integral Fast Reactor (IFR) and the International Thermonuclear Experimental Reactor (ITER).
    - The successful operation of an IFR and associated fuel pyroprocessing and fabrication technologies will prototype the fission physics and technologies.
    - The successful operation of ITER and its blanket test program will prototype the fusion physics and technologies.
2. Be conservative insofar as possible.
    - Modest plasma, power density, etc. performance parameters.
    - Adapt IFR and ITER component designs, and use IFR and ITER design guidelines on stress margins, structure fractions, etc.
    - Use conservative 99% actinide—fission product separation efficiency.

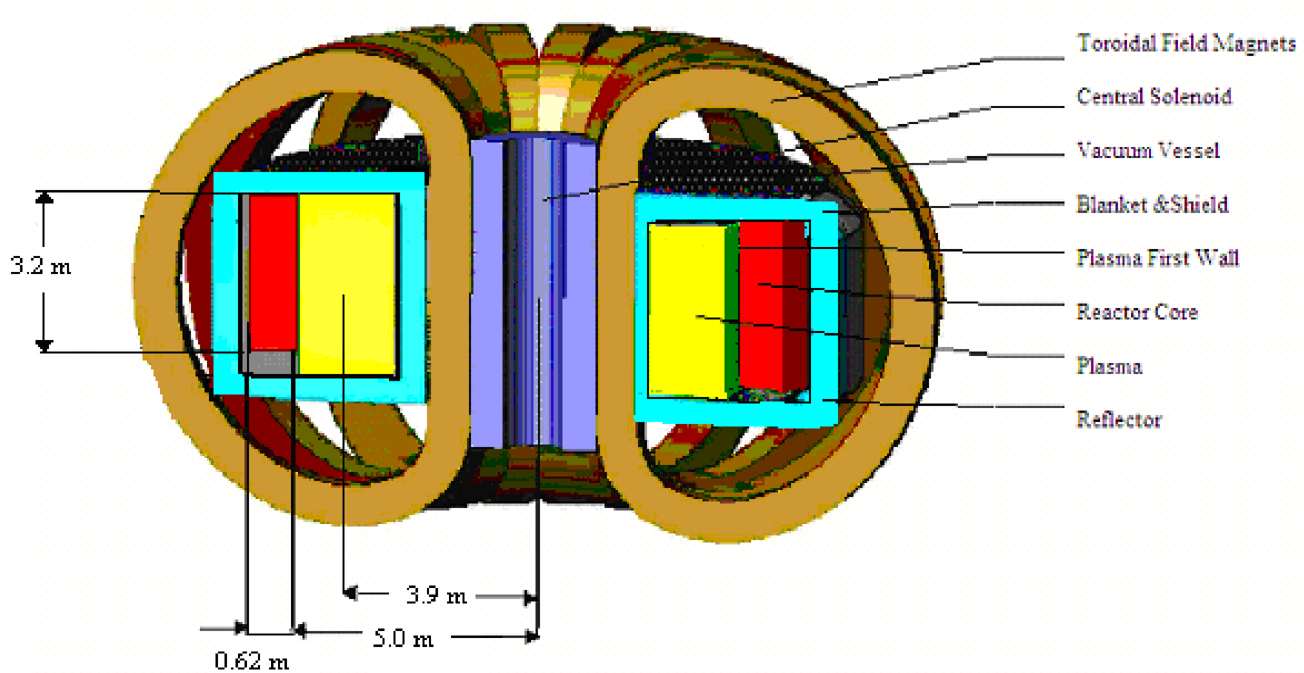
SUB-CRITICAL ADVANCED BURNER REACTOR (SABR)

ANNULAR FAST REACTOR (3000 MWth)

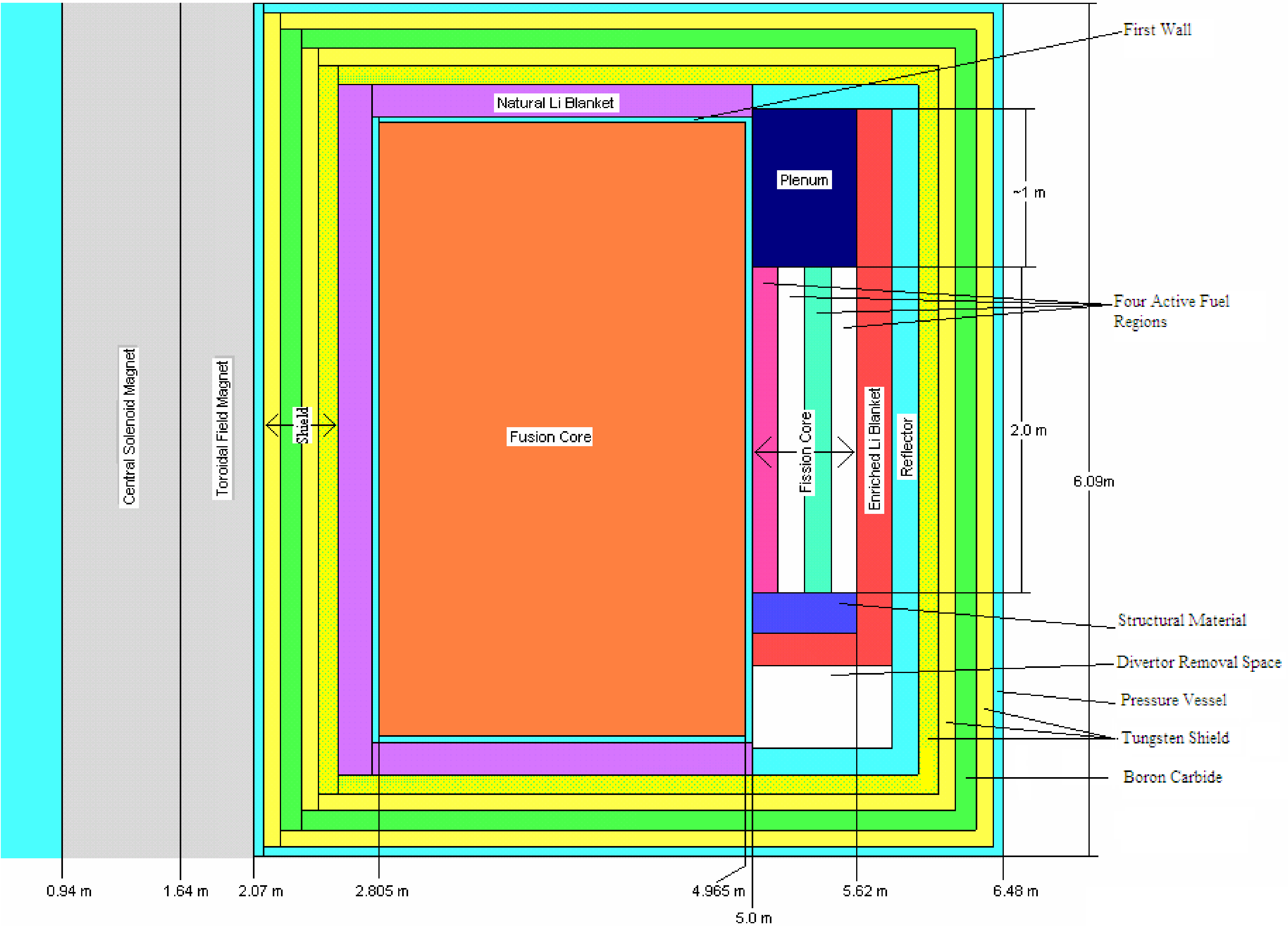
- Fuel—TRU from spent nuclear fuel. TRU-Zr metal being developed by ANL.
- Sodium cooled, loop-type fast reactor.
- Based on fast reactor designs being developed by ANL in Nuclear Program.

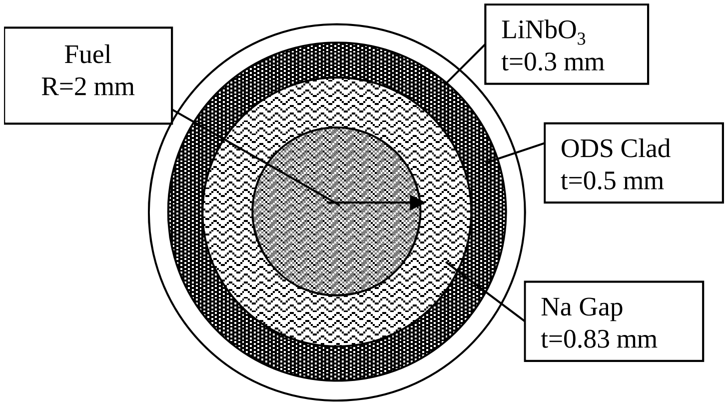
TOKAMAK D-T FUSION NEUTRON SOURCE (200-500 MWth)

- Based on ITER plasma physics and fusion technology.
- Tritium self-sufficient (Li_4SiO_4).
- Sodium cooled.

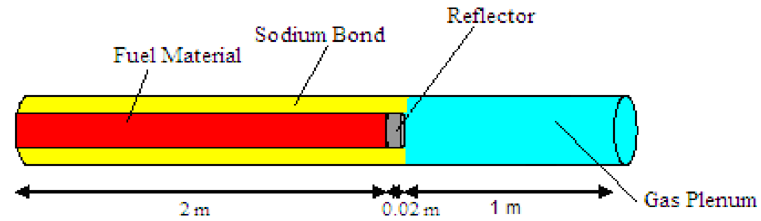


R-Z cross section SABR calculation model





FUEL

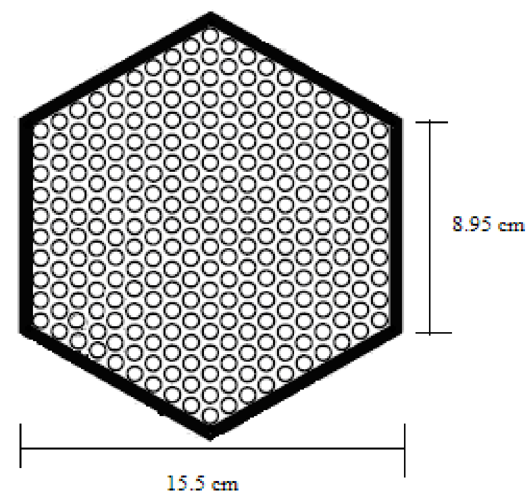


Axial View of Fuel Pin

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

Core Thermal Analysis

Core Thermal and Heat Removal Parameters

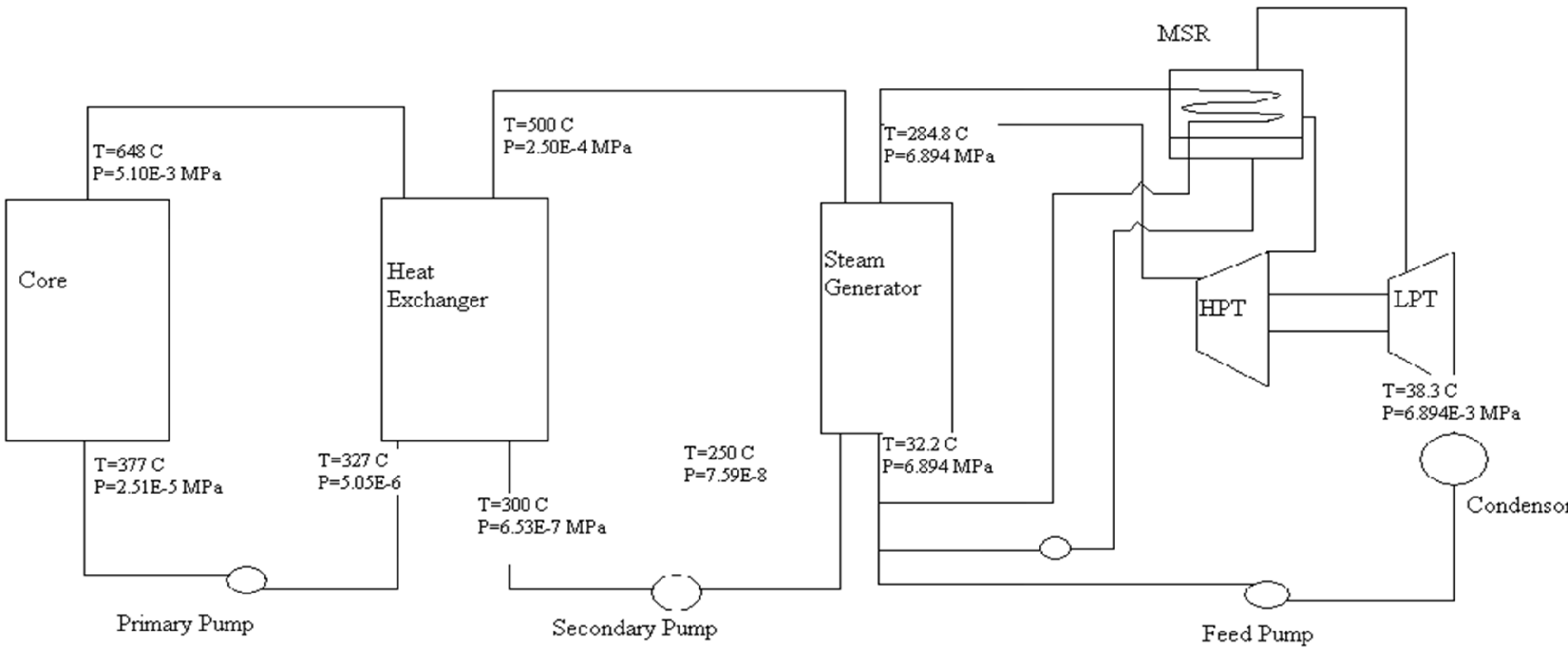
Power Density	73 MW/m^3
Linear Pin Power	6 kW/m
Coolant T_{in}	377 °C
Coolant T_{out}	650 °C
Min. Centerline Temp	442 °C
Max Centerline Temp	715 °C
Mass Flow Rate($\dot{m}$)	8700 Kg/s
Coolant Velocity(v)	1.4 m/s
Total Pumping Power	454 KW*

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

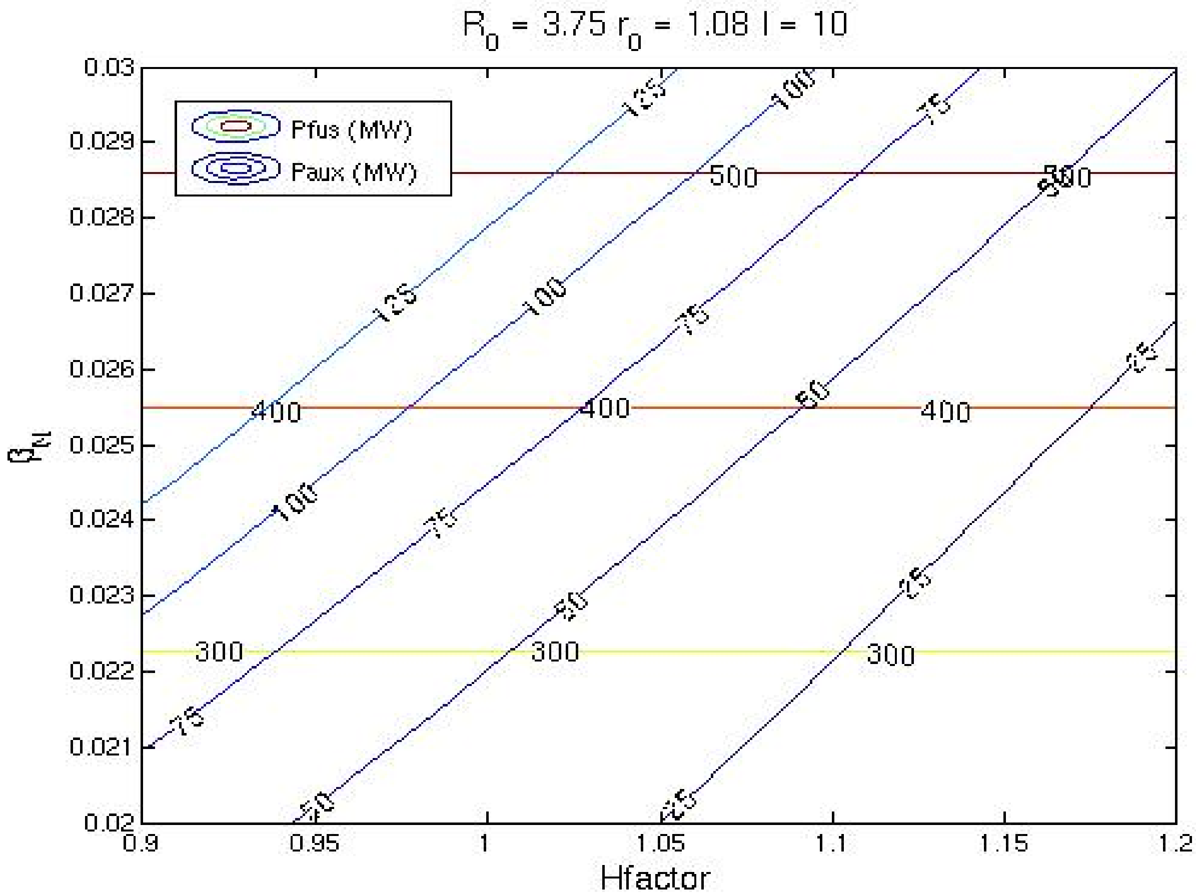
- **Primary and intermediate Na loops**
- **Secondary water Rankine cycle**



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 %**

Fusion Neutron Source

400-500 MW Operation Space at 10 MA

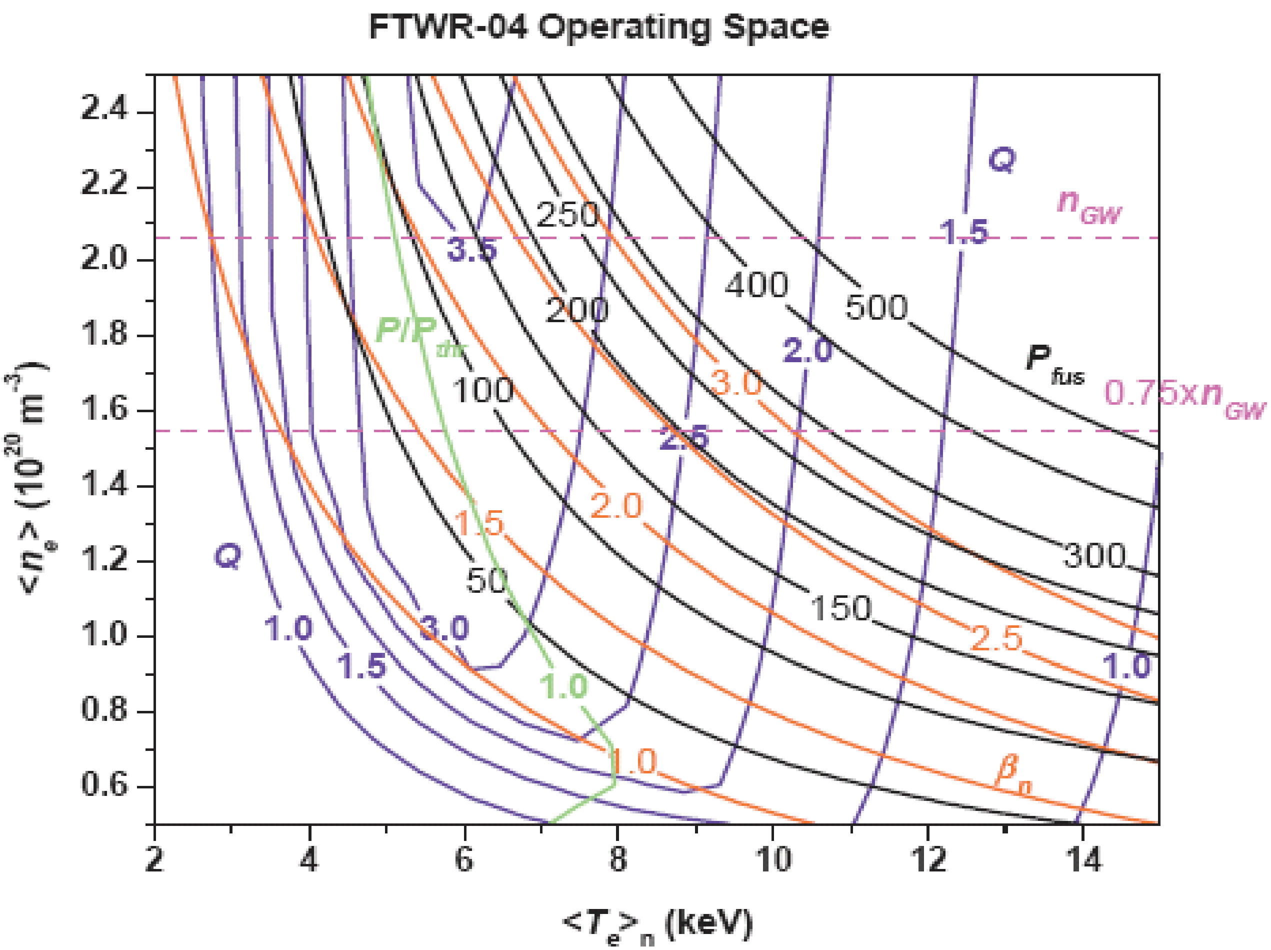


Operational space of SABR at 10 MA
(*Horizontal lines indicate* Pf_{us} *and slanted lines* P_{aux})

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.

Neutron Source Design Parameters

SABR TOKAMAK NEUTRON SOURCE PARAMETERS

Parameter	**SABR Low power**	**SABR High power**	**ITER**	**Pure Fusion Electric ARIES-AT**
Current, I (MA)	8.3	10.0	15.0	13.0
P_{fus} (MW)	180	500	400	3000
Major radius, R (m)	3.75	3.75	6.2	5.2
Magnetic field, B (T)	5.7	5.7	5.3	5.8
Confinement $H_{IPB98}(y,2)$	1.0	1.06	1.0	2.0(?)
Normalized beta, β_N	2.0	2.85	1.8	5.4
Energy Mult, Q_p	3	5	5-10	>30
Htg&CD Power, MW	100	100	110	35
Neutron Γ_n (MW/m^2)	0.6	1.8	0.5	4.9
CD ηcd/fbs	.61/.31	.58/.26	?/?	?/.91
Availability (%)	75	75	25(4)	>90

Heat Removal from Fusion Neutron Source

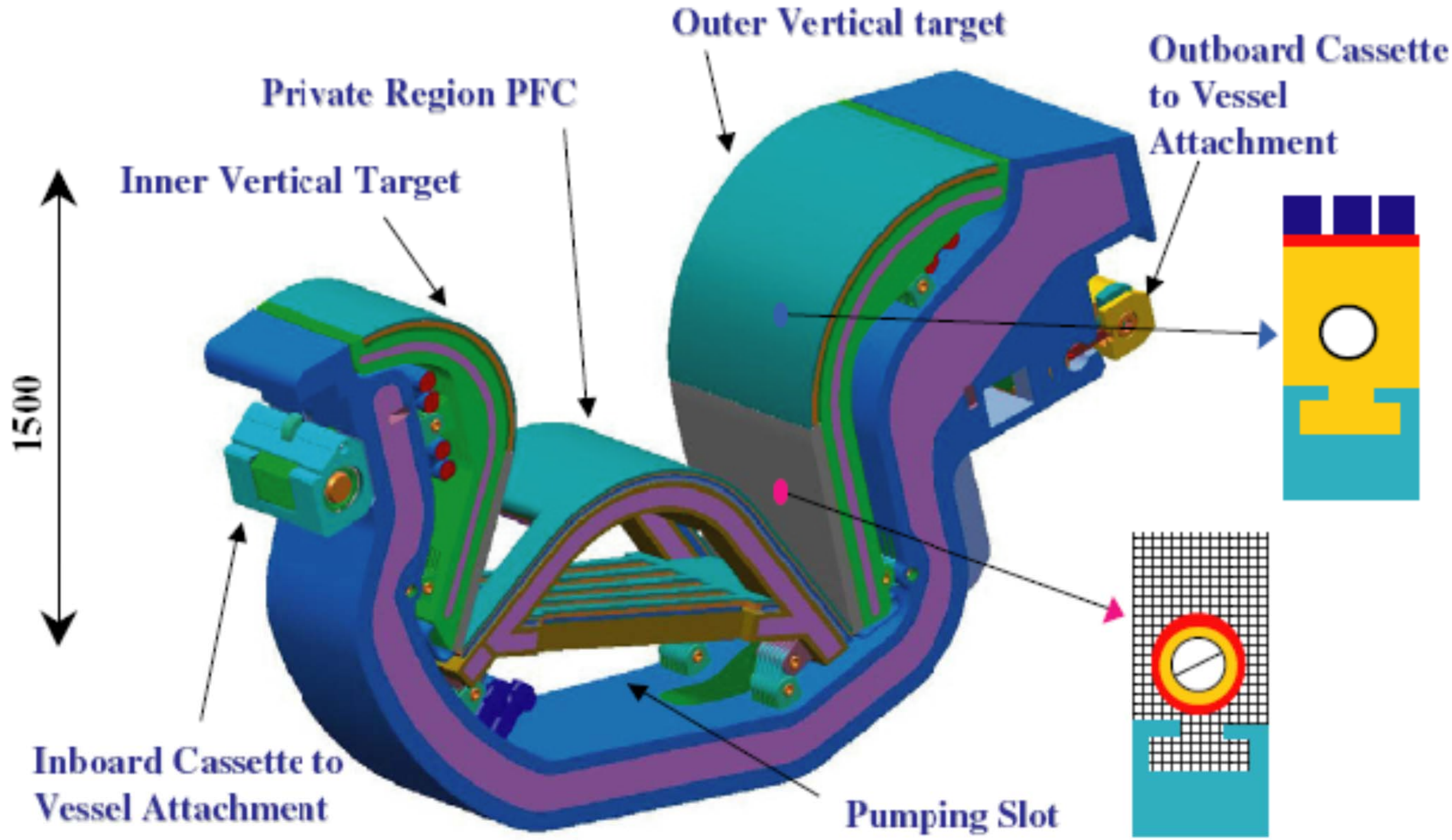
<u>First Wall</u>

- 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%

<u>Divertor Module</u>

- 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

Heat Removal from Fusion Neutron Source



-- Design for 500 MWt plasma

-- ITER designs adapted for Na

-- 50%/50% first wall/divertor

-- FLUENT/GAMBIT calculations

SABR S/C Magnet Design Adapted from ITER

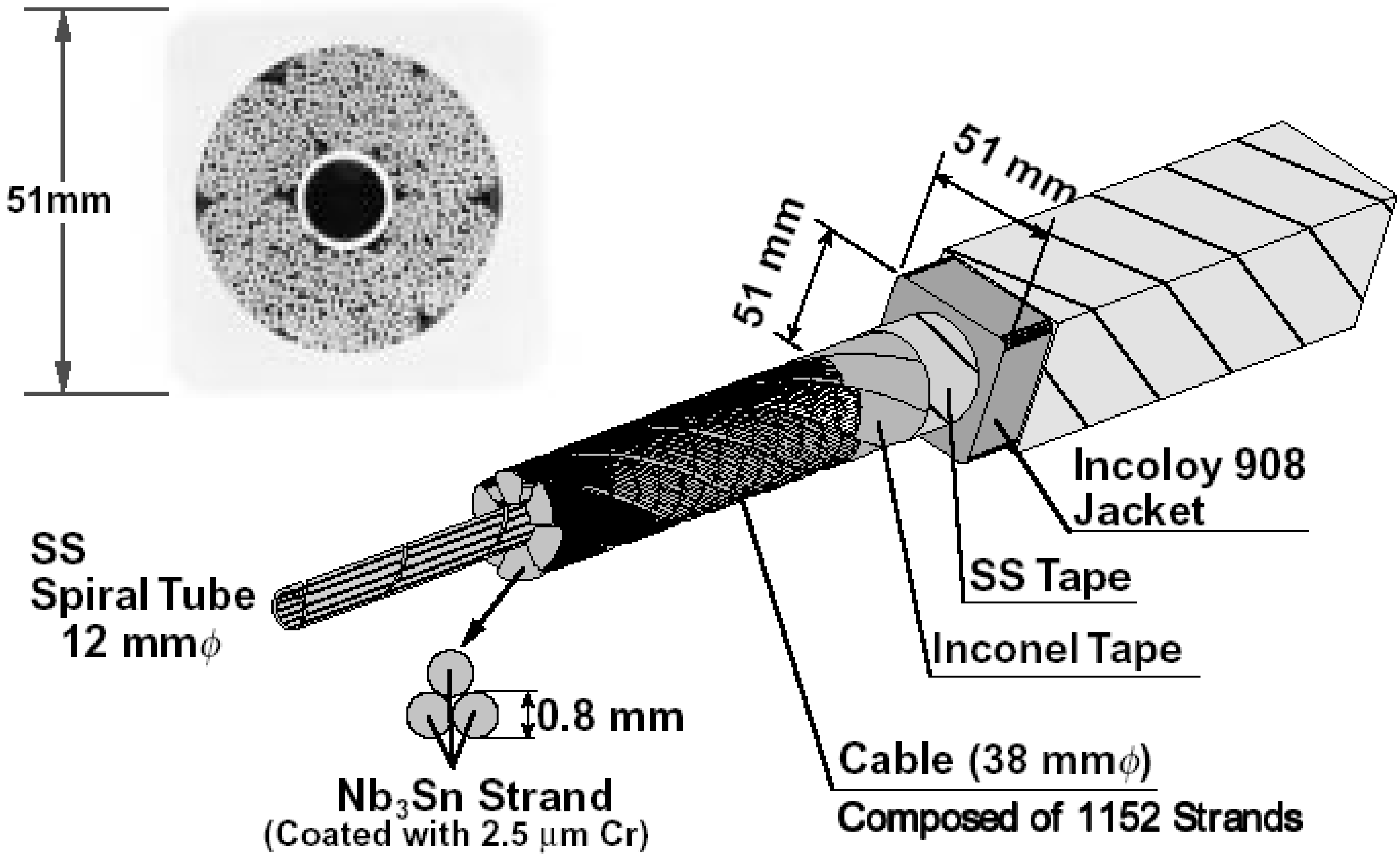
Central Solenoid Parameters

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, Δ_{OH} (m)	0.70
VS_{start} (V-s) design/needed	87.7/82.5
σ_{CS} (MPa) IM/EOB	194. / 230.
σ_{max} (MPa) (ITER)	430.
f_{struct}	0.564

TF coil parameters

Parameters	
Radial Thickness, Δ_{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

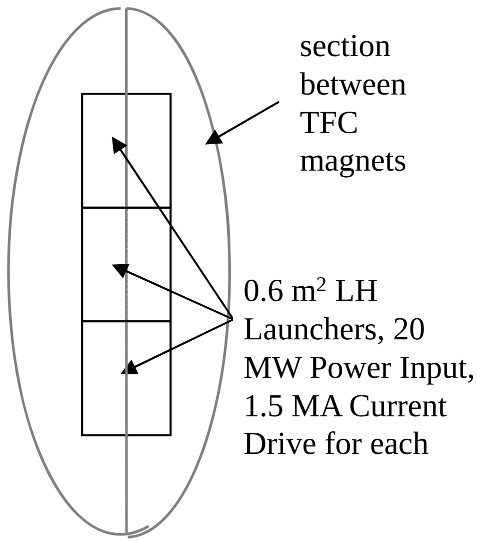
SABR S/C Magnet Design Adapted from ITER



Detailed cross section of CS cable-in-conduit conductor

SABR Lower Hybrid Heating & CD System

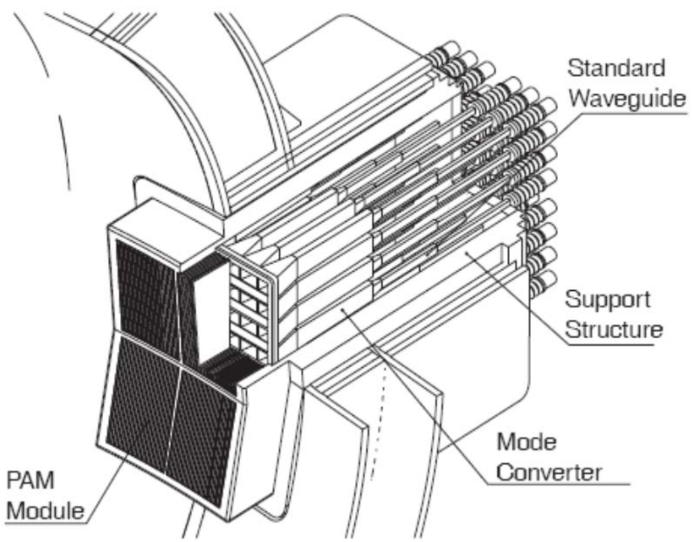
2 SETS of 3 PORTS @ 180°
20 MW Per 0.6 m² PORT



H&CD SYSTEM PROPERTIES

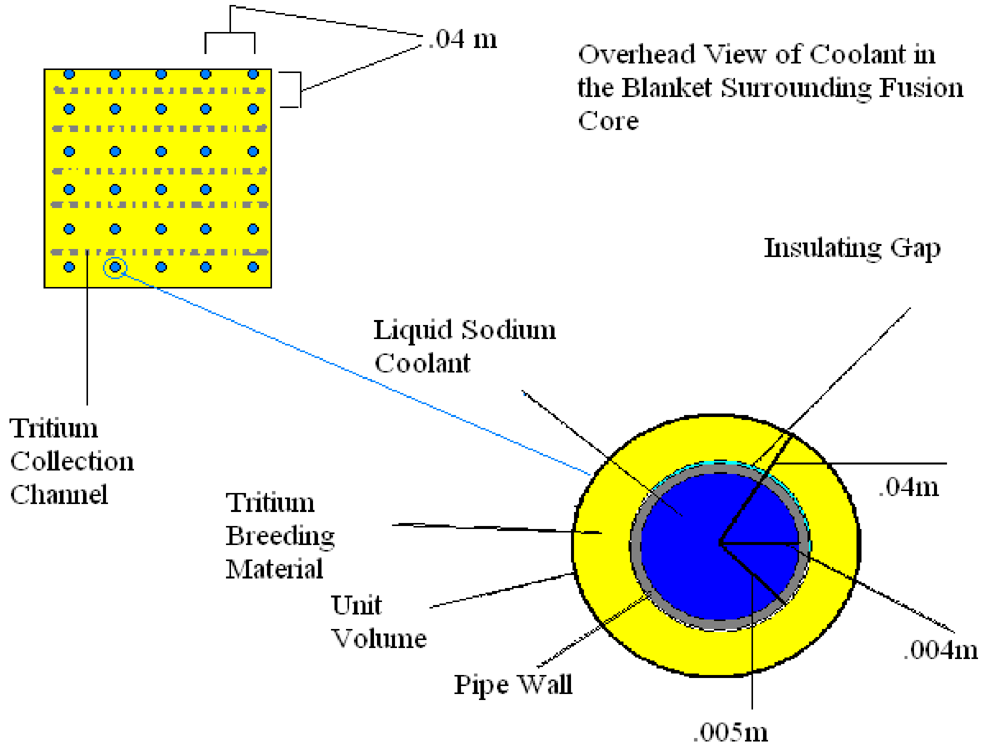
Property	SABR	ITER
I_{bs} (MA)	2.5	~7.5
f_{bs} (%)	25	~50
I_p (MA)	10	15
P_{aux}(MW)	100	110
P_{tot}(MW)	120	130
# Port Plugs	6	10*
PD (MW/m²)	33	9.2 **

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



Used ITER LH Launcher Design

Li_4SiO_4 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 ERANOS Tritium Inventory Calculations for 700 d Burn Cycle, 60 d Refueling Indicated More Than Adequate Tritium Production.

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 = 6.9×10^{18} n/cm^2

ABSORBED DOSE TO INSULATOR = 7.2×10^7 RADS

What are the TECHNICAL ISSUES?

1. Fusion Physics
   - Current drive efficiency and bootstrap current. Plasma heating with LHR.
   - Disruption avoidance/mitigation.
2. Fusion Technology
   - Tritium retention.
   - Tritium breeding and recovery.
   - A 100-200 dpa structural material (ODS).
3. Fission Technology
   - MHD effects on Na flow in magnetic field. (molten salt coolant backup?)
   - Refueling in tokamak geometry.

SABR FUEL CYCLE STUDIES

2 BURNER FUEL CYCLES

1. TRU BURNER—all TRU (ANL 65.8%Pu & 34.2% MA) from LWR SNF fabricated into fast burner reactor fuel.

2. MA BURNER---some Pu saved and remaining MA-rich TRU (EU 45.7%Pu & 54.3%MA) fabricated into fast burner reactor fuel.

- Burner reactor fuel recycled.
- 4-batch fuel cycles, out-to-in shuffling.
- Fuel residence time limited by 200dpa radiation damage limit to ODS clad.
- 1% separation efficiency assumed.

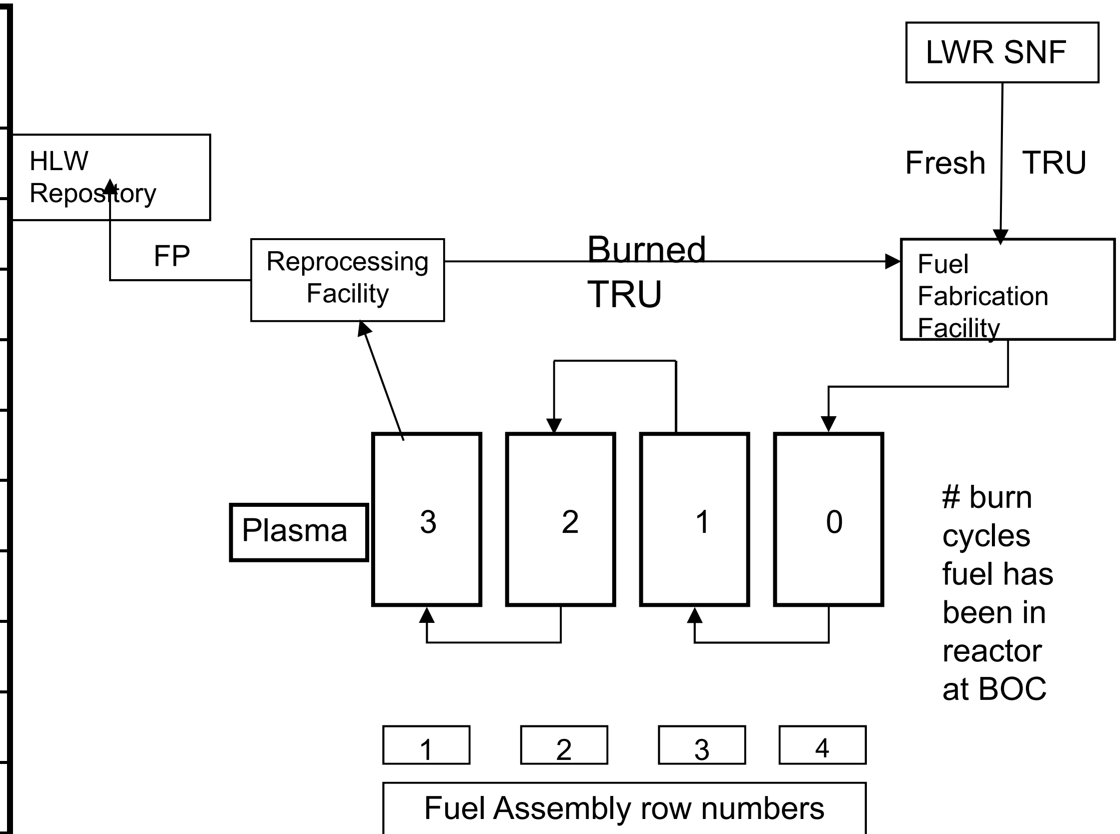
NEUTRONICS CALCULATION MODEL

- ERANOS Neutron Transport & Fuel Cycle Code
- 1968 P1 lattice calculation collapsed to 33 group homogenized assembly cross sections from 20 MeV to 0.1 eV. JEFF 2.0 Nuclear Data
  - 2D, 33 group, RZ, S8 discrete ordinates calculation with 91 radial and 94 axial mesh points
  - Source calculation with volumetric fusion neutron source adjusted to achieve 3000MWth thermal power in core.
  - For the fuel depletion, the flux and number densities are calculated every 233 days, with new multi-group cross sections being generated every 700 days.

SABR TRU BURNER Fuel Cycle

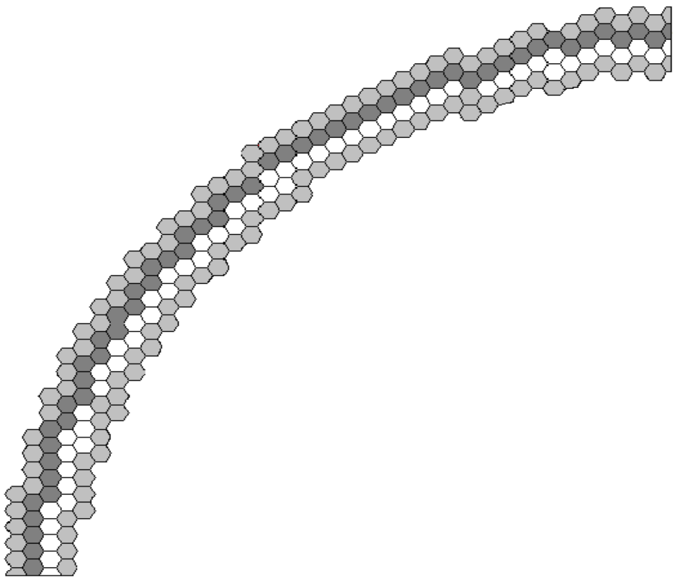
ANL Fuel Composition

	Mass Percent	Mass Percent
Isotope	BOL	BOC
Np^{237}	17.0	8.53
Pu^{238}	1.4	12.62
Pu^{239}	38.8	21.71
Pu^{240}	17.3	26.83
Pu^{241}	6.5	6.22
Pu^{242}	2.6	6.95
Am^{241}	13.6	8.32
Am^{242}	0.0	0.54
Am^{243}	2.8	2.96
Cm^{242}	0.0	0.40
Cm^{243}	0.0	0.08
Cm^{244}	0.0	2.25
Cm^{245}	0.0	0.57



4-BATCH TRU BURNER FUEL CYCLE

- Fuel cycle constrained by 200 dpa clad radiation damage lifetime. 4 (700 fpd) burn cycles per 2800 fpd residence
- OUT-to-IN fuel shuffling
- BOL k_{eff} = 0.945, P_{fus} = 172MW, 30.3 MT TRU
- BOC k_{eff} = 0.878, P_{fus}= 312MW, 28.8 MT TRU
- EOC k_{eff} = 0.831, P_{fus}= 409MW, 26.8 MT TRU
- 25.6% FIMA TRU burnup per 4-batch residence, >90% with repeated recycling
- 1.06 MT TRU/FPY fissioned
- 3000 MWth SABR supports 3.2 1000 MWe LWRs (0.25 MT TRU/yr) at 75% availability during operation (2 mo refueling).



ANNULAR CORE CONFIGURATION

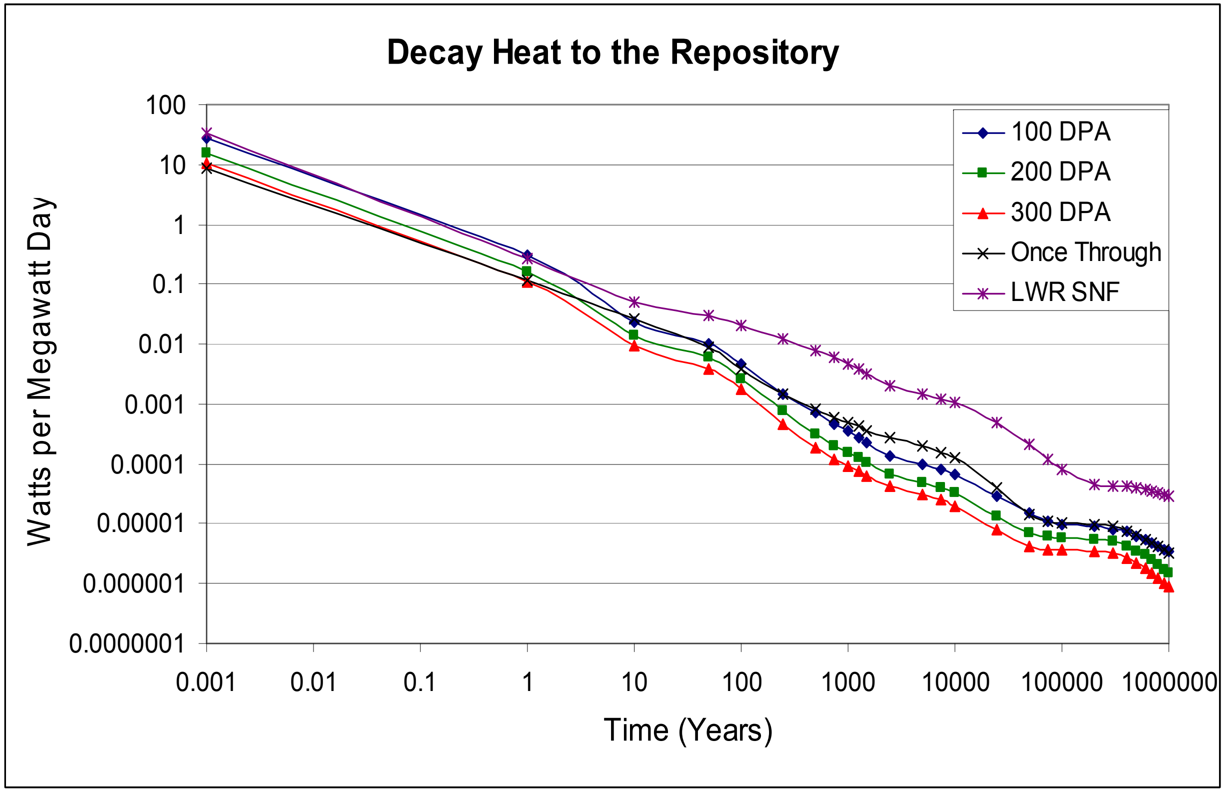
SABR TRU FUEL COMPOSITION (w/o)

ANL Composition 40Zr-10Am-10Np-40Pu (w/o)

Isotope	Fresh Fuel	BOC Input	To Re-Process	Core Av EOC/BOC
Np-237	17.0	8.53	7.25	9.1/8.3
Pu-238	1.4	12.62	17.3	14.6/17.3
Pu-239	38.3	21.71	18.3	21.9/20.3
Pu-240	17.3	26.83	29.2	27.2/28.2
Pu-241	6.5	6.22	7.31	5.55/5.55
Pu-242	2.6	6.95	7.45	6.50/6.99
Am-241	13.63	8.32	7.45	8.87/8.35
Am-242	0.00	0.54	0.84	0.71/0.74
Am-243	2.8	2.96	2.79	2.82/2.85
Cm-242	0.00	0.40	0.59	0.33/0.35
Cm-243	0.00	0.08	0.10	.075/.080
Cm-244	0.00	2.25	2.51	2.01/2.24
Cm-245	0.00	0.57	0.56	0.42/0.49

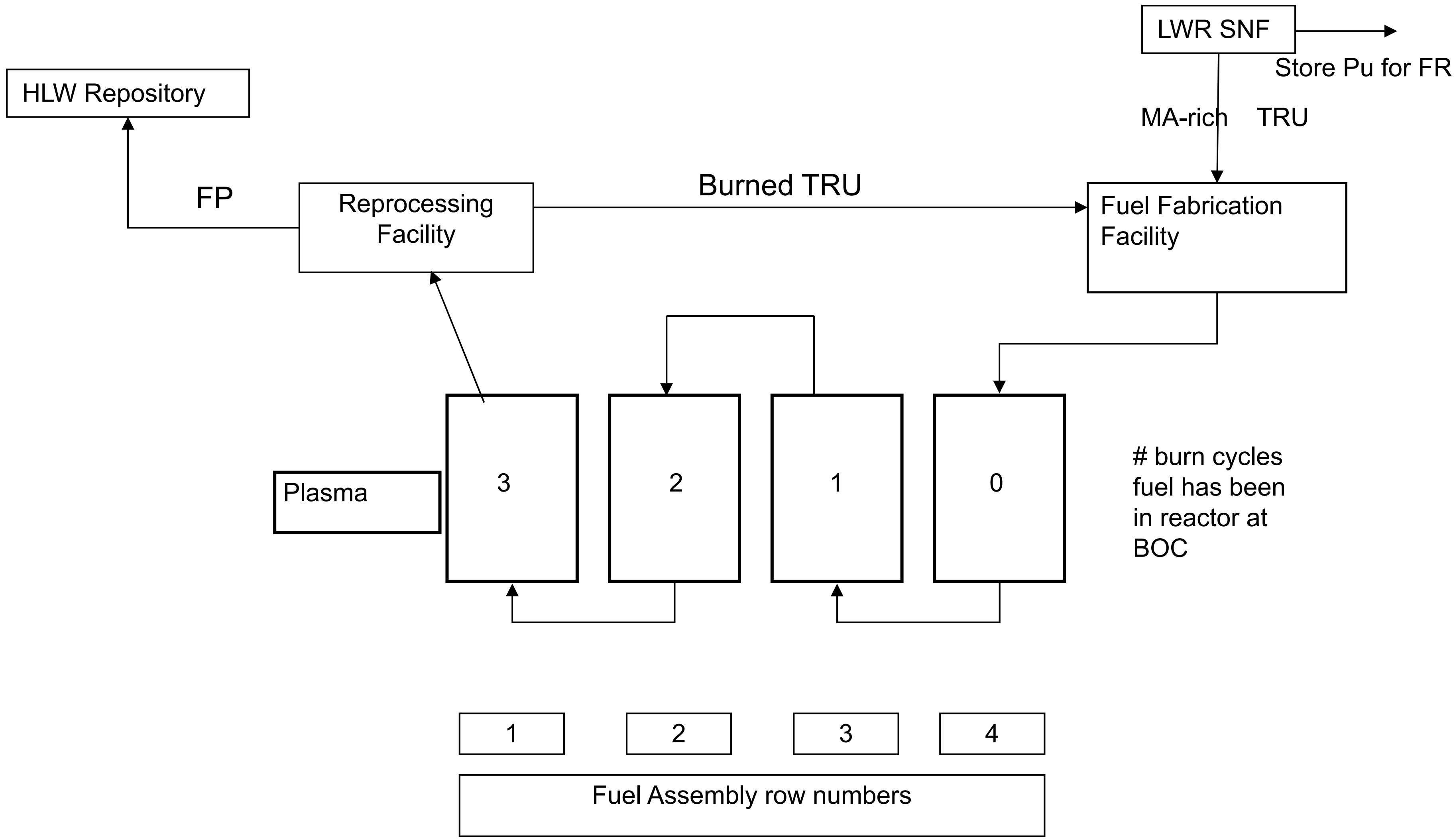
Effect of Clad Radiation Damage Limit on Fuel Cycle Transmutation Performance

Parameter	Units	100 DPA	200 DPA	300 DPA
TRU Burned per Residence	%	16.7%	25.6%	31.6%
TRU Burned per Year	MT/FPY	1.04	1.064	0.909
TRU Burned per Residence	MT	1.01	2.04	2.49
Ratio of Decay Heat to LWR SNF Decay Heat at 100,000 Years		0.063	0.035	0.024
Kilograms of TRU to repository per year (1% sep. efficiency)		67.68	31.39	19.71
LWR Support Ratio (75% availability)		2.9	3.2	3.6
DPA	Displacements per atom	97	214	294



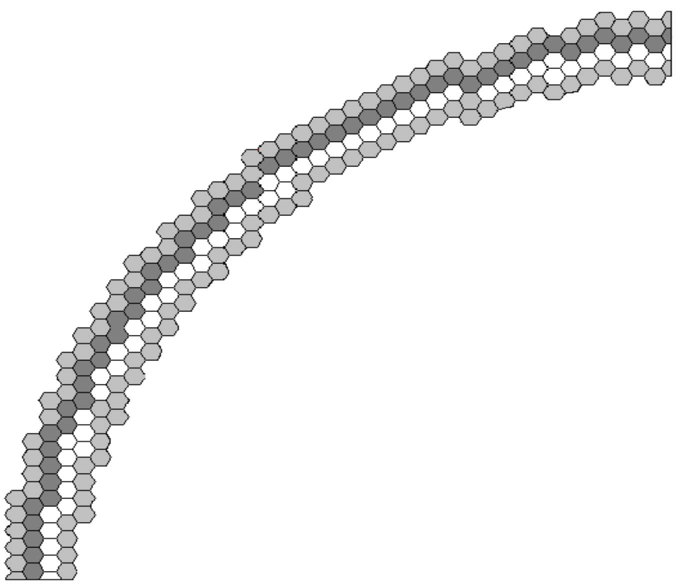
Decay Heat to the Repository
100
10
1
0.1
0.01
0.001
0.0001
0.00001
0.000001
0.0000001
Watts per Megawatt Day
0.001
0.01
0.1
1
10
100
1000
10000
100000
1000000
Time (Years)
100 DPA
200 DPA
300 DPA
Once Through
LWR SNF

SABR MA BURNER Fuel Cycle



4-BATCH MA BURNER FUEL CYCLE

- Fuel cycle constrained by 200 dpa clad radiation damage lifetime. 4 (700 fpd) burn cycles per 2800 fpd residence
- OUT-to-IN fuel shuffling
- BOL k_{eff} = 0.889, P_{fus} = 470 MW, 50.0 MT TRU
- BOC k_{eff} = 0.949, P_{fus} = 195 MW, 48.5 MT TRU
- EOC k_{eff} = 0.932, P_{fus} = 289 MW, 46.5 MT TRU
- 15.5% FIMA TRU burnup per 4-batch residence, >90% with repeated recycling
- 1.08 MT TRU/FPY (850 kg MA/FPY) fissioned
- 3000 MWth SABR supports 25.5 1000 MWe LWRs (25 kg MA/yr) at 75% availability during operation (2 mo refueling).

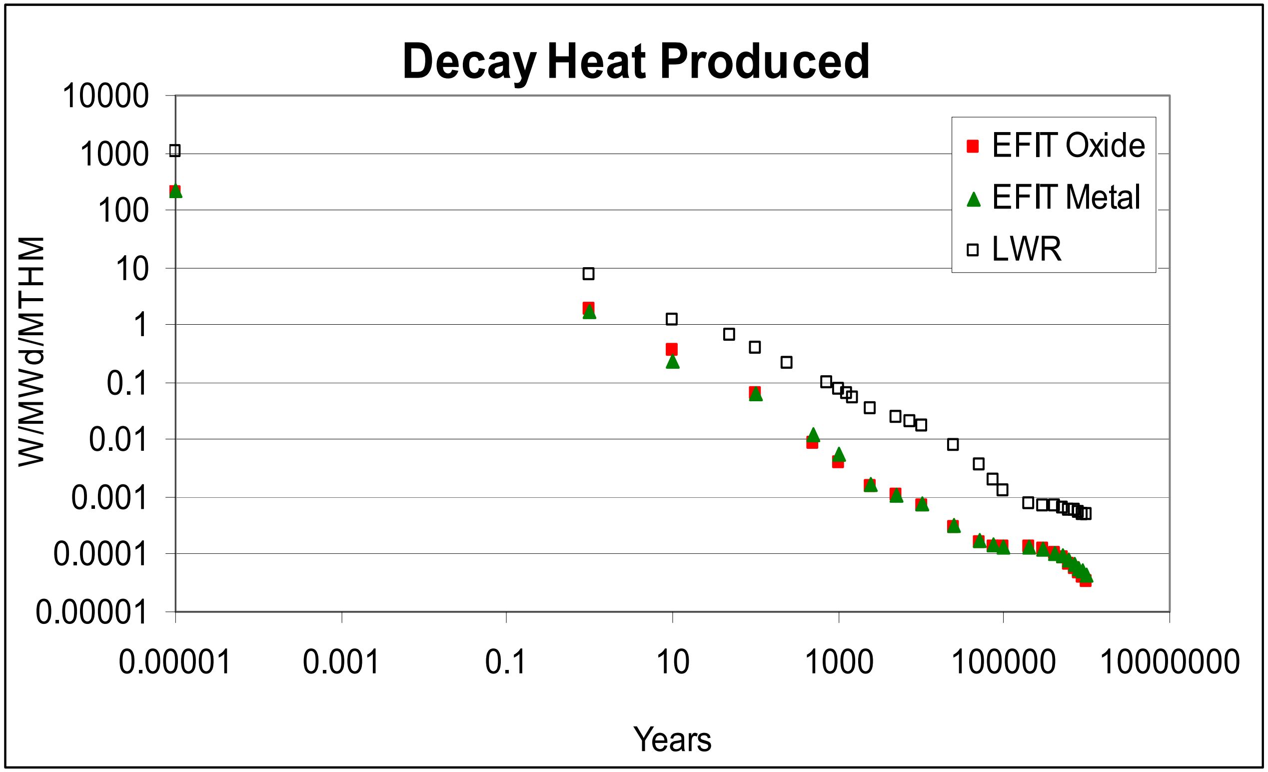


ANNULAR CORE CONFIGURATION

SABR MA TRU FUEL COMPOSITION (w/o)

EU Composition 13MgO-40Pu-43Am-2Np-2Cm

Isotope	Fresh Fuel	BOC Input	To Re-Process	Core Av EOC/BOC
Np-237	2.11	1.94	30.02	1.92/1.95
Pu-238	1.71	18.82	10.29	12.18/10.55
Pu-239	21.23	16.14	15.98	14.71/15.68
Pu-240	15.59	17.11	17.86	18.53/18.02
Pu-241	1.76	2.51	2.28	2.39/2.25
Pu-242	5.42	7.40	7.65	8.36/7.84
Am-241	41.00	31.49	30.02	27.48/29.46
Am-242	0.14	1.18	1.47	1.63/1.52
Am-243	8.72	7.64	7.38	7.20/7.37
Cm-242	0.00	1.19	0.65	0.69/0.77
Cm-243	0.03	0.12	0.12	0.12/0.12
Cm-244	1.63	3.25	2.51	3.97/3.69
Cm-245	0.62	0.78	0.76	0.82/0.76
Cm-246	0.05	0.06	0.01	0.02/0.01


Decay Heat Produced
W/MWd/MTHM
10000
1000
100
10
1
0.1
0.01
0.001
0.0001
0.00001
0.00001
0.001
0.1
10
1000
100000
10000000
Years
EFIT Oxide
EFIT Metal
LWR

SABR Neutronics
Fuel Cycle Comparison

	SABR TRU Burner ANL Metal Fuel	SABR-MA Burner EU-Metal Fuel	SABR-MA Burner EU-Oxide Fuel
Power Peaking	1.69/1.89	1.46/1.62	1.34/1.51
BOL Pfus (MW)	172	489	515
BOC Pfus (MW)	302	190	195
EOC Pfus (MW)	401	246	325
BOL Keff	0.945	0.889	0.909
BOC Keff	0.878	0.949	0.959
EOC Keff	0.831	0.932	0.936

SABR Mass Balance
Fuel Cycle Comparison

	SABR TRU Burner ANL Metal Fuel	SABR-MA Burner EU-Metal Fuel	SABR-MA Burner EU-Oxide Fuel
BOL Mass HM (kg)	30254	49985	47359
BOC Mass HM (kg)	28846	48468	45658
EOC Mass HM (kg)	26803	46441	43542
Delta Mass (kg)	2042	2027	2110
Loading outer (kg)	7887	13040	12345
HM Out (kg)	5862	11013	10234
FIMA (%)	25.6	15.5	17.1

FUEL CYCLE CONCLUSIONS
SABR FFH BURNER REACTORS

- A SABR TRU-burner reactor would be able to burn all of the TRU from 3 LWRs of the same power. A nuclear fleet of 75% LWRs (% nuclear electric power) and 25% SABR TRU-burner reactors would reduce geological repository requirements by a factor of 10 relative to a nuclear fleet of 100% LWRs.

- A SABR MA-burner reactor would be able to burn all of the MA from 25 LWRs of the same power, while setting aside Pu for future fast reactor fuel. A nuclear fleet of 96% LWRs and 4% SABR MA-burners would reduce HLWR needs by a factor of 10.

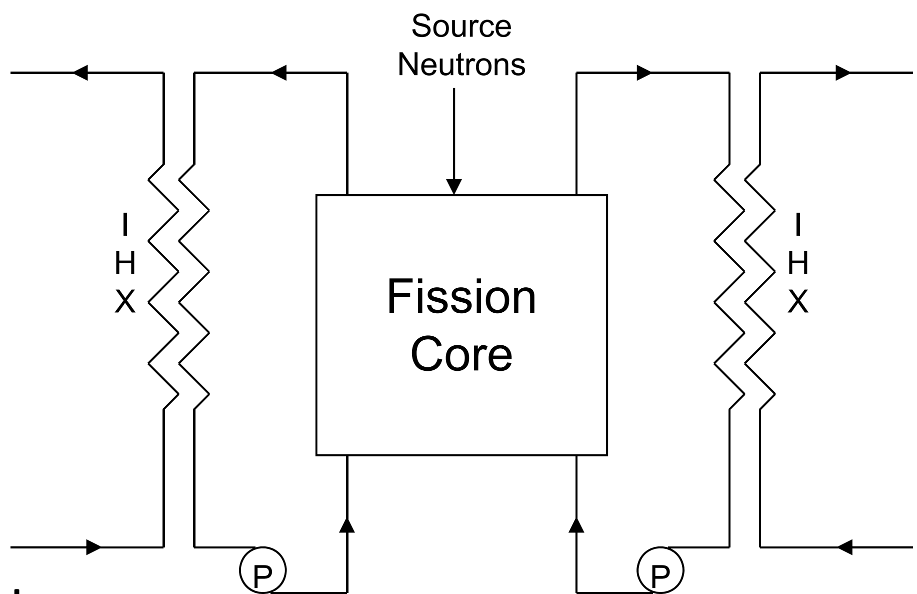
Comparison with ADS & Critical Burners

[a]A 1000MWe LWR produces 25 kg/yr MA. [b] present LWR fleet produces 25,000 kg/yr MA.

	SABR MA-metal	SABR MA-oxide	EFIT (ADS) MA-oxide	LCRFR (critical) MA-oxide/U
Power (MWth)	3000	3000	384	1000
MA fissioned (kg/yr)	853	674	135	261 (net)
Discharge burnup (%)	15.5	17.1	10.7	13.2
Fuel residence time (d)	2800	2800	1095	2100
LWR support ratio[a]	34.1	27.0	5.4	10.4
# units for USA LWR fleet [b]	3	4	19	10

RELAP5 DYNAMIC SAFETY ANALYSES

Accident Simulations



- **Accidents simulated:**
  - **Loss of Power Accident (LOPA),**
  - **Loss of Flow Accident (LOFA),**
  - **Loss of Heat Sink Accident (LOHSA), and**
  - **Accidental Increase in Fusion Neutron Source Strength.**
- **Coolant Boiling Temperature: 1,156 K, Fuel Melting Temperature: 1,473 K**
- **Small < 0 Doppler and >0 sodium coefs. Large < 0 fuel expansion reactivity coefficient not included in calculations.**

RESULTS---ACCIDENT ANALYSES

- Loss of plasma heating power leads to shutdown of SABR neutron source in 1-2 s, making this a good “scram” mechanism.

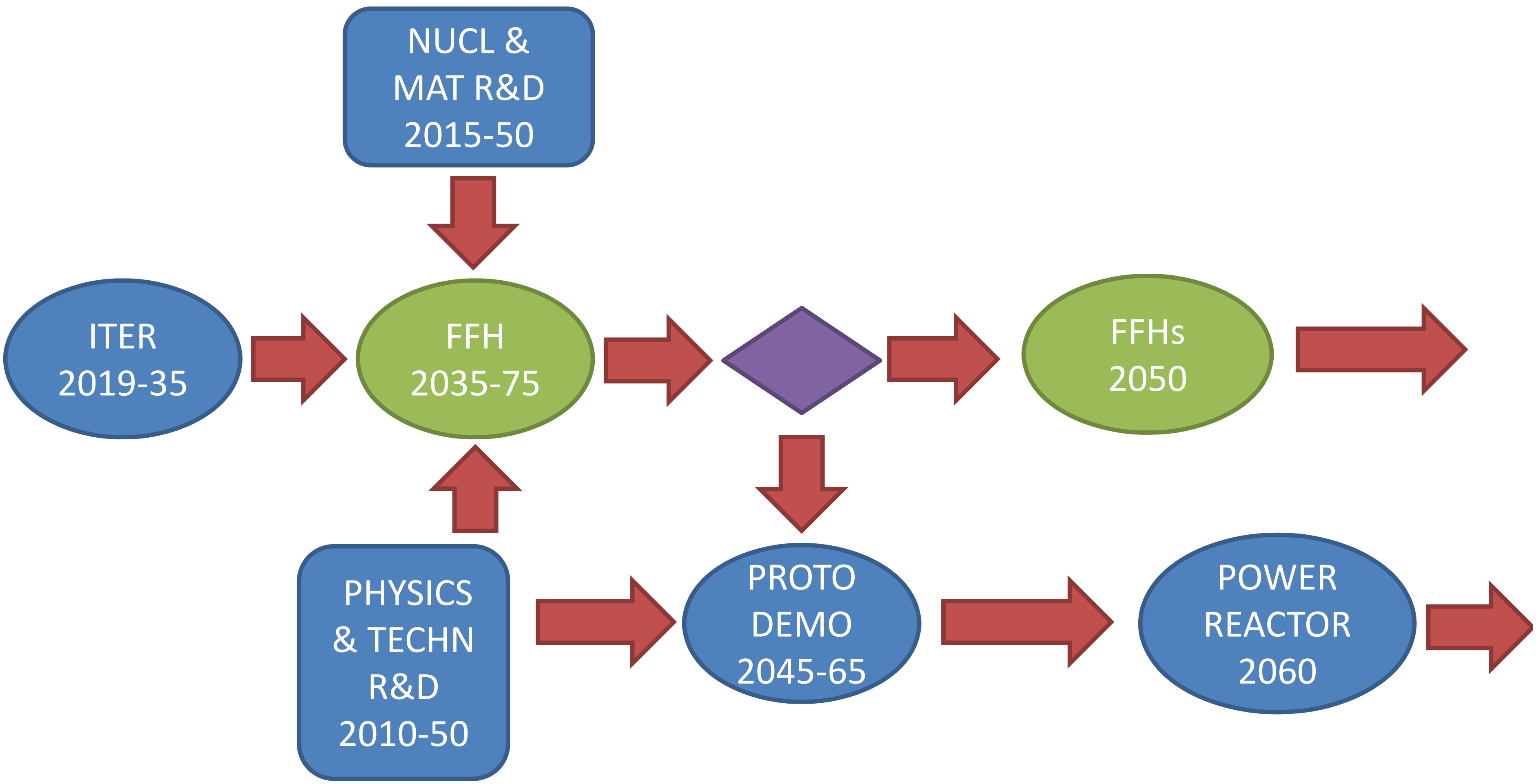
- Analyses (w/o negative fuel bowing coef) indicate loss of 50% flow (LOFA) or 50% heat removal (LOHSA) can be tolerated (w/o control action).

- Negative fuel bowing/expansion reactivity should lead to IFR/EBR-II passive safety (not yet modeled).

- If the plasma operates just below ‘soft’ instability limits, any neutron source surges should be self-limited by plasma pressure and density limits.

FUSION POWER DEVELOPMENT WITH A DUAL FUSION-FISSION HYBRID PATH

FUSION POWER DEVELOPMENT WITH A DUAL FUSION-FISSION HYBRID PATH



Plasma Physics Advances Beyond ITER

- PROTODEMO must achieve reliable, long-pulse plasma operation with *plasma parameters* (β,τ) *significantly more advanced* than ITER.

- FFH must achieve *highly reliable, very long-pulse plasma operation* with plasma parameters similar to those achieved in ITER.

Fusion Technology Advances Beyond ITER

- FFH must operate with moderately higher surface heat and neutron fluxes and with *much higher reliability* than ITER.
- PROTODEMO must operate with *significantly higher surface heat and neutron fluxes* and with higher reliability than ITER.
- PROTODEMO and FFH would have similar magnetic field, plasma heating, tritium breeding and other fusion technologies.
- PROTODEMO and FFH would have a similar requirement for a radiation-resistant structural material to 200 dpa.

FUSION R&D FOR A SABR FFH IS ON THE PATH TO FUSION POWER

FFH PLASMA PHYSICS R&D for FFH or PROTODEMO

1. Control of instabilities.
2. Reliable, very long-pulse operation.
3. Disruption avoidance and mitigation.
4. Control of burning plasmas.

FFH FUSION TECHNOLOGY R&D for FFH or PROTODEMO

1. Plasma Support Technology (magnets, heating, vacuum,etc.)—improved reliability of the same type components operating at same level as in ITER.
2. Heat Removal Technology (first-wall, divertor)—adapt ITER components to Na coolant and improve reliability.
3. Tritium Breeding Technology—develop reliable, full-scale blanket & tritium processing systems based on technology tested on modular scale in ITER.
4. Advanced Structural (200 dpa) and Other Materials.
5. Configuration for remote assembly & maintenance.

ADDITIONAL FUSION R&D BEYOND FFH FOR TOKAMAK ELECTRIC POWER

1. Advanced plasma physics operating limits (β,τ).
2. Improved components and materials.

INTEGRATION OF FUSION & FISSION TECHNOLOGIES IS NEEDED FOR FFH

- For Na, or any other liquid metal coolant, the magnetic field creates heat removal challenges (e.g. MHD pressure drop, flow redistribution). Coating of metal surfaces with electrical insulation is one possible solution. This is also an issue for a PROTODEMO with liquid Li or Li-Pb.
- Refueling is greatly complicated by the tokamak geometry, but then so is remote maintenance of the tokamak itself, which is being dealt with in ITER and must be dealt with in any tokamak reactor. However, redesign of fuel assemblies to facilitate remote fueling in tokamak geometry may be necessary.
- The fusion plasma and plasma heating systems constitute additional energy sources that conceivably could lead to reactor accidents. On the other hand, the safety margin to prompt critical is orders of magnitude larger in SABR than in a critical reactor, and simply turning off the plasma heating power would shut the reactor down to the decay heat level in seconds.
- Etc.

PROs & CONs of Supplemental FFH Path of Fusion Power Development

- Fusion would be used to help meet the USA energy needs at an earlier date than is possible with ‘pure’ fusion power reactors. This, in turn, would increase the technology development and operating experience needed to develop economical fusion power reactors.

- FFHs would support (may be necessary for) the full expansion of sustainable nuclear power in the USA and the world.

- An FFH will be more complex and more expensive than either a Fast Reactor (critical) or a Fusion Reactor.

- However, a nuclear fleet with FFHs and LWRs should require fewer burner reactors, reprocessing plants and HLWRs than a similar fleet with critical Fast Burner Reactors.

RECOMMENDATIONS

- Perform an in-depth *conceptual design* of the burner reactor-neutron source-reprocessing-repository system to determine if it is technically feasible to deploy a SABR FFH Advanced Burner Reactor within 25 years and identify needed R&D.

- Perform comparative *dynamic safety and fuel cycle studies* of critical and sub-critical ABRs to quantify any transmutation performance advantages of a SABR because of the relaxation of the criticality constraint and the much larger reactivity margin of safety to prompt critical.

- Perform *comparative systems and scenario studies** to evaluate the cost-effectiveness of various combinations of Critical, FFH and ADS Advanced Burner Reactors disposing of the legacy spent fuel TRU and the spent fuel TRU that will be produced by an expanding US LWR fleet. The cost of HLWRs and fuel separation and refabrication facilities, as well as the cost of the burner reactors, should be taken into account.

*Small studies ongoing at ANL and KIT.

The Issues to be Studied for the FFH Burner Reactor System

- ***Is a FFH Burner Reactor Technically Feasible*** *and on what timescale?* A detailed *conceptual design study* of an FFH Burner Reactor and the fuel reprocessing/ refabrication system should be performed to identify: a) the readiness and technical feasibility issues of the separate fusion, nuclear and fuel reprocessing/refabricating technologies; and b) the technical feasibility and safety issues of integrating fusion and nuclear technologies in a FFH burner reactor. This study should involve experts in all physics and engineering aspects of a FFH system: a) fusion; b) fast reactors; c) materials; d) fuel reprocessing/refabrication; e) high-level radioactive waste (HLW) repository; etc. The study should focus first on the most advanced technologies in each area; e.g. the tokamak fusion system, the sodium-cooled fast reactor system.

- ***Is a FFH Burner Reactor needed*** *for dealing with the accumulating inventory of spent nuclear fuel (SNF)discharged from LWRs?* First, *dynamic safety and fuel cycle analyses* should be performed to quantify the advantages in transmutation performance in a FFH that result from the larger reactivity margin to prompt critical and the relaxation of the criticality constraint. Then, a *comparative systems study of several scenarios* for permanent disposal of the accumulating SNF inventory should be performed, under different assumptions regarding the future expansion of nuclear power. The scenarios should include: a) burying SNF in geological HLW repositories without further reprocessing; b) burying SNF in geological HLW repositories after separating out the uranium; c) reprocessing SNF to remove the transuranics for recycling in a mixture of critical and FFH burner reactors (0-100% FFH) and burying only the fission products and trace transuranics remaining after reprocessing; d) scenario "c" but with the plutonium set aside to fuel future fast breeder reactors (FFH or critical) and only the "minor actinides" recycled; e) scenarios (c) and (d) but with pre-recycle in LWRs; etc. Figures of merit would be: a) cost of overall systems; b) long-time radioactive hazard potential; c) long-time proliferation resistance; etc.

- ***What additional R&D is needed for a FFH Burner Reactor*** *in addition to the R&D needed to develop the fast reactor and the fusion neutron source technologies?* This information should be developed in the conceptual design study identified above.

GEORGIA TECH SABR DESIGN TEAM 2000-11

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Relation Between Fusion and Fission Power

Sub-critical operation increases fuel residence time in Burner Reactor before reprocessing is necessary

$$N_{fis} = \frac{Sk}{\upsilon\Sigma_{rem}\left(1-k\right)}, \quad S = \frac{P_{fus}}{E_{fus}}, \quad k = \frac{\nu\Sigma_{fis}}{\Sigma_{rem}}$$

$$P_{fis} = \upsilon N_{fis}\Sigma_{fis}E_{fis}, \; P_{fis} = \frac{E_{fis}}{E_{fus}}\nu\frac{\left(1-k\right)}{k}P_{fus}$$

As *k* decreases due to fuel burnup, Pfus can be increased to compensate and maintain Pfis constant.

Thus, sub-critical operation enables fuel burnup to the radiation damage limit before it must be removed from the reactor for reprocessing.

Sub-critical operation provides a larger margin of safety against accidental reactivity insertions that could cause prompt critical power excursions.

$$\frac{dn}{dt} = \frac{\rho - \beta}{\Lambda} n + \lambda C, \quad \rho \equiv \frac{k-1}{k}, \quad \Lambda \equiv \left(\upsilon \Sigma_{fis}\right)^{-1} \quad neutron\,kinetics$$

$$\frac{dC}{dt} = \frac{\beta}{\Lambda} n - \lambda C, \quad delayed\ neutron\ precursors$$

$$n(t) = n_0 \left[\frac{\rho}{\rho - \beta} \exp\left(\frac{\rho - \beta}{\Lambda} t \right) - \frac{\beta}{\rho - \beta} \exp\left(\frac{-\lambda \rho}{\rho - \beta} t \right) \right]$$

$$\Lambda \approx 10^{-6}\, s \ for\ fast\ reactors, \ \lambda^{-1} \approx 0.1 - 10\, s$$

$$for\ critical\ reactor,\ accidental\ reactivity\ insertion\ margin\ of\ safety\ \rho_{acc} = \beta$$

$$\beta \sim 0.007\ for\ U\ fuel, \ \sim 0.002 - 0.003\ for\ TRU\ fuel$$

$$for\ sub-critical\ reactor\ \ \rho \Rightarrow \rho + \rho_{sub}, \ \rho_{sub} = \frac{k_{sub} - 1}{k_{sub}} < 0, \ \left|\rho_{sub}\right| >> \beta$$

$$margin\ of\ safety\ \rho_{acc} = \left|\rho_{sub}\right| + \beta, \quad for\ k = 0.9, \left|\rho_{sub}\right| \sim 0.1$$