The ARIES Designs for a Tokamak Power Plant

With particular emphasis on the ARIES-I and PULSAR Designs

presented by

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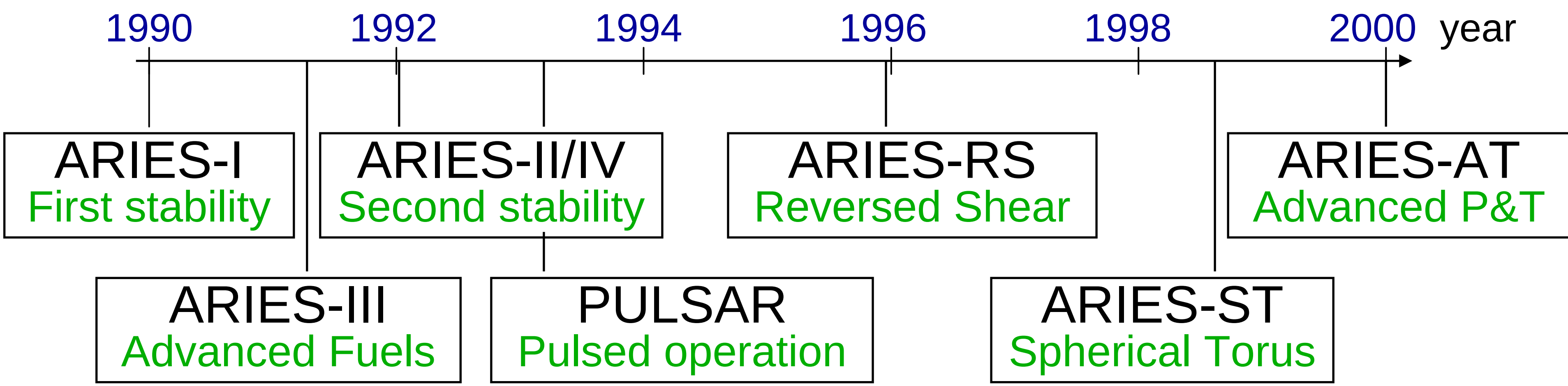
Outline

- **Overview of the ARIES studies**
- **Description of ARIES-I**
- **Description of PULSAR**
- **Relation of ARIES-I and PULSAR to the other ARIES designs**
- **Comparison of ARIES-I and PULSAR**
- **Some of the lessons learned from the studies**
- **Summary and conclusions**

ARIES

- **Advanced Reactor Innovation Evaluation Study**
  - ~ 10 year program to identify and begin evaluation of attractive concepts for a Magnetic Fusion Energy (MFE) power plant
  - Led by Profs. R. Conn and F. Najmabadi (UCLA/UCSD)
  - Involved over 50 fusion scientists and engineers from about 15 institutions…University, Laboratory, Private sector
- **Documented 7+ tokamak power plant designs**



Philosophy of the ARIES Studies

A new technology (FUSION) can penetrate the market only if it is significantly better than any existing technology (FISSION)

- Attractive Safety Features
  - Eliminate "N-stamp" requirements (extensive, expensive design certification)
  - Low radioactive inventory (no 3-mile island or Chernobyl)
  - Minimal weapons proliferation and security costs
- Attractive Environmental Features
  - Waste disposal advantages – "Class-C" shallow-land burial (no Yucca Mountain)
- The assumption is that if these advantages are factored into the "true cost", then fusion will have an advantage over fission
  - (Corollary is that the designs must be such as to keep these advantages)
- The physics and engineering assumptions used in the ARIES designs were sometimes very aggressive in order to get an attractive design:
  - "Theoretically possible", not necessarily "experimentally demonstrated"

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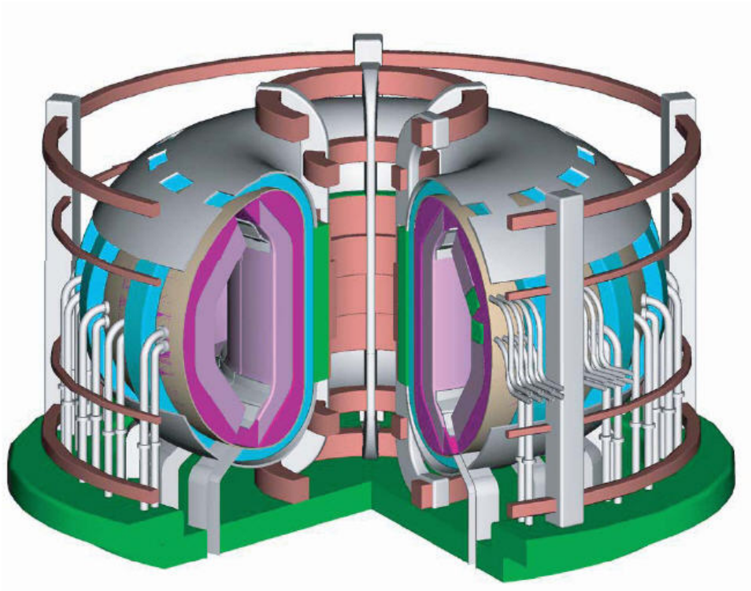
ARIES-I

First-Stability Regime, Steady State Plasma
MHD Stable to kink modes without a Conducting Wall

ARIES
PULSAR
STARLITE

- **ARIES-I design was to have "present day" physics ("first stability regime"), aggressive engineering, but keeping safety and environmental advantages**
- **Because RF current drive is relatively inefficient, the fraction of self-generated current (bootstrap current) must be large…68% in ARIES-I**
- **The constraint of "first stability" and high bootstrap current leads to relatively low β = 1.9%, and modest normalized β_N = 3.0,**
- **Since fusion power ~ $\beta^2 B^4$, this is compensated by high B (21 T at coil, 11.3 T at plasma center)***



R=6.75 m, a = 1.5 m,
B_T = 11.3 T, I_P = 10 MA

1 GW net power

*(Redesign, A-1', has 16 T and 9 T)

Engineering features of the ARIES-I design:

- Advanced superconductor Nb_3Sn alloys toroidal field magnets producing 21 (16) T at magnet and 11.3 (9) T at plasma center
- ARIES-I blanket is He-cooled (at 10 MPa) design with SiC composite structural material, and Li_2ZrO_3 solid tritium breeders with beryllium neutron-multiplier
  - SiC composites are high strength, high temperature structural materials with very low activation and very low decay afterheat
- An advanced Rankine power conversion cycle as proposed for future coal-burning plants (49% gross efficiency).
- Folded wave-guide launcher made of SiC composite with 0.02 mm Cu coating (for RF current drive)
- Fusion power core modular for easy maintenance using a vertical lift approach

Major Parameters of ARIES-I

		A-1'	ARIES PULSAR STARLITE
Aspect Ratio	4.5	4.5	High A to decrease I_P and divertor heat loads, and increase f_{BS},
Major Radius (m)	6.75	7.9	Required for power balance
Vertical Elongation κ	1.8	1.8	Minimizes PF energy, vertically stable
Plasma Current (MA)	10.2	10	Provides adequate confinement
Toroidal Field on Axis (T)	11.3	9	At limit of advanced alloy conductor
Toroidal beta	1.9%	1.9%	At first stability limit without wall
Neutron Wall load (MW/m^2)	2.5	2.0	20-MWy/m^2 lifetime
Fusion power (MW)	2564		97 MW ICRF CD power,
Net electric power (MW)	1000	1000	Same for all ARIES designs
Net plant efficiency	39%		Advanced Rankine steam power cycle with 49% efficiency

ARIES-I

First-Stability Regime, Steady State Plasma
MHD Stable with no Conducting Wall

ARIES
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Troyon Limit: $$\beta \leq C_T \left(\frac{\mu_0}{40\pi}\right)\left(\frac{I_P}{aB_T}\right)$$ *C_T=3.0*

or, $$(\beta/\varepsilon)(\varepsilon\beta_P) \leq \left(\frac{C_T}{20}\right)^2 \frac{(1+\kappa^2)}{2}$$

I_P = plasma current (MA)
B_T = toroidal field (T)
ε = a/R = inverse aspect ratio
$\beta = 2\langle p\rangle/B^2$, $\beta_p = 2\langle p\rangle/B_p^2$
κ = plasma elongation

Bootstrap fraction: $$\frac{I_{BS}}{I_P} = \frac{1}{\varepsilon^{1/2}}(\varepsilon\beta_P)C_{BS}$$ *C_{BS}=0.5*

it follow that ➔ $$(\beta/\varepsilon) \leq \frac{1}{\varepsilon^{1/2}\frac{I_{BS}}{I_P}} C_{BS} \left(\frac{C_T}{20}\right)^2 \frac{(1+\kappa^2)}{2}$$

Bootstrap alignment: *Need to have $q_0 > 1$ for $I_{BS}/I_P > 0.5$ to avoid local bootstrap overdrive. This tends to lower C_T*

=> tradeoff between high β and high Bootstrap fraction

A=R/a=4.5, I_{BS}/I_P=.68, β=1.9%, q_0=1.3

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Objectives of the PULSAR study

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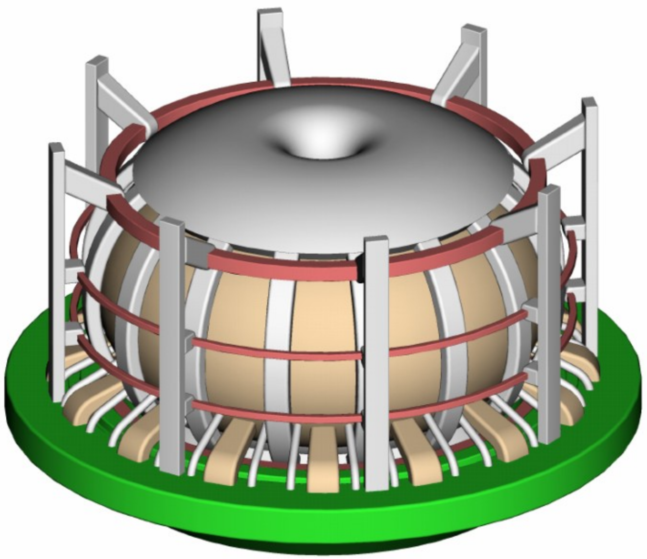
- **Study the feasibility and potential features of a tokamak with a pulsed mode of plasma operation as a fusion power plant.**
- **Identify trade-offs which lead to the optimal regime of operation.**
- **Identify critical and high-leverage issues unique to a pulsed-plasma tokamak power plant.**
- **Compare steady-state and pulsed tokamak power plants.**



R=8.6 m, a = 2.15 m,
B_T = 7.5 T, I_P = 15 MA

1 GW net power

PULSAR Plasma Regime of Operation

- The loop voltage induced by the "inductive" current-drive system is constant across the plasma (stationary state):

$$\frac{\partial \vec{B}}{\partial t} = 0 \quad \Rightarrow \quad \nabla \times \vec{E} = 0 \quad \Rightarrow \quad E = \frac{V_L}{2\pi} \nabla \phi$$

  – In this stationary state, plasma current-density profiles (induced and bootstrap) are determined by *n* and *T* profiles;

$$\vec{J} = \frac{V_L}{2\pi R \eta(n,T)} + J_{BS}(n,T)$$

  – Pressure profile is *n* x *T*

- Thus, Equilibrium is completely determined from $n(\psi)$, $T(\psi)$, I_P
  – *No additional freedom to tailor the current profile to improve stability limits*

PULSAR Plasma Regime of Operation(2)

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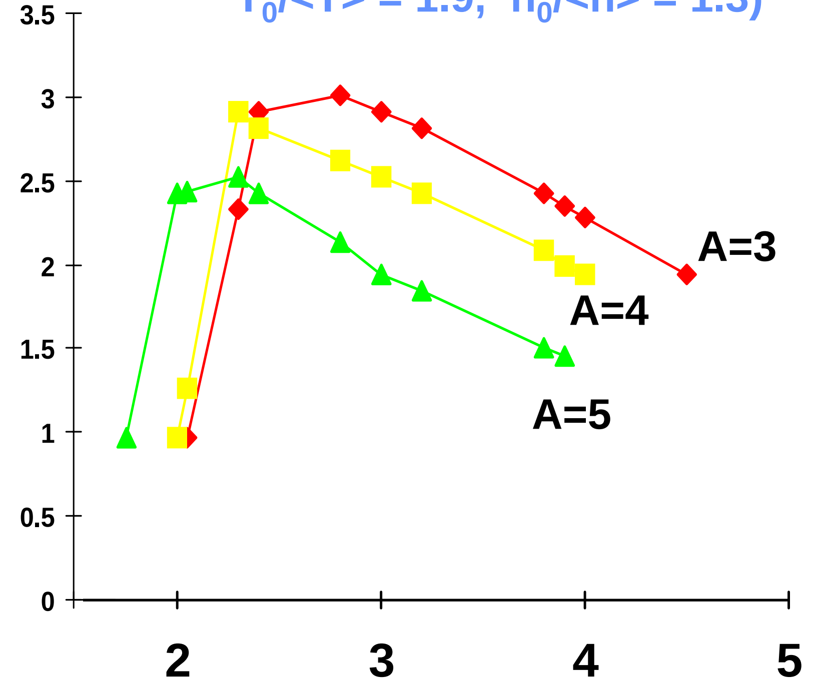
$$\beta_N \equiv \frac{\beta a B_T}{I_P}$$

- It follows that the current-density profile cannot be tailored to achieve the highest possible β
  - β_N is limited to ≤ 3.0 (for most favorable profiles…broad)
  - Bootstrap fraction is not large (~30% to 40%, maximum)
  - Second stability operation is not possible
- A large scan of stable ohmic equilibrium was made and a fit to the data base was used in the systems analysis

Aspect ratio and q* variation:
($T_0/\langle T\rangle = 1.9$, $n_0/\langle n\rangle = 1.3$)



$$q^* \equiv \frac{B_T \pi a^2 (1+\kappa^2)}{R I_P}$$

Power Flow in a Pulsed Tokamak

- **Utilities require a minimum electric output for the plant to stay on the grid**
- **Grid requires a slow rate of change in introducing electric power into the grid**
- **Large thermal power equipment such as pumps and heat exchangers cannot operate in a pulsed mode. In particular, the rate of change of temperature in the steam generator is ~ 2^{o}C/min in order to avoid boiling instability and induced stress.**

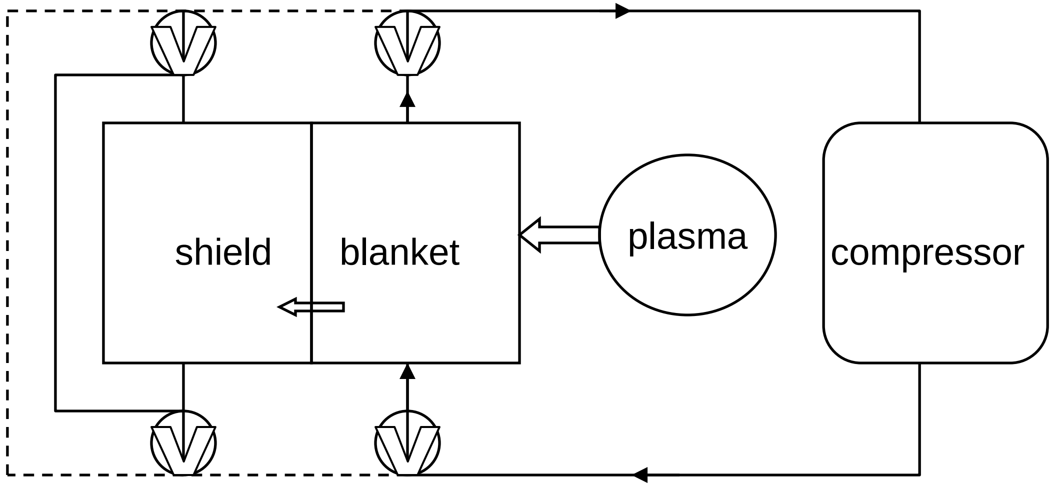
➔ **Therefore, Steady Electric Output is Required and an Energy storage system is needed.**

PULSAR Energy Storage System

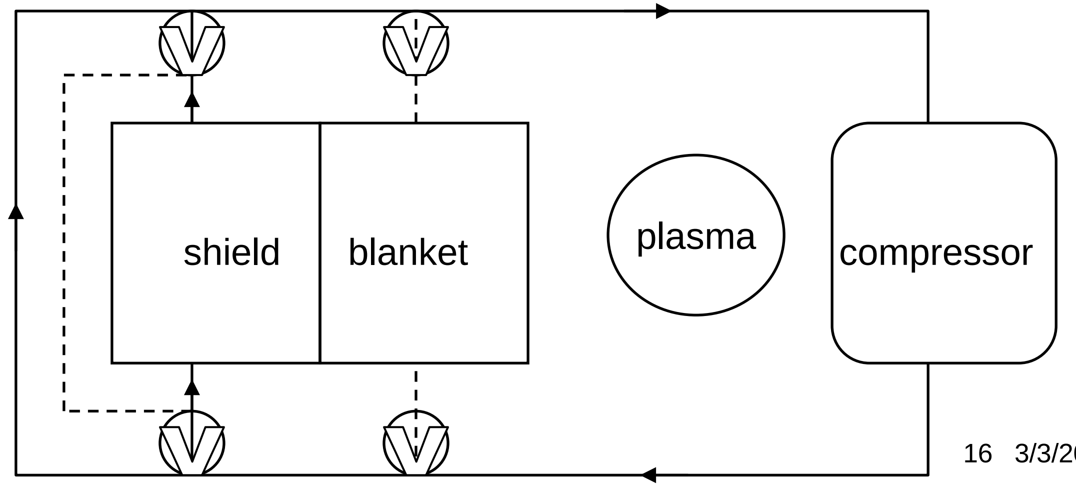
- An external energy storage system which uses the thermal inertia is inherently very large:
  - During the burn, $T_{coolant} > T_{storage}$
  - During the dwell, $T_{coolant} < T_{storage}$
  - But the coolant temperature should not vary much. Therefore, thermal storage system should be very large
- PULSAR uses the outboard shield as the energy storage system and uses direct nuclear heating during the burn to store energy in the shield;
  - This leads to a low cost energy storage system but the dwell time is limited to a few 100's of seconds

Energy is accumulated in outer shield during burn phase, regulated by mass flow control during dwell phase

Burn Phase



Dwell Phase



PULSAR cycle

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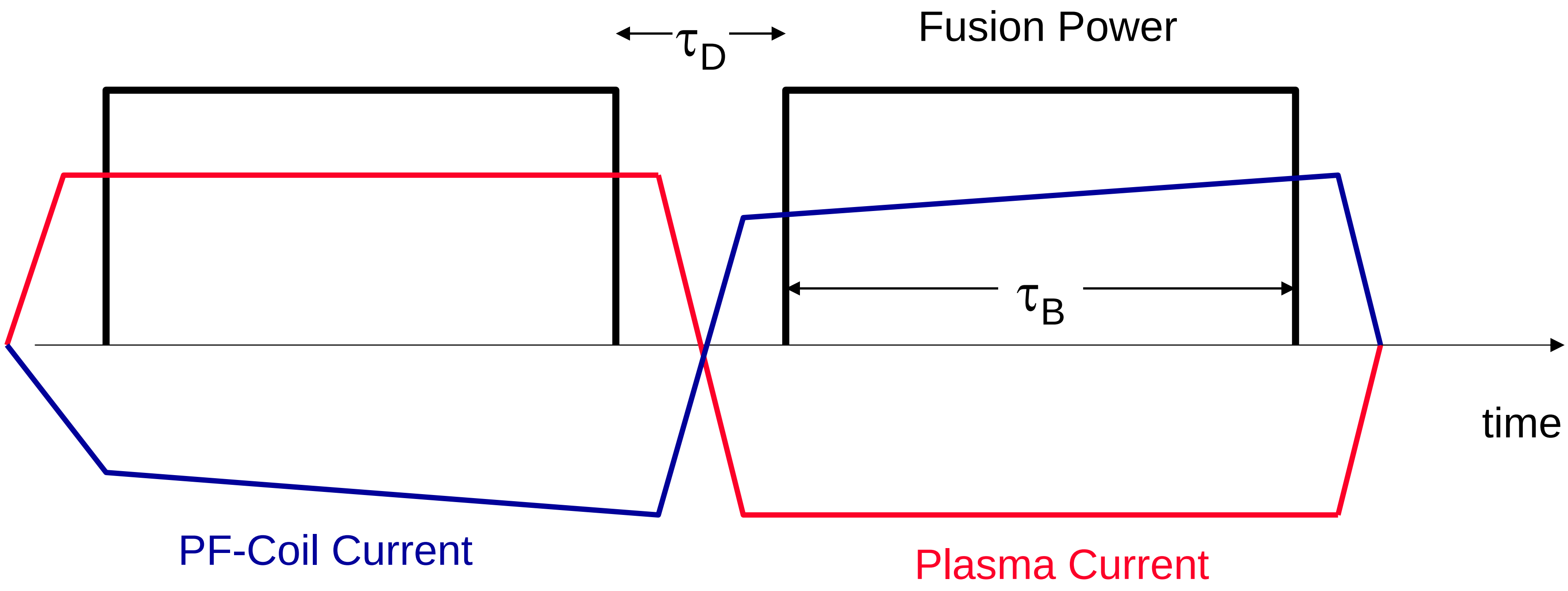


- Plasma physics sets lower limit on dwell time;
  - upper limit set by thermal storage system.
- Burn time determined through trade-off between size of OH system and number of cycles.
  - COE insensitive to burn times between 1 and 4 Hrs

$\tau_D \sim 200$ sec

$\tau_B \sim 9000$ sec

Pulsar Dwell Time Calculation

Current Rampup time	**54 s**
Plasma ignition time,	**53 s**
Plasma de-ignition time	**38 s**
Plasma shutdown time	**54 s**
Total Dwell time:	**200 s**

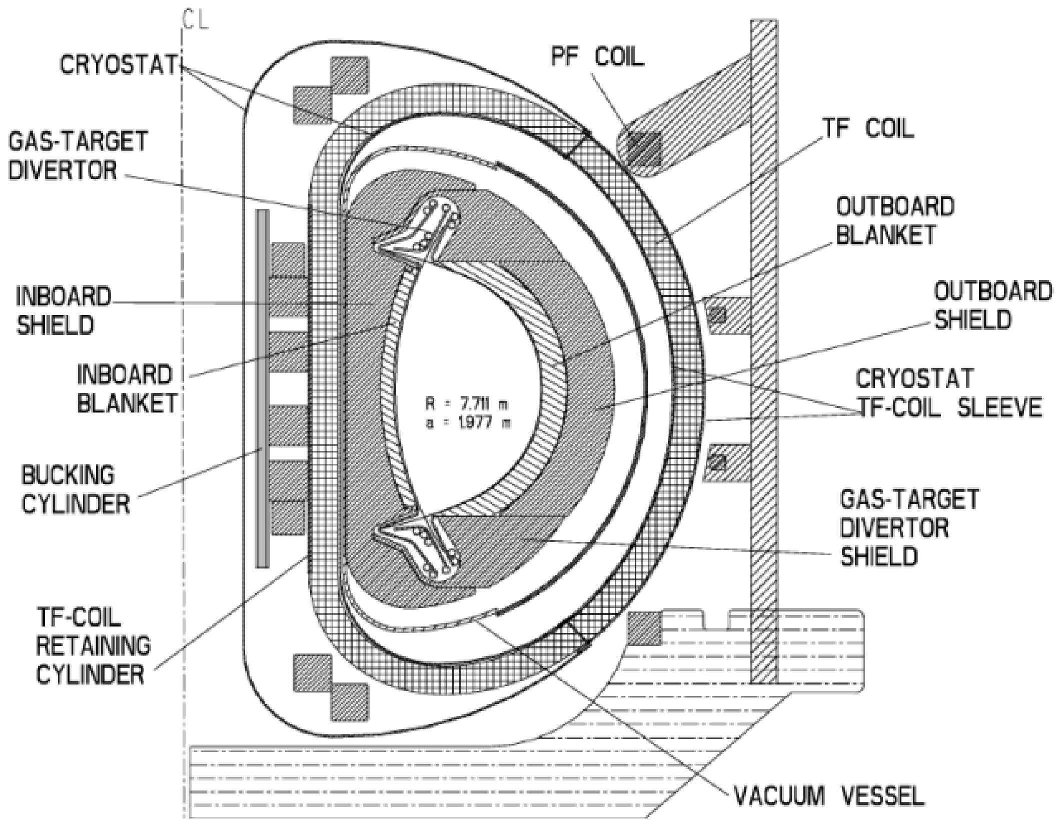
Burn time, 9000 s

Number of cycles 2,700 / year

PULSAR magnet system

- **The PULSAR TF magnet system is similar to the old ITER design**
- **OH solenoid is located between the TF coil and the bucking cylinder**
- **Shear panels are used between the TF coils**
- **Inner legs of the TF coils are keyed together to support the shear loads**
- **Because of the elaborate key system, the supportable stress in the inner leg of the TF coils is reduced**

=> ~15% Lower Toroidal field strength than in ARIES I



Major Parameters of PULSAR

A-1'

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Aspect Ratio	4.0	(4.5)	Optimizes at slightly lower A since f_{BS} does not weigh as heavily
Major Radius (m)	8.5	7.9	Required for power balance
Vertical Elongation κ	1.8	1.8	Minimizes PF energy, vertically stable
Plasma Current (MA)	13	10	Provides adequate confinement
Toroidal Field on Axis (T)	6.7	9	Lower due to cyclic PF induced stresses
Toroidal beta	2.8%	1.9%	First stability limit without wall, lower β_P
Neutron Wall load (MW/m^2)	1.3	2.0	20-MWy/m^2 lifetime
Net electric power (MW)	1000	1000	Same for all ARIES designs

PULSAR

ITER-like design with current driven by OH-coils + Bootstrap current

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Troyon Limit: $$(\beta/\varepsilon)(\varepsilon\beta_P) \le \left(\frac{C_T}{20}\right)^2 \frac{(1+\kappa^2)}{2}$$ *C_T=3.0*

Operate at higher I_P (lower β_P) to maximize β/ε

- However, no freedom in current profile…no non-inductive current drive
  - Current profile J determined from T and n profiles by stationary constraint

$$\frac{\langle \eta(J - J_{BS}) \bullet B \rangle}{\langle B \bullet \nabla\varphi \rangle} = \frac{V_L}{2\pi}$$

- Using this constraint, stability boundaries can be mapped out
  - Depend only on ε, q*, and density and temp. profile form factors

9000 s burn with 200 s OH recharge, during which thermal reservoir is tapped

$A=R/a=4$, $I_{BS}/I_P=.34$, $\beta=2.8\%$, $q_0=0.8$

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Reactor Operating Modes

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	1ST STABILITY REGIME-wall stabilization not required	2ND STABILITY REGIME-wall stabilization of kink modes
STEADY STATE	MODERATE β MODERATE β_p ARIES-I	HIGH β HIGH β_p ARIES RS, AT, ST
PULSED	HIGH β LOW β_p PULSAR	NOT POSSIBLE

Dimensionless Parameter Space
$\beta/\varepsilon \times 2/(1+\kappa^2)$ vs $\varepsilon\,\beta_P$
for Tokamak Reactor Regimes

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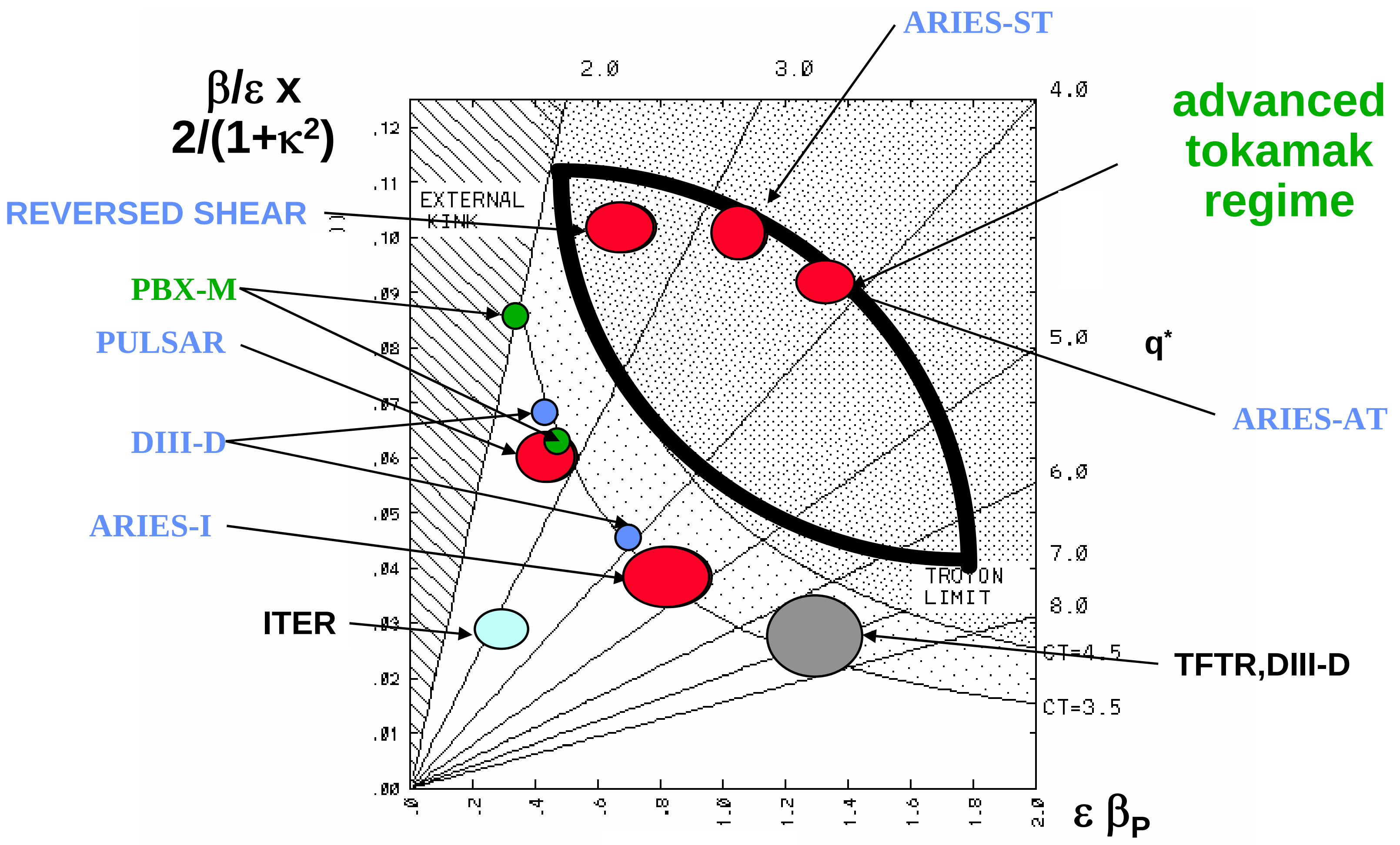
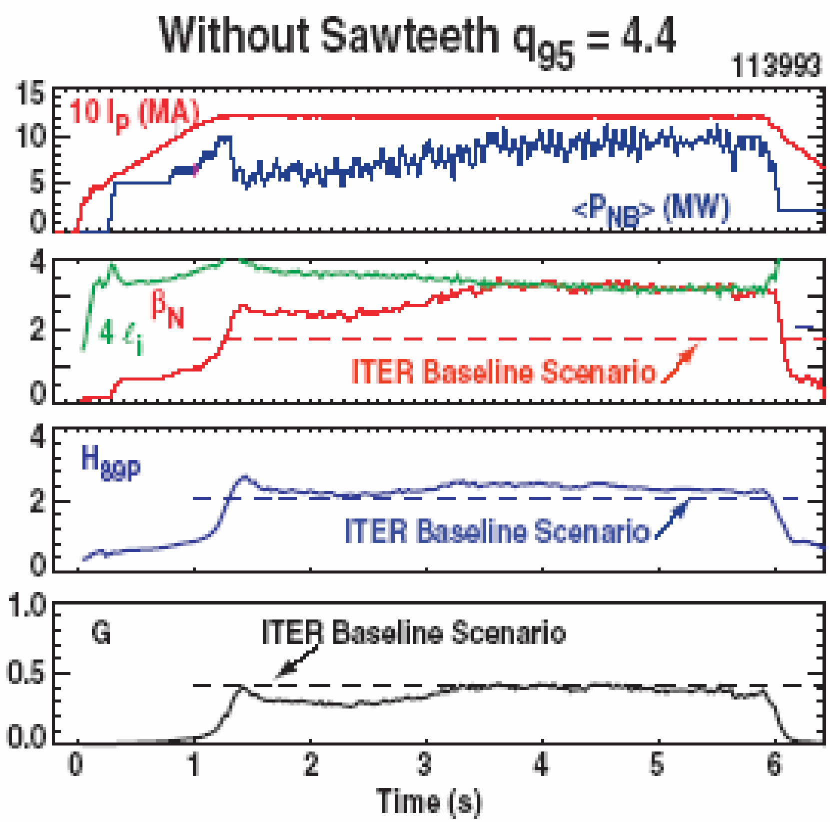


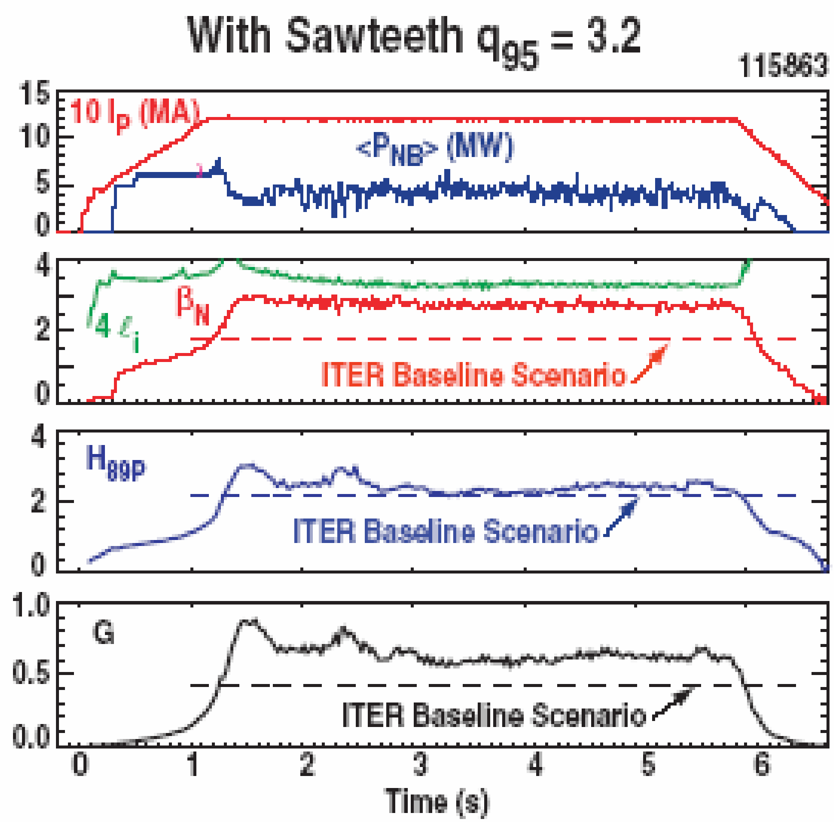
Bootstrap current →

Both the AIRES-I and PULSAR operating modes have demonstrated stationary high performance on DIII-D

ARIES-I like



PULSAR like



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Comparison Chart between PULSAR and ARIES-I'

Physics assumptions of the two first stability devices are the same (except non-inductive current-drive physics).

	PULSAR	ARIES-I'
Current-drive system	PF system very expensive, but efficient, separate system for heating	Non-inductive drive Expensive & inefficient, used also for heating
Recirculating power	Low	High due to RF
Optimum Plasma Regime	Moderate Bootstrap, High A, Low I	High Bootstrap, Higher A, Lower I
Current Profile Control	No, 30%-40% bootstrap fraction β_N~ 3, $\beta \sim 2.8\%$	Yes, 65-%-75% bootstrap fraction β_N~ 3.3, $\beta \sim 1.9\%$
Toroidal-field Strength	Lower because of interaction with cycling PF (B ~ 14 T on coil)	Higher (B ~ 16 T on coil)
Power Density	Low	Medium
Size and Cost	High (~ 9 m major radius)	Medium (~ 8 m major radius)
Energy Storage	Yes, Shield	No need
Disruptions	More frequent	fewer

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Many Critical Issues and dependencies have been uncovered by the ARIES Studies

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MHD Regime:

- **tradeoff β for I_{BS}/I_P (and alignment) and hence circulating power**
- **operate at 90% of β-limit to reduce disruption frequency**
- **severe constraints on close-fitting shell and n>0 feedback**
- **effect of ohmic-profiles on stable β in non-CD machine**
  - **has implications for ITER**

Plasma Shaping:

- **plasma elongation limited by control-coil power and location**
- **plasma triangularity restricted by divertor geometry**

Current Drive:

- **need for efficient off-axis CD (other than LHCD)**
- **CD frequency also important for wall-plug efficiency**
- **minimize coverage of RF launchers to avoid affecting tritium breeding**

Divertors:

- **radiated power needed to reduce power to divertor**

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Summary

- Both the ARIES-I and PULSAR designs are very close to the achieved physics data base
- Both steady-state and pulsed power plants tend to optimize at larger aspect ratio and low currents
- Even though the plasma β is larger in a pulsed tokamak, the fusion power density (wall loading,etc) would be lower because the magnetic field at the coil would be lower
- A major innovation of the PULSAR study is the low-cost thermal storage system using the outboard shield
- The magnet system and fusion power core are much more complex in a pulsed-plasma tokamak, but there is no CD system
- Assuming the same availability and unit costs, PULSAR is about 25% more expensive than a comparable ARIES-I class device
- These designs provide an important backup if the more aggressive "Advanced Tokamak" designs prove impractical

AST 558: Graduate Seminar - "Prospects for Fusion Energy"

February 7	A Brief History of Fusion and Magnetic Fusion Basics - Meade
February 14	Recent JET Experiments and Science Issues - Strachan
February 21	Advanced Tokamaks FIRE to ARIES - Meade
February 28	The ARIES Power Plant Studies – Jardin
March 7	IFE basics and NIF - Mark Herrmann(LLNL)
Midterms and Spring Break	
March 21	The FESAC Fusion Energy Plan - Goldston
March 28	Fusion with High Power Lasers – Sethian(NRL)
April 4	ITER Physics and Technology- Sauthoff
April 11	Stellarator Physics and Technology - Zarnstorff
April 18	“New” Mirror Approaches for Fusion - Fisch
April 25	ST Science and Technology – Peng
May 2	FRC Science and Technology - Cohen

Lesson # 1:
It's β/ε (i.e. $\beta R_0/a$) that's important, not β !

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MHD Theory

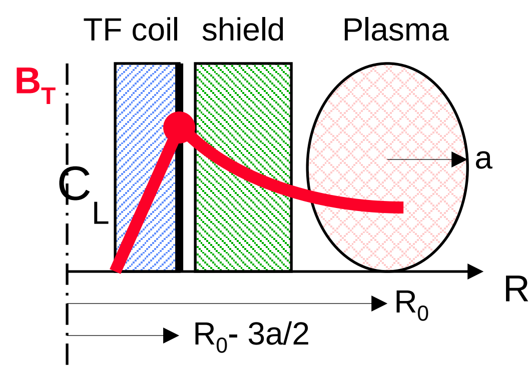
1. Large aspect ratio expansion of MHD perturbed energy δW shows that β enters only as β/ε (reduced MHD)

2. Troyon scaling may be written in dimensionless form as:

$$\beta/\varepsilon < C_T S/(20q^*)$$

Here , the right hand side is independent of ε. C_T = 3.5 is the Troyon coefficient, q^* > 2 is the cylindrical safety factor, and $S=(1+\kappa^2)/2$ is the shape factor.

SC Reactors



$B_T = \mu_0 I_{TF}/2\pi R$ is limited by it's value at the edge of the TF coil, $R \sim R_0$- 3a/2

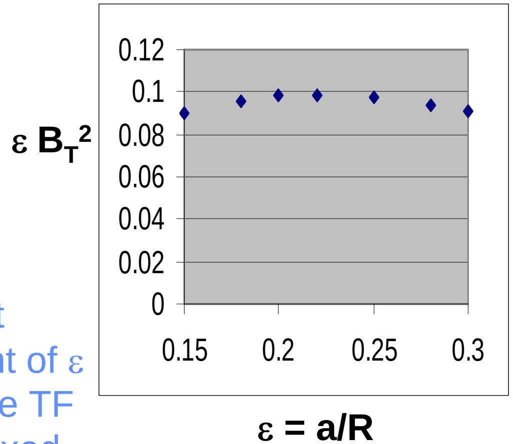
Power Density:

$P \sim \beta^2 B_T^4$

$= (\beta/\varepsilon)^2(\varepsilon B_T^2)^2$

MHD Figure of merit

Almost independent of ε for B_T at the TF coil held fixed



Lesson # 2:
Non-Inductive current drive is very costly !

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$$I_{CD} = \gamma_{CD} (P_{CD}/n_e R)$$

I_{CD} = Total non-inductively driven current (A)

P_{CD} = Power to plasma by CD system (W)

n_e = average density (in units of $10^{20}/m^3$)

R = major radius (m)

γ_{CD} = CD figure of merit

- **Theoretical calculations show $\gamma_{CD} \alpha T_e^n$ with 0.6 < n < 0.8**
- **Highest values to date for γ_{CD} are 0.45 (JET with ICRF+LH) and 0.34 (JT-60 with LHCD). Note that for a Reactor with I_P=20 MA, n_e = 1.5 x 10^{20}, R = 8 m, γ_{CD} = 0.34, this gives**

P_{CD} = 700 MW to the plasma.

- **This is unrealistic for a 1000 MW Power plant, since wall plug power is much higher (several efficiencies involved)**

♏ **most of the plasma current must be self-generated (bootstrap) for a non-inductive reactor**