Heavy Water, Gas and Liquid Metal Cooled Reactors

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22.06: Engineering of Nuclear Systems

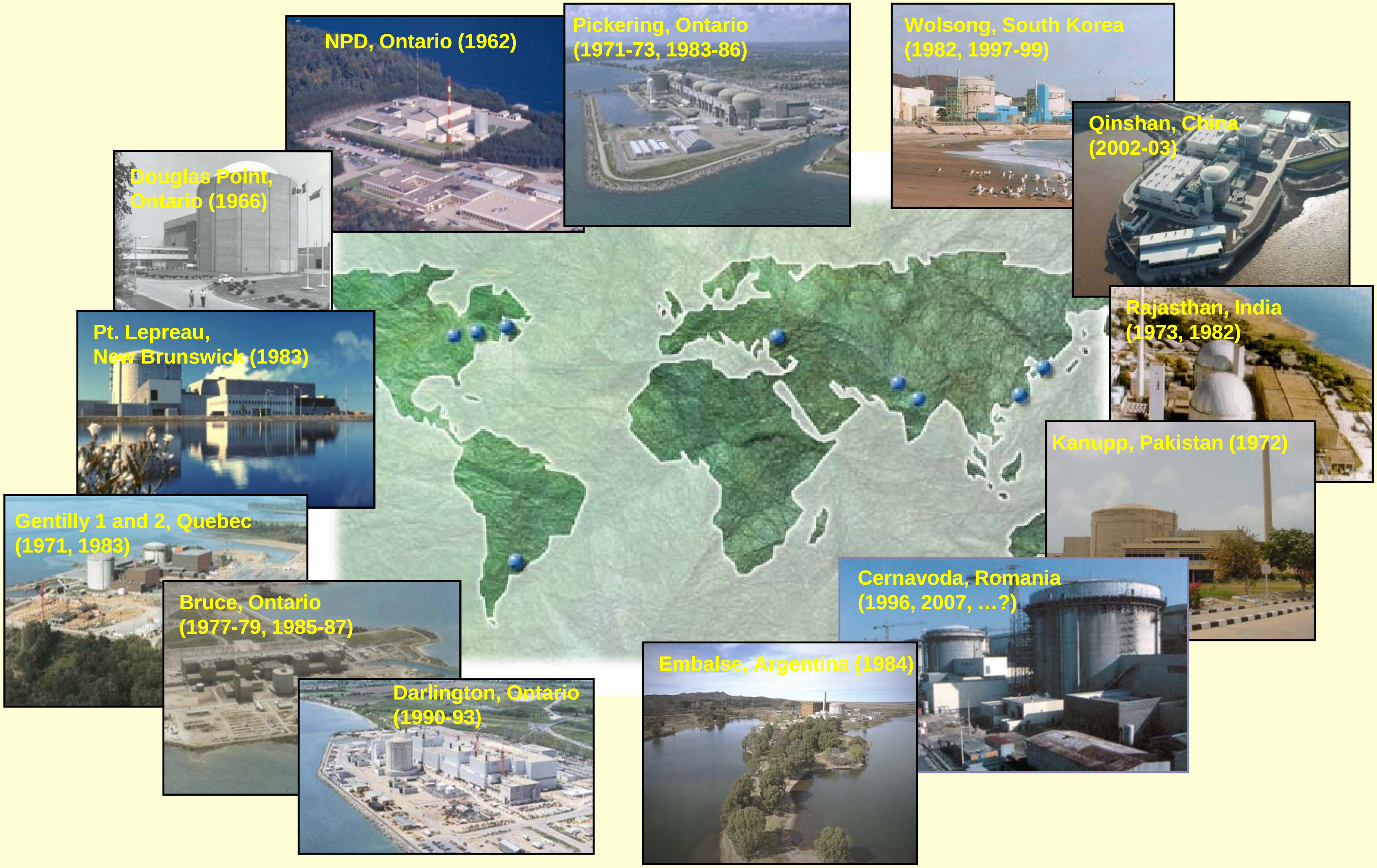
CANES CENTER FOR ADVANCED NUCLEAR ENERGY SYSTEMS

Heavy Water Cooled Reactors (CANDU)

Key CANDU Features

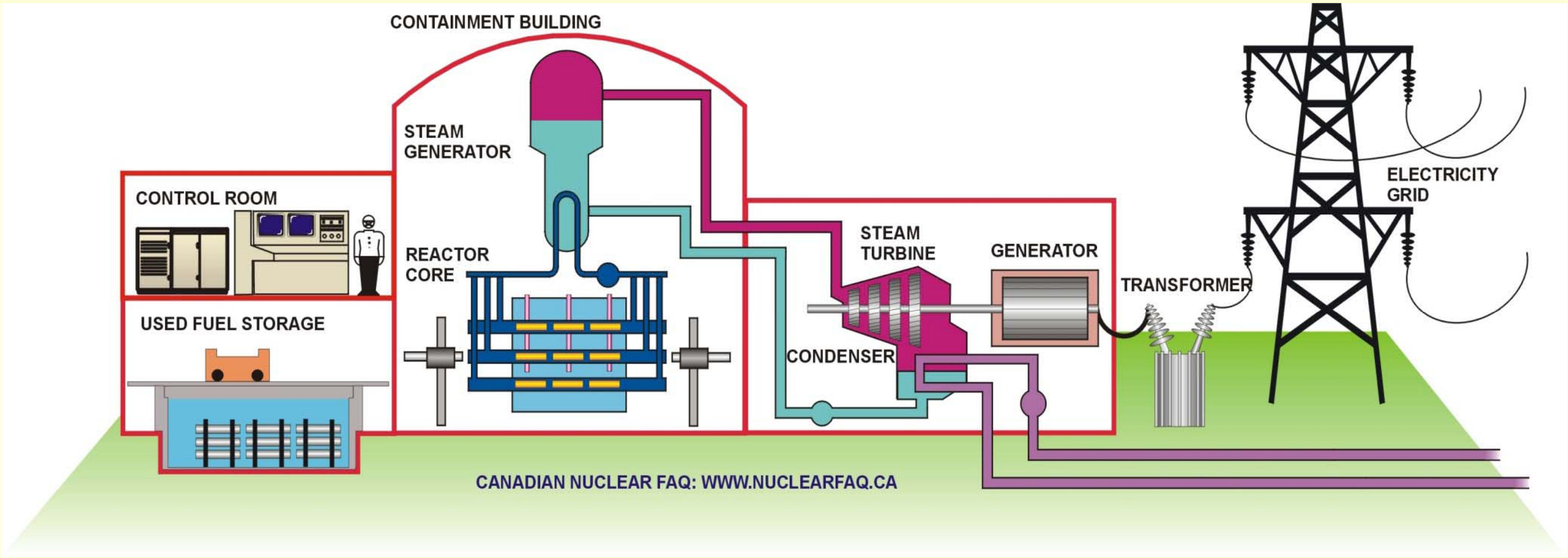
- CANada Deuterium Uranium
- Designed for natural uranium fuel (no enrichment needed)
- Heavy water (D_2O) moderated
- Pressure tube reactor (no pressure vessel)
- Moderator & coolant separated
- Pressurized coolant and steam generators (similar to PWR)
- On-power refuelling
- High resource utilization (150 tons mined uranium per GW_eyr, compared to 200 tons mined U per GW_eyr for a typical PWR)

Source: Jeremy Whitlock, AECL Chalk River Labs, 4/16/07



20 in Canada **12 offshore**

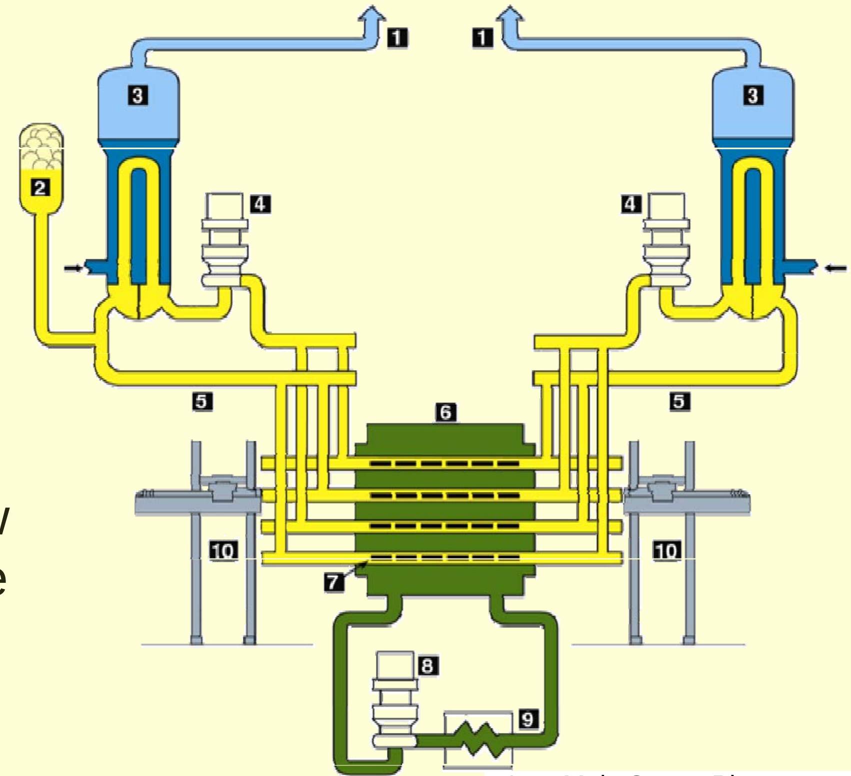
CANDU STATION OVERVIEW



Power cycle similar to PWR and BWR

CANDU PRIMARY SYSTEM

- Natural uranium fuel and D_2O moderator
- Fuel contained in individual fuel channels (pressure tubes) filled with high pressure (>10 MPa) and high temperature (~300°C) D_2O coolant
- Pressure tubes contained in a large cylindrical tank (calandria) filled with low pressure (<1 MPa) and low temperature (<80°C) D_2O moderator
- Fuel clad and pressure tubes are made of Zr alloys
- Fuelling machines connect to individual pressure tubes for refuelling
- Conventional turbine/generator and auxiliary systems



Steam
Feedwater
Heavy Water Coolant
Heavy Water Moderator

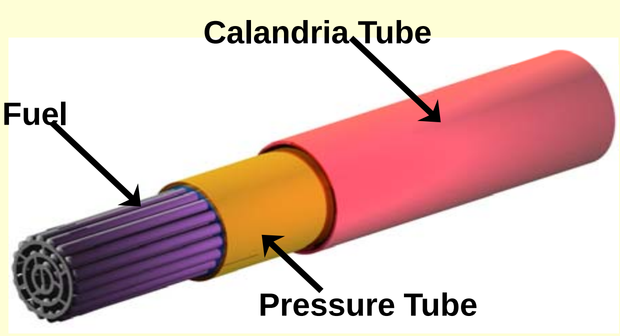
1 Main Steam Pipes
2 Pressurizer
3 Steam Generators
4 Heat Transport Pumps
5 Headers
6 Calandria
7 Fuel
8 Moderator Pumps
9 Moderator Heat Exchangers
10 Fuelling Machines

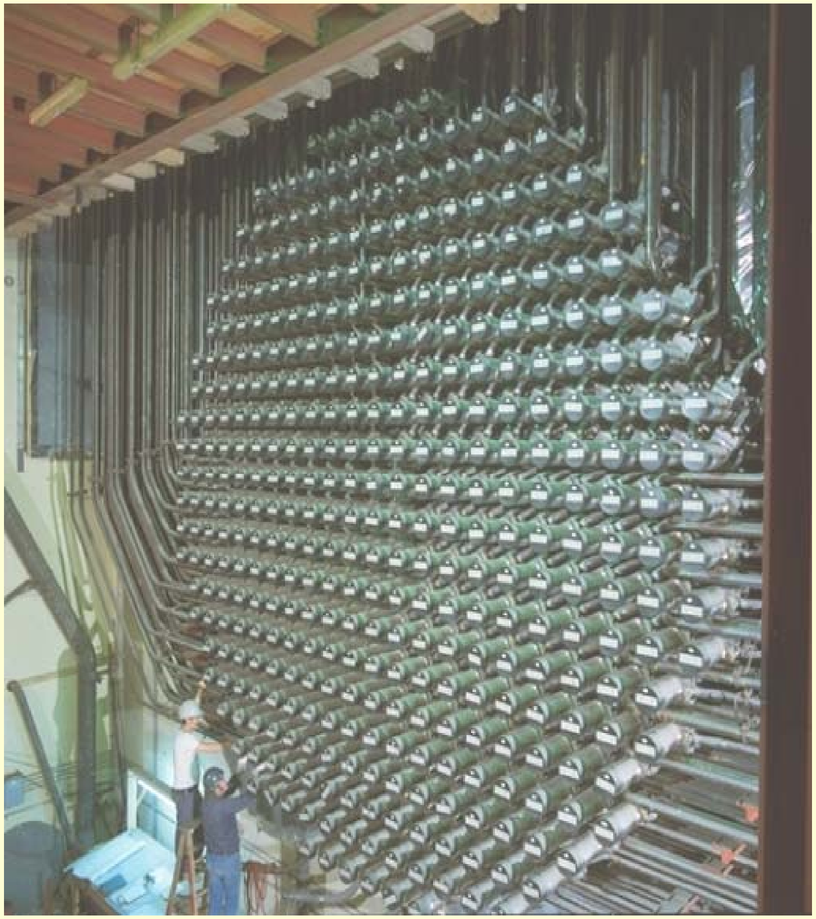
CANDU FUEL BUNDLES

- UO_2 pellets in Zircaloy cladding (0.38 mm thick)
- 28 or 37 pins form a fuel bundle (pins have a 13.08 mm outside diameter)
- Pins held together by end plates.
- Pins separated by spacers. Outer pins have bearing pads.
- Bundles: 495.3 mm long and 102.5 mm in diameter
- Average burnup: 7500-8500 MWd/ton)

PRESSURE TUBES (OR FUEL CHANNELS)

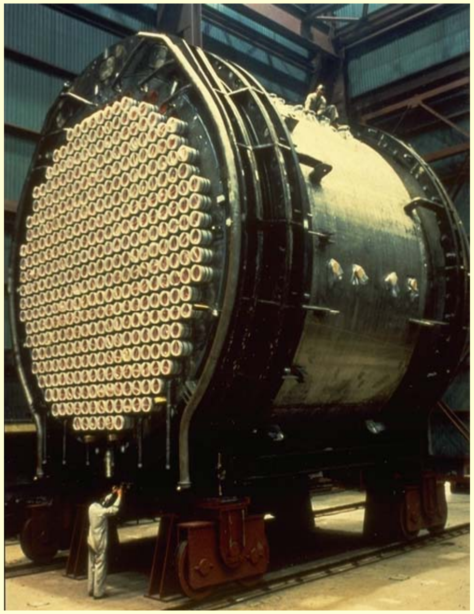
- Each fuel channel consists of a pressure tube and two end-fittings (primary pressure boundary), plus a calandria tube
- Pressure tube - calandria tube separated by a gas-filled annulus; gap maintained by “garter” springs
- Low neutron cross section
- Total channel length: 11.56 m (~6 m fuelled)

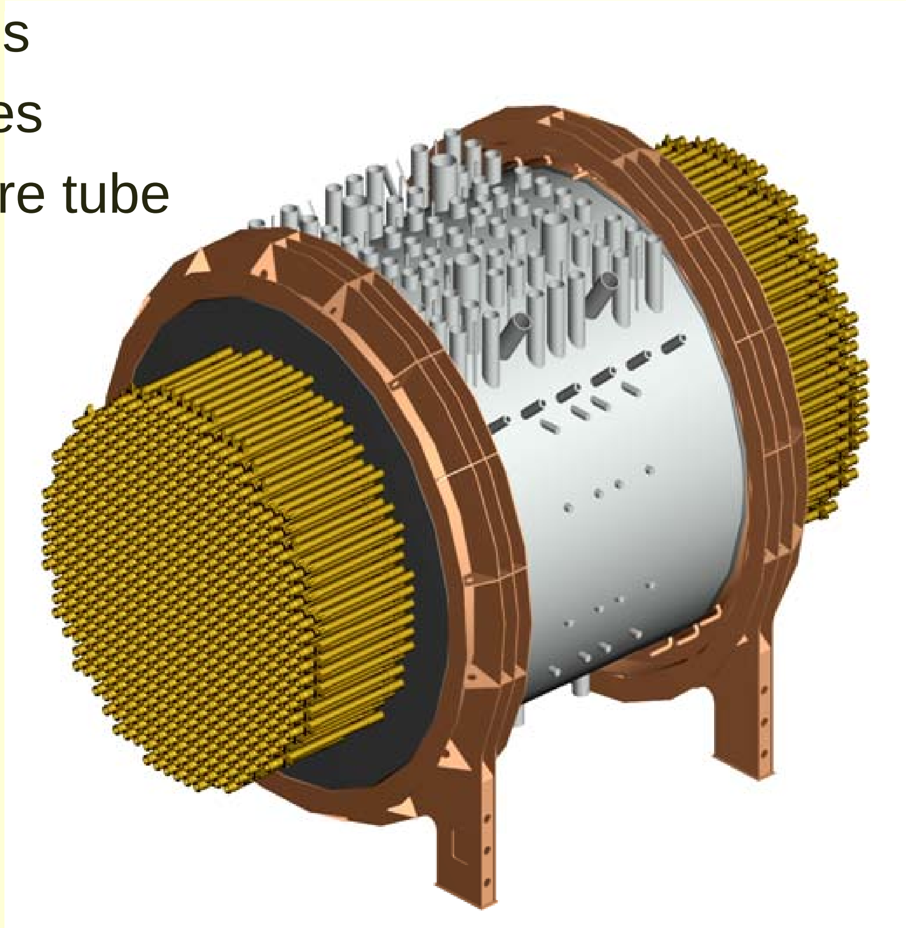




CALANDRIA ASSEMBLY

- Holds the heavy water moderator
- Penetrated horizontally by pressure tubes, and vertically by reactivity devices
- 380-480 horizontal pressure tubes
- 12 or 13 fuel bundles per pressure tube
- Not a pressure vessel





REPRESENTATIVE PARAMETERS FOR ADVANCED CANDU (ACR-700)

Parameter	Value
Thermal power (MWth)	1980
Gross electric power (nominal) (MWe)	731
Reactor pressure (MPa)	12.6
Nominal coolant inlet temperature (^{o}C)	279
Nominal coolant outlet temperature (^{o}C)	325
Nominal moderator temperature (^{o}C)	74
Length of fuel bundle (mm)	495.3
Core length (mm)	5940
Number of bundles per fuel channel	12
Number of fuel channels (Pressure tubes)	284
Pressure tube inner radius (mm)	51.689
Pressure tube outer radius (mm)	58.169
Number of fuel elements per channel	43
Pressure tube lattice pitch (mm)	220

Connection of CANDU Core Design to Neutronics

What enables a CANDU reactor to operate with natural uranium?

What determines the pressure tube spacing?

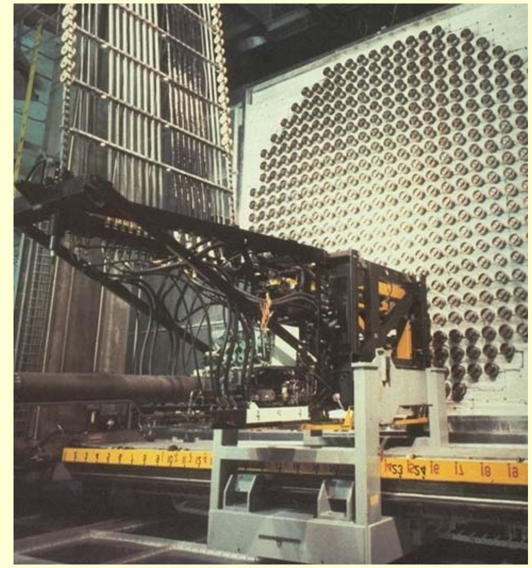
Is the power density in a CANDU core <, = or > than a PWR?

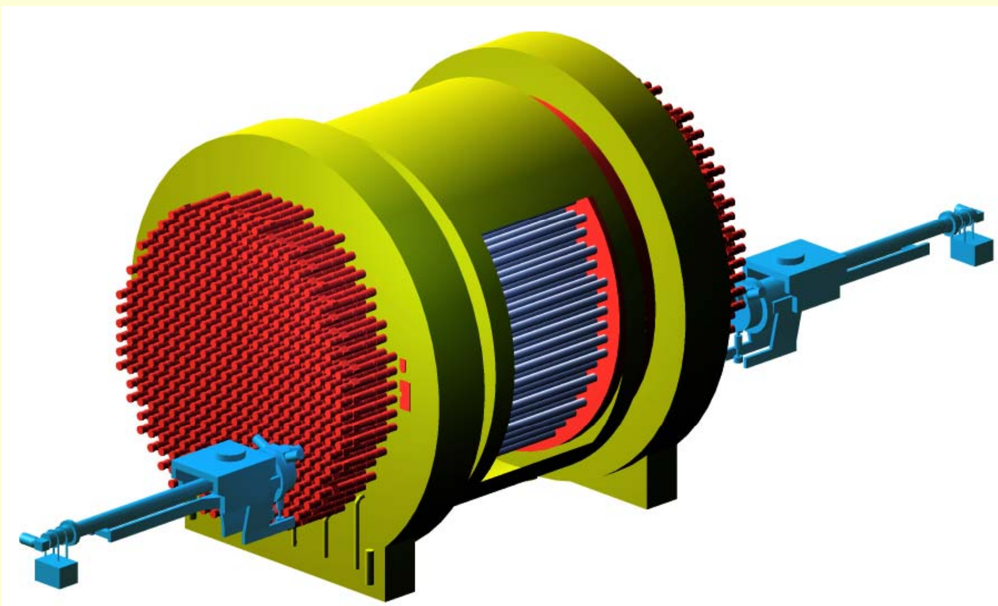
What would happen if the calandria tank were drained?

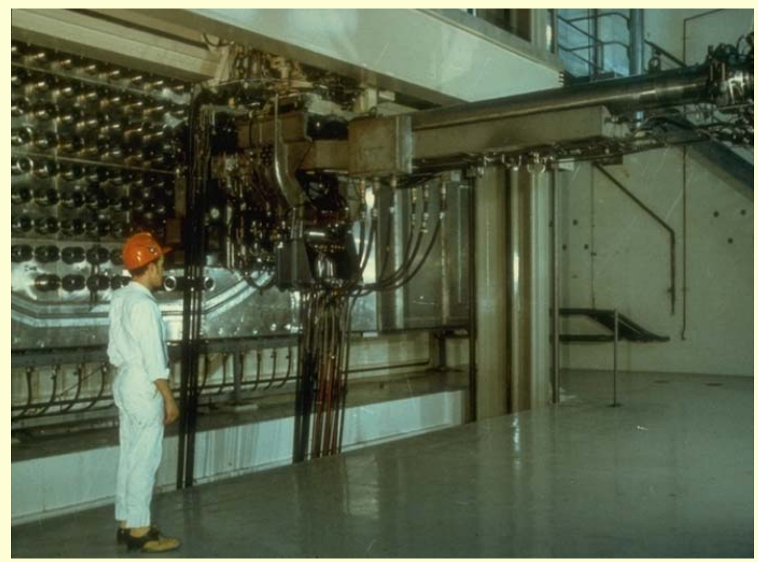
What happens to reactivity if some voiding (boiling) occurs in a CANDU pressure tube?

FUELLING MACHINES

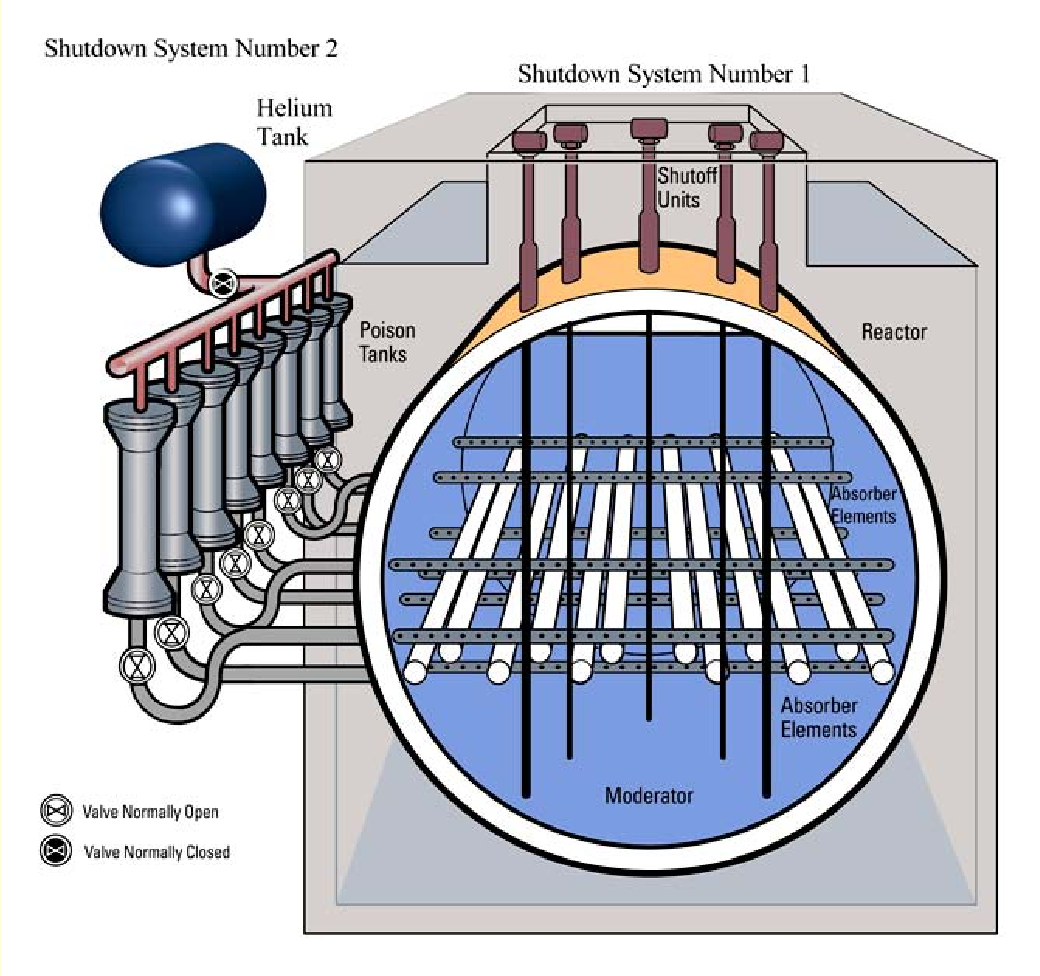
- Two fuelling machines operate simultaneously accepting or loading fuel
- Remotely operated from control room







TWO FAST-ACTING SHUTDOWN SYSTEMS



High Temperature Gas Reactors (HTGR)

HTGR Overview

- Small modular units: 125-300 MWe
- Helium cooled, 850-900°C outlet T, <9 MPa pressure
- Thermal efficiency >40%
- Graphite moderated
- Microsphere UO_2 or UCO fuel
- Electricity and process heat
- Passive decay heat removal
- Two “flavors”: block core or pebble bed

Block Core HTGR

TRISO fuel particle

TRISO Coated fuel particles (left) are formed into cylindrical fuel compacts (center) and inserted into hexagonal graphite fuel elements (right).

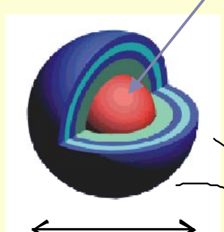
TRISO PARTICLES

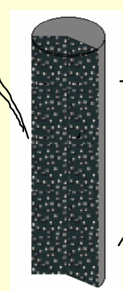
CYLINDRICAL COMPACTS

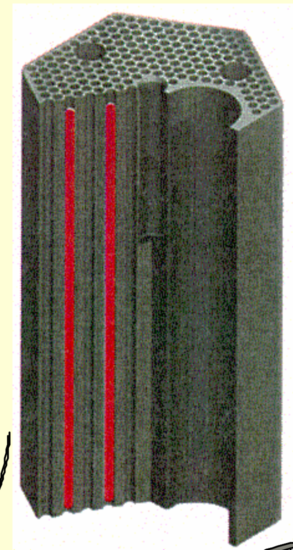
HEXAGONAL FUEL ELEMENTS

Block Core HTGR (2)

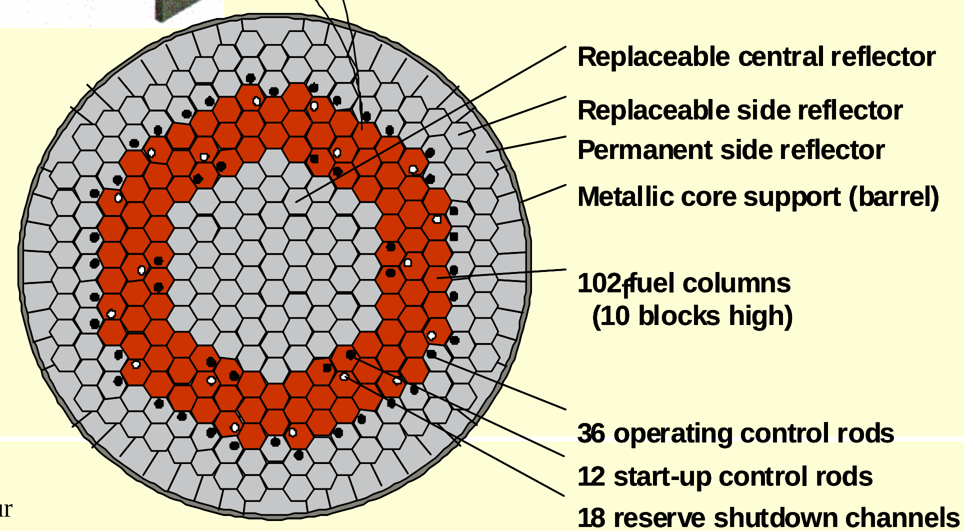
U 17-19% enriched
Requires mixed enrichment, burnable poisons







80-cm tall blocks stacked
Core 10 blocks high
102 columns of fuel



Block Core HTGR (3)

Being developed by AREVA, General Atomics and Japan.

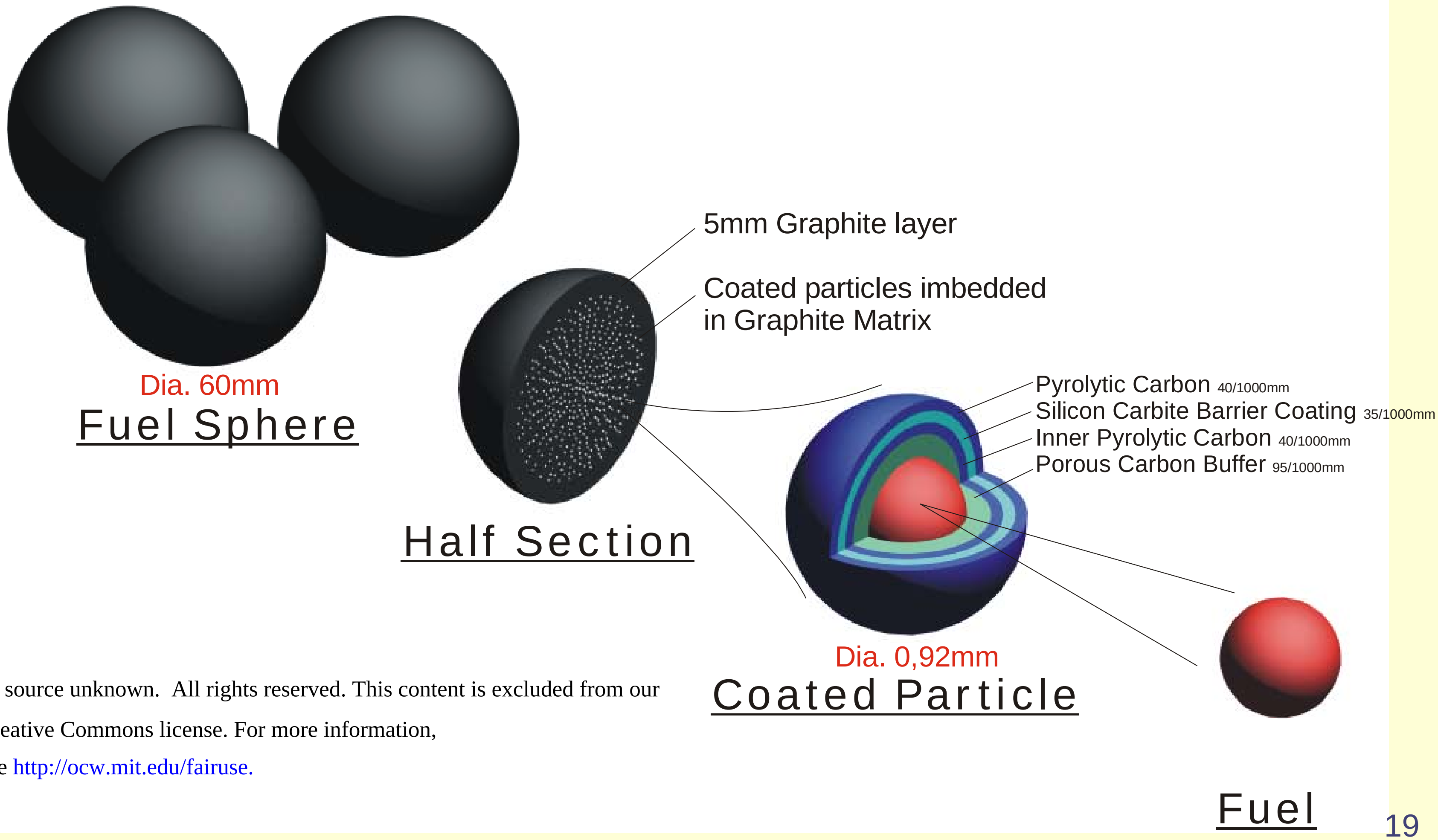
Experience in US (Ft. Saint Vrain) and Japan (HTTR)

330 MWe
Operated from 1979 to 1989
U/Th fuel
Poor performance, mechanical problems, decommissioned

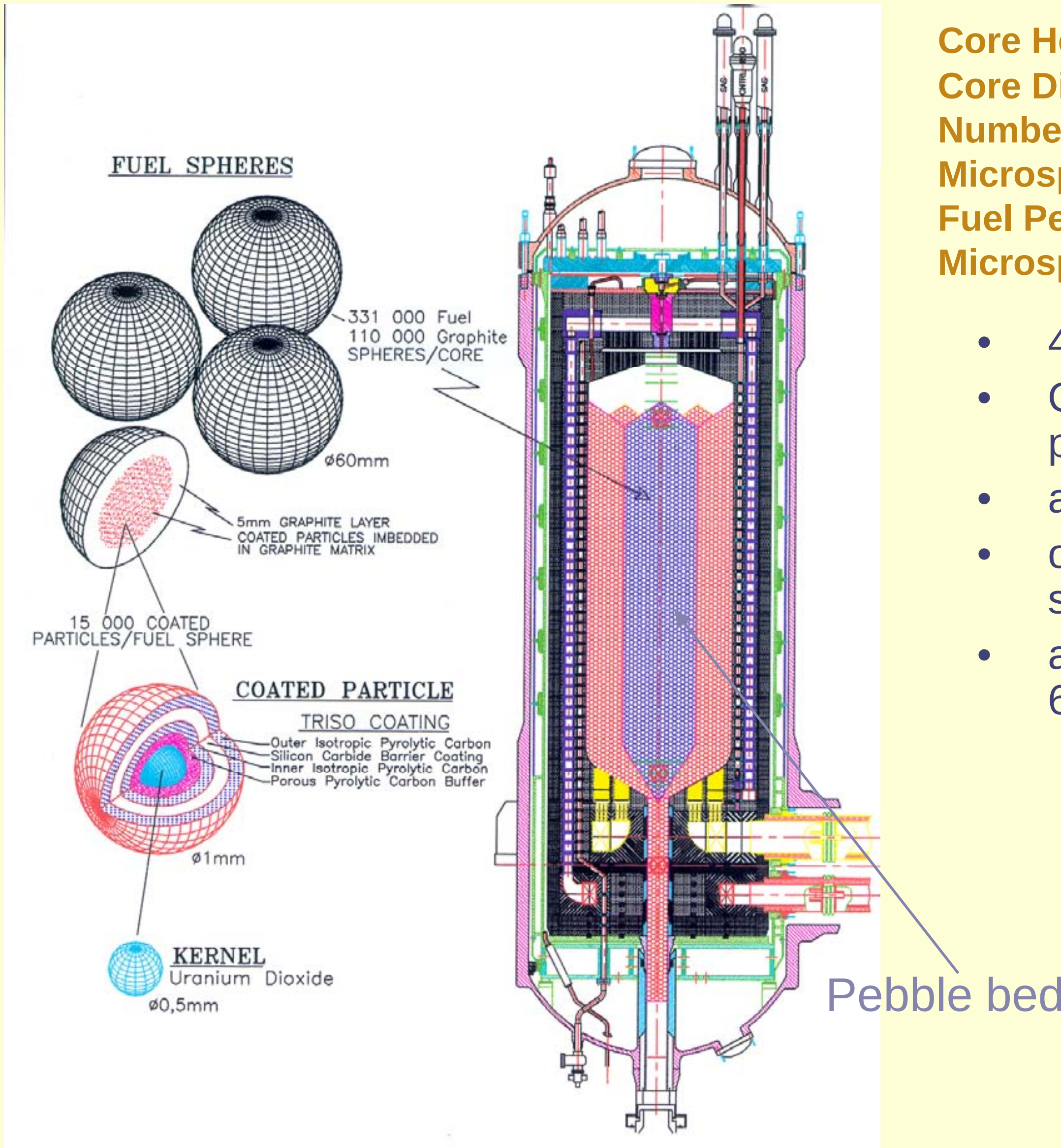
40 MWth Test Reactor at JAERI
First Critical 1999
Intermediate Heat Exchanger
Currently in Testing for Power Ascension

Pebble Bed HTGR

FUEL ELEMENT DESIGN FOR PBMR



Pebble Bed HTGR (2)



Core Height	**10.0 m**
Core Diameter	**3.5 m**
Number of Fuel Pebbles	**360,000**
Microspheres/Fuel Pebble	**11,000**
Fuel Pebble Diameter	**60 mm**
Microsphere Diameter	**~ 1mm**

- 400,000 pebbles in core
- Online refueling, about 3,000 pebbles handled each day
- about 350 discarded daily
- one pebble discharged every 30 seconds
- average pebble cycles through core 6 times

•Enrichment 8 - 9% - constant - no burnable poisons
•Low excess reactivity - lower peak operating temperatures (200°C lower)

Pebble Bed HTGR (3)

Being developed by PBMR Ltd. and China.

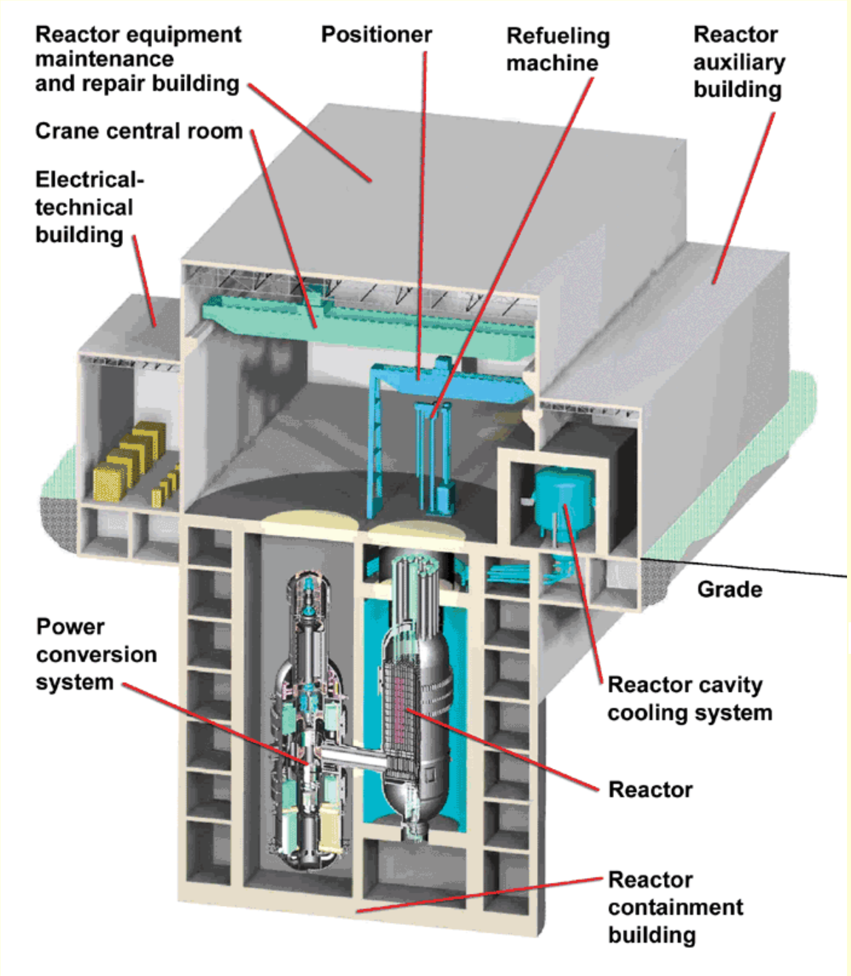
Experience in Germany (AVR, THTR) and China (HTR-10)

15 MWe research reactor
UO_2 fuel
Operated for 22 years

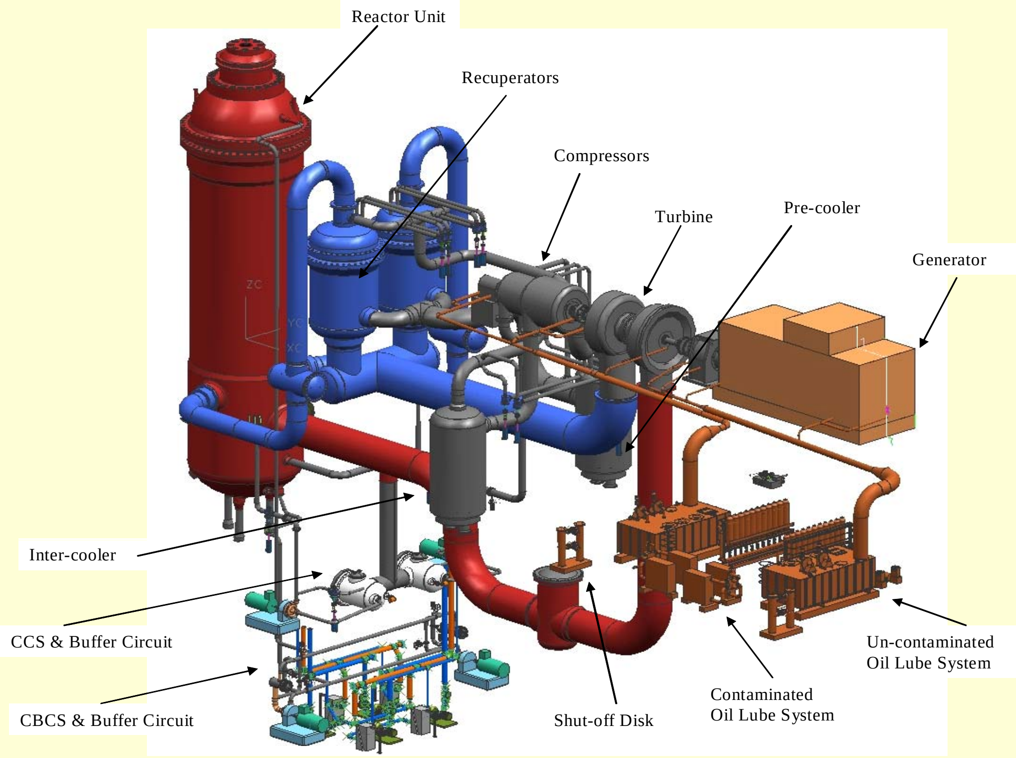
300 MWe demo plant at
Hamm-Uentrop
U/Th fuel

10 MWth - 4 MWe Electric
First criticality Dec 1, 2000
Intermediate Heat Exchanger
- Steam Cycle

HTGR Layouts – Direct Cycle



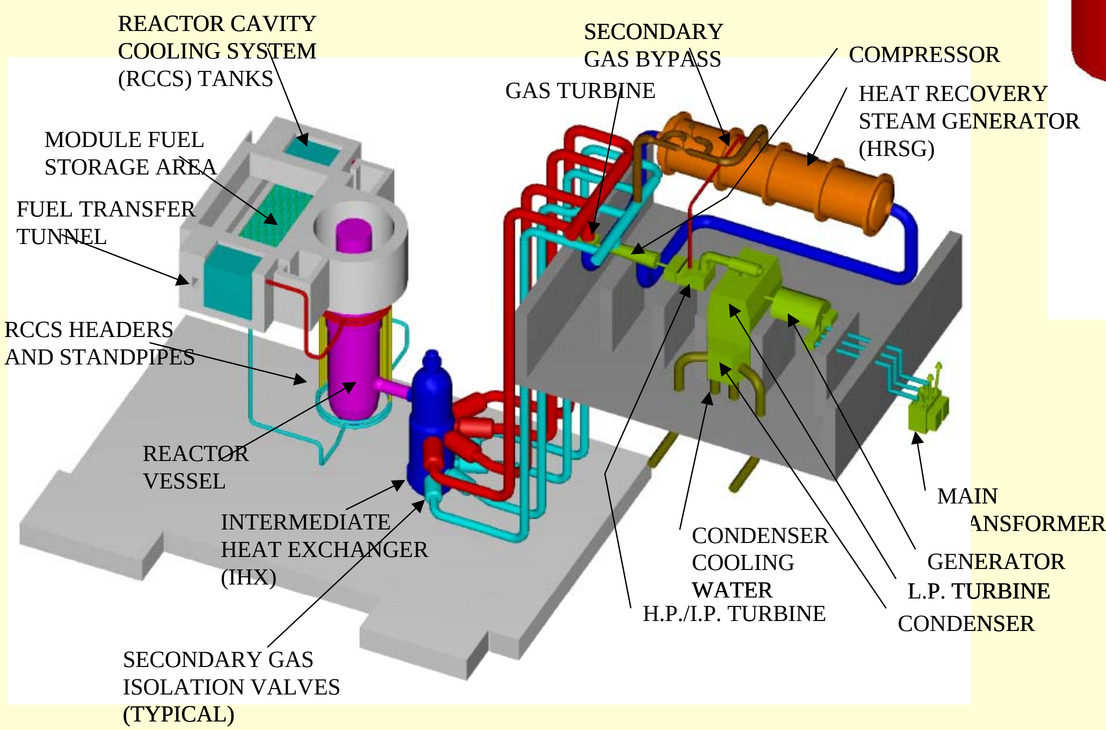
General Atomics, block core, vertical turbo generator

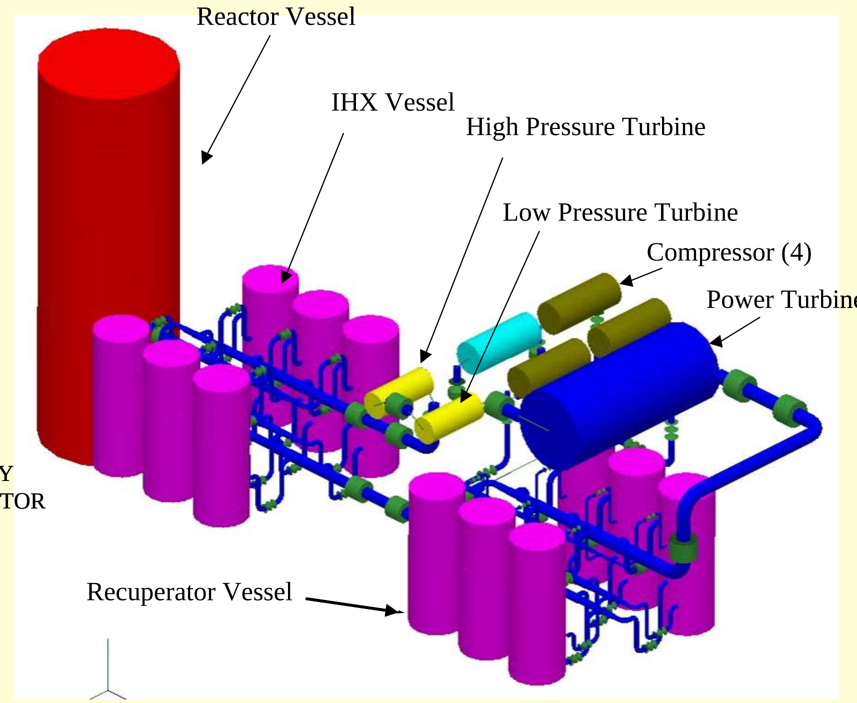


PBMR Ltd., pebble bed core, horizontal turbo-generator

HTGR Layouts – Indirect Cycle

AREVA, block core, combined cycle





MIT, pebble bed core, 3-shaft turbo-generator

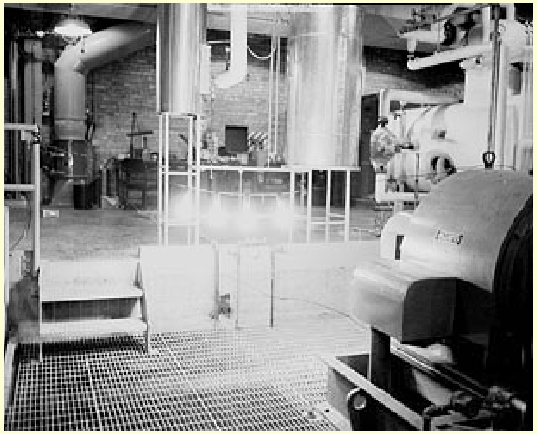
HTGR Safety

- No fission product release from TRISO fuel at up to 1600°C
- Low power density (due to use of graphite moderator) makes it possible to remove decay heat by radial conduction and radiation
- In case of unprotected Loss of Coolant Accident (LOCA) with loss of on-site and off-site power (a very serious event) there is no fuel melting
- Concerns for air ingress (graphite “burns” at high temperature)

Liquid Metal (Sodium) Cooled Fast Reactors

Fast Reactors – the concept

- Fast reactor is a system in which neutrons are not moderated
- The number of neutrons emitted per neutron absorbed is higher for fast fissions, so the extra neutrons can be absorbed in a U-238 blanket to produce Pu-239, thus “breeding” new fuel
- If properly designed, fast reactors can actually breed more fuel than they consume (multiple fuel recycles become possible)
- Needs a coolant that does not moderate neutrons, typically a liquid metal such as sodium
- Interestingly, the first nuclear reactor to produce electricity was a fast reactor in Idaho.

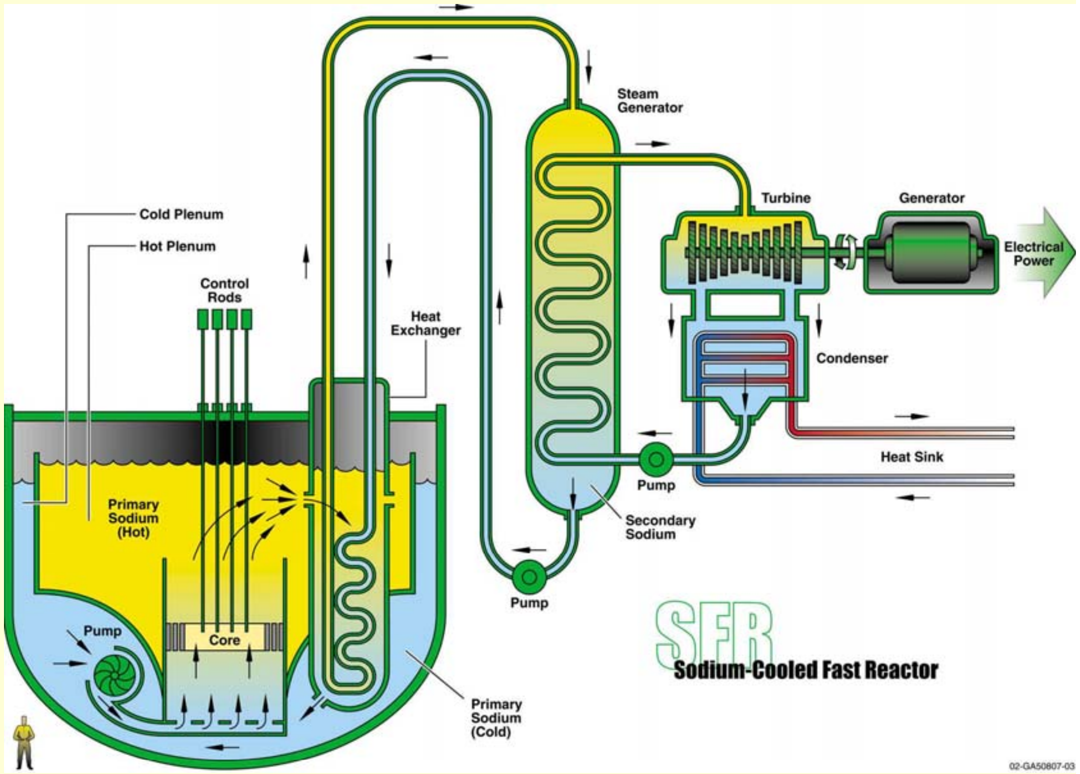


EBR-I

Sodium-Cooled Fast Reactor (SFR)

Characteristics

- **Sodium coolant**
- **>500°C Outlet Temp**
- **150 to 1300 MWe**
- **Metal or oxide fuel possible**



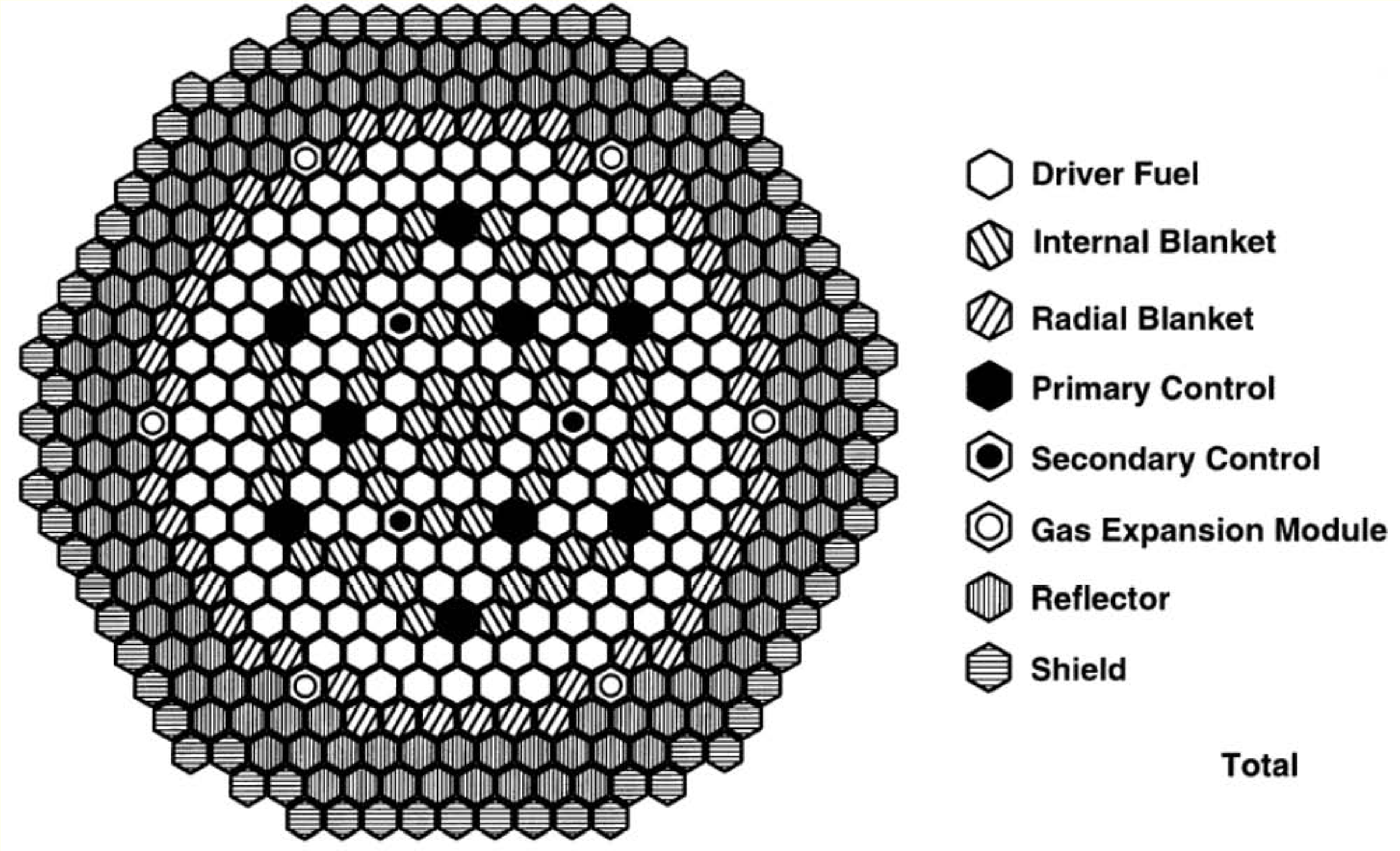
Benefits

- **Efficient fissile material generation (breeding)**
- **Sodium is excellent heat transfer fluid and has high boiling point (880°C)**
- **Relatively high temperature (good for efficiency ~40%), but low pressure system (good for safety)**

Drawbacks

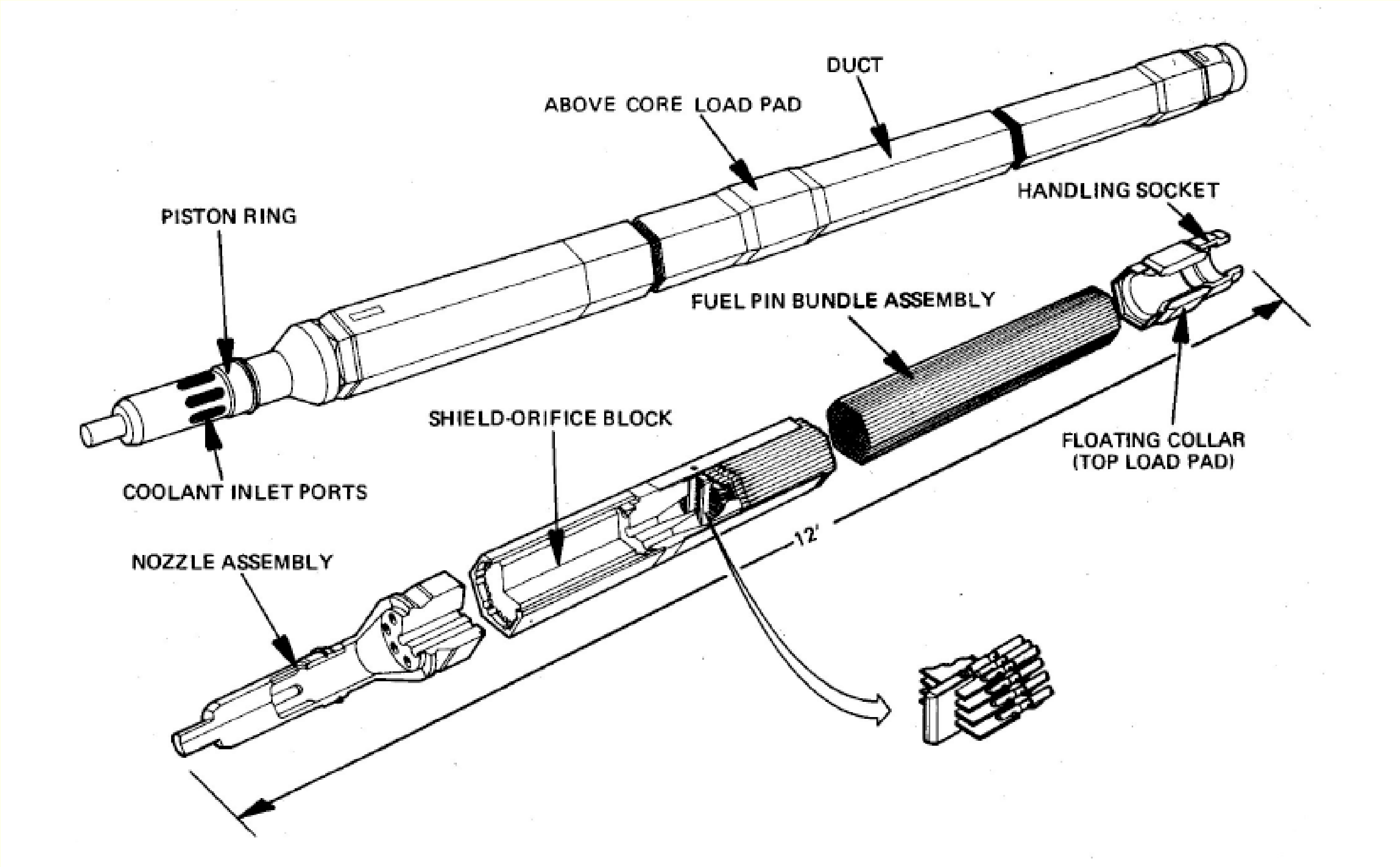
- **Sodium is reactive with air and steam, hence the intermediate loop and special fire protection measures, which add to cost and complexity**
- **Requires higher initial enrichment to get started (why?)**
- **Has positive void reactivity feedback (why?)**
- **Generates weapons-grade Pu (proliferation concern)**

SFR Core



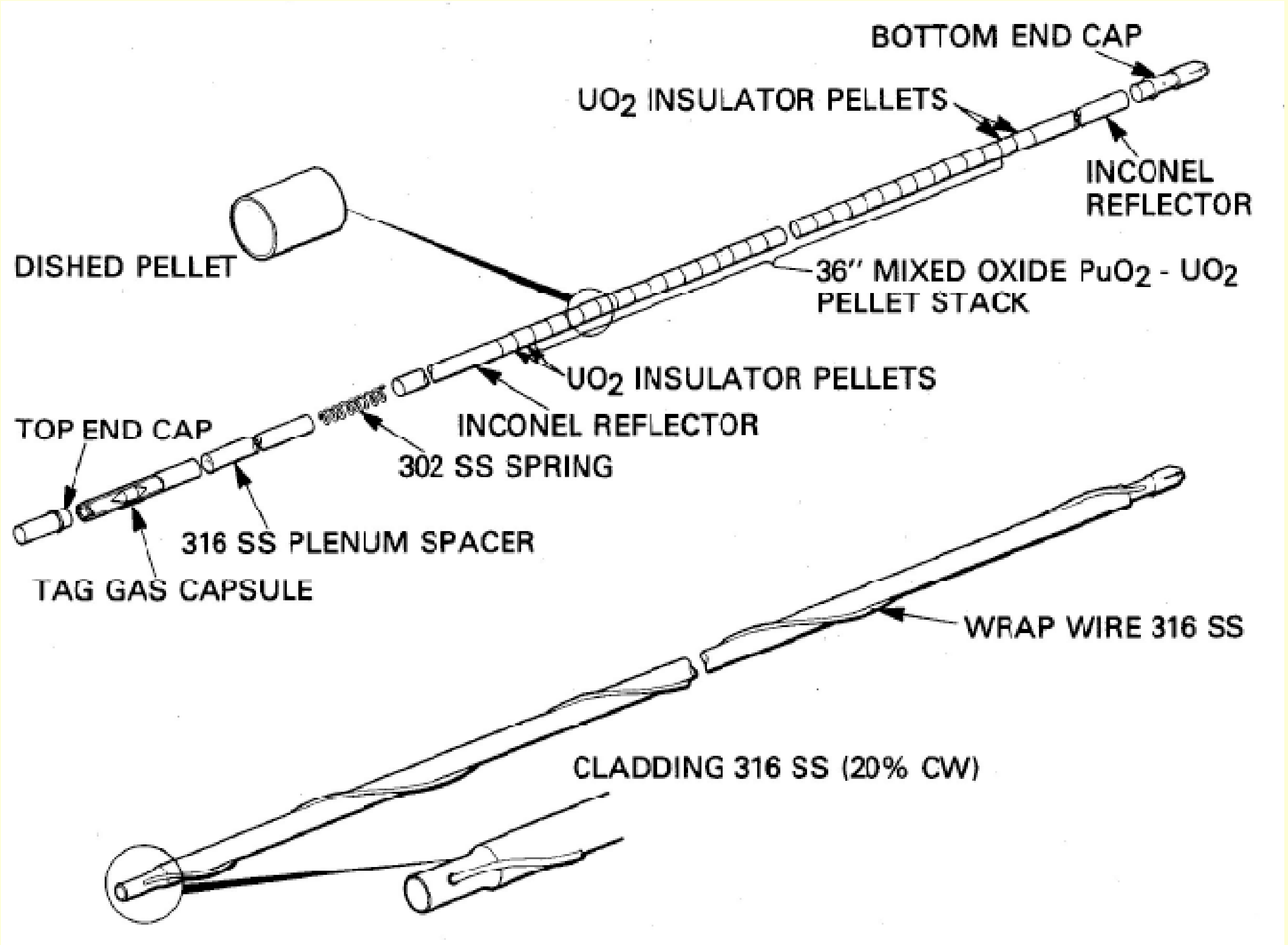
- Uses U-238 blankets to absorb neutrons that escape from the driver fuel core region

SFR Fuel Assembly



- Hexagonal fuel assemblies with duct

SFR Fuel Rod



Very tight lattice requires use of wire wrap to keep fuel rods separated

Representative parameters for SFR core

- Data for GE's SuperPRISM, 1000 MWt core, T_{in}=371°C, T_{out}=510°C

SuperPRISM Fuel and Blanket Assembly Cross-Section Dimensional Data

Fuel Type	Oxide				Metal			
Assembly type	Fuel		Blanket		Fuel		Blanket	
	(in)	(mm)	(in)	(mm)	(in)	(mm)	(in)	(mm)
Assembly pitch	6.355	161.42	6.355	161.42	6.355	161.42	6.355	161.42
Duct gap	0.170	4.32	0.170	4.32	0.170	4.32	0.170	4.32
Duct wall thickness	0.155	3.94	0.155	3.94	0.155	3.94	0.155	3.94
Load pad gap	0.010	0.25	0.010	0.25	0.010	0.25	0.010	0.25
Pin count	**217**		**127**		**271**		**127**	
Pin outer diameter	0.335	8.51	0.473	12.01	0.293	7.44	0.473	12.01
Pin cladding wall thickness	0.0250	0.635	0.0220	0.559	0.022	0.559	0.022	0.559
Fuel outer diameter	0.2779	7.059	0.4236	10.759	0.2156	5.477	0.3955	10.046
Pin spacer type	**SSWW***		**SSWW***		**SSWW***		**SSWW***	
Spacer pitch	8.0	203.2	8.0	203.2	8.0	203.2	8.0	203.2
Spacer wire diameter	0.055	1.397	0.037	0.940	0.056	1.422	0.037	0.940
Fuel fabrication density (% of Theoretical density)	89.4		95.4		100.0		100	
Fuel smeared density (% of Theoretical density)	85		93		75		85	
Volume fractions (%, Before irradiation)								
Fuel	37.63		51.17		28.30		44.61	
Bond (Fuel-cladding annulus)	1.95		1.32		9.43		7.87	
Coolant	34.57		26.54		36.57		26.54	
Structure	25.85		20.97		25.70		20.97	

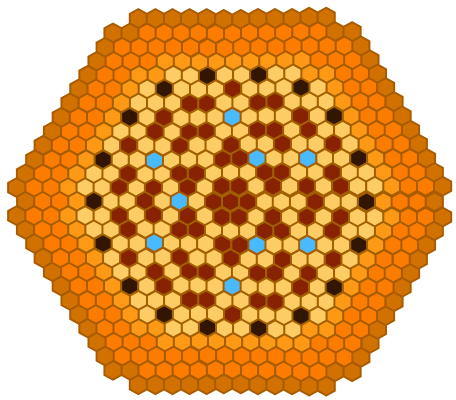
*SSWW Straight start wire wrap

Image by MIT OpenCourseWare.

From Dubberley et al., Proc. of ICONE-8, Baltimore, 2000

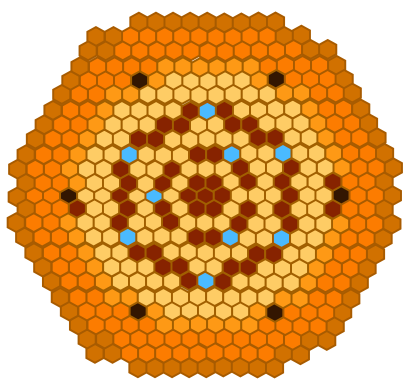
Representative parameters for SFR core (2)

Driver fuel	162
Internal blanket	73
Radial blanket	60
Primary control	9
Secondary control	3
Gas expansion module	18
Reflector	138
Shield	78
Total	541



SuperPRISM oxide core configuration

Driver fuel	138
Internal blanket	49
Radial blanket	48
Primary control	9
Secondary control	3
Gas expansion module	6
Reflector	126
Shield	72
Total	451



SuperPRISM metal core configuration

Image by MIT OpenCourseWare.

	Cost optimized MOX	Cost optimized metal	High breeding MOX	High breeding Metal	Limit
Cycle average breeding ratio	1.03	1.05	1.17	1.22	
Cycle burnup reactivity loss (% dk/kk')	0.98	0.12	0.81	-0.31 (gain)	3.4
Core inventory at BOC *Fissile PU (kg - kg/MWt)* *Total TRU (kg)* *Total U (kg)*	 3469.4 - 3.47 5207.7 29718.5	 2336.1 - 2.34 3078.2 23014.2	 3612.2 - 3.61 5341.0 45939.5	 2458.8 - 2.46 3195.9 33052.7	
Feed enrichment (wt.%, Total Pu in U-TRU)	29.81	21.42	29.61	21.29	33
Supplied fissile Pu - kg/year - kg/GWDt	411.20 1.32	366.16 1.18	408.40 1.32	363.97 1.17	
Fissile Pu gain (kg/year)	11.15	19.25	57.10	69.91	
TRU consumption rate (kg/year)	-38.60 (gain)	-33.60 (gain)	-85.60 (gain)	-84.63 (gain)	
Cycle average spatial power peaking factor	1.54	1.41	1.54	1.42	
Average linear power (kW/m, Cycle average)	15.97	18.90	15.66	18.32	
Peak linear power (kW/m) - Fuel - Internal blanket - Radial blanket	30.14 27.16 17.76	30.42 40.25 30.70	29.65 26.45 17.33	29.77 38.30 29.80	40
Peak neutron flux (10^{15} n/cm^2-s) - Total - Fast	2.38 1.38	3.62 2.47	2.33 1.36	3.49 2.37	
Average fuel burnup (MWd/kg) Peak fuel burnup (MWd/kg) Peak fast fluence, fuel-blanket (10^{23} n/cm^2)	116 178 2.96 - 2.44	106 149 3.71 - 3.90	114 175 2.91 - 2.40	103 145 3.61 - 3.79	 180 4.0
Core thermal hydraulic performance *Pressure drop (MPa)* *Maximum assembly outlet temp. (C)* *Maximum subchannel coolant temp. (C)* *Thermal striping potential (C)* *Thermal constraints satisfied* *GEM full-core stroke*	Good 0.31 619 678 197 Yes Yes	Good 0.41 595 648 189 Yes Yes	Good 0.31 620 679 197 Yes Yes	Good 0.43 594 648 194 Yes Yes	 0.48 621 887 206
Peak fuel pin steady state performance (HT9M) *Maximum creep rupture damage fraction* *Maximum total diametrial growth (%)* *Maximum thermal creep strain (%)* *Minimum power to melt at centerline (%)* *Maximum power to melt at surface (%)*	Good 0.0026 0.69 0.37 150	Good 0.00003 0.42 0.07 138 113	Good 0.0023 0.76 0.37 150	Good 0.00006 0.49 0.08 133 113	 0.2 2.0 1.0
Duct structural performance (HT9) *Maximum radial growth (mm)*	Good 1.7	Acceptable 2.3	Good 1.2	Acceptable 2.2	2.2 (Cons) 3.2 (Exp)

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From Dubberley et al., Proc. of ICONE-8, Baltimore, 2000.

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