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Overview of Nuclear Reactor Thermal hydraulic

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Follow Up Training Course on Reactor Engineering and Safety: High-Temperature Gas-Cooled Reactor

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Biodata

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- Pendidikan:
  - ✓S1 Teknik Energi: Georg Simon-Ohm, Nurnberg, Germany 1997
  - ✓S2 Teknik Mesin - Konversi Energi: Universitas Indonesia, 2024

- Pelatihan / Training:
  - ❖Longterm Training on Nuclear Safety and Safety Regulation 1997, Japan, 3 months
  - ❖Scientist Exchange Program, Japan, 2000 - 2001
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  - ❖Instructor Training Course, Japan, 2 months - 2011

Outlines

- Basic concept of nuclear reactor thermal hydraulic
- Overview of PWR and BWR Cooling System
- Thermal Hydraulic Safety Margin in PWR and BWR Core
- Analysis Tool and Method, Experiences in TH analysis

Basic concept of nuclear reactor thermal hydraulic

- Three fundamental knowledges:
  - Thermodynamics
  - Fluids mechanics
  - Heat transfer
- how **heat is generated, transferred, and removed** inside a nuclear reactor using flowing coolant.

Main purposes:

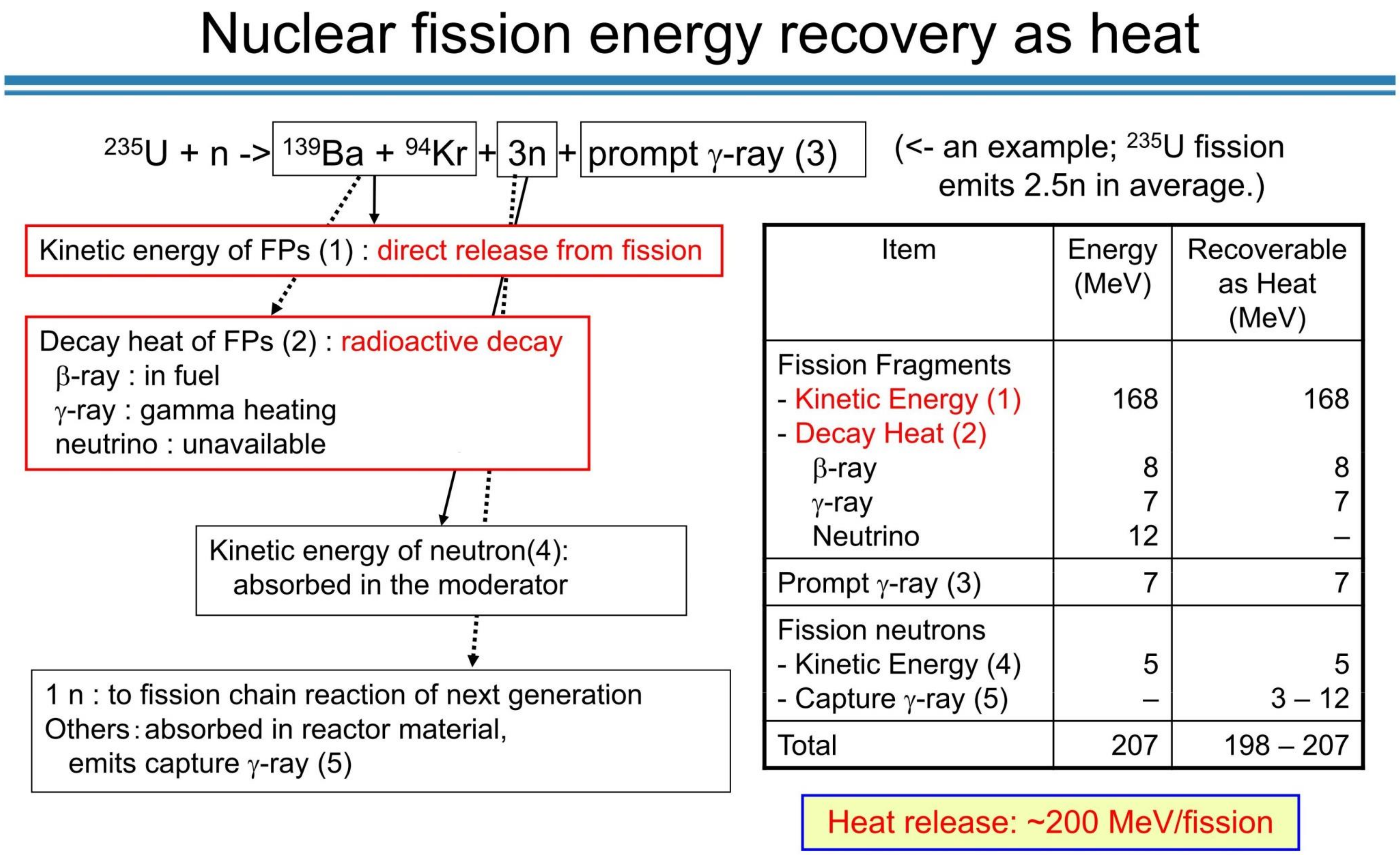
- Fuel temperature stays below damage limits
- Coolant can remove the generated heat
- Temperature, pressure and flow remain stable
- The reactor remains safe during accidents

Operational temperatures in Light water reactor (LWR) and High temperature gas reactor (HTGR)

Parameter	LWR	HTGR
Coolant	Water (PWR: not boiling) (BWR: boiling)	Helium
Core outlet temperature	~290–330 °C	~700–950 °C
Fuel type	UO_2 pellets	TRISO
Fuel peak temperature	~ 1500–2000 °C (2800 °C: melting)	~1200 – 1600 °C (> 1600: increased particle coating damage)
Moderator	Water	Graphite
Pressure	Very high (15 MPa in PWR) (6.9 MPa)	Moderate (3 – 7 MPa)
Thermal efficiency	~33% (core heat to steam generation)	~40–50% possible (core heat to useful heat)

Basic concept of nuclear reactor thermal hydraulic

Heat generation in the reactor core



Nuclear fission energy recovery as heat

$^{235}\mathrm{U} + n \rightarrow \boxed{{}^{139}\mathrm{Ba} + {}^{94}\mathrm{Kr}} + \boxed{3n} + \boxed{\text{prompt } \gamma\text{-ray (3)}}$

(<- an example; ^{235}U fission emits 2.5n in average.)

Kinetic energy of FPs (1) : direct release from fission

Decay heat of FPs (2) : radioactive decay
β-ray : in fuel
γ-ray : gamma heating
neutrino : unavailable

Kinetic energy of neutron(4):
absorbed in the moderator

1 n : to fission chain reaction of next generation
Others : absorbed in reactor material,
emits capture γ-ray (5)

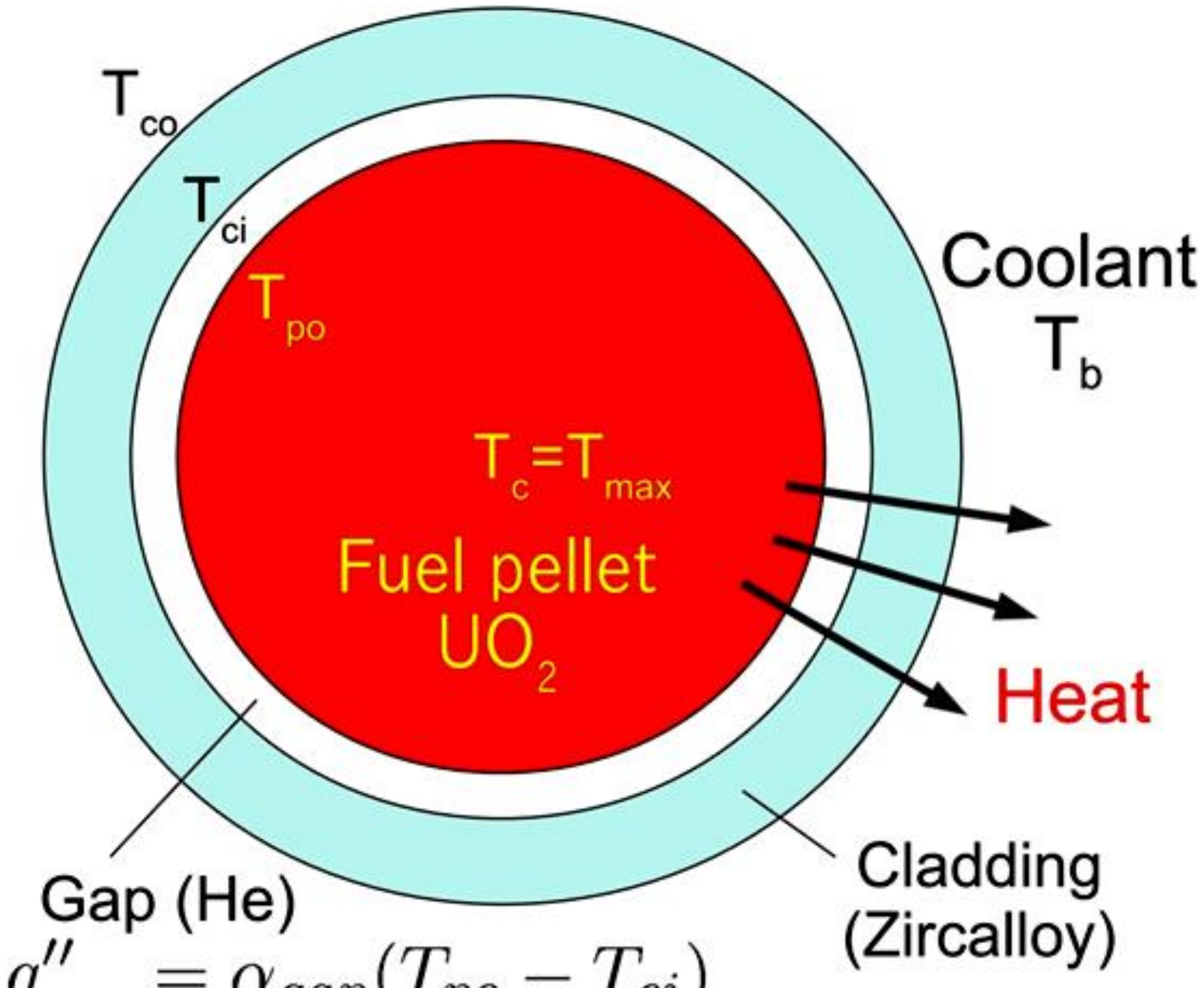
Item	Energy (MeV)	Recoverable as Heat (MeV)
Fission Fragments		
- Kinetic Energy (1)	168	168
- Decay Heat (2)		
β-ray	8	8
γ-ray	7	7
Neutrino	12	–
Prompt γ-ray (3)	7	7
Fission neutrons		
- Kinetic Energy (4)	5	5
- Capture γ-ray (5)	–	3 – 12
Total	207	198 – 207

Heat release: ~200 MeV/fission

- ~ 93 % as kinetic energy of fission fragments: heating the fuel directly
- 200 MeV ≈ 3.2x10^{11} Joule
- (Energy per fission)

Core thermal power:
Energy per fission x Fission rate = Joule / sec = Watt

Fuel rod cross section

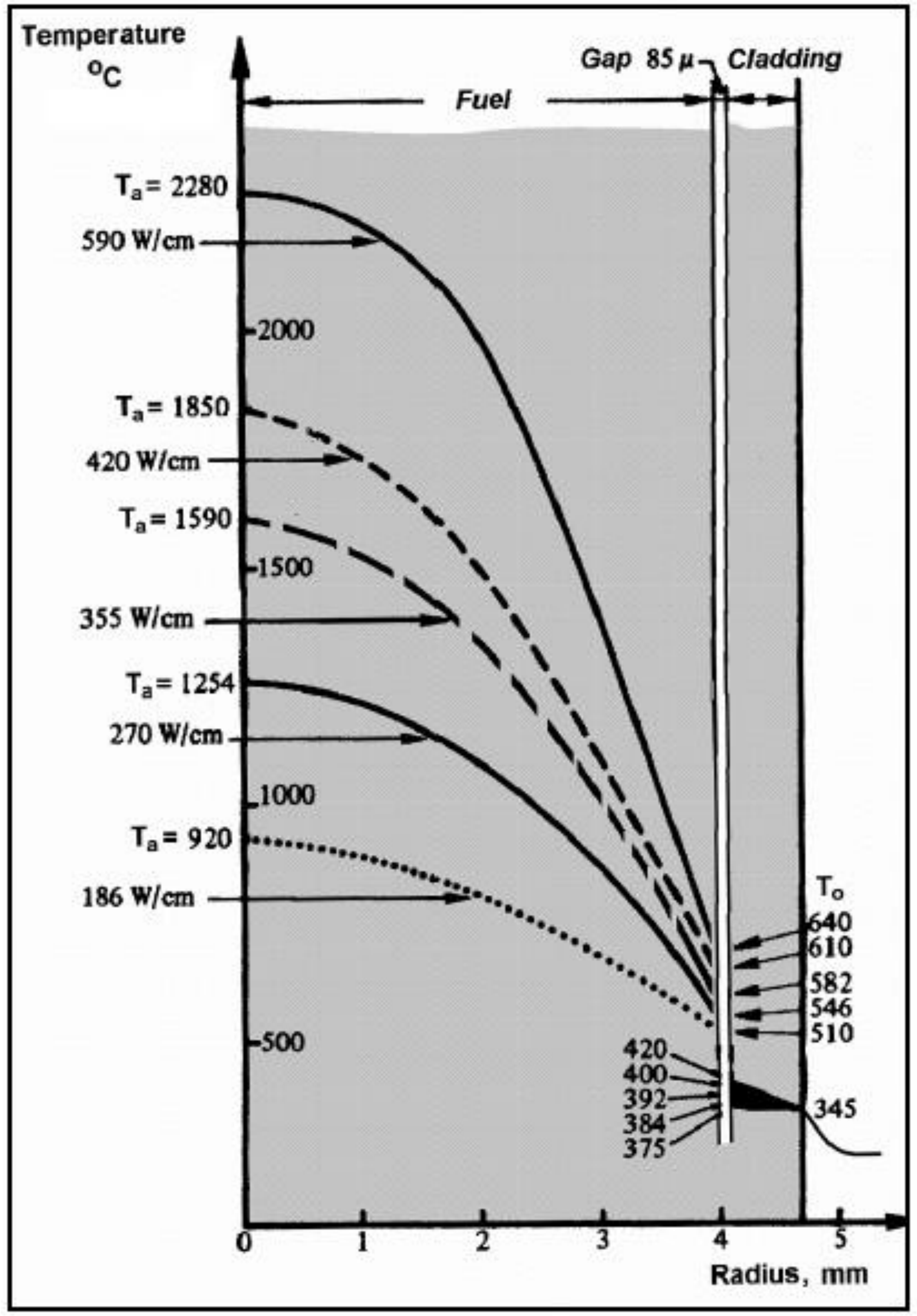


$$q''_{gap} = \alpha_{gap}(T_{po} - T_{ci})$$

- Power (P) ≈ Volumetric heat generation (q‴, Watt/m3)
- Assuming heat is generated from the fuel centre to the entire volume, the $T_C = T_{max}$
- Because fuel rod is multilayered heat-conduction medium (heat conductivity, k):
- T in radial = $T_C - \frac{q'''.\, r^2}{4k}$
- Looking at $\Delta T = T_C - T_S = \frac{q'''.\, r^2}{4k} = \frac{q'}{4k\pi}$
- Then $q' = \pi.\, R^2 q'''$ = linear heat generation rate (W/m)

- Temperature profile is nearly parabolic
- ΔT depends on heat conductivity and linear heat generation rate

Axial Fuel Rod Temperature Distribution



- ΔT depends on heat conductivity and linear heat generation rate
- The thermal conductivity of UO2, varies significantly with temperature, and depends on its density and on burnup.

Temperature distribution inside gap (between rod and cladding):

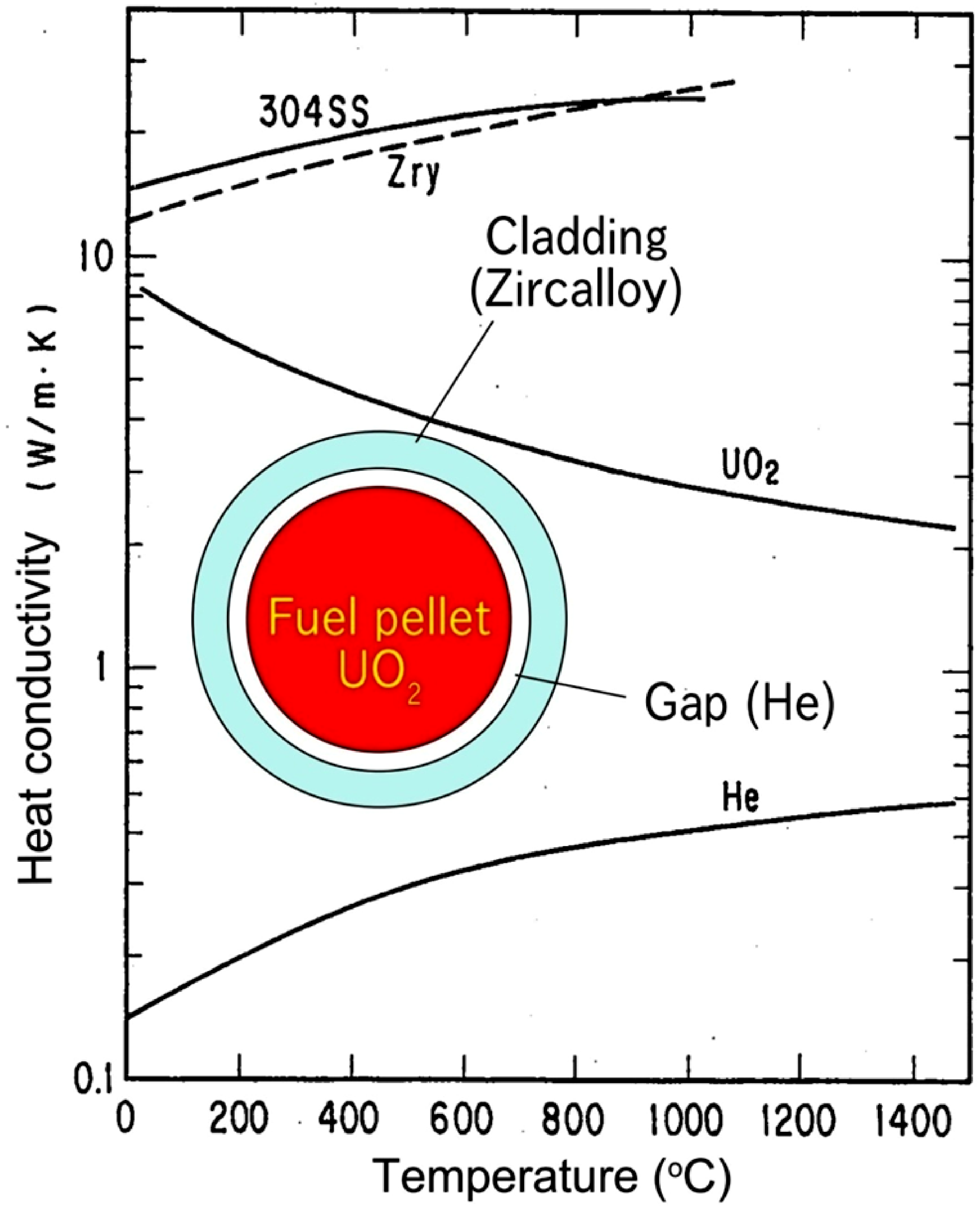
- Heat leaves the fuel rod surface as heat flux
- Lower k to decrease the temperature (ΔT_{gap}= 50 – 300 °)
- Normally as Helium
- Heat transfer occurs by gas conduction, radiation (and solid contact)
- = h_g (gap conductance)

$h_{gas} \sim k_{He}/\delta$: Gap conductance, ~6kW/m²K (a rough estimation for fresh fuel)

The gap is not uniform nor constant, affected by deformation or cracking of the fuel pellet.
=> A large uncertainty in the T jump in the gap (100--200K)

- If linear heat generation rate is known: $q'' = \dfrac{q'}{2\pi . R}$
- Cladding conduction (For cylindrical conduction): $T - T_b = -\dfrac{q'}{2\pi k} \ln \dfrac{r}{b}$

Dependence of Fuel Rod Heat Conductivity on Temperature



UO$_2$ k decreases with T [K]

$$k = \frac{3820}{T + 129.4} + 4.79\text{x}10^{-11}\,T^3$$

He k of gases $k = a\sqrt{T/M}$

a : constant
M: molecular weight

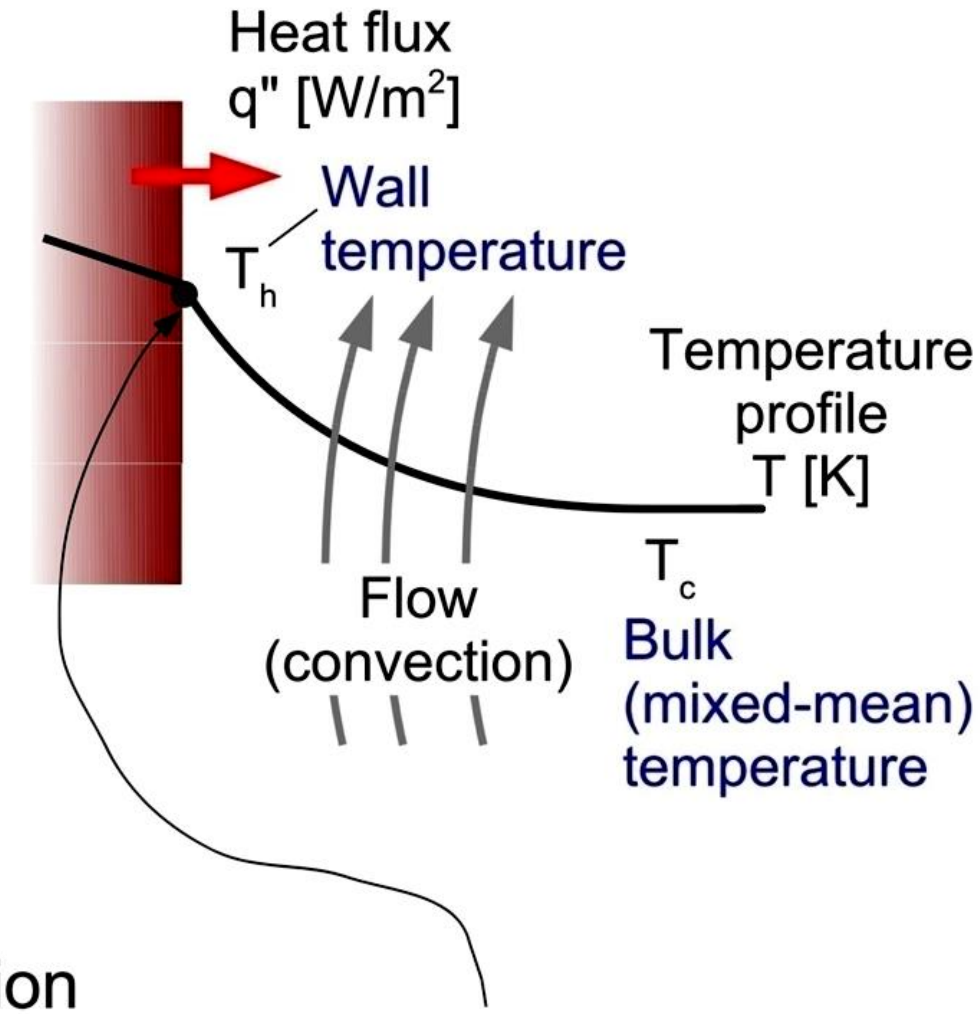
Heat Transfer to coolant

$$q'' = \alpha(T_h - T_c)$$

q" [W/m²]: Heat flux
α [W/m²K]: Heat transfer coefficient
T_h [K]: Wall temperature
T_c [K]: Fluid (coolant) temperature (bulk, mean)

"**Heat transfer coefficient**" is a math expression of an overall effectiveness of heat flow versus temperature gap (engineering, macroscopic)

It actually involves microscopic processes of heat conduction, flow, or radiation.



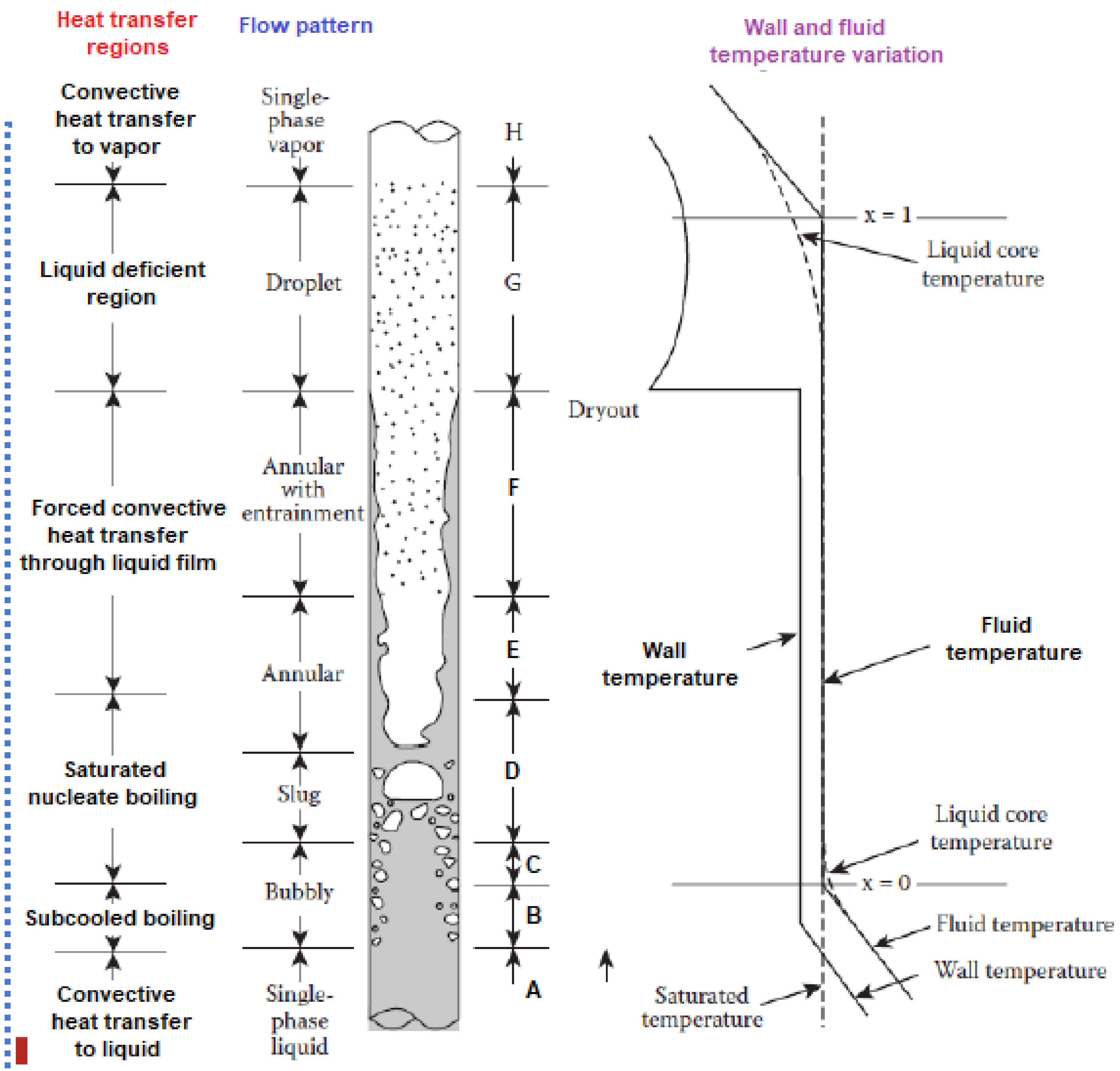
Heat Conduction

$$q'' = k\frac{dT}{dx} \Longrightarrow q'' = k_{wall}\,(dT/dx)_{wall} = k_{fluid}\,(dT/dx)_{fkuid}$$

HT coefficient in:

- PWR:
  - ✓ mostly single-phase forced convection
  - ✓ h≈20,000−100,000 W/m2K,
  - ✓ relative stable due to pressure
  - ✓ T coolant rises gradually
- BWR:
  - ✓ nucleate boiling heat transfer
  - ✓ h≈50,000−300,000 W/m2K,
  - ✓ Changing
  - ✓ Boiling, vapor fraction, flow regime

Boiling heat transfer regimes in a heated channel



Critical point (Departure from nucleate boiling / DNB)
No liquid film, small HT of steam

BWR fluid region (esp. annular)
high vapor fraction, significant boiling
Annular: liquid film wall, steam center

PWR fluid region
SNB allowed locally, increases HT coefficient

Core Power Density

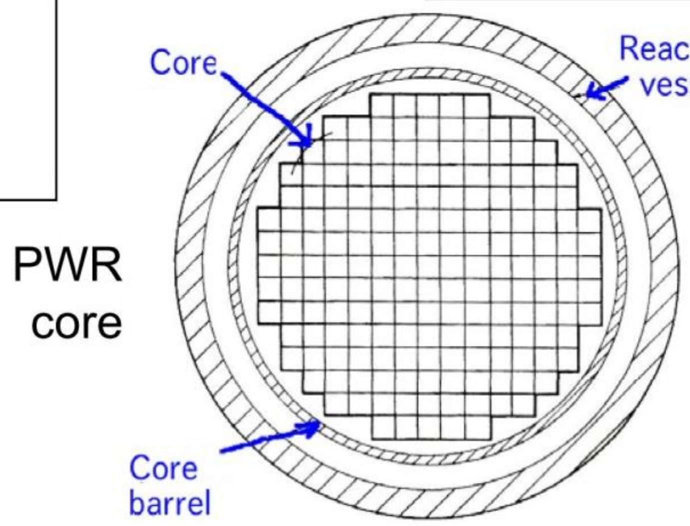
- ➢ Divided the total power by core volume yields the power density.

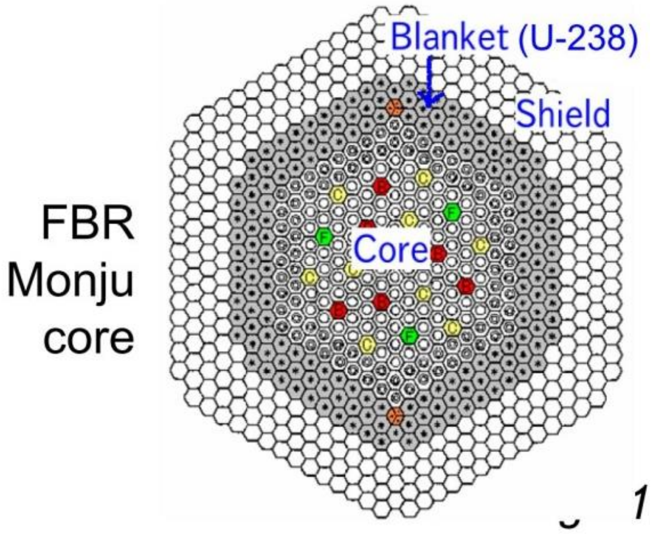
$$\frac{P}{V} = E_f \frac{M_f}{A} N_A \sigma_f \phi \frac{1}{V} \quad (\mathrm{W/m^3})$$

PWR > BWR due to tight lattice bundle with small diameter rods

Due to a good heat transfer by sodium coolant. (small Pr = ν/α -> relatively narrow flow path is enough)
Including the "blanket" volume, it is comparable to PWR.

Reactor type	Power density ($\mathrm{MW/m^3}$)
PWR	100
BWR	50
FBR (Monju)	302
HGR (HTTR)	2.5
Research Reactor	30-500



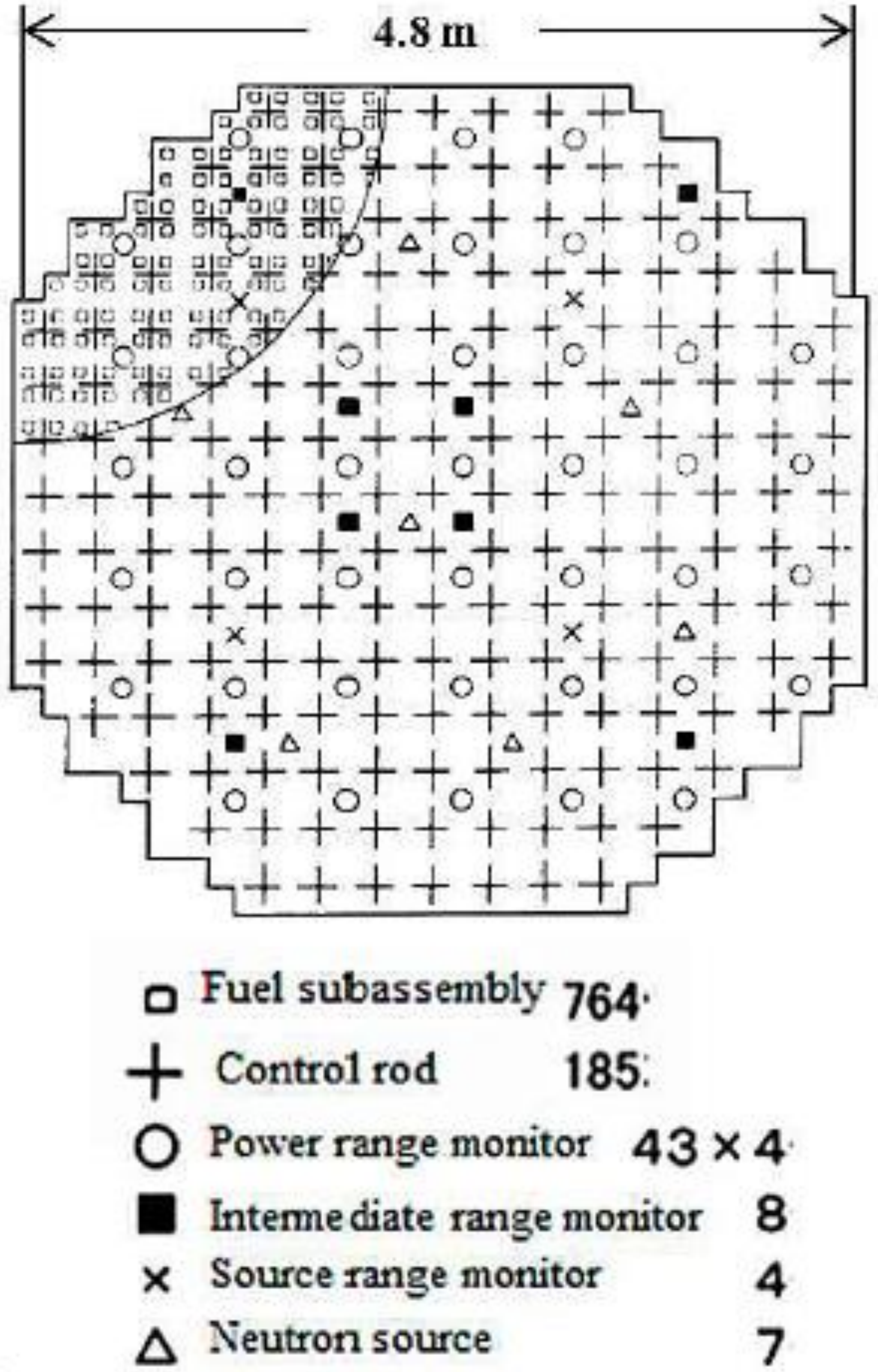


Core density due to:

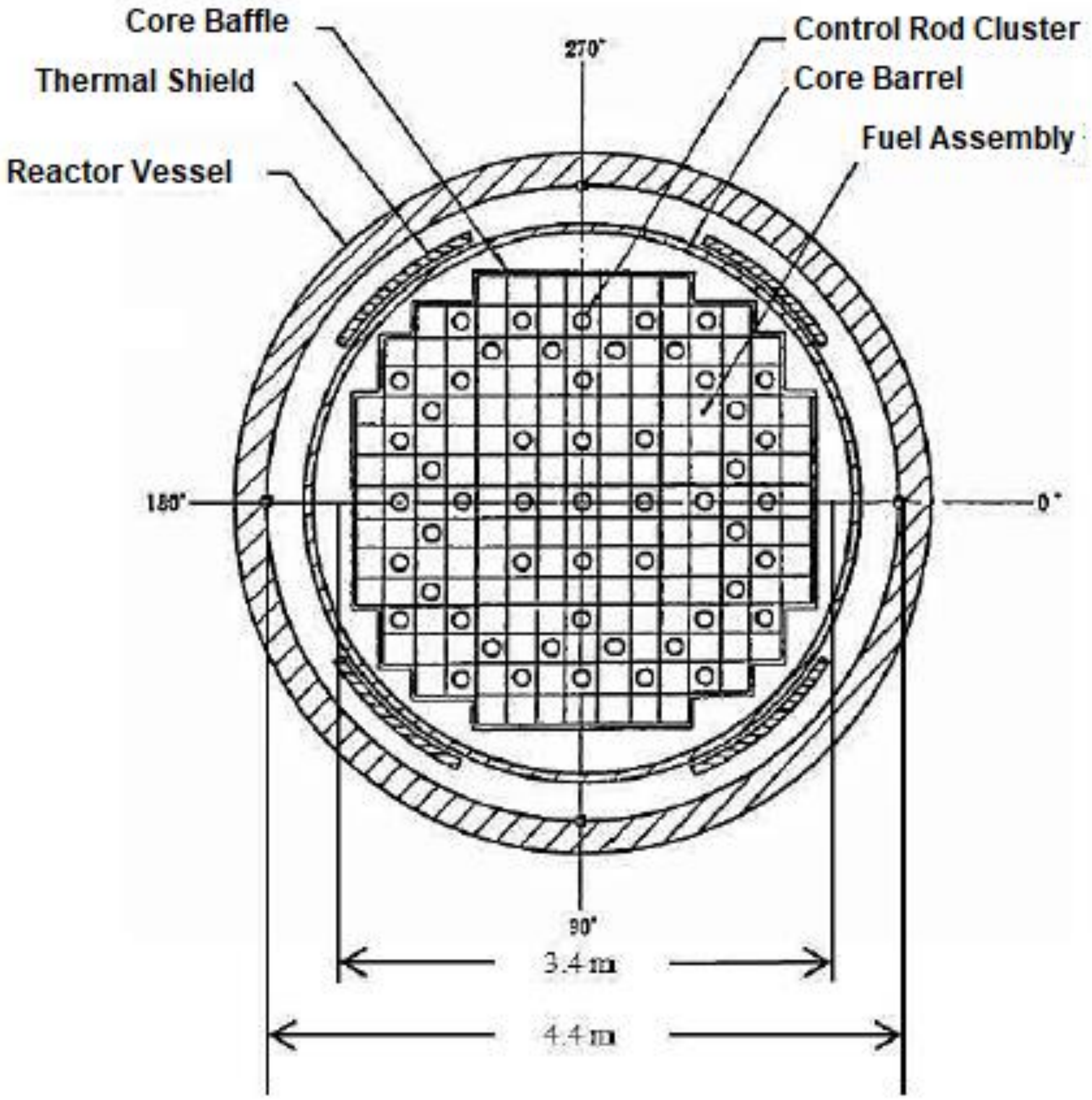
- Core size
  - Compact (PWR)
  - Larger (HTGR > BWR)
- Coolant phase:
  - Single, suppressed (PWR)
  - Boiling, low density (BWR)
  - Gas phase (HTGR)

Overview of PWR and BWR Cooling System

PWR and BWR Core Map

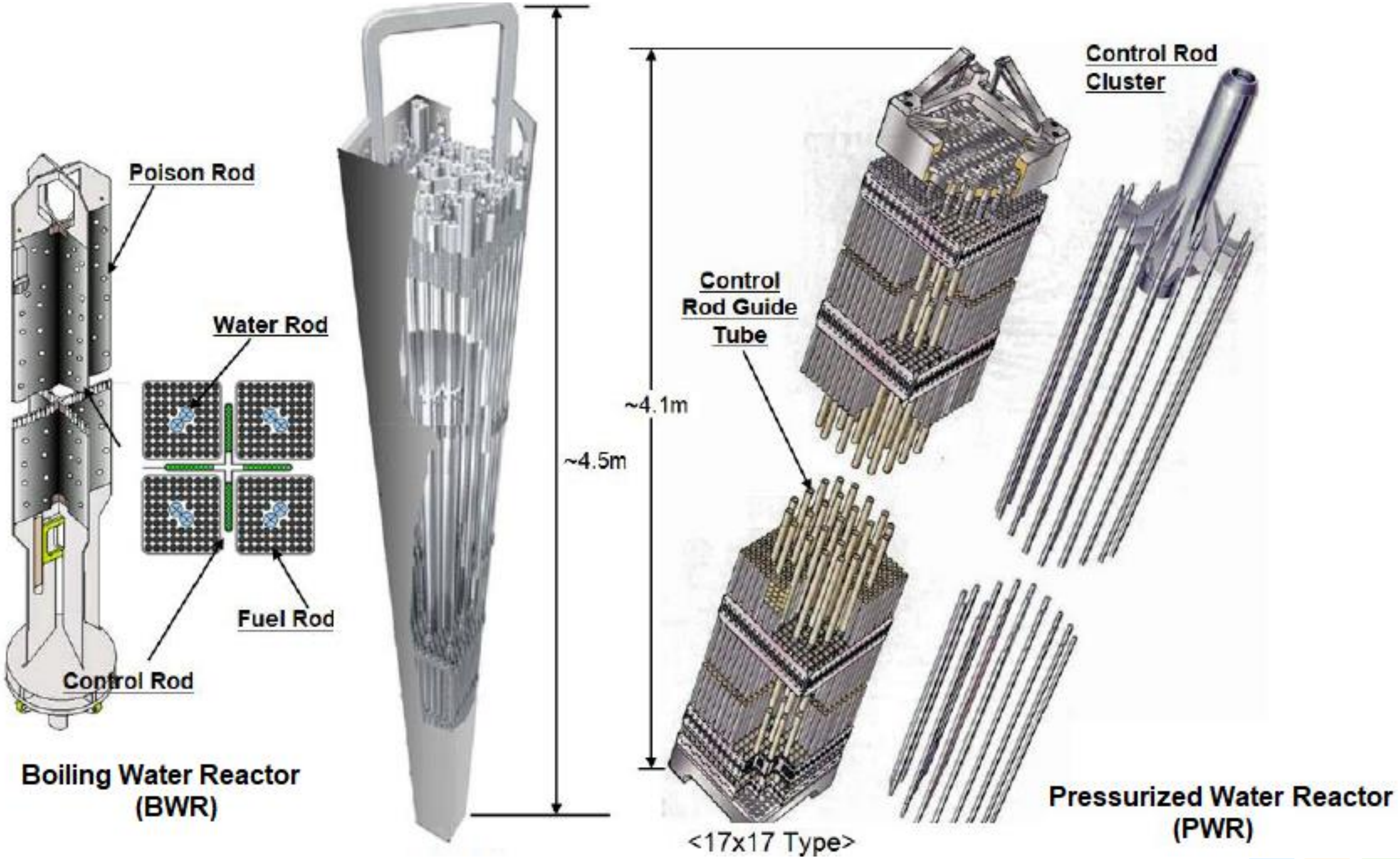


BWR

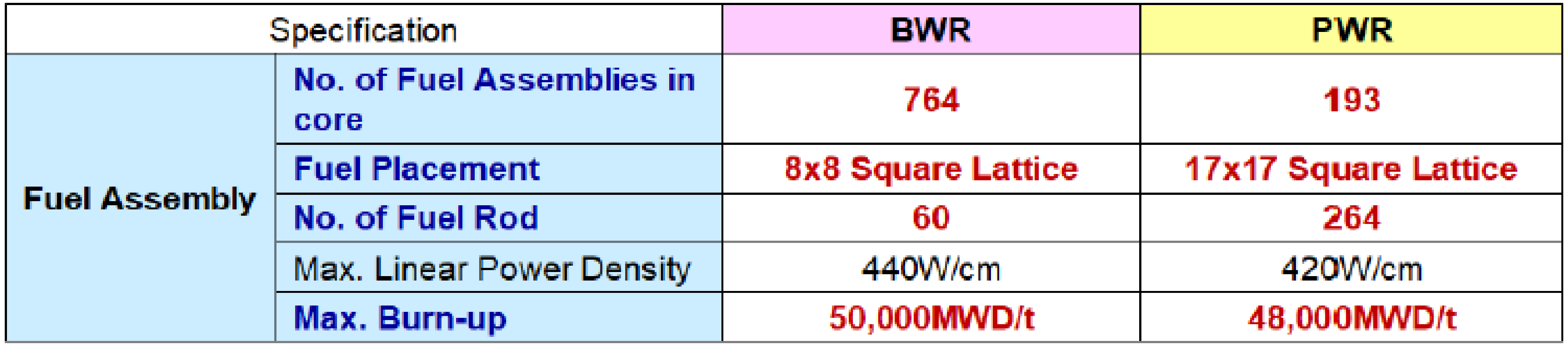


PWR

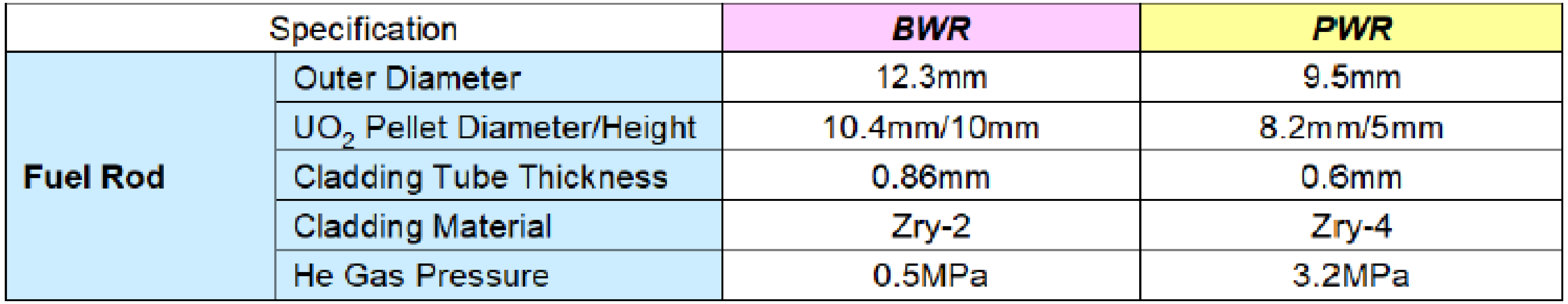
PWR and BWR Fuel Assembly



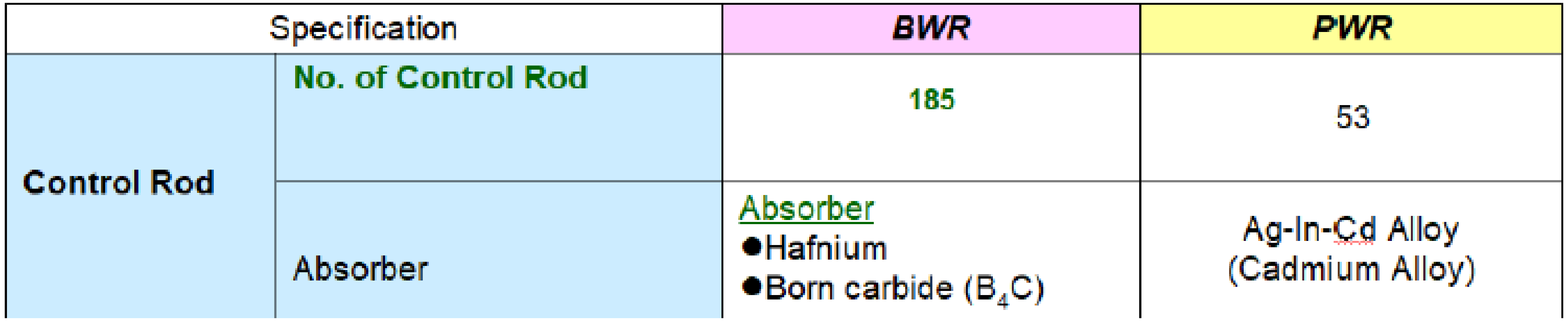
PWR and BWR Fuel Assembly Specifications



Specification		BWR	PWR
Fuel Assembly	**No. of Fuel Assemblies in core**	**764**	**193**
	Fuel Placement	**8x8 Square Lattice**	**17x17 Square Lattice**
	No. of Fuel Rod	**60**	**264**
	Max. Linear Power Density	440W/cm	420W/cm
	Max. Burn-up	**50,000MWD/t**	**48,000MWD/t**

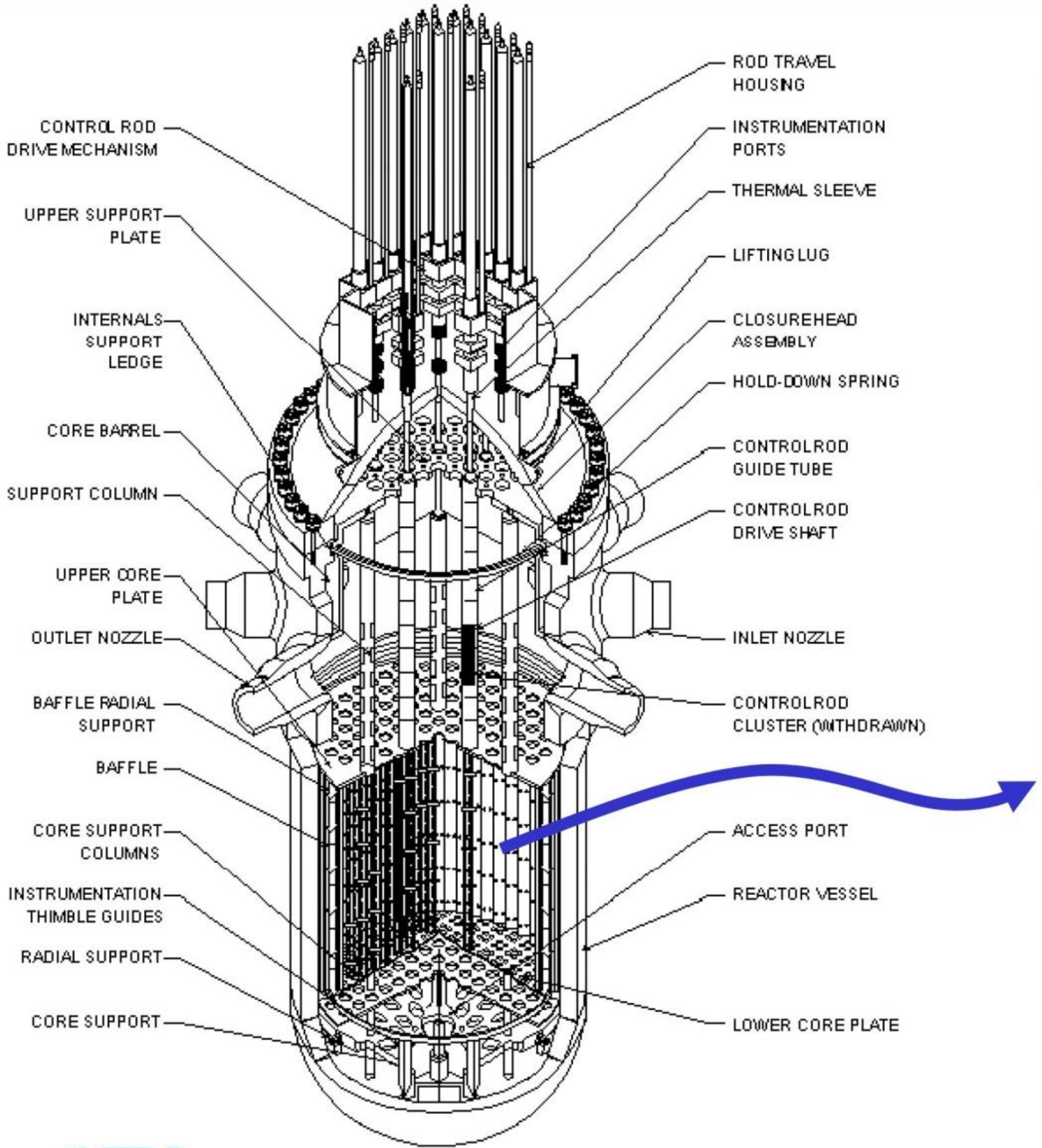


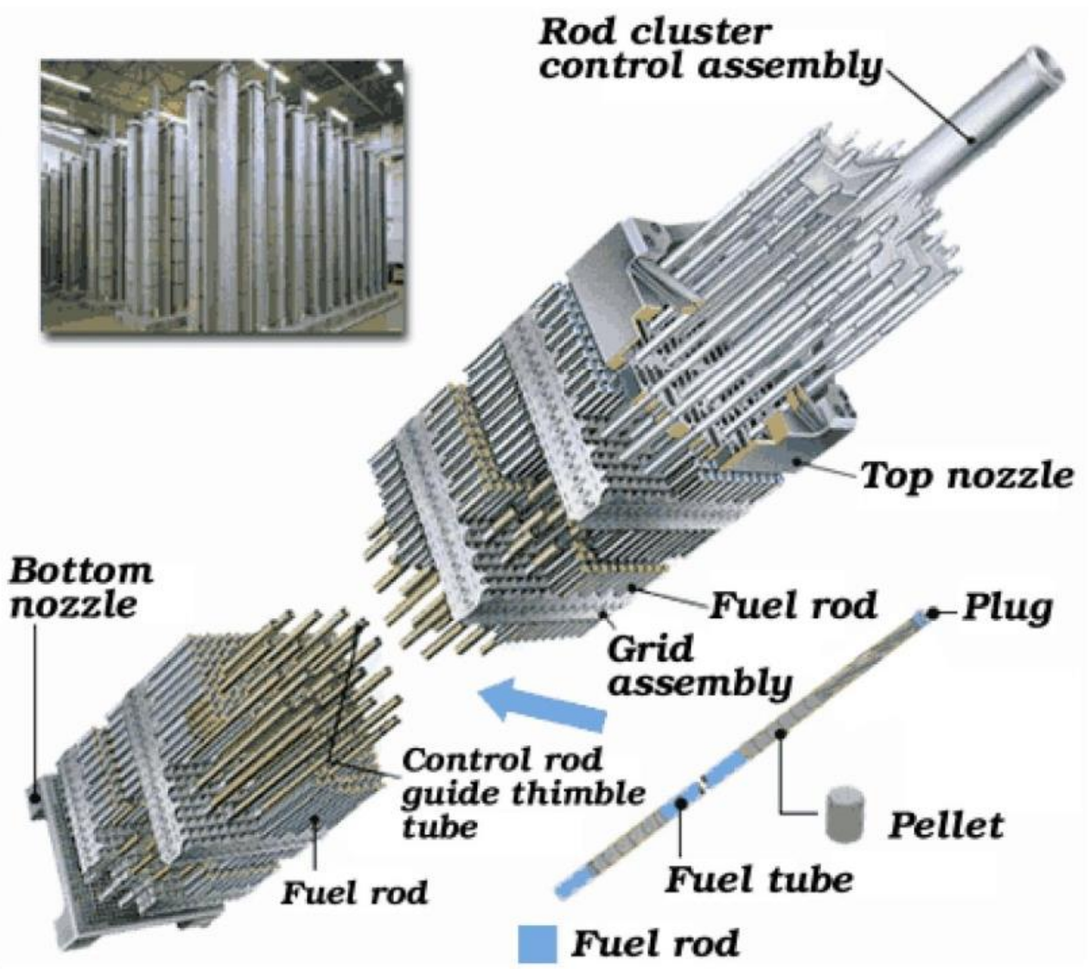
Specification		***BWR***	***PWR***
Fuel Rod	Outer Diameter	12.3mm	9.5mm
	UO_2 Pellet Diameter/Height	10.4mm/10mm	8.2mm/5mm
	Cladding Tube Thickness	0.86mm	0.6mm
	Cladding Material	Zry-2	Zry-4
	He Gas Pressure	0.5MPa	3.2MPa



Specification		***BWR***	***PWR***
Control Rod	**No. of Control Rod**	**185**	53
	Absorber	Absorber ●Hafnium ●Born carbide (B_4C)	Ag-In-Cd Alloy (Cadmium Alloy)

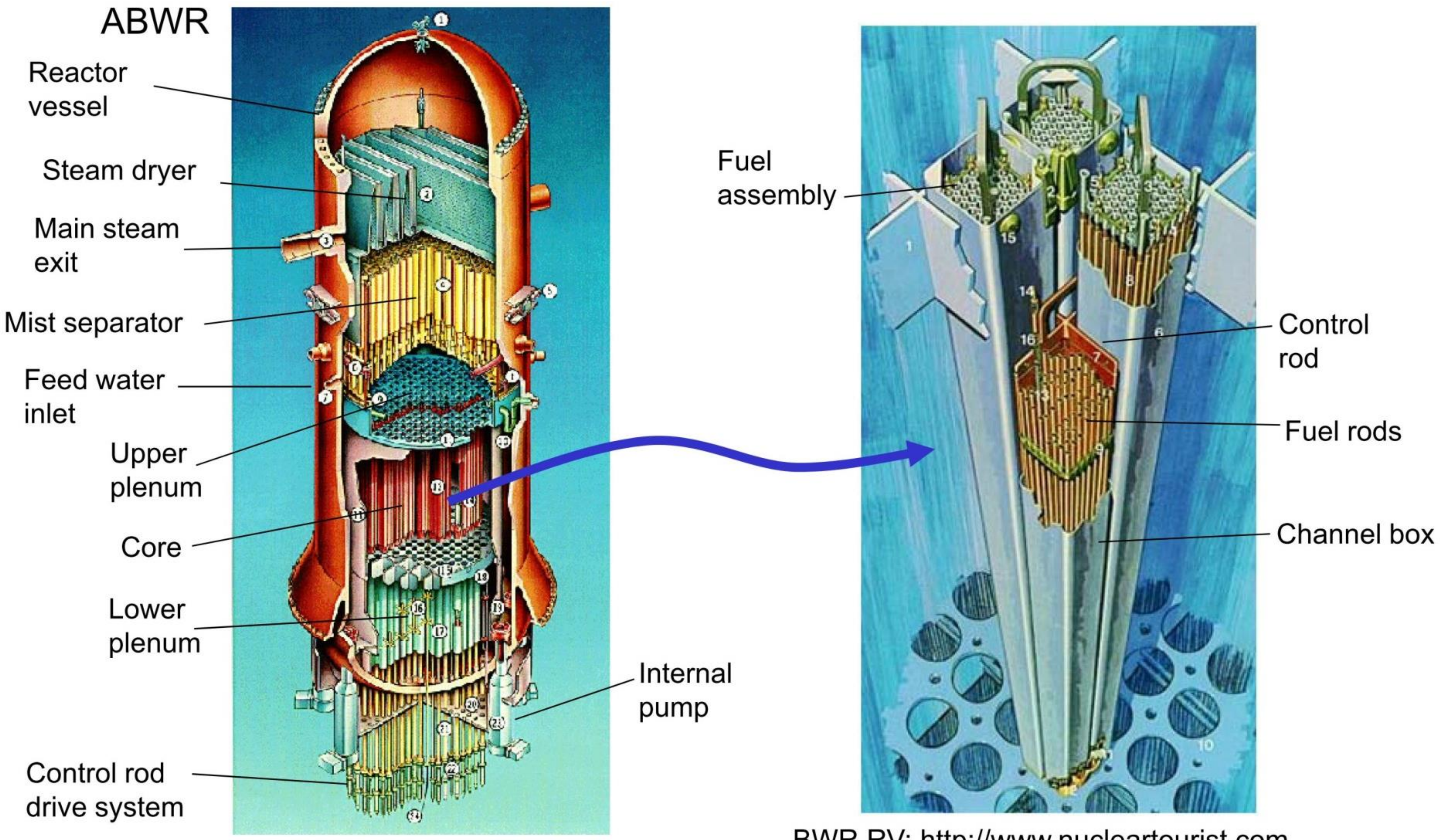
PWR Core and Reactor Pressure Vessel





PWR RV: http://en.wikipedia.org
PWR fuel: http://www.mnf.co.jp

BWR Core and Reactor Pressure Vessel



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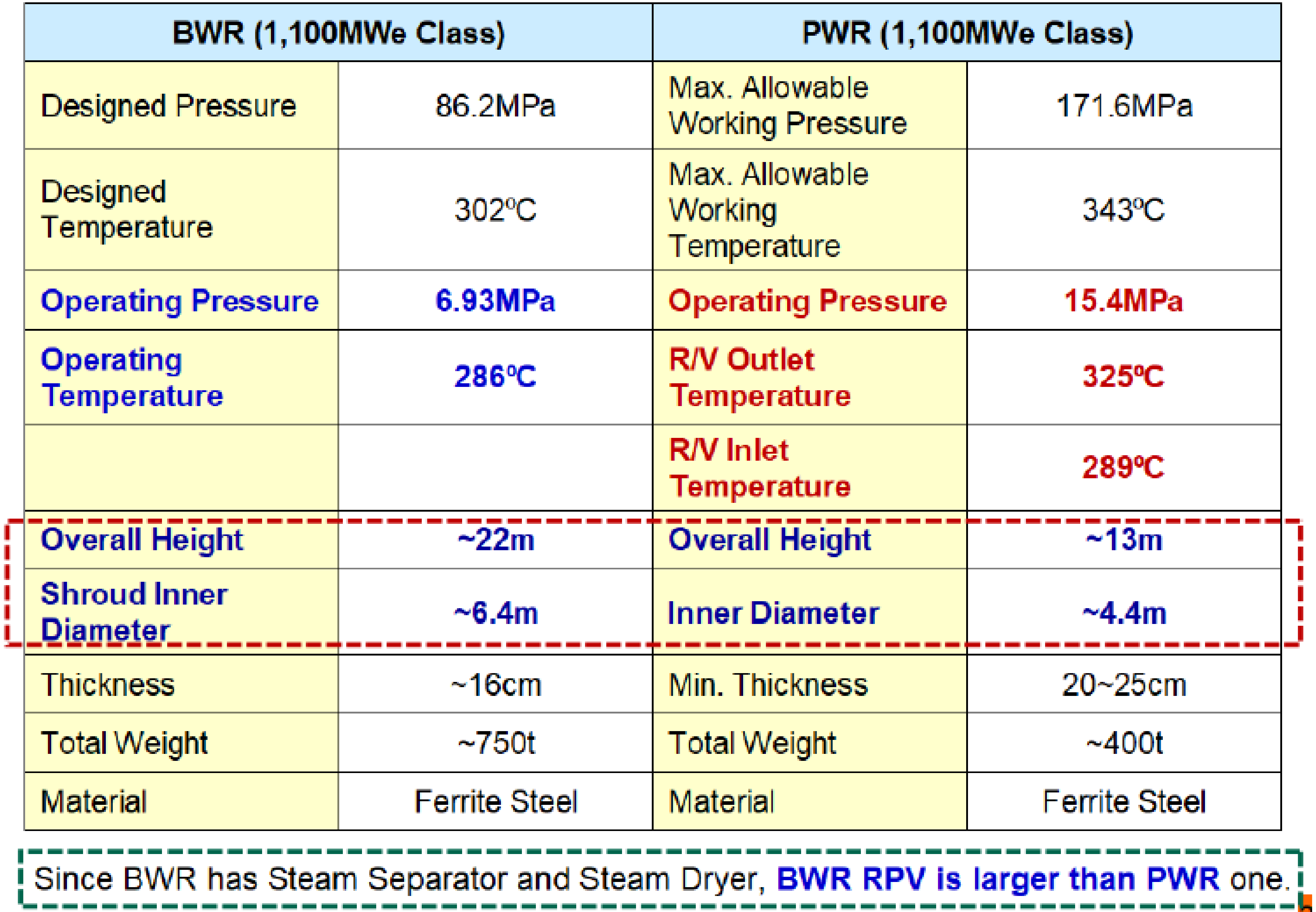
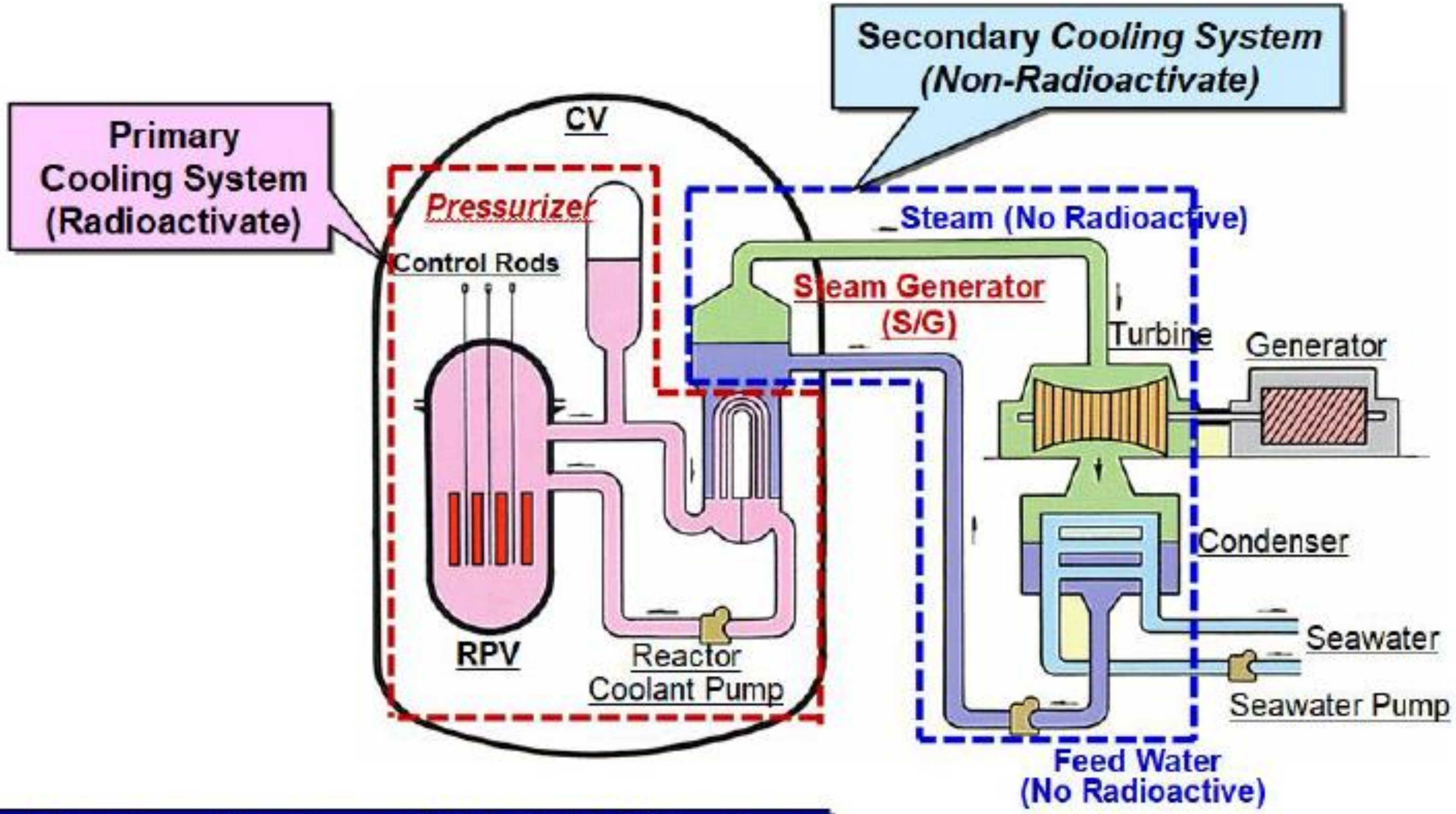
PWR and BWR Reactor Pressure Vessel Specifications



BWR (1,100MWe Class)		PWR (1,100MWe Class)	
Designed Pressure	86.2MPa	Max. Allowable Working Pressure	171.6MPa
Designed Temperature	302ºC	Max. Allowable Working Temperature	343ºC
Operating Pressure	**6.93MPa**	**Operating Pressure**	**15.4MPa**
Operating Temperature	**286ºC**	**R/V Outlet Temperature**	**325ºC**
		R/V Inlet Temperature	**289ºC**
Overall Height	**~22m**	**Overall Height**	**~13m**
Shroud Inner Diameter	**~6.4m**	**Inner Diameter**	**~4.4m**
Thickness	~16cm	Min. Thickness	20~25cm
Total Weight	~750t	Total Weight	~400t
Material	Ferrite Steel	Material	Ferrite Steel

Since BWR has Steam Separator and Steam Dryer, **BWR RPV is larger than PWR** one.

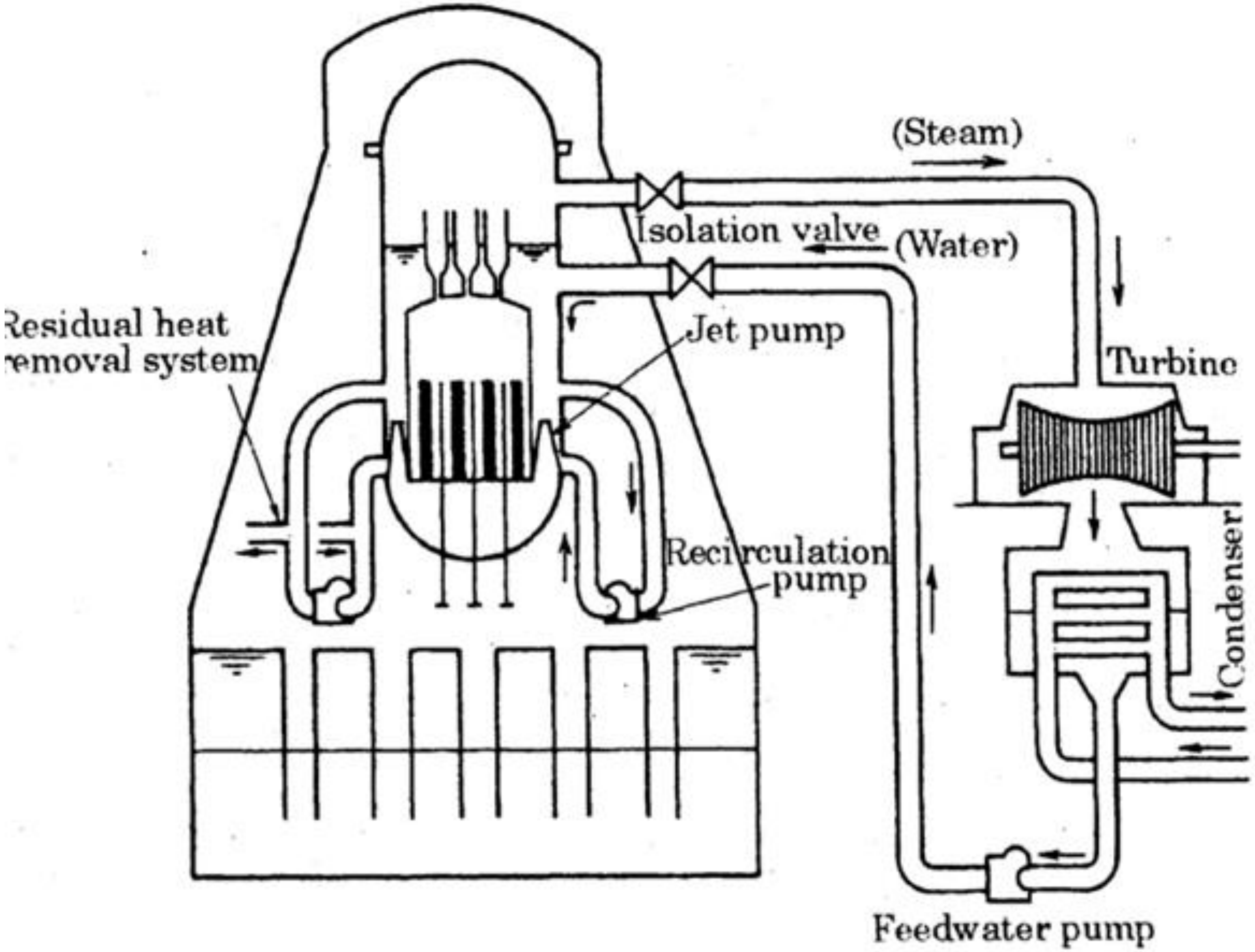
PWR Cooling System

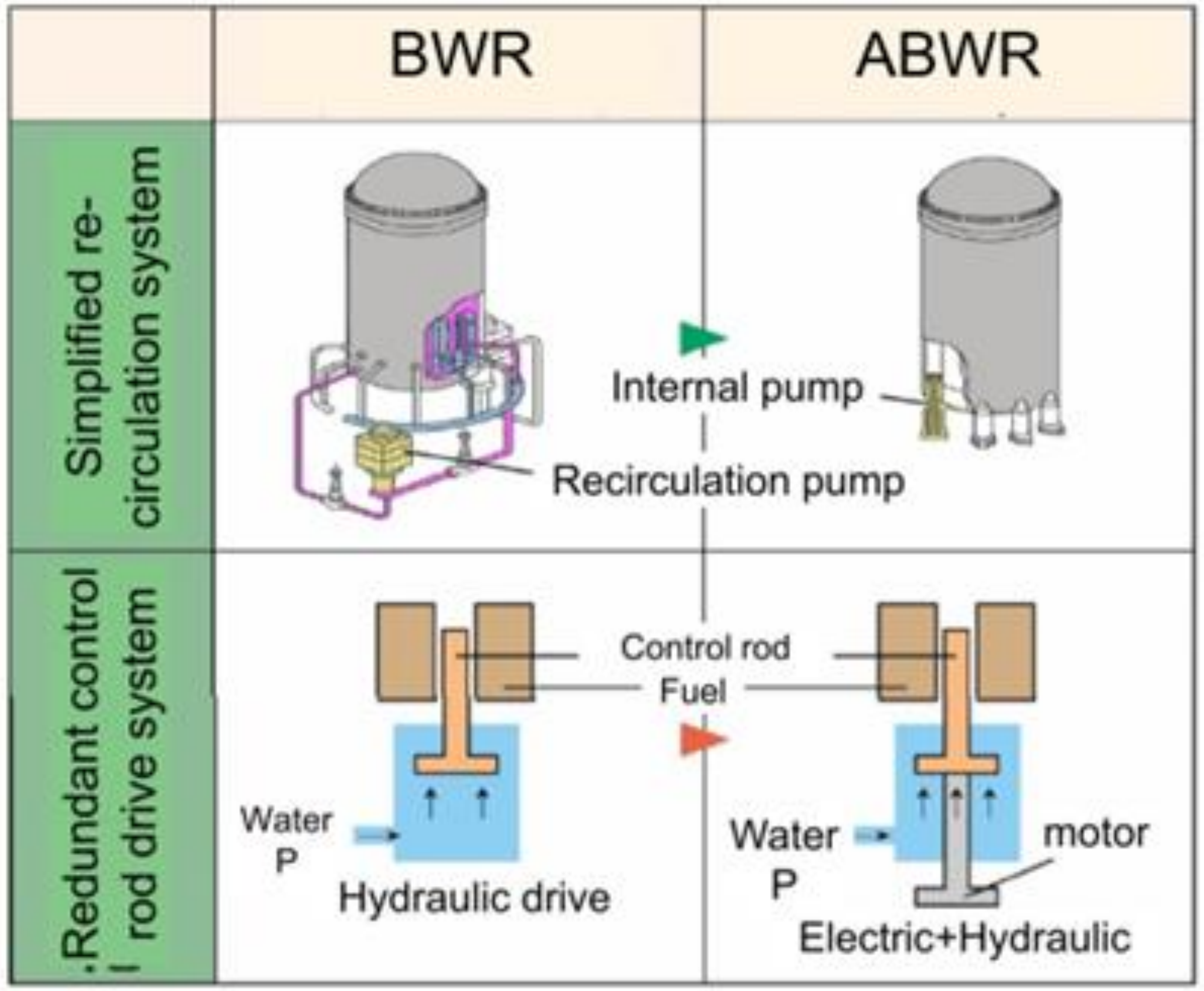


Main Plant Specifications of PWR (1,100MWe class)

RPV Pressure	Core Flow rate	Core Outlet Temp.	Core Inlet Temp.	Steam Flow Rate	Feed Water Temp.
15.4MPa	*60,100t/h*	*325°C*	*289°C*	*6,760t/h*	*223°C*

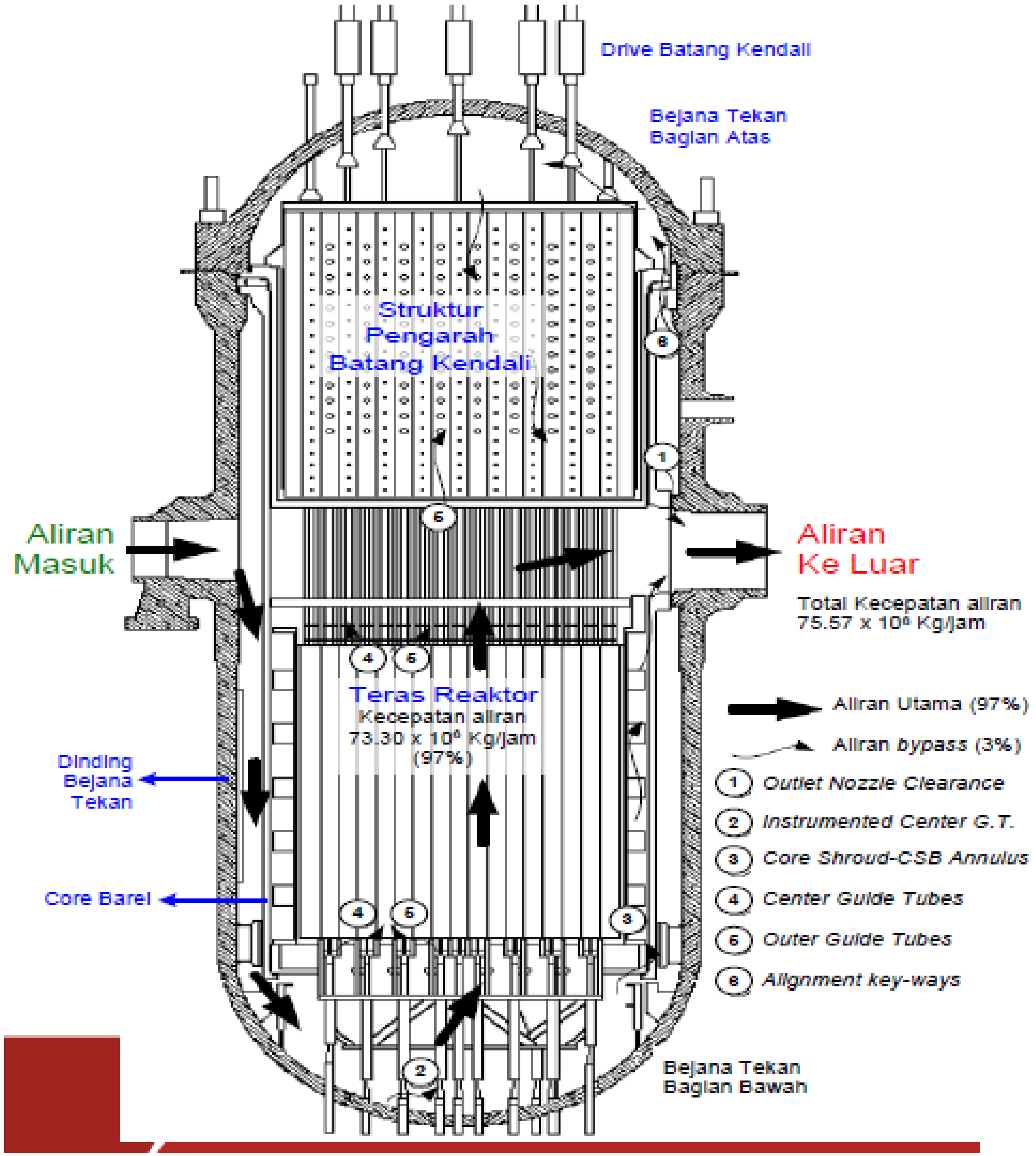
BWR Cooling System





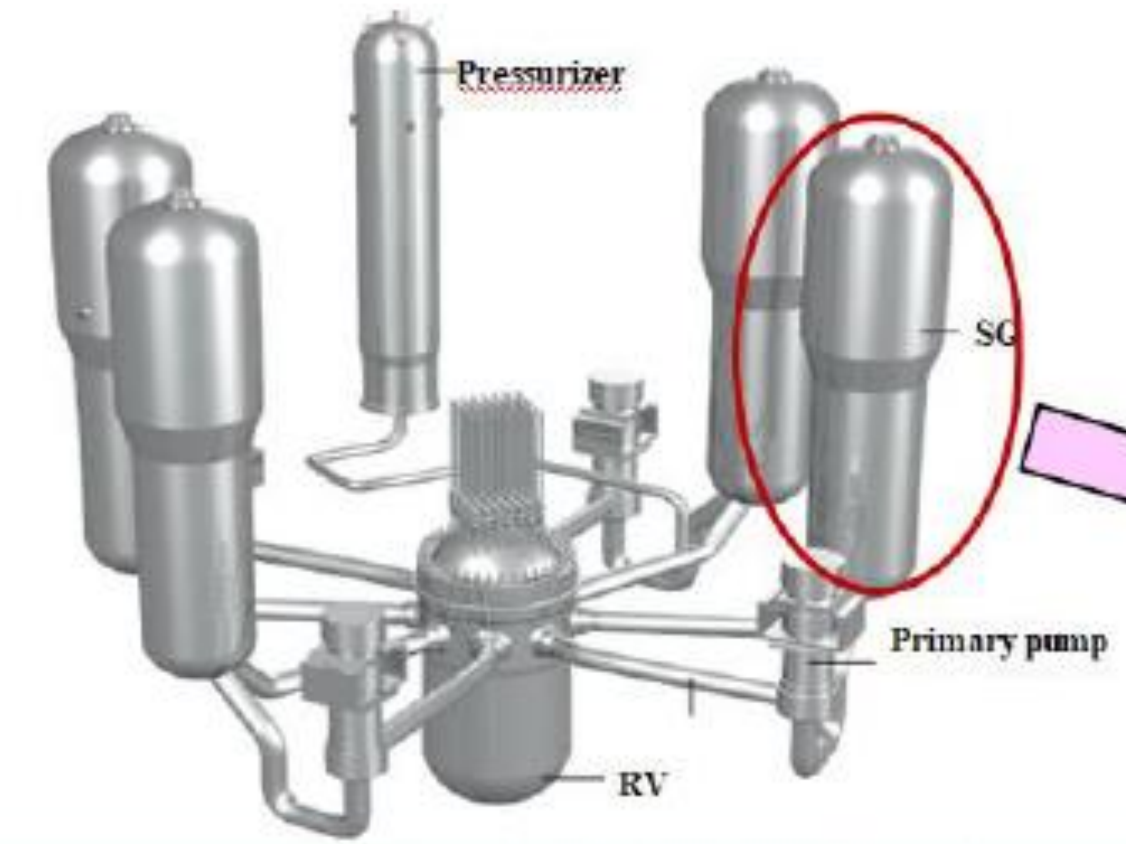
Elimination of the pipes reduces the possibility of the large break LOCA in the re-circulation system.

PWR Reactor Pressure Vessel Flow Pattern

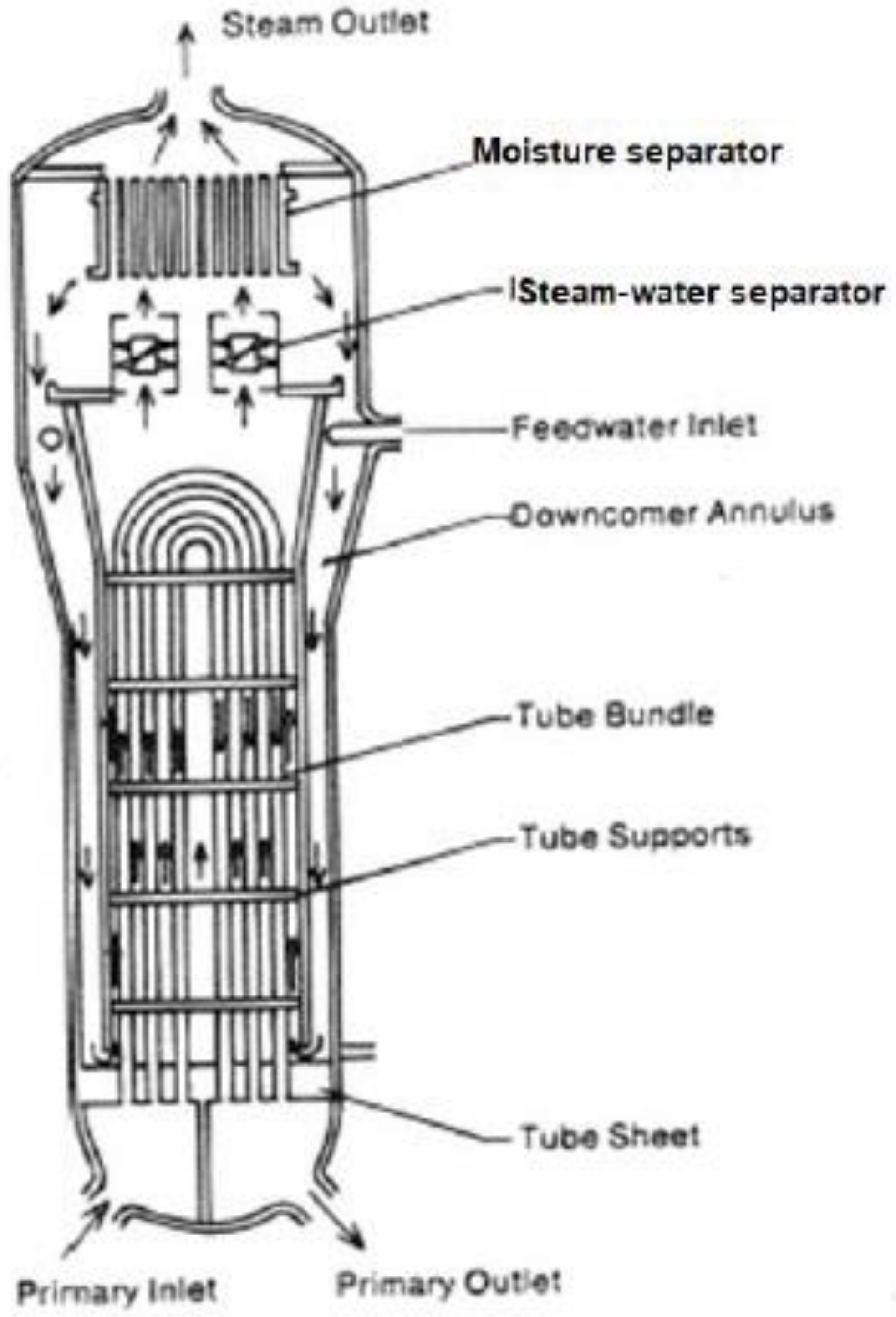


- Inlet
- Down through Narrow Channel
- Mixing in Downcomer
- Reactor Core
- Outlet

PWR Nuclear Steam Supply System



Specification of S/G (4 S/G Types)	
Height	~20m
Diameter	~5m
Design Pressure	~8.2MPa
Shape	U Shape Transfer Tube
Primary Side (Inside)	325°C, 15.4MPa
Secondary Side (Outside)	277°C, 6.0MPa
Material	TTI Inconel 690 (High Corrosion Resistant)
No. of Tubes	~3,300
Tube Diameter	~2cm
Tube Thickness	~1.3mm



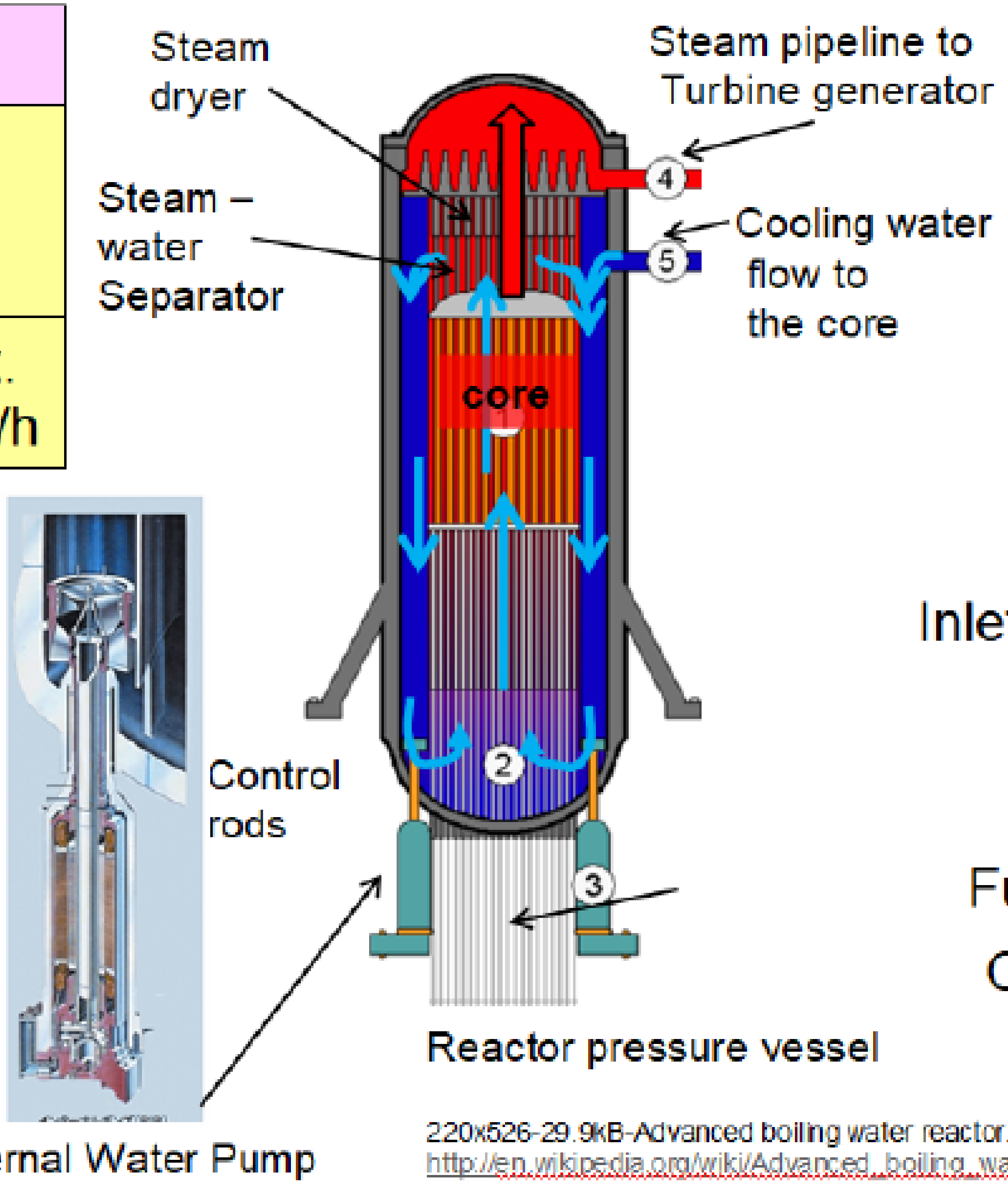
BWR Nuclear Steam Supply System

1350MWe ABWR	
No. of Circulation Pumps	10
Core Flow Rate	Approx. 8300m³/h

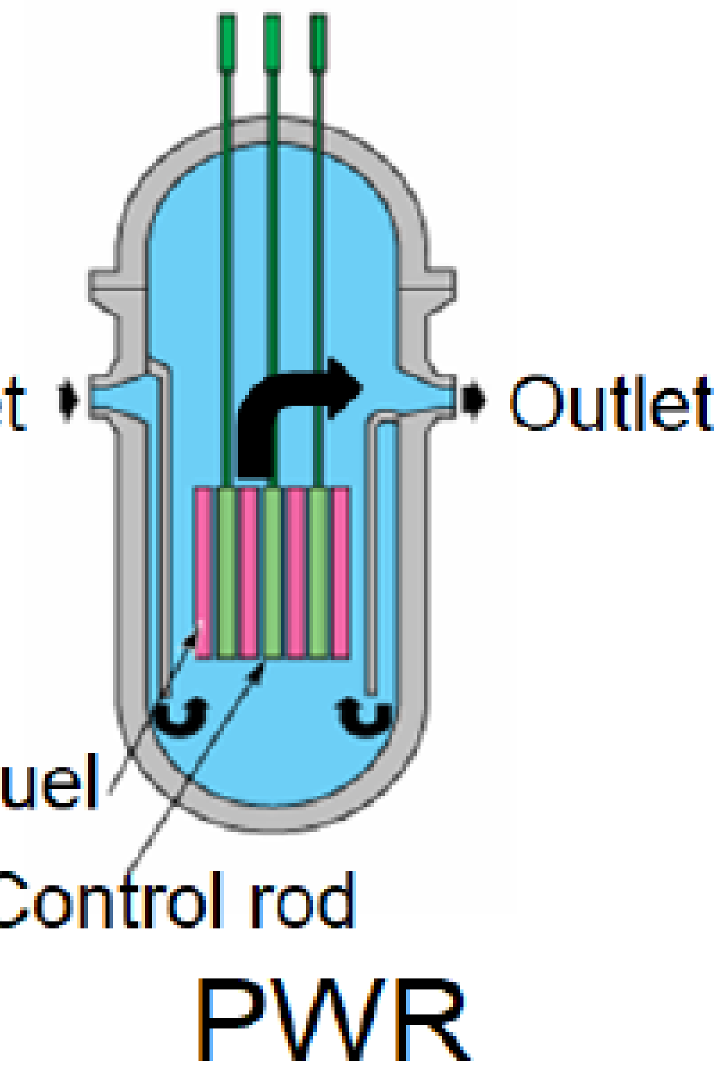
Coolant –water flown from core is mixed with feed-water.

They are recalculated with internal Pumps.

Thermal power can be controlled by internal pump with the recirculation



220x526-29.9kB-Advanced boiling water reactor...
http://en.wikipedia.org/wiki/Advanced_boiling_wate...

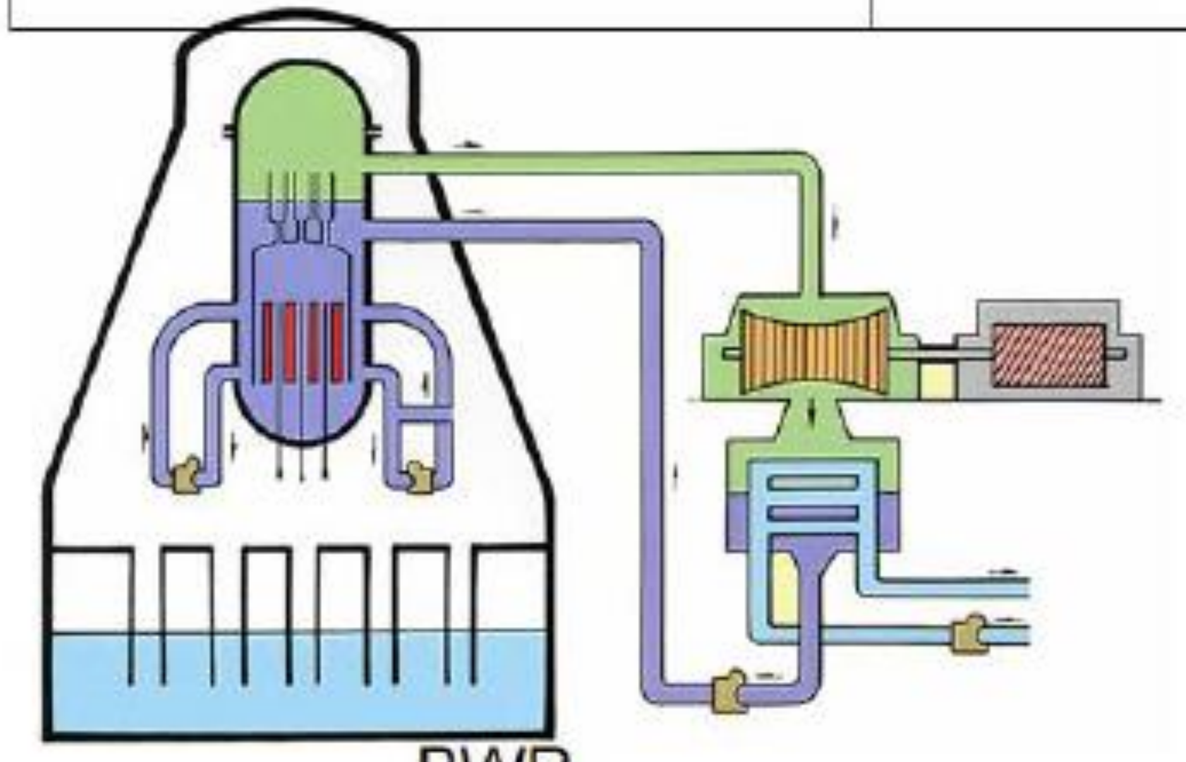


PWR

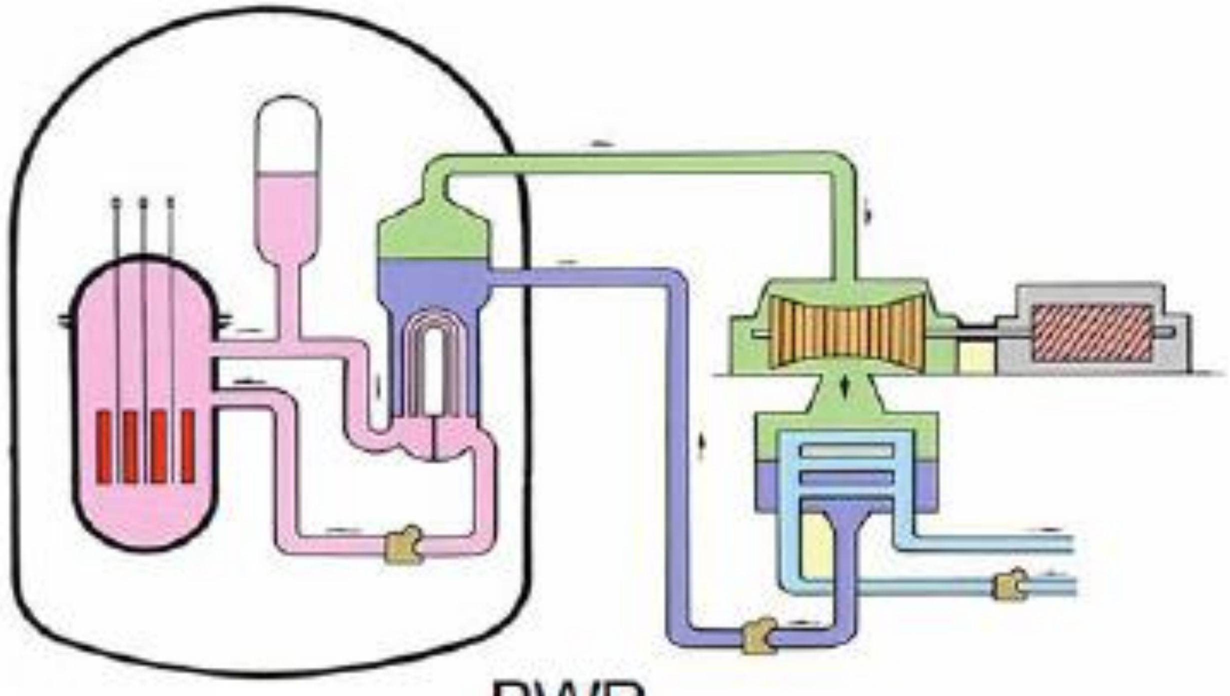
PWR and BWR Nuclear Steam Supply System

The ***steam condition of BWR is superior to PWR*** because BWR can directly use the ***steam produced in a core***. While, PWR's steam is generated via S/G (secondary).

	BWR	PWR
Steam Pressure	6.9MPa	6.0MPa
Steam Temperature (T_H)	286ºC	277ºC
Feed Water Temperature (T_L)	216ºC	223ºC
Steam Flow Rate	6.41×10^6kg/h	6.76×10^6kg/h
$P_{eff} = P_e / P_t$ (%)	**33.4% (ABWR:34.6%)**	**32.2% (APWR:34.4%)**



BWR



PWR

Heat Balance of PWR Core

The increase of the coolant enthalpy between the inlet and outlet of the core equals the heat generation in the core.

No boiling case (single phase flow)
-> The enthalpy increase is expressed by
the temperature rise.

$$Q = W\Delta h$$

$$\Downarrow$$

$$Q = WC_p\Delta T$$

Existing PWR
inlet : ~ 285 °C
outlet : ~ 320 °C on average
~ 345 °C for high power peaking

W = coolant mass flow rate (kg/sec)
$\Delta h = h_{out} - h_{in}$ (kJ/kg)
Cp = coolant specific heat (kJ/kg. K)

Typical commercial Thermal power:

W = 17000 kg/s
Cp = 5 kJ/kg
ΔT = 35 °C
Δh = 175 kJ/kg
$Q \approx 3000$ MWt

Heat Balance of BWR Core

Boiling flow (water-vapor two-phase flow)
-> The enthalpy increase of the coolant includes
- the apparent heat (temperature rise) and
- the latent heat (vaporization).

$$Q = W\Delta h = W(\underbrace{\Delta h_{sub}}_{\text{Energy for heat up to the saturation}} + \underbrace{x\Delta h_{fg}}_{\text{Energy for vapor generation}})$$

x: quality $x = \frac{W_g}{W} \sim x_e = \frac{h - h_{ls}}{h_{gs} - h_{ls}} = \frac{h - h_{ls}}{\Delta h_{fg}}$

The original meaning of the "quality" is the fraction of vapor mass flow rate. Practically, it is near the "equilibrium quality", meaning the enthalpy level of the two-phase mixture. (0 at saturation liquid, 1 at saturation vapor)

W = coolant mass flow rate (kg/sec)
$\Delta h = h_{out} - h_{in}$
$\Delta h_{sub} = h_{is} - h_{\text{in}}$ = sensible heating
Δh_{fg} = latent heat vaporization
h_{is} = saturated liquid enthaply
x = vapor fraction at core outlet

Typical commercial Thermal power:
W = 14000 kg/s (P = 7 MPa)
$h_{out} / h_{in} \approx 1450 / 1200$ kJ/kg
$h = h_{out}$
Δh_{fg} = 1500 kJ/kg
h_{is} = 1260 kJ/kg
$x \approx 13$ %
Q = 3500 MWt

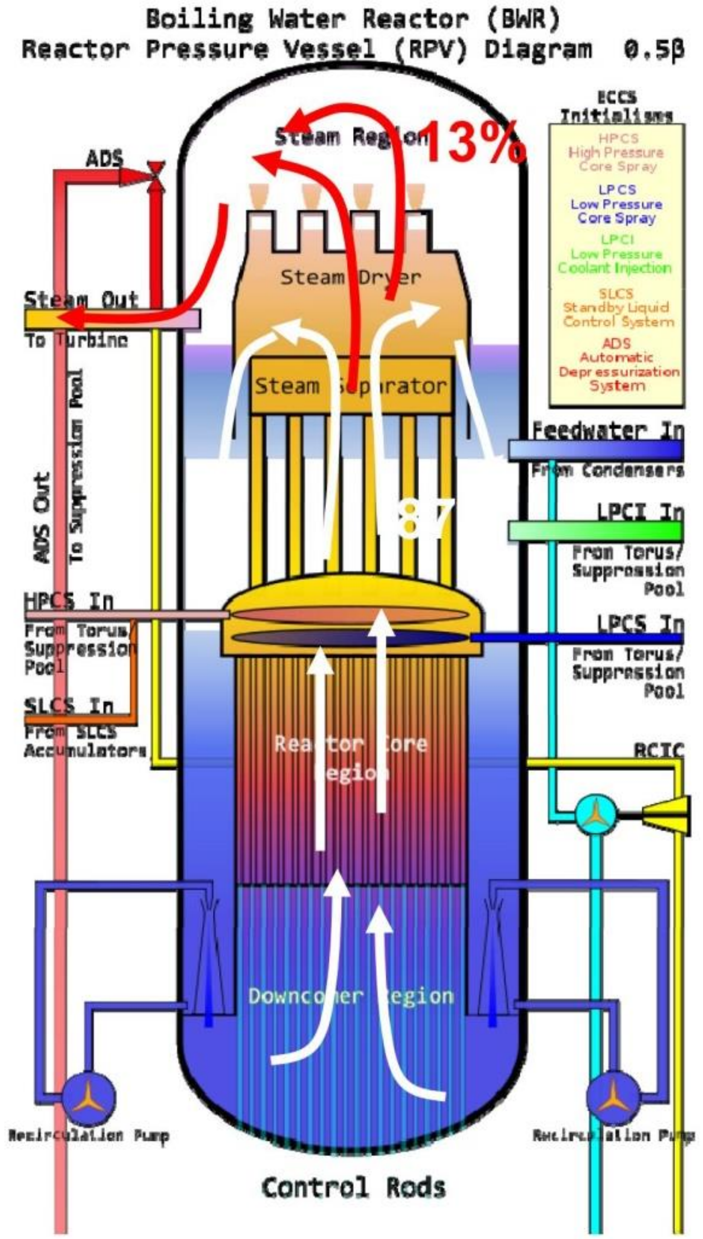
Vapor and Water in BWR Core

Existing BWR: Core exit quality = 13%

- 13% of the inlet water becomes vapor, the rest 87% is re-circulated as saturation water.
- The vapor works at the turbine, and is fed back to the core as condensed water.

The core power is transferred to the turbine only by the vapor. Recirculation water is neglected from the view point of energy balance.

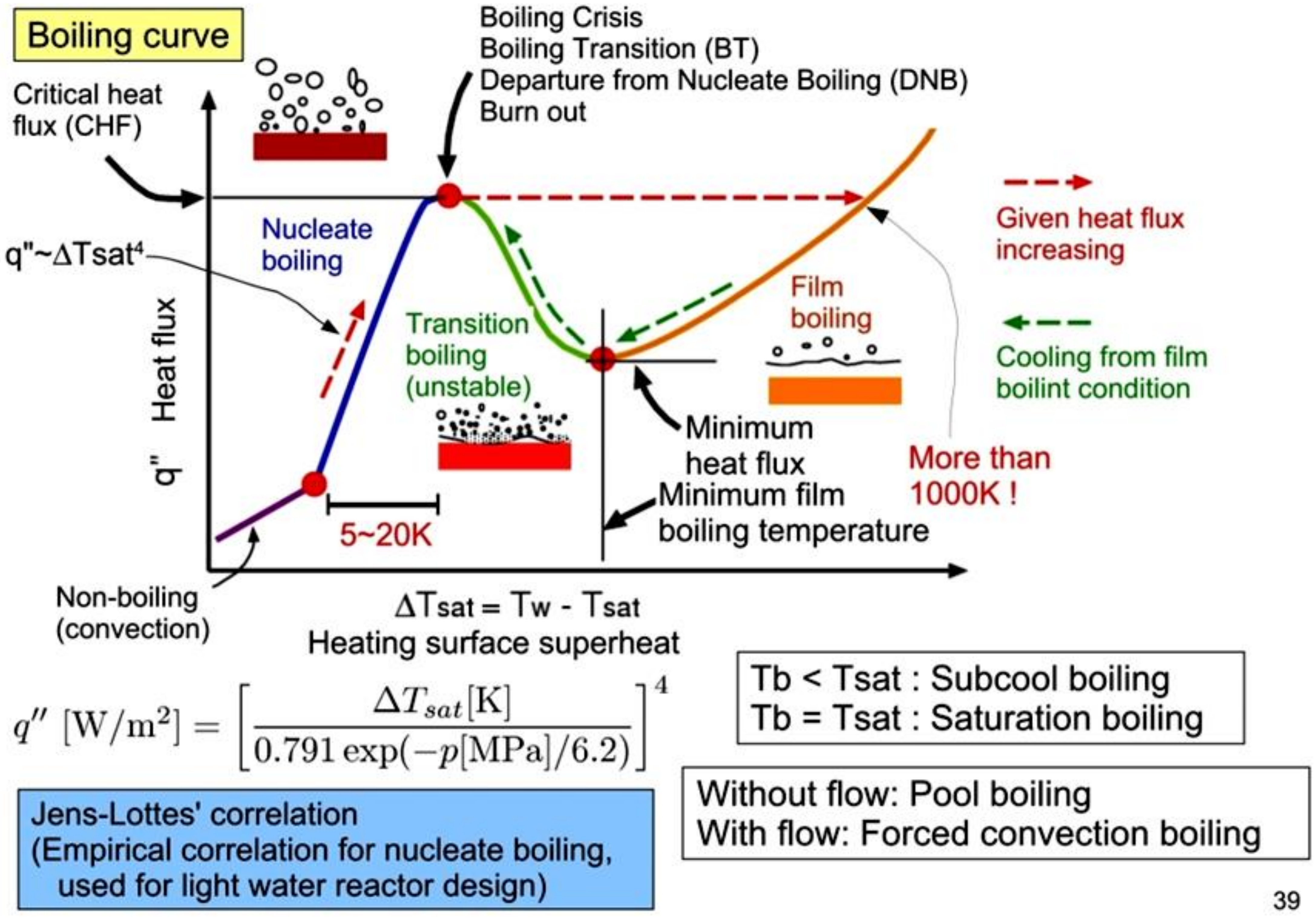
(The same holds in the secondary loop of a PWR)



BWR thermal power:

W = 14000 kg/s (P = 7 MPa)
h_{out} / h_{in} ≈ 1450 / 1200 kJ/kg
$h = h_{out}$
Δh_{fg} = 1500 kJ/kg
h_{is} = 1260 kJ/kg
x ≈ 13 %
Q = 3500 MWt

Thermal Hydraulic Safety Margin in PWR and BWR Core



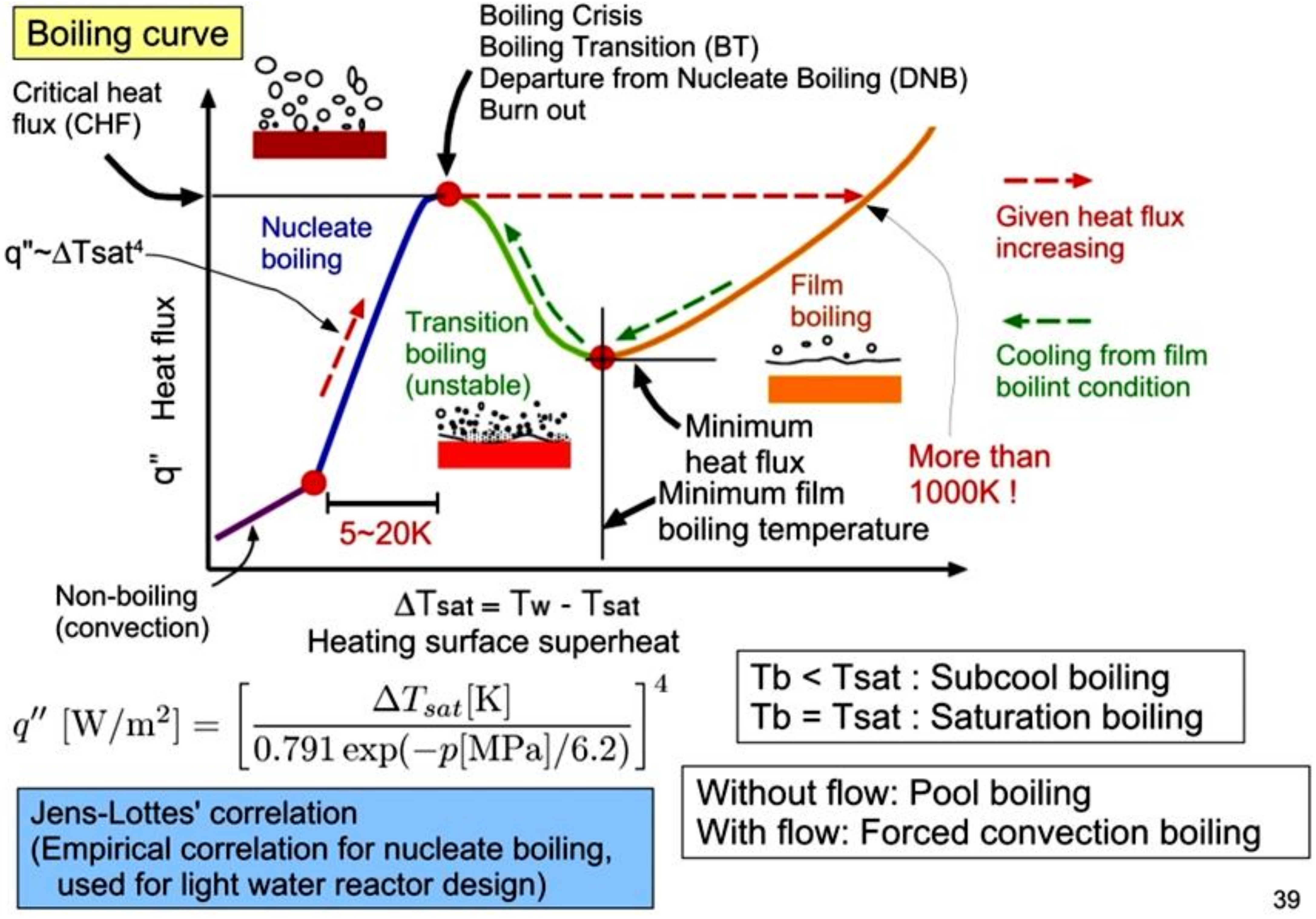
$$q'' \ [\mathrm{W/m^2}] = \left[\frac{\Delta T_{sat}[\mathrm{K}]}{0.791\exp(-p[\mathrm{MPa}]/6.2)}\right]^4$$

Jens-Lottes' correlation
(Empirical correlation for nucleate boiling, used for light water reactor design)

Tb < Tsat : Subcool boiling
Tb = Tsat : Saturation boiling

Without flow: Pool boiling
With flow: Forced convection boiling

- q'' = heat flux from rod to coolant
- CHF = maximum heat removal capability
- ΔT_{sat} = should stable or below CHF point
- Nucleate boiling: most efficient region for heat transfer and fuel cooling
  - ✓ Subcooled boiling (PWR): $T_{bulk} < T_{sat}$
  - ✓ Nucleate boiling (BWR):
- After CHF: T_w high, wall (cladding) overheating
  - ➢ Point of safety in PWR and BWR



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Tb = Tsat : Saturation boiling

Without flow: Pool boiling
With flow: Forced convection boiling

TH safety margin in PWR:

- Minimum Departure from Nucleate Boiling (MDNBR)
- DNBR: $\frac{CHF}{q''actual}$
- $q''\ actual > CHF$: lower than MDNBR
  - ✓ Typical PWR: 1.30 – 1.35
  - ✓ Advanced PWR: 1.4 – 1.8

TH safety margin in BWR:

- Minimum Critical Power Ratio (MCPR)
- CPR: $\frac{critical\ power}{actual\ power}$
- Power as dryout occurs
- Actual power defines the annular flow (liquid film covering the wall)
  - Typical BWR: ~ 1.25–1.3

Thermal Hydraulic Safety Limits

MDNBR

Minimum Departure from Nucleate Boiling Ratio
(The most critical TH safety limit in PWRs)

- Ratio of CHF to local heat flux.
- MDNBR > 1.3–1.5 to prevent film boiling and fuel damage..

Peak Fuel Temperature

T_fuel,max < T_melt

UO_2 melts at ~2,865 °C. Peak centerline temperature must stay well below this, typically designed for < 2,200 °C to allow margin.

Peak Cladding Temperature

PCT in LOCA Scenarios

During a Loss-of-Coolant Accident (LOCA), PCT must stay below 1,204 °C (2,200 °F) per NRC regulations to prevent cladding oxidation and failure.

Coolant Void Fraction

Steam quality limit

In BWRs, high void fractions reduce moderator density and neutron moderation. TH and neutronics are tightly coupled for stability.

Analysis Tools and Methods

System Codes

RELAP5 / RELAP-7 — transient TH analysis
TRACE — NRC best-estimate code
CATHARE — European system code
Used for LOCA, transient, safety analysis

Sub-channel Codes

COBRA-TF / VIPRE — detailed core analysis
FLICA — French sub-channel code
Resolve local heat flux and DNB conditions
Used for fuel assembly design

CFD Codes

ANSYS CFX / Fluent — general CFD
OpenFOAM — open-source CFD
Star-CCM+ — industrial simulations
Detailed 3-D flow and heat transfer

Coupled Codes

Neutronics + TH coupling essential
PARCS/RELAP5, SIMULATE-3K
Multi-physics for transient analysis
Required for Anticipated Transient Without Scram (ATWS)

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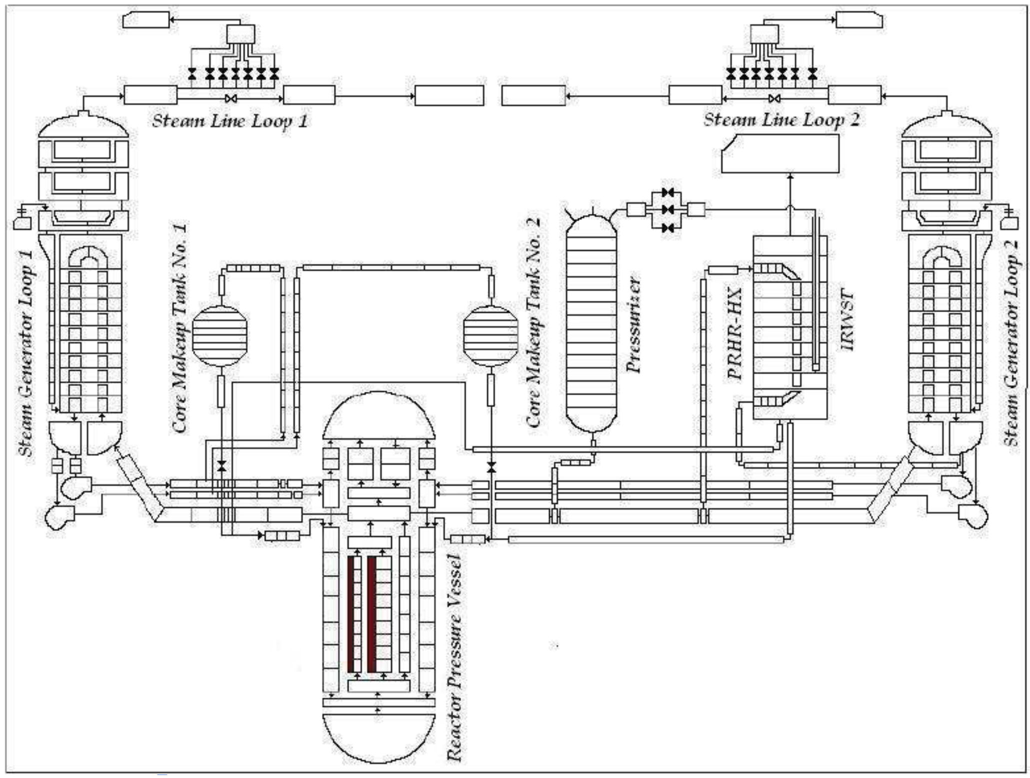
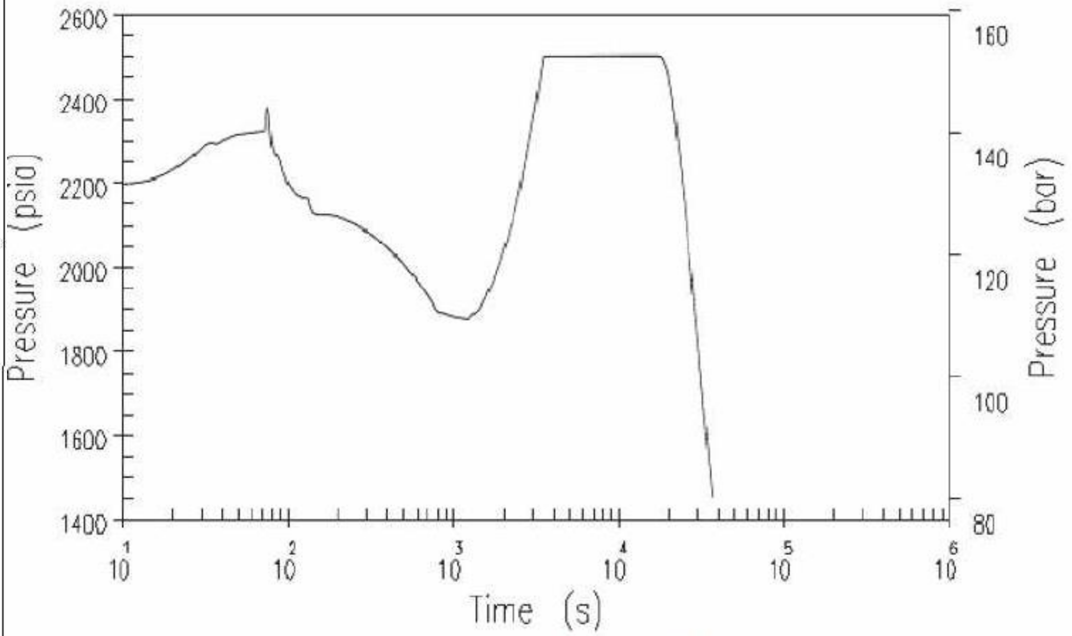
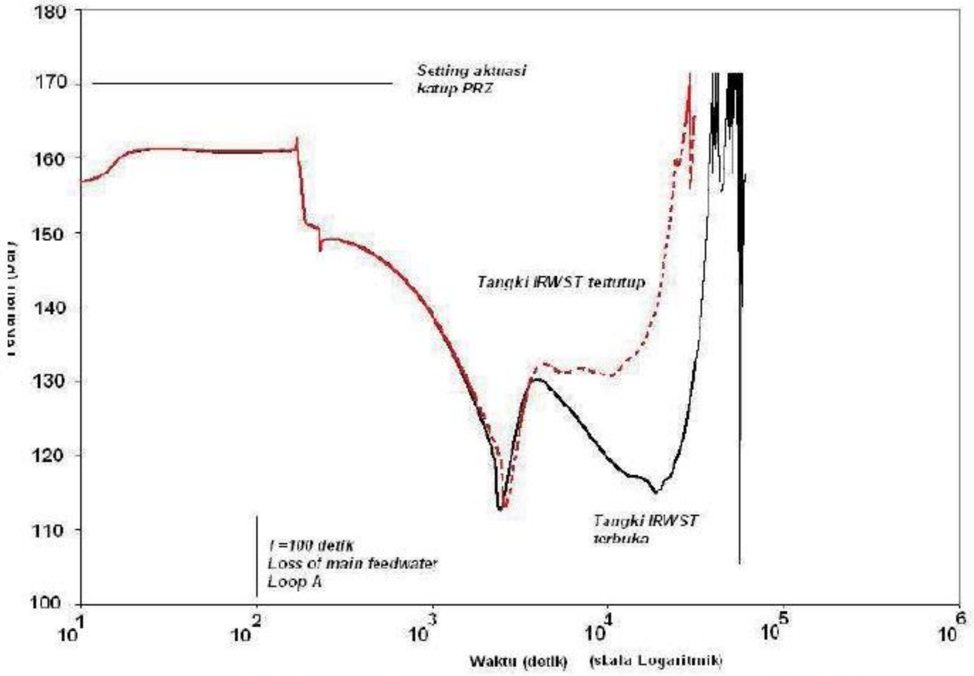
Pengembangan model AP1000 untuk simulasi kecelakaan





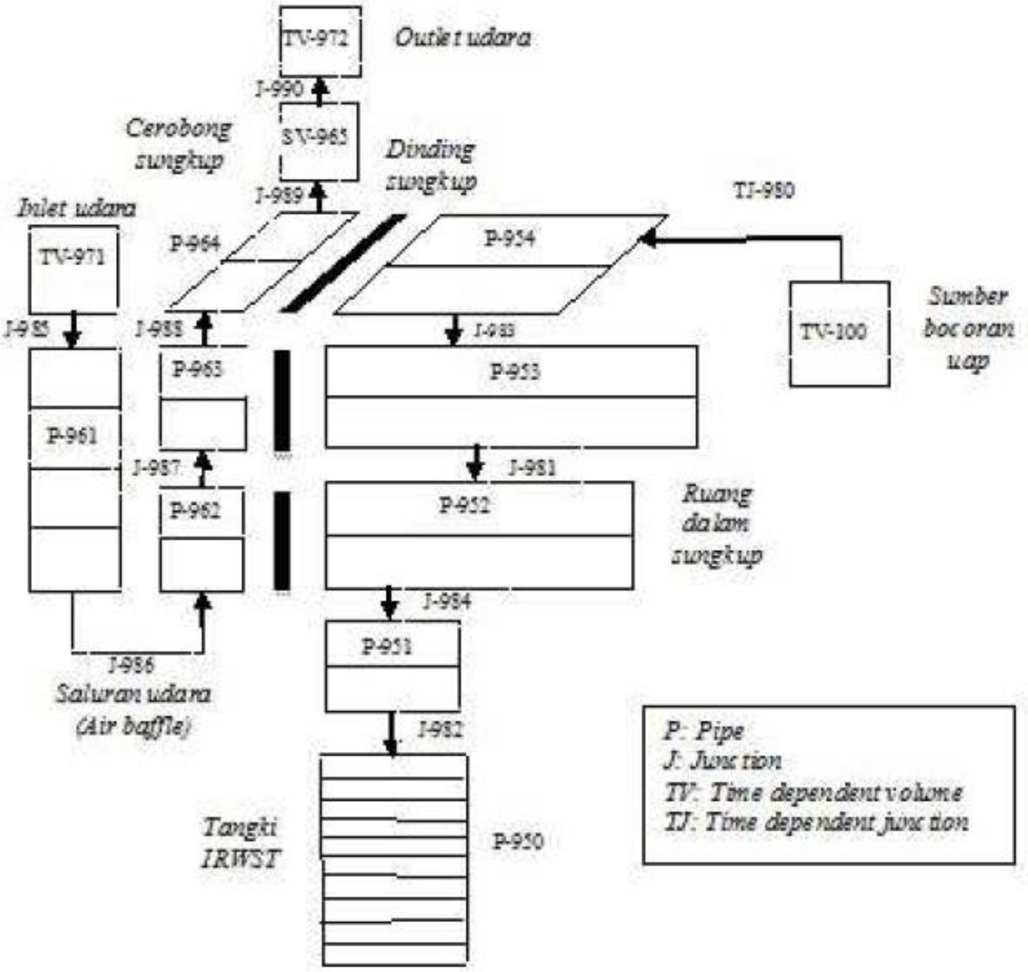
Gambar 5. Perubahan tekanan primer hasil perhitungan LOFTRAN [8]

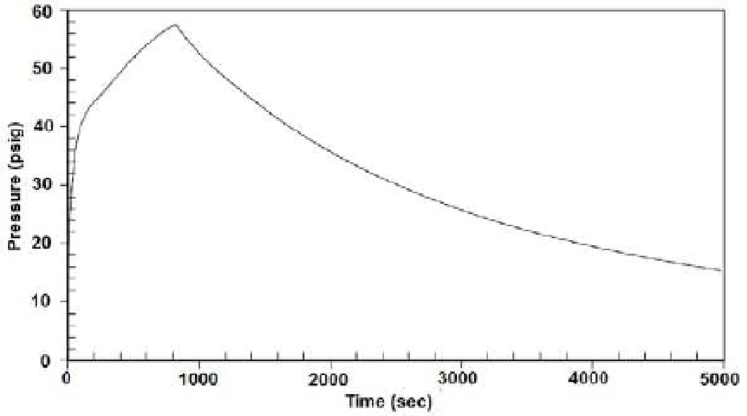


Gambar 6. Perubahan tekanan primer hasil perhitungan RELAP5

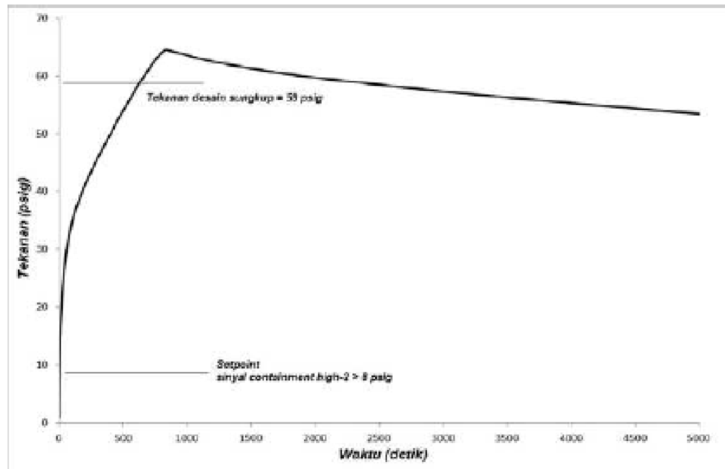
Penambahan sistem pendingin pasif:

- ADS, CMT,
- PRHR-HX, IRWST





Gambar 5: Respon tekanan sungkup dari WGOTHIC code [7]

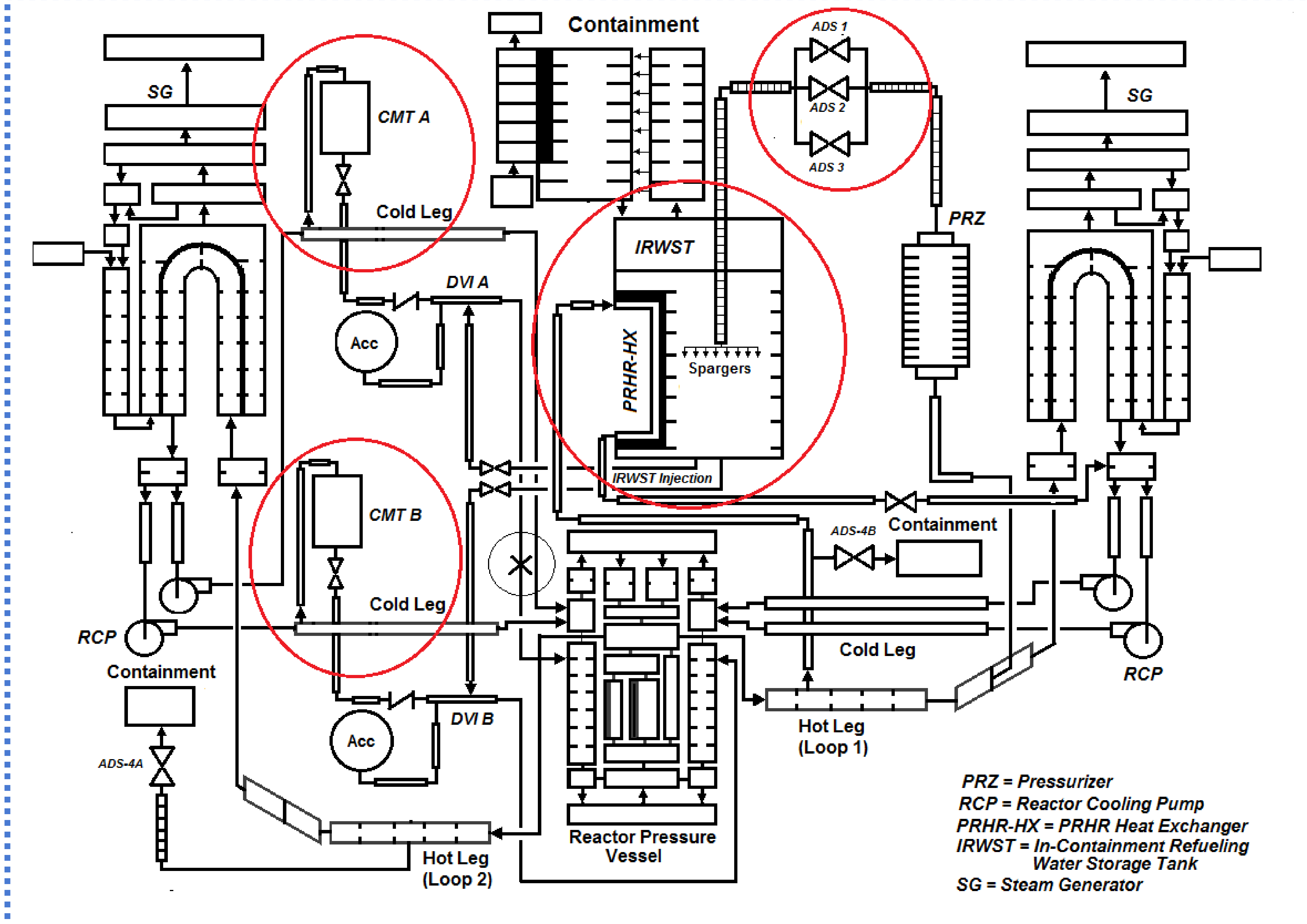


Gambar 6: Respon tekanan sungkup dari RELAP5

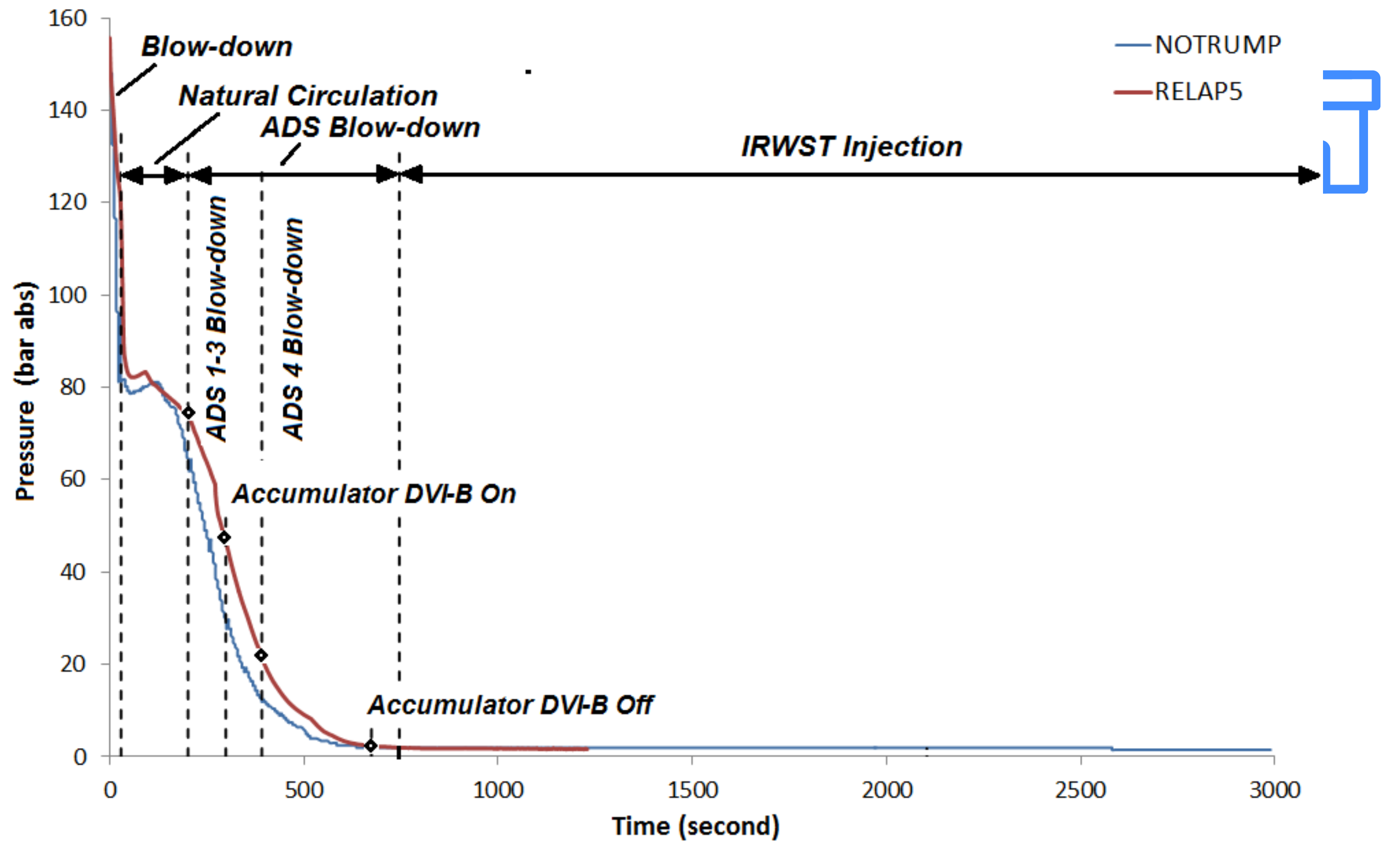
- Verifikasi Kecelakaan Hilangnya Aliran Air Umpan pada Reaktor Daya PWR Maju - Jurnal Tri Dasa Mega 2012
- Pemodelan Sistem Pendinginan Sungkup secara Pasif menggunakan RELAP5 - Jurnal Tri Dasa Mega 2012

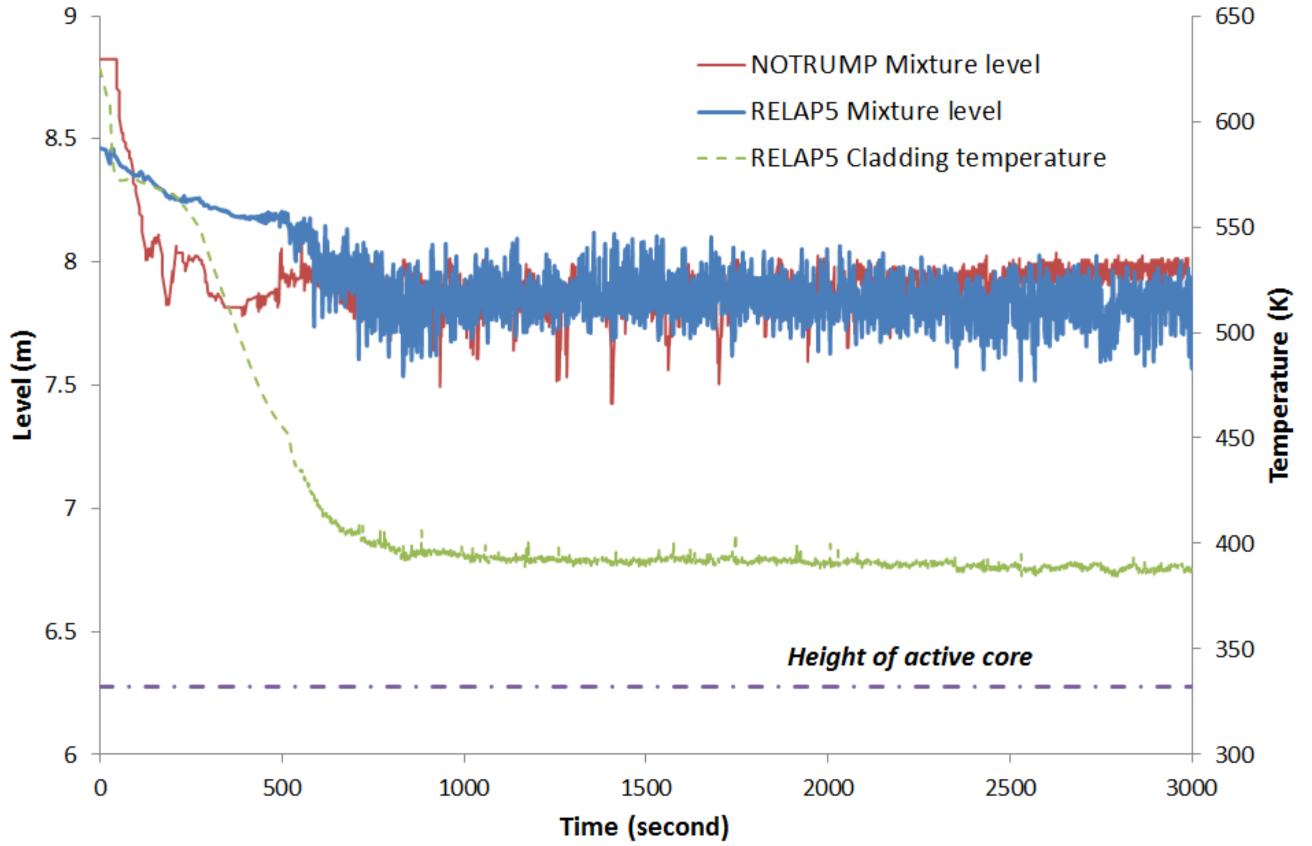
Pengembangan model AP1000 untuk simulasi kecelakaan

Performance Analysis of AP1000 Passive Systems During Direct Vessel Injection (DVI) Line Break- Atom Indonesia 2016



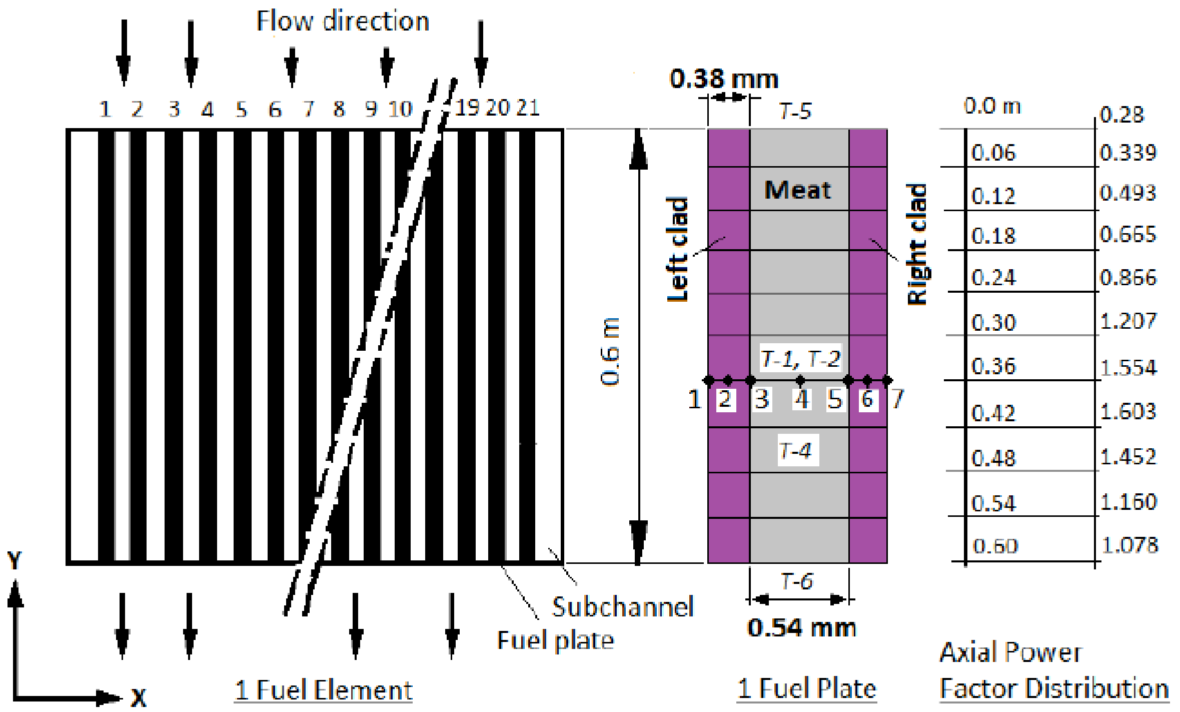
PRZ = Pressurizer
RCP = Reactor Cooling Pump
PRHR-HX = PRHR Heat Exchanger
IRWST = In-Containment Refueling Water Storage Tank
SG = Steam Generator

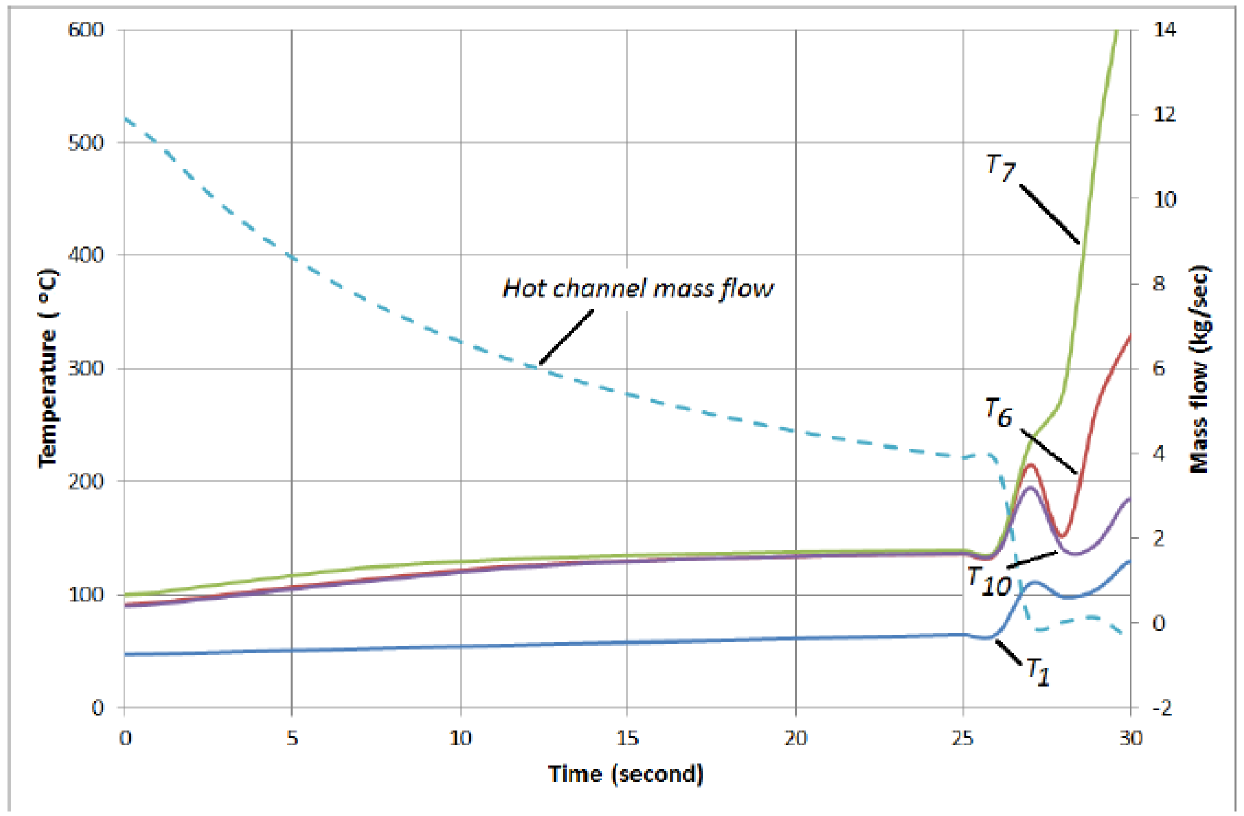


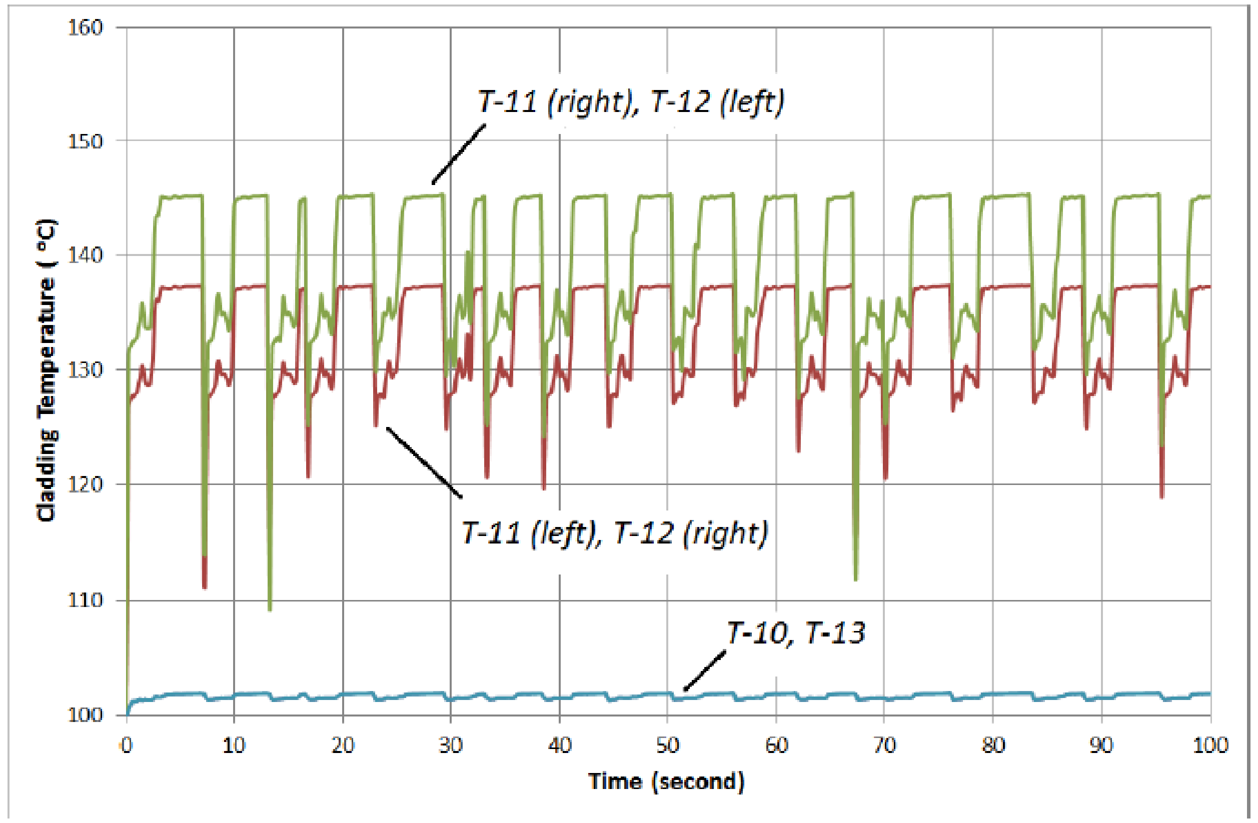


Pengembangan model RSG-GAS untuk simulasi kecelakaan PARAH

RELAP5 Simulation For Severe Accident Analysis of RSG-GAS Reactor – Jurnal Tri Dasa Mega 2018

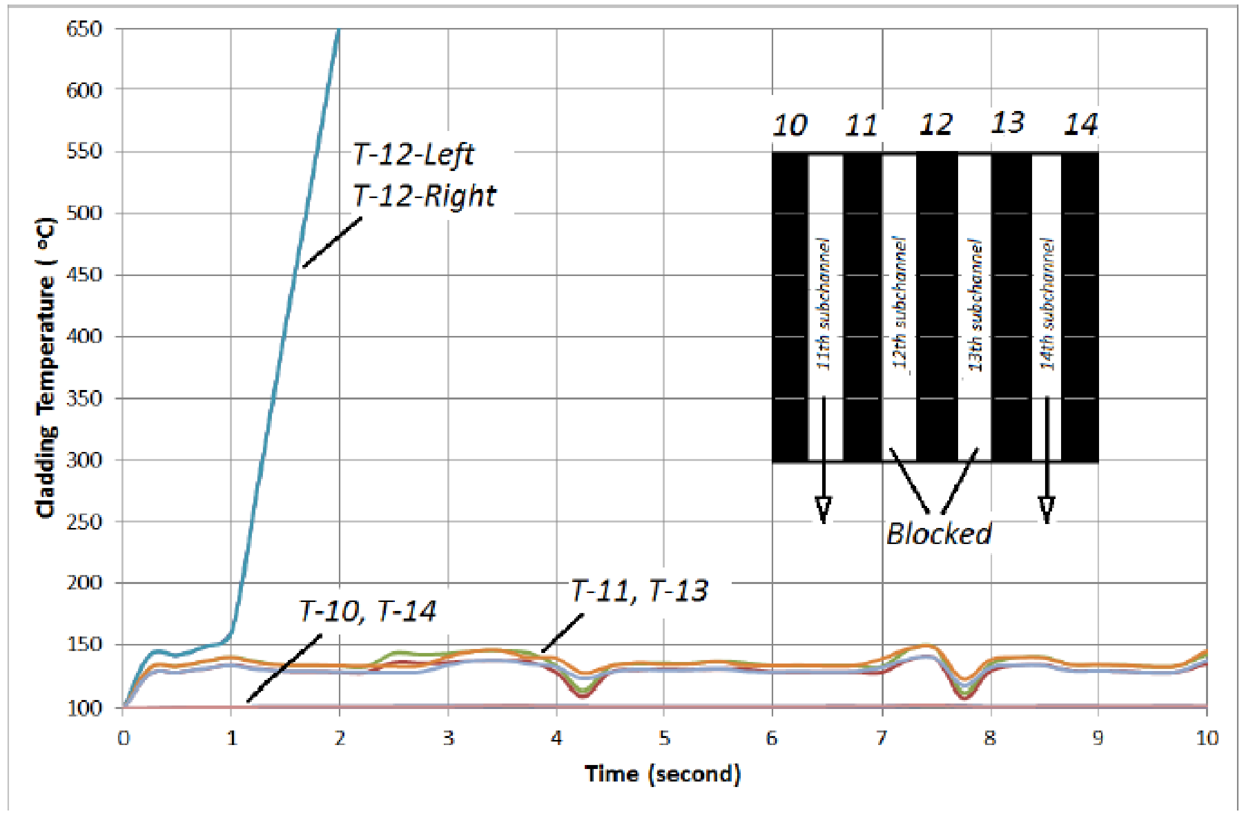






T cladding during LOFA, Full Power, no Trip

T cladding during 1 blocked subchannel, Full Power, no Trip



T cladding during 2 blocked subchannel, Full Power, No Trip

Country	Operational	Under Construction / Planned	Permanently Shutdown
United States	~94	0 planned large units	~ 42
France	57	0	14
China	~57	35+	Few
Russia	36	5	~8
Japan	14 operational	2 under construction	~30
South Korea	26	2	2
Canada	19	0	~6
Ukraine	15*	2	4
India	24	8	0
United Kingdom	9	2	~35
Sweden	6	0	7
Spain	7	0	3
Belgium	5	0	2
Switzerland	4	0	1
Finland	5	0	0
Czech Republic	6	Planned SMR/new units	0
Slovakia	5	1	0
Hungary	4	1	0
Romania	2	Planned	0
Bulgaria	2	Planned	4
Pakistan	6	1	0
Iran	1	1	0
Bangladesh	0	2	0
Belarus	2	0	0
Brazil	2	1	0
Argentina	3	1	2
Mexico	2	0	0
South Africa	2	Planned	0
Armenia	1	Planned replacement	0
Slovenia	1	Planned expansion	0
Netherlands	1	Planned	0
UAE	4	Planned expansion	0
Turkey	0	4	0

Status (2025)	Approximate Number
Operational reactors	415–417
Under construction	72–75
Permanently shutdown	210+

Reactor Type	Approximate Number Operating	Share of World Fleet
PWR	~300–310	~70%
BWR	~60–65	~15%
PHWR / CANDU	~45	~10%
Gas-cooled reactors (AGR/GCR/HTGR)	~15	~3%
Fast reactors (FBR)	~2–4	<1%
LWGR / RBMK	~8–10	~2%
Other types	Few	<1%

Terima kasih