Research and Development of Supercritical-pressure light water cooled reactors, Super LWR and Super FR

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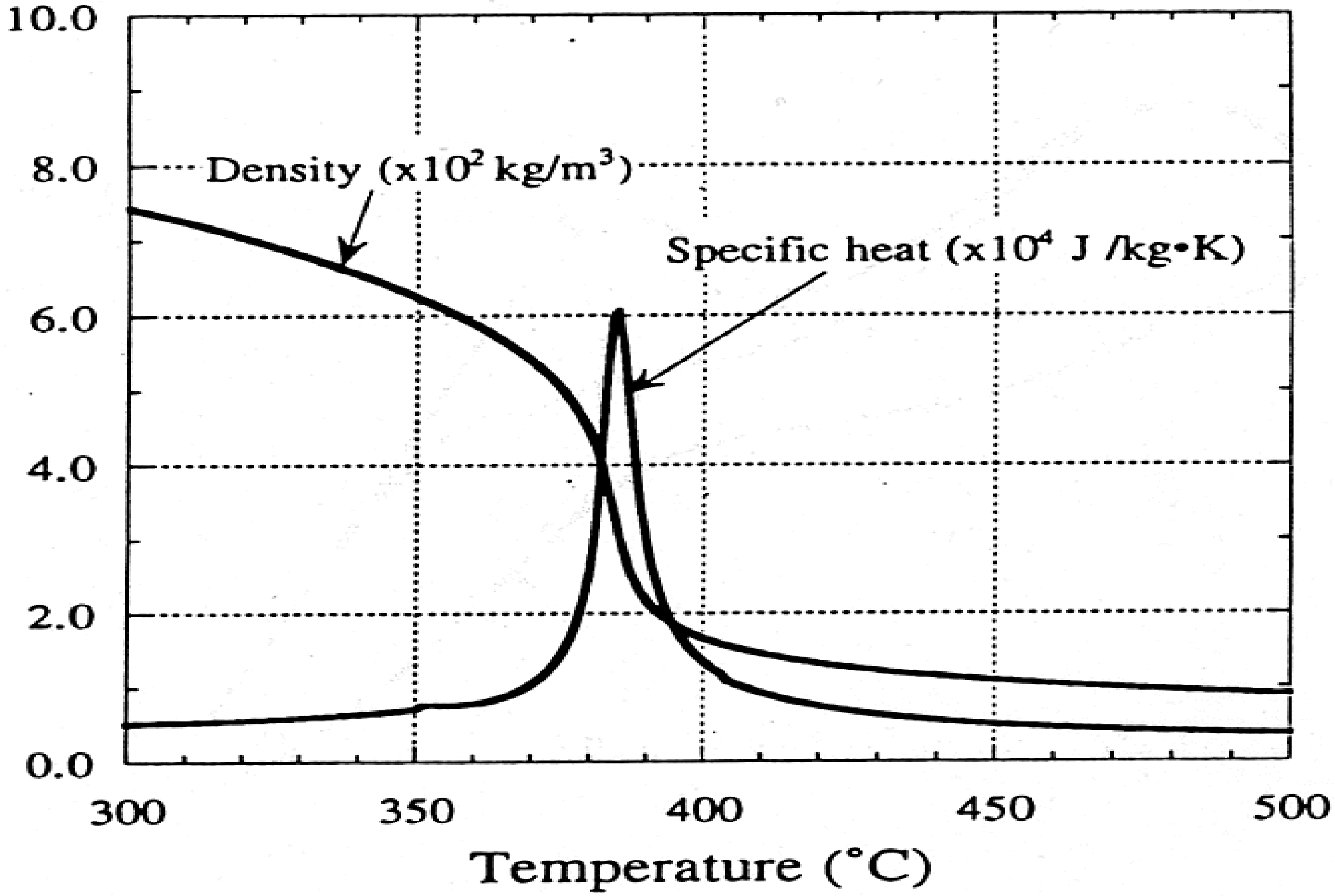
Outline

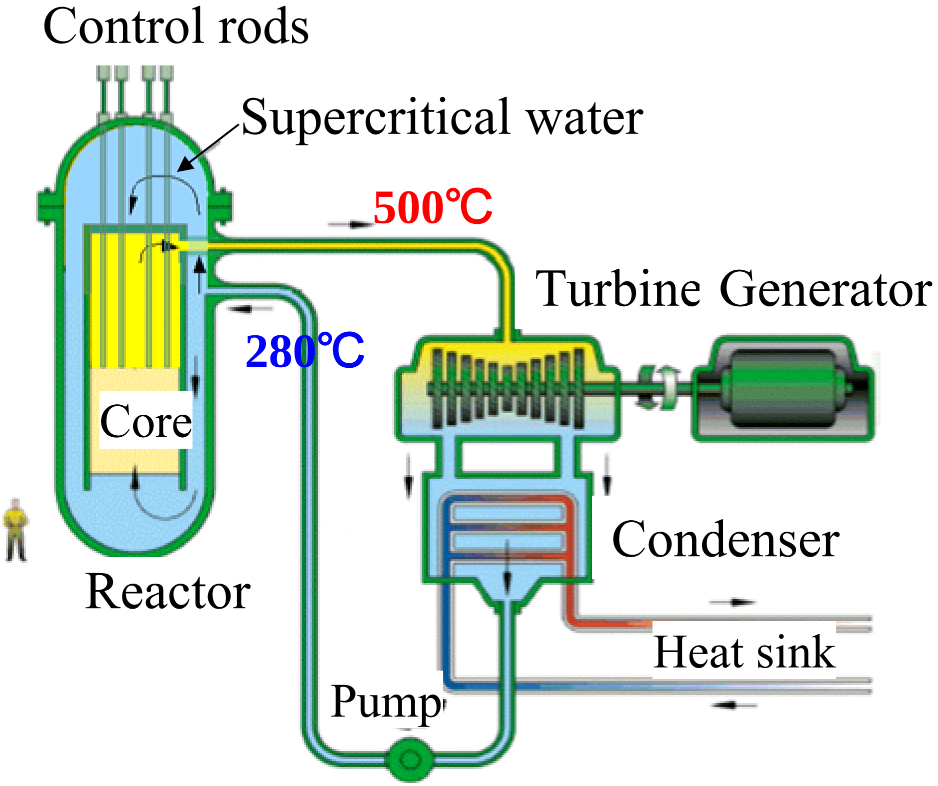
1. Introduction
2. Fuel and core design
3. Safety
4. Fast reactor
5. R&D



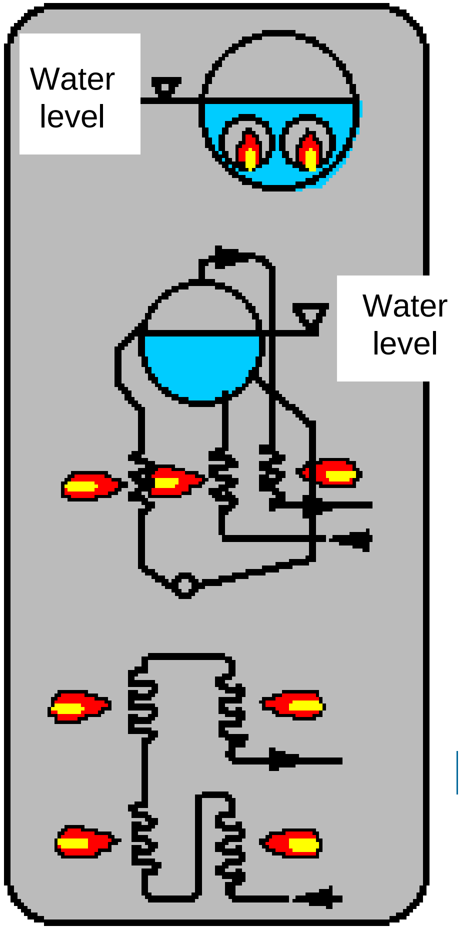
Change of density and specific heat of water with temperature at supercritical pressure (25 MPa)

Super LWR

- Super LWR: Supercritical-pressure light water cooled and moderated reactor developed at Univ. of Tokyo
- Once-through direct cycle thermal reactor



- Pressure: 25 MPa
- Inlet: 280℃
- Outlet (average): 500℃
- Flow rate: 1/8 of BWR



Circular Boiler

↓

Water tube boiler

↓

Once-through boiler

LWR

↓

Super LWR, Super FR (SCWR)

Evolution of boilers

Supercritical fossil-fired power plants

Once-through boilers

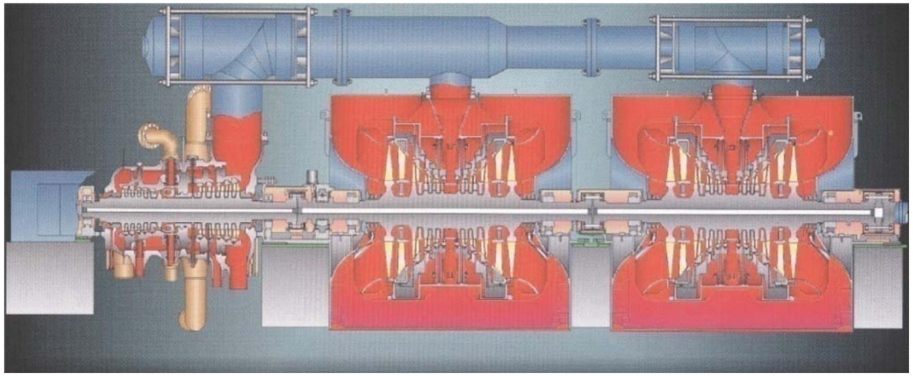
Number of units are larger than that of LWRs.

Proven technologies; turbines, pumps, piping etc.

USA; developed in 1950’s, Largest unit is 1300MWe.

Japan; deployed in 1960’s and constantly improved.

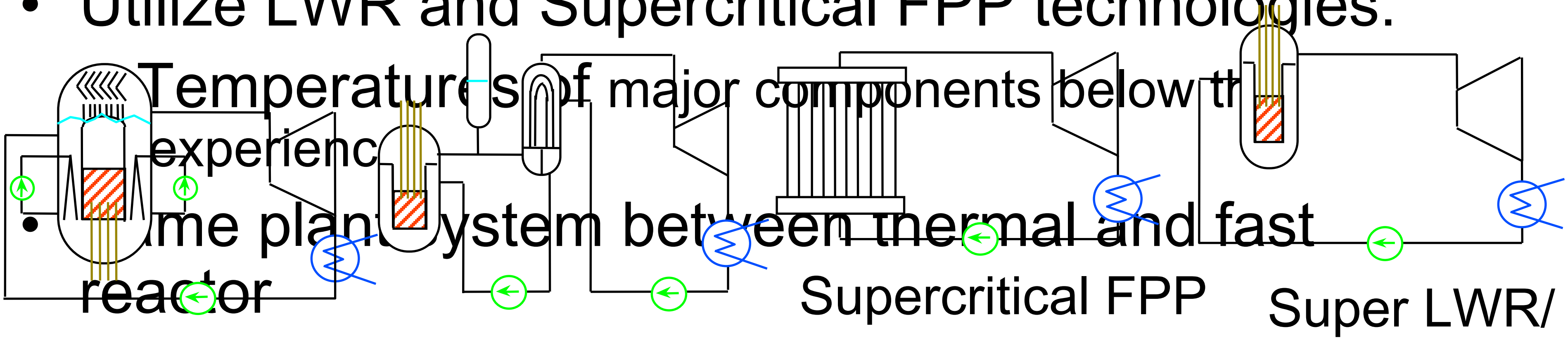
Many plants in Russia and Europe.



Compact SC turbine (700MWe,31.0MPa,566°C)

Features of Super LWR/Super FR

- Compact & simple plant systems; Capital cost reduction
  - No steam/water separation and no SGs: Coolant enthalpy inside CV is small.
  - High specific enthalpy & low flow rate: Compact components
- High temperature & thermal efficiency (500C, ~44%)
- Utilize LWR and Supercritical FPP technologies:
  - Temperatures of major components below th experienc
- ame plant ystem between thermal and fast reactor



BWR PWR Supercritical FPP (once-through boiler) Super LWR/ Super FR

Fuel and core design

Core design criteria

Thermal design criteria

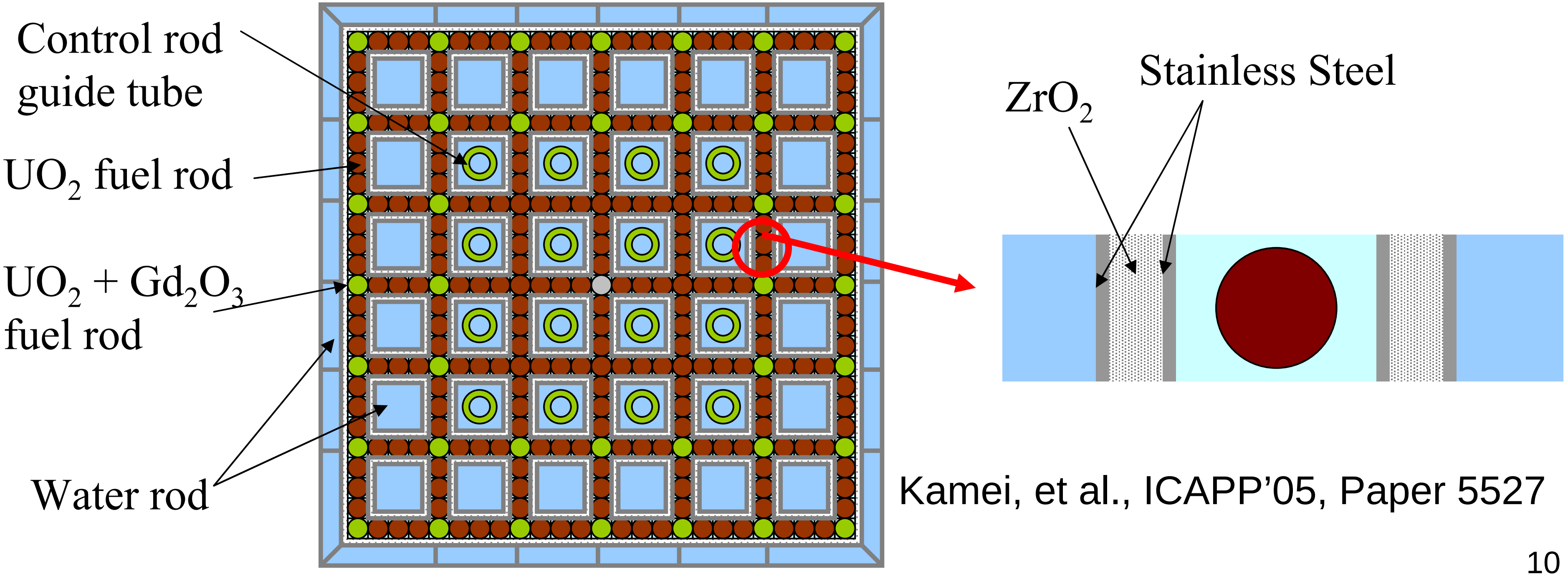
- Maximum linear heat generation rate (MLHGR) at rated power ≦ 39kW/m
- Maximum cladding surface temperature at rated power ≦ 650C for Stainless Steel cladding
- Moderator temperature in water rods ≦ 384C (pseudo critical temperature at 25MPa)

Neutronic design criteria

- Positive water density reactivity coefficient (negative void reactivity coefficient)
- Core shutdown margin ≧ 1.0%ΔK/K

Fuel assembly (example)

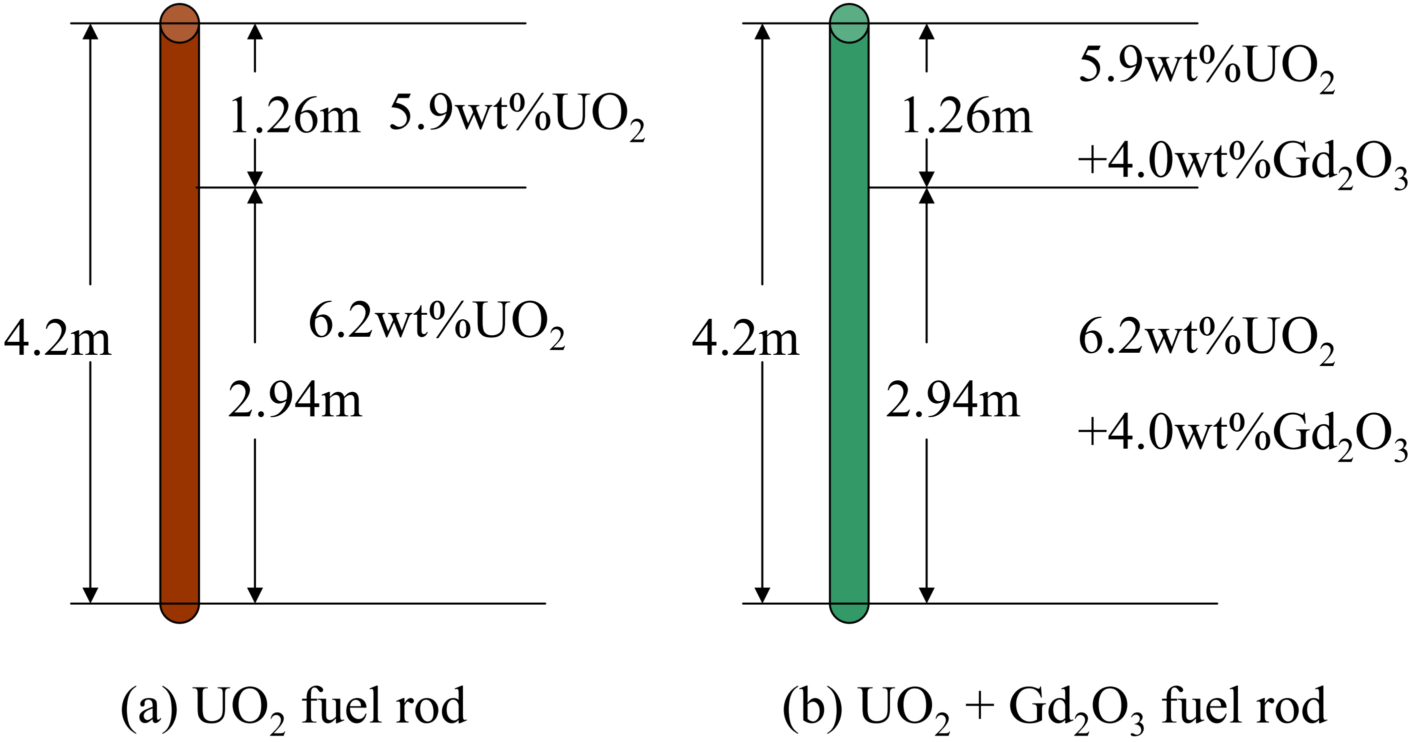
Design requirements →	Solution
Low flow rate per unit power (< 1/8 of LWR) due to large ⊿T of once-through system	Narrow gap between fuel rods to keep high mass flux
Thermal spectrum core	Many/Large water rods
Moderator temperature below pseudo-critical	Insulation of water rod wall
Reduction of thermal stress in water rod wall	
Uniform moderation	Uniform fuel rod arrangement



Kamei, et al., ICAPP’05, Paper 5527

Fuel enrichment

- Fuel enrichment is divided into two regions to prevent top axial power peak
- Average fuel enrichment 6.11wt%



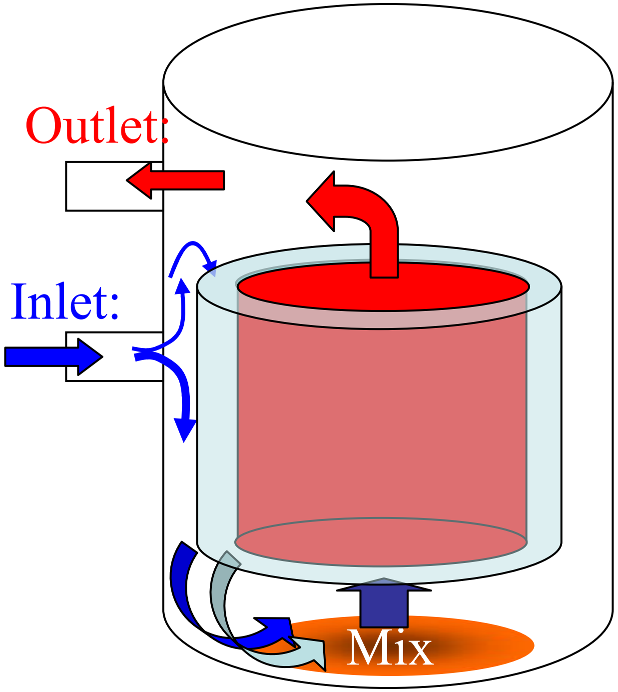
(a) UO_2 fuel rod (b) $UO_2 + Gd_2O_3$ fuel rod

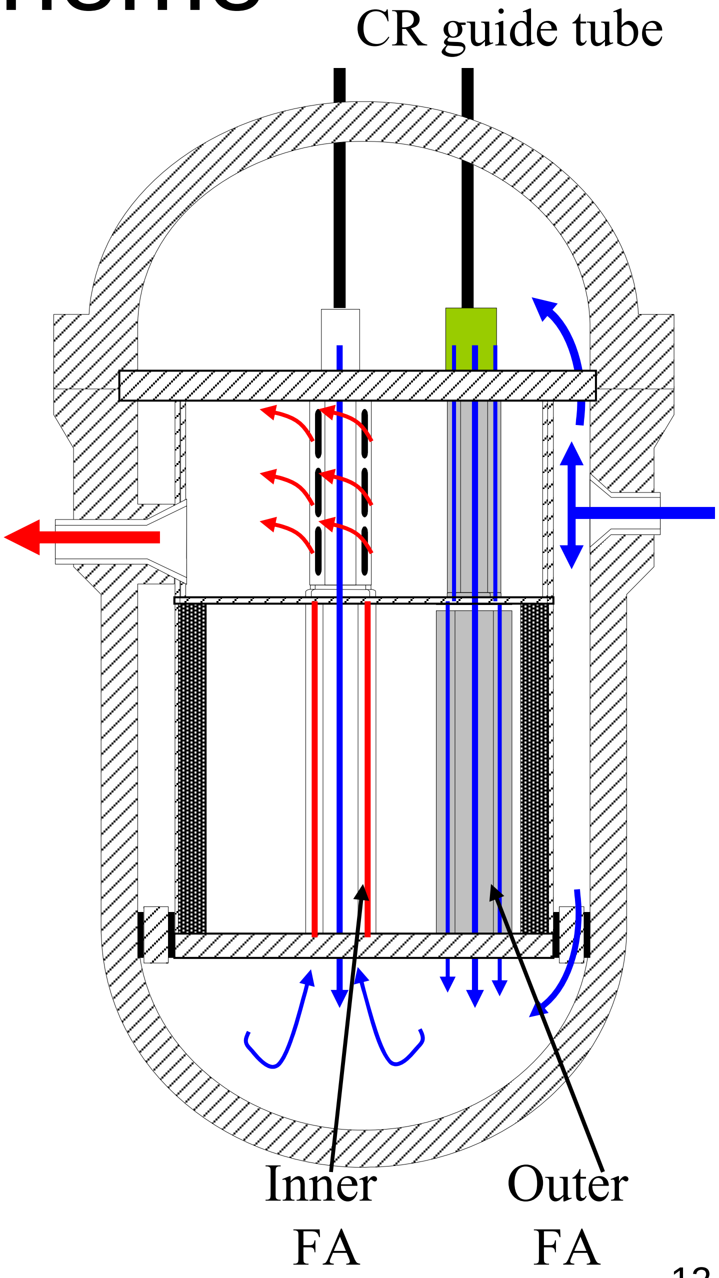
Coolant flow scheme

Flow directions

	Coolant	Moderator
Inner FA	Upward	Downward
Outer FA	Downward	Downward

To keep high average coolant outlet temperature



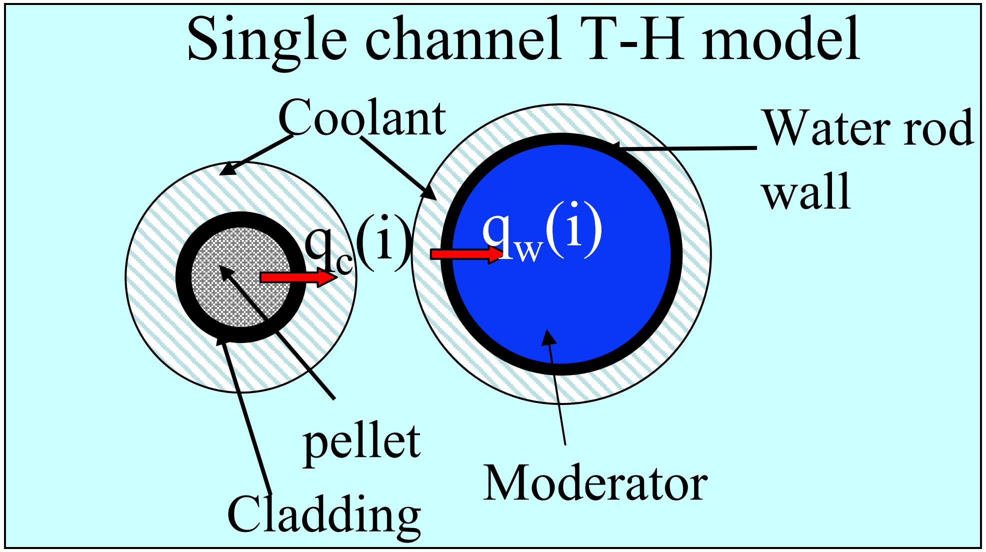


Kamei, et al., ICAPP’05, Paper 5527

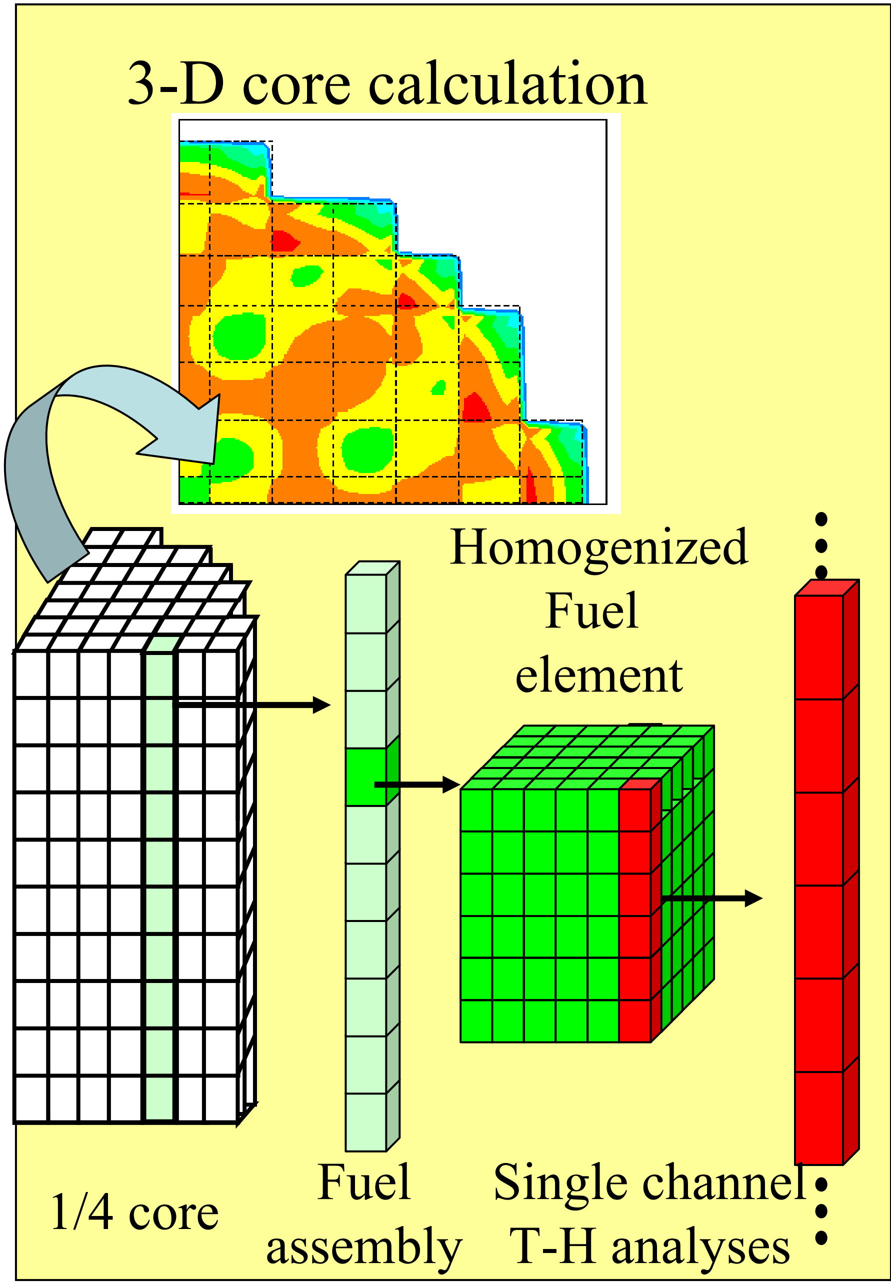
3-D N-T Coupled Core Calculation

- T-H calculation based on single channel model
- Neutronic calculation; SRAC

Core consists of homogenized fuel elements



Single channel T-H model

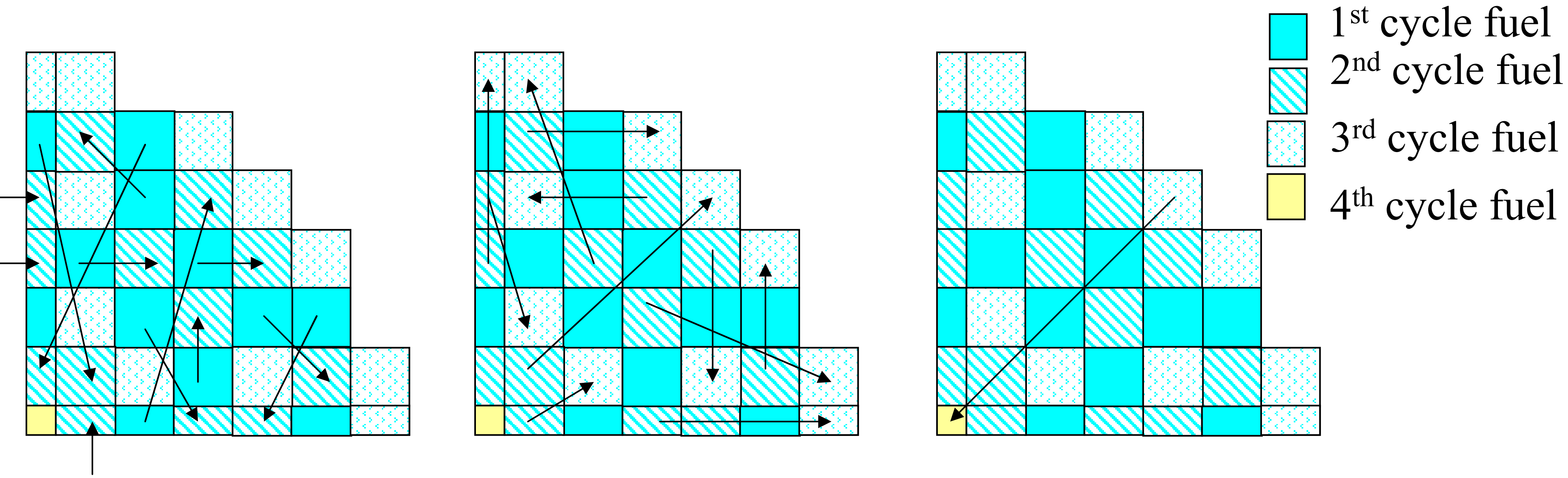


3-D core calculation

Fuel load and reload pattern

- 120 FAs of 1st ,2nd and 3rd cycle fuels and one 4th cycle FA
- 3rd cycle FAs which have lowest reactivity are loaded at the peripheral region of the core to reduce the neutron leakage

The low leakage core with high outlet temperature is made possible by downward flow cooling in peripheral FAs

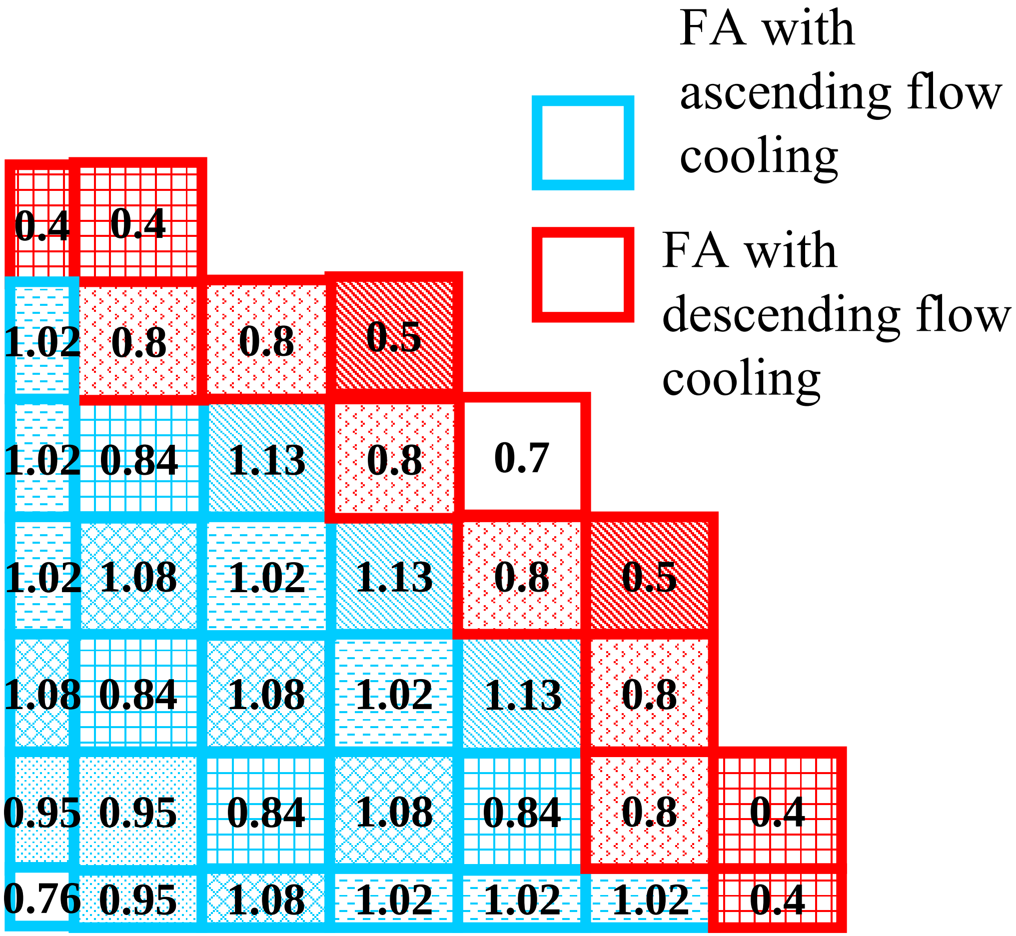


(a) 1st → 2nd cycle (b) 2nd → 3rd cycle (c) 3rd → 4th cycle

¼ symmetric core

Coolant flow rate distribution

- Flow rate to each FA is adjusted by an inlet orifice
- 48 out of 121FAs are cooled with descending flow



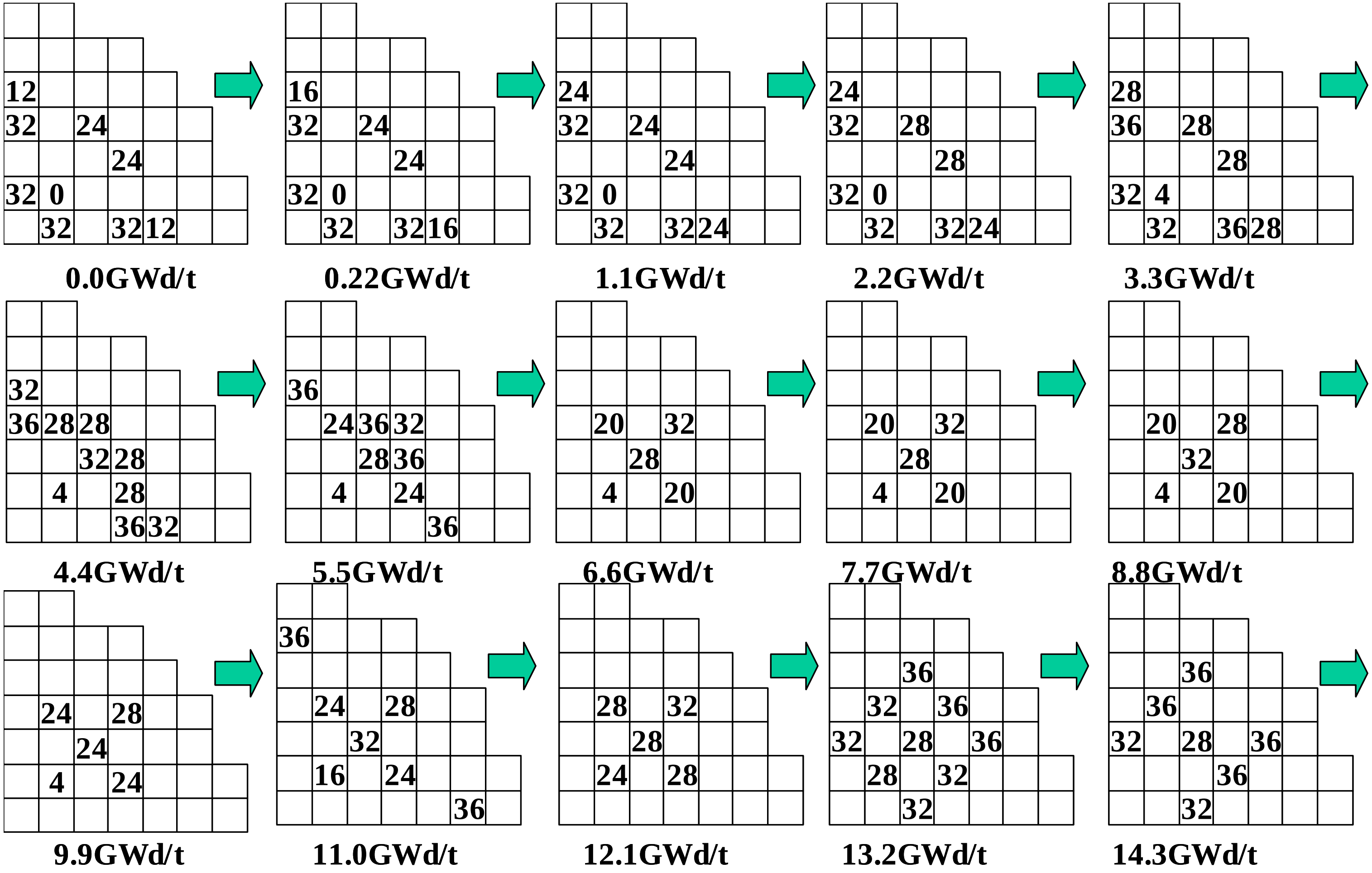
0.4	0.4					
1.02	0.8	0.8	0.5			
1.02	0.84	1.13	0.8	0.7		
1.02	1.08	1.02	1.13	0.8	0.5	
1.08	0.84	1.08	1.02	1.13	0.8	
0.95	0.95	0.84	1.08	0.84	0.8	0.4
0.76	0.95	1.08	1.02	1.02	1.02	0.4

Relative coolant flow distribution (1/4 core)

Control rod patterns

- X : withdrawn rate (X/40)   Blank box : complete withdrawal (X=40)

- At the EOC, some CRs are slightly inserted to prevent a high axial power peak near the top of the core



12						
32		24				
			24			
32	0					
	32		32	12		

0.0GWd/t

16						
32		24				
			24			
32	0					
	32		32	16		

0.22GWd/t

24						
32		24				
			24			
32	0					
	32		32	24		

1.1GWd/t

24						
32		28				
			28			
32	0					
	32		32	24		

2.2GWd/t

28						
36		28				
			28			
32	4					
	32		36	28		

3.3GWd/t

32						
36	28	28				
		32	28			
	4		28			
			36	32		

4.4GWd/t

36						
	24	36	32			
		28	36			
	4		24			
				36		

5.5GWd/t

	20		32			
		28				
	4		20			

6.6GWd/t

	20		32			
		28				
	4		20			

7.7GWd/t

	20		28			
		32				
	4		20			

8.8GWd/t

	24		28			
		24				
	4		24			

9.9GWd/t

36						
	24		28			
		32				
	16		24			
					36	

11.0GWd/t

	28		32			
		28				
	24		28			

12.1GWd/t

		36				
	32		36			
32		28		36		
	28		32			
		32				

13.2GWd/t

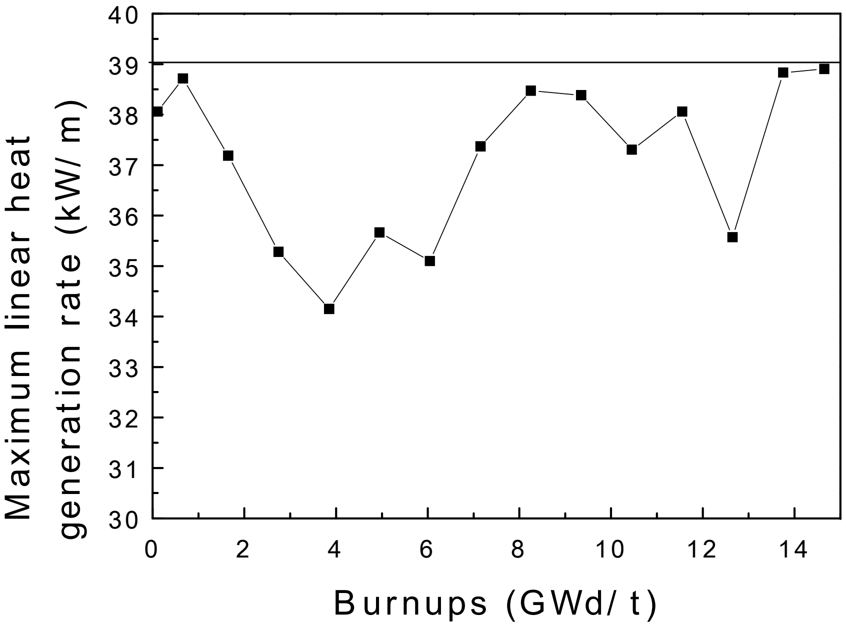
		36				
	36					
32		28		36		
			36			
		32				

14.3GWd/t

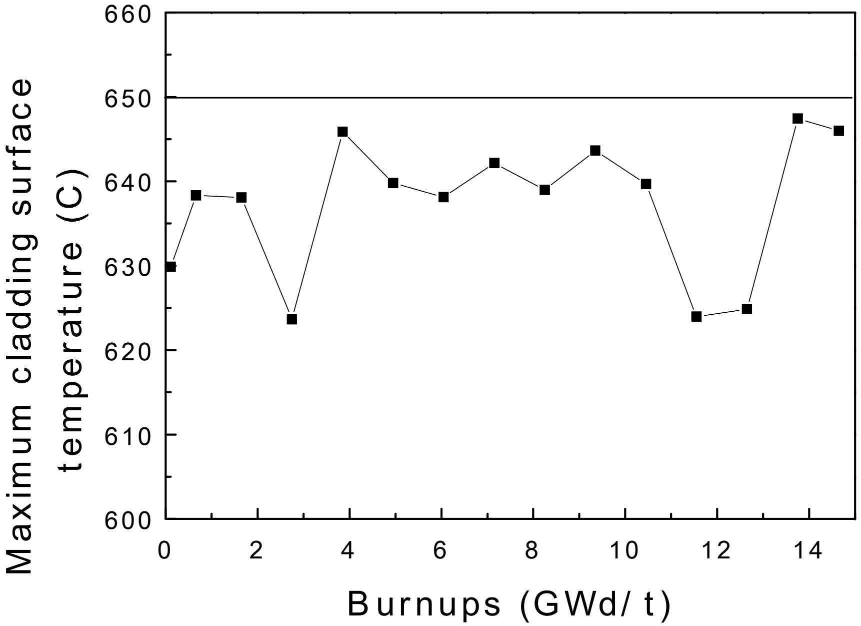
MLHGR and MCST

MLHGR and MCST are kept below 39kW/m and 650C throughout a cycle.

→ Thermal design criteria are satisfied



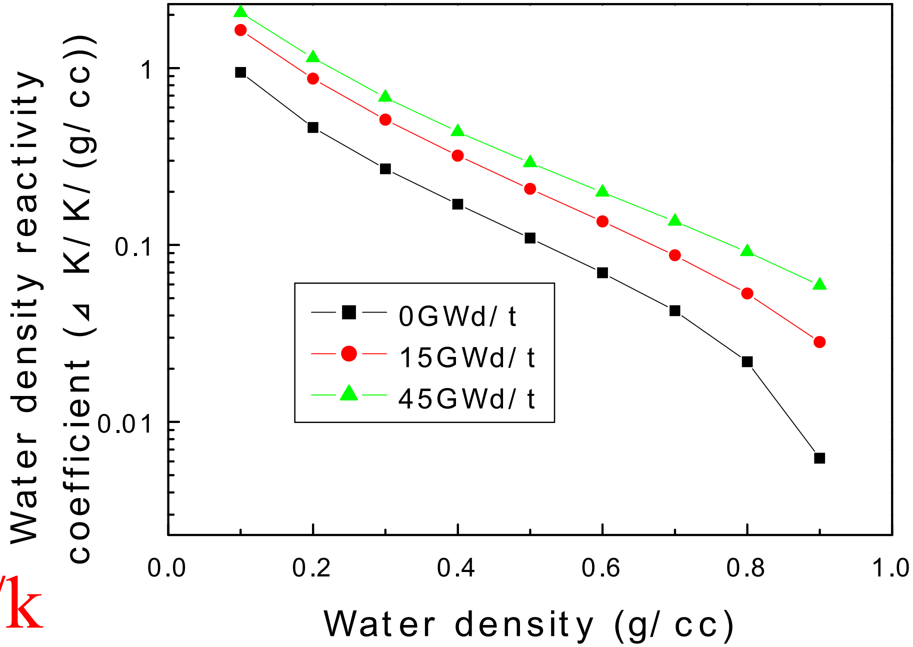
(a) MLHGR



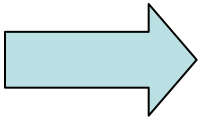
(b) MCST

Water density reactivity coefficient and Shutdown margin

- Positive water density reactivity coefficient (Negative void reactivity coefficient)



- Shutdown margin is 1.27 %dk/k
  - One rod stuck
  - Cold and clean core



Neutronic design criteria are satisfied

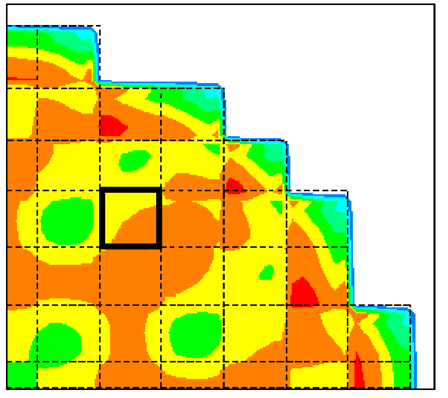
Super LWR characteristics summary

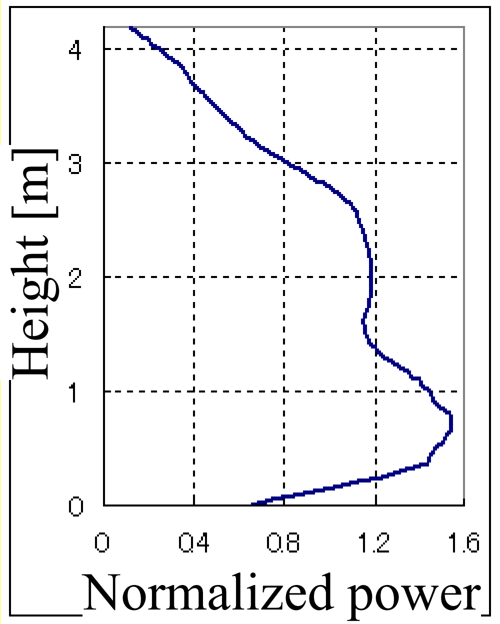
Core	Super LWR
Core pressure [MPa]	25
Core thermal/electrical power [MW]	2744/1200
Coolant inlet/outlet temperature [C]	280/500
Thermal efficiency [%]	43.8
Core flow rate [kg/s]	1418
Number of all FA/FA with descending flow cooling	121/48
Fuel enrichment bottom/top/average [wt%]	6.2/5.9/6.11
Active height/equivalent diameter [m]	4.2/3.73
FA average discharged burnup [GWd/t]	45
MLHGR/ALHGR [kW/m]	38.9/18.0
Average power density [kW/l]	59.9
Fuel rod diameter/Cladding thickness (material) [mm]	10.2/0.63 (Stainless Steel)
Thermal insulation thickness (material) [mm]	2.0 (ZrO_2)

Sub-channel analysis coupled with 3D core calculation

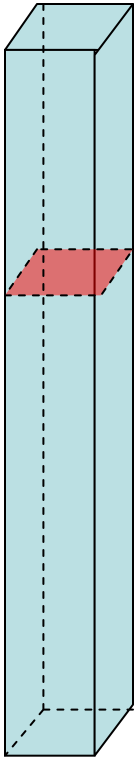
Reconstruction of pin power distributions

Core power distributions
(3-D core calculations)



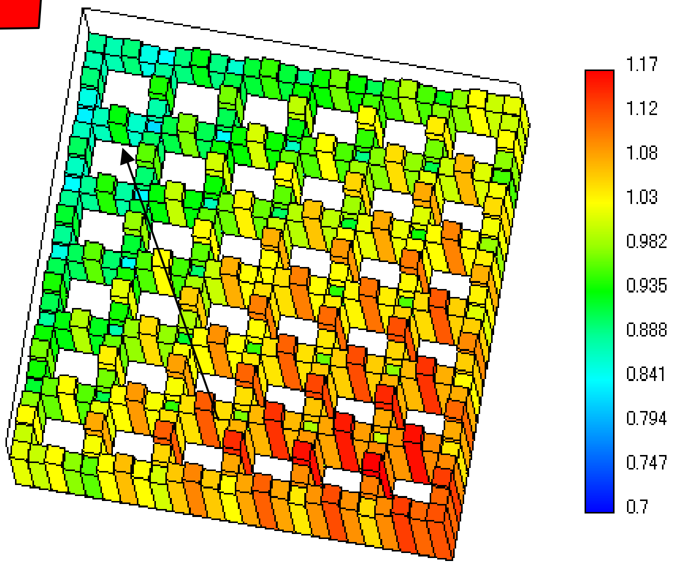


Homogenized
FA



Coupled subchannel analyses

Pin power distribution
f(burnup history,
density, CR insertion)

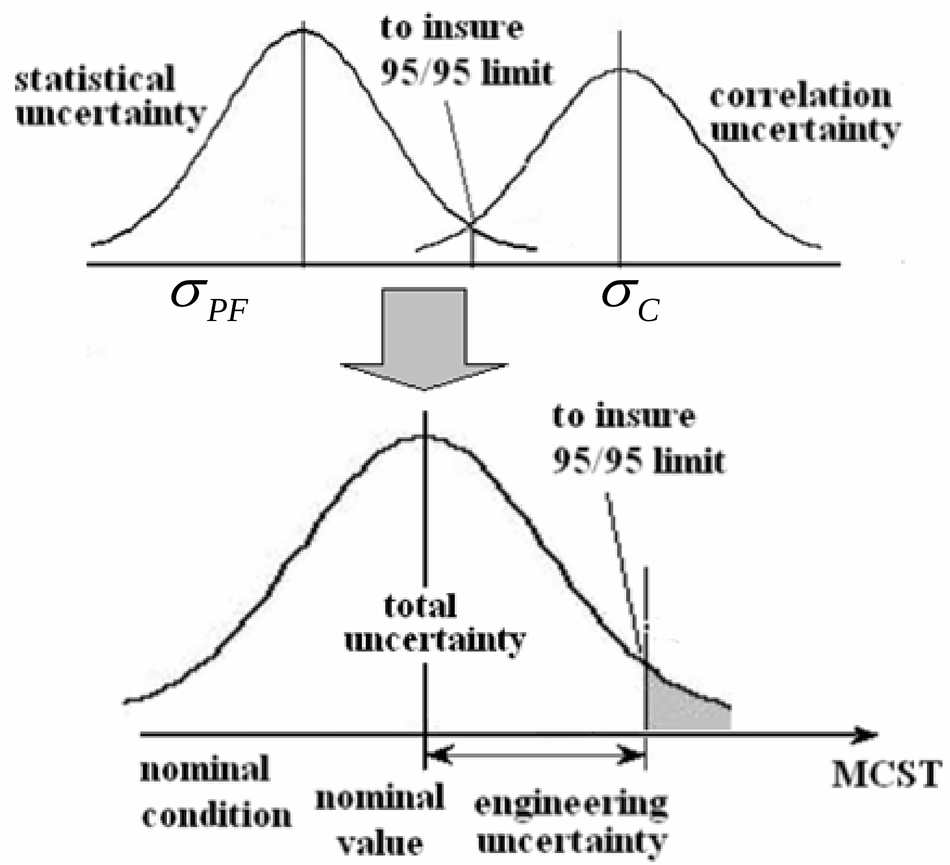


Reconstructed pin power distribution

Statistical thermal design

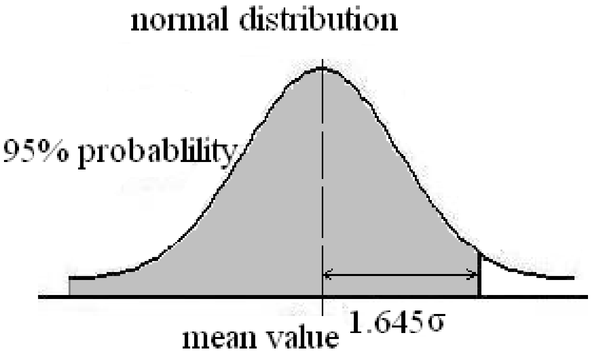
- Taking uncertainties into evaluation of peak cladding temperature

➢Monte Carlo statistical procedure

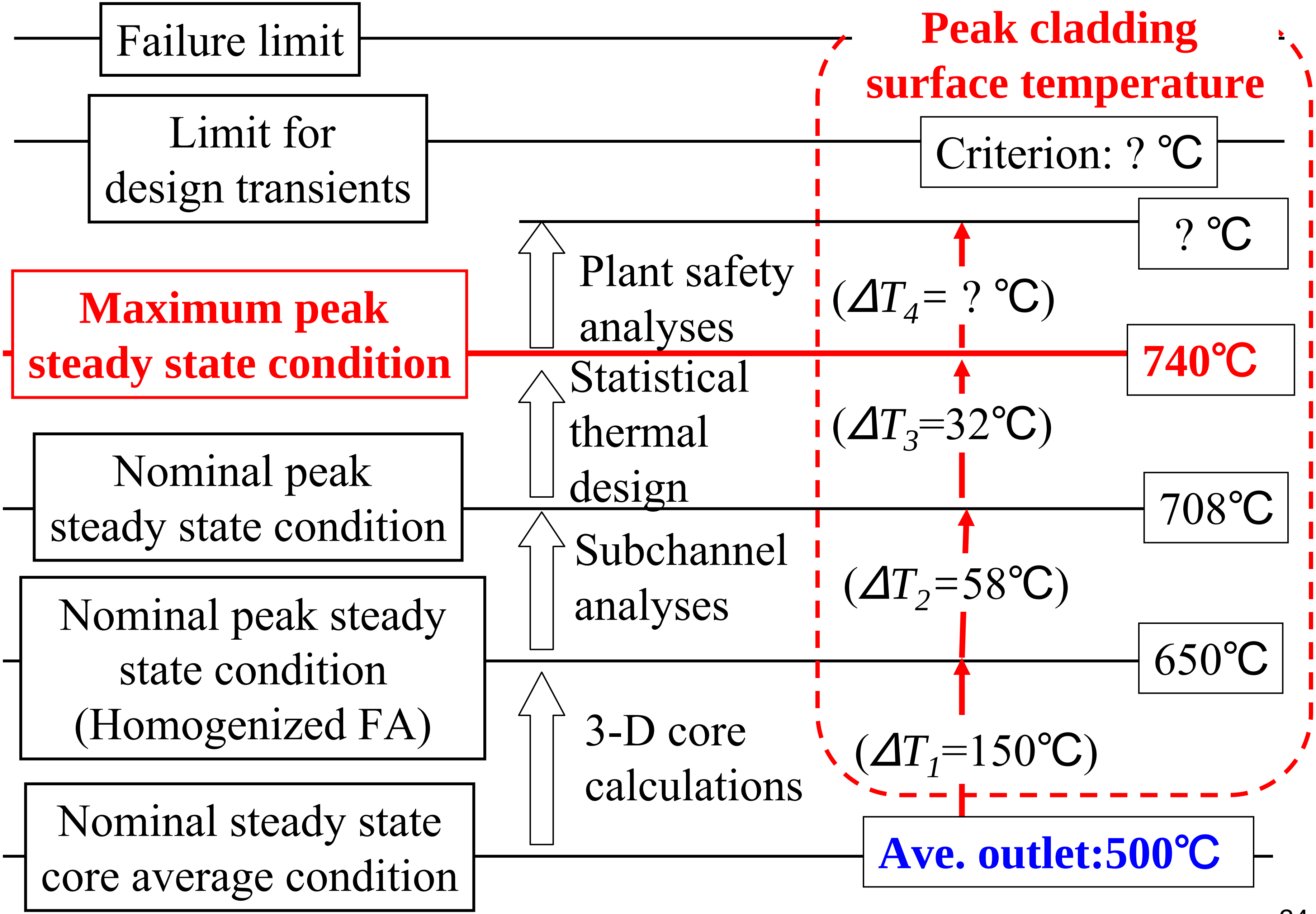


$$\sigma_T^2 = \sigma_{PF}^2 + \sigma_C^2$$

Engineering uncertainty is evaluated as: $k\sigma_T$
k=1.645 is to ensure 95/95 limit.



Peak Cladding Surface Temperature



Failure limit

Limit for design transients

Maximum peak steady state condition

Nominal peak steady state condition

Nominal peak steady state condition (Homogenized FA)

Nominal steady state core average condition

Plant safety analyses

Statistical thermal design

Subchannel analyses

3-D core calculations

Peak cladding surface temperature

Criterion: ? ℃

? ℃

(ΔT_4= ? ℃)

740℃

(ΔT_3=32℃)

708℃

(ΔT_2=58℃)

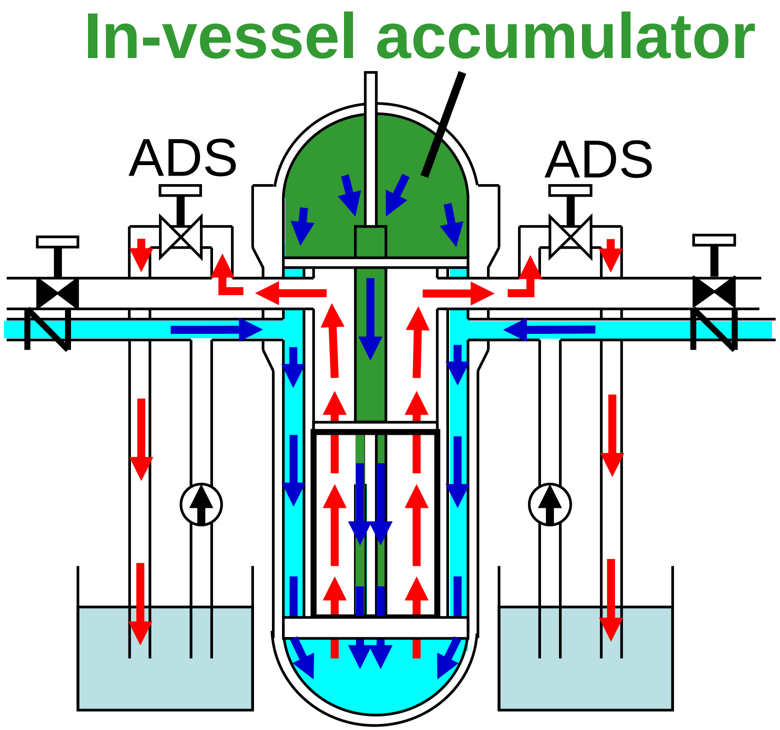
650℃

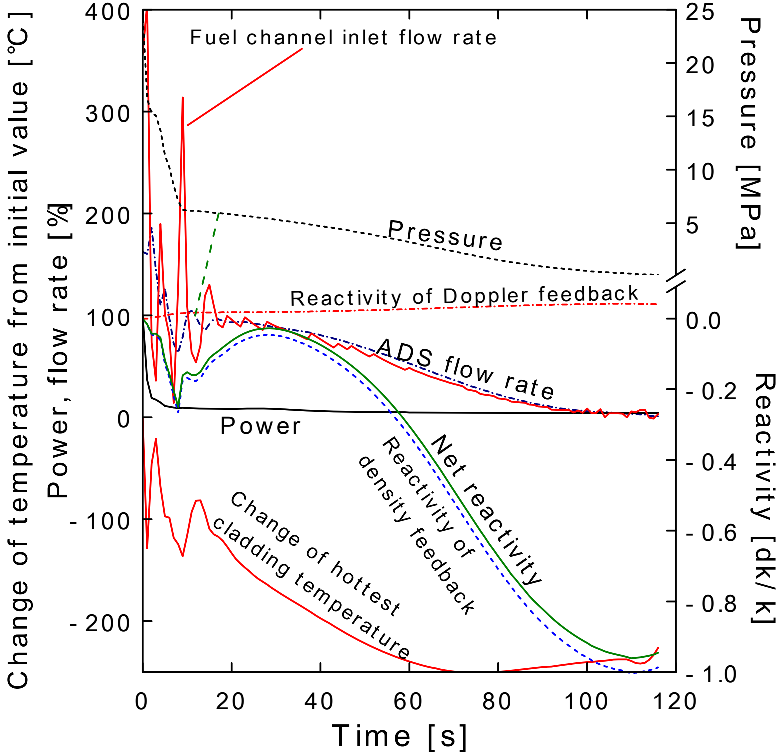
(ΔT_1=150℃)

Ave. outlet:500℃

Safety

Depressurization induces core coolant flow of the once-through cycle reactor





Once-through system ⇒ Coolant flow induced in the core

Large water inventory of Top dome ⇒ In-vessel accumulator

Negative void reactivity ⇒ Power decreasing

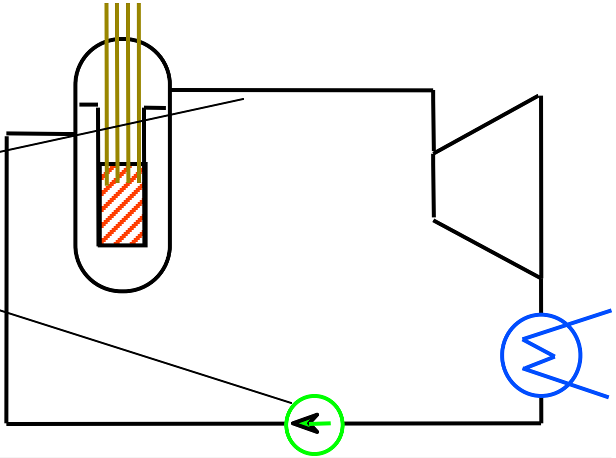
Safety principle of Super LWR

- Keeping coolant inventory is not suitable due to no water level and large density change.

Safety principle is keeping **core coolant flow rate**.

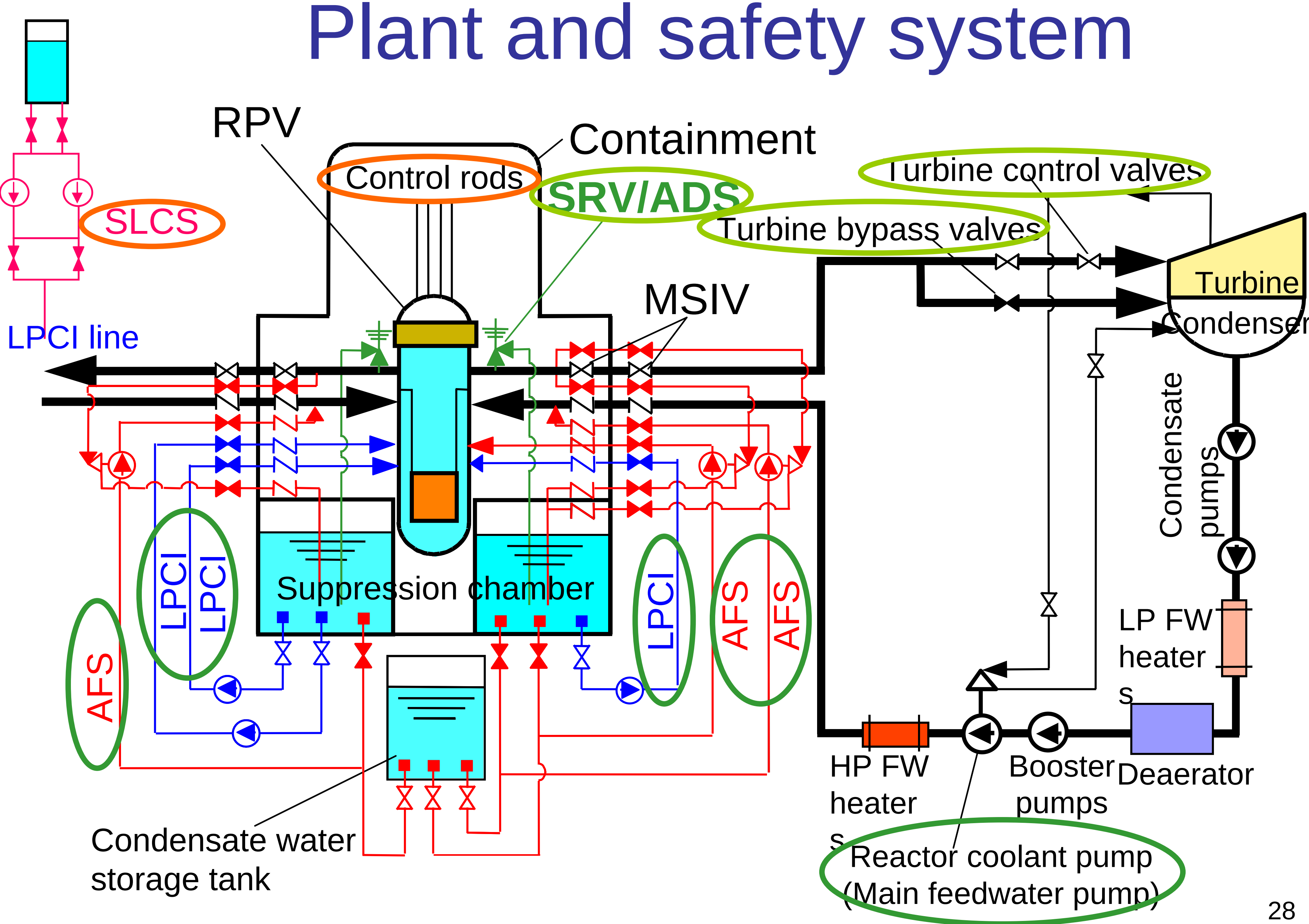
Coolant supply (main coolant flow rate)

Coolant outlet (pressure)



	BWR	PWR	Super LWR
Requirement	RPV inventory	PCS inventory	Core flow rate
Monitoring	RPV water	Pressurizer	Main coolant flow rate,

Plant and safety system



Abnormal levels and actuations

Flow rate low (⇔Coolant flow from cold-leg)

Level 1 (90%)*	**Reactor scram**
Level 2 (20%)*	**AFS**
Level 3 (6%)*	**ADS/LPCI**

Pressure high (⇔Coolant outlet at hot-leg)

Level 1 (26.0 MPa)	**Reactor scram**
Level 2 (26.2 MPa)	**SRV**

Pressure low (⇔Valve opening, LOCA)

Level 1 (24.0 MPa)	**Reactor scram**
Level 2 (23.5 MPa)	**ADS/LPCI**

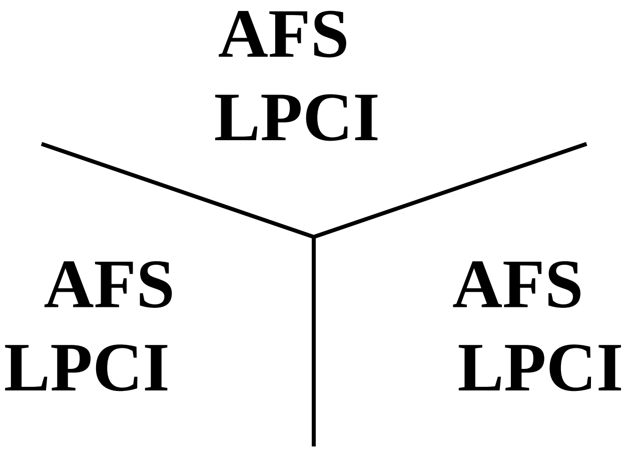
*100% corresponds rated flow rate

Safety system design

Capacity:

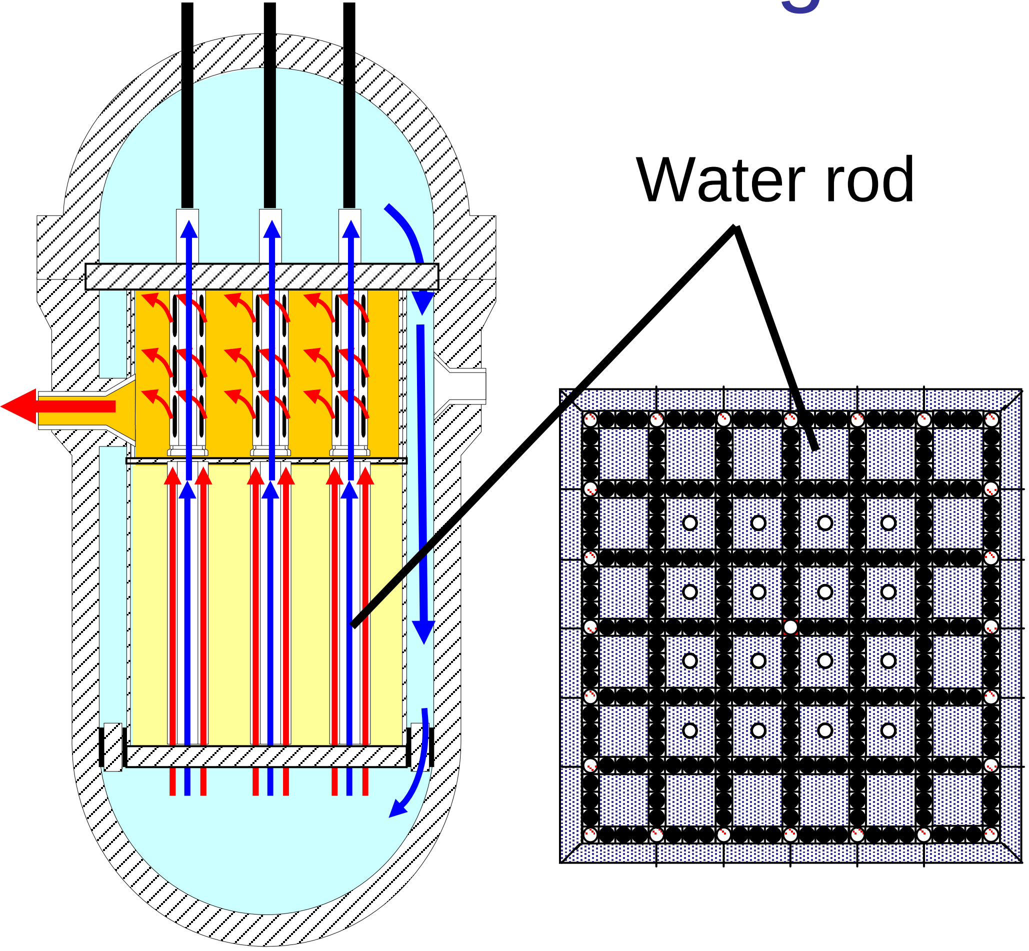
AFS	**TD 3 units: 50kg/s/unit (4%)* at 25MPa**
LPCI/RHR	**MD 3 units: 300kg/s/unit (25%)* at 1MPa**
SRV/ADS	**8 units: 240kg/s/unit (20%)* at 25MPa**

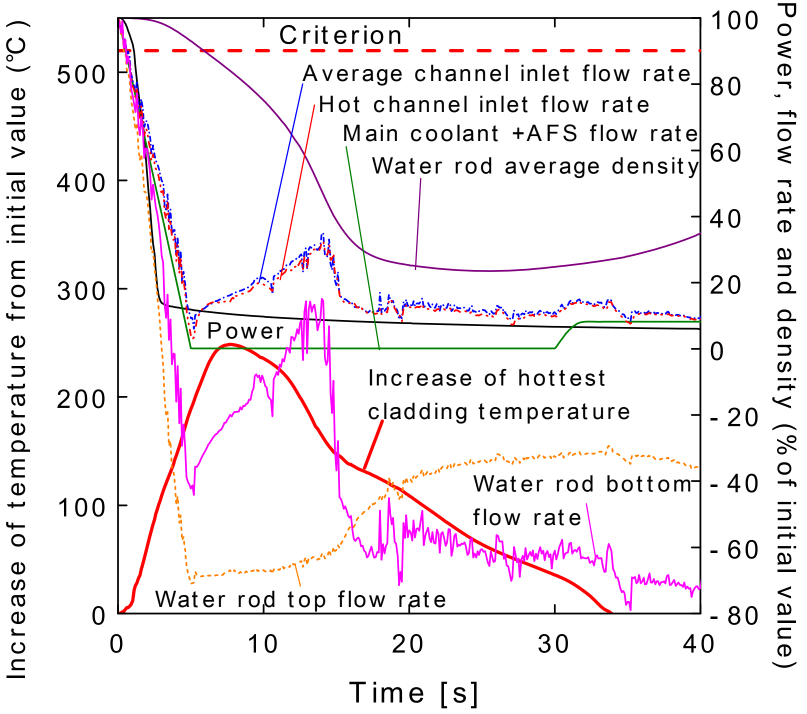
Configuration:



*100% corresponds to rated flow rate

Water rods mitigate loss-of-flow events.





Total loss of reactor coolant flow

⊿MCST≃250℃

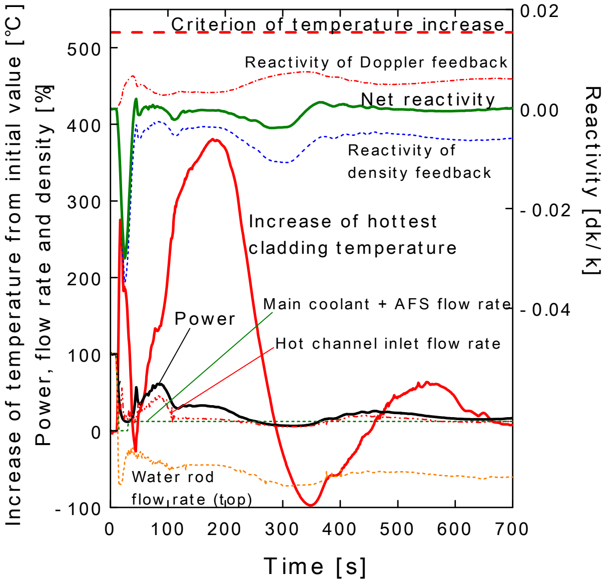
Under loss-of-flow condition:

Heat conduction to water rods increases. → “Heat sink” effect

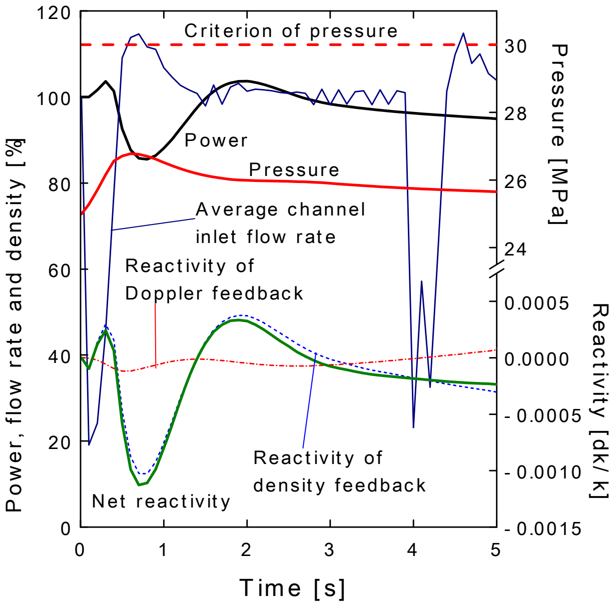
Water rods supply their inventory to fuel channels due to thermal expansion. → “Water source” effect

Alternative action is not necessary under ATWS conditions (Super LWR)

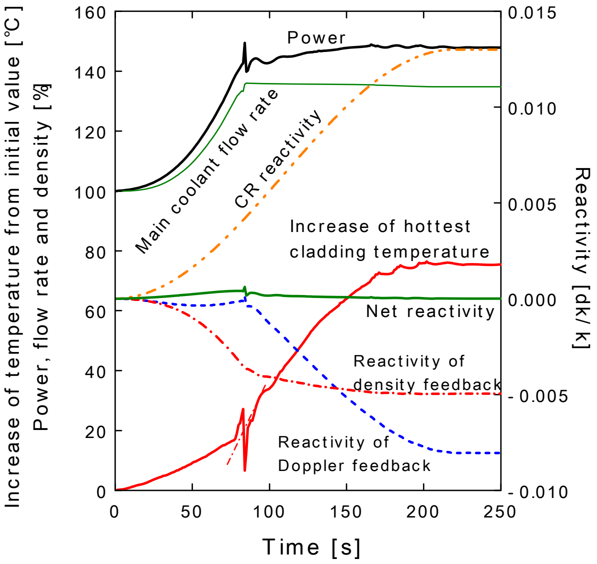
Analysis results for ATWS events without an alternative action



Loss of offsite power



Loss of turbine load without bypass



Uncontrolled CR withdrawal at normal operation

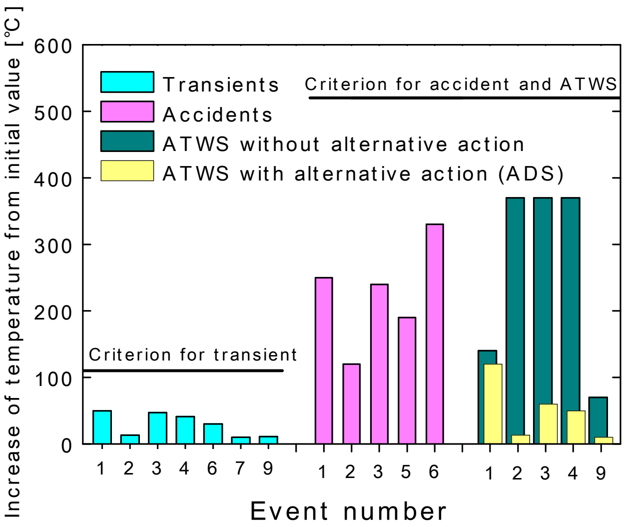
Good inherent safety characteristics of Super LWR

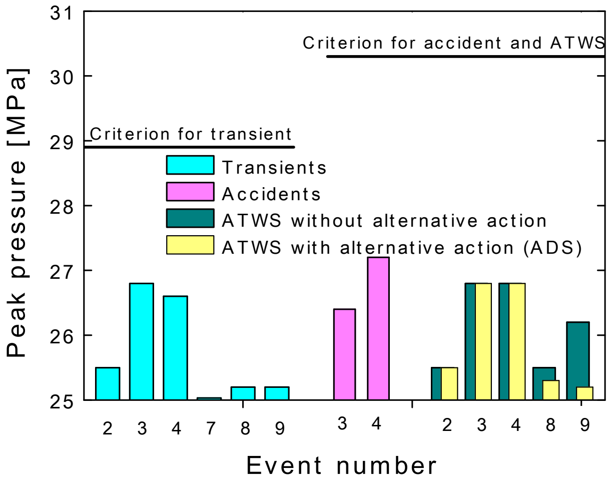
Why ATWS is mild?

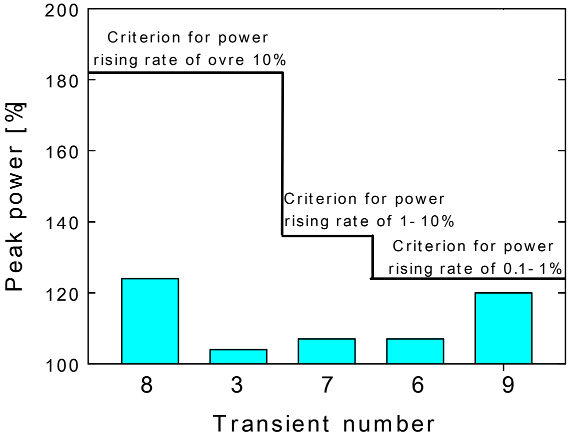
1. Small power increase by valve closure.
    - flow stagnation mitigates density increase
    - no void collapse
2. Power decreases with core flow rate due to density feedback.

Good ATWS behavior without alternative action inserting negative reactivity

Summary of safety analysis results

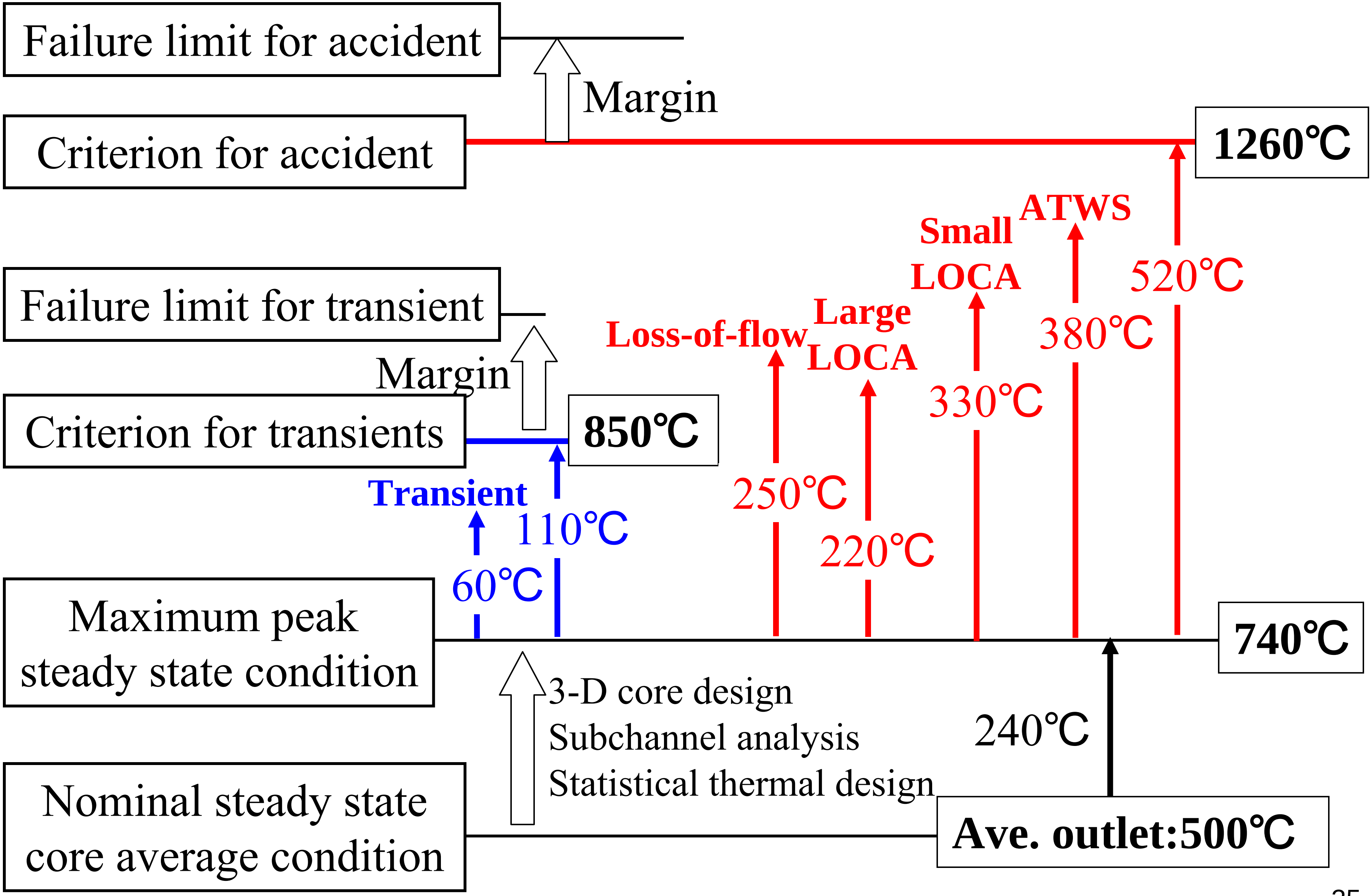






Transients	**Accidents**
1. Partial loss of reactor coolant flow 2. Loss of offsite power 3. Loss of turbine load 4. Isolation of main steam line 5. Pressure control system failure 6. Loss of feedwater heating 7. Inadvertent startup of AFS 8. Reactor coolant flow control system failure 9. Uncontrolled CR withdrawal at normal operation 10. Uncontrolled CR withdrawal at startup	1. Total loss of reactor coolant flow 2. Reactor coolant pump seizure 3. CR ejection at full power 4. CR ejection at hot standby 5. Large LOCA 6. Small LOCA

ΔMSCT for abnormal events



Failure limit for accident

Margin

Criterion for accident

1260℃

Failure limit for transient

Margin

Criterion for transients

850℃

Maximum peak steady state condition

740℃

Nominal steady state core average condition

Ave. outlet:500℃

3-D core design
Subchannel analysis
Statistical thermal design

Summary of safety characteristics of Super LWR

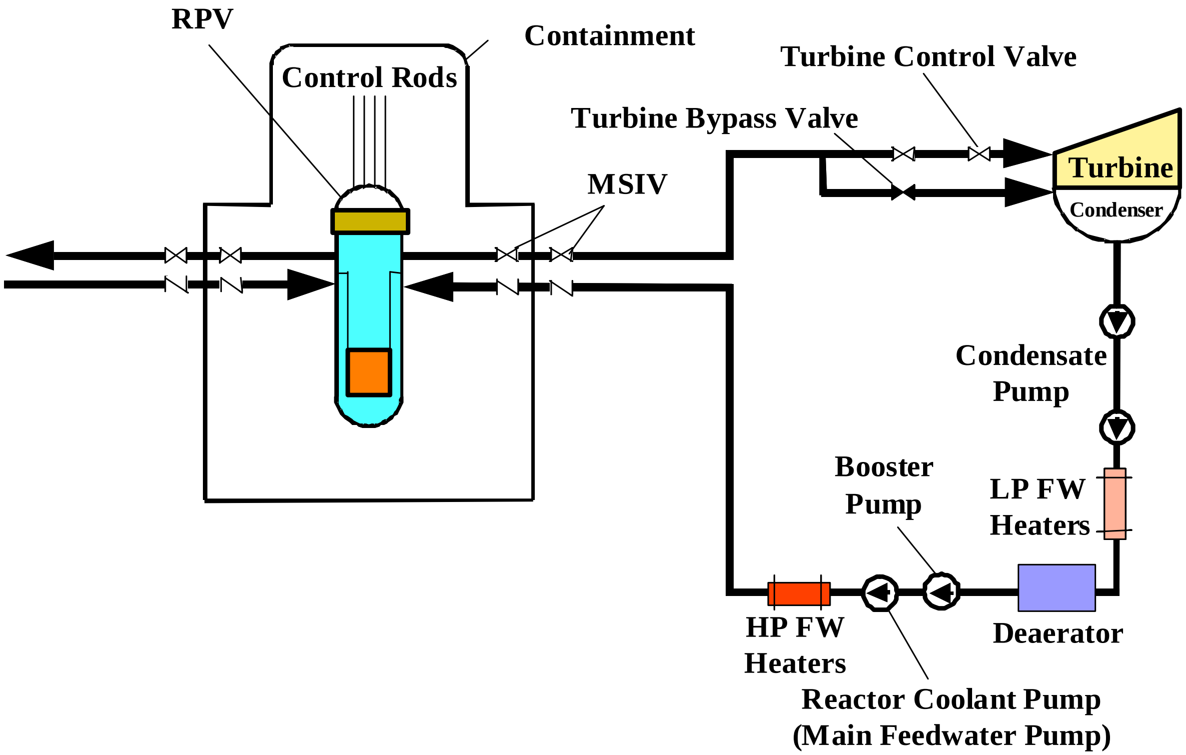
- Core cooling by depressurization
- Top dome and water rods serve as an “in-vessel accumulator”
- Loss of flow mitigated by water rods
- Short period of high cladding temperature at transients
- Mild behavior at transients, accidents and ATWS
- Simple safety principle (keeping flow rate) due to once-through cooling cycle

Super fast reactor

Tight fuel lattice

Supercritical-pressure light water cooled fast reactor

Same plant system as Super LWR



Plant system of Super LWR and Super FR

Advantages of Super Fast Reactor

Low reactor coolant flow rate due to high enthalpy rise
High head pumps of the once-through direct cycle plant

- ➢ Compatible with tight fuel lattice core of Super FR, a light water cooled fast reactor
- ➢ No pumping power increase and instability problems of high conversion LWR

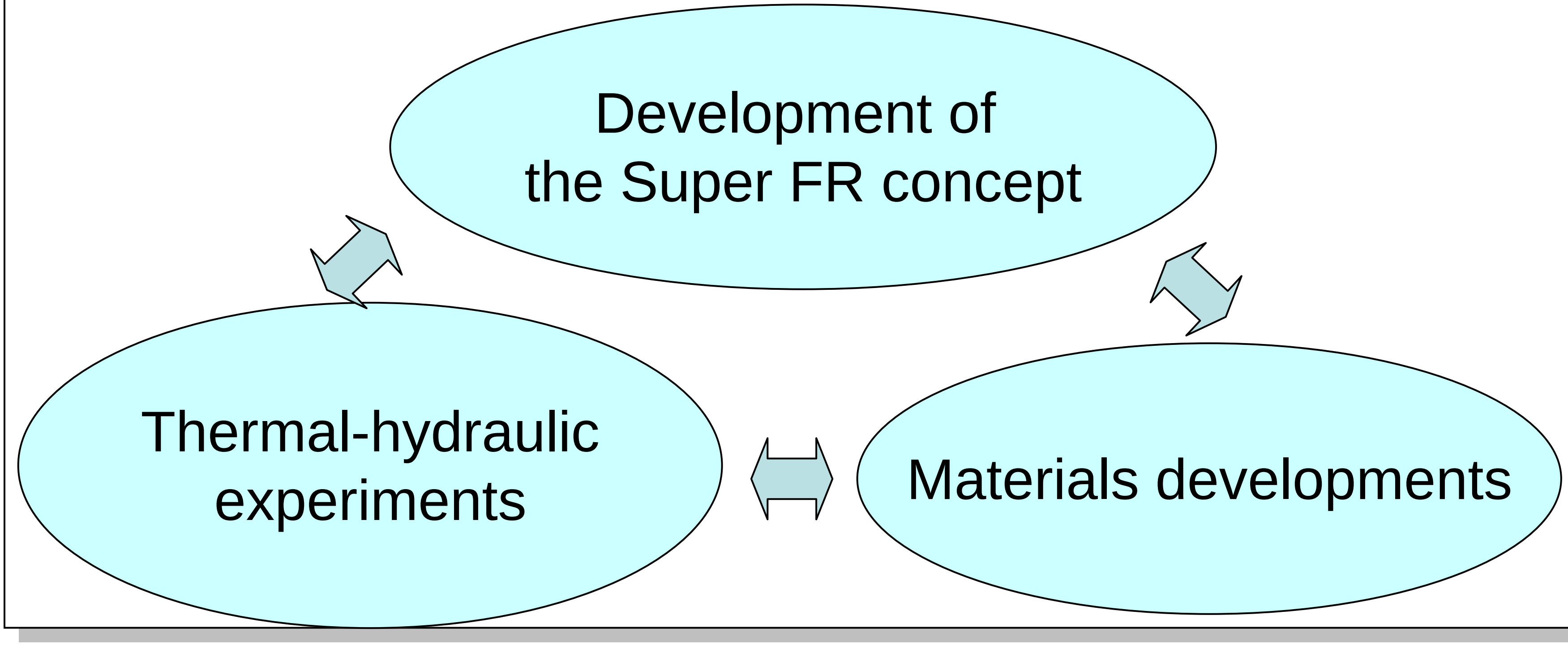
Same plant system as Super LWR, the thermal reactor
Fast reactors have higher power densities than thermal reactors due to no moderator necessary.

- ➢ Making capital cost of Super FR lower than LWRs
  (Capital cost; Super FR< Super LWR< LWRs)

R&D of Super Fast Reactor

University of Tokyo, JAEA, Kyusyu Univ. and TEPCO
entrusted by MEXT as one of the Japanese NERI,
5 years, Dec. 2005-March 2010

Leader: Y. Oka (University of Tokyo)



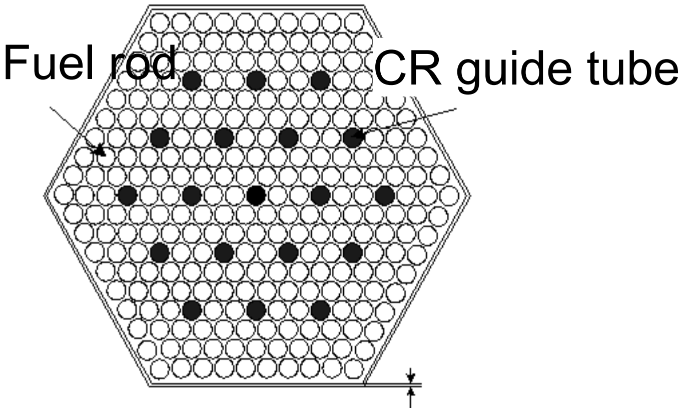
Development of the Super FR concept

Thermal-hydraulic experiments

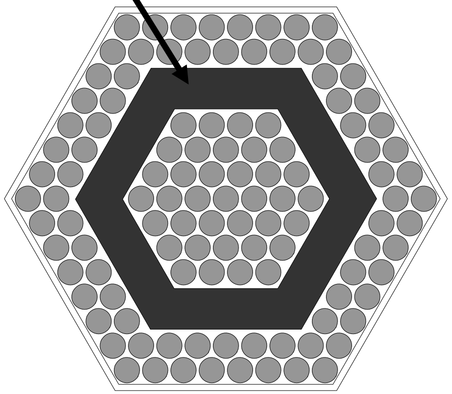
Materials developments

Fuel and Core (example)

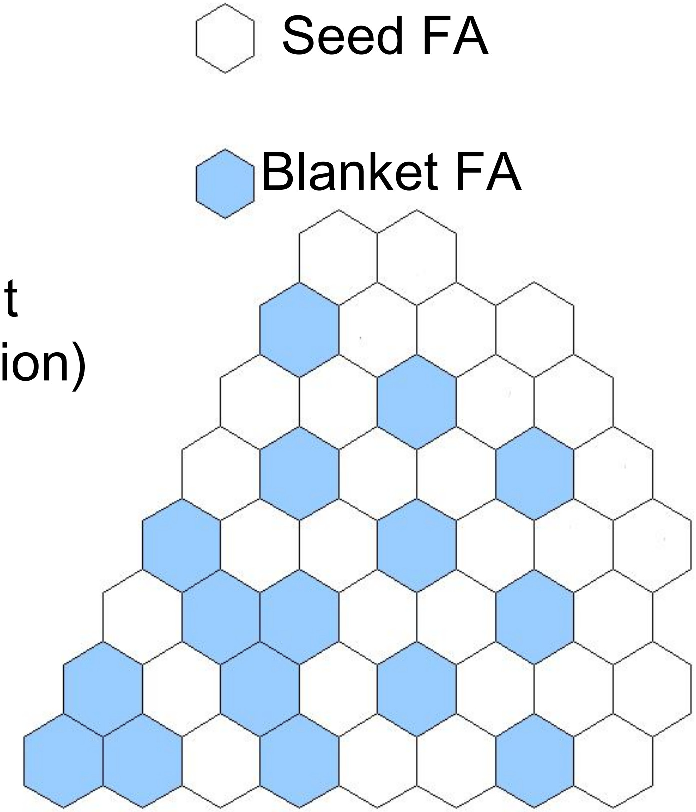
- MOX fuel with SS cladding (Fuel rod analysis)
- Core design: 3-D N-TH coupled core burn-up calculation, subchannel analysis



Seed FA



Blanket FA

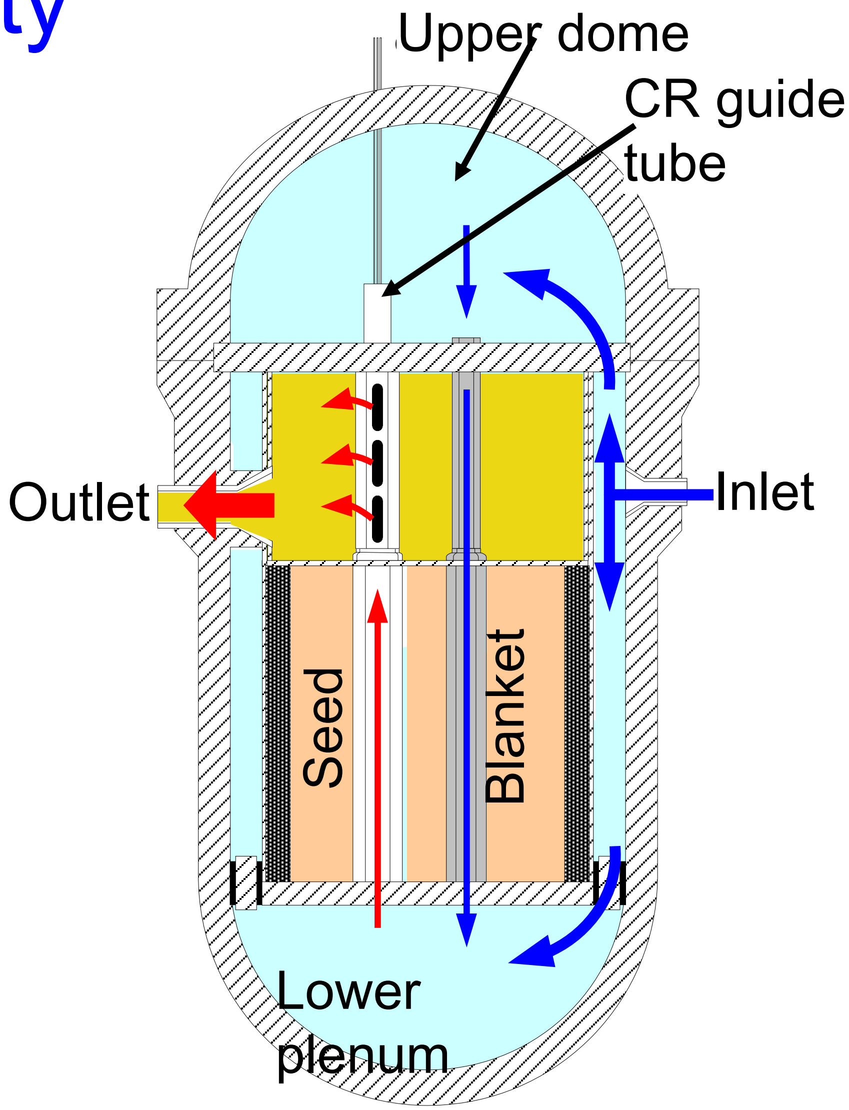


1/6 Core (example)

Core Structure and Plant Control and Safety

Core characteristics (700MWe)

	Core1	Core 2
Fuel		
Fuel (Seed/Blanket)	MOX/dep.UO_2	
Fuel pellet density	95%TD	
Rod OD[mm]	7.0	5.5
Pitch/ OD	1.16	1.19
Cladding Material	SUS304	
Thickness [mm]	0.43	0.4
Effective heating length [cm]	300	200
Core		
No. of seed fuel assemblies	126	162
No. of blanket fuel assemblies	73	
Pitch of FA	14.2	11.6



RPV and the coolant flow

Thermal hydraulic experiments

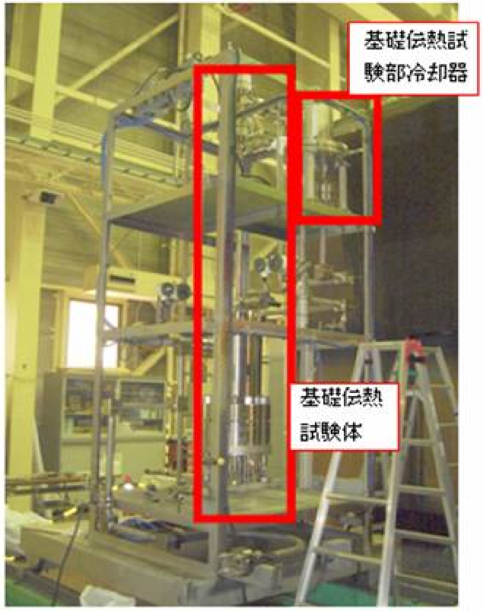
Kyusyu University ;HCFC22 (Freon)

(1) single tube and 7-rod bundle

(2) critical heat flux near critical pressure

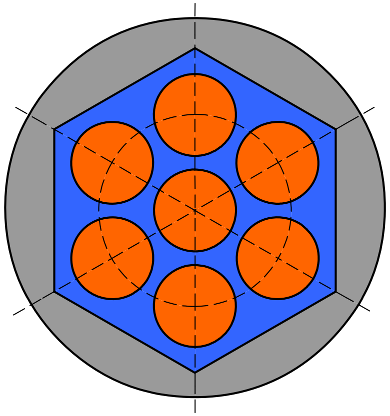
(3) critical flow and condensation

JAEA Naka-lab; Supercritical Water



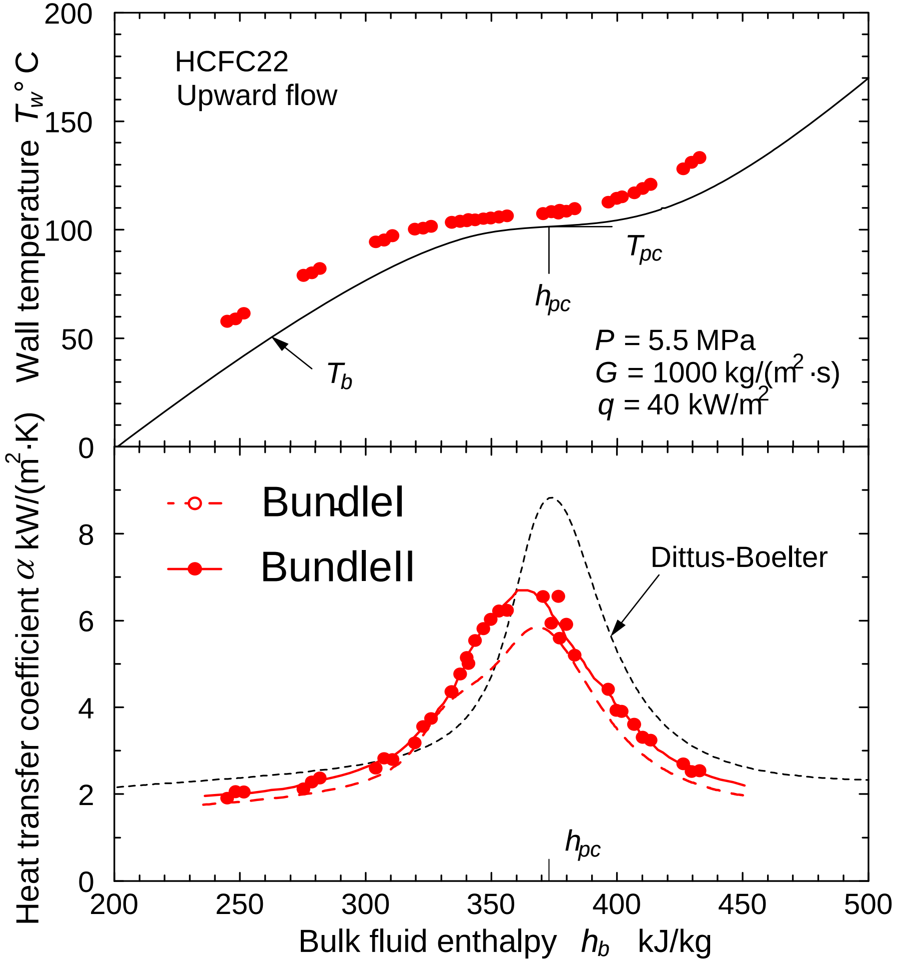
Heater rods and spacers

Single rod 7-rod bundle

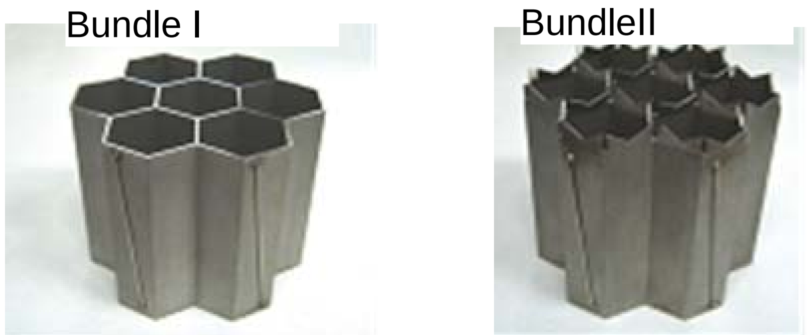


Experimental results; HCFC22(Freon)

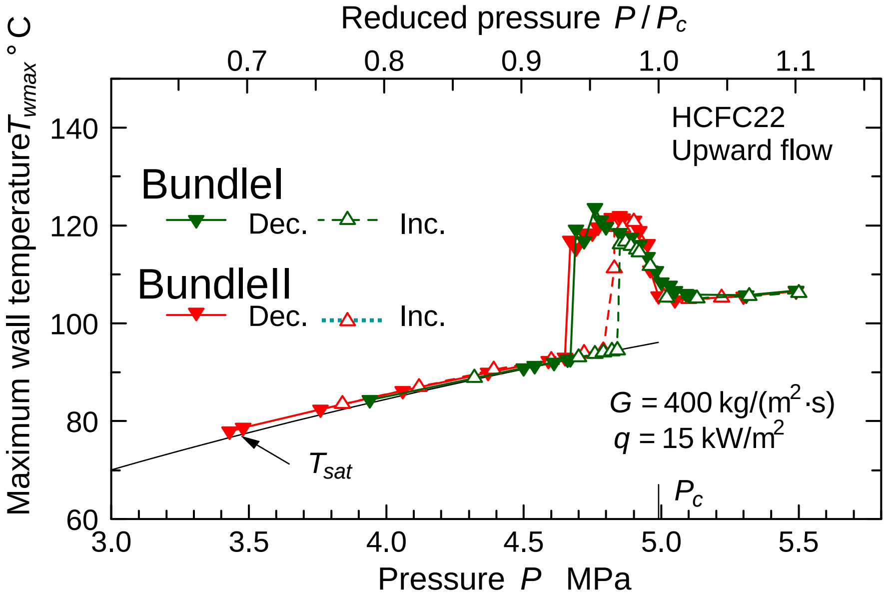
Grid spacer effect on heat transfer coefficients and critical heat flux



Wall temperature and heat transfer coefficient of 7-rod bundle test



Grid spacers



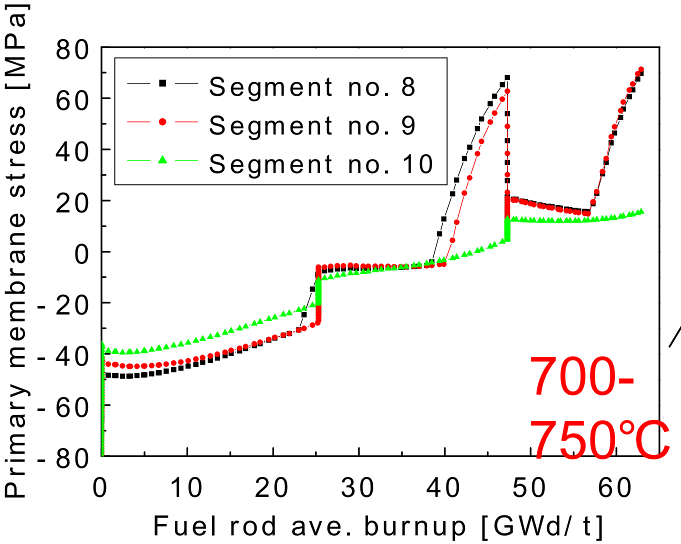
Maximum wall temperature at critical heat flux

Need for Developing High Creep Strength Clad

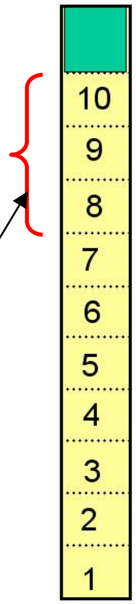
- Max. stress on clad at peak T (700-750℃): 70-100MPa
  - Exceed creep strength of SS for LWR (SUS316L)
  - Advanced SS for LMFBR (PNC1520) almost satisfies the requirement but SCC susceptibility, corrosion and neutron absorption properties need to be improved

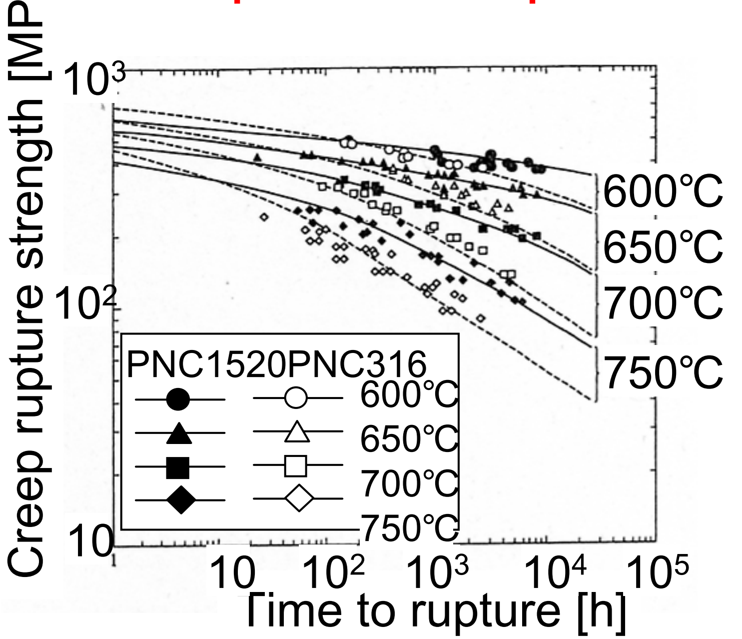
⇩

- High creep strength clad needs to be developed for Super FR



Fuel rod analysis results (Super LWR)

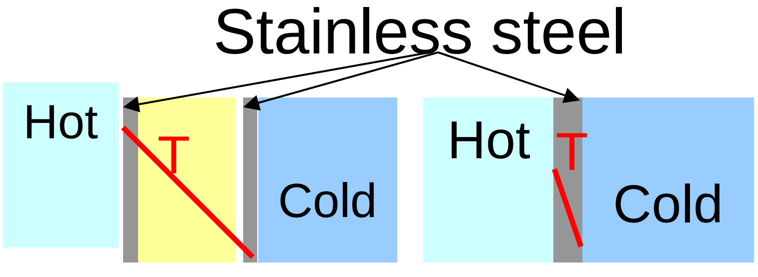




Creep rupture strength of advanced SS

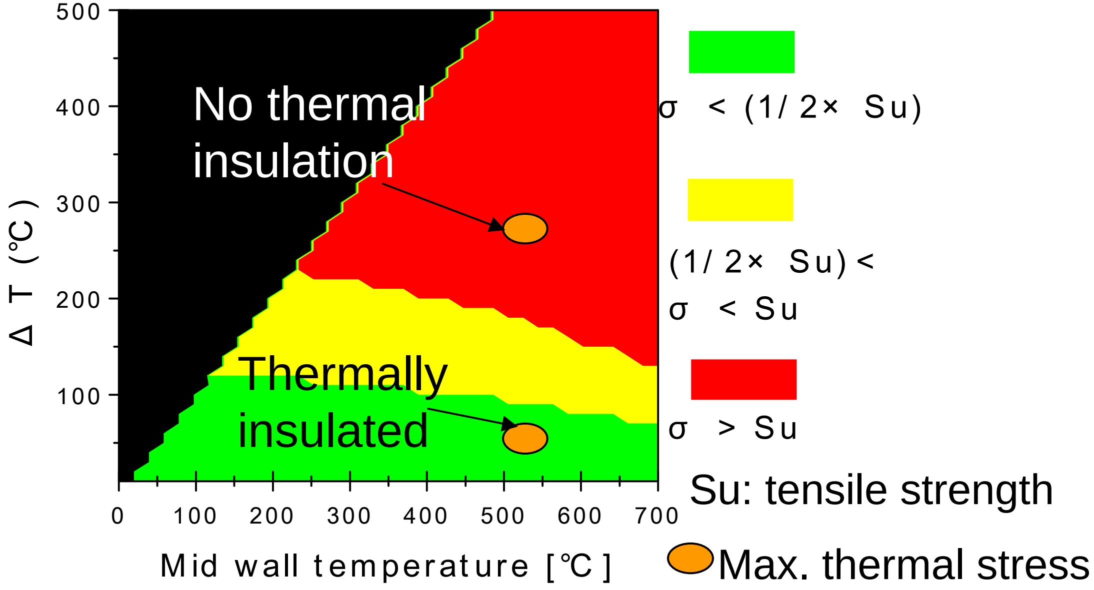
Developed Good Thermal Insulator
Yttria stabilzed zirconia (YSZ)

- Large ΔT (～250℃)
- Thermal insulator is required for:
  - reduction of thermal stress
  - maintaining coolant temperature

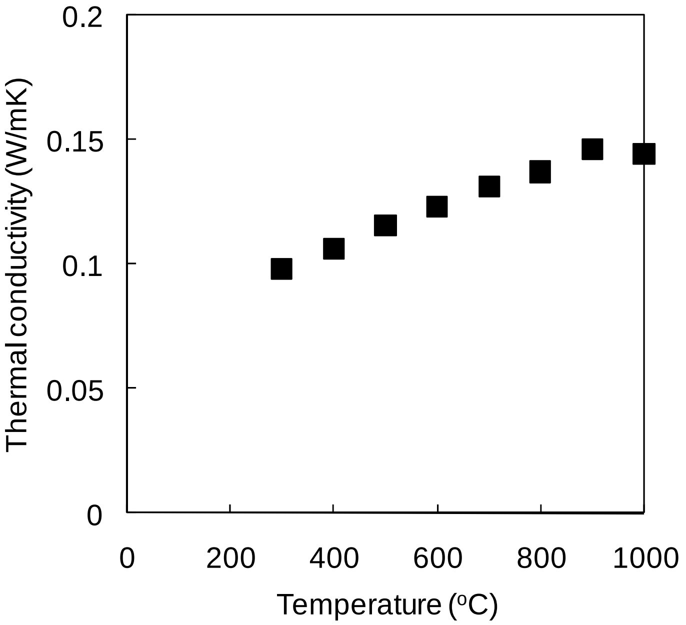


Insulated

No insulation



Thermal stress on the wall



Thermal conductivity of YSZ

~1/20 of Zirconia

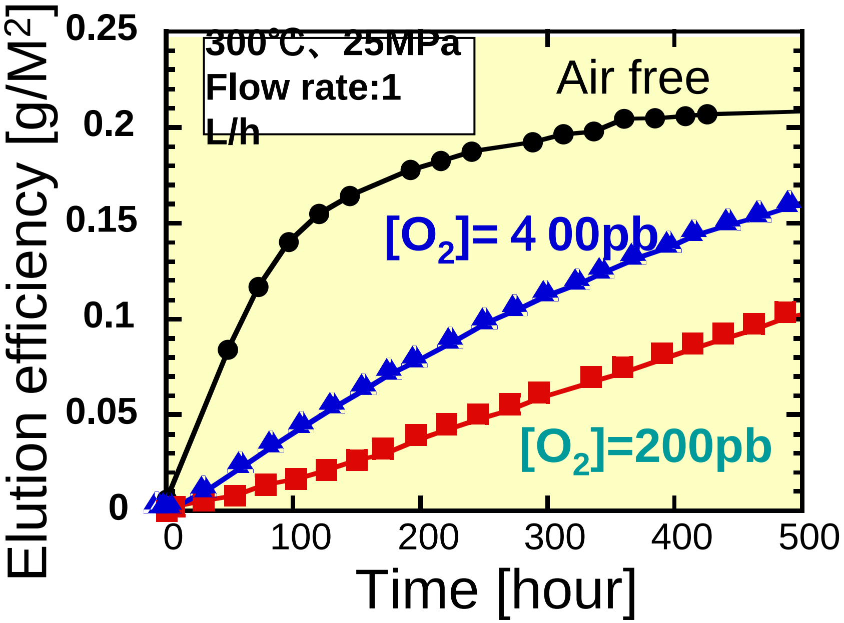
Elution of structural material in SC water

Experimental devices

Elution decreases with temperature (at 25 MPa)

	Absolute value (g / m^2)		Relative value (Normalized at 300 °C)	
	Deaerated	200 ppb O_2	Deaerated	200 ppb O_2
300 °C	0.203	0.102	1.0	1.0
400 °C	0.0098	0.0085	0.048	0.083
450 °C	0.0045	0.0045	0.022	0.045
550 °C	< 0.002	0.0062	< 0.01	0.060

Elution depends on O_2



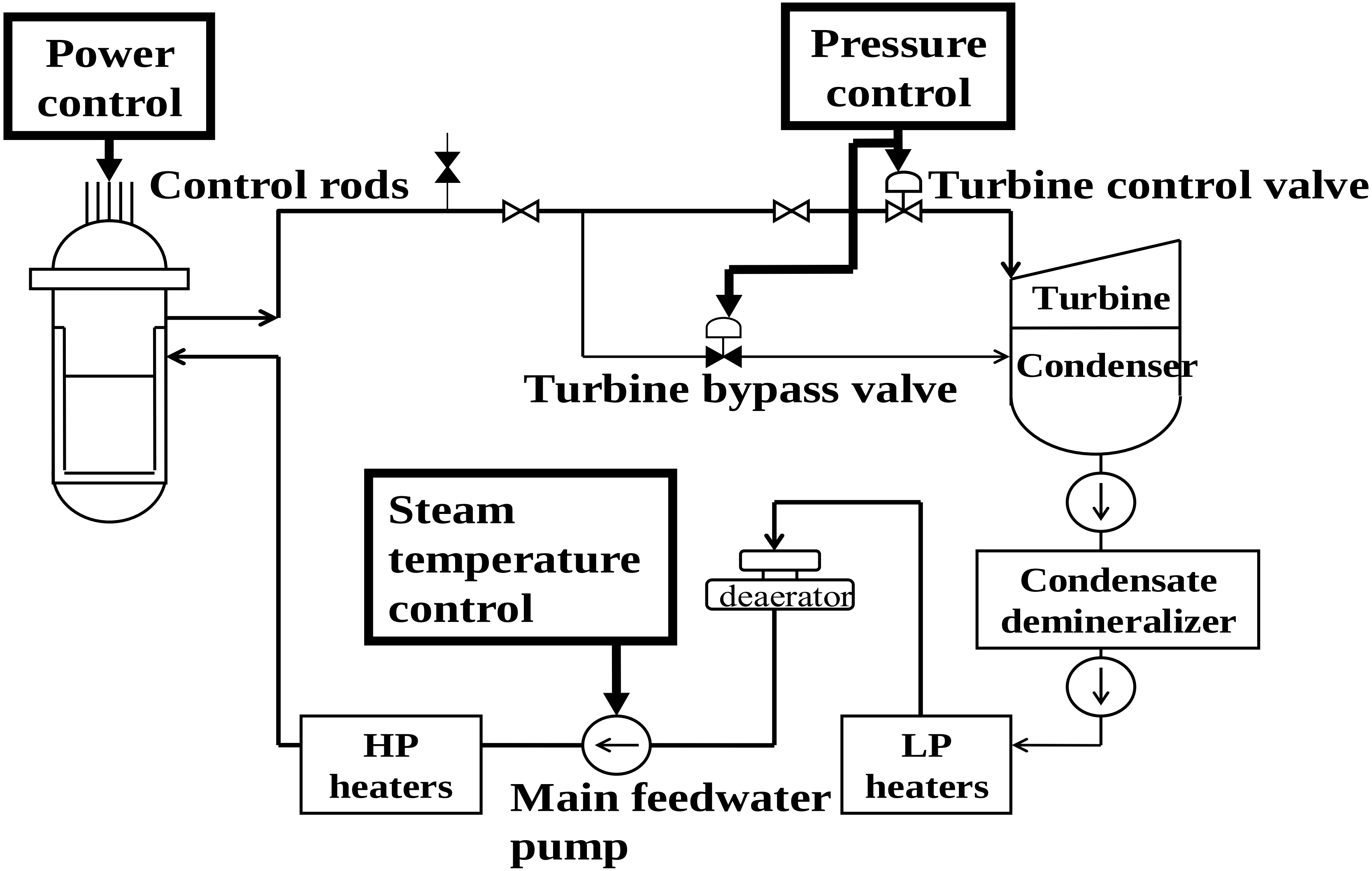
SCWR R&D in the world

- Japan: University of Tokyo; Super LWR concept (since 1989), Super FR R&D (2005-2010). Toshiba; SCPR R&D, Consortium for GIF R&D
- China; Shanghai JTU (8 organizations) SCWR R&D (2007-2012), CGNPC announced the plan of constructing an experimental SCWR from 2016.
- EU; HPLWR phase 1 (FZK, 2000-2), phase 2 (FZK, 10 organizations of 8 countries 2006-9), planning of phase 3
- Canada: pressure tube type SCWR R&D：NSERC/NRCan/AECL-Universities program
- Korea: thermal hydraulics (KEARI)
- Russia: SC thermal hydraulic loops of IPPE, WS at NIKIET in 2008
- USA: TH and materials at Univ. Wisconsin and Univ. Michigan (finished)
- GIF SCWR OECD/NEA (Canada, EU, Japan and other countries) phase 2
- IAEA: CRP of supercritical thermal hydraulics

SCR symposiums; 1st and 2nd at University of Tokyo in 2000 and 2003, 3rd at Shanghai JTU in 2007 and 4th in Heidelberg in 2009

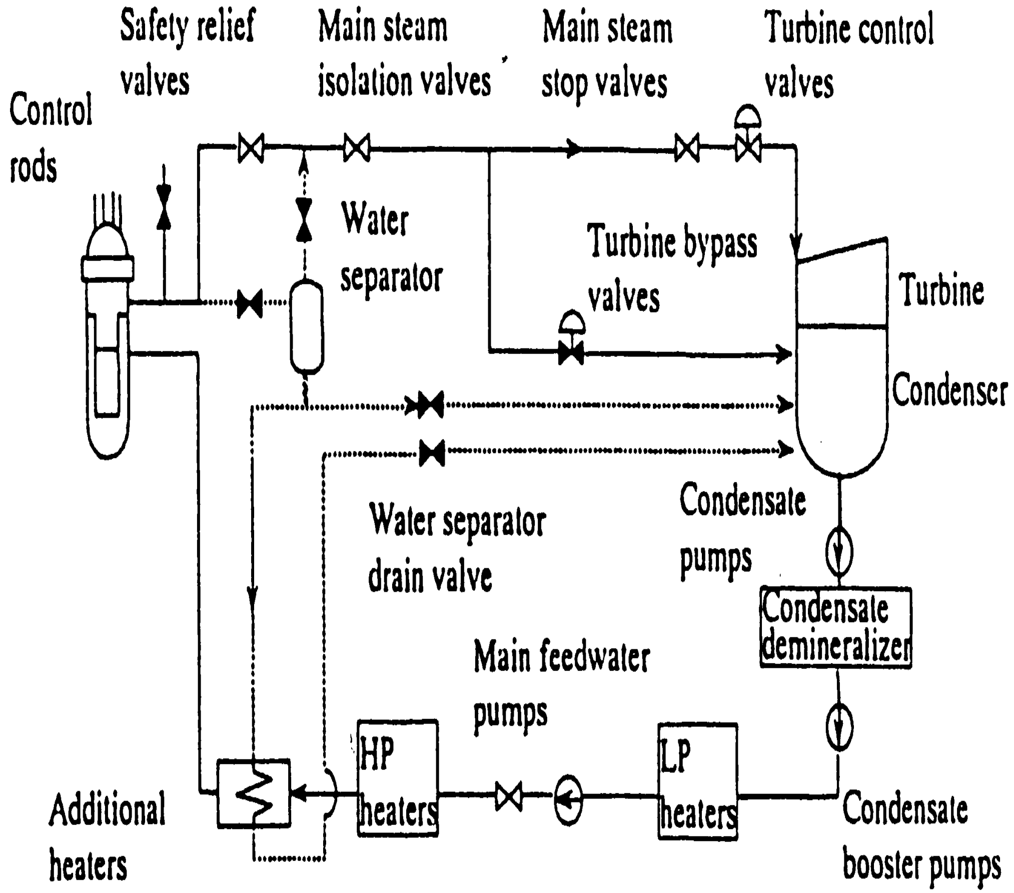
Thank you

Control and start up



Plant control system

Sliding Pressure Startup System



Sliding pressure supercritical water-cooled reactor

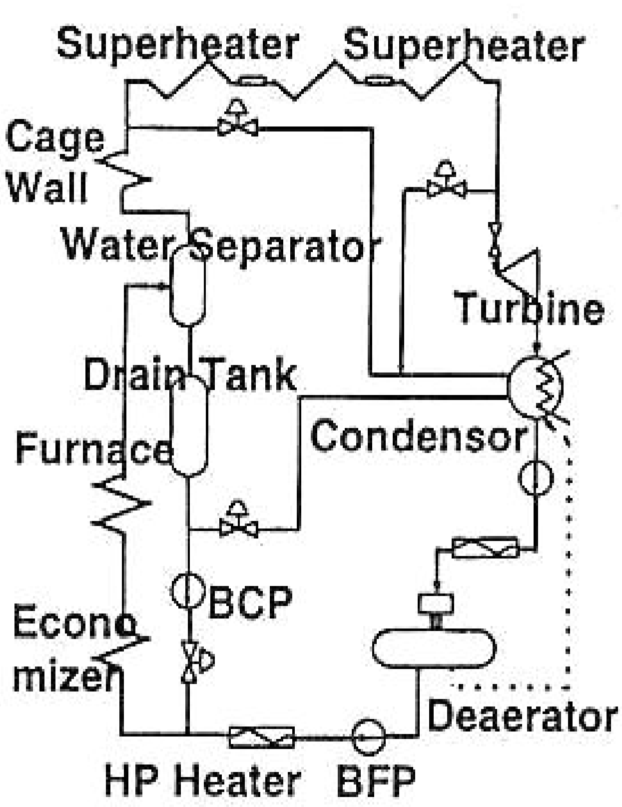
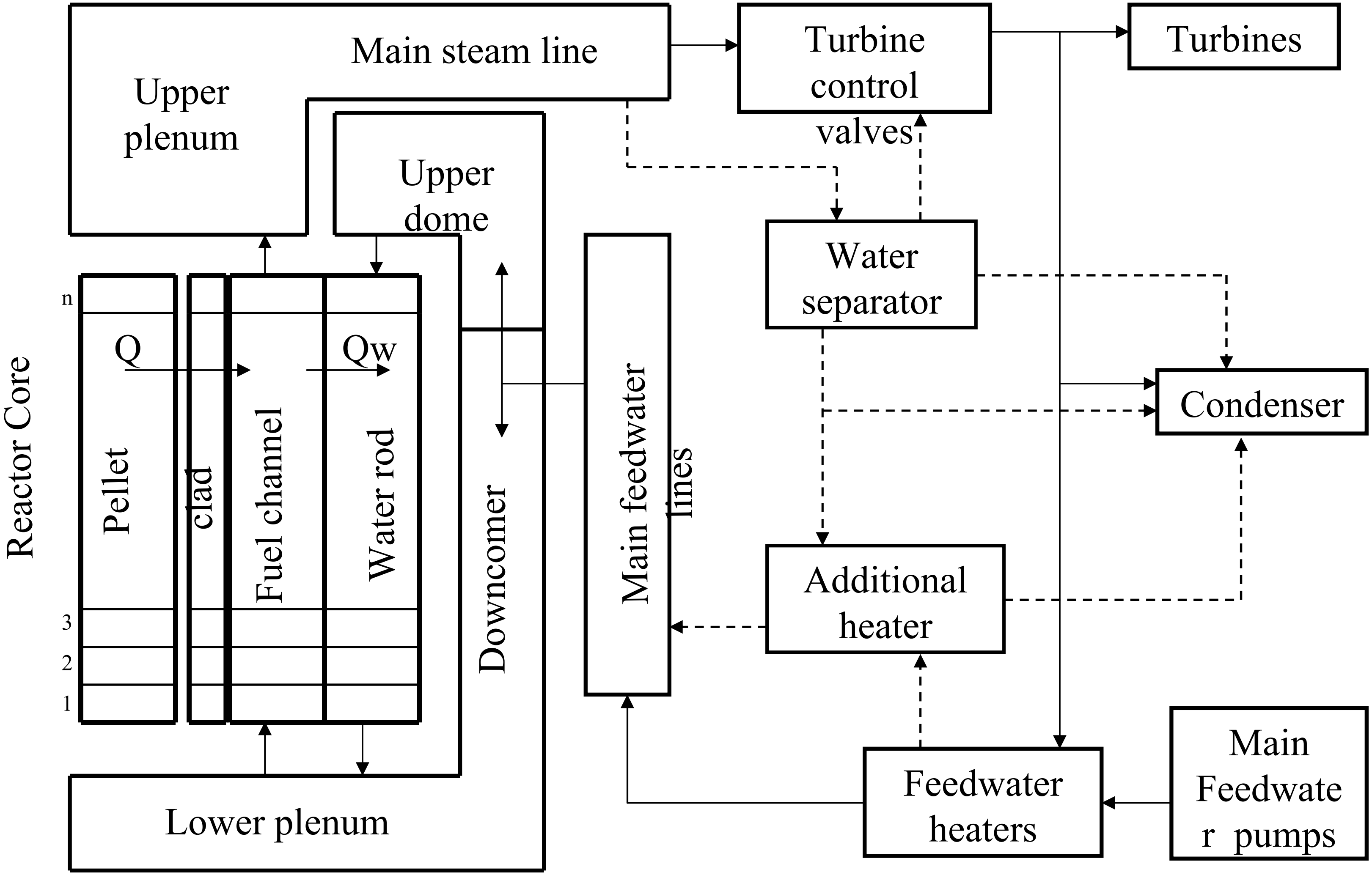


Fig.18 Sliding pressure fossil fired power plant

Nuclear heating starts at subcritical pressure.

Water separator is installed on a bypass line.



Calculation Model for Sliding Pressure Startup

Sliding Pressure Startup Procedure

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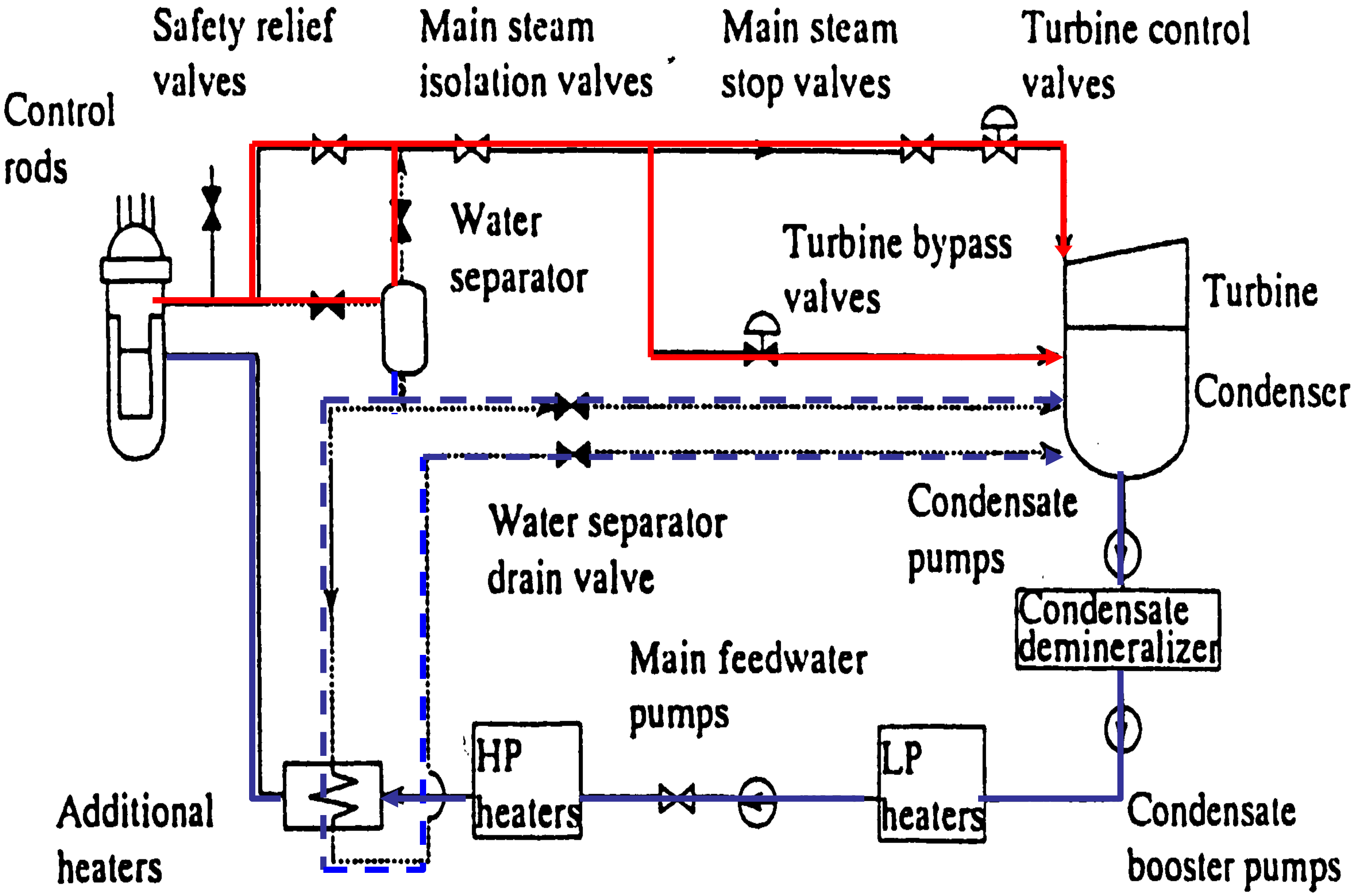
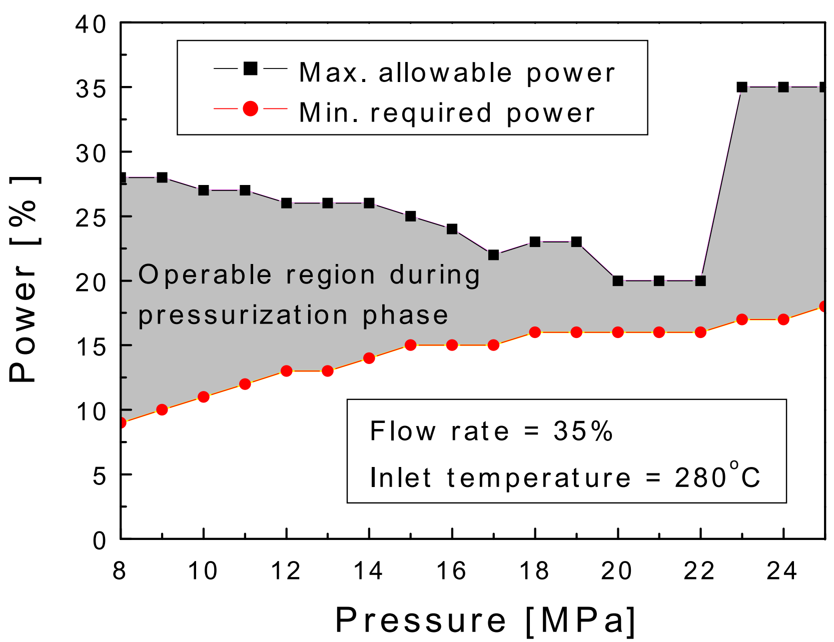
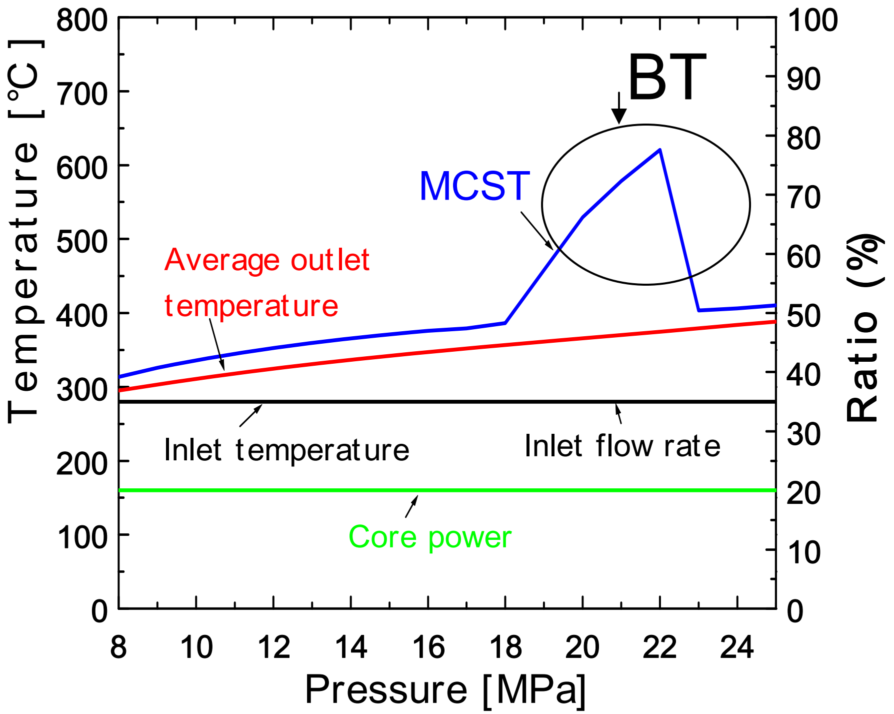


Pressurization phase

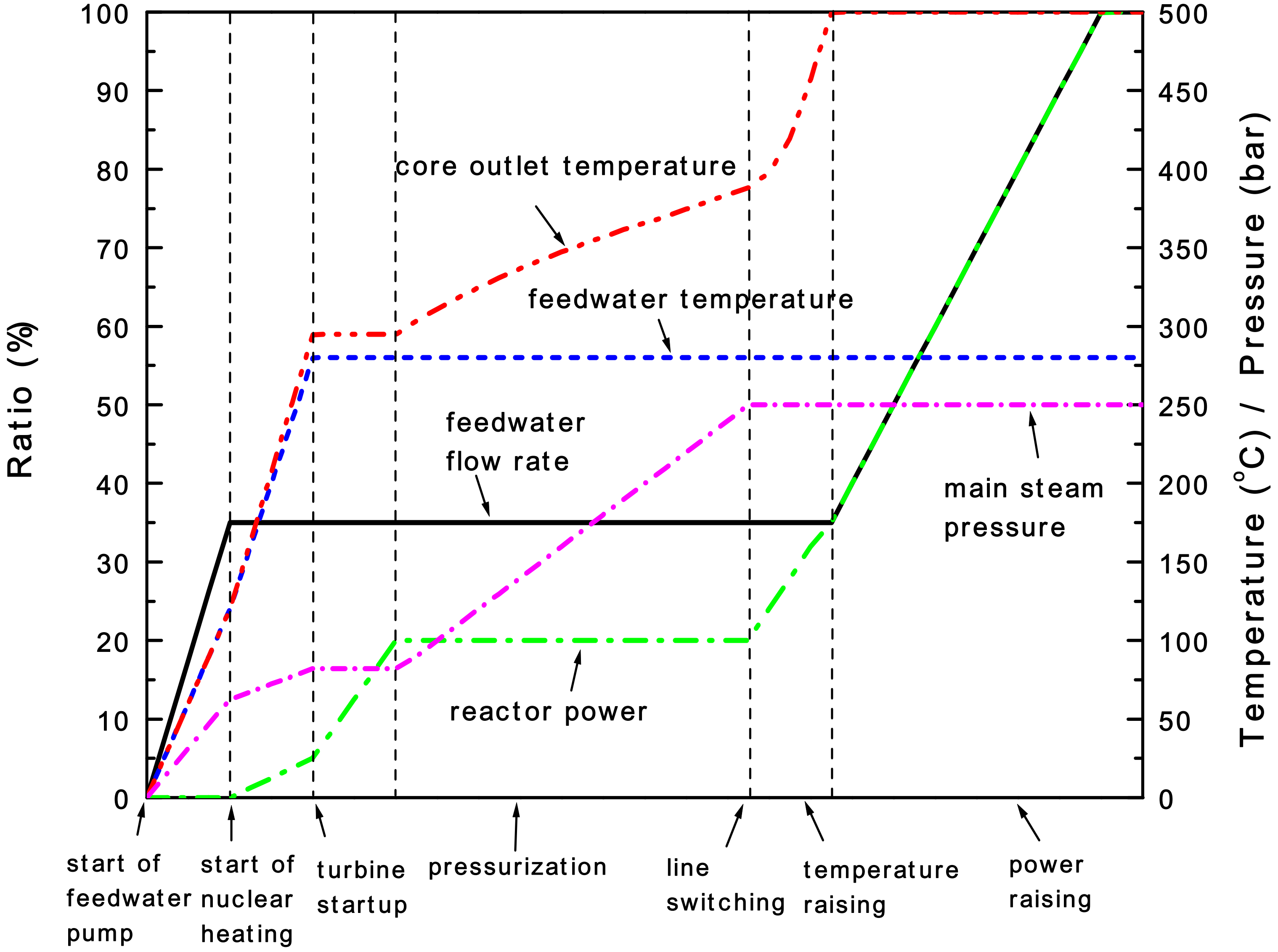




- Sliding pressure startup system (nuclear heating starts at subcritical pressure)
- Clad temperature increase in pressurization phase is due to BT
- Power / flow region is limited by CHF
- CHF may be increased by grid spacers

Sliding Pressure Startup Curve

(By Thermal-Hydraulic Analysis)

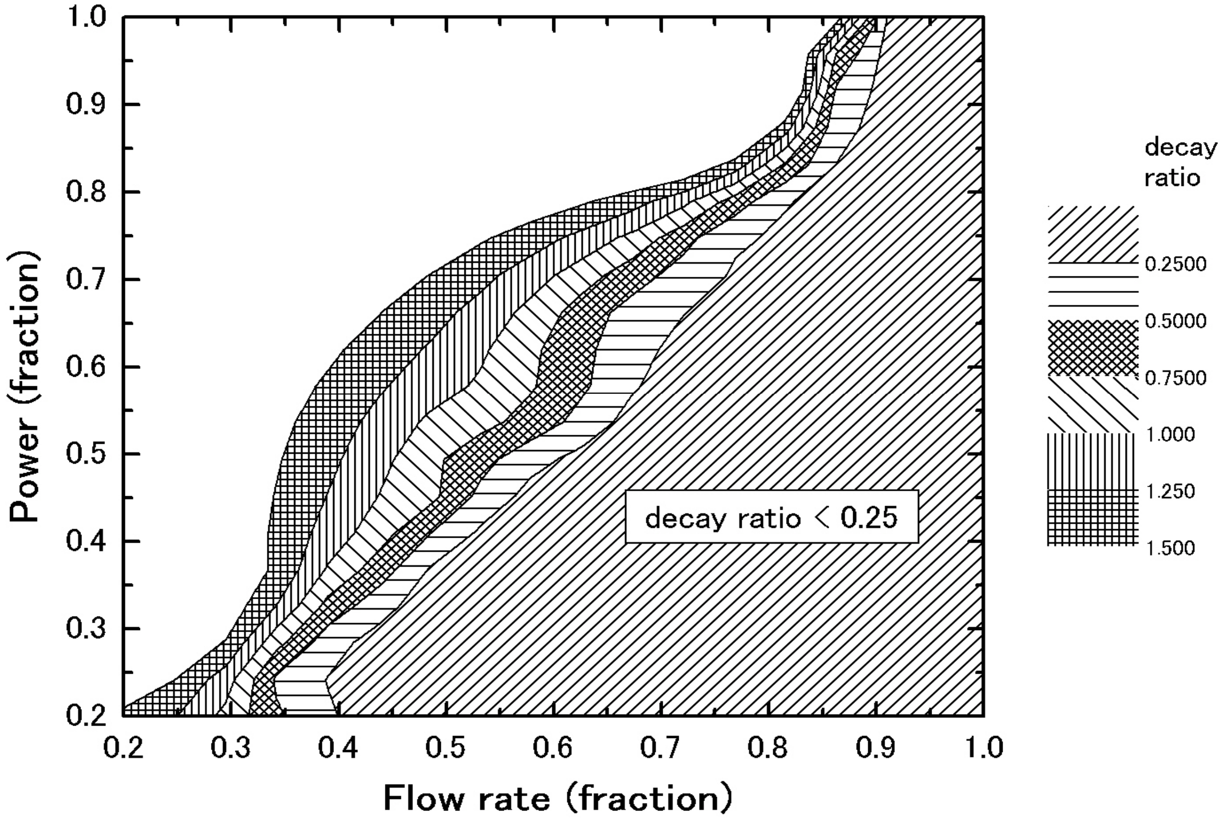


Linear Stability Analysis (for Supercritical Pressure)

Thermal-Hydraulic Stability (Supercritical pressure)

Coupled Neutronic Thermal-Hydraulic Stability (Supercritical pressure)

Decay Ratio Map for Coupled Neutronic Thermal-hydraulic Stability

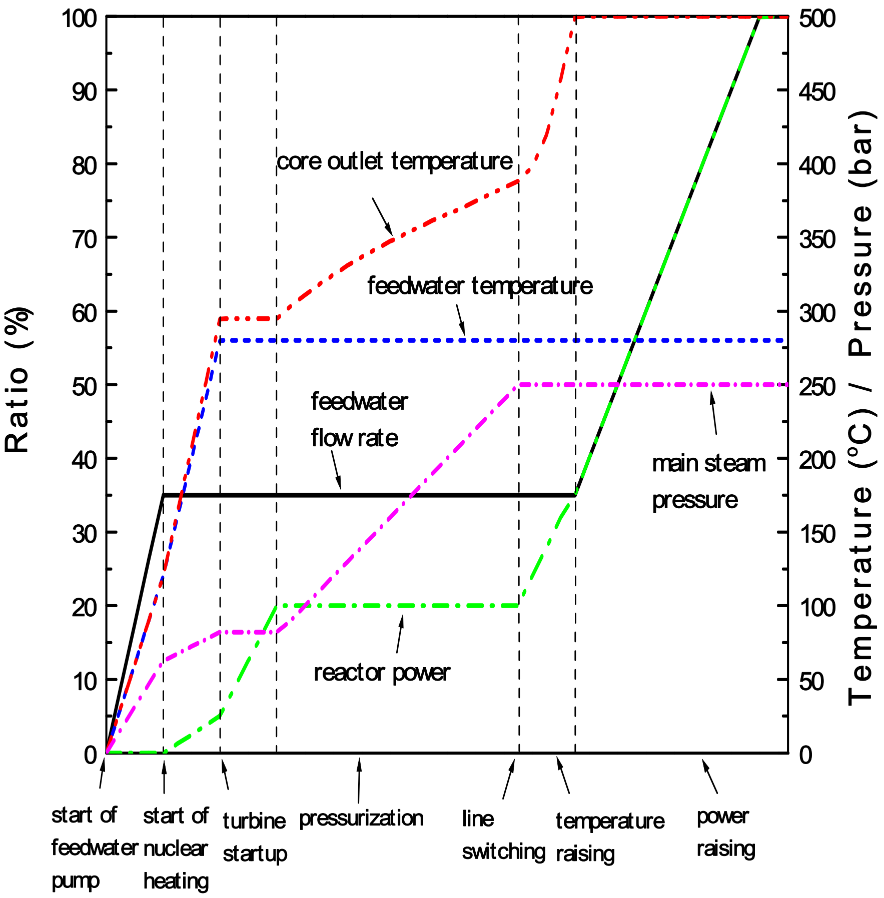


Decay ratio increases with power to flow rate ratio.

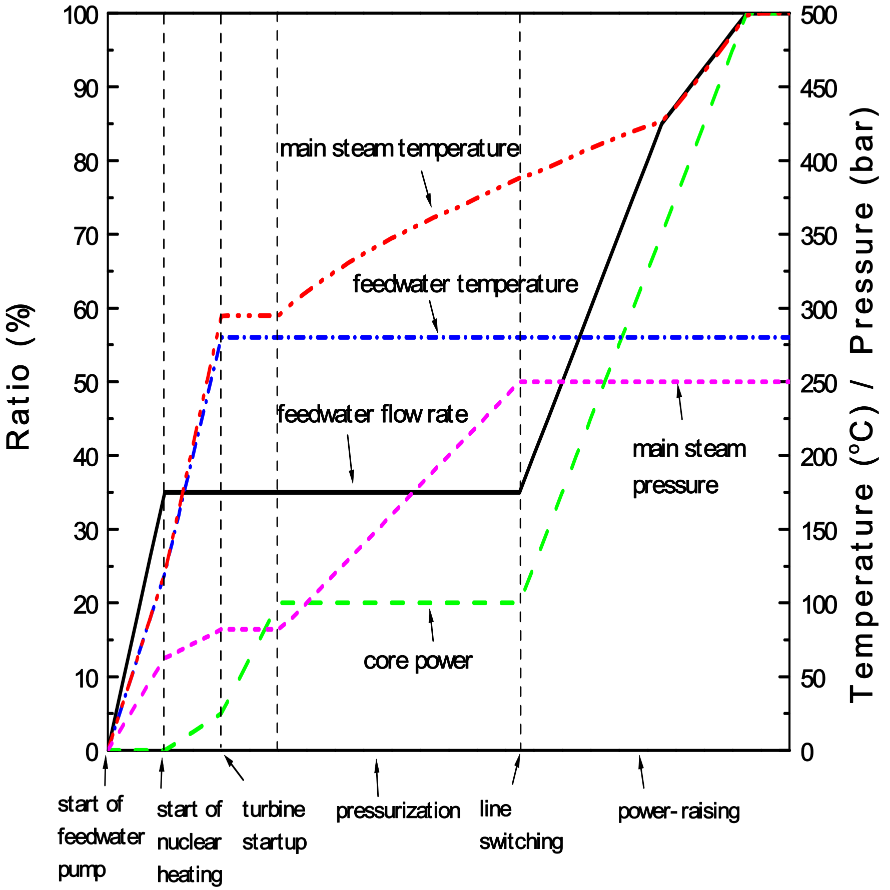
Stability Analysis during Sliding-Pressure Startup

- Coupled neutronic thermal-hydraulic stability analysis
- Thermal-hydraulic stability analysis
- Thermal-hydraulic analysis
- Sliding pressure startup procedures

Sliding pressure startup curve (Thermal criteria only)



Sliding pressure startup curve (Both Thermal and Stability criteria)

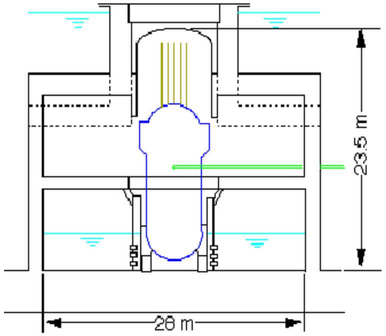


Scope of studies and Computer codes

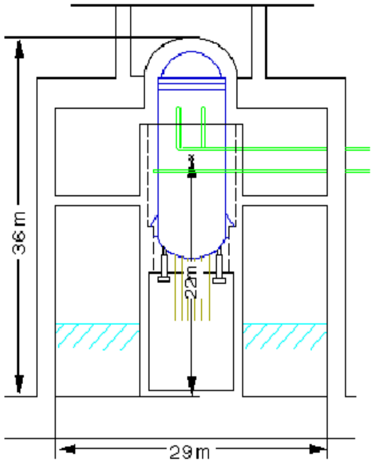
1. Fuel and core

   Single channel thermal hydraulics (SPROD), 3D coupled core neutronic/thermal-hydraulic (SRAC-SPROD), Coupled sub-channel analysis, Statistical thermal design method, Fuel rod behavior (FEMAXI-6), Data base of heat transfer coefficients of supercritical water
2. Plant system; Plant heat balance and thermal efficiency
3. Plant control
4. Safety; Transient and accident analysis at supercritical-and subcritical pressure, ATWS analysis, LOCA analysis (SCRELA)
5. Start-up (sliding-pressure and constant-pressure)
6. Stability (TH and core stabilities at supercritical and subcritical-pressure)
7. Probabilistic safety assessment

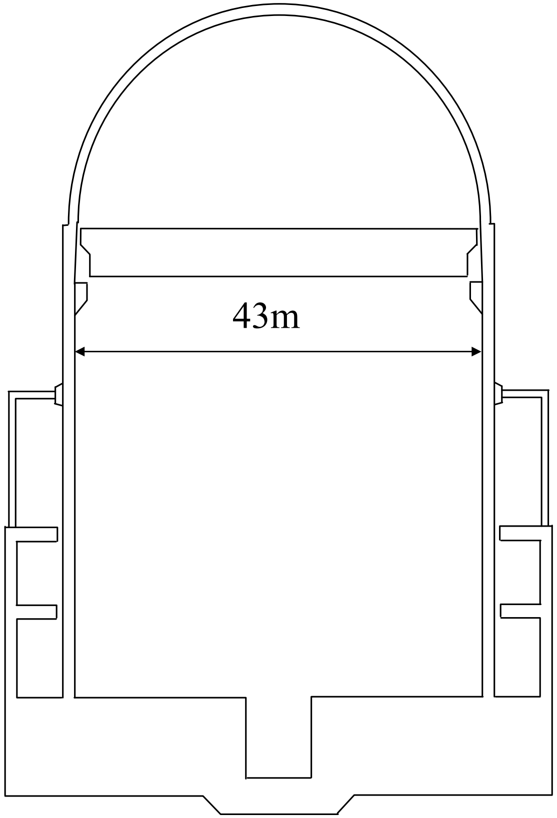
Economic potential



SCLWR-H(1700MWe)



ABWR(1350MWe)



PWR(1100MWe)

Comparison of containments