Joint ICTP-IAEA Course on Science and Technology of Supercritical Water-Cooled Rectors (SCWRs)
International Center for Theoretical Physics, Trieste, Italy, 27 June to 1 July, 2011

SC19
Plant dynamics and control

Yoshiaki Oka

Professor, Joint Department of Nuclear Energy, Waseda University

Emeritus Professor, University of Tokyo

Objectives

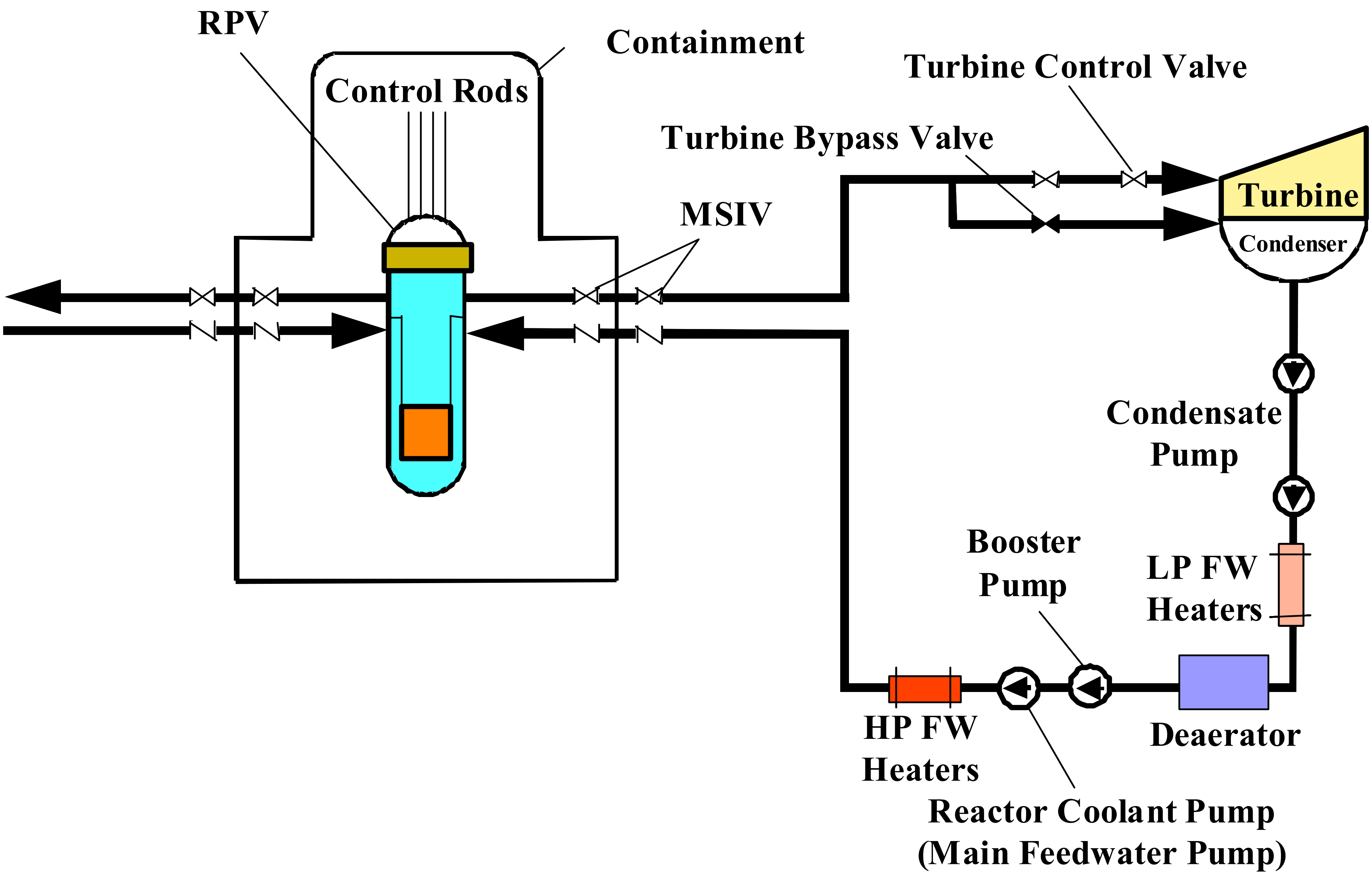
Objectives of the lecture are for understanding of

- Plant dynamics calculation models
- Plant control system
- Plant start-up system and thermal consideration during start-up
- Linear stability analysis method and analyses of thermal hydraulic and NT coupled stability at rated power and during start-up

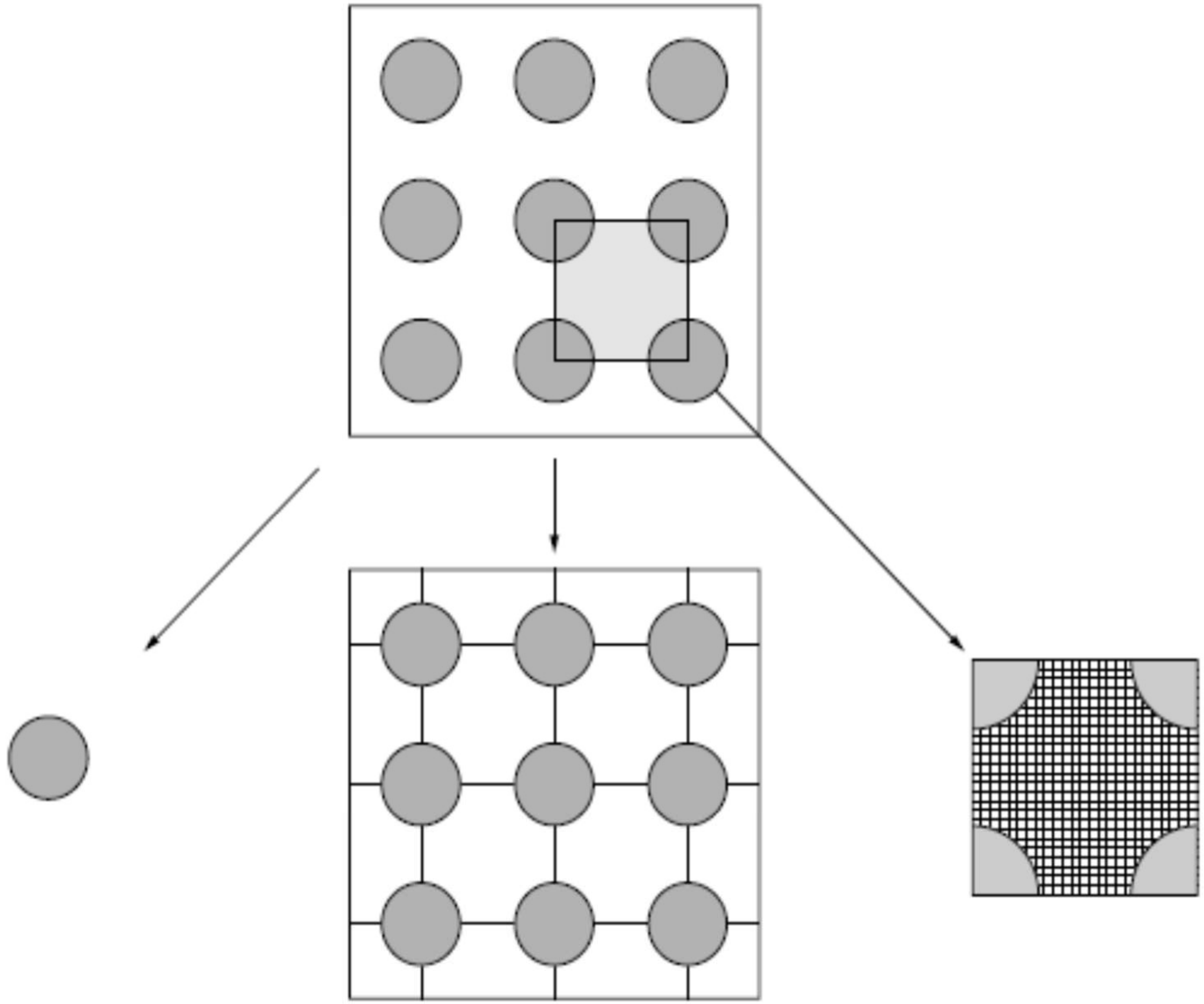
Plant dynamics calculation

1. Single channel calculation model
2. Node junction model
3. Plant dynamics calculation model

Main plant system of Super LWR



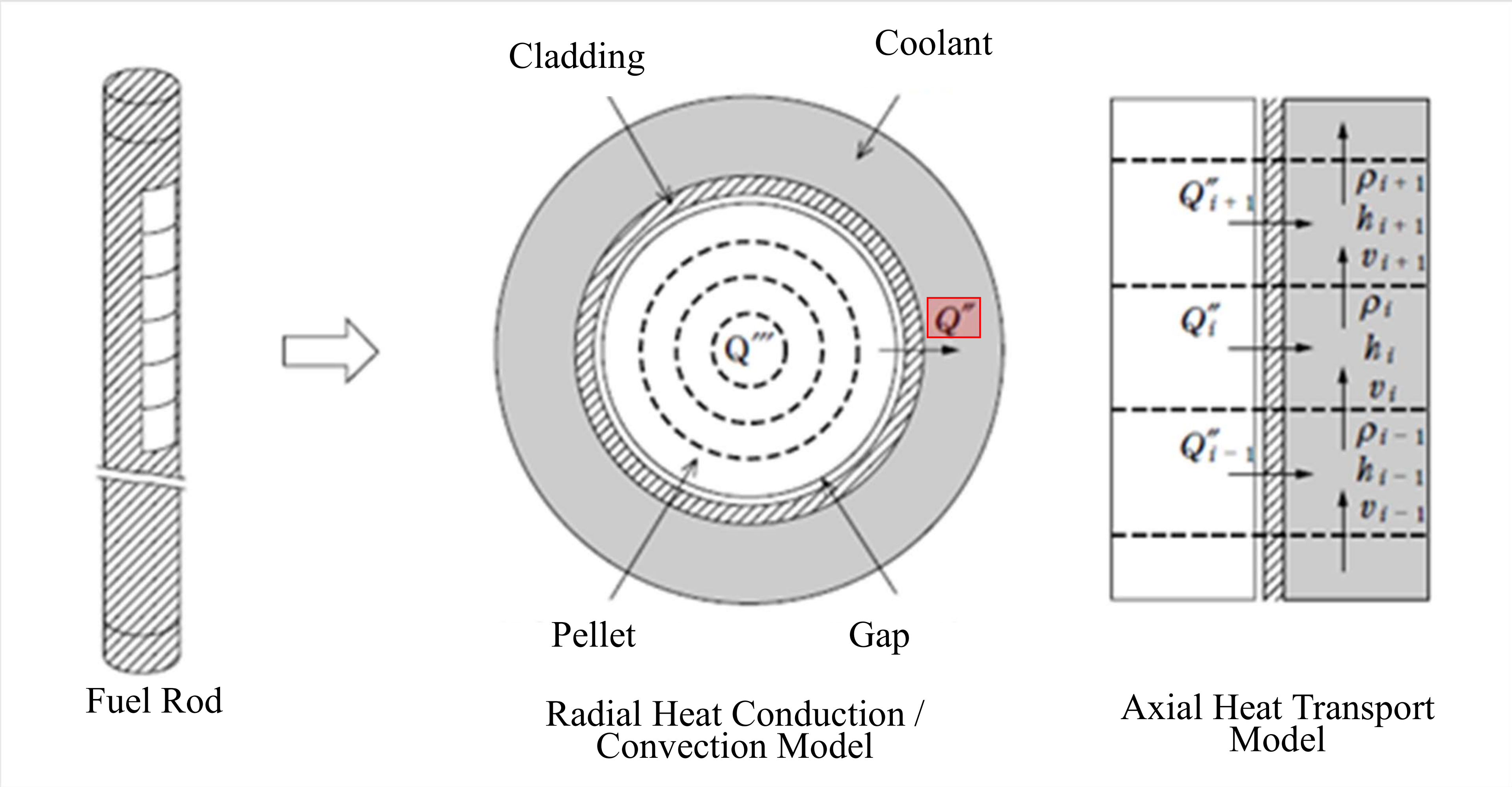
Models of fuel and core for thermal hydraulic analysis



(a) Single channel analysis model of a fuel rod and a core

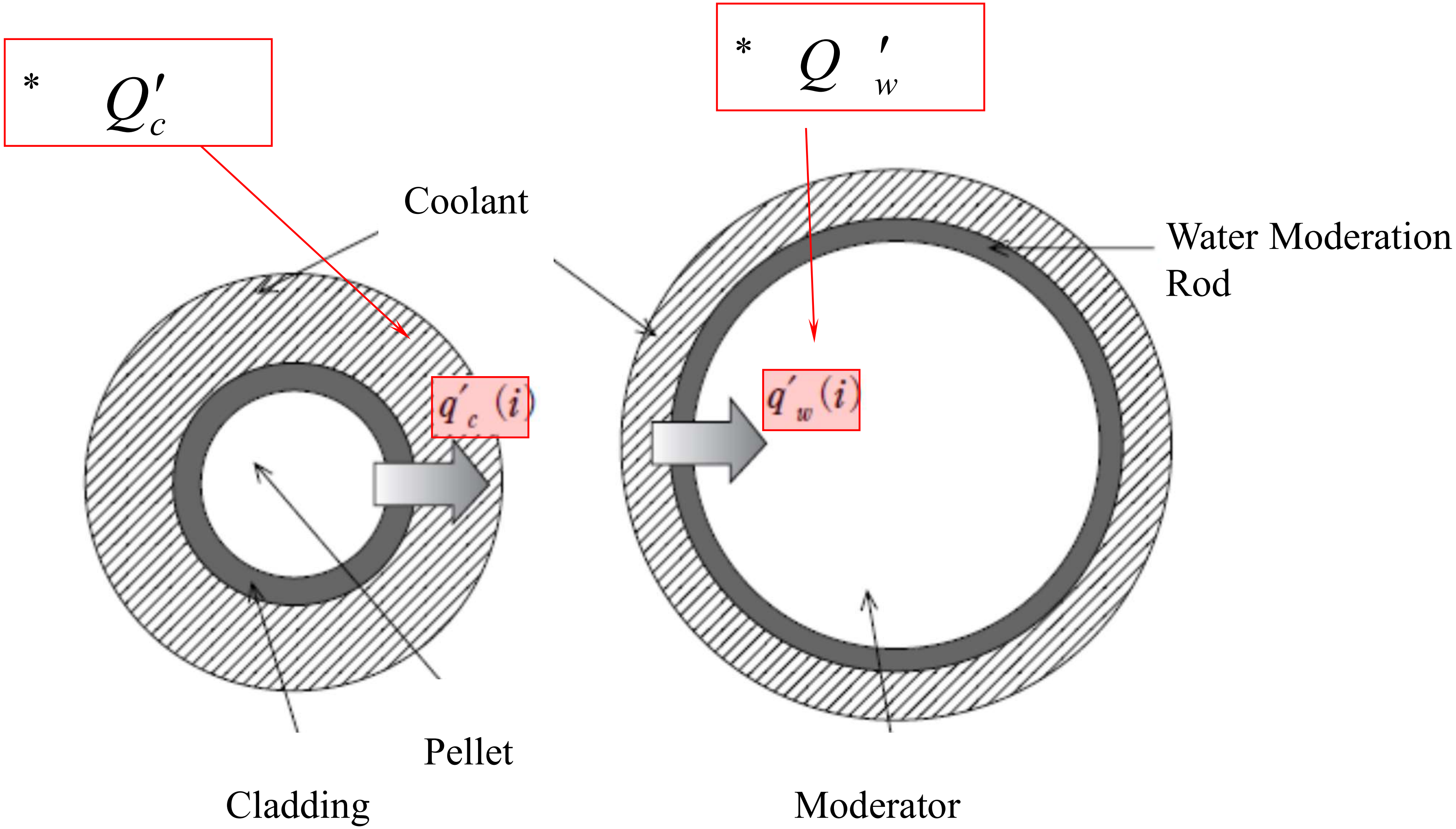
(b) Subchannel analysis model of fuel assembly

(c) CFD analysis model of a fuel channel

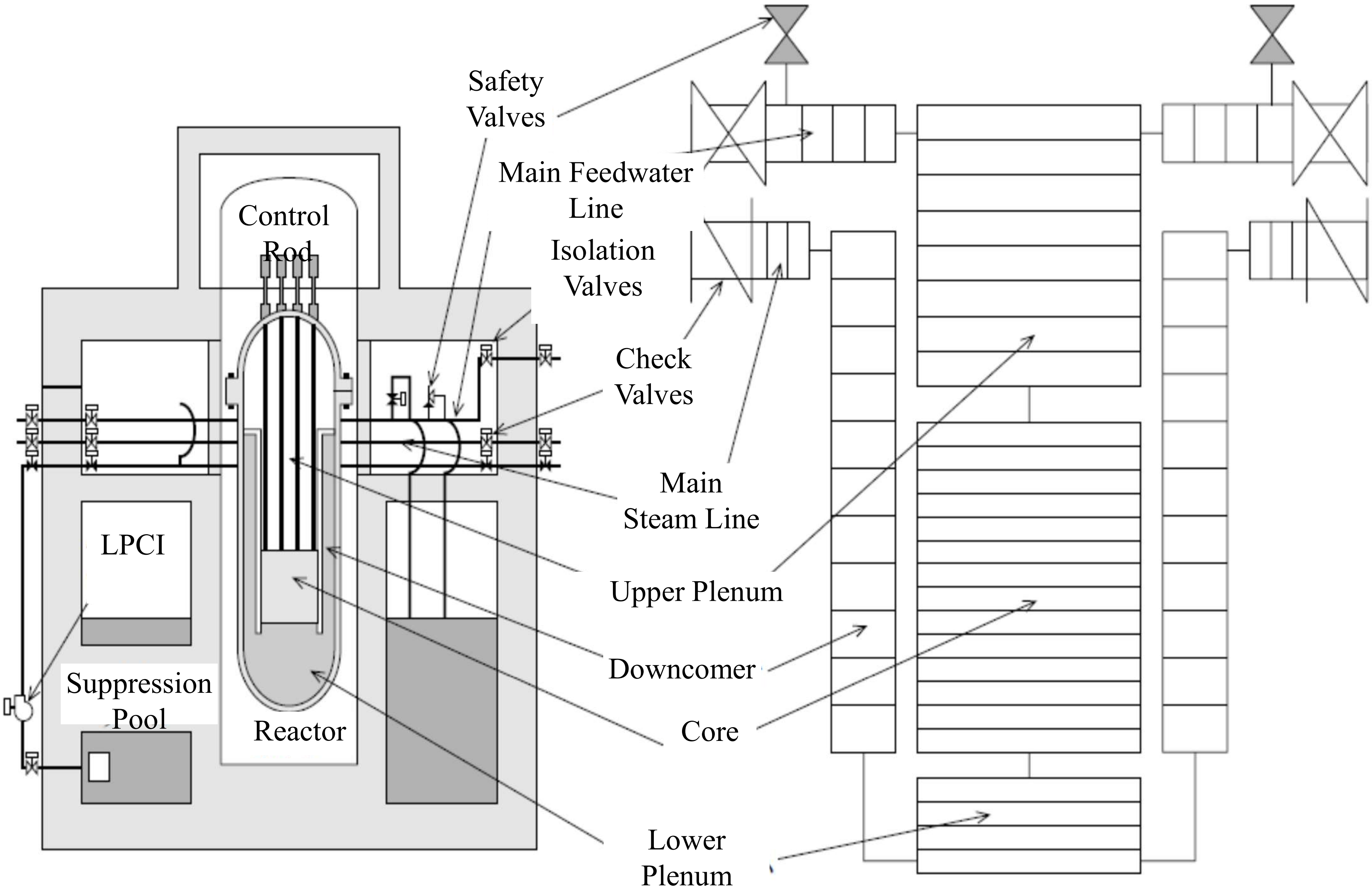


Single-Channel (Heat Transfer Calculation) Model

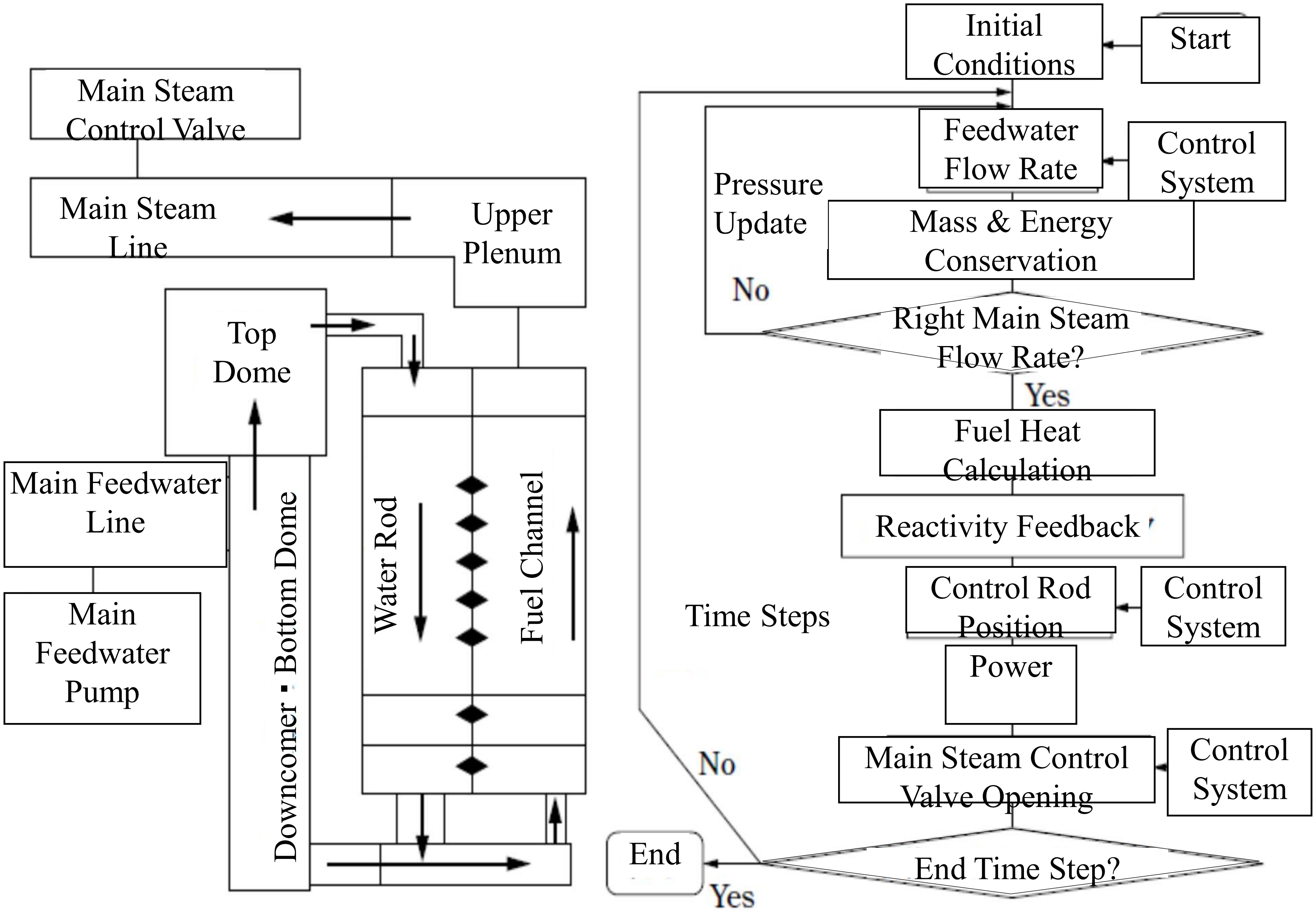
Single-Channel Thermal-Hydraulic Calculation Model of Fuel and Water Rod



Node Junction Model



Plant Dynamics Analysis Model and Calculation Flow Chart



Calculation model

- Neutron kinetics model
- Fuel rod heat transfer model
- Fuel channel thermal-hydraulic model
- Ex-core circulation model

Neutron kinetics model

- point kinetics model with six delayed neutron precursor groups
- Doppler and density reactivity feedback

$$\frac{\partial n(t)}{\partial t} = \frac{\Delta\rho - \beta}{\Lambda} n(t) + \sum_{i=1}^{6} \lambda_i C_i(t)$$

$$\frac{\partial C_i(t)}{\partial t} = \frac{\beta_i}{\Lambda} n(t) - \lambda_i C_i(t)$$

$$\delta\Delta\rho(t) = \left(\frac{\partial \Delta\rho}{\partial T_f^{ave}}\right) \delta T_f^{ave}(t) + \left(\frac{\partial \Delta\rho}{\partial \rho}\right) \delta\rho(t)$$

where, $n(t)$: number of neutrons

$C_i(t)$: precursor concentration of delayed neutron group i

t: time

β_i: fraction of delayed neutron group i

β: $\sum_{i=1}^{6}\beta_i$

$\Delta\rho$: reactivity

Fuel Rod Heat Transfer Model

- Only radial heat transfer is considered.

$$\rho_f C_p \frac{\partial}{\partial t}\left(T_f\right) = \frac{1}{r}\frac{\partial}{\partial r}\left(r k_f \frac{\partial T_f}{\partial r} \right) + q'''$$

$$T_f^{ave} - T_s = \left(\frac{r_f + t_c}{r_f} \right) \left[\frac{r_f}{4k_f} + \frac{1}{h_g} + \frac{t_c}{k_c} \right] q''$$

$$q''(r_c, t) = h_c \left(T_{r_c} - T \right)$$

where, C_p : specific heat of fuel pellet [J/kg-K]

k_f : thermal conductivity of fuel pellet [W/m-K]

q''' : power density [W/m^3]

r : radial distance [m]

T_f : fuel pellet temperature [K]

ρ_f : fuel pellet density [kg/m^3].

where, k_f : average thermal conductivity of pellet [W/m-K]

h_g : thermal gap conductance [W/m^2-K]

k_c : thermal conductivity of cladding [W/m-K]

q'' : heat flux from fuel pellet [W/m^2]

r_f : fuel pellet radius [m]

t_c : cladding thickness [m]

T_f^{ave} : fuel pellet average temperature [K]

T_s : cladding surface temperature [K].

where, h_c : heat transfer coefficient between cladding surface and coolant [W/m^2-K]

r_c : cladding radius [m]

T_s : cladding surface temperature [K]

T : coolant bulk temperature [K].

Thermal conductivity of LWR fuel pellet

$$k_f = \frac{3\,824}{402.4 + T_f^{ave}} + 6.1256 \times 10^{-11} \left(T_f^{ave} + 273\right)^3$$

where, T_f^{ave}: average temperature [C]

k_f : thermal conductivity of fuel pellet [W/m-K]

Thermal-Hydraulic Model

(Fuel Channel and Water Rod)

- Single-channel Single-phase One-dimensional Model
- Forward finite difference method for axial nodalization

Mass Conservation:

$$\frac{\partial \rho}{\partial t} + \frac{\partial (\rho u)}{\partial z} = 0$$

Energy Conservation:

$$\frac{\partial (\rho h)}{\partial t} + \frac{\partial (\rho u h)}{\partial z} = \frac{P_e}{A} q''$$

Momentum Conservation:

$$-\frac{\partial P}{\partial z} = \frac{\partial (\rho u)}{\partial t} + \frac{\partial (\rho u^2)}{\partial z} + \rho g \cos\theta + \frac{2f}{D_h} \rho u^2$$

$$f = 0.0791 \times \mathrm{Re}^{-0.25} \qquad \text{(Blasius equation)}$$

State Equation:

$$\rho = \rho(P, h)$$

where, t : time [s]

z : position [m]

ρ : coolant density [kg/m^3]

u : fluid velocity [m/s]

h : specific enthalpy [J/kg]

q'' : heat flux at fuel rod surface [W/m^2]

A : flow path area of fuel channel [m^2]

P_e : wetted perimeter of fuel rod [m]

P : pressure [Pa]

g : gravitational acceleration

D_h : hydraulic equivalent diameter of fuel channel [m]

Re : Reynolds number

θ : vertical angle of fuel channel

f : frictional coefficient

for example, $f = 0.0791 \times \mathrm{Re}^{-0.25}$ (Blasius equation).

Ex-core Circulation Model

Orifice Model:

$$\Delta P = \zeta \frac{\rho u^2}{2}$$

Feedwater pump model:

$$\delta P = C_{pump} \delta u$$

Feedwater pipe model:

$$-\frac{dP}{dz} = \frac{d}{dt}\rho u + \frac{d}{dz}\rho u^2 + \frac{2f}{D}\rho u^2$$

Exit valve model:

$$\Delta P = \zeta \frac{\rho u^2}{2}$$

where,

P :	pressure [Pa]
ΔP :	pressure drop [Pa]
δP :	pressure change of pump [Pa]
ζ :	form pressure drop coefficient
ρ :	coolant density [kg/cm^3]
u :	fluid velocity [m/s]
δu :	fluid velocity change of pump
C_{pump} :	pressure drop coefficient of pump
z :	position [m]
t :	time [s]
f :	friction pressure drop coefficient
D :	diameter of feedwater pipe [m]

Plant control

Q5. Plant contol system?

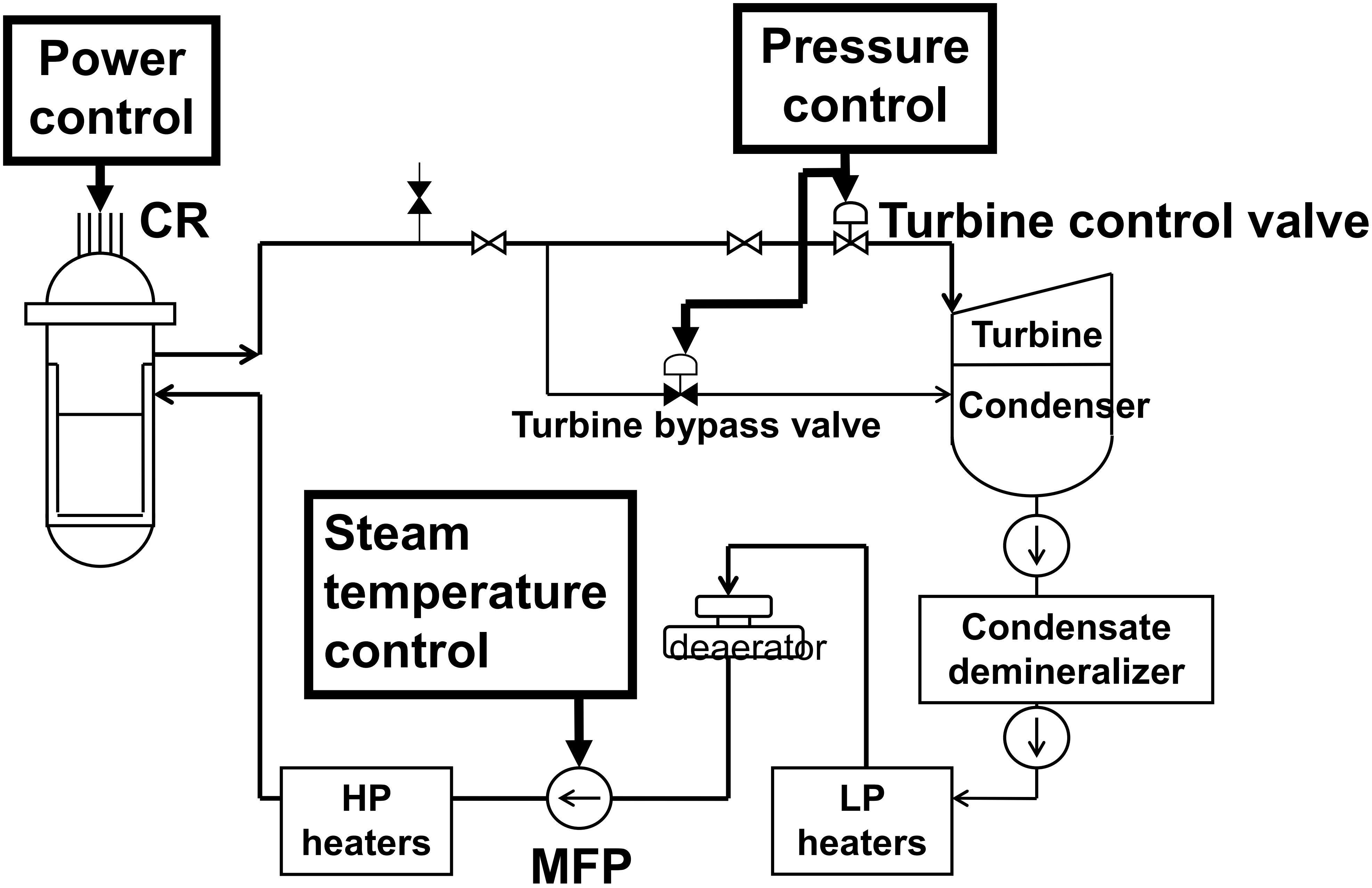
How to control reactor power,
steam (outlet coolant) temperature,
and reactor pressure?

A5. Study / follow BWR control system design
prepare plant dynamics code.

procedure

1. Find sensitivity of control parameters
2. Design control system
3. Assess stable response against pertubations

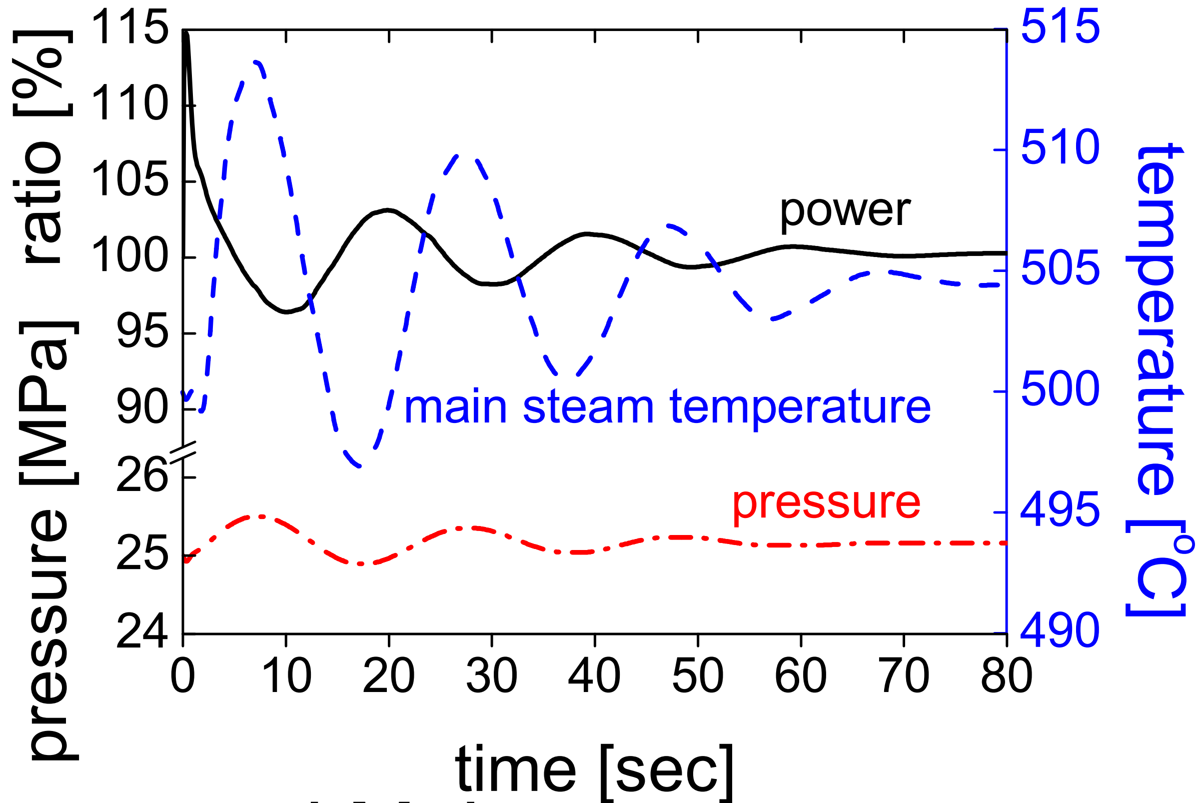
Plant control system



Plant dynamics without a control system

Step response – Reactivity Insertion -

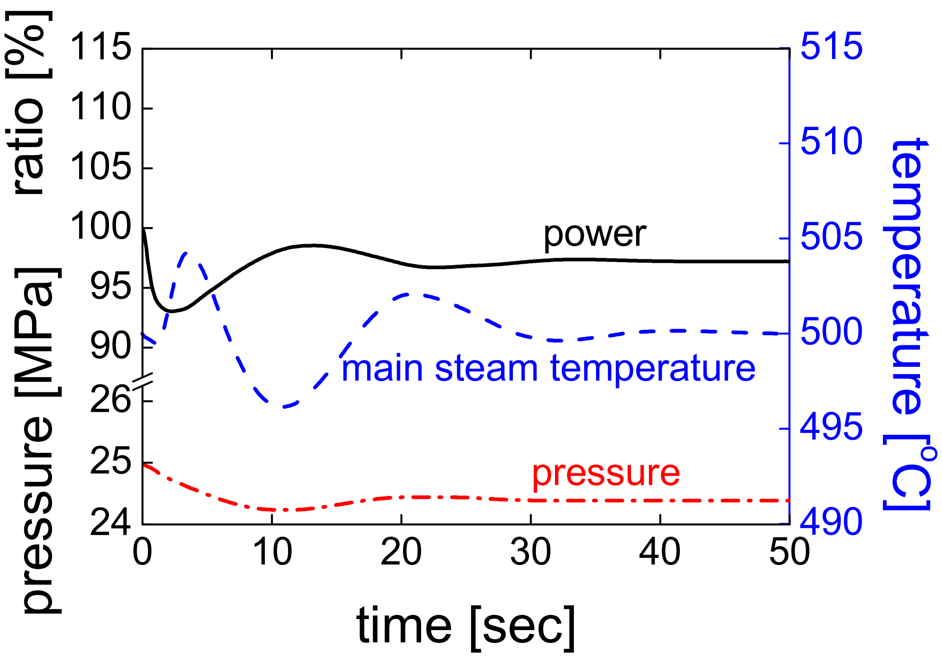
$0.1 inserted stepwise resulting from CR withdrawal



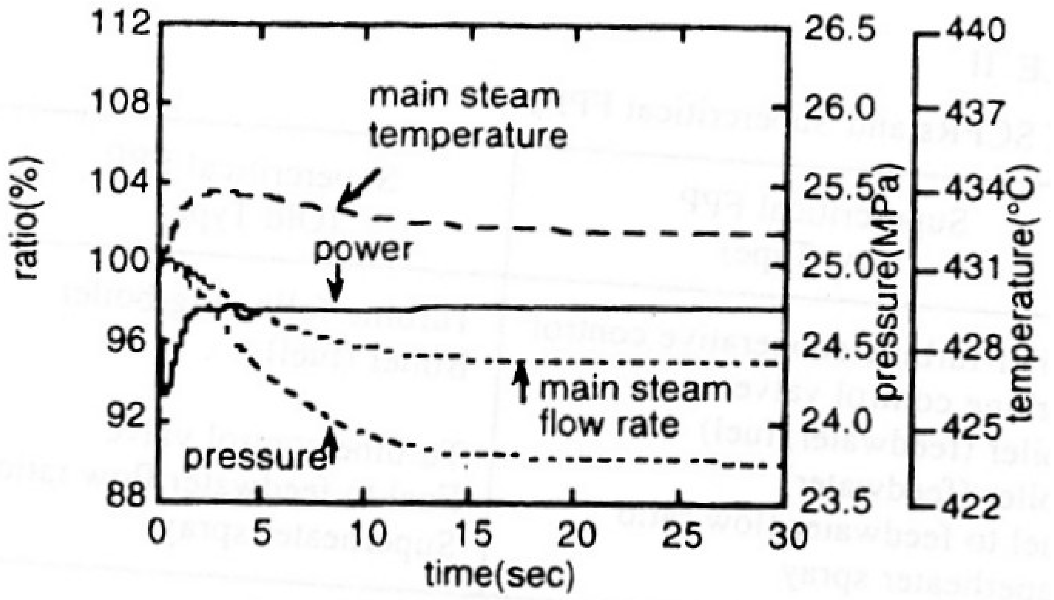
<u>Reactor power and Main steam temperature sensitive to CR position</u>

Step response - Change of FW flow rate -

FW flow rate decreases stepwise to 95%.



SCLWR-H



SCFR

Nakatsuka, et. al., Nucl. Tech., vol. 121 (1998)

Reactor power is not so sensitive to FW flow rate.

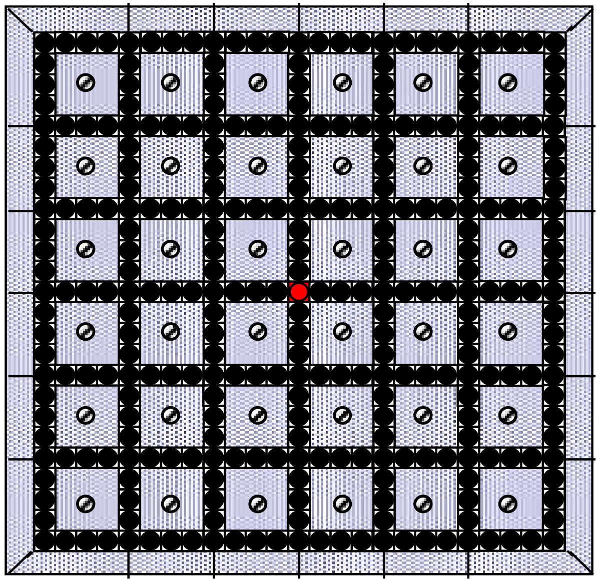
Why reactor power is not sensitive to flow rate?

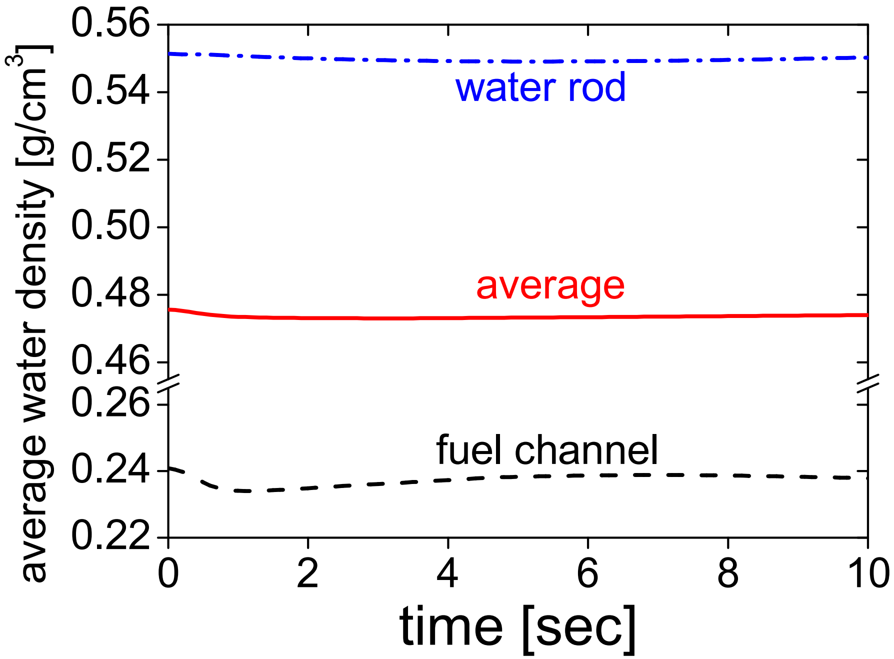
• Density change is small in WR because heat transfer is small between fuel channels and WR.

• More than 70% (volume) of water is in WR.

Change of average water density is small.

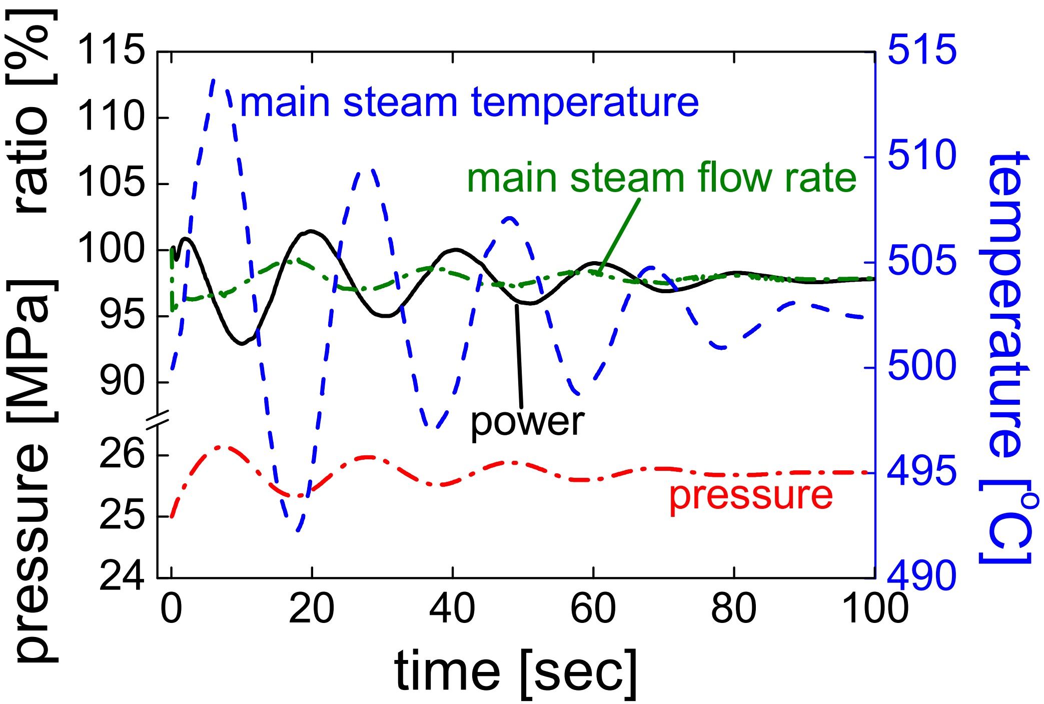
→ Reactivity change is small.





Step response
- Change of turbine control valve opening -

Main steam flow rate is decreased to 95%.



<u>Pressure and Main steam temperature are sensitive to turbine control valve opening.</u>

Power control

$$v = \begin{cases} v_{\max} e / b & e \le b \\ v_{\max} & e > b \end{cases}$$

v: control rod drive speed [cm/s]
v_{max}: maximum speed [cm/s]
e : deviation of the power from the setpoint
b : coefficient that converts power deviation to maximum drive speed

Steam temperature control

$$\frac{du(t)}{dt} = K_P e(t) + K_I \int_0^t e(t)dt$$

u(t): Feedwater demand signal [%]

e(t): Deviation of main steam temperature from setpoint

K_p: Proportional gain [%]

K_I: Integral gain [% s^{-1}]

K_p and K_I are optimized as 0.5 and 0.0, respectively.

Pressure control system

Same as BWR

$$Vr(t) = 100 - \frac{P_{set} - P(t)}{K} \qquad V(t) = Vr(t) + 2\frac{dVr(t)}{dt} - 5\frac{dV(t)}{dt}$$

Vr(t): Demand signal of opening [%]

V(t): Valve Opening [%]

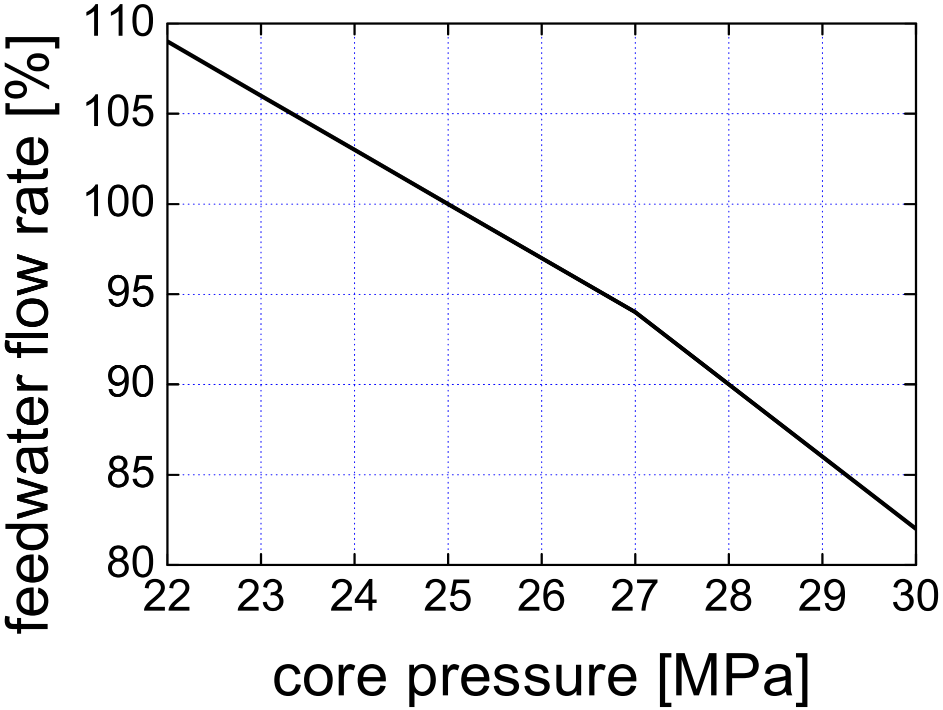
P_{set}: Pressure setpoint [MPa]

P(t): Turbine inlet pressure [MPa]

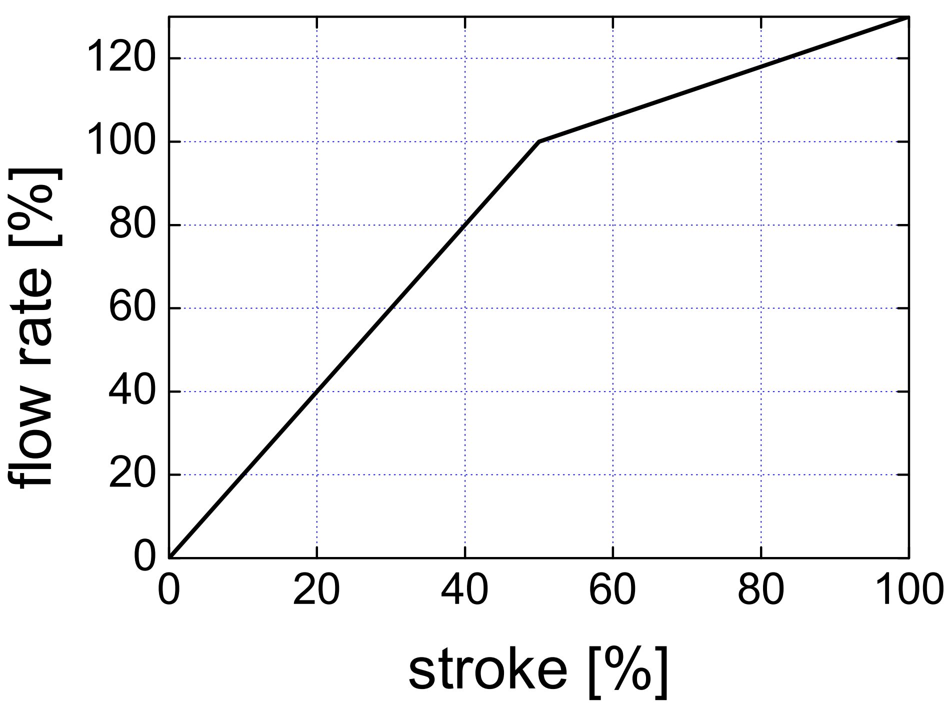
K: Gain converting pressure deviation into valve opening [MPa]

K is optimized as 0.25MPa.

Characteristics of FW pumps and turbine control valves



FW flow rate vs. Core pressure

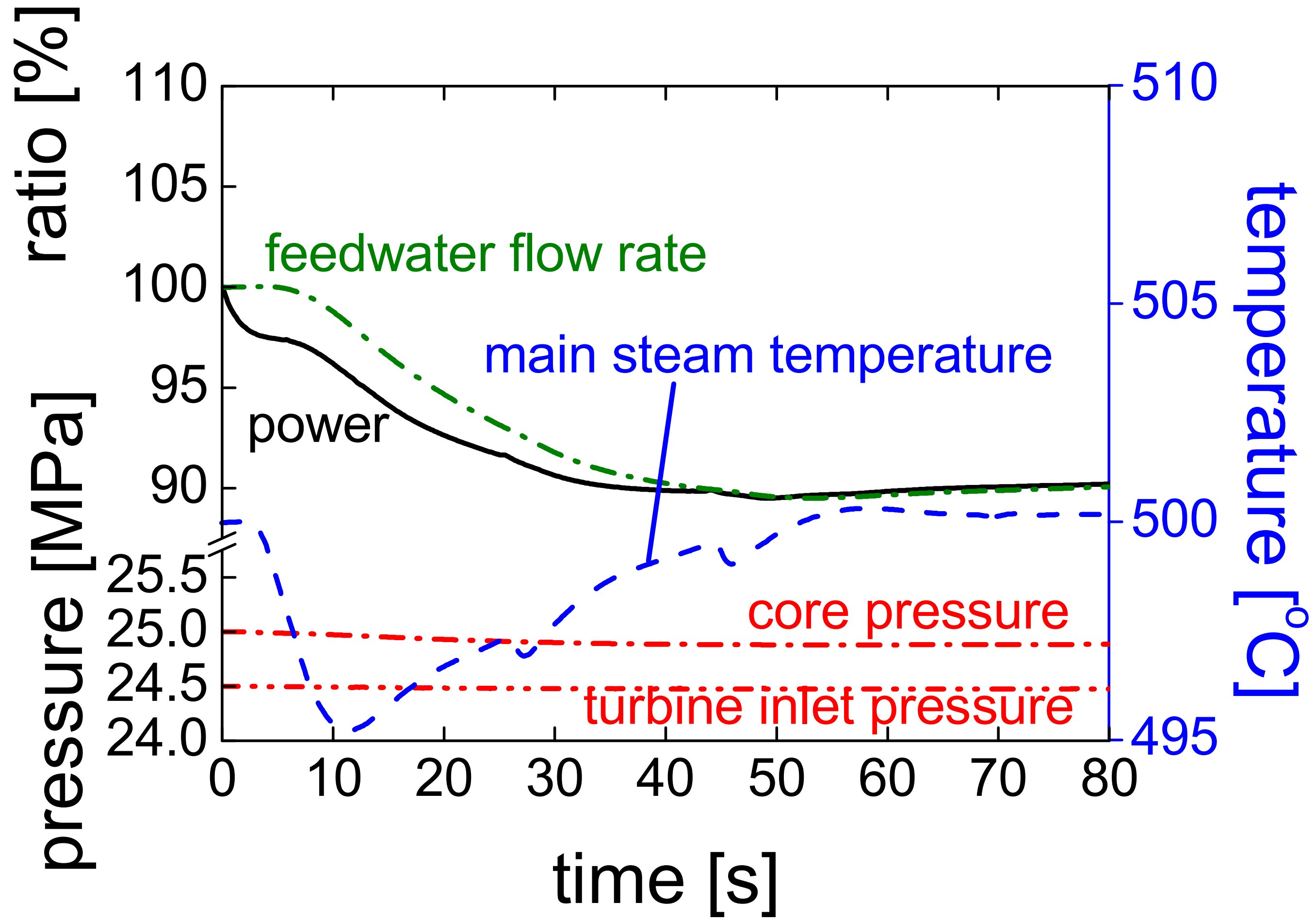


Main steam flow rate vs.
Turbine control valve stroke

Plant dynamics with control system

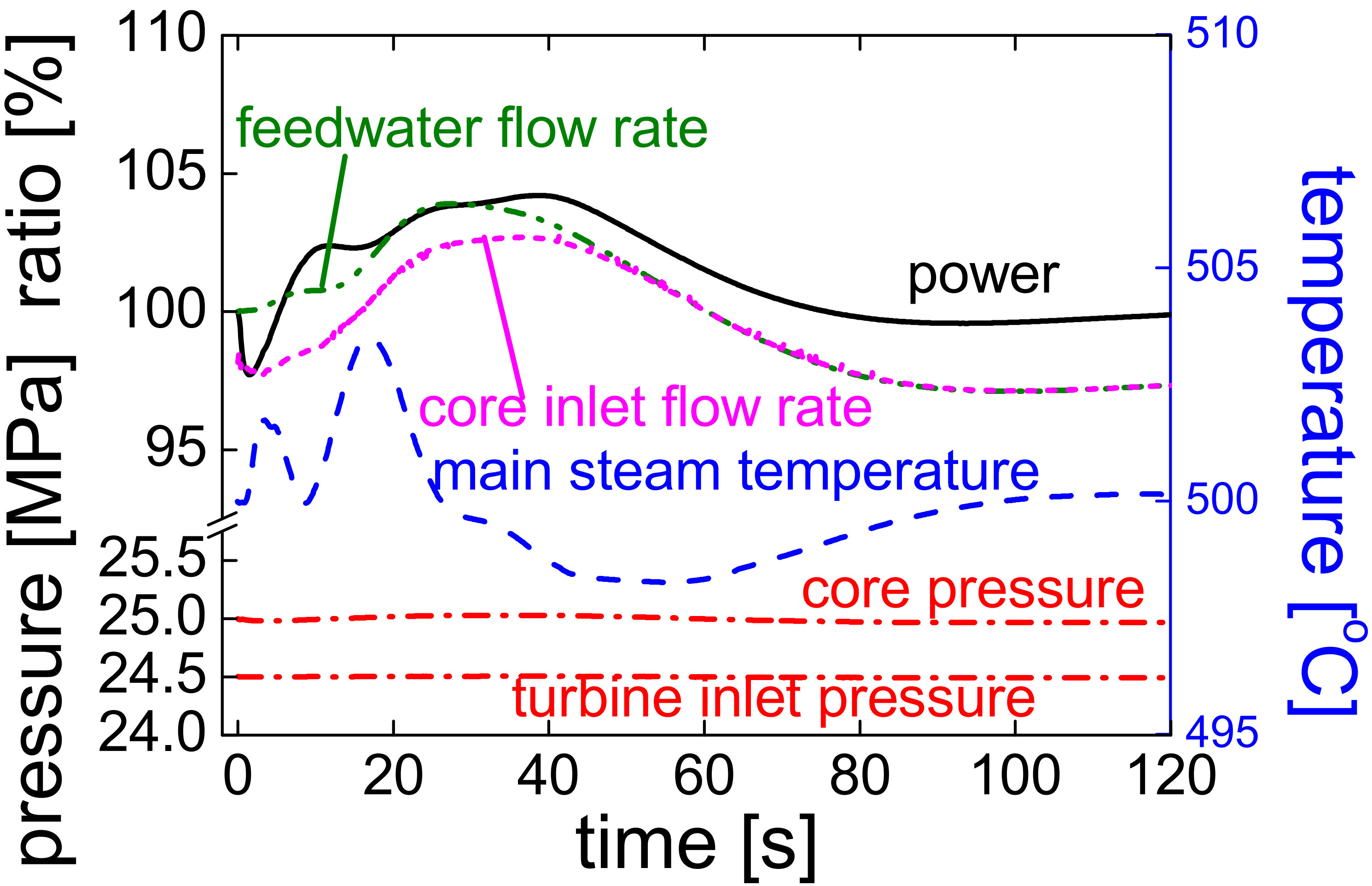
- Setpoint change of reactor power -

Setpoint of reactor power: 100 → 90%

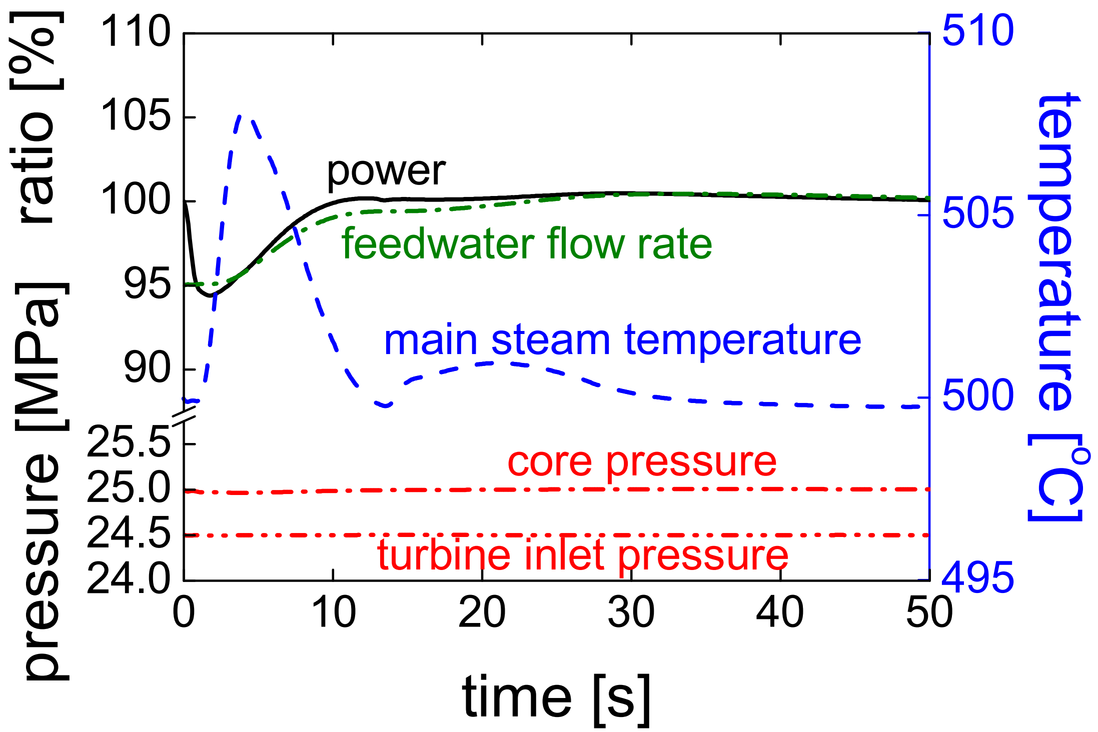


- Decrease in feedwater temperature -

Feedwater temperature: 280 → 270°C



- Decrease in feedwater flow rate -

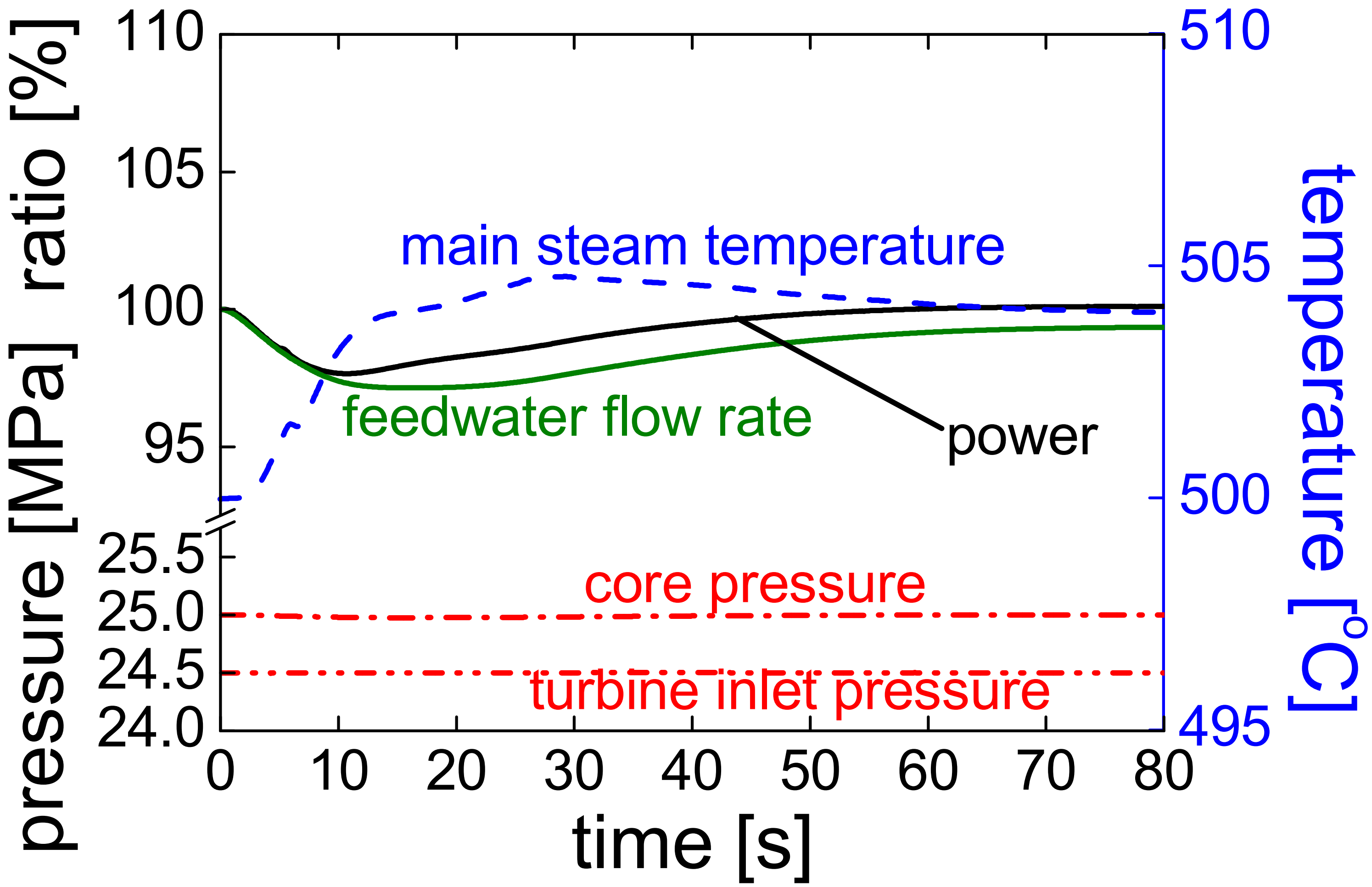


Decrease in feedwater flow rate
100 → 95%

Coolant density coefficient [dk/k/(g/cm^3)]	0.04	0.1	**0.2**	0.4	0.6	1.0
Power variation [%]	2	3	**6**	11	18	31
Steam temperature variation [°C]	11	10	**8**	7	6	-
Settling time [s]	60	40	**30**	60	90	-

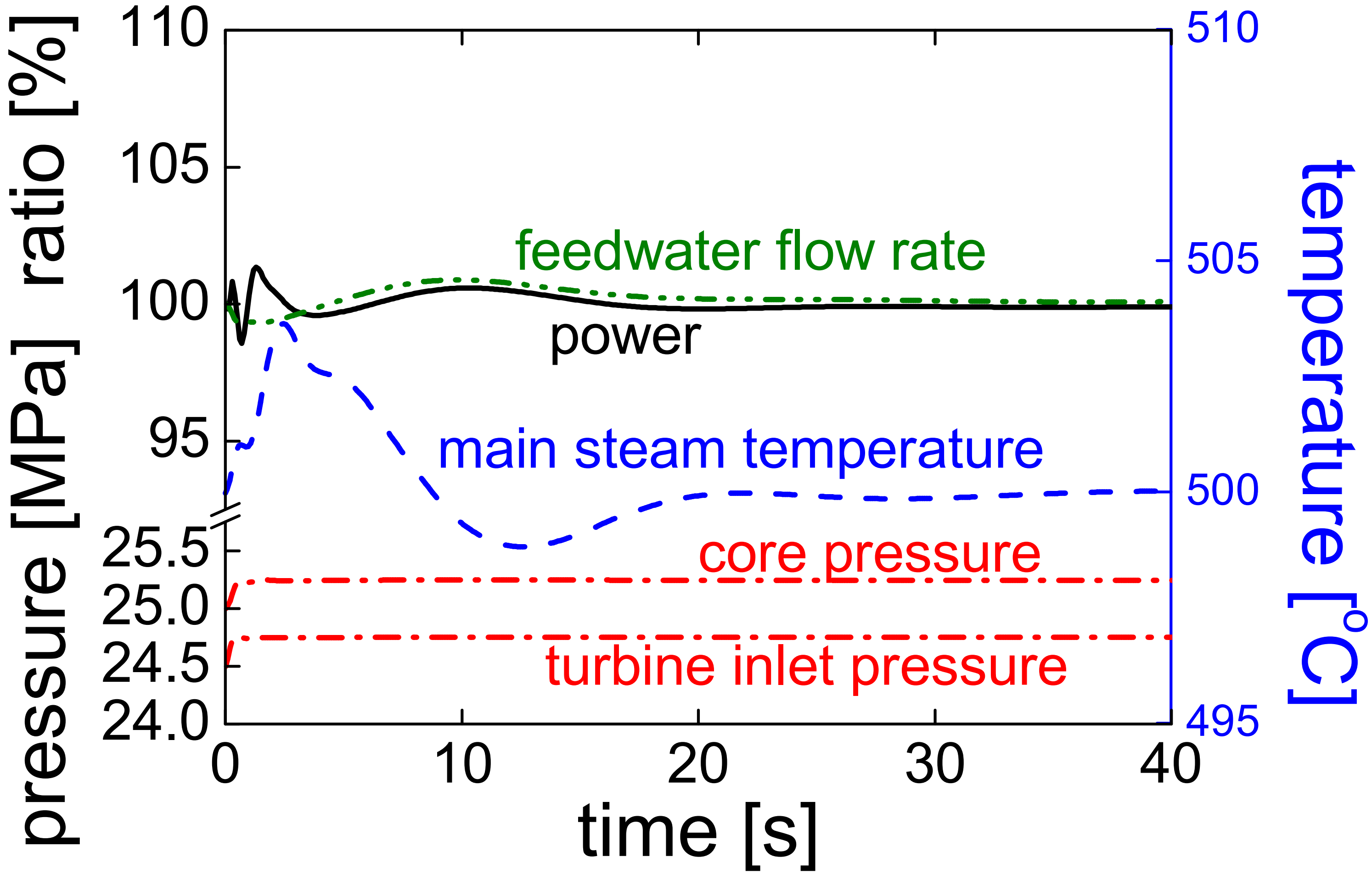
- Setpoint change of steam temperature -

Setpoint of main steam temperature: 500 → 504℃



- Setpoint change of pressure -

Setpoint of turbine inlet pressure: 24.5 → 24.75MPa



Comparison of plant control strategy

BWR: Turbine-following-Reactor

PWR: Reactor-following-Turbine

FPP: Turbine-Boiler-coordination

SCR: Turbine-following-Reactor

<table>
<tr><th></th><th>BWR</th><th>PWR</th><th>FPP</th><th>SCR</th></tr>
<tr><td>Electric power</td><td>Reactor power</td><td>Turbine control valve</td><td rowspan="3">Turbine control valve and Boiler input</td><td>Reactor power</td></tr>
<tr><td>Steam pressure</td><td>Turbine control valve</td><td>Reactor power</td><td>Turbine control valve</td></tr>
<tr><td>Reactor Power</td><td>Recirculation (and CR)</td><td>Boron (and CR)</td><td>CR</td></tr>
</table>

Plant start-up

1. Thermal criteria:

Maximum cladding temperature should not exceed the limit at full power.

2. Mechanical stress criteria (not discussed here):

Thermal stress should not be excessive.

Ex. Coolant temperature rise of the reactor pressure vessel should be below 55C/h for BWR

Q6: Plant Start-up system?

Q7: How to start-up the reactor and turbine?

Q8: What are limiting parameters during start-up?

Q9: When starting from subcritical-pressure
boiling transition occurs inevitably in once-through reactor,
How to deal with it?

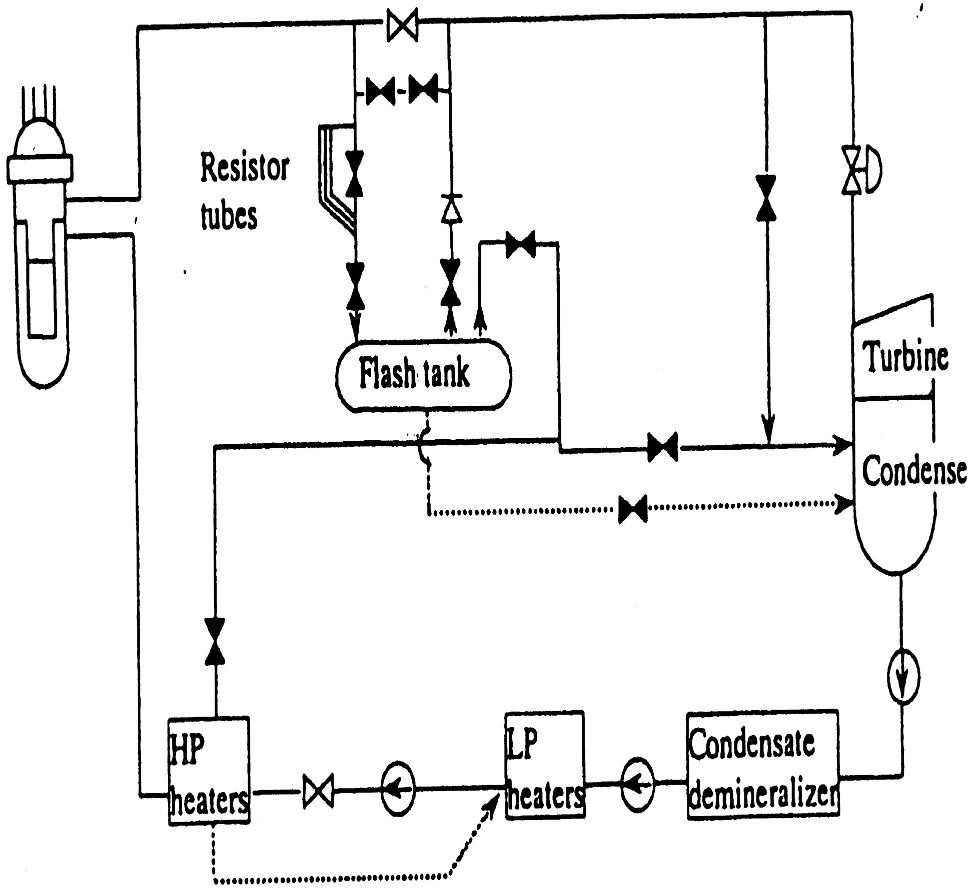
A6: Refer to supercritical-fossil-fired power plants(FPP)

A7: Plant start-up analysis and stability analysis

A8: Maximum cladding temperature /
moisture content in steam (not to damage tubine blades)/
(thermal-hydraulic and NTcoupled) stability

A9: Keep cladding temperature below that of the rated power/
same as the FPP

Constant Pressure Startup System



Constant pressure supercritical water-cooled reactor

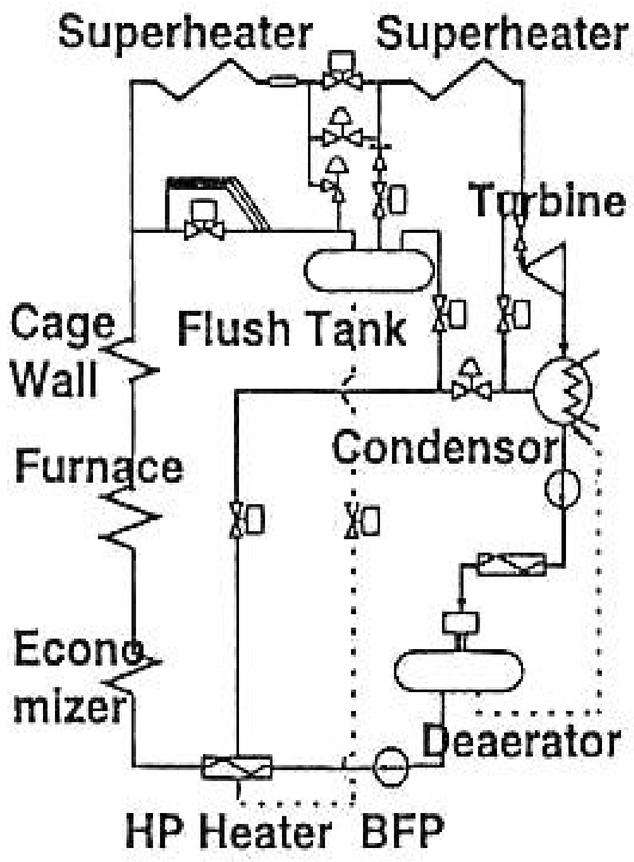
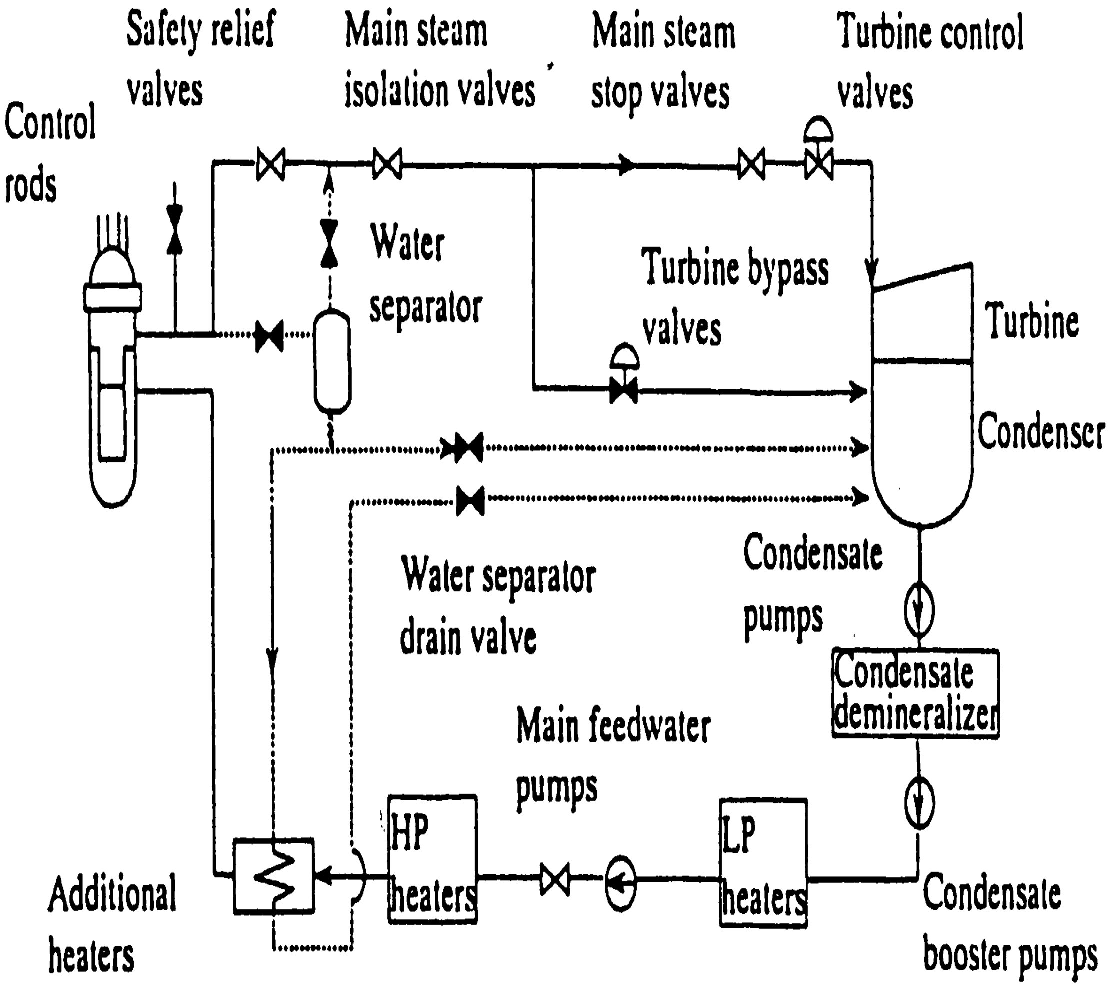


Fig.17 Constant pressure, supercritical fossil-fired power plant

Nuclear heating starts at supercritical pressure.

Sliding Pressure Startup System



Sliding pressure supercritical water-cooled reactor

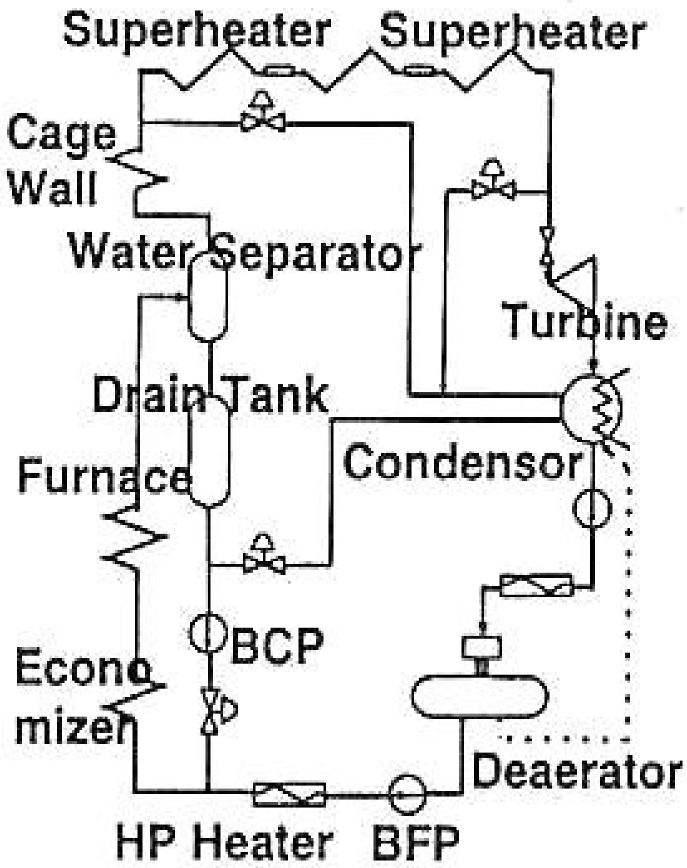


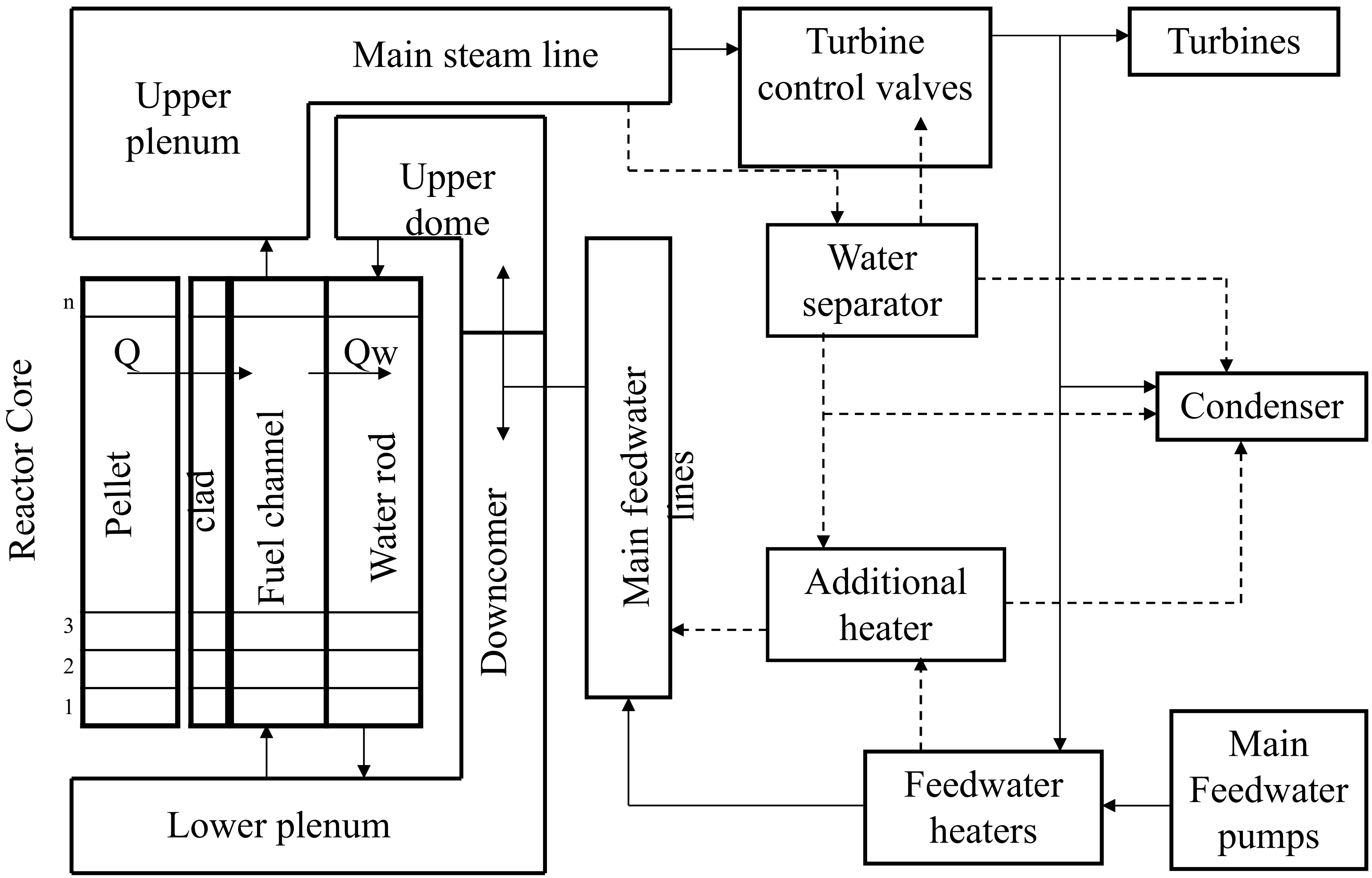
Fig.18 Sliding pressure fossil fired power plant

Nuclear heating starts at subcritical pressure.

Water separator is installed on a bypass line.

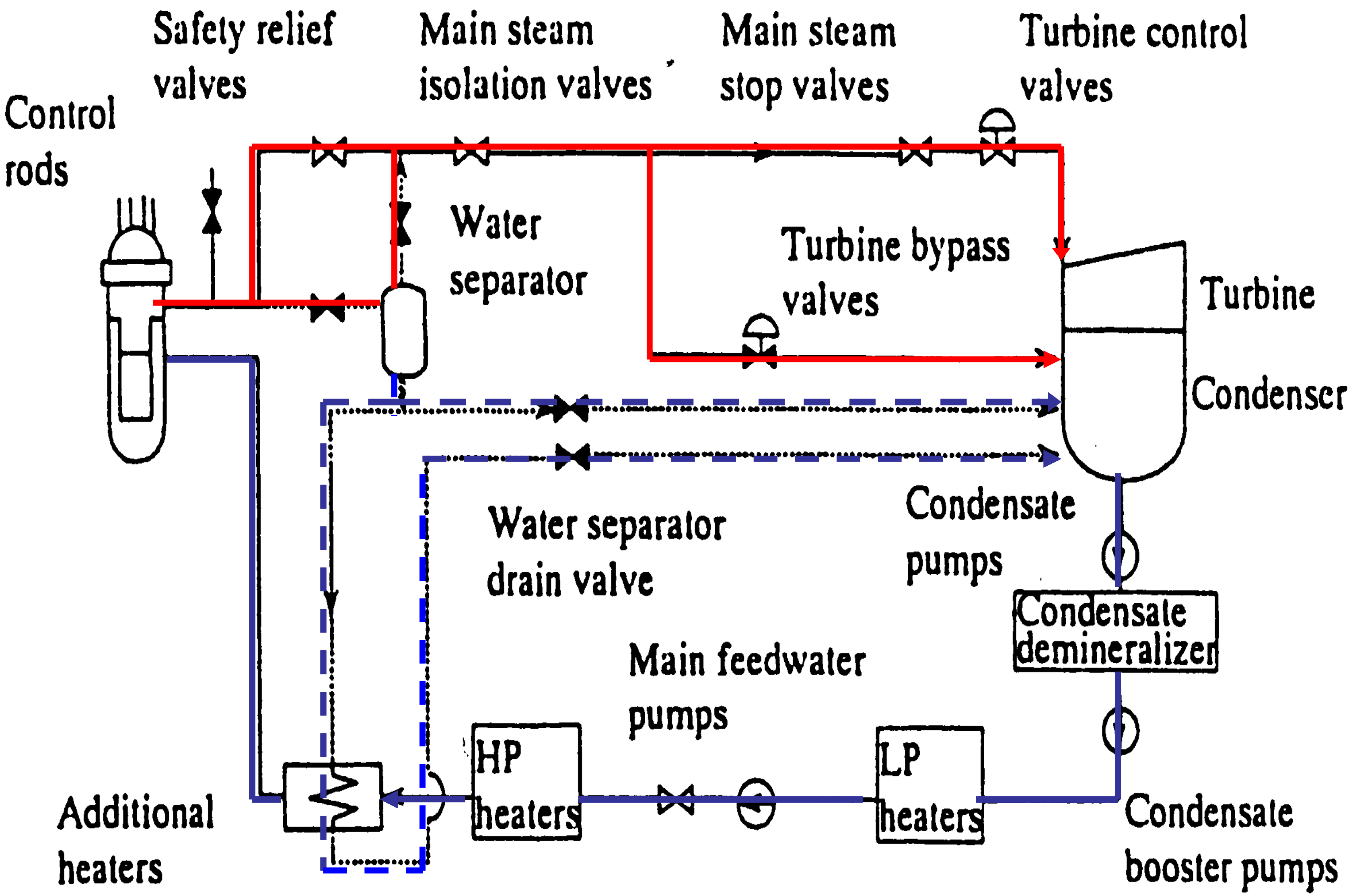
Startup Systems

- Constant pressure startup system
  - nuclear heating starts at supercritical pressure
  - constant pressure during load change
  - a flash tank and pressure-reducing valves required
- Sliding pressure startup system
  - nuclear heating starts at a subcritical pressure
  - sliding pressure operation at low load
  - steam-water separator, drain tank, drain valves and additional heaters required



Calculation Model for Sliding Pressure Startup

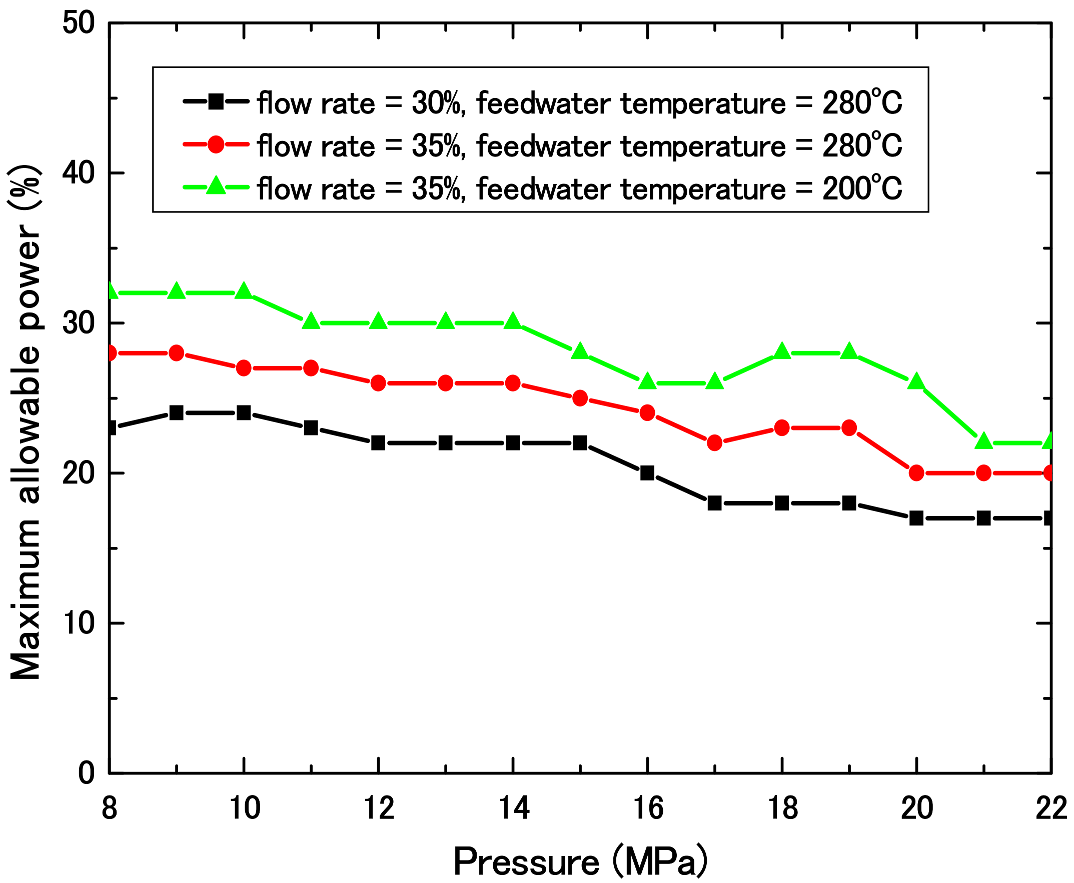
Sliding Pressure Startup Procedure



Pressurization Phase

Maximum allowable power

- maximum cladding surface temperature must be less than 620 °C.
- boiling must not occur in the water rods.

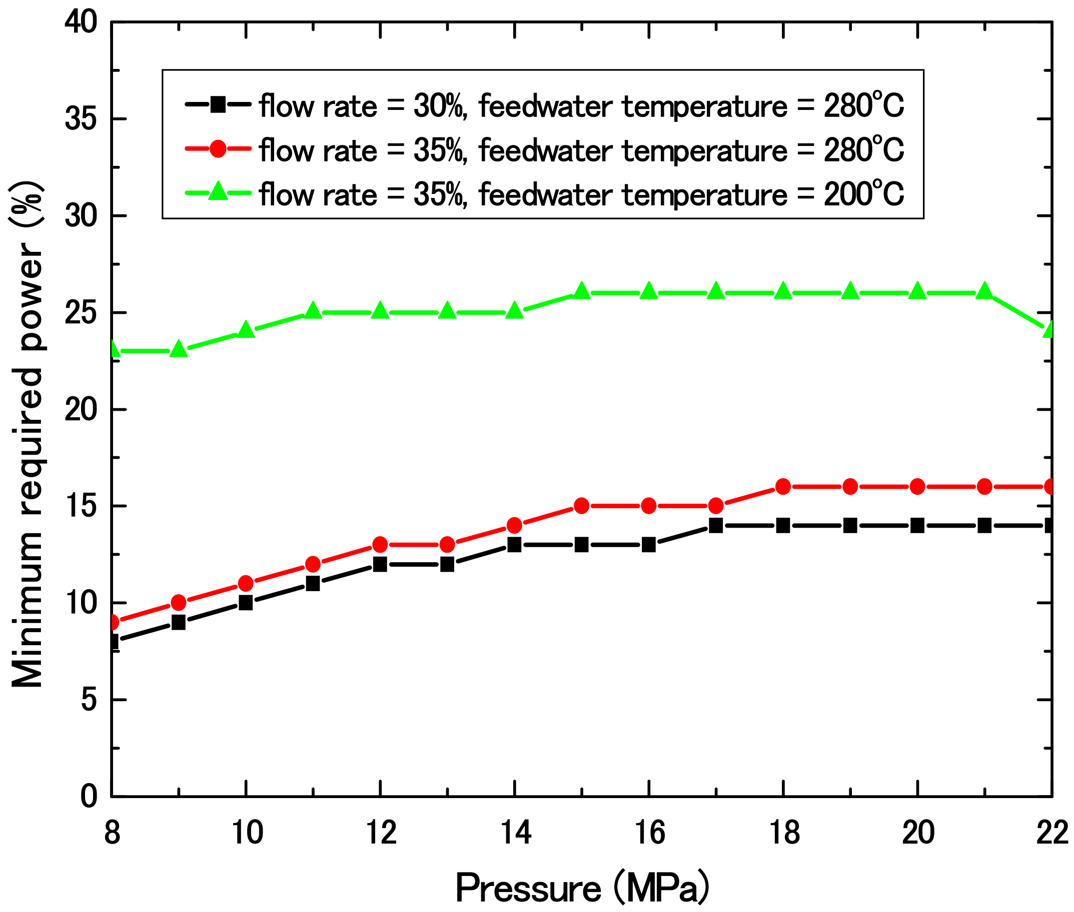


- Increasing the flow rate can increase the maximum allowable power.
- Decreasing the feedwater temperature can increase the maximum allowable power.

Pressurization Phase

Minimum required power

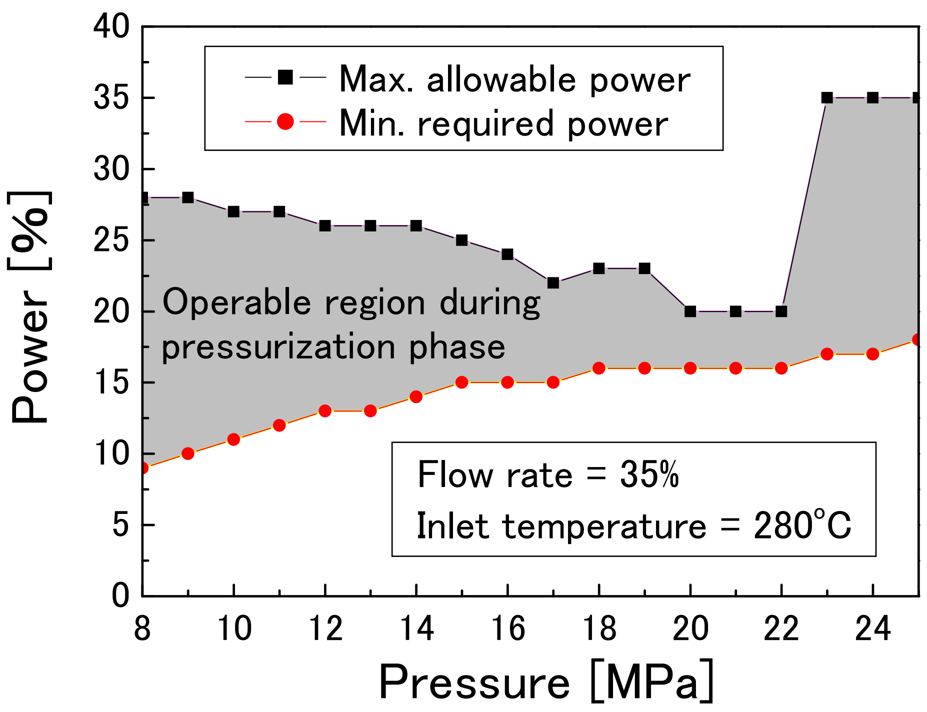
• core outlet enthalpy must be high enough to provide the required turbine inlet steam enthalpy.

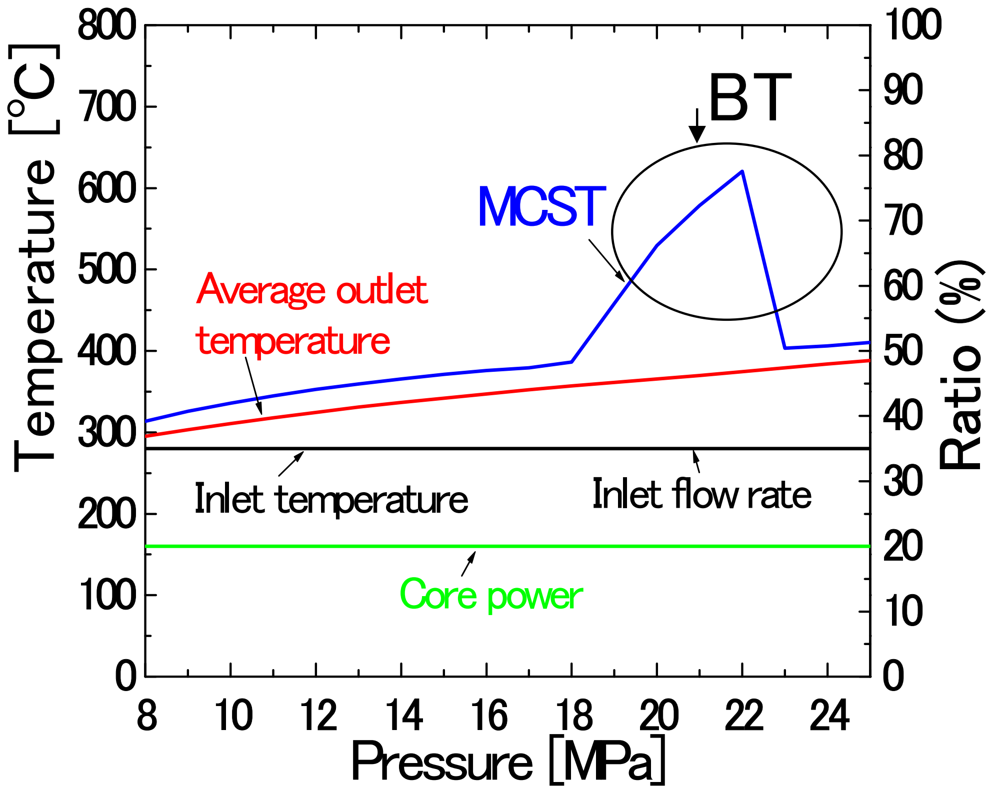


• Decreasing the flow rate can decrease the minimum required power.

• Increasing the feedwater temperature can decrease the minimum required power.

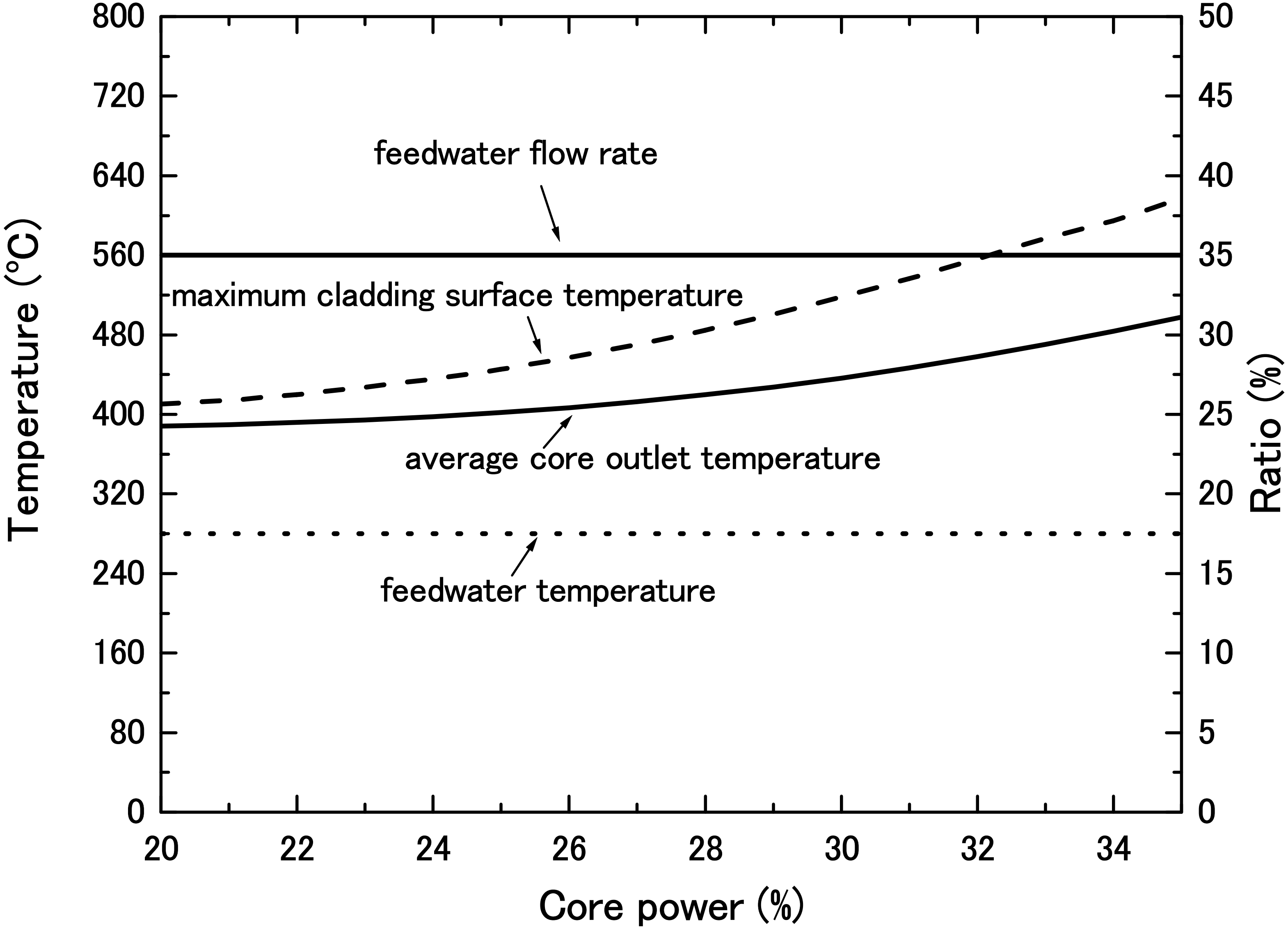
Pressurization phase





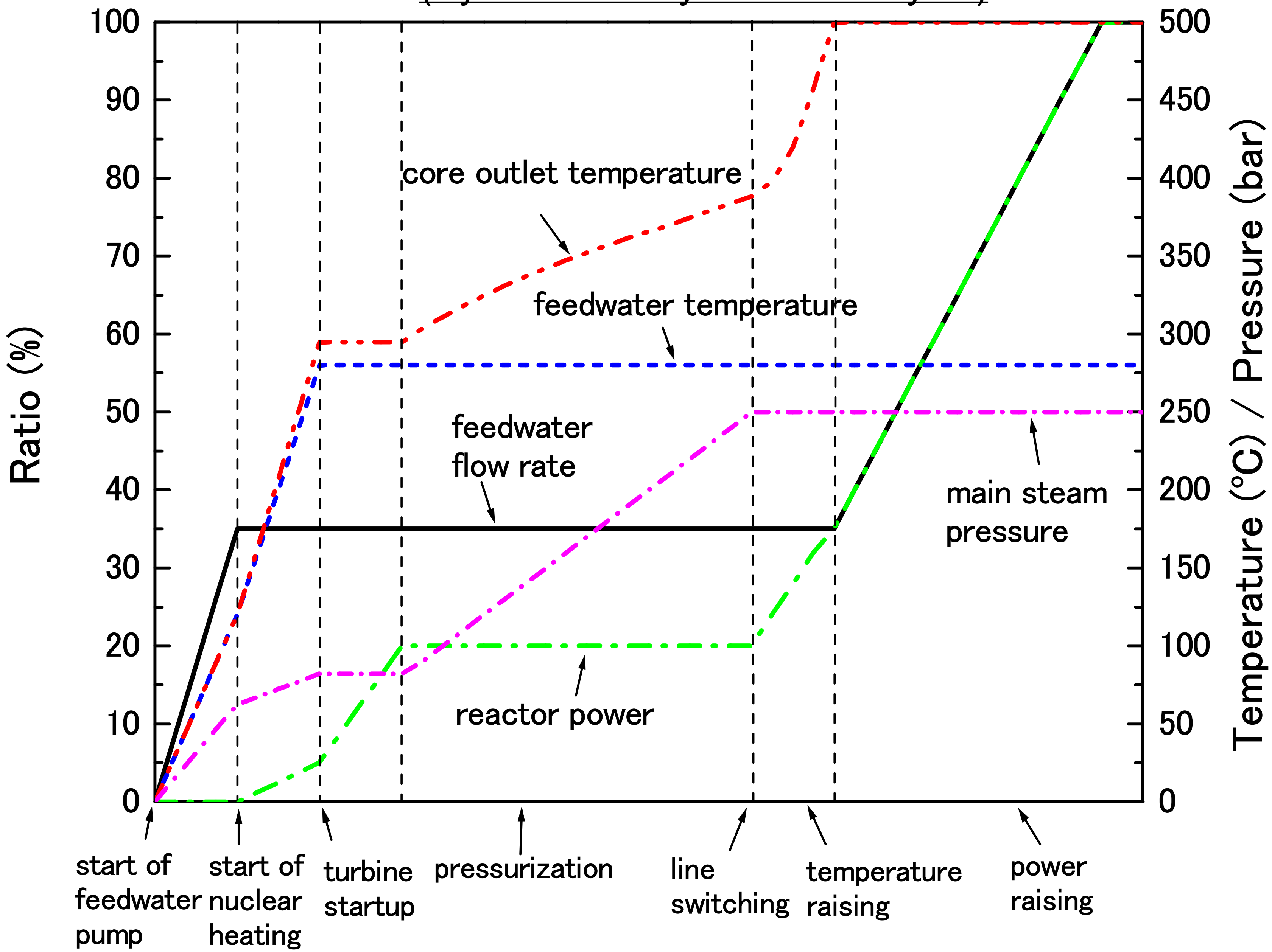
- Sliding pressure startup system (nuclear heating starts at subcritical pressure)
- Clad temperature increase in pressurization phase is due to BT
- Power / flow region is limited by CHF
- CHF may be increased by grid spacers

Temperature Raising Phase



Sliding Pressure Startup Curve

(By Thermal-Hydraulic Analysis)



Component weights required for startup

	Constant Pressure Startup	Sliding Pressure Startup		
	Flash tank (bypass)	Separator (main line)	Separator (bypass)	Separator (bypass)
Flow rate	25%	100%	35%	35%
Design pressure (MPa)	7.6	27.5	25	25
Design temperature (^{o}C)	291	500	400	400
Material	SBV2	SCMV4	SBV2	SCMV4
Shell length/thickness(m)	4.0 / 0.1	3.9 / 0.26	3.9 / 0.12	3.9 / 0.19
Inner diameter (m)	3.4	1.283	1.08	1.56
Cross-sectional area (m^2)	9.08	1.293	0.91	1.91
No. of separators needed	1	4	2	1
Unit weight (kg)	52300	40500	15750	37600
Total weight (kg)	**52300**	**162000**	**31500**	**37600**

Stability

Linear Stability Analysis

Frequency Domain Linear Stability Analysis

Write governing equations (core thermal-hydraulics, neutron kinetics, fuel dynamics, ex-core systems)

↓

Linearize governing equations by perturbation

↓

Perform Laplace transform

↓

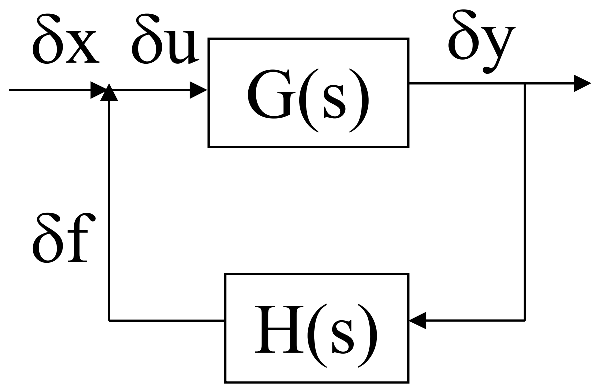
Obtain overall system transfer functions from open loop transfer functions

↓

Determine the roots of characteristic equation: (1+G(s) H(s) = 0)

↓

Calculate decay ratio from the dominant pole

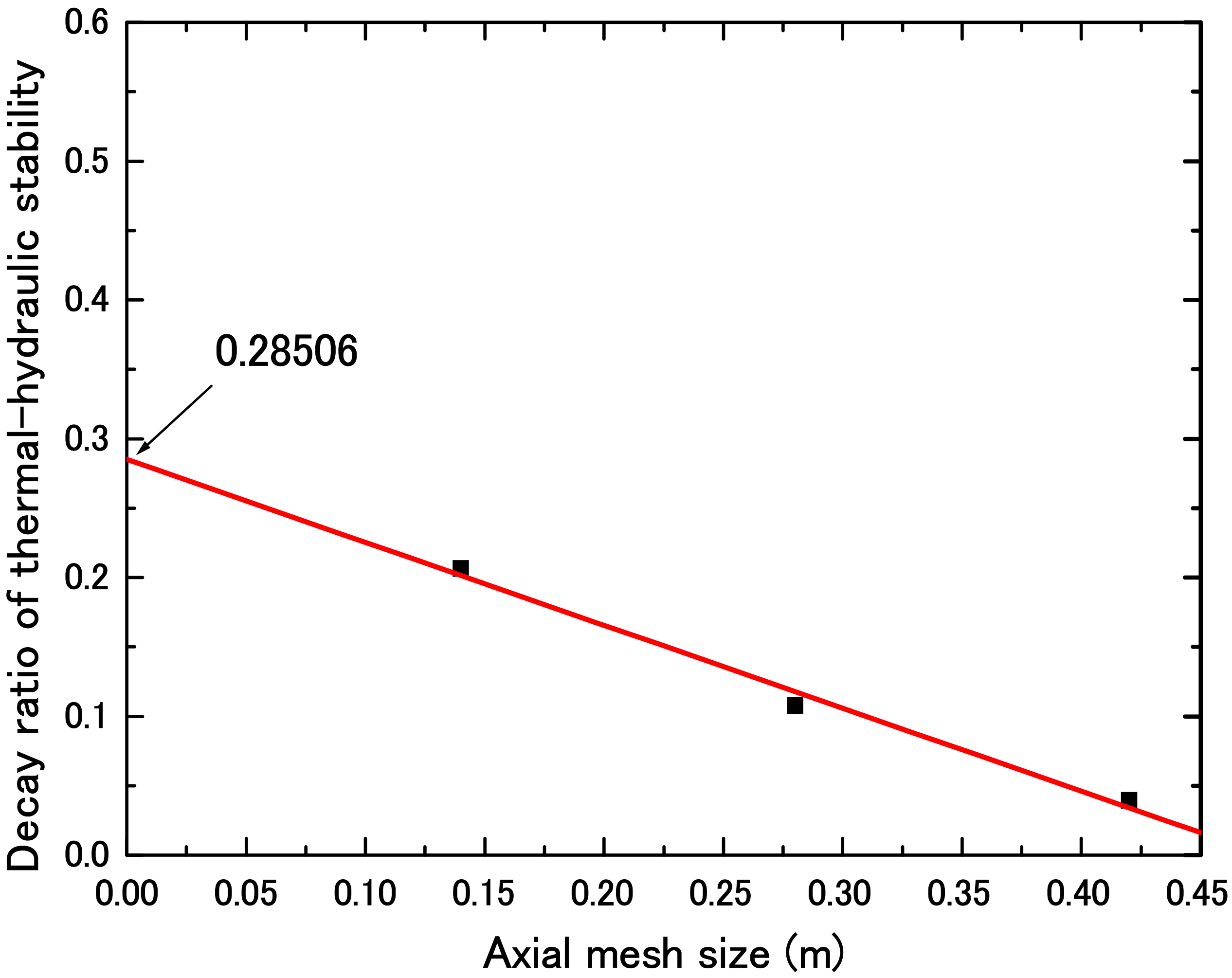


$$\frac{\delta y}{\delta x} = \frac{G(s)}{1 + G(s)H(s)}$$

Dependence of decay ratio on the axial mesh size

• Decay ratio generally increases as the axial mesh size decreases.

• Thus, the decay ratio is determined by extrapolating to the zero mesh size.



• Pressure = 8 MPa

• Power = 20%

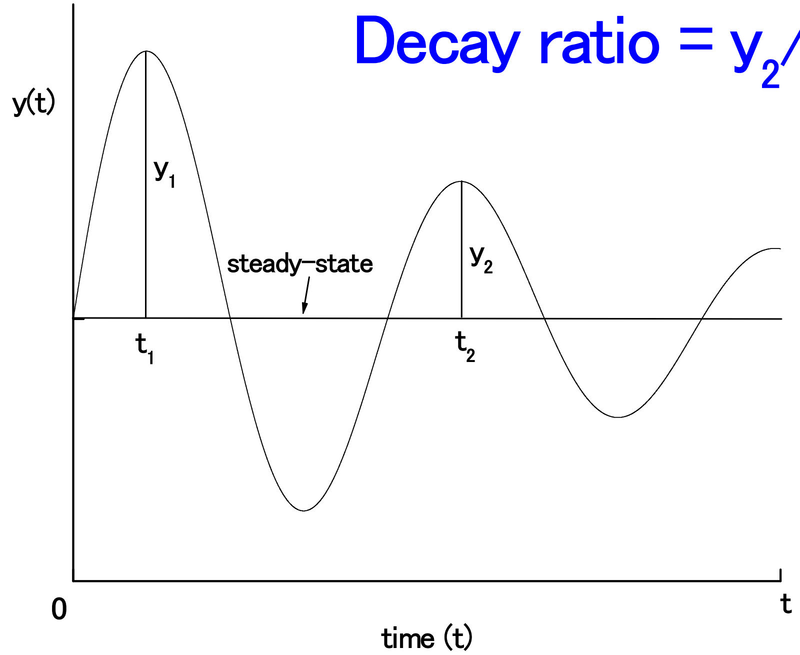
• Flow rate = 35%

• Orifice pressure drop coefficient = 30

Stability Criteria

The same stability criteria for BWR are used for Super LWR.

Decay ratio = y_2/y_1



	Normal operating conditions	All operating conditions
Thermal-hydraulic stability	Decay ratio $\leq$ 0.5 (damping ratio $\geq$ 0.11)	Decay ratio < 1.0 (damping ratio > 0)
Coupled neutronic thermal-hydraulic stability	Decay ratio $\leq$ 0.25 (damping ratio $\geq$ 0.22)	Decay ratio < 1.0 (damping ratio > 0)

Linear Stability Analysis Code (Supercritical pressure)

- Neutron kinetics model
- Fuel rod heat transfer model
- Water rod heat transfer model
- Fuel channel thermal-hydraulic model
- Water rod thermal-hydraulic model
- Ex-core circulation model

Fuel Channel Thermal-hydraulics Model (subcritical pressure)

- One-dimensional single-channel model
- Homogeneous equilibrium two-phase mixture is assumed.
- Phasic slip and subcooled boiling are not considered.
- Constant inlet subcooling is assumed.
- Constant system pressure and constant phasic properties are assumed.

$$\frac{\partial \rho}{\partial t}+\frac{\partial(\rho u)}{\partial z}=0$$

$$\frac{\partial(\rho h)}{\partial t}+\frac{\partial(\rho u h)}{\partial z}=\frac{P_e}{A}q''-\frac{N_f}{N_w A_w}Q_w$$

$$-\frac{\partial P}{\partial z}=\frac{\partial(\rho u)}{\partial t}+\frac{\partial(\rho u^2)}{\partial z}+\rho g\cos\theta+\frac{2f}{D_h}\rho u^2$$

$$f=0.0791\times \mathrm{Re}^{-0.25} \quad \text{(Blasius equation)}$$

$$\rho=\rho(P,h)$$

$$f_{SP}=0.079\left(\frac{GD_h}{\mu}\right)^{-1/4}=f_{fo}$$

$$f_{TP}=0.079\left(\frac{GD_h}{\bar{\mu}}\right)^{-1/4}=f_{fo}\left[1+x\left(\frac{\mu_{fg}}{\mu_g}\right)\right]^{-1/4}$$

$$\frac{1}{\bar{\mu}}=\frac{x}{\mu_g}+\frac{(1-x)}{\mu_f}$$

(Mc Adams et al.)

Fuel Rod Heat Transfer Model

- Lumped parameter model
- one-dimensional radial heat transfer equations

$$\frac{\partial}{\partial t}\left(\rho_f C_p T_f\right) = \frac{1}{r}\frac{\partial}{\partial r}\left(r k_f \frac{\partial T_f}{\partial r}\right) + q'''$$

$$T_f^{ave} - T_s = \left(\frac{r_f + t_c}{r_f}\right)\left[\frac{r_f}{4k_f} + \frac{1}{h_g} + \frac{t_c}{k_c}\right] q''$$

$$q''(r_c, t) = h_c\left(T_{r_c} - T\right)$$

Single-phase heat transfer	Dittus- Boelter equation
Nucleate boiling	Thom equation
Forced convective heat transfer	Schrock-Grossman correlation
Film boiling	Groeneveld correlation
Supercritical-pressure heat transfer	Oka-Koshizuka correlations

Water Rod Heat Transfer Model

- axial heat transfer neglected
- heat conduction through the water rod wall not considered
- Since water rod wall temperature is less than saturation temperature, boiling does not happen on the water rod outer surface, as well as on the water rod inner surface.
- single phase heat transfer is assumed for water rod heat transfer
- Dittus-Boelter equation is used to calculate heat transfer coefficient.

$$T - T_w = \frac{N_f}{N_w} Q_w \left[\frac{1}{\pi D_w h_{s1}} + \frac{1}{\pi \left(D_w - 2t_{ws} \right) h_{s2}} \right]$$

Heat Transfer at Supercritical Pressure

$$\mathrm{Nu} = 0.015\ \mathrm{Re}^{0.85}\ \mathrm{Pr}^{C}$$

$$C = 0.69 - \frac{81000}{q''_{\mathrm{det}}} + f_c \times q''$$

For $0 < H \le 1.5$ MJ/kg

$$f_c = 2.9 \times 10^{-8} + \frac{0.11}{q''_{\mathrm{det}}}$$

For 1.5 MJ/kg $< H \le 3.3$ MJ/kg

$$f_c = -8.7 \times 10^{-8} - \frac{0.65}{q''_{\mathrm{det}}}$$

For 3.3 MJ/kg $< H \le 4.0$ MJ/kg

$$f_c = -9.7 \times 10^{-7} + \frac{1.30}{q''_{\mathrm{det}}}$$

q''_{det} is determined by using Yamagata's relation.

$$q''_{\mathrm{det}} = 200\ G^{1.2}$$

Parameter range

G : 1000 ~ 1750 kg/m^{2}s

q" : 0 ~ 1800 kW/m^2

h : 0.1 ~ 3.3 MJ/kg
(T=20 ~ 550°C)

OK correlation (Oka-Koshizuka-Kitoh): for plant dynamics calculation such as safety, stability, control etc.

Heat transfer correlations at subcritical pressure

Pre-CHF Heat Transfer Correlations

Nucleate boiling (Thom correlation)

$$h = \frac{1}{dT}\left(\frac{\Delta T_{sat} e^{P/1260}}{0.072}\right)^2$$

Pressure:	5.2 – 14.0 MPa
Mass flux:	1000 – 3800 kg/m^{2}s
Heat flux:	0 ~ 1600 kW/m^2

Forced convective vaporization (Schrock-Grossman Correlation)

$$h = (2.50)(0.023)\left(\frac{k_f}{D_e}\right)\left(\Pr_f\right)^{0.4}\left[\frac{GD_e(1-X)}{\mu_f}\right]^{0.8}\left(\frac{1}{X_{tt}}\right)^{0.75}$$

$$\frac{1}{X_{tt}} = \left(\frac{X}{1-X}\right)^{0.9}\left(\frac{\rho_f}{\rho_g}\right)^{0.5}\left(\frac{\mu_g}{\mu_f}\right)^{0.1}$$

Pressure:	0.3 – 3.5 MPa
Mass flux:	240 – 4500 kg/m^{2}s
Heat flux:	190 ~ 4600 kW/m^2

Forced convection in subcooled liquid and superheated vapor (Dittus-Boelter Correlation)

$$h = 0.023\left(\frac{k_f}{D_e}\right)\left(\Pr_f\right)^{0.4}\left(\frac{GD_e}{\mu_f}\right)^{0.8}$$

$$h = 0.023\left(\frac{k_g}{D_e}\right)\left(\Pr_g\right)^{0.4}\left(\frac{GD_e}{\mu_g}\right)^{0.8}$$

Heat transfer correlations at subcritical pressure

Post-CHF Heat Transfer Correlations

Transition Boiling (McDonough, Milich, and King Correlation)

$$h = \frac{q_{CHF} - const\,(T_w - T_{w,CHF})}{dT}$$

Pressure:	5.5 – 14.0 MPa
Mass flux:	270 – 1900 kg/m^{2}s
Wall temperature	~ 555 ^{o}C

Stable Film Boiling (Groeneveld Correlation)

$$h = 0.00327\left(\frac{k_g}{D_e}\right)\left(\Pr_{v,w}\right)^{1.32}\left[\left(\frac{GD_e}{\mu_g}\right)\left(X + \frac{\rho_g}{\rho_f}(1 - X)\right)\right]^{0.901} Y^{-1.50}$$

$$Y = \max\left[1 - 0.1(1 - X)^{0.4}\left(\frac{\rho_f}{\rho_g} - 1\right)^{0.4}, 0.1\right]$$

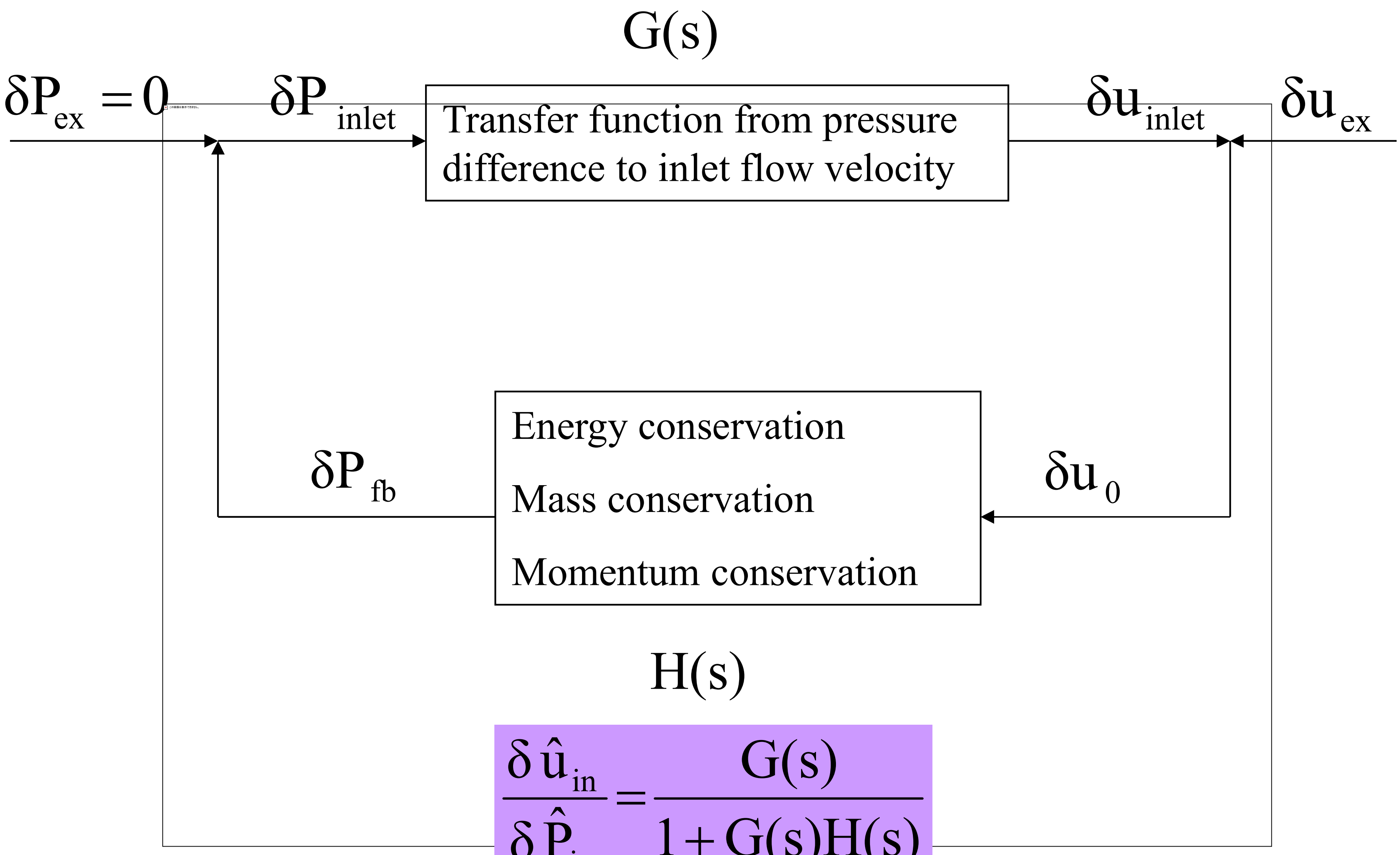
Pressure:	6.8 – 21.5 MPa
Mass flux:	700 – 5300 kg/m^{2}s
Quality:	0.1 – 0.9
Diameter:	1.5 – 25.4 mm

Critical heat flux

- In order to predict dryout, it is important to determine CHF as accurately as possible.
- 1995 CHF lookup table (Groeneveld et al.) is used in this study because of its wider applicable ranges of parameters compared with other CHF correlations.
- The CHF lookup table is based on the CHF data between 0.1 MPa and 20 MPa. The CHF data for near critical pressures are limited.
- The spacer effects are not considered in this study.
- Consideration of spacers will increase CHF and enhance heat transfer.

Thermal-Hydraulic Stability
(Supercritical pressure)

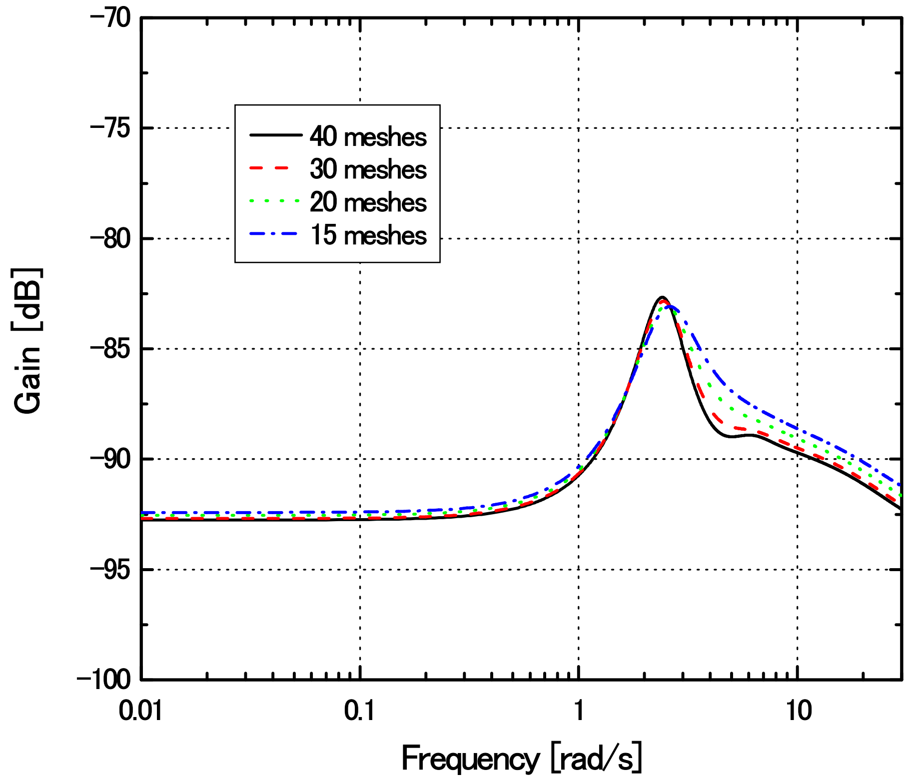
Block diagram for thermal-hydraulic stability



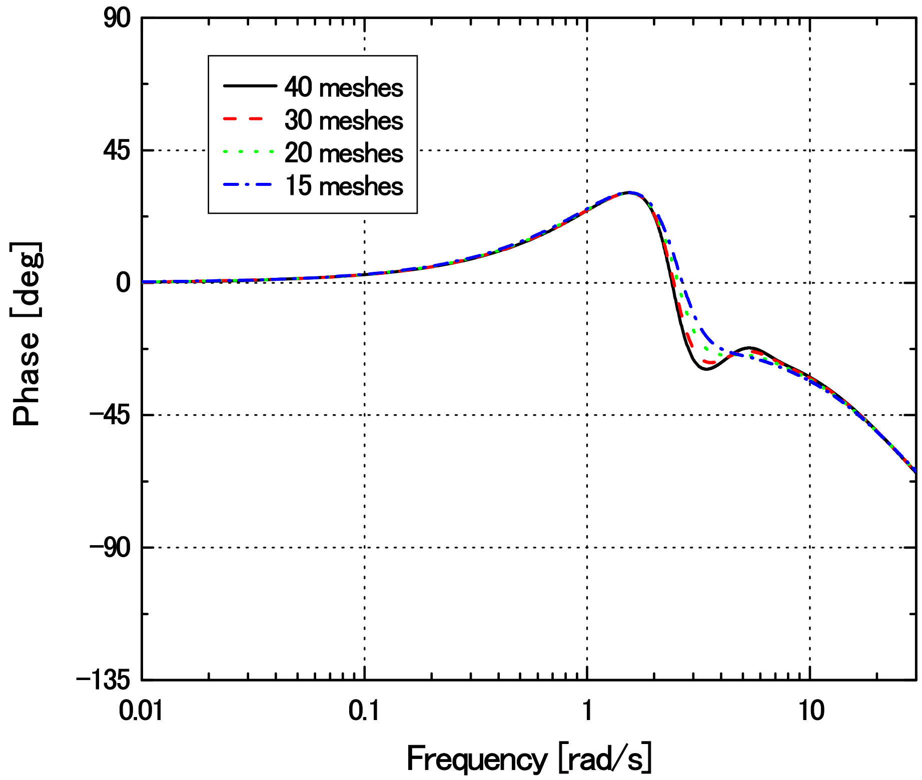
$$\frac{\delta \hat{u}_{in}}{\delta \hat{P}_{in}} = \frac{G(s)}{1+G(s)H(s)}$$

Frequency Response of Thermal-Hydraulic Stability (100% Average power channel)

Gain Response



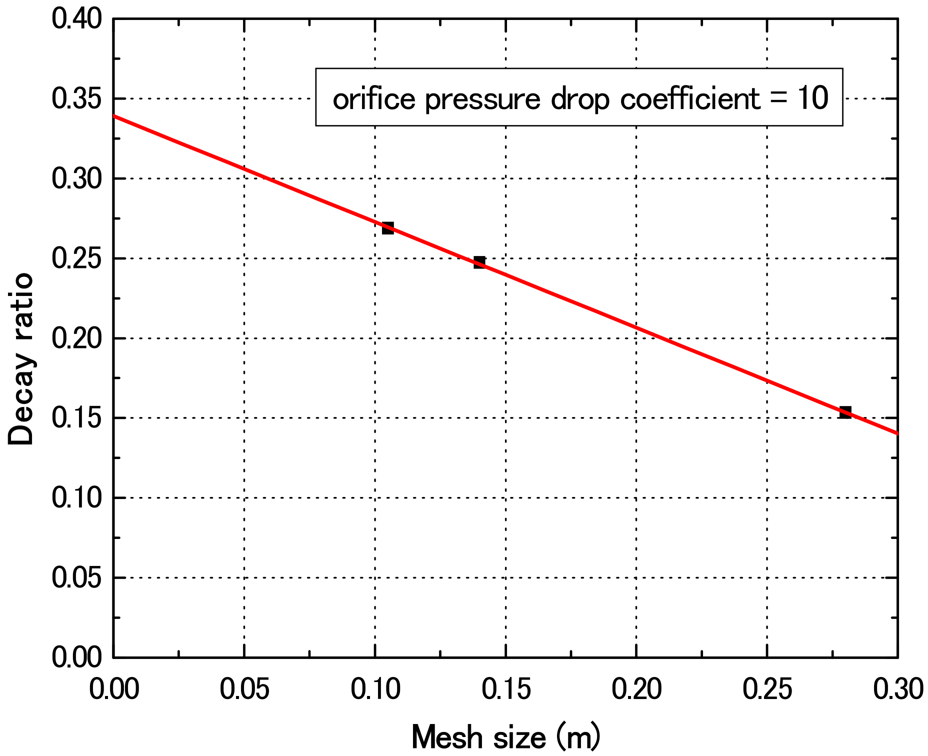
Phase Response



- A resonant peak occurs due to the thermal-hydraulic feedback effect.

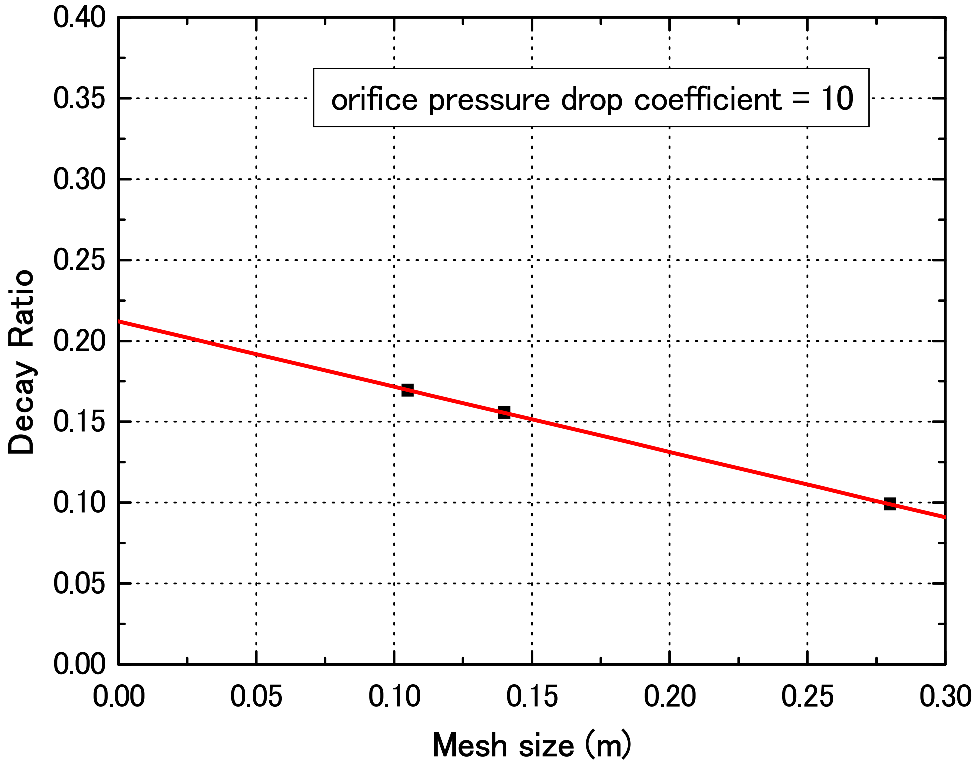
Thermal-Hydraulic Stability : Decay Ratio

Hottest Channel



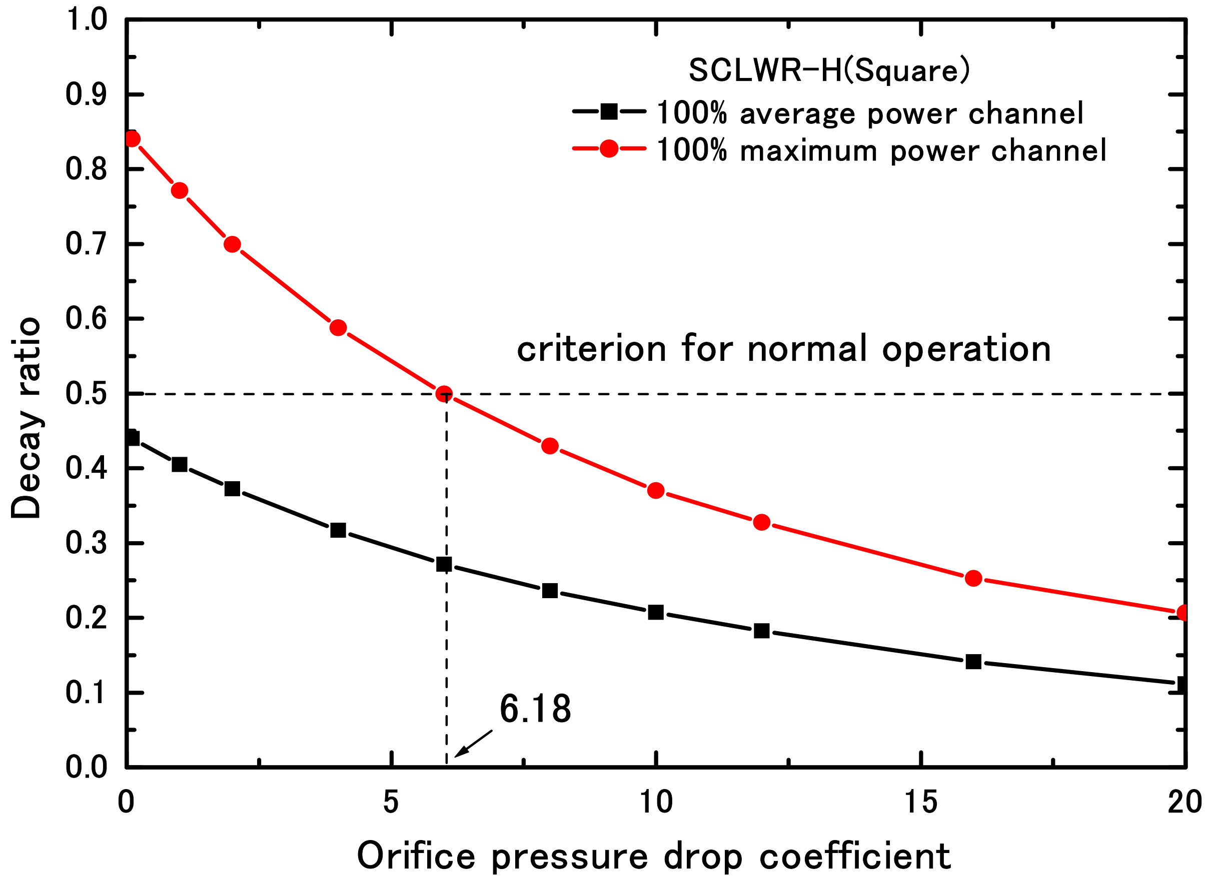
Decay ratio = 0.34

Average Channel



Decay ratio = 0.21

Decay ratios for thermal-hydraulic stability of Super LWR (full power operation)

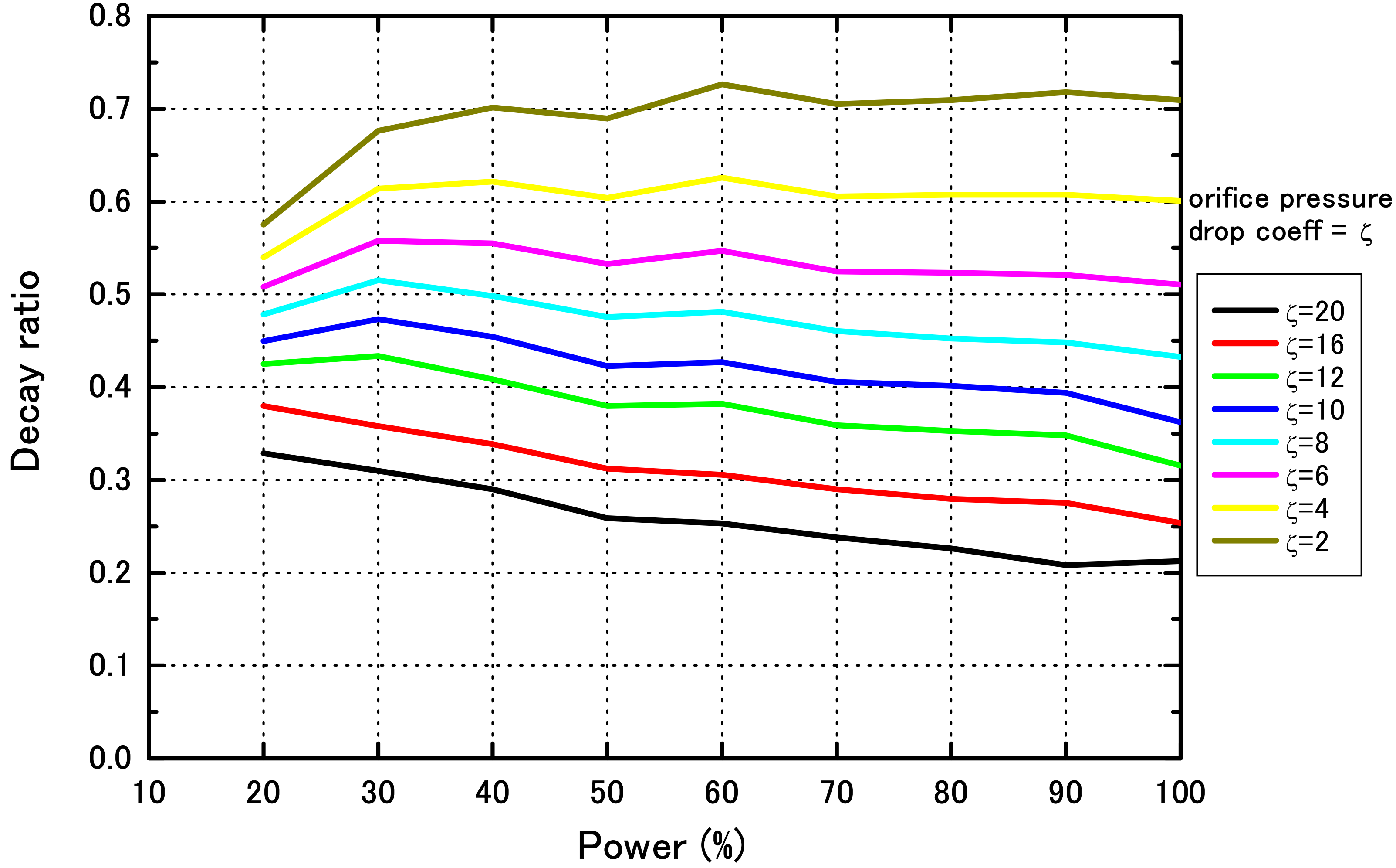


Required orifice pressure drop coefficient = 6.18

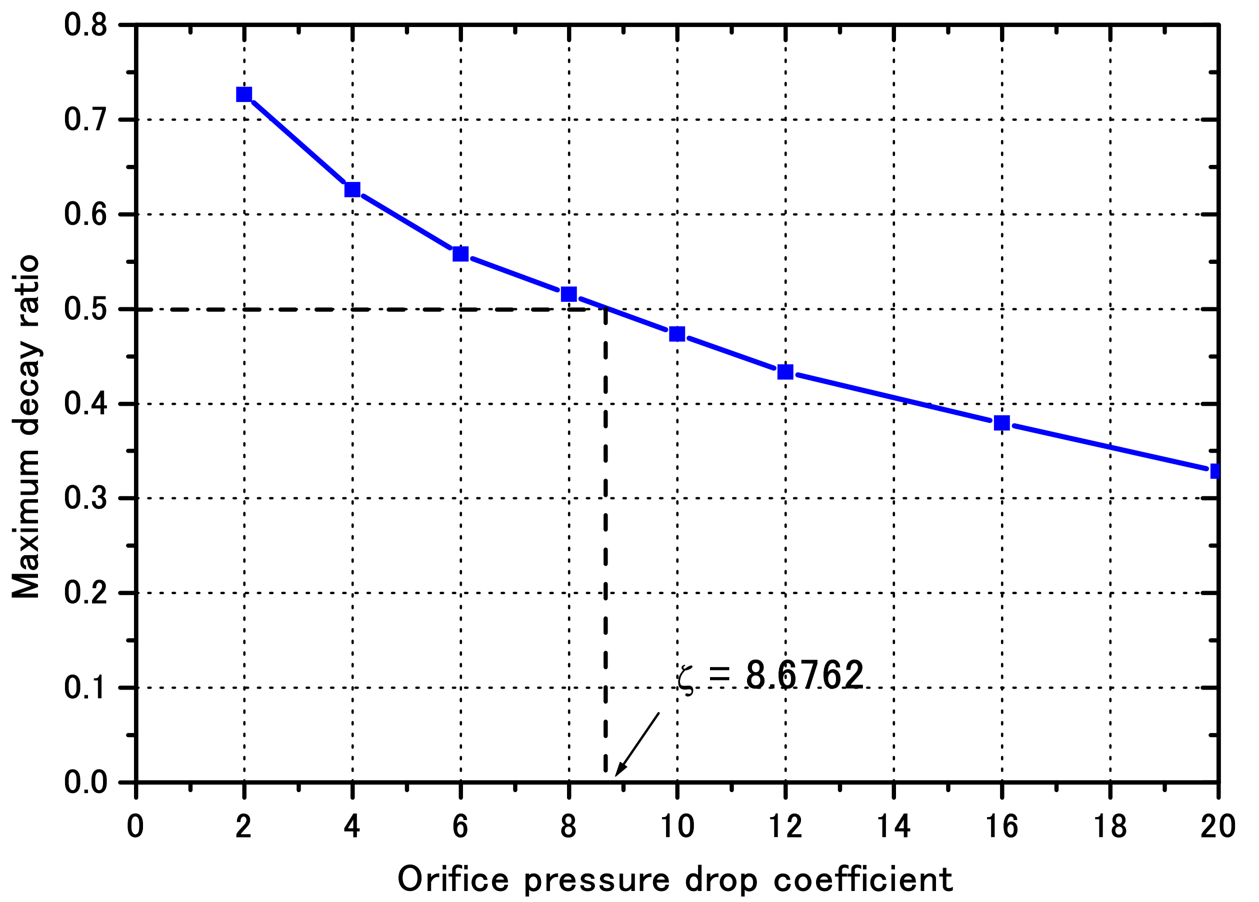
Required orifice pressure drop = 0.0054 MPa

(Core pressure drop = 0.133 MPa)

Decay ratios of thermal-hydraulic stability of Super LWR at partial power operations (maximum power channel)



Relation between maximum decay ratio and orifice pressure drop coefficient for thermal-hydraulic stability (partial power operations)

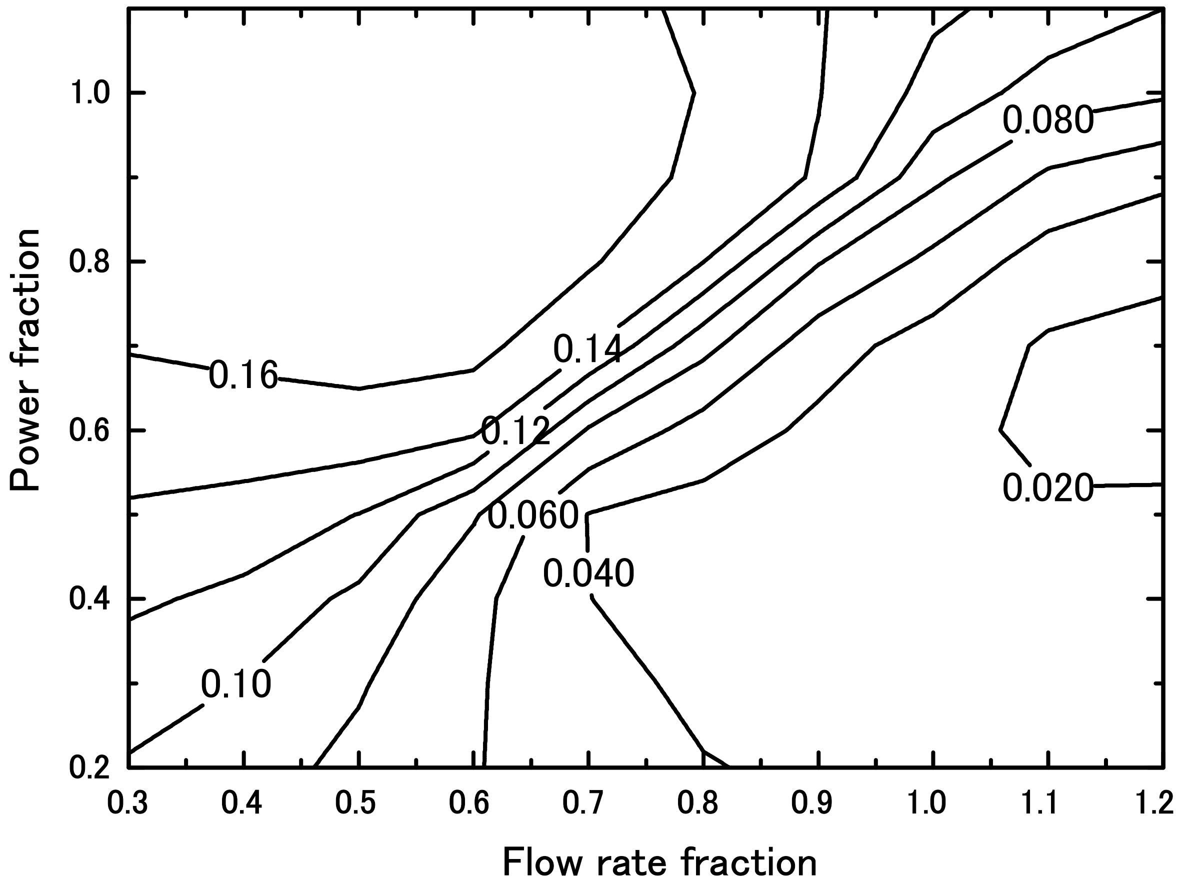


Required orifice pressure drop coefficient = 8.68

Required orifice pressure drop = 0.0075 MPa

Decay Ratio Map for Thermal-hydraulic Stability

Orifice pressure loss coefficient = 20; Axial mesh size = 0.105 m

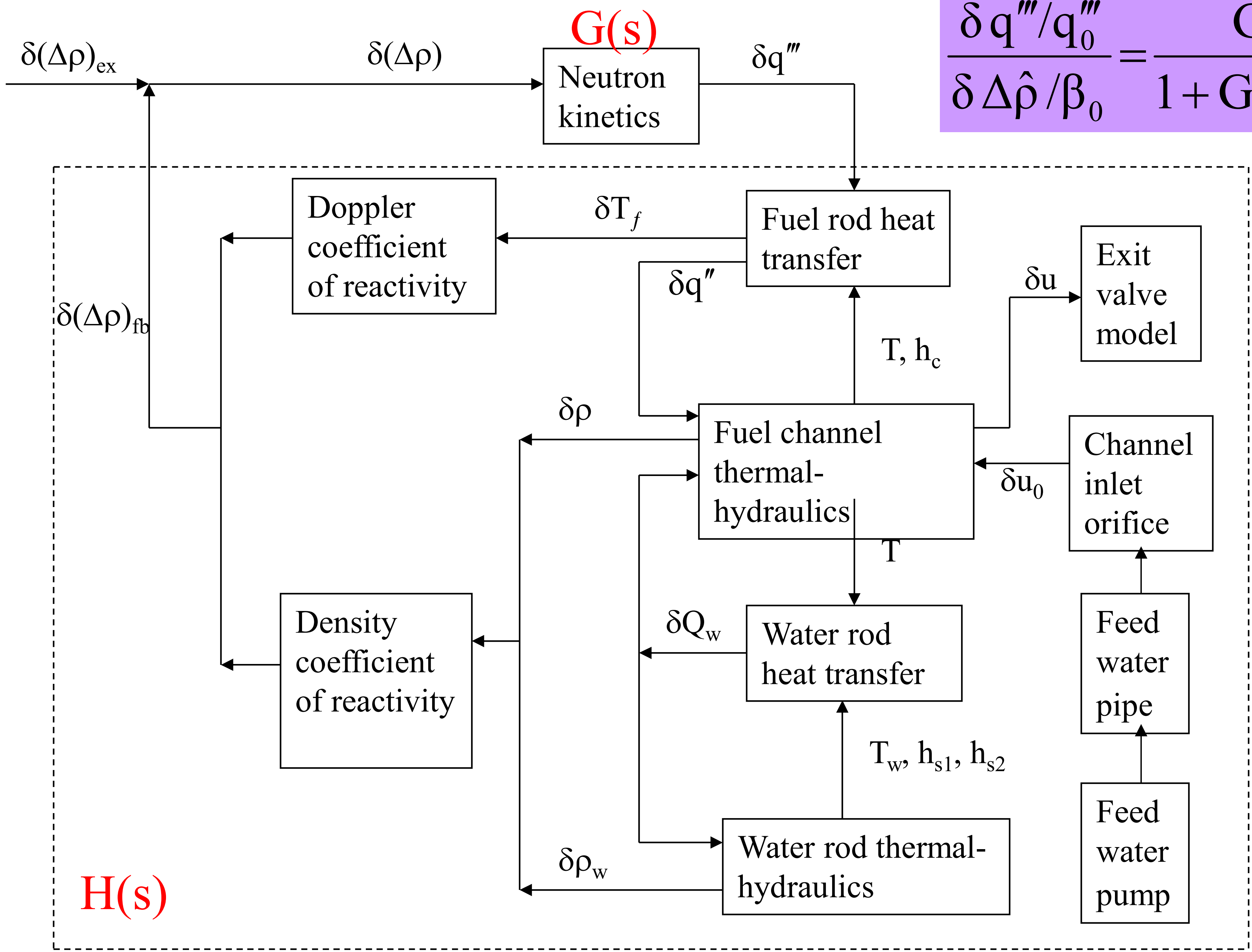


The decay ratio increases as the power to flow rate ratio increases.

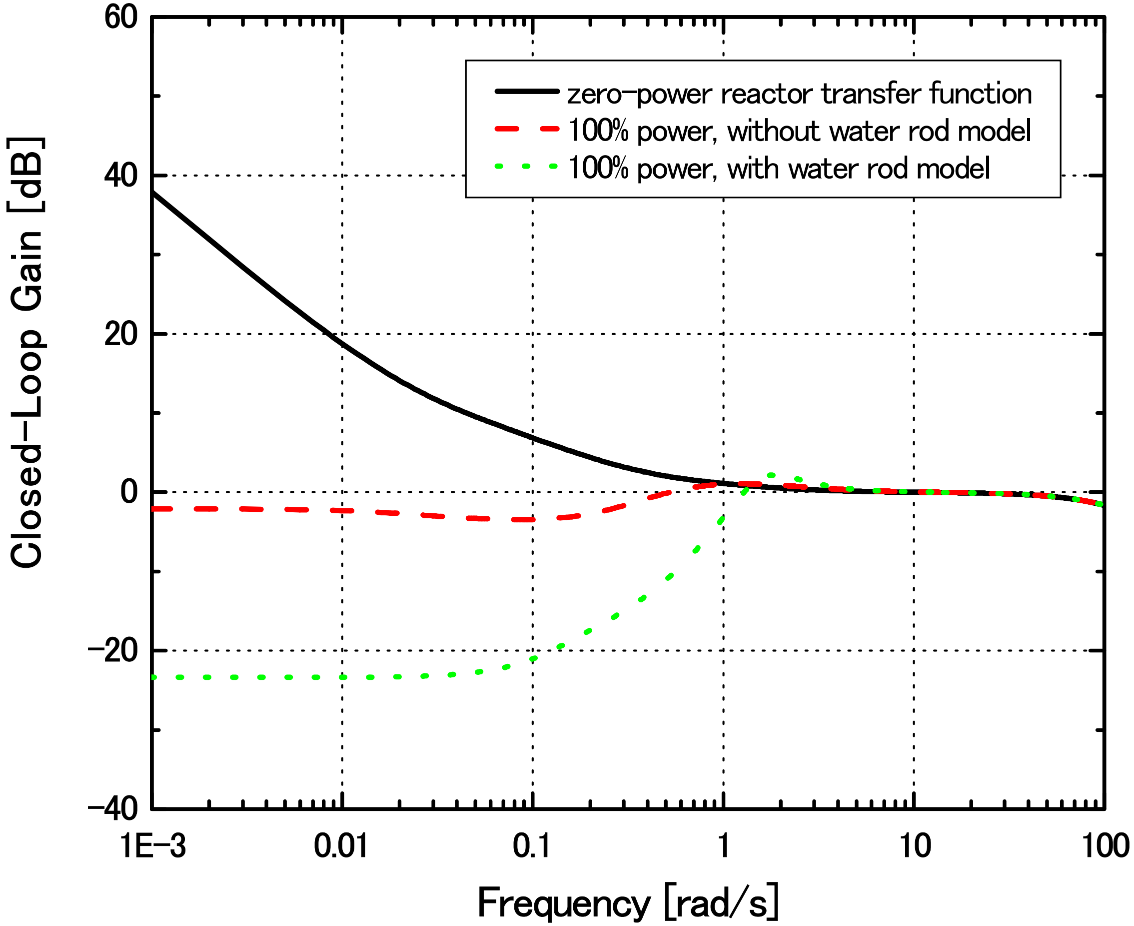
Coupled Neutronic Thermal-Hydraulic Stability (Supercritical pressure)

Block diagram for coupled neutronic thermal–hydraulic stability

$$\frac{\delta q'''/q_0'''}{\delta \Delta\hat{\rho}/\beta_0} = \frac{G(s)}{1+G(s)H(s)}$$



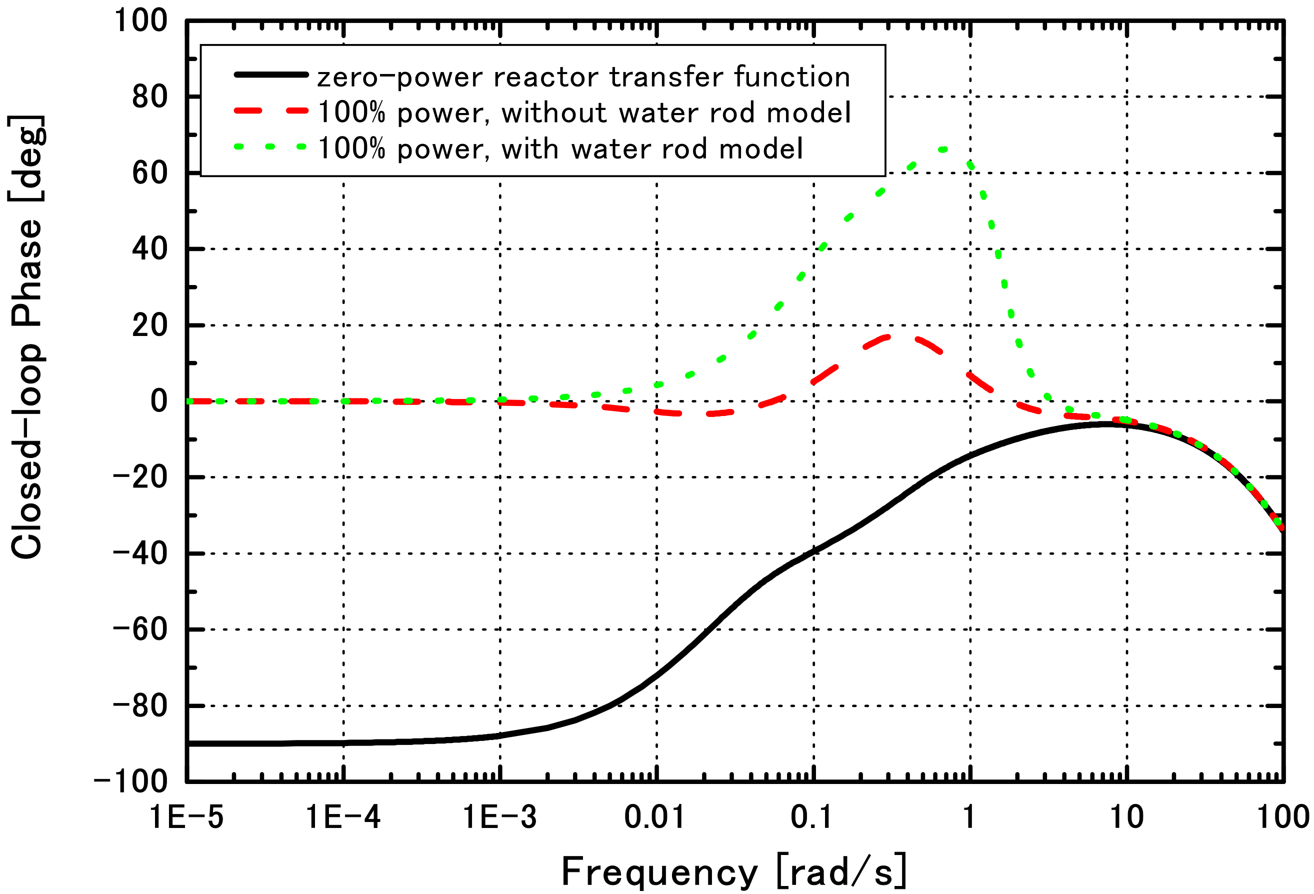
Gain Response of Coupled Neutronic Thermal-Hydraulic Stability (100% Average Power Channel)



The presence of downward flowing water rods

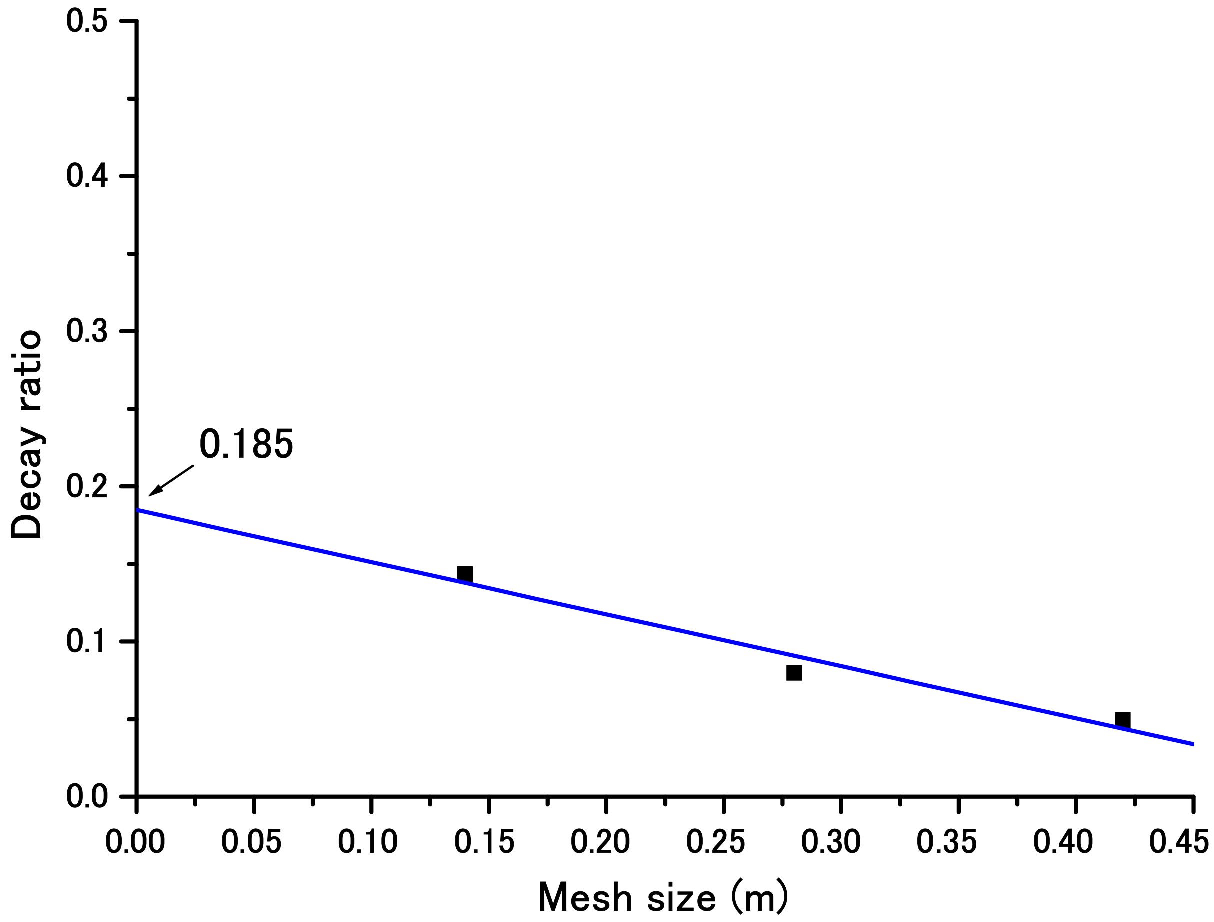
- reduces the density reactivity feedback effect
- increases the resonant peak
- increases the resonant frequency
- thus makes the reactor system less stable

Phase Response of Coupled Neutronic Thermal-Hydraulic Stability (100% Average Power Channel)



- The presence of water rods increases the phase lag of the closed loop transfer function. (due to the time delay in the heat transfer to the water rods)

Coupled neutronic thermal-hydraulic stability of Super LWR for 100% average power channel

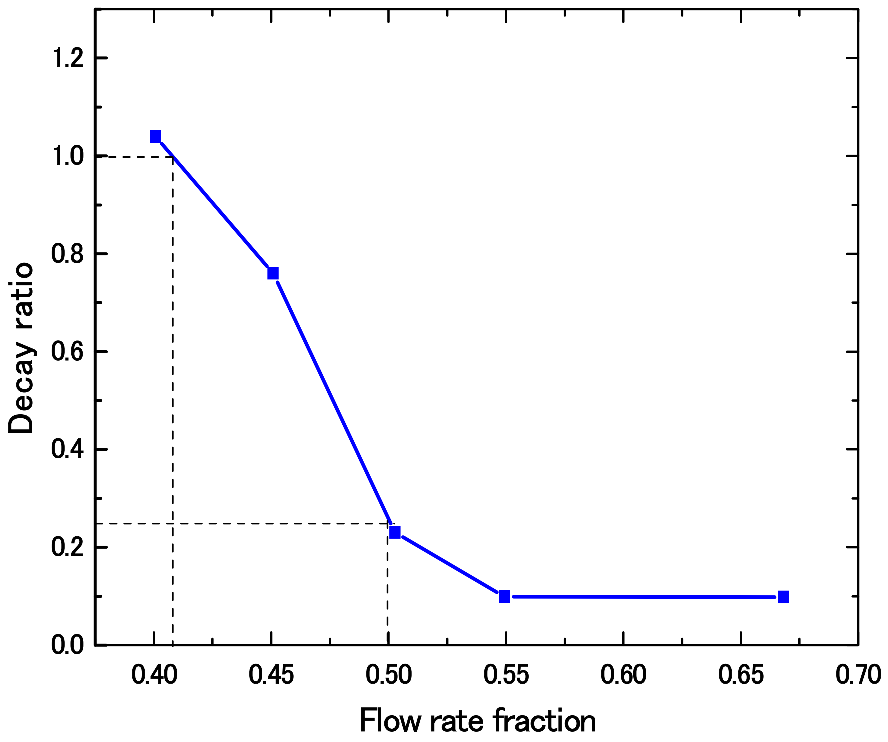


Decay ratio for 100% average power channel = 0.185

Coupled Neutronic Thermal-hydraulic stability

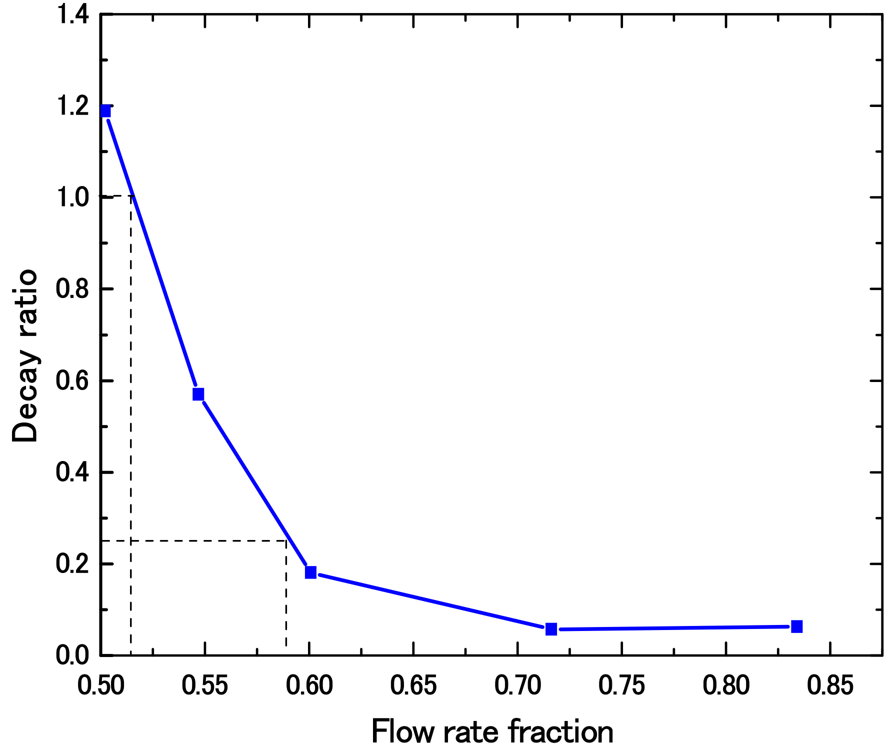
(Partial Power Operations)

40% Core Power



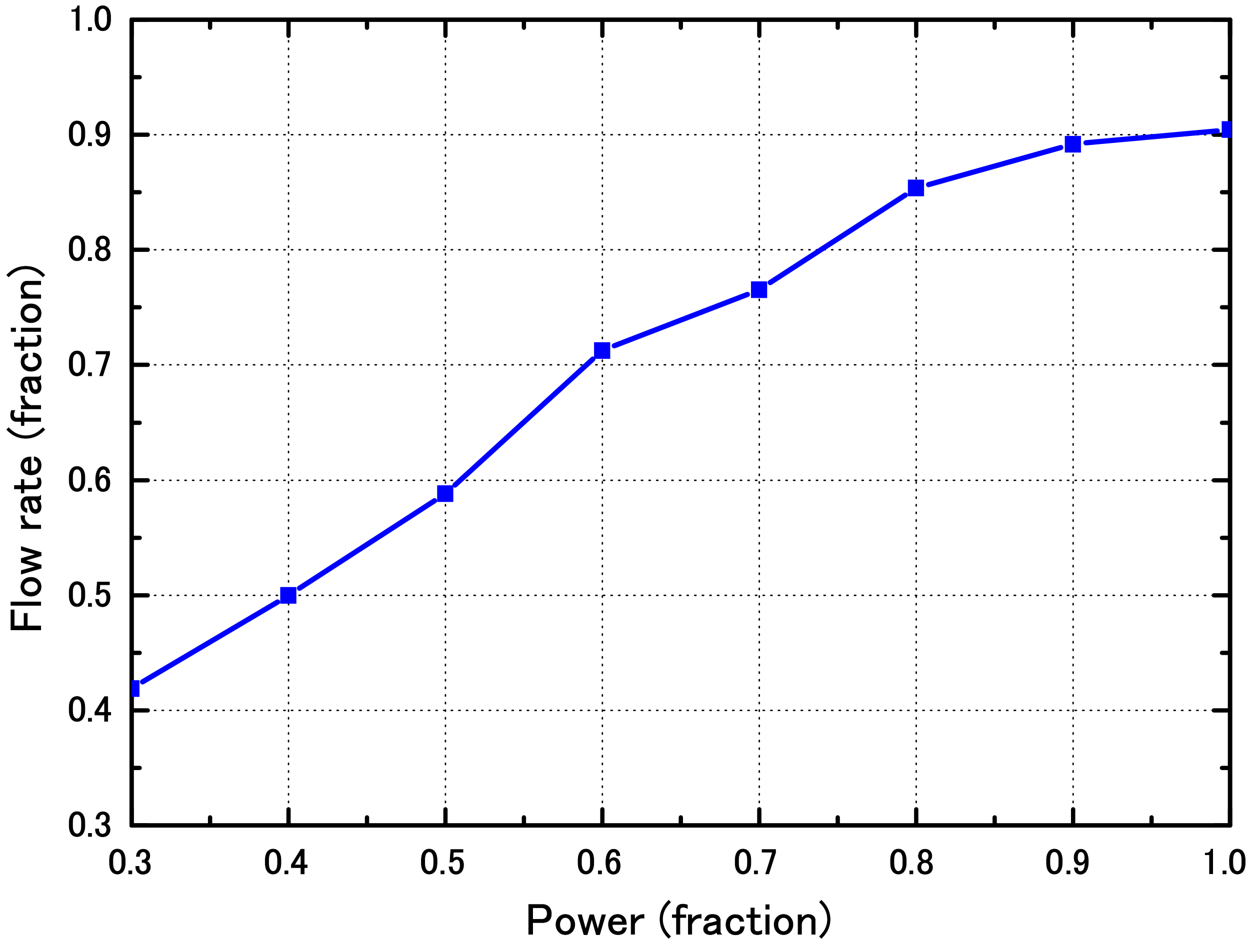
Minimum required flow rate = 50%

50% Core Power

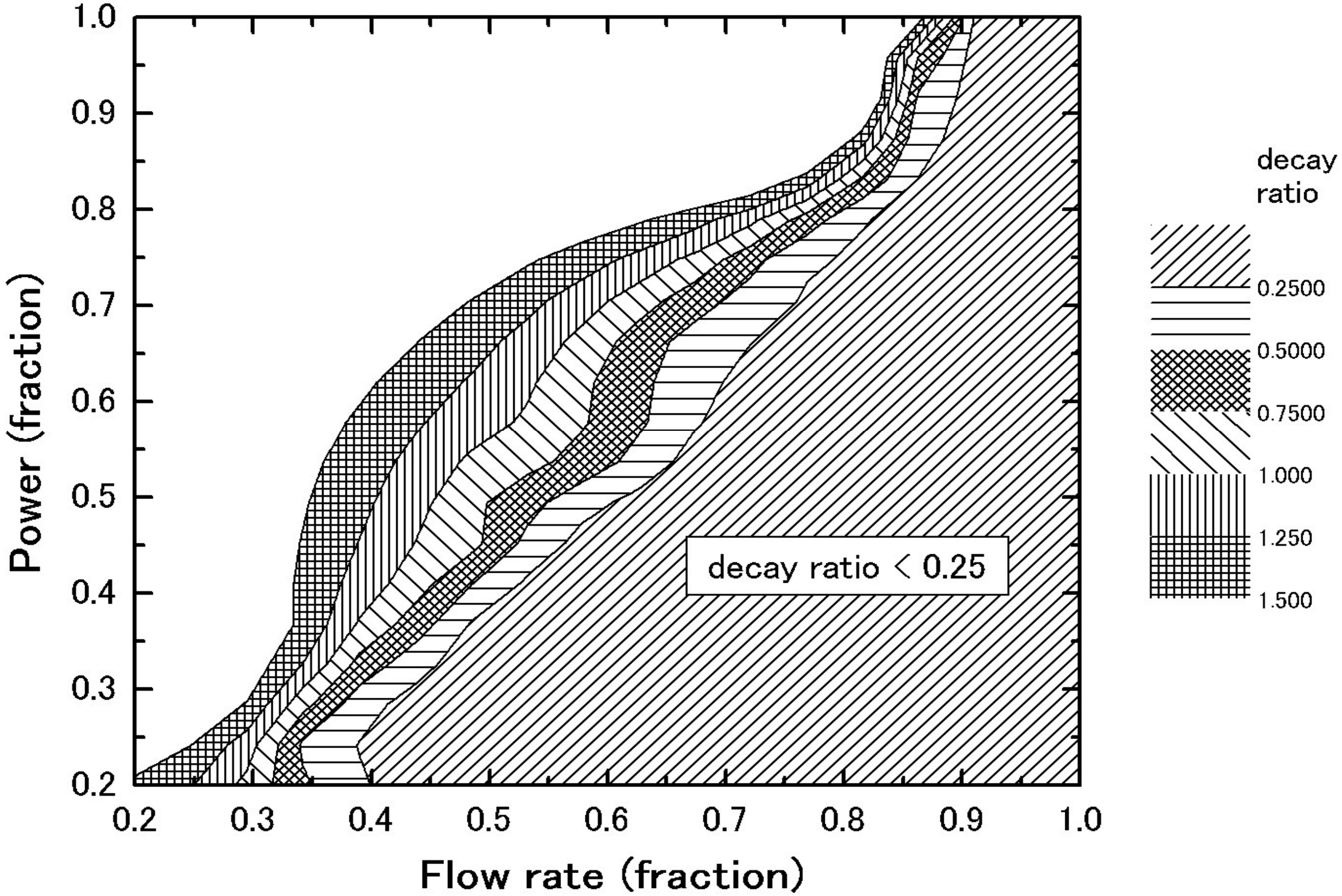


Minimum required flow rate = 58.9%

Flow rate required for coupled neutronic thermal-hydraulic stability at partial power operation



Decay Ratio Map for Coupled Neutronic Thermal-hydraulic Stability



Decay ratio increases with power to flow rate ratio.

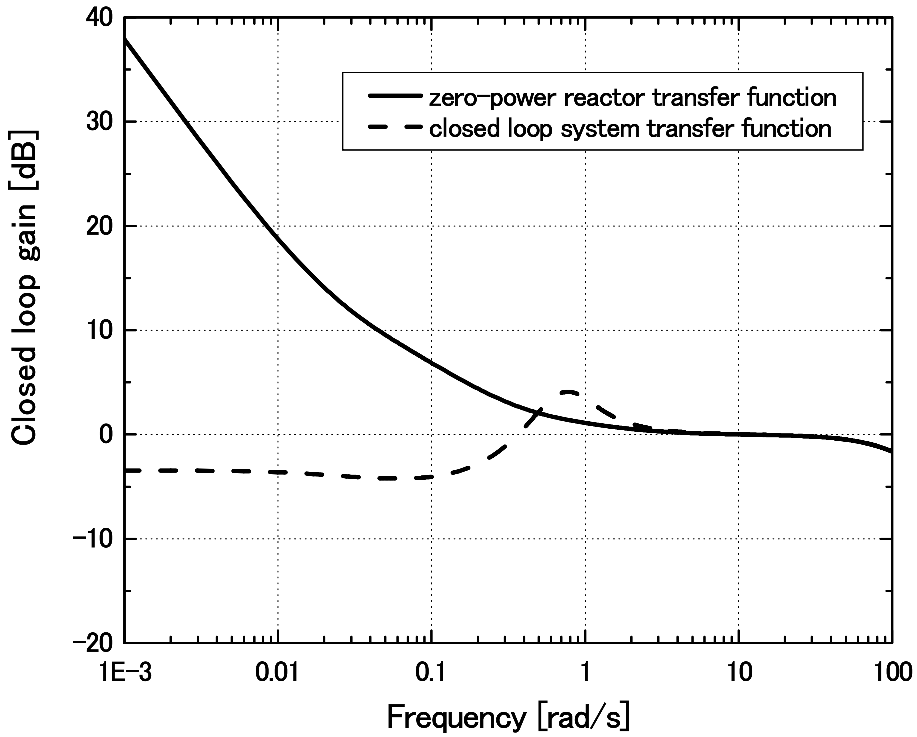
Stability Analysis during Sliding-Pressure Startup

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

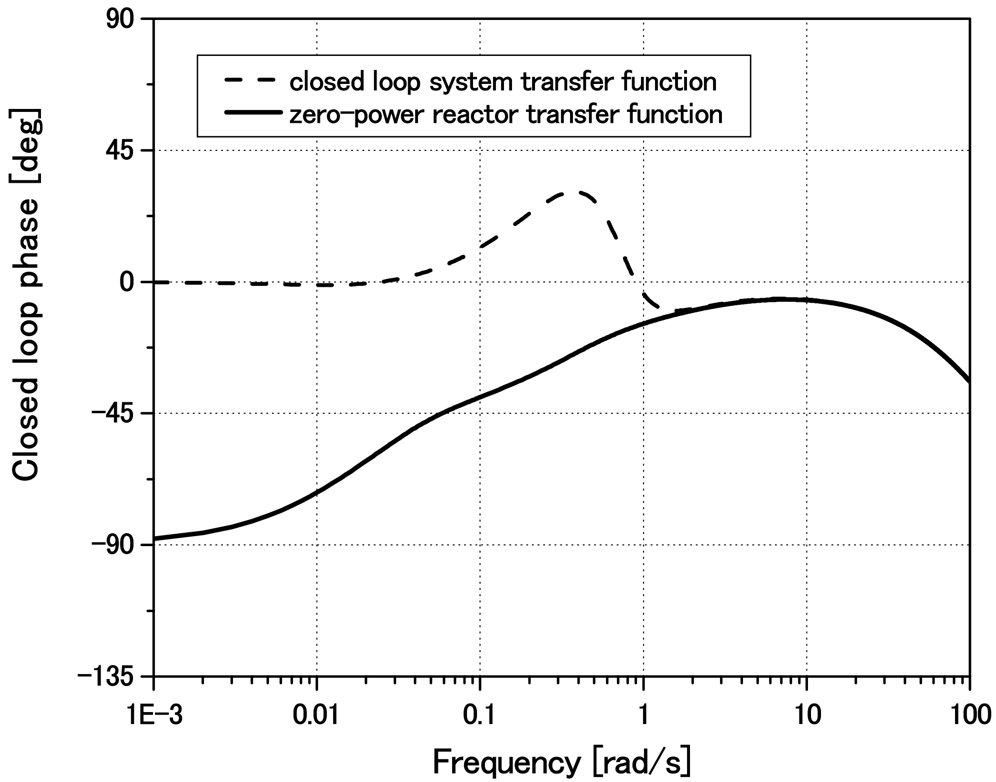
Gain and phase response of coupled neutronic thermal-hydraulic stability of Super LWR (at subcritical pressure)

(8 MPa pressure; 20% power; 35% flow rate)

Gain response

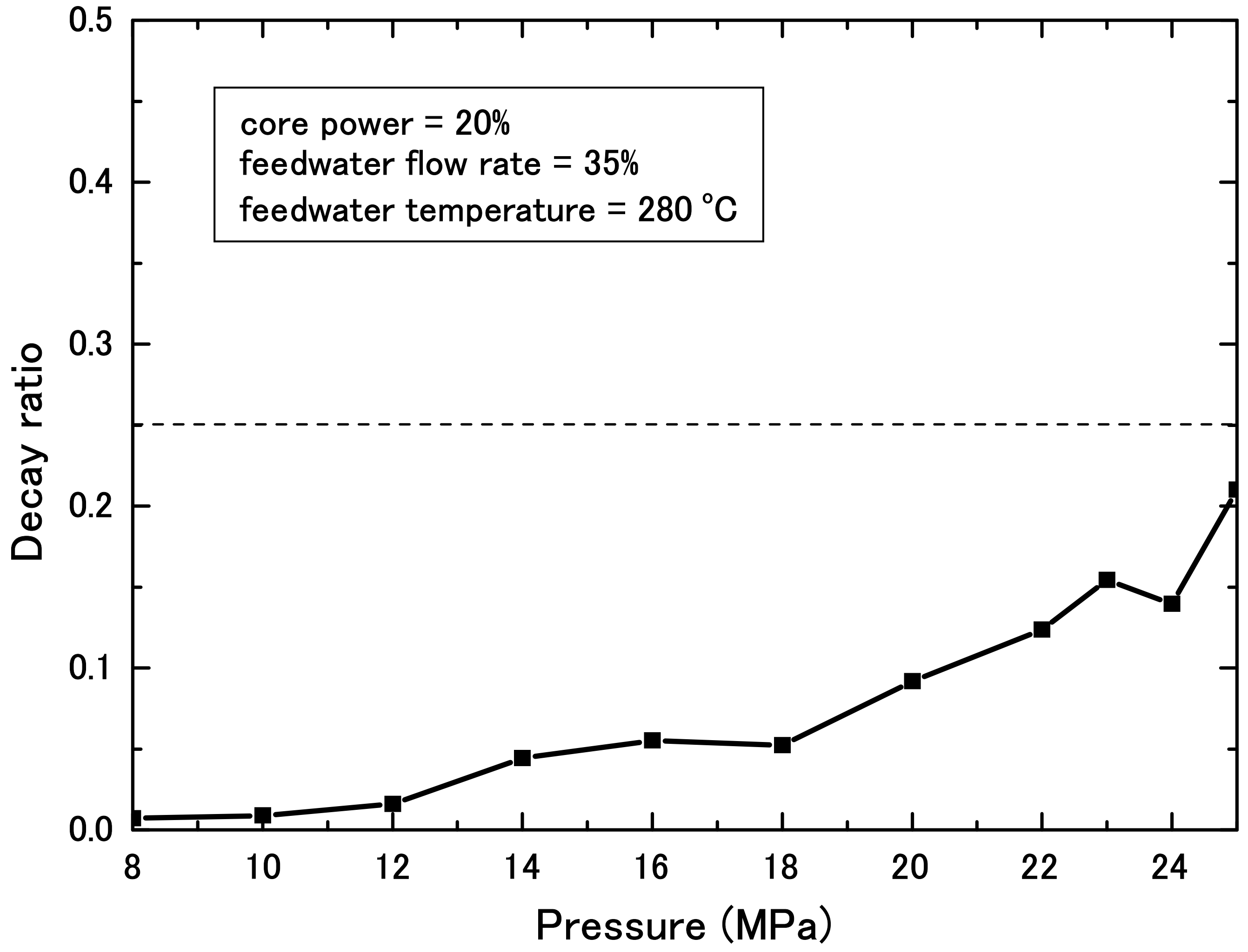


Phase response

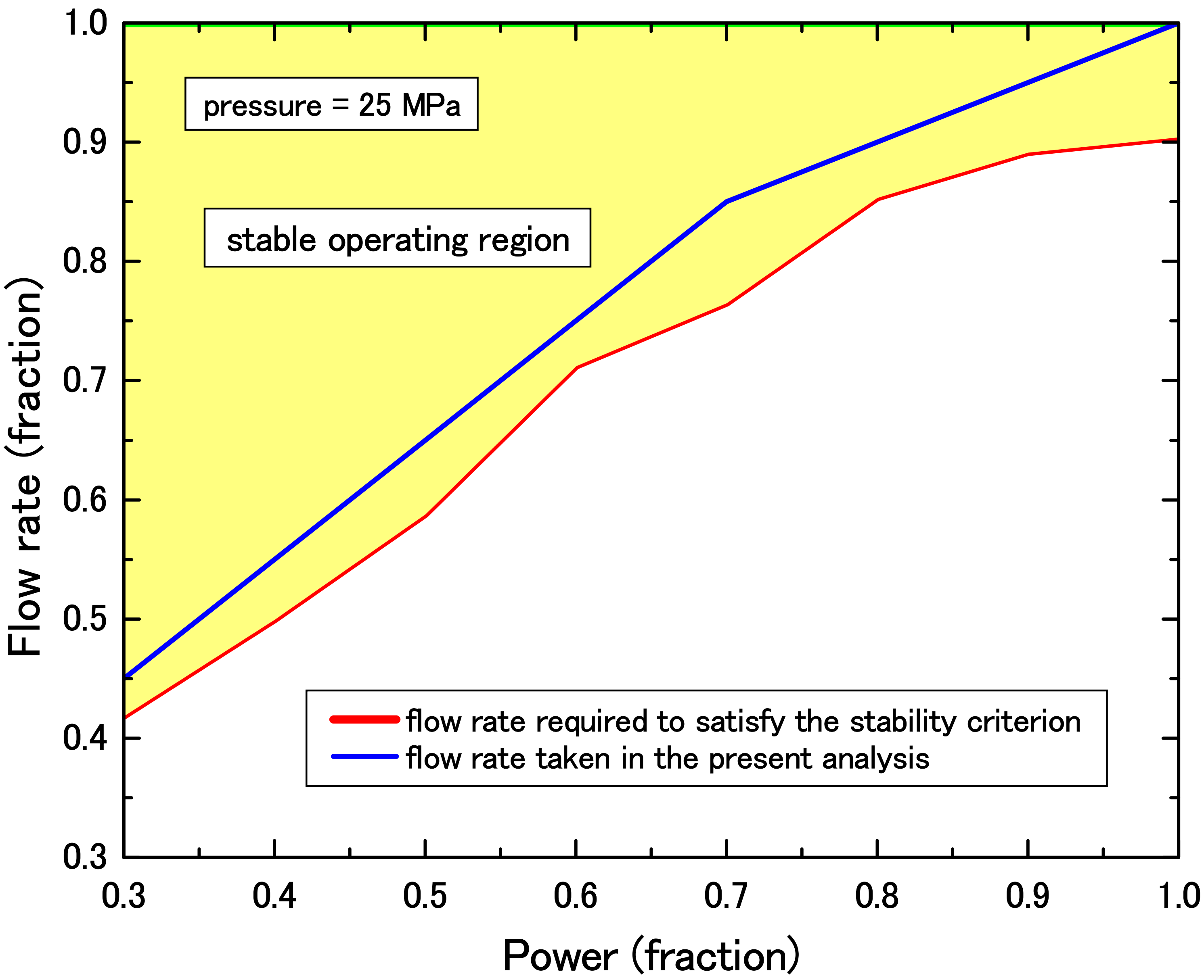


A gain peak occurs around 0.8 rad/s due to neutronic feedback (coolant transit time ~ 3.2 s).

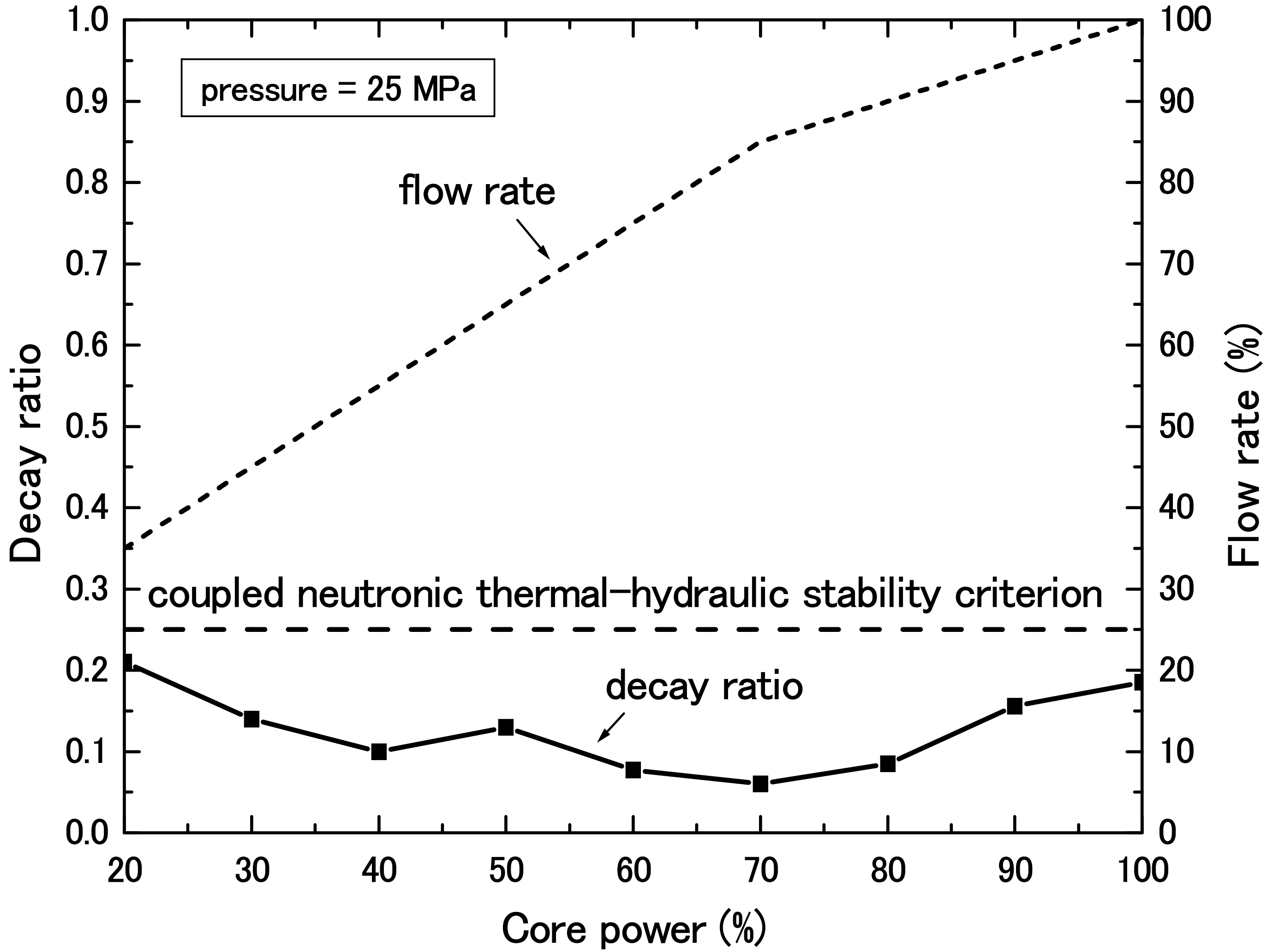
Coupled neutronic thermal-hydraulic stability (Pressurization phase)



Flow rate during power raising phase



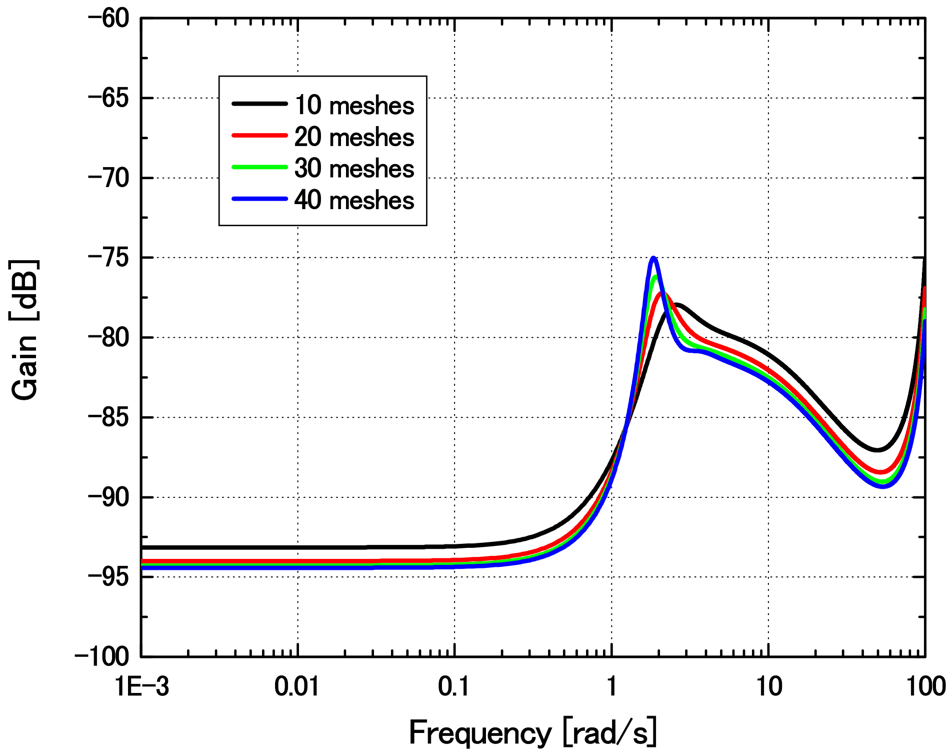
Coupled neutronic thermal-hydraulic stability (Power-raising phase)



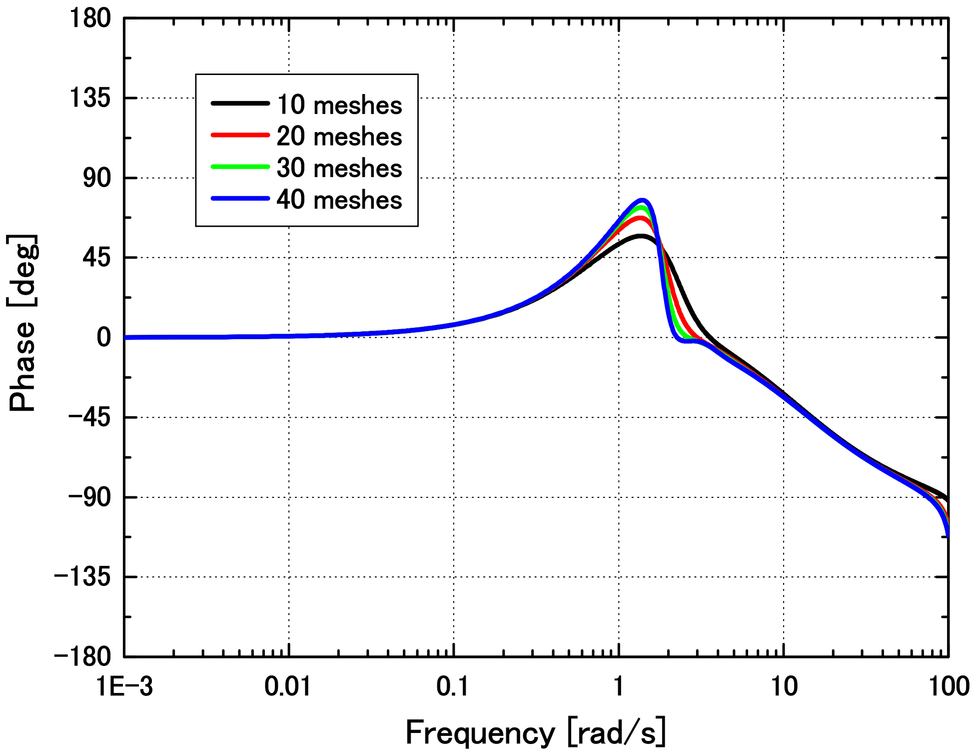
Gain and phase response of thermal-hydraulic stability of Super LWR (at subcritical pressure)

(8 MPa pressure; 20% power; 35% flow rate)

Gain response

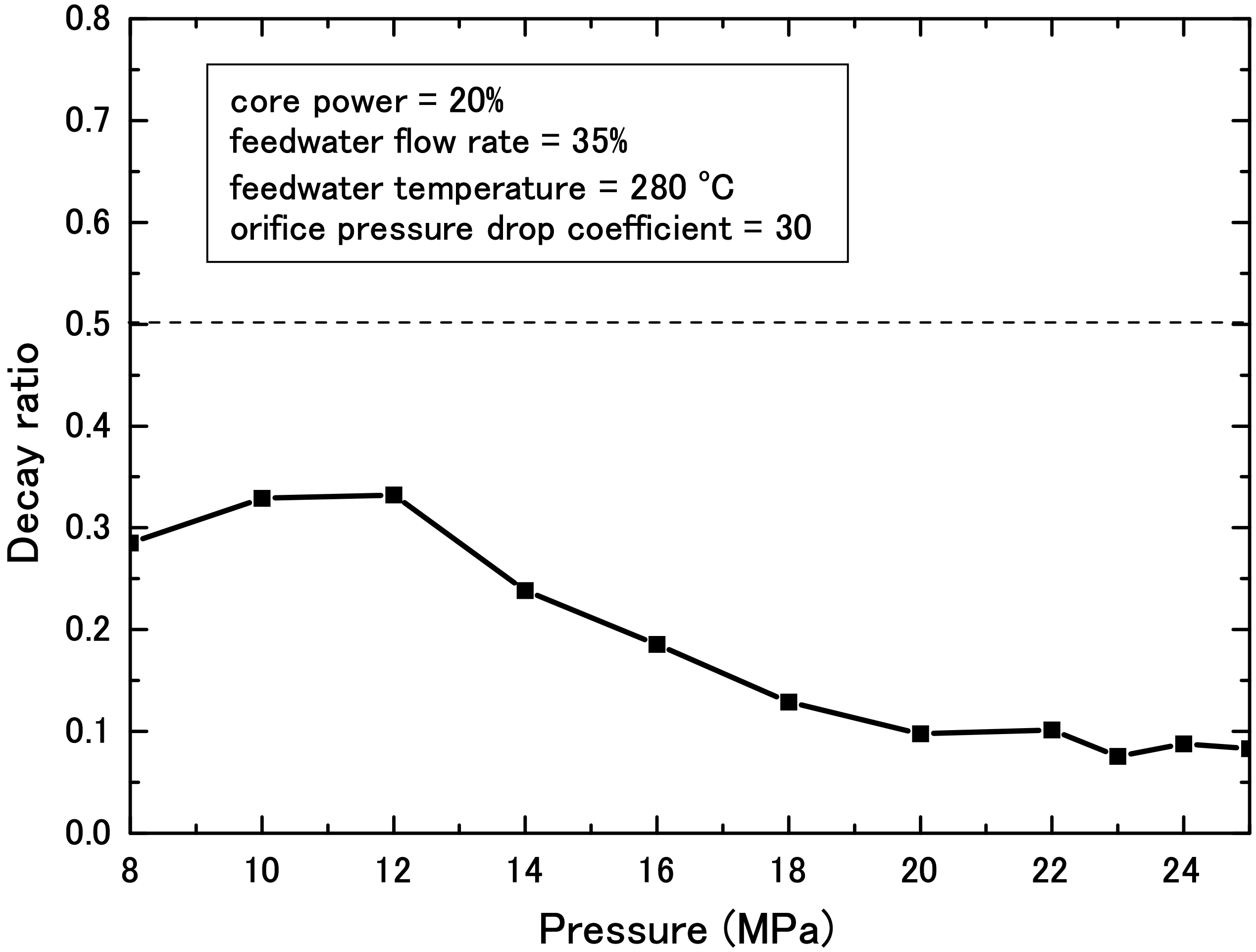


Phase response



A gain peak occurs around 2 rad/s due to flow feedback (coolant transit time ~ 3.2 s).

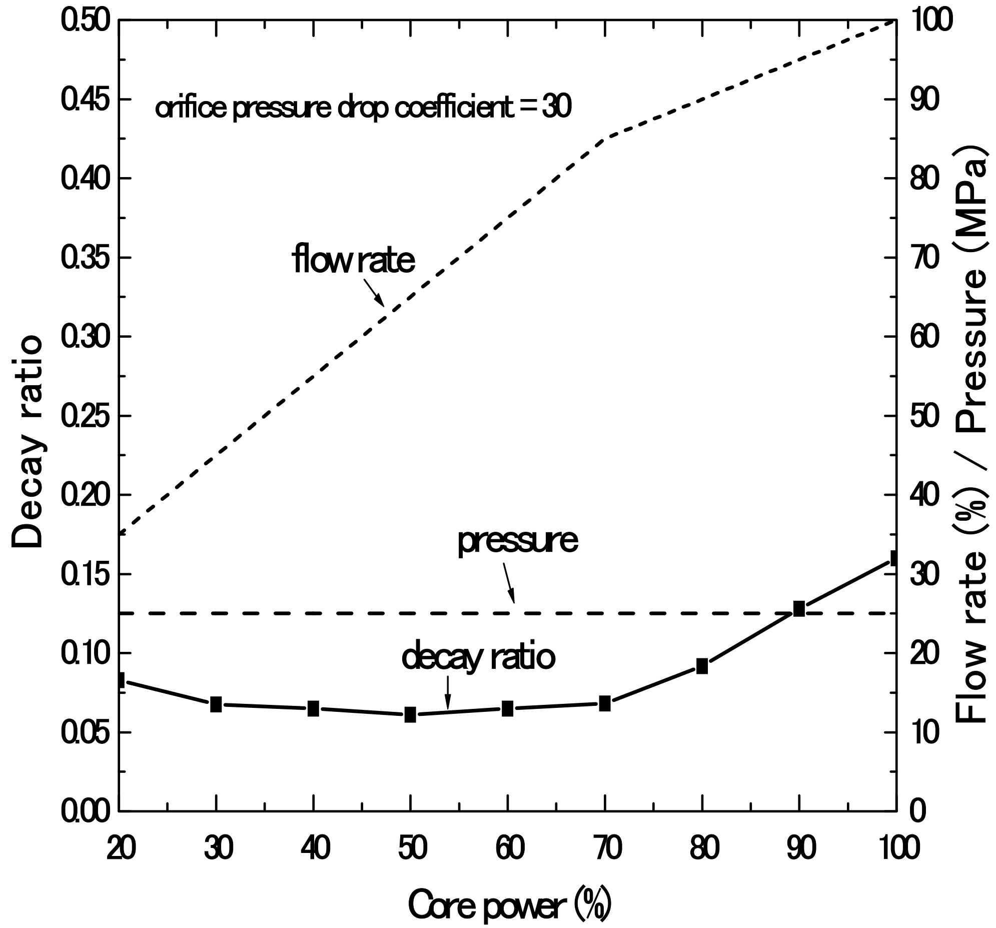
Thermal-hydraulic stability (Pressurization phase)

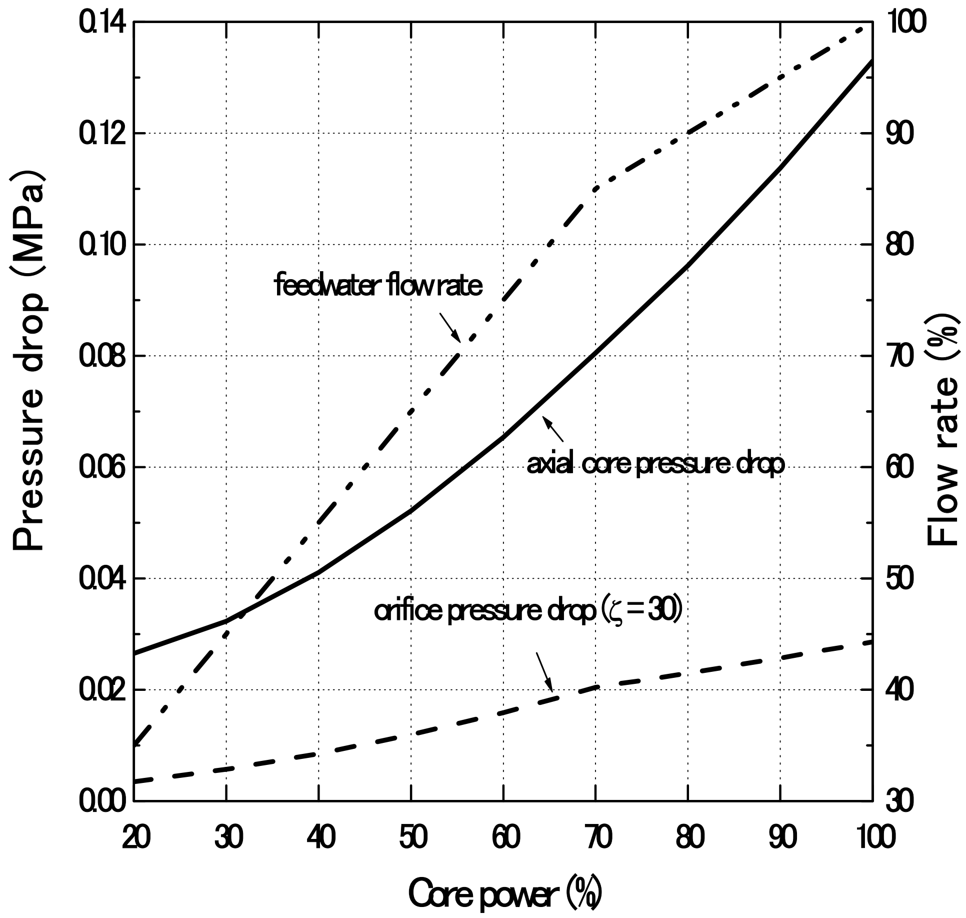


Axial core pressure drop = 0.027~0.037 MPa

Orifice pressure drop = 0.0034 MPa

Thermal-hydraulic stability (Power-raising phase)

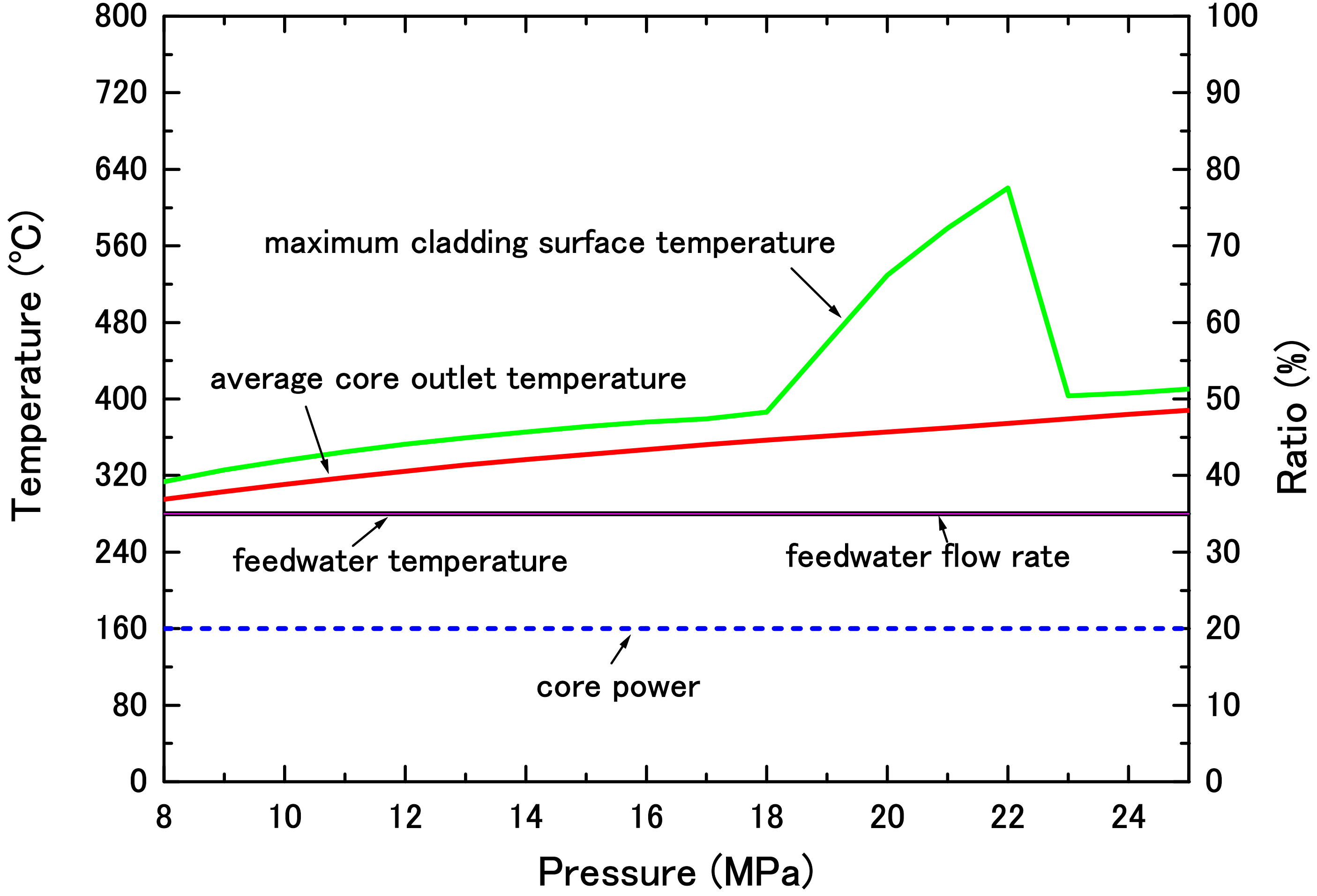




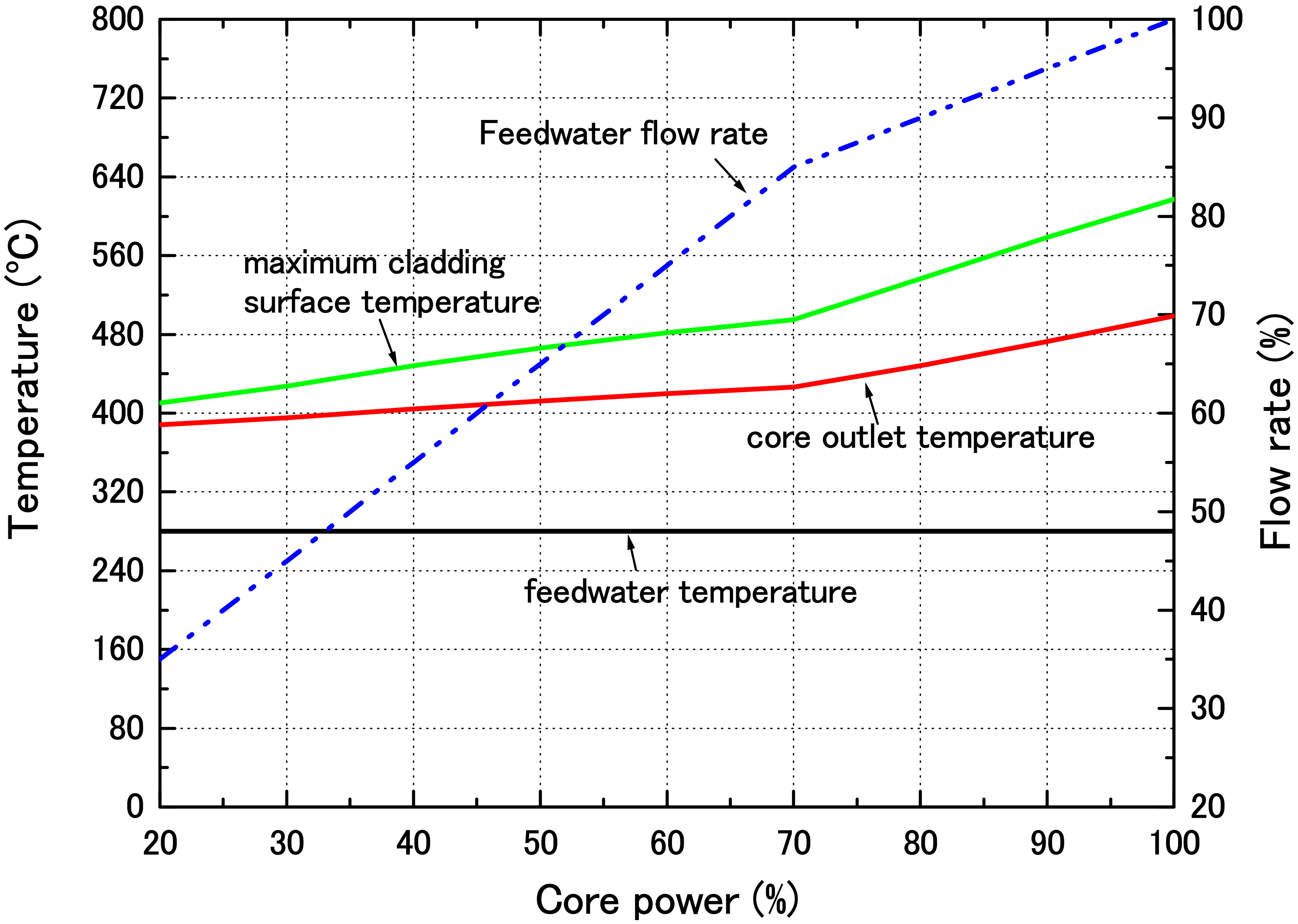
Axial core pressure drop (100% maximum power) = 0.133 MPa

Orifice pressure drop (100% maximum power) = 0.028 MPa

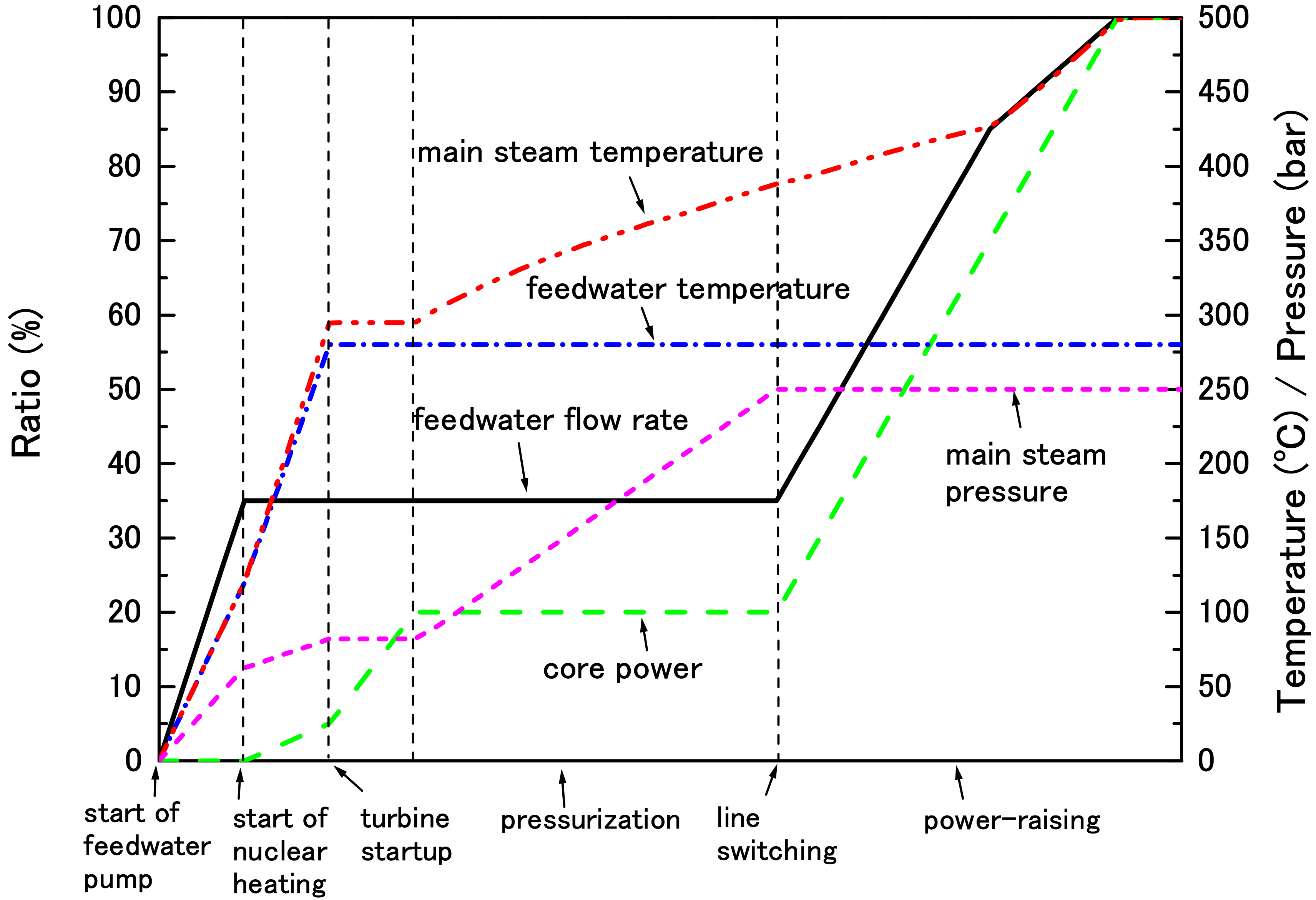
Thermal-Hydraulic Analysis (Pressurization phase)



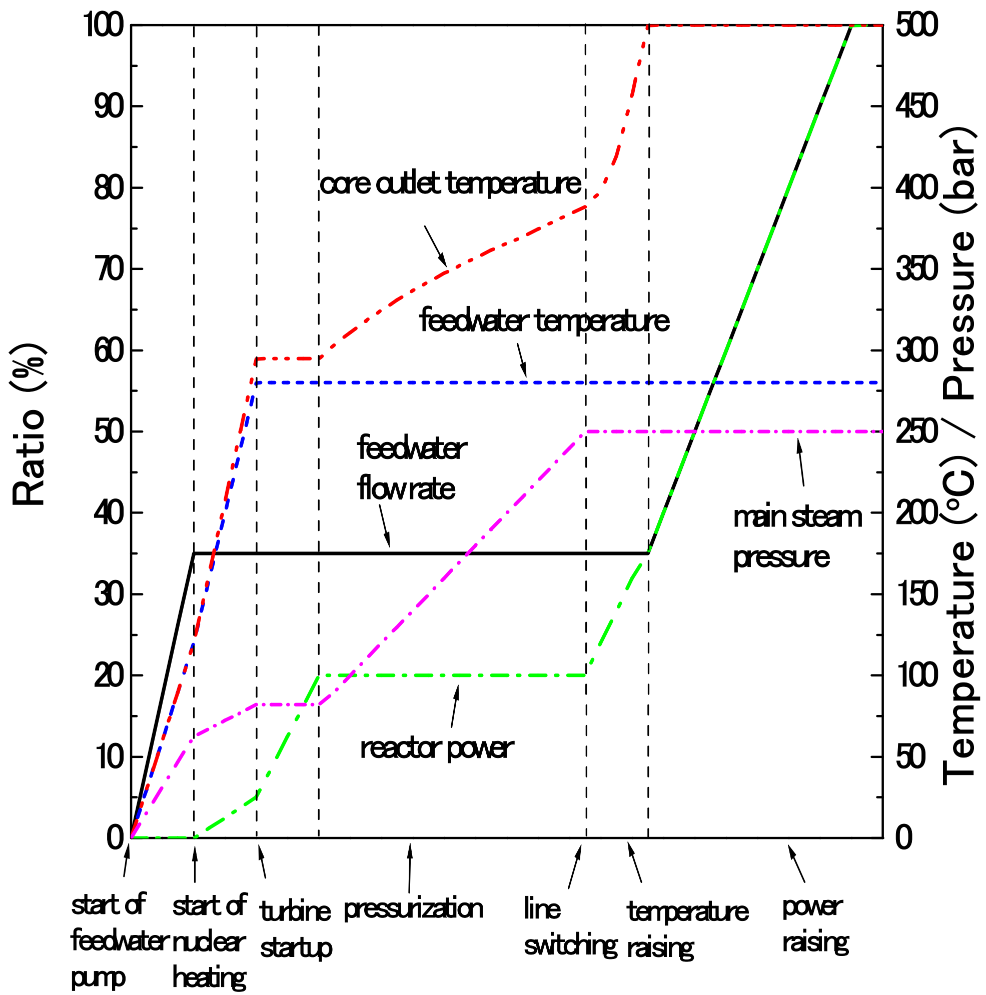
Thermal-Hydraulic Analysis (Power-raising phase)



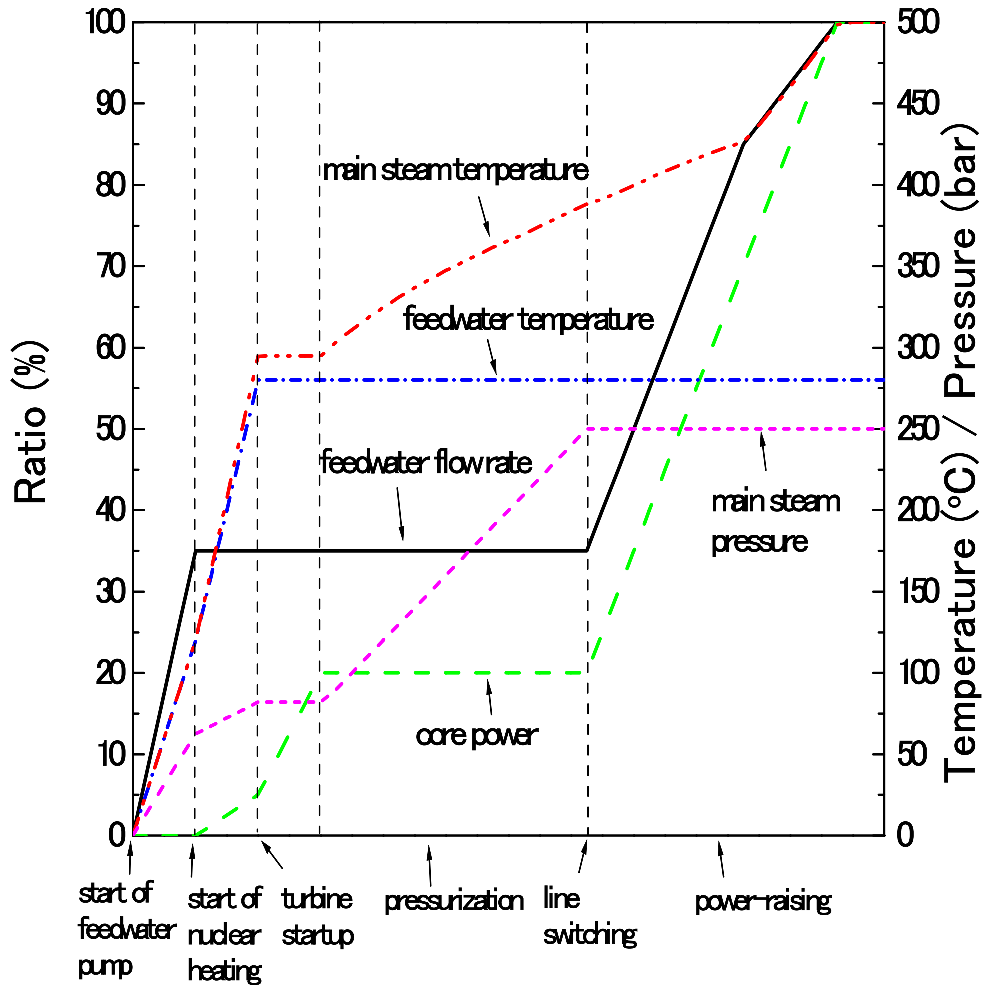
Sliding Pressure Startup Curve



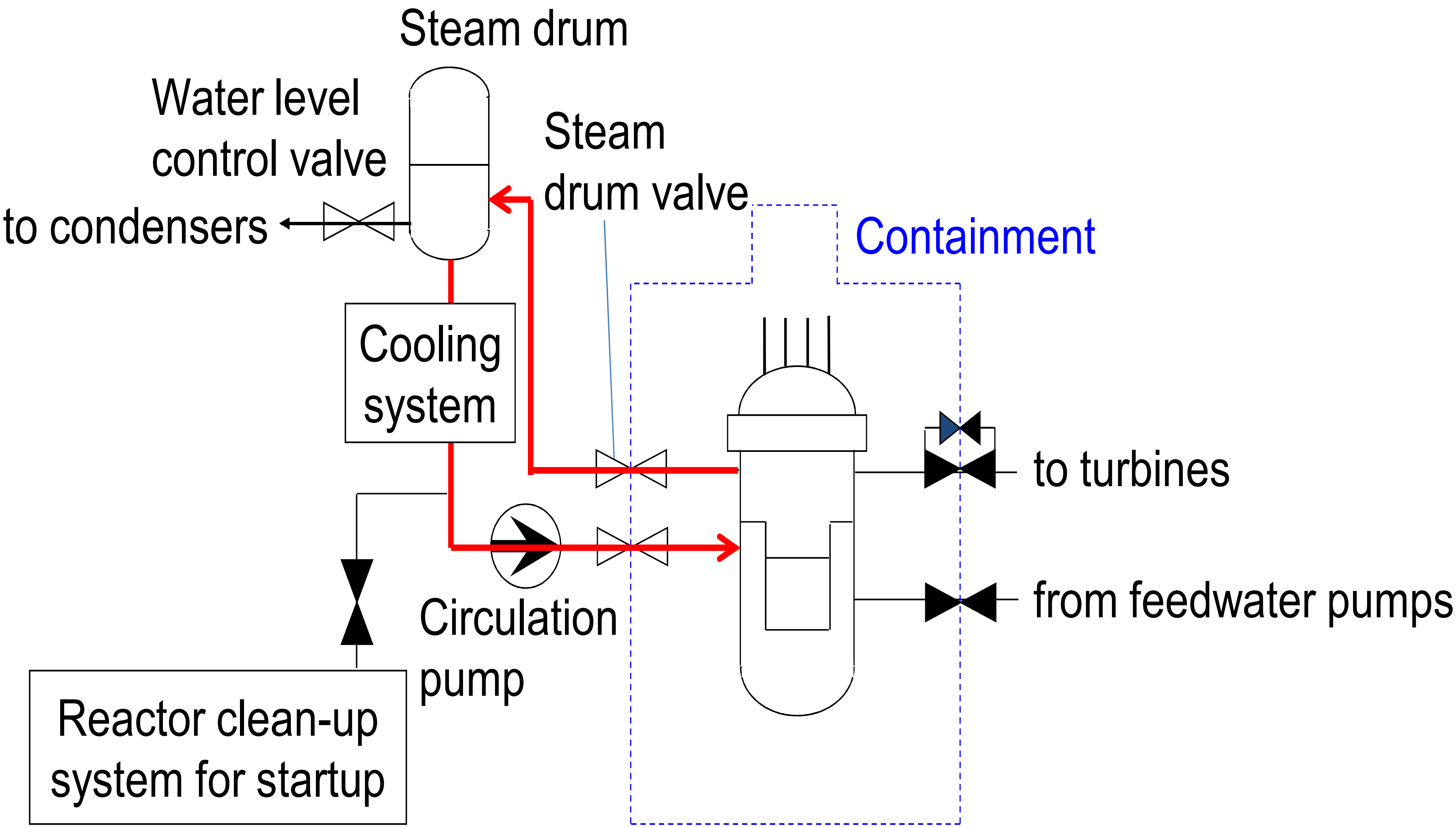
Sliding pressure startup curve (Thermal criteria only)



Sliding pressure startup curve (Both Thermal and Stability criteria)



Revised startup system of Super LWR and the Super FR



Thank you