Overview of Research at Yamaji Lab WASEDA University

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Akifumi Yamaji

- 1997 – 2006 : University of Tokyo, Dr. Engineering

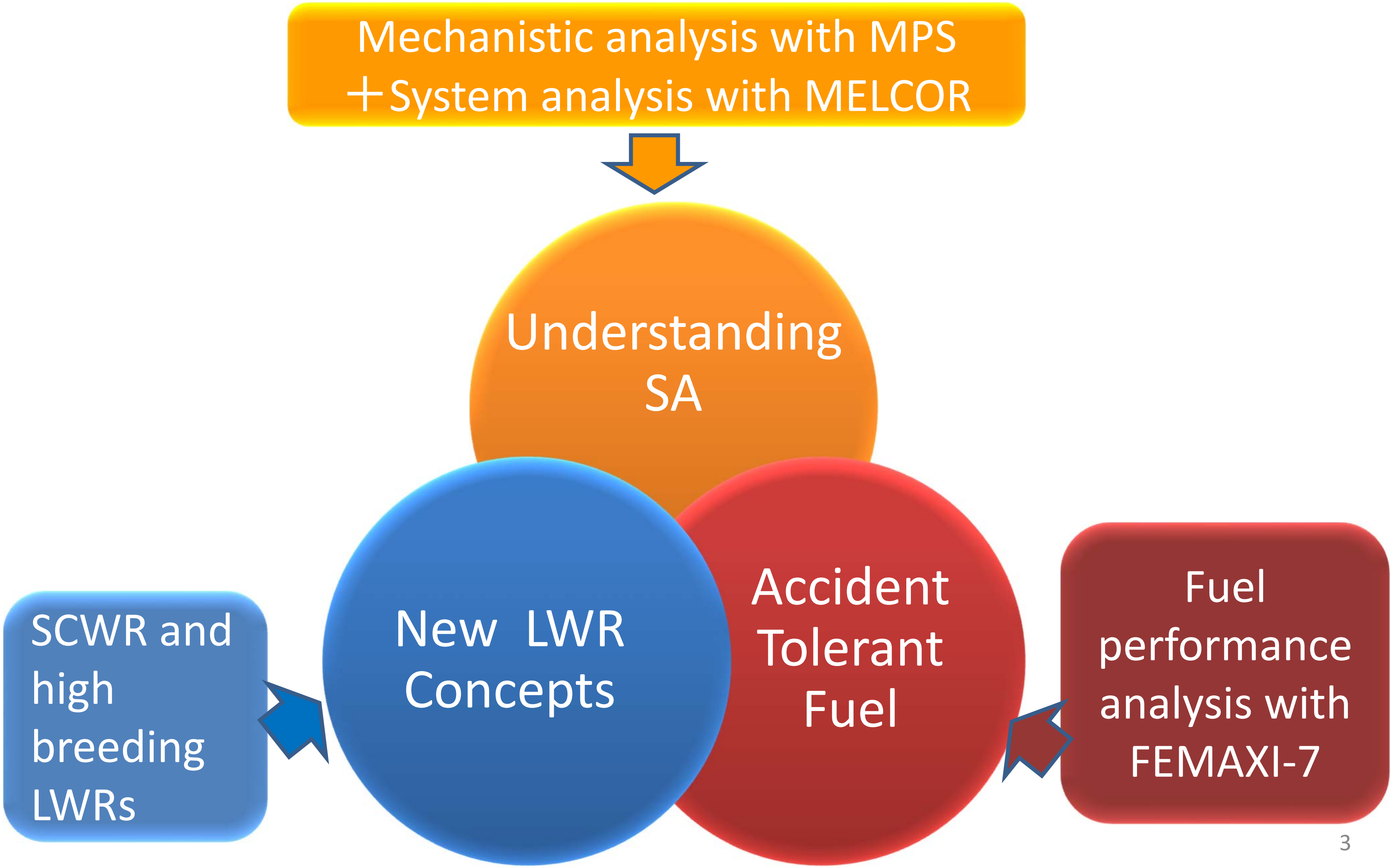
- 2006 – 2011: Research Engineer, Japan Atomic Energy Agency (JAEA)

- 2011 – 2014: Nuclear Scientist, OECD Nuclear Energy Agency (NEA)

- 2014 –: Assistant Professor (tenure), Waseda University

Waseda University

Research Scope of Yamaji Lab



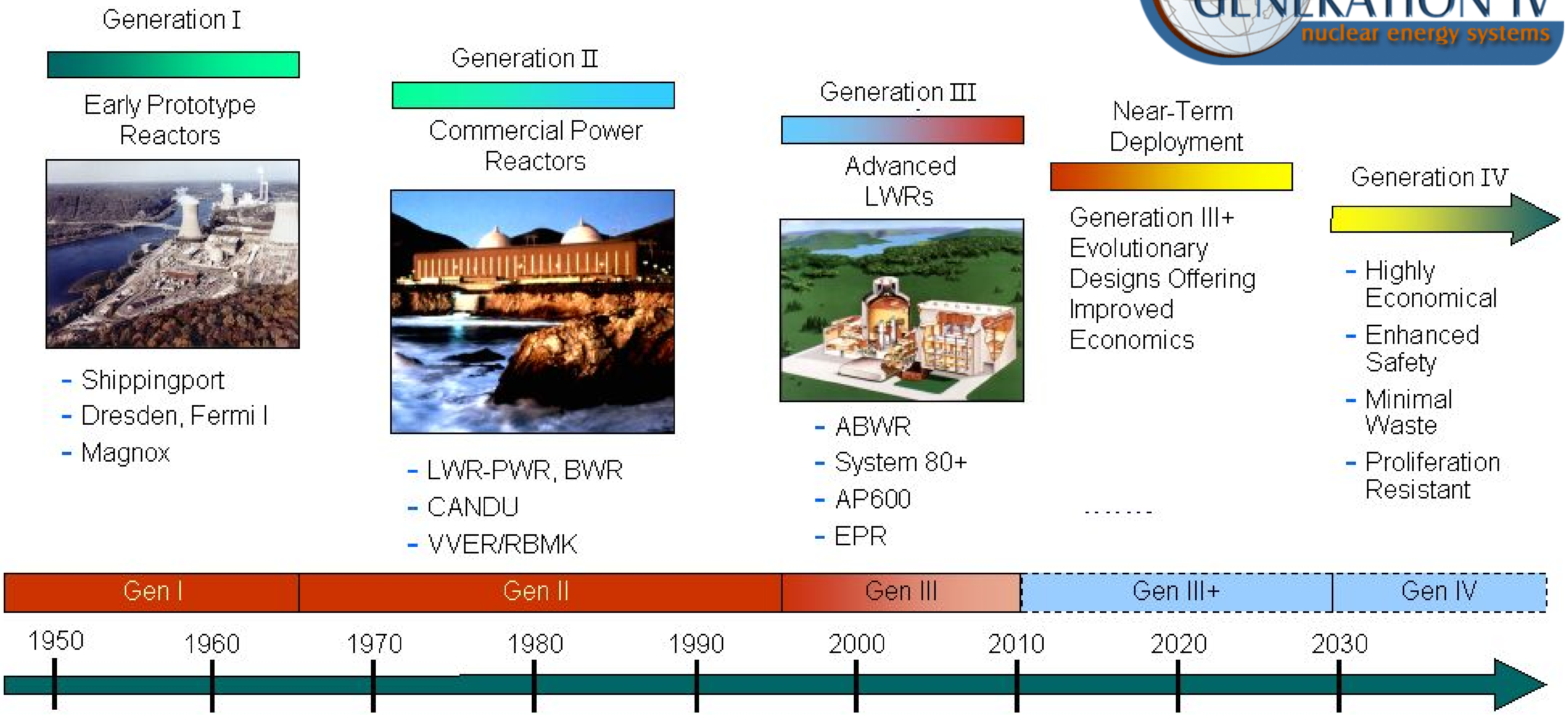
New LWR Reactor Concept Studies

SCWR

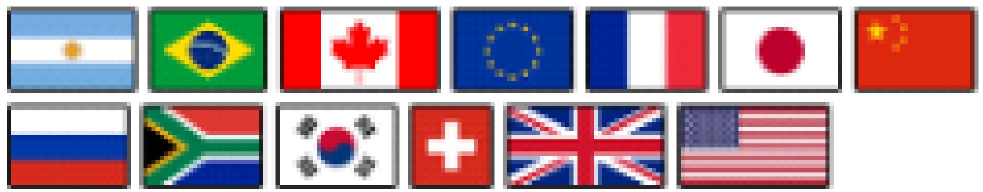
High breeding LWR

New LWR Concepts

Generation IV Reactors（GEN-IV）



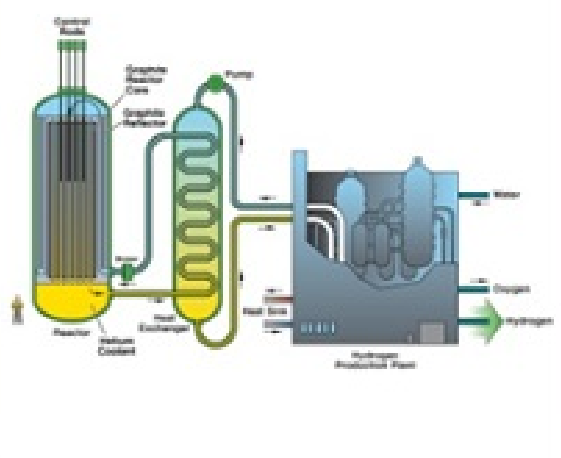
- Sustainability
- Economics
- Safety & reliability
- Proliferation resistance & physical protection



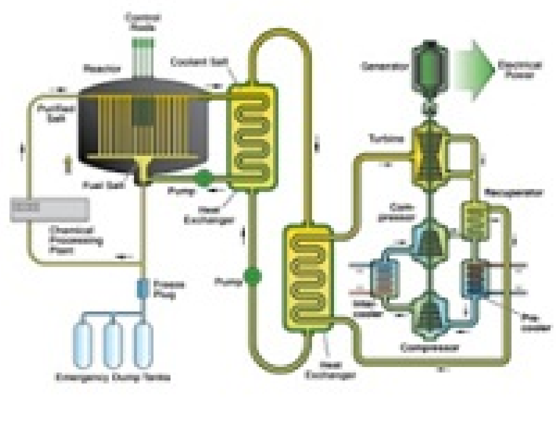
Source：Wikipedia
https://ja.wikipedia.org/wiki/%E7%AC%AC4%E4%B8%96%E4%BB%A3%E5%8E%9F%E5%AD%90%E7%82%89

GEN-IV Concepts

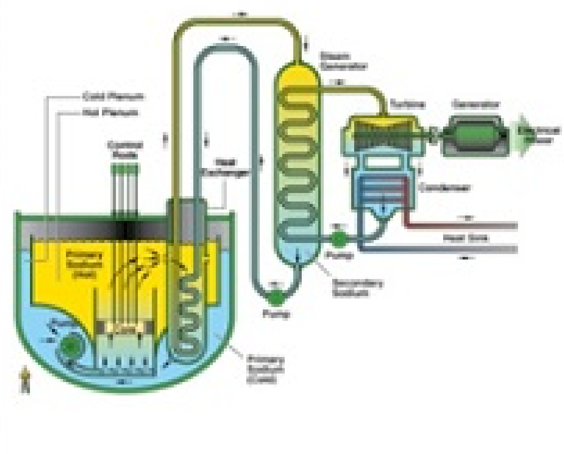
Very-High-Temperature Reactor



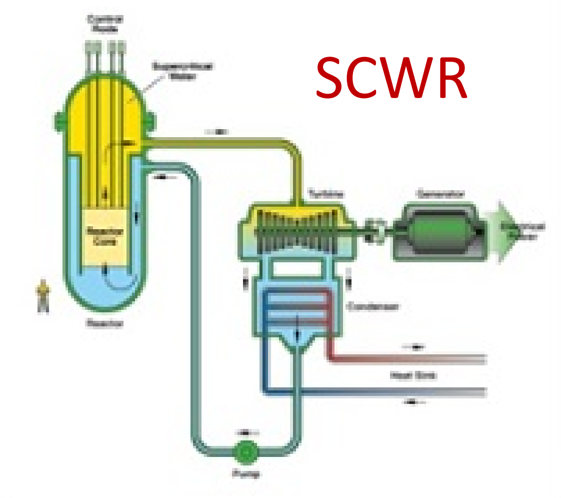
Molten Salt Reactor



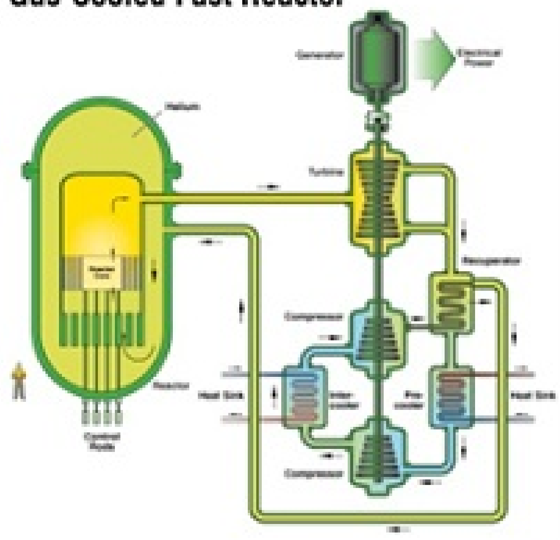
Sodium-Cooled-Fast Reactor



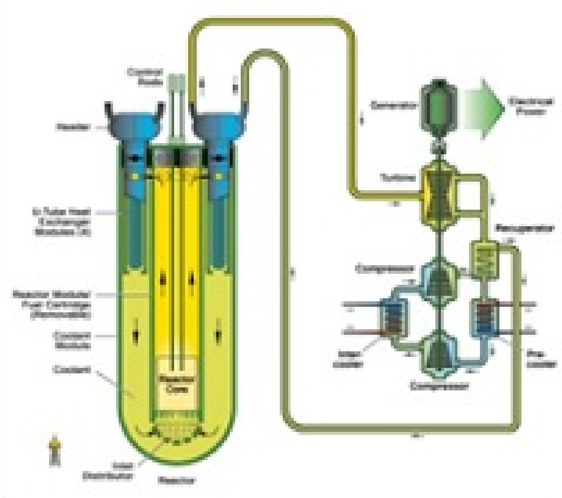
Supercritical-Water-Cooled Reactor



Gas-Cooled Fast Reactor



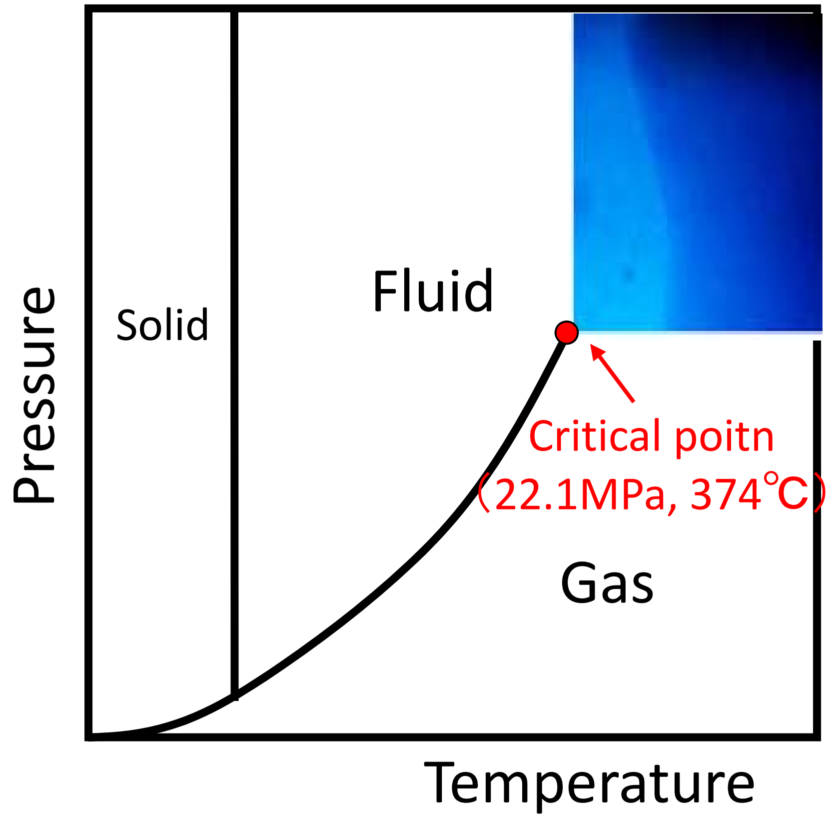
Lead-Cooled Fast Reactor



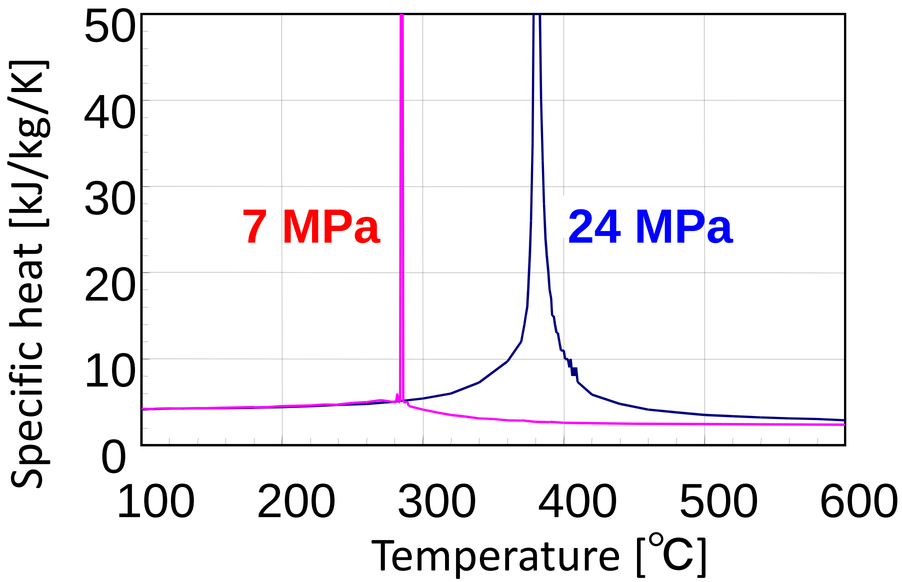
Source： Wikipedia
https://ja.wikipedia.org/wiki/%E7%AC%AC4%E4%B8%96%E4%BB%A3%E5%8E%9F%E5%AD%90%E7%82%89

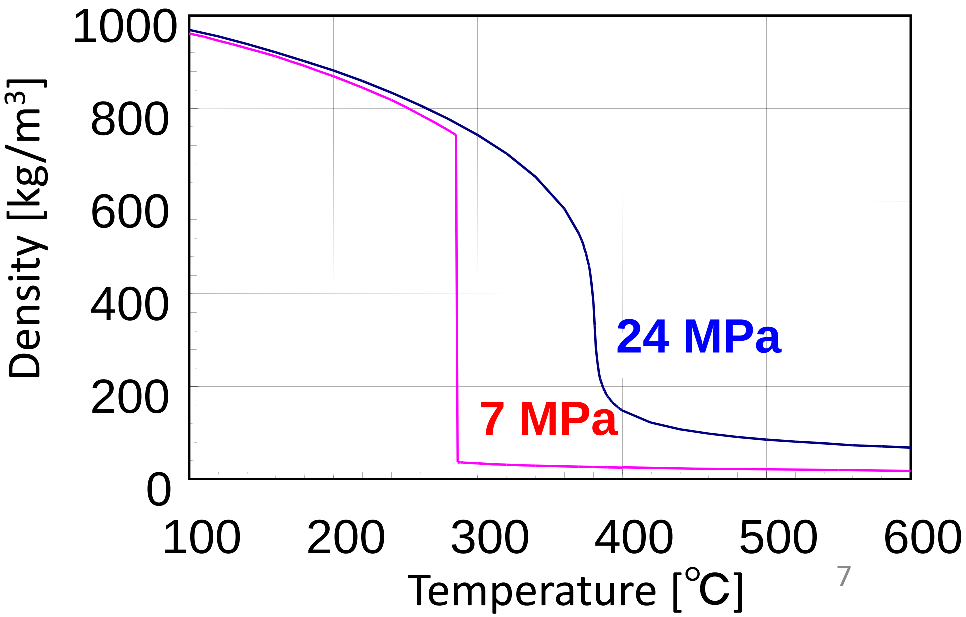
What is Supercritical Water?

- High temperature & pressure
- No phase change (boiling)
- Matured technology (coal fire power plants since 1960s)



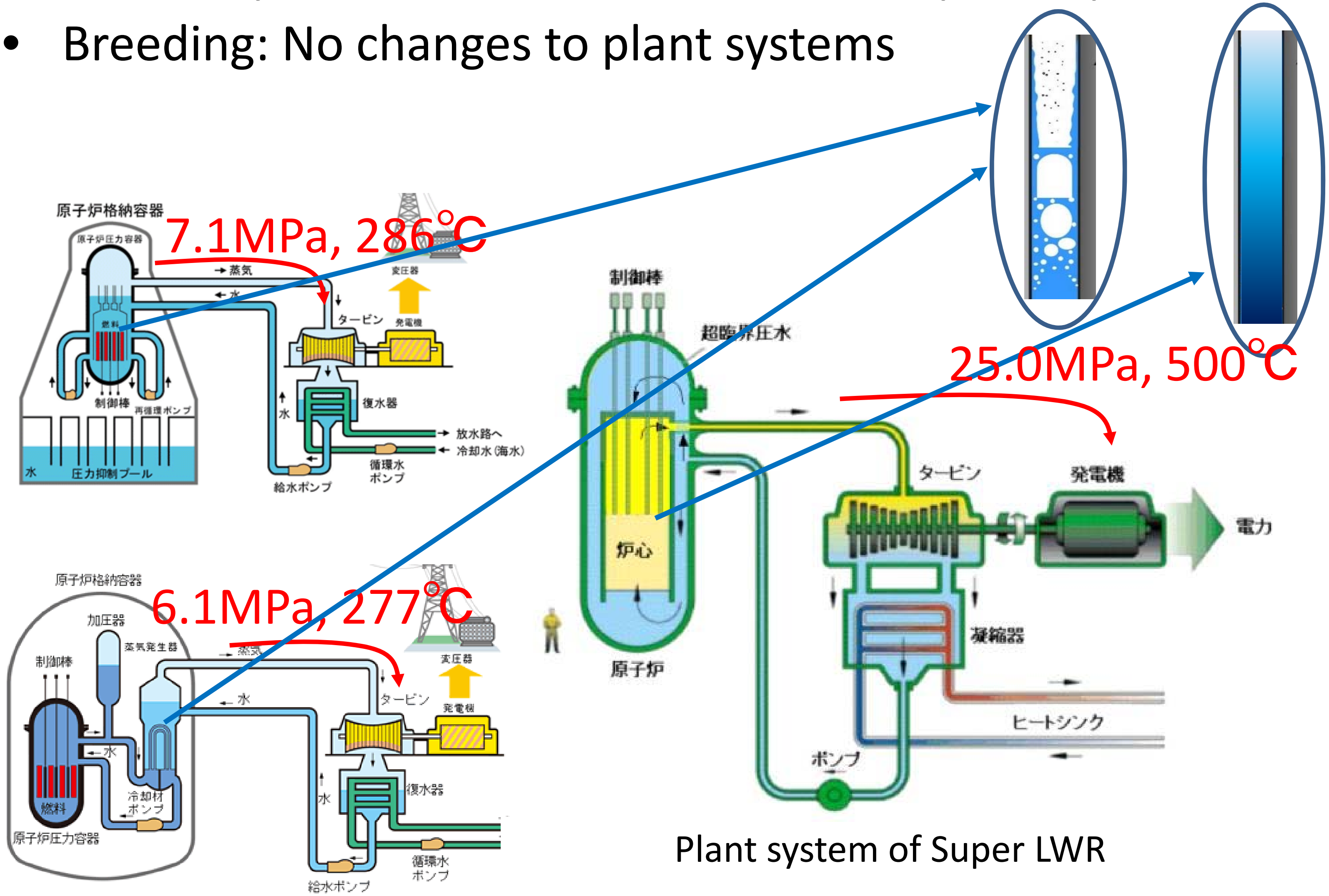
Phase diagram of water





What is SCWR (Super LWR)?

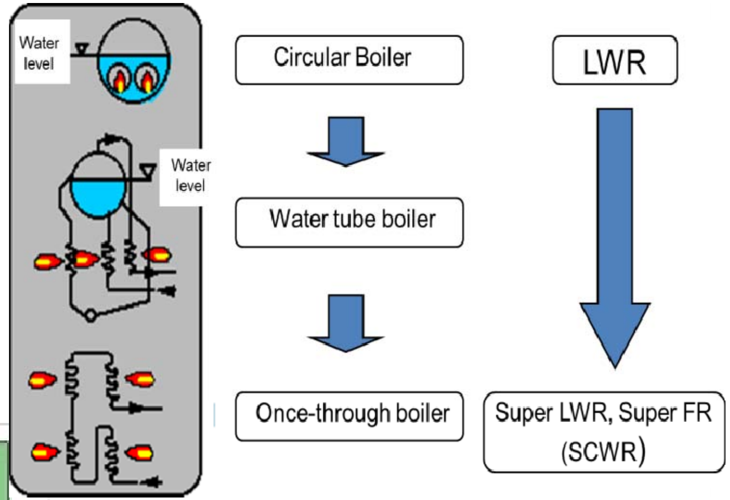
- Economy : High efficiency, compact & simple (low capital cost)
- Reliability : Matured LWR and fossil fired power plant technologies
- Breeding: No changes to plant systems

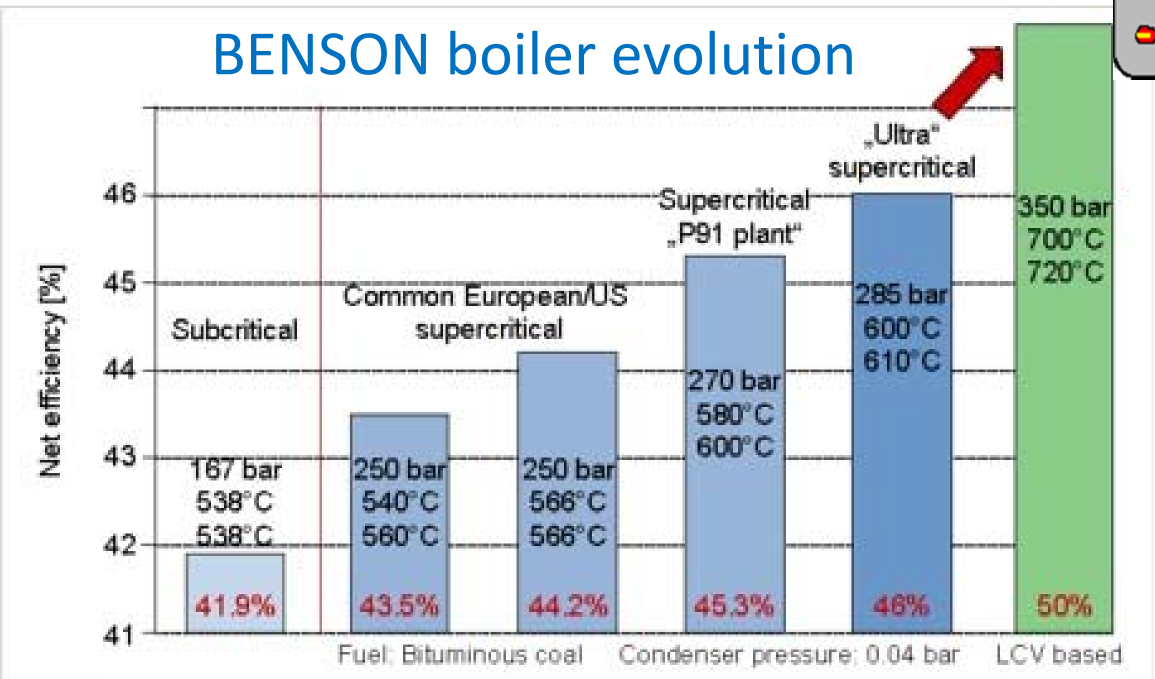


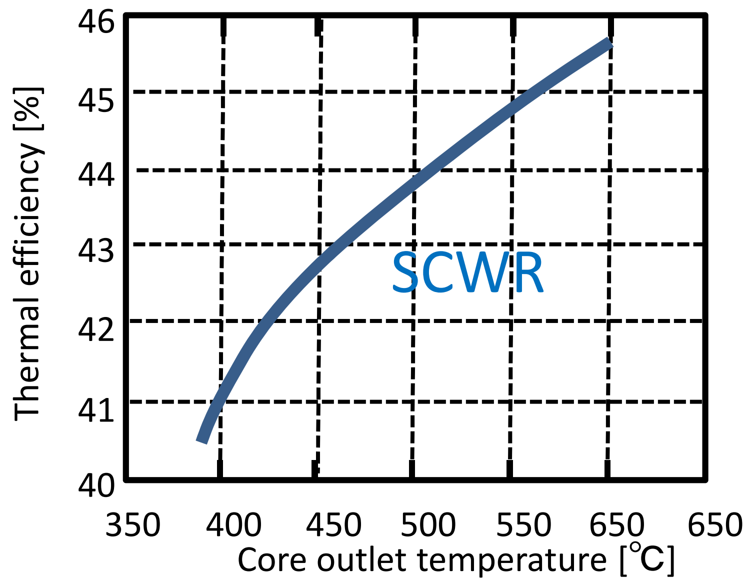
Plant system of Super LWR

Evolution of Boilers and Thermal Efficiency

- SC: Supercritical
- USC: Ultra-supercritical

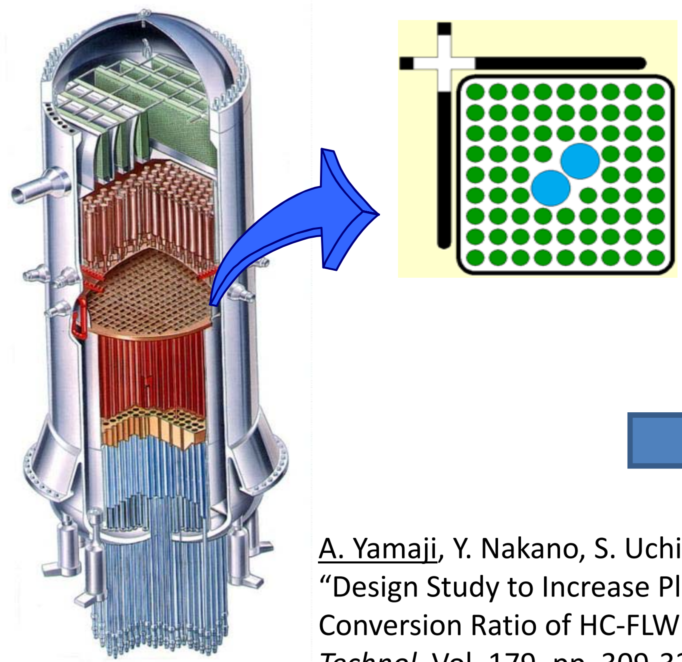




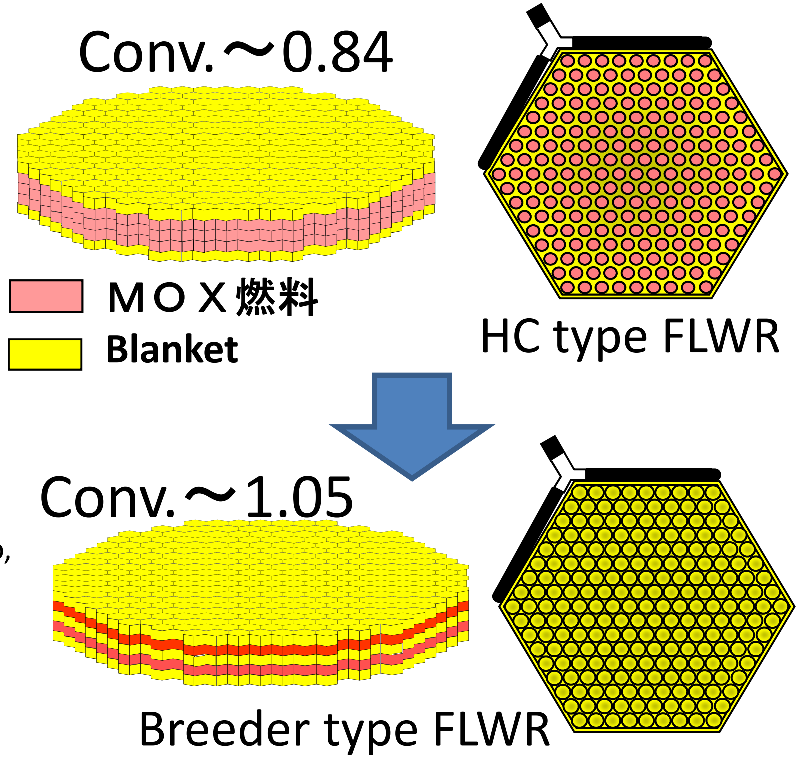


Source: BENSON® Boiler (SIEMENS)
http://www.energy.siemens.com/mx/en/fossil-power-generation/power-plants/steam-power-plants/benson.htm#content=Efficiency

Breeding with LWR



A. Yamaji, Y. Nakano, S. Uchikawa, T. Okubo, "Design Study to Increase Plutonium Conversion Ratio of HC-FLWR Core," *Nucl. Technol.* Vol. 179, pp. 309-322 (2012)

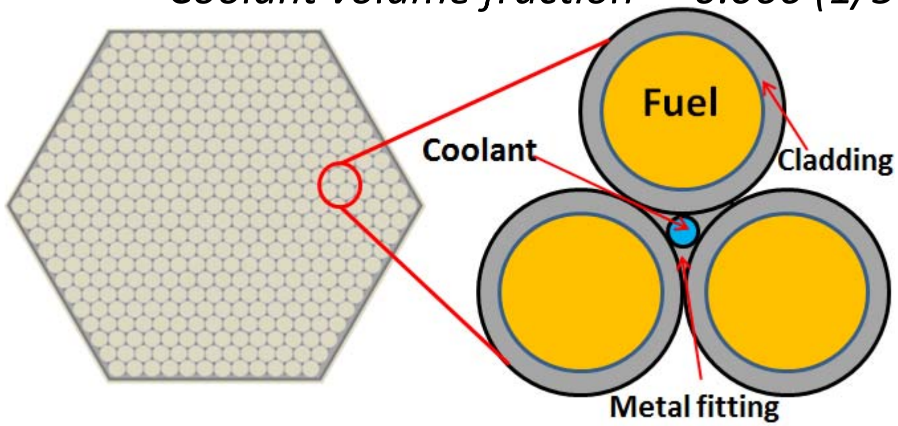


- BWR type RMWR (Reduced Moderation Water Reactor)
  - Fissile Plutonium Surviving Ratio (FPSR)～*1.007*
  - Compound System Doubling Time (CSDT) ～245 years
- Average energy consumption in advanced countries is expected to double in the next 50 years

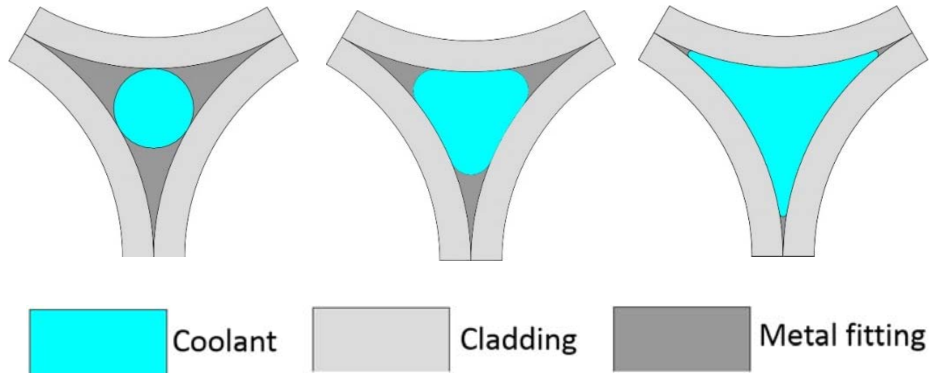
→ **Design target of CSDT < 50 years**

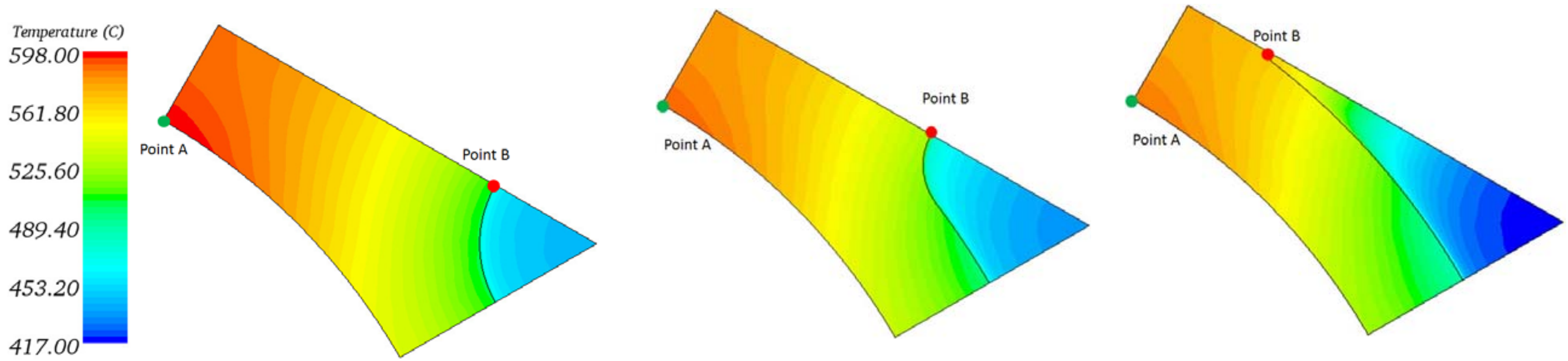
Breeding with LWRs

Coolant volume fraction ~0.066 (1/5 of RMWR)



Tightly packed fuel assembly





Analysis with CFD code (STAR-CCM+)

Design area identified with respect to Maximum Linear Heat Rate, Maximum Cladding Temperature, Pressure drop

R. Guo, Y. Oka. CFD analysis of coolant channel geometries for a tightly packed fuel rods assembly at subcritical pressure. Nuclear Engineering and Design, Vol. 284, pp. 115-129.

R. Guo, Y. Oka. CFD analysis of coolant channel geometries for a tightly packed fuel rods assembly of Super FBR. Nuclear Engineering and Design. With editor.

BWR Core Design

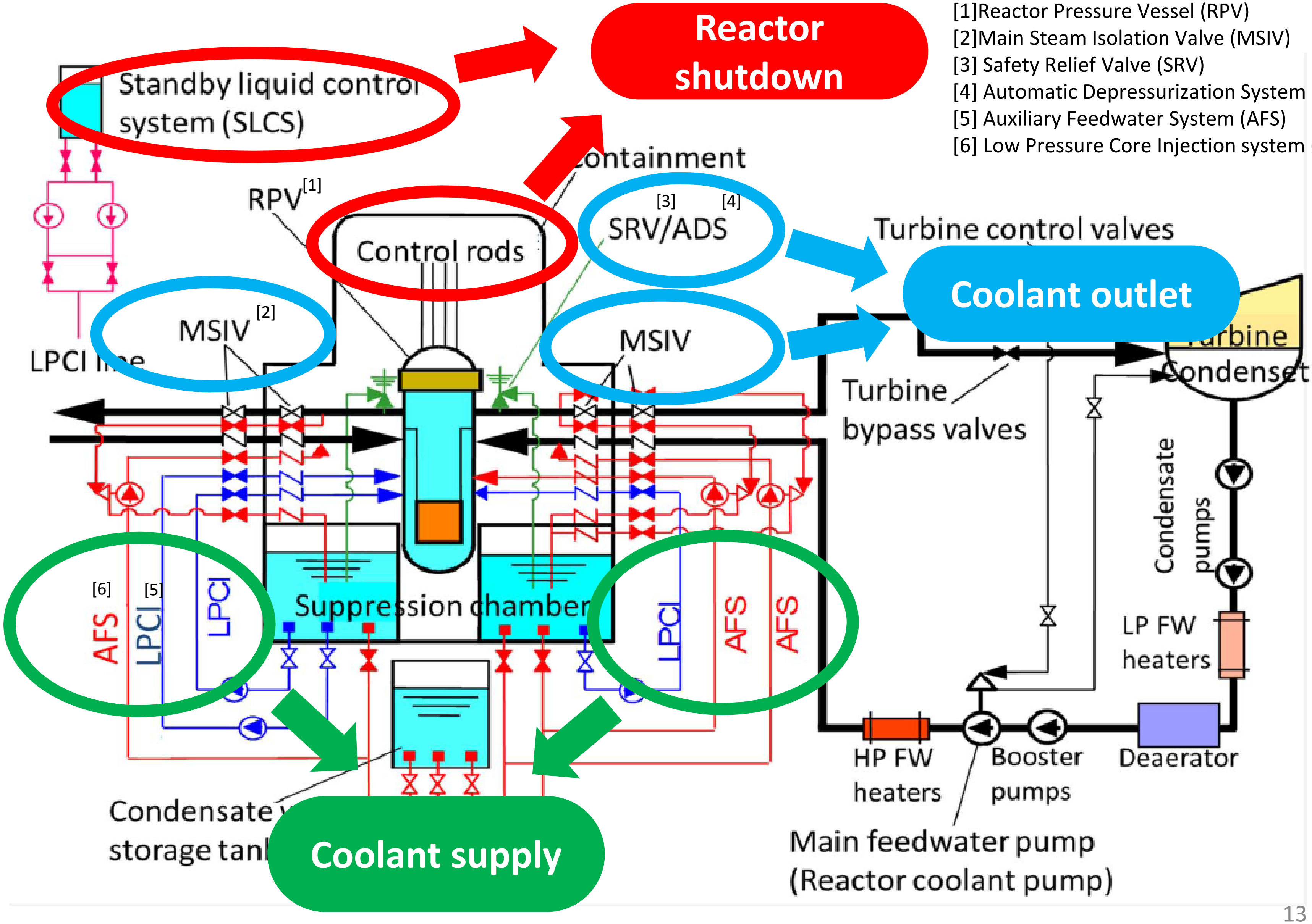
Performance of breeding BWR core	
Ave. Pu enrichment	*27.5 wt%*
Operating cycle length	*700 d*
Ave. discharge burnup (seed)	*48.25 GWd/tHM*
Fuel batch (Seed/Blanket)	*4/1*
Maximum Mass flux	*9,950 kg/m2s*
Power(th)	*1,017 MW*
Average exit quality (%)	*0.226*
Core pressure drop	*3.31 MPa*
Maximum cladding temperature in Seed (BOEC/EOEC)	*307.2 ° C/306.1 ° C*
Maximum cladding temperature in Blanket (BOEC/EOEC)	*293.4 ° C/296.2 ° C*
Void reactivity (BOEC/EOEC)	***-0.29%/-0.49%***
MCHFR (BOEC/EOEC)	***1.9/1.9***
FPSR	***1.061***
CSDT	***41.6y***

- ***The breeding goals are achieved***
- ***The design criteria are satisfied***

- ***Power and burnup need to be increased, pressure drop is too high***

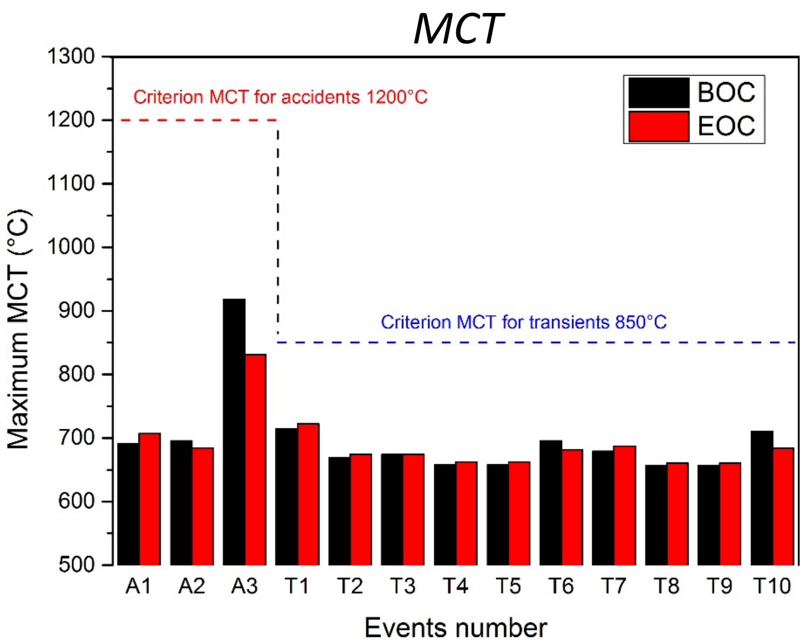
Plant Safety System

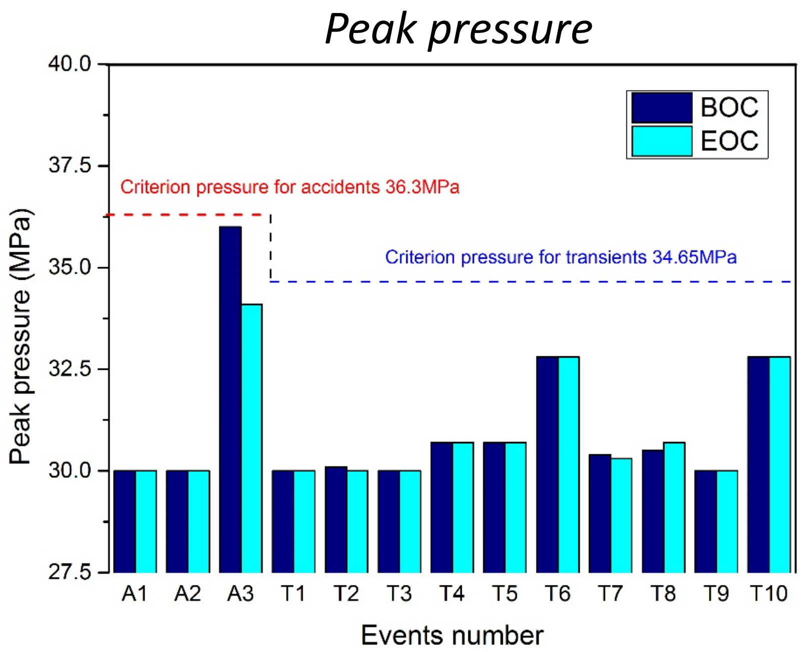
[1]Reactor Pressure Vessel (RPV)
[2]Main Steam Isolation Valve (MSIV)
[3] Safety Relief Valve (SRV)
[4] Automatic Depressurization System (ADS)
[5] Auxiliary Feedwater System (AFS)
[6] Low Pressure Core Injection system (LPCI)



Source: Oka, Y. et al., 2010, Super Light Water Reactors and Super Fast Reactors. Springer. Chap. 6.

Summary of safety analysis





Accident

1. *Total loss of feed water flow*
2. *Reactor coolant pump seizure*
3. *Control rod ejection (operation)*

Transient

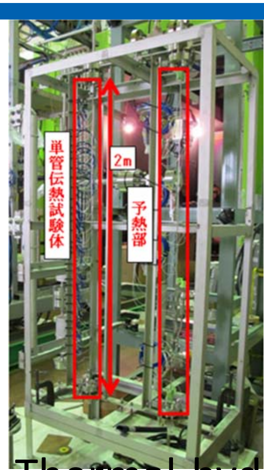
1. *Loss of feedwater heating*
2. *Inadvertent startup of AFS*
3. *Partial loss of feedwater flow*
4. *Loss of offsite power*
5. *Loss of turbine load (with bypass valves open)*
6. *Loss of turbine load (without bypass valves open)*
7. *Control rod abnormality (operation)*
8. *Feed water flowrate control system failure*
9. *Pressure control system failure*
10. *MSIV closure*

- *All the safety criteria are satisfied at the accidents and transients at supercritical pressure at both BOC and EOC.*
- *On account of the small fraction of coolant volume, the pressure and MCTs are more sensitive to "core heat-up" that leads to change of coolant temperature and flow rate.*
- *The "control rod ejection" is the most important event.*

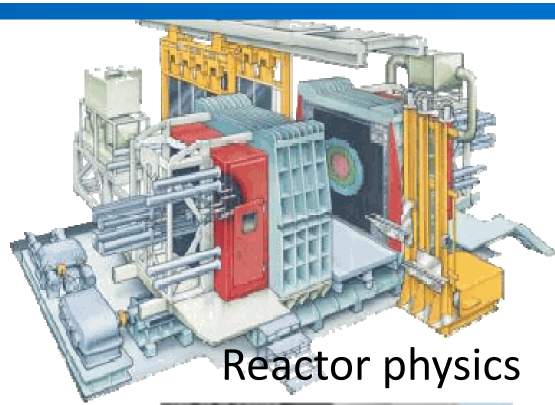
R. Guo and A. Yamaji, "Analysis of accidents and abnormal transients of a high breeding fast reactor cooled by supercritical-pressure light water," Analysis of accidents and abnormal transients of a high breeding fast reactor cooled by supercritical-pressure light water. Nuclear Engineering and Design. (accepted for publication)

National Projects (2005-2009/2010-2013)

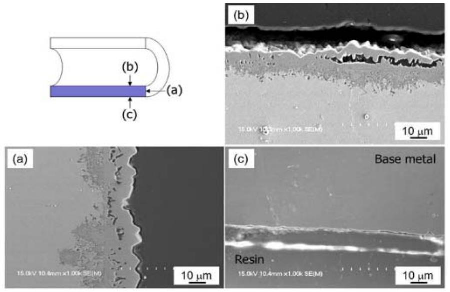
- Led by Prof. Oka
  - 7 universities and institutes



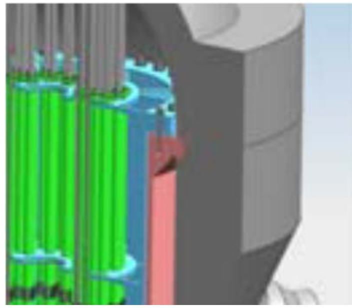
Thermal-hydraulics



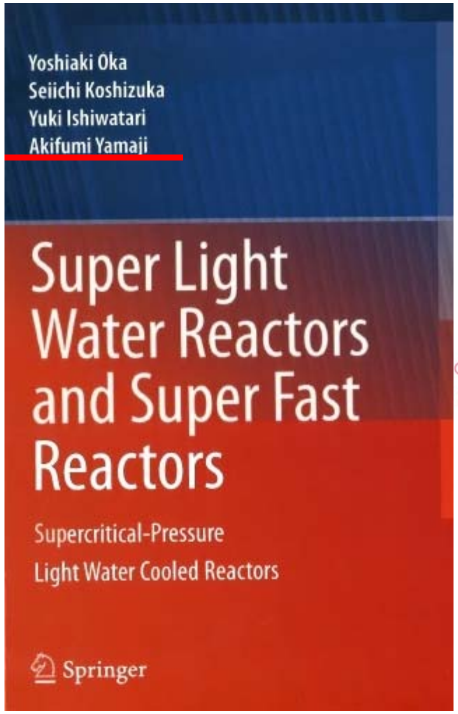
Reactor physics



Material development

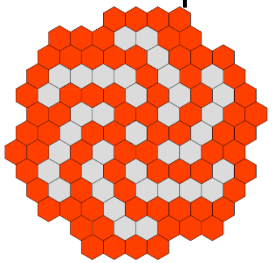


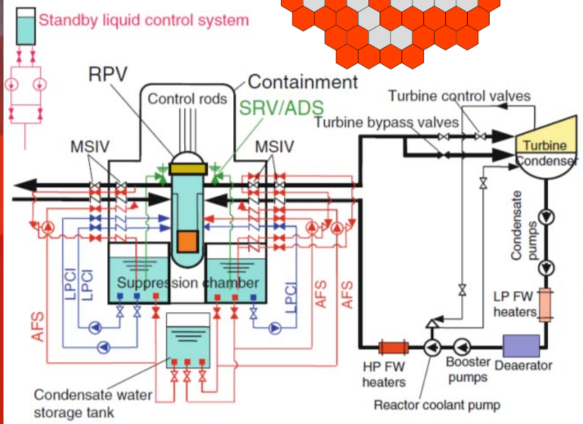
structures



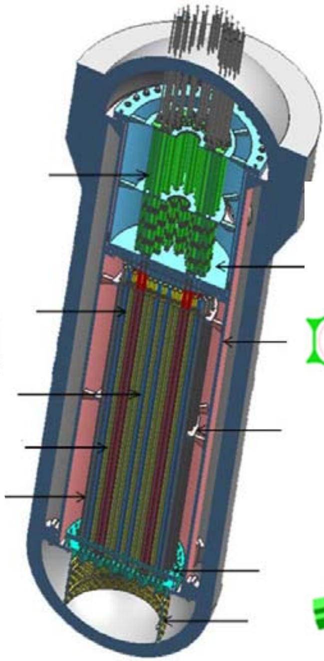
Published in 2010

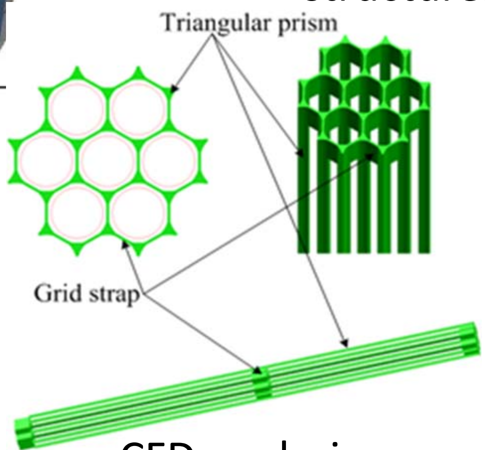
Core design





Plant safety and control

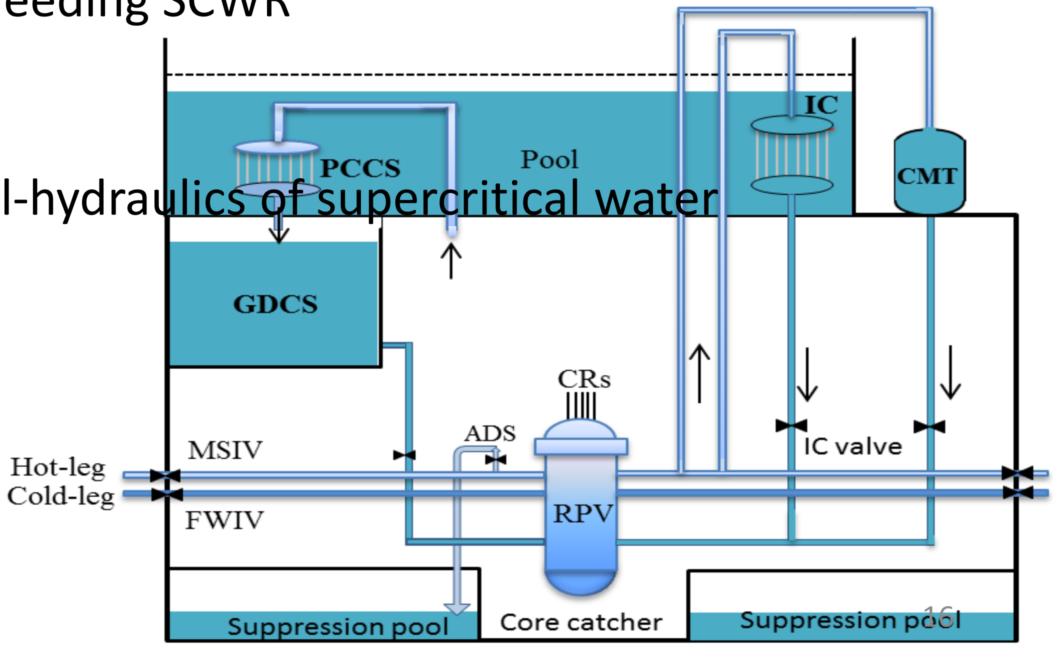




CFD analysis

Potential Subjects for Doctor Course Students

- High breeding LWR (BWR and SCWR)
  - Core design
  - Safety studies (including LOCA, ATWS)
- Severe accident study of SCWR
  - In-Vessel Retention (IVR) VS Core catcher concepts for SCWR
  - Passive safety
  - SA analysis with MELCOR
- Other subjects
  - Startup and stability of high breeding SCWR
- Available codes
  - SRAC for core neutronics
  - Home-made codes for thermal-hydraulics of supercritical water
  - STAR-CCM+
  - MELCOR
  - (RELAP)
  - MPS



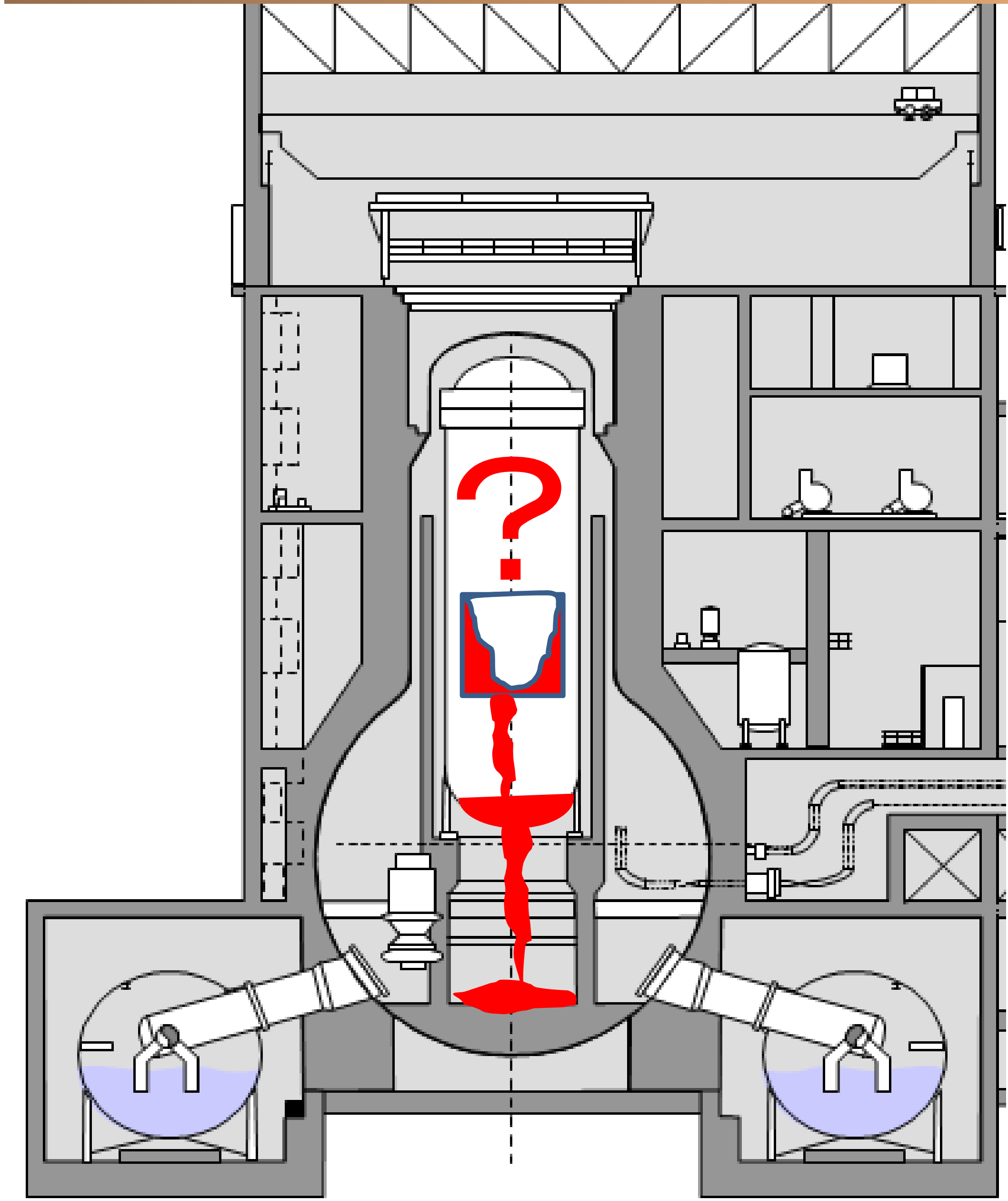
Understanding Severe Accidents

MPS Method

MELCOR

Understanding SA

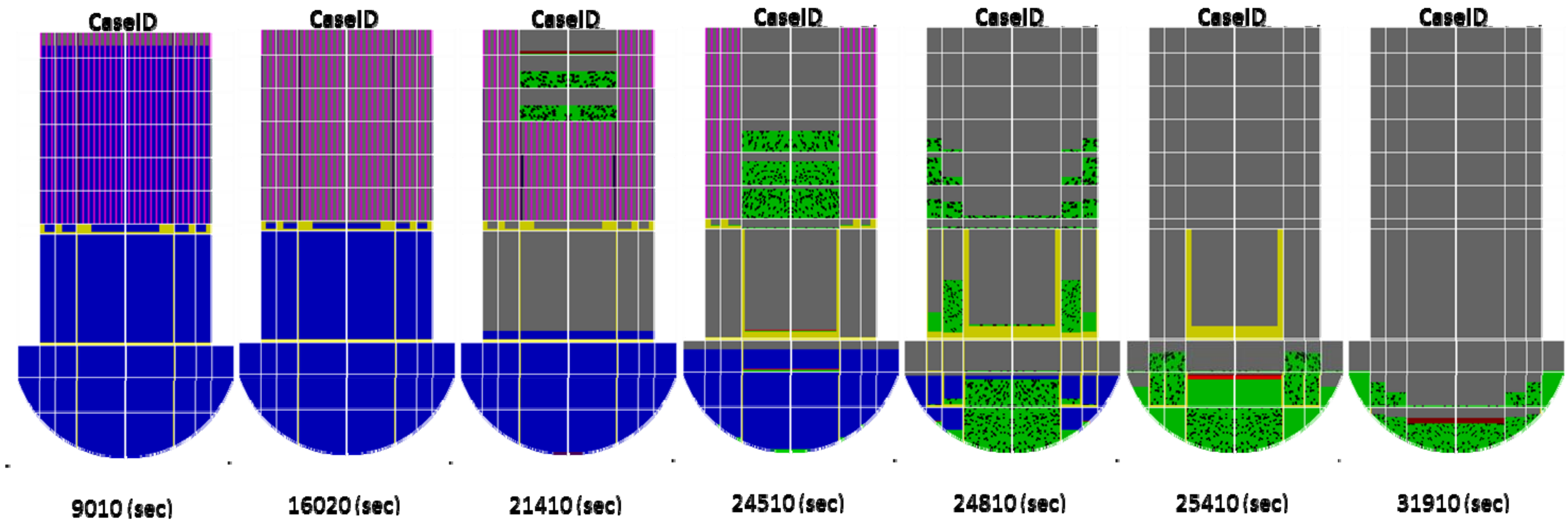
Understanding Melt Propagation, Debris Distributions, Status



- By investigation
- By experiment
- By simulation (computer science)
  - System analysis (empirical models)
  - MPS method (mechanistic models)

Understanding Accident Propagation with MELCOR

- Good for capturing overall accident propagation
- Dependent on empirical models and various assumptions (failure modes)
- Limited experience

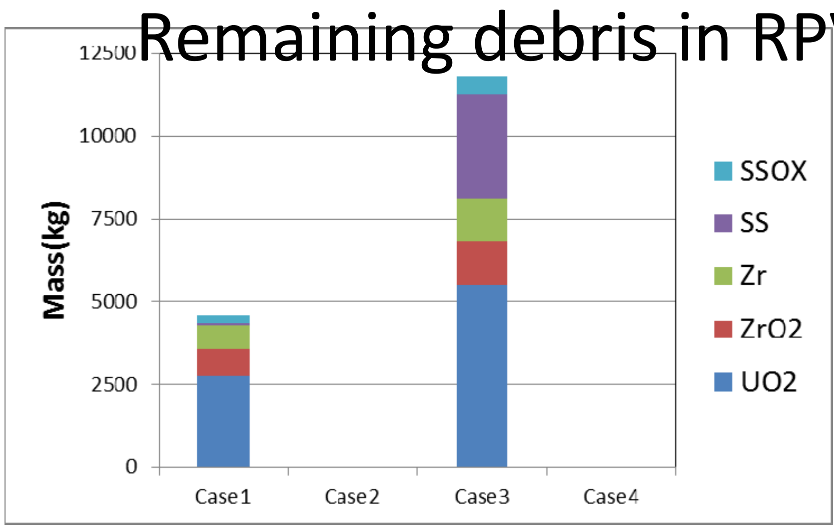


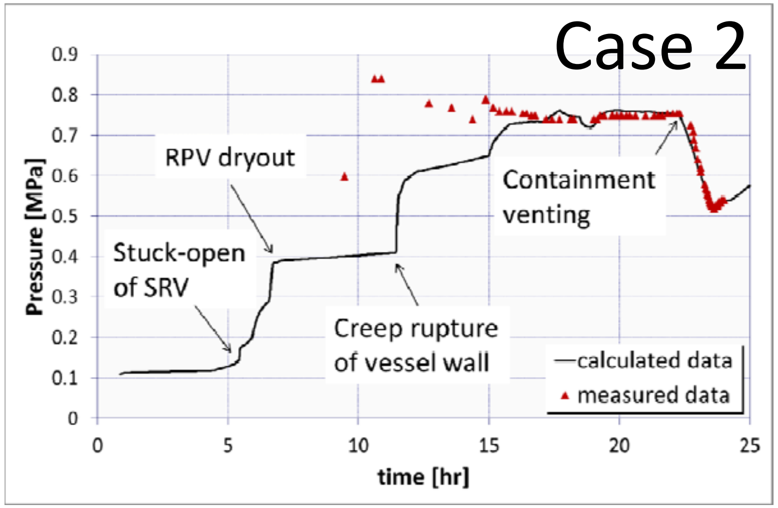
Example of core degradation simulated for 1F1 type accident

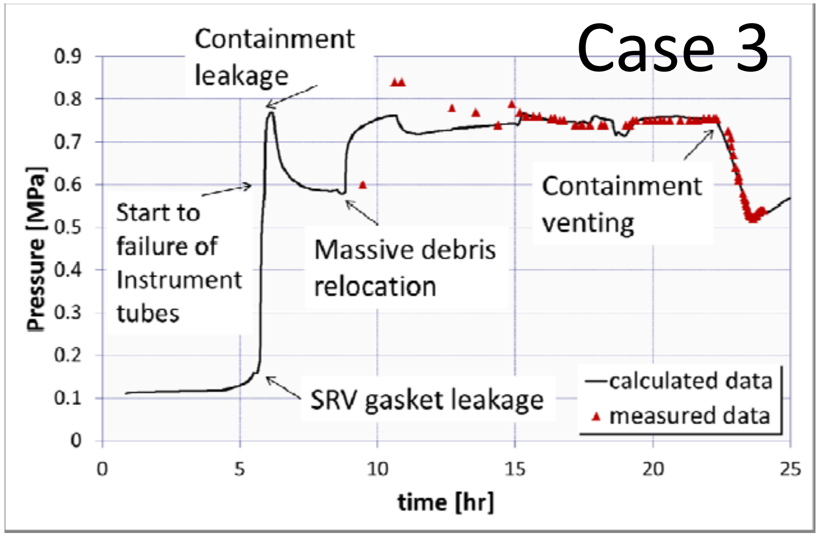
Understanding Accident Scenario of 1F1

Table I. Sensitivity cases

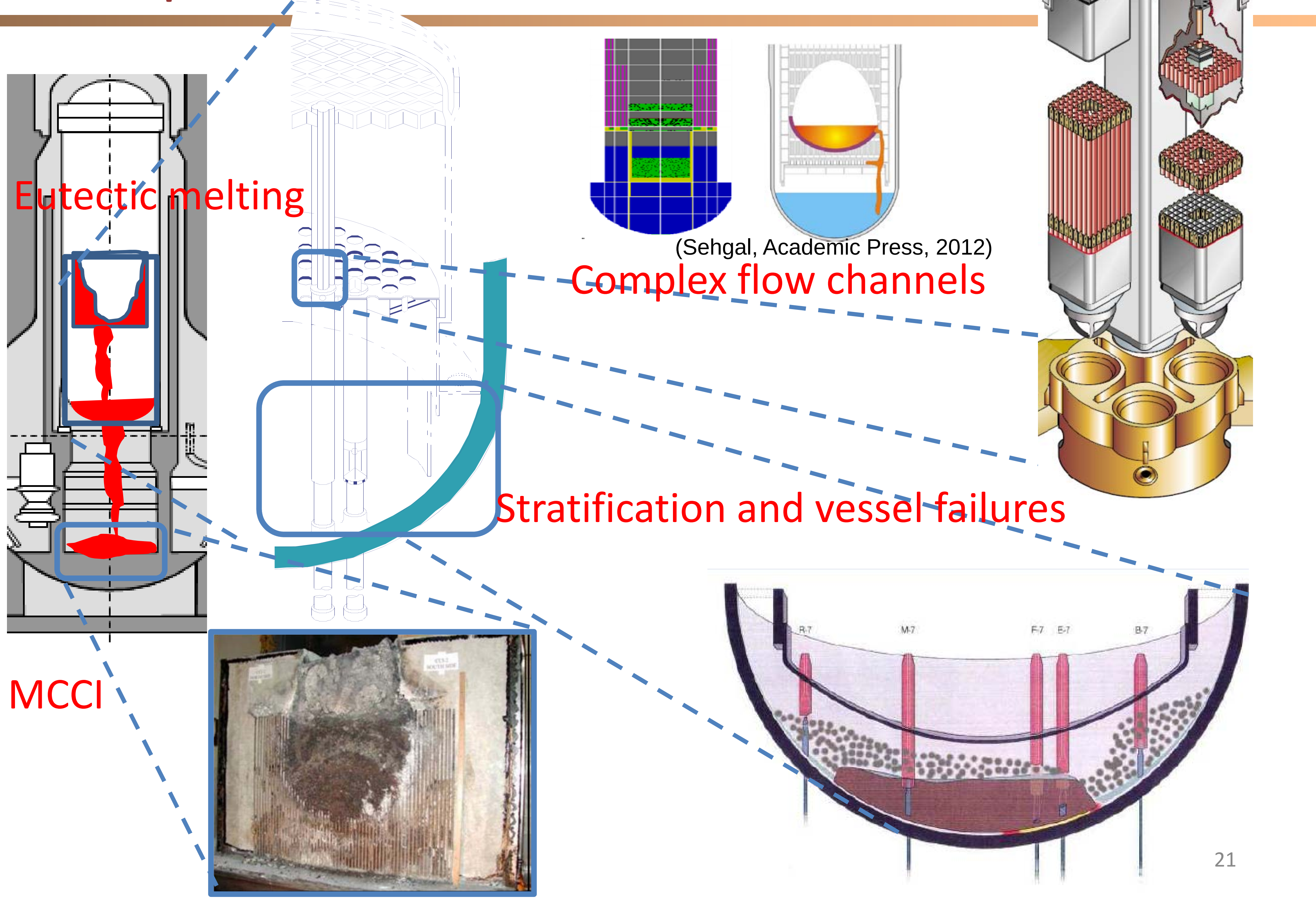
	Depressurization scenario	RPV failure scenario
Case1	Stuck-open of SRV	Instrument tubes failure
Case2	Stuck-open of SRV	Creep rupture of vessel wall
Case3	Gasket leakage of SRV	Instrument tubes failure
Case4	Gasket leakage of SRV	Creep rupture of vessel wall

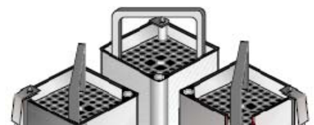






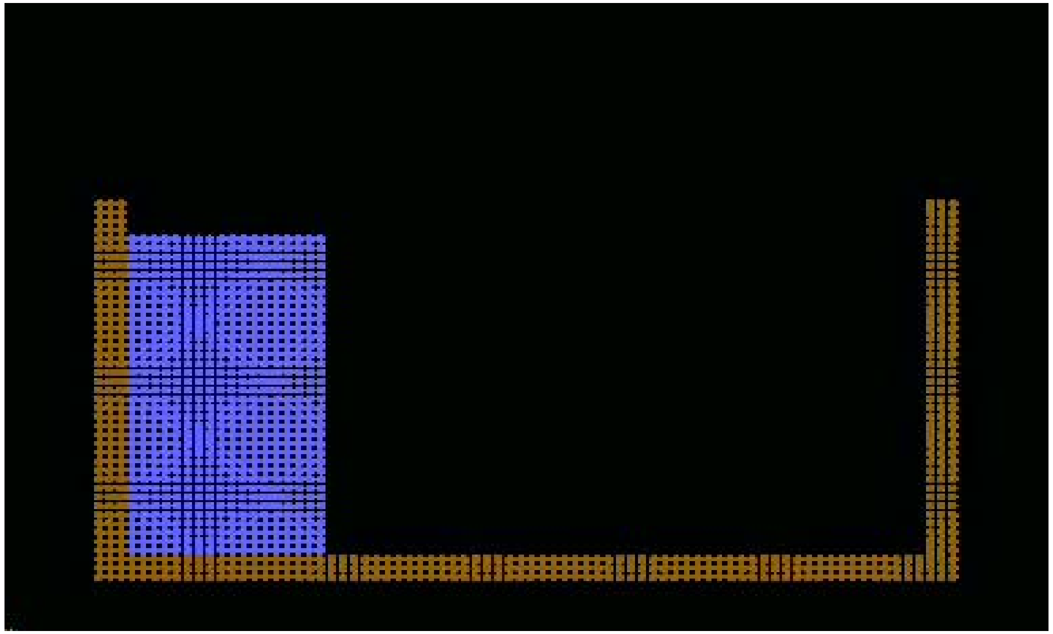
Complex Phenomena

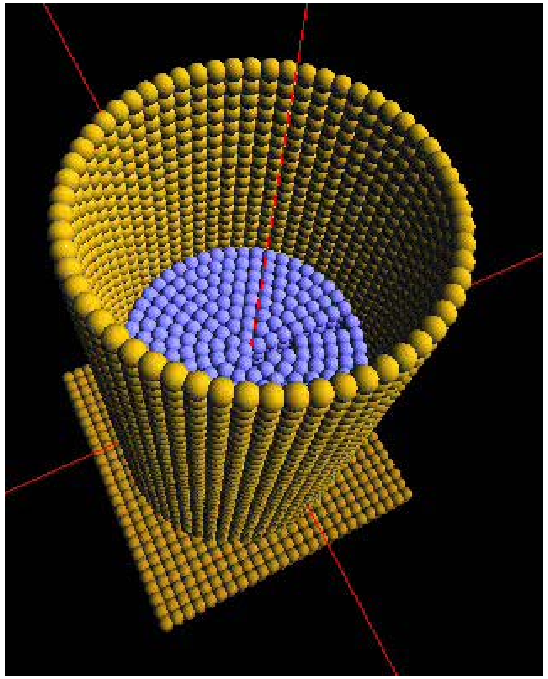




Moving Particle Semi-implicit (MPS) Method

- Developed at Oka-Koshizuka Lab. in 1990s (Univ. of Tokyo)
- Based on Lagrangian method
- No grids or meshes (c.f., Eulerian method)
- Incompressible flows can be simulated without empirical correlations and scaling models (conservations of mass, energy, momentum)





The MPS code of the present study was developed based on MPS-SW-MAIN-Ver.2.0 which was kindly provided by S. Koshizuka and K. Shibata(University of Tokyo).

S. Koshizuka, Y. Oka, “Moving –particle semi-implicit method for fragmentation of incompressible fluid,” Nuclear Science and Engineering., 123, 421-434 (1996).

Discretization of Governing Equations

Governing Equations

Mass conservation

$$\frac{D\rho}{Dt} + \rho \nabla \cdot \mathbf{u} = 0$$

Momentum conservation

$$\frac{D\mathbf{u}}{Dt} = -\frac{1}{\rho}\nabla P + \nu \nabla^2 \mathbf{u} + \frac{1}{\rho}\mathbf{f}$$

Energy conservations

$$\frac{Dh}{Dt} = k\nabla^2 T + S$$

Particle interaction models

Divergence

$$\nabla \bullet \mathbf{u} = \left(\frac{\partial}{\partial x}\mathbf{i} + \frac{\partial}{\partial y}\mathbf{j} + \frac{\partial}{\partial z}\mathbf{k}\right) \bullet (u\mathbf{i} + v\mathbf{j} + w\mathbf{k})$$

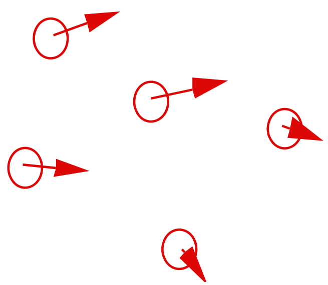
Gradient

$$\nabla \times f = \frac{\partial f}{\partial x}\mathbf{i} + \frac{\partial f}{\partial y}\mathbf{j} + \frac{\partial f}{\partial z}\mathbf{k}$$

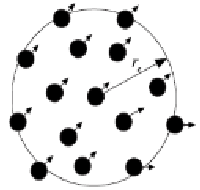
Laplacian

$$\triangle = \frac{\partial^2}{\partial x^2} + \frac{\partial^2}{\partial y^2} + \frac{\partial^2}{\partial z^2}$$

Particle interactions

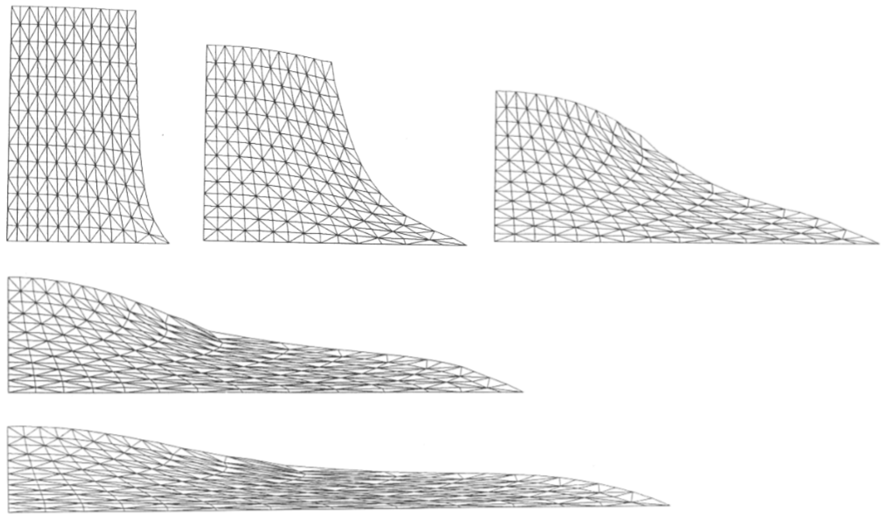


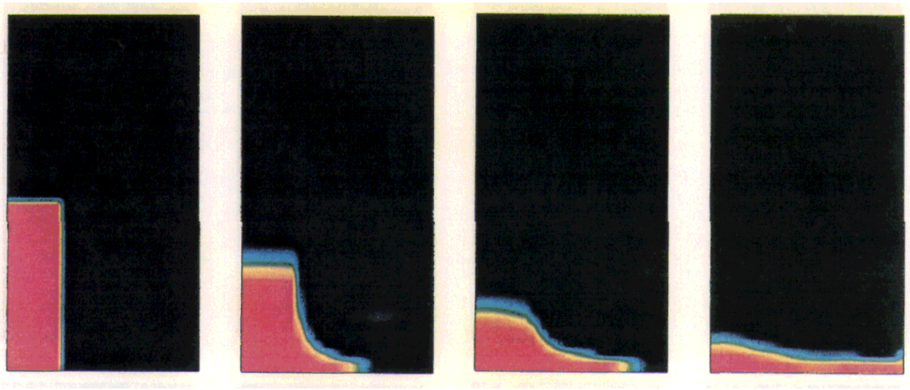
Weight function



$$w(r) = \begin{cases} \frac{r_e}{r} - 1 & (0 \le r < r_e) \\ 0 & (r_e \le r) \end{cases}$$

Tracking Free Surfaces

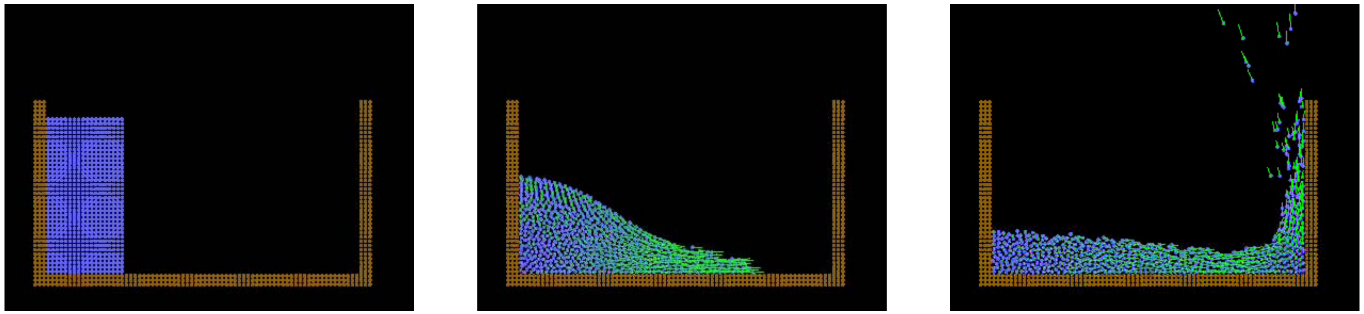




Volume-Of-Fluid (VOF) method

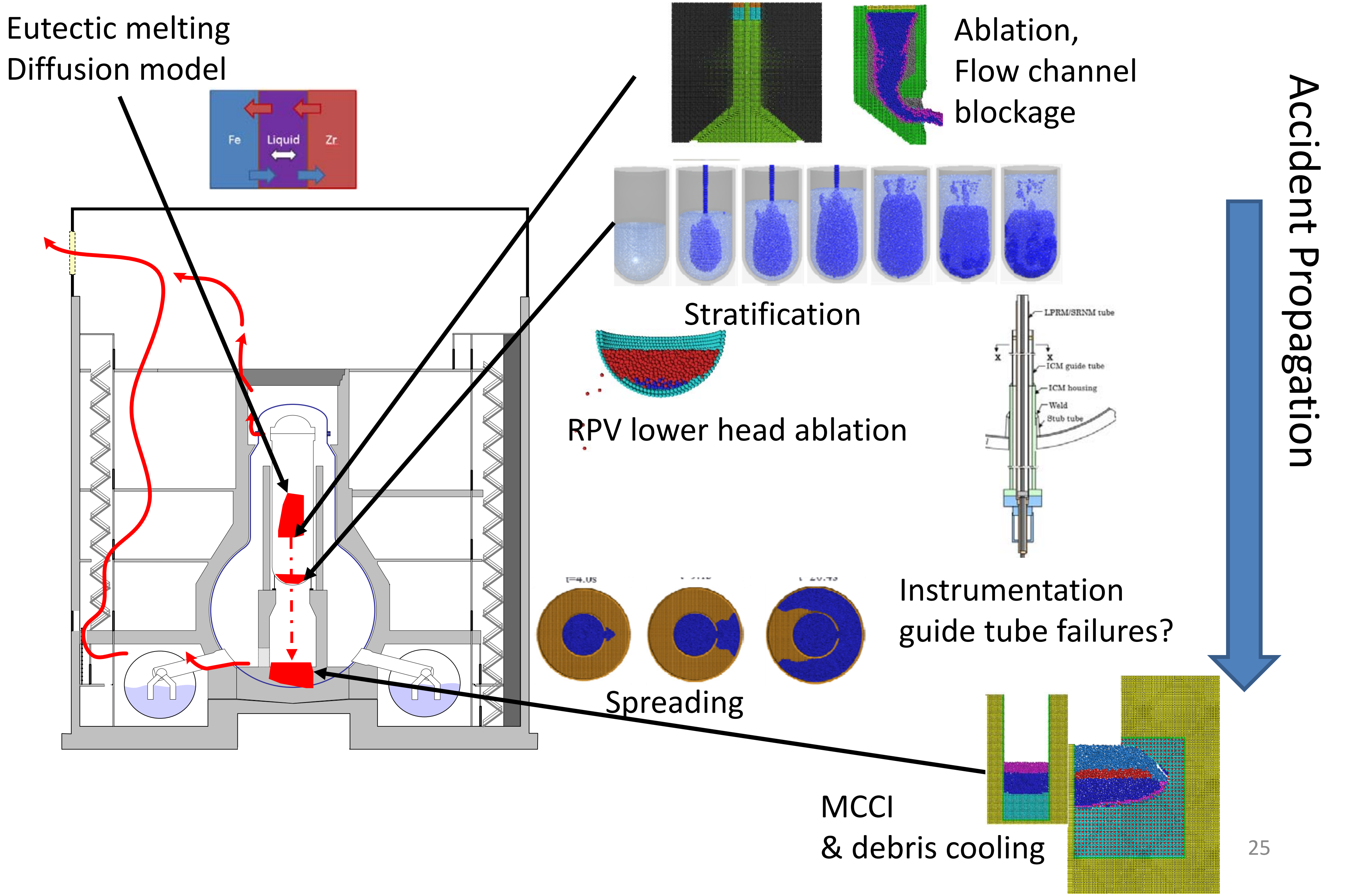
Finite Element Method (FEM)
Mesh distortion

Numerical divergence at boundaries

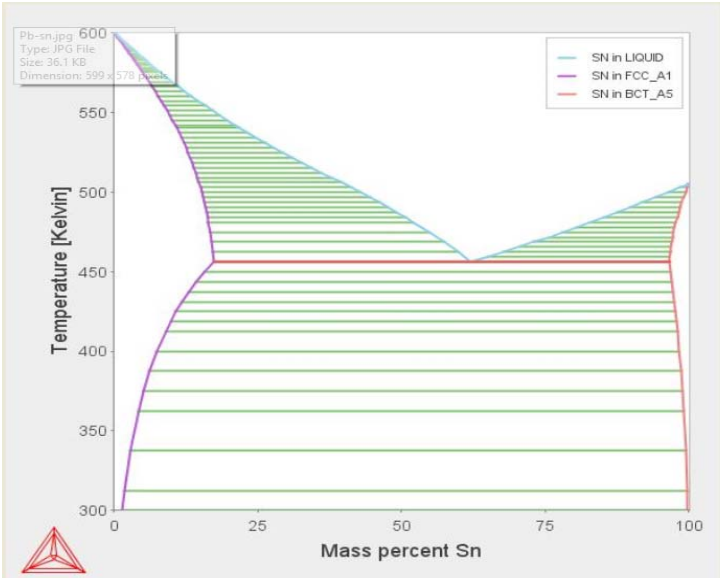


MPS method：No meshes needed

Applications of MPS Method at Waseda University



Eutectic reaction



- Eutectic system: the lowest possible melting temperature over all of the mixing ratios for the involved component species.
- Eutectic system (severe accident): melting temperature of material below its melting point at particular mixing ratio for the involved component species.

Eutectic Melting Validation

By Dr. Asril Pramutadi Andi Mustari

- TREAT experiment
- Fe(alloy)-U system

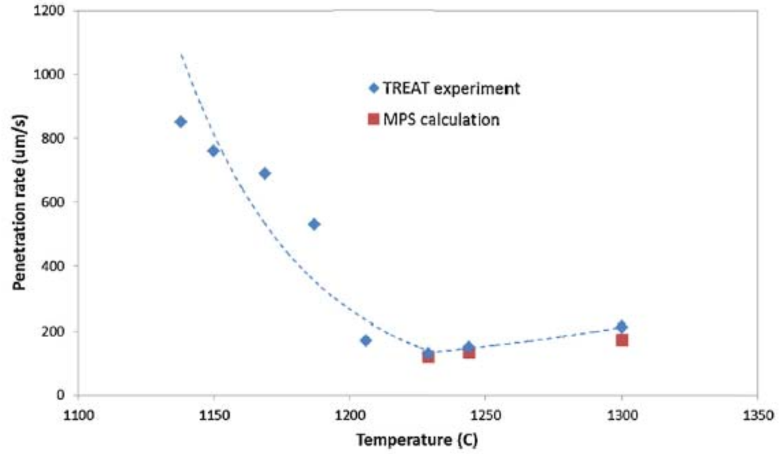


Fig. 10. Comparison of penetration rate of Armco iron by uranium between experiment and MPS simulation.

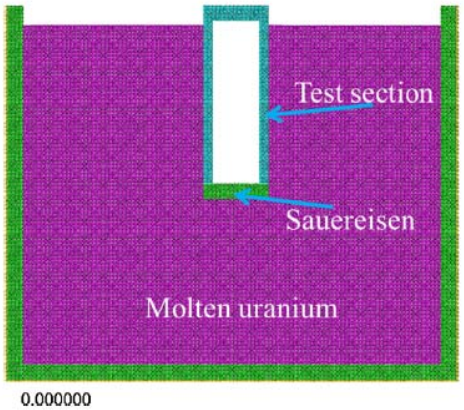


Fig. 7. Two-dimensional calculation model of the MPS calculation.

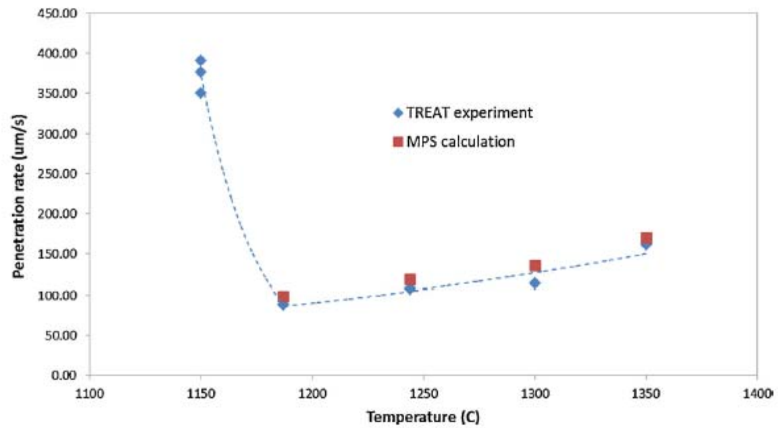
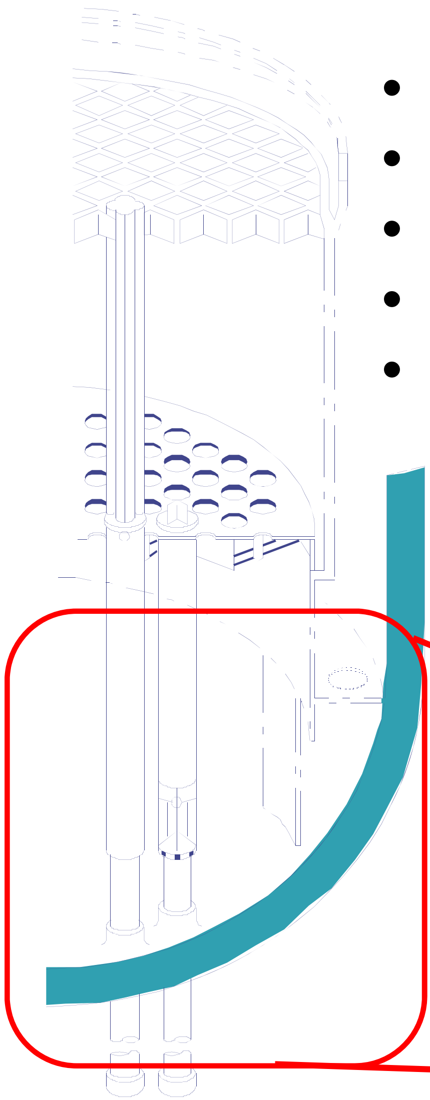


Fig. 11. Comparison of penetration rate of SS 304 by uranium between experiment and MPS simulation.

Nuclear Engineering and Design 278 (2014) 387–394

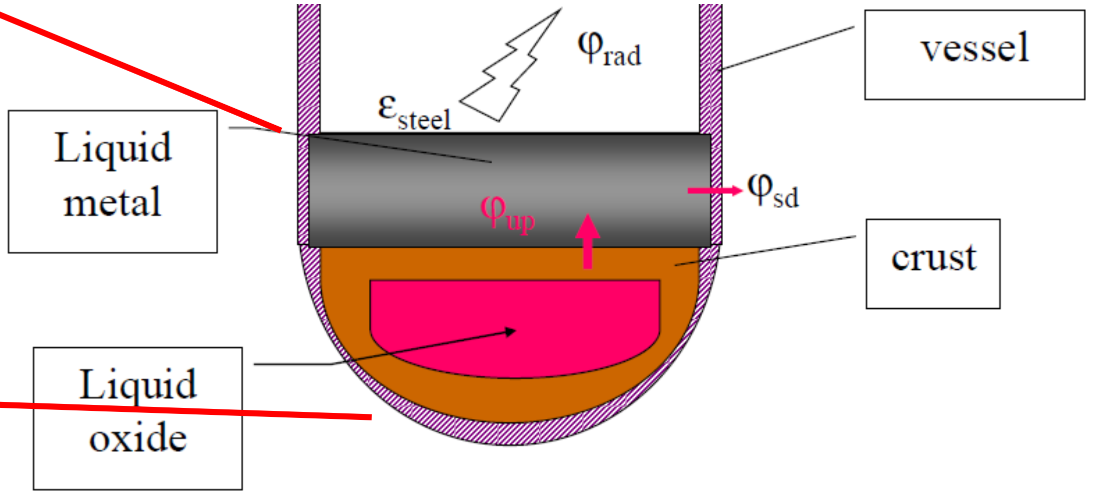
A part of this study is the result of "Mechanistic study of melt behavior in lower RPV head" carried out under the Strategic Promotion Program for Basic Nuclear Research by the Ministry of Education, Culture, Sports, Science and Technology of Japan.

RPV Lower Head Failure

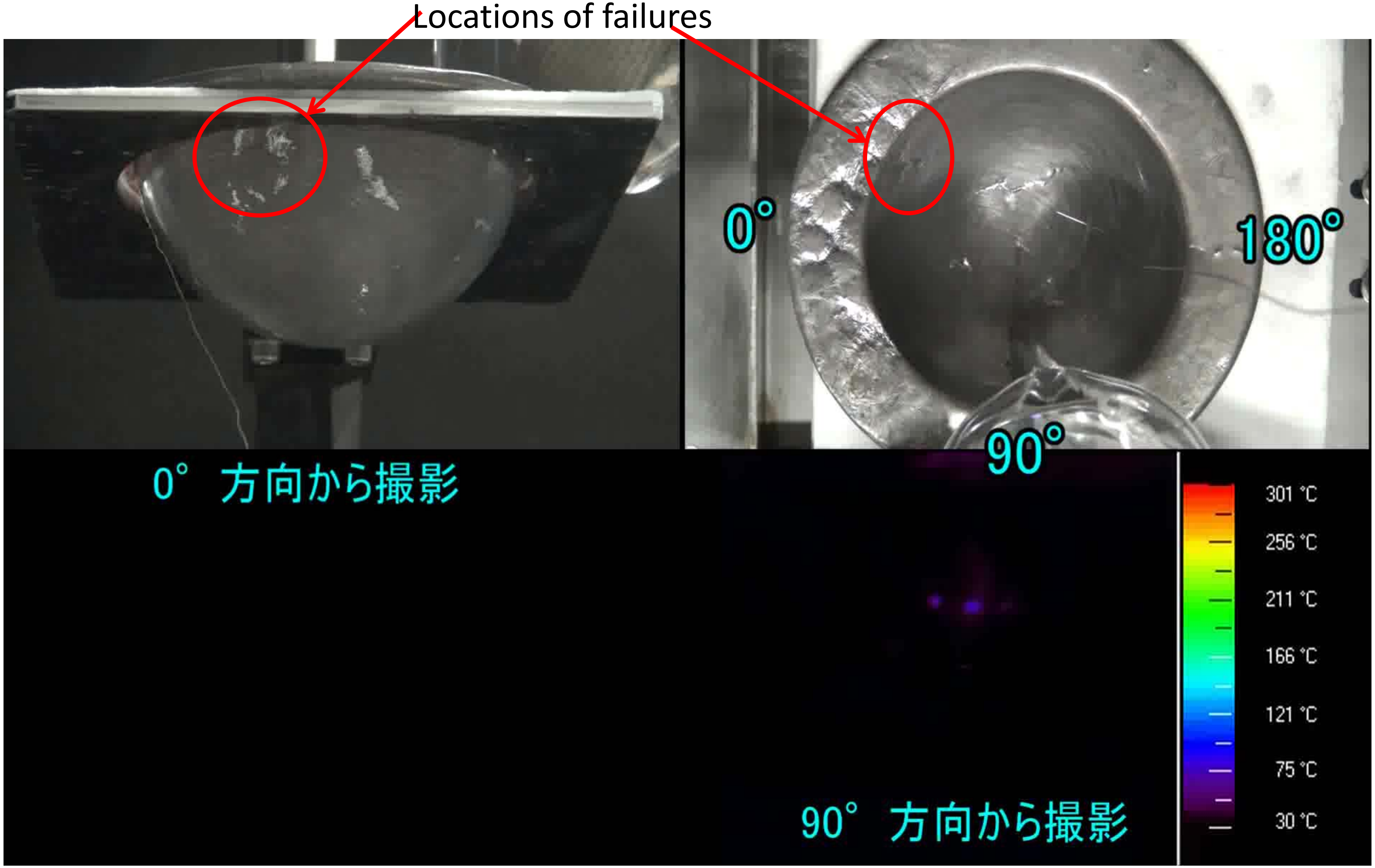


- Convection
- Stratification
- Focusing effect
- Vessel wall ablation
- Vessel wall creep

FOREVER(1/10 scale experiment)
Wolfgang Luther, "Late In Vessel Phenomina", SARnet Short Course on Severe Accident Phenomenology, 2011.

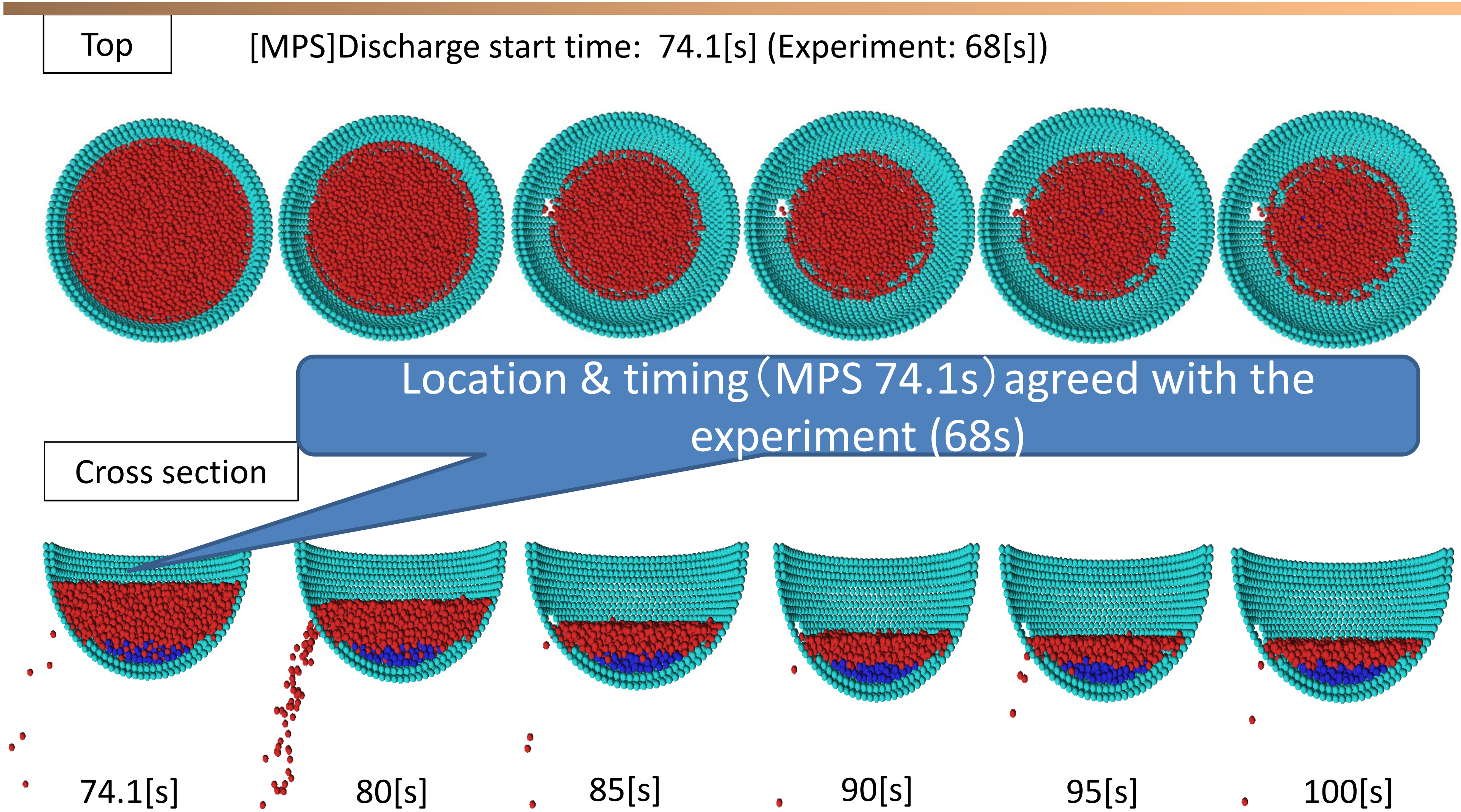


Pb-Bi Vessel Ablation



Experiment conducted at Central Research Institute of Electric Power Industry (CRIEPI) Masahiro Furuya et al.,
A part of this study is the result of “Mechanistic study of melt behavior in lower RPV head” carried out under the Strategic Promotion Program for Basic Nuclear Research by the Ministry of Education, Culture, Sports, Science and Technology of Japan.

MPS Simulation for Pb-Bi Vessel Ablation



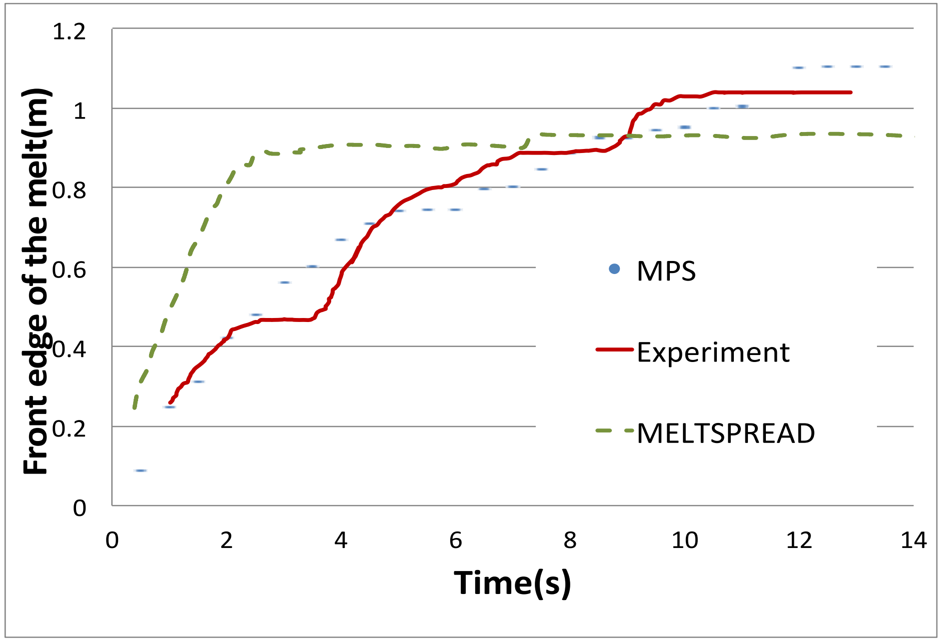
D. Masumura et al., "Analysis of Metal Vessel Wall Ablation Experiment with High Temperature Liquid by MPS Method," *Proc.* NURETH-16, Chicago, USA, August 30-September 4, 2015

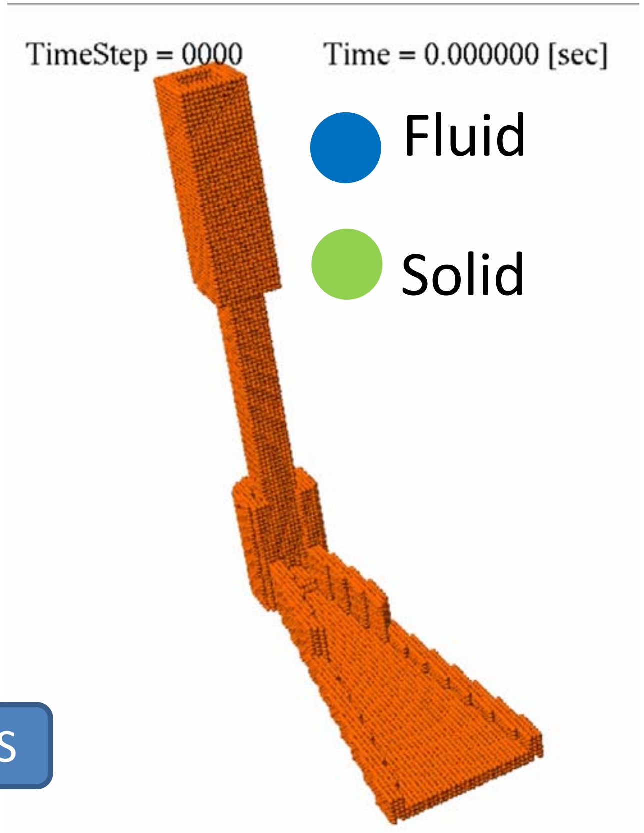
A part of this study is the result of "Mechanistic study of melt behavior in lower RPV head" carried out under the Strategic Promotion Program for Basic Nuclear Research by the Ministry of Education, Culture, Sports, Science and Technology of Japan.

Spreading- Analysis of FARO L-26S

FARO: Karlsruhe in Germany in 1997

	Melt	Floor
Material	UO_2-ZrO_2 （8 ： 2）	Stainless steel
Initial temperature [K]	2950	290
Solidus – liquidus T [K]	2860 - 2910	



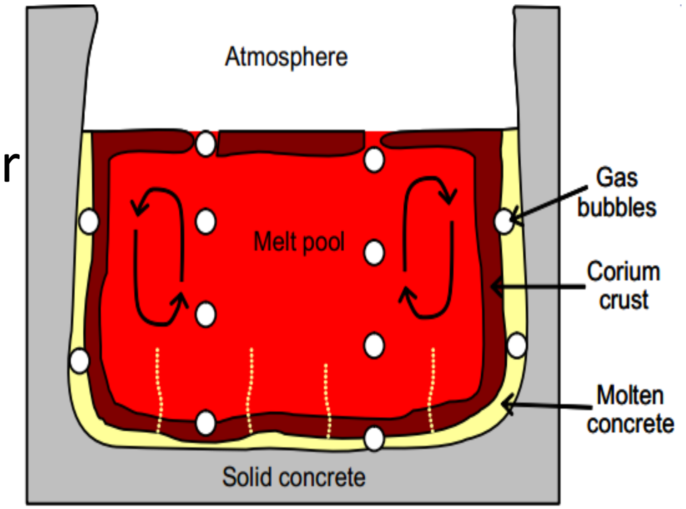


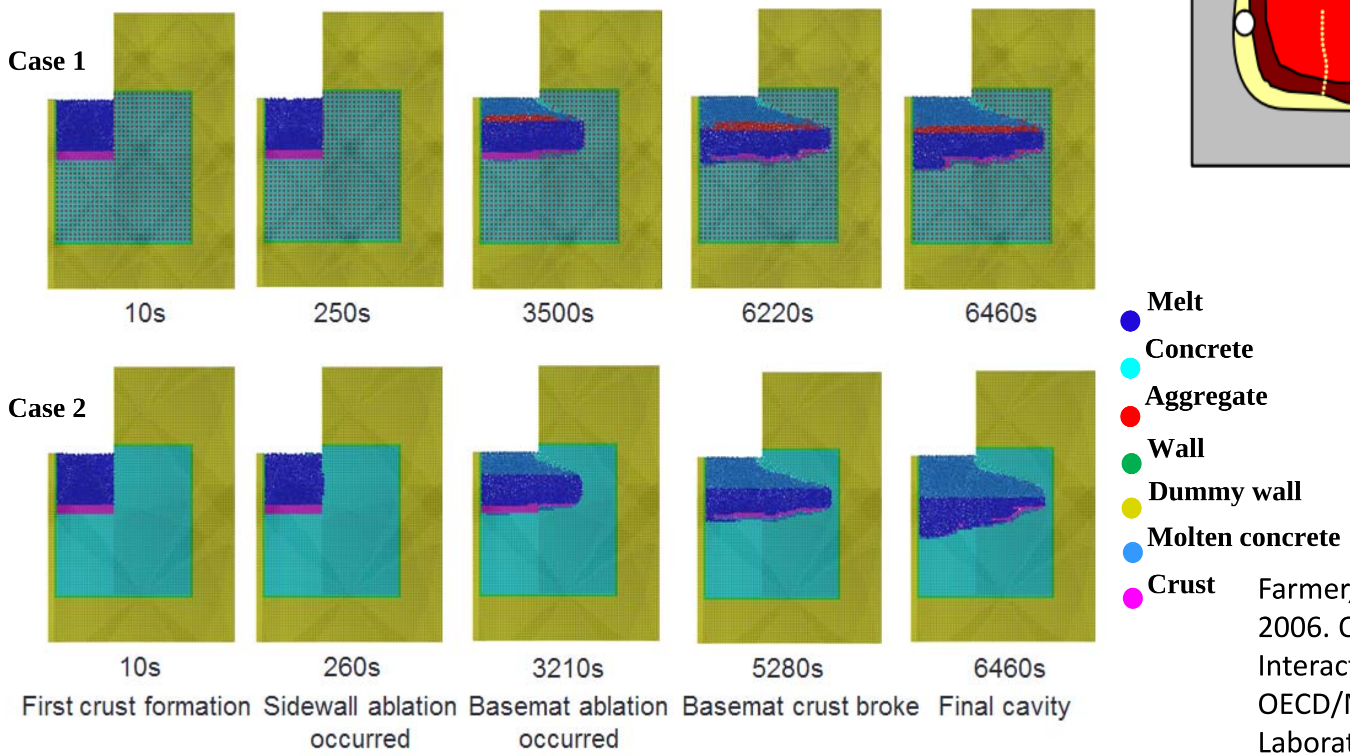
Stepwise profile of the front edge is well reproduced by MPS

A. Yamaji and X. Li, "Development of MPS Method for Analyzing Melt Spreading Behavior and MCCI in Severe Accidents," *Proc.* 6th APS, Bandung, Indonesia, 19-20 August (2015).

Anisotropic Ablation during MCCI

- Effect of Zr oxidation (chemical heat)
- When and how the molten corium will melt through the basemat (anisotropic ablation), or
- Termination of MCCI (top flooding, bottom flooding, sacrificial materials)





Farmer, M.T., Lomperski, S., Kilsdonk, D.J., et al., 2006. OECD MCCI Project 2-D Core Concrete Interaction (CCI) Tests: Final Report. OECD/MCCI-2005-TR05, Argonne National Laboratory, Argonne, USA.

X. Li and A. Yamaji "Numerical Simulation of Anisotropic Ablation of Siliceous Concrete – Analysis of CCI-3 MCCI Experiment by MPS Method," *Proc.* NURETH-16, August 30-September 4, Chicago, USA (2015)

Potential Subjects for Doctor Course Students

- RPV failure
  - Creep rupture and ablation
  - Corium composition
  - Effect of penetration tubes
- MCCI
  - Top flooding
  - Bottom flooding
  - Gas generation and mixing phenomena, stratification of metal and oxide layers
  - Crust breach
  - Scaling to real plant
- Joint research with Japan Atomic Energy Agency (JAEA)
  - Fukushima related projects

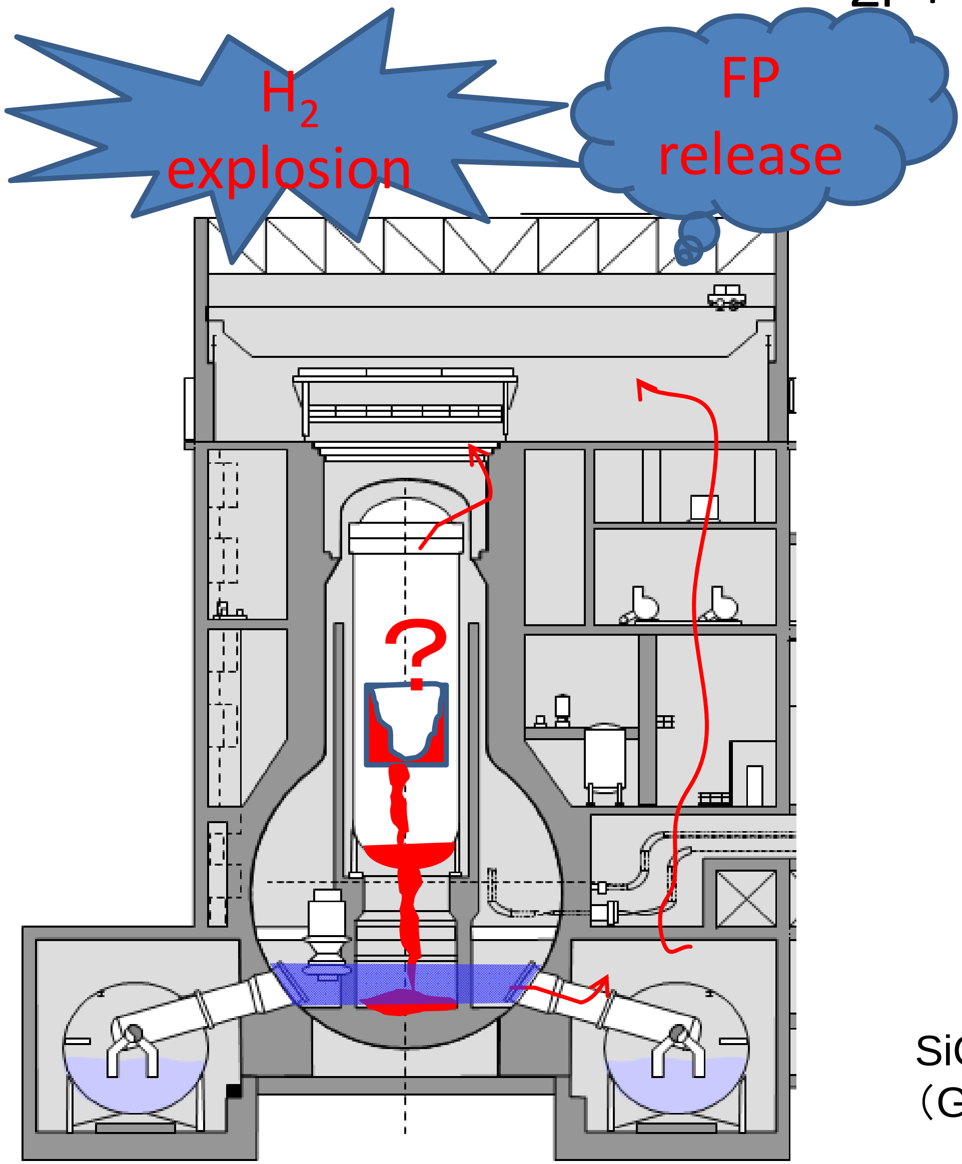
Accident Tolerant Fuel

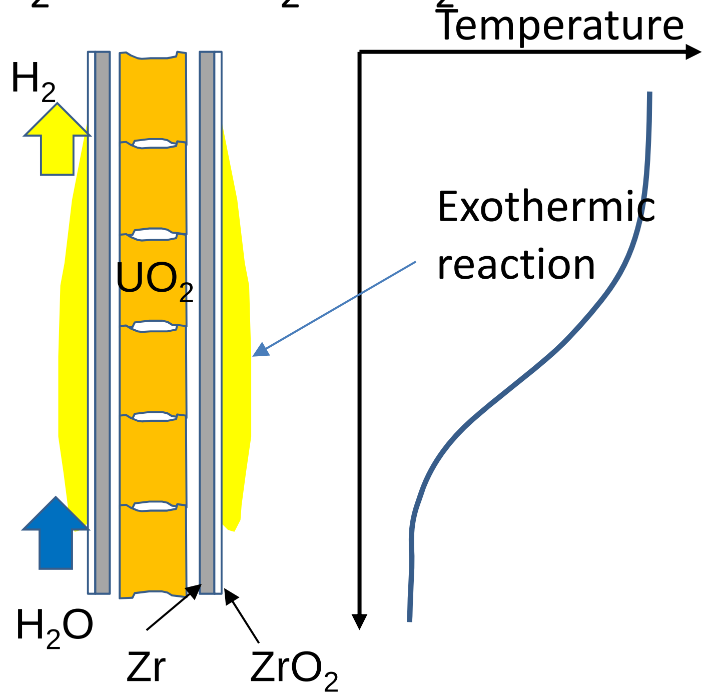
FEMAXI-7

ATF

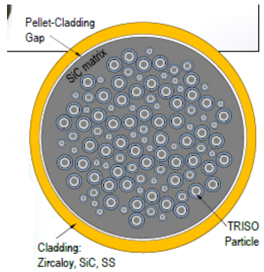
What is Accident Tolerant Fuel?

$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 + 5.8 \times 10^8 \text{ J/kg}$



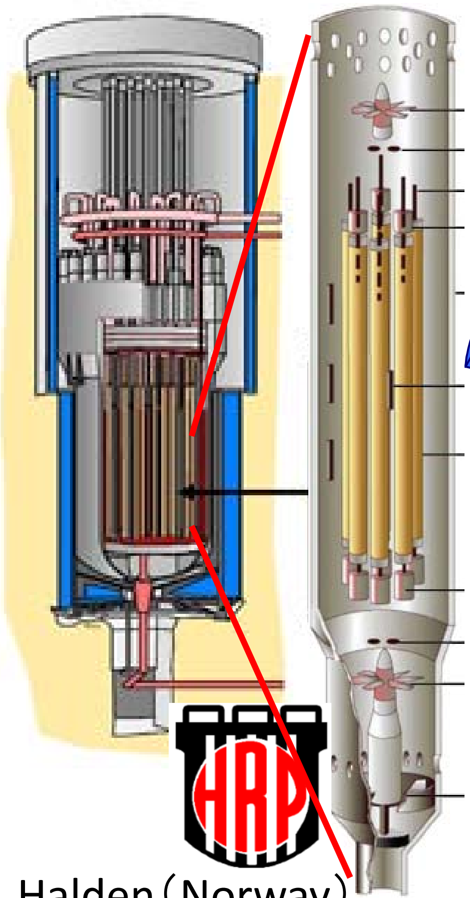


SiC/SiC cladding (General Atomic)

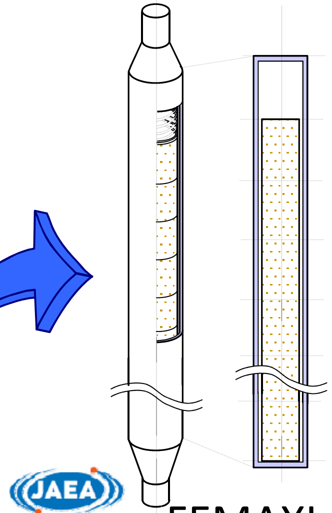


Microencapsulated fuels (ORNL)

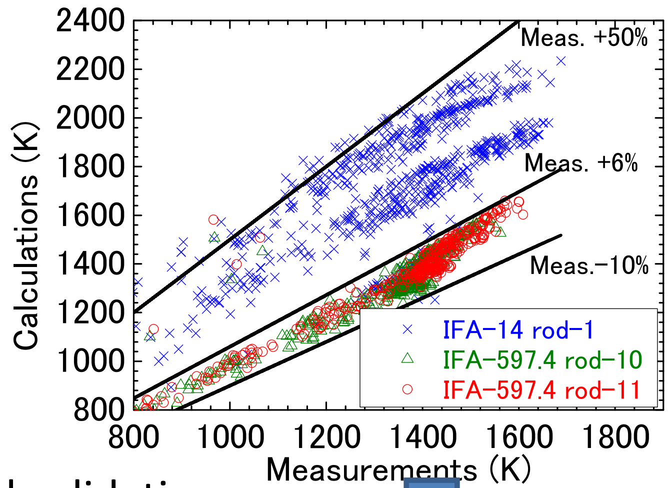
Validating FEMAXI-7 and Evaluating ATF Performance



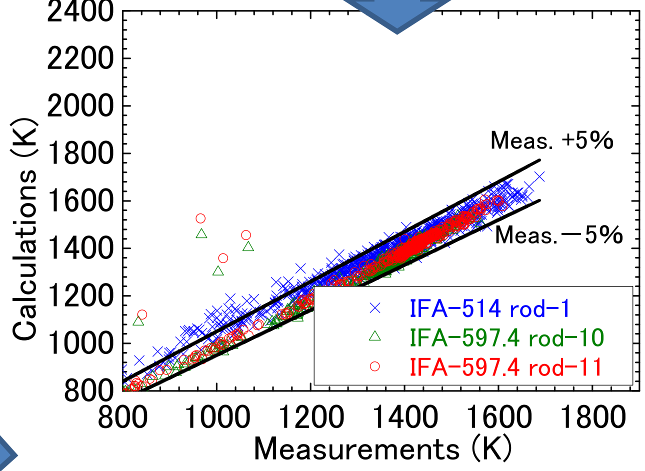
Halden (Norway)



FEMAXI-7



Model validations



Fuel performance evaluation of ATF under

- Normal operation
- Anticipated transient
- Accident