원자력계통 모델 및 시뮬레이션 연구실
Nuclear System Model & Simulation Lab

2022. 2.

정재준
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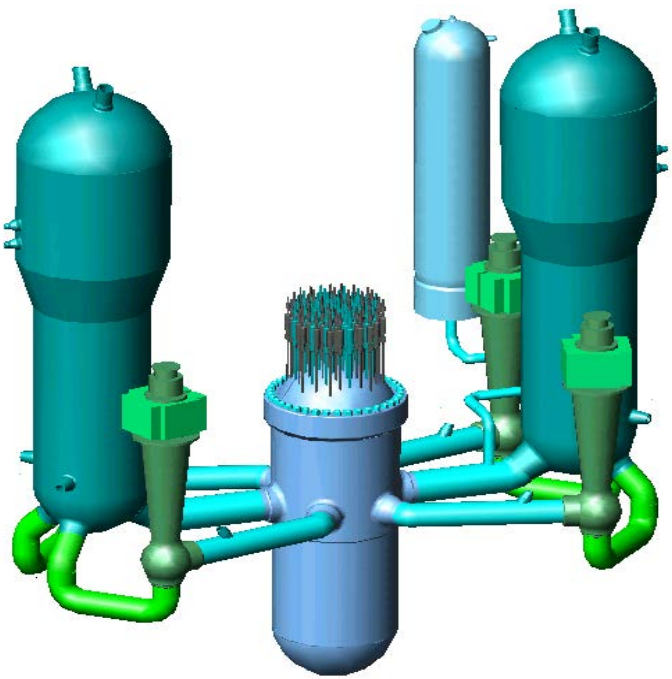
원자력발전소의 열수력학적 특성

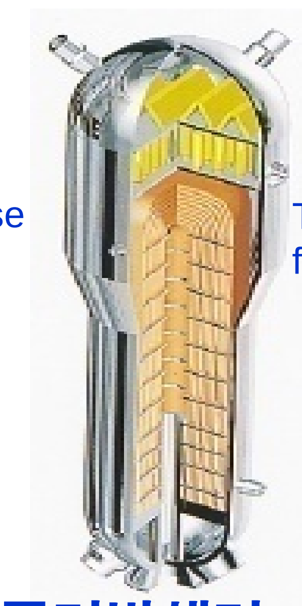
- 대형 시스템, 복잡한 내부 구조
- 단상 및 2상 유동
- 다분야 연계: 노심동특성, 유체역학, 열전달, 제어, 수화학



OPR1000 (2775 MWt PWR)

Single-phase flow

원자로



Two-phase flow

증기발생기

http://www.doosanheavy.com/

원자력 안전

- 공공과 환경을 방사선 위해(Ionizing radiation hazard)로부터 보호
- 핵연료 건전성 확보 – 지속적이고 충분한 냉각
- 다양한 가상적 사고(설계기준사고)에 대비함
- 원자력 시스템 코드(계통 열수력 분석 코드)로 평가

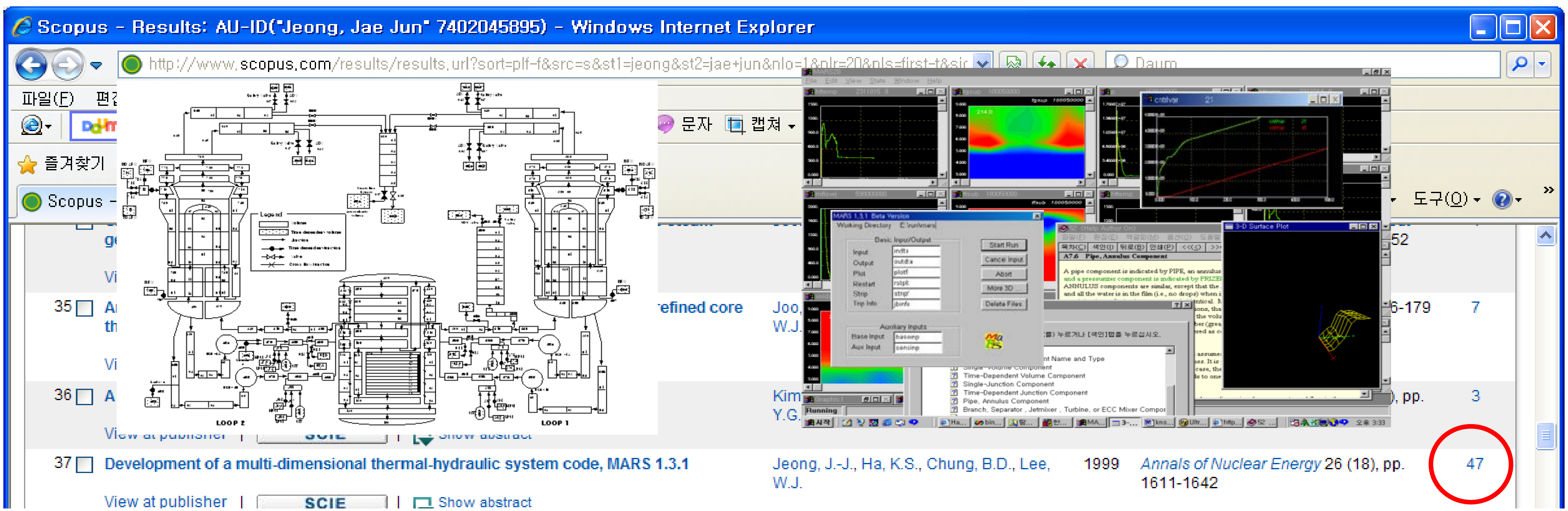
주요 연구분야

- **원자력발전소 안전성 평가**
- **원자력계통 해석**
  - 원자력 계통 열수력해석 코드 및 시뮬레이터
  - 원자로 기기스케일 열수력 분석코드
- **원전 안전계통 개발**
- **사용후핵연료 건식저장 열해석**

- **다차원 다상유동 및 열전달 모델링·수치해석**
- **시스템에어컨 해석**

열수력 코드 개발(계통/기기)

- "COBRA/RELAP5; A Merged Version of the COBRA-TF and RELAP5/MOD3 Codes," *Nuclear Technology*, vol. 99, pp. 177-187 (1992).
- "Improvement and Assessment of the CATHARE2 Three-dimensional Module compared with the UPTF Downcomer Test 7," *Nuclear Technology*, vol. 117, pp. 267-280 (1997).
- "Development of A Multi-dimensional Thermal-Hydraulic System Code, MARS 1.3.1," *Annals of Nuclear Energy*, vol. 26, no. 18, pp. 1611-1642 (1999).
- "The CUPID Code Development and Assessment Strategy," *Nuclear Engineering and Technology*, vol. 42, 636-655 (2010).
- Recent Improvements in the CUPID Code for a Multi-Dimensional Two-Phase Flow Analysis of Nuclear Reactor Components, Nuclear Engineering and Technology, Vol. 46, No.5, pp.655-666 (2014).



원자력계통 안전성 평가

- "Hot Channel Analysis Capability of the Best-Estimate Multi-Dimensional System Code, MARS 3.0," *Nuclear Engineering and Technology*, Vol. 37 No.5 pp. 469-478 (2005).
- "Simulation of A Main Steam Line Break Accident Using A Coupled ‘System Thermal-Hydraulics, Three-dimensional Reactor Kinetics, and Hot Channel’ Analysis Code, *Annals of Nuclear Energy*, Vol. 33, pp. 820-828 (2006).
- "A Coupled Analysis of System Thermal-Hydraulics and Three-Dimensional Reactor Kinetics for a 12-finger Control Element Assembly Drop Event in a PWR Plant," *Annals of Nuclear Energy*, **37**, 1580-1587 (2010).
- Transient Hydraulic Response of a Pressurized Water Reactor Steam Generator to a Feedwater Line Break Using the Nonflashing Liquid Flow Model, Journal of Pressure Vessel Technology, Vol. 139 / 031302-1 ~ 9 (2017).
- Assessment of the MARS Code Using the Two-Phase Natural Circulation Experiments at a Core Catcher Test Facility, Science and Technology of Nuclear Installations, Volume 2017, Article ID 5731420 (2017).

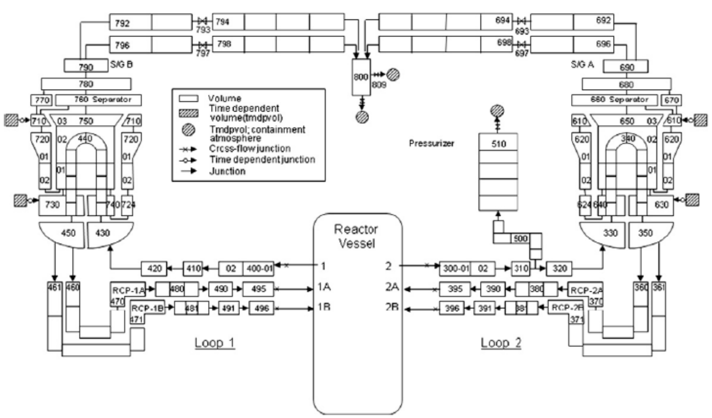


Fig. 3. The MARS nodalization for the OPR1000 plant.

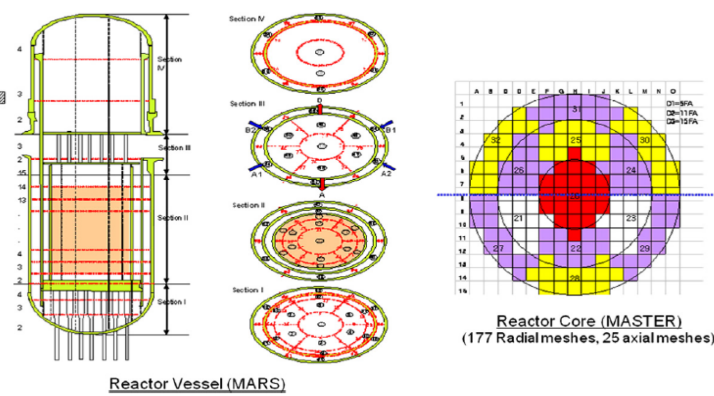
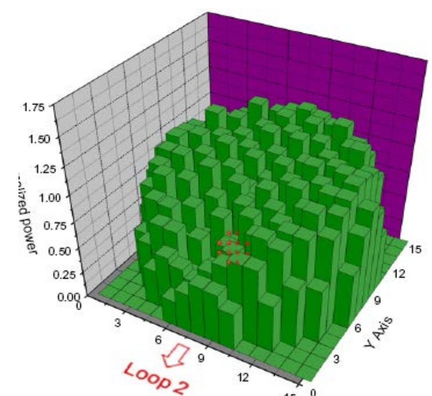


Fig. 4. Nodal diagrams of the reactor vessel for MARS and the core for MASTER.

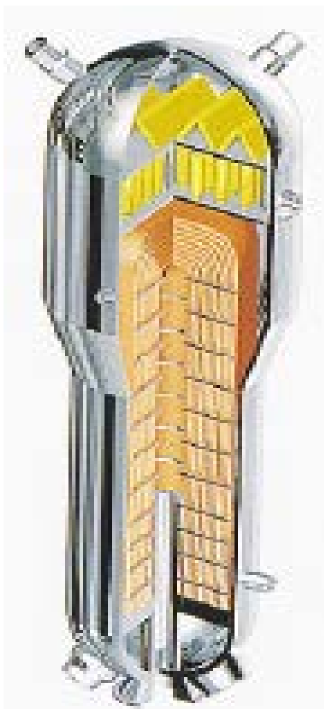


(b) At 4.2 s.

ıtions at 0 s and 4.2 s.

열수력 현상 모델링

- "Non-uniform Flow Distribution in the Steam Generator U-Tubes of A Pressurized Water Reactor Plant During Single- and Two-Phase Natural Circulations," *Nuclear Engineering and Design*, vol. 231(3), pp. 303-314 (2004).
- Development of an empirical correlation for the onset of flow instability in narrow rectangular channels, Nuclear Engineering and Design, 375, 111090 (2021).
- Improvement of the heat transfer enhancement model considering the droplet-wall heat transfer downstream of the flow blockage in the reflood phase, Nuclear Engineering and Design, 380 (2021) 111277.
- Application of the machine learning technique for the development of a condensation heat transfer model for a passive containment cooling system, Accepted for Publication in Nuclear Engineering and Technology (Dec. 2021).



U-tube Steam Generator (UTSG)
http://www.doosanheavy.com/

Flow excursion in a U-tube occurs

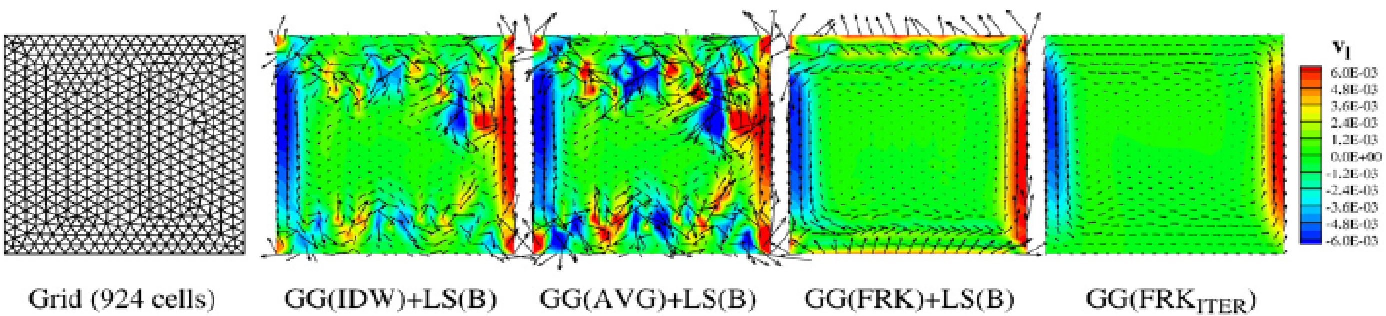
if $\boxed{|\dot{m}| < \dot{m}_C}$**,**

where

$$\dot{m}_C = \frac{\pi D_i^4 g \bar{\rho}_{1\phi} \rho_{f,sat}^{\sec} \left\{ \frac{\beta C_p \Delta T_{2\phi}}{U_{1\phi}} + \frac{h_{fg}}{U_{2\phi}\Delta T_{2\phi}} \left[x_i - \frac{v_{f,sat}^{\sec}}{v_{fg}} \log\left(1 + \frac{v_{fg}}{v_f} x_i\right)\right]\right\}}{16 D_o \left(fL/D_i + K \right)}$$

수치해법 개발

- "A Semi-Implicit Numerical Scheme for Transient Two-Phase Flows on Unstructured Grids," *Nuclear Engineering and Design*, Vol. 238, 3403–3412 (2008).
- "Numerical Effects of the Semi-conservative Form of Momentum Equations for Multi-Dimensional Two-Phase Flows," *Nuclear Engineering and Design, vol.* 239, pp. 2365–2371 (2009).
- "A Continuity Based Semi-Implicit Scheme For Transient Two-Phase Flows," *Journal of Nuclear Science and Technology,* vol. 47, No. 9, pp.779-789 (2010).
- "An Improved Numerical Scheme to Evaluate the Pressure Gradient on Unstructured Meshes for Two-Phase Flow Analysis," *International Communications in Heat and Mass Transfer,* 37, 1273–1279 (2010).
- Numerical Investigation of the CANDU Moderator Thermal-Hydraulics using the CUPID Code, Progress in Nuclear Energy, vol. 85, pp. 541-547 (2015).



(c) Velocity distributions: unstructured grid

Robustness

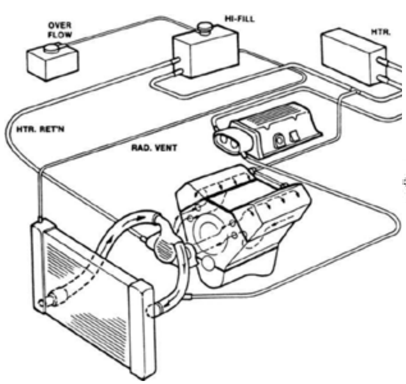
Conservation

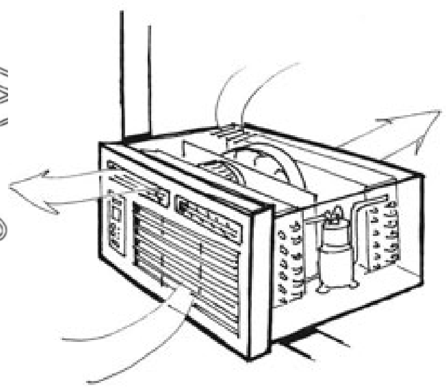
Accuracy

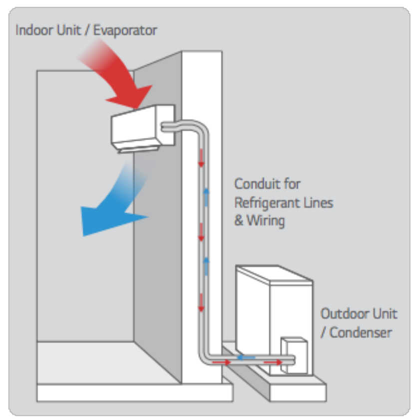
시스템 에어컨 시뮬레이션

- Many thermo-hydraulic experiments of refrigerants aim to depict various operating conditions of air-conditioner and refrigerator.
- However, the experiments for the cycle analysis are not cost-effective expensive.
- It is necessary to develop a program for analysis of the cooling cycles for design and optimum control.
- Most programs can simulate the cycles in a simplified form and can not accurately calculate two-phase flows in the cycle.
- We are trying to analyze the refrigeration cycle by using the MARS code.

- Preliminary assessment of the nuclear thermal–hydraulic system code MARS for the application to a refrigeration cycle, Nuclear Engineering and Design 367 (2020) 110798

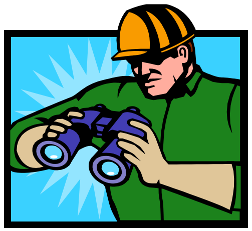




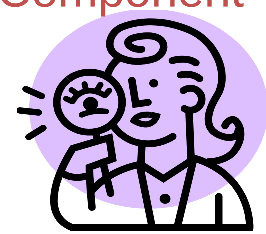


다중스케일 열수력 해석체계 개발

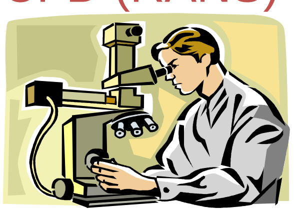
System

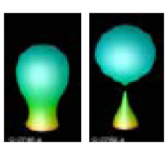


Component



CFD (RANS)





DNS
(Direct Numerical Simulation)

Macro (System) scale

Boundary conditions

Meso (Component) scale

Local flow conditions

Closure laws

Micro (Turbulent) scale

[G. Yadigaroglu, NED, 2005]

$$\frac{\partial}{\partial t}(\alpha_k \rho_k) + \nabla \cdot (\alpha_k \rho_k \underline{u}_k) = \Omega_k$$

$$\frac{\partial}{\partial t}(\alpha_k \rho_k \underline{u}_k) + \nabla \cdot (\alpha_k \rho_k \underline{u}_k \underline{u}_k) = -\alpha_k \nabla P + \nabla \cdot [\alpha_k (\tau_k + \tau_k^T)] + \alpha_k \rho_k \underline{g} + u_{ki}\Omega_k + M_{ik} + M_{ik}^{ND} - \nabla \alpha_k \cdot \tau_{ki}$$

$$\frac{\partial}{\partial t}(\alpha_k \rho_k e_k) + \nabla \cdot (\alpha_k \rho_k e_k \underline{u}_k) = -\nabla \cdot [\alpha_k (q_k + q_k^T)] - P \frac{\partial}{\partial t}\alpha_k - P \nabla \cdot (\alpha_k \underline{u}_k) + \Phi_k^T + \Phi_k^{\mu} + Q_{ik} + \Gamma_k h_{ki} + M_{ik} \cdot (\underline{u}_{ki} - \underline{u}_k) + \nabla \alpha_k \cdot \tau_k \cdot (\underline{u}_{ki} - \underline{u}_k) + q_{wk}$$

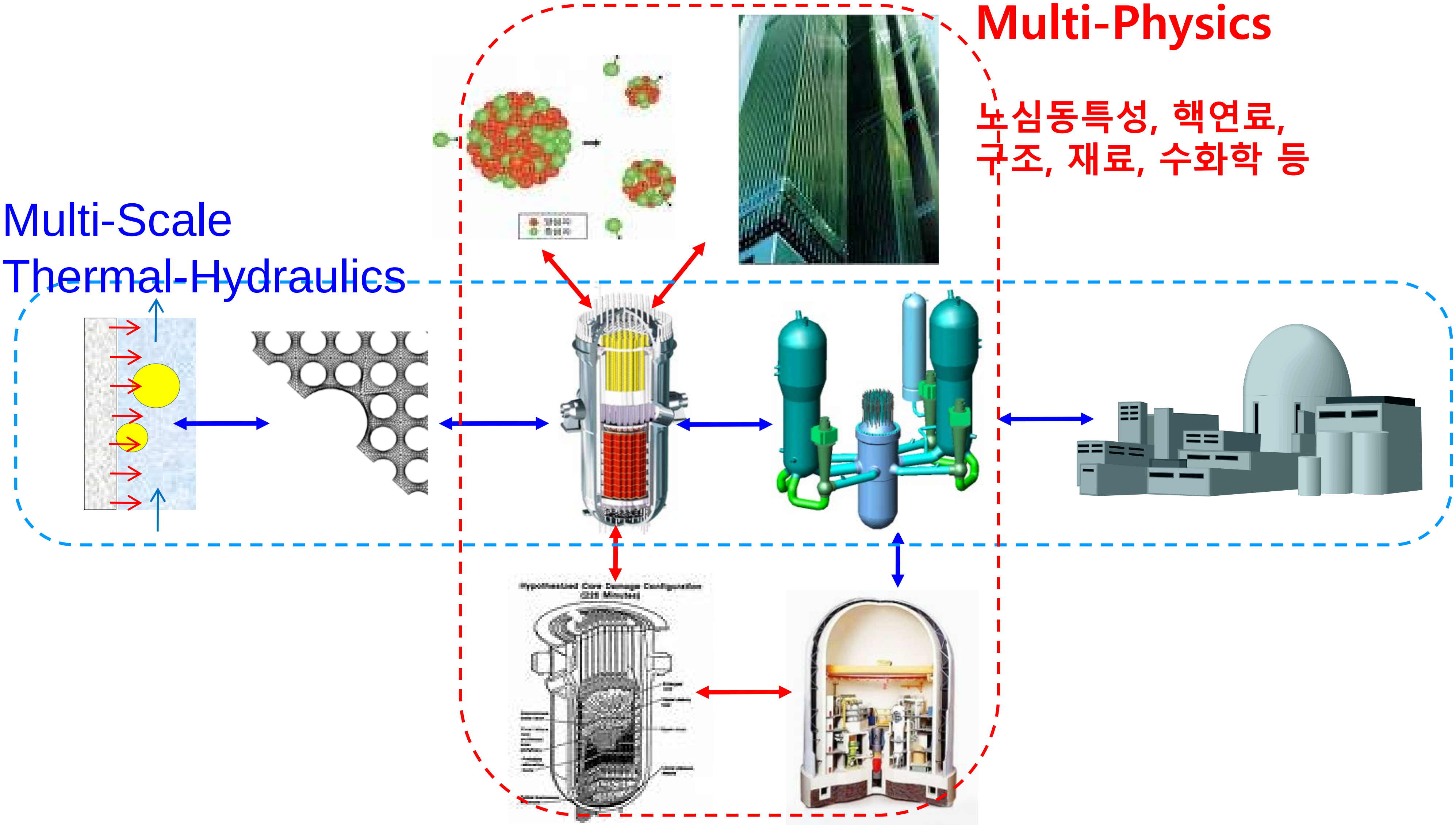
PTS
(Pressurized Thermal Schock)

DNB
(Departure from Nucleate Boiling)

LOCA
(Loss of Coolant Accident)

…

다분야 통합 원자로계통 해석체계 개발



재학생 및 졸업생 현황

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Thank you!