LARGE-EDDY SIMULATION OF THERMAL STRIPING IN THE UPPER INTERNAL STRUCTURE OF THE PROTOTYPE GEN-IV SODIUM-COOLED FAST REACTOR

Detailed modelling and simulation with optimal flow region and integrated simulation with component simplification

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00 Contents

01 Introduction

Benefits and challenges for thermal-hydraulics in fast SMRs

- Many SMRs employ passive safety systems, and main components are generally installed inside reactor coolant system boundary to improve safety and efficiency.
- Natural circulation system requires to be verified. Some reactors with longer fuel cycle have to be analyzed thoroughly concerning about structural deterioration by thermal fatigue induced by thermal transients.
- Some of the main thermal hydraulic items
  - Internal subassembly: determination of the clad temperature in the sub-channel for safety
  - Global core behavior: global core thermal hydraulic behavior study to ensure the core outlet temperature as uniform as possible, to ensure the decay heat removal performance in natural circulating conditions.
  - Upper and lower plena: possible thermal stratification between the upper and lower plena in steady-state condition
  - Gas entrainment: positive reactivity
  - Decay heat removal systems: natural circulation with 3D effects

01 Introduction

Research direction for modelling and simulation in fast SMRs

- The SMRs' design features require more specific modelling and simulation for design evaluation and safety analysis.
- Multi-dimensional multi-physics phenomena such as coolant mixing and heat transfer, including convection, conduction, and radiation, need to be investigated thoroughly from a design stage to ensure performance and safety.
- For the system codes generally using a 1D approach, some difficulties may appear when 3D phenomena occur due to nonsymmetrical situations or important buoyancy effects like thermal stratification. 3D code has fine resolution but cannot cover whole plant due to its heavy computing load.
- So, a new development is the dynamic coupling of the system and CFD codes to take into account 3D effects on the global system behavior during transient situations.
- Advanced multi-physics methods in LWRs: CASL (Consortium for Advanced Simulation of LWRs, USA), NURESIM (European Platform for Nuclear Reactor Simulation, 2005-2007), NURISP (Nuclear Reactor Integrated Simulation Project, 2007–2012), CUPID (KOREA).

01 Introduction

Research direction for modelling and simulation in fast SMRs

- Computational Fluid Dynamics (CFD), coupled with experiments using improved measurement techniques, is already helping in the design and licensing of advanced PWRs, e.g., EPRTM and AP1000. The capability of systems analysis code is also being expanded to include 3D effects and more coupling with neutronics, sub-channel and CFD codes.
- Except for the sodium–water reaction issue, the R&D efforts for the advanced SFR have some similarities with those for the advanced PWRs since coupling of fine-scale CFD code with systems code seems to be the main focus of computational activities in both cases. However, there are profound differences because of very different fluids, i.e., liquid sodium vs. water. Finally, all sophisticated analytical results must be validated with appropriate experiments.
- Efficient fine-scale CFD coupled with 1D system codes as well as structural analysis codes could be helpful to enhance design evaluation and safety analysis for fast SMRs.

01 Introduction

Need for the fine-scale flow simulations

- The core outlet region is subjected to two main challenges: the reliability of the subassembly outlet temperature measurements and the risk of thermal fatigue on the structures due to temperature fluctuations in mixing zones at different temperatures. The numerical analysis of such phenomena requires a refined modeling of the concerned region. The use of Large Eddy Simulation (LES) is recommended for the estimation of the amplitude and frequency range of the temperature fluctuations.
- Decay heat removal is a major challenge for all types of nuclear reactors. For sodium cooled fast reactors, passive decay heat removal based on natural convection is possible. This is one of the important advantages of these reactors. Several flow regimes must be considered corresponding to different operating conditions from forced convection to natural convection. The system codes cannot reflect 3D effects such as nonsymmetrical flow features or buoyancy effects like thermal stratification.
- Heat transfer through solids includes several connected flow regions. Temperatures of the reactor support structures including the reactor vessel (RV) are critical to structure integrity. To evaluate the temperatures of the support structures, all the connected flow regions have to be simulated in detail simultaneously.

02 Large Eddy Simulation

Need for LES

- The creep-fatigue is a mechanism that can deteriorate the structural integrity for SFRs.
- Repeated thermal cycles by thermal striping can accelerate the creep-fatigue damage on the structures.
- In a thermal striping region, the turbulent flow fluctuates with high frequencies.
- The Reynold-averaged Navier-Stokes (RANS) approach is not sufficient to resolve the flow physics
- Compared to the time-averaged RANS turbulence models, the LES can resolve time dependent fluctuations of flow variables more accurately.

02 Large Eddy Simulation

LES

- Unlike the RANS model, the governing equations for the LES model performs spatial filtering instead of taking a time average value for a special physical quantity. The solution ϕ consists of filtered value $\tilde{\phi}$ and sub-filtered value ϕ'. The fileted value $\tilde{\phi}$ can be written as follows:

$$\phi = \tilde{\phi} + \phi$$

$$\tilde{\phi}(t,x) = \iiint_{-\infty}^{\infty} G(x-x',\Delta)\phi(t,x')dx'$$

- The filtered continuum and momentum conservations for the LES model are expressed by the following equations:

$$\frac{\partial \rho}{\partial t} + \nabla \cdot (\rho \tilde{V}) = 0$$

$$\frac{\partial}{\partial t}(\rho \tilde{V}) + \nabla \cdot (\rho \tilde{V} \otimes \tilde{V}) = -\nabla \cdot \tilde{p} I + \nabla \cdot (T + T_t) + f_b$$

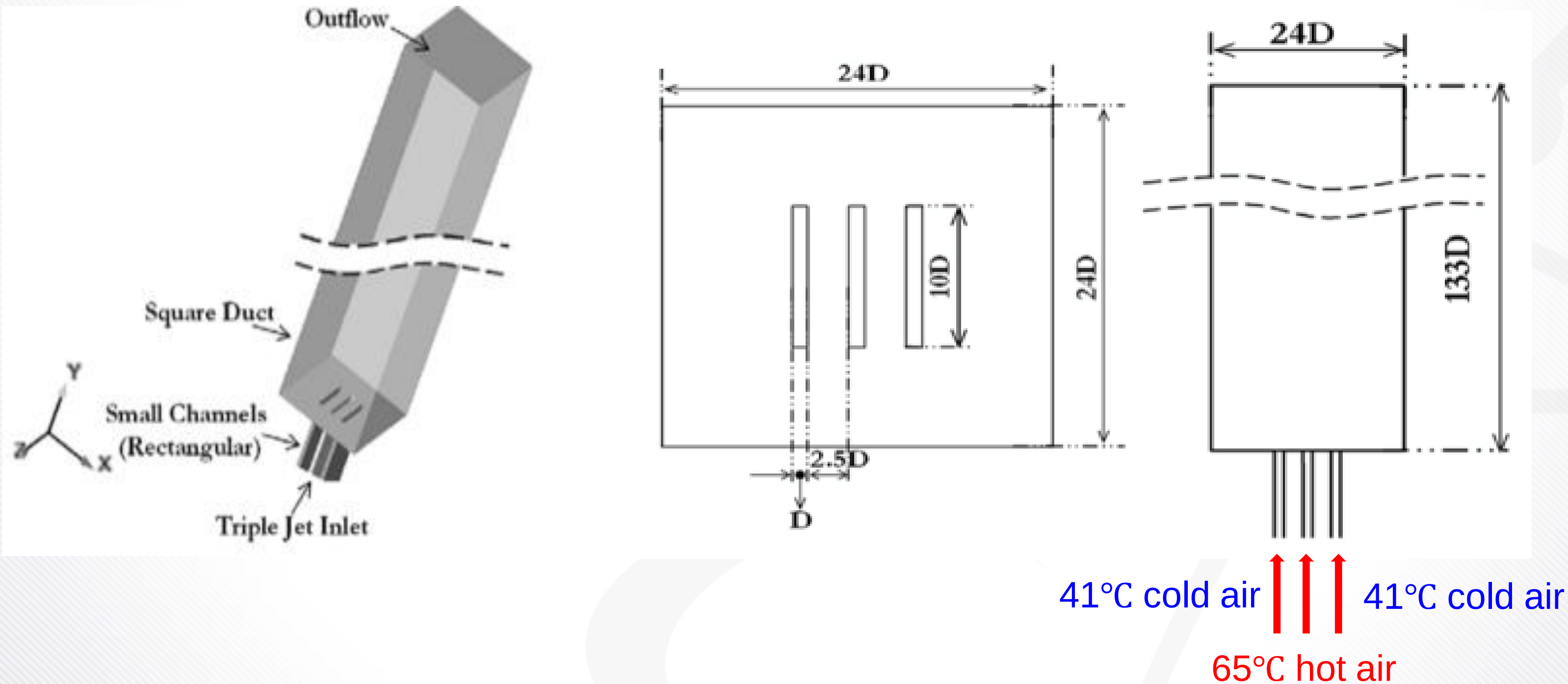
$$T_t = 2\mu_t S - \frac{2}{3}(\mu_t \nabla \cdot \tilde{v} + \rho k) I$$

- where the turbulent stress tensor T_t represents a sub-grid scale stress and is obtained Boussinesq approximation. The turbulent viscosity parameter μ_t is defined based on the sub-grid-scale WALE (wall-adapting local eddy viscosity) model.

02 Large Eddy Simulation

Preliminary simulation using LES

- Before applying the LES to thermal striping in the PGSFR upper internal structure (UIS) region, the LES with WALE model was validated through a triple jet experiment.

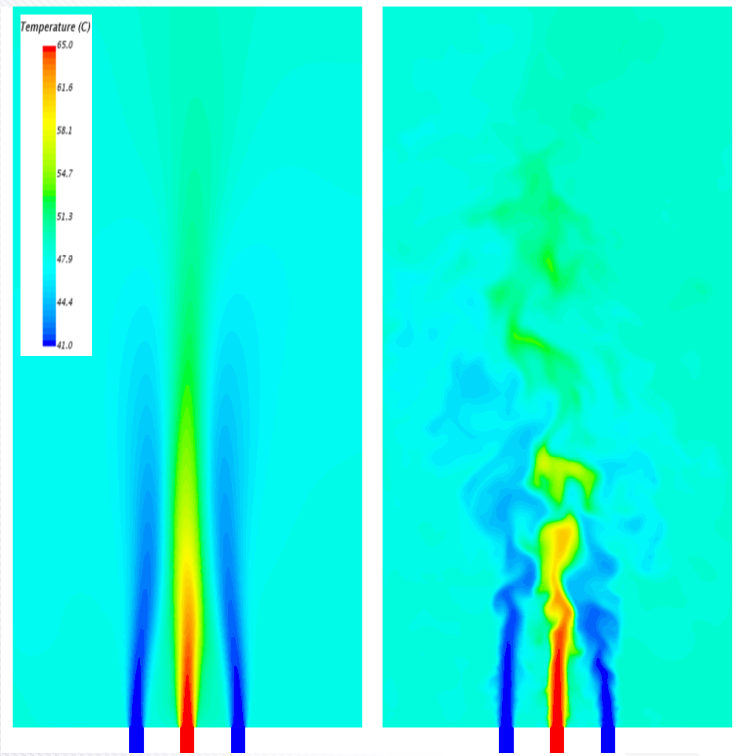


Schematic diagram of triple jet geometry

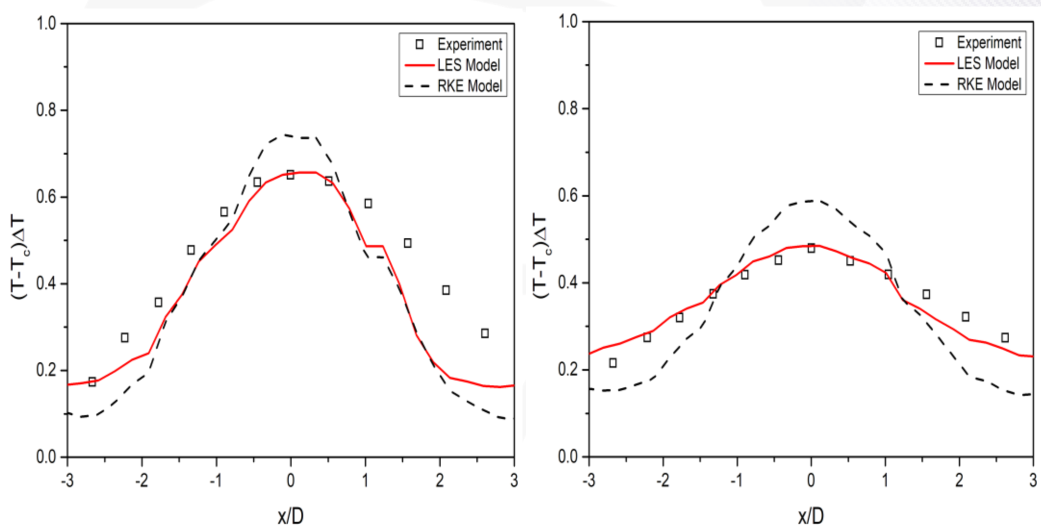
02 Large Eddy Simulation

Preliminary simulation using LES

- The LES model shows good agreement with the experiment. The results demonstrate that the present LES model can be employed to predict the thermal stripping in SFRs.



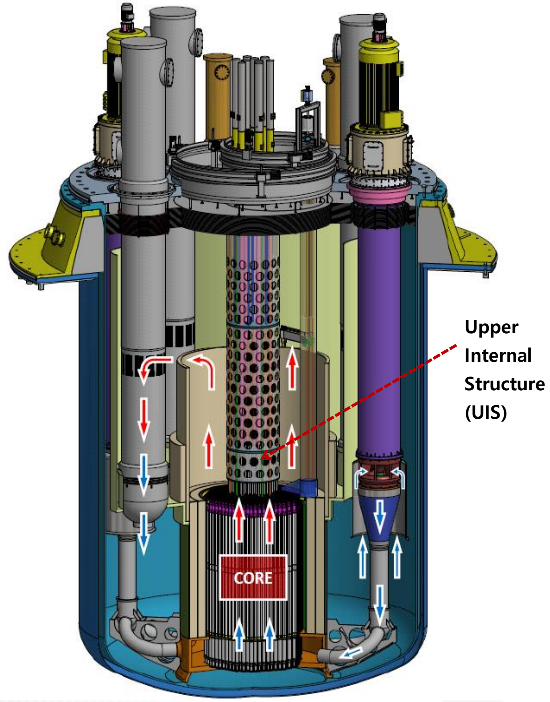
Instantaneous temperature contour in the mid-plane
RANS with realizable k-ε model (left), LES model (right)

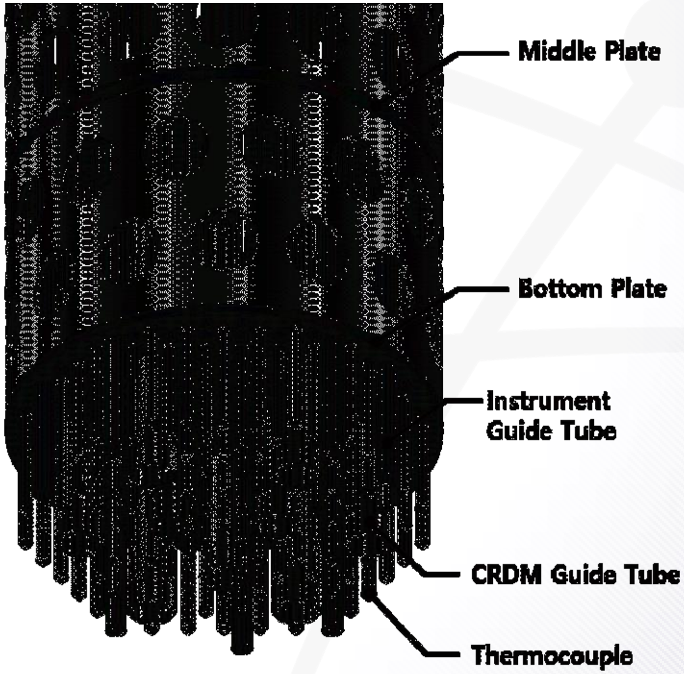


Time-averaged temperature profiles in the mid-plane
(y/D=12 (left), y/D=18(right))

LES study of thermal striping along the CRDM guide tube in the PGSFR

- The upper internal structure (UIS) faces directly the hot sodium discharged from the core exit, and thermal striping occurs by sodium with different temperatures.
- The control rod drive mechanism (CRDM) guide tubes may suffer the thermal striping due to the strong inflow of the surrounding fuel subassemblies.



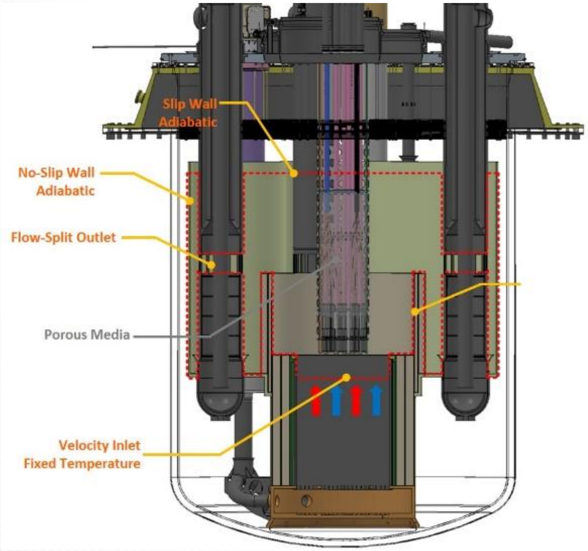


Schematic diagram of the reactor core and close-up view of the upper internal structure

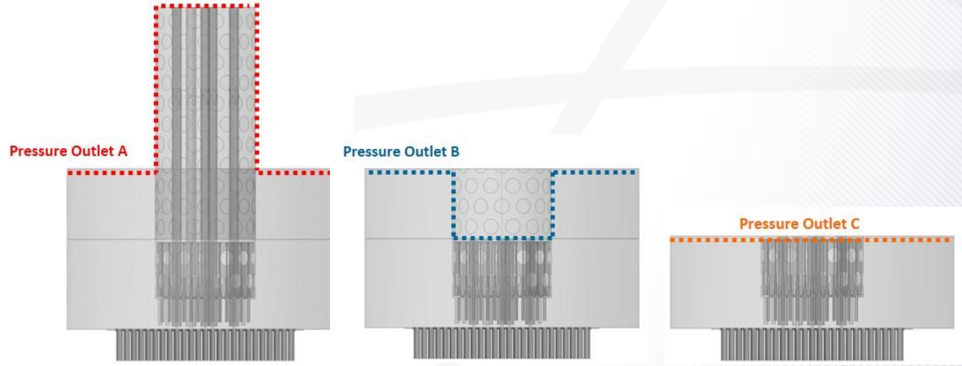
02 Large Eddy Simulation

LES study of thermal striping along the CRDM guide tube in the PGSFR

- It is difficult to apply the LES to the full upper plenum due to the enormous number of meshes. A way to set up a computational domain affordable for LES simulation by pre-evaluation using RANS simulation was applied.
- The RANS simulation results for reduced domains were compared with the results obtained from the full domain to decide the computational domain to which boundary conditions don't have much influences.



Full domains for simulation

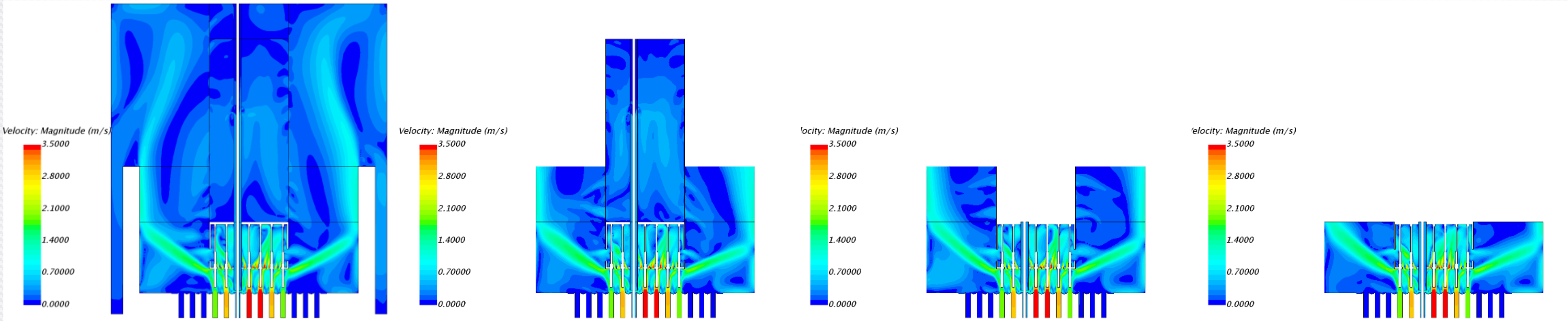


Candidate reduced computational domains

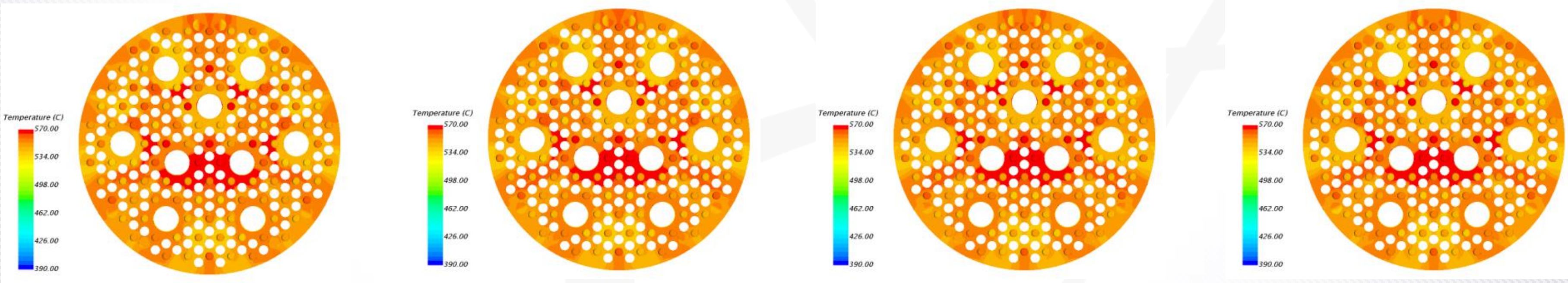
02 Large Eddy Simulation

LES study of thermal striping along the CRDM guide tube in the PGSFR

- Velocity and temperature distributions at various vertical and horizontal section



Case 01 Case 02 Case 03 Case 04

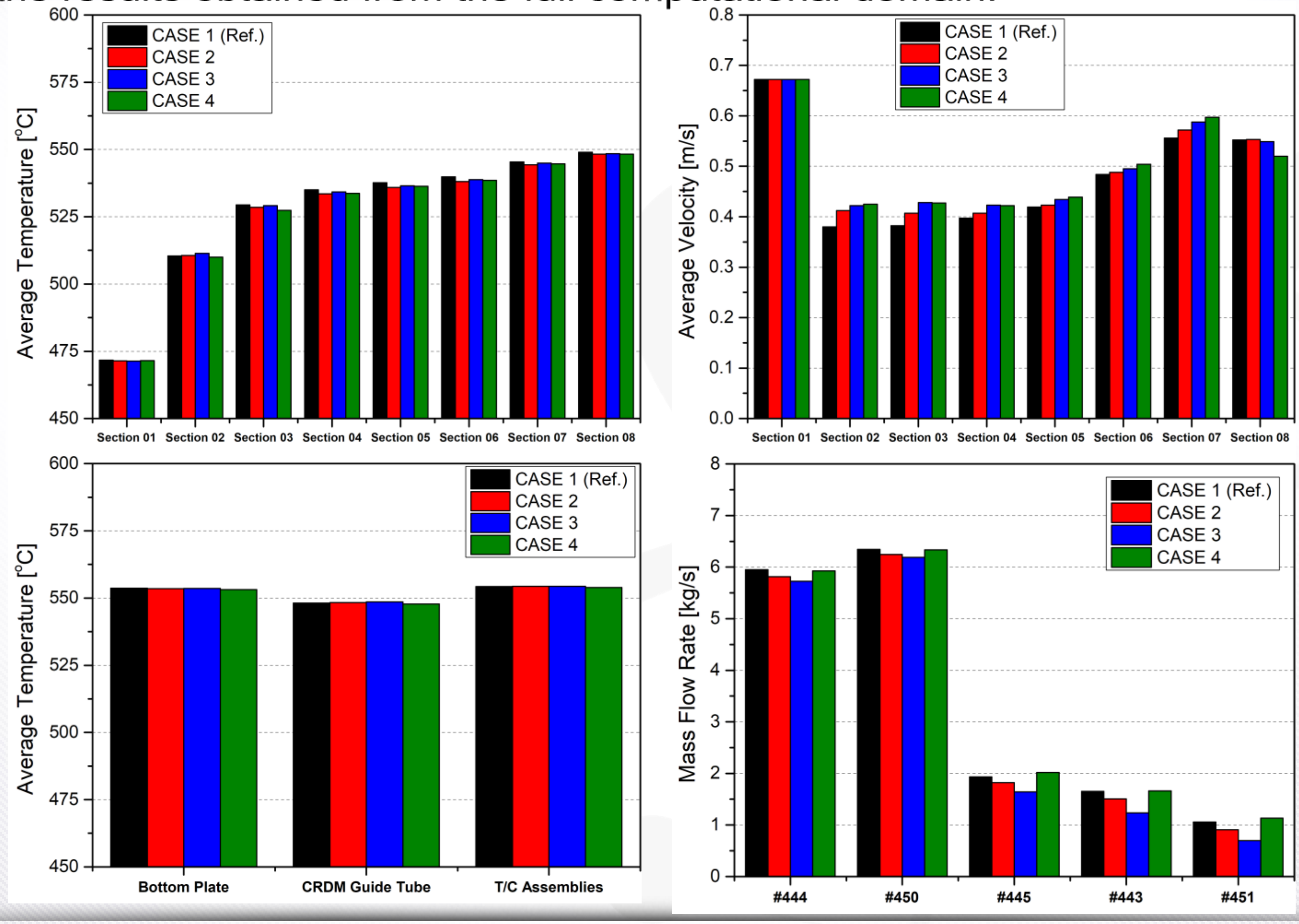


Case 01 Case 02 Case 03 Case 04

02 Large Eddy Simulation

LES study of thermal striping along the CRDM guide tube in the PGSFR

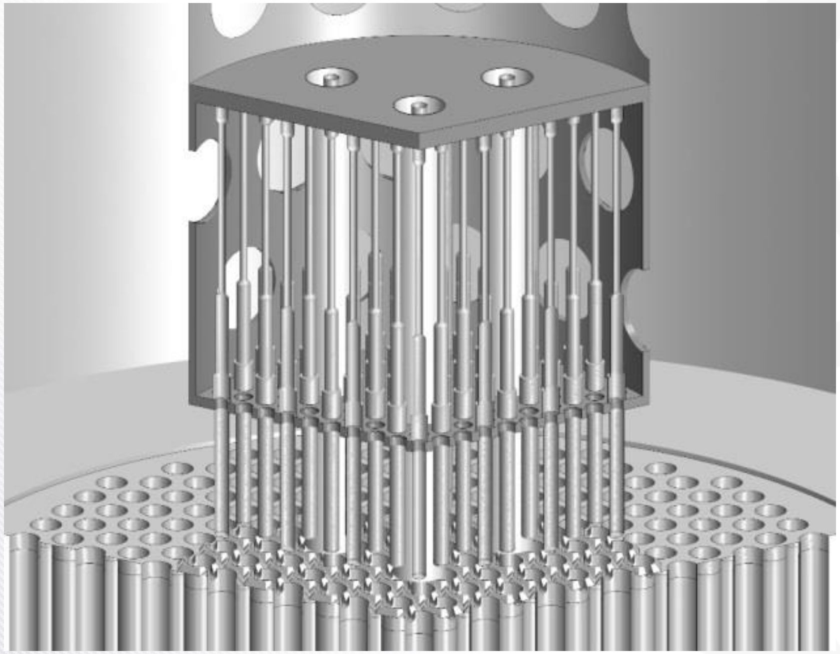
- Results from three reduced computational domain cases were quantitatively compared with the results obtained from the full computational domain.



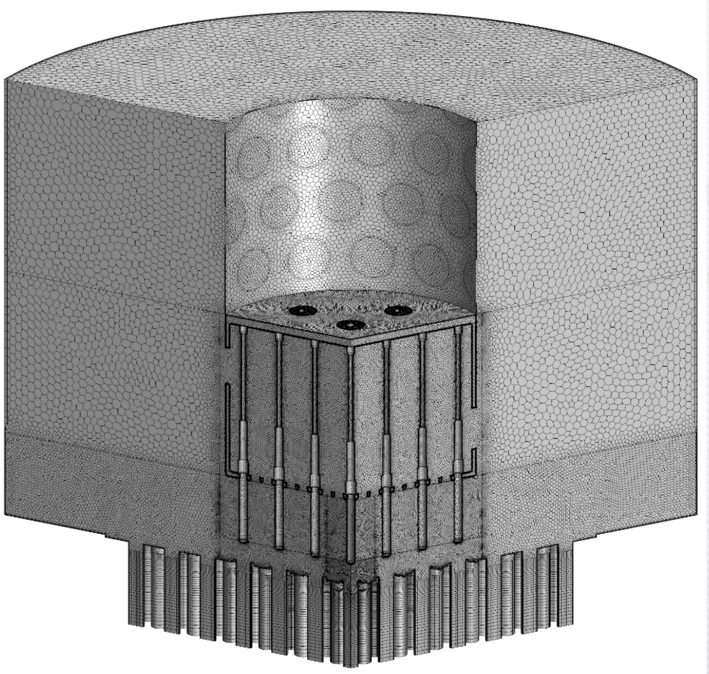
02 Large Eddy Simulation

LES study of thermal striping along the CRDM guide tube in the PGSFR

- The geometry of the IHX inlet region and the upper core shield structure did not affect the flow characteristics of the core exit region.
- The UIS geometry was found to be crucial for thermal striping.
- Reduced computational domain covers 120˚ region of the core, UIS, and hot sodium plenum which involves18 fuel subassemblies and 3 control rod subassemblies.



Computation domain for the thermal striping analysis

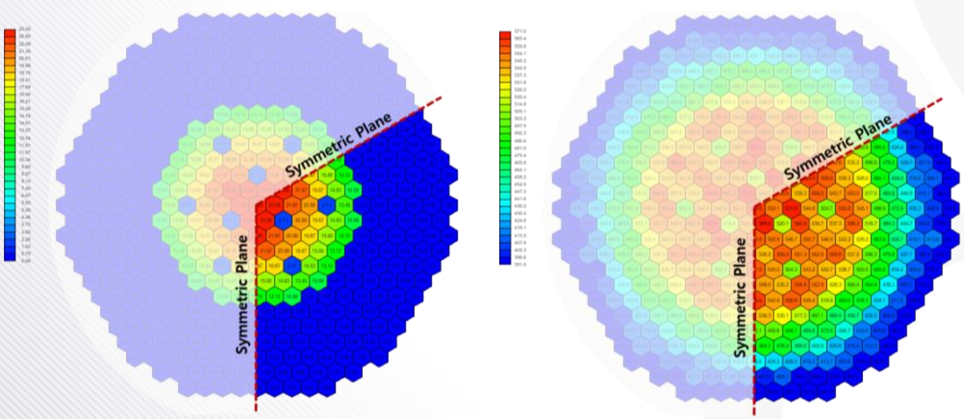


Grid system

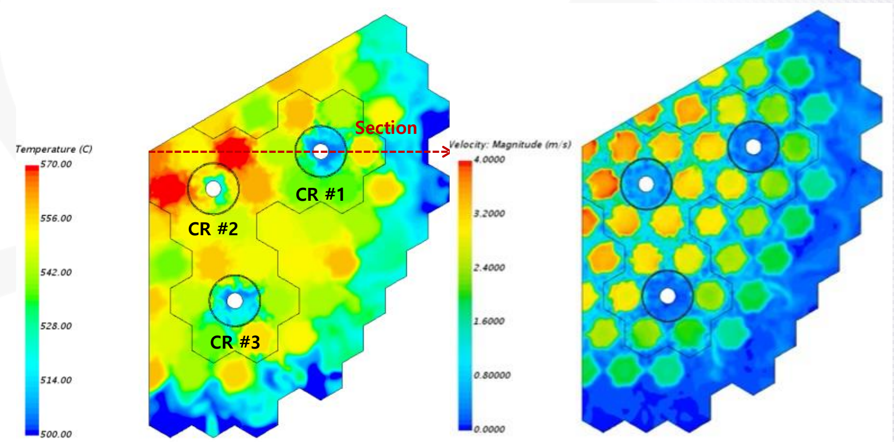
02 Large Eddy Simulation

LES study of thermal striping along the CRDM guide tube in the PGSFR

- The flow rate through each control subassembly during normal operation is 1/5 ~ 1/10 of fuel subassemblies.
- The large temperature differences between the two neighboring subassemblies ranged from 50 to 55℃.
- At a horizontal plane around the tip of the CRDM guide tubes, which were located 50 mm away from the core exit, the temperature and velocity distribution were calculated.



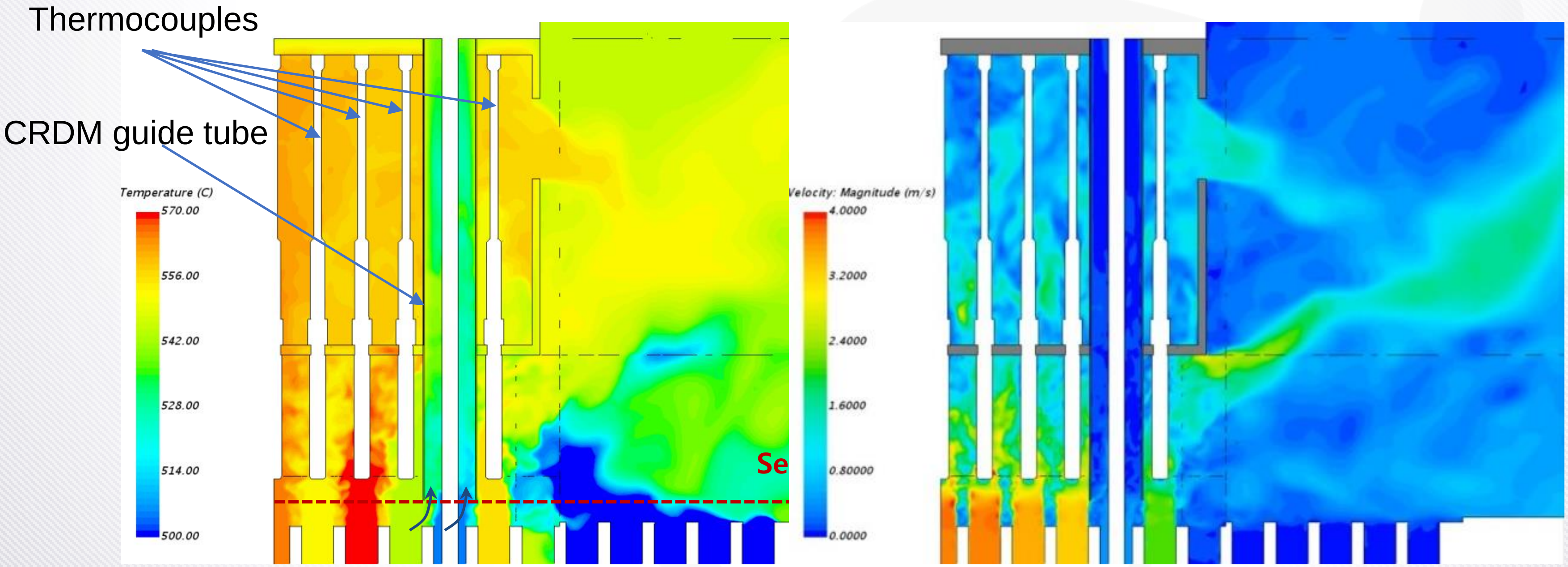
Core exit flow conditions of computation region (Left: Flow rates, Right: Temperatures)



Temperature and velocity distribution at the horizontal section (50 mm off from the core exit)

LES study of thermal striping along the CRDM guide tube in the PGSFR

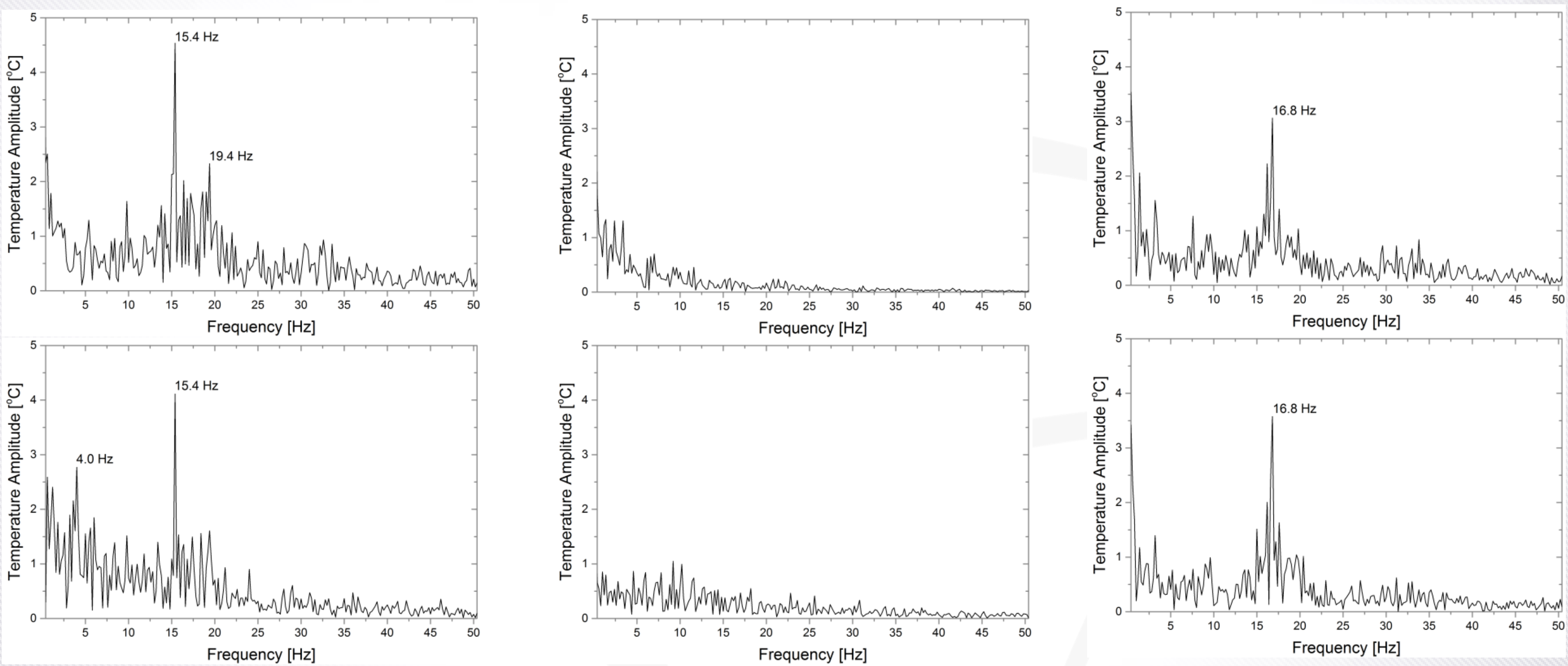
- Since the flow rates from the neighboring fuel and control subassemblies were similar for the CR #1, the flow instability was higher than the CR #2.
- The amplitude of the oscillating temperature around CR #1 was found to be about 40℃ at the CR #1.



Temperature and velocity distribution at the vertical section through CR #1

LES study of thermal striping along the CRDM guide tube in the PGSFR

- FFT of temperature fluctuations on CR#1, CR#2, CR#3



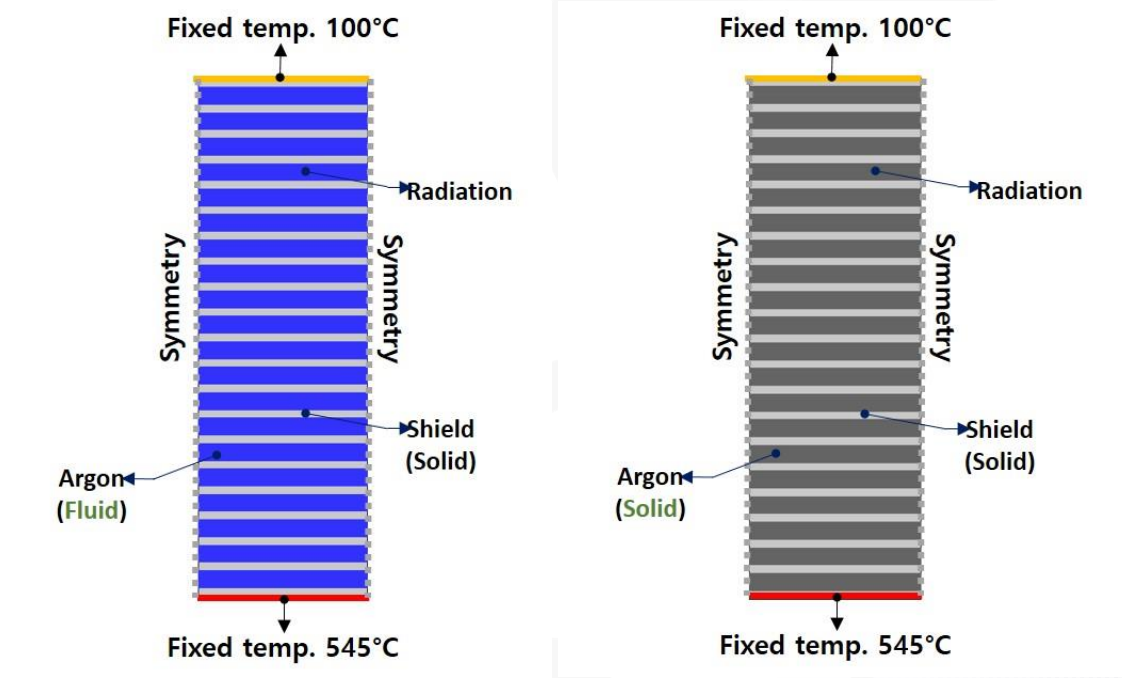
FFT of the temperature difference fluctuation at the CR#1 (Left), CR#2 (Middle), CR#3 (Right)
(Top: Inner wall, Bottom: Outer wall)

Integrated modelling and simulation for RVCS design evaluation

- The RVCS of the PGSFR is a passive heat removal system that operates during normal operation and severe accidents. The RVCS protects the vault and concrete cavity from the core heat during normal operation and removes the decay heat in the case of severe accidents. The RVCS removes the heat from the CV surface via natural convection.
- The RVCS is connected to the containment vessel (CV) and the reactor support structures. The CV and the reactor vessel (RV) are also connected to the head access area (HAA) through the reactor head (RH).
- The CV and the RV of the PGSFR are supported by the structure connected to the RH. The temperature of the reactor support structure, the vessels, and the concrete cavity are critical for the reactor integrity.
- For thermal-hydraulic analysis, the thermal boundary conditions for only a local region cannot be given accurately without considering the connected regions. Thus, all the connected regions needs to be modelled and simulated to assess the RVCS performance and examine the temperature of the reactor support structure, the vessels, and the concrete.
- However, full-scale simulation that reflects exact physics and geometries is impractical. Therefore, simplified models for the components inside the reactor vessel were applied.

Integrated modelling and simulation for RVCS design evaluation

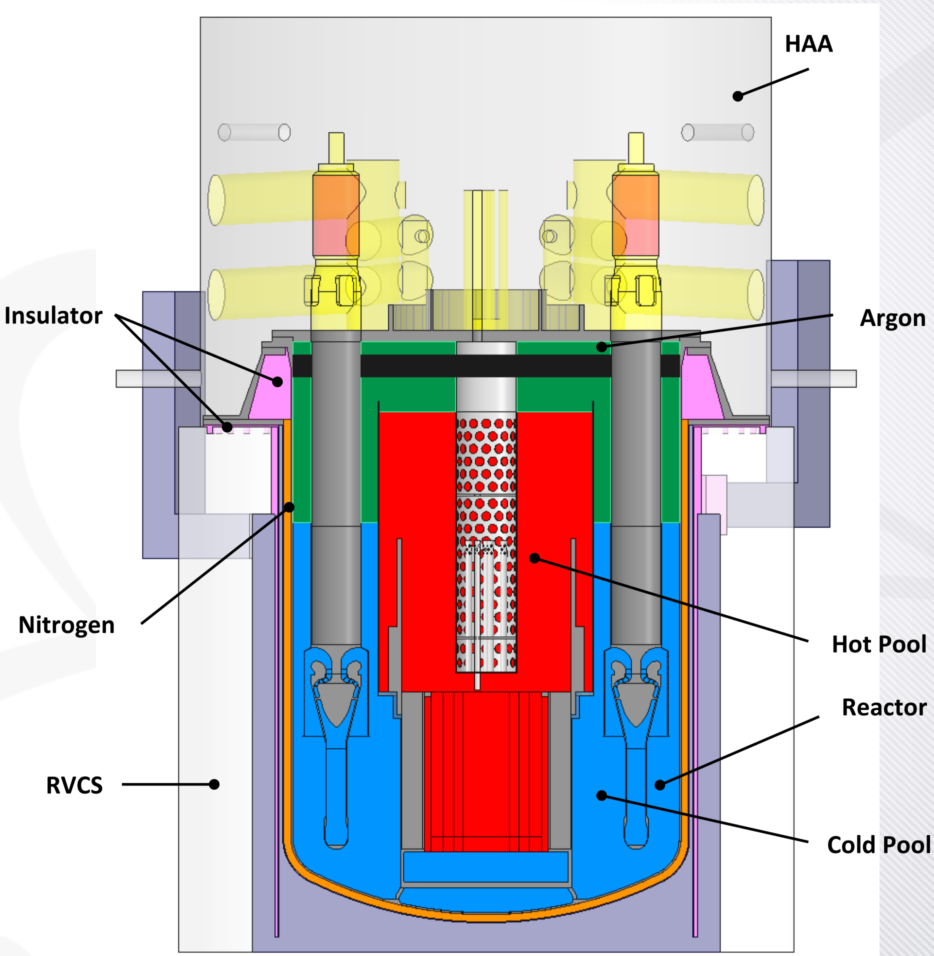
- Porous media: intermediate heat exchangers (IHXs), decay heat exchangers (DHXs), and the reactor core inside the RV
- Conductive material : upper shield structure in the cover gas region
- Dramatic reduction of number of grids while holding the heat transfer rate difference between the two models within 1%



Simplified modelling of the upper shield structure (Left: Original, Right: Argon region as solid)

Integrated modelling and simulation for RVCS design evaluation

- HAA, thermal-hydraulics inside the RV, RVCS
- Solver (STAR-CCM+ ver.11.02.009)
  - ✓ 3D steady-state
  - ✓ k-ω SST turbulent model
  - ✓ Buoyance force considered
- Conditions
  - ✓ Pump Impeller: momentum source, mixing plane model (841 RPM)
  - ✓ Radiation heat transfer activated
    - RVCS/HAA Air, Nitrogen Gas, Argon Gas

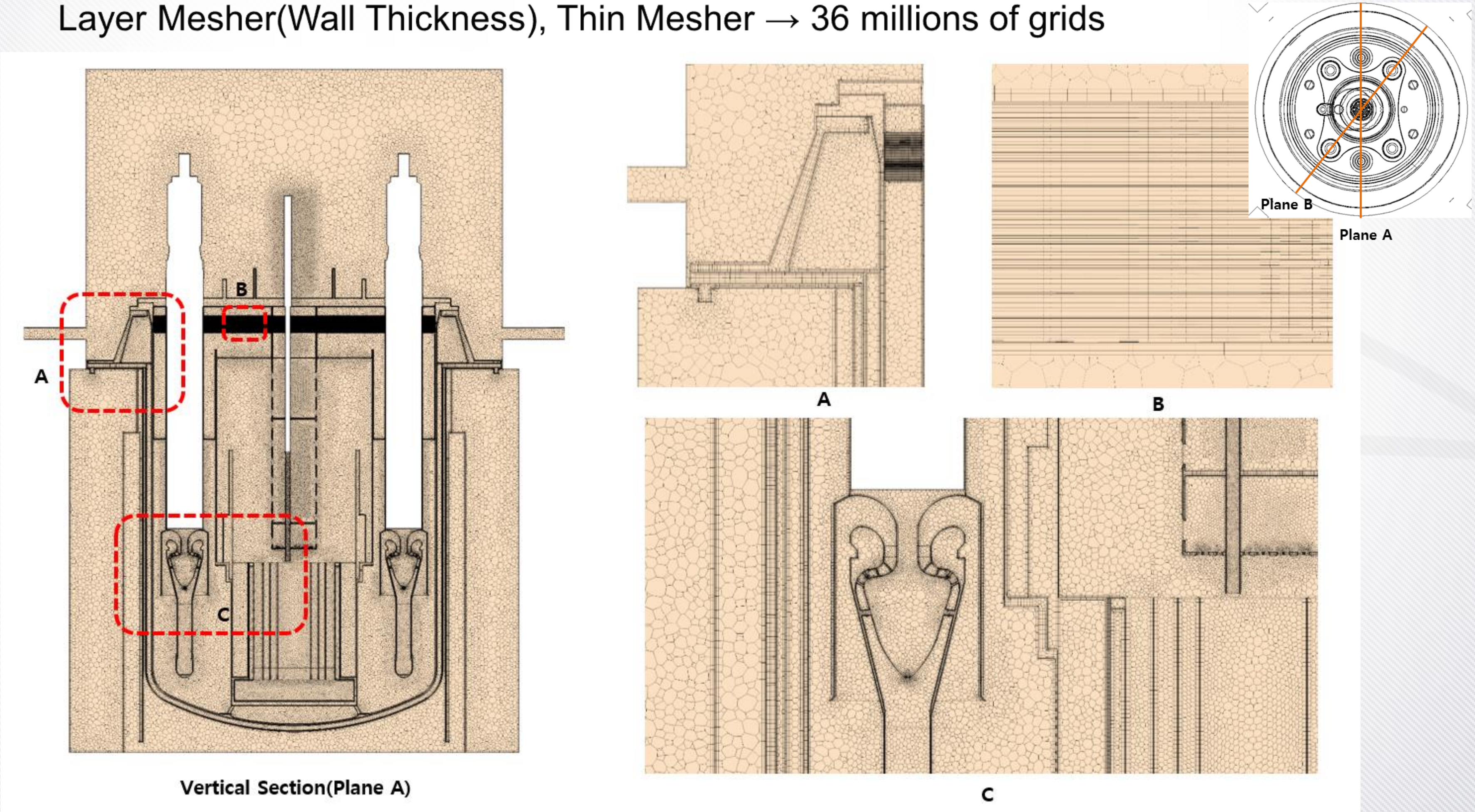


Computational domain for integrated simulation

03 Integrated Modelling and Simulation

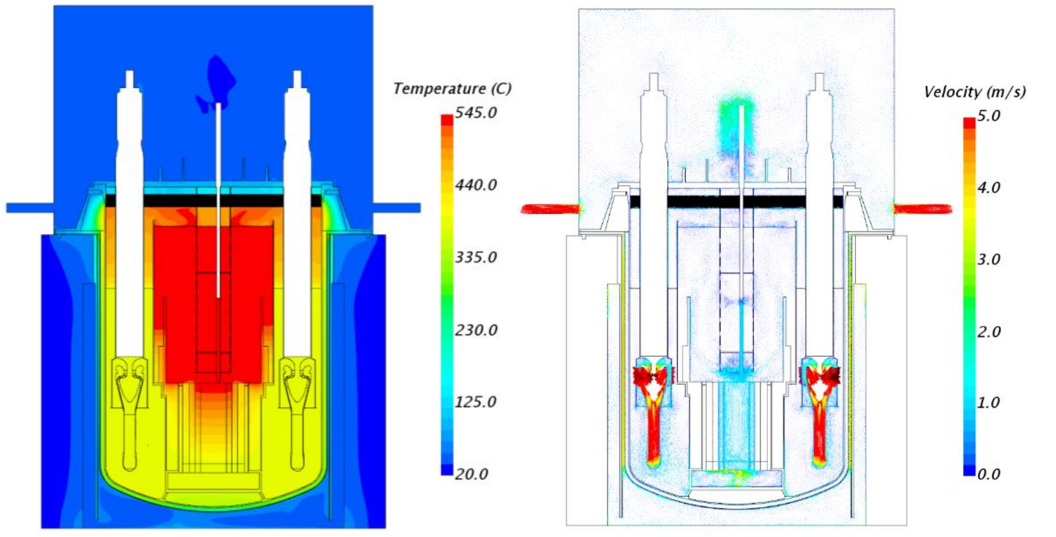
Integrated modelling and simulation for RVCS design evaluation

- Grid system by Surface Remesher, Automatic Surface Repair, Polyhedral Mesher, Prism Layer Mesher(Wall Thickness), Thin Mesher → 36 millions of grids



Integrated modelling and simulation for RVCS design evaluation

- The averaged temperatures of the main flow paths such as the core inlet and outlet were in good agreement with the design values within less than 1% difference.
- 20℃ air fed into the HAA area was discharged with increased temperature of 43℃~ 47℃.
- Hot pool sodium temperature decreased to below the design limit 150℃ at the reactor head.
- The temperatures of the vessels and support structures were within the design limits.

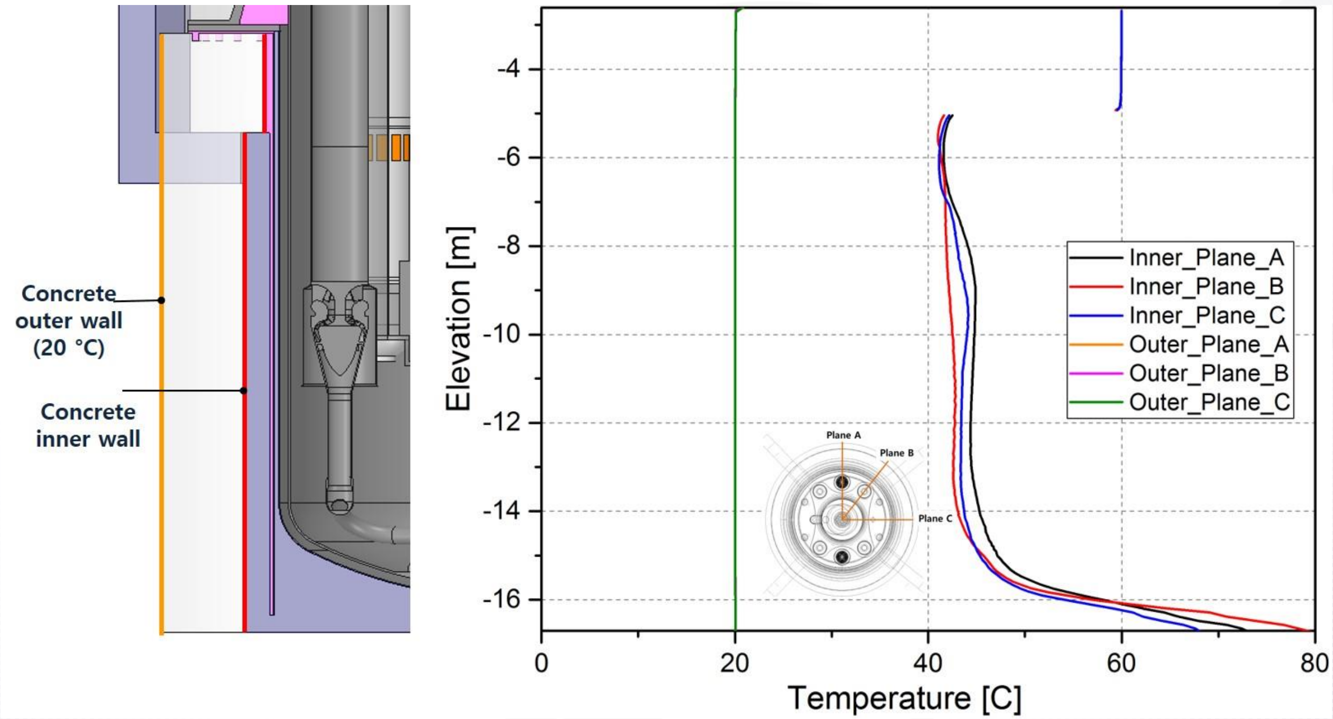


Temperature distribution (left) and velocity distribution (right)

03 Integrated Modelling and Simulation

Integrated modelling and simulation for RVCS design evaluation

- The temperature at the concrete wall of the RVCS should be below the ASME limit of 65℃ during the plant normal operation.
- The concrete wall temperatures were around 40℃, except for the upper and lower regions.
- For the lower region, an insulator needs to be added to eliminate hot spots.



Vertical temperature distribution of inner & outer concrete in the RVCS

04 Concluding Remarks

- Generally, SMRs utilize a passive heat removal system, and components are arranged compactly inside the reactor vessel for modularization. A smaller inventory, but complicated arrangement requires detailed modelling and simulations. A refined CFD simulation or system codes coupled with multi-dimensional CFD can be very helpful for design evaluation and safety analysis.
- The thermal striping phenomena in the UIS region of the PGSFR was investigated by using the LES model with an optimally reduced computational domain. Temperature fluctuations due to thermal striping in a short time scale were well resolved.
- An integrated simulation that included all the connected flow regions in the primary heat transport system was carried out to evaluate the RVCS performance and to examine the temperatures of the support structures. To this end, refined simplification models for the upper shield structure, heat exchangers, pump, and reactor core were developed. Through this work, temperature distributions in the HAA, CV, RV, RH, RVCS, and reactor support structures were clearly resolved.

THANK YOU