SEOUL NATIONAL UNIVERSITY

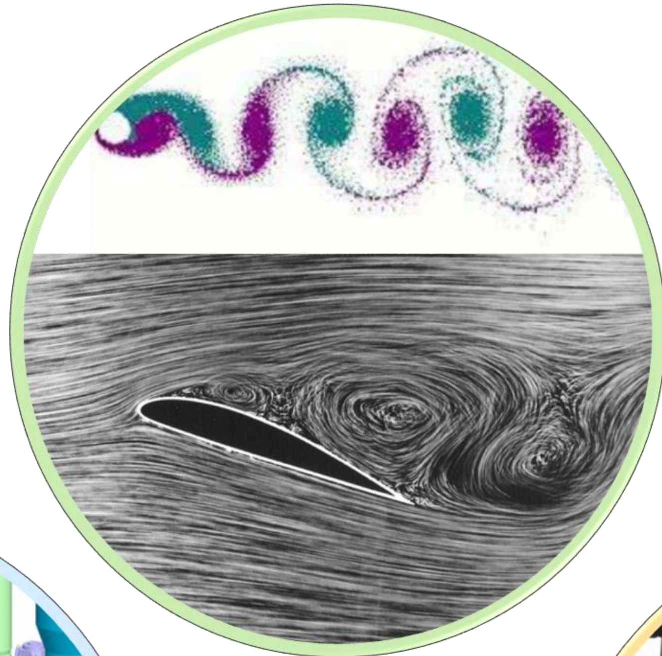
Nuclear Thermal Hydraulic Engineering Lab.

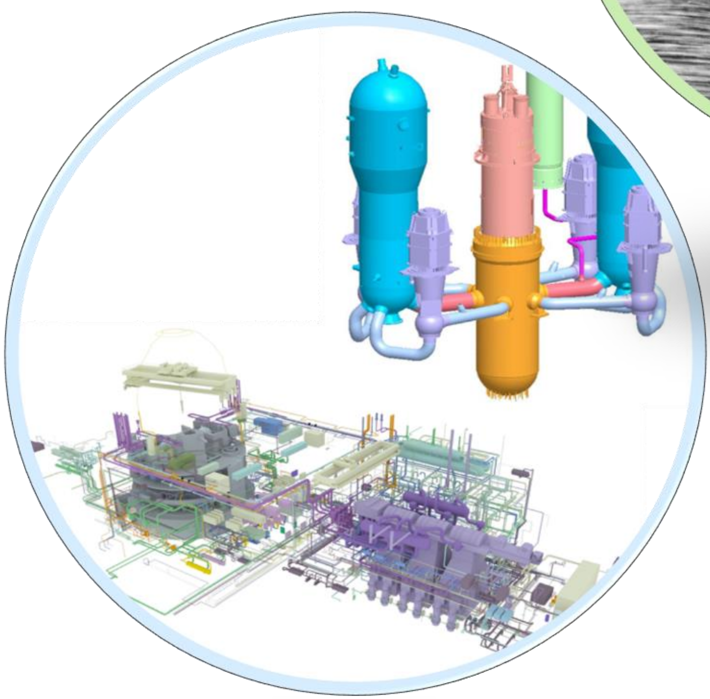
원자로 열유체 실험

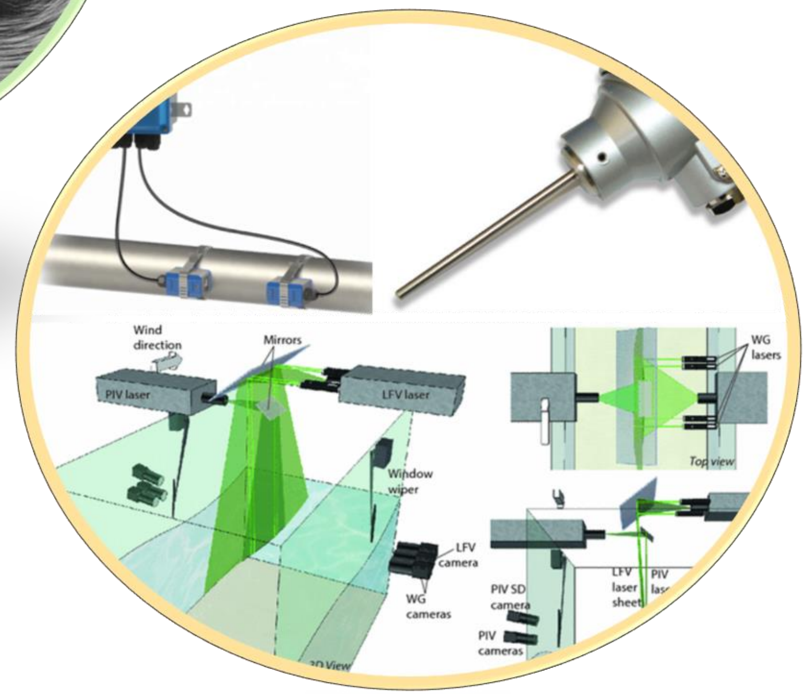
Department of Nuclear Engineering, Seoul National Univ.
Hyoung Kyu Cho

강의 개요

❖ 원자로 열유체 실험 ?

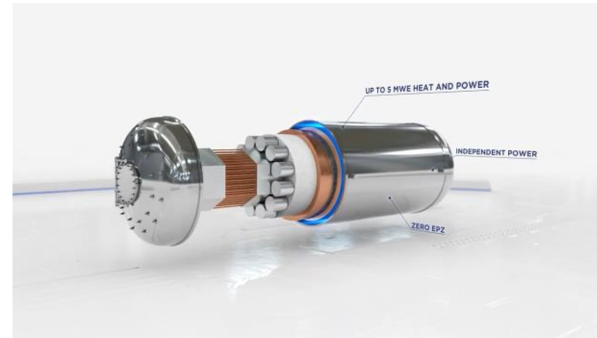


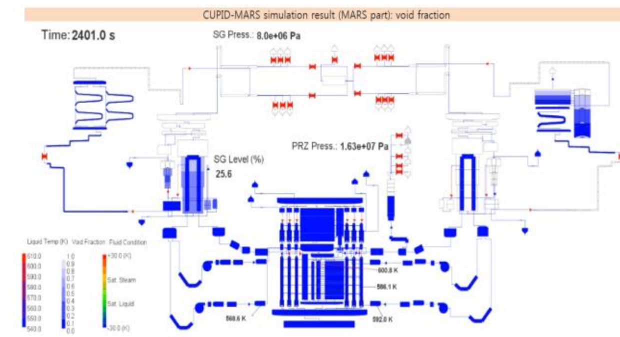


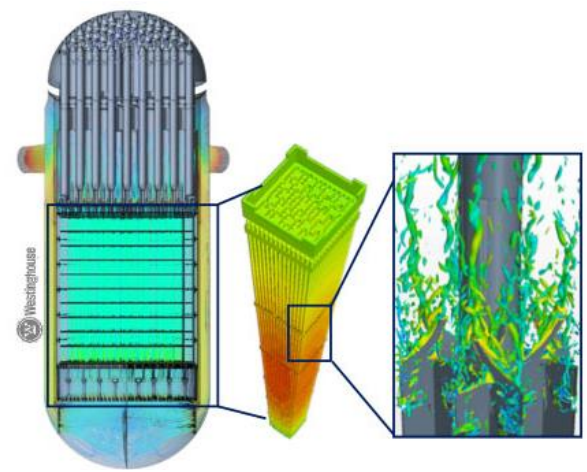


강의 개요

- 원자로 열수력학 (Nuclear Thermal-hydraulics)
- 원자로 열유체역학 (Nuclear thermo-fluid dynamics)
  - ✓ 유동 (fluid flow) + 열전달 (heat transfer)
  - ✓ Ex) 정상 상태 ? 사고 조건 ?
  - ✓ 필요 분야
    - 시스템 설계
    - 신형 원자로 개발
    - 원자로 안전 해석
  - ✓ 관련 업무
    - 전산유체코드 및 계통 해석 코드
      - – 원자로 설계 및 안전성 평가, 코드 개발
    - 열수력 실험
      - – 신형 원자로 및 계통 개발, 이론 개발







강의 개요

❖ 원자력연구원 조직

강의 개요

❖ 원자력연구원 조직

강의 개요

❖ 원자력안전기술원 조직

https://blog.naver.com/PostView.nhn?blogId=kins20&logNo=221918207875&redirect=Dlog&widgetTypeCall=true&directAccess=false

규제 현장에 필요한 기술을 선제적으로 연구하여
원자력 안전에 이바지하는 규제검증평가실,
원자력 안전의 백신입니다!

Q5. 규제검증코드(MARS-KS) 개선도 주요 업무 중 하나입니다. 원전 안전해석 및 계통 열수력 분야에 서 중요하게 쓰이는 규제검증코드란 무엇인가요?

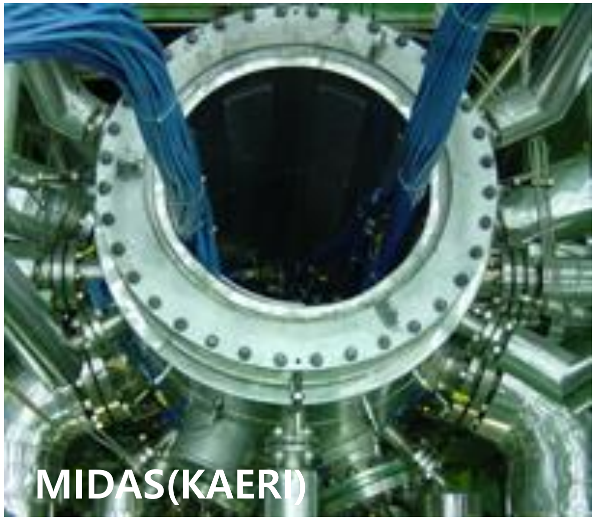
우리나라에서 가장 많이 쓰는 소프트웨어 중 하나는 한글 워드프로세서입니다. 간단한 메모부터 결재문서, 보고서, 논문 등 많은 결과물을 만들어 내는 중요한 프로그램이죠. 1989년 아래한글 1.0에서 시작해 한컴 2018에 이르기까지 지속적으로 오류를 수정하고 기능을 개선한 결과 입니다. 원전 안전해석 및 계통 열수력 분야에서 비슷한 역할을 하는 프로그램이 바로 MARS-KS 코드입니다. 실험설비 평가나 원자력발전소의 설계기준사고 및 과도해석을 수행하여 기준 만족성 여부를 판단할 때마다 사용됩니다. 2007년 한국원자력연구원(KAERI)이 개발한 MARS 코드를 KINS가 코드의 품질 등을 확인 후 규제 검증 해석을 목적으로 도입하여 MARS-KS ver 1.0으로 명명한 것이 코드의 시작점입니다.

Q6. 규제검증코드(MARS-KS) 개선이 중요한 이유는 무엇인가요?

현재 사용하는 컴퓨터에 과거 초기에 개발되었던 아래한글 1.0을 사용할 수 있을까요? 불가능할 것입니다. 운영환경이 맞지 않아 설치조차 불가능 합니다. 해석코드도 비슷한 상황입니다. PC 운영체계, 코드를 만드는 컴파일러의 변경 및 기 능지원 중단 등 여러 가지 환경변화에 맞춰 따라 가지 않으면 코드를 기반으로 개발된 입력자료, 기술 등이 전혀 사용할 수 없는 상황에 이를 테니까요. 변화하지 않는 기술은 사라집니다. 현재의 높은 기술을 유지하기 위해서는 지속적인 변화에 발맞춰 나가야 합니다. MARS-KS 역시 평가 대상이 새로 개발되거나 변경될 때에는 관련 입력모델도 개발하거나 개선해야 합니다. 현재는 MARS-KS ver1.5까지 발견된 오류를 수정하고 새롭고 효율적인 기능을 추가·발전시켜 사용 중 입니다. 코드 개선 발전이 원자력 안전해석기술 발전의 핵심기반기술을 유지하고 있는 셈입니다.

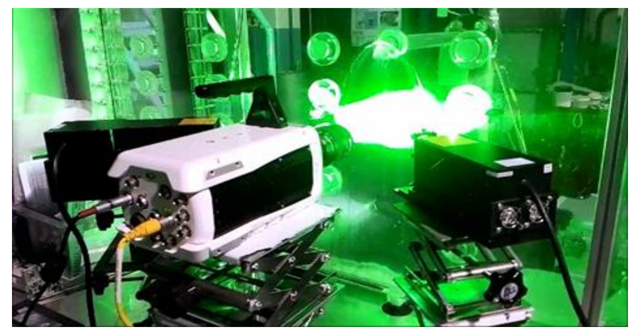
강의 개요

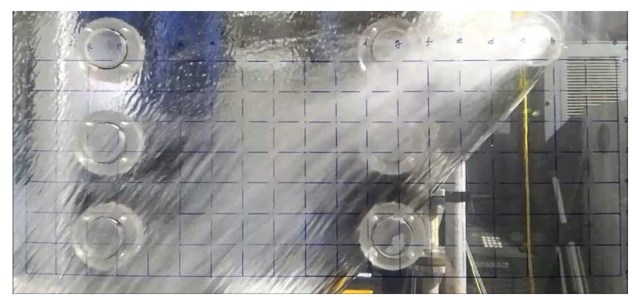
❖ **원자로 열유체 실험**



강의 개요

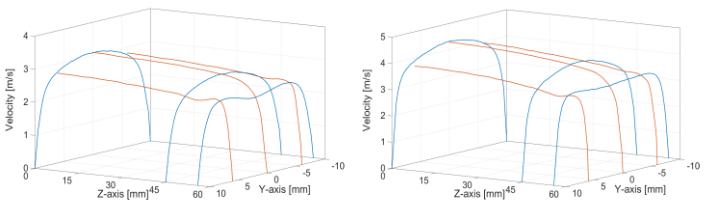
❖ 원자로 열유체 실험

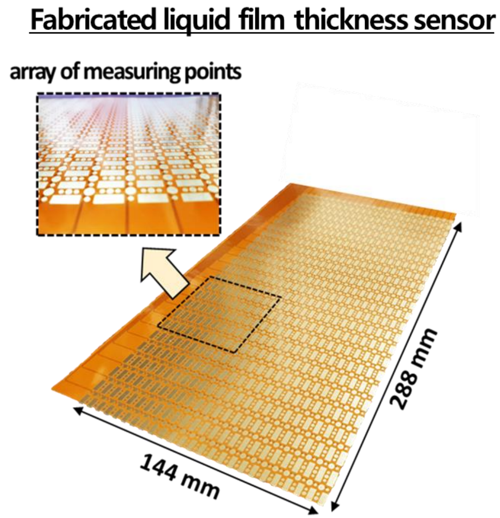


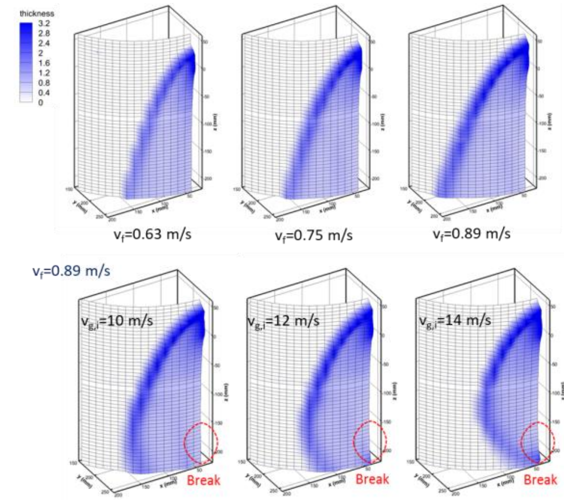


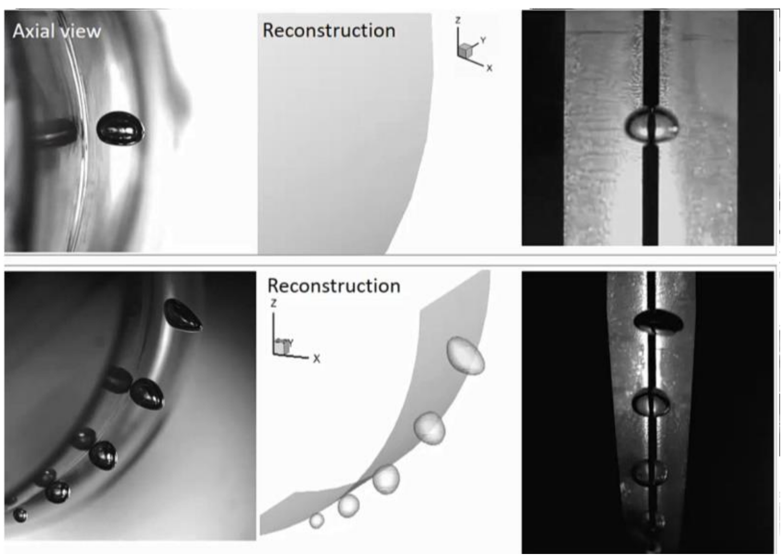
강의 개요

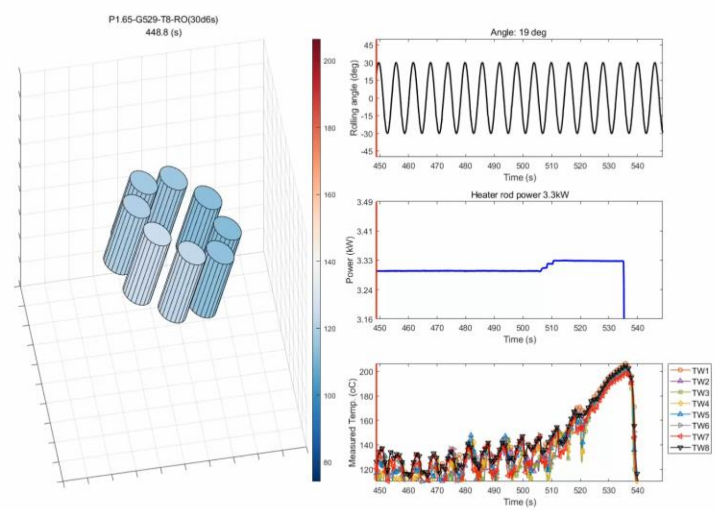
❖ 원자로 열유체 실험: SNU-DNE-NUTHEL





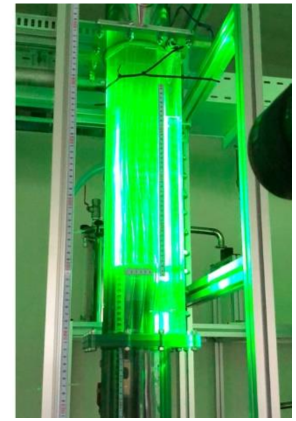


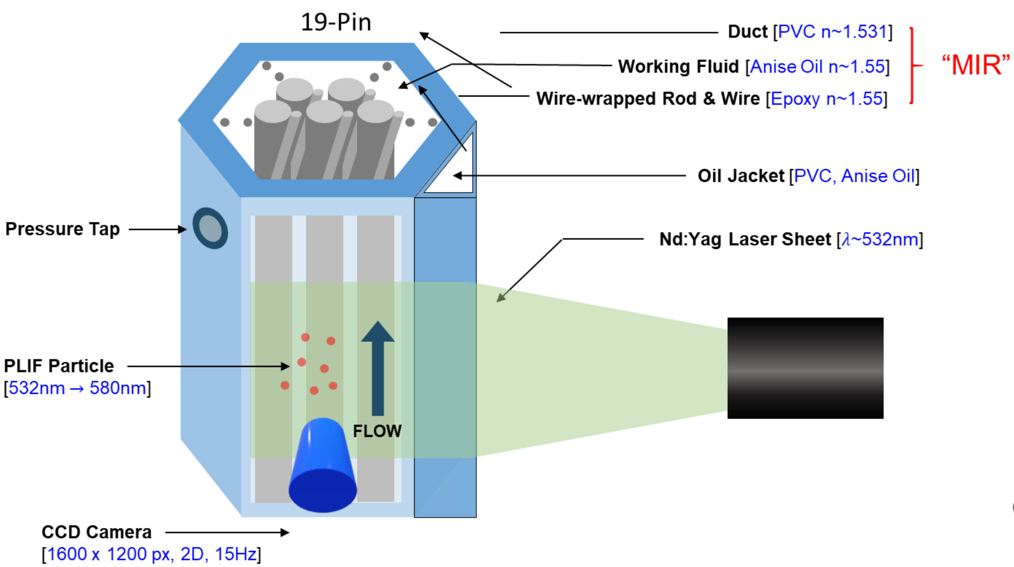


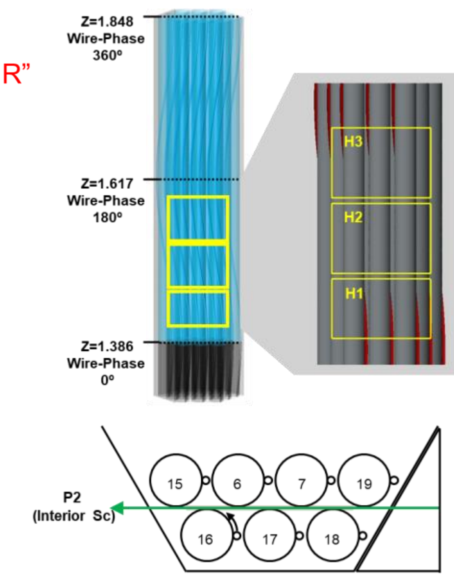


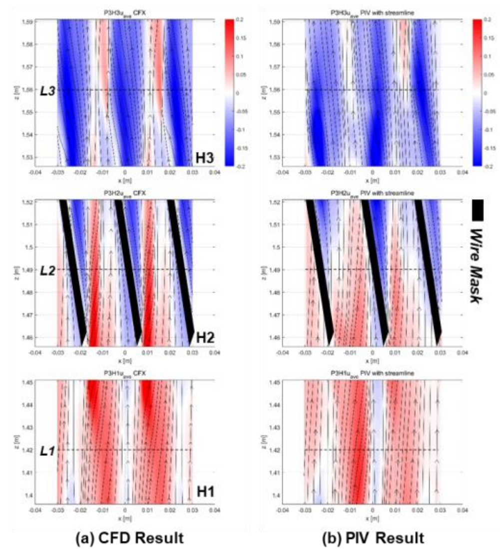
강의 개요

❖ **원자로 열유체 실험: SNU-DNE-ESLAB**







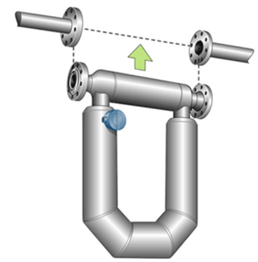


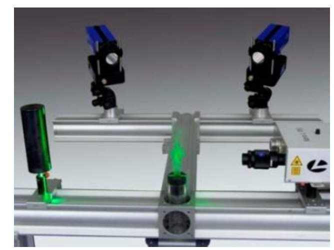
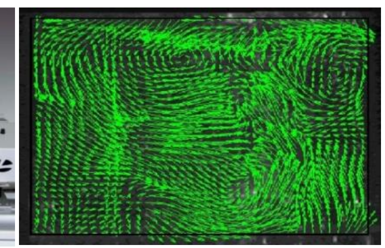
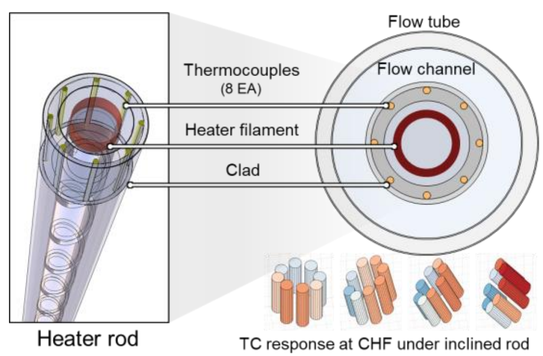
P2-Transverse Velocity

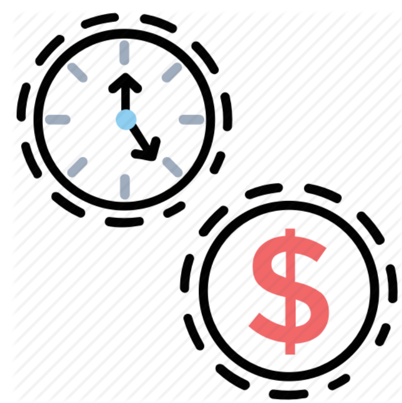
강의 개요

❖ **열수력 실험 비용**
- ✓ **대형 실험장치 구축 비용 및 기간**
  - 3~5년, 수십 ~ 수백 억
- ✓ **주요 계측기기 가격**
  - Thermocouple
  - Differential pressure transmitter
  - Hot film sensor
  - Coriolis mass flow meter
  - High speed camera
  - PIV system
- ✓ **히터 가격**






강의 개요

- **가동원전에 대해 안전성 평가 및 안전목표 요구**
  - 중대사고시 격납건물 건전성 확보
  - 선량기준 : EAB 선량 250mSV 이하
  - 안전목표 : CDF 10^{-4} 이하, LERF 10^{-5} 이하, Cs-137 100 TBq 방출빈도 10^{-6} 이하

Reactor Safety Analysis Code (원자로 안전해석 코드)

- ✓ 원자로에 내에서 발생하는 현상을 예측할 수 있는 컴퓨터 프로그램
- ✓ 원자로 내부의 복잡한 현상을 예측하기 위해 다양한 모델 및 실험 상수 필요

DBA (Design Based Accident, 설계 기준 사고)

- ✓ 원자로 설계 시, 발생할 수 있다 가정하고 안전 설비 구축
- ✓ LOCA (Loss of Coolant Accident)

Severe Accident (중대 사고)

- ✓ 사고로 인해 원자로 노심 용융 발생
- ✓ 설계 기준 사고는 아니나 사고관리 필요

열수력 실험의 필요성

- ✓ 사고 발생 시 원자로 내부에서 발생하는 현상 규명
- ✓ 안전해석코드 검증 및 개선
- ✓ 사고를 대비한 신 안전계통 개발 및 성능 평가
- ✓ 보다 효율적인 원자력발전소 개발

강의 개요

❖ **안전해석코드 모델 개발을 위한 실험 사례**

✓ **2상 유동 지배 방정식 (질량 보존식)**

$$\frac{\partial}{\partial t}(\alpha_g \rho_g) + \nabla \cdot (\alpha_g \rho_g \vec{u}_g) = \Gamma_v$$

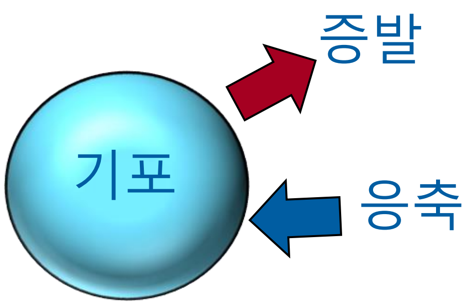
$$\int \Gamma_v dV = -\frac{V}{h_g^* - h_f^*}\left[\left(\frac{P_s}{P}\right) H_{ig}\left(T^{s,n+1} - T_g^{n+1}\right) - H_{if}\left(T^{s,n+1} - T_l^{n+1}\right)\right]$$

$$H_{ig} = a_i h_{ig} \qquad H_{if} = a_i h_{if}$$

계면면적밀도 ↗ ↖ 계면열전달계수

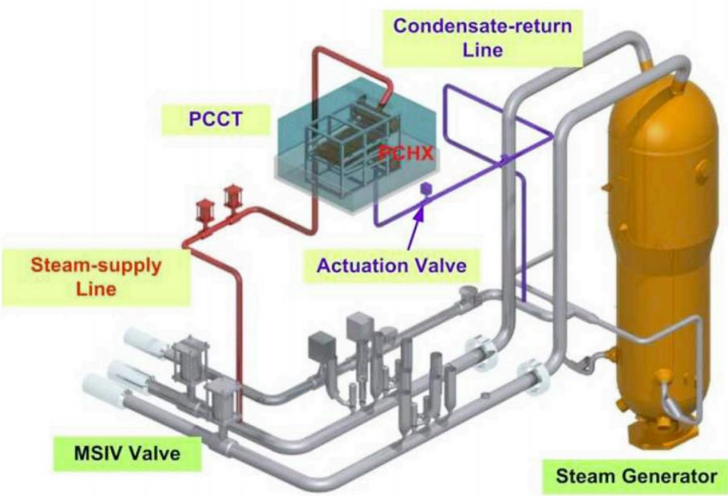
$$a_i = \frac{6\alpha}{D_{bubble}}$$

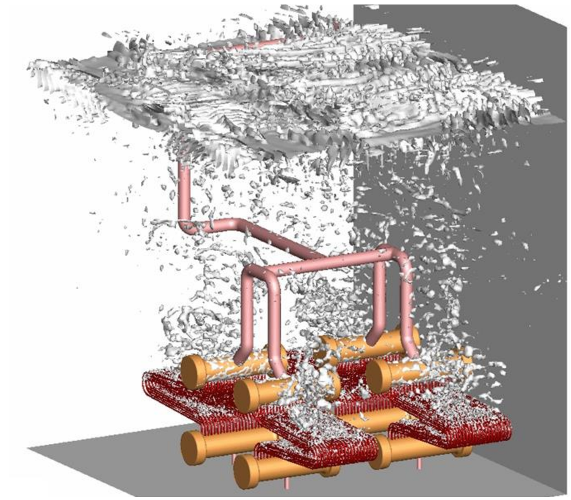
← 기포 직경

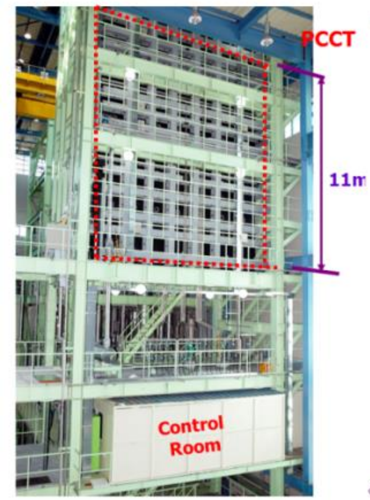


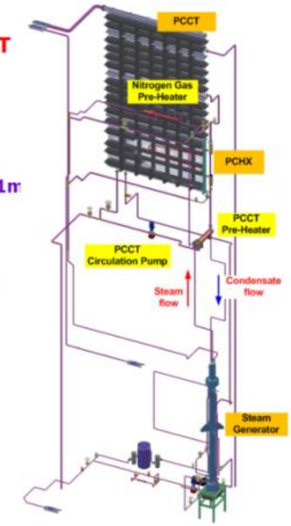
강의 개요

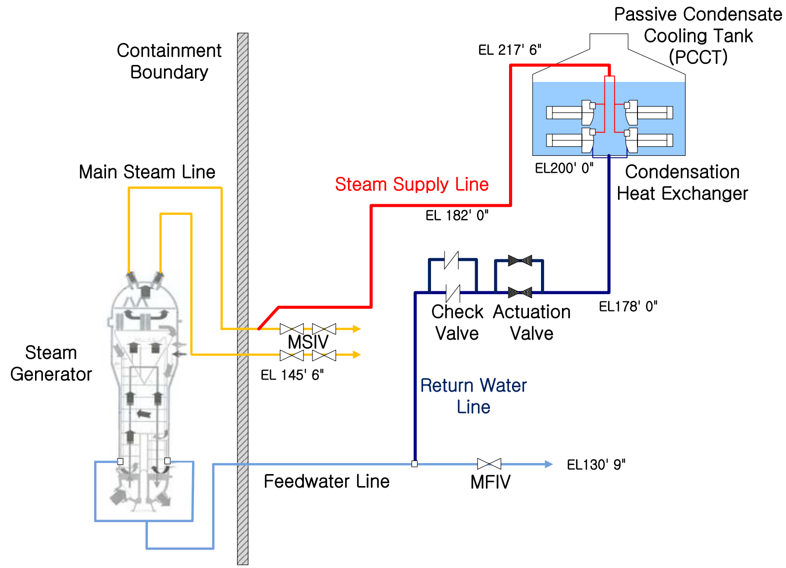
❖ **신형원자로 개발을 위한 열수력 실험 사례 (PASCAL 실험, APR+)**

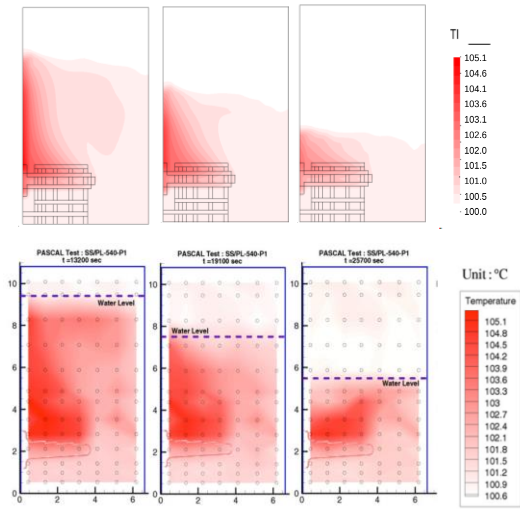


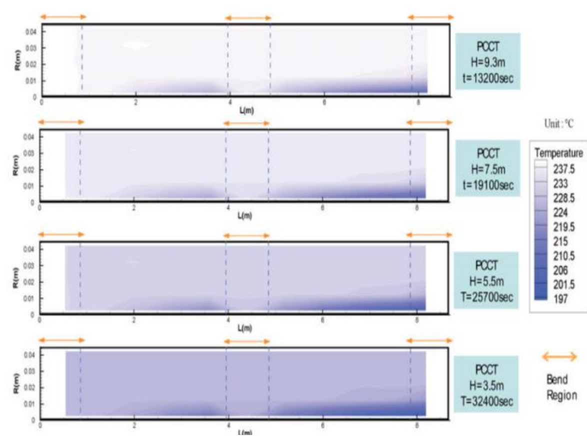












강의 개요

❖ 열수력 실험 과정

- ✓ **예비 해석**
  - 실험 전, 대상 현상에 대한 **예비 해석** 수행
  - 실험 **장치 설계** 및 실험 **조건 선정**에 활용
- ✓ **설계 및 제작**
  - **척도 해석 (scaling analysis)**: 원형 크기의 실험이 불가능한 경우
  - 예산, 공간, 전기용량, 제작 기술 한계 등에 대한 고려 필요
  - 다양한 배경 지식이 요구됨
- ✓ **측정 방법 수립**
  - **측정 방법** 선정: 가격, 오차, 적용 범위 등에 대한 고려 필요
  - **계측기 설치**: 정확한 측정이 가능하도록 규정 준수
  - **데이터 취득 시스템** 구성
  - **오차 분석**
- ✓ **실험 수행**
  - 실험 결과 분석
  - 실험 결과의 활용: 안전해석코드 평가, 모델 개발, 신규 이론 제시

강의 개요

❖ 주별 강의 및 실습 내용

주차	실습 내용	비고	조교
1 (3/3)	• CFD 기초 (동영상 강의)	• 모델링, 격자 생성	• 이창원/김신엽/유진성
2 (3/10)	• CFD 기초 (동영상 강의)	• 유체 문제	• 이창원/김신엽/유진성
3 (3/17)	• CFD 기초 (동영상 강의)	• 열전달 문제	• 이창원/김신엽/유진성
4 (3/24)	• DAQ 기초	• 열전대 설치, 히트파이프 vs 구리튜브	• 이산, 정명진
5 (3/31)	• 유량계	• 유량계 설치, DAQ 작성 • 전자기-코리올리 비교 • 공기 유량계 (압력, 온도 병행)	• 이산, 정명진
6 (4/7)	• 압력계, PID 제어	• 계장 기초, 압력계 수위, 유량별 압력	• 이산, 정명진

강의 개요

❖ 주별 강의 및 실습 내용

A

B

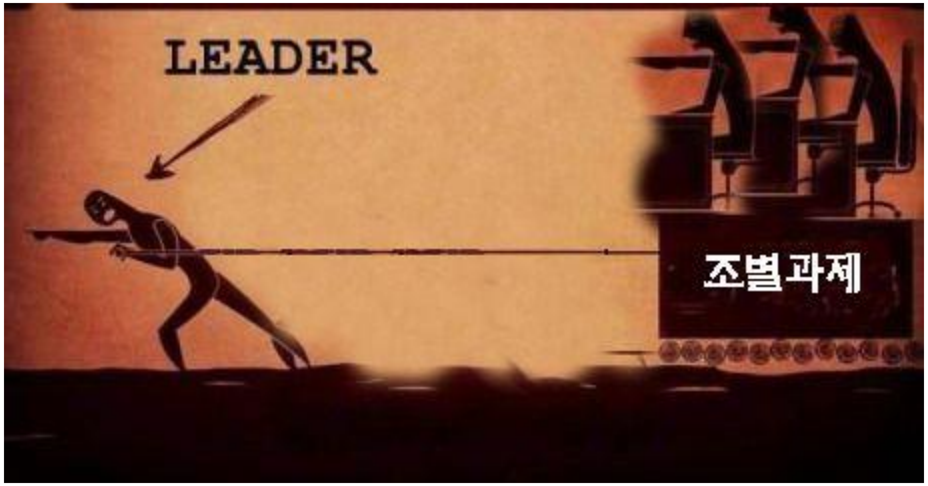
C

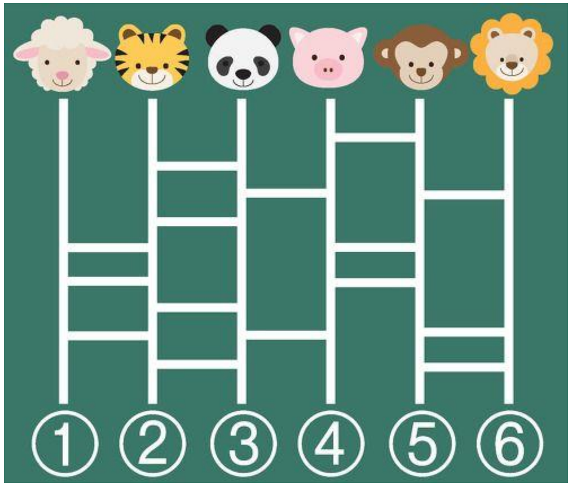
주차	실습 내용	비고	조교
7 (4/14)	• CHF 기초	• 비등 열전달 이론 • 수조 CHF 촬영, NEOUL 장치 시운전	• 김건우/유진성/홍희표
8 (4/21)	• CHF 실험-1	• 정지상태 실험	• 김건우/유진성/홍희표
9 (4/28)	• CHF 실험-2	• 요동조건 실험	• 김건우/유진성/홍희표
10 (5/12)	• ECC bypass 실험-1	• DAQ 작성, 예비 실험	• 최치진/이산
11 (5/19)	• ECC bypass 실험-2	• 기체유량, 액체유량, 비대칭 효과 실험	• 최치진/이산
12 (5/27)	• IR 카메라 실습	• IR 이론, Leidenfrost 촬영 (HSV + IR)	• 김신엽/이산
13 (6/2)	• 기포거동 실험-1	• 센서 제작, DAQ 작성, 기포율 측정	• 조형규/정명진
14 (6/9)	• 기포거동 실험-2	• HSV 촬영, 영상 처리	• 조형규/정명진
15 (6/16)	• 기포거동 실험-3	• PIV + HSV, 영상 처리	• 조형규/정명진/김신엽

강의 개요

❖ **평가**

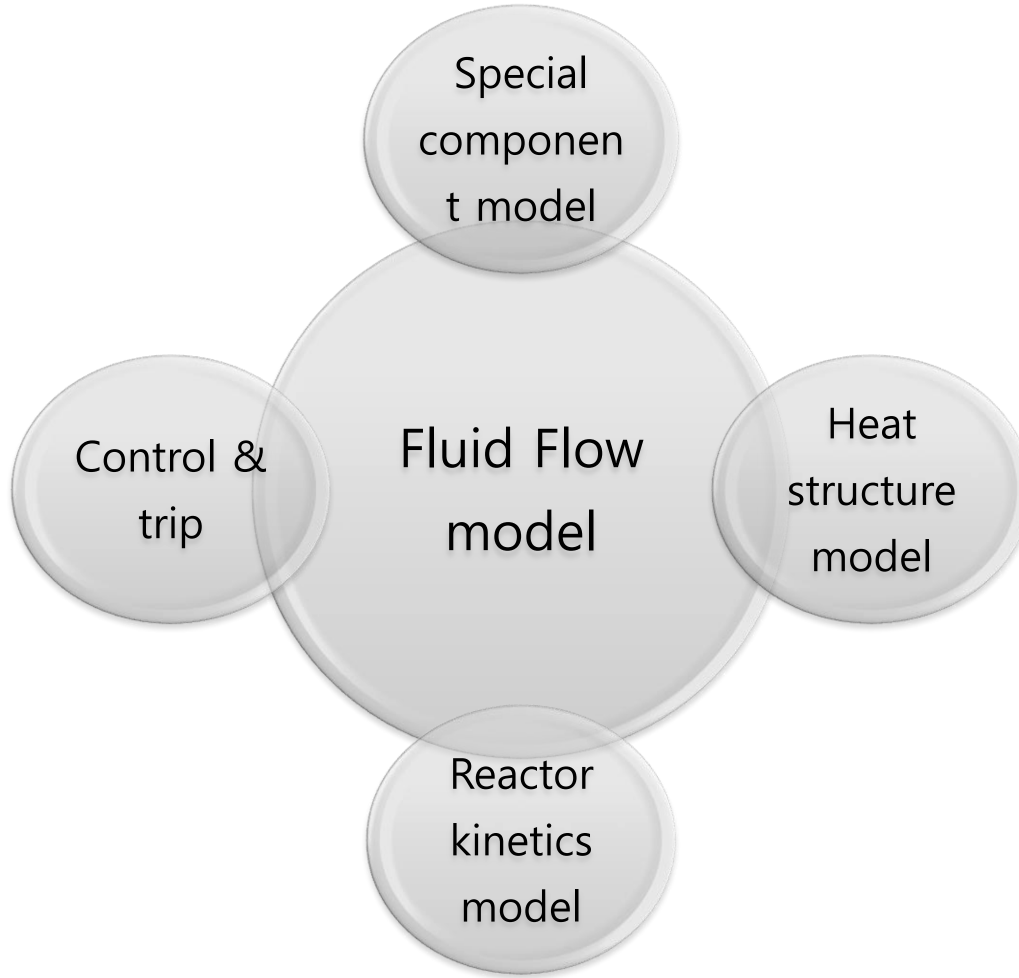
- ✓ **중간 고사 (20 %), 기말고사 (20 %), 과제 (50 %), 기타 (10 %)**
- ✓ **과제**
  - 각 실험 종료 후 이론 및 실험 결과에 대한 요약 보고서
  - 측정 데이터 및 분석 포함
  - 실습 종료 시 진행 상황 포함





Models in System Codes

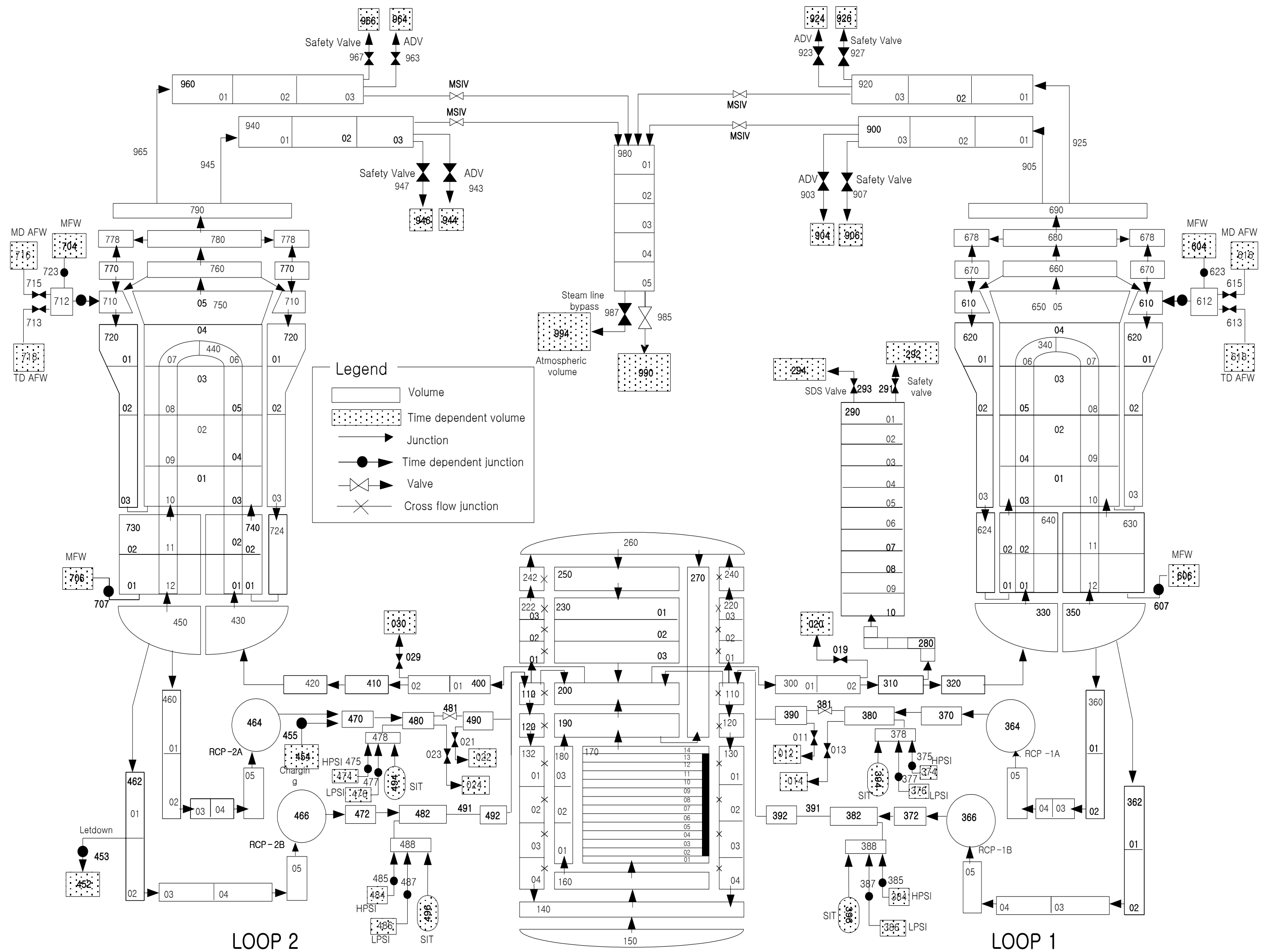
- **Fluid flow model (hydrodynamic model)**
  - 1D two-phase model
  - 3D two-phase model (with limited capability)
  - Boron and non-condensable gases
- **Special component model**
  - Pump, valve, turbine, jet pump, etc.
- **Heat structure model**
  - Heat conduction model for solid
  - Fuel rod, structure, etc.
  - 1D/2D heat conduction model
- **Reactor kinetics model**
  - Usually, point-kinetics model
  - Recently, coupled with 3D reactor kinetics model
- **Control & trip system model**



System Code Applications

- Safety analysis of NPPS
- NSSS design
- Rulemaking, licensing and audit calculation for NPP transients and accidents
- Accident management strategy development
- Basis for nuclear plant analyzer or simulator

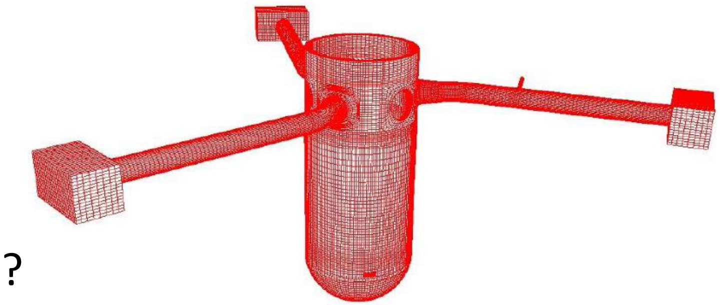
RELAP5 Nodalization for 2-Loop PWR



STH and CFD

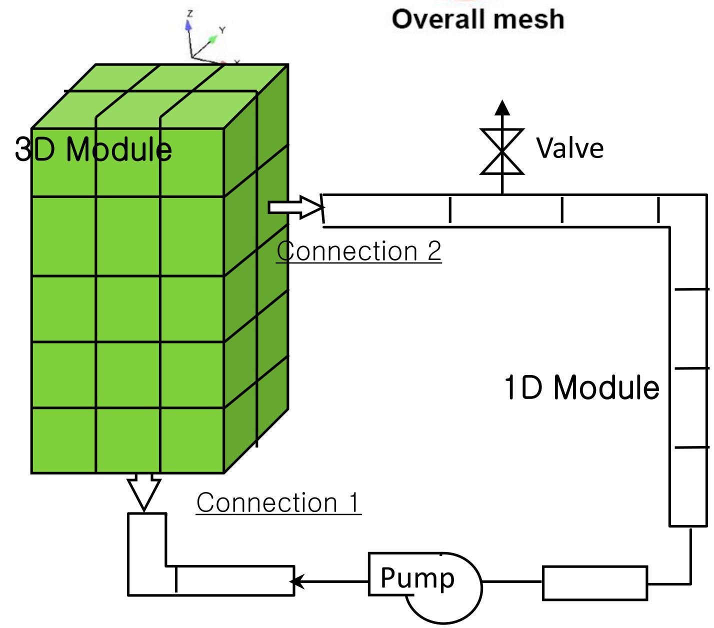
- MARS (Multi-dimensional Analysis of Reactor Safety)
  - **Developed by KAERI**
  - **Based on the unified version of RELAP5 (1D) and COBRA-TF (3D)**
  - **Nuclear long-term R&D program of Korean government since 1997**
    - 1997~2006: Development stage
    - 2007~ present: QA and regulatory utilization stage
  - **Main features**
    - Multi-D module
    - Coupled calculation capabilities
      - 3D reactor kinetics code: MASTER
      - Containment analysis code: CONTEMPT4 and CONTAIN
    - Code restructuring
    - Written in FORTRAN90
    - User friendly feature
    - Systematic verification and validation (V&V)

Copyright problem ?

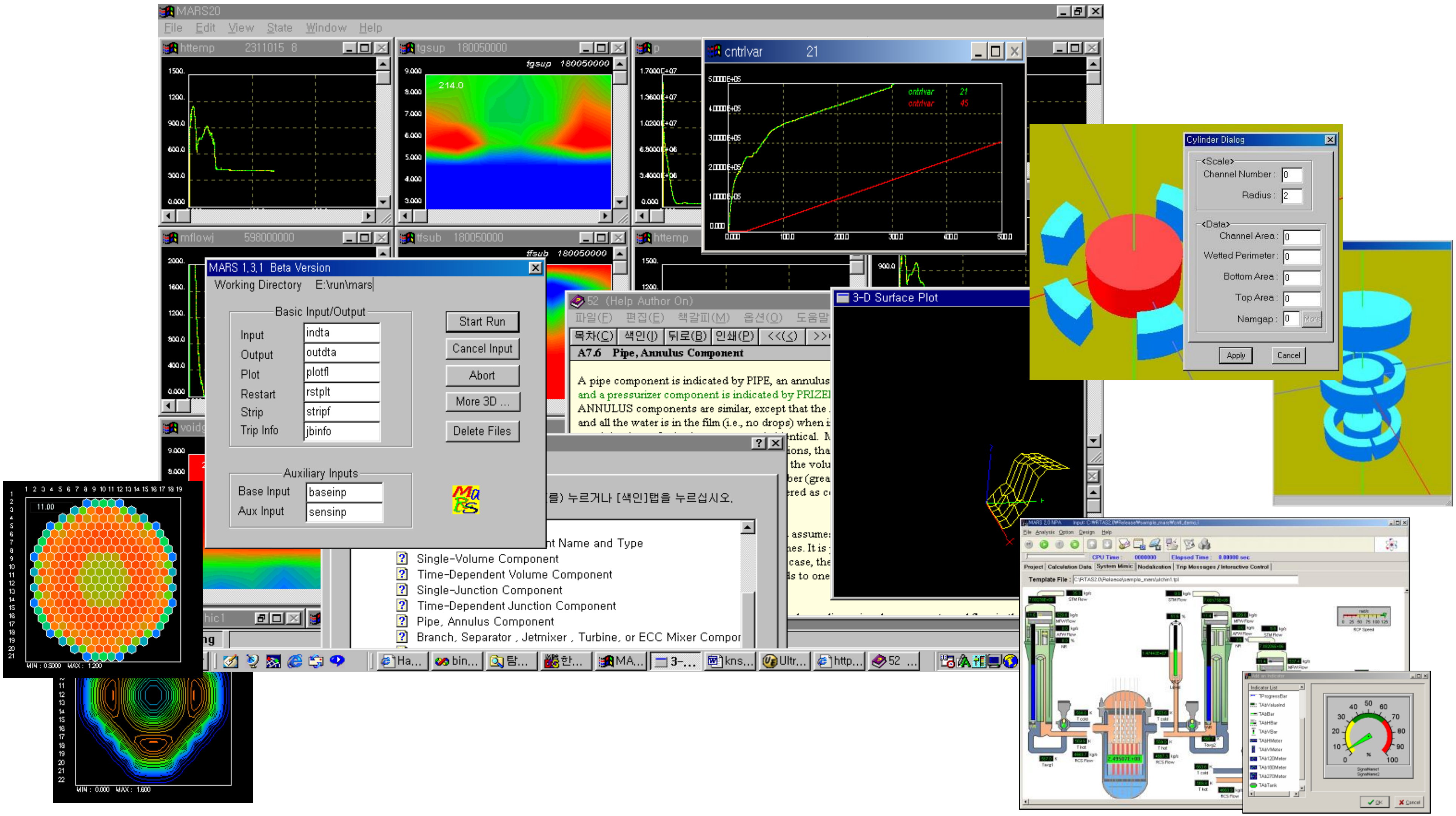


Overall mesh

3D ?

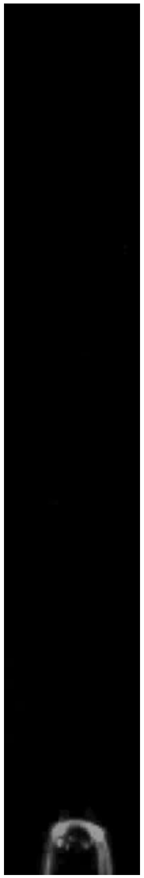


❖ MARS (Multi-dimensional Analysis of Reactor Safety)



STH and CFD

- Shortcomings of current system codes
  - **Flow regime map approach for two-phase flow**
    - Developed under steady-state, fully developed flow conditions
    - Used for unsteady, developing flow conditions
  - **Subcooled boiling model**
    - Complicated nature, wide application rage
  - **Droplet model**
    - Difficulty in measurement
  - **Multi-dimensional two-phase flow**
    - One-D governing equation
  - **Condensation under the presence of non-condensable gases**
  - **Numerical diffusion of first-order scheme**



❖ US R&D Roadmap

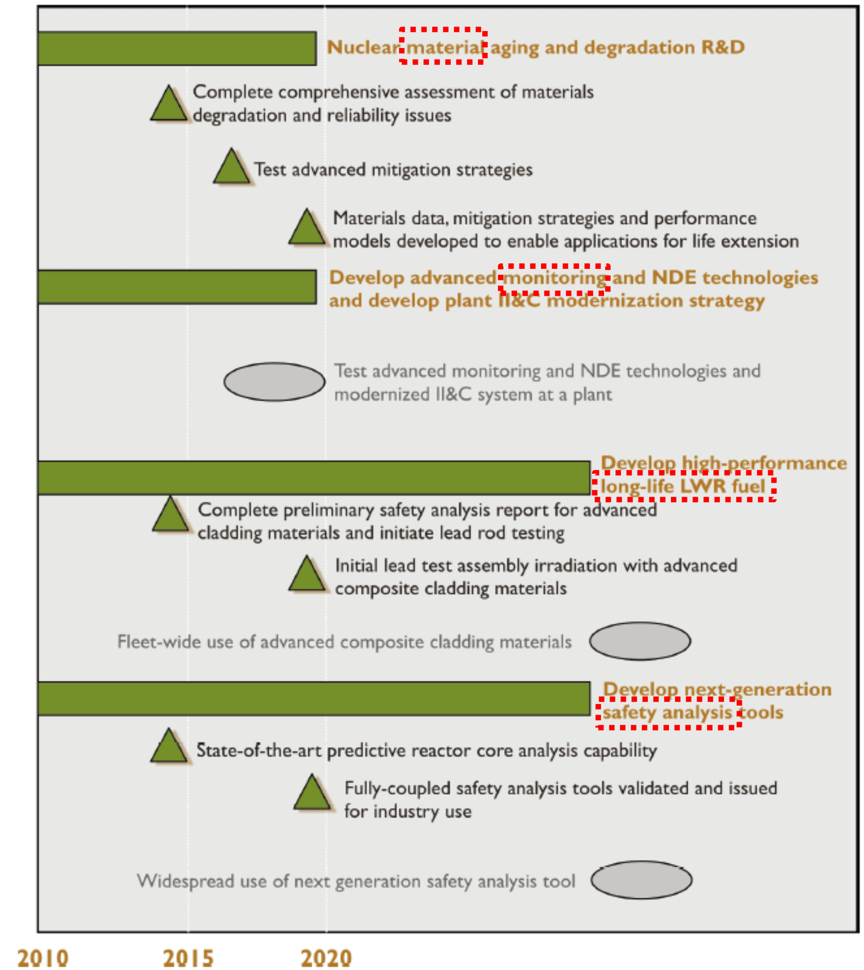
Challenges

- Aging and degradation of system structures and components
- Fuel reliability and performance issues.
- Obsolete analog instrumentation and control technologies
- **Design and safety analysis tools based on 1980s vintage knowledge bases and computational capabilities**

R&D Topics

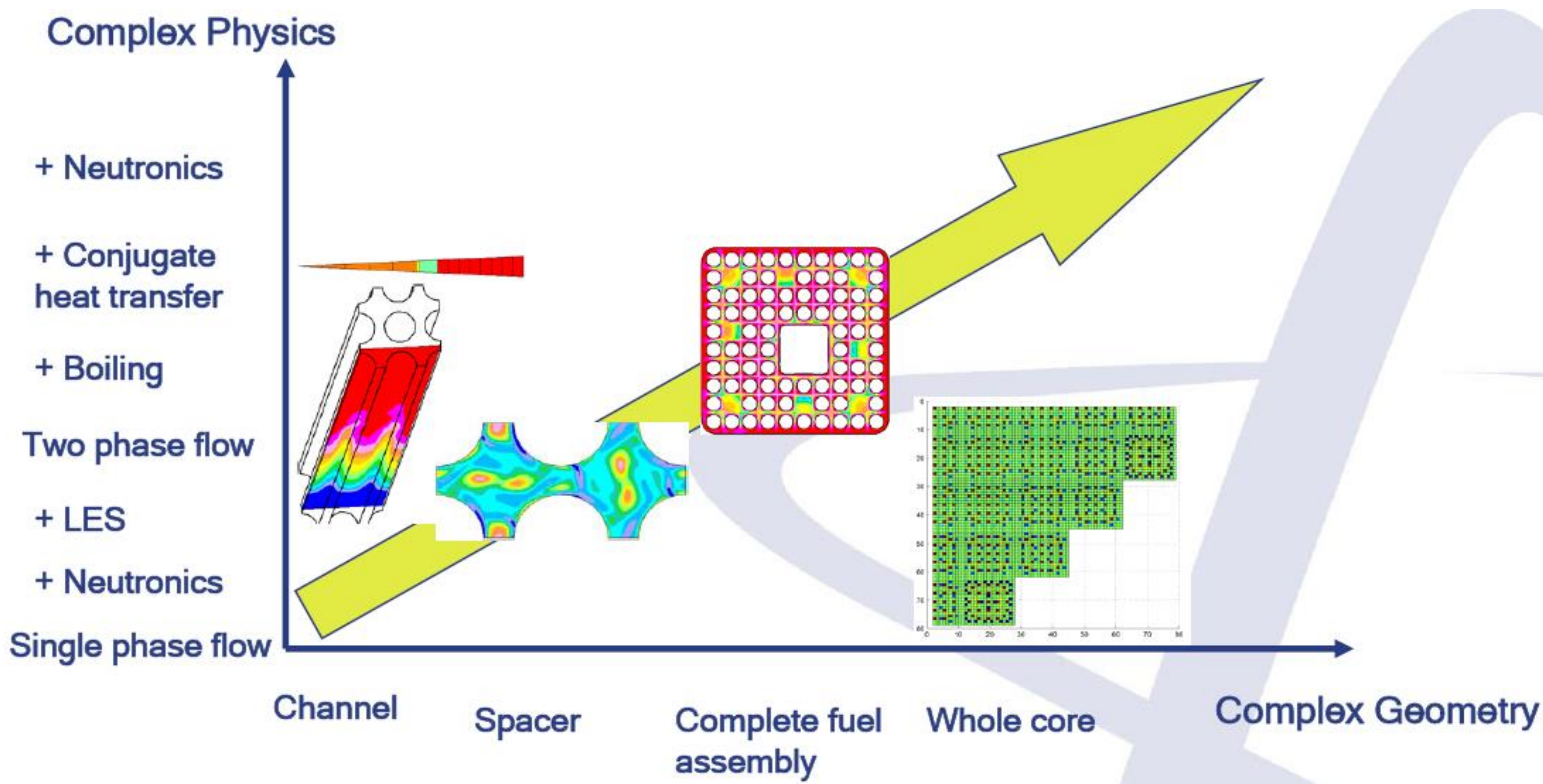
- Nuclear Materials Aging and Degradation
- Advanced LWR Nuclear Fuel Development
- Advanced Instrumentation, Information, and Control (II&C) System Technologies
- Risk-Informed Safety Margin Characterization (RISMC)
- Efficiency Improvement
- **Advanced Modeling and Simulation Tools**

R&D for current reactors



Nuclear Energy Research and Development Roadmap, U.S. Department Of Energy, 2010

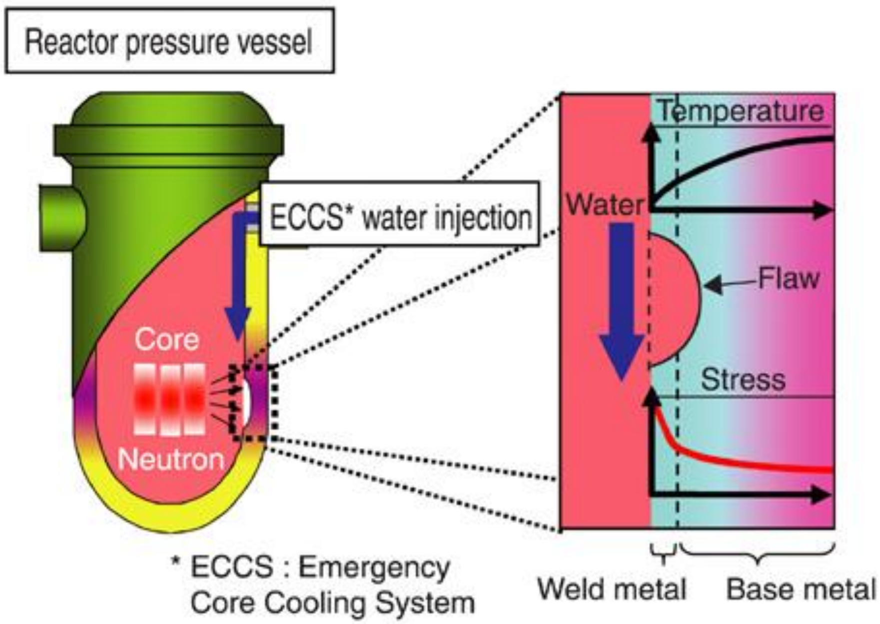
- Extension of CFD codes to nuclear reactor safety problems



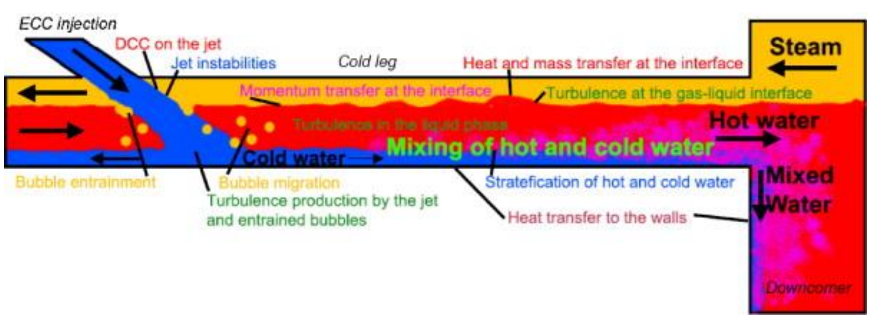
Identification of NRS problems where extension of CFD capabilities to two-phase flow may bring real benefit

	NRS problem	Maturity of present CFD tools
1	**DNB, dry out and CHF investigations**	M
2	Subcooled boiling	M
3	**Two-phase pressurized thermal shock**	M
4	Direct contact condensation: steam discharge in a pool	M
5	Pool heat exchangers: thermal stratification and mixing problems	H
6	Corrosion Erosion deposition	L
7	Containment thermal-hydraulics	H
8	Two-phase flow in valves, safety valves	L
9	ECC bypass and downcomer penetration during refill	L
10	Two phase flow features in BWR cores	M
11	Atmospheric transport of aerosols outside containment	M
12	DBA reflooding	M
13	Reflooding of a debris bed	L
14	Steam generator tube vibration	L
15	Upper plenum injection	L
16	Local 3-D effects in singular geometries	L
17	Phase distribution in inlet and outlet headers of steam generators	L
18	Condensation induced waterhammer	L
19	Components with complex geometry	L
20	Pipe Flow with Cavitation	M
21	External reactor pressure vessel cooling	M
22	Behaviour of gas-liquid interfaces	M
23	Two-phase pump behaviour	L
24	Pipe Break-In vessel mechanical load	M
25	Specific features in Passive reactors	M

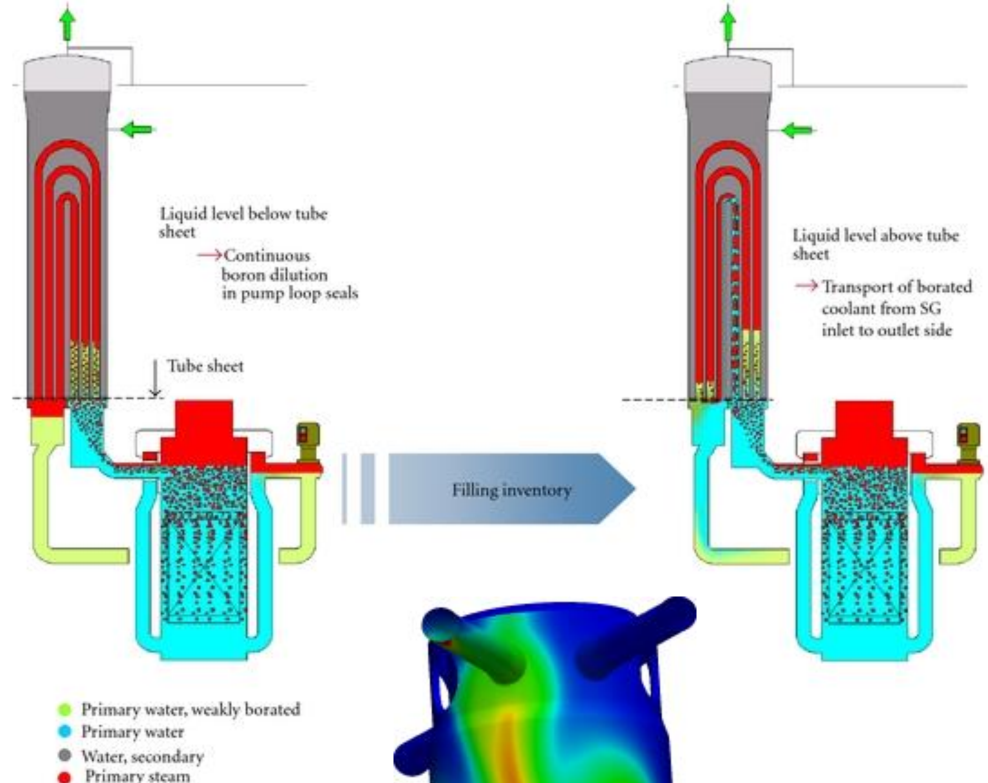
STH and CFD



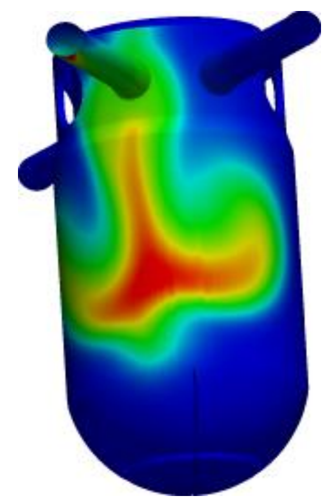
Reactor pressure vessel
ECCS* water injection
Core
Neutron
Temperature
Water
Flaw
Stress
Weld metal
Base metal
* ECCS : Emergency
Core Cooling System



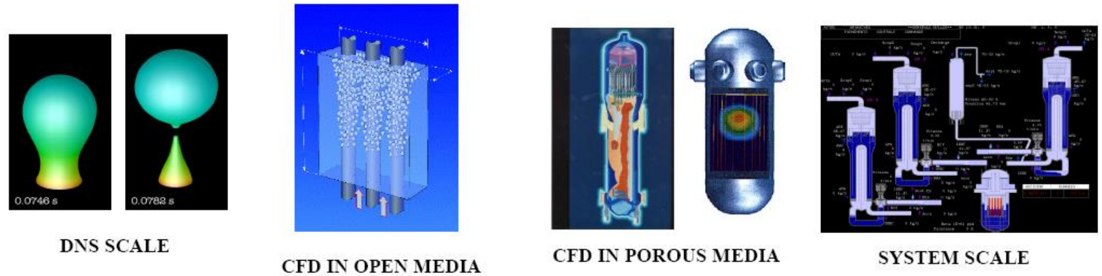
ECC injection
DCC on the jet
Jet instabilities
Cold leg
Heat and mass transfer at the interface
Steam
Momentum transfer at the interface
Turbulence at the gas-liquid interface
Turbulence in the liquid phase
Hot water
Mixing of hot and cold water
Cold water
Bubble entrainment
Bubble migration
Turbulence production by the jet
and entrained bubbles
Stratefication of hot and cold water
Heat transfer to the walls
Mixed
Water
Downcomer

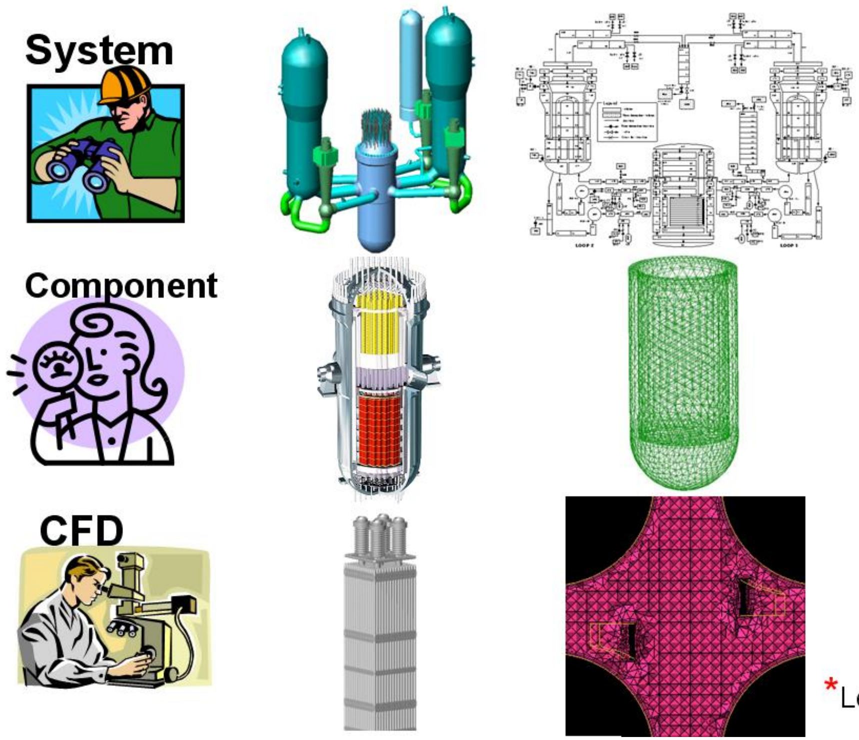


Liquid level below tube
sheet
→Continuous
boron dilution
in pump loop seals
Tube sheet
Filling inventory
Liquid level above tube
sheet
→ Transport of borated
coolant from SG
inlet to outlet side
Primary water, weakly borated
Primary water
Water, secondary
Primary steam



❖ Multi-Scale Approach





~10^0 m, # ~$10^{\sim3}$ *

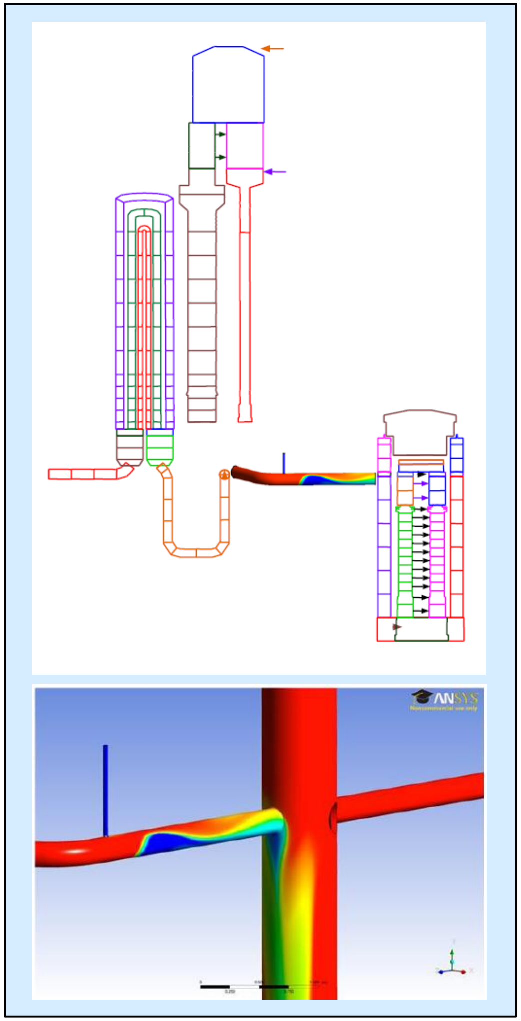
~10^{-1} m, # ~$10^{\sim5}$

>10^{-2} m, # ~$10^{\sim9}$

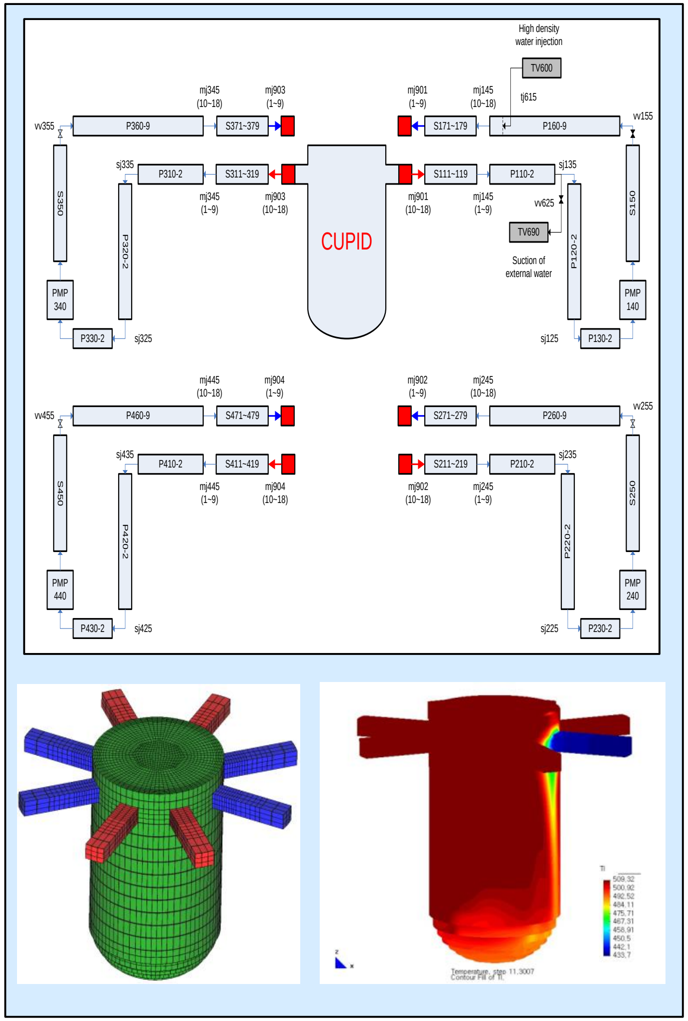
*Length Scale, # of meshes

Multi-Scale Approach

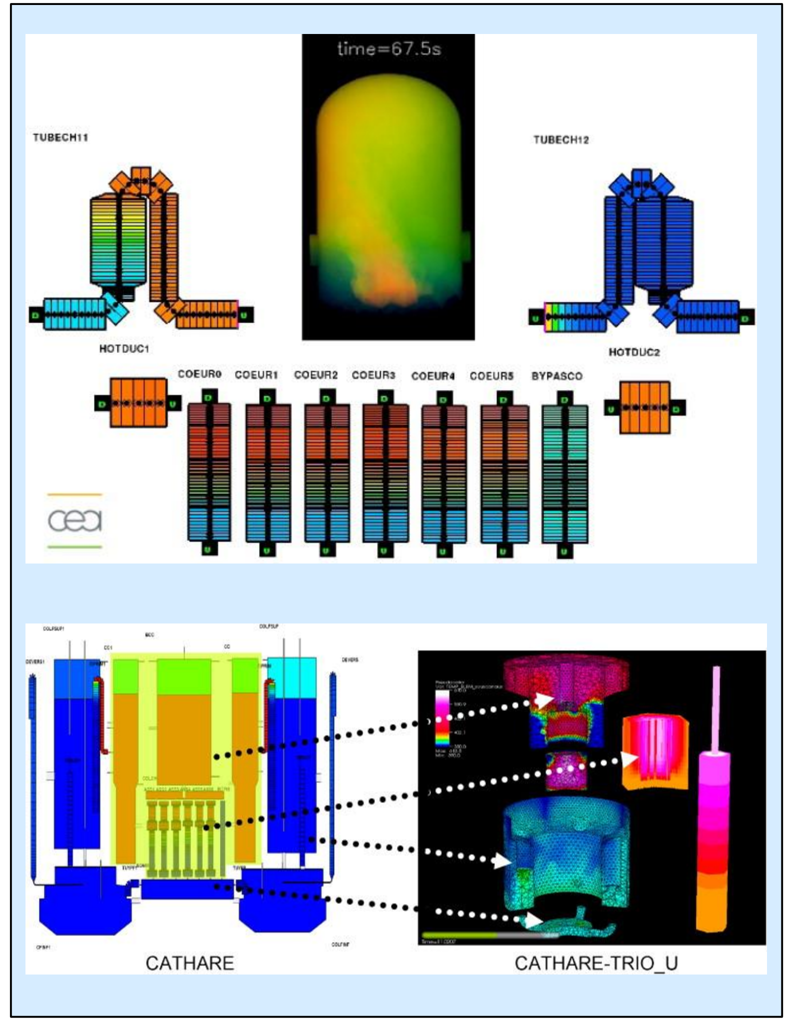
ATHLET + CFX



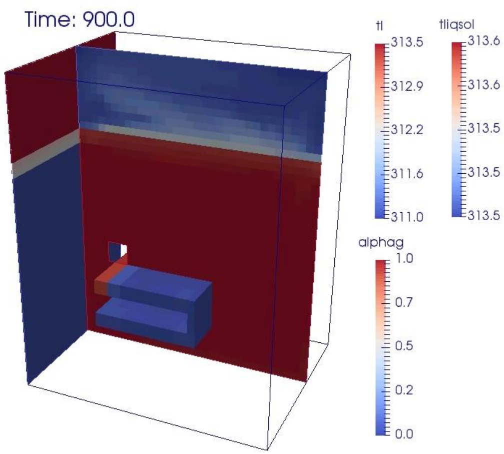
CUPID + MARS

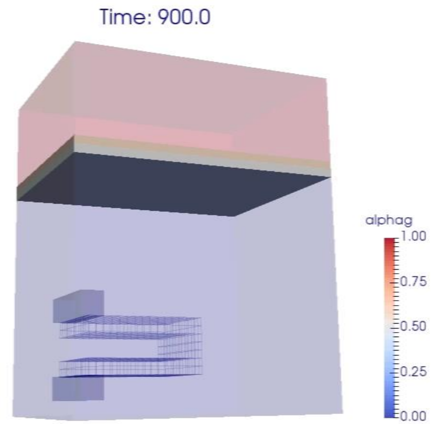


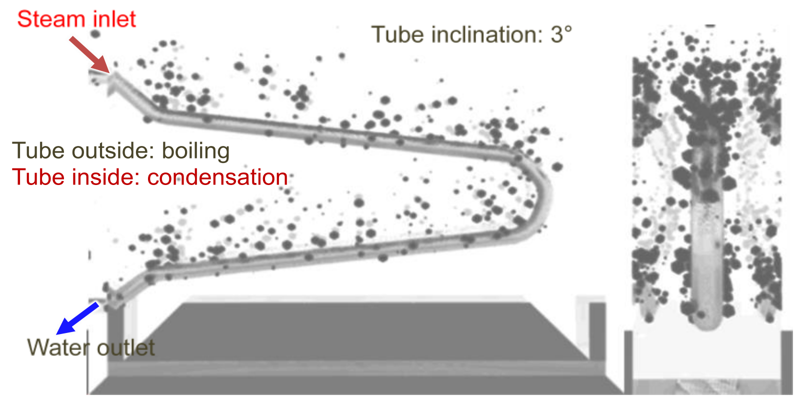
CATHARE + TrioU

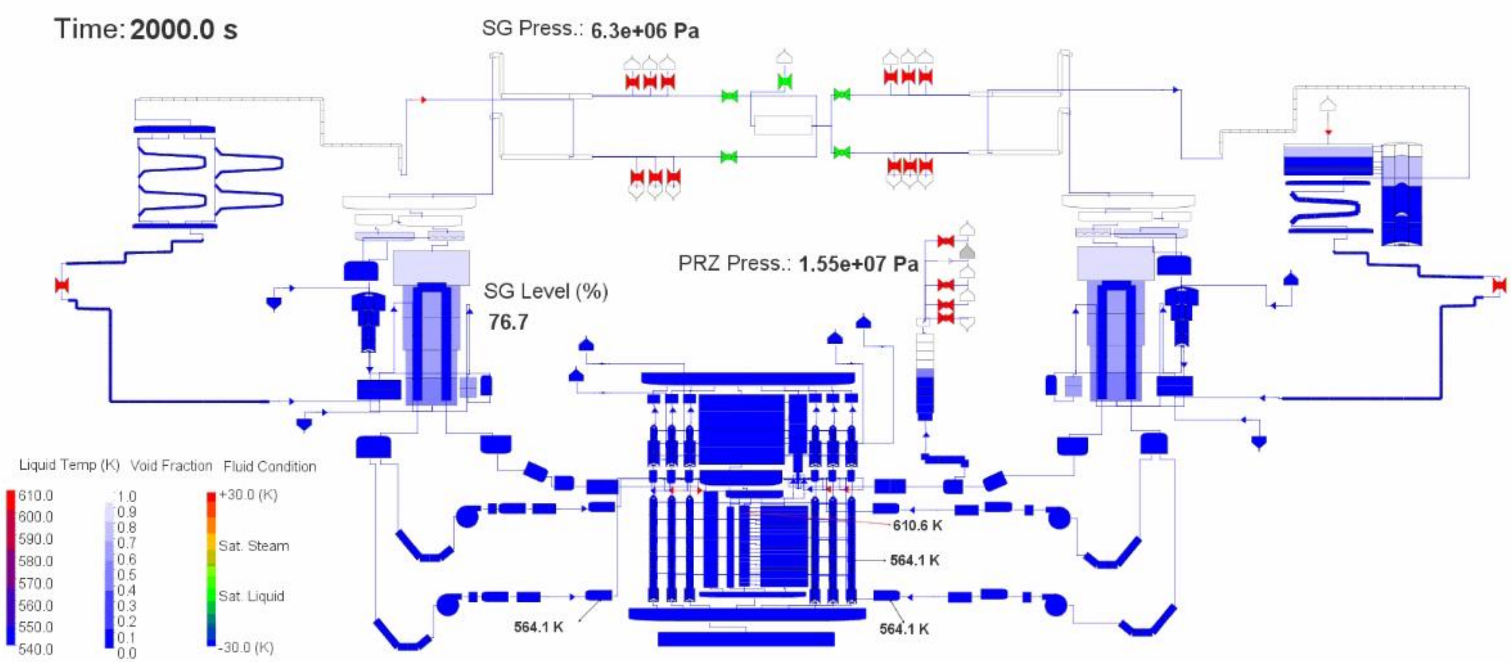


Multi-Scale Approach









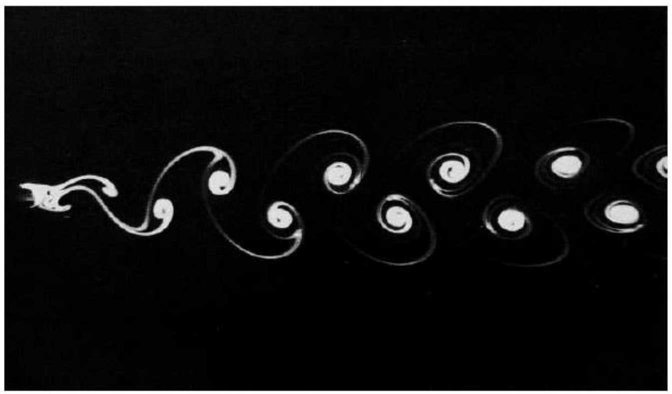
- Multi-Scale Approach

Scales	Models	
System Scale System Codes	0D, 1D 3D Porous (Coarse)	Simplified geometry Classical flow regime map Wall friction term Form loss term Wall heat transfer term
Component Scale Component Codes	3D Porous (finer)	
Local Scale CFD Codes	3D CFD Open Medium	Realistic geometry Local flow regime map Non-drag force terms
Micro Scale	DNS VOF, Level Set	Realistic geometry Interface tracking

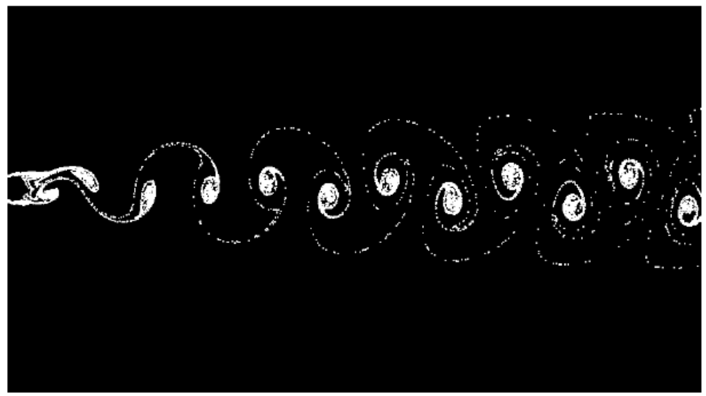
Computational Fluid Dynamics

- A branch of computer-based science that provides predictions of fluid flows
- Mathematical modeling (typically a system on non-linear, coupled PDEs)
- Numerical methods (discretization and solution techniques)
- Software tools

- CFD enables scientists and engineers to perform ‘numerical experiments’ (i.e. computer simulations) in a ‘virtual flow laboratory’.



Real experiment



CFD simulation

Computational Fluid Dynamics

Experiments vs. simulations

- CFD gives an insight into flow patterns that are difficult, expensive or impossible to study using traditional (experimental) techniques

Experiments	Simulations
Quantitative description of flow phenomena using measurements • for one quantity at a time • at a limited number of points and time instants • for a laboratory-scale model • for a limited range of problems and operating conditions Error sources: measurement errors, flow disturbances by the probes	Quantitative prediction of flow phenomena using CFD software • for all desired quantities • with high resolution in space and time • for the actual flow domain • for virtually any problem and realistic operating conditions Error sources: modeling, discretization, iteration, implementation

Computational Fluid Dynamics

- As a rule, CFD does not replace the measurements completely but the amount of experimentation and the overall cost can be significantly reduced.

Experiments	Simulations
• expensive • slow • sequential • single-purpose • Difficult to transport	• cheap(er) • fast(er) • parallel • multiple-purpose • Portable, easy to use and modify

- **The results of a CFD simulation are never 100% reliable because**
  - the input data may involve too much guessing or imprecision
  - the mathematical model of the problem at hand may be inadequate
  - the accuracy of the results is limited by the available computing power

- **It is not a magic tool !**

Computational Fluid Dynamics

Classification of fluid flows

• Viscous • Compressible • Steady • Laminar • Single -phase	• Inviscid • Incompressible • Unsteady • Turbulent • Multiphase

The reliability of CFD simulations is greater

- For laminar/slow flows than for turbulent/fast ones
- For single-phase flows than for multi-phase flows
- For steady than for unsteady flow

Nuclear thermal-hydraulics

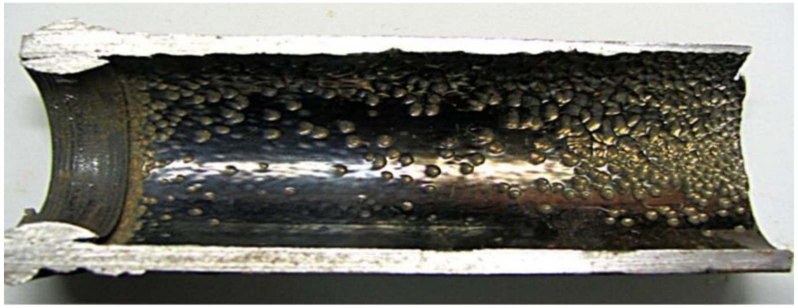
- Unsteady flow: fast transient
- Highly turbulent flow
- Mutliphase flow: various flow patterns

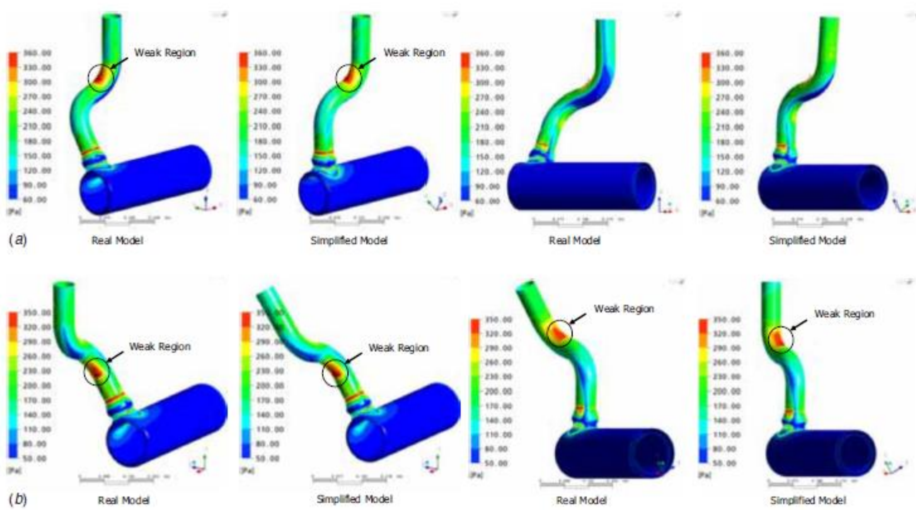
\+ complexity of geometry
\+ insufficient validation data
\+ lack of universal governing equations

STH and CFD

CFD in NRS

FAC





18" elbow wall thickness decreased from 12.7 to 1.5 mm on feed-water pump inlet at Surry, 1986

Wall Thickness reduced from 10 to 1.5 mm on Feed-water piping at Mihama unit 3, 2004

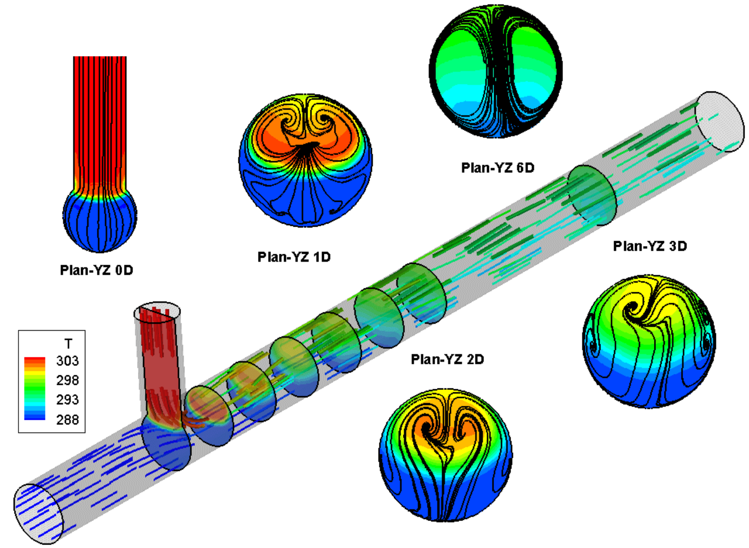
Failure in a high pressure extraction line at Fort Calhoun in 1997

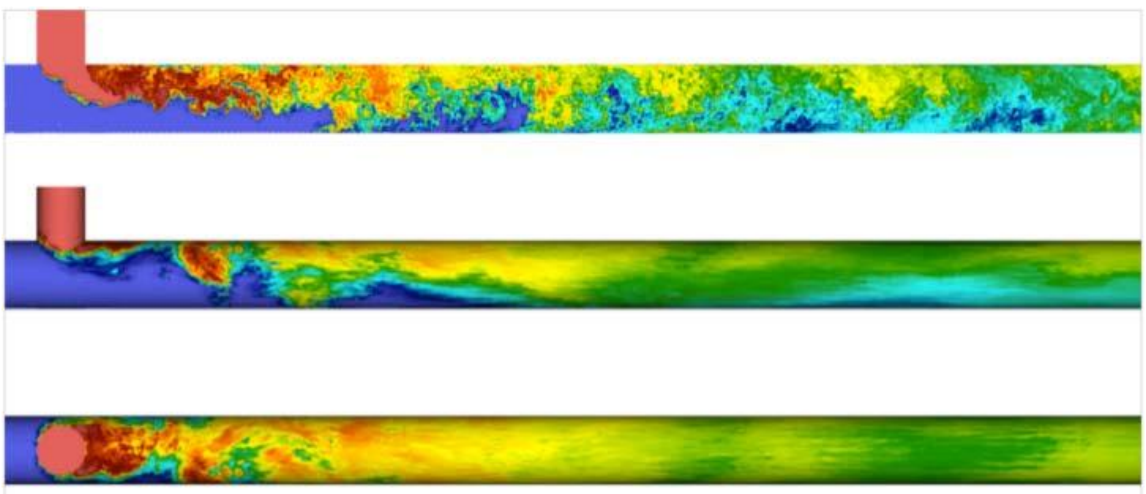
Failure downstream of the LCV in the reheater drain line at Millstone unit 2, 1991.

Failure of 14" heater drain extraction line to high pressure heater at Arkansas unit 2, 1986

Failure of the Feed-water Heater Point Beach Unit 1, 1999

- CFD in NRS
  - **Thermal cycling**
    - Thermal stripping
    - High frequency thermal fluctuation on the inner surface of a component
    - Can cause the propagation of deep cracks
    - Civaux N4 class reactor
      - shut down in May 1998 following a leak of primary coolant from a pipe in the Residual Heat Removal (RHR) system





CFD in NRS

Natural convection

- Passive mechanism of heat removal
- Small modular reactors
- Passive safety systems