Texas A&M University
Thermal-Hydraulic Research Activity

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2011 RELAP5 International
User's Group Seminar and Meeting

September 12-13, 2013
Idaho Falls, ID

Table of Content

- Generation IV
- Generic Safety Issue - 191

U.S. DEPARTMENT OF ENERGY

NEUP Nuclear Energy University Programs
U.S. Department of Energy

Generation IV
Reactor Cavity Cooling System

WISCONSIN
UNIVERSITY OF WISCONSIN–MADISON

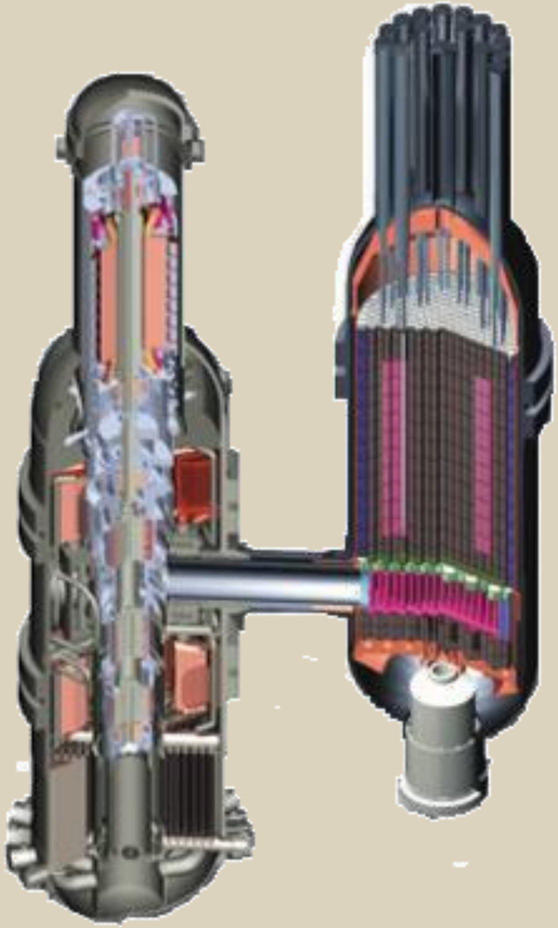
University of Idaho
College of Engineering

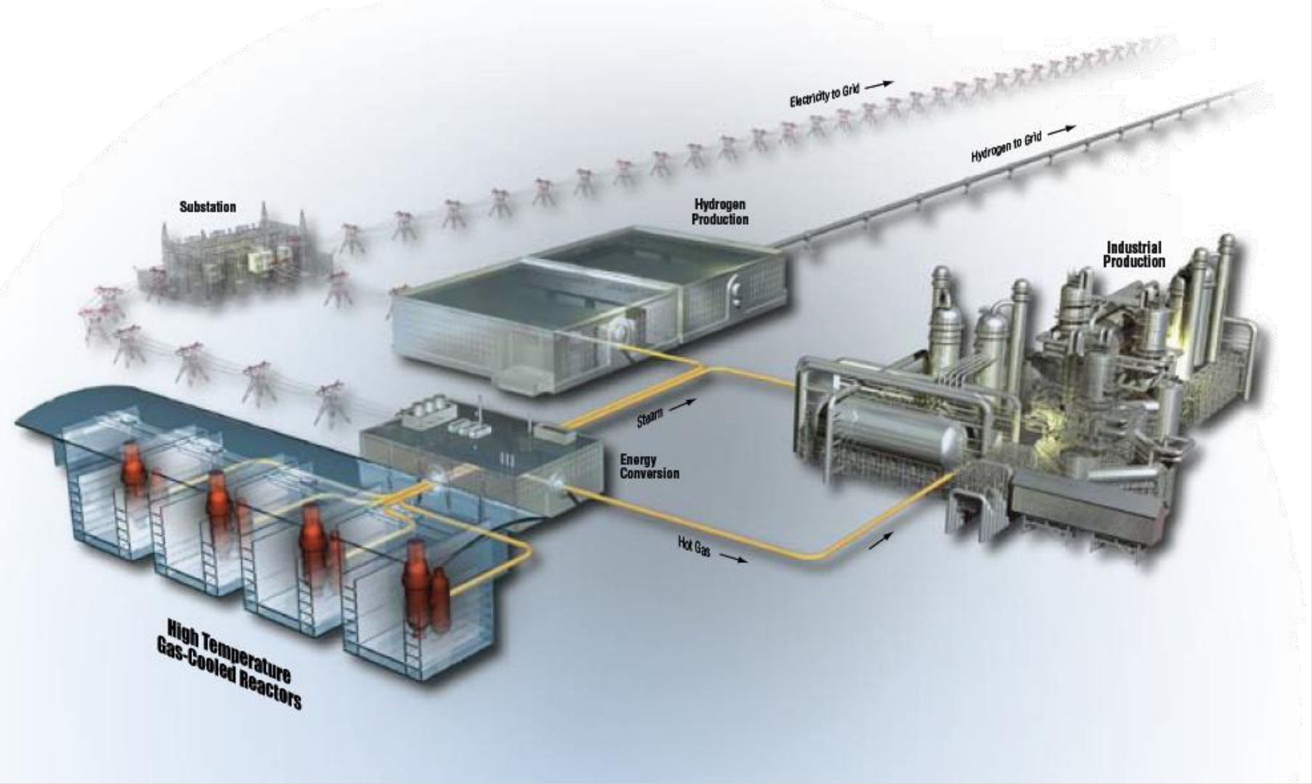
Argonne NATIONAL LABORATORY

INTRODUCTION – High Temperature Gas-Cooled Reactors

One of the Six Generation IV Designs Proposed

- Outlet Temperatures: 700ºC - 850ºC
- Modules of 200MWt – 625MWt

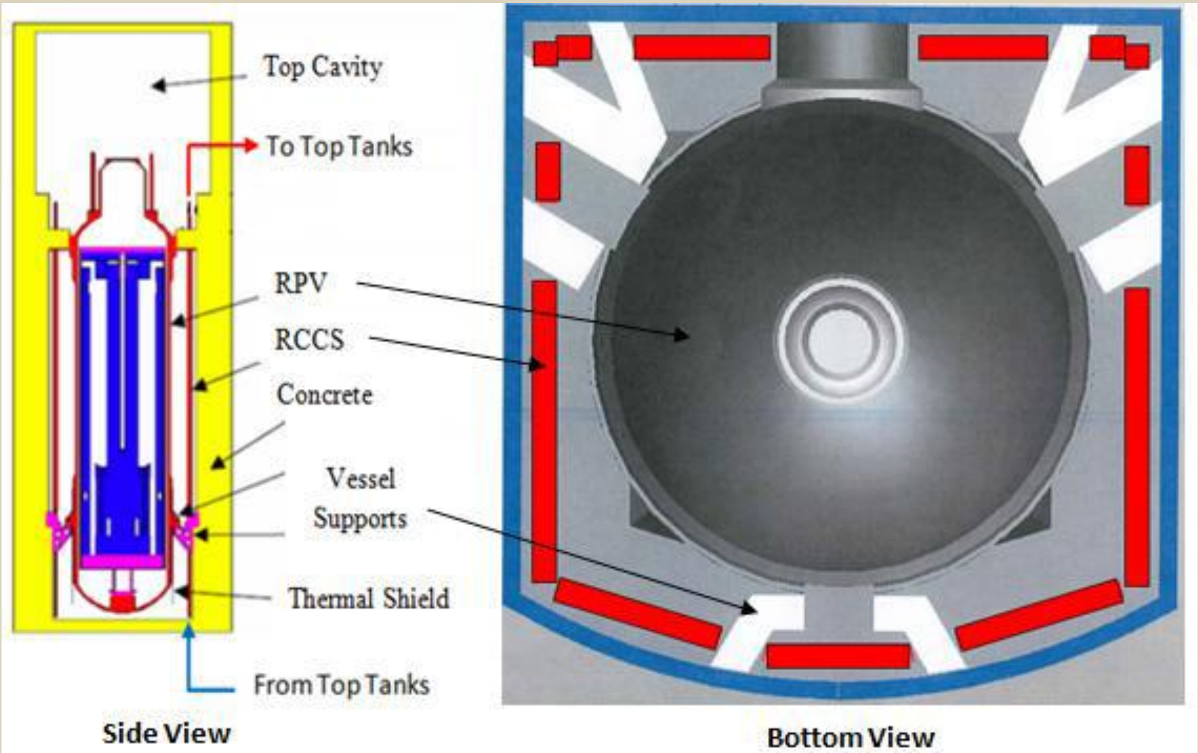




INTRODUCTION – Reactor Cavity Cooling System

New Safety Features: RCCS

- Designed to passively remove the heat from the reactor cavity during normal operation and under accident conditions;
- Two Proposed Coolants:
  - Air (Open Loop)
  - <u>Water (Closed Loop)</u>



25 Riser's Panels
9 Risers per Panel

OBJECTIVES

1. Conduct scaled test to study the thermal-hydraulic behavior of a water-cooled RCCS under different operating conditions;
2. Identify and Analyze specific phenomena occurring during the single-phase and the two-phase flow stages of the operation;
3. Develop and refine computational models (systems codes and Computational Fluid Dynamics codes) to analyze these phenomena;
4. Produce experimental data to be used for computational codes validation.

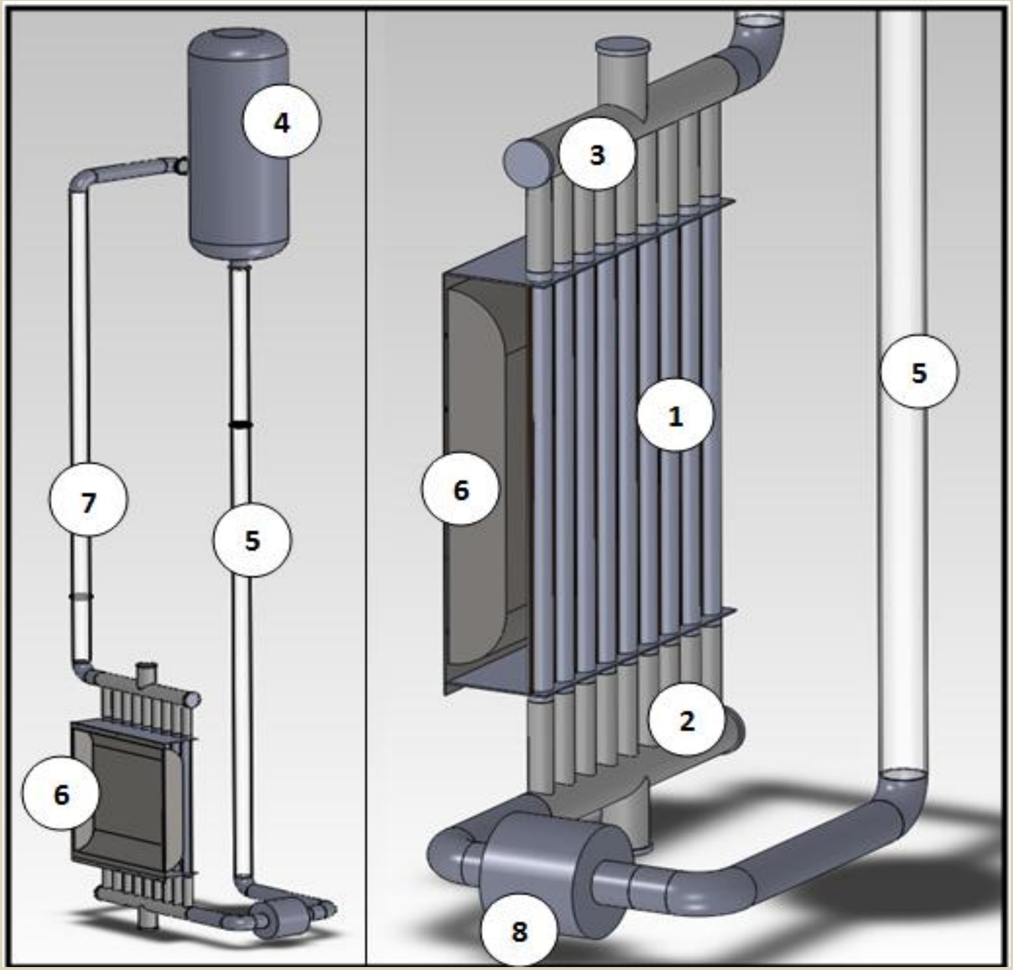
EXPERIMENTAL

Scaling, designing, building and operating a small-scale water-cooled RCCS to be used to conduct single-phase (steady-state) and two-phase (transient) experiments.

COMPUTATIONAL

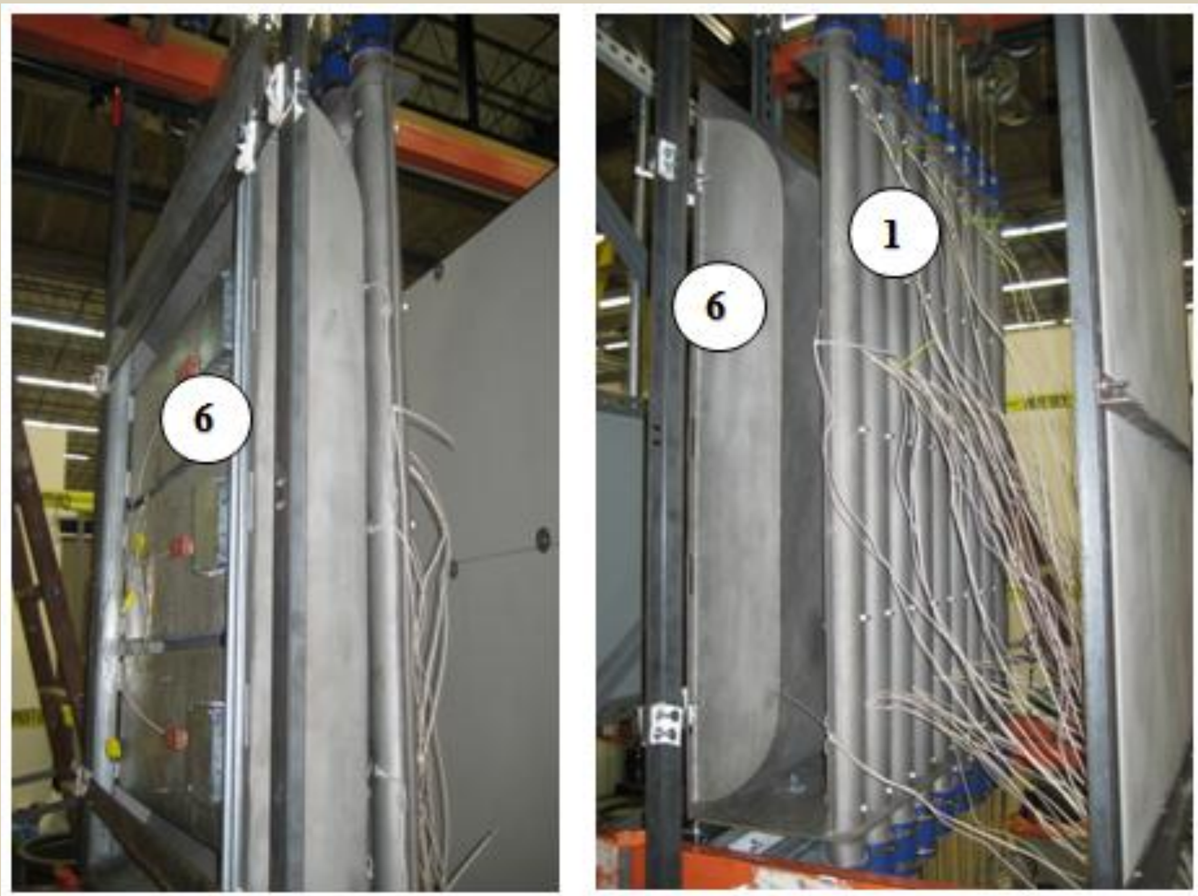
Selecting a system code, developing and refining a dedicated model to conduct the simulations of the full-scale power plant and the experimental facility.

EXPERIMENTAL FACILITY OVERVIEW



1. Risers Panel
2. Bottom Manifold
3. Top Manifold
4. Water Tank
5. Downcomer
6. Vessel
7. Upward Pipeline
8. Magnetic Flowmeter

EXPERIMENTAL FACILITY OVERVIEW



1 Risers Panel

6 Vessel/Heaters

3 Electric Radiant Heaters. Total Power Installed: 24 kW

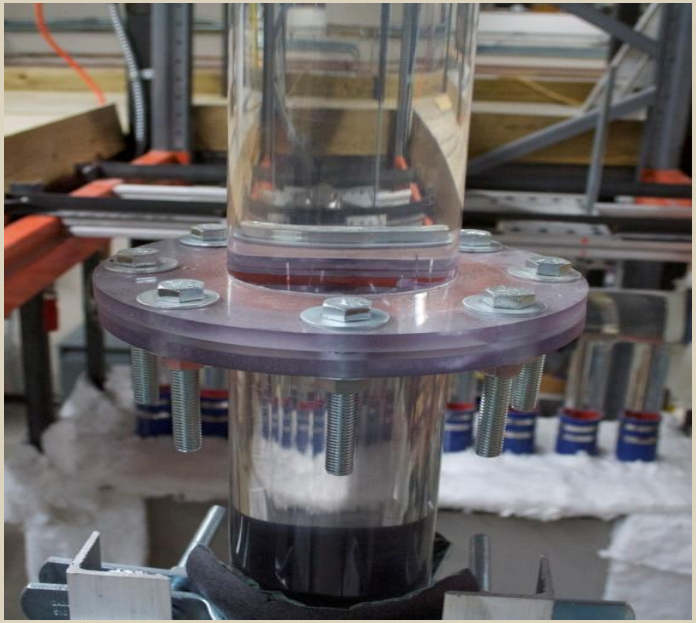
EXPERIMENTAL FACILITY OVERVIEW

4 Water Tank

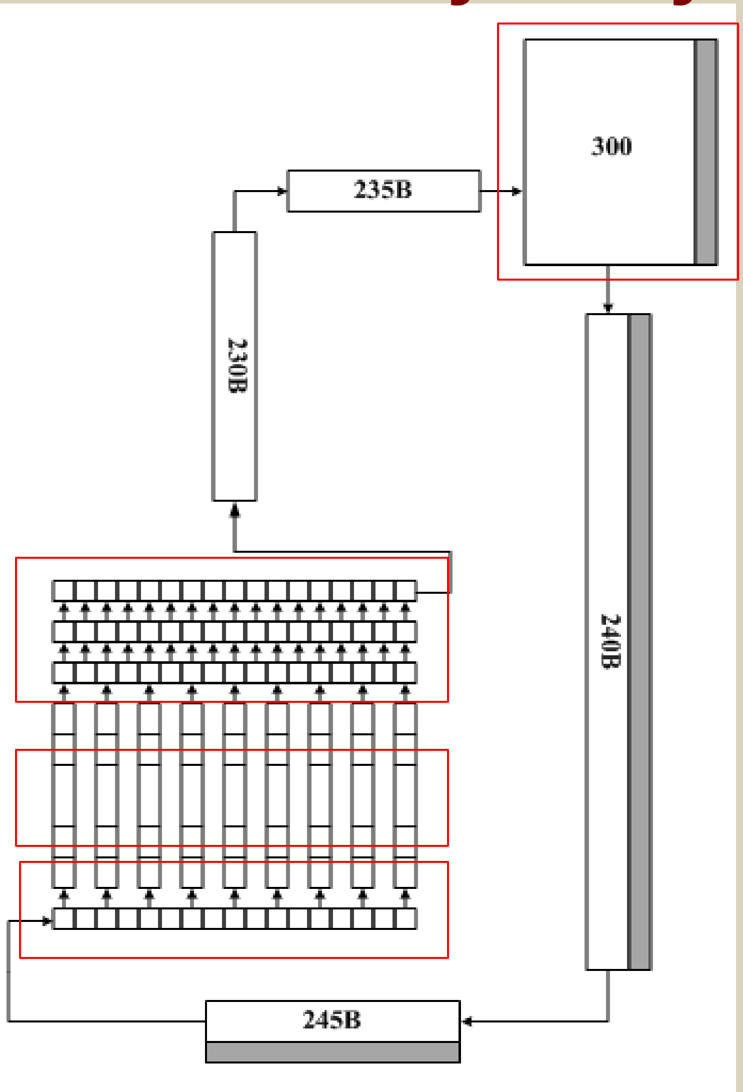
5 Downcomer

7 Upward Pipeline

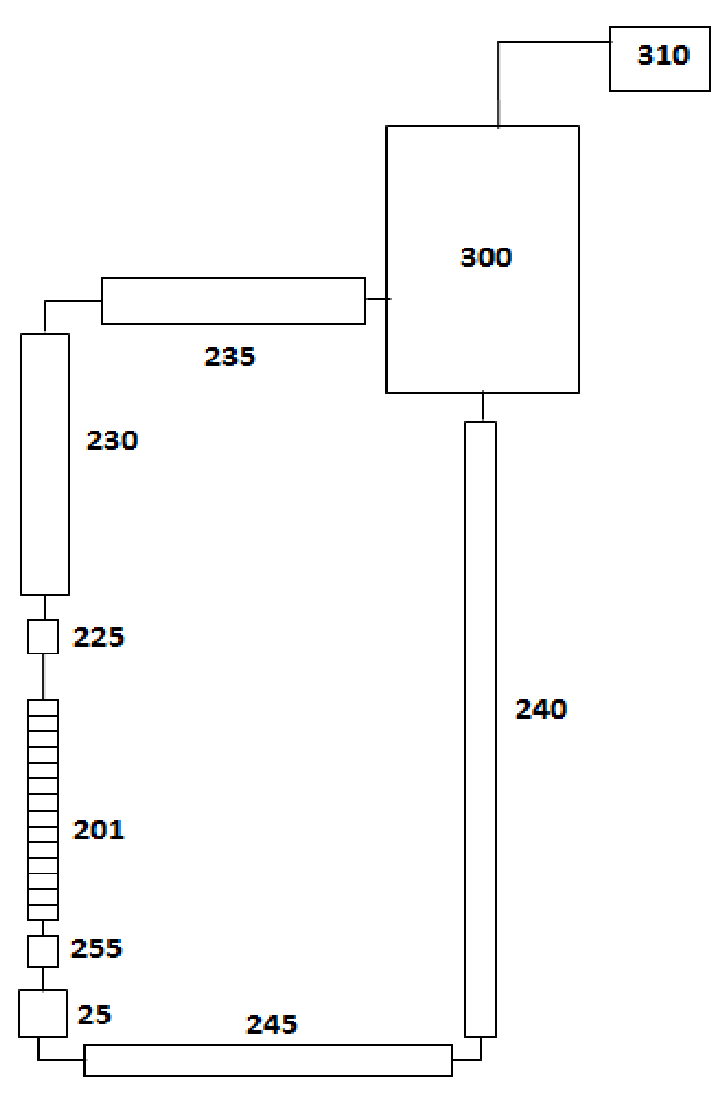
EXPERIMENTAL FACILITY OVERVIEW



RELAP5-3D Hydrodynamic Model

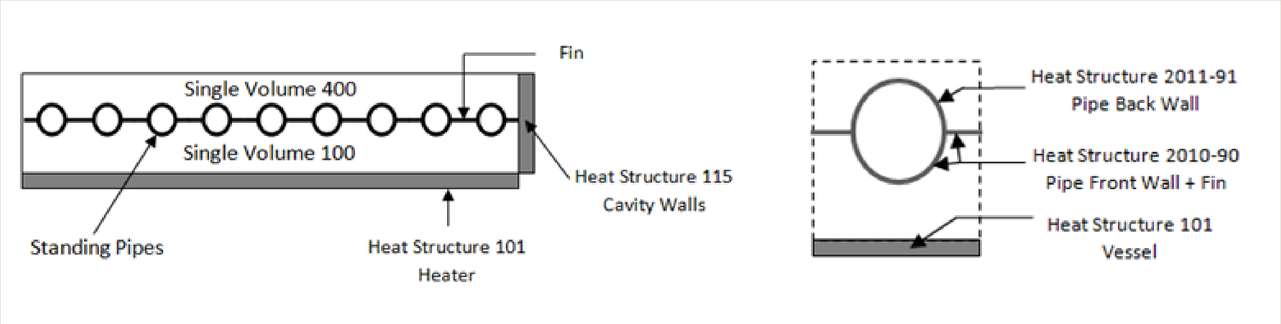


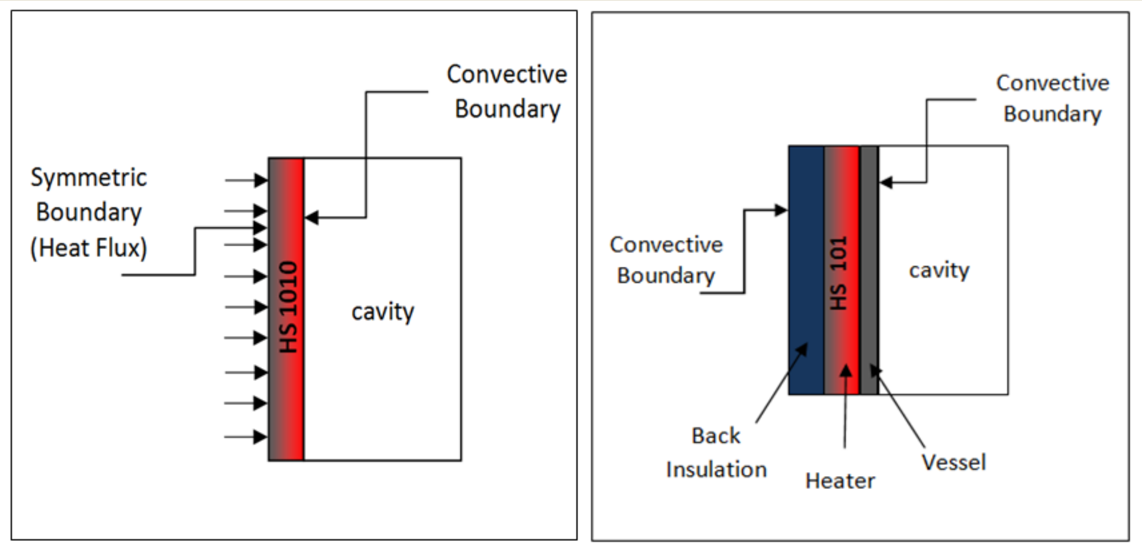
Experimental Facility



Full Plant

Model of the Cavity

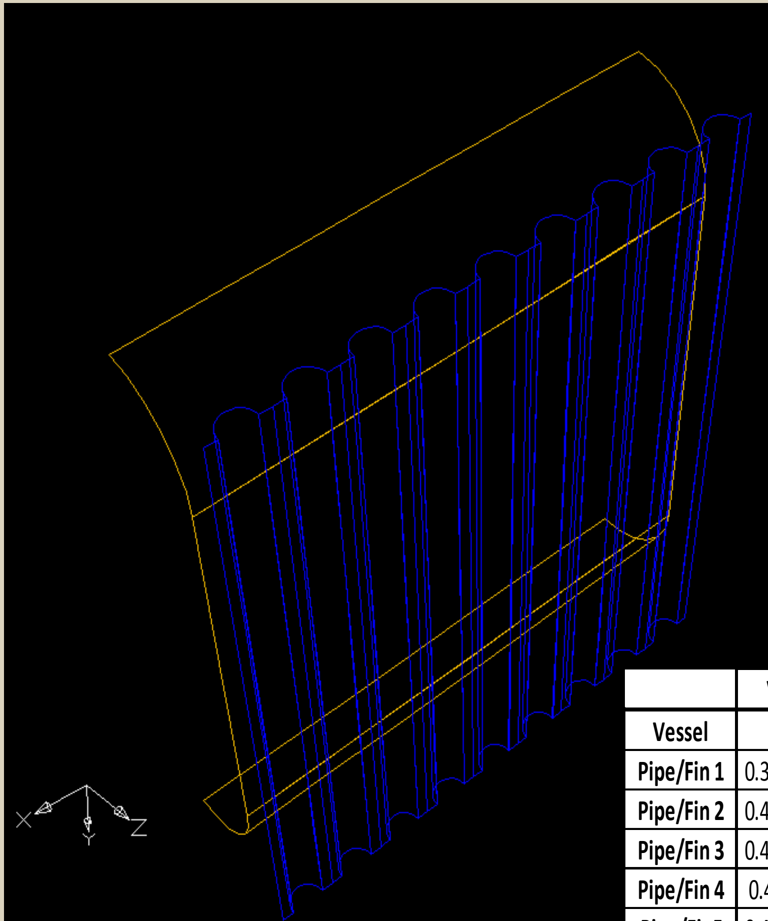




R.Vaghetto, S.Lee, Y.A.Hassan," REACTOR CAVITY COOLING SYSTEM FACILITY SHAKEDOWN AND RELAP5-3D MODEL VALIDATION", Proceedings of the 20th International Conference on Nuclear Engineering ICONE20 July 30-August 3, 2012, Anaheim, California, USA

Nevada™ View Factor Calculations

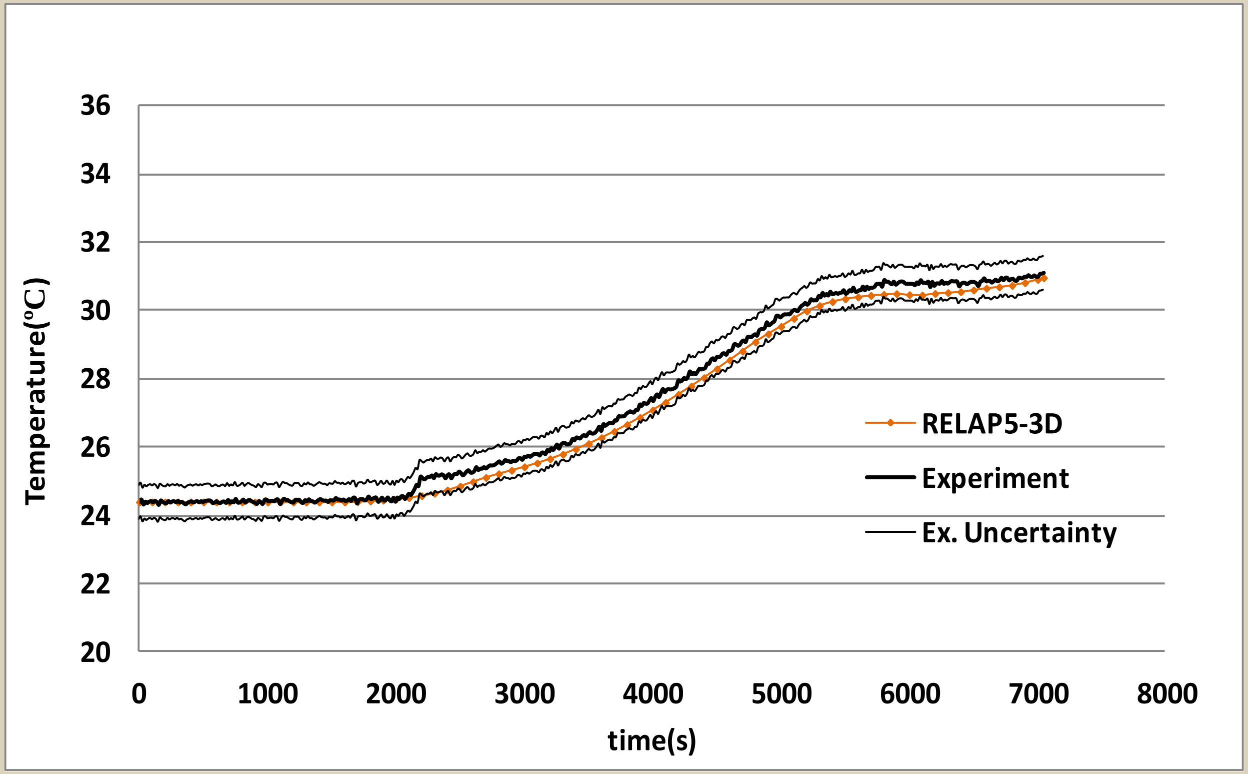
Input Geometry



View Factors

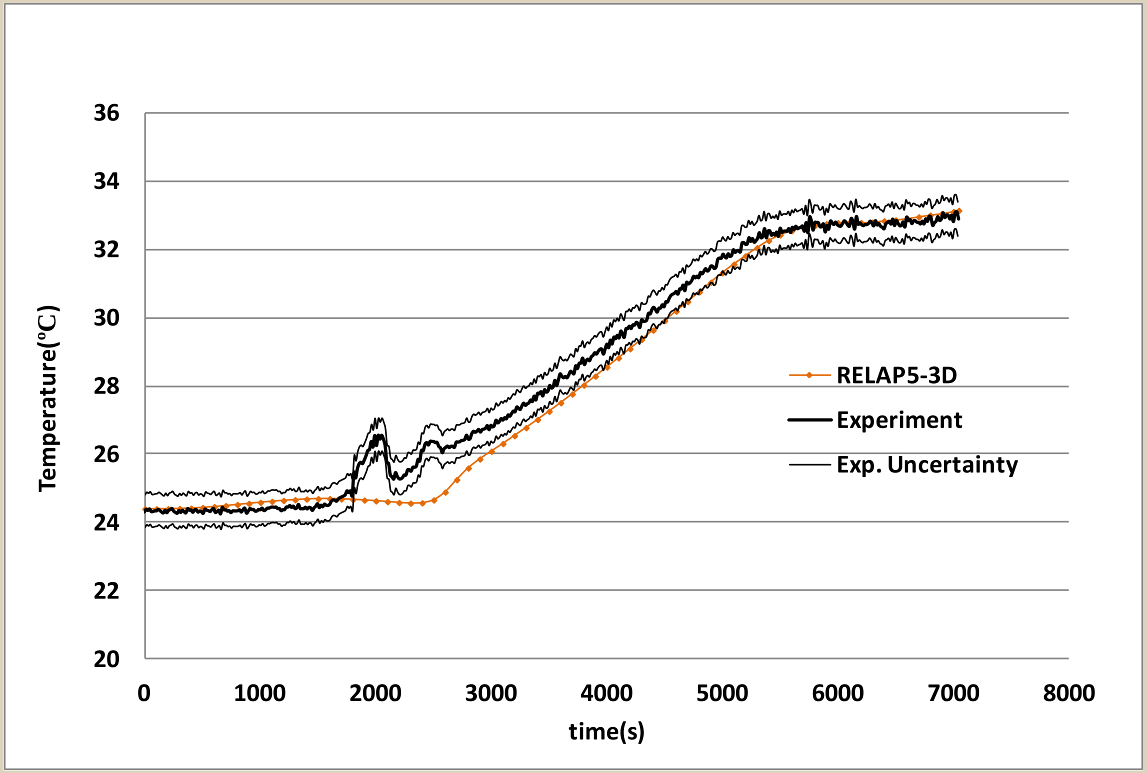
	Vessel	Pipe/Fin 1	Pipe/Fin 2	Pipe/Fin 3	Pipe/Fin 4	Pipe/Fin5	Pipe/Fin 6	Pipe/Fin 7	Pipe/Fin 8	Pipe/Fin 9	Cavity Walls
Vessel	0	0.085056	0.106983	0.109572	0.109981	0.110126	0.109981	0.109572	0.106983	0.085056	0.06669
Pipe/Fin 1	0.362298801	0.116379	0.066142	0	0	0	0	0	0	0	0.455153
Pipe/Fin 2	0.455697571	0.066142	0.116379	0.066142	0	0	0	0	0	0	0.295739
Pipe/Fin 3	0.466725501	0	0.066142	0.116379	0.066142	0	0	0	0	0	0.284561
Pipe/Fin 4	0.46846765	0	0	0.066142	0.116379	0.066142	0	0	0	0	0.282655
Pipe/Fin5	0.469085282	0	0	0	0.066142	0.116379	0.066142	0	0	0	0.282279
Pipe/Fin 6	0.46846765	0	0	0	0	0.066142	0.116379	0.066142	0	0	0.282655
Pipe/Fin 7	0.466725501	0	0	0	0	0	0.066142	0.116379	0.066142	0	0.284561
Pipe/Fin 8	0.455697571	0	0	0	0	0	0	0.066142	0.116379	0.066142	0.295739
Pipe/Fin 9	0.362298801	0	0	0	0	0	0	0	0.066142	0.116379	0.455153
Cavity Walls	0.088695504	0.1421138	0.092339506	0.0888494	0.088254248	0.088137	0.088254248	0.0888494	0.092339506	0.1421138	0

Results – Cavity Inlet Coolant Temperature



R.Vaghetto, Y.A.Hassan,” ANALYIS OF THE STEADY-STATE PHASE OF THE REACTOR CAVITY COOLING SYSTEM EXPERIMENTAL FACILITY AND COMPARISON WITH RELAP5-3D SIMULATIONS”, The 15th International Topical Meeting on Nuclear Reactor Thermal-hydraulics, NURETH-15 Pisa, Italy, May 12-15, 2013

Results – Cavity Outlet Coolant Temperature



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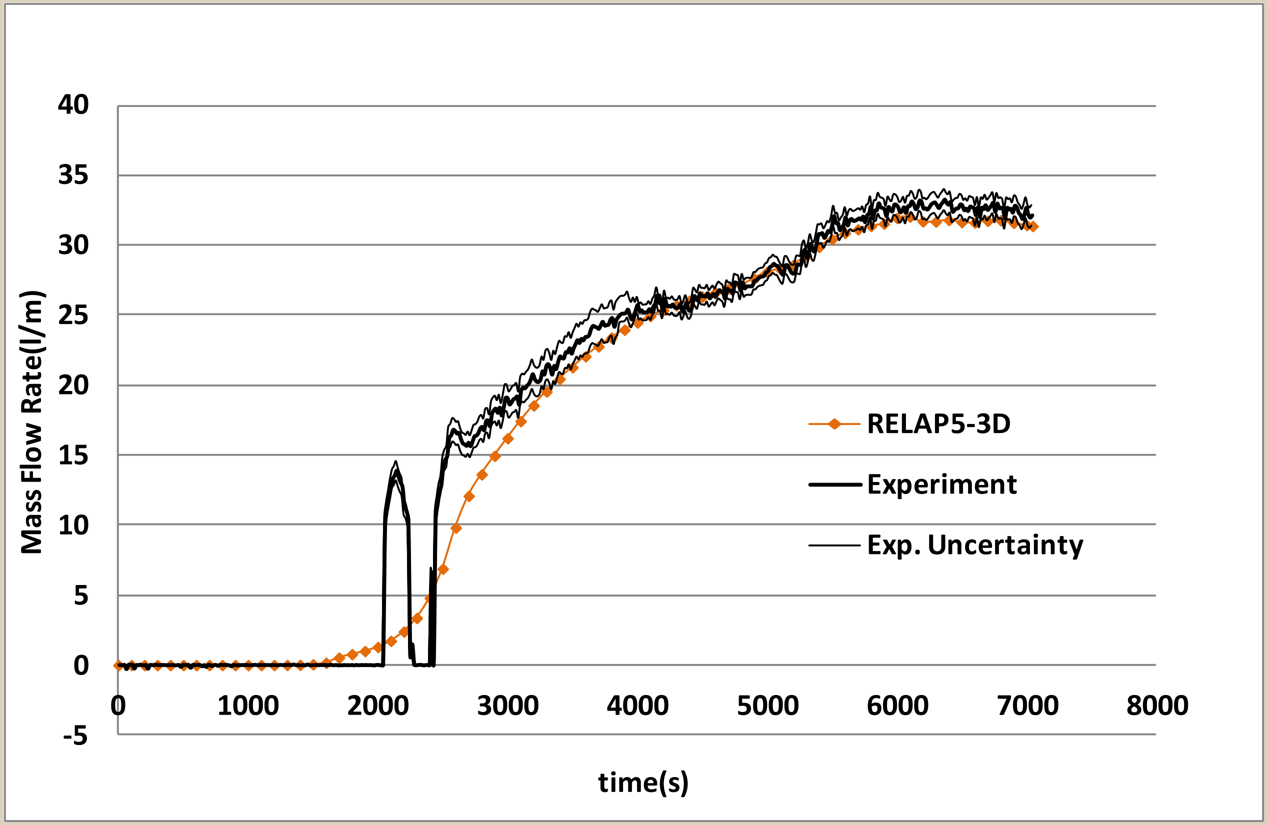
Results – Main Coolant Flow Rate



R.Vaghetto, Y.A.Hassan," ANALYIS OF THE STEADY-STATE PHASE OF THE REACTOR CAVITY COOLING SYSTEM EXPERIMENTAL FACILITY AND COMPARISON WITH RELAP5-3D SIMULATIONS", The 15th International Topical Meeting on Nuclear Reactor Thermal-hydraulics, NURETH-15 Pisa, Italy, May 12-15, 2013

Generic Safety Issue - 191

INTRODUCTION

The Emergency Core Cooling System (ECCS) is designed to cool down the reactor during postulated accidents such as Loss of Coolant Accidents (LOCA).

During the first phase of the accident (***Safety Injection***) cold water from the Refueling Water Storage Tank (RWST) is injected into the primary system through Safety Injection (SI) pumps.

In a later phase (***Long-Term Cooling***), the cooling process continues using the water discharged from the break into the reactor containment and collected in the sump.

INTRODUCTION

During a Loss of Coolant Accident (LOCA) debris may be produced and transported in different ways through the Reactor Containment.

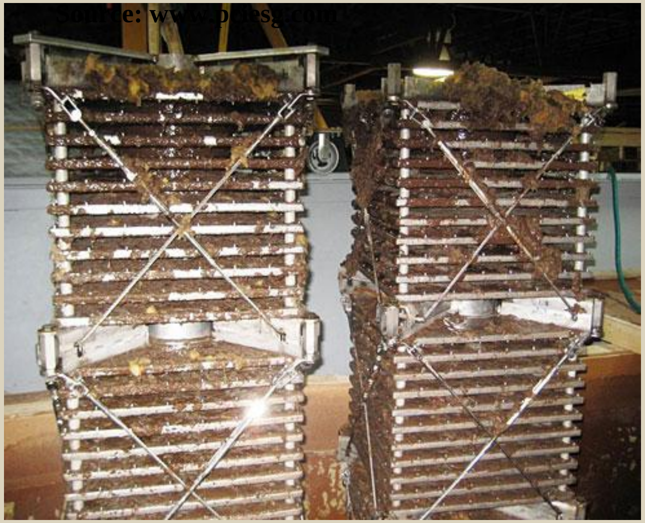
A set of sump screens are typically installed in the containment to minimize the amount of debris that could be injected into the primary system and its impact on the required core cooling (***downstream effects***).

Source: www.pciesg.com

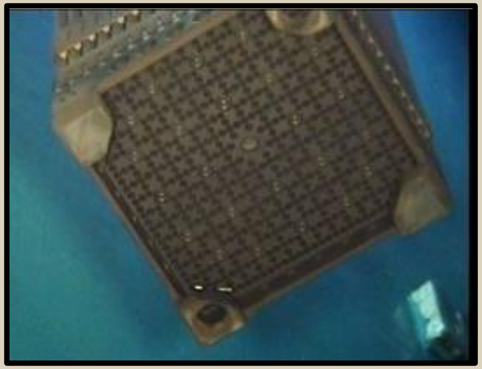
INTRODUCTION

Downstream Effects

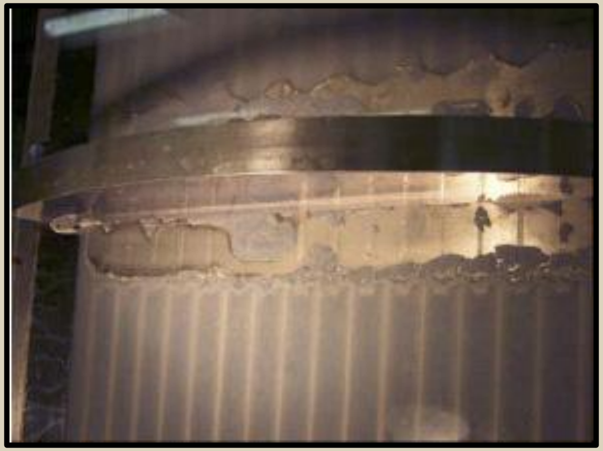
Source: www.pciesg.com



Some debris (fines) may pass thought the sump screen, especially during the early stage of the Long-Term Cooling Phase (clean Screen) and may be transported into the core.



The coolant flow may be perturbed by the debris deposition and accumulation in the fuel assemblies.



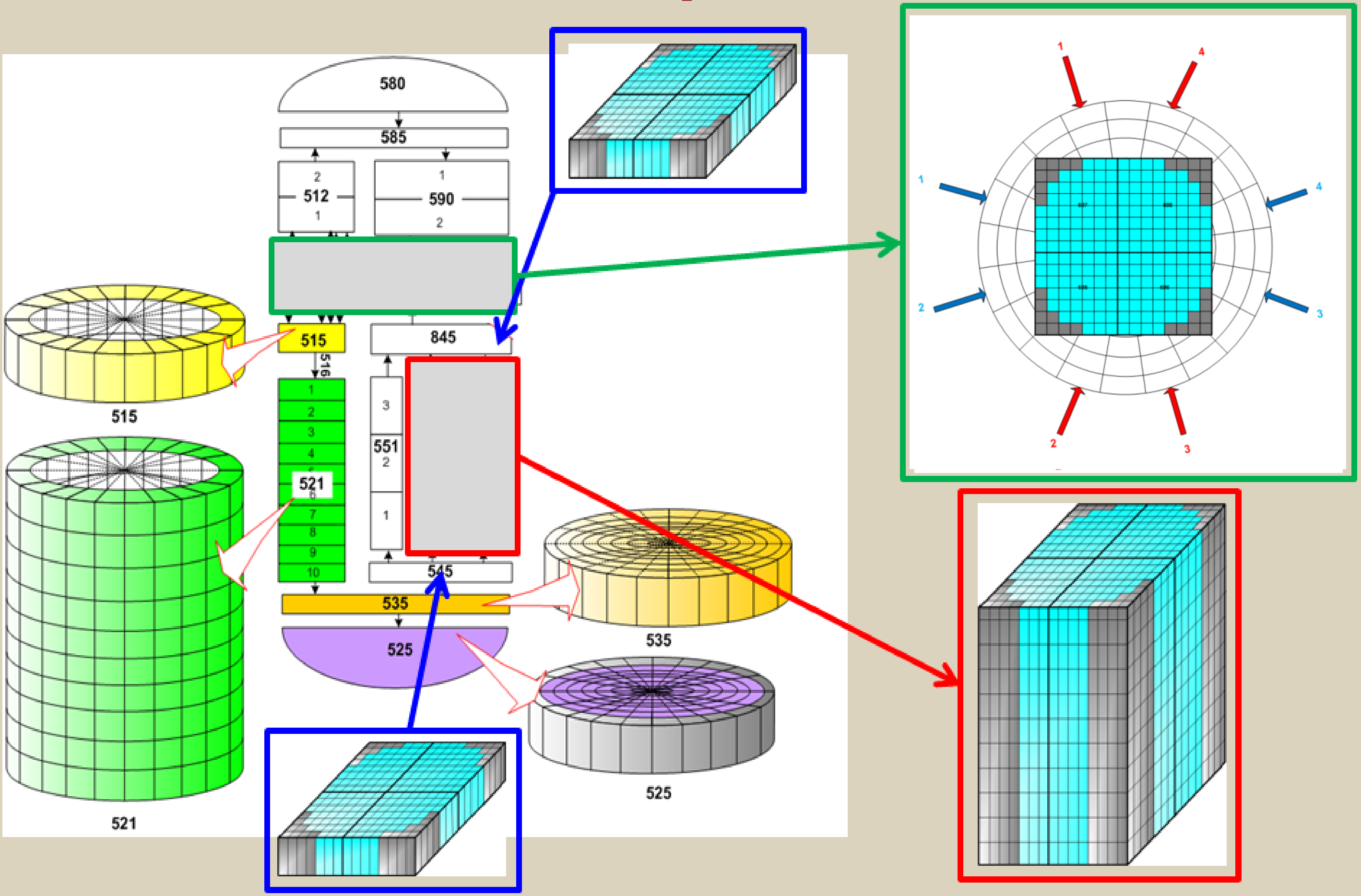
Core blockage may occur and core cooling degradation may lead to core damage.

Source: PWROG Website

OBJECTIVES

1) Analyze LOCA scenarios of different break sizes and locations
2) Confirm whether alternative flow paths may guarantee the core cooling even under such conservative conditions for specific scenarios.
3) Identify critical scenarios that may lead to core damage.
4) Study the Flow Paths in the core

Multi-Dimensional Input Model



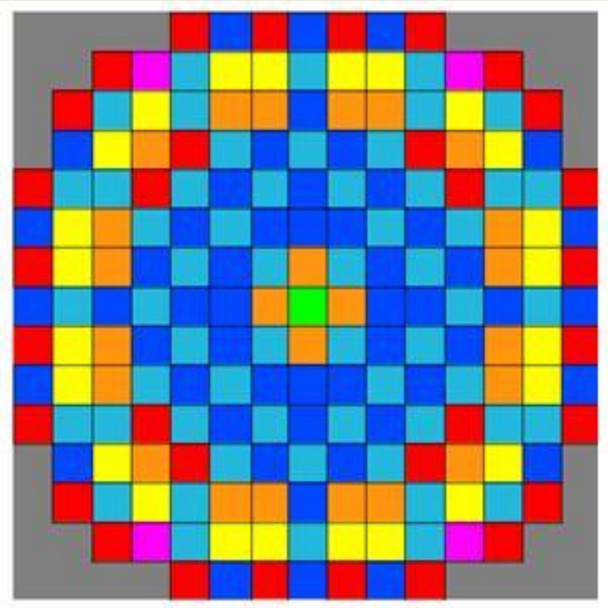
Actual Cold and Hot Legs layout, with angles and relative location to the core taken from CAD Drawings.

193 Fuel Channels with cross flow individually simulated

R.Vaghetto, Y.A.Hassan,"STUDY OF DEBRIS-GENERATED CORE BLOCKAGE SCENARIOS DURING LOSS OF COOLANT ACCIDENTS USING RELAP5-3D", Nuclear Engineering and Design 261 (2013) 144– 155

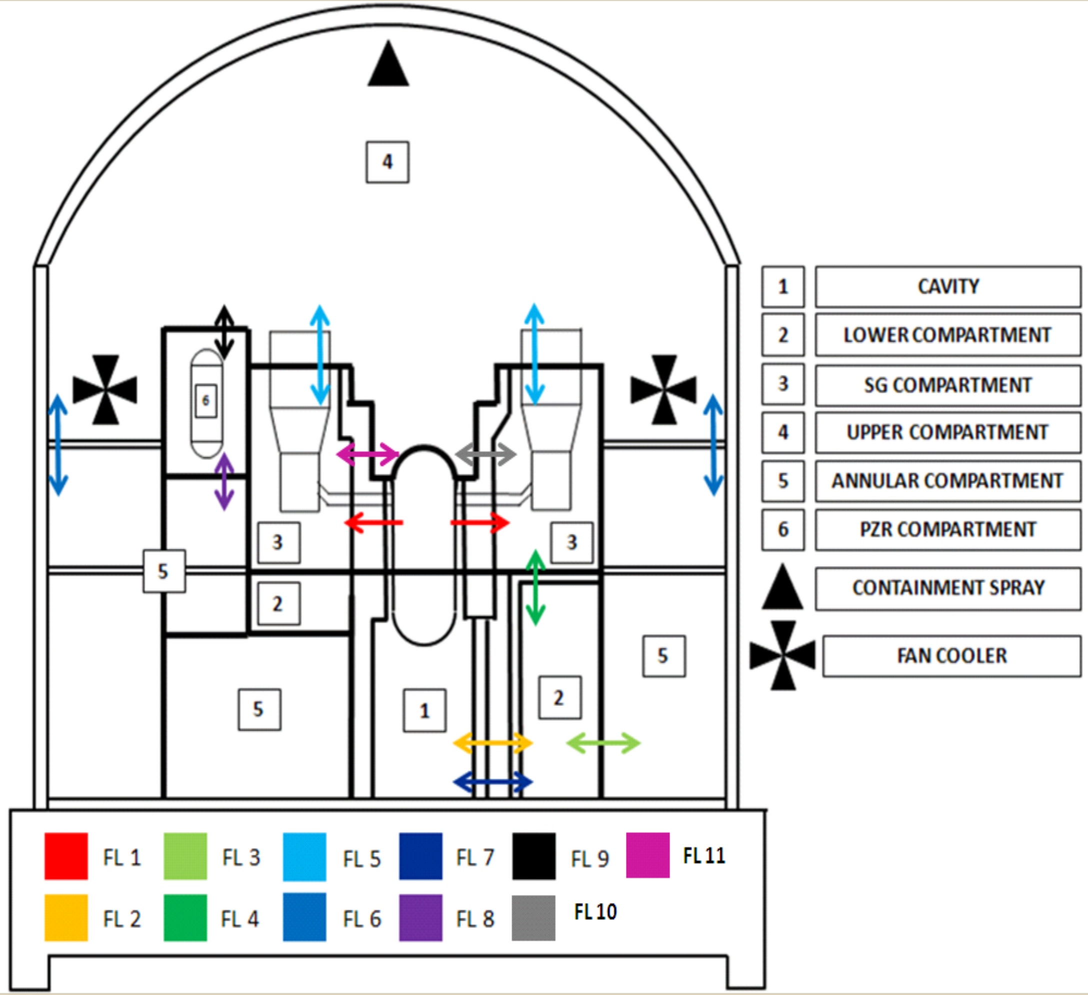
3D-Vessel, 3D-Core Model

- 193 Fuel Channels individually simulated.
- Each channel has 11 axial nodes
- Cross flow junctions between adjacent channels
- Typical PWR core fuel arrangement.
- 193 Heat structures to represent the power generation in each assembly
- Typical axial and radial power distribution (17^{th} cycle, EOL)
- Total number of nodes adopted to model the core = 2123



Each color represents a different power sharing

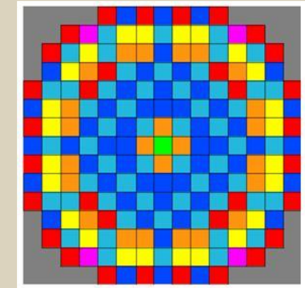
MELCOR Model of the Reactor Containment



- 6 control volumes
- 11 flow paths
- 49 heat structures
  - Floors, ceilings, and walls
- Engineered safety features
  - Containment Sprays
  - Fan Coolers

R.Vaghetto, B.A.Beeny,Y.A.Hassan,K.Vierow, "Analysis of Long-Term Cooling of a LOCA by Coupling RELAP5-3D and MELCOR", 2012 ANS Annual Meeting Chicago, IL, June 24-28, 2012

EXAMPLES OF 3D CORE FLOW VISUALIZATION
6” CL Break – No Core Blockage

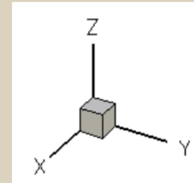


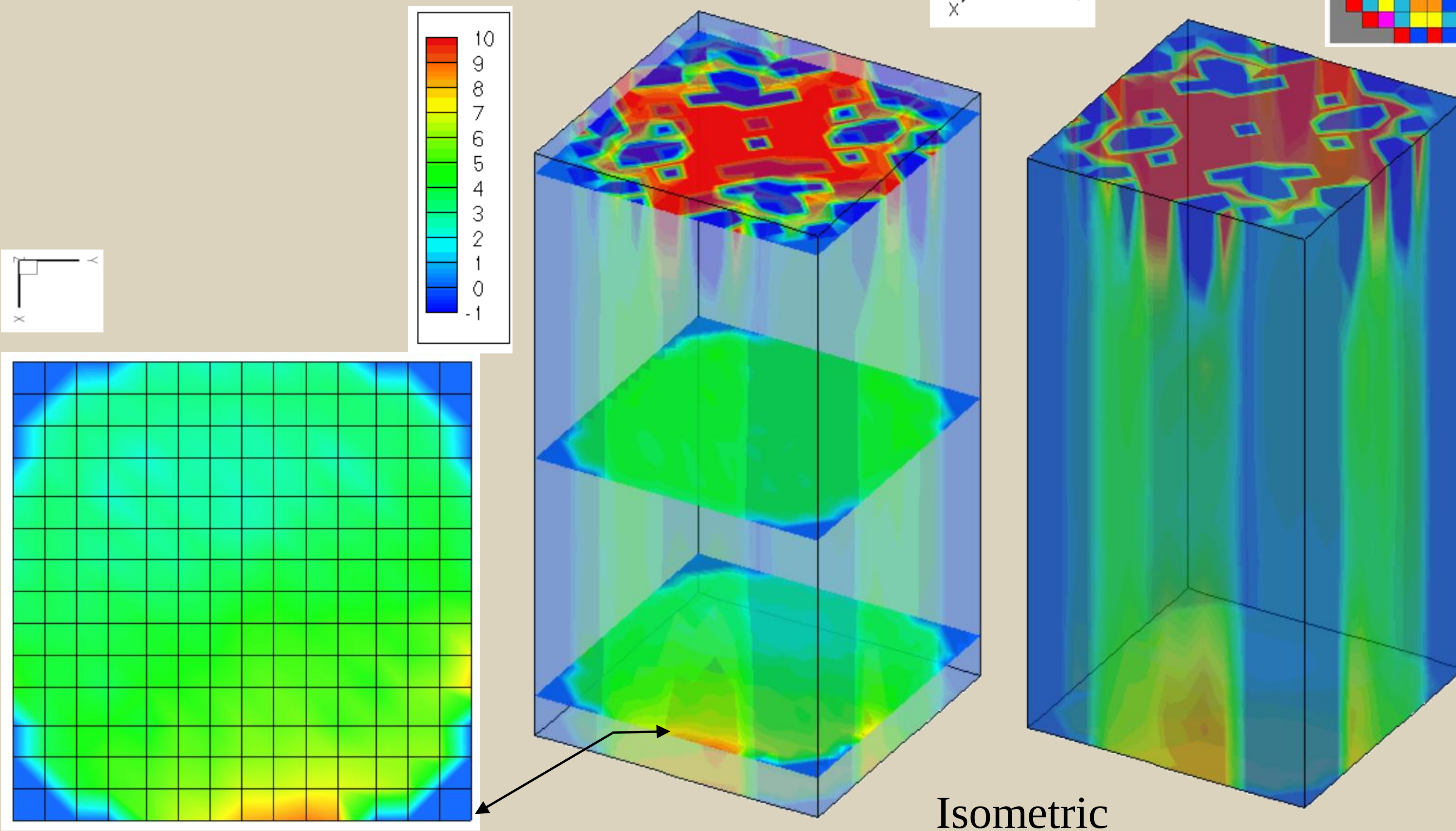
Vertical Flow Rate (lbm/s)

Broken Loop : 3

Sump Switchover Time = 2500 s

Snapshot at t = 2000 s





Core Inlet

TEXAS A&M ENGINEERING

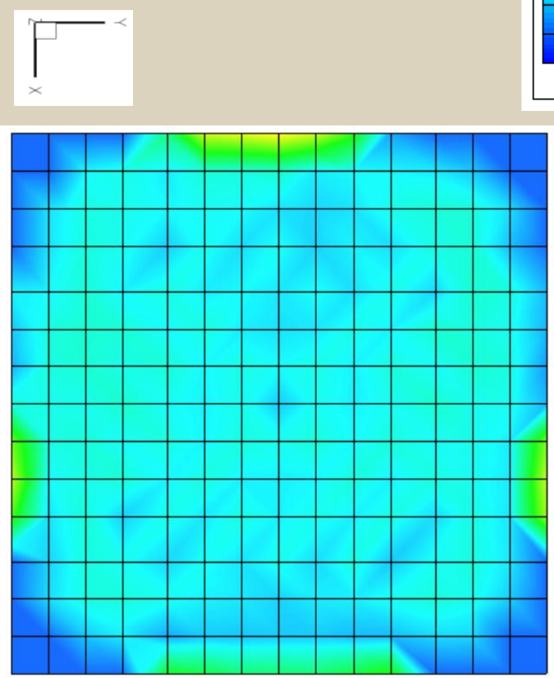
Vertical Flow Rate (lbm/s)

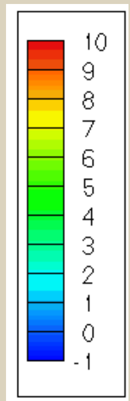
Broken Loop : 3

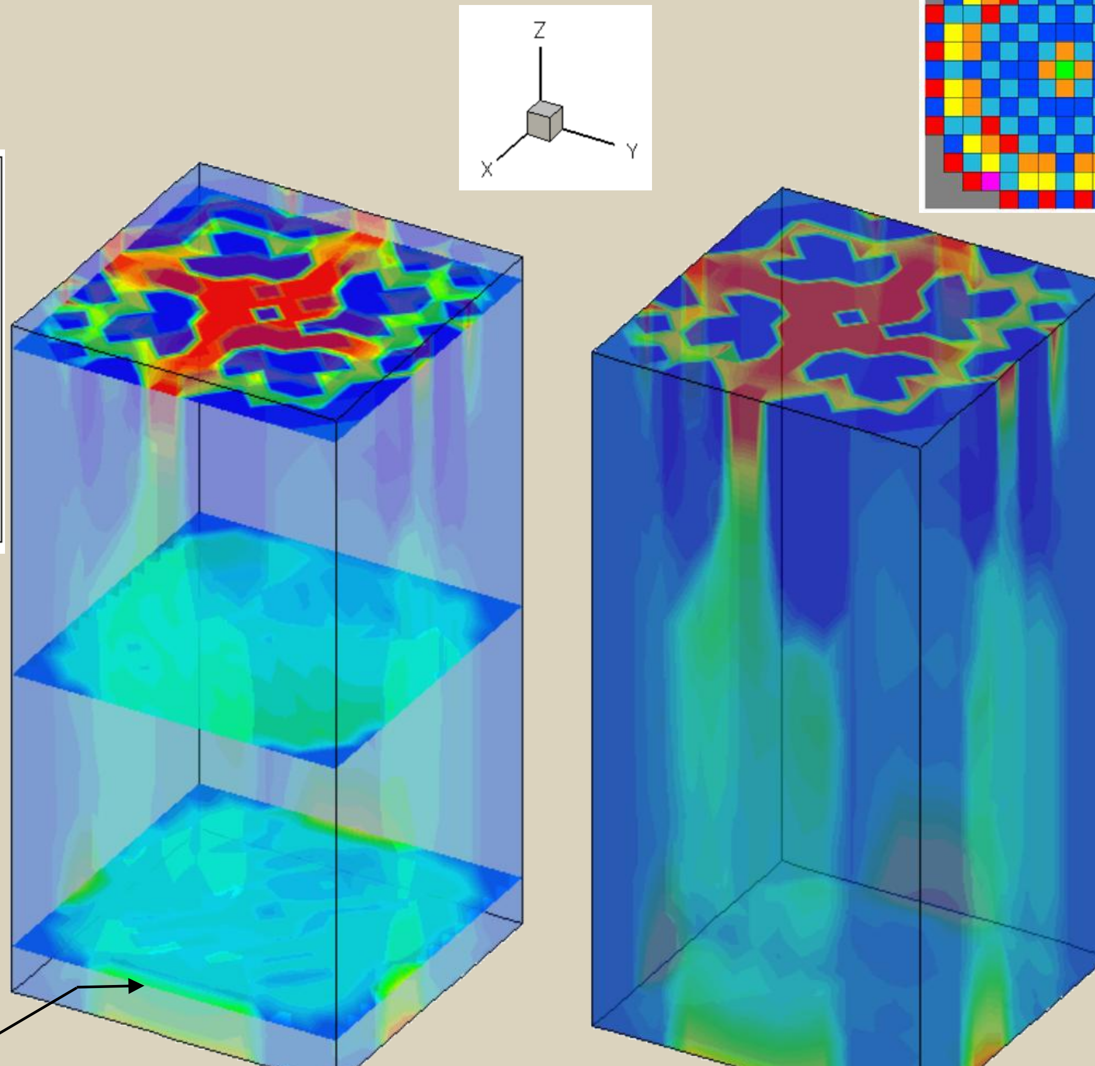
Sump Switchover Time = 2500 s

Snapshot at t = 2500 s

(at Sump Switchover)



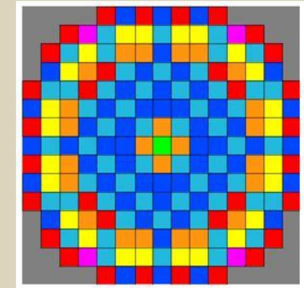




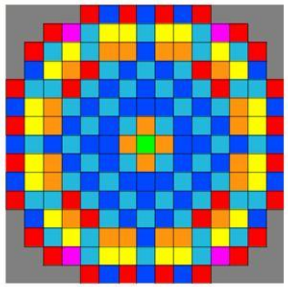
Core Inlet

Isometric

Power Distribution

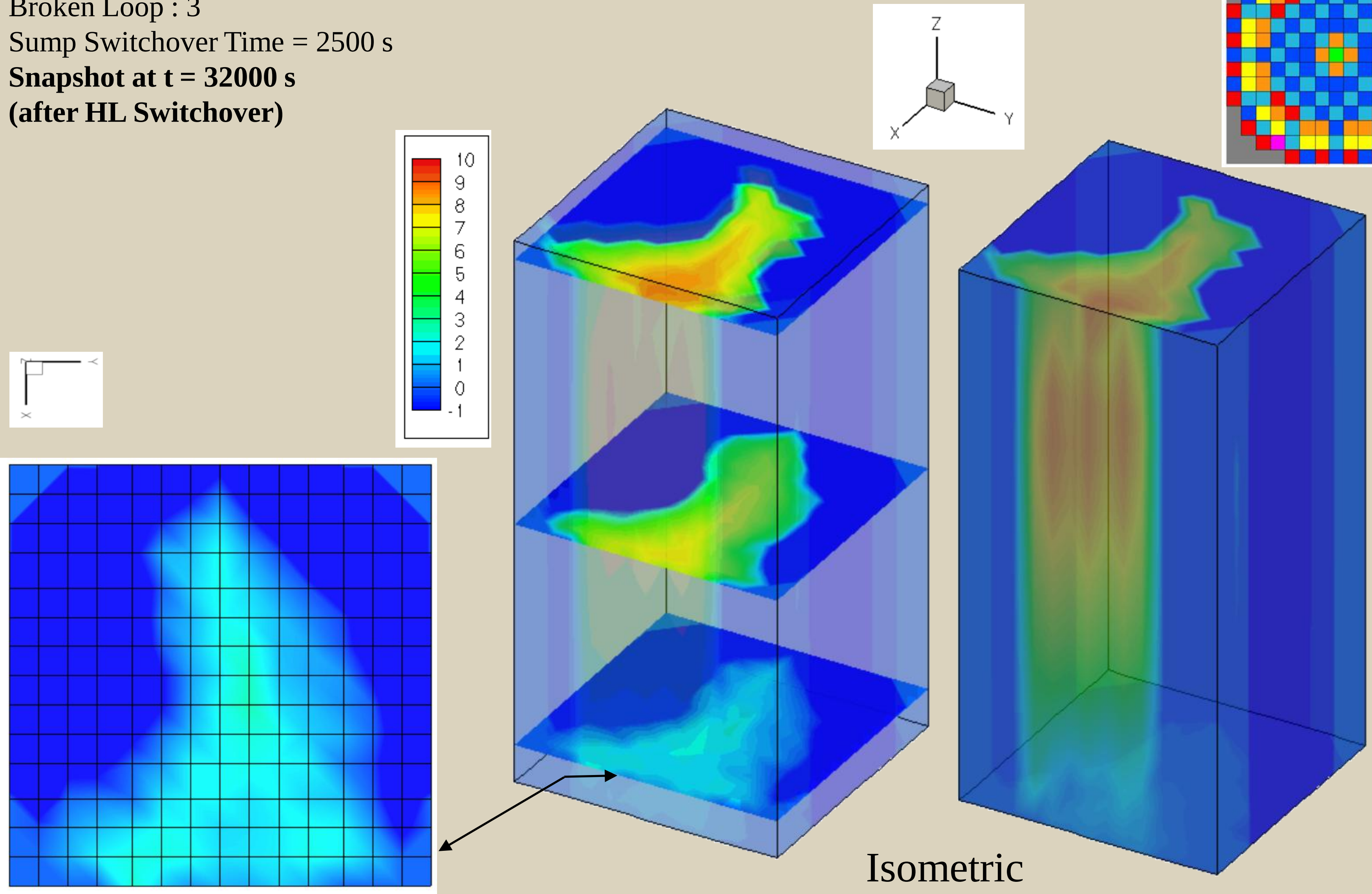


Power Distribution



Vertical Flow Rate (lbm/s)

Broken Loop : 3
Sump Switchover Time = 2500 s
Snapshot at t = 32000 s
(after HL Switchover)



Core Inlet

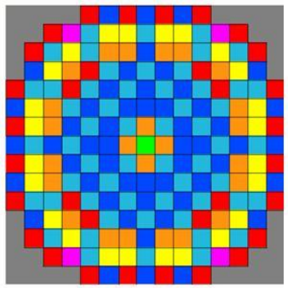
Isometric

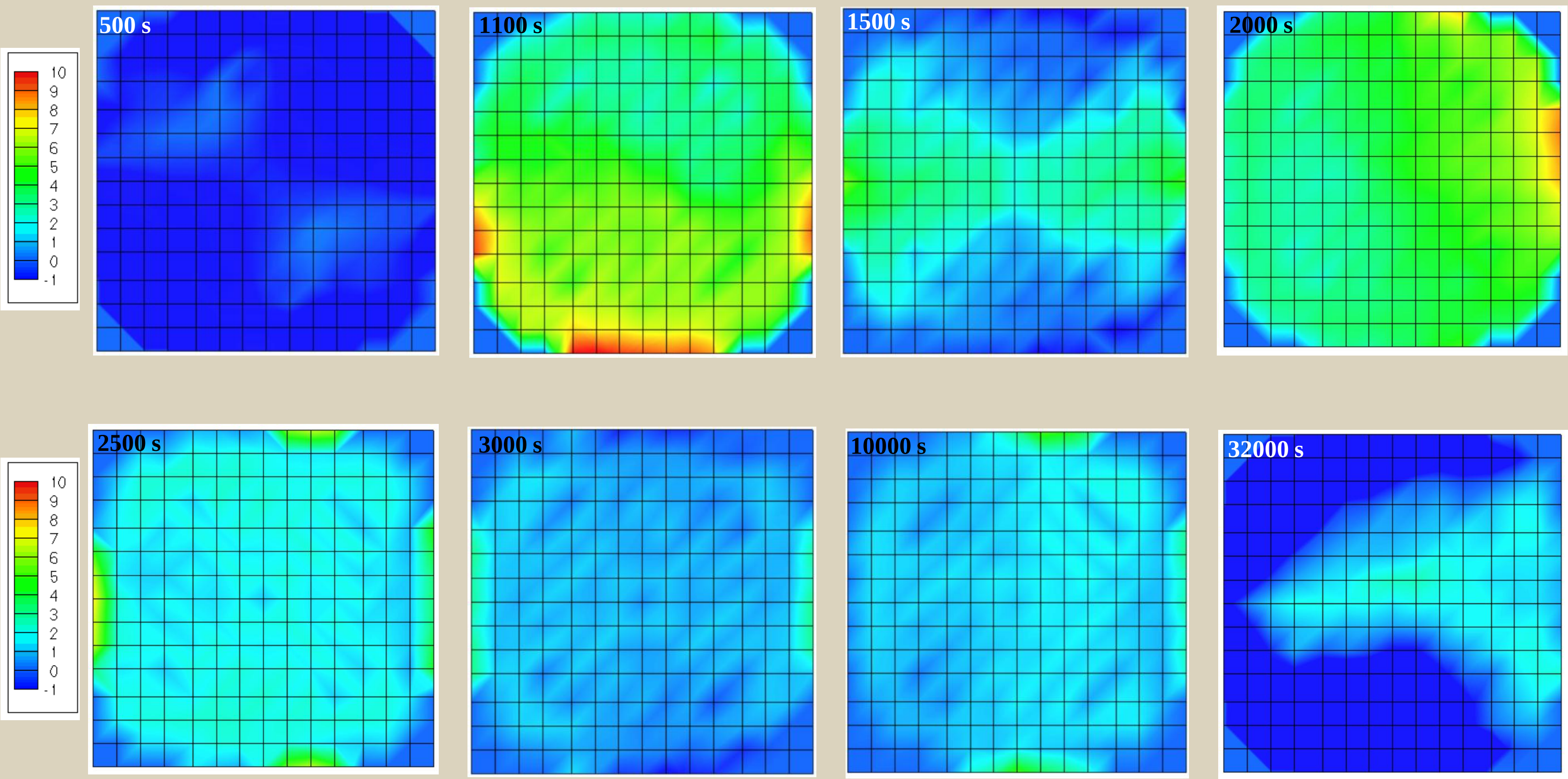
Core Inlet Flow Maps (lbm/s)

Broken Loop : 3

Sump Switchover Time = 2500 s

Snapshots



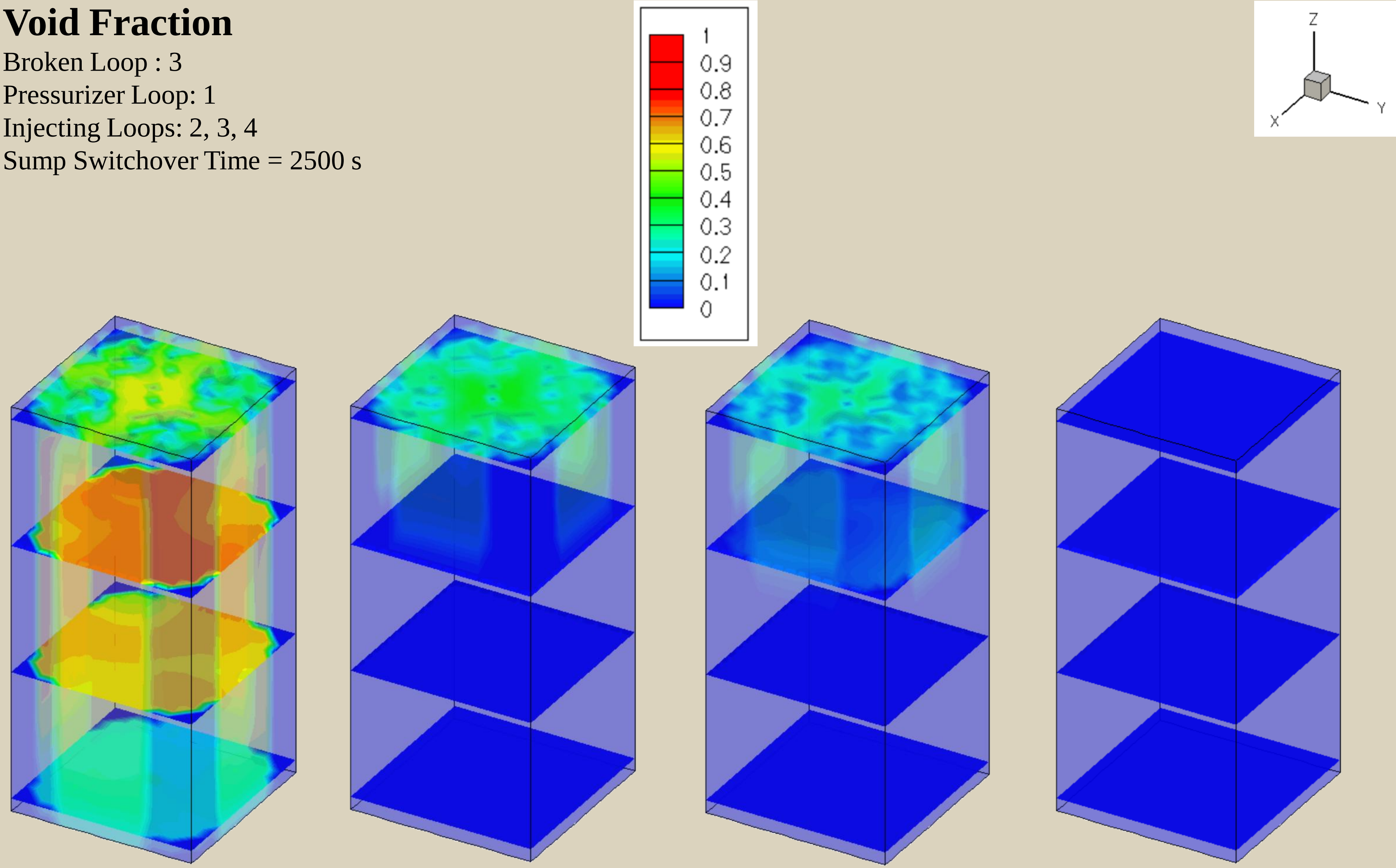


Sump Switchover

After HL Switchover

Void Fraction

Broken Loop : 3
Pressurizer Loop: 1
Injecting Loops: 2, 3, 4
Sump Switchover Time = 2500 s

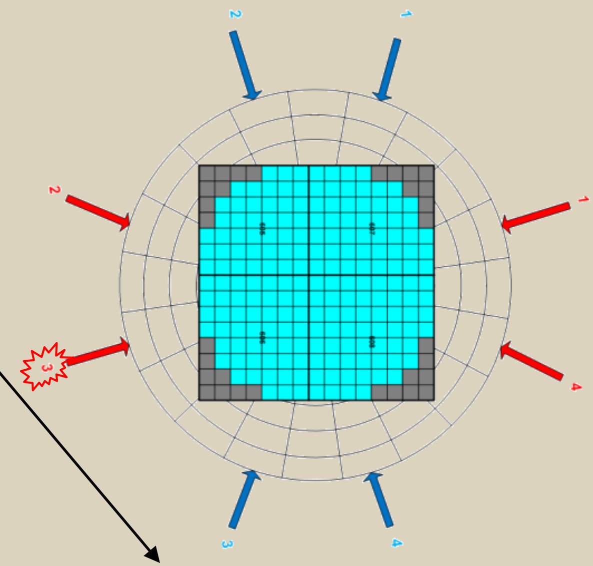


t = 650 s   t = 2000 s   t = 2500 s   t = 32000 s

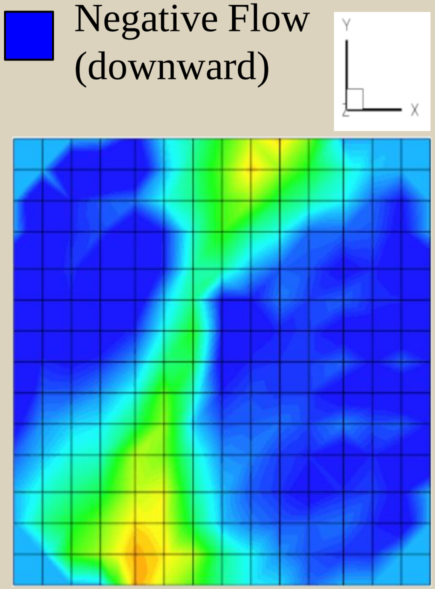
EXAMPLES OF 3D CORE FLOW VISUALIZATION
DEG HL Break Simulation Results
Full Core Blocked, Free Core Bypass

Vertical Flow Rate (lbm/s)
Broken Loop : 3
Pressurizer Loop: 1
Injecting Loops: 2, 3, 4
Snapshot at t = 2500 s

Nodalization (top View)

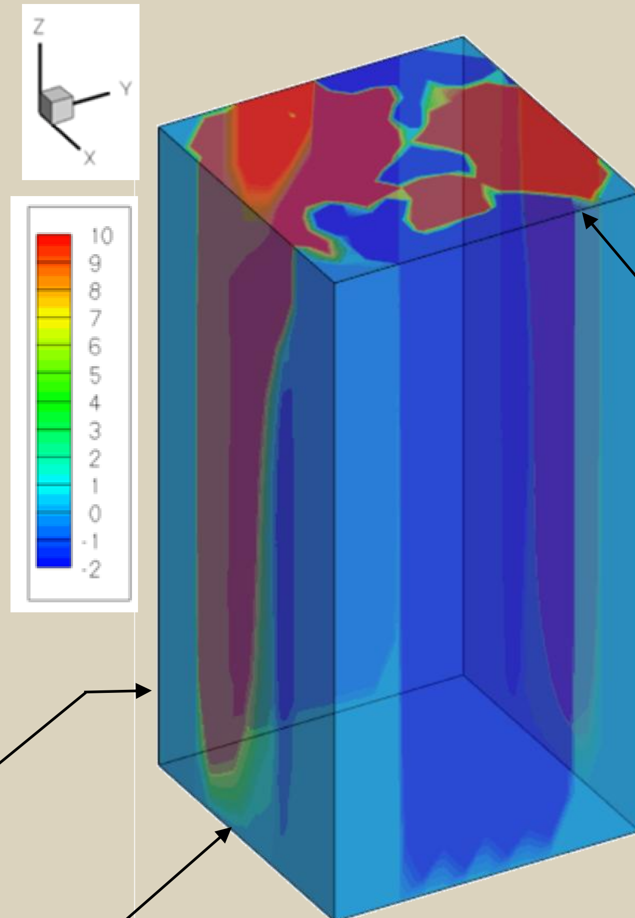


Negative Flow (downward)

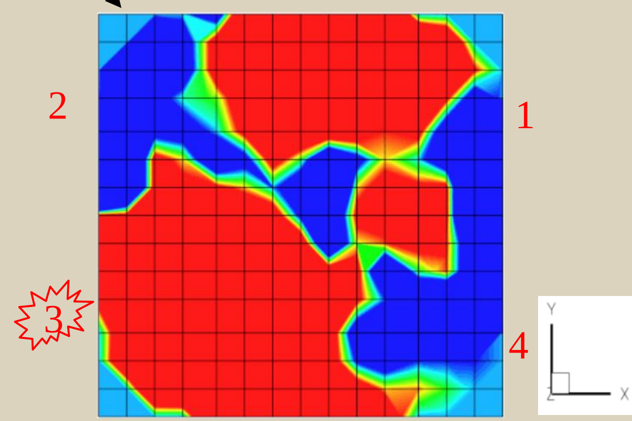


Core Inlet

Blockage Plane



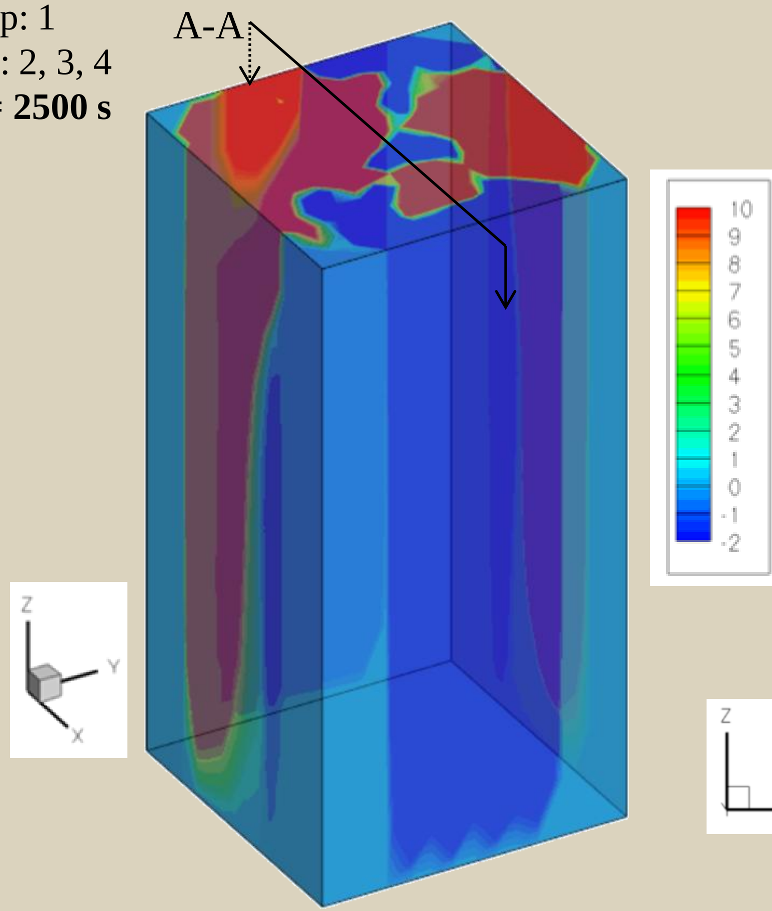
Isometric



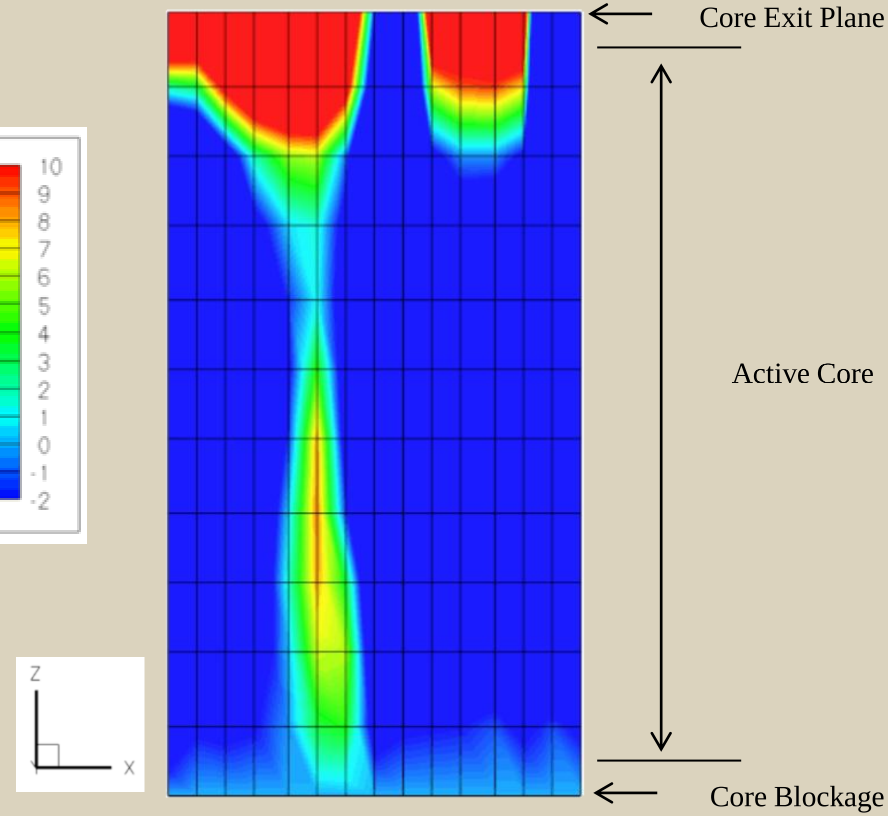
Core Outlet

- Coolant flow reaches the top of the core from hot legs (SGs spillover) and core baffle.
- Flow patterns found to be related to the hot leg injection configuration and break location.

Vertical Flow Rate (lbm/s)
Broken Loop : 3
Pressurizer Loop: 1
Injecting Loops: 2, 3, 4
Snapshot at t = 2500 s



Isometric



Longitudinal
Section View (A-A)

Downward coolant flow reached the bottom of the core and then proceeds upward toward the broken leg.