The Role of Computational Fluid Dynamics & Safety System Codes for Nuclear Reactor Predictions

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Why pursue advanced modeling and simulation capabilities?

- When integrated with theory and experiment, modeling & simulation **enhances opportunities for new insights into the complex phenomena occurring in the nuclear reactor**

- Advanced modeling & simulation offers the ability to improve the performance and safety of nuclear energy; **computational of CFD/system codes provides new capabilities & tools** for doing so

- These advancements can be utilized in **smart** simulators – **will impact the decisions of management reactor accidents.**

Experimentation – Extremely Difficult & Expensive

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Measurement Techniques, Sensors, Instrumentation not yet Developed

- Measurement Volume Inaccessible, Small for Intrusive/Non-intrusive measurement
- Intermittent/Unsteady, Multicomponent, Multiphase Flow
- Hostile Environment: High Temperature, Contaminants (Dirt), radioactive

CFD Appears to be a Logical Scheme to Complement Experimentation

The last decade has seen an increasing use of three-dimensional CFD codes to predict steady state and transient flows in nuclear reactors. (*A CFD hurricane hits nuclear industry!*!!).

The reason for the increased use of multidimensional CFD methods is that a number of important phenomena such as pressurized thermal shocks, boron mixing and thermal striping cannot be predicted by traditional one-dimensional system codes with the required accuracy and spatial resolution.

CFD codes contain empirical models for simulating turbulence, heat transfer, multi-phase flows, and chemical reactions. Such models must be *validated* before they can be used with sufficient confidence in *Nuclear Reactor Safety* applications.

Beyond Design Basis Accident

- Accidents occurring outside the NPP design basis
- It may or not involve core degradation

- Severe Accident event: a BDBA involving core degradation (melting of reactor core)

Accident Management

- Actions taken during evolution of events leading to a BDBA
  - Prevent evolution towards severe accident plant conditions
  - Mitigate consequences of a severe accident
  - Achieve long term safe stable state

Severe accident phenomena

- In-vessel
  - Core heat-up and uncovering
  - Fuel degradation
  - Core debris relocation in lower head
  - RPV failure
  - Etc.
- Ex-vessel
  - Primary containment pressurization
  - Primary containment / core debris interaction
  - Fission product transport
  - Hydrogen distribution
  - Primary containment failure
  - Etc.

BDBA Code General Classification

- Mechanistic
  - Best estimate phenomenological models as much as possible
  - Also commonly known as Detailed Codes
- Parametric
  - Combination of phenomenological and user defined parametric models
  - Also commonly known as Fast Running Codes

Note: the codes commonly referred as to Special or Dedicated Codes belong to either of the above.

Thermal-Hydraulic Codes

Selected Thermal-Hydraulic Codes

- Codes for LWR System Analysis
  - RELAP5 (INL)
  - TRAC (LANL)
  - RAMONA (BNL)
  - COBRA (PNL)
  - CATHARE (CEA)
  - ATHLET (GRS)
  - CATHENA (AECL)
  - TRACE (NRC)
  - RETRAN (EPRI)
  - FATHOM (ANSYS)
- Computational Fluid Dynamics
  - COMMIX (ANL)
  - FLUENT (ANSYS)
  - STAR-CD (CD-Adapco)
- Codes for Fuel Performance Analysis
  - Westinghouse proprietary
  - GE proprietary
  - AREVA proprietary
  - FRAP (PNL/INL)
  - FALCON (EPRI)
- Codes for Containment Analysis
  - CONTEMPT (INL)
  - MELCOR (SNL)
  - GOTHIC (EPRI)
- Codes for Severe Accidents
  - SCDAP/RELAP (INL)
  - MELCOR (SNL)
  - MAAP (EPRI)

Detailed Codes

- detailed information of performance of different plant components and systems during BDBA and severe accident, core degradation and/or containment performance, fission product transport, etc.
- ATHLET-CD (Analysis of Thermal-Hydraulics of Leaks and Transients – Core Degradation): GRS, Germany.
- ICARE/CATHARE (Code Avancé de THermohydraulique pour les Accidents de Réacteurs à Eau): IRSN, France.
- RELAP/SCDAP (Reactor Excursion and Leak Analysis Program / Severe Core Damage Analysis Package): USNRC/INEEL, USA.
- COCOSYS (Containment Code System): GRS, Germany.
- SAMPSON (Severe Accident analysis code with Mechanistic, Parallelized Simulations Oriented towards Nuclear field), NUPEC, Japan.
- CONTAIN: USNRC/SNL, USA.
- VICTORIA: USNRC/SNL, USA.

Parametric Codes

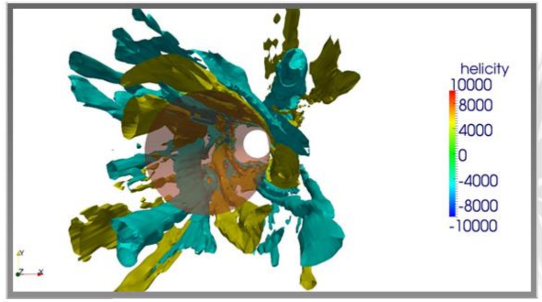
- Close to real time simulations. Applications to support Level 2 PSA, validation of accident management strategies, and severe accident prevention and mitigation actions.
- ASTEC (Accident Source Term Evaluation Code): IRSN/GRS, France/Germany.
- MELCOR (Methods for Estimation of Leakages and Consequences of Releases): USNRC/SNL, USA.
- MAAP (Modular Accident Analysis Program): EPRI, USA.

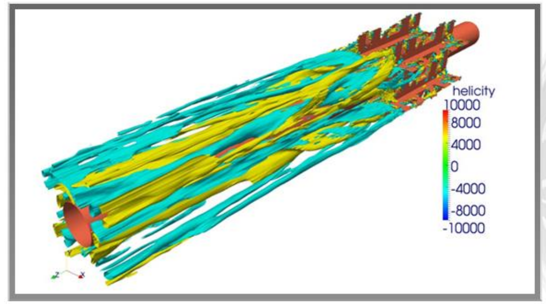
Dedicated Codes

- Deal with a single specific phenomenon. Several of them are based on CFD techniques.
- MC3D. Thermo-hydraulic multiphase flow code used for steam explosion due to fuel-coolant interaction, IRSN, France.
- GASFLOW. Hydrogen transport, mixing and combustion in containments, IKET/KIT, Germany.
- GOTHIC. 0 – 3D single and multiphase thermalhydraulics, NAI/EPRI, USA.
- TRIO. Single and two-phase thermalhydraulics, CEA, France.

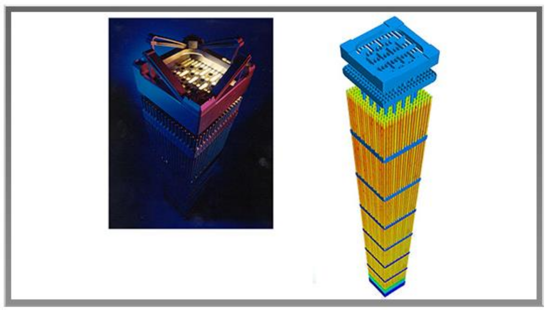
Cladding wear due to grid-to-rod-fretting (GTRF) can lead to fuel rod leaks in pressurized water reactors. The manifestation of this failure mechanism is complex, involving many factors ranging from basic material properties to operating conditions.

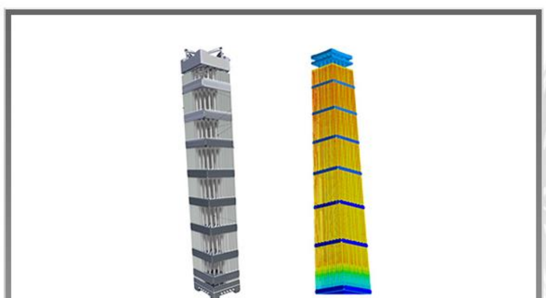
U.S. DEPARTMENT OF ENERGY

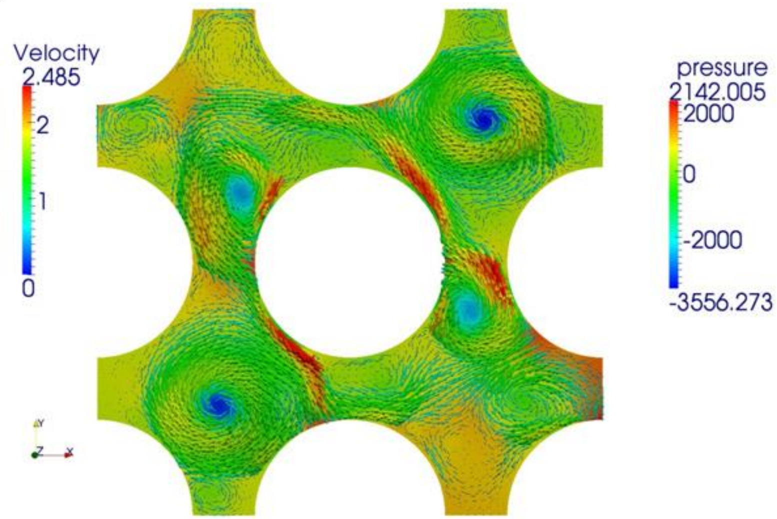




Westinghouse PWR17x17 Fuel Assembly and Temperature Distribution from Modeling Results

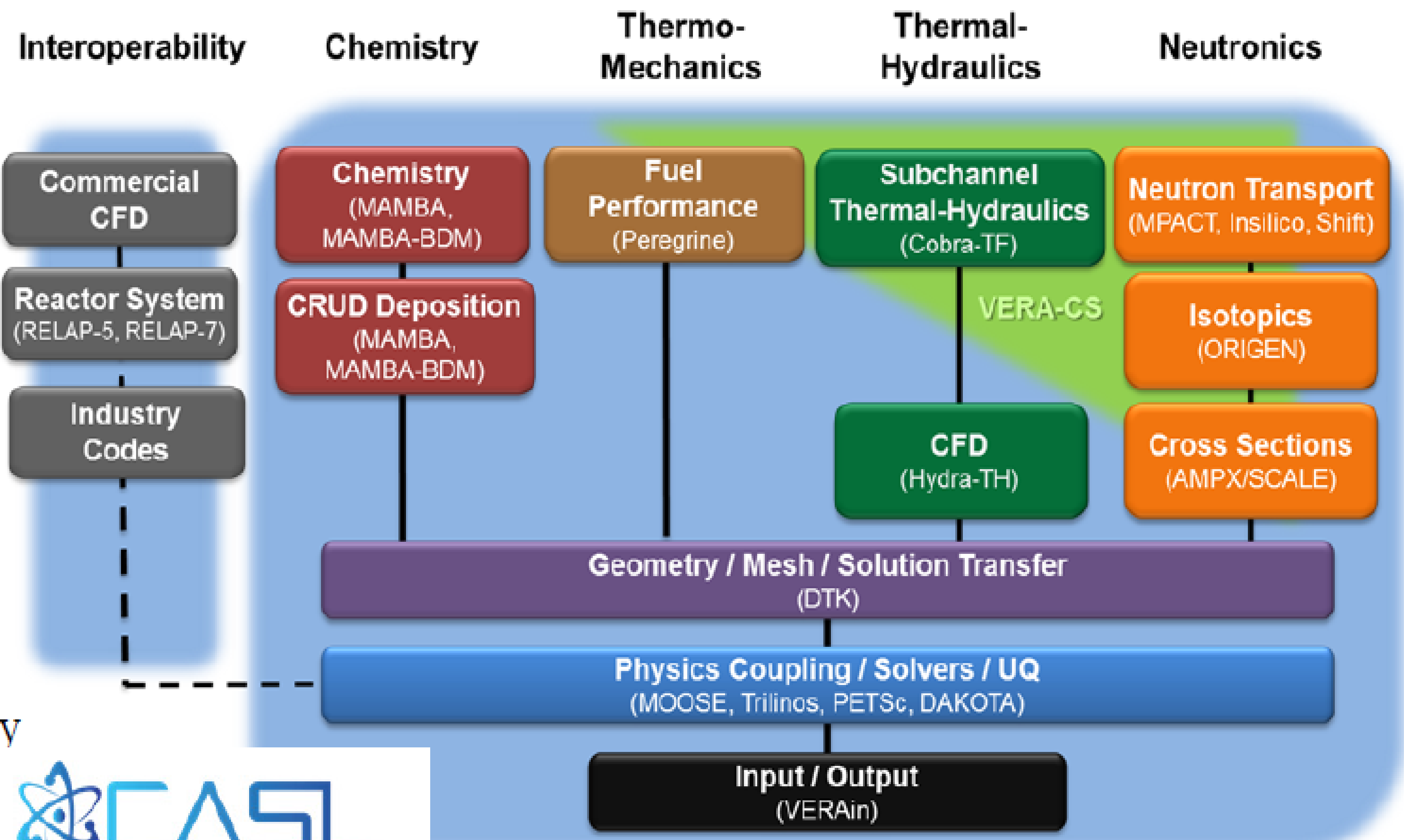






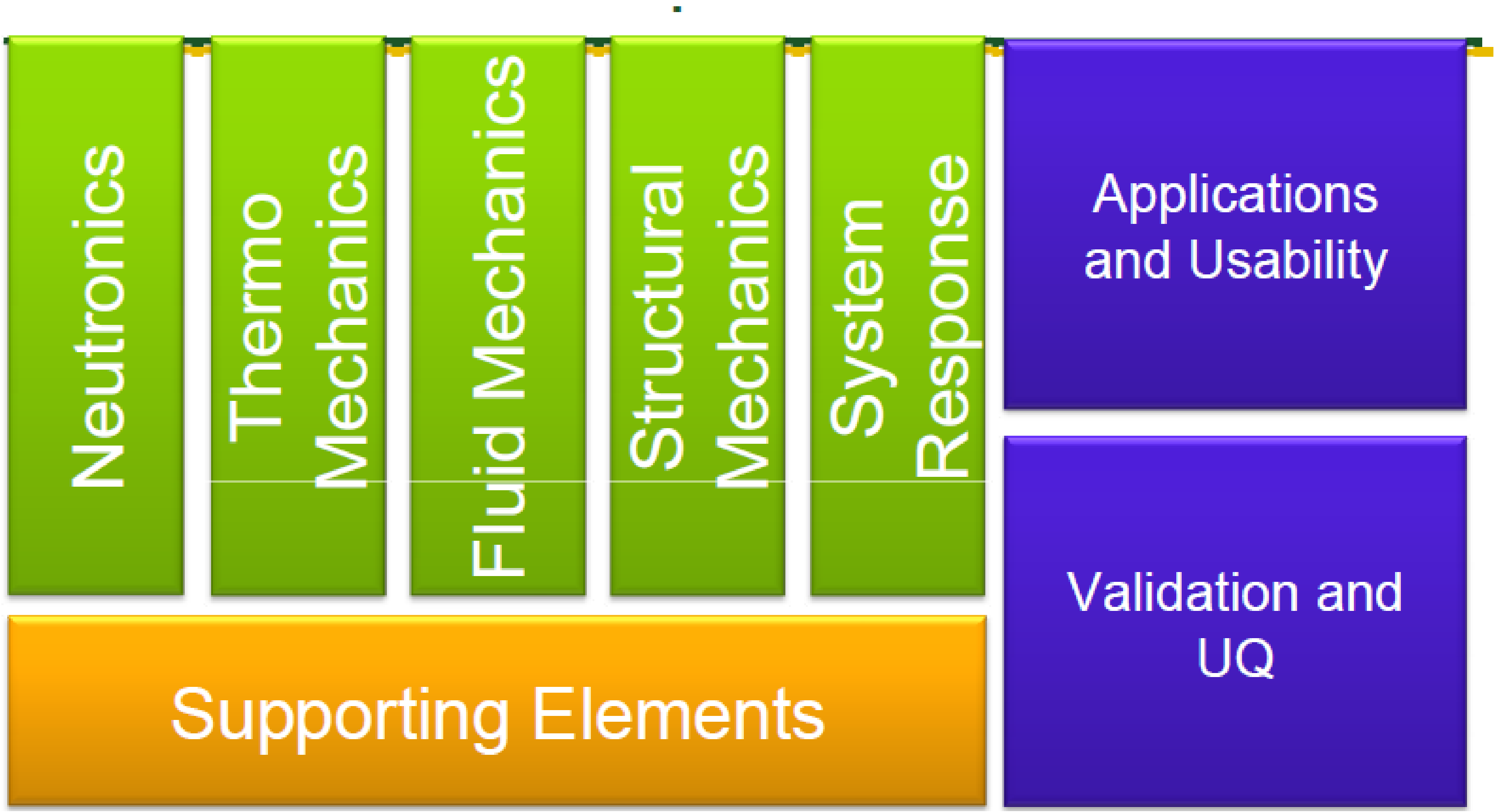
Top 20% bi-power rods in PWR Core. (Image courtesy of Jozsef Bakosi, Nate Barnett, Mark Christon, Marianne Francois, and Rob Lowrie. This image was provided for use relating to the Consortium for Advanced Simulation of Light Water Reactors (CASL), which is managed by Oak Ridge National Laboratory, U.S. Dept. of Energy.)

Images from http://www.casl.gov/



V

Capabilities of Virtual Environment for Reactor Application (VERA) for modeling and simulation



Multiphysics / Multiscale Development Approach

CFD Analysis Procedure

1: Understand the Problem and Formulate the Simulation Strategy

2: Create the Geometry

3: Generate the Mesh

4: Establish the Boundary and Initial Conditions

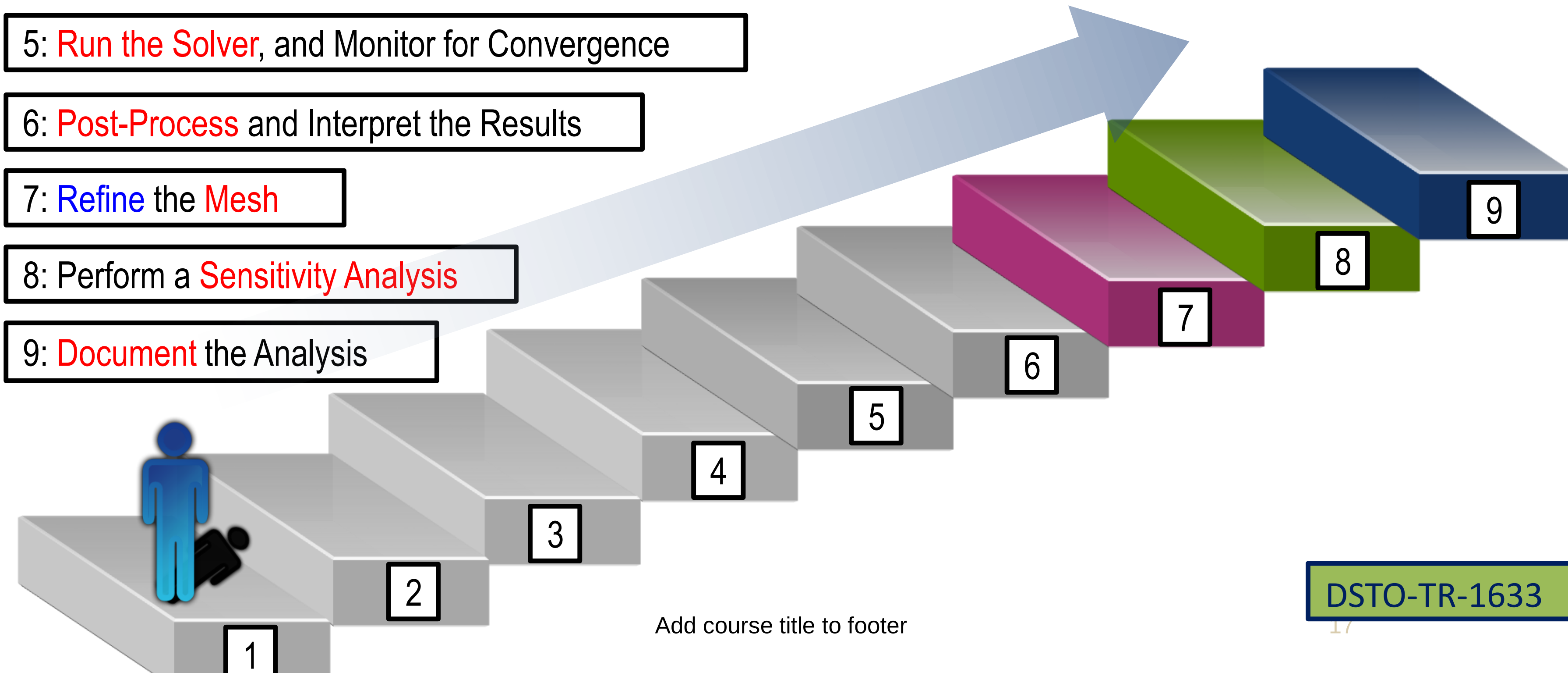
5: Run the Solver, and Monitor for Convergence

6: Post-Process and Interpret the Results

7: Refine the Mesh

8: Perform a Sensitivity Analysis

9: Document the Analysis



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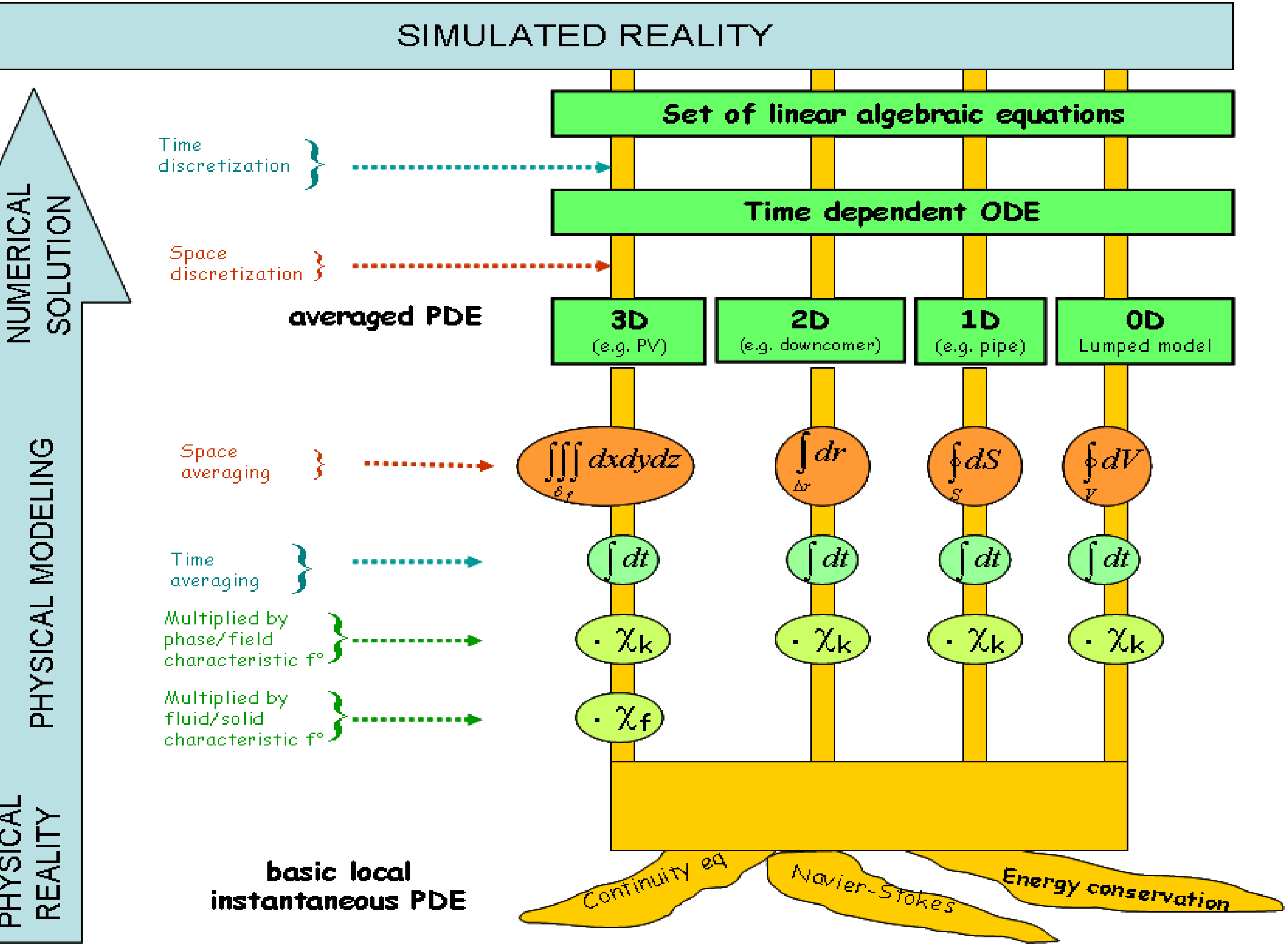
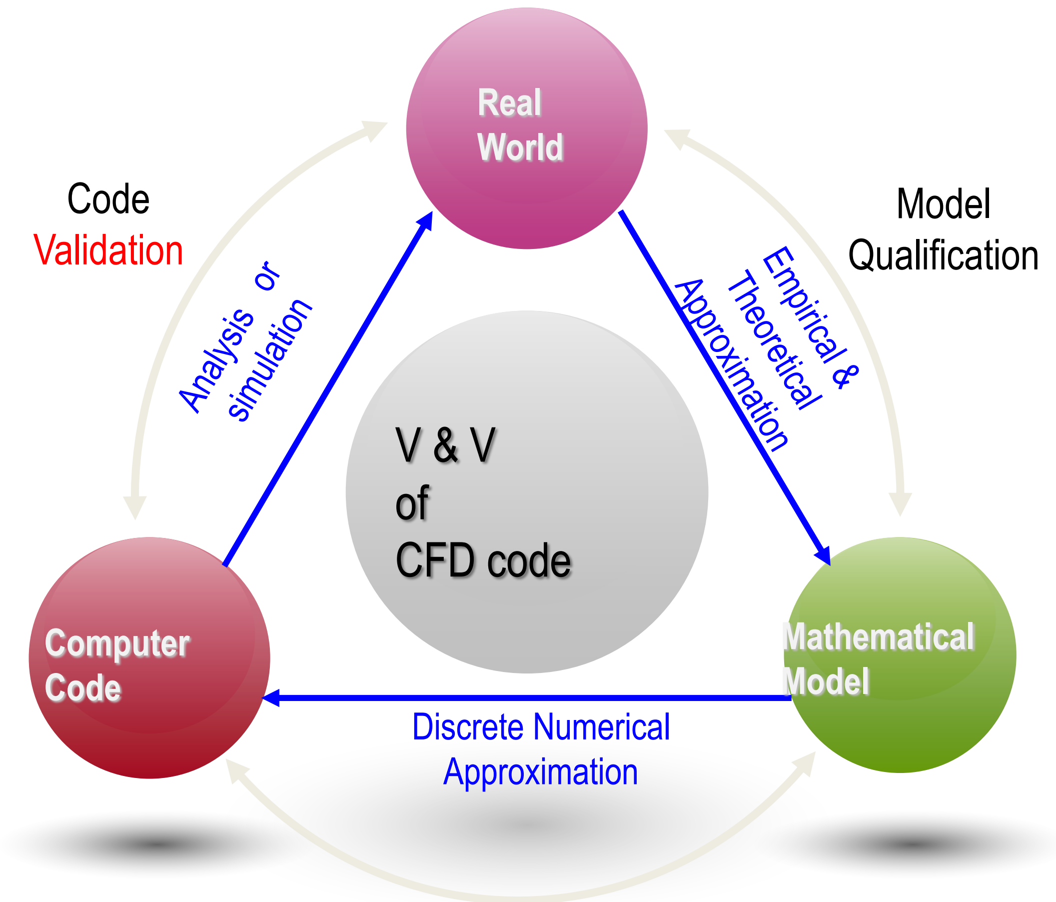


The successive steps for establishing and solving the equations

Verification & Validation (V&V)



(AIAA guide, 1998)

Verification

- Verification is the assessment of the accuracy of the solution to a computational model by comparison with known solutions.
- The relationship of the simulation to the real world is not an issue.
- Verification is primarily a mathematics issue.

Validation

- Validation is the assessment of the accuracy of a computational simulation by comparison with experimental data.
- The relationship between computation and the real world, i.e., experimental data, is the issue.
- Validation is primarily a physics issue.

V&V in CFD

Csni-r2007-13

NRS problems requiring CFD with/without coupling to system codes (1/2)

	NRS problem	System classification	Incident classification	Single- or multi-phase
1	Erosion, corrosion and deposition	Core, primary and secondary circuits	Operational	Single/Multi
2	Core instability in BWRs	Core	Operational	Multi
3	Transition boiling in BWR/determination of MCPR	Core	Operational	Multi
4	Recriticality in BWRs	Core	BDBA	Multi
5	Reflooding	Core	DBA	Multi
6	Lower plenum debris coolability/melt distribution	Core	BDBA	Multi
7	Boron dilution	Primary circuit	DBA	Single
8	Mixing: stratification/hot-leg heterogeneities	Primary circuit	Operational	Single/Multi
9	Heterogeneous flow distribution (e.g. in SG inlet plenum causing vibrations, HDR expts., etc.)	Primary circuit	Operational	Single
10	BWR/ABWR lower plenum flow	Primary circuit	Operational	Single/Multi

V&V in CFD

NRS problems requiring CFD with/without coupling to system codes (2/2)

	NRS problem	System classification	Incident classification	Single- or multi-phase
11	Waterhammer condensation	Primary circuit	Operational	Multi
12	PTS (pressurised thermal shock)	Primary circuit	DBA	Single/Multi
13	Pipe break – in-vessel mechanical load	Primary circuit	DBA	Multi
14	Induced break	Primary circuit	DBA	Single
15	Thermal fatigue (e.g. T-junction)	Primary circuit	Operational	Single
16	Hydrogen distribution	Containment	BDBA	Single/Multi
17	Chemical reactions/combustion/detonation	Containment	BDBA	Single/Multi
18	Aerosol deposition/atmospheric transport (source term)	Containment	BDBA	Multi
19	Direct-contact condensation	Containment/ Primary circuit	DBA	Multi
20	Bubble dynamics in suppression pools	Containment	DBA	Multi
21	Behaviour of gas/liquid surfaces	Containment/ Primary circuit	Operational	Multi
22	Special considerations for advanced (including Gas-Cooled) reactors	Containment/ Primary circuit	DBA/BDBA	Single/Multi

DBA – Design Basis Accident; BDBA – Beyond Design Basis (or Severe) Accident; MCPR – Minimum Critical Power Ratio

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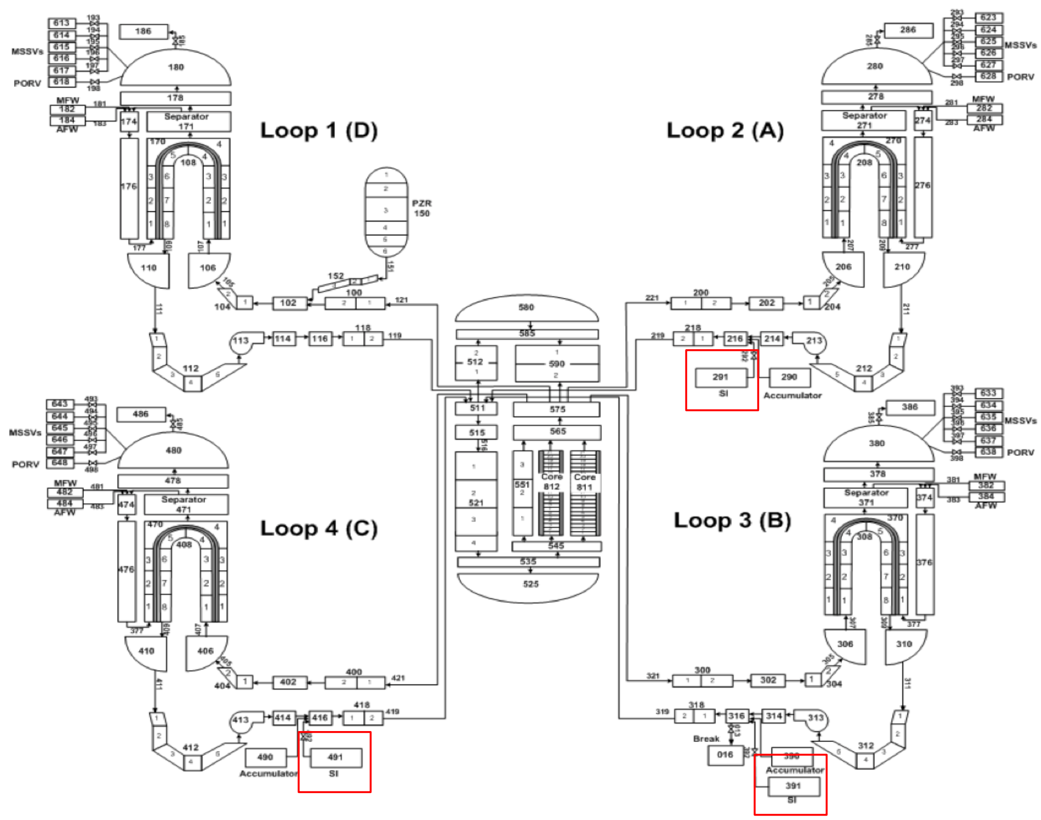
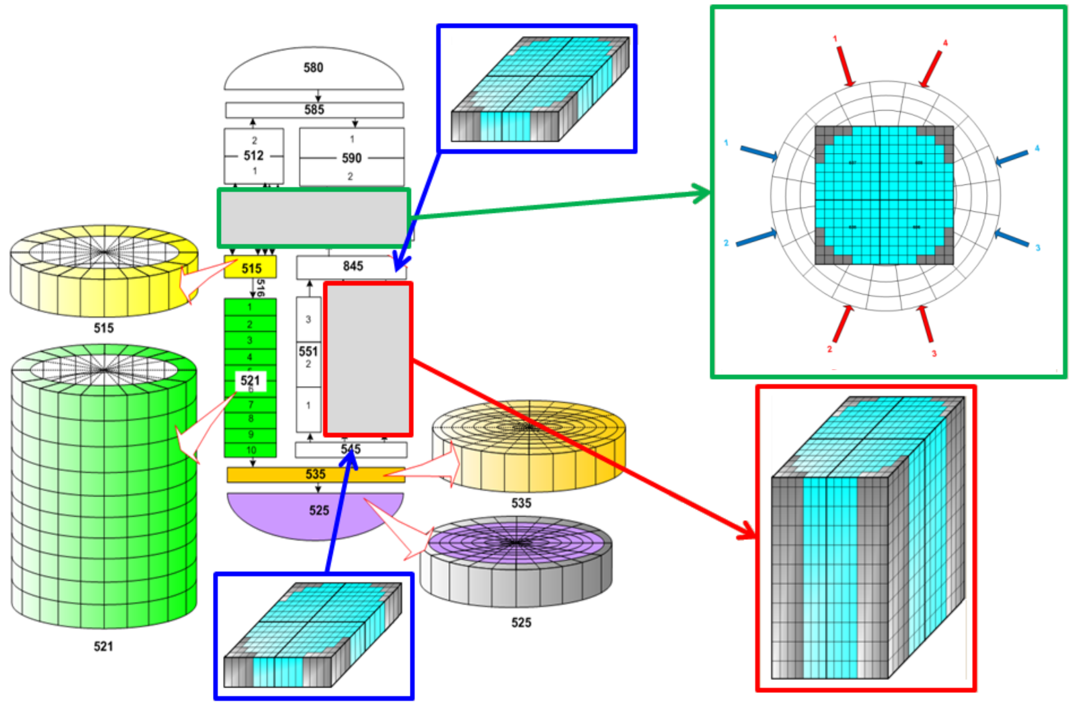
RELAP5-3D Nodalization Diagram (1D Model)



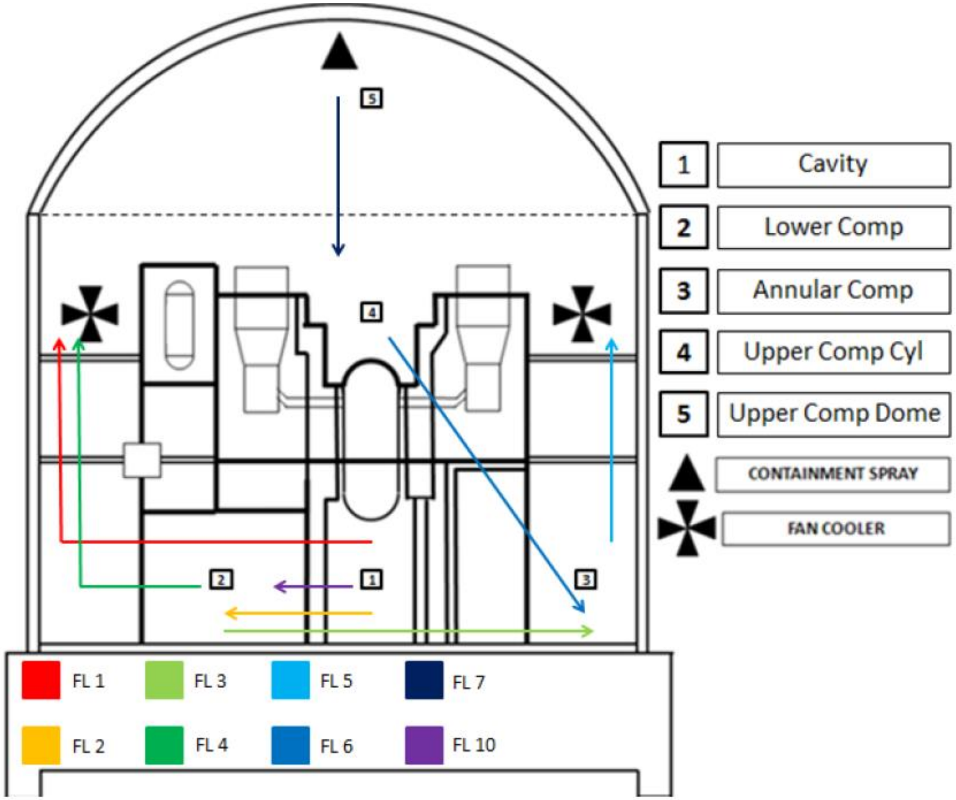
RELAP5-3D Nodalization Diagram (3D Model)



Realistic <u>Cold and Hot Legs Layout</u> (from CAD Drawings)

<u>193 Fuel Channels</u> with cross flow individually simulated

MELCOR Nodalization Diagram



Analysis of Core Blockage Scenarios

Preliminary Analysis (1D Core)

Purpose:
- Identify possible alternative flow paths for core cooling
- Identify critical scenarios that may lead to core damage

Conditions:
- Instantaneous full core and core bypass blocked at sump switchover
- No core cross flow (1D model)

Larger breaks in cold leg may lead to core damage

	Break Location	
Break Size	**Cold Leg**	**Hot Leg**
Small (2")	Pass	Pass
Medium (6")	Fail	Pass
Large (DEG)	Fail	Pass

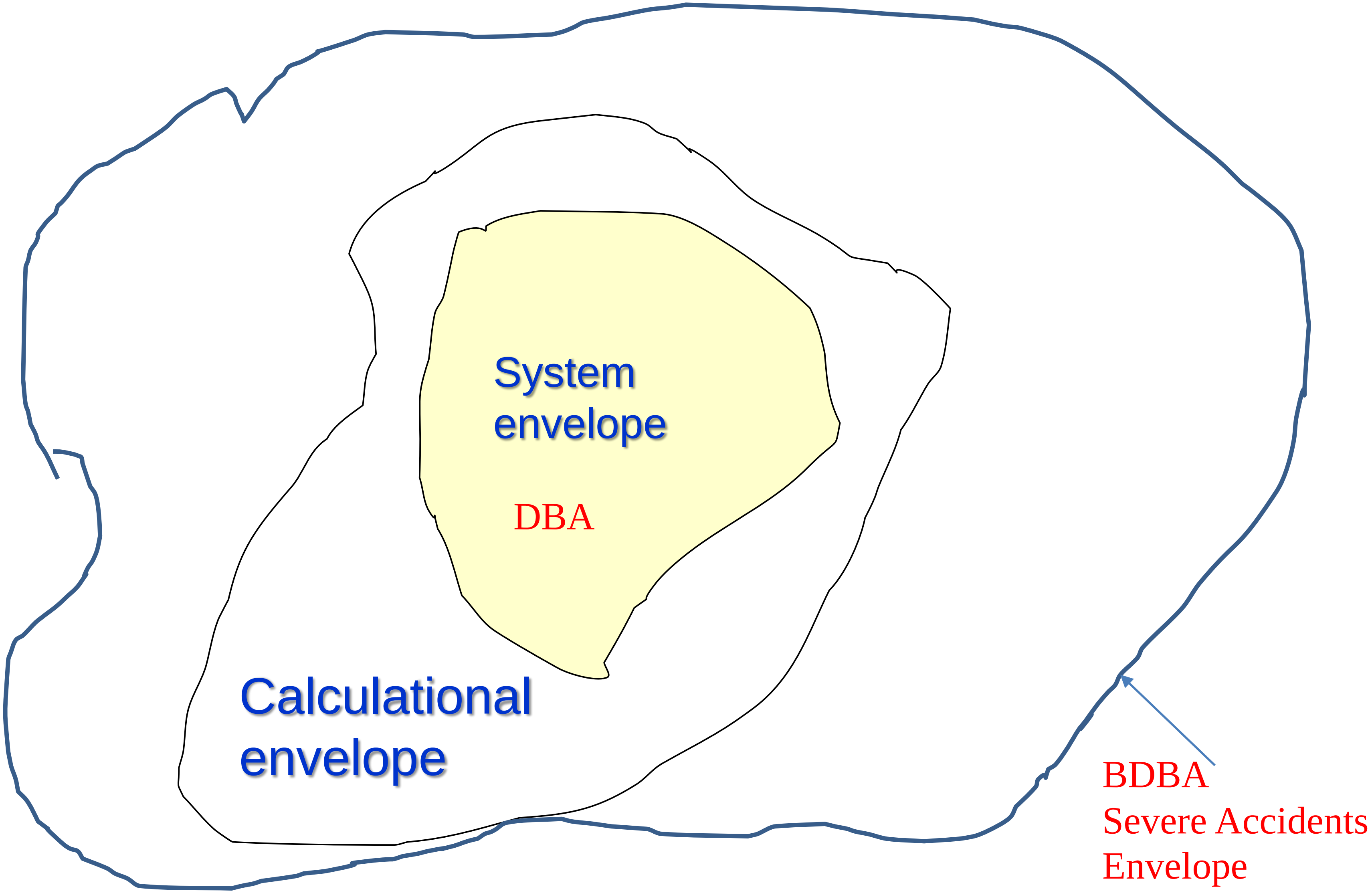


Diagram diagram of system and calculation envelopes

Incorporate lessons learned from recent experience and previous works + CFD/System Code Calculations + Experimental Data + ….. = BIG DATA

BIG DATA → SMART SIMULATOR

neural networks can be utilized to demonstrate the feasibility of ANNs to predict the event progress based on the available data

Mitigating Strategies for Beyond Design Basis Events

- **Change operational mode to avoid severe states**

- **Alter the unfavorable events (How we get here?)**
- **Interpretation and collection of the old and recent data & benchmarks & Identify uncertainty**

- **Mitigate the consequences using simulators**

- **Lower the cost**

- **Conduct many analyses to identify vulnerability**

- **Support / use predictive multi-scale multi-physics modeling and simulation**

Nuclear Power is Different *Because* Accidents are Rare, But Can be High-Consequence

Plants Must Operate at a High Level of Safety for 60 years – perhaps longer

Most Difficult of all Industrial Activities to Explain to the General Public

Success will be achieved by international collaboration