Karlsruher Institut für Technologie

Importance of LWR Best-Estimate Safety Codes for the Analysis of Fukushima-like Accidents

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- Safety Assessment and Role of Numerical Codes
- Multi-scale Best-Estimate Thermal Hydraulic Methods
- Best-Estimate Methodologies for LWR Including Uncertainty
- Toward High-Fidelity Multi-physics LWR Simulations
- Summary

Motivation

- Fukushima unfortunately happen even though
  - Lesson learnt from TMI-1 and Chernobyl
  - Periodic safety assessment based on probabilistic and deterministic methods
  - National and international evaluation of NPP events
  - Worldwide upgrade of nuclear power plants to meet today's safety requirements

- Accident analysis starts with Postulated Initiating Events (PIE), categories of events and bounding analysis
  - Fukushima sequence of events not really anticipated in accident analysis
  - Assumptions on initiators and availability of safety systems did not cover Fukushima accident

- Fukushima accident progression revealed the importance of
  - Operational and emergency handbook
  - AMM and Severe Accident Management guidelines (SAM)

- Development of proper countermeasures to control accidental sequences (prevention) and to limit their consequences (mitigation) requires a deep understanding of physical phenomena
  - Numerical safety analysis codes can help a lot
  - Predictive capability of "safety analysis codes" increased significantly last decades

- Safety demonstration relies on "validated" numerical codes and must be based on state-of-the-art

Defense in Depth Concept (DiD)

Safety level (DiD Level)			Goal	Deployed systems	Procedures
1	Intended operation	Normal operation	prevention of abnormal operation and failures	operational systems, limiting and protection systems	conservative design, quality assurance
2		Operational occurrences	prevention of DBAs	operational systems, limiting and protection systems	inherent safety by design (negative temperature feedback to reactivity)
3	Design Basis Accidents (DBA)		control of accident and prevention of BDBA	passive und active safety systems	inherent safety by design
4	Beyond Design Basis Accidents (BDBA)	a: very unlikely events e.g. ATWS, etc.	control of event	safety systems and/or still available operational systems	accident management incl. operator actions:
		b: multiple failure of safety systems	prevention of core damage		preventive
		c: severe accident incl. core damage	mitigation of consequences		mitigative

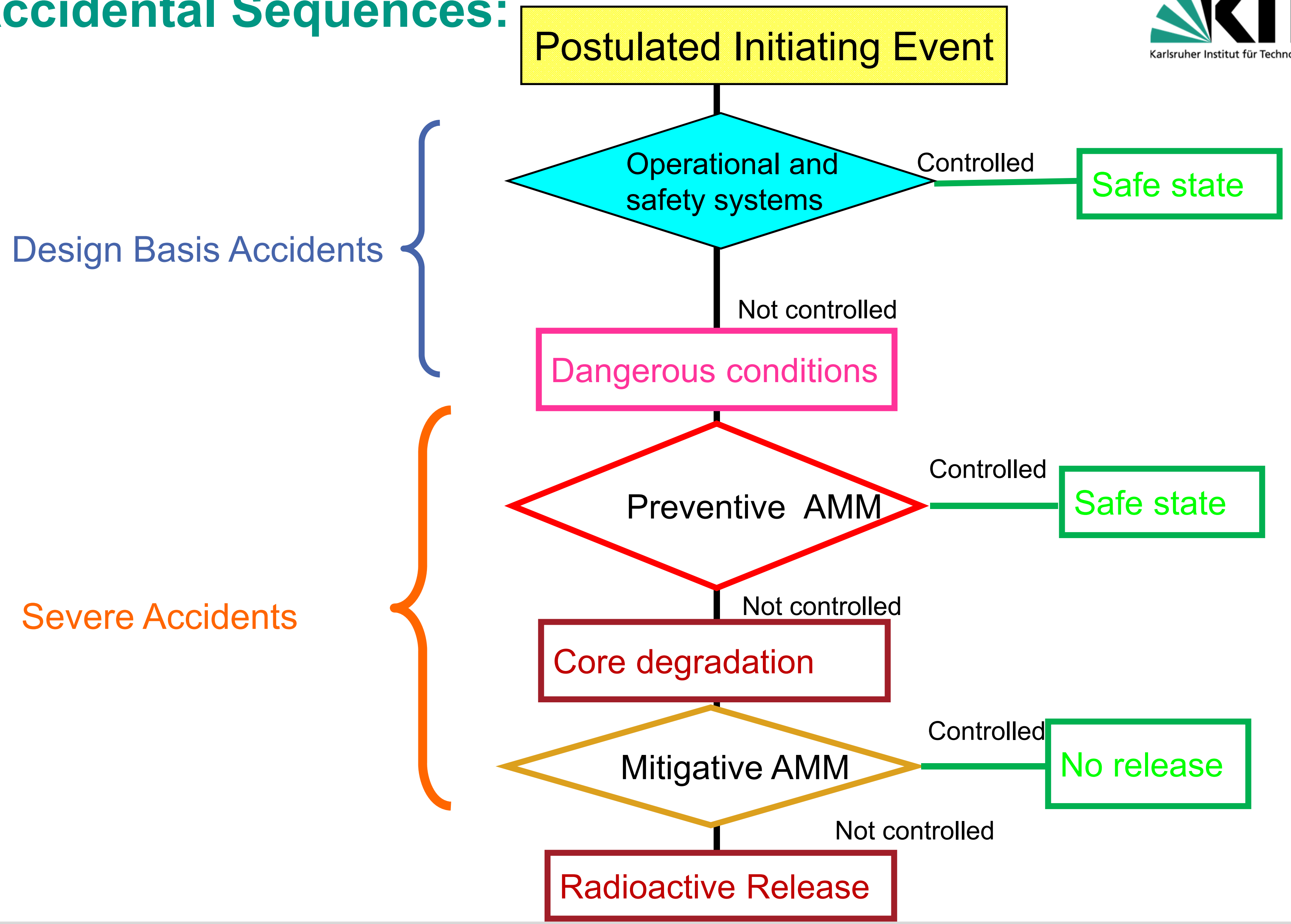
DiD: Defense in Depth

DBA Scenarios: level 3 and 4a: DBA + ATWS[1)]

Beyond DBA: scenarios of interest at level 4b and 4c

[1)]: **A**nticipated **T**ransient **W**ithout **S**CRAM (ATWS) considered as level 4a event, but traditionally deterministically analysed for LWRs

Accidental Sequences:



Safety Demonstration
Based mainly on Numerical Simulation Codes

- Neutron physical and thermal hydraulic, structural mechanic design tools
    - **Compliance of** inherent safety feature and safety limits during normal operation
- Neutron physical/thermal hydraulic system codes
    - **Compliance of** safety limits during transients and DBA
- TH system codes coupled with models for chemo-physical material behavior of severe accidents
    - **Compliance of** safety limits during beyond DBA
- Thermal hydraulic phenomena in Containment
    - **Compliance of** containment design limits (last barrier)
- Dispersion of radioactive material in case of hypothetical core meltdown accidents
    - **Compliance of** radiological limits in case of DBA and hypothetical accidents

Multi-scale
Best-Estimate Thermal Hydraulic Methods

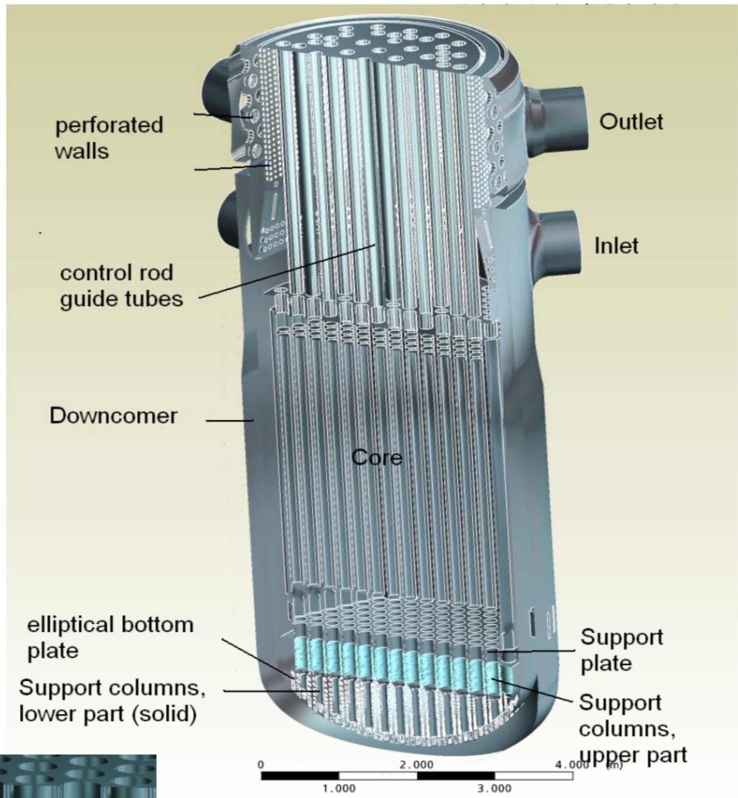
- Main goals:
  - Better description of phenomena inside primary circuit including the reactor core (3D effects)
  - Improvement of two-phase flow models of sub-channel codes (KTF) and CFD codes
  - Multi-scale coupling of thermal-hydraulic models

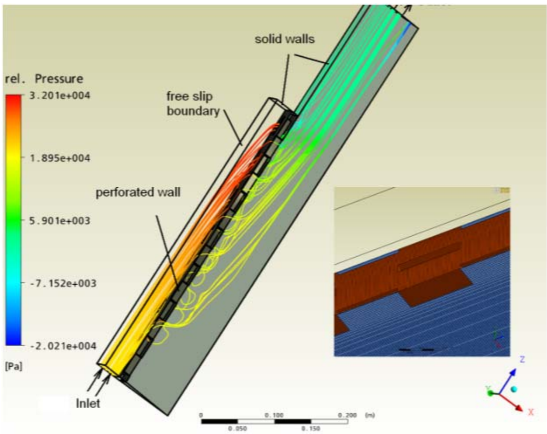
- Focus on multi-scale modelling of nuclear power plants (fuel assemblies, core, primary system, secondary systems)
  - CFD Codes: NEPTUNE CFD
  - Sub-channel codes: SUBCHANFLOW and KTF
  - System codes: TRACE, RELAP5, ATHLET

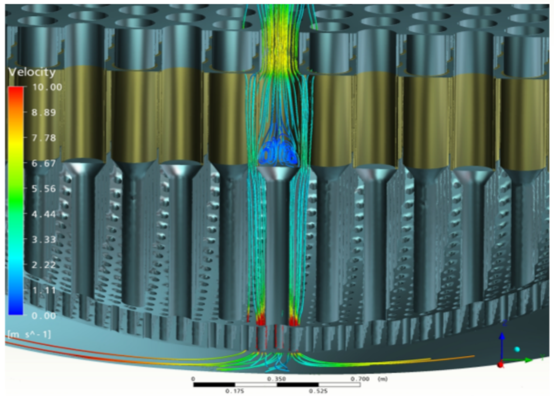
- Validation and Verification:
  - Int. benchmarks e.g. V1000-CT Benchmark, OECD BFBT, PSBT Benchmark

PWR RPV
CFD Multi-scale Modeling

- Downcomer and lower plenum:
  - Elliptical bottom plate resolved by grid
  - Loss coefficient derived from detail model
- Development chain
  - Δp obtained from stand-alone full detail model (3 Mio. cells)
  - Implementation of Δp coefficient in the coarser RPV model (5000 cells)
- Computing effort:
  - Linux Cluster Xeon 2.4 GHz: 2 weeks CPU time (12 processes parallel) for 1800s transient
  - Model development at Windows PC (XP64 Bit-system)







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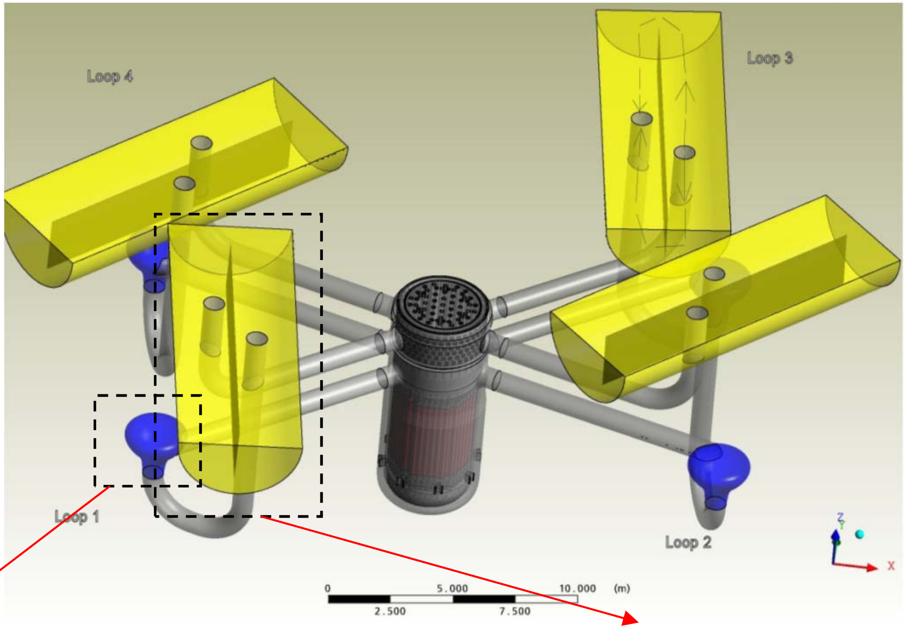
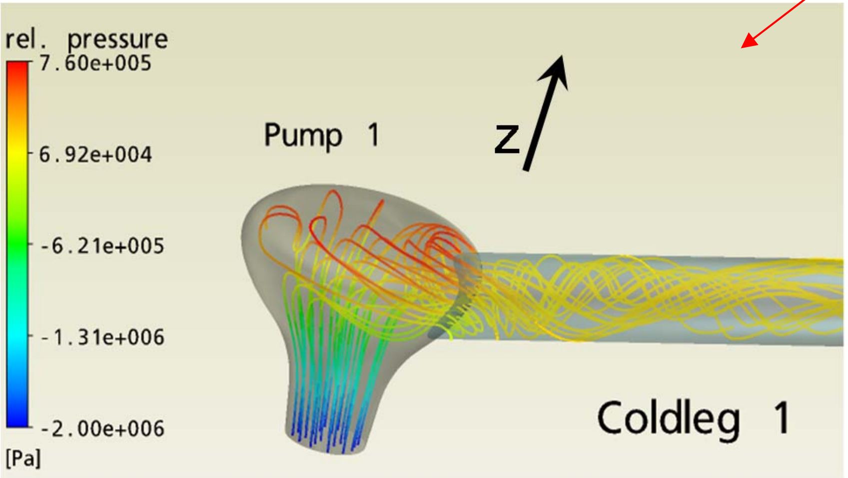
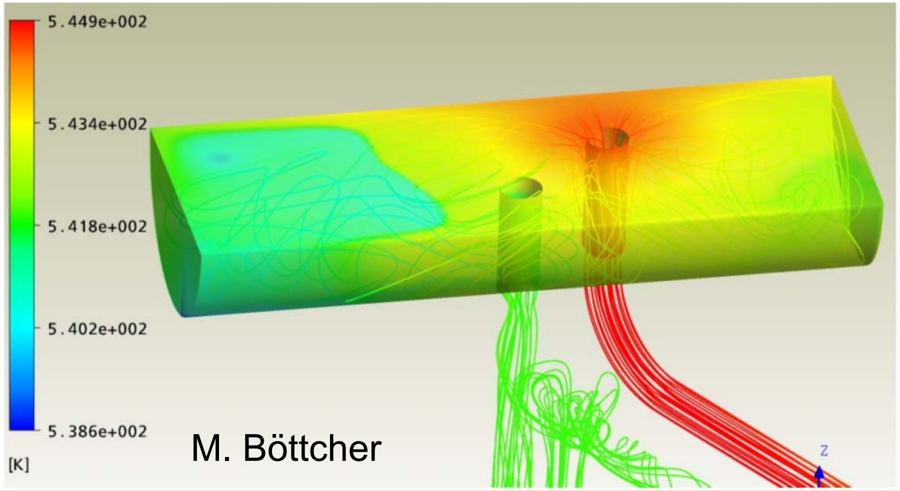
PWR Plant: CFD Integral Primary Loop Model: Multi-scale Approach

- Primary loops:
  - Steam generators and pumps: porous media approach; heat and momentum exchange by volumetric source terms
  - 34 000 000 cells (for all), Δ~50mm

- Model assumptions:
  - Parts of the geometry are not available like pump impellers or several details from the pump housing
  - Construction without CAD geometry (partially)







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Best-Estimate Methodologies for Transient Analysis of BWR Including U&S Quantification

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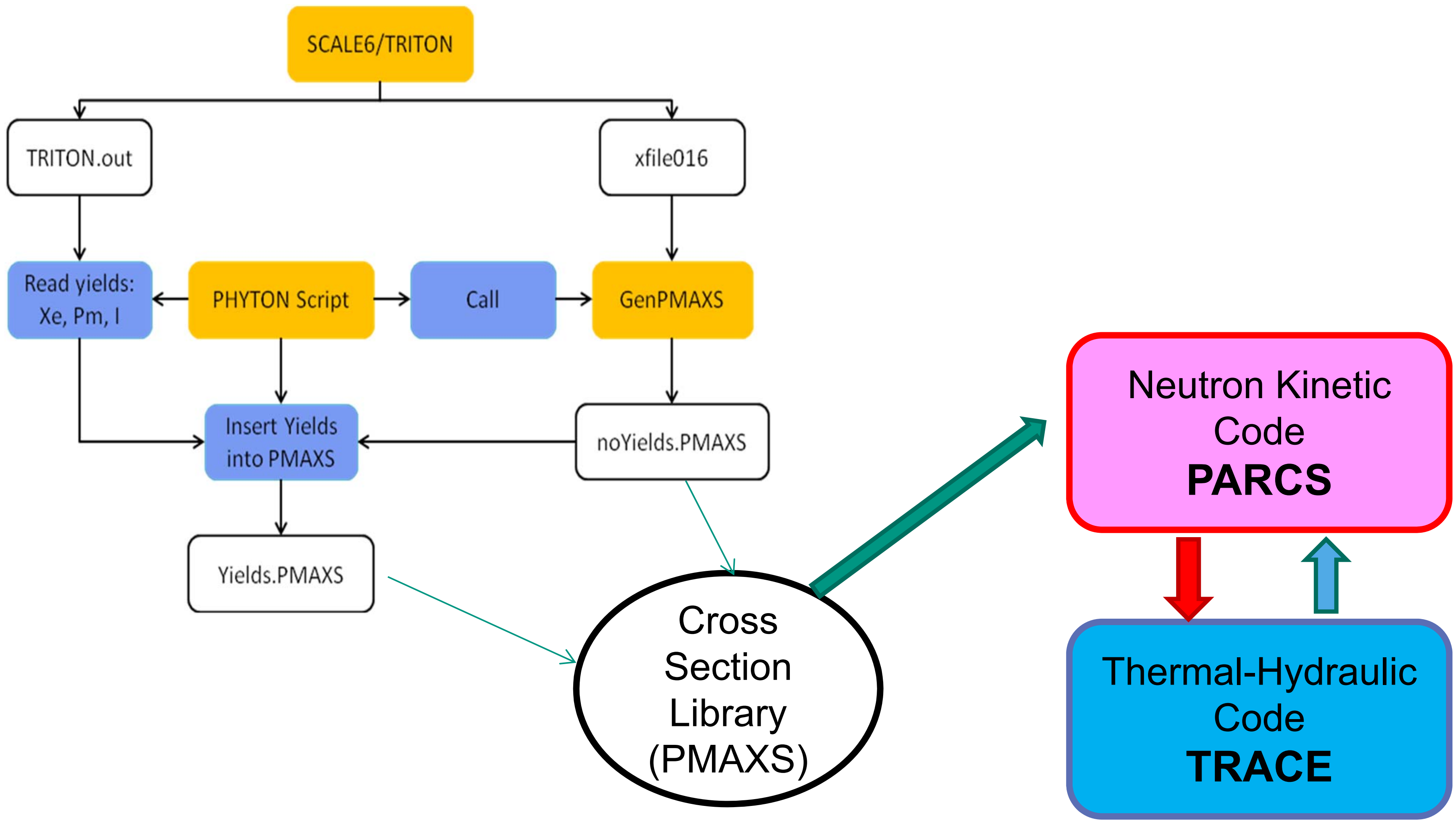
- Goal: Realistic description of thermal hydraulic and neutron kinetics processes taking into account interactions between TH and NK

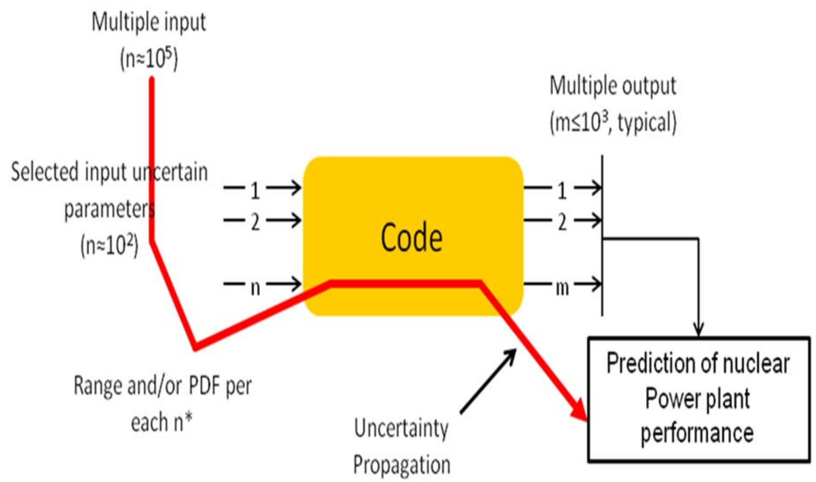
- Approach is based on following steps:
  - XS generation with the lattice code SCALE6/TRITON for any core loading
  - Multidimensional thermal hydraulic and neutron kinetic core models
  - Transient simulation with coupled TH / NK codes such as TRACE/PARCS
  - Quantification of code's uncertainties

Computational Route
Based on SCALE/PARCS/TRACE

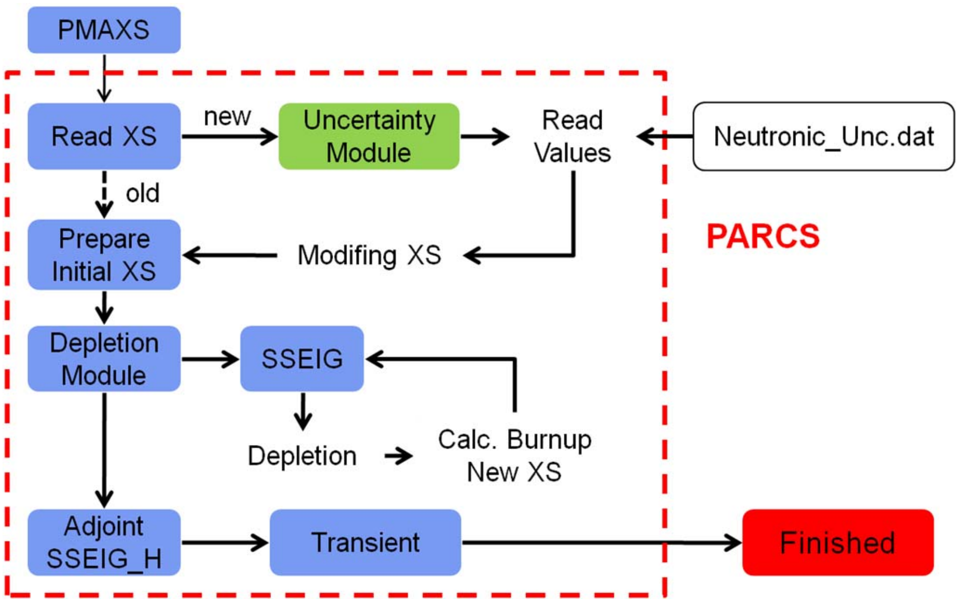


Uncertainty and Sensitivity Quantification (1)

- **Coupling of PARCS with SUSA: Propagation of uncertainty of input parameters to output ones by Monte Carlo sampling**



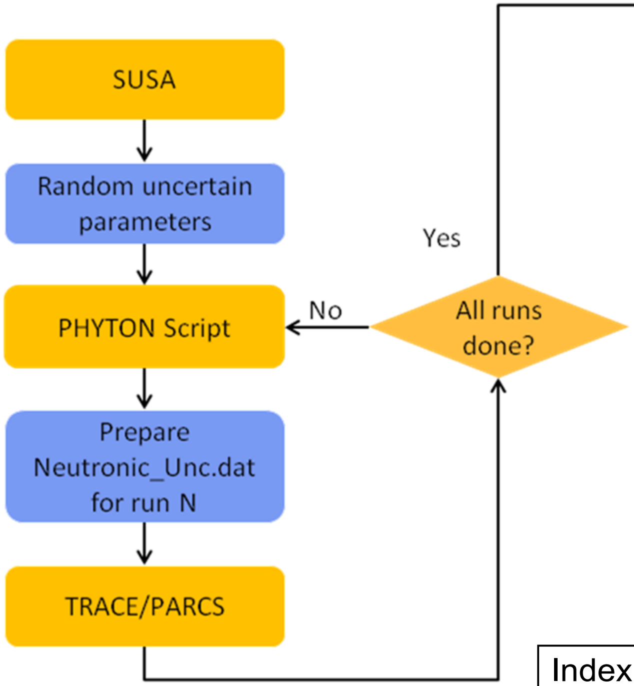
Scheme of Uncertainty Propagation



New developed PARCS/SUSA Scheme

Uncertainty and Sensitivity Quantification (2)

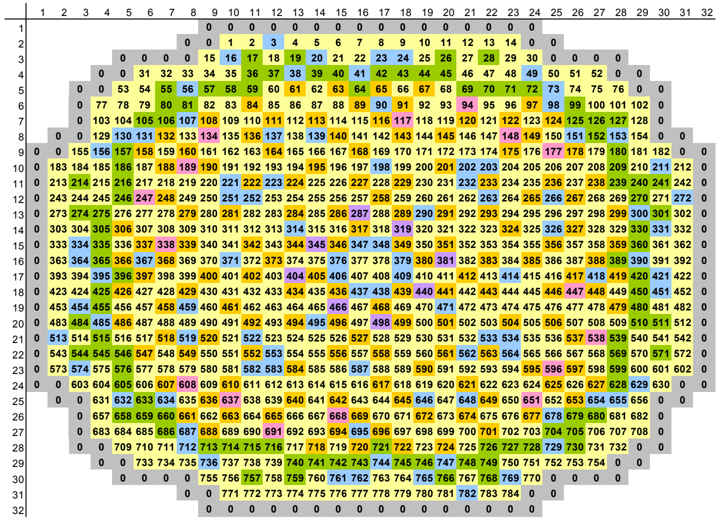
Flow chart of uncertainty analysis for TRACE/PARCS with SUSA

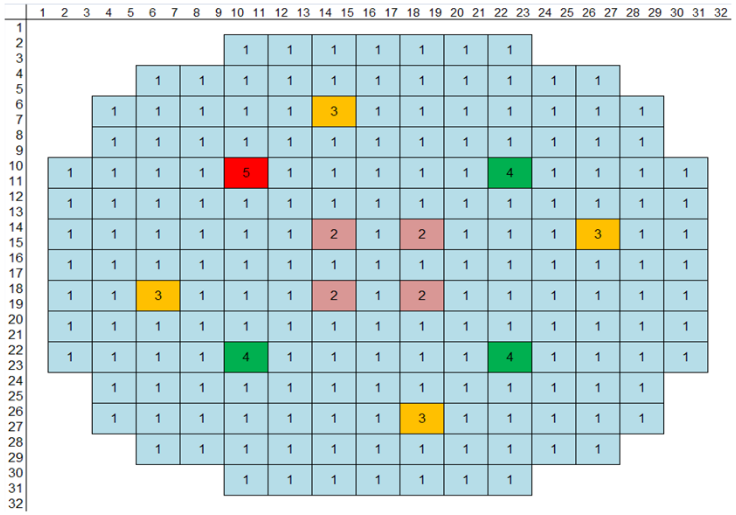


Index	1	2	3	4	5	6	7	8	9	10	11
Parameter	$\Sigma_{t,1}$	$\Sigma_{t,2}$	$\Sigma_{a,1}$	$\Sigma_{a,2}$	$\nu\Sigma_{f,1}$	$\nu\Sigma_{f,2}$	$\kappa\Sigma_{f,1}$	$\kappa\Sigma_{f,2}$	$\sigma_{Xe,1}$	$\sigma_{Xe,2}$	$\sigma_{Sm,1}$
Index	12	13	14	15	16	17	18	19	20	21	22
Parameter	$\sigma_{Sm,2}$	$\Sigma_{f,1}$	$\Sigma_{f,2}$	Σ_{s12}	ADF_1	ADF_2	InV_1	InV_2	yield	β	λ

List of important parameters for S&U

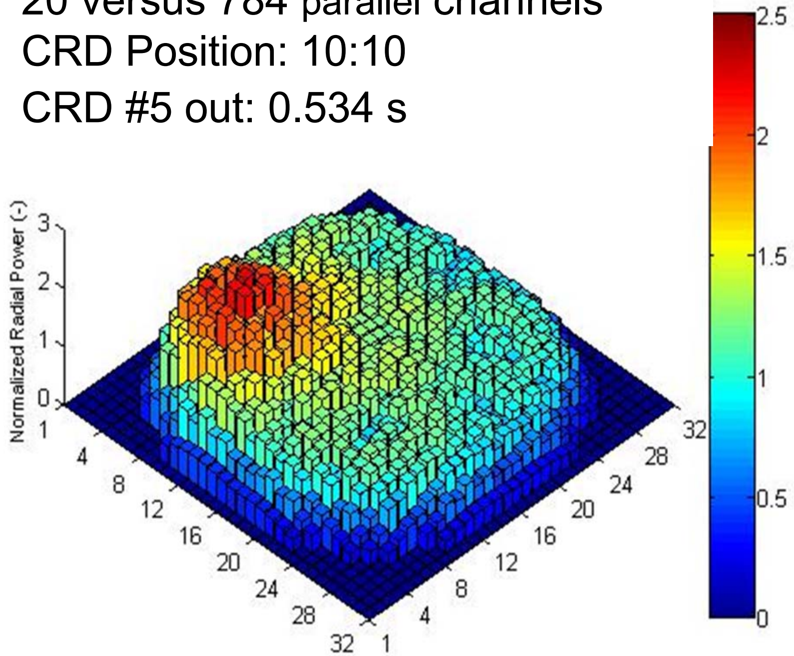
BWR Control Rod Drop Analysis





Models:

- BWR core
- Power: 3840 MWth
- 20 versus 784 parallel channels
- CRD Position: 10:10
- CRD #5 out: 0.534 s

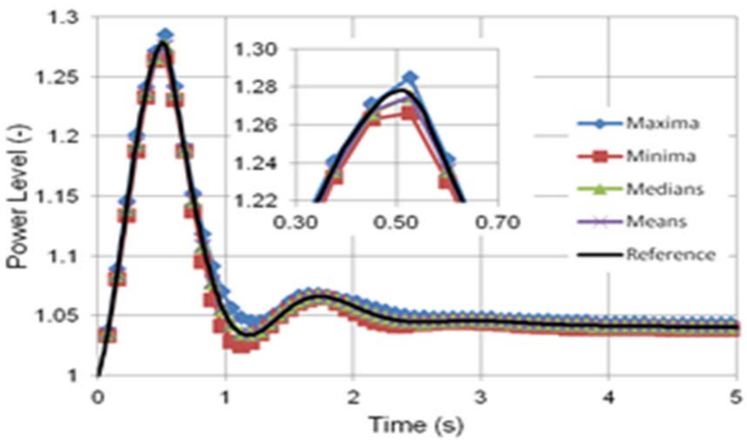


Radial power distribution at P_{max} (0.510s)

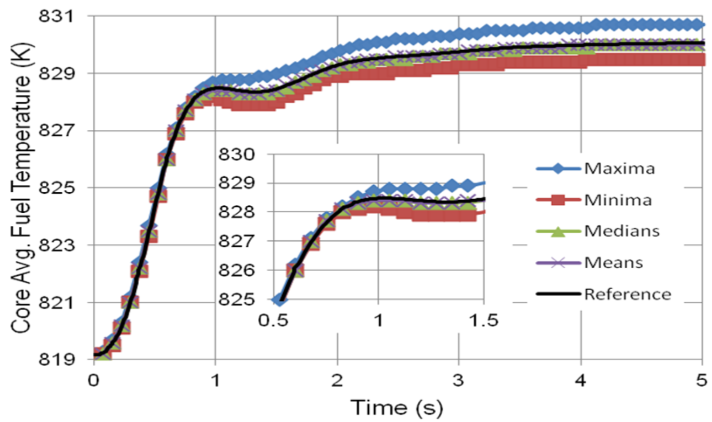
Control Rod Drop Accident (CDA):

- TH Model with 20 CHAN: 128 % of P
- TH Model with 784 CHAN: 125 %

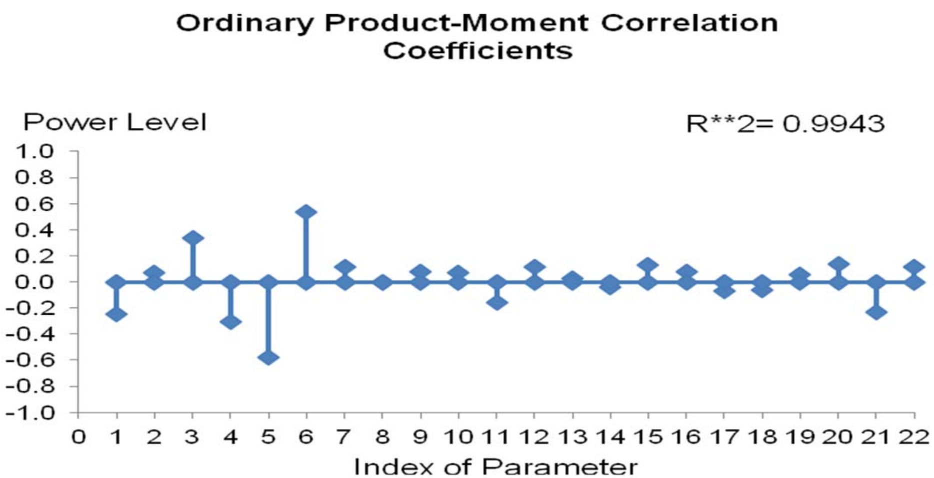
BWR CDA
Analysis including U & S Analysis



Rel. Power Evolution during Rod Drop Accident with uncertainty band



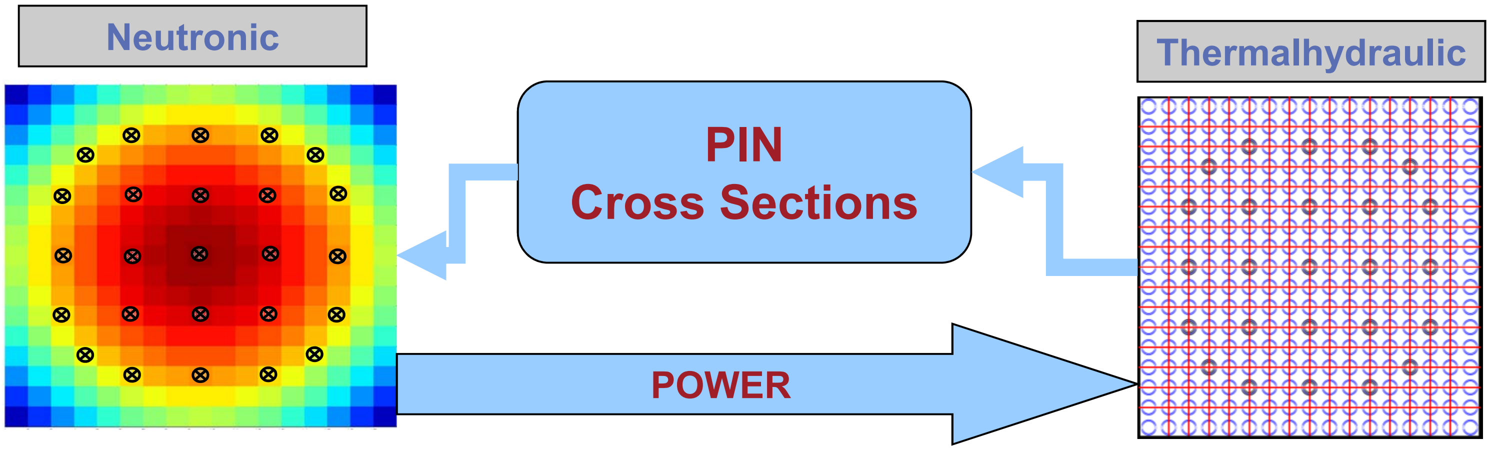
Core average fuel temperature uncertainty band



- **Most influencing parameters:**
  - **Transport cross sections Σt**
  - **Fission ν Σf and**
  - **Absorption Σa= Σc and Σf**
- **↑ thermal νΣf2 ➔ ↑Pmax**

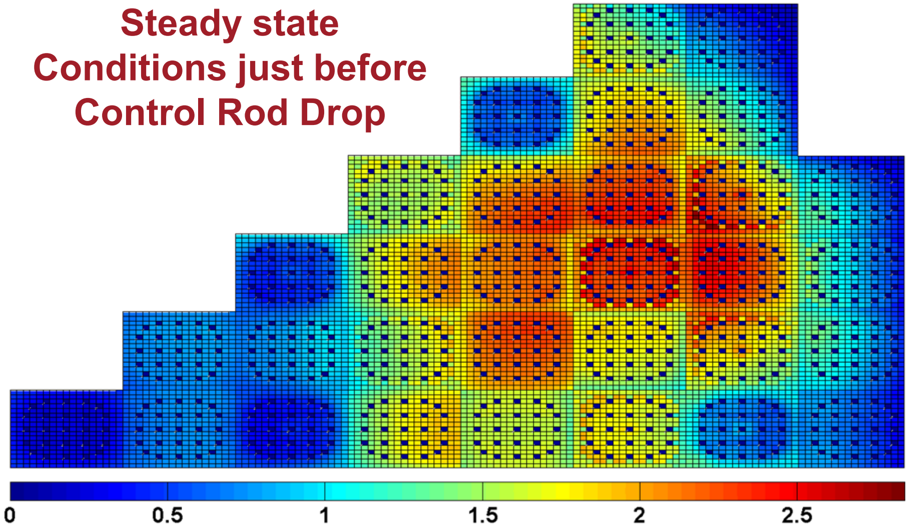
LWR
Advanced Methods for Safety Assessment of DBA

- **EU NURISP Project**
- **USA CASL Project**
- **High-fidelity / multi-physics developments: From FA to pin-based simulations**
  - **Direct prediction of local safety parameters at cell level**
  - **Reduction of conservatism**
  - **Coupling of a time dependent SP_3 Transport with a sub-channel code: DYNSUB**

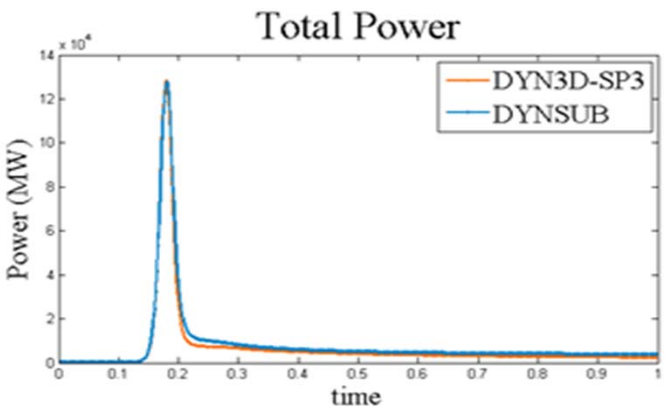


DYNSUB: PWR MOX REA Benchmark

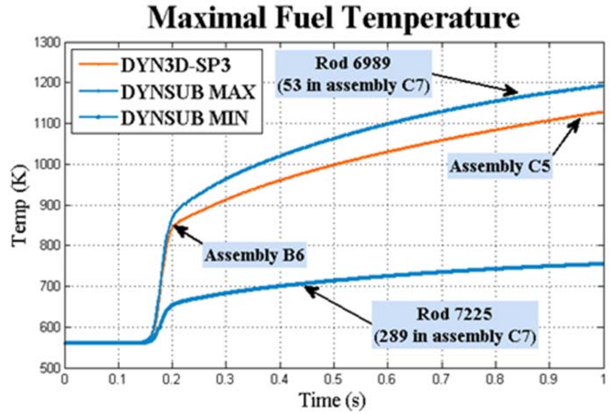
Steady state
Conditions just before
Control Rod Drop

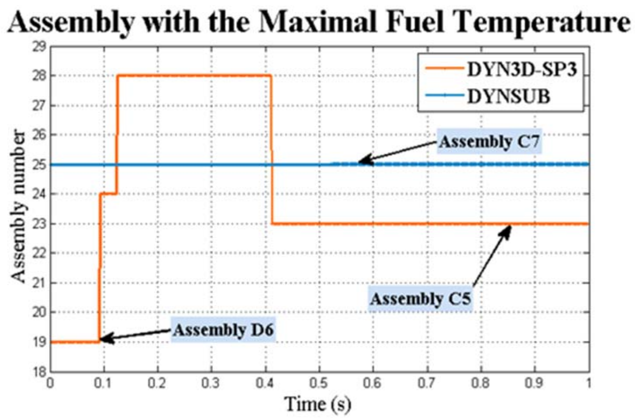


DYNSUB: rel. Pin Power of hottest layer 9



DYNSUB: Global Parameters





DYNSUB: Prediction of local safety parameters

Summary

- **Continuous evaluation of "plant safety status" of operating reactors is very important**
  - **Identification of weakness for upgrades and reduction of residual risk**
  - **Rethinking of PIE categories and bounding approach needed**

- **Best-estimate numerical simulation codes may help to**
  - **Better understand the sequence of accidents**
  - **Develop preventive and mitigative measures**
  - **Characterize the fuel composition in reactor and fuel storage pool at any operation time**
    - **Potential radioactive release**
    - **Expected dose rates inside / outside the plant**

- **Enhanced predictive capability of codes due to the advances in computer science**

- **International cooperation focused on harmonisation of safety requirements and safety assessment urgently needed**

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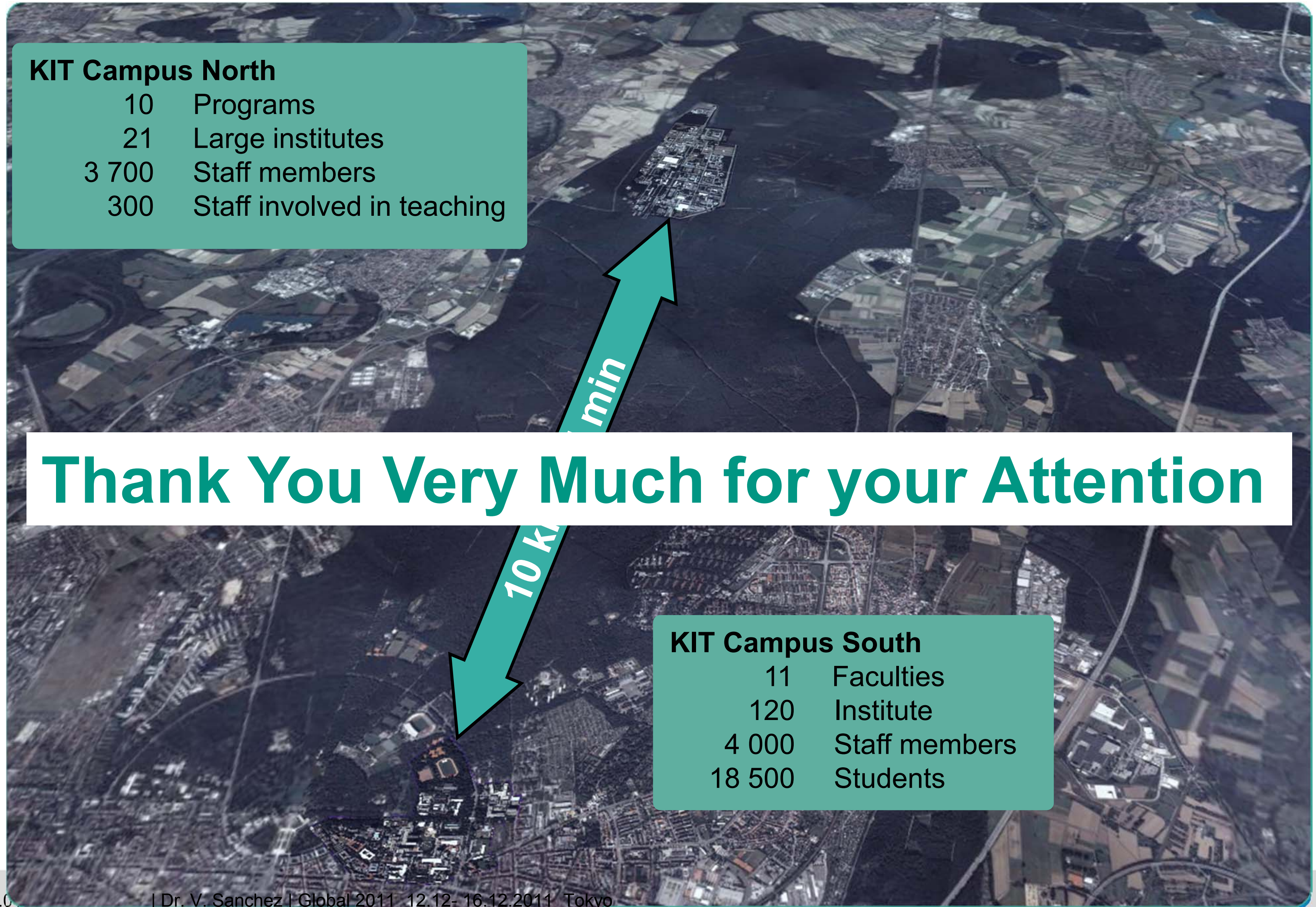

Thank You Very Much for your Attention