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NEUTRONIC AND THERMALHYDRAULIC COUPLING

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Neutronic and Thermalhydraulic Coupling

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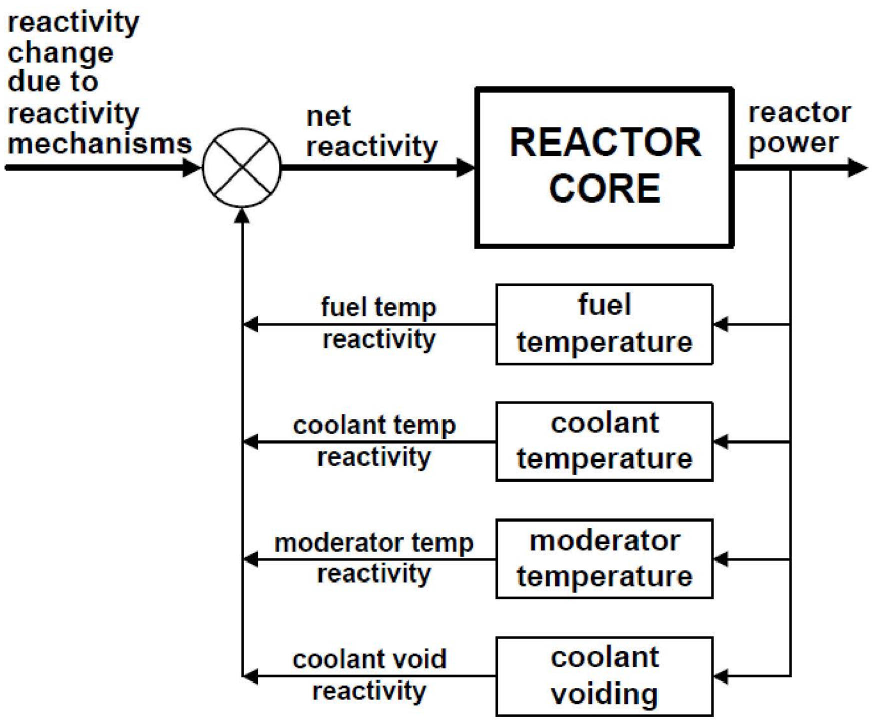
Outline

- Evolution of analysis methods
- Importance of coupling in nuclear safety calculations
  - Dependency of thermalhydraulics on power.
  - Dependency of neutronics on thermal characteristics
  - Density and moderator temperature feedback effects
  - Doppler broadening and Doppler feedback
- SCWR vs. BWR features
- Coupling methodology
  - Hierarchy
  - Spatial mesh, grouping and mapping
  - Temporal stepping
  - Variables, control and relaxation
- Examples (BWR and SCWR)

Evolution of Safety Analysis Methods

- Safety analysis methodologies have in general evolved
  - purely deterministic and conservative approaches using coarse tools
  - Move toward higher fidelity methods
    - e.g., 2 and multigroup diffusion, transport solutions, resonance and spatial self shielding, sub channel hydraulics and CFD
  - "mixed" methods
    - e.g., best estimate codes with conservative assumptions
  - Best Estimate Plus Uncertainty (BEPU) type approaches modelling multiphysics.
    - Coupling of discipline specific codes.
    - New multiphysics integrated codes
- History can be traced within each jurisdiction.
- Current practice for best estimate work usually involves coupled code or multiphysics simulations.
  - Realistic and spatially resolved feedback effects
  - TH-NK coupling

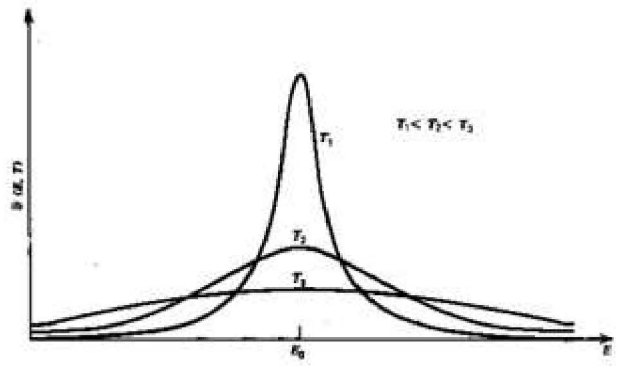
Important Feedback Mechanisms



- Geometrical/expansion based reactivity feedback (thermomechanical effects)
- Fuel burn-up (poisons, breeding, isotopic)

Fuel Temperature Feedback - Need for TH Coupling

- fuel-temperature feedback; that is, the reactivity effect of a change in fuel temperature.
  - This is a major component of the power coefficient of reactivity, since fuel temperature is directly linked to reactor power
- Doppler broadening "extends" the range of a resonance, and results in increased capture in the resonance.
  - Increased capture in U-238 resonances is a **negative** reactivity effect, leading to a **negative fuel-temperature reactivity coefficient**.
  - Pu-239 is present, the low-lying fission resonance at 0.3 eV must also be considered. In this case, increased capture is a **positive** reactivity effect.
  - The fuel-temperature coefficient then becomes **less negative as Pu content increases**.



Doppler Broadening of Resonance with Temperature
[from **Nuclear Reactor Analysis**, by James J. Duderstadt and Louis J. Hamilton, John Wiley & Sons, 1976]

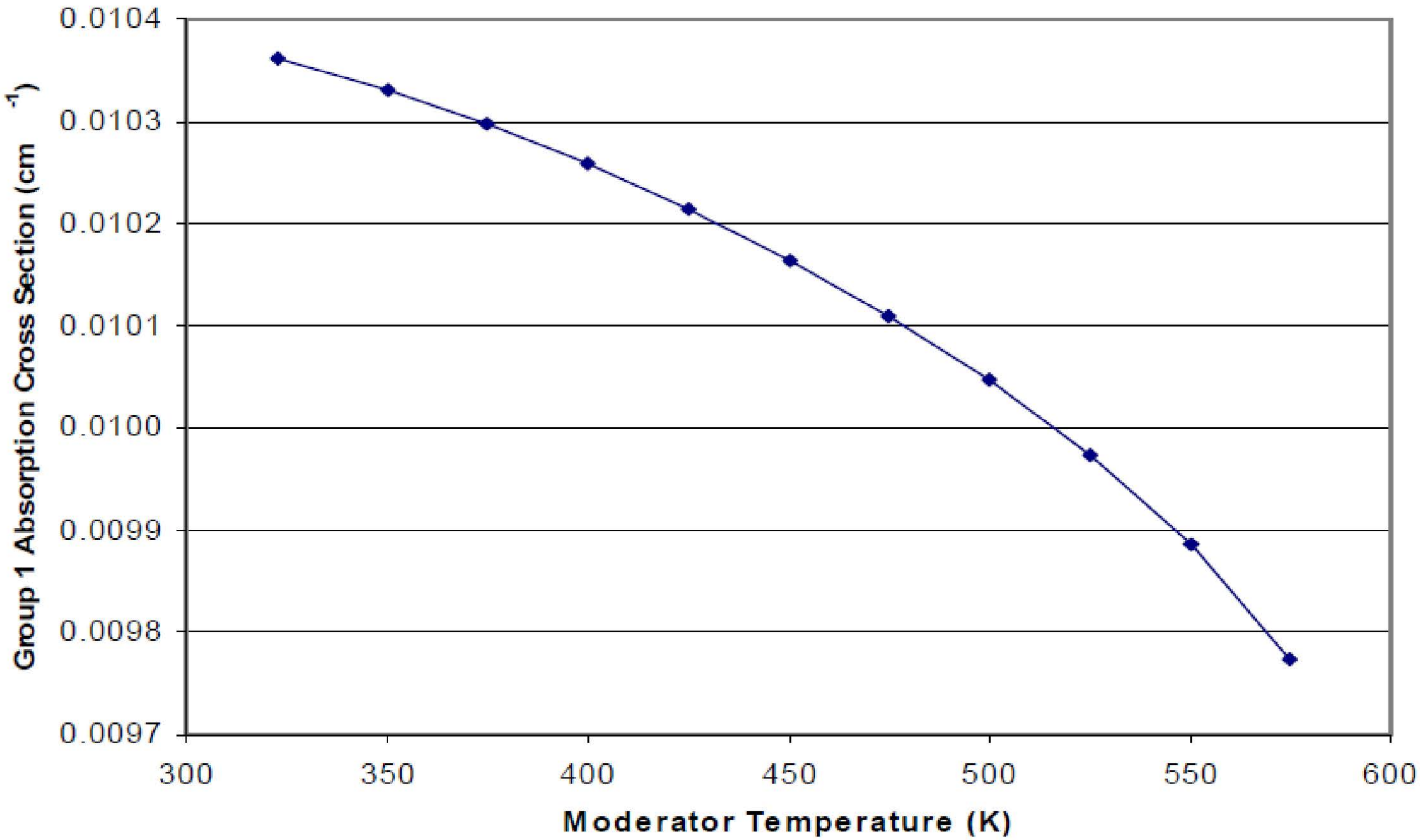
Moderator Temperature Feedback - Need for NK-TH Coupling

- density changes with temperature
  - A decrease in moderator density decreases the effectiveness by which neutrons are slowed down through the resonance region.
    - Hence the resonance absorption increases, causing the resonance escape probability to decrease.
    - The lower moderator density, however, causes the thermal utilization to increase, resulting in a positive temperature effect
  - in liquid-moderated reactors the decreasing moderator density is the dominant effect and causes **<u>the moderator temperature coefficient to be negative</u>**.
    - exceptions may occur under some conditions
    - PWR beginning of cycle with heavy boron loads
- changes in the thermal *neutron energy spectrum play a secondary role*
  - Potentially more pronounced in SCWR

Moderator Temperature Effects PWR Example

Figure 9. Cross-section dependence on moderator temperature

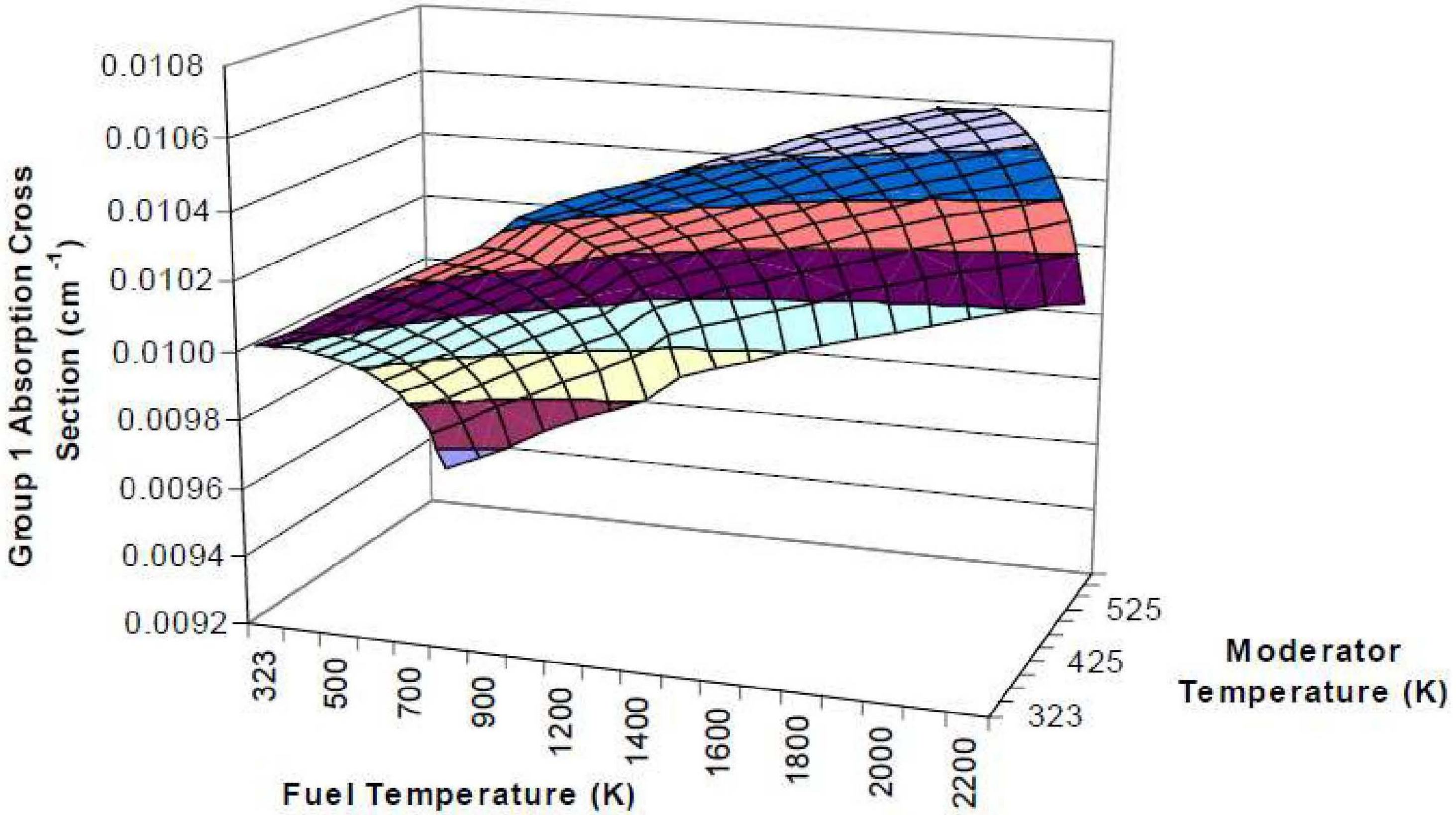
Obtained at PSU with CASMO-3



JOANNA PELTONEN, Thesis, KTH 2009.

Macroscopic Effects PWR Example

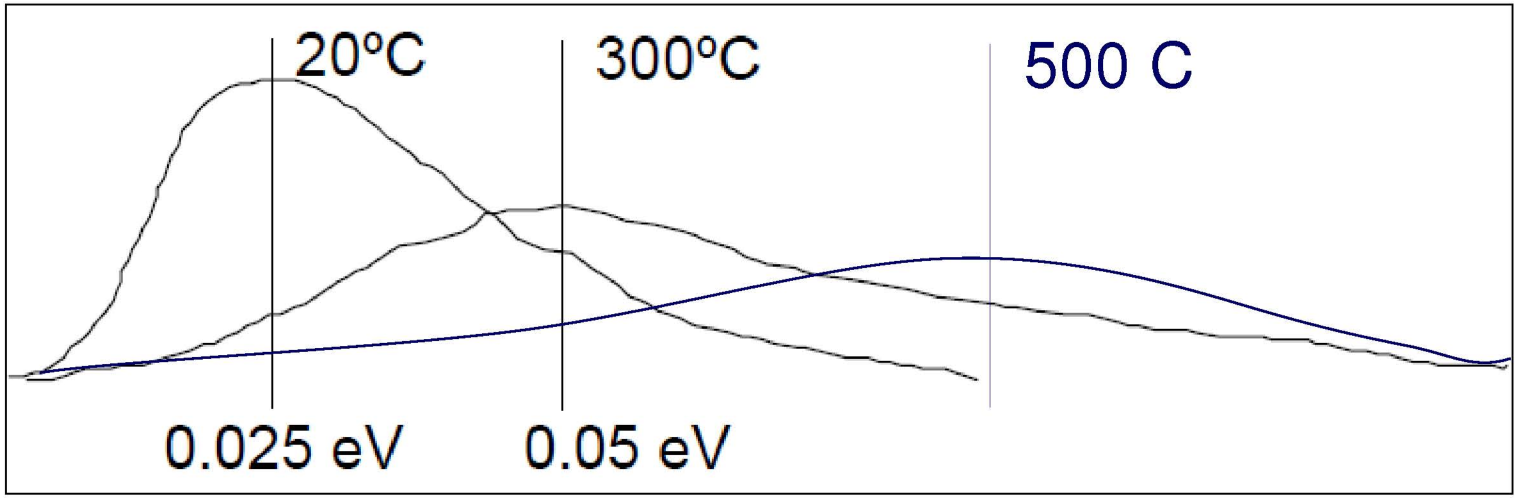
Obtained at PSU with CASMO-3



JOANNA PELTONEN, Thesis, KTH 2009.

Neutron Energy Spectrum

- Hardening the spectrum
  - Shift in the thermal portion of the neutron spectrum to higher energies.
    - i.e., "thermal peak" median value moves to higher energy.



Neutronic Power – Thermal Lag

- Time scale effects:
- The delayed reactivity effect from the moderator temperature variation is a dominant link between the neutronic and thermal-hydraulic behaviour
- There exists a time delay between changes in fission heating in the fuel and the temperature response in the coolant;
  - heat transport from the fuel,
  - across the fuel-clad gap,
  - through the cladding,
  - and into the coolant takes a measurable amount of time.

Thermal Response

- Heat is created in the fuel through the fission process.
- Fission Process (fast time scale, delayed time scale)
  - Fuel coolant and moderator temperatures
  - Geometry
  - Neutron flux, materials, interaction cross sections….
  - Power production
- Heat Transfer Processes (slow time scale)
  - Volumetric power production
  - Radial (axial) conduction in fuel, gap and sheath.
  - Convection to coolant/moderator (plus radio-heating effects).
    - Coolant velocities, geometry, pressure temperature….

High Level Objective

- Since the neutronic properties are influenced by the heat transfer behaviour and vice versa → the “true” solution should include the feedback effects which occur at each spatial location and within each time step.
  - Temperature → density
  - Temperature → microscopic cross sections
  - Thermal spectrum hardening
  - Influence on local and total power.
  - Complete solution involves TH-RP-Fuel (thermomechanical effects)

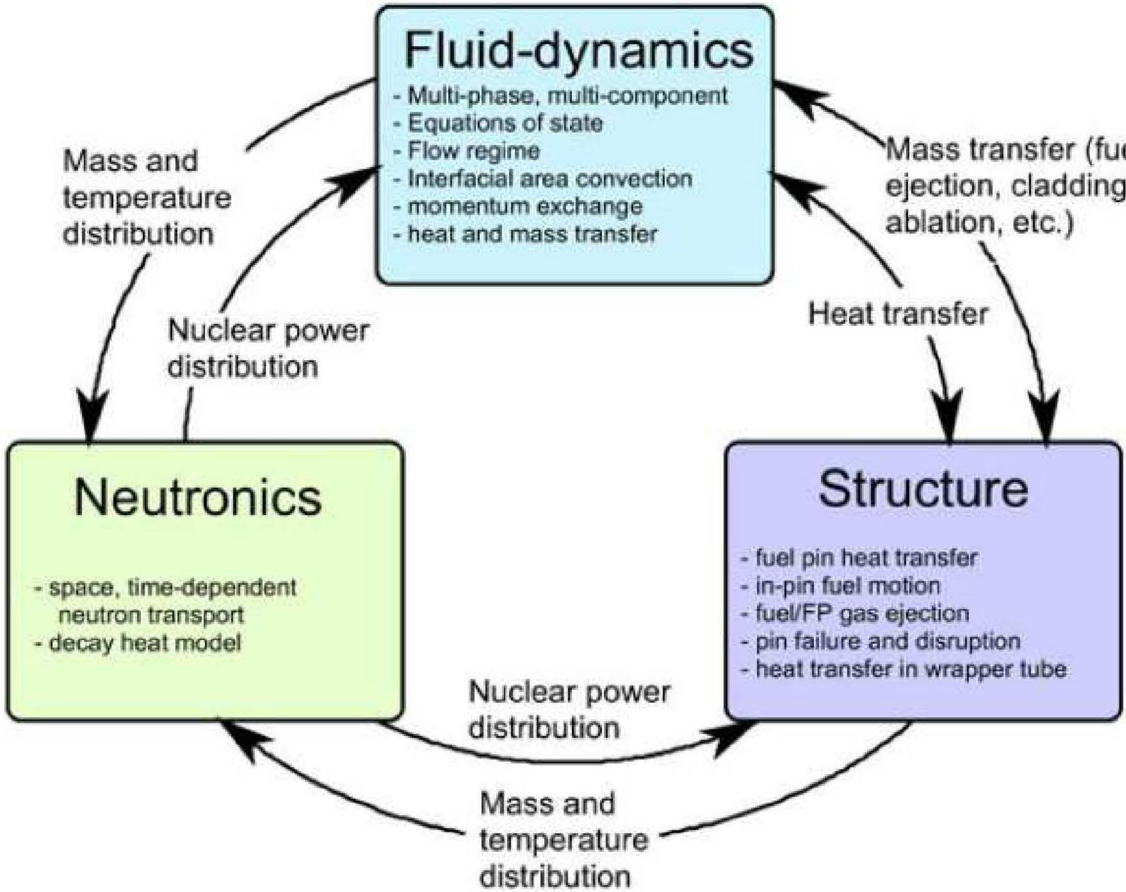
LWR Drivers to Coupling

- Coupling of computer codes would improve the quality of accident analyses for transients where either a strong or uneven feedback effect exists or different solution domains have to be taken into account.
  - Inadvertent control rod withdrawal (uneven feedback);
  - Control rod ejection (strong local feedback);
  - Start-up of a cold or boron free loop (uneven feedback);
  - External asymmetrical boron dilution (uneven feedback);
  - Transients with potential for inherent boron dilution (uneven feedback);
  - Anticipated transients without scram (uneven feedback);
  - Cool-down transients with re-criticality potential (steam or feed lines break (uneven feedback);
  - LOCA with strong influence from containment processes (different solution domains);
  - Severe accident progression and radioactive material transport in the containment (different solution domains).

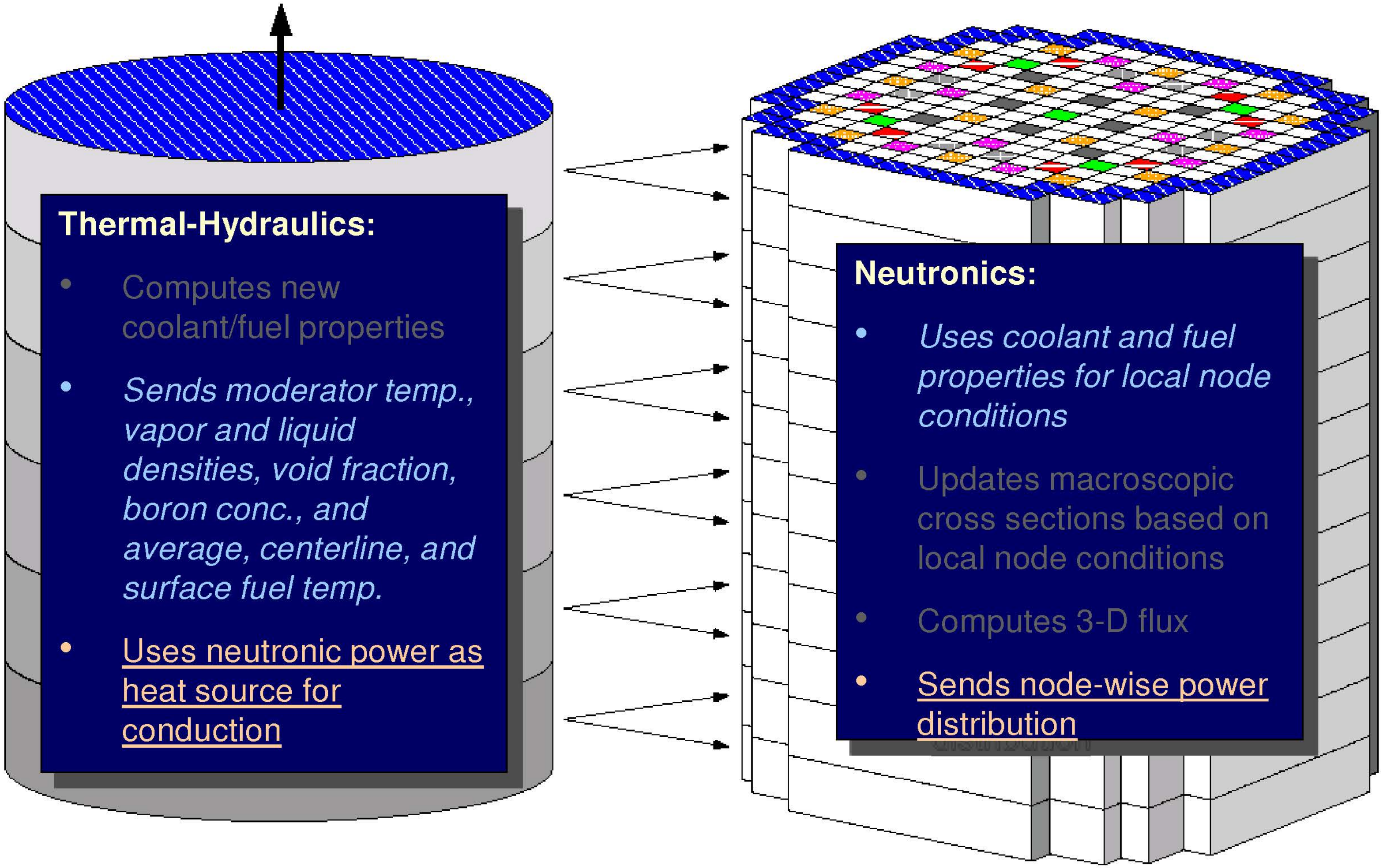
IAEA TECDOC-1539, "Use and Development of Coupled Computer Codes for the Analysis of Accidents at Nuclear Power Plants"

Coupling

- A total interactive coupled solution:



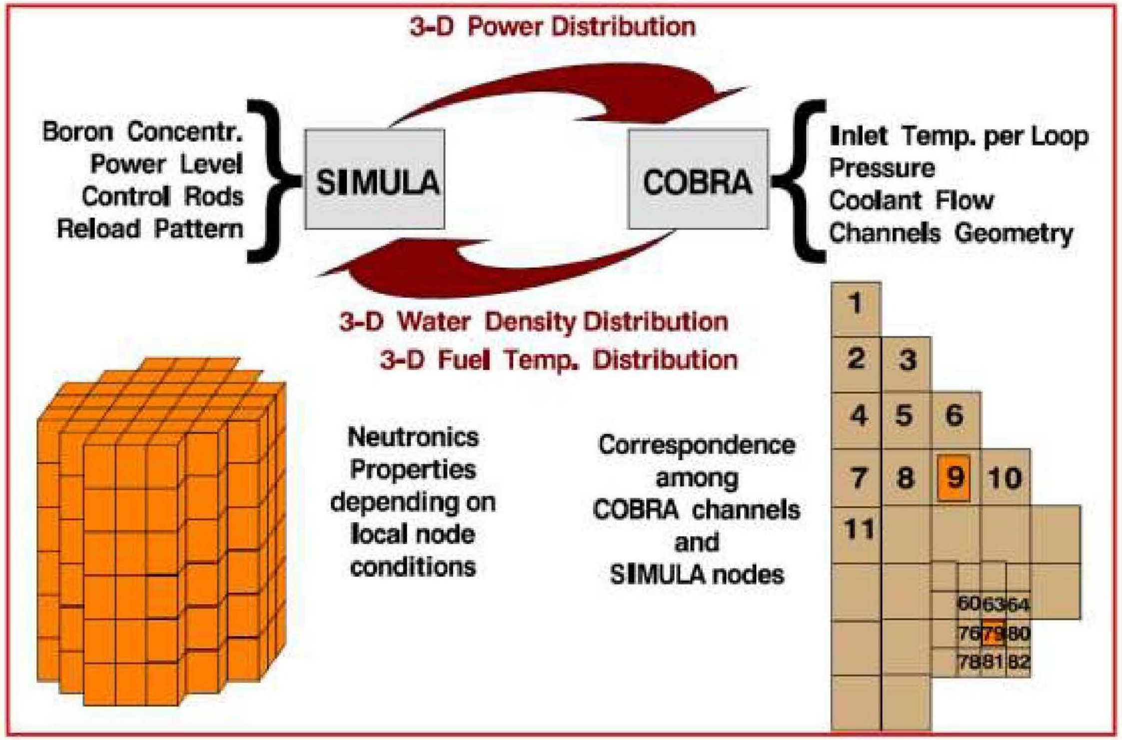
PARCS Coupling Example



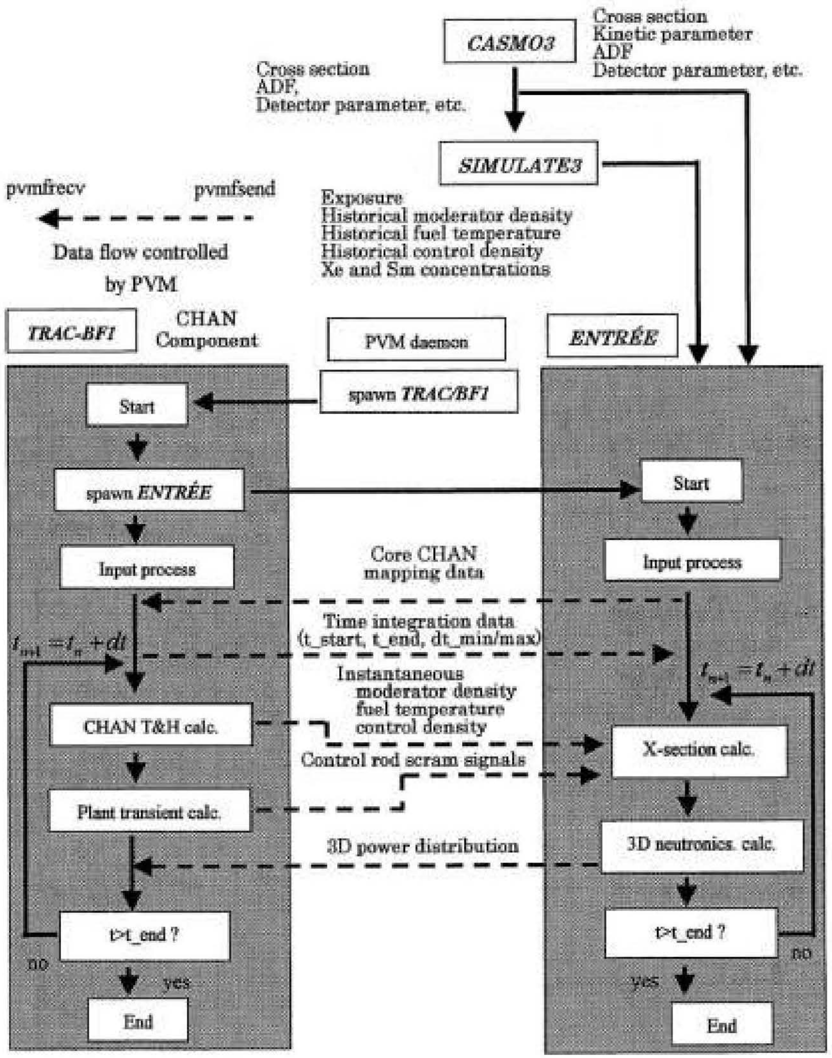
T. Kozlowski and T. Donner, Presentation to USNRC, 2006.

SIMTRAN Example

- SIMTRAN example



More Detailed Example

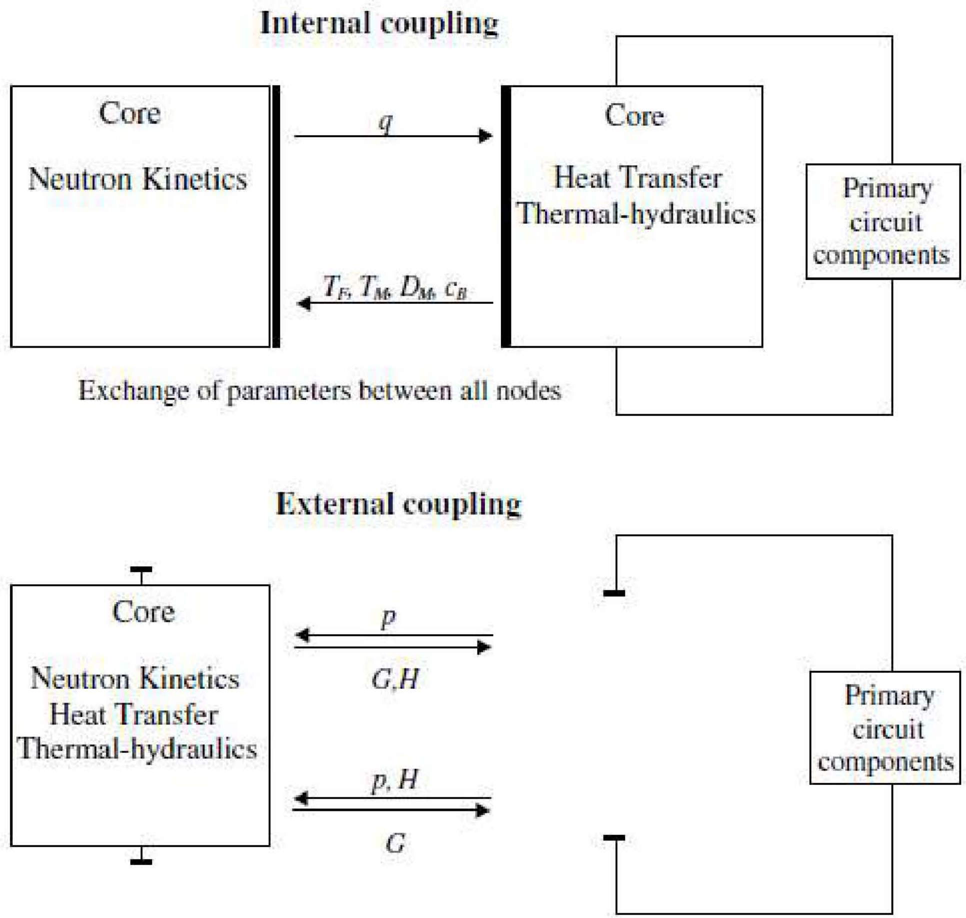


Hotta, J of Nuc Sci and Technology, 2000

Important Aspects to Coupling

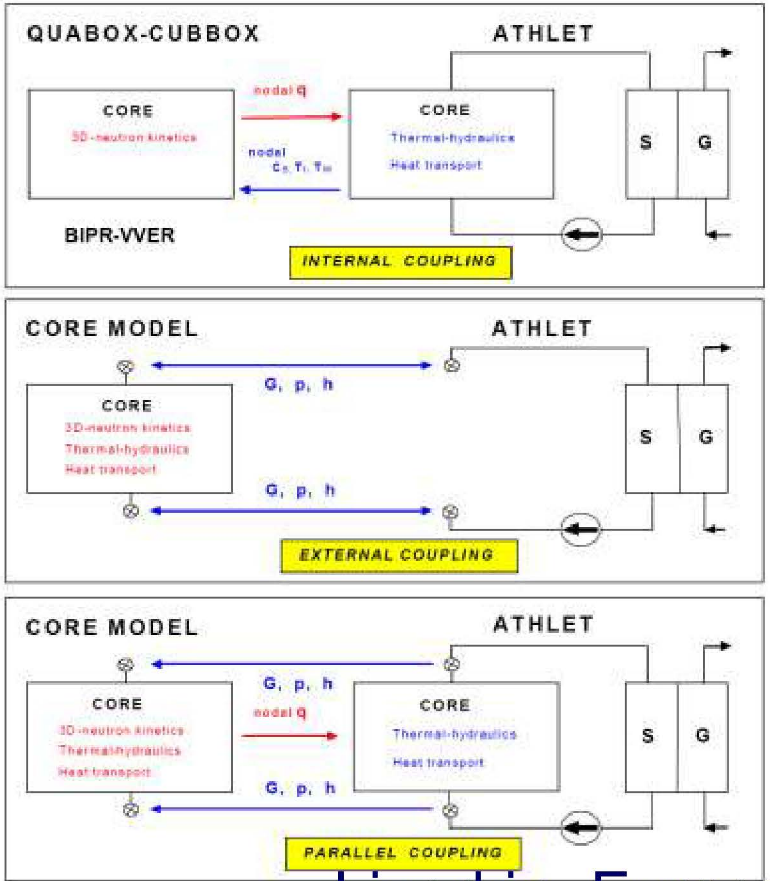
- Overall coupling scheme
- Variable transfer method
- Initial conditions, boundary conditions, time step control and convergence
- Mapping and Grouping (Spatial Mapping)
- Time interval for transfers (Temporal Mapping)
- Validation, qualification and quality assurance

Example Coupling Schemes



Ivanov, Annal of Nuc Energy, 2007

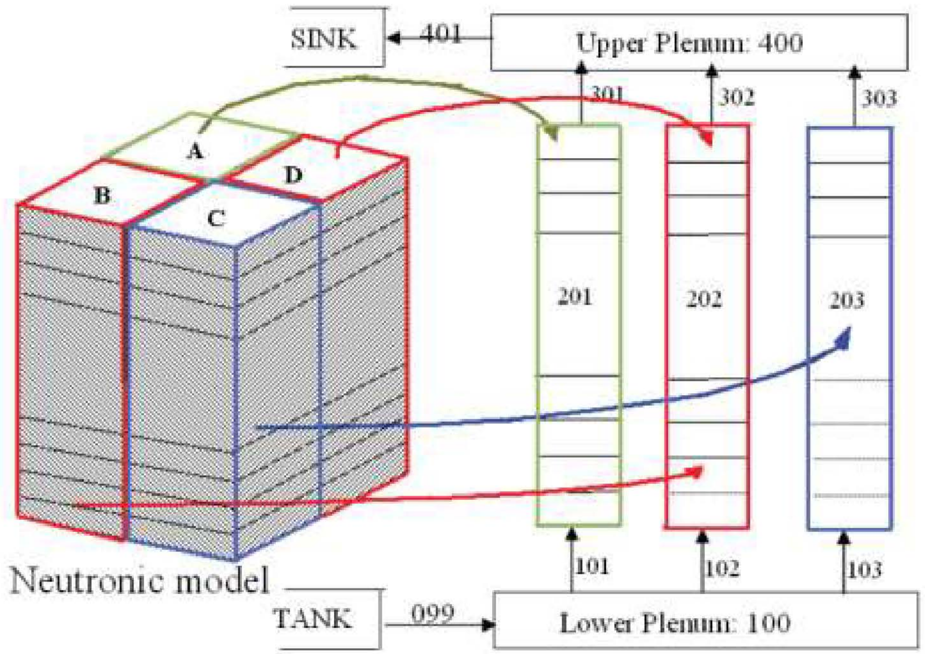
Example Coupling Schemes



Lizorkin, Eurosafe Forum, 2007

Mapping

- Mapping is the term used to specify how a variable is transferred across the multi-physics simulation
  - While NK calculates power distribution at each node, often these nodes do not necessarily match the TH nodal scheme, or may have to be processed before passing.
- Mapping is also a term used to “group” similar fuel assemblies into a single simulation channel → speed up TH simulations.
  - Reduces the complexity of the TH model



JOANNA PELTONEN, Thesis, KTH 2009.

Spatial Coupling: General Idea of Spatial Mapping

- Mapping determines
  - where to deposit NK node power in the T/H code
    - Part of NK power goes to TH cell for direct moderator heating
    - Part of NK power goes to HS cell for fuel heat source
  - where to deposit TH cell properties in NK code
    - T/H and HS sends temperatures/densities feedback to NK node
- Weight specifies the fraction of the total neutronic power generated in a particular node that is deposited into its associated TH and HS cell
- Weights are usually geometric volume fractions determined by what fraction of neutronic node lies in the volume of corresponding TH and HS cell

Spatial Coupling: Important notes on mapping

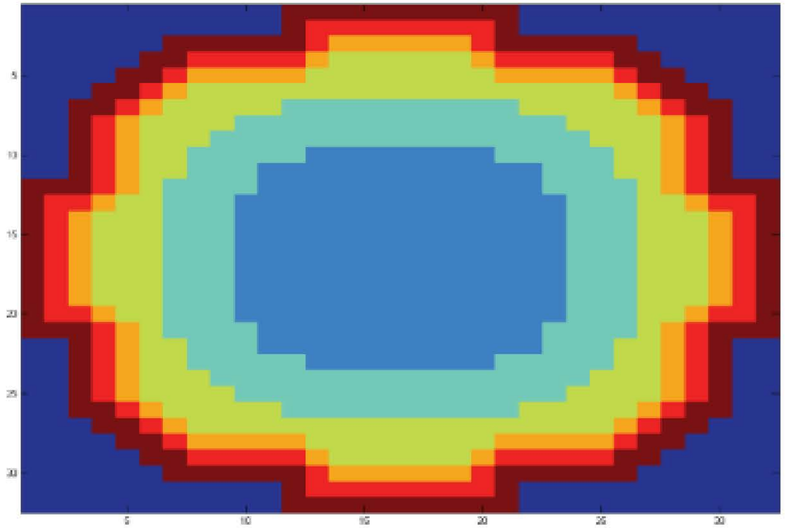
- Sum of weights for each neutronic node MUST sum up to 1.0
- All neutronic nodes have to be mapped somewhere in the T/H code
- Mapping non-conforming meshes is possible, but makes mapping difficult,
  - very coarse T/H mesh to very fine neutronic mesh
- 1-1 mapping is ideal but not practicable

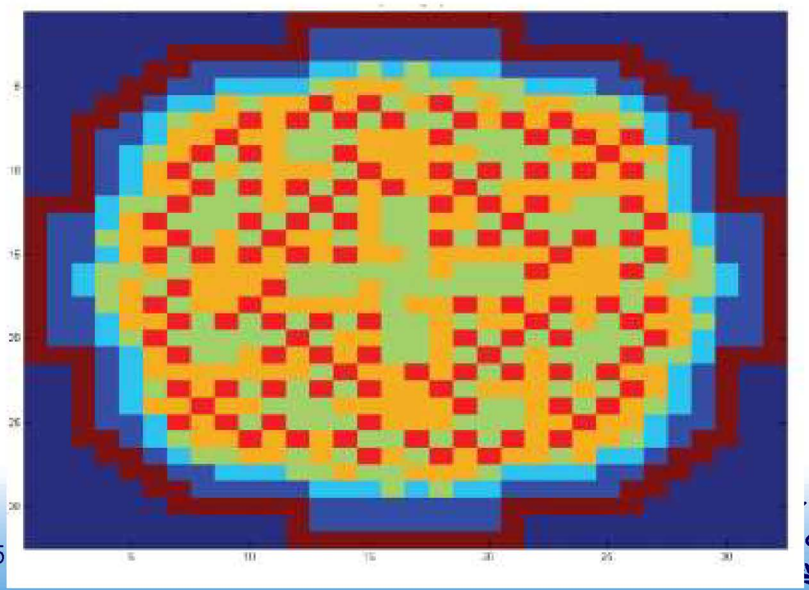
Spatial Coupling: Reflector mapping

- Radial reflector:
  - additional heat structure with no power mapped to the TH (bypass) channel should be used
- Axial reflector
  - Heat structures should have additional axial evevation, but the upper and lower ones will have no power (representing axial reflector for mapping)
  - TH-channels should also have additional axial elevations but only from 2 to N-1 represents the active core

Possible Mapping/Grouping Philosophy

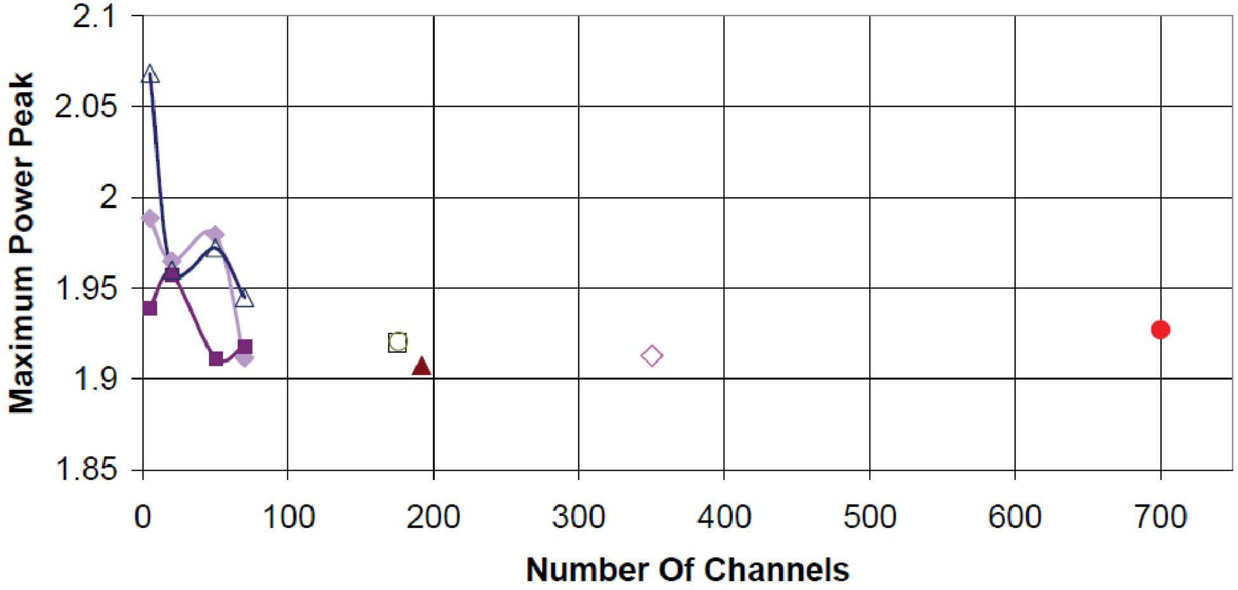
- Geometry
  - the fuel bundles are grouped together according their geometry e.g. inlet orifices.
- Thermal-Hydraulic
  - the cluster of channels represents similar thermal-hydraulic characteristics e.g. void fraction or flow
- Exposure
  - channel grouping according to burnup.
- Power
  - grouping according peaking factors - neutron flux fundamental mode (need of steady-state calculations).
- Higher flux mode
  - grouping according to first azimuthal and higher harmonic modes (need of steady-state eigenvalues calculations).





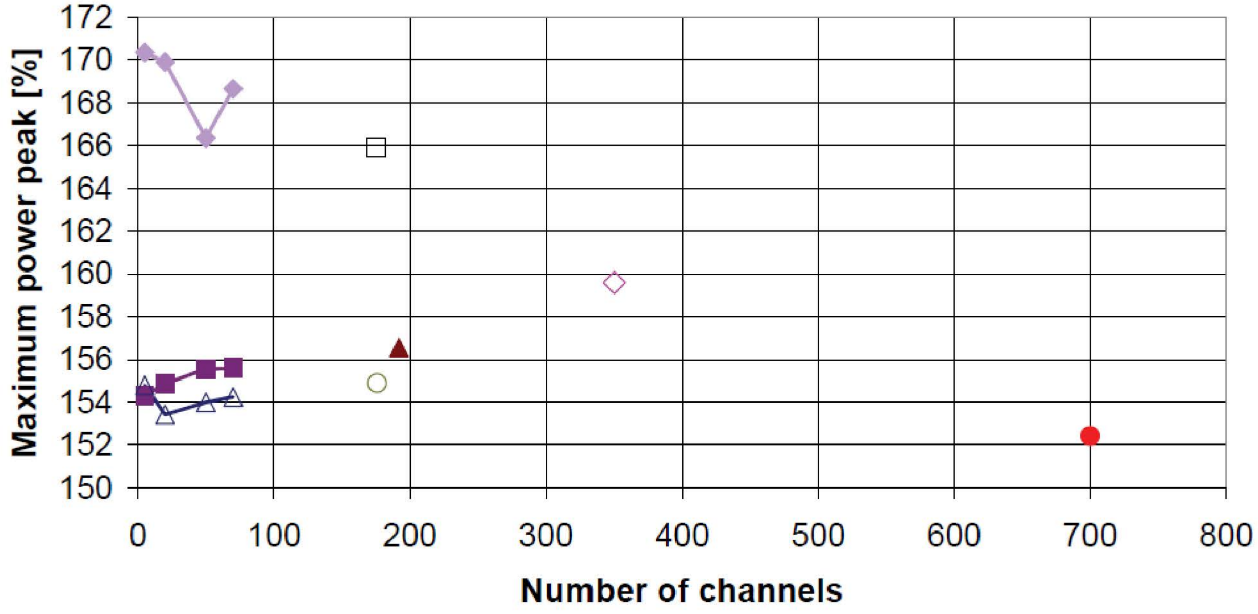
Importance → physically based grouping may reduce the uncertainty in the results.

BWR CR DROP – Effect of Grouping



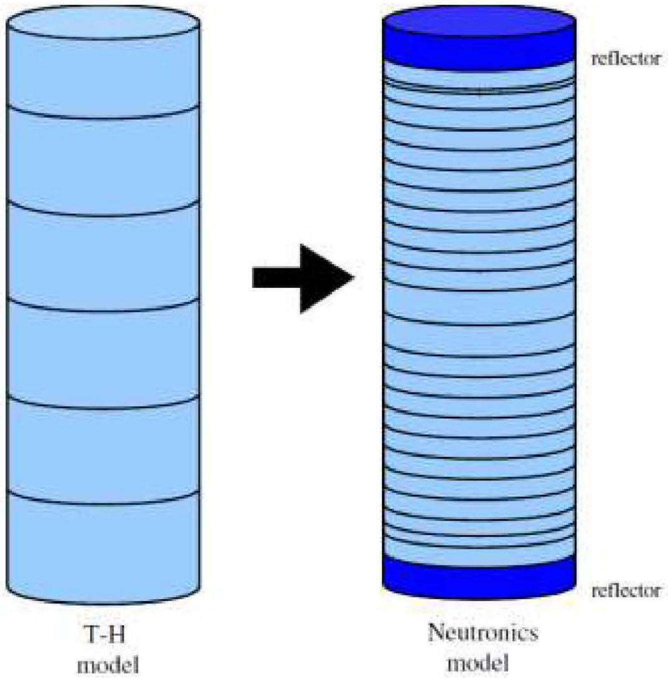
JOANNA PELTONEN, Thesis, KTH 2009.

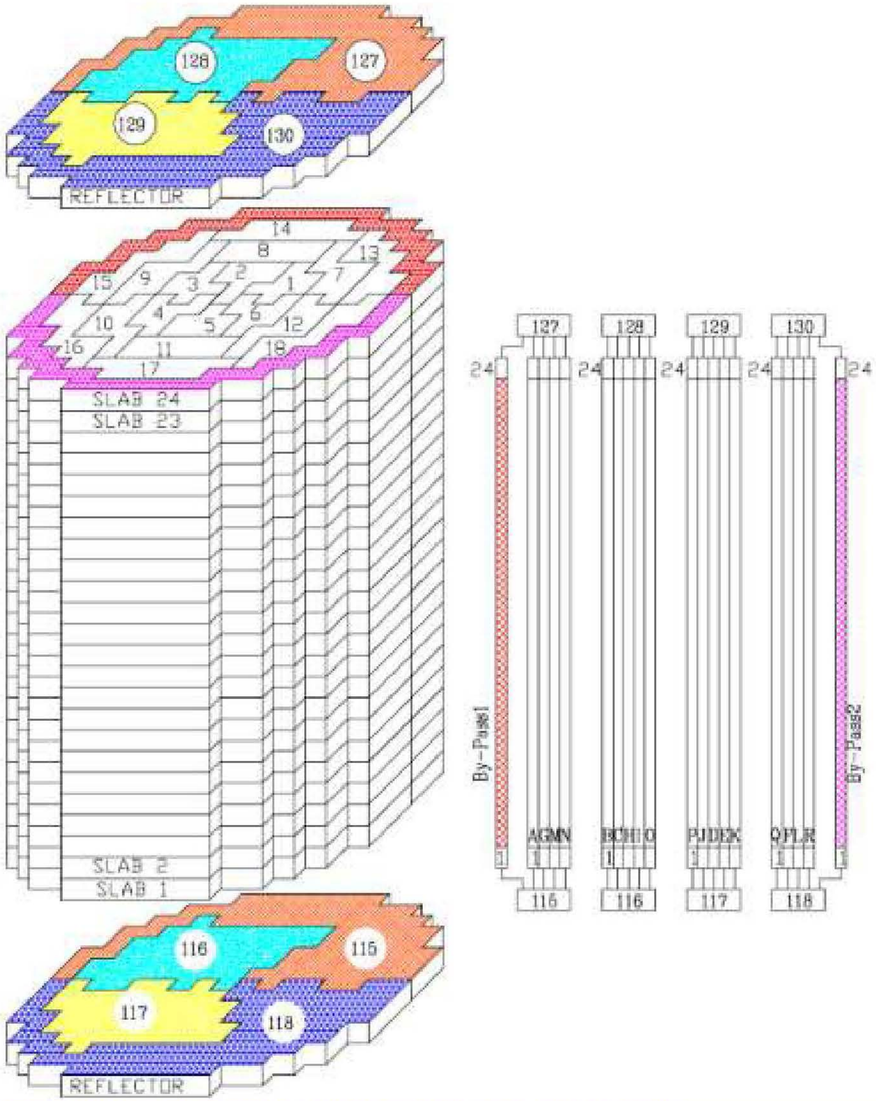
BWR Turbine Trip – Effect of Grouping



JOANNA PELTONEN, Thesis, KTH 2009.

Example of Direct Mapping

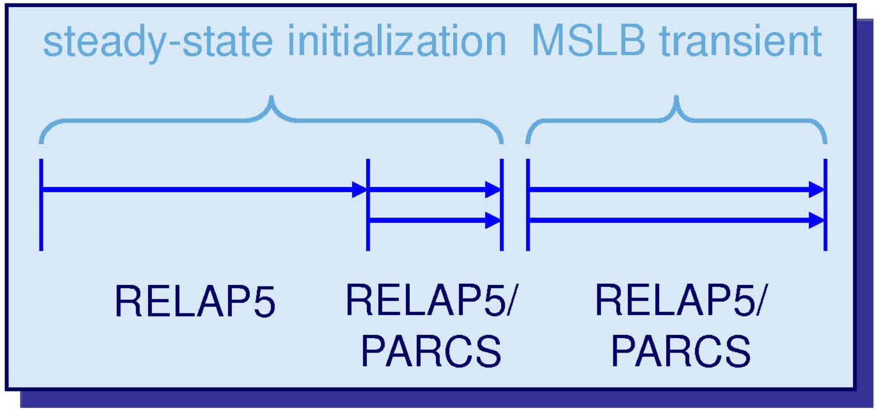




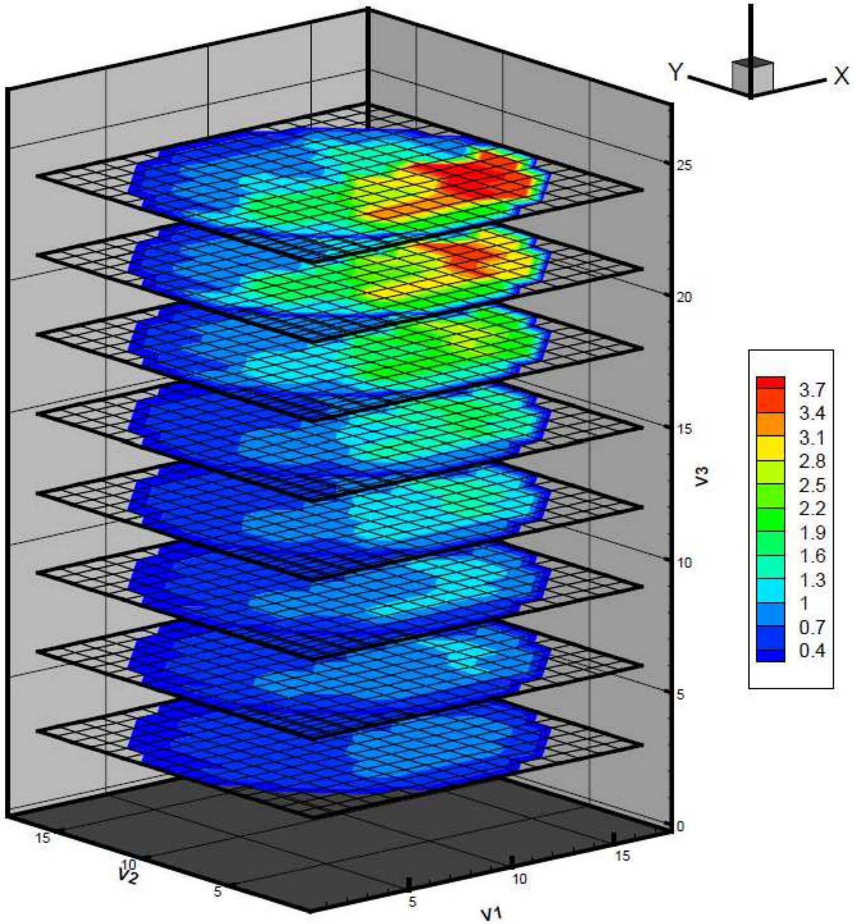
Bovalini et al. INL Report, **TMI-1 MSLB COUPLED 3-D NEUTRONICS/THERMALHYDRAULICS ANALYSIS:**

OECD/NEA MSLB Benchmark: Example Execution Sequence

- Flow Development
  - RELAP5 *transnt* stand-alone
- Steady-State Eigenvalue Calculation
  - RELAP5 restarted in *stdy-st* mode
  - PARCS performs eigenvalue calculation
- Transient Calculation
  - RELAP5 restarted in *transnt* mode
  - PARCS obtains transient flux solution

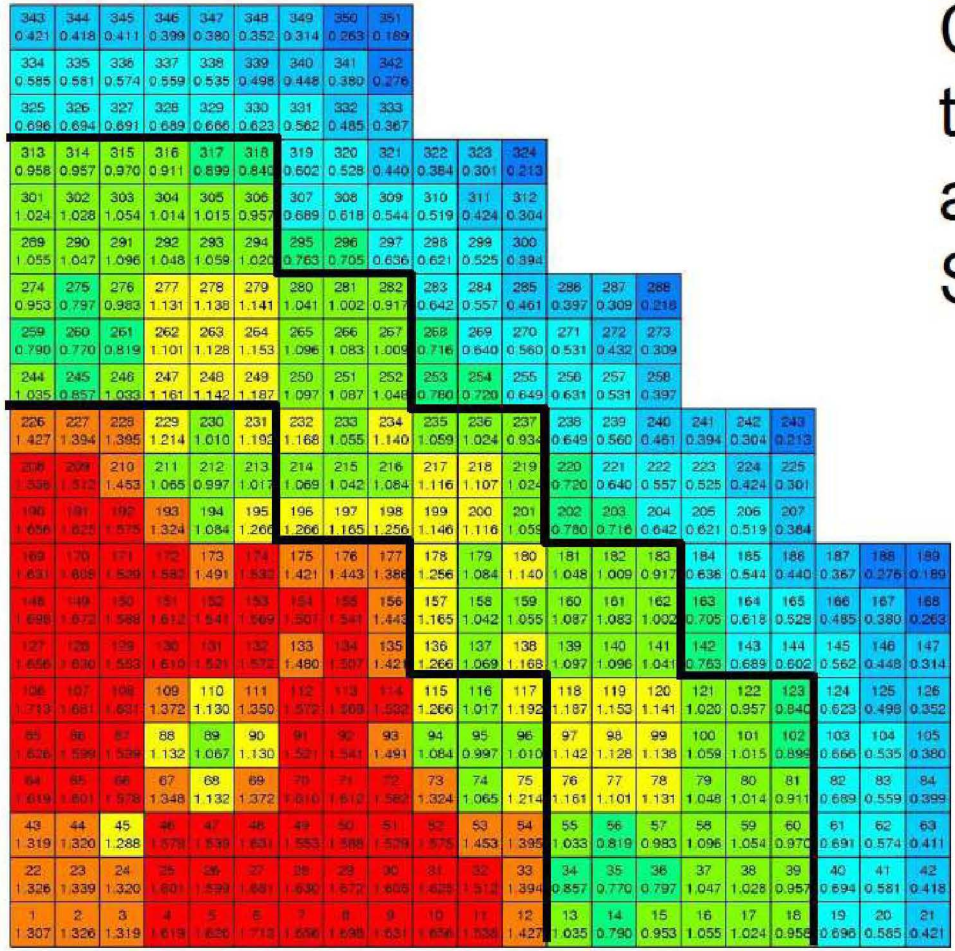


Example Results: TMI



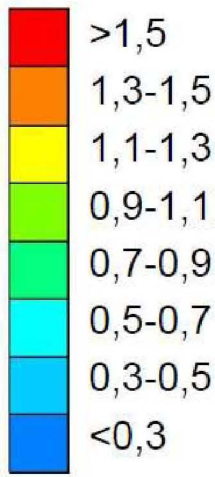
Bovalini et al. INL Report, TMI-1 MSLB COUPLED 3-D NEUTRONICS/THERMALHYDRAULICS ANALYSIS:

LWR Example



Coupled neutronics thermal-hydraulics analysis with KARATE-SPROD

C. Maraczy, AEKI-KFKI, 2008



BWR – SCWR Comparison

- The BWR concepts are similar to SCWR
  - High Moderator/Coolant density changes in the core
  - Coupling between density-reactivity effects
  - Possible no SG – direct cycle
  - Similar safety systems and features.
  - Stability
- Moving forward we can study BWR, ABWR, ESBWR to learn coupling impacts and methods for SCWR.

BWR Stability

- The stability of BWR reactor systems has been a concern from the inception of this reactor type, and extensive experimental and theoretical studies have been performed to design a stable fuel and core configuration.
- A wide review of reported instability events can be found in [D'Auria, SOAR BWR Stability Report].
  - "The reactor core and associated coolant, control and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed".

BWR Instabilities

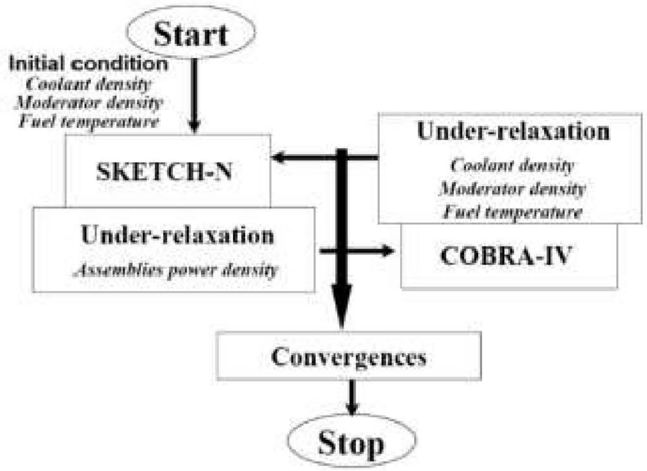
- 2-phase instabilities in BWRs are considered in BWR operations and safety.
  - First, BWR can experience in-phase (core wide) or out-of-phase (regional) oscillations.
  - Second, some NCTH instabilities in reactor cores have resulted in stable, finite amplitude oscillations,
  - others lead to increasing amplitude oscillations until scram.

Need for Coupled Analysis in SCWR

- 3D-kinetic-thermalhydraulics analysis of:
  - Design Assist
  - Normal Operating Conditions (accurate channel powers, margins, flows)
  - Stability Analyses (similar to BWR)
  - AOO (asymmetrical behaviour either within the postulate accident, or as a result of strong feedback effects)
  - DBAs

Steady State SCWR Coupled Analysis

- Steady State coupled solution (XIAO-JING LIU and XU CHENG, 2009)



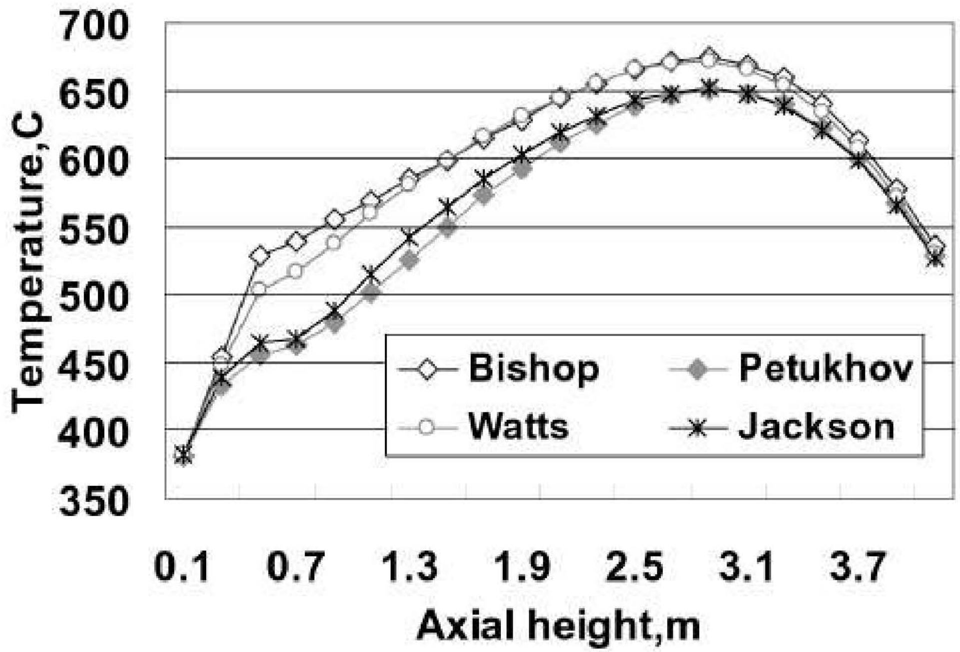
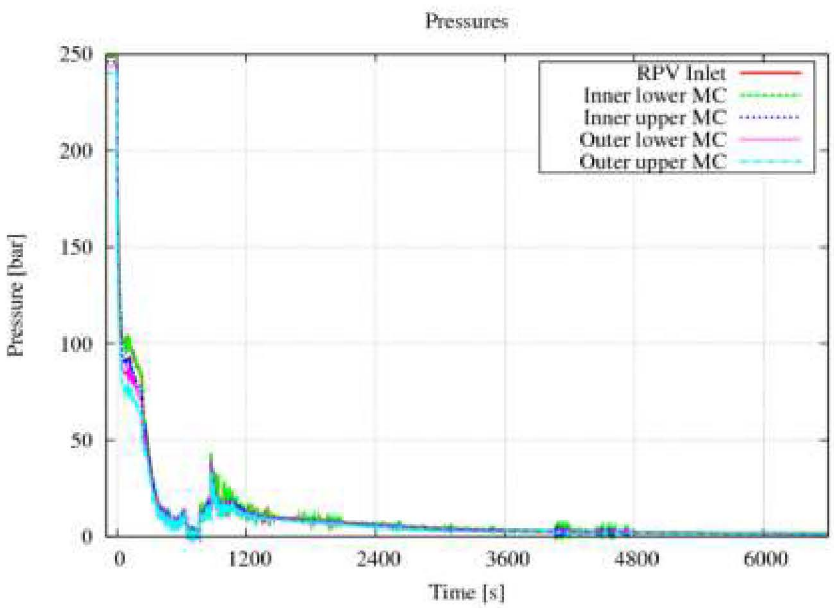
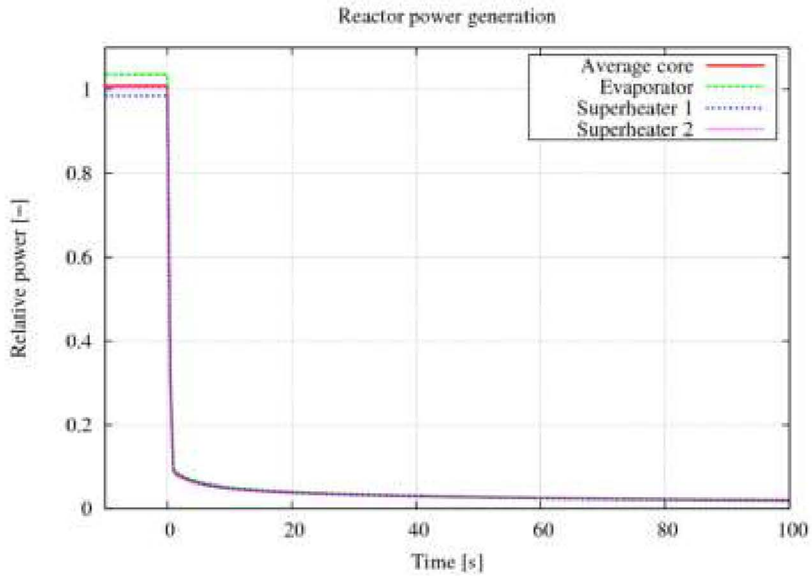


Fig. 21. Cladding Temperature Distribution of Fuel Rod 45

SCWR Coupled Analysis - LOCA





J. Kurki, M. Hänninen, VTT Publication TM-38683-34

Evolution of Coupling

- In traditional reactor core analysis for both steady-state and transient calculations of LWR conventional nuclear power plants, condensed few-group two-dimensional (2-D) cross-section sets are used as input data.
  - These cross-section sets are generated by separate database calculations using characteristic weighting spectra and are parameterised in terms of burn-up and thermal-hydraulic feedback parameters.
  - Under the real reactor conditions, especially in transient situations, these spectra change and the 2-D cross-section modelling based on a parameterisation model only approximately describes the effects of neutron flux distributions, which change in space, time and energy.
- This so-called 2-D off-line cross-section generation and modelling constitutes a basic input data uncertainty affecting the results of coupled 3-D neutronics/thermal-hydraulic calculations.

Coupling Limitations

- Often the initial conditions (i.e., core burn-up level) over a fuel cycle can lead to different results → necessitates several simulations.
  - Cross-sections are dependent on burn-up, control variables and thermal-hydraulic properties.
    - The burn-up dependence of cross-sections is a three-dimensional vector of exposure, spectral history and burnable poison history (for PWR) or control rod history (for BWR). It is based on isotopic depletion.
  - As fuel is burnt the isotopic content is changed in the fuel and, therefore the cross-section behaviour changes.
    - For example, with the production of Pu isotopes there is a hardening of the cross-section spectrum due to the increase of Pu in the fuel.
    - Other changes occur due to the decay and production of fission products.
  - This means that even if all other thermal-hydraulic properties are constant (*i.e.* steady-state conditions) there is still a change in the cross-section behaviour due to the long-term change in isotopes in the fuel while it is being depleted or (as it is usually called) changes in nodal isotopic.

Subtle Limitations

- A strong history effect can be seen if the depletion calculation is performed at core average thermal-hydraulic conditions.
  - This is called a density history effect.
  - The density of the coolant at the core outlet is smaller than the density of the core inlet (*Tout* > *Tave* > *Tin*), meaning that if a calculation is performed at core average conditions the neutron behaviour will be modelled inaccurately.
  - At the core inlet, the actual cross-sections contain more moderation than the values calculated at average conditions and less moderation at the top of the core.
- Further, if the approximate values are used by the nodal code the calculated power is shifted by the density of the water when thermal-hydraulic feedback is considered in the calculation.
  - The density history has a direct effect on the axial power shape and produces a power shape that is skewed towards the bottom of the core.
  - The cross-sections at the bottom of the core are generated based on under-moderated conditions and on over-moderated conditions at the top of the core.
- The modelling of the cross-sections for the axial temperature distribution in a reactor core is very important to accurately predict the axial power distribution.

Conclusions

- 3D Neutronics coupled to Thermalhydraulic is an important feature in existing LWR, HWR and future SCWR designs.
- Methodologies continue to evolve.
- Good coupling should consider:
  - Method (internal, external, tight, loose…)
  - Mapping (spatial and temporal) & Grouping
  - Relaxation and convergence
  - No “best-practices” yet → but maybe soon to come.
    - Similar complexity as the CFD world which has developed a best practices guideline.