The Abdus Salam International Centre for Theoretical Physics

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Joint ICTP-IAEA Course on Science and Technology of Supercritical Water Cooled Reactors

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SCWR CORE DESIGN - GENERAL

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SCWR Core Design – General (SC07)

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Joint ICTP-IAEA Course on Science and Technology of SCWRs, Trieste, Italy
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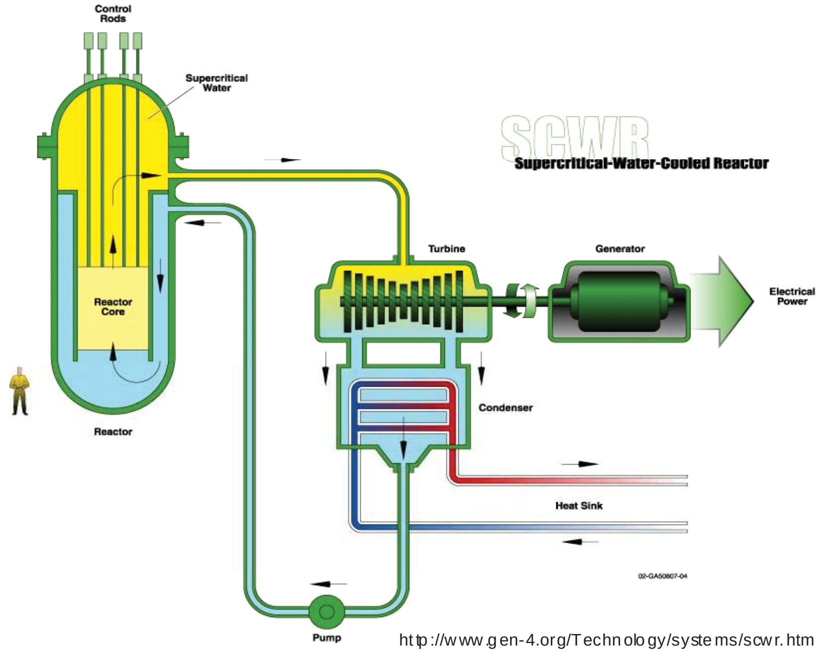
Objectives

- To present an overview of current SCWR core designs
- To highlight core-design criteria for SCWR
- To introduce the importance of coupling between neutronics and thermal-hydraulics in SCWR core design
- To demonstrate applications of analytical tools (subchannel and CFD codes) in support of SCWR core design

Introduction

- Overview of current SCWR core designs
  - Similarities and differences
- Design Goals
  - Generation IV International Forum (GIF)
- Design Requirements
  - International standards and local guides
- Core design criteria
  - Limits for normal operations and abnormal scenarios
- Reactor design process
  - Five categories

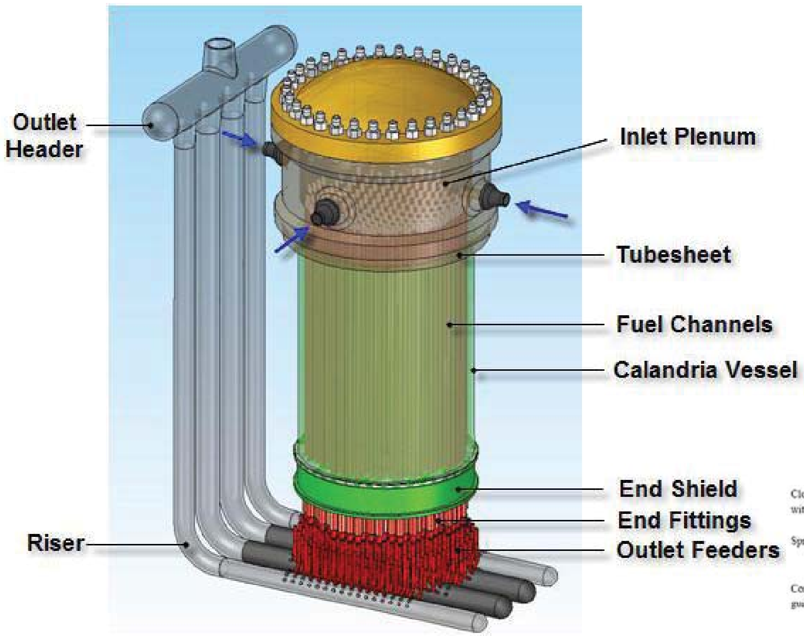


http://www.gen-4.org/Technology/systems/scwr.htm

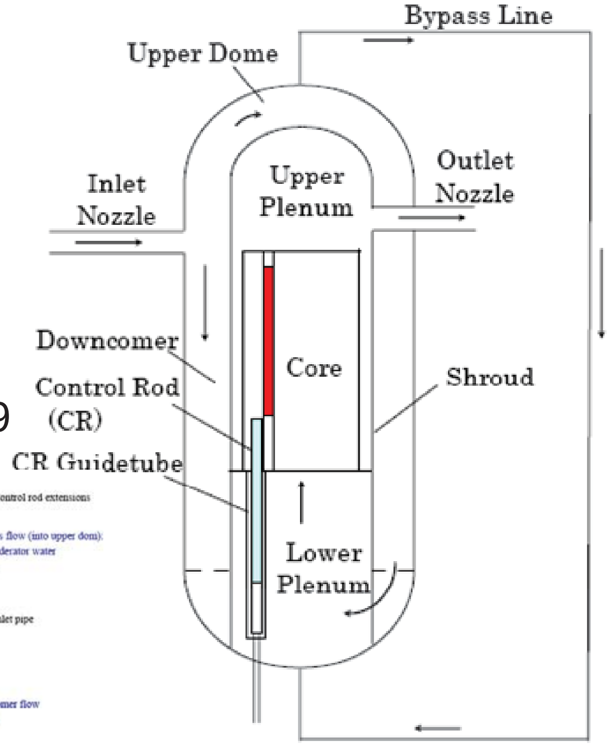
Overview of SCWR Design Status

- Global development of SCWR design concepts was introduced in Lecture S04
  - Korea and United States are no longer pursuing their design concepts
- Canada completed their pre-conceptual thermal-spectrum core design
  - Just started the next phase to complete the conceptual design
- China is working on the conceptual mixed-spectrum core design
  - A thermal-spectrum pressure-tube concept is also being pursued separately
- EU (HPLWR) and Japan (JSCWR) completed their conceptual thermal-spectrum core designs
  - Awaiting for their government approval of the next phase
- Japan is also working on the conceptual fast-spectrum core design (Super Fast Reactor)
- Russia completed their conceptual fast-spectrum core design
  - Unclear about the next phase

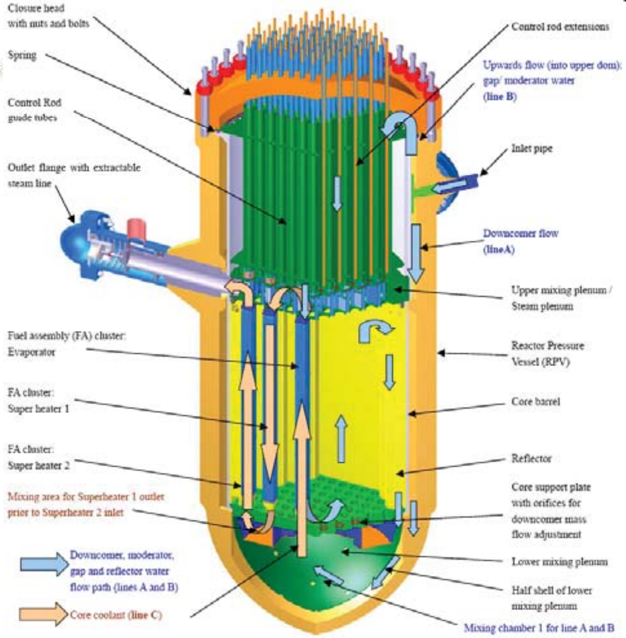
Current SCWR Thermal Core Designs



M. Yetisir et al., 2011



T. Schulenberg et al., 2009

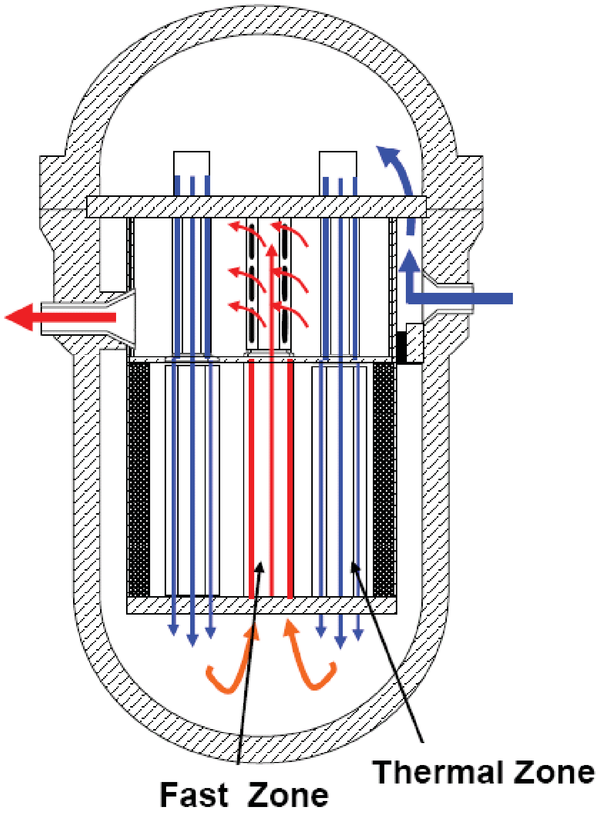


K. Yamada et al., 2011

General Similarities and Differences in Thermal Core Designs

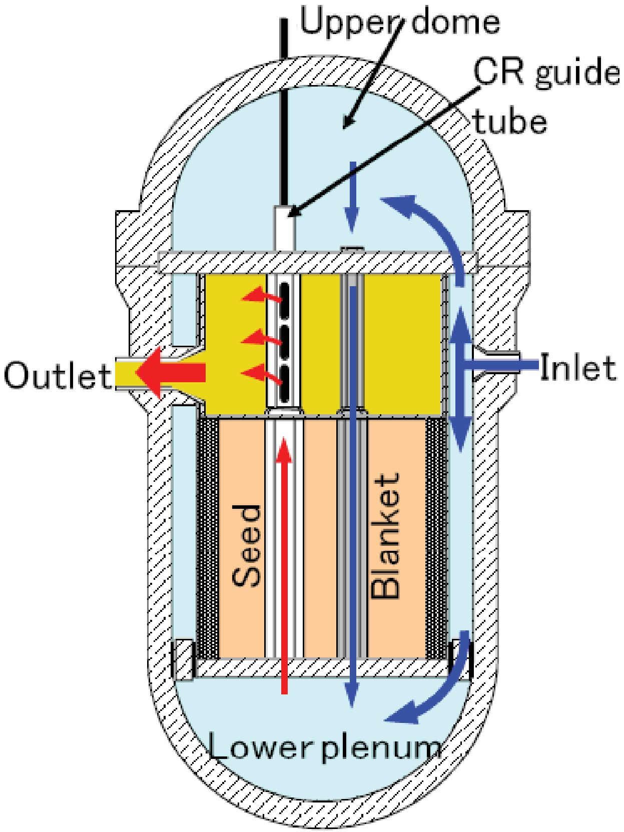
- Similarities
  - Direct cycle
  - Pressure at 25 MPa
  - Light-water coolant
  - Modified stainless-steel cladding
  - Vertical core
  - Batch fuelling (3 cycles)
- Differences
  - Reactor type
    - Pressure tube vs pressure vessel
  - Reactor power
  - Moderator
    - Light water vs heavy water
  - Inlet and outlet coolant temperatures
  - Fuel design
    - Uranium vs Pu-Thorium
  - Flow pass
    - Single vs 2 or 3 passes

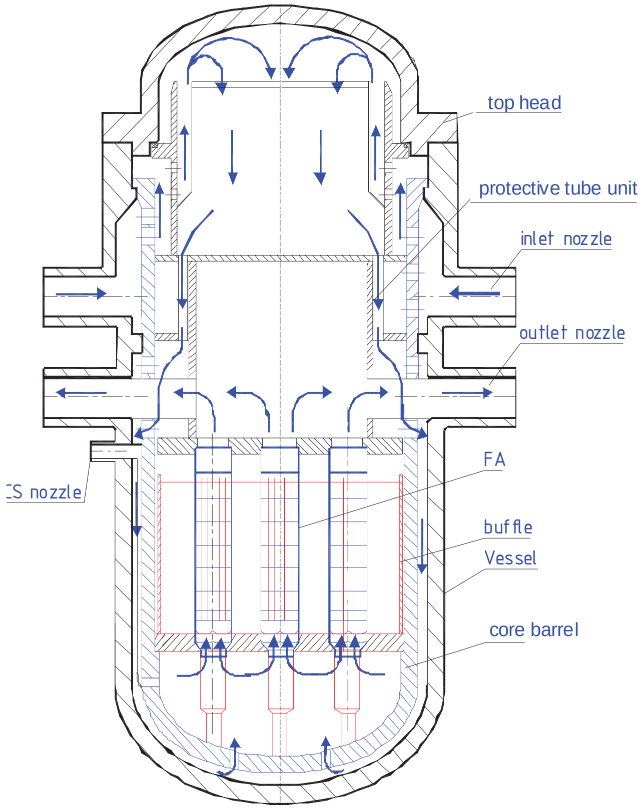
Current SCWR Mixed and Fast Core Designs



X. Cheng et al., 2007

T. Nakatsuka et al., 2010





S. Ryzhov et al., 2011

General Similarities and Differences in Mixed and Fast Core Designs

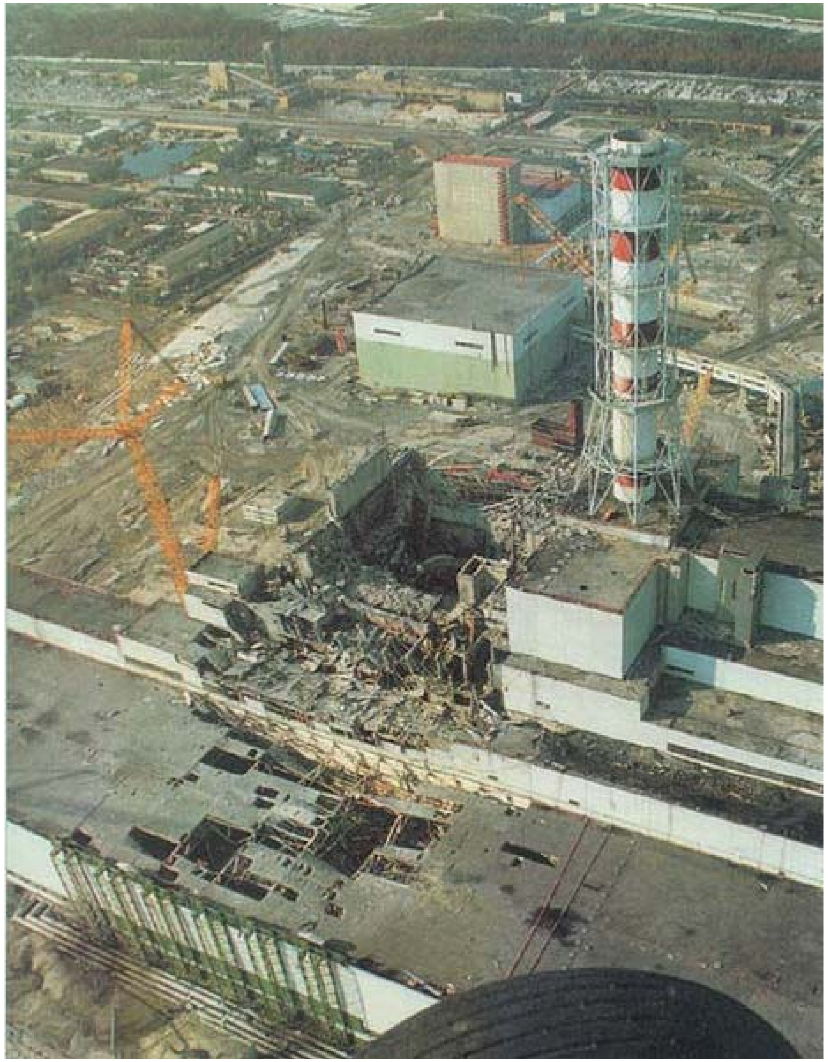
- Similarities
  - Direct cycle
  - Pressure vessel type
  - Pressure at ~25 MPa
  - Light-water coolant
  - Modified stainless-steel cladding
  - Vertical core
  - MOX fuel

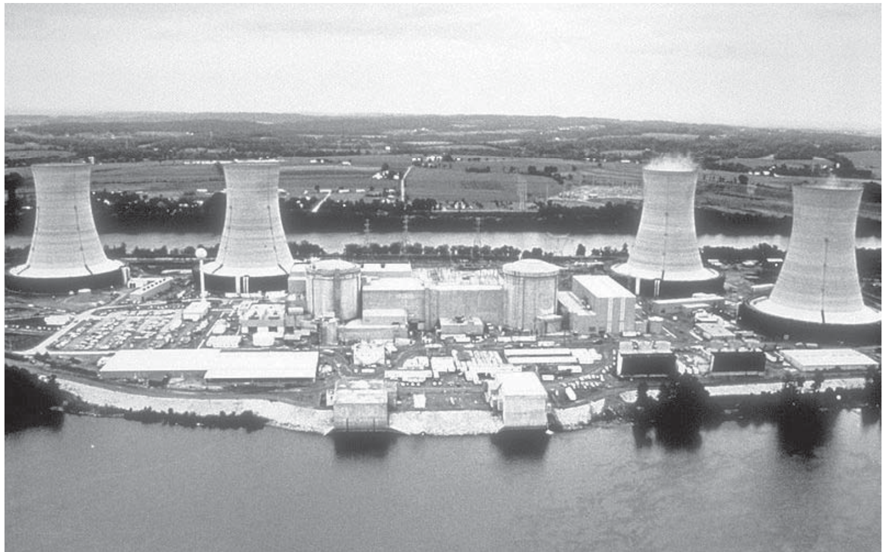
- Differences
  - Reactor power
  - Inlet and outlet coolant temperatures
  - Fuel design (seed and blanket configuration)
    - Layer assembly vs separate assemblies
  - Flow pass
    - Single vs 2 passes

Design Goals

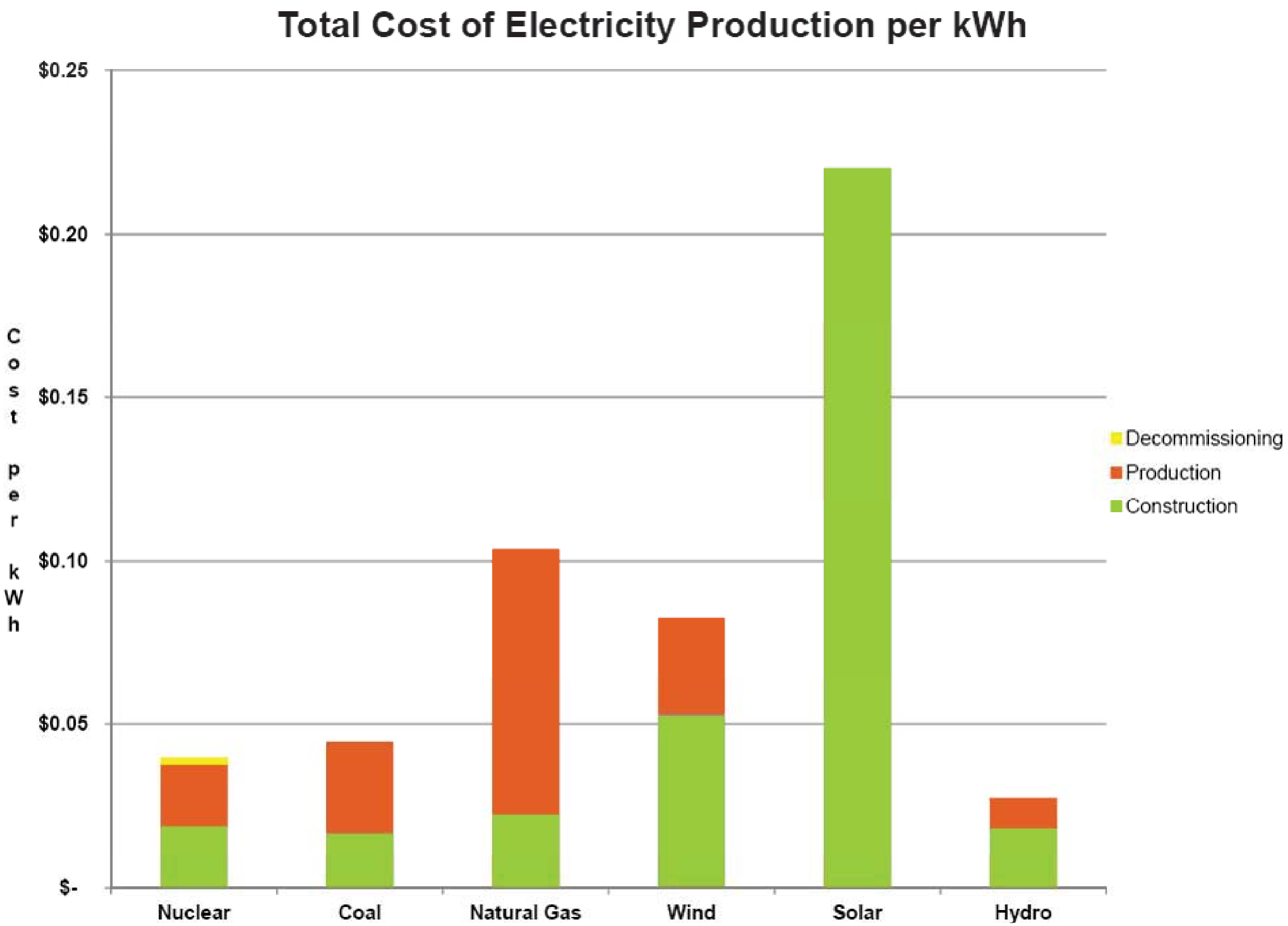
- GIF
  - Enhanced safety (strengthen public acceptance)
  - Improved economic (reduce capital and operating costs)
  - Enhanced sustainability (reduce waste and improve resource utilization)
  - Increased proliferation resistance (strengthen public confidence)
- Competitiveness to other power generation sources and systems
  - Base load compared to peak load
  - Supply stability (e.g., oil price fluctuation)

Enhanced Safety is the Top Priority





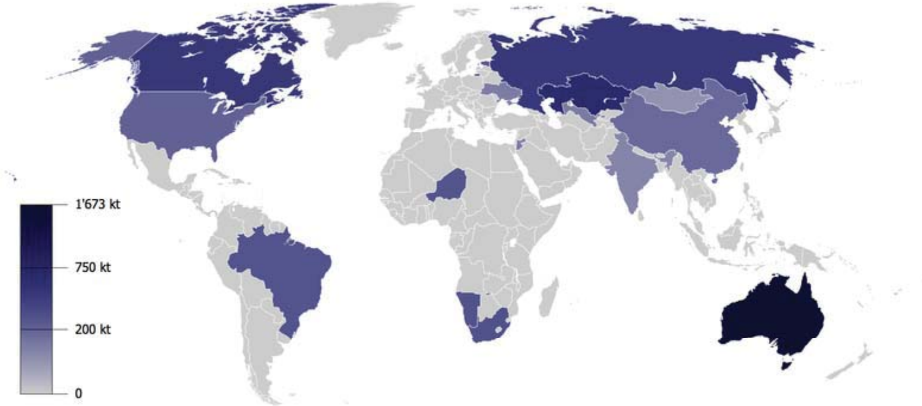
Economic

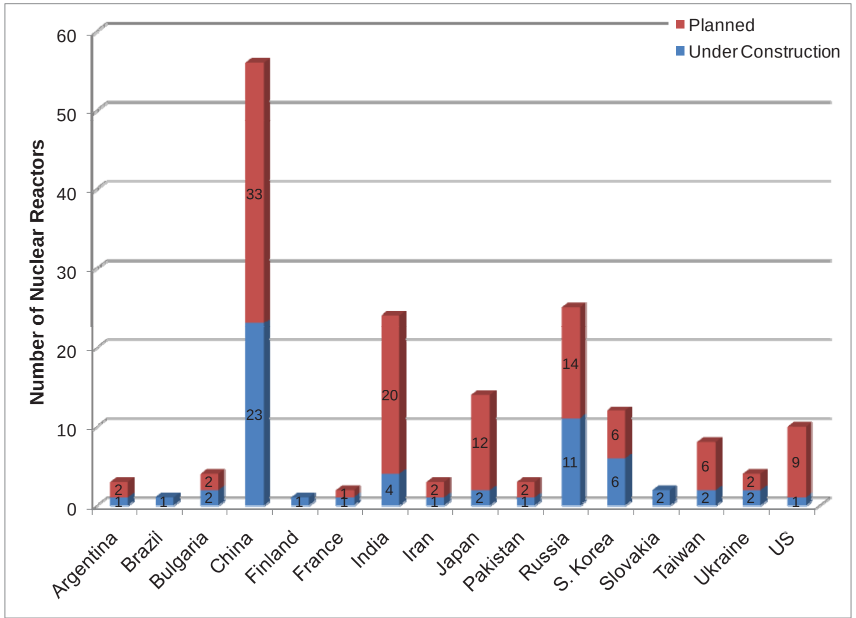

Total Cost of Electricity Production per kWh
$0.25
$0.20
$0.15
$0.10
$0.05
$-
Cost per kWh
Decommissioning
Production
Construction
Nuclear
Coal
Natural Gas
Wind
Solar
Hydro

Sustainability

- Uranium fuel is used in all power reactors
  - Natural
  - Enriched
- Largest uranium reserves (over 100,000 tons)
  - Australia, Brazil, Canada, China, Jordan, Kazakhstan, Namibia, Niger, Russia, South Africa, Ukraine, United States, Uzbekistan
- Rapid growth in nuclear plant construction
  - 61 units under construction
  - 111 units planned/ordered

Global Uranium Reserves





Design Requirements

- Functional Requirements
- Performance Requirements
- Safety Requirements
- Applicable Codes, Standards and Classification
- Environmental Conditions
- Overpressure Protection
- Inspection and Testing
- Reliability and Maintainability
- Layout and Modulization
- Interfacing Systems
- Decontamination and Decommissioning
- Materials and Chemistry
- Loads, Loads Combinations and Service Limits
- Human Factors and other Design Requirements and Constraints

- International Standards
  - IAEA Safety Standards Series (Safety of Nuclear Power Plant: Design) Requirements, NS-R-1
- Local Standards/Guides
  - Applicable to installation at individual country

Five Categories in Design Process

- Nuclear Design
  - Neutron economy consideration
  - Fuel cycle, core configuration, fuel geometry, fuel-to-moderator ratio
- Thermalhydraulics Design
  - Efficient/effective heat removal from fuel and core
  - Fuel rod, fuel assembly, core configuration
- Economics
  - Minimize energy costs and enhance plant reliability
  - Capital costs (materials, fabrication and construction requirements)
  - Net electrical output (minimize in-plane energy consumption)
  - Operating and maintenance costs (system reliability, waste handling, fuel, burnup, fuel cycle, spent-fuel storage)
  - Decommissioning
- Materials Selection
  - Minimize corrosion, chemical reaction, and corrosion product transport
  - Enhanced mechanical properties (e.g., high strength)
  - Chemistry control strategy
  - Structural, cladding, fuel, coolant, moderator, insulator (key to SCWR)
- Safety and Control
  - Facilitate steady-power operation
  - Limit severity and/or mitigate consequences of potential accidents
  - Minimize radiation exposures for operators and general public
  - Monitoring and protective systems, shut-down system, containment
  - Control rods, trips, poisons, fuel, core configuration

Core Design Criteria Introduction

- All five categories in the design process are linked to the core design/configuration
- No SCWR cores in operation
  - No defined standards/guidelines
  - No operating experience
- Evolved from existing design criteria for PWRs
  - Maximum cladding surface temperature at nominal conditions
  - Maximum fuel centerline temperature for nominal fuel rod dimension at 100% Power
  - Minimum departure from nucleate boiling ratio (DNBR) at nominal operating conditions
    - Irrelevant to SCWR since there is no phase change

SCWR Cladding Temperature Criteria

- Minimum Nominal Cladding Temperature Ratio
  - Ratio of maximum calculated design cladding temperature and the defined cladding limit temperature
  - Define the maximum allowable cladding temperature at normal operating conditions
    - Ensure optimal performance of the fuel rod
    - Limit corrosion rates to acceptable values
- Minimum Safety Cladding Temperature Ratio
  - Ratio between the maximum calculated transient cladding temperature and the defined cladding safety limit temperature
  - Define the maximum allowable cladding safety limit temperature at transient conditions in safety analyses
    - Based on 95% probability and 95% confidence for the fuel cladding integrity
    - Higher than normal operation cladding temperature for short periods of time

SCWR Fuel Temperature Criteria

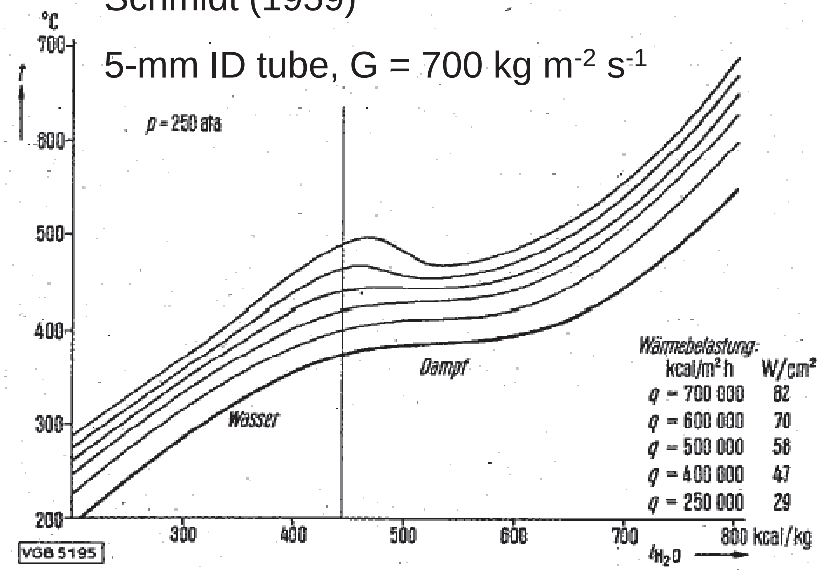
- Maximum Fuel Temperature for Nominal Conditions and Transient Analyses
  - Fuel centreline temperature criteria
    - Prevent fuel melting
    - Fuel and composition dependent
  - More crucial for SCWR
    - High operating temperature

Deteriorated Heat Transfer Phenomena

- DNB-like phenomena with sharp cladding temperature rises at the pseudo-critical point
  - Observed at low flows and high heat fluxes in tubes and annuli
  - Unclear whether it can be encountered in bundles
    - Spacing devices enhance turbulence suppressing this phenomena

Schmidt (1959)

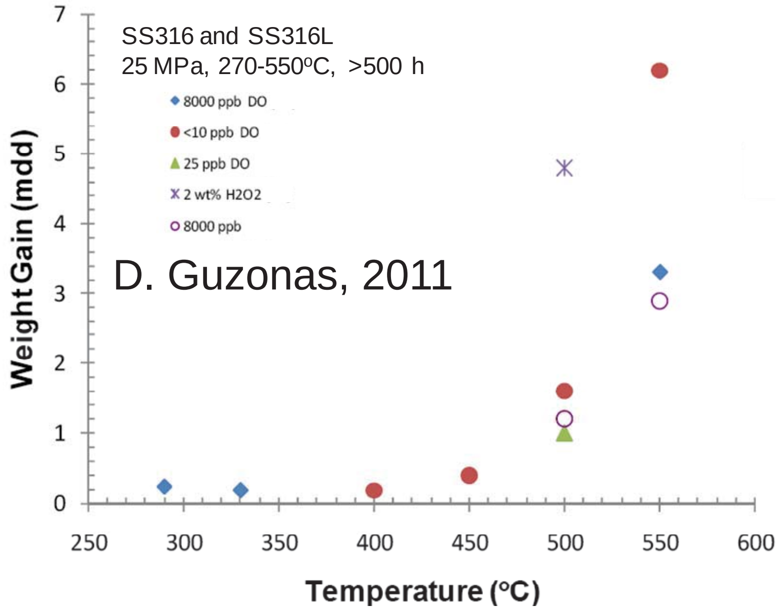
5-mm ID tube, G = 700 kg m^{-2} s^{-1}



- Ongoing debate on its importance
  - Corresponding peak cladding temperatures could be lower than those at high bulk-fluid temperatures
    - The phenomena is ignored in some preliminary safety analyses
  - Concerns:
    - Its occurrence at axial peak power location
    - Non-uniform occurrences around fuel rod leading to bowing
    - Material corrosion

Common SCWR Design Challenges: Materials

- In-core components (internals and fuel cladding):
  - No single alloy with sufficient information to confirm its performance
  - Zirconium-based alloys are not viable material
  - Need thermal and corrosion-resistant barrier

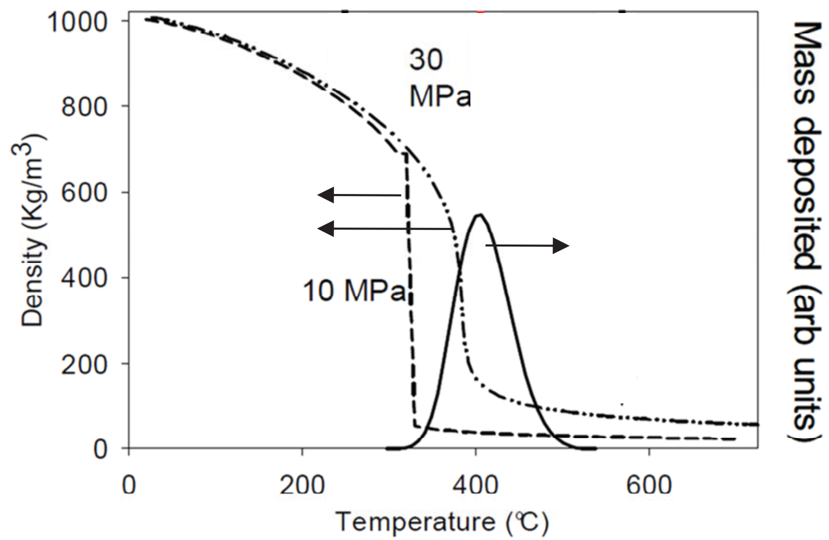


- Out-of-core components
  - Based on material characteristics for fossil-fire power plants
  - Different acceptance requirements on corrosion for nuclear power plants
- Demonstrate performance in key areas:
  - Corrosion and stress corrosion cracking; strength, embrittlement and creep resistance; and dimensional and microstructural stability
  - Quantify irradiation effect

Common SCWR Design Challenges: Chemistry

- Changes in chemical properties due to marked change in SCW density through the critical point
- SCWR In-core radiolysis is markedly different from those of conventional water-cooled reactors
  - Extrapolation of the behavior is inappropriate
  - Strong impact on corrosion and stress corrosion cracking
- Identification of an appropriate water chemistry to minimize
  - Corrosion rates
  - Stress corrosion cracking
  - Deposition of deposits on fuel cladding and turbine blades
- Establish a chemistry-control strategy

D. Guzonas, 2011



Common SCWR Design Challenges: Thermalhydraulics

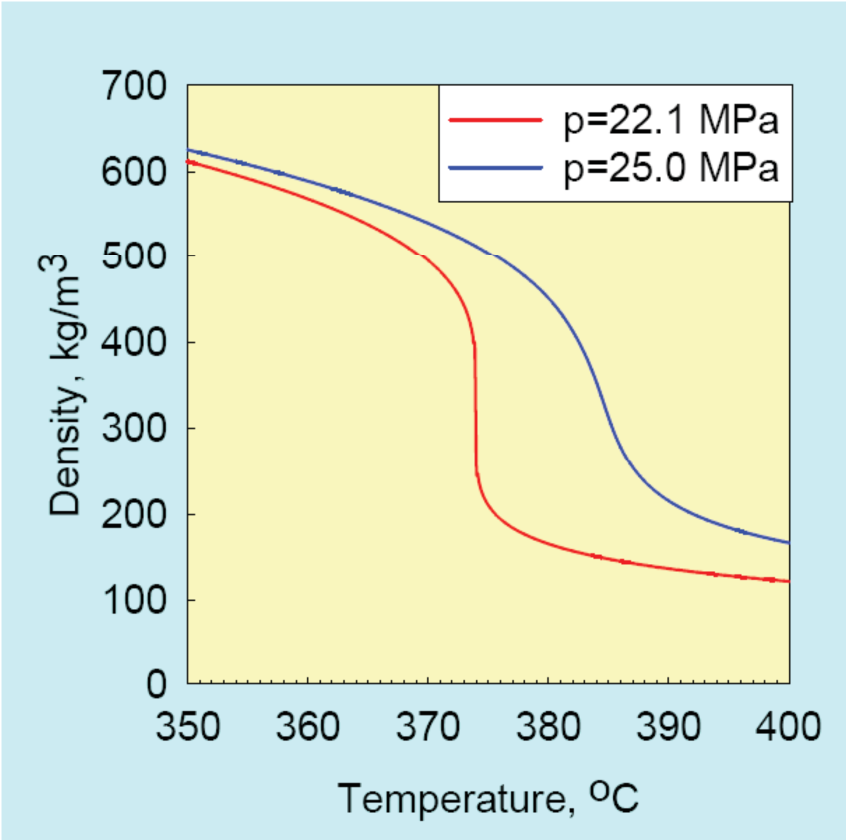
- Cladding temperature limits have been adopted as the design criteria
  - Traditional CHF criteria is no longer appropriate due to the lack of phase change
- Accurate prediction of supercritical water (SCW) heat transfer is essential to establish the power output and safety margin
  - Need experimental data for relevant bundle geometry at conditions of interest
- Most experimental data on SCW heat transfer were obtained with tubes
  - Applicable to the fossil-plant boiler but not directly to SCWR geometry and conditions
- No experimental data on SCW heat transfer in relevant bundle geometry
  - Only a few bundle-subassembly data
  - No information on separate effects

Common SCWR Design Challenges: Safety

- Establishment of design-basis accidents and potential initiators for severe accidents
- Transient experimental data on supercritical heat transfer
  - Pressure transient through the pseudo-critical point
- Experimental SCW data on critical flow
  - Designs of safety/relief valve and depressurization system
  - Support of large break loss-of-coolant accident analyses
- Susceptibility to dynamic oscillations
  - Large variation of coolant density in the axial direction
  - Strong coupling of the neutronic and thermal-hydraulic behavior
  - Need experimental data and analytical model to predict the onset of instability
- Applicability of safety analysis code
  - Validations were performed for subcritical applications only
  - Need integral test data at supercritical conditions

Nuclear and Thermalhydraulics Designs

- Core coolant density variations at normal operation
  - Small in LWRs and HWRs
  - Large in SCWRs
- Large density variation in the core affecting
  - Moderation and absorption
  - Neutron spectrum
  - Fission power distribution
  - In turn, impacting the coolant and moderator density profiles
- Coupled neutronics and thermalhydraulics analysis is required in design calculations



J.Q. Shan et al., 2010

Neutronics and Thermalhydraulics Coupling

- Coupling analyses are performed using two suites of analytical toolsets
  - Neutronics codes (such as MCNP) to establish radial and axial distributions of power
  - Thermalhydraulics codes (subchannel code) to determine fuel temperature and coolant temperature (and in turn the coolant density)
- An iterative approach is applied in the calculations

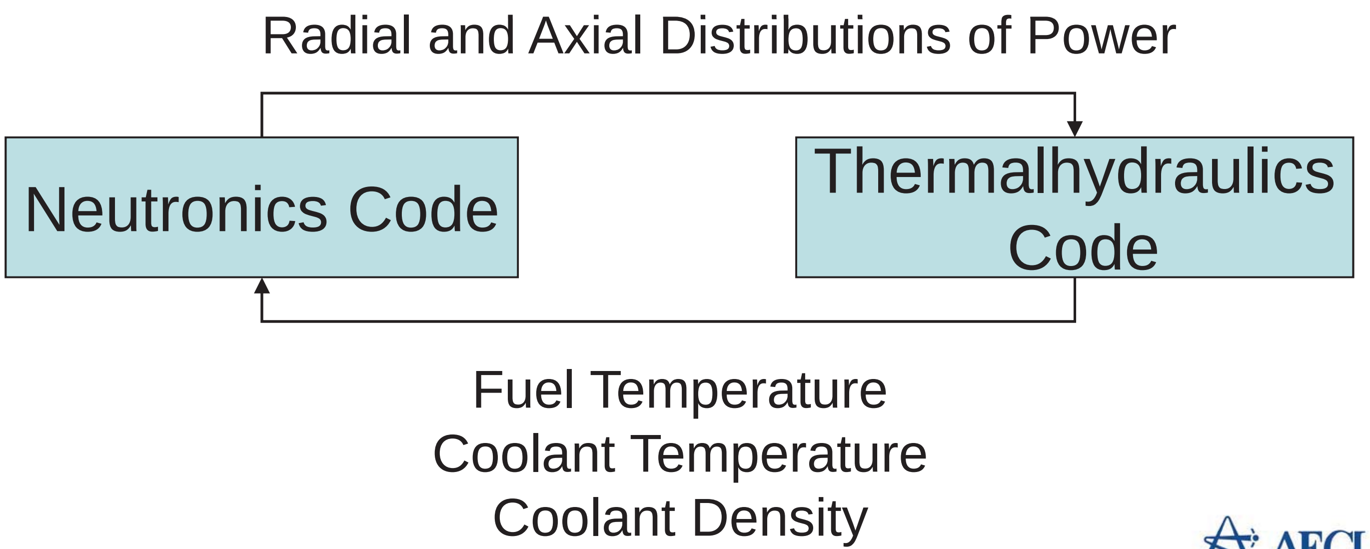
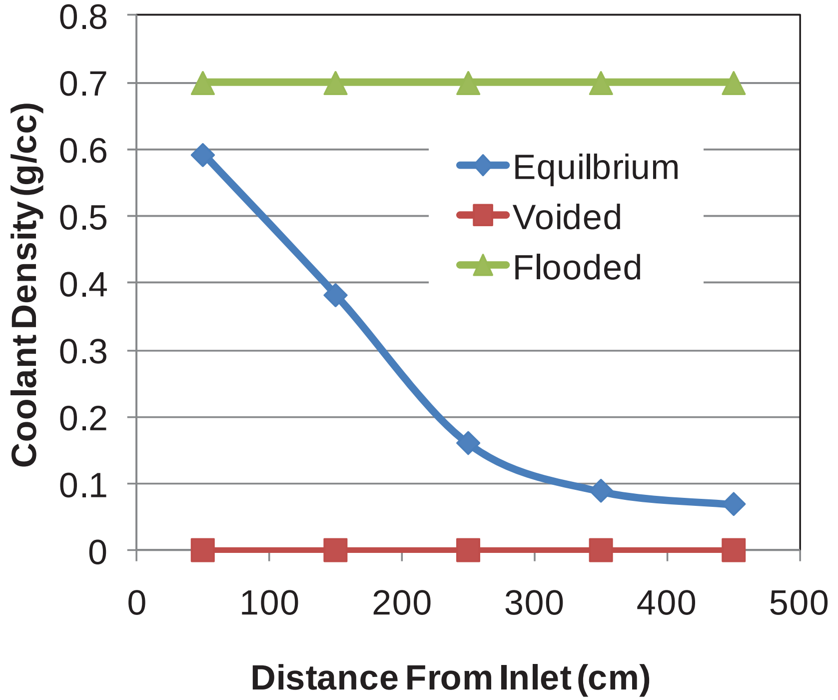


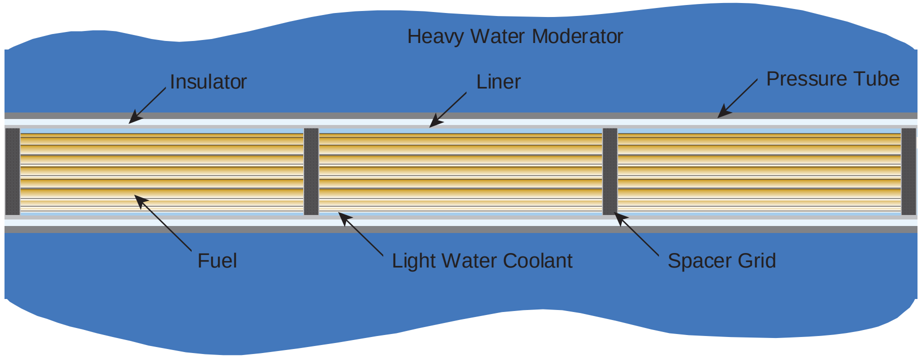
Illustration of Fuel Channel Density Variations

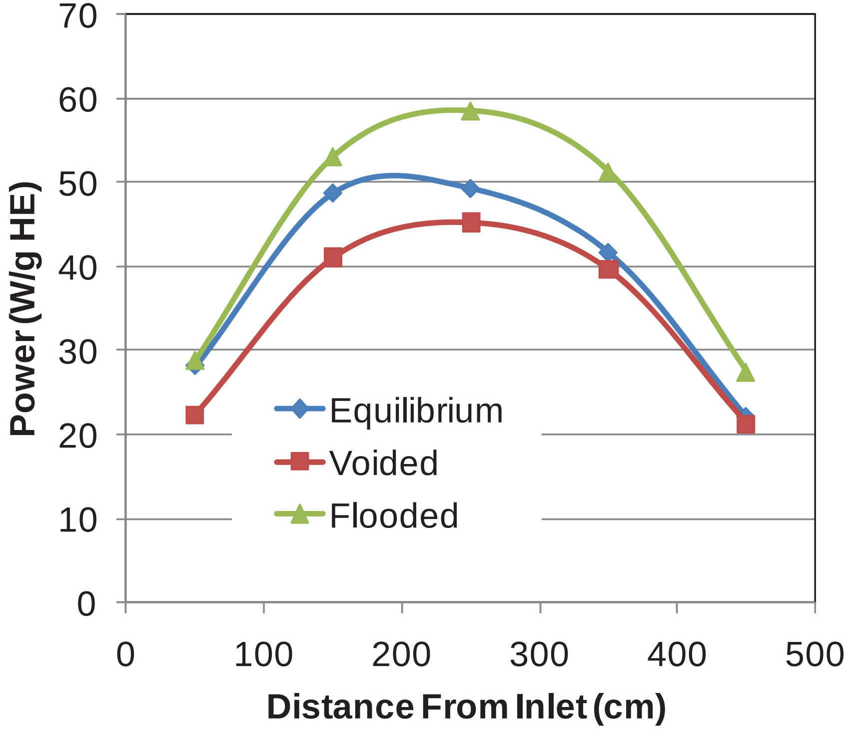
- Established core power profiles at BOC conditions using the RFSP code
  - Coolant flowing uni-directionally inside the fuel channel
  - Large moderator pool
- Coolant properties are defined at 5 set positions along the fuel channel
- Consider Three Cases:
  - Equilibrium Coolant Density (and Temperature) Distribution
  - Uniform Coolant Density, Inlet Conditions
  - Uniform Coolant Density, Outlet Conditions



Impact of Density Variation on Axial Power Profiles in Fuel Bundle

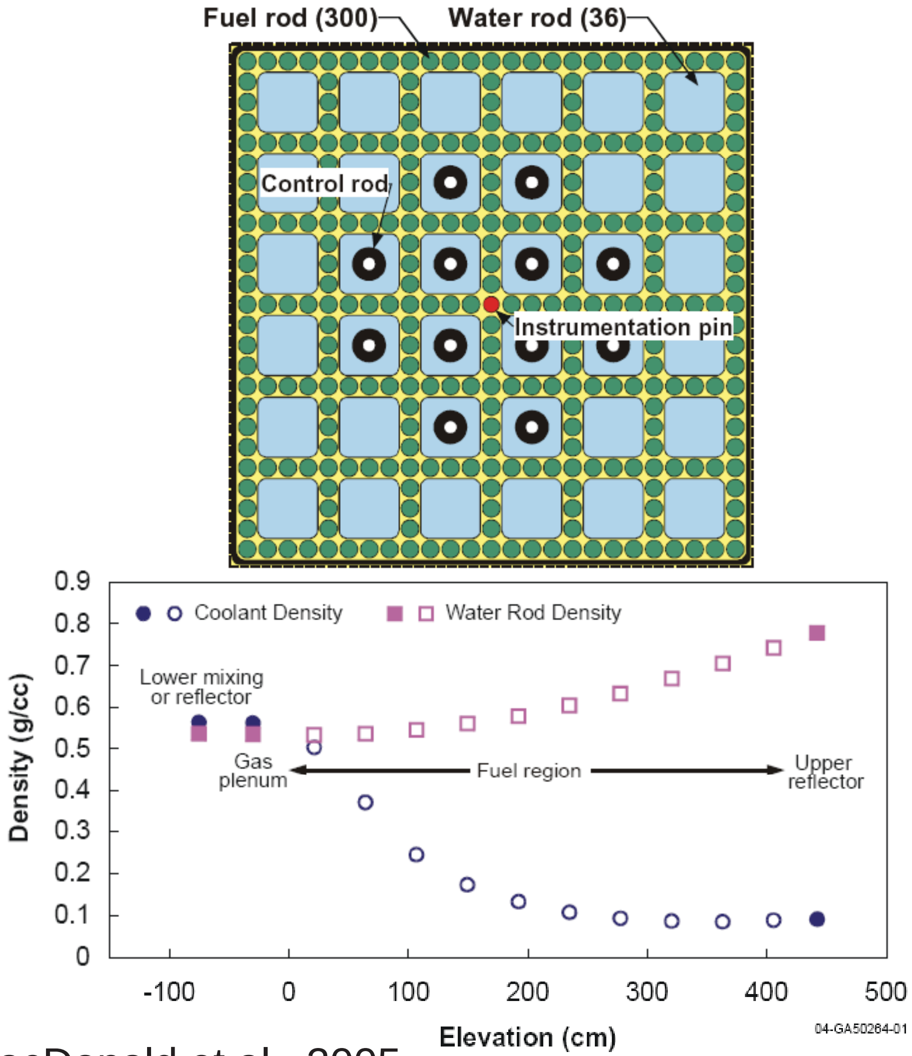
- Channel power changes determined using WIMS with a fixed total neutron flux
- Voided channel
  - Similar to channel outlet conditions
  - Overall power decrease
- Flooded channel
  - Similar to channel inlet conditions
  - Overall power increase
- Equilibrium channel
  - Transition from flooded channel to void channel case
  - Power peak close to the upstream end





Impact of Water Rods on Density Profiles

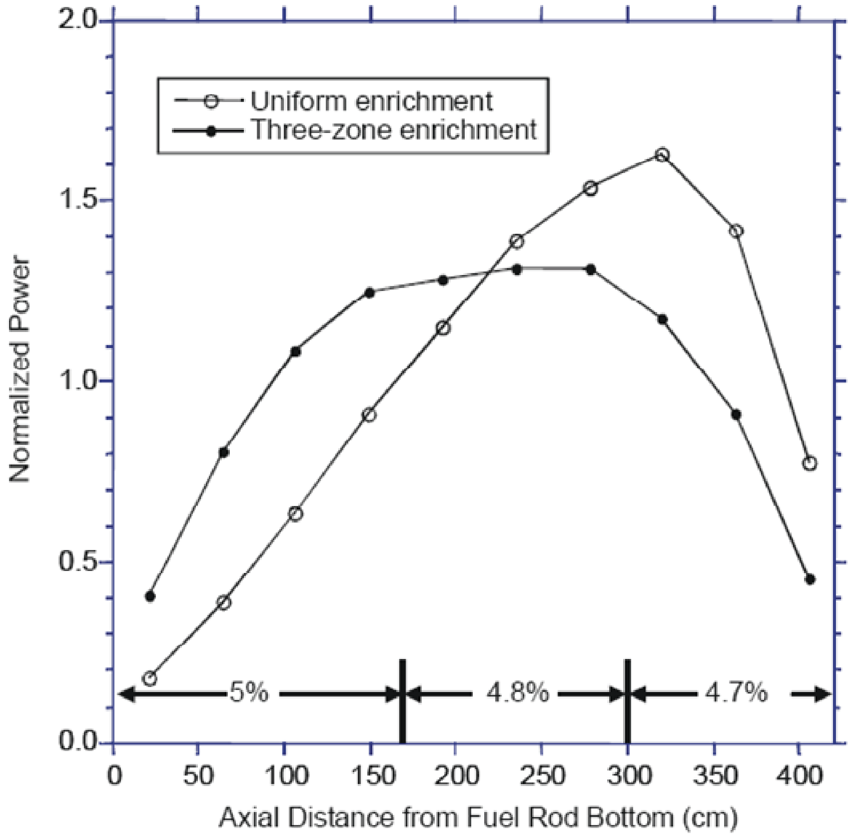
- INEEL performed coupled neutronic and thermal-hydraulics analysis
- SC Water flows down inside water rods (as moderator) and up along the fuel rods (as coolant) of the fuel assembly
- Water temperature increases continuously and water density decreases
  - The decrease is steeper along the fuel rods than the water rods



P. MacDonald et al., 2005

Impact of Density Variation on Axial Power Profiles in Fuel Assembly

- INEEL applied MCNP, ORIGEN2, and MOCUP codes in neutronic calculations
- Reference case with 5% uniform enrichment
  - Power peak at the downstream end of the core
  - Influence of water-rod density variations
- Power peak reduction using three enrichment zones
  - Flatten the axial power profile



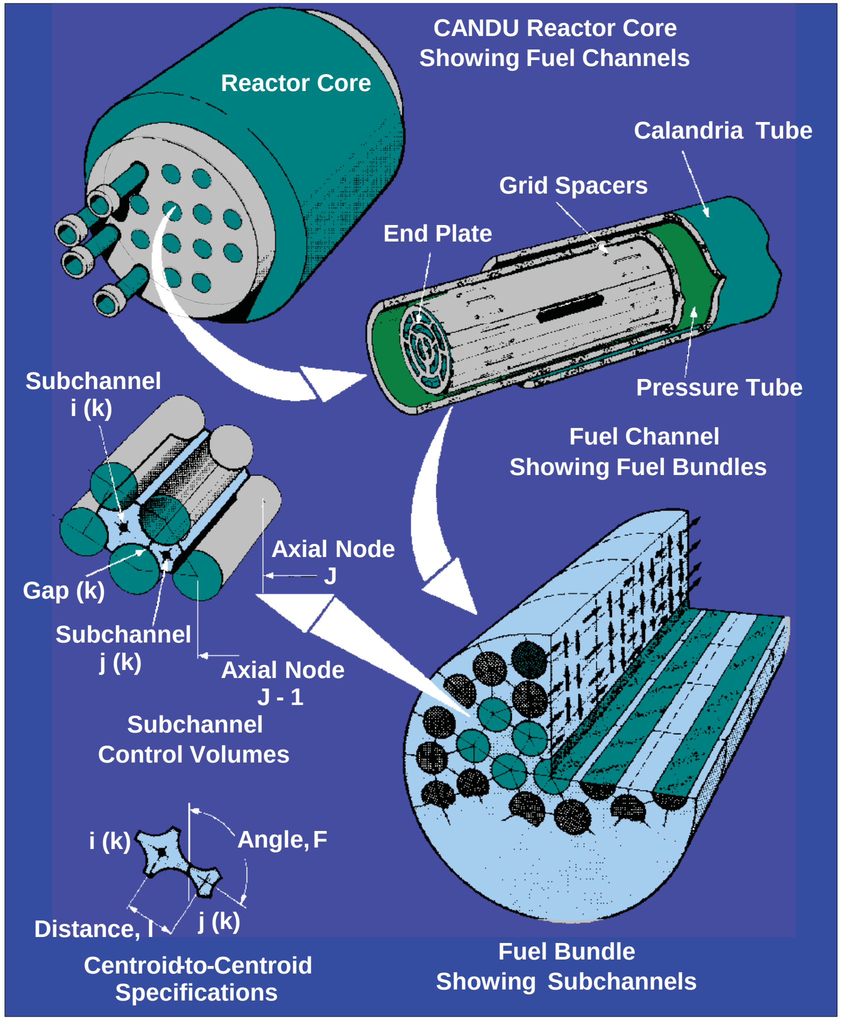
P. MacDonald et al., 2005

Thermalhydraulics Tools

- Density distributions were based on cross-sectional averaged calculations
- Local-conditions variations are important due to flow and enthalpy imbalances within bundles
  - Internal, side, and corner rods in a fuel assembly
  - Rods in various rings of a fuel bundle
- Detailed calculations are required using analytical tools
  - Subchannel codes
    - Essential for fuel design and optimization
  - Computational-fluid dynamics tools
    - Support fuel design (e.g., spacers) and subchannel-code development
    - Design support of other components (e.g., outlet nozzles)

Subchannel Codes

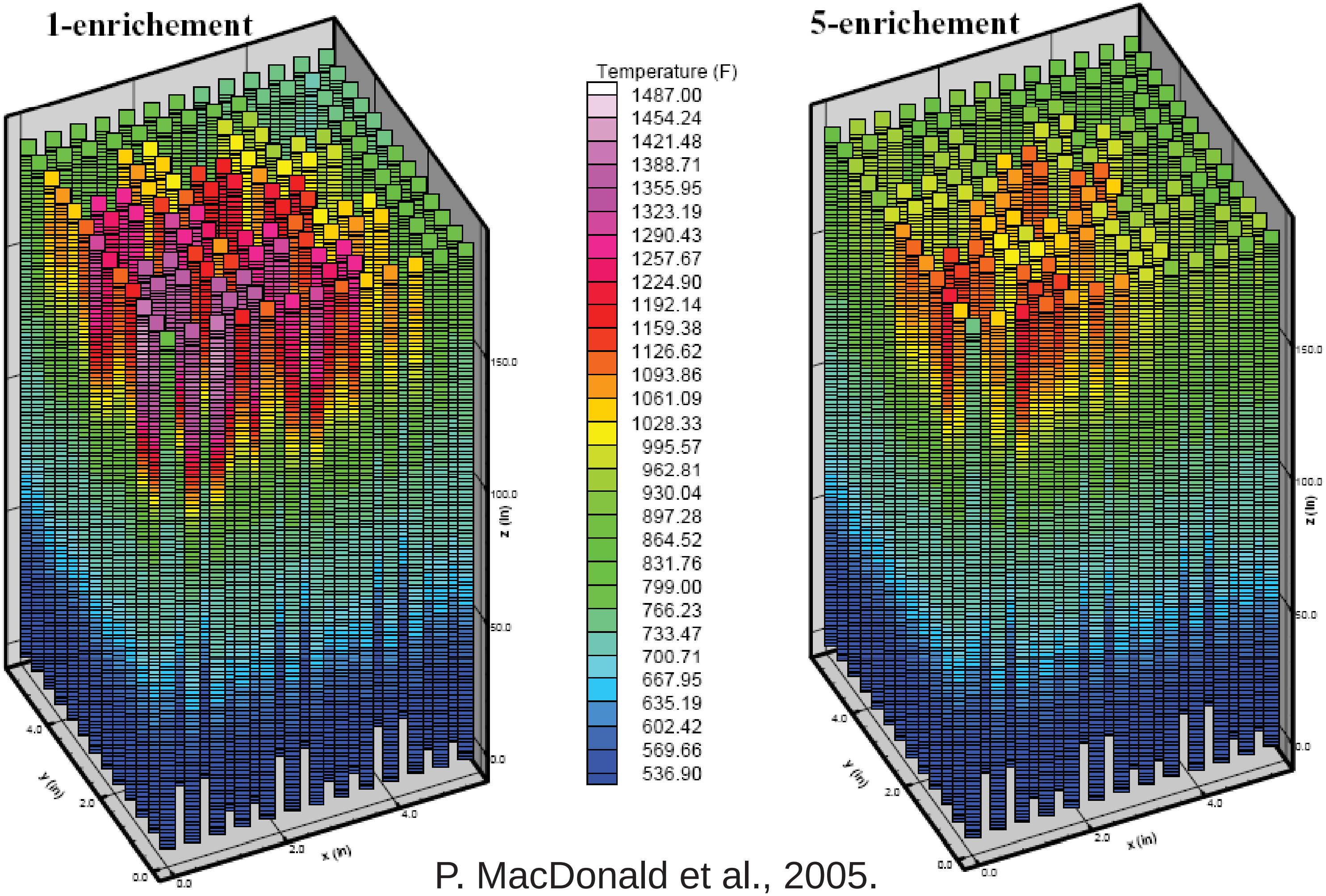
- Local flow conditions and wall temperature predictions
  - Separate the flow area in the fuel bundle or assembly into interconnected subchannels
  - Each subchannel is treated as a tube-equivalent flow channel
  - Local flow conditions inside each subchannel are calculated using equations of conservation and boundary conditions
  - Constitutive equations are introduced to establish heat transfer, transition points, mixing between subchannels, void drift, buoyancy, etc.
- Subchannel flow modelling:
  - Sub-critical conditions
    - Homogeneous flow
    - Drift-flux model
    - Two fluids
  - Super-critical conditions
    - No phase change
    - Pseudo-liquid and pseudo-vapor flows



Subchannel Analysis for SCWR Fuel Assemblies

- Mainly for normal operating conditions
  - Relatively simpler than analyses for current fleets of reactors
    - No phase change
  - Provide detailed distributions of local flow conditions and cladding temperature in the fuel assembly and bundle
  - Support fuel design and optimization
- Issues
  - Lack of relevant data to establish constitutive equations (e.g., mixing)
  - Reflecting mainly the parametric trends of the implemented correlations
    - Observed effect in tubes may not be present in bundles
  - Lack of prediction methods for separate effects (e.g., spacer, gap size, etc.)
  - Lack of relevant bundle data for validation

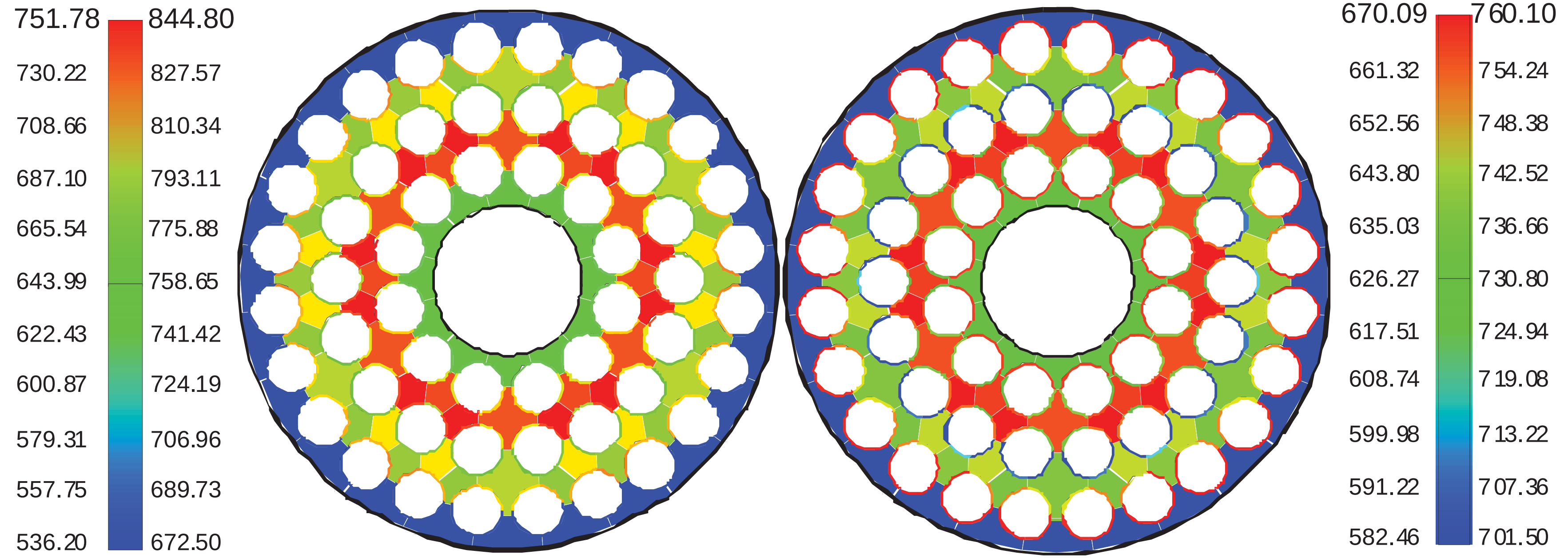
Illustrations of Subchannel-Code Predictions of Cladding Temperatures



P. MacDonald et al., 2005.

Fuel Bundle Design Optimization Using Subchannel Code

Exit Pressure	25 MPa
Inlet Temperature	350°C
Outlet Temperature	625°C
Mass Velocity	806 kg/(m^2·s)
Average Heat Flux	0.788 MW/m^2



Reference Bundle　　Optimized Bundle

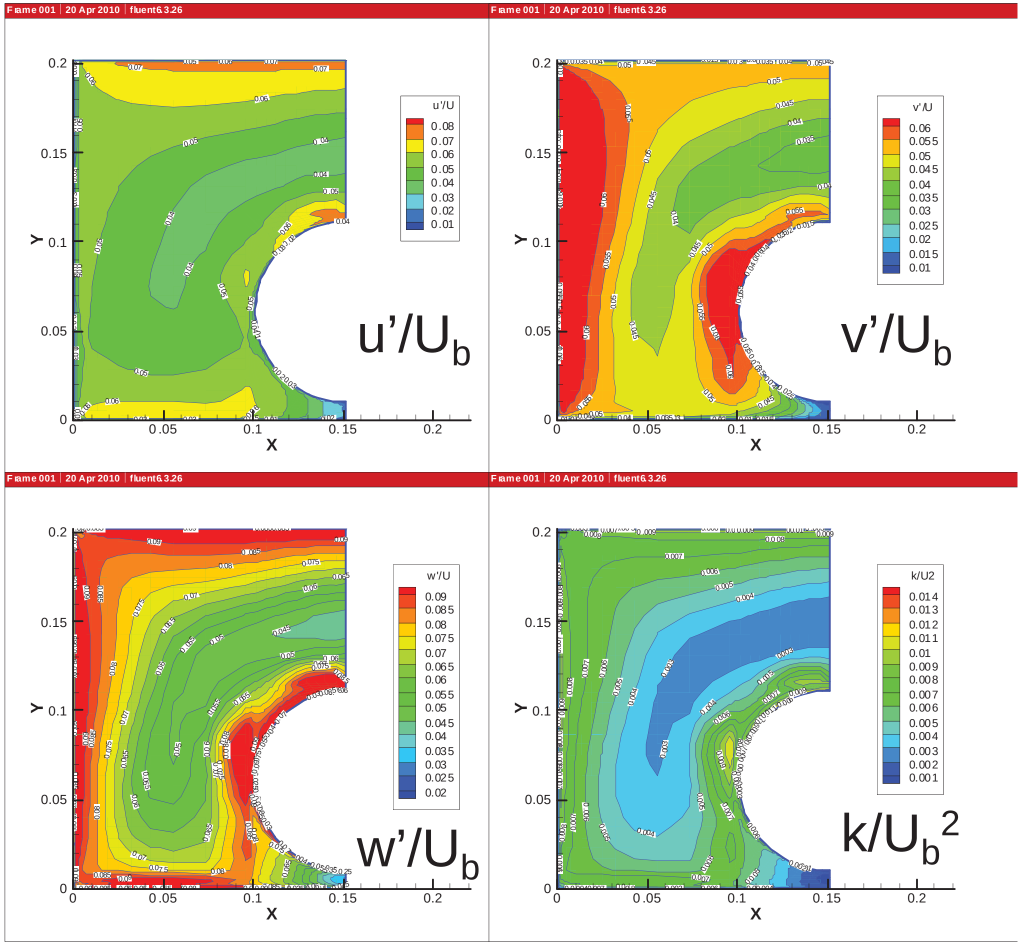
Needs of Computational Fluid Dynamics Analyses

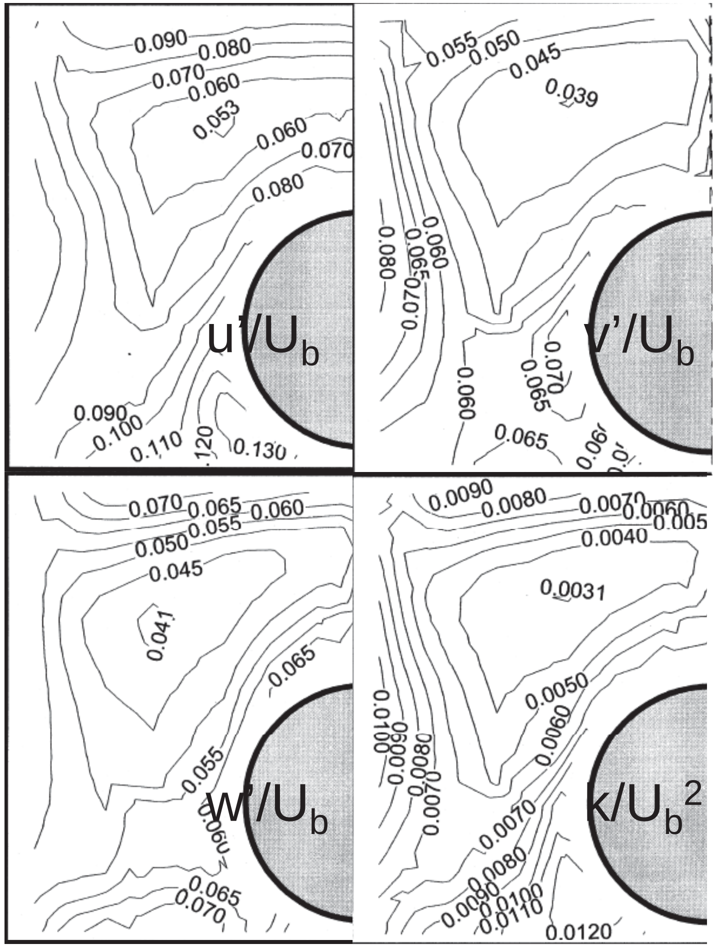
- Issues related to subchannel code applications prompt the need of further details in thermal-hydraulics calculations
- In addition, supercritical fluid heat transfer depends strongly on the near-wall characteristics
  - Turbulence structure due to buoyancy and acceleration of near-wall fluid
    - Deteriorated heat-transfer phenomena
- Subchannel code is based on cross-sectional averaged conditions within a subchannel
  - Cannot capture such details
- Applications of computational fluid dynamic tools to support fuel design and optimization and supplement information to subchannel analyses

CFD Analysis for SCWR Fuel

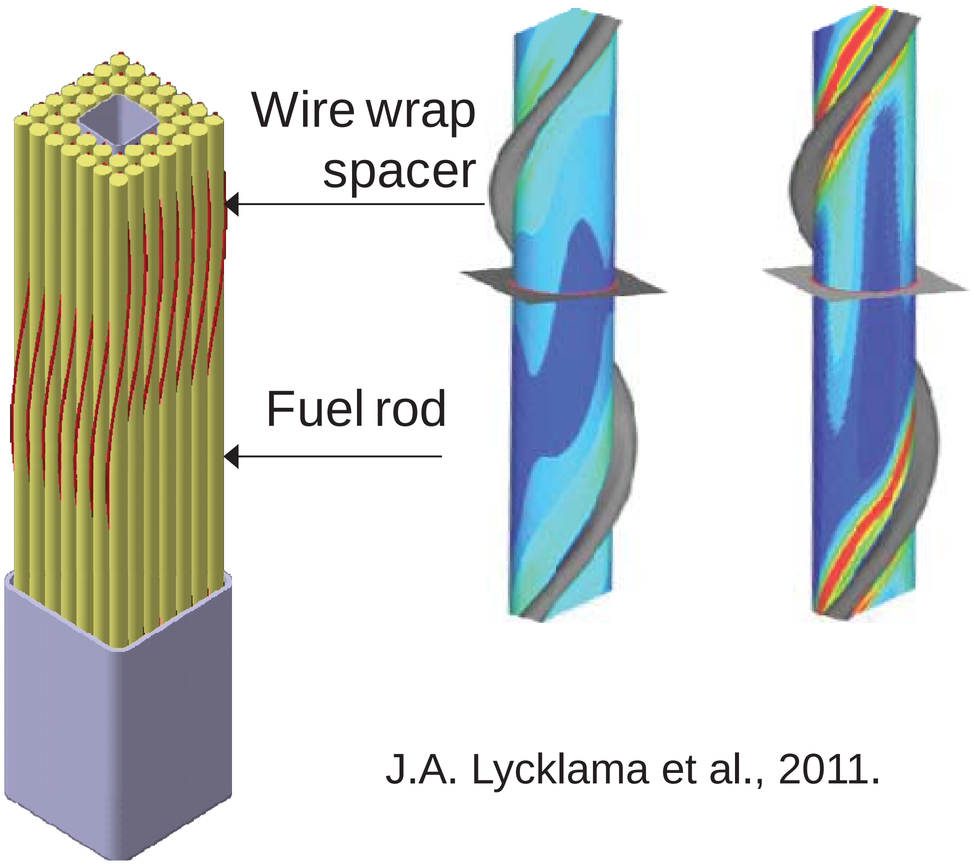
- Computational fluid dynamics (CFD) tools apply numerical methods and algorithms to evaluate fluid flow and heat transfer in open or closed channels
- Turbulence modelling
  - Model the length and time scales associated with turbulent flow
  - Examples: Direct numerical simulation, Large eddy simulation, Reynolds-averaged Navier–Stokes equations, etc.
- Mainly for normal operating conditions
  - Relatively simpler than analyses for current fleets of reactors
    - No phase change
    - CFD for two-phase flow is still under development
  - Provide detailed distributions of flow conditions and cladding temperature inside a subchannel
- Issues
  - Require considerable computing resources limiting its applications to simple geometry or small sections of fuel assembly at this point
  - Lack of turbulence measurements to establish appropriate turbulence models
  - Lack of detailed measurements of flow and temperature of relevant fixtures or geometry for validation

Illustrations of CFD Modeling of Turbulent Flow near a Rod

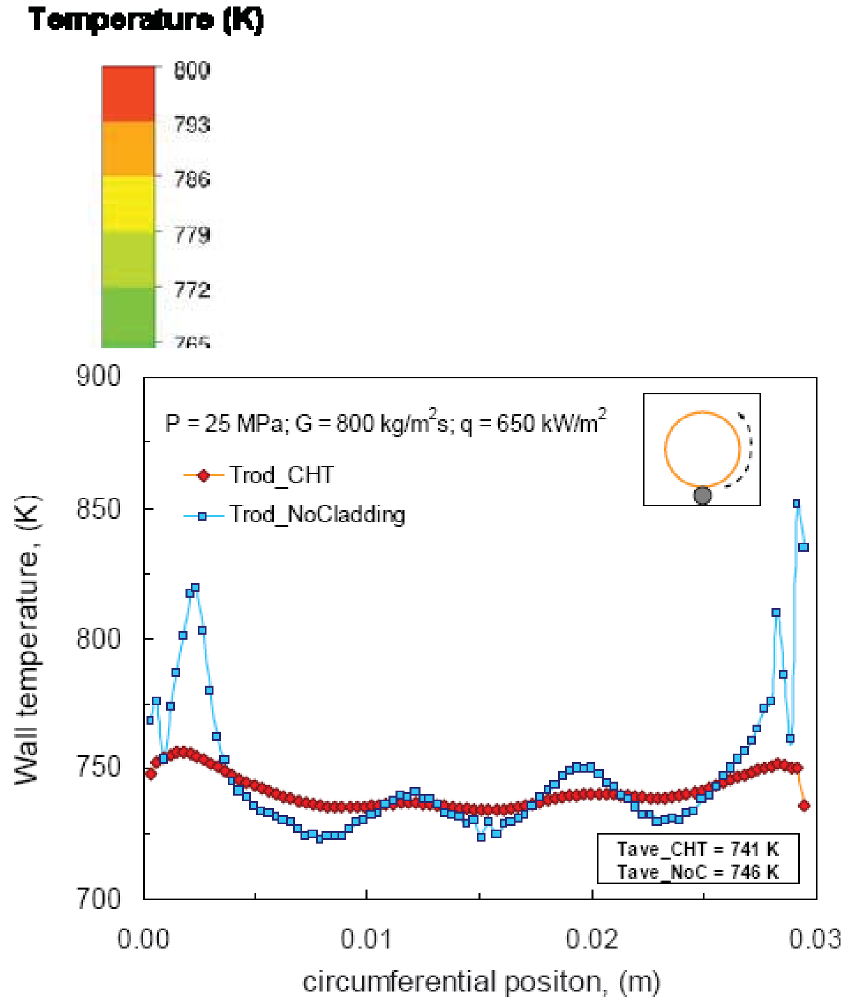




Illustrations of CFD Modeling of SCW Heat Transfer of a Wire-Wrap Spacer



J.A. Lycklama et al., 2011.



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Conclusions

- Several SCWR concepts are being pursued and have been completed at different stages
- SCWR reactor design goals match closely to those established by GIF
- Core design criteria are mainly related to cladding and fuel temperature
- Strong challenges remain in key design areas
- Coupled neutronic and thermalhydraulics analysis is necessary in view of the large variation in fluid density in the core impacting power distribution and stability
- Subchannel and CFD codes are important tools to assist in SCWR core and fuel designs
  - Further developments are still necessary

References (1)

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