



SuperCritical Water-cooled Reactor

***GIF-Symposium
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SCWR System Steering Committee

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1 General information on SCWR

General Features of SCWR

- *Evolutionary Innovative development on the base of the current LWRs' and fossil power plants' technologies*
- *Cooled with light water and moderated with light or heavy water System coolant of supercritical pressure (> 22.1 MPa) (supercritical)*
 - *Focus on thermal neutron spectrum with option on fast spectrum*
 - *Two options of power conversion system*
 - Direct (with once-through steam cycle, no coolant recirculation in the primary system, no steam generators, compact containment with pressure suppression pools)*
 - Indirect (with like-PWR two-circuit power conversion system)*
 - *High steam enthalpy, enabling compact turbines*
 - *Plant net efficiency $> 40\%$ for Indirect, $>44-45\%$ for Direct*
- *Decreased capital, O&M costs per installed MW and fuel consumption per MW-h, High availability (95%) and capacity factor ($> 85\%$) (improved economics)*
 - *Improved safety, proliferation resistance & sustainability*

1 General information on SCWR

General Challenges of SCWR compared with conv. LWR

- *Minimization of temperature non-uniformities in reactor core providing high averaged coolant enthalpy rise and coolant outlet temperature*
 - Coolant enthalpy rise in the core up to 10x hotter
 - Intermediate coolant mixing in the core?
 - Higher coolant core outlet temperatures $> 500^{\circ}\text{C}$
- *Development of water chemistry strategy minimizing corrosion, radioactive mass transport, dissolution of structure materials, deposition of impurities at FEs and served equipment surfaces, as well as suppressing radiolysis*
 - *Hotter peak cladding temperatures $> 600^{\circ}\text{C}$*
 - *Fuel cladding material integrity*
 - *Stainless steel instead of Zircalloy claddings*
- *Development of models prediction of heat transfer in pseudo-critical area cladding temperatures*
- *Provision of thermohydraulic and neutronic stability under condition of big compressibility of SCW coolant in pseudo-critical area*
 - *Development of different safety strategy for*
 - *Control of coolant mass flow rate instead of control of coolant inventory*
 - *Demonstration and use of passive safety systems*
- *Substantiation of proliferation resistance, e.g. in case of fast neutron spectrum*

1 General information on SCWR

SCWR System Agreement (year of sign.) and Representatives

- **Canada** (2006) *L. Leung, D. Brady*
- **Euratom** (2006) *T. Schulenberg, J. Starflinger*
- **Japan** (2006) *H. Matsui*
- **Russia** (2011) *A. Sedov, A. Churkin*
- **China** (2014) *Y.P. Huang, L.F. Zhang*

Projects:

- ***Thermal-Hydraulics and Safety, TH&S***, signed (EU, CA, JP), CN 's joining procedure is ongoing, and RU also expressed interest to join
- ***Materials and Chemistry, M&C***, signed (EU, CA, JP), CN 's joining procedure is ongoing,
- ***Fuel Qualification Testing, FQT***, provisional (EU, CA, JP, RU, CN)
- ***System Integration and Assessment, SI&A***, provisional (EU, CA, JP, RU, CN)

2 "Thermal-Hydraulics and Safety" Project

Project Status in 2015

- *Heat transfer data at supercritical pressures for rod bundles with prototypical spacer geometry have been obtained with*
 - *Supercritical water*
 - *Supercritical CO₂*
 - *Supercritical refrigerants.*
- *These data can now be used to validate codes and to improve prediction methods.*
- *A new joint benchmark exercise is being prepared to start in 2015*
- *The Project Plan is being updated to capture potential contributions of Canada, Euratom and China from 2015-2019*

2 Thermal-Hydraulics and Safety Project

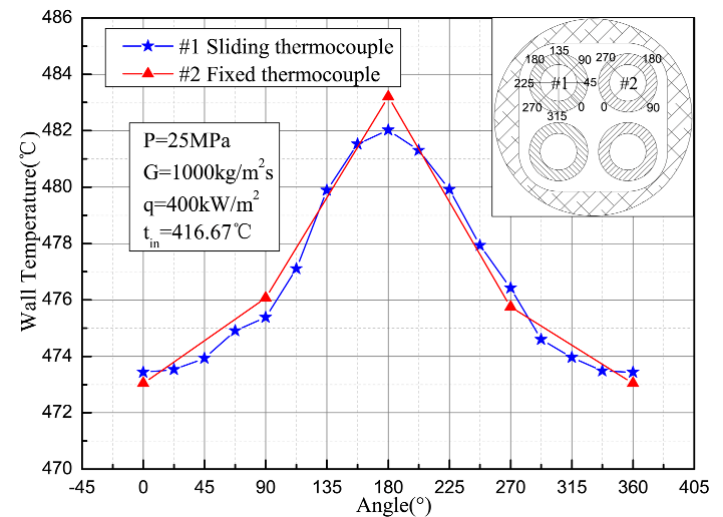
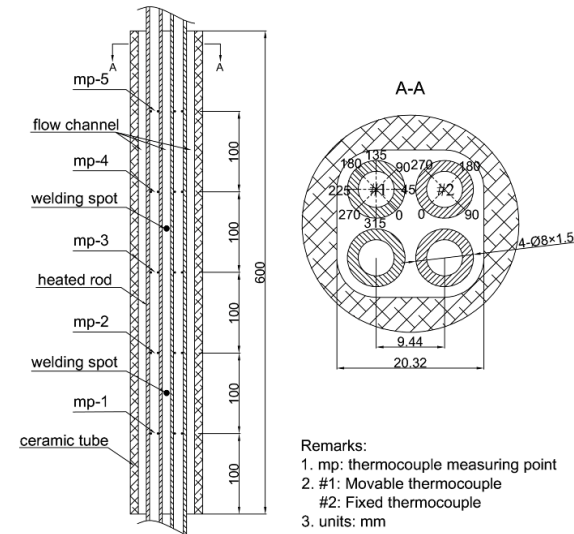
Canada contributions to TH&S Project

- Experimental wall temperature data obtained with supercritical water, carbon dioxide, and refrigerant-134a flow through tubes, annuli and bundles.
- Experimental flow data obtained with natural circulation of supercritical carbon dioxide in single and parallel channels (stability analyses)
- Experimental data on critical flow of supercritical water through sharp-edged orifices with 1-mm and 1.4-mm openings
- A prediction method for water in tubes at sub- and super-critical conditions
- Stability boundary for super-critical water in channels
- Critical-flow model

2 Thermal-Hydraulics and Safety Project

Canada contributions to “TH&S” Project

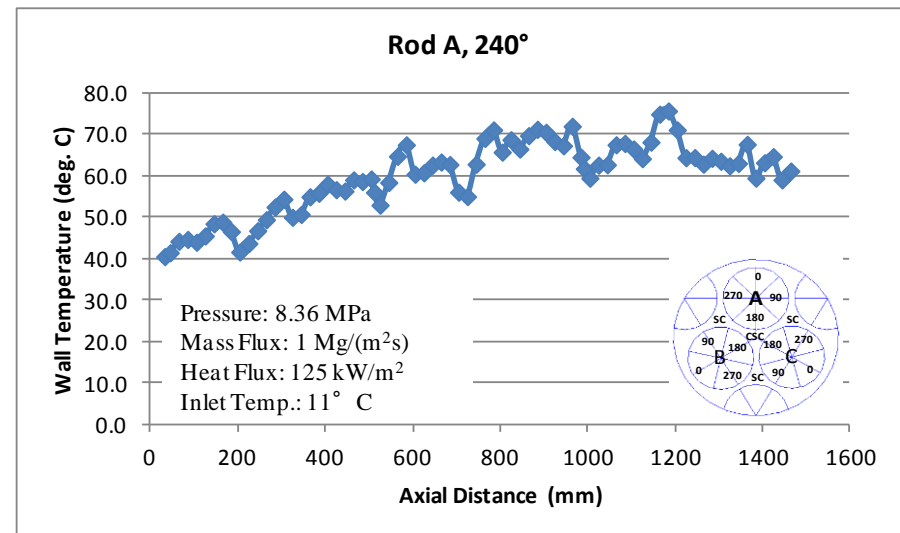
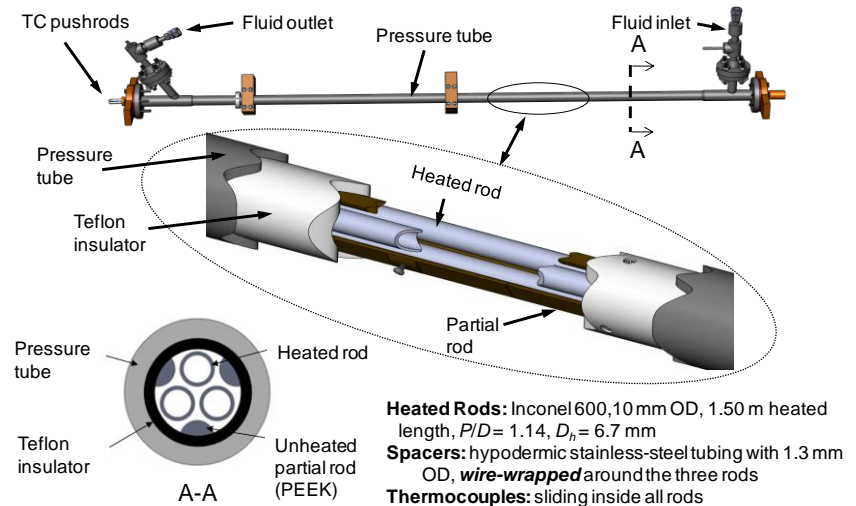
- 2x2 bundle tests with supercritical water flow
 - Four 8-mm OD rods
 - 600-mm heated length
 - Square flow channel with rounded corners
 - Moveable thermocouples in one rod and fixed thermocouples in another one to measure wall temperature distributions
- Two testing phases
 - Bundle with no spacer
 - Bundle with wire-wrapped spacers



2 Thermal-Hydraulics and Safety Project

Canada contributions to “TH&S” Project

- 3-rod bundle tests with upward flow of carbon dioxide
 - 10-mm Inconel-600 tubes of 1.5-m heated length
 - 1.4-mm gap between tubes (pitch/diameter = 1.14)
 - Unheated filler rod segments to minimize flow mal-distribution
 - Moveable thermocouples
 - Wide range of flow conditions at sub-critical and super-critical pressures
- Detailed circumferential temperature measurements along the heated length



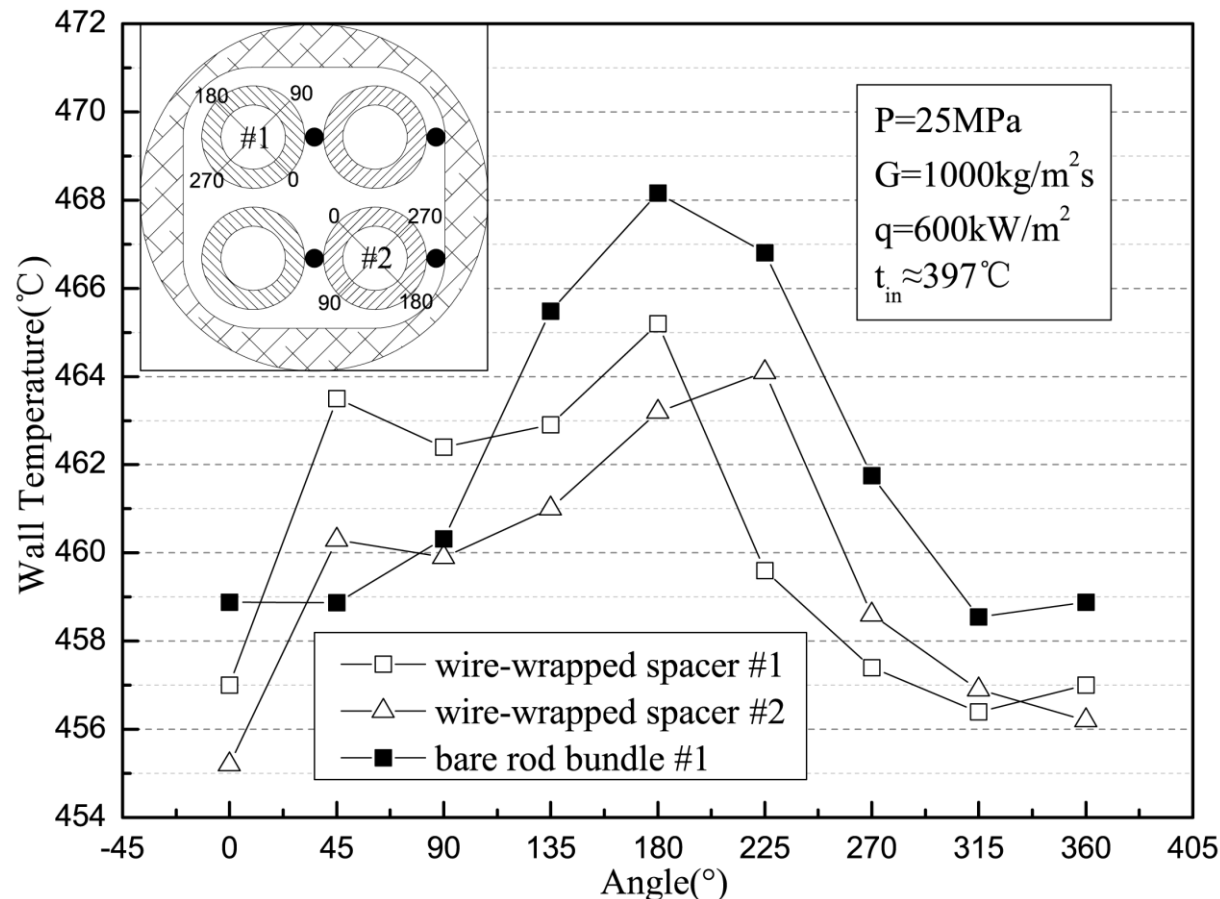
2 Thermal-Hydraulics and Safety Project

Chinese potential contributions to TH&S Project

- Experimental wall temperature data obtained with supercritical water flow through tubes, annuli and 2X2 bundles.
- Experimental flow data obtained with supercritical water flow through parallel channels for instability issue.
- Experimental flow data obtained with natural circulation of supercritical carbon dioxide in single channel (stability analyses)
- A prediction method for water in tubes at sub- and supercritical conditions
- A prediction method for stability boundary for supercritical water in channels
- Critical-flow models

2 Thermal-Hydraulics and Safety Project

Heat transfer experiment with supercritical water in a 2x2 rod bundle with *wire-wrapped spacer* from CANADA

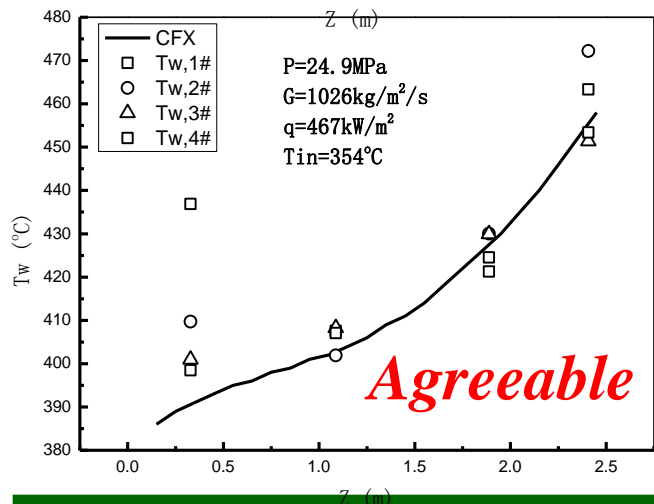
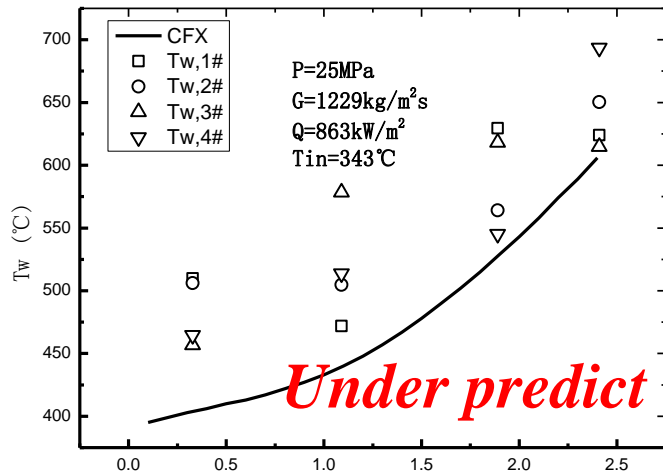


Improved coolant mixing due to the wrapped wire

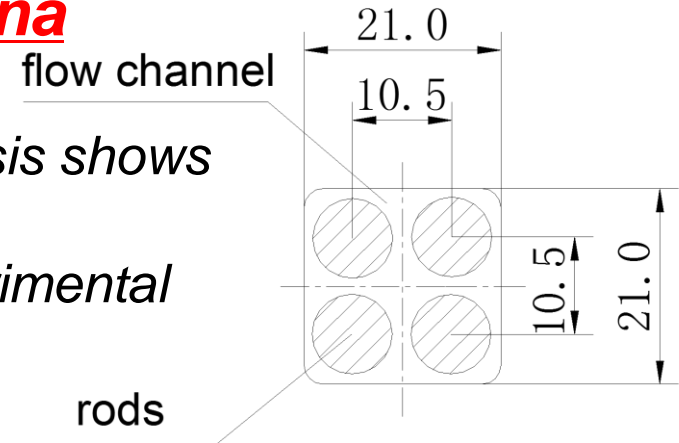
L. Leung, Y. Rao, ISSCWR7-2031

2 Thermal-Hydraulics and Safety Project

Heat transfer experiment with supercritical water in a **smooth 2x2 rod bundle from China**



Preliminary analysis shows
CFD performance
depends on experimental
parameters



Design pressure	30MPa
Design temp.	550°C
O.D. of rod	Φ9.5mm
Rod displacement	2×2, square arranged
Rod Pitch	10.5mm
Heated Length	2500mm
Channel dimension	square 21×21mm

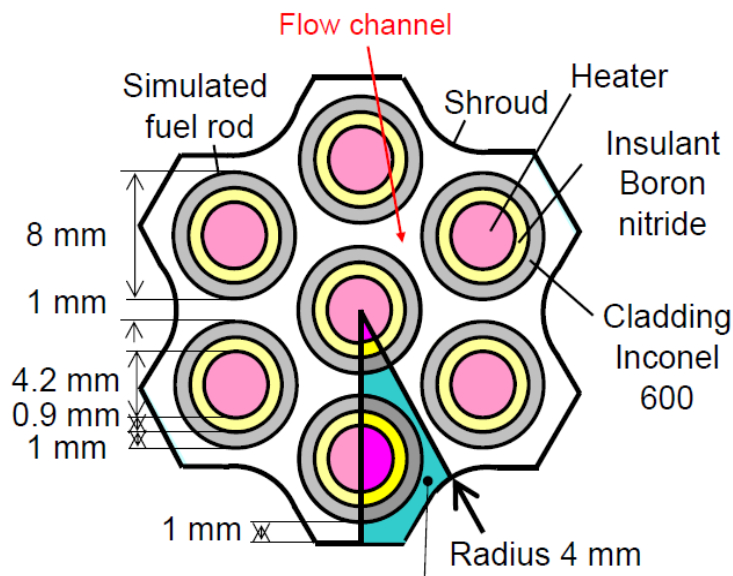
2 Thermal-Hydraulics and Safety Project

Joint benchmark exercise

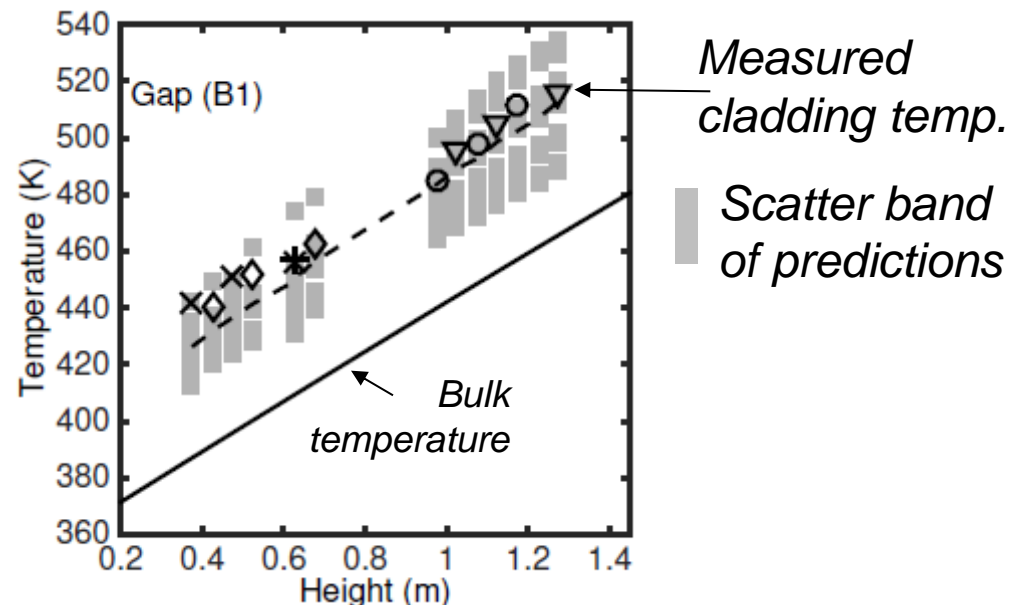
M. Rohde et al., ISSCWR7-2044

- *Flow and heat transfer of supercritical water in a 7 rod bundle*
- *Experimental data contributed by JAEA, Japan*
- *Blind predictions by 10 organizations from EU and Canada*
- *Organized by M. Rohde, TU Delft*

Test geometry



Typical benchmark result



2 Thermal-Hydraulics and Safety Project

TH&S Updated Project Plan

Planned future contributions 2015 to 2019, e.g.

- Heat transfer to supercritical water in tubes, annuli, sub-channels and rod bundles (CA, CN, RF)*
- Heat transfer to supercritical CO₂ and Freon in tubes, annuli and rod bundles; analysis of fluid-to-fluid scaling laws (CA, CN, EU, RF)*
- Pressure loss of supercritical water flow in rod bundles (CN, RF)*
- Test of rod cladding ballooning (RF)*
- Blow-down experiments with supercritical water (CA, CN, RF)*
- Flow instabilities (CA, CN, EU, RF)*
- SCWR safety requirements and evaluation (CA,EU, CN, RF)*
- System code development (CA, CN)*
- CFD and turbulence modelling (CA, CN, EU, RF)*
- Already 96 deliverables proposed in total (by CA, CN and EU)***

3 Materials and Chemistry Project

Progress in 2014:

- *EU: Commissioning tests of out-of-pile supercritical water loop at CVR/Rez completed*
- *CA, EU, JP: Joint deliverable on results of round-robin corrosion tests*
CA, EU: Development of Materials Databases
- *CA, EU: Development of coatings, surface modification*
- *CA, EU: Selection and qualification of commercial alloys in terms of general corrosion, stress corrosion cracking susceptibility and structural integrity*
- *CA, EU: Assessment of physico-chemical properties of SCW on materials corrosion behavior and general corrosion mechanism in SCW*
- *EU: Development work on reference electrodes and test facilities capable of working under in-situ reactor conditions*
- *CA: Specification of water chemistry control strategy, water radiolysis model*

3 Materials and Chemistry Project

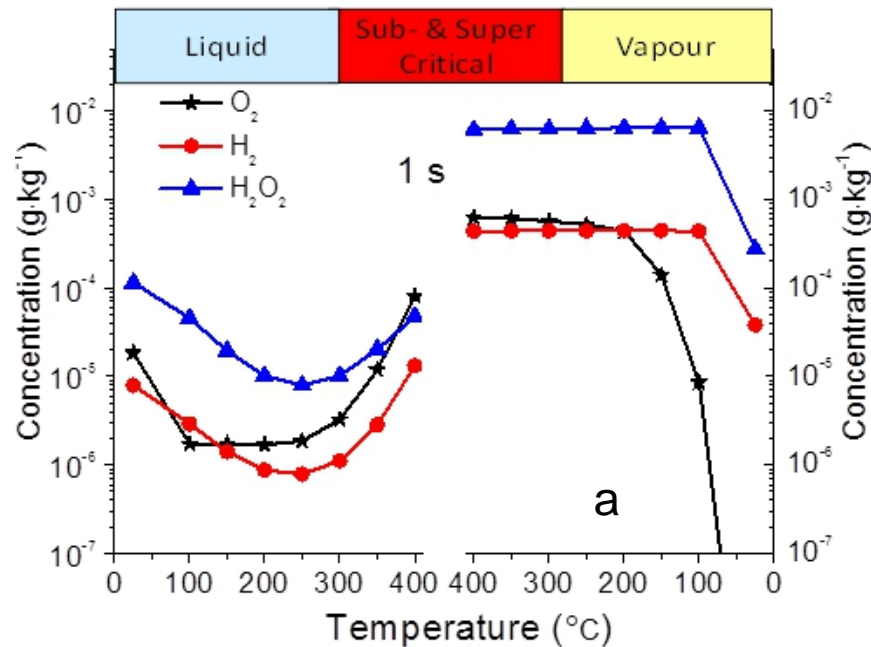
Project Status in 2015:

- *Study of the effect of surface finish and water chemistry on corrosion behaviour in supercritical water*
- *An iron/iron oxide reference electrode development work for in-situ corrosion monitoring up to 700 C in supercritical water*
- *Creep tests of SS 347H and SS 310S*
- *Stress corrosion cracking susceptibility tests of irradiated 310S and Alloy 800H at 625 C*
- *Interactive modelling of fuel cladding degradation mechanisms*
- *Round robin on general corrosion / stress corrosion cracking susceptibility tests in Europe, Canada and China to further assess facility-dependent effects*

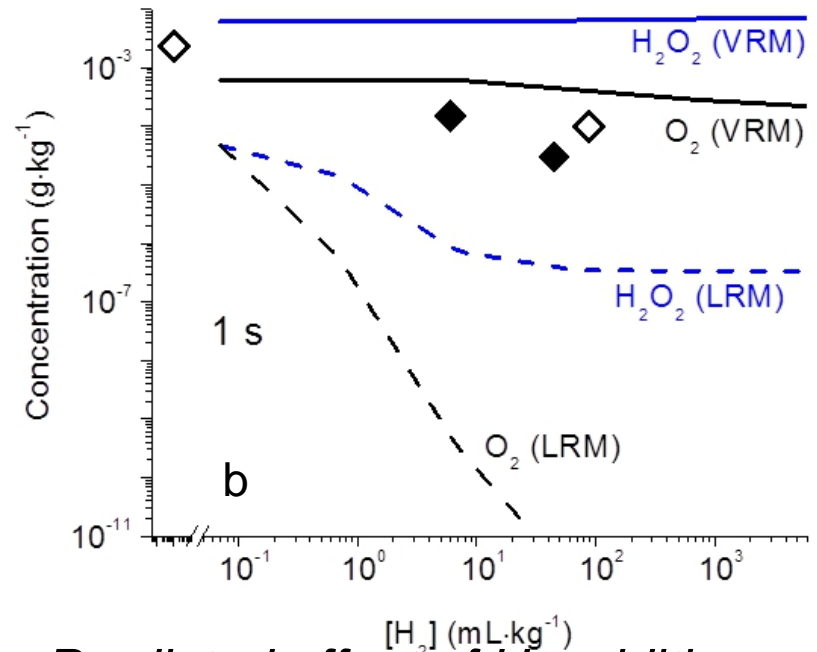
3 Materials and Chemistry Project

Project “Materials and Chemistry”, Example of model predictions

V. Subramanian et al., ISSCWR7-2083



Predicted concentrations of oxidizing species produced by water radiolysis in the Canadian SCWR core



Predicted effect of H_2 addition on the concentration of oxidizing species and comparison with measurements from Beloyarsk NPP in superheated steam

3 Materials and Chemistry Project

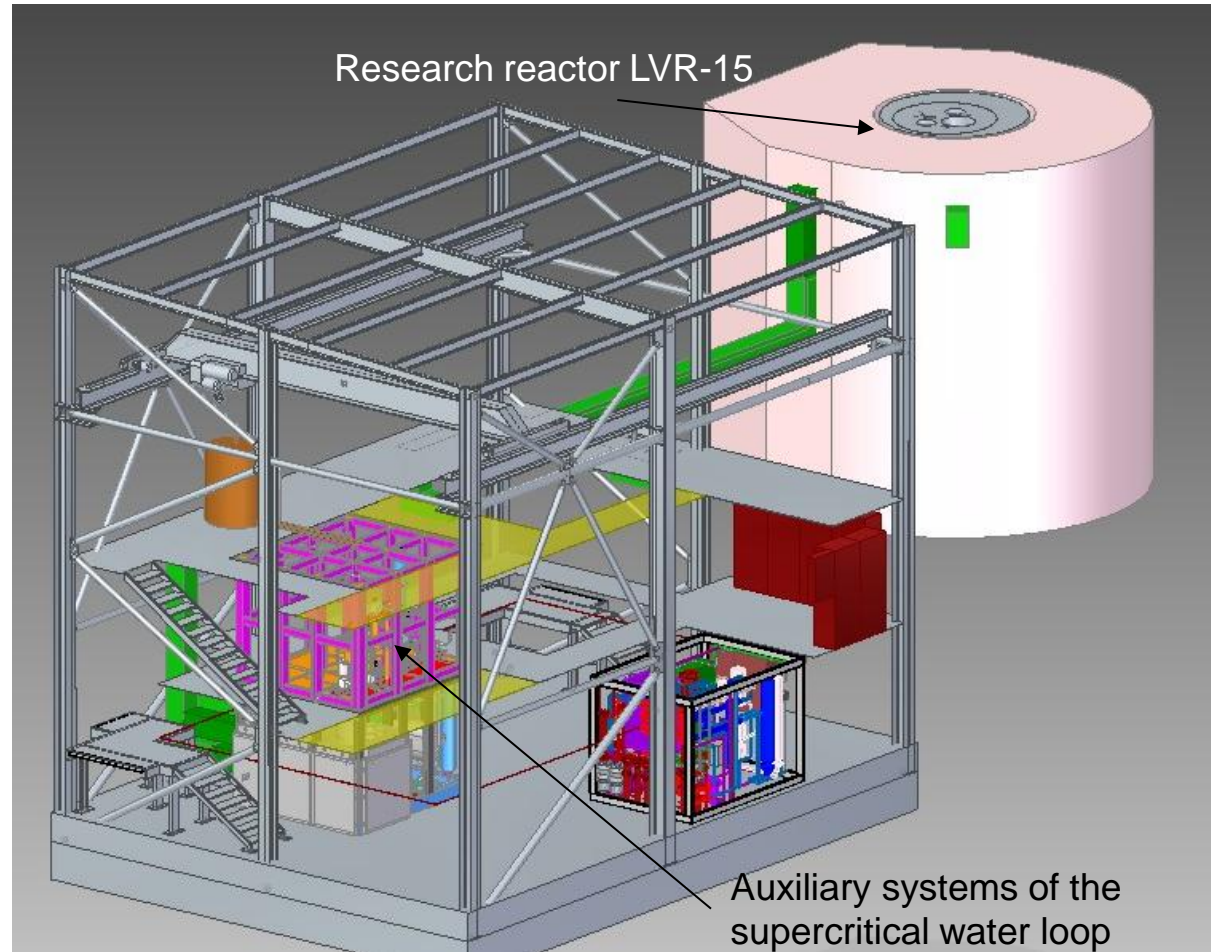
Project “Materials and Chemistry”

The project plan is being updated to capture potential contributions of Canada, Euratom and China from 2015 – 2019

- Tests of un-irradiated material: corrosion, stress corrosion cracking, creep, effect of coatings and surface modification, ODS materials (CA, CN, EU)*
- Radiolysis and water chemistry: corrosion tests with an in-pile supercritical water loop (EU), supported by modelling (CA), and out-of-pile test (CN)*

4 Fuel Qualification Testing Project

Planned FQT facility at CVR Rez, Czech Republic



*M. Ruzickova et al.,
ISSCWR7-2054*

4 Fuel Qualification Testing Project

Objectives of the Fuel Qualification Testing

The first time to use supercritical water in a nuclear reactor

- *Test of the licensing procedure, identify general problems*
- *Validation of thermal-hydraulic predictions*
- *Validation of transient system code predictions*
- *Validation of material performance*
- *Validation of stress and deformation predictions*
- *Qualification of fuel rod and spacer manufacturing processes*
- *Test of measurement systems for supercritical water*
- *Test of fuel-cladding interaction*
- *... etc.*

4 Fuel Qualification Testing Project

FQT test section inside the reactor core

Dimensions of the fuel assembly:

Rod diameter

8 mm

Cladding thickness

0.5 mm

Rod pitch

9.44 mm

Wire thickness

1.44 mm

Wire pitch

200 mm

UO₂ enrichment

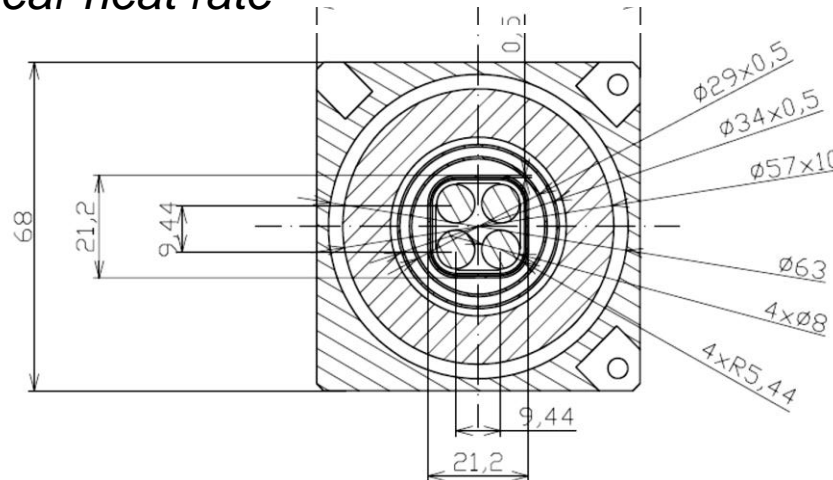
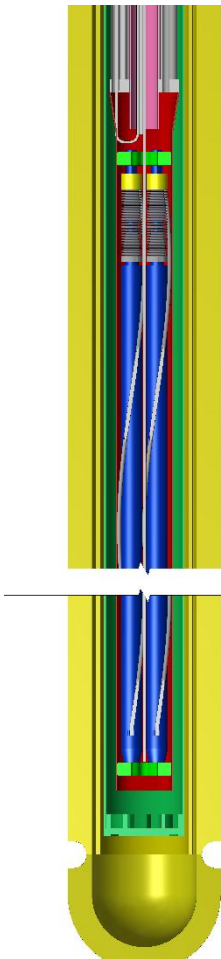
19.75%

Fissile power

63.6 kW

Linear heat rate

39 kW/m

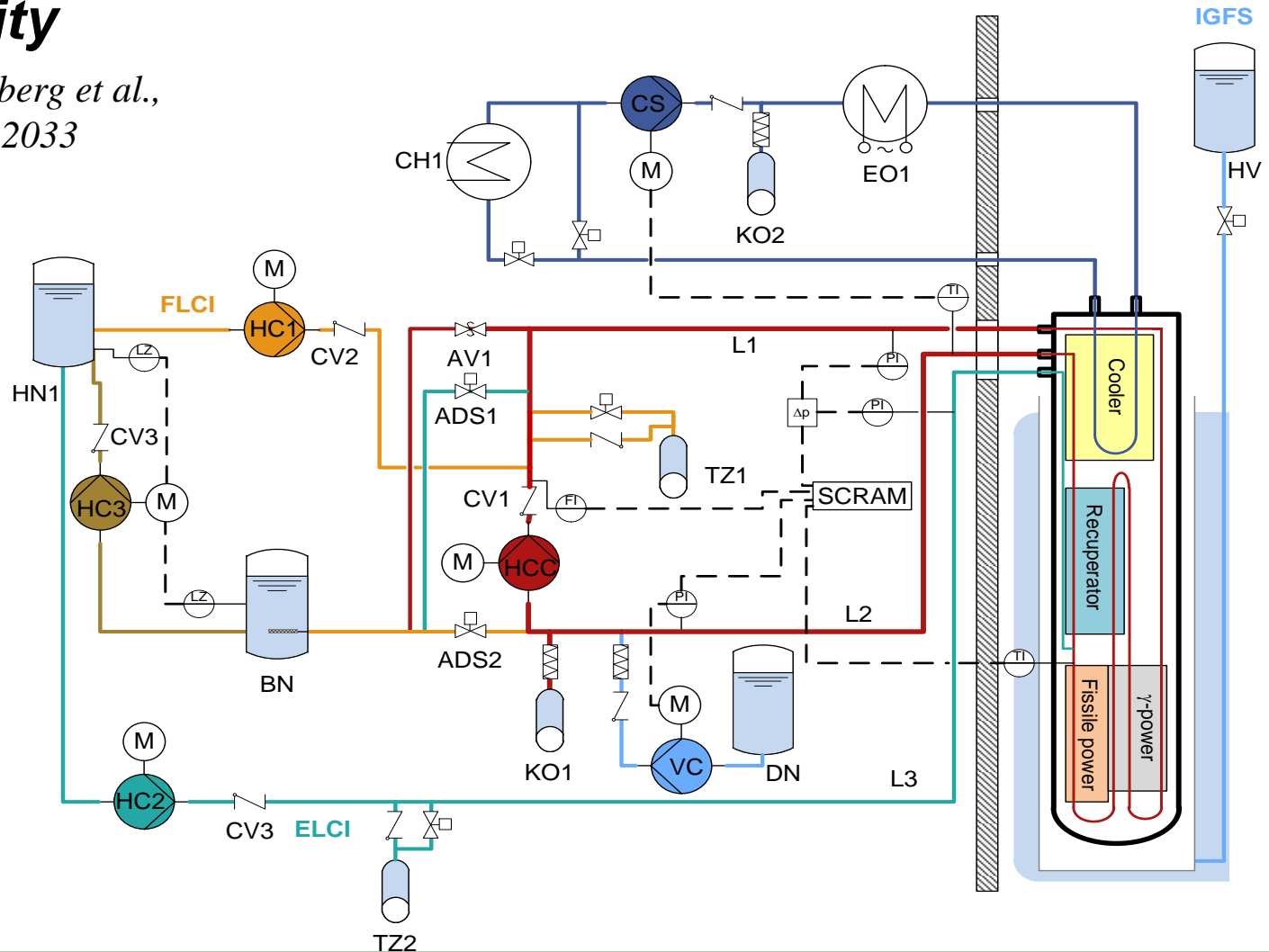


*M. Ruzickova et al.,
ISSCWR7-2054*

4 Fuel Qualification Testing Project

Safety system of the FQT facility

*T. Schulenberg et al.,
ISSCWR7-2033*



4 Fuel Qualification Testing Project

Out-of-pile tests of the test section for FQT Heat Transfer Experiments from SJTU, China

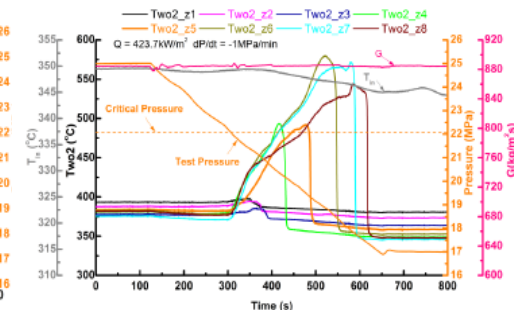
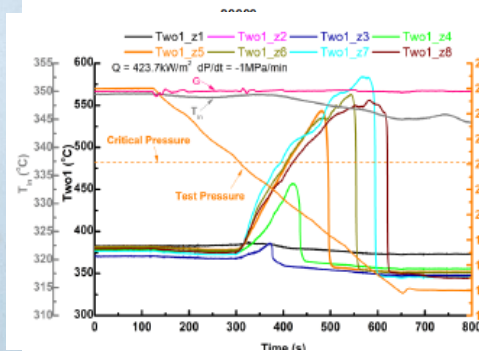
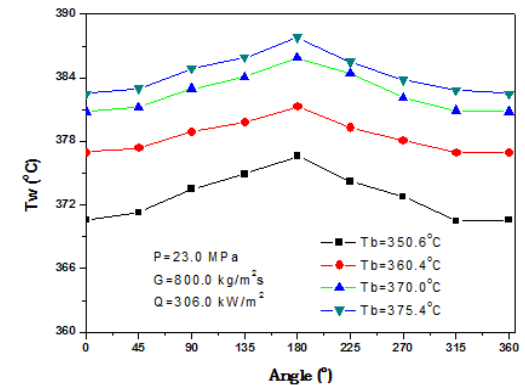
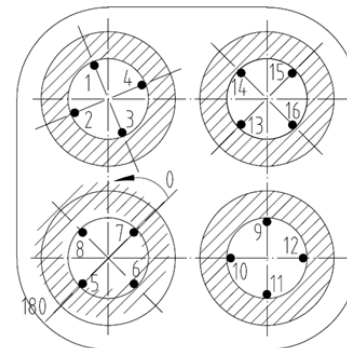


SWAMUP Supercritical Water Loop



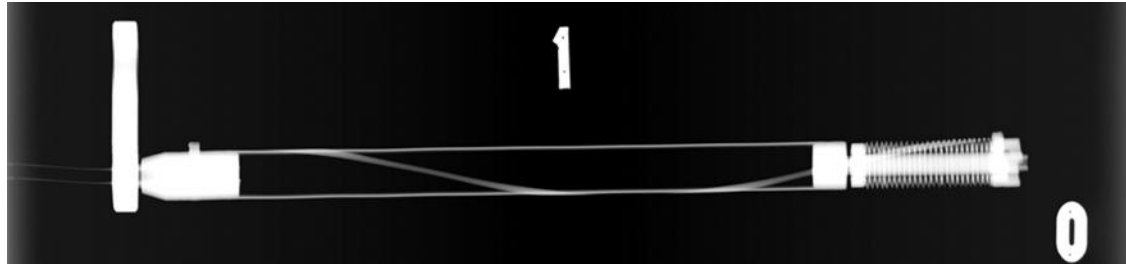
at SJTU, China

- Steady experiments to measure the wall temperature for CFD validation
- Depressurization transient experiments to validate the system code



4 Fuel Qualification Testing Project

FQT fuel pin mock-up tests



Radiographic 2D X-ray image of the fuel pin mock-up after successful test



Collapsed fuel rod in case of non-successful test

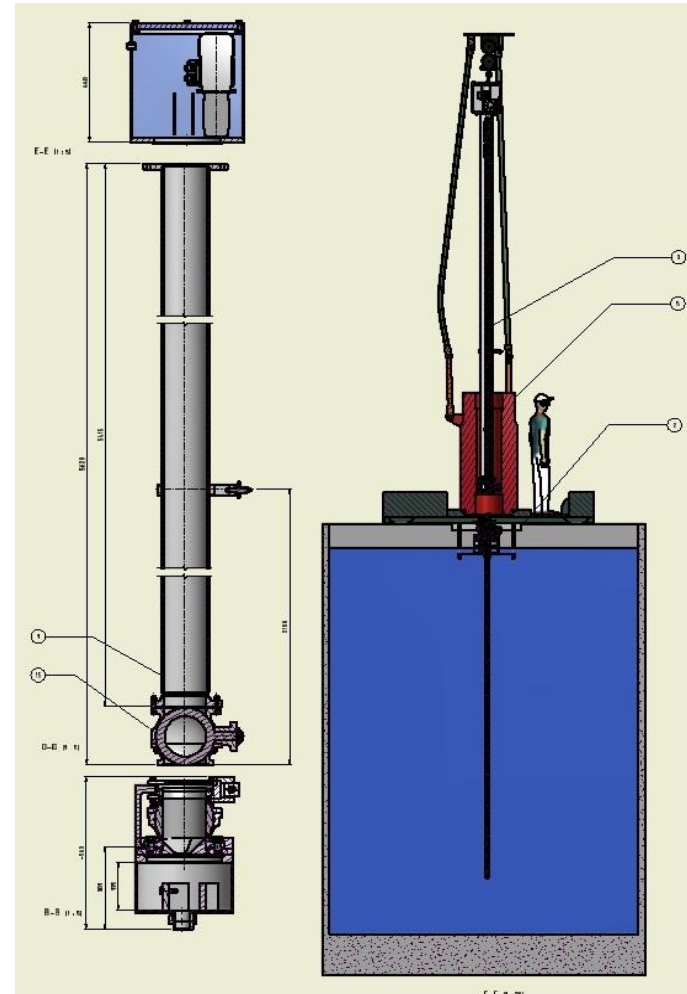
R. Novotny et al., ISSCWR7-2080

4 Fuel Qualification Testing Project

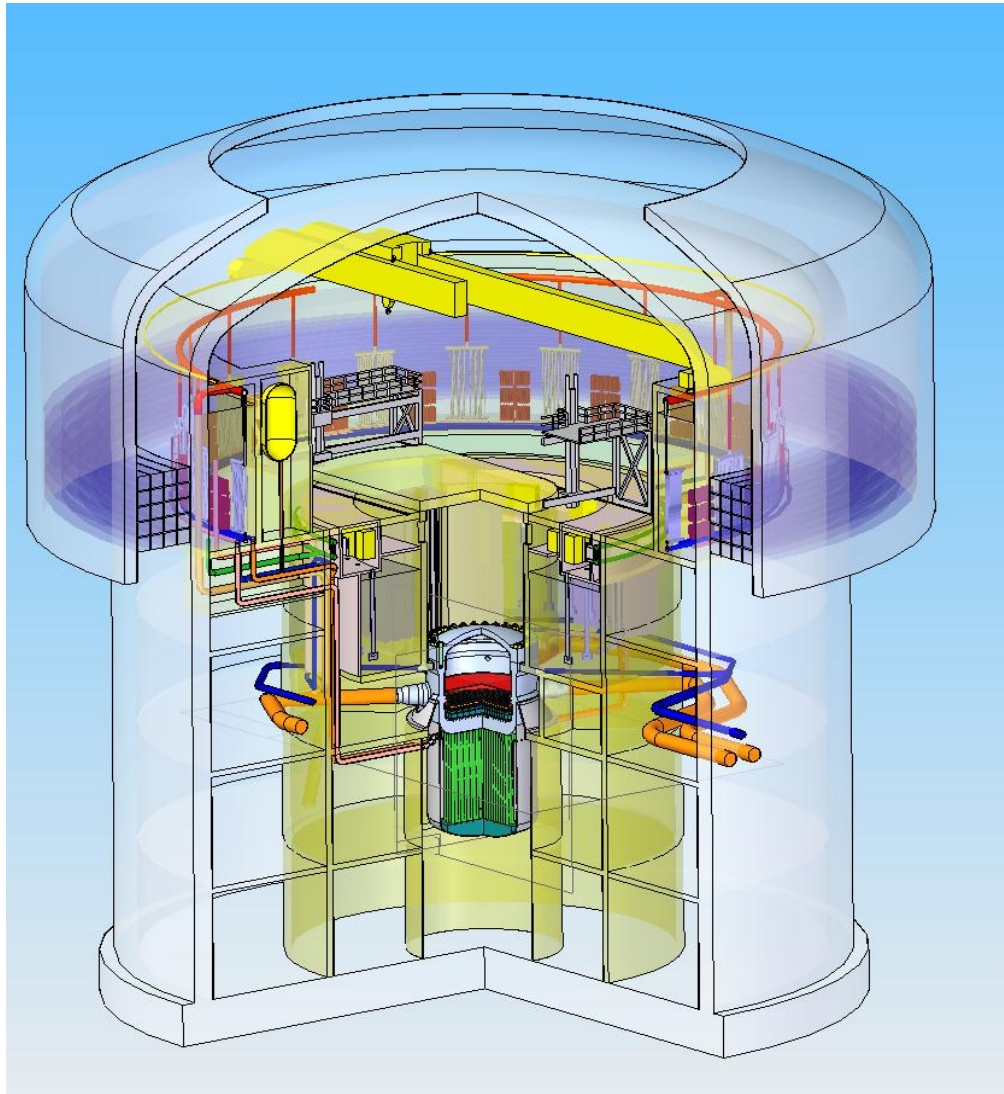
Design of the fuel handling system for the fuel qualification test

This system is needed to remove the fuel in case of failure of fuel rods during the in-pile tests

*M. Ruzickova et al.,
ISSCWR7-2054*



5 System Integration & Assessment Project



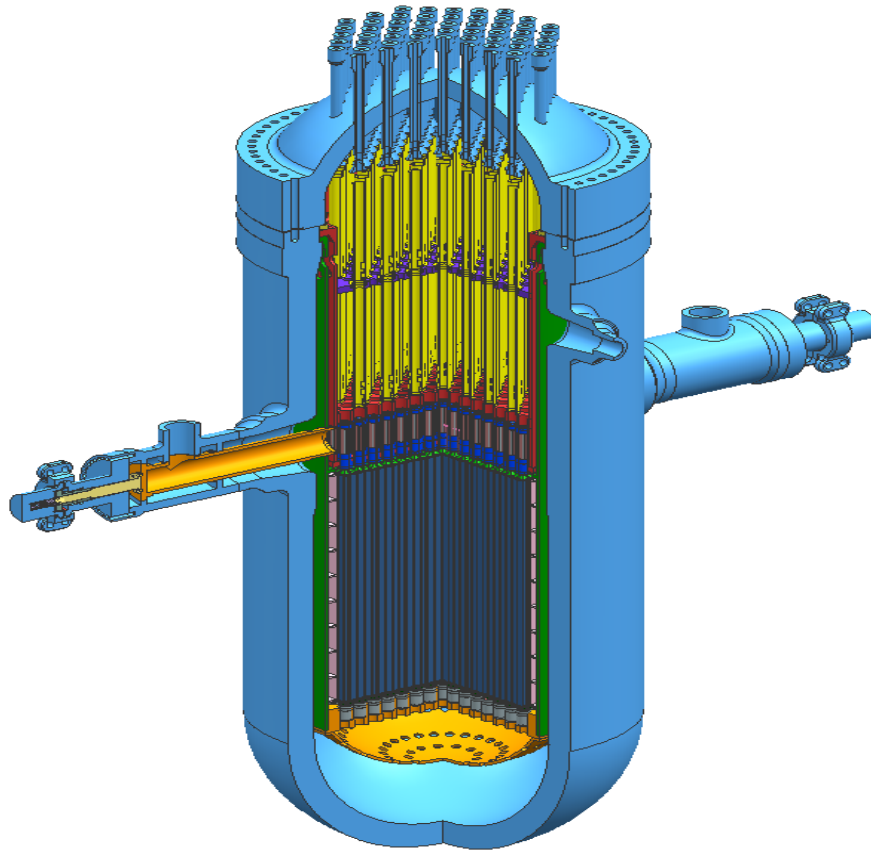
*Canadian SCWR
design concept with
pressure tubes*

*336 vertical fuel channels
2500 MW thermal power
1200 MW electric power
625 C core outlet temp.
48% efficiency*

*Design to be
completed and will
be assessed in Oct.
2015*

5 System Integration & Assessment Project

*China SCWR design concept with
pressure vessel-CSR1000*



**CSR1000 technical
parameters**

<i>parameters</i>	<i>value</i>
<i>thermal power</i>	<i>2300MW</i>
<i>electric power</i>	<i>~1000MWe</i>
<i>efficiency</i>	<i>~ 43%</i>
<i>operating pressure</i>	<i>25MPa</i>
<i>design pressure</i>	<i>27.5MPa</i>
<i>reactor inlet temperature</i>	<i>280°C</i>
<i>reactor outlet temperature</i>	<i>500°C</i>
<i>reactor flow rate</i>	<i>4284 t/h (1190 kg/s)</i>
<i>loop number</i>	<i>2</i>
<i>cycle</i>	<i>direct once- through</i>
<i>coolant flow-path</i>	<i>two-pass</i>
<i>design lifetime</i>	<i>60 years</i>

Design to be completed and assessed in 2017

6 ISSCWR-7 Conference

- *Successfully held on March 15-18, 2015 in Helsinki*
 - *Hosted by VTT Technical Research Centre of Finland*
 - *in co-operation with the Finnish Network for Generation Four Nuclear Energy Systems (GEN4FIN), the Generation IV International Forum (GIF) , the International Atomic Energy Agency (IAEA) and the Canadian Nuclear Society (CNS).*
- *Provided a forum for discussion of advancements and issues, sharing information on technical achievements, and establishing future collaborations on research and development for SCWR between research organisations.*
- *About 90 participants took part in the symposium from 14 different countries and 92 talks were given during the symposium week. All symposium papers were published as conference proceedings.*
- *Selected papers will be published in international journal for archival.*

6 ISSCWR-7 Conference

***For further information,
please visit (<http://www2.vtt.fi/sites/isscer7>)***



**Thank you for
your attention!**