SCWRs: THERMODYNAMIC CYCLE OPTIONS AND THERMAL ASPECTS OF PRESSURE-CHANNEL DESIGN

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HIGHLIGHTS

- Introduction to SCWRs
- General Considerations on SCW NPP Cycles
- SCW NPP Cycles Description
- SCW NPP Cycles Analysis and Results
- SCWR Fuel-Channel Calculations
- SCWR Fuel-Channel Results
- New HT Correlation for SC Water
- Conclusions

INTRODUCTION TO SCWRS

- Super-Critical Water Reactor (SCWR) is one of the six Generation IV concepts chosen for further investigation and development.

- Generation IV technology is expected to:
  - Increase thermal efficiency
  - Reduce capital costs
  - Minimize generation of nuclear waste
  - Reduce risk of weapon proliferation

- Currently operating Nuclear Power Plants (NPPs) thermal efficiencies range between 30 — 33%.

- As such, more competitive designs, with higher thermal efficiencies (45 — 50 %) need to be developed and implemented.

INTRODUCTION TO SCWRS

- SCWR concept dating back to late 1950s.
- Successful deployment of supercritical coal-fired thermal power plants triggers research in the area of SCW NPPs.

Parameter	Value
Critical Pressure (MPa)	22.1
Critical Temperature (°C)	374.1
Critical Density (kg/m^3)	315.0

- Proposed SCWR operating parameters:
  - Pressure - 25 MPa
  - Reactor outlet temperature - 625°C

GENERAL CONSIDERATIONS ON SCW NPP CYCLES

Review of SC Turbines

Major Parameters of Selected Current and Upcoming Hitachi SC Turbines

First Year of Operation	Power Rating (MW_{el})	P (MPa)	T_{main} / T_{reheat} (°C)
2011	495	24.6	566 / 566
2010	677	25.5	566 / 566
	809	25.4	579 / 579
	790	26.4	600 / 620
2009	677	25.5	566 / 566
	600	25.5	600 / 620
2008	1000	24.9	600 / 600
	870	24.7	566 / 593
	870	24.7	566 / 593
2007	1000	24.9	600 / 600
	870	25.3	566 / 593

GENERAL CONSIDERATIONS ON SCW NPP CYCLES

SCW NPP Cycle Options

- Direct Cycle
  - SC “steam” from nuclear reactor fed directly to a SC turbine.
  - Eliminates need for complex and expensive equipment such as steam generators.
  - Potentially highest cycle efficiency for given parameters.
- Indirect and Dual Cycles
  - Heat exchangers used to transfer heat from a reactor coolant (primary loop) to the secondary loop coolant.
  - Safety benefits of containing potential radioactive particles inside the primary loop coolant.
  - Reduced maximum temperature of the secondary loop coolant lowers the cycle efficiency.

GENERAL CONSIDERATIONS ON SCW NPP CYCLES

Reheating Options for SCW NPP

- **No-Reheat Cycle**
  - Simplified layout, contributing to lower capital costs.
  - Lowest efficiency of all considered configurations.
- **Single-Reheat Cycle**
  - Reduced development costs due to wide variety of SC turbines with single reheat currently manufactured.
  - Increased complexity associated with introduction of Steam-ReHeat (SRH) channels to the reactor core.
- **Double Reheat Cycle**
  - Highest thermal efficiency, but complicated nuclear-steam reheat configuration that would significantly increase design and construction costs.

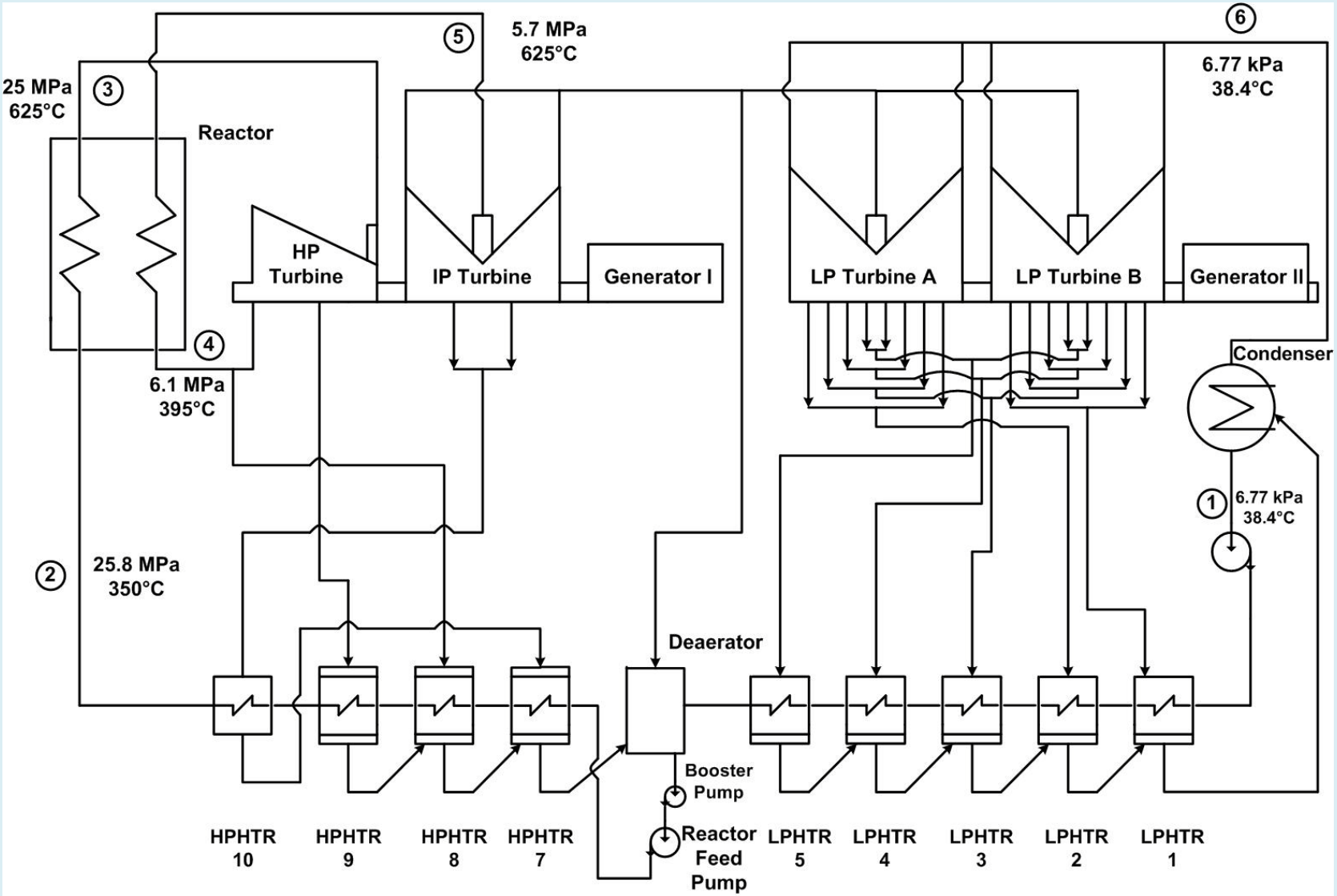
GENERAL CONSIDERATIONS ON SCW NPP CYCLES

SCW NPP Model Assumptions and Simplifications

- **Gland-Steam System and auxiliary steam consumers neglected.**
- **Performance losses due to mechanical equipment, generator and piping-pressure drops also neglected.**
- **Steady-state, steady-flow processes with negligible potential and kinetic effects and no chemical reactions.**
- **System parameters calculated for NPP power output of 1200 MW_{el}.**

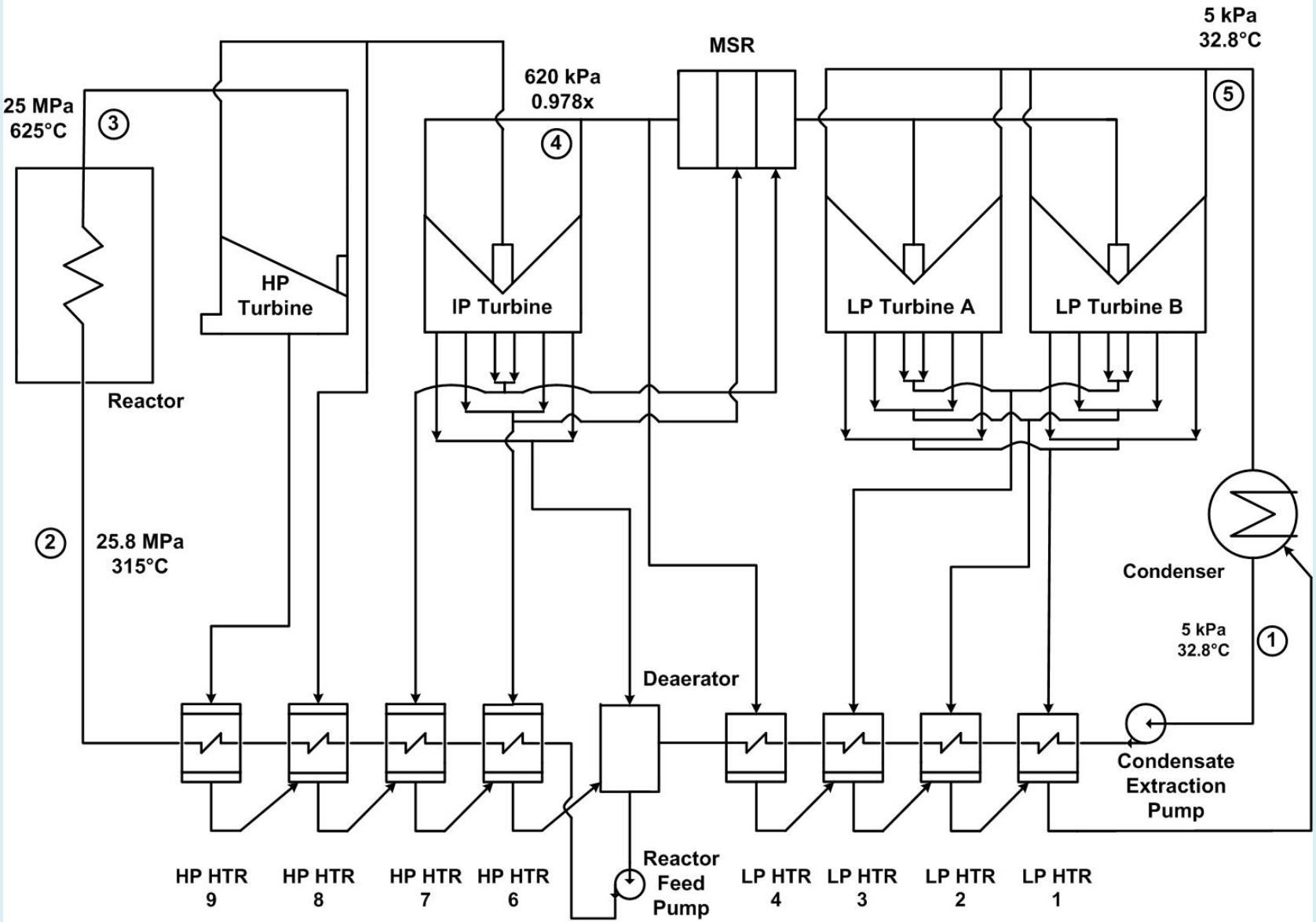
SCW NPP CYCLES DESCRIPTION

Single-Reheat Cycle A



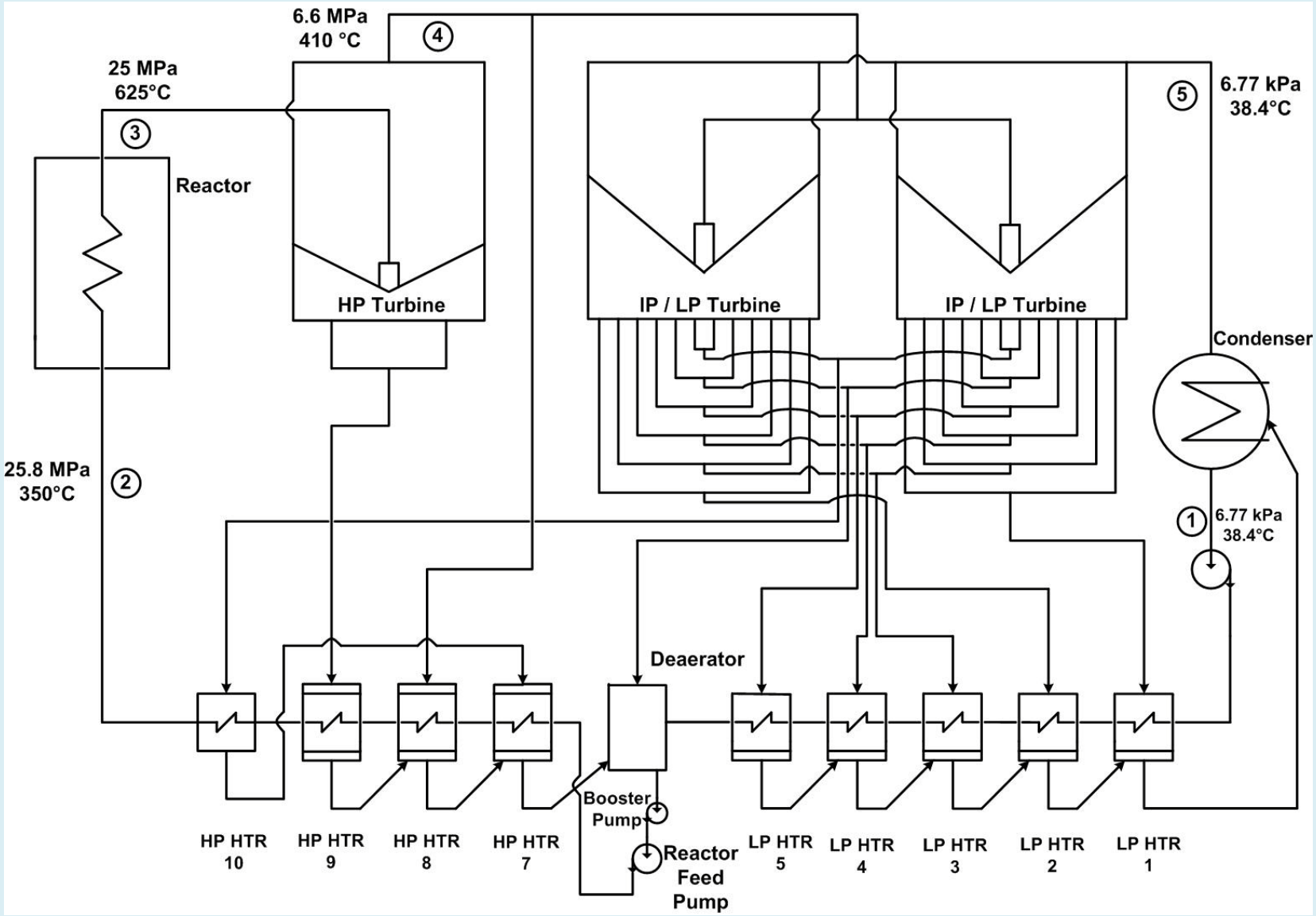
SCW NPP CYCLES DESCRIPTION

Single-Reheat Cycle B (with Moisture Separator Reheat unit)



SCW NPP CYCLES DESCRIPTION

No-Reheat Cycle C

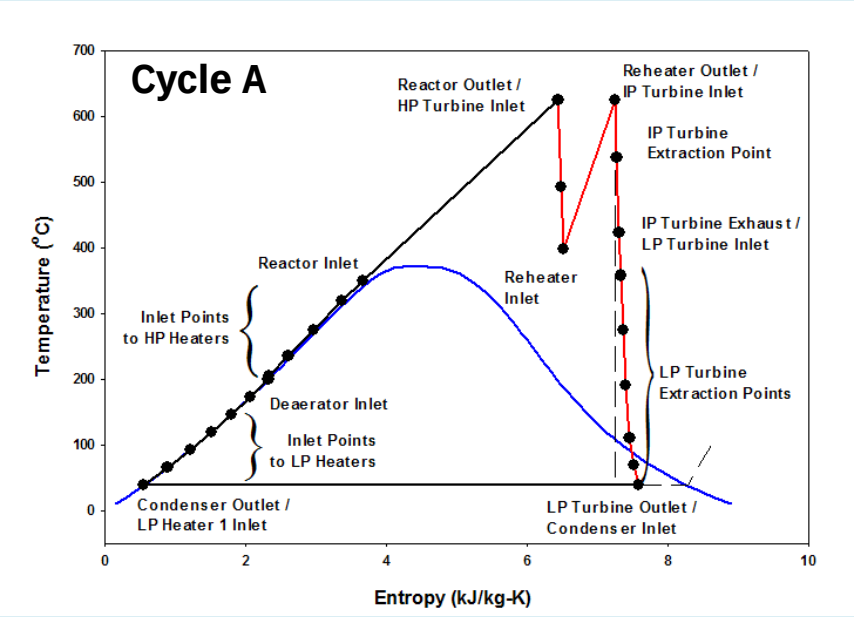


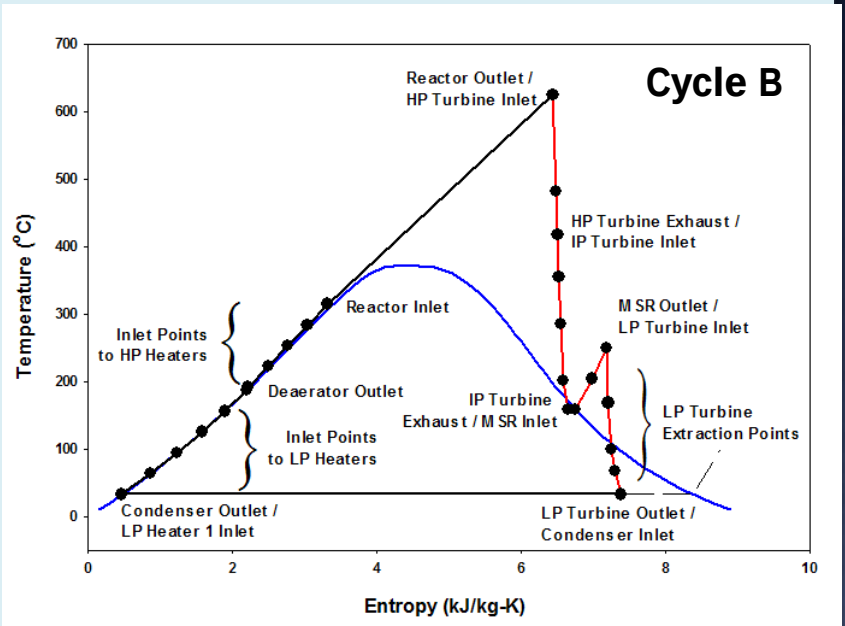
SCW NPP CYCLES ANALYSIS AND RESULTS

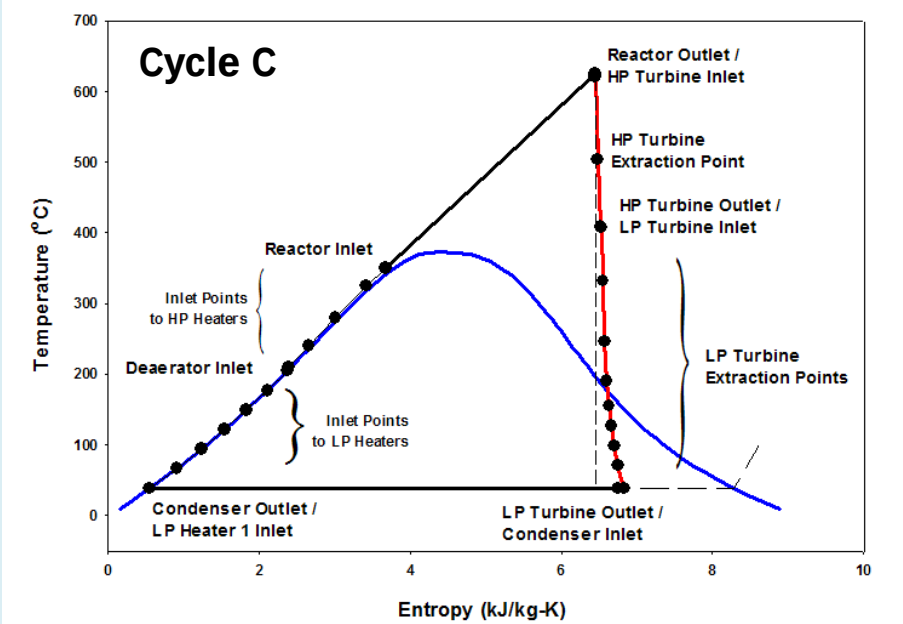
Thermal Efficiency of Proposed SCW NPP Cycles

SCW NPP Cycle	Thermal Efficiency (%)
A (Single-Reheat)	52
B (Single-Reheat with MSR unit)	52
C (No-Reheat)	51

T-s Diagrams Associated with the Proposed SCW NPP Cycles







Selected Parameters of Proposed SCW NPP Cycles A and C

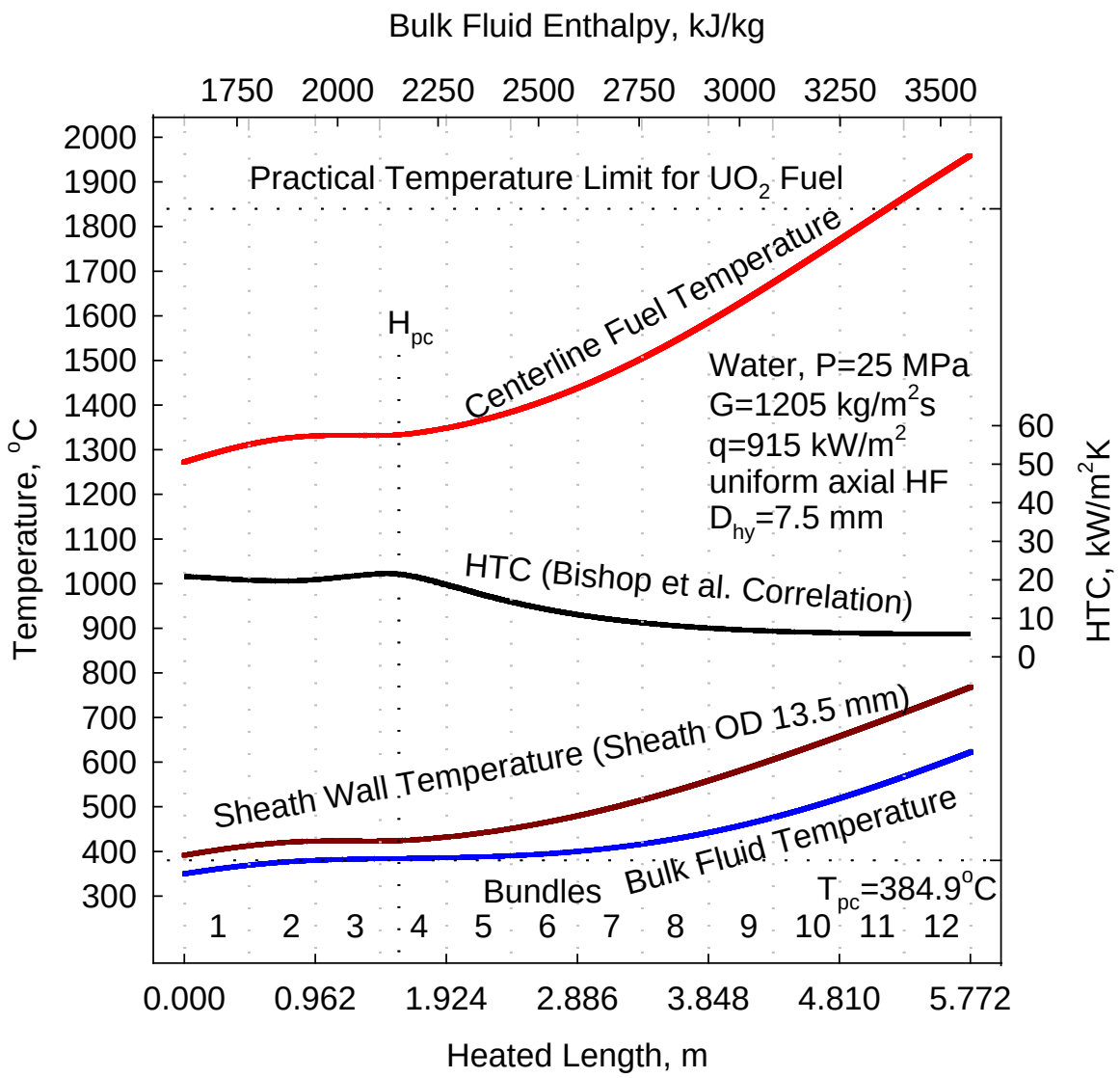
Parameters	Unit	Description / Value	Description / Value
Cycle Type	—	Single-Reheat (A)	No-Reheat (C)
Reactor Type / Spectrum	—	Pressure Tube / Thermal	
Fuel	—	UO_2 (ThO_2)	
Cladding Material	—	Inconel or Stainless Steel	
Reactor Coolant	—	H_2O	
Thermal Power	—	2300	2340
Pressure of SCW at Outlet	MPa	25	25
T_{in} Coolant (SCW)	°C	350	350
T_{out} Coolant (SCW)	°C	625	625
Pressure of SHS at Inlet	MPa	6.1	—
T_{in} Coolant (SHS)	°C	400	—
T_{out} Coolant (SHS)	°C	625	—
Thermal Power SCW Channels	MW_{th}	1870	2340
Thermal Power SRH Channels	MW_{th}	430	—
Thermal Power SCW /SRH Channel	MW_{th}	8.5 / 5.5	8.5 / —
Number of SCW / SRH Channels	—	220 / 80	270 / —
Total Flow Rate of SCW / SHS	kg/s	960 / 780	1190 / —
Flow Rate SCW / SRH Channel	kg/s	4.37 / 10	4.37 / —

Selected Parameters of Proposed SCWR fuel channels

Parameters	Unit	Description / Value		
T_{max} cladding (design value)	°C	850		
T_{max} fuel centerline (industry limit)	°C	1850		
Heated fuel-channel length	m	5.772		
Bundles per fuel channel	—	12		
Number of fuel rods per bundle	—	43		
Bundle type	—	CANFLEX	Variant-18	Variant-20
Number of heated fuel rods	—	43	42	42
Number of unheated fuel rods	—	—	1	1
OD of heated fuel rods (# of rods)	mm	11.5(35) & 13.5 (8)	11.5	11.5
Diameter of unheated fuel rods	mm	-	18	20
D_{hy} of fuel channel	mm	7.52	7.98	7.83
Heated area of fuel bundle string	m^2	9.26	8.76	8.76
Flow area of fuel channel	mm^2	3625	3788	3729
Pressure tube ID	mm	103.45		
Heat flux in SCW channel (A & C cycles)	kW/m^2	918	970	970
Heat flux in SRH channel (A cycle)	kW/m^2	594	628	628
Mass flux in SCW channel (A & C cycles)	kg/m^2s	1206	1154	1172
Mass flux in SRH channel (A cycle)	kg/m^2s	2759	2640	2682

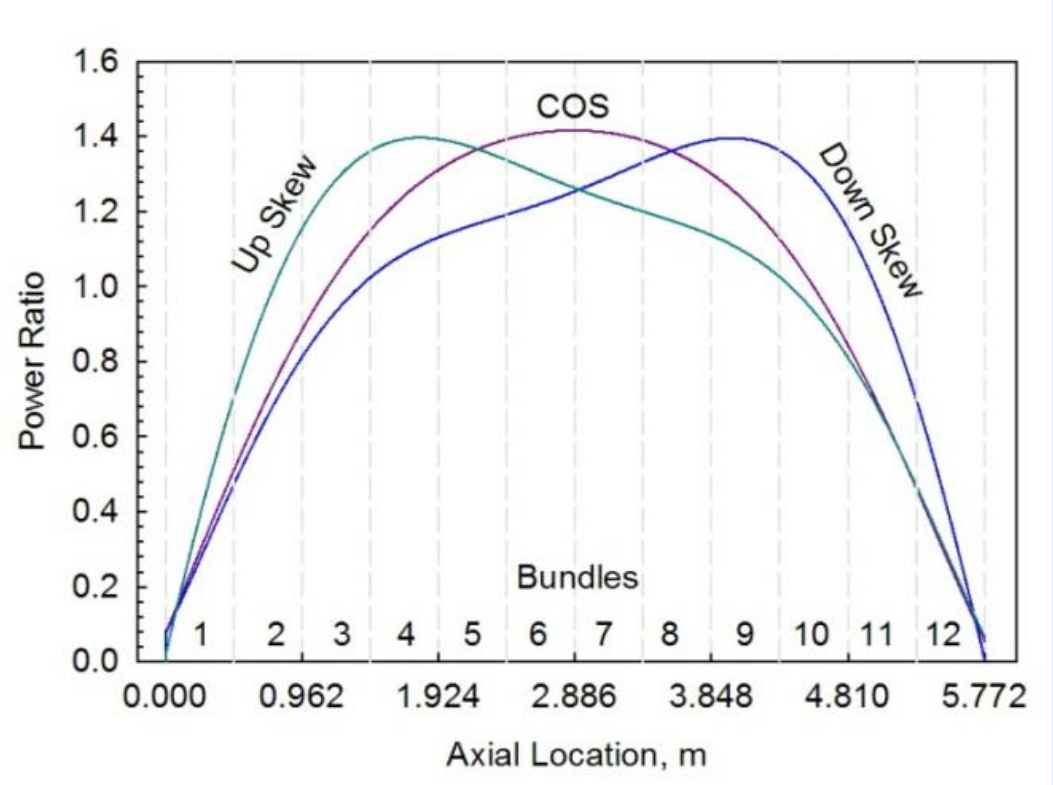
SCWR FUEL-CHANNEL CALCULATIONS

- **Previous study with uniform Axial Heat Flux Profile (AHFP) and average fuel thermal conductivity showed that fuel centerline temperature might exceed industry accepted limit of 1850°C for UO_2.**



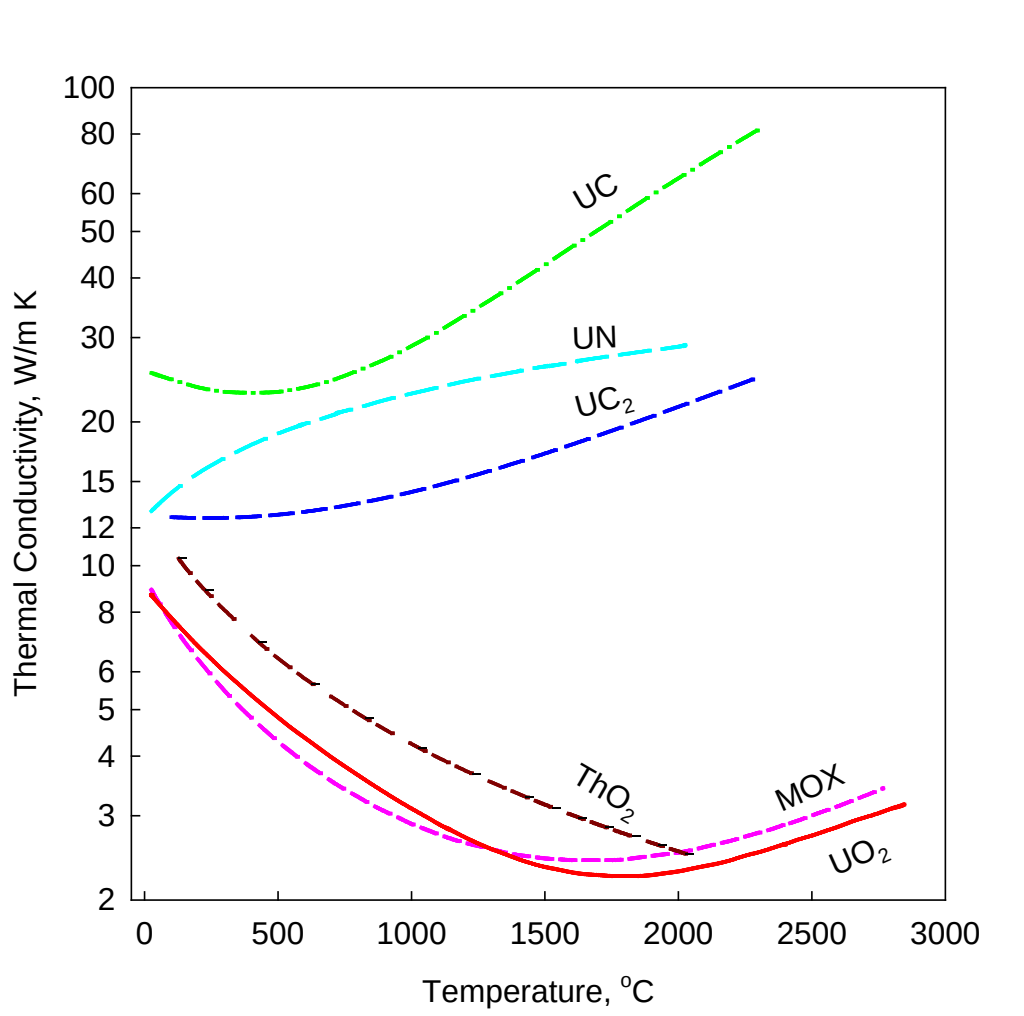
SCWR FUEL-CHANNEL CALCULATIONS

- **More realistic AHFPs, such as cosine, upstream-skewed and downstream-skewed cosine profiles are analyzed.**



SCWR FUEL-CHANNEL CALCULATIONS

- **Since UO_2 (commonly used in currently operating reactors) has a very low thermal conductivity, alternative nuclear fuels were considered (UC, UN, UC_2, ThO_2, MOX).**



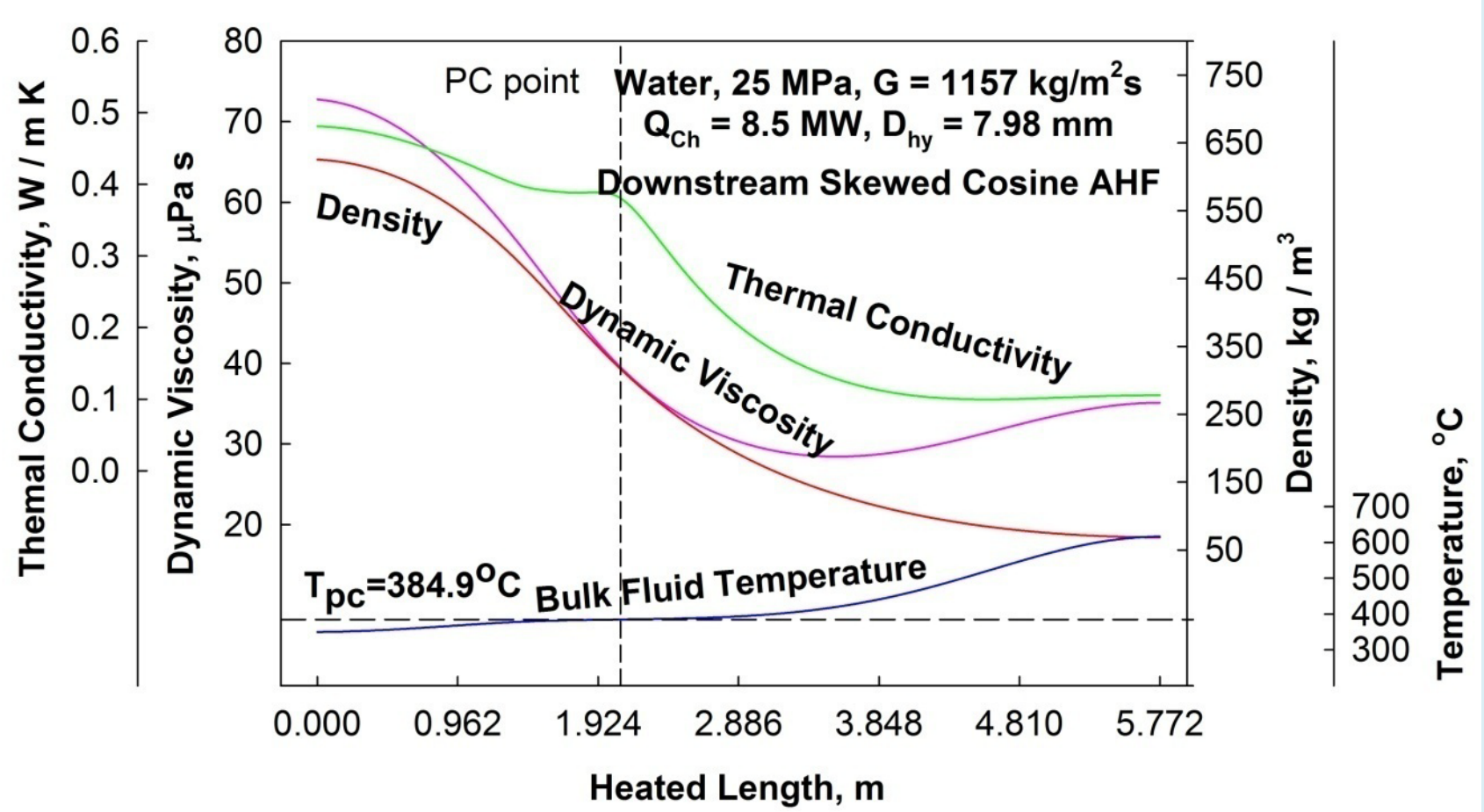
SCWR FUEL-CHANNEL CALCULATIONS

- In general, many parameters (density, porosity, manufacturing method, etc.) affect thermal conductivity of any potential fuel.
- As such, only generic thermal conductivities of nuclear fuels were used in the calculations.

Property	Units	Fuel					
		UO_2	MOX	ThO_2	UN	UC	UC_2
Molar mass	kg/kmol	270.3	271.2	264	252	250	262
Theoretical density	kg/m_3	10960	11,074	10,000	14,300	13,630	11,700
Melting temperature	°C	2850	2750	3227	2850	2365	2800
Boiling temperature	°C	3542	3538	> 4227	—	4418	—
Heat of fusion	kJ/kg	259	286	—	—	196	—
Specific heat	kJ/kg·K	0.235	0.240	0.235	0.190	0.200	0.162
Thermal conductivity	W/m·K	8.68	7.82	9.7	13.0	25.3	13

SCWR FUEL-CHANNEL CALCULATIONS

- **All thermophysical properties (calculated using NIST REFPROP 2007) undergo significant and drastic changes within the pseudocritical region.**
- **The pseudocritical region location along the fuel channel length depends on the AHFP applied.**



SCWR FUEL-CHANNEL CALCULATIONS

Methodology and Calculations

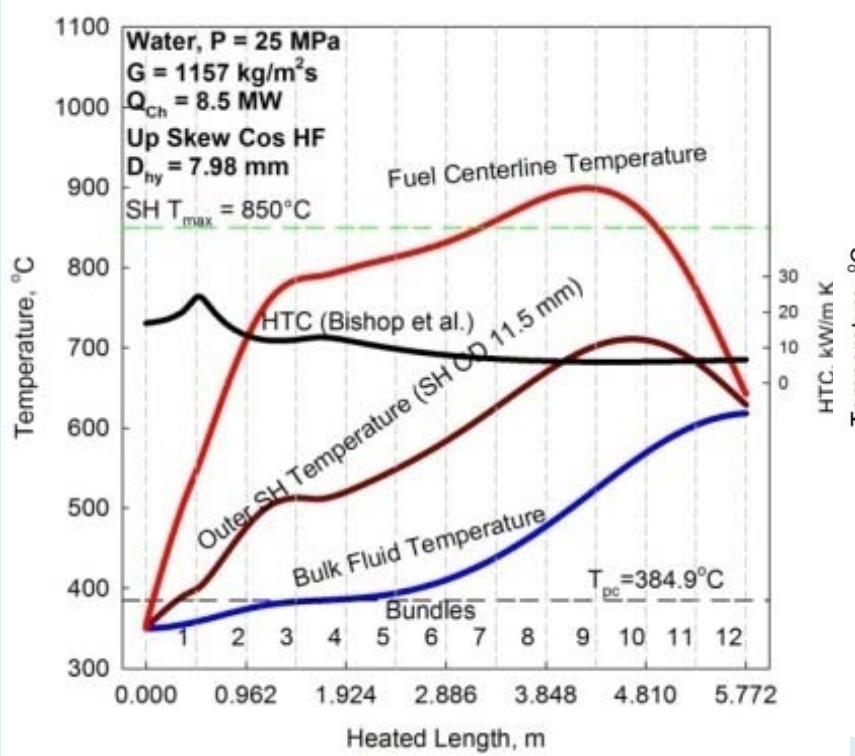
- **Thermophysical properties of coolant at sheath temperature, and thermal conductivities of sheath and fuel calculated using an iterative method.**
- **Coolant properties estimated on bulk-fluid temperature (i.e., average coolant temperature in cross-section) and wall temperature.**
- **Calculations performed along heated-bundle length with 1-mm increment.**
- **Bishop et al. correlation used to determine HTC along fuel channel:**

$$Nu_b = 0.0069\,\mathrm{Re}_b\,\overline{\mathrm{Pr}}^{0.66}\left(\frac{\rho_w}{\rho_b}\right)^{0.43}\left(1+2.4\frac{D}{x}\right)$$

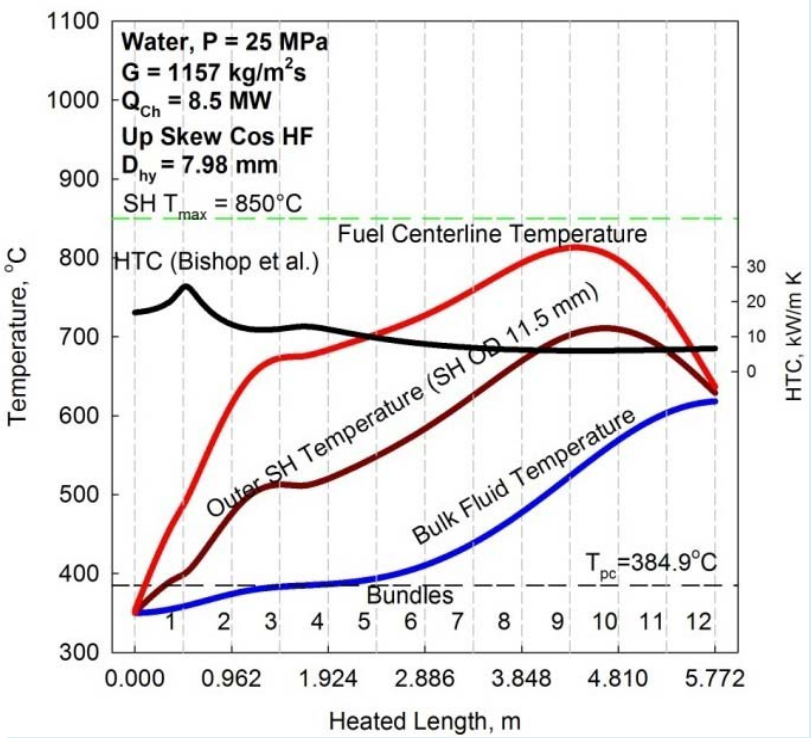
- **Term representing entrance effect in a bare tube was neglected due to the various appendages attached to the fuel bundle (endplates, etc.).**
- **Fuel centerline temperature determined by small radial increments with variable thermal conductivity.**

SCWR FUEL-CHANNEL CALCULATION RESULTS

Upstream Skewed Cosine AHFP



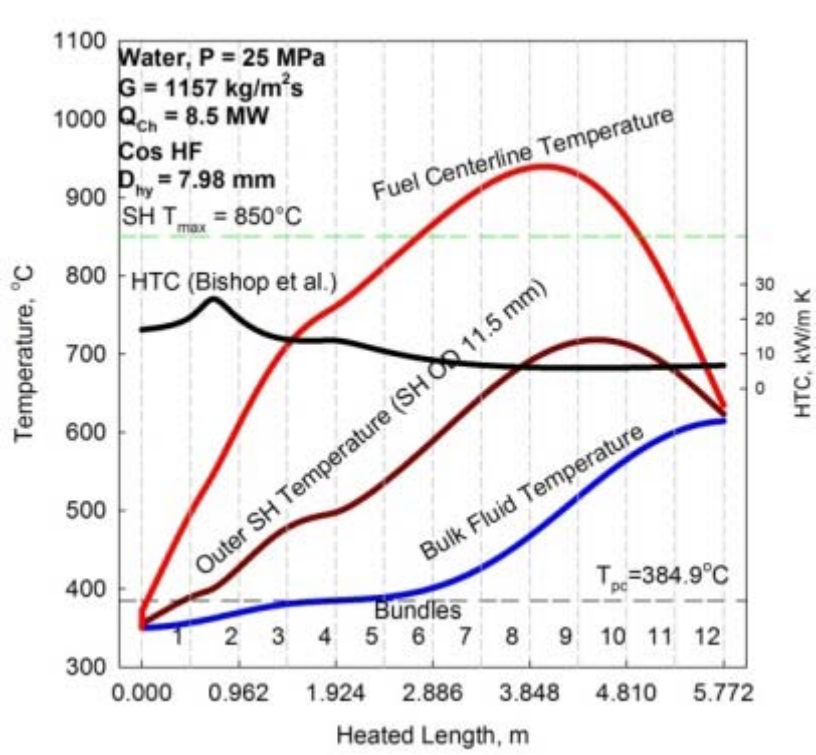
Temperature and HTC profiles for UC_2 fuel



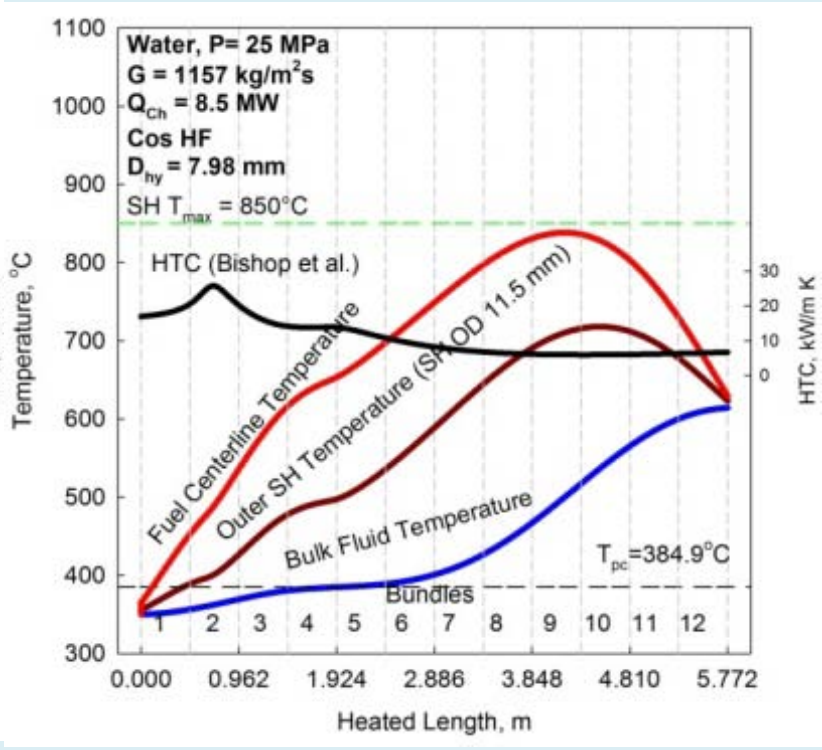
Temperature and HTC profiles for UC fuel

SCWR FUEL-CHANNEL CALCULATION RESULTS

Cosine AHFP



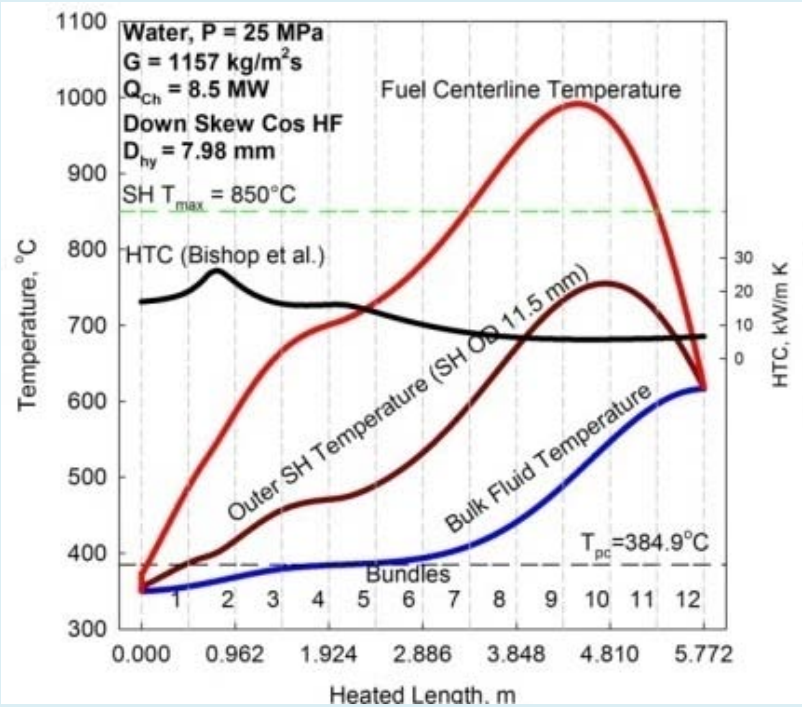
Temperature and HTC profiles for UC_2 fuel



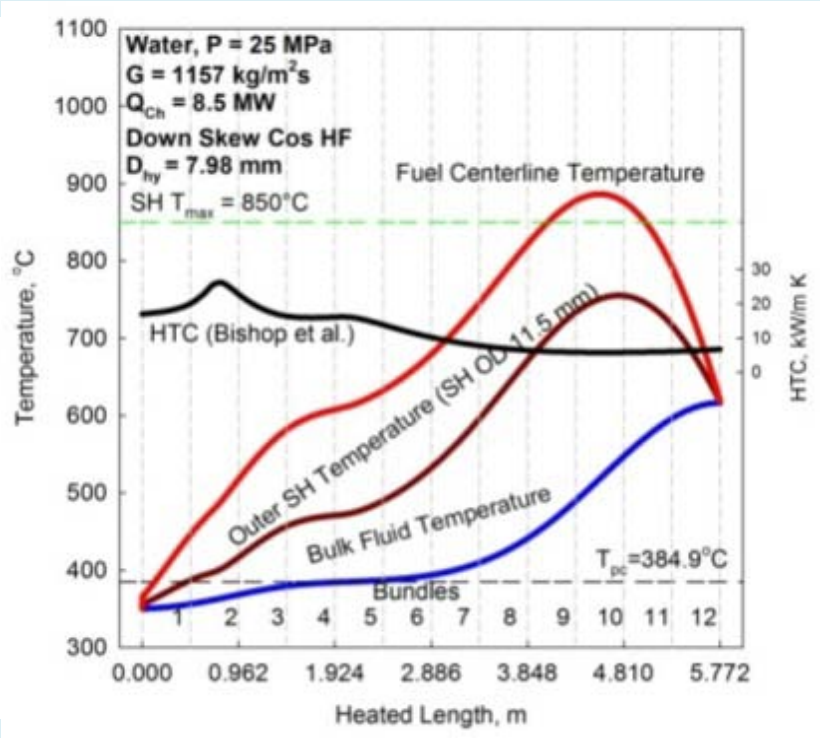
Temperature and HTC profiles for UC fuel

SCWR FUEL-CHANNEL CALCULATION RESULTS

Downstream Skewed Cosine AHFP



Temperature and HTC profiles for UC_2 fuel



Temperature and HTC profiles for UC fuel

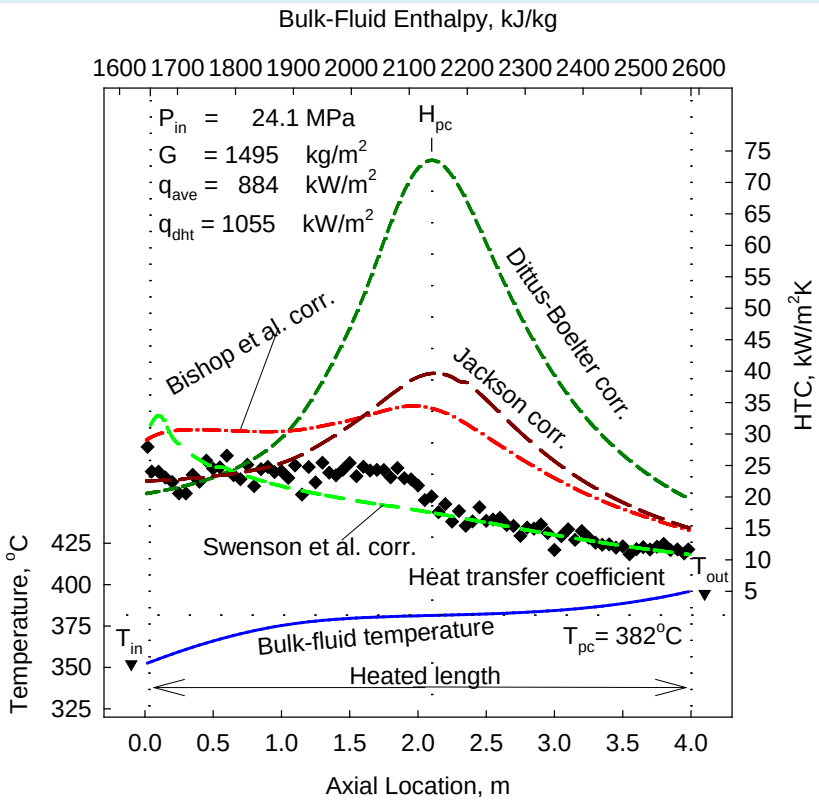
NEW HT CORRELATION FOR SUPERCRITICAL WATER

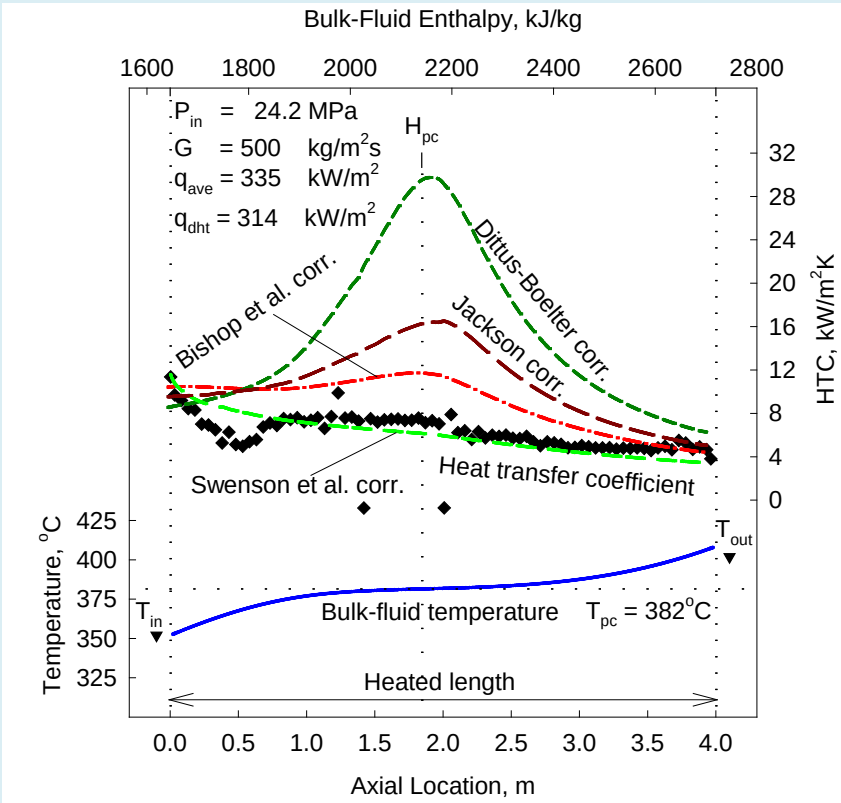
- **Large set of experimental data obtained in Russia was analyzed with the objective of developing a new heat-transfer correlation for supercritical water.**

- **Experimental conditions for the dataset:**
  - **4-m long vertical tube with 10-mm ID**
  - **Pressure: 24 MPa**
  - **Inlet temperatures: 320 to350°C**
  - **Mass fluxes: 200 to 1500 kg/m^{2}s**
  - **Heat fluxes up to 1250 kW/m^2**

NEW HT CORRELATION FOR SCW

- **Previously developed correlations do not match the experimental dataset for supercritical water:**
  - Dittus-Boelter correlation significantly overestimates experimental HTC values within pseudocritical range.
  - Bishop et al. and Jackson correlations deviated substantially from experimental data within the pseudocritical range.
  - Swenson et al. correlation provided a good fit for certain flow conditions, but not for others.





NEW HT CORRELATION FOR SUPERCRITICAL WATER

- A dimensional analysis was conducted using the Buckingham Π-theorem to derive a general form of empirical supercritical water heat-transfer correlation for the Nusselt number.

- Based on the dataset obtained in Russia, the new heat-transfer correlation is:

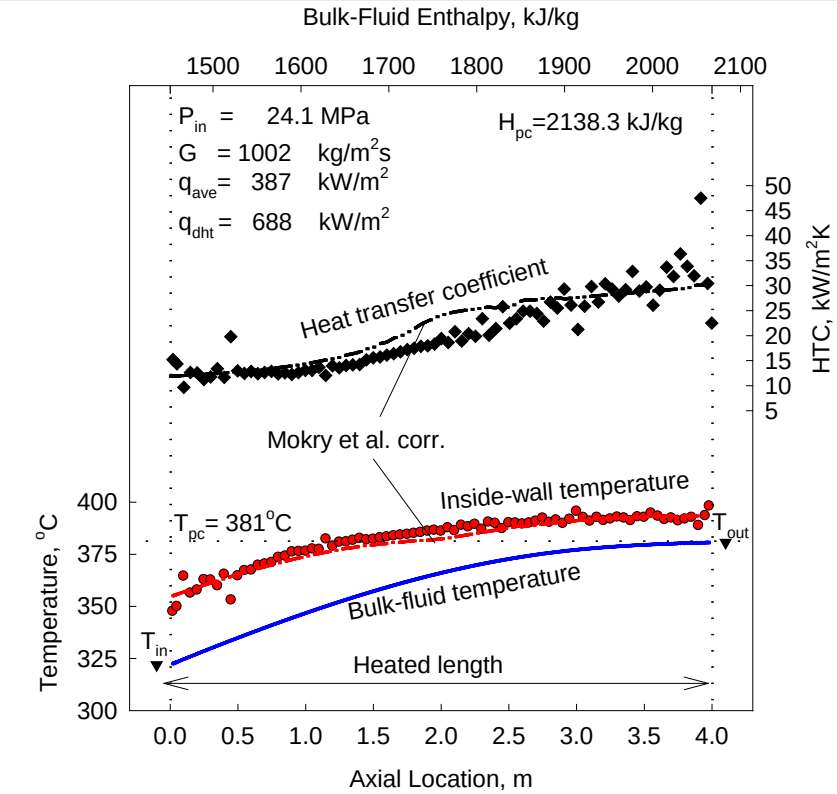
$$Nu_b = 0.0061\,\mathrm{Re}_b^{0.904}\,\overline{\mathrm{Pr}}^{0.684}\left(\frac{\rho_w}{\rho_b}\right)^{0.564}$$

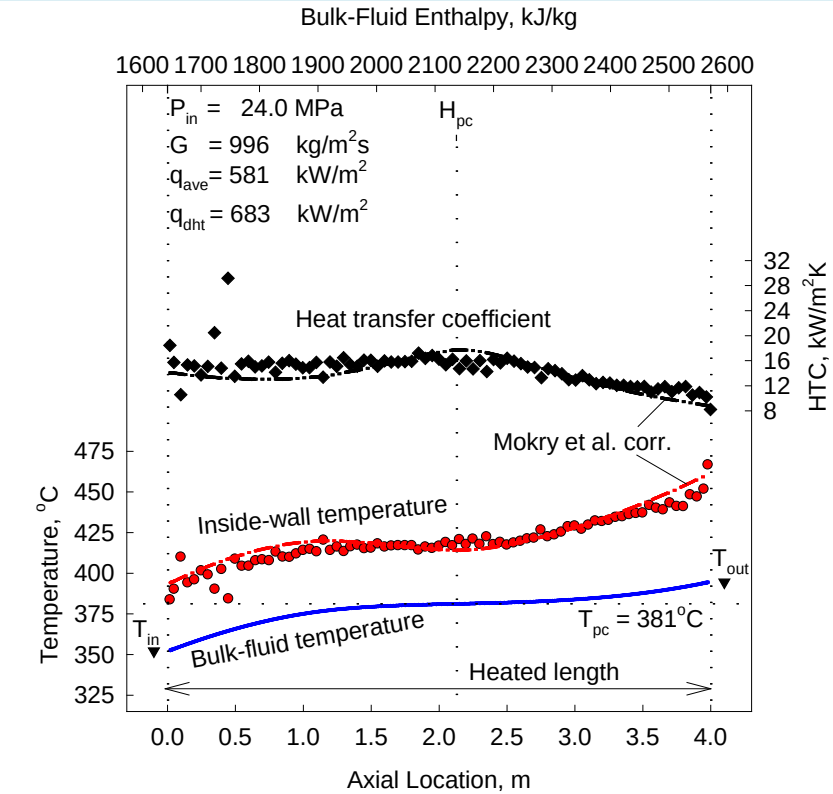
- It has uncertainty of about ±25% for HTC values and about ±15% for calculated wall temperatures.

- The derived correlation can be used for HTC calculations of SCW heat exchangers, for preliminary HTC calculations in SCWR fuel bundles, for future comparison with other datasets, for verification of computer codes and scaling parameters between water and modeling fluids.

NEW HT CORRELATION FOR SUPERCRITICAL WATER

Temperature and HTC variations at various heat fluxes along 4-m circular tube (D=10 mm, P_{in}=24.0 MPa and G=1500 kg/m^{2}s).





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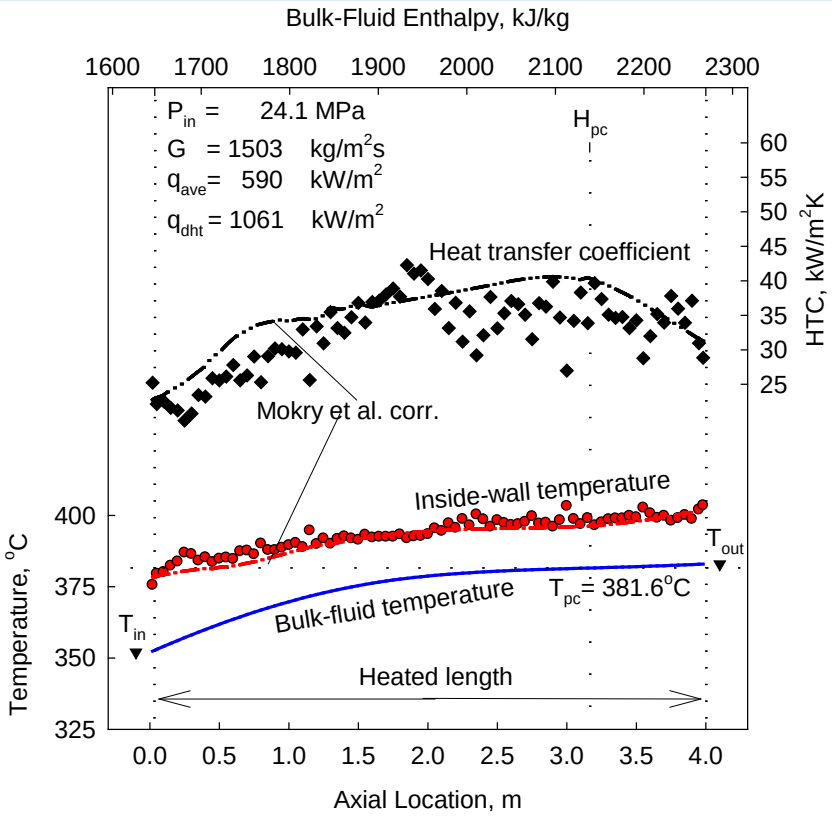

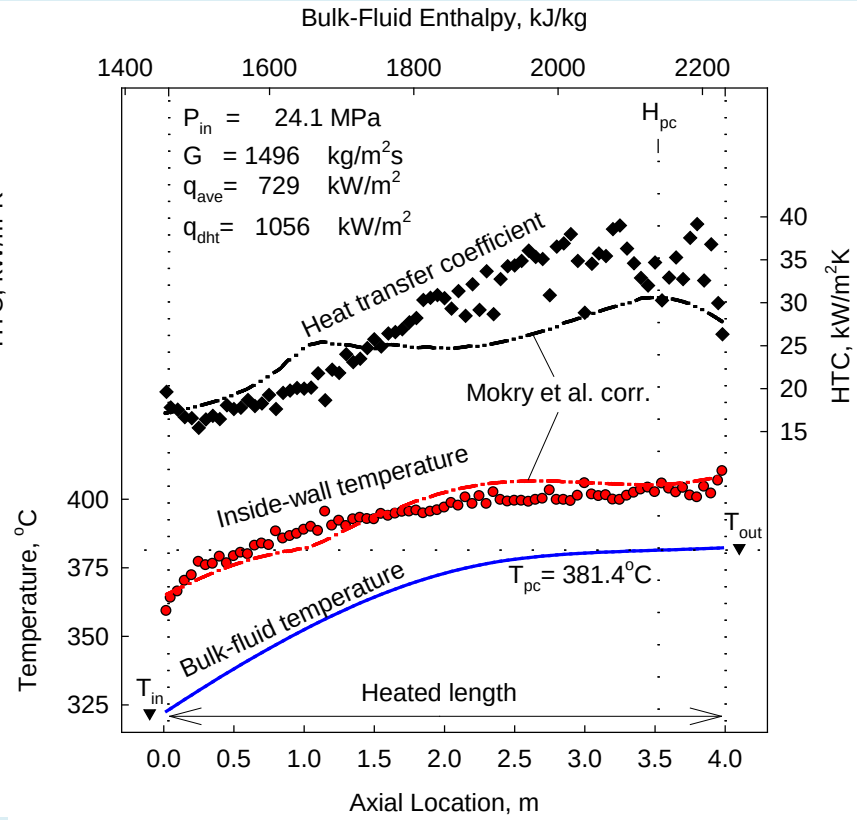

CONCLUSIONS

- **The vast majority of the modern SC turbines are single-reheat-cycle turbines. Just a few double-reheat-cycle SC turbines have been manufactured and put into operation. However, despite their efficiency benefit double-reheat-turbines have not been considered economical.**

- **In order to maximize the thermal-cycle efficiency of the SCW NPPs it would be beneficial to include nuclear steam reheat. Advantages of a single-reheat cycle in application to SCW NPPs are:**

  - **High thermal efficiency (45 – 50%), which is the current level for SC thermal power plants and close to the maximum thermal efficiency achieved in the power industry at combined-cycle power plants (up to 55%).**
  - **High reliability through proven state-of-the-art turbine technology; and**
  - **Reduced development costs accounting on wide variety of SC turbines manufactured by companies worldwide.**

CONCLUSIONS

- The major disadvantage of a single-reheat cycle implementation in SCW NPPs is the requirement for significant changes to the reactor-core design due to addition of the nuclear steam-reheat channels at subcritical pressures.
- Based on the abovementioned analysis, the single-reheat cycle with heat regeneration and the corresponding arrangement appear to be the most advantageous as a basis for a SCW NPP with the co-generation of hydrogen.
- In general, UO_2 nuclear fuel might not be a good choice for SCWRs, because at certain conditions the fuel centerline temperature exceeds the industry accepted limit of 1850°C.
- UC, UN and UC_2 fuels with significantly higher thermal conductivities compare to that of UO_2, MOX and ThO_2 should be considered as potential alternatives. However, further investigation would be required into their properties as they are new fuels.
- UC nuclear fuel with its highest thermal conductivity values compared to that of other nuclear fuels (UO_2, MOX, ThO_2, UN and UC_2) will have the largest safety margin for the fuel centerline temperature.

CONCLUSIONS

- **The following supercritical-water heat-transfer dataset obtained in a vertical bare tube was used for development of a new heat-transfer correlation and its comparison with the experimental data, with other correlations from the open literature and with FLUENT CFD code within: P=24 MPa, T_{in} = 320 – 350°C, G = 200 – 1500 kg/m^{2}s and q ≤ 1250 kW/m^2. This dataset was obtained within the SCWR operating conditions.**

- **The derived correlation showed the best fit for the experimental data within a wide range of flow conditions. This correlation has uncertainty about ±25% for HTC values and about ±15% for calculated wall temperature.**

THANK YOU!

Questions?