



# **Use of EPICS and python technology for development of a computational toolkit for high heat flux testing of plasma facing components**

***R. Sugandhi, R. Swamy and S. Khirwadkar  
Institute for Plasma Research***

10<sup>th</sup> IAEA technical meeting on control, data acquisition and remote participation for fusion research, 20-24 Apr 2015, Ahmedabad

# Outline of the talk

- **Introduction**
  - Requirements of high heat flux testing of PFC
  - Critical heat flux phenomena
  - Context for Parametric optimization
- **Computational toolkit description**
  - Design
  - Implementation: EPICS , Python and pyepics
- **Results**
- **Future scope of work and conclusion**

# Computational Simulations and virtual Experimentation

- Computer simulation facilitate testing of all the feasible test cases. It is a useful aid for the predicting experiments where operational cost is very high.
- Provides flexibility of parameters variation and understading of phenomena and operational regimes
- Open source technologies are matured and provides rich programming APIs

# Divertors in a tokamak

- Divertors are important plasma facing components.
- Used to exhaust He ash and heat flux and control of impurities and fuel density.
- Absorb high heat load to improve performance of tokamak

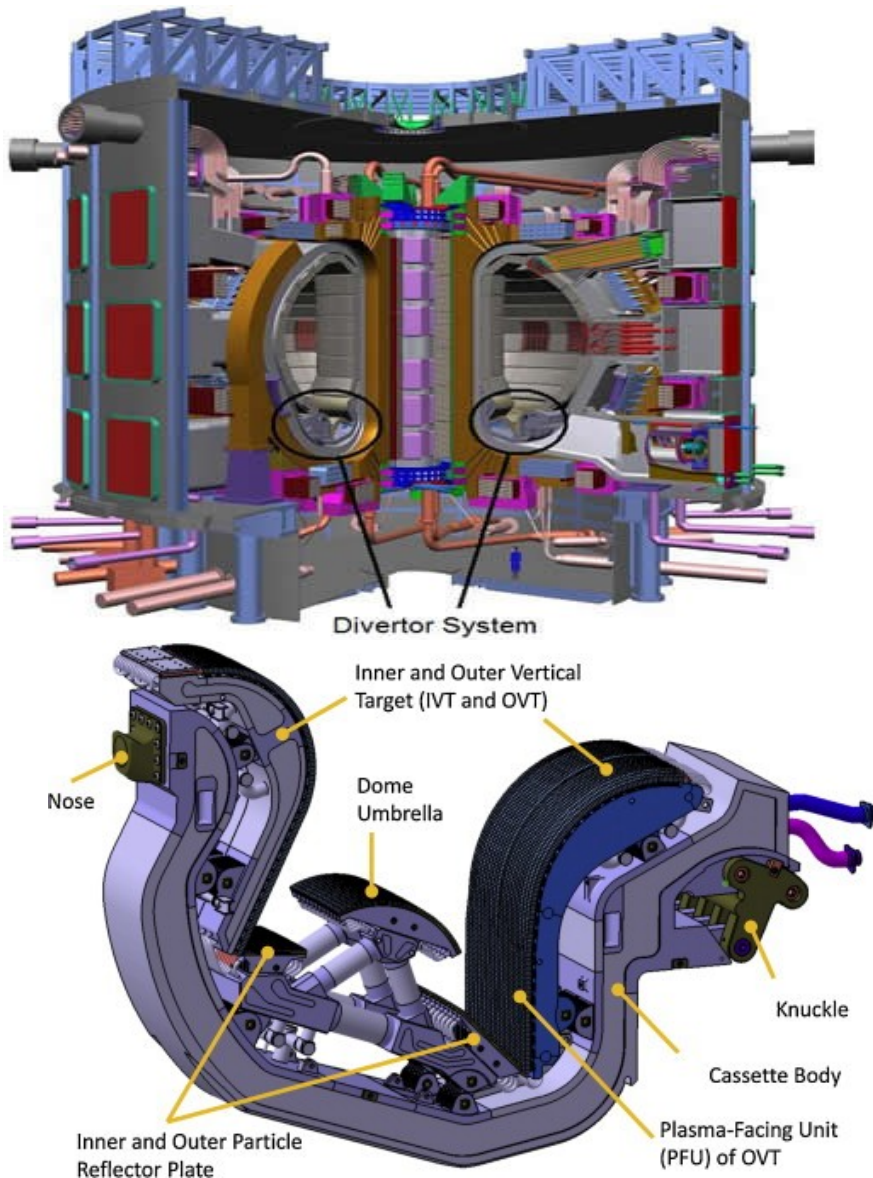
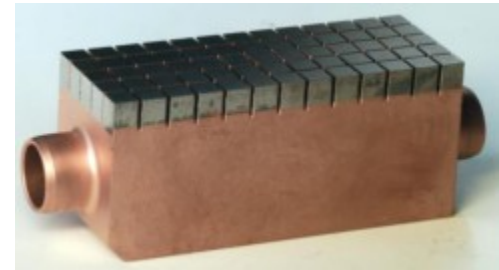


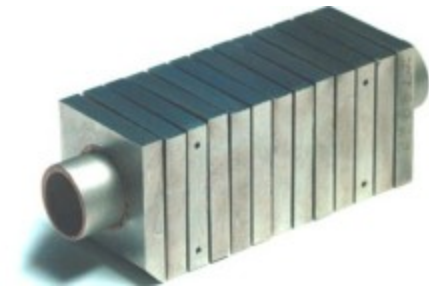
Figure 1: Main components of a ITER tungsten divertor cassette [1]

# Complexities in Divertor design

- Subject to high heat load of 5-20 MW/m<sup>2</sup>
- Design challenges:
  - Requires materials to withstand intense heat load
  - **Cooling system to protect system from burnout and environment issues**
- Operational stability under:
  - Static load conditions of plasma
  - Transient load conditions  
(in case of instability)

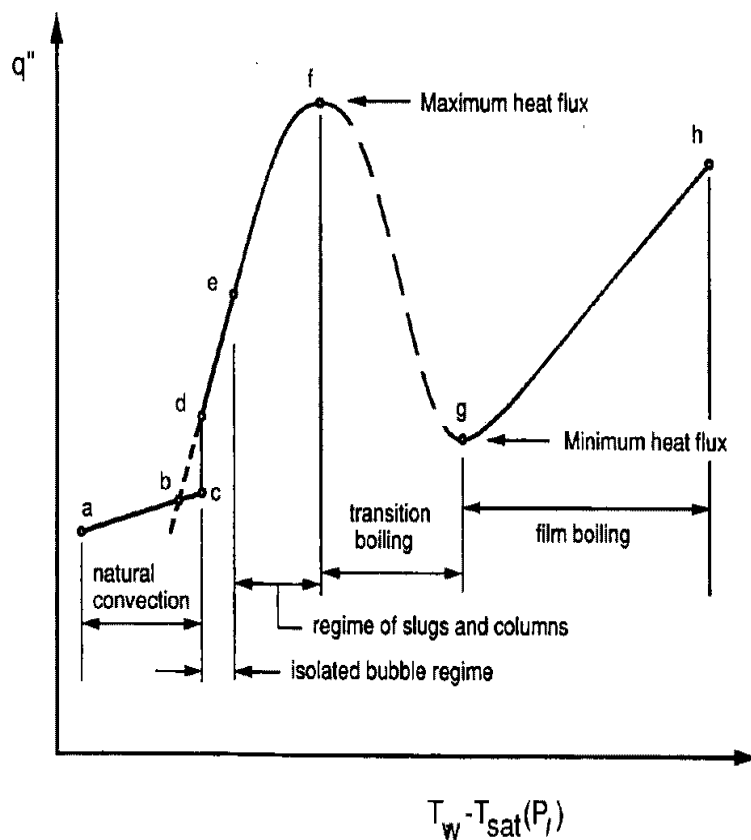
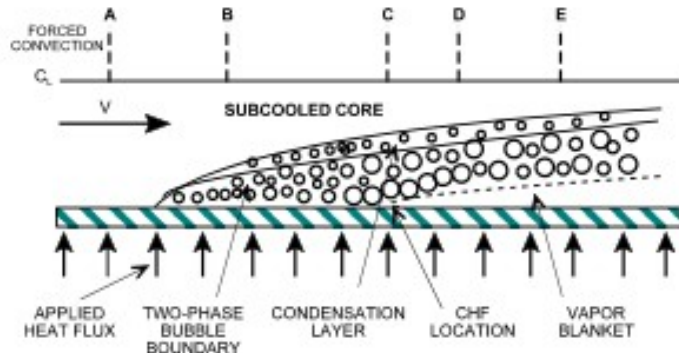


**Tungsten Macro brush**



**Tungsten Monoblock**

# Critical Heat Flux (CHF) phenomena



- Describes loss of liquid layer or phase change at the wall which can lead to decrease the efficiency of head transfer thus burn out.
- Accurate prediction of CHF is must for a safe design.

# Thermal hydraulic correlations

- Convection heat transfer coefficient is given by

$$h = \text{Nu} \cdot k / d \quad Q = m' C_p \Delta T$$

Where,

**Nu** - Nusselt number

**k** - Conductivity of coolant (W/mK)

**d** - Inside diameter of the tube (m);

**Q** = Rate of heat energy removed (J/s)

**m'** = mass flow rate (Kg/s)

**C<sub>p</sub>** = Specific Heat Capacity (J/Kg.K)

**ΔT** = coolant temperature rise

- The Reynolds number (Re), Prandtl number (Pr) and the Nusselt Number (Nu) are given by the relations

$$\text{Re} = \rho V d / \mu \quad \text{Pr} = \mu C_p / k \quad \text{Nu} = 0.023 \text{Re}^{0.8} \text{Pr}^{0.4}$$

Where,

**ρ** – Density of the fluid

**μ** – Dynamic viscosity (Kg/m.s)

**V** – Velocity of the fluid (m/s)

**d** – Inside diameter of the tube (m)

**k** – Conductivity of coolant (W/mK)

# Tong-75 CHF correlation [3]

➤ CHF model for one sided heating condition of fusion devices are modeled by many relations. Tong-75 correlation has shown good agreement with experiments. It is a semi empirical model and also used for thermo-hydraulic analysis of ITER divertors.

$$CHF_w = 0.23 f G H_{fg} \left(1 + 0.00216 \left(\frac{P}{P_C}\right)^{1.8} \text{Re}^{0.5} Ja\right)$$

$$f = 8 \text{Re}^{-0.06} \left(\frac{d_h}{d_o}\right)^{0.32}$$

$$Ja = \frac{\rho_f C_p (T_{Sat} - T)}{\rho_g H_{fg}}$$

$$\text{Re} = \frac{GD}{\mu_f}$$

Where  $CHF_w$  is the critical heat flux at the tube wall,  $G$  is the coolant mass velocity,  $T$  is the local coolant temperature,  $P$  is the local coolant pressure,  $T_{sat}$  is the saturation temperature corresponding to  $P$ ,  $H_{fg}$  is the latent heat of vaporization of water at  $T_{sat}$ ,  $P_C$  is the critical pressure,  $\text{Re}$  is the Reynold number,  $d_h$  is the hydraulic diameter,  $\mu_f$  is the water viscosity at  $T$ ,  $Ja$  is the Jakob number,  $\rho_f$  is the water density at  $T$ ,  $\rho_g$  is the vapour density at  $T_{sat}$ ,  $C_p$  is water specific heat,  $d_o$  is reference diameter



# Computational complexity and parametric Optimization

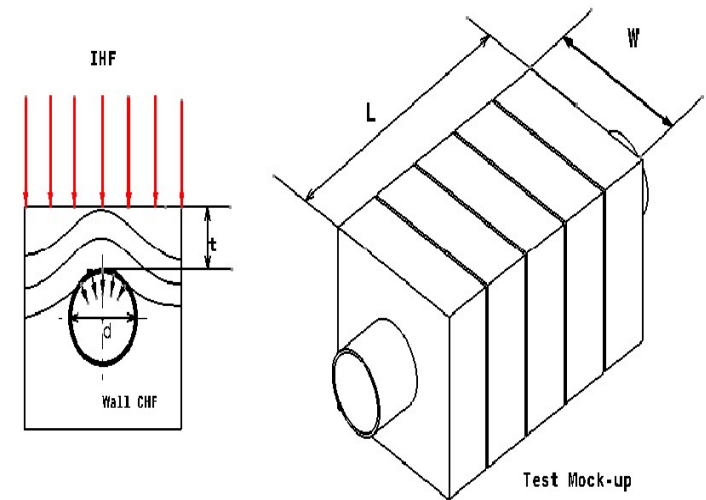
- **Computational complexity:**

- Non-linear inter parameter dependency and curve fitting required for the CHF computation
- Thermo physical properties for water are taken from NIST database

$$\{T, C_p\}, \{T, \rho\}, \{T_{sat}, \mu_f\}, \{P, T_{sat}\}, \{P, H_{fg}\}, \{\rho, T_{sat}\}$$

- **Parametric Optimization:**

- Find best local cooling condition viz. Pressure, flow and temperature for maximum heat transfer using parametric optimization of CHF relation such that steady state wall heat flux is maintained.
- Constraint Optimization by linear approximation (COBYA) technique is used for optimization for the parametric optimization.



**Figure 3: Schematic illustrating the peaking of heat flux to the coolant for a given incident heat flux**

# High Heat flux Test facility (HHFTF) at IPR

- High heat flux facilities is commissioned to test the thermal performance of divertor mock up and cooling system under intense heat exposure.
- It will use electron gun (200KW) as source and high pressure and temperature water cooling system (under procurment).

**Need to design an integrated toolkit enriched with computational routines and an experimental framework for simulation and interface to the sensors and transducers**

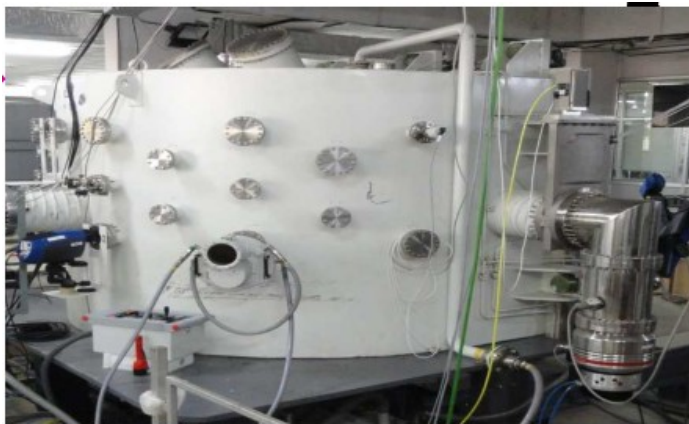


Figure 4: Vacuum System of HHFTF

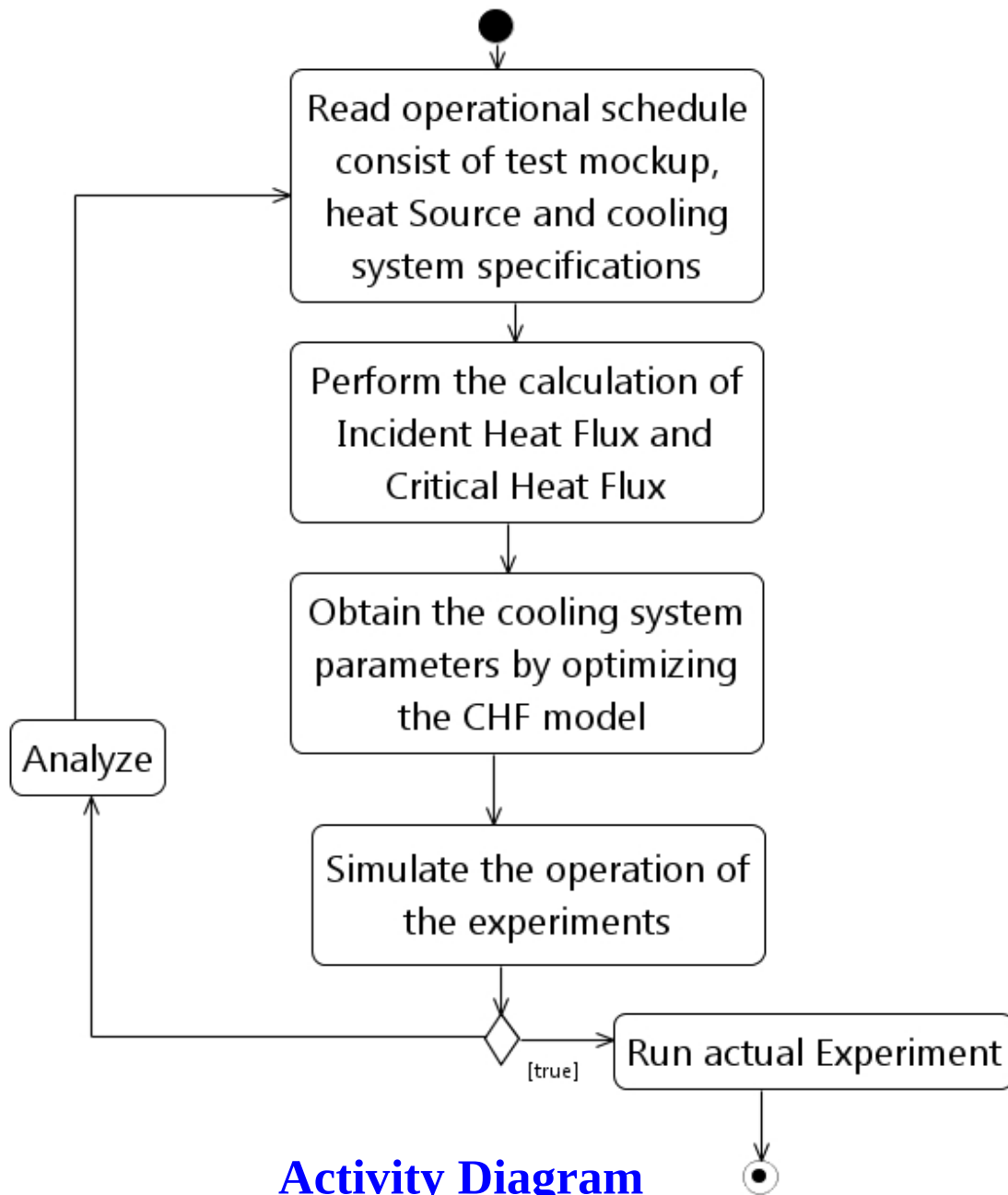


Figure 5: 200 kW Electron Gun

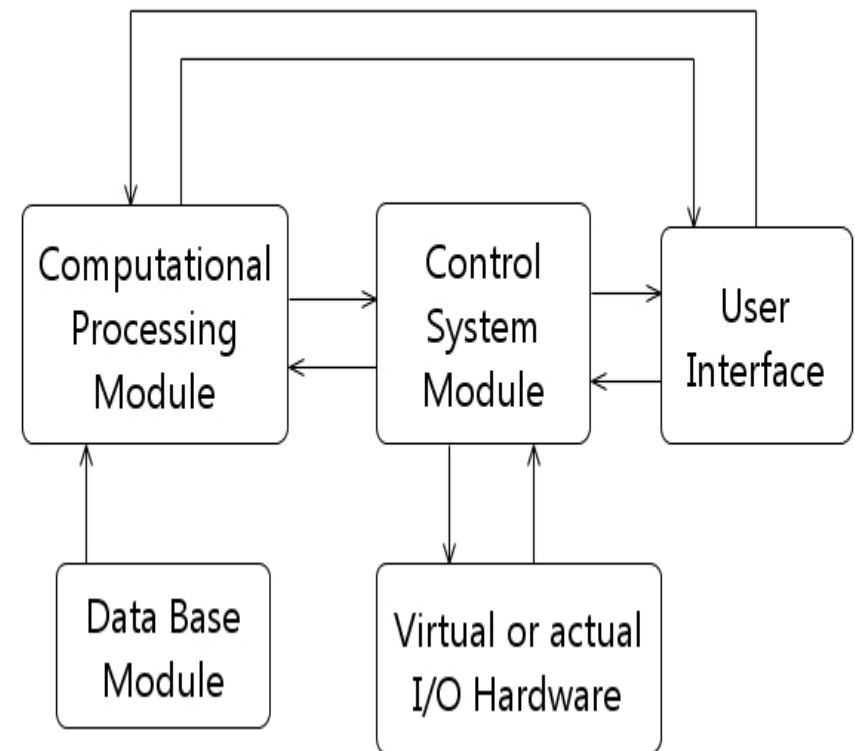
# Software development targets

- Develop computation code to predict the optimum cooling system parameters of pressure, temperature and flow.
- Graphical user interface
- I&C hardware integration and simulation flexibility.
- Provide a virtual simulation of the system operation using optimized parameters.
- Development using open source softwares and relevant to fusion technology road map

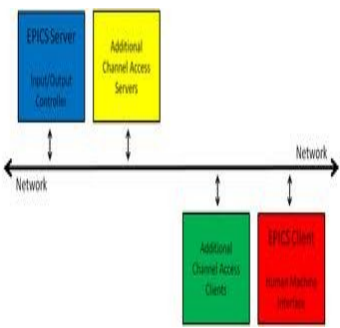
# Design Description



**Activity Diagram**



**Architectural Diagram**



# Implementation (1/3)



## EPICS

### ( Experimental Physics and Industrial Control System )

- A rich control system development framework for I&C integration , Open source, Used at around ~350 labs world wide (including ITER)
- Rich tools for data display, archivals and alarms are available.
- Support a good user interface toolkit like control system studio, which is based on eclipse and has python interface.
- I/O simulation support

# Implementation (2/3)

- **Python:**

- Used for computational processing module development
- Support object oriented and modular programming
- Scripting language, clear indentation, popular and open source
- Rich computational libraries: Numpy , Scipy Matplotlib
- Support test framework e.g. NOSE

- **Postgress Database:**

- Used to hold NIST database.

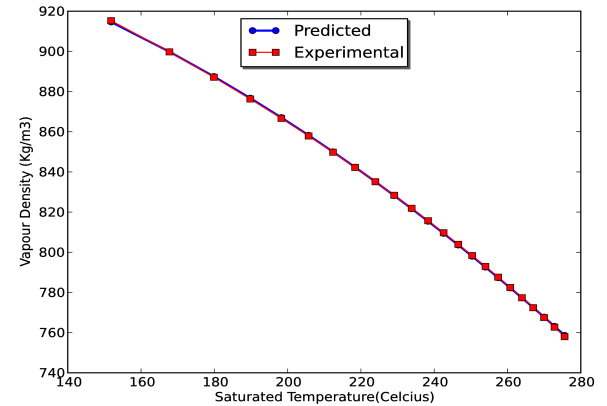
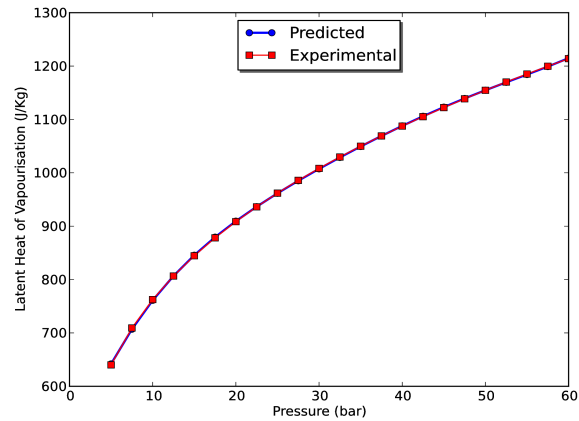
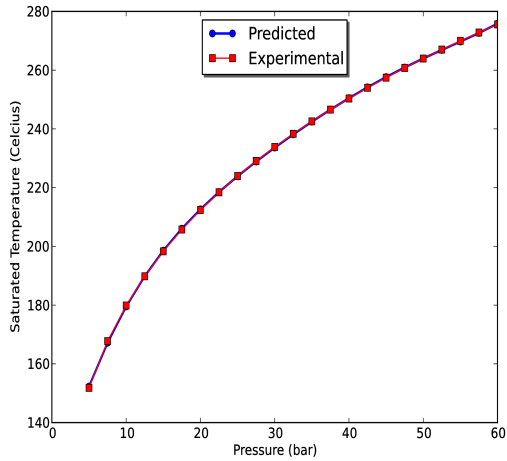
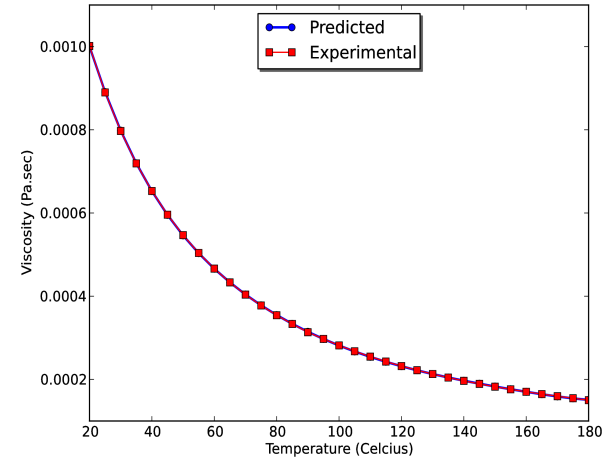
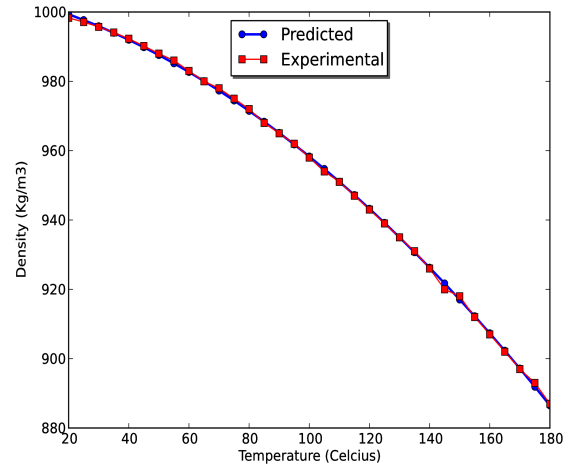
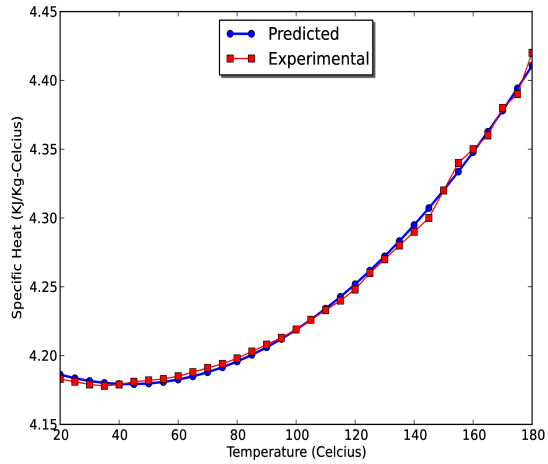


# Implementation (3/3)

- **pyepics:**

- This library is used to provide the interface between EPICS and python .
- Used at university of chicago
- offers object oriented and functional form of interface.
- And process variable processing capabilities.
- Well documented and can be used where extensive simulation is required.

# Results : Curve fitting





# Results : Heat flux calculations

user\_interface\_module1.opi Simulator Project(SIMP) X

**INTRODUCTION COMPUTATIONS MEASURE ANALYSIS**

**CHFw**

- (Pressure)
- (Temperature)
- (Flow Rate)
- (Tube Diameter)
- (Pressure,Temp)
- (Pressure,Flow)
- (Pressure, Tube dia)
- (Temp, flow)
- (Temp,Tube dia)
- (Flow,Tube dia)

CHF CALC

**COOLING SET POINTS :**

Pressure (Bar) = 15 Temperature (Celc) = 82 Flow Rate (lpm) = 150 Tube Diameter (mm) = 10

**SOURCE POWER (ELECTRONIC BEAM)**

Power(kW) = 15

**TEST MOCKUP DIMENSIONS**

Width = 30 Length (mm) = 50

**Tube Type**

- SmoothTube
- Swirl Tube
- Screw Tube
- Hypervapotron Tube

**Area of surface (mm<sup>2</sup>)** 1500.00 mm<sup>2</sup>

**Area of tube (mm<sup>2</sup>)** 785.00 mm<sup>2</sup>

**Incident Heat Flux (IHF)** 10.00 MW/m<sup>2</sup>

**Inerwall Heat Flux (IWHF)** 19.11 MW/m<sup>2</sup>

**Peaking Facto(fp)** 1.40

**Total Heat Flux** 26.75

**Tong 75 Coorelation**

$$CHF_w = 0.23 f G H_{fg} (1 + 0.00216 (P/P_c)^{1.8} Re^{0.5} Ja) C_f$$

26.724043 MW/m<sup>2</sup>

# Results: CHF vs Pressure and optimization

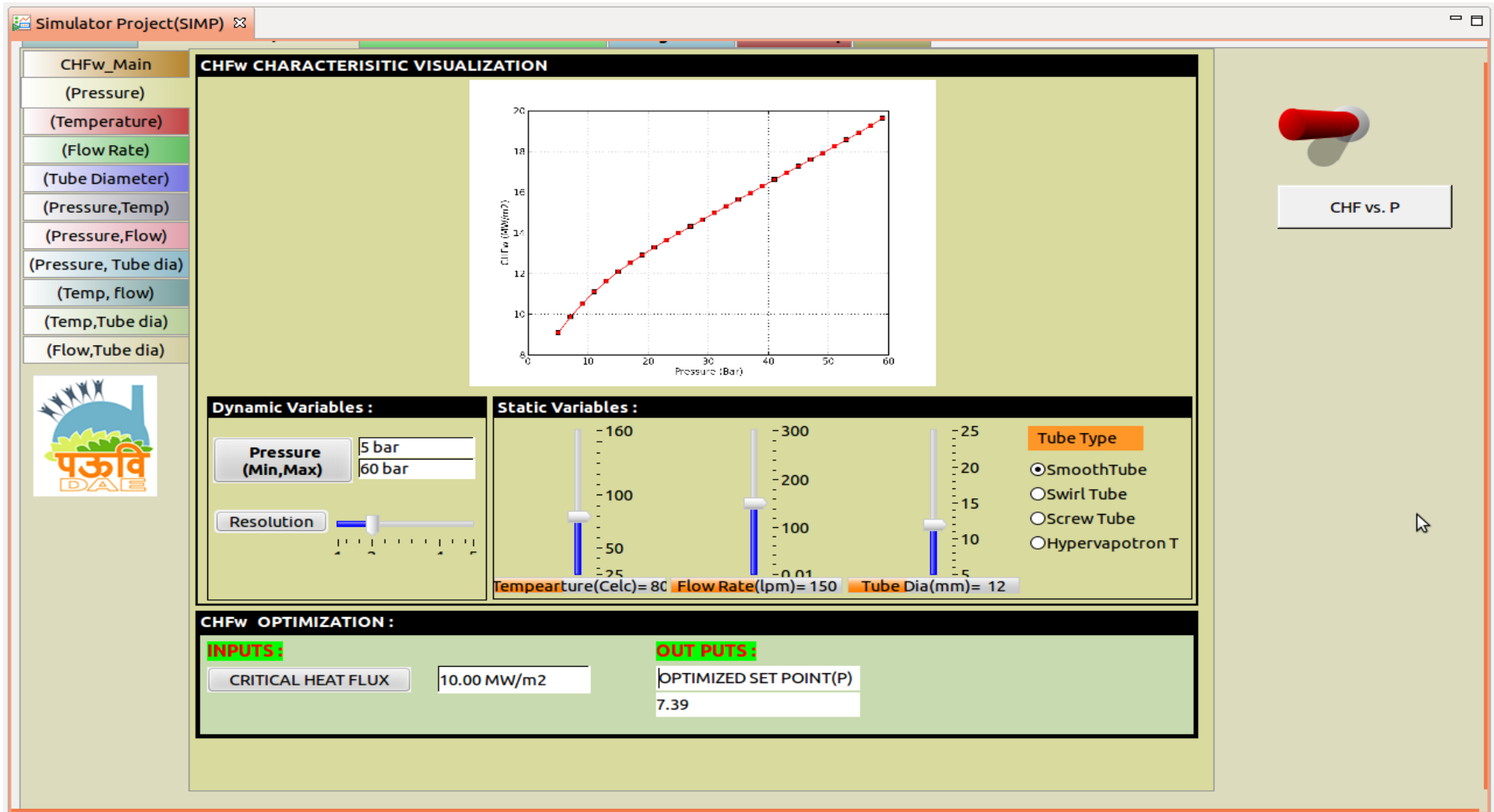


Figure: CHF Optimization and Pattern Visualization

# Results: Parametric Optimization

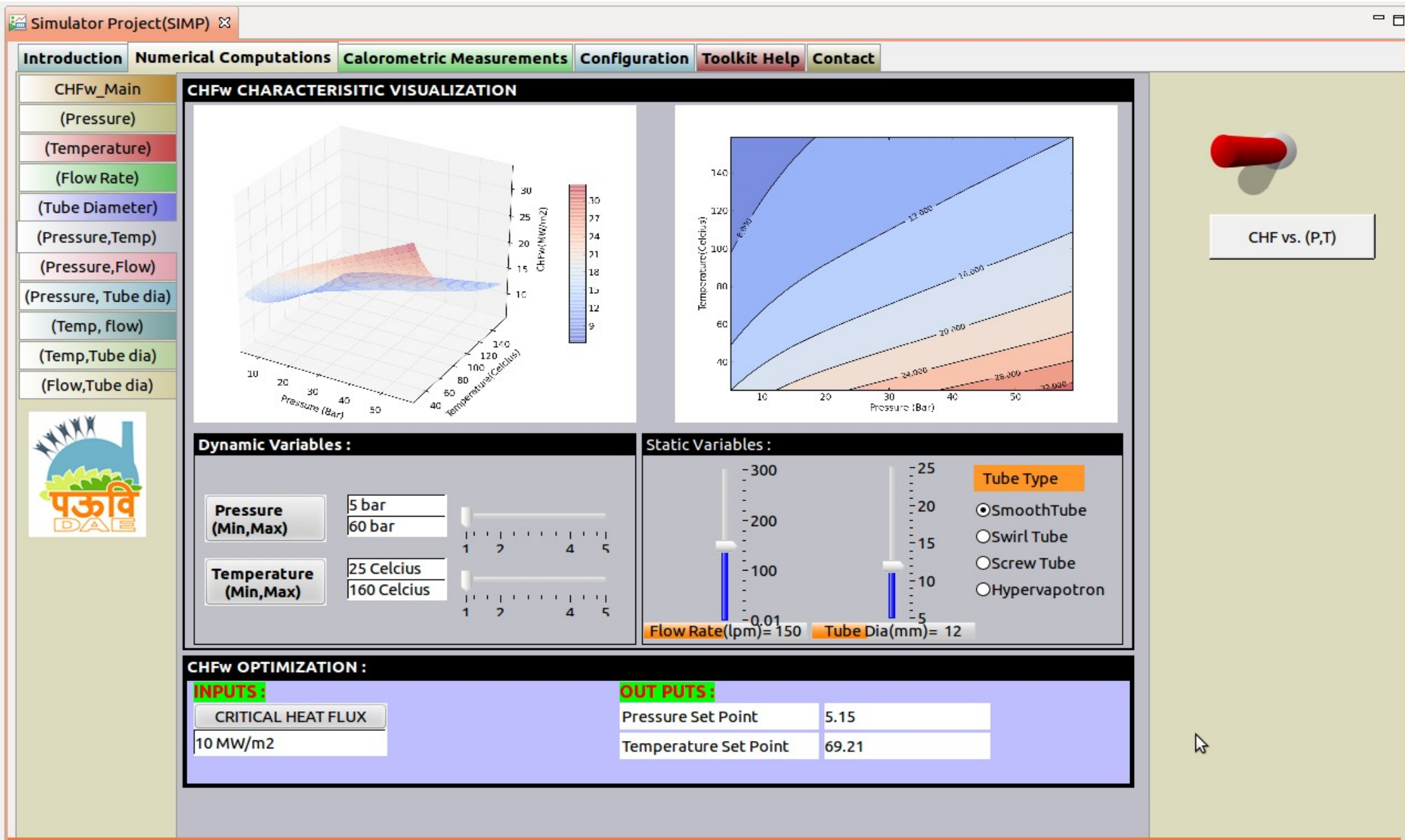


Figure: Parametric Optimization

# Results: Heat transfer simulation on optimised set points



# Validation

- NOSE frame work of python is used for automated testing of simulated test cases and heat transfer coorelations.
- Published data of international 2006 CHF database table is used as a referance for validation.
- Results of optimization are validated using graph plotting.

# Conclusion and Future directions

- A integrated tool kit having experimental and computational features is presented. Useful for the high heat flux test experiments of similar nature
- The parametric optimization offers the required parameters for the operation.
- The toolkit can be extended for
  - Advanced cooling tube geometries
  - Simulation capabilities
  - Multi-objective optimization features.

# References

1. “ITER tungsten divertor design development and qualification program”, T.Hirai et.al, Fusion Engineering and Design 88 (2013) 1798-1801.
2. “Modeling the Nukiyama curve for water cooled Fusion Divertor channels”, Theron D.Marshall, Technical report
3. L.S.Tong “ A Phenomenological study of Critical Heat Flux”, ASME Paper 75-HT-68 (1975).
4. “The 1995 look-up table for critical heat flux in tubes”, D.C. Groeneveld et.al., Nuclear Engineering and Design 163 (1996).
5. “ The 2006 CHF Lookup Table”, D.C. Groeneveld et.al, Nuclear Engineering and Design 237 (2007) 1909–1922.
6. “Assessment of a heat transfer correlations package for water-cooled plasma-facing components in fusion reactors ‘ , S.T.Yin et.al, Nuclear Engineering and Design 146 (1994) 311-323 .
7. “Safety implications of an integrated boiling curve model for water-cooled divertor channels”, T. Marshall, Fusion Engineering and Design 63-64 (2002) 235-242.
8. “ Understanding, predicting and enhancing Critical Heat Flux”, Soon Heung Chang et.al, The 10<sup>th</sup> International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-10) Seoul, Korea, October 5-9, 2003.
9. “Critical heat flux analysis and R&D for the design of the ITER divertor”, A.R. Raffray, Fusion Engineering and Design 45 (1999) 377-407.

**Thanking you**